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

3.1 CONFORMANCE WITH NRC GENERAL DESIGN CRITERIA

3.1.1 Summary Description

This section contains an evaluation of the design bases of the plant as measured against the NRC General Design Criteria for Nuclear Power Plants, Appendix A of 10 CFR 50 effective May 21, 1971, and subsequently amended July 7, 1971. The General Design Criteria, which are divided into six groups and total 55 in number, are intended to establish minimum requirements for the design of nuclear power plants.

It should be noted that the General Design Criteria were not written specifically for the BWR; 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 a variety of interpretations. For this reason, there are some cases where conformance to a particular criterion is not directly measurable. In these cases, the conformance of plant design to the interpretation of the criterion is discussed. For each of the 55 criteria, a specific assessment of the plant design is made and a complete list of references are included to identify where detailed design information pertinent to each criterion is treated in this document.

Based on the content herein, SERI concludes that the nuclear power plant fully satisfies and is in compliance with the General Design Criteria.

3.1.2 Criterion Conformance

3.1.2.1 Group I - Overall Requirements

3.1.2.1.1 Criterion 1 - Quality Standards and Records

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

Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be

supplemented or modified as necessary to assure a quality product in keeping with the required safety function. A quality assurance program shall be established and implemented in order to provide adequate assurance that these structures, systems, and components will satisfactorily perform their safety functions.

Appropriate records of the design, fabrication, erection, and testing of structures, systems, and components important to safety shall be maintained by or under the control of the nuclear power unit licensee throughout the life of the unit.

Discussion

Structures, systems, and components important to safety are listed in Table 3.2-1 of Chapter 3. The total quality assurance program is described in Chapter 17 and is applied to the items contained in this table indicated with a Q. The intent of the quality assurance program is to assure sound engineering in all phases of design and construction through conformity to regulatory requirements and design bases described in the license application. In addition, the program assures adherence to specified standards of workmanship and implementation of recognized codes and standards in fabrication and construction. It also includes the observance of proper preoperational and operational testing and maintenance procedures as well as the documentation of the foregoing by keeping appropriate records. The total quality assurance program of the applicant and its principal contractors is responsive to and satisfies the intent of the quality-related requirements of 10 CFR 50, including Appendix B.

Structures, systems, and components are first classified in Chapter 3 with respect to their location and service and their relationship to the safety function to be performed. Recognized codes and standards are applied to the equipment in these classifications as necessary to assure a quality product in keeping with the required safety function.

Documents are maintained which demonstrate that all the requirements of the quality assurance program are being satisfied. This documentation shows that appropriate codes, standards, and regulatory requirements are observed, specified materials are used, correct procedures are utilized, qualified personnel are provided and that the finished parts and components meet the applicable specifications for safe and reliable

operation. These records are available so that any desired item of information is retrievable for reference. These records will be maintained during the life of the operating license.

The detailed quality assurance program developed by the applicant and its contractors satisfies the requirements of Criterion 1.

For further discussion, see the following sections:

a.	Principal Design Criteria	1.2
b.	Plant Description	1.2
c.	Classification of Structures, Components, and Systems	3.2
d.	Quality Assurance Program	17.2

3.1.2.1.2 Criterion 2 - Design Bases for Protection Against Natural Phenomena

Structures, systems, and components important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, 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:

a. 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 data that have been accumulated

b. Appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena

c. The importance of the safety functions to be performed

Discussion

All safety-related structures, systems, and components have been either protected from or designed to withstand the effects of natural phenomena. The natural phenomena and their magnitude are selected in accordance with their probability of occurrence at

this specific site. The designs are based upon the most appropriate margin to account for uncertainties in the historical data presented in Sections 2.3, 2.4, and 2.5. The design criteria for the structures, systems, and components affected by each natural phenomenon are presented in Sections 3.2, 3.3, 3.4, 3.5, 3.7, and 3.8. Those combinations of natural phenomena and plant-oriented accidents that are considered in the design are identified in Sections 3.8, 3.9, 3.10, and 3.11.

3.1.2.1.3 Criterion 3 - Fire Protection

Structures, systems, and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions. Noncombustible and heat resistant materials shall be used wherever practical throughout the unit, particularly in locations such as the containment and 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.

Discussion

Structures, systems, and components important to safety have been designed to meet the requirements of Criterion 3. Fire protection systems meeting the requirements of Criterion 3 have been provided.

The plant is designed to minimize the probability and effect of fires. Noncombustible and fire-resistant materials have been used in the containment, control room, components of safety features systems, and throughout the unit wherever practical to reduce fire potential. Equipment and facilities for fire protection, including detection, alarms and extinguishments, have been provided to protect both plant equipment and personnel from fire and the resultant release of toxic vapors. Both automatic and manual types of fire-fighting equipment have been provided.

Fire protection has been provided by automatic deluge, water spray, sprinkler, clean agent suppression (a term used to encompass Halon 1301, Novec 1230, or similar fire suppression agent), manual hose stations, and portable extinguishers.

Fire-fighting systems have been designed to assure that their rupture or inadvertent operation will not significantly impair systems important to safety.

The fire protection system consists of a reliable system, designed and installed in accordance with the requirements of the National Fire Protection Association, The Nuclear Energy Liability Property Insurance Association, and the applicable local codes and regulations.

The fire protection system has been provided with test valves and facilities for periodic testing. All equipment is accessible for periodic inspection.

The fire protection system is discussed in detail in subsection 9.5.1.

3.1.2.1.4 Criterion 4 - Environmental and Missile Design Bases

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

Discussion

Environmental and missile design bases are in accordance with General Design Criterion 4. Structures, systems, and components important to safety have been designed to withstand 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, assuming that nonrelated cataclysmic events do not occur simultaneously. These structures, systems, and components are 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 both inside and outside the nuclear power unit.

Discussion of dynamic effects associated with the postulated rupture of piping is contained in Section 3.6. Missile protection is discussed in Section 3.5. Section 3.11 contains a discussion of design environmental conditions.

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

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.

Discussion

Unit 2 has been canceled; therefore, this criterion is not applicable.

As noted in Section 1.1.1, after Unit 1 had received its Commercial Operating License, Entergy Operations, Inc. formally requested the NRC to revoke the Construction Permit and officially cancel the second unit at Grand Gulf Nuclear Station. Thus, General Criterion 5, Sharing of Structures, Systems, and Components is not applicable to the Grand Gulf Nuclear Station.

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3.1.2.2 Group II - Protection by Multiple Fission Product Barriers

3.1.2.2.1 Criterion 10 - Reactor Design

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

Discussion

The reactor core components consist of fuel assemblies, control rods, in-core ion chambers, neutron sources, and related items. The mechanical design is based on conservative application of stress limits, operating experience and experimental test results. The fuel is designed to provide high integrity over a complete range of power levels including transient conditions. The core is sized with sufficient heat transfer area and coolant flow to ensure that fuel design limits are not exceeded under normal conditions or anticipated operational occurrences.

The reactor protection system is designed to monitor certain reactor parameters, sense abnormalities, and to scram the reactor thereby preventing fuel design limits from being exceeded when trip points are exceeded. Scram trip set points are selected on operating experience and by the safety design basis. There is no case in which the scram trip set points allow the core to exceed the thermal hydraulic safety limits. Power for the reactor protection system is supplied by two independent ride-through ac power supplies. An alternate power source is available for each bus.

An analysis and evaluation has been made of the effects upon core fuel following adverse plant operating conditions. The results of abnormal operational transients are presented in Chapter 15 and show that MCPR does not fall below the transient MCPR limit, thereby satisfying the transient design basis.

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

For further discussion, see the following sections:

a.	Principal Design Criteria	1.2
b.	Plant Description	1.2
c.	Fuel Mechanical Design	4.2
d.	Nuclear Design	4.3
e.	Thermal and Hydraulic Design	4.4
f.	Reactor Recirculation System	5.4
g.	Reactor Core Isolation Cooling System	5.4
h.	Residual Heat Removal System	5.4
i.	Accident Analysis	15

3.1.2.2.2 Criterion 11 - Reactor Inherent Protection

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

Discussion

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

The inherent dynamic behavior of the core is characterized in terms of:

a. Fuel temperature or Doppler coefficient.

b. Moderator void coefficient.

c. Moderator temperature coefficient. The combined effect of these coefficients in the power range is termed the power coefficient.

Doppler reactivity feedback occurs simultaneously with a change in fuel temperature and opposes the power change that caused it; it contributes to system stability. Since the Doppler reactivity opposes load changes, it is desirable to maintain a large ratio of moderator void coefficient to Doppler coefficient for optimum

load-following capability. The boiling water reactor has an inherently large moderator-to-Doppler coefficient ratio which permits use of coolant flow rate for load following.

In a boiling water reactor, the moderator void coefficient is of importance during operation at power. Nuclear design requires the void coefficient inside the fuel channel to be negative. The negative void reactivity coefficient provides an inherent negative feedback during power transients. Because of the large negative moderator coefficient of reactivity, the BWR has a number of inherent advantages, such as:

a. The use of coolant flow as opposed to control rods for load following

b. The inherent self-flattening of the radial power distribution

c. The ease of control

d. The spatial xenon stability

The reactor is designed so that the moderator temperature coefficient is small and positive in the cold condition; however, the overall power reactivity coefficient is negative. Typically, the power coefficient at full power is about $-0.04k/\Delta k/\Delta P/P$ at the beginning of life and about $-0.03\Delta k/k/\Delta P/P$ at 10,000 MWD/T. These values are well within the range required for adequate damping of power and spatial xenon disturbances.

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

For further discussion, see the following sections:

a.	Principal Design Criteria	1.2
b.	Nuclear Design	4.3
c.	Thermal and Hydraulic Design	4.4

3.1.2.2.3 Criterion 12 - Suppression of Reactor Power Oscillations

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

Discussion

The reactor core is designed to ensure that no power oscillation will cause fuel design limits to be exceeded. The power reactivity coefficient is the composite simultaneous effect of the fuel temperature or Doppler coefficient, moderator void coefficient, and moderator temperature coefficient to the change in power level. It is negative and well within the range required for adequate damping of power and spatial xenon disturbances. Analytical studies indicate that for large boiling water reactors, under-damped, unacceptable power distribution behavior could only be expected to occur with power coefficients more positive than about -0.01Δk/k/ΔP/P. Operating experience has shown large boiling water reactors to be inherently stable against xenon induced power instability. The large negative operating coefficients provide:

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

b. Load following with recirculation flow control.

c. Strong damping of spatial power disturbances.

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

To prevent power oscillations caused by neutronic/thermal-hydraulic instability, GGNS has implemented NEDC-33075P-A, Detect and Suppress Solution-Confirmation Density [DSS-CD] licensing Topical Report, Revision 7 and NEDC-33075P-A, Detect and

Suppress Solution-Confirmation Density [DSS-CD] licensing Topical Report, Revision 8. This includes licensing features that prevent power oscillations due to neutronic/thermal-hydraulic instabilities from developing considering reasonably limiting anticipated operational occurrences, and defense-in-depth features that provide diverse methods to prevent the onset of power oscillations and provide protection against unanticipated and hypothetical events.

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

For further discussion see the following sections:

a.	Principal Design Criteria	1.2
b.	Nuclear Design	4.3
c.	Thermal and Hydraulic Design	4.4
d.	Rod Control and Information System	7.7
e.	Accident Analysis	15

3.1.2.2.4 Criterion 13 - Instrumentation and Control

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

Discussion

The fission process is monitored and controlled for all conditions from source range through power operating range. The intermediate and power ranges of the neutron monitoring system detect core conditions that threaten the overall integrity of the fuel barrier due to excess power generation and provide a signal to the reactor protection system. Fission detectors, located in

the core, are used for neutron detection. The detectors are located to provide optimum monitoring in the intermediate and power ranges.

The IRM monitors neutron flux from the upper portion of the SRM to the lower portion of the Local Power Range Monitor (LPRM) subsystem. The IRM is capable of generating a trip signal to scram the reactor.

The Local Power Range Monitor (LPRM) subsystem consists of fission chambers located throughout the core, the signal conditioning equipment, and trip functions. LPRM signals are also used to block rod withdrawal and to generate the necessary trip signal for reactor scram (APRM).

The reactor protection system protects the fuel barriers and the nuclear process barrier by monitoring plant parameters and causing a reactor scram when predetermined set points are exceeded. Separation of the scram and normal rod control function prevents failures in the reactor control circuitry from affecting the scram circuitry.

To provide protection against the consequences of accidents involving the release of radioactive materials from the fuel, reactor coolant pressure boundary, and the containment, the reactor vessel isolation control system initiates automatic isolation of appropriate pipelines whenever monitored variables exceed preselected operational limits.

Nuclear system leakage limits are established so that appropriate action can be taken to ensure the integrity of the reactor coolant pressure boundary. Nuclear system leakage rates are classified as identified and unidentified, which correspond respectively to the flow to the equipment drain and floor drain sumps. The permissible total leakage rate limit to these sumps is based upon the makeup capabilities of various reactor component systems. The unidentified leakage rate as established in subsection 5.2.5 is less than the value that has been conservatively calculated to be a minimum leakage from a crack large enough to propagate rapidly, but which still allows time for identification and corrective action before integrity of the process barrier is threatened.

The process radiation monitoring system monitors radiation levels of various processes and provides trip signals whenever pre-established limits are exceeded.

As noted above, adequate instrumentation has been provided to monitor system variables in the reactor core, reactor coolant pressure boundary, and reactor containment. Appropriate controls have been provided to maintain the variables in the operating range and to initiate the necessary corrective action in the event of abnormal operational occurrence or accident.

3.1.2.2.5 Criterion 14 - Reactor Coolant Pressure Boundary

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

Discussion

The piping and equipment pressure parts within the reactor coolant pressure boundary through the outer isolation valve(s) are designed, fabricated, erected, and tested to provide a high degree of integrity throughout the plant lifetime. Chapter 3.2 classifies systems and components within the reactor coolant pressure boundary as Quality Group A. The design requirements and codes and standards applied to this Quality Group ensure a quality product in keeping with the safety functions to be performed.

In order to minimize the possibility of brittle fracture within the reactor coolant pressure boundary, the fracture toughness properties and the operating temperature of ferritic materials are controlled to ensure adequate toughness. Subsection 5.2.3, "Reactor Coolant Pressure Boundary Materials," describes the methods utilized to control toughness properties. Materials are to be impact tested in accordance with ASME Boiler and Pressure Vessel Code, Section III, where applicable. Where reactor coolant pressure boundary piping penetrates the containment, the fracture toughness temperature requirements of the reactor coolant pressure boundary materials apply.

Piping and equipment pressure parts of the reactor coolant pressure boundary are assembled and erected by welding unless applicable codes permit flanged or screwed joints. Welding procedures are employed which produce welds of complete fusion and free of unacceptable defects. All welding procedures, welders, and welding machine operators used in producing pressure-containing welds are qualified in accordance with the requirements of Section IX of the ASME Boiler and Pressure Vessel Code for the materials to be welded. Qualification records,

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

Section 5.2 contains the detailed material and examination requirements for the piping and equipment of the reactor coolant pressure boundary prior to and after its assembly and erection. Leakage testing and surveillance are accomplished as described in the evaluation against Criterion 30 of the General Design Criteria.

The design, fabrication, erection, and testing of the reactor coolant pressure boundary assure an extremely low probability of failure of abnormal leakage, thus satisfying the requirements of Criterion 14.

For further discussion, see the following sections:

a.	Principal Design Criteria	1.2
b.	Design of Structures, Components, Equipment, and Systems	3.0
c.	Overpressurization Protection	5.2
d.	Reactor Vessel and Appurtenances	5.3
e.	Reactor Recirculation System	5.4
f.	Accident Analysis	15

3.1.2.2.6 Criterion 15 - Reactor Coolant System Design

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

Discussion

The reactor coolant system consists of the reactor vessel and appurtenances, the reactor recirculation system, the nuclear system pressure relief system, the main steam lines, the reactor core isolation cooling system, and the residual heat removal system. These systems are designed, fabricated, erected, and tested to stringent quality requirements and appropriate codes and standards which assure high integrity of the reactor coolant pressure boundary throughout the plant lifetime. The reactor

coolant system is designed and fabricated to meet the requirements of the ASME Boiler and Pressure Vessel Code, Section III, as indicated in Chapter 3.

The auxiliary, control, and protection systems associated with the reactor coolant system act to provide 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. As described in the evaluation of Criterion 13, instrumentation is provided to monitor essential variables to ensure that they are within prescribed operating limits. If the monitored variables exceed their predetermined settings, the auxiliary, control, and protection systems automatically respond to maintain the variables and systems within allowable design limits.

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

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

For further discussion, see the following sections:

a.	Principal Design Criteria	1.2
b.	Design of Structures, Components, Equipment, and Systems	3.0
c.	Overpressurization Protection	5.2.2
d.	Reactor Coolant Pressure Boundary Leakage Detection System	5.2.5
e.	Reactor Vessel	5.3
f.	Reactor Recirculation System	5.4
g.	Accident Analysis	15

3.1.2.2.7 Criterion 16 - Containment Design

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

Discussion

The containment described in subsections 3.8.1 and 6.2.1 is the major barrier to uncontrolled release of radioactivity after a major accident. As outlined in the Technical Specifications, this structure is periodically tested for leakage. Test provisions and the descriptions are described in subsection 6.2.6.

The enclosure building and auxiliary building are combined to form a secondary containment to provide a controlled environment in which radiation leakage from the primary containment can be collected, filtered, and released to the atmosphere in a controlled manner. The pressure within the secondary containment is maintained at -1/4 inch of water when the standby gas treatment system is in operation. This negative pressure prevents the unfiltered escape of the primary containment leakage to the environment. For more details on the secondary containment design, refer to subsection 6.2.3.

3.1.2.2.8 Criterion 17 - Electrical Power Systems

An onsite electric power system and an offsite electrical 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:

a. Specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded as a result of anticipated operational occurrences.

b. 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 limit 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 the 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.

Discussion

A minimum of two physically independent transmission lines are required from network distribution to the plant switchyard. There will be three physically independent transmission lines provided. Therefore, the minimum requirement is exceeded by one extra line.

A minimum of two separate offsite feeds are required for the engineered safety feature buses. There are three engineered safety feature transformers, each of which can feed the engineered safety feature buses. The minimum requirement is exceeded by one extra feed. Engineered safety feature buses will normally be fed from a transformer connected to the off site transmission system. Loss of this normally connected offsite power source will initiate the starting of the diesel generators.

The three onsite diesel generators, each of which feeds a separate engineered safety features bus, are capable of starting and accepting load within 10 seconds of a start command. Two out of three diesel generators are required to assure that core cooling, containment integrity, and other vital safety functions are maintained. Therefore, a single failure of any one of the diesel generator systems will not prevent the safe shutdown of the reactor.

Each engineered safety features bus is controlled from a separate dc battery system, completely isolated from each other. A single failure in any one of the battery systems will not impair control of the ESF required to maintain the reactor in a safe shutdown.

Independence is provided by physical separation of components and cables to minimize vulnerability of redundant safety systems to any single credible accident.

The above systems are designed in accordance with IEEE 308-1971. Further details of operation are explained in subsections 8.2 and 8.3.1.

3.1.2.2.9 Criterion 18 - Inspection and Testing of Electric Power Systems

Electric power systems important to safety shall be designed to permit appropriate periodic inspection and testing of important areas and features, such as wiring, insulation, connections, and

switchboards, to assess the continuity of the systems and the condition of their components. The systems shall be designed with a capability to test periodically:

a. The operability and functional performance of the components of the systems such as onsite power sources, relays, switches, and buses

b. The operability of the systems as a whole and, under conditions as close to design as practical, the full operational sequence that brings the systems into operation, including operation of applicable portions of the protection system, and the transfer of power among the nuclear power unit, the offsite power system, and the onsite power system

Discussion

Class IE electric power systems have been designed as described below in order that the following aspects of the system can be periodically tested:

a. The operability and functional performance of the components of Class IE electric power systems (diesel generators, emergency buses, dc system)

b. The operability of these electric power systems as a whole, and under conditions as close to design as practical including the full operational sequence that actuates these systems

The 500-kV switchyard breakers will be inspected, maintained, and tested on a routine basis without affecting the rest of the system.

The presence of three independent network sources allows periodic transmission line maintenance and protective relay testing without reducing below minimum requirements of GDC 17.

Any one of the engineered safety features transformers and its circuits to the engineered safety feature buses can be taken out of service and tested periodically and still meet the minimum requirement of two power sources. The protective relaying of these circuits will also be tested during routine tests, along with the lines and transformers, during outages.

The 4160-volt and 480-volt circuit breakers and the associated equipment will be tested one at a time while redundant equipment is operational.

The dc system has detectors to indicate when there is a ground existing on any part of the system. Each battery system is monitored for continuity with a detecting device. This device initiates an alarm if any radical changes occur in an individual system. Except for normal maintenance during operation, battery supply cannot be interrupted or load tested unless an outage is taken in a battery system.

Important areas and features such as wiring, insulation, connections, and switchboards on equipment important for safety are designed to permit periodic inspection during equipment shutdown.

The engineered safety features power supply buses and associated diesel generator are arranged for periodic testing of each load group independently. The testing procedure will simulate a bus voltage loss to start the diesel, bring it to operating condition, and synchronize it to the bus. Full load testing of the diesel generator can be performed by manually synchronizing to the normal power supply. These tests can be performed periodically together with other equipment not included in the balance of plant which will prove the engineered safety systems' operability.

The design of the emergency power systems provides testability in accordance with the requirements of Criterion 18.

For further discussion, see the following sections:

a.	Onsite AC Power Systems	8.3.1
b.	Onsite DC Power Systems	8.3.2
c.	Initial Test Program	14

3.1.2.2.10 Criterion 19 - Control Room

A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents. Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel

receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident.

Equipment at appropriate locations outside the control room shall be provided: (1) with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown; and (2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures.

Discussion

The control room and its post-accident ventilation systems have been designed to satisfy seismic Category I requirements, as discussed in subsection 3.2.1. Under accident conditions, sufficient shielding and ventilation are provided to permit occupancy of the control room for eight hours per day for 30 days, plus once-daily ingress and egress, without receiving more than 5 rem whole body or 30 rem thyroid. The shielding is described in subsection 12.3.2; ventilation is discussed in subsection 9.4.1. The accident analyses are discussed in Chapter 15. Habitability is discussed in Section 6.4.

As discussed in subsection 7.4.1.4, the reactor plant can be brought to cold shutdown conditions from outside the control room.

3.1.2.3 Group III - Protection and Reactivity Control System

3.1.2.3.1 Criterion 20 - Protection System Functions

The protection system shall be designed:

a. 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

b. To sense accident conditions and to initiate the operation of systems and components important to safety

Discussion

The reactor protection system is designed to provide timely protection against the onset and consequences of conditions that threaten the integrity of the fuel barrier and the reactor coolant pressure boundary barrier. Fuel damage is prevented by initiation of an automatic reactor shutdown if monitored nuclear system variables exceed pre-established limits of anticipated operation occurrences. Scram trip settings are selected and verified to be far enough above or below operating levels to provide proper protection but not be subject to spurious scrams. The reactor protection system includes the high inertia motor-generator power system, sensors, bypass circuitry, and switches that signal the control rod system to scram and shut down the reactor. The scrams initiated by neutron monitoring system variables, nuclear system high pressure, turbine stop valve closure, turbine control valve fast closure, main steam line isolation valve closure, and reactor vessel low water level will prevent fuel damage following abnormal operational transients. Specifically, these process parameters initiate a scram in time to prevent the core from exceeding thermal-hydraulic safety limits during abnormal operational transients. Additional scram trips are initiated by drywell high pressure, reactor vessel high and low water level, and scram discharge volume high water level. Response by the reactor protection system is prompt and the total scram time is short. Control rod scram motion starts in about 200 milliseconds after the high flux set point is exceeded.

In addition to the reactor protection system which provides for automatic shutdown of the reactor to prevent fuel damage, protection systems are provided to sense accident conditions and initiate automatically the operation of other systems and components important to safety. Systems such as the emergency core cooling system are initiated automatically to limit the extent of fuel damage following a loss-of-coolant accident.

Other systems automatically isolate the reactor vessel or the containment to prevent the release of significant amounts of radioactive materials from the fuel and the reactor coolant pressure boundary. The controls and instrumentation for the emergency core cooling systems and the isolation systems are initiated automatically when monitored variables exceed preselected operational limits.

The design of the protection systems satisfy the functional requirements as specified in Criterion 20.

For further discussion, see the following sections:

a.	Principal Design Criteria	1.2
b.	Reactivity Control Mechanical Design	4.6
c.	Control Rod Drive Housing Supports	4.6
d.	Overpressurization Protection	5.2
e.	Main Steam Line Isolation Valves	5.4
f.	Emergency Core Cooling System	6.3
g.	Reactor Protection System	7.2
h.	Engineered Safety Feature Systems	7.3
i.	Neutron Monitoring System	7.6
j.	Process Radiation Monitoring System	7.6
k.	Leak Detection System	7.6

3.1.2.3.2 Criterion 21 - Protection System Reliability and Testability

The protection system shall be designed for high functional reliability and inservice testability commensurate with the safety functions to be performed. Redundancy and independence designed into the protection system shall be sufficient to assure that:

a. No single failure results in loss of the protection function, and

b. 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.

Discussion

Reactor protection system design provides assurance that, through redundancy, each channel has sufficient reliability to fulfill the single-failure criterion. No single component failure, intentional bypass maintenance operation, calibration operation,

or test to verify operational availability will impair the ability of the system to perform its intended safety function. Additionally, the system design assures that when a scram trip point is exceeded there is a high scram probability. However, should a scram not occur, other monitored components will scram the reactor if their trip points are exceeded. There is sufficient electrical and physical separation between channels and between logics monitoring the same variable to prevent environmental factors, electrical transients, and physical events from impairing the ability of the system to respond correctly.

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

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

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

The main steam line isolation valves may be tested during full reactor operation. Individually, they can be closed to 90 percent of full open position without affecting the reactor operation. If

reactor power is reduced sufficiently, the isolation valves may be fully closed. During refueling operation, valve leakage rates can be determined.

Residual heat removal system testing can be performed during normal operation. Main system pumps can be evaluated by taking suction from the suppression pool and discharging through test lines back to the suppression pool. System design and operating procedures also permit testing the discharge valves to the feedwater lines. The low pressure coolant injection mode can be tested after reactor shutdown.

Each active component of the emergency core cooling systems provided to operate in a design basis accident is designed to be operable for test purposes during normal operation of the nuclear system.

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

For further discussion, see the following sections:

a.	Principal Design Criteria	1.2
b.	Reactivity Control System	4.6
c.	Main Steam Line Isolation Valves	5.4.5
d.	Residual Heat Removal System	5.4.7
e.	Containment Systems	6.2
f.	Emergency Core Cooling Systems	6.3
g.	Reactor Protection System	7.2
h.	Engineered Safety Feature Systems	7.3
i.	Accident Analysis	15

3.1.2.3.3 Criterion 22 - Protection System Independence

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

Discussion

The components of protection systems are designed so that the mechanical and thermal environment resulting from any emergency situation in which the components are required to function will not interfere with the operation of that function. Wiring for the reactor protection system outside of the control room enclosures is run in rigid metallic wireways. Reactor protection system wiring may be run in the same wireway system with other ESF wire of the same division. The wires from duplicate sensors on a common process tap are run in separate wireways. The system sensors are electrically and physically separated. Only one trip actuator logic circuit from each trip system may be run in the same wireway.

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

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

For further discussion, see the following sections:

a.	Principal Design Criteria	1.2
b.	Main Steam Line Isolation Valves	5.4.5
c.	Residual Heat Removal System	5.4.7
d.	Emergency Core Cooling Systems	6.3
e.	Reactor Protection System	7.2
f.	Engineered Safety Feature Systems	7.3
g.	Accident Analysis	15

3.1.2.3.4 Criterion 23 - Protection System Failure Modes

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

Discussion

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

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

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

For further discussion, see the following sections:

a.	Principal Design Criteria	1.2
b.	Emergency Core Cooling Systems	6.3
c.	Reactor Protection System	7.2
d.	Engineered Safety Feature Systems	7.3

3.1.2.3.5 Criterion 24 - Separation of Protection and Control Systems

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

Discussion

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

Containment, drywell, and reactor vessel isolation control systems are designed so that any one failure, maintenance operation, calibration operation, or test to verify operational availability will not impair the functional ability of the isolation control system to respond to essential variables.

Process radiation monitoring is provided in the area of main steam lines and/or ventilation lines that may serve as discharge routes for radioactive materials. Four instrumentation channels are used to prevent an inadvertent action as a result of instrumentation malfunctions.

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

For further discussion, see the following sections:

a.	Principal Design Criteria	1.2
b.	Reactivity Control Mechanical Design	4.6
c.	Emergency Core Cooling System	6.3

d.	Reactor Protection System	7.2
e.	Engineered Safety Feature Systems	7.3
f.	Neutron Monitoring System	7.6
g.	Process Radiation Monitoring	7.6

3.1.2.3.6 Criterion 25 - Protection System Requirements for Reactivity Control Malfunctions

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

Discussion

The reactor protection system provides protection against the onset and consequences of conditions that threaten the integrity of the fuel barrier and the reactor coolant pressure boundary. Any monitored variable which exceeds the scram set point will initiate an automatic scram and not impair the remaining variables from being monitored, and if one channel fails, the remaining portions of the reactor trip system shall function.

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

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

For further discussion, see the following sections:

a.	Principal Design Criteria	1.2

b.	Reactivity Control System	4.6
c.	Nuclear Design	4.3
d.	Thermal and Hydraulic Design	4.4
e.	Reactor Protection System	7.2
f.	Rod Control and Information System	7.7
g.	Accident Analysis	15

3.1.2.3.7 Criterion 26 - Reactivity Control System Redundancy and Capability

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

Discussion

Two independent reactivity control systems utilizing different design principles are provided. The normal method of reactivity control employs control rod assemblies which contain boron carbide (B4C) powder or boron carbide (B4C) powder and hafnium. Positive insertion of these control rods is provided by means of the control rod drive hydraulic system. The control rods are capable of reliably controlling reactivity changes during normal operation (e.g., power changes, power shaping, xenon burnout, normal startup and shutdown) via operator-controlled insertions and withdrawals. The control rods are also capable of maintaining the core within acceptable fuel design limits during anticipated operational occurrences via the automatic scram function. The unlikely occurrence of a limited number of stuck rods during a scram will not adversely affect the capability to maintain the core within fuel design limits.

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

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

The control rod system is capable of holding the reactor core subcritical under cold conditions, even when the control rod of highest worth is assumed to be stuck in the fully withdrawn position. This shutdown capability of the control rod system is made possible by designing the fuel with burnable poison (Gd_2O_3) to control the high reactivity of fresh fuel. In addition, the standby liquid control system is available to add soluble boron to the core and render it subcritical, as discussed in subsection 3.1.2.3.8.

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

For further discussion, see the following sections:

a.	Principal Design Criteria	1.2
b.	Reactivity Control System	4.6
c.	Engineered Safety Feature Systems	7.3
d.	Standby Liquid Control System - Instrumentation and Control	7.4
e.	Rod Control and Information System	7.7

3.1.2.3.8 Criterion 27 - Combined Reactivity Control Systems Capability

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

Discussion

There is no credible event applicable to the BWR which requires combined capability of the control rod system and poison additions by the emergency core cooling network. The BWR design is capable of maintaining the reactor core subcritical, including allowance for a stuck rod, without addition of any poison to the reactor coolant. The primary reactivity control system for the BWR during postulated accident conditions is the control rod system. Abnormalities are sensed, and, if protection system limits are reached, corrective action is initiated through automatic insertion of control rods. High integrity of the protection system is achieved through the combination of logic arrangement, actuator redundancy, power supply redundancy, and physical separation. High reliability of reactor scram is further achieved by separation of scram and manual control circuitry, individual control units for each control rod, and fail-safe design features built into the rod drive system. Response by the reactor protection system is prompt and the total scram time is short.

In the very unlikely event that more than one control rod fails to insert, and the core cannot be maintained in a subcritical condition by control rods alone as the reactor is cooled down subsequent to initial shutdown, the standby liquid control system (SLCS) will be actuated to insert soluble boron into the reactor core. The SLCS has sufficient capacity to ensure that the reactor can always be maintained subcritical; and hence, only decay heat will be generated by the core which can be removed by the Residual Heat Removal system, thereby ensuring that the core will always be coolable.

The design of the reactivity control systems assures reliable control of reactivity under postulated accident conditions with appropriate margin for stuck rods. The capability to cool the core is maintained under all postulated accident conditions; thus, Criterion 27 is satisfied.

For further discussion, see the following sections:

a.	Principal Design Criteria	1.2
b.	Reactivity Control System	4.6
c.	Nuclear Design	4.3
d.	Thermal and Hydraulic Design	4.4
e.	Reactor Protection System	7.2
f.	Rod Control and Information System	7.7
g.	Accident Analysis	15

3.1.2.3.9 Criterion 28 - Reactivity Limits

The reactivity control systems shall be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither:

a. Result in damage to the reactor coolant pressure boundary greater than limited local yielding nor

b. 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.

Discussion

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

The control rod mechanical design incorporates a hydraulic velocity limiter in the control rod which prevents rapid rod ejection. This engineered safeguard protects against a high reactivity insertion rate by limiting the control rod velocity to less than five ft/sec. Normal rod movement is limited to six-inch increments and the rod withdrawal rate is limited through the hydraulic valve to three in./sec.

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

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

For further discussion, see the following sections:

a.	Principal Design Criteria	1.2
b.	Control Rod Drive Systems	3.9.4
c.	Reactor Core Support Structures and Internals Mechanical Design	4.2
d.	Reactor Control System	4.6
e.	Nuclear Design	4.3
f.	Control Rod Drive Housing Supports	4.6
g.	Overpressurization Protection	5.2
h.	Reactor Vessel and Appurtenances	5.3
i.	Main Steam Line Flow Restrictor	5.4.4
j.	Main Steam Line Isolation Valves	5.4.5
k.	Accident Analysis	15

3.1.2.3.10 Criterion 29 - Protection Against Anticipated Operational Occurrences

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

Discussion

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

An extremely high reliability of timely response to anticipated operational occurrences is maintained by a thorough program of inservice testing and surveillance. Active components can be tested or removed from service for maintenance during reactor operation without compromising the protection or reactivity control functions even in the event of a subsequent single failure. Components important to safety such as control rod drives, main steam isolation valves, residual heat removal pumps, etc., are tested during normal reactor operation. Functional testing and calibration schedules are developed using available

failure rate data, reliability analyses, and operating experience. These schedules represent an optimization of protection and reactivity control system reliability by considering, on one hand, the failure probabilities of individual components and, on the other hand, the reliability effects during individual component testing on the portion of the system not undergoing test. The capability for inservice testing ensures the high functional reliability of protection and reactivity control systems should a reactor variable exceed the corrective action set point.

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

For further discussion, see the following sections:

a.	Principal Design Criteria	1.2
b.	Main Steam Line Isolation Valves System	5.4.5
c.	Residual Heat Removal System	5.4.7
d.	Containment Systems	6.2
e.	Emergency Core Cooling Systems	6.3
f.	Reactor Protection System	7.2.3.1
g.	Engineered Safety Feature Systems	7.3
h.	Accident Analysis	15

3.1.2.4 Group IV - Fluid Systems

3.1.2.4.1 Criterion 30 - Quality of Reactor Coolant Pressure Boundary

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

Discussion

By utilizing conservative design practices and detailed quality control procedures, the pressure retaining components of the reactor coolant pressure boundary are designed and fabricated to retain their integrity during normal and postulated accident

conditions. Accordingly, components which comprise the reactor coolant pressure boundary are designed, fabricated, erected, and tested in accordance with recognized industry codes and standards listed in Chapter 5. Further, product and process quality planning was provided in Chapter 17 of the Grand Gulf PSAR to assure conformance with the applicable code, and standards, and to retain appropriate documented evidence verifying compliance. Because the subject matter of this criterion deals with aspects of the reactor coolant pressure boundary, further discussion on this subject is treated in the response to Criterion 14, Reactor Coolant Pressure Boundary.

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

The reactor coolant pressure boundary and the leak detection system are designed to meet the requirements of Criterion 30.

For further discussion, see the following sections:

a.	Principal Design Criteria	1.2
b.	Overpressurization Protection	5.2
c.	Reactor Coolant Pressure Boundary Leakage Detection System	5.2

d.	Reactor Vessel and Appurtenances	5.3
e.	Reactor Recirculation System	5.4
f.	Reactor Vessel Instrumentation	7.7
g.	Leak Detection System	7.6
h.	Technical Specification	TRM Section 3.8
i.	Nuclear Design	4.3

3.1.2.4.2 Criterion 31 - Fracture Prevention of Reactor Coolant Pressure Boundary

The reactor coolant pressure boundary shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions:

a. The boundary behaves in a nonbrittle manner

b. 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:

a. Material properties

b. The effects of irradiation on material properties

c. Residual, steady-state and transient stresses

d. Size of flaws

Discussion

Brittle fracture control of pressure-retaining ferritic materials is provided to ensure protection against non-ductile fracture. To minimize the possibility of brittle fracture failure of the reactor pressure vessel, the reactor pressure vessel is designed to meet the requirements of ASME Code, Section III, Appendix G which considers material properties, steady-state and transient stresses, and the size of flaws.

The nil-ductility transition (NDT) temperature is defined as the temperature below which ferritic steel breaks in a brittle rather than ductile manner. The NDT temperature increases as a function of neutron exposure at integrated neutron exposures greater than

about 1 x 10^{17} nvt with neutron of energies in excess of 1 MeV.

The reactor assembly design provides an annular space from the outermost fuel assemblies to the inner surface of the reactor vessel that serves to attenuate the fast neutron flux incident up on the reactor vessel wall. This annular volume contains the core shroud, jet pump assemblies, and reactor coolant. Assuming plant operation at rated power and availability of 100 percent for the plant lifetime, the neutron fluence at the inner surface of the vessel causes a slight shift in the transition temperature. Expected shifts in transition temperature during design life as a result of environmental conditions, such as neutron flux, are considered in the design. Operational limitations assume that NDT temperature shifts are accounted for in the reactor operation.

The reactor coolant pressure boundary is designed, maintained, and tested such that adequate assurance is provided that the boundary will behave in a non-brittle manner throughout the life of the plant. Therefore, the reactor coolant pressure boundary is in conformance with Criterion 31.

For further discussion, see the following sections:

a.	Design of Structures, Components, Equipment, and Systems	3.0
b.	Integrity of Reactor Coolant Pressure Boundary	5.2

3.1.2.4.3 Criterion 32 - Inspection of Reactor Coolant Pressure Boundary

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

Discussion

The Grand Gulf design conforms with the intent of Criterion 32.

Consideration has been given to the inspectability of the reactor coolant system, in the design of the components, in the equipment layout, and in the support structures to permit access for inspection. Access to the various components of the reactor coolant system is explained in subsection 5.2.4.

Surveillance specimens of the reactor vessel weld, heat-affected zone, and base materials will be encapsulated and installed in the reactor vessel to enable periodic monitoring of material properties with exposure in accordance with Appendix H to 10 CFR 50.

For further discussion, see the following sections:

a.	In-Service Inspection and Testing	5.2.4
b.	Mechanical Systems and Components	3.9
c.	Details on Material Surveillance	5.3.1.6

3.1.2.4.4 Criterion 33 - Reactor Coolant Makeup

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

Discussion

Means are provided for detecting reactor coolant leakage. The leak detection system consists of sensors and instruments to detect, annunciate, and in some cases isolate the reactor coolant pressure boundary from potential hazardous leaks before predetermined limits are exceeded. Small leaks are detected by temperature changes, sump level changes, and by measuring fission product concentration in the containment atmosphere. In addition to these means of detection, large leaks are detected by changes in flow rates in process lines, and changes in reactor water

level. The allowable leakage rates have been based on predicted and experimentally determined behavior of cracks in pipes, the ability to make up coolant system leakage, the normally expected background leakage due to equipment design, and the detection capability of the various sensors and instruments. The total leakage rate limit is established so that, in the absence of normal ac power concomitant with a loss of feedwater supply, makeup capabilities are provided by the RCIC system.

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

For further discussion, see the following sections:

a.	Detection of Leakage Through Reactor Coolant Pressure Boundary	5.2.5
b.	Reactor Core Isolation Cooling System	5.4.6
c.	Emergency Core Cooling System	6.3

3.1.2.4.5 Criterion 34 - Residual Heat Removal

A system to remove residual heat shall be provided. The 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.

Discussion

Residual heat removal system provides the means to remove decay heat and residual heat from the nuclear system so that refueling and nuclear system servicing can be performed.

The major equipment of the RHR system consists of heat exchangers, main system pumps, and service water pumps. The equipment is connected by associated valves and piping, and the controls and instrumentation are provided for proper system operation.

Two independent loops are located in separate protected areas.

Both normal ac power and the auxiliary onsite power system provide adequate power to operate all the auxiliary loads necessary for plant operation. The power sources for the plant auxiliary power system are sufficient in number, and of such electrical and physical independence, that no single probable event could interrupt all auxiliary power at one time.

The plant auxiliary buses supplying power to engineered safety features and reactor protection systems and those auxiliaries required for safe shutdown are connected by appropriate switching to standby diesel-driven generators located in the plant. Each power source, up to the point of its connection to the auxiliary power buses, is capable of complete and rapid isolation from any other source.

Loads important to plant operation and safety are split and diversified between switchgear sections, and means are provided for detection and isolation of system faults.

The plant layout is designed to effect physical separation of essential bus sections, standby generators, switchgear, interconnections, feeders, power centers, motor control centers, and other system components.

Full capacity standby diesel generators are provided to supply a source of electrical power which is self-contained within the plant and is not dependent on external sources of supply. The standby generators produce ac power at a voltage and frequency compatible with the normal bus requirements for essential equipment within the plant. Each of the diesel generators has sufficient capacity to start and carry the essential loads it is expected to drive.

The residual heat removal system is adequate to remove residual heat from the reactor core to assure fuel and reactor coolant pressure boundary design limits are not exceeded. Redundant reactor coolant circulation paths are available to and from the vessel and RHR system. Redundant onsite electric power systems

are provided. The design of the residual heat removal system, including its power supply, meets the requirements of Criterion 34.

For further discussion see the following sections:

a.	Residual Heat Removal System	5.4.7
b.	Emergency Core Cooling Systems	6.3
c.	Auxiliary Power System	8.3
d.	Water Systems	9.2
e.	Accident Analysis	15

3.1.2.4.6 Criterion 35 - Emergency Core Cooling

A system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate such that:

a. Fuel and clad damage that could interfere with continued effective core cooling is prevented

b. 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.

Discussion

The emergency core cooling systems (ECCS) consist of the following:

a. High Pressure Core Spray System (HPCS)

b. Automatic Depressurization System (ADS)

c. Low Pressure Core Spray (LPCS) System

d. Low Pressure Coolant Injection (LPCI) (an operating mode of the RHRS)

The emergency core cooling systems are designed to limit fuel cladding temperature over the complete spectrum of possible break sizes in the reactor coolant pressure boundary including a complete and sudden circumferential rupture of the largest pipe connected to the reactor vessel.

The HPCS system consists of a single motor driven pump, system piping, valves, controls and instrumentation. The HPCS system is provided to assure that the reactor core is adequately cooled to prevent excessive fuel clad temperatures for breaks in the nuclear system which do not result in rapid depressurization of the reactor vessel. The HPCS continues to operate when reactor vessel pressure is below the pressure at which LPCI operation or LPCS operation maintains core cooling. A source of water is available from either the condensate storage tank or the suppression pool.

The automatic depressurization system functions to reduce the reactor pressure so that flow from LPCI and the LPCS enters the reactor vessel in time to cool the core and prevent excessive fuel clad temperature. The automatic depressurization system uses several of the nuclear system pressure relief valves to relieve the high pressure steam to the suppression pool.

The low pressure core spray system consists of: a centrifugal pump that can be powered by normal auxiliary power or the standby ac power system; a spray sparger in the reactor vessel, above the core (separate from the HPCS sparger); piping and valves to convey water from the suppression pool to the sparger; and associated controls and instrumentation. In case of low water level in the reactor vessel or high pressure in the drywell, the LPCS system automatically sprays water onto the top of the fuel assemblies in time and at a sufficient flow rate to cool the core and prevent excessive fuel temperature. The LPCI system starts from the same signals which initiate the LPCS system and operates independently to achieve the same objective by flooding the reactor vessel.

In case of low water level in the reactor or high pressure in the drywell, the LPCI mode of operation of the RHR system pumps water into the reactor vessel in time to flood the core and prevent excessive fuel temperature. Protection provided by LPCI extends to a small break where the Automatic Depressurization System has operated to lower the reactor vessel pressure.

Results of the performance of the emergency core cooling systems for the entire spectrum of liquid line breaks are discussed in Section 6.3. Peak cladding temperatures are well below the 2200 F design basis.

Also provided in subsection 6.3.3 is an analysis to show that the emergency core cooling systems conform to 10 CFR 50, Appendix K. This analysis shows complete compliance with the Final Acceptance Criteria with the following results:

a. Peak clad temperatures are well below the 2200 F NRC acceptability limit.

b. The amount of fuel cladding reacting with steam is nearly an order of magnitude below the 1 percent acceptability limit.

c. The clad temperature transient is terminated while core geometry is still amenable to cooling.

d. The core temperature is reduced and the decay heat can be removed for an extended period of time.

The redundancy and capability of the onsite electrical power systems for the emergency core cooling systems are represented in the evaluation against Criterion 34.

The emergency core cooling systems provided are adequate to prevent fuel and clad damage which could interfere with effective core cooling and to limit clad metal-water reaction to a negligible amount. The design of the emergency core cooling systems, including their power supply, meets the requirements of Criterion 35.

For further discussion, see the following sections:

a.	Residual Heat Removal System	5.4.7
b.	Emergency Core Cooling Systems	6.3
c.	Auxiliary Power System	8.3
d.	Water Systems	9.2
e.	Accident Analysis	15

3.1.2.4.7 Criterion 36 - Inspection of Emergency Core Cooling System

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

Discussion

The emergency core cooling systems discussed under Criterion 35 include inservice inspection considerations. The RPV core spray sparger will be inspected in accordance with the reactor vessel internals management program. Removable plugs in the reactor shield and/or panels in the insulation provide access for examination of nozzles. Removable insulation is provided on the emergency core cooling systems piping out to and including the first isolation valve outside the drywell. Inspection of the emergency core cooling systems is in accordance with the intent of Section XI of the ASME Code. Subsection 5.2.4 defines the Inservice Inspection Plan, access provisions, and areas of restricted access.

During plant operations, the pumps, valves, piping, instrumentation, wiring, and other components outside the drywell can be visually inspected at any time. Components inside the drywell can be inspected when the drywell is open for access. When the reactor vessel is open, for refueling or other purposes, the spargers and other internals can be inspected. Portions of the ECCS which are part of the reactor coolant pressure boundary are designed to facilitate inservice inspection to detect defects which might affect the cooling performance. Particular attention has been given to the reactor nozzles, core spray, and feedwater spargers. The design of the reactor vessel and internals for inservice inspection, and the plant testing and inspection program ensures that the requirements of Criterion 36 will be met.

For further discussion, see subsections 5.2.4 and 6.6.

3.1.2.4.8 Criterion 37 - Testing of Emergency Core Cooling System

The emergency core cooling system shall be designed to permit appropriate periodic pressure and functional testing to assure:

a. The structural and leaktight integrity of its components

b. The operability and performance of the active components of the system

c. 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

Discussion

The emergency core cooling system consists of the high pressure core spray (HPCS) system, automatic depressurization system (ADS), low pressure coolant injection (LPCI) mode of the RHR system, and low pressure core spray (LPCS) system. Each of these systems is provided with sufficient test connections and isolation valves to permit appropriate periodic pressure testing to assure the structural and leaktight integrity of its components.

The HPCS, LPCS, LPCI and the ADS are designed to permit periodic testing to assure the operability and performance of the active components of each system.

The pumps and valves of these systems will be tested periodically to verify operability. Flow rate tests will be conducted on LPCS, LPCI, and HPCS systems.

The emergency core cooling systems are subjected to tests to verify the performance of the full operational sequence that brings each system into operation. The operation of the associated cooling water systems is discussed in the evaluation of Criterion 46. It is concluded that the requirements of Criterion 37 are met.

For further discussion, see Section 6.3 and subsection 3.9.6.

3.1.2.4.9 Criterion 38 - Containment Heat Removal

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

Discussion

The containment heat removal function is accomplished by the residual heat removal system (RHR). Following a loss-of-coolant accident, one or both of the following operating modes of the RHR system would be initiated:

a. Containment Spray - Condenses steam within the containment.

b. Suppression Pool Cooling - Limits the temperature within the containment by removing heat from the suppression pool water via the RHR heat exchangers. Either or both redundant RHR heat exchangers can be manually activated.

The redundancy and capability of the offsite and onsite electrical power systems for the residual heat removal system is presented in the evaluation against Criterion 34.

For further discussion, see the following sections:

a.	Residual Heat Removal System	5.4.7
b.	Containment Systems	6.2
c.	Standby AC Power Supply and Distribution	8.3
d.	Water Systems	9.2
e.	Accident Analysis	15

3.1.2.4.10 Criterion 39 - Inspection of Containment Heat Removal System

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

Discussion

Provisions are made to facilitate periodic inspections of active components and other important equipment of the containment heat removal systems. During plant operations, the pumps, valves, piping, instrumentation, wiring and other components outside the drywell can be visually inspected at any time and will be inspected periodically. The testing frequencies of most components will be correlated with the component inspection.

The suppression pool is designed to permit appropriate periodic inspection.

The containment heat removal system is designed to permit periodic inspection of major components. This design meets the requirements of Criterion 39.

For further discussion, see the following sections:

a.	Residual Heat Removal System	5.4.7
b.	Containment Systems	6.2
c.	Emergency Core Cooling Systems	6.3
d.	Water Systems	9.2

3.1.2.4.11 Criterion 40 - Testing of Containment Heat Removal System

The containment heat removal system shall be designed to permit appropriate periodic pressure and functional testing to assure:

a. The structural and leaktight integrity of its components

b. The operability and performance of the active components of the system

c. 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.

Discussion

The containment heat removal function is accomplished by a cooling mode of the residual heat removal system (RHR).

The RHR system is provided with sufficient test connections and isolation valves to permit periodic pressure and flow rate testing.

The pumps and valves of the RHR will be operated periodically to verify operability. The cooling mode is not automatically initiated, but operation of the components is periodically verified. The operation of associated cooling water systems is discussed in the response to Criterion 46. It is concluded that the requirements of Criterion 40 are met.

For further discussion refer to subsection 6.2.2.

3.1.2.4.12 Criterion 41 - Containment Atmosphere Cleanup

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

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

Discussion

Fission products and all other materials are confined in the containment. Leakage from the containment is to the auxiliary building, from which the standby gas treatment system (SGTS) draws air, as described in subsections 6.2.3, 6.5.1, and 6.5.3. The SGTS discharges this air through HEPA and deep-bed charcoal filters to reduce the levels of radiation before exhausting to the environment. The SGTS meets seismic Category I requirements.

The hydrogen concentration is controlled by (1) completely mixing the drywell and containment volume after a LOCA to reduce the hydrogen concentration and (2) sometime later, when hydrogen concentrations in both regions reach the control limit, initiating the hydrogen control system. This system is described in subsection 6.2.5 and is also classed as seismic Category I.

The systems are operable from both normal and emergency power sources. Compressors, fans, and other active elements of both systems are redundant so that a single failure of any active component will not prevent safe functioning of passive components and will not prevent safe operation of the system.

3.1.2.4.13 Criterion 42 - Inspection of Containment Atmosphere Cleanup System

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.

Discussion

All parts of the systems described for Criterion 41 can be inspected periodically (except small lengths of piping or ducting passing through concrete shielding) for visible indications of damage or potential failure. Satisfactory operation of the systems during periodic testing demonstrates the integrity of the lengths of piping or ducting passing through the concrete shielding.

For further discussion refer to subsection 5.2.4.

3.1.2.4.14 Criterion 43 - Testing of Containment Atmosphere Cleanup Systems

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

Discussion

Inspection and testing (including filter dioctyl phthalate and freon testing for the standby gas treatment system) are discussed in subsections 6.2.3 and 6.5.1, and, for the compressors and components of the combustible gas control system, in subsection 6.2.5.4. Inspection and testing of the containment are discussed in subsections 3.8.1.7, 3.8.3.7, and 6.2.6.

3.1.2.4.15 Criterion 44 - Cooling Water

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

Suitable redundancy in components and features and suitable 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.

Discussion

The standby service water system, described in detail in subsection 9.2.1, transfers heat from structures, systems, and safety-related components to the ultimate heat sink during conditions when the normal heat sink is unavailable. The ultimate heat sink is described in detail in subsection 9.2.5.

Two independent, redundant systems with interties, system leak detection, and isolation capabilities have been provided to assure that, for onsite electrical 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. A single-failure analysis of the standby service water system is provided in Tables 9.2-1 and 9.2-2.

A third SSW Loop is provided for Division III. SSW Loop C shall be designed only to support the HPCS system and the HPCS diesel.

3.1.2.4.16 Criterion 45 - Inspection of Cooling Water System

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

Discussion

The standby service water system has been designed to permit periodic inspection of all important components, including pumps, cooling tower fans, heat exchangers, and isolation valves, to assure the integrity and capability of the system as described in subsection 9.2.1.

3.1.2.4.17 Criterion 46 - Testing of Cooling Water System

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

including operation of applicable portions of the protection system and the transfer between normal and emergency power sources.

Discussion

The standby service water system has been 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.

Refer to subsection 9.2.1 for further discussion.

3.1.2.5 Group V - Reactor Containment

3.1.2.5.1 Criterion 50 - Containment Design Basis

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

Discussion

The principal design parameters and the safety-design bases described in subsection 6.2.1 resulted from the calculation of post-LOCA pressure and temperature conditions described in subsection 6.2.1.3 and based on NRC guides. The design internal

pressure is at least 15 percent greater than the peak calculated pressure. The resulting structural design is described in subsection 3.8.1.

3.1.2.5.2 Criterion 51 - Fracture Prevention of Containment Pressure Boundary

The reactor containment boundary shall be designed with sufficient margin to assure that, under operating, maintenance, testing, and postulated accident conditions, (1) its ferritic materials behave in a nonbrittle manner, and the (2) 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:

a. Material properties

b. Residual, steady-state, and transient stresses

c. Size of flaws

Discussion

Operational testing and postulated accident temperatures are combined with appropriate pressures and other loads in the load combination equations of subsection 3.8.1.3. The materials used in the construction of safety-related structures and components provide (in compliance with the codes and standards listed in subsection 3.8.1.2) adequate safety margins to account for variations in properties, residual stresses, and size of flaws. The implementation of the requirements of the codes and standards was supplemented by the QA/QC program discussed in Chapter 17 of the Grand Gulf PSAR.

3.1.2.5.3 Criterion 52 - Capability for Containment Leakage Rate Testing

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

Discussion

As described in "Technical Specifications," the primary containment can and will be tested periodically to determine the actual leakage rate at appropriate pressures. Provisions in containment design for the performance of the tests are described in subsections 6.2.1.6 and 6.2.6.

3.1.2.5.4 Criterion 53 - Provisions for Containment Testing and Inspection

The reactor containment shall be designed to permit:

a. Appropriate periodic inspection of all important areas, such as penetrations

b. An appropriate surveillance program

c. Periodic testing at containment design pressure of the leaktightness of penetrations which have resilient seals and expansion bellows

Discussion

A capability exists for containment testing and inspection in conformance with General Design Criterion 53.

Containment design includes provisions for testing of penetrations, liner plate areas, and areas of seals and bellows on applicable penetrations. Penetrations are visually inspected and pressure tested for leaktightness at periodic intervals. Inspections are performed as specified in, and at the required intervals per, Appendix J of 10 CFR 50. Types B and C penetration tests are conducted in accordance with 10 CFR 50, Appendix J, with the exceptions discussed in Technical Specifications.

Testing and inspection are discussed in subsections 3.8.1.7 and 6.2.6.

3.1.2.5.5 Criterion 54 - Piping Systems Penetrating Containment

Piping systems penetrating primary reactor containment shall be provided with leak detection, isolation, and containment capabilities having redundancy, reliability, and performance capabilities which reflect the importance to safety of isolating these piping systems. Such piping systems shall be designed with

a capability to test periodically the operability of the isolation valves and associated apparatus and to determine if valve leakage is within acceptable limits.

Discussion

Piping systems penetrating the primary reactor containment are provided with containment isolation valves. Penetrations which must be closed for containment isolation have redundant valving and associated apparatus. Automatic isolation valves with air or motor operators, which do not restrict normal plant operation, are periodically tested to assure operability. The isolation valve arrangements are discussed in subsection 6.2.4.

All piping penetrating the containment has been equipped with test connections and test vents or has other provisions to allow periodic leak rate testing to ensure that leakage is within the acceptable limit as defined by the Technical Specifications and Appendix J of 10 CFR 50, as described in subsection 6.2.6. Containment isolation features are seismic Category I and are protected against potential environmental effects including missiles. All containment penetrations are designed according to the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Class 2, and are inspected periodically as required by the ASME Boiler and Pressure Vessel Code, Section XI.

3.1.2.5.6 Criterion 55 - Reactor Coolant Pressure Boundary Penetrating Containment

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

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

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

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

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

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

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

Discussion

Subsection 6.2.4 discusses containment isolation. In general, the four items listed in this criterion are followed. In certain cases, other acceptable defined bases for isolation that deviate from items (1) through (4) of this criterion are followed. Due to the large number of containment isolation valves, subsection 6.2.4 and the associated tables and figures are referenced to show how these four items or other acceptable design bases are met. The isolation valving is shown schematically in Figures 6.2-76 through 6.2-79, and is summarized, with valve actuation, in Table 6.2-44. In addition, all such piping meets the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Nuclear Power Plant Components, Class 2 (or better), and are periodically inspected in accordance with section XI of the Code. Penetration piping is designated as seismic Category I and is protected against possible environmental effects including missiles.

Instrument lines are designed according to the criteria of Regulatory Guide 1.11.

3.1.2.5.7 Criterion 56 - Primary Containment Isolation

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

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

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

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

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

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

Discussion

Subsection 6.2.4 discusses containment isolation. In general the four items listed in this criterion are followed. In certain cases, other acceptable defined bases for isolation which deviate from items (1) through (4) of this criterion are followed. Due to the large number of containment isolation valves, subsection 6.2.4 and the associated tables and figures are referenced to show how these four items or other acceptable design bases are met. The isolation valving is shown schematically in Figures 6.2-76 through 6.2-79 and is summarized, with valve actuation, in Table 6.2-44. Simple check valves are not utilized as automatic isolation valves outside the containment. Instrument lines satisfy the criteria set by Regulatory Guide 1.11.

3.1.2.5.8 Criterion 57 - Closed System Isolation Valves

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

Discussion

As discussed in subsection 6.2.4, the Grand Gulf Nuclear Station containment isolation valve arrangement does not utilize the provisions of Criterion 57.

3.1.2.6 Group VI - Fuel and Reactivity Control

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

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

Discussion

Waste handling systems have been incorporated in the plant design for processing and/or retention of radioactive wastes from normal plant operations to ensure that the effluent releases to the environment are as low as practicable and within the limits of 10 CFR 20. The plant is also designed with provisions to prevent radioactivity releases during accidents from exceeding the limits of 10 CFR 100, 10 CFR 50.67 and 10 CFR 50 GDC 19.

As discussed in Chapter 11.0, radioactive wastes are processed in accordance with 10 CFR 50, Appendix I requirements.

Potentially radioactive liquid leakage and/or spills are collected by drain systems and directed to the liquid radwaste system.

Gaseous leakage within the drywell will be confined there. Gaseous leakage in the containment, if significant, will cause the process radiation monitors (described in subsection 11.5) to isolate the ventilation system. Significant gaseous leakage in the radwaste or turbine building will cause the area radiation monitoring system to alarm in the control room, as described in subsection 12.3.4, Area Radiation Monitoring System.

Should any condition exist which could prevent safe release of liquid waste, the liquid radwaste system has ample tank capacity to permit deferring the release.

The principal gaseous effluents from the plant during normal operation are the noncondensible gases from the air ejectors. These gases are exhausted through a holdup system and a low temperature offgas treatment system including charcoal adsorbers. The effluent from this system is continuously monitored and controlled and the system will be shut down and isolated in the event of abnormally high radiation levels. The offgas system is capable of safely processing for release considerably more radioactive gas than would be expected during normal plant conditions. A ten-minute holdup line has been provided to allow for decay of short-lived isotopes, and charcoal adsorbers have been provided for holdup of xenon and krypton. Refer to Section 11.3 of this report for a description of the offgas system. No additional gas storage is required.

3.1.2.6.2 Criterion 61 - Fuel Storage and Handling and Radioactivity Control

The fuel storage and handling, radioactive waste, and other systems which may contain radioactivity shall be designed to assure adequate safety under normal and postulated accident conditions. These systems shall be designed:

a. With a capability to permit appropriate periodic inspection and testing of components important to safety

b. With suitable shielding for radiation protection

c. With appropriate containment, confinement, and filtering systems

d. A residual heat removal capability having reliability and testability that reflects the importance to safety of decay heat and other residual heat removal

e. To prevent significant reduction in fuel storage coolant inventory under accident conditions

Discussion

New Fuel Storage

New fuel is placed in dry storage in the new fuel storage vault which is located inside the auxiliary building. The storage vault within the auxiliary building provides adequate shielding for radiation protection. Storage racks preclude accidental criticality (see evaluation against Criterion 62). The new fuel storage racks do not require any special inspection and testing for nuclear safety purposes.

Spent Fuel Handling and Storage

Irradiated fuel is also stored in the auxiliary building. Spent fuel is stored in the auxiliary building spent fuel pool and in the upper containment pool. Although no fuel is stored in the upper containment pool during plant operation, other components such as control rods, control rod guide tubes, fuel sipping containers, blade guides, and defective fuel storage canisters (without fuel) may be permanently stored in one end of the pool. Defective spent fuel canisters (with fuel), and fuel can be temporarily stored in the upper pool during refueling operations. Prior to reactor startup, all spent fuel will be removed from the upper containment pool, and transferred to the spent fuel pool for storage. Fuel pool water is circulated through the fuel pool cooling and cleanup system (FPCC) to maintain fuel pool water temperature, purity, water clarity, and water level. Storage racks preclude accidental criticality (see evaluation against Criterion 62).

Reliable decay heat removal is provided by the closed loop FPCC system. It consists of two circulating pumps, two heat exchangers, two filter-demineralizers, one fuel pool drain tank, and the required piping, valves, and instrumentation. The pool water is circulated through the system, suction is taken from the drain tank, flow passes through the heat exchanger and filters, and it is discharged into the spent fuel pool and/or the upper containment pool. Cooling flow through the racks and spent fuel

assemblies is provided by natural circulation within the pools. The thermal driving head is sufficient to preclude nucleate boiling in the rack cells with the greatest hydraulic resistance. Unit 1 pool water temperatures are maintained below 140 F by limiting the number of spent fuel assemblies stored in the pools, utilizing the decrease in decay heat over time, and using the RHR system to remove decay heat when the FPCC system cannot maintain the pool temperature below the 140 F limit.

The FPCC system can be aligned in a "one pump/two heat exchanger" mode to support the maximum postulated heat load on the system. This mode consists of one FPCC pump supplying both FPCC heat exchangers in parallel, with cooling supplied by one SSW pump. Whether or not the one-pump/two heat exchangers configuration shall be required is contingent upon the number of fuel assemblies in the pool and their decay times (i.e. heat load). This mode of operation shall be capable of providing adequate cooling to the Spent Fuel Pool (SFP) with fuel offloads consisting of normal batch sizes and normal design heat loads, up to and including the offload that fills the SFP to capacity.

High and low pool water level is alarmed in the control room. Fission product concentration in the pool water is minimized by use of the filter-demineralizer. This minimizes the release from the pool to the fuel handling area environment.

Tests and inspections required to maintain FPCC system operability are discussed in 9.1.3.

Radioactive Waste Systems

The radioactive waste systems provide all equipment necessary to collect, process, and prepare for disposal all radioactive liquids, gases, and solid waste produced as a result of reactor operation.

Liquid radwastes are classified, contained, and treated as high or low conductivity, chemical, detergent, sludges or concentrated wastes. Processing includes filtration, ion exchange, analysis, and/or dilution. Liquid wastes are also decanted and sludge is accumulated for disposal as solid radwaste. Wet solid wastes are packaged in containers approved by the disposal site and DOT. Dry solid radwastes are packaged in steel or fiber drums, cartons, or boxes. Gaseous radwastes are monitored, processed, recorded, and

controlled so that radiation doses to persons outside the Restricted Area are below those allowed by applicable regulations.

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

The radwaste systems are used on a routine basis and do not require specific testing to assure operability. Performance of the liquid radwaste system is monitored by conductivity monitors during operation. A high conductivity is indicative of breakdown of the filter-demineralizer beds.

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

For further discussion, see the following sections:

a.	Residual Heat Removal System	5.4.7
b.	Containment	6.2
c.	Radioactive Waste Systems	11.0
d.	New Fuel Storage Racks	9.1.1
e.	Spent Fuel Storage Racks	9.1.2
f.	Fuel Pool Cooling and Cleanup System	9.1.3
g.	Heating, Ventilating and Air Conditioning System	9.4
h.	Radiation Protection Design	12.0

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

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

Discussion

Appropriate plant fuel handling and storage facilities are provided to preclude accidental criticality for new and spent fuel. Criticality in new and spent fuel storage is prevented by the geometrically safe configuration of the storage rack combined with the use of neutron absorber (Boraflex) material in the high density storage racks. There is sufficient spacing or neutron poison material between the assemblies to assure that the array when fully loaded is substantially subcritical. Fuel elements are limited by rack design to only top loading and fuel assembly positions. The new and spent fuel racks are seismic Category I structures.

New fuel is placed in dry storage in the top-loaded new fuel storage vault. This vault contains a drain to prevent the accumulation of water. The new fuel storage vault racks (located inside the auxiliary building) are designed to prevent an accidental critical array, even in the event the vault becomes flooded or subjected to seismic loadings. The center-to-center new fuel assembly spacing limits the effective multiplication factor of the array to not more than 0.90 for new dry fuel. K_{eff} will not exceed 0.95 if the new fuel is flooded.

Spent fuel is stored under water in the spent fuel pool or temporarily in the upper containment pool. New fuel may also be stored in these racks temporarily during refueling outage activities. The racks in which spent or new fuel assemblies are placed are designed and arranged to ensure subcriticality in the storage pool. Spent and new fuel is maintained at a subcritical multiplication factor K_{eff} of less than 0.95 under normal and abnormal conditions. Abnormal conditions may result from an earthquake, accidental dropping of equipment, or damage caused by the horizontal movement of fuel handling equipment without first disengaging the fuel from the hoisting equipment. The spent fuel pool and upper containment pool racks are designed to use Boraflex to maintain subcriticality except that some regions of the racks may credit only assembly spacing limitations due to Boraflex degradation.

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

The use of geometrically safe configurations for new fuel storage racks, the use of neutron absorbing material for the spent fuel pool and upper containment pool racks, and the design of fuel handling systems precludes accidental criticality in accordance with Criterion 62.

For further discussion, see the following sections:

a.	Refueling Interlocks	7.6.1.1
b.	New Fuel Storage Racks	9.1.1
c.	Spent Fuel Storage Racks	9.1.2

3.1.2.6.4 Criterion 63 - Monitoring Fuel and Waste Storage

Appropriate systems shall be provided in fuel storage and radioactive waste systems and associated handling areas:

a. To detect conditions that may result in loss of residual heat removal capability and excessive radiation levels

b. To initiate appropriate safety actions

Discussion

Appropriate systems have been provided to meet the requirements of this criterion. A malfunction of the fuel pool cooling and cleanup system, which could result in loss of residual heat removal capability and excessive radiation levels, is alarmed in the control room. Alarmed conditions include high/low fuel pool cooling water pump discharge pressure and high/low level in the fuel storage pool and fuel pool drain tanks. System temperature is also continuously monitored and alarmed in the control room. The containment and drywell ventilation exhaust radiation monitoring system detects abnormal amounts of radioactivity and initiates appropriate action to control the release of radioactive material to the environs.

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

For further discussion, see the following sections:

a.	Liquid Radwaste System	11.2

b.	Solid Radwaste System	11.4
c.	Gaseous Radwaste System	11.3
d.	Process Radiation Monitoring	7.6, 11.5
e.	Fuel Pool Cooling and Cleanup System	9.1.3

3.1.2.6.5 Criterion 64 - Monitoring Radioactivity Releases

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

Discussion

Means have been provided for monitoring radioactivity releases resulting from normal and anticipated operational occurrences. The following station releases are monitored:

a. Gaseous releases

b. Liquid discharge

In addition, the containment atmosphere is monitored.

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

a.	Detection of Leakage Through Reactor Coolant Pressure Boundary	5.2.5
b.	Radioactive Waste Management	11.0

3.2 CLASSIFICATION OF STRUCTURES, COMPONENTS, AND SYSTEMS

This section provides a guide to the classification method of structures, components, and systems.

It is intended that the seismic classification of structures, systems, and components and the system quality group classification for pressure-containing components of fluid systems comply with the guidelines of Regulatory Guides 1.26 and 1.29 (see Appendix 3A for exceptions).

In this classification method (Table 3.2-1), structures, systems, and components are designated as seismic or nonseismic Category I depending on whether or not an item is safety-related. Class IE electrical equipment is seismic Category I equipment. The seismic classification of systems instrumentation is discussed in Chapter 7.For further information on instrument and electrical equipment classification see Section 3.10.

In addition, in fluid systems, components are grouped into four Quality Groups, which are listed to correspond with the guidelines of Regulatory Guide 1.26.In general, three of these Quality Groups, A through C, are seismic Category I, and Quality Group D is not.

3.2.1 Seismic Classification

General Design Criterion 2 of Appendix A to 10 CFR 50, General Design Criteria for Nuclear Power Plants, and Appendix A to 10 CFR 100, "Seismic and Geologic Siting Criteria for Nuclear Power Plants," require that nuclear power plant structures, systems, and components important to safety be designed to withstand the effects of natural phenomena such as earthquakes without loss of capability to perform necessary safety functions.

3.2.1.1 Definitions

Seismic Category I structures, components, and systems are defined in accordance with NRC Regulatory Guide 1.29 as those necessary to assure:

a. The integrity of the reactor coolant pressure boundary

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

c. The capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the guideline exposures of 10 CFR Part 100

Seismic Category I structures, components, and systems are designed to withstand the appropriate seismic loads as discussed in Section 3.7, Seismic Design, and other applicable loads without loss of function. Seismic Category I structures are sufficiently isolated or protected from other structures to ensure that their integrity is maintained at all times.

Components (and their supporting structures) that are not seismic Category I and whose collapse could result in loss of required function through impact or flooding of seismic Category I structures, equipment, or systems required after a safe shutdown earthquake are analytically checked and designed to confirm their integrity against collapse when subjected to seismic loading resulting from the safe shutdown earthquake.

3.2.1.2 Classifications

Table 3.2-1 provides a listing of structures, components, and systems and identifies those that are seismic Category I.

Where only portions of systems are identified as seismic Category I in Table 3.2-1, the boundaries of the seismic Category I portions of the system are shown on the piping and instrumentation diagrams, identified by the symbol S/I or Q on the diagram, in appropriate sections of the FSAR. Compliance of the above seismic classifications with NRC Regulatory Guide 1.29 is discussed in Appendix 3A, Conformance with NRC Regulatory Guides.

3.2.2 System Quality Group Classification

This subsection provides the system quality group classification for each fluid system pressure-containing component. Components are classified according to safety-related importance as dictated by service and functional requirements and by the consequences of failure. The design, fabrication, inspection, and testing requirements fixed for each classification provide the required degree of conservatism in assuring pressure integrity.

3.2.2.1 Quality Group Classifications

Equipment quality group classifications are indicated in Table 3.2-1. Figure 3.2-1 is a diagram which depicts the relative locations of these components along with their quality group classification. System quality group classifications and interfaces between classifications in systems with components of different classifications are indicated on the system piping and instrumentation diagrams that are found in the pertinent sections of this report. Compliance of quality group classification with NRC Regulatory Guide 1.26 is discussed in Appendix 3A, Conformance with NRC Regulatory Guides.

3.2.2.2 Code Requirements

The code requirements applicable to each quality group classification are identified in Table 3.2-2. Design code requirements correspond to those indicated in Section 50.55a of 10 CFR 50 and NRC Regulatory Guide 1.26, Table 1, except for Quality Group D pumps. The design standard for these pumps is discussed in the footnote of Table 3.2-2.

3.2.3 Quality Assurance Program

To fulfill the requirements of Chapter 17, those items that fall under the Quality Assurance Program are identified in Table 3.2-1 as Q-listed.

3.2.4 System Safety Classifications

Structures, systems, and components are classified as Safety Class 1, Safety Class 2, Safety Class 3, or Other in accordance with the importance to nuclear safety. Equipment is assigned a specific safety classification, recognizing that components within a system may be of differing safety importance. A single system may thus have components in more than one safety class. Supports shall be appropriate for the component supported, as defined by the ASME Code, Section III.

The safety classes are defined in this subsection and examples of their broad application are given. Because of specific design considerations, these general definitions are subject to interpretation and exceptions. Table 3.2-1, Equipment Classification, provides a summary of the safety classes for the principal structures, systems, and components of the plant.

Design requirements for components of safety classes are also delineated in this section. Where possible, reference is made to accepted industry codes and standards which define design requirements commensurate with the safety function(s) to be performed. In cases where industry codes and standards have no specific design requirements, the locations of the appropriate subsections that summarize the requirements to be implemented in the design are indicated.

3.2.4.1 Safety Class 1

3.2.4.1.1 Definition of Safety Class 1

Safety Class 1, SC-1, applies to components of the reactor coolant pressure boundary whose failure could cause a loss of reactor coolant at a rate in excess of the normal makeup system capability.

3.2.4.2 Safety Class 2

3.2.4.2.1 Definition of Safety Class 2

Safety Class 2, SC-2, applies to those structures, systems, and components, other than service water systems, that are not Safety Class 1 but are necessary to accomplish the safety functions of:

a. Inserting negative reactivity to shut down the reactor

b. Preventing rapid insertion of positive reactivity

c. Maintaining core geometry appropriate to all plant process conditions

d. Providing emergency core cooling

e. Providing and maintaining containment

f. Removing residual heat from the reactor and reactor core

g. Storing spent fuel

Safety Class 2 includes the following:

a. Reactor protection system

b. Those components of the control rod system which are necessary to render the reactor subcritical

c. Systems or components which restrict the rate of insertion of positive reactivity

d. The assembly of components of the reactor core which maintain core geometry including the fuel assemblies, core support structure, and core grid plate, as examples

e. Other components within the reactor vessel such as jet pumps, core shroud, and core spray components which are necessary to accomplish the safety function of emergency core cooling

f. Emergency core cooling systems

g. Containment vessel

h. Secondary containment and standby gas treatment system

i. Post-accident containment heat removal systems

j. Deleted

k. Initiating systems required to accomplish safety functions, including emergency core cooling initiating system and containment isolation initiating system

l. At least one of the systems which recirculates reactor coolant to remove decay heat when the reactor is pressurized and the system to remove decay heat when the reactor is not pressurized

m. Spent fuel storage racks and spent fuel pool

n. Electrical and instrument auxiliaries necessary to operation of the above

o. Pipes having a nominal pipe size of 3/4 in. or smaller, that are connected to the reactor coolant pressure boundary

p. Hydrogen igniter assemblies

Structures, systems, and components in Safety Class 2 are listed in Table 3.2-1.

3.2.4.3 Safety Class 3

3.2.4.3.1 Definition of Safety Class 3

Safety Class 3, SC-3, applies to those structures, systems, and components that are not Safety Class 1 or Safety Class 2, but

a. Whose function is to process radioactive wastes and whose failure would result in release to the environment of gas, liquid, or solids resulting in a single-event whole body dose to a person at the site boundary greater than 500 mrem.

b. Which provide or support any safety system function.

Safety Class 3 includes the following:

1. Gaseous waste disposal system components, unless it can be demonstrated that a single component failure would not result in calculated potential exposures in excess of 500 mrem whole body at the site boundary

2. Those portions of the radwaste equipment or structures required to prevent leakage to the environs, at an excessive rate, of liquids from the liquid waste disposal system

3. Service water systems required for the purpose of:

   (a) Removal of decay heat from the reactor

   (b) Emergency core cooling

   (c) Post-accident heat removal from the suppression pool

   (d) Providing cooling water needed for the functioning of emergency systems

4. Fuel supply for the onsite emergency electrical system

5. Emergency equipment area cooling

6. Compressed gas or hydraulic systems required for support, control, or operation of safety systems

7. Electrical and instrumentation auxiliaries necessary for operation of the above

3.2.4.4 Other Structures, Systems, and Components

3.2.4.4.1 Definition of Other Structures, Systems, and Components

A boiling water reactor has a number of structures, systems, and components in the power conversion or other portions of the facility which have no direct safety function but which may be connected to or influenced by the equipment within the safety classes defined above. Such structures, systems, and components are designated as "other."

3.2.4.4.2 Design Requirements for Other Structures, Systems, and Components

The design requirements for equipment classified as "other" is specified by the designer with appropriate consideration of the intended service of the equipment and expected plant and environmental conditions under which it will operate. Where possible, design requirements are based on applicable industry codes and standards. Where these are not available, the designer utilizes accepted industry or engineering practice.

3.2.5 Correlation of Safety Classes with Industry Codes

The design of plant equipment is commensurate with the safety importance of the equipment. Hence, the various safety classes have a gradation of design requirements. The correlation of safety classes with other design requirements is summarized in Table 3.2-3.

Table 3.2-4 presents industry codes with their edition and addenda that apply to Quality Group A, B, and C components performing pressure boundary functions.

3.2.6 Replacement and Modification of ASME Section III, Class 1, 2, and 3 Components

Components and component parts that are Quality Group A, B or C and ASME Section III, Class 1, 2, or 3, respectively, that require replacement or modification shall comply with the rules of ASME Section XI, IWA-4000. In accordance with ASME Section XI, IWA-4000 items used in replacements or modifications are required to

meet the requirements of the original construction code, including code stamping. Under the guide lines of Generic Letter 89-09 exception is provided that allows the use of non-stamped items to replace items that were originally stamped provided the staff positions of the generic letter are complied with.

3.2.6.1 Generic Letter 89-09 Requirements

In order to use the guidance provided by Generic Letter 89-09, it must first be established that an equivalent Section III stamped replacement is not available. Cost cannot be used as a justification for purchasing non-stamped items. Where replacements are no longer available in full compliance with the stamping and documentation requirements of Section III of the Code they are procured under the Operational Quality Assurance Program, meeting the requirements 10CFR Part 50, Appendix B. Suppliers qualified under these provisions are listed in the Qualified Suppliers List. Replacements obtained under the guidance of Generic Letter 89-09 meet all other applicable requirements of Section III (including third party inspection by an Authorized Nuclear Inspector) endorsed by NRC regulations except that the Code N-symbol need not be applied.

Before an item is obtained under the provisions of Generic Letter 89-09, an evaluation is performed to determine compliance with the generic letter and to document the basis for the actions taken. The evaluation is maintained for the lifetime of each replacement item.

Table 3.2-5 lists the items obtained under the guidance of Generic Letter 89-09. The installed location of each item is listed in the records required by ASME Section XI and the Operational Quality Assurance Program.

3.2.6.2 Generic Letter 89-09 Implementation

The guidance given in Generic Letter 89-09 is applicable to components and component parts (both welded and non-welded). Non-welded component parts include only those parts that have specific design or fabrication processes that makes the part unique to the component.

When an item is obtained from a non-N-Certificate holder, an Authorized Nuclear Inspector (ANI) will be assigned to the supplier's facility when the item is manufactured with processes that require witnessing by an ANI. When items are manufactured

with processes that do not require witnessing by an ANI, the site ANI's review of all associated documentation will satisfy the requirements of Section III.

For a replacement component to be deemed as equivalent, as used in Section 3.2.6.1, it must fulfill the same safety function without physical design changes to the facility. Also, to qualify as equivalent, the replacement components operating parameters should imitate the original components to the extent that a new system or safety analysis is not required.

The equivalency of a component part may not be definable because the component manufacturer maintained all specific design and fabrication information relevant to the part. Although a Section III certificate holder may be available that will manufacture a substitute part, the replacement part is obtained from the original manufacturer to ensure form, fit and function with the component to which it will be installed. Component parts obtained under the guidance of Generic Letter 89-09 are obtained from the original manufacturer provided all other requirements of the generic letter are complied with. When the original manufacturer is not available, or cannot supply the part in compliance with the Operational Quality Assurance Program, the component part is obtained from a Section III certification holder.

TABLE 3.2-1: CLASSIFICATION OF SYSTEMS, COMPONENTS AND STRUCTURES

Principal Component(a)		Scope Supply (b)	Safety Class(c)	Location(d)	Quality Group Classification (e)	Principal Construction Code(f)	Seismic Category(g)	Q-List(h)	Comment
I	Reactor System (Sections 4.0 and 5.3) (ff)								
1.	Reactor vessel	GE	1	C	A	III-1	I	Y	
2.	Reactor vessel support skirt	GE	1	C	-	III-1	I	Y	
3.	Reactor vessel appurtenances, pressure retaining portions	GE	1	C	A	III-1	I	Y	
4.	CRD housing supports	GE	2	C	-	See Sect.3.9.4.3.1	I	Y	
5.	Reactor internal structures, engineered safety features	GE	2	C	-	Other	I	Y	
6.	Reactor internal structures, other	GE	Other	C	-	Other	NA	N	
7.	Control rods	GE	2	C	-	Other	I	Y	
8.	Control rod drives	GE	2	C	-	See Sect. 3.9.4.3	I	Y	
9.	Core support structure	GE	2	C	-	III-NG	I	Y	
10.	Power range detector hardware	GE	2	C	B	III-2	I	Y	(i)
11.	Fuel assemblies	GE	2	C	-	Other	I	Y	
II.	Nuclear Boiler System (Sections 5.2, 5.4.4, and 5.4.5) (ff) (mm)								
1	Vessels, level instrumentation condensing chambers	GE	1	C	A	III-1	I	Y	(i)
2	Vessels, air accumulators	P	3	C,A	C	III-3	I	Y	

TABLE 3.2-1: CLASSIFICATION OF SYSTEMS, COMPONENTS AND STRUCTURES (CONTINUED)

Principal Component(a)		Scope Supply (b)	Safety Class(c)	Location(d)	Quality Group Classification (e)	Principal Construction Code(f)	Seismic Category(g)	Q-List(h)	Comment
3	Piping, relief valve discharge	GE,P	3, Other	C	C,	DIII-3, B31.1	I,NA	Y,N	(j)
4	Piping, main steam, within outer isolation valve	GE	1	C,A	A	III-1	I	Y	(i)
5	Pipe supports, main steam	GE	1	C	-	III-1	I	Y	
6	Piping, other within outer isolation valves	P	1	C,A	A	III-1	I	Y	(i)
7	Piping, instrumentation beyond outermost isolation valves	P	2	C,A,T	B	III-2	I	Y	
8	Safety relief valves	GE	1	C	A	III-1	I	Y	
9	Valves, main steam isolation valves	GE	1	C	A	III-1	I	Y	
10	Valves, other, isolation valves and within	P	1	C	A	III-1	I	Y	(i)
11	Valves, instrumentation beyond outermost isolation valves	P	2	C,A,T	B	III-2	I	Y	
12	Mechanical modules, instrumentation, with safety function	GE	2	C	-	Other	I	Y	
13	Electrical modules with safety function	GE	2	C	-	Other	I	Y	

TABLE 3.2-1: CLASSIFICATION OF SYSTEMS, COMPONENTS AND STRUCTURES (CONTINUED)

Principal Component(a)		Scope Supply (b)	Safety Class(c)	Location(d)	Quality Group Classification (e)	Principal Construction Code(f)	Seismic Category(g)	Q-List(h)	Comment
14	Piping, main steam between outer isolation valve and main steam shutoff valve	P	2	A	B	III-2	I	Y	(i)
15	Valves, main steam, from outer isolation valve up to and including main steam shutoff valve	P	2	A	B	III-2	I	Y	(i)
16	Cable, with safety function	P	2	-	-	Other	NA	Y	(k)
17	Reactor feedwater piping and valves, RPV up to and including outer containment isolation check valve	P	1	C,A	A	III-1	I	Y	(i,v)
18	Reactor feedwater piping and valves, from outer containment isolation check valve up to and including feedwater shutoff valve	P	2	A	B	III-2	I	Y	(i)
19	Pipe whip restraints, main steam	GE	2	C	-	Other	I	Y	
III.	Recirculation System (Section 5.4.1) (ff) (mm)								

TABLE 3.2-1: CLASSIFICATION OF SYSTEMS, COMPONENTS AND STRUCTURES (CONTINUED)

Principal Component(a)		Scope Supply (b)	Safety Class(c)	Location(d)	Quality Group Classification (e)	Principal Construction Code(f)	Seismic Category(g)	Q-List(h)	Comment
1.	Piping	GE	1	C	A	III-1	I	Y	(i)
2.	Pipe supports, recirculation line	GE	1	C	-	III-1	I	Y	
3.	Pipe whip restraints, recirculation line	GE	2	C	-	Other	I	Y	
4.	Pumps (Recirculation)	GE	1	C	A	III-1	I	Y	
5.	Valves	GE	1	C	A	III-1	I	Y	(i)
6.	Motors, pump (HPU Skid)	GE	2	C	-	Other	I	Y	
7.	Electrical modules, with safety function	GE	2	C	-	Other	I	Y	
8.	Cable with safety function	P	2	-	-	Other	NA	Y	(k)
IV.	CRD Hydraulic System (Section 4.6.1) (ff) (mm)								
1.	Deleted								
2.	Valves, scram discharge volume lines	GEP	2 2	C C	B B	III-2 III-2	I I	Y Y	(i) (i)
3.	Valves, insert and withdraw lines	GE P	2 2	C C	B B	III-2 III-2	I I	Y Y	(m) (m)
4.	Valves, other	P	Other	C,A	D	B31.1	NA	N	
5.	Deleted								
6.	Piping, scram discharge volume lines	P	2	C	B	III-2	I	Y	
7.	Piping, insert and withdraw lines	P	2	C	B	III-2	I	Y	
8.	Piping, other	P	Other	C,A	D	B31.1	NA	N	
9.	Hydraulic control unit	GE	2	C	(n)	III-2	I	Y	(n)

TABLE 3.2-1: CLASSIFICATION OF SYSTEMS, COMPONENTS AND STRUCTURES (CONTINUED)

Principal Component(a)	Scope Supply (b)	Safety Class(c)	Location(d)	Quality Group Classification (e)	Principal Construction Code(f)	Seismic Category(g)	Q-List(h)	Comment
10. Electrical modules with safety function	FE	2	C	-	Other	I	Y	
11. Cable, with safety function	P	2	-	-	Other	NA	Y	(k)
12. CRD Pumps	GE/P	Other	A	-	API	NA	N	
13. Filters	GE/P	Other	A	D	VIII-1	NA	N	
V. Standby Liquid Control System (mm)								
1. Standby liquid control tank	GE	2	C	B	III-2	I	Y	
2. Pump	GE	2	C	B	III-2	I	Y	
3. Pump motor	GE	2	C	-	Other	I	Y	
4. Valves, explosive	GE	2	C	B	III-2	I	Y	
5. Valves, drywell isolation and within	P	1	C	A	III-1	I	Y	(i)
6. Valves, beyond drywell isolation valves (excluding explosive valves)	P	2	C	B	III-2	I	Y	(i)
7. Piping, downstream of outboard check valves	P	1	C	A	III-1	I	Y	(i)
8. Piping, upstream of outboard check valves	P	2	C	B	III-2	I	Y	(i)
9. Electrical modules, with safety function	GE	2	C	-	Other	I	Y	
10. Cable, with safety function	P	2	C	-	Other	NA	Y	(k)
11. Pulsation Dampener	P	2	C	B	III-2	I	Y	

TABLE 3.2-1: CLASSIFICATION OF SYSTEMS, COMPONENTS AND STRUCTURES (CONTINUED)

Principal Component(a)	Scope Supply (b)	Safety Class(c)	Location(d)	Quality Group Classification (e)	Principal Construction Code(f)	Seismic Category(g)	Q-List(h)	Comment
VI. Neutron Monitoring System (Sections 7.1.2.1.4, 7.2, and 7.6.1.5) (ff) (mm)								
1. Detector dry tube assy	GE	2	C	B	III-2	I	Y	
2. Electrical modules, IRM and APRM	GE	2	C,Z,A	-	Other	I	Y	
3. Cable, IRM and APRM, with safety function	P	2	C,Z,A	-	Other	NA	Y	(k)
VII. Reactor Protection System (Sections 7.1.2.1.1 and 7.2) (ff) (mm)								
1. Electrical modules	GE	2	C,T,Z,A	-	Other	I	Y	
2. Cable, with safety function	P	2	C,T	-	Other	NA	Y	(k)
VIII. Process Radiation Monitors (mm)								
1. Electrical modules, main steam line, containment and drywell ventilation exhaust, aux. bldg. fuel handling area pool sweep exhaust, aux. bld. fuel handling area ventilation exhaust, and control room vent	GE	2	C,Z,A	-	Other	I	Y	
2. Cable, main steam line and ventilation monitors, with safety	P	2	C,Z,A	-	Other	NA	Y	(k)

TABLE 3.2-1: CLASSIFICATION OF SYSTEMS, COMPONENTS AND STRUCTURES (CONTINUED)

Principal Component(a)	Scope Supply (b)	Safety Class(c)	Location(d)	Quality Group Classification (e)	Principal Construction Code(f)	Seismic Category(g)	Q-List(h)	Comment
3. SSW Radiation Monitor	GE	3	A	-	Other	I	Y	(zz17)
IX. RHR System (Sections 5.4.7, 6.2.2, 6.3, and 6.5.2) (ff) (mm)								
1. Heat exchangers, primary side	GE	2	A	B	III-2 & TEMA C	I	Y	
2. Heat exchangers, secondary side	GE	3	A	C	III-3 & TEMA C	I	Y	
3. Piping, within outermost isolation valves	P	1	C,A	A	III-1	I	Y	(i)
		2	C,A	B	III-2	I	Y	(i)
4. Piping, beyond outermost isolation valves	P	2	A	B	III-2	I	Y	(i)
5. Pumps	GE	2	A	B	III-2	I	Y	
6. Pump motors	GE	2	A	-	Other	I	Y	
7. Valves, isolation LPCI line	P	1	C,A	A	III-1	I	Y	
8. Valves, isolation, other	P	1	C,A	A	III-1	I	Y	(i)
	P	2	C,A	B	III-2	I	Y	(i)
9. Valves, beyond isolation valves	P	2	A	B	III-2	I	Y	(i)
10. Containment spray nozzles	P	2	C	-	Other	NA	Y	
11. Suppression pool strainers	P	2	C	B	III-2	I	Y	(jj)
12. Mechanical modules	GE	2	A	-	Other	I	Y	
13. Electrical modules with safety function	GE	2	A	-	Other	I	Y	
14. Cable, with safety function	P	2	-	-	Other	NA	Y	(k)
15. ADHRS heat exchangers	GE/P	3	A	c	III-3	I	Y	(yy)
16. ADHRS process piping	P	3	A	c	III-3	I	Y	(zz)

TABLE 3.2-1: CLASSIFICATION OF SYSTEMS, COMPONENTS AND STRUCTURES (CONTINUED)

Principal Component(a)		Scope Supply (b)	Safety Class(c)	Location(d)	Quality Group Classification (e)	Principal Construction Code(f)	Seismic Category(g)	Q-List(h)	Comment
17.	ADHRS pumps	GE/P	3	A	C (yy),(zz1)	III-3	I	Y	
18.	ADHRS pump motors	GE/P	Other	A	-	Other	NA	N	(yy)
19.	ADHRS valves	P	2/3	A	B/C	III-2/3	I	Y	(zz1)
20.	ADHRS electrical modules	P	-	A	-	Other	NA	N	
X.	Low Pressure Core Spray (mm)								
1	Piping, within outermost isolation valves	P	1	C,A	A	III-1	I	Y	(i)
			2	A	B	III-2	I	Y	(i)
2	Piping, beyond outermost isolation valves	P	2	A	B	III-2	I	Y	(i)
3	Pumps	GE	2	A	B	III-2	I	Y	
4	Pump motors	GE	2	A	-	Other	I	Y	
5	Valves, isolation and within	P	1	C,A	A	III-1	I	Y	(i)
			2	A	B	III-2	I	Y	(i)
6	Valves, beyond outermost isolation valves	P	2	A	B	III-2	I	Y	(i)
7	Electrical modules with safety function	GE	2	A	-	Other	I	Y	
8	Cable, with safety function	P	2	A	-	Other	NA	Y	(k)
9	Suppression pool strainers	P	2	C	B	III-2	I	Y	(jj)
XI.	High Pressure Core Spray (mm)								
1.	Piping, within outermost isolation valve	P	1	C,A	A	III-1	I	Y	(i)
			2	A	B	III-2	I	Y	

TABLE 3.2-1: CLASSIFICATION OF SYSTEMS, COMPONENTS AND STRUCTURES (CONTINUED)

Principal Component(a)		Scope Supply (b)	Safety Class(c)	Location(d)	Quality Group Classification (e)	Principal Construction Code(f)	Seismic Category(g)	Q-List(h)	Comment
2.	Piping, return test line to condensate storage tank beyond second isolation valve	P	Other	O	D	B31.1	NA	N	
3.	Piping, beyond outermost isolation valve, other	P	2	A	B	III-2	I	Y(i)	
4.	Pumps	GE/P	2	A	B	III-2	I	Y	
5.	Pump motors	GE/P	2	A	-	Other	I	Y	
6.	Valves, outer isolation and within	GE	1	C	A	III-1	I	Y(i)	
		P	1	C	A	III-1	I	Y	(i)
			2	A	B	III-2	I	Y	(i)
7.	Valves, beyond isolation valves, motor operated	GE	2	A	B	III-2	I	Y(i)	
8.	Valves, other	P	2	A	B	III-2	I	Y	(i)
9.	Electrical modules, with safety function	GE	2	A	-	Other	I	Y	
10.	Electrical auxiliary equipment	GE	3	A	-	Other	I	Y	
11.	Cable with safety function	P	2	A	-	Other	NA	Y	(k)
12.	Suppression pool strainers	P	2	C	B	III-2	I	Y	(jj)
13.	Valves, return test line to condensate storage tank beyond second isolation valve	P	Other	O	D	B31.1	NA	N	

TABLE 3.2-1: CLASSIFICATION OF SYSTEMS, COMPONENTS AND STRUCTURES (CONTINUED)

Principal Component(a)		Scope Supply (b)	Safety Class(c)	Location(d)	Quality Group Classification (e)	Principal Construction Code(f)	Seismic Category(g)	Q-List(h)	Comment
14.	Piping, HPCS flushing water drainline to RHR	P	Other	O	D	B31.1	NA	N	
15.	Valves, HPCS flushing water drainline to RHR	P	Other	O	D	B31.1	NA	N	
16.	Piping downstream of the last isolation valve, on the drain, vent and test lines	P	Other	O	D	B31.1	NA	N	
XII.	RCIC System (Section 5.4.6) (ff) (mm)								
1.	Piping, within outer most isolation valves	P	1 2	C,A A	A B	III-1 III-2	I I	Y Y	(i) (i)
2.	Piping, beyond outer most isolation valves	P	2	A	B	III-2	I	Y	(i)
3.	Piping and valves, return test line to condensate storage tank beyond second isolation valve	P	Other	O,A	D	B31.1	NA	N	
4.	Pumps	GE	2	A	B	III-2	I	Y	
5.	Valves, isolation and within	P	1 2	C, A A	A B	III-1 III-2	I	Y	(i)
6.	Valves, other	P	2	A	B	III-2	I	Y	(i)
7.	Turbine	GE	2	A	-	Other	I	Y	(o)
8.	Electrical modules, with safety function	GE	2	A	-	Other	NA	Y	

TABLE 3.2-1: CLASSIFICATION OF SYSTEMS, COMPONENTS AND STRUCTURES (CONTINUED)

Principal Component(a)		Scope Supply (b)	Safety Class(c)	Location(d)	Quality Group Classification (e)	Principal Construction Code(f)	Seismic Category(g)	Q-List(h)	Comment
9.	Cable, with safety function	P	2	A	-	Other	NA	Y	(k)
10.	Gland seal system air compr.	GE	2	A	D	Other	I	N	
11.	Suppression pool strainers	P	2	C	B	III-2	I	Y	(jj)
12.	Gland seal system piping and valves	P	Other	A	D	B31.1	NA	N	
XIII.	Fuel Service Equipment								
1.	Fuel preparation machine	GE	3	C,A	-	Other	I	Y	
2.	General purpose grapple	GE	2	C,A	-	Other	I	Y	
3.	Horizontal fuel transfer system	GE, P	2, Other	C,A	B,D, NA	III-2 & III-MC & B31.1	I,NA	Y,N	(p)
4.	New fuel inspection stand	GE	3	A	-	Other	I	Y	
XIV.	Reactor Vessel Service Equipment								
1.	Steam line plugs	GE	Other	C	-	Other	NA	N	
2.	Head strongback carousel	GE	3	C	-	Other	I	Y	
3.	Dryer and separator strongback	GE	Other	C	-	Other	NA	N	
XV.	In-Vessel Service Equipment								
1.	Control rod grapple	GE	3	C	-	Other	I	Y	
XVI.	Refueling Equipment								
1.	Refueling equipment platform assembly	GE	2	C	-	Other	I	Y	
2.	Refueling bellows	P	Other	C	-	Other	NA	N	

TABLE 3.2-1: CLASSIFICATION OF SYSTEMS, COMPONENTS AND STRUCTURES (CONTINUED)

Principal Component(a)	Scope Supply (b)	Safety Class(c)	Location(d)	Quality Group Classification (e)	Principal Construction Code(f)	Seismic Category(g)	Q-List(h)	Comment
3. Deleted								
4. Deleted								
5. Auxiliary Platform	GE	2	C	-	Other	I	Y	
6. Fuel Handling Platform	GE	2	A	-	Other	I	Y	
XVII. Storage Equipment								
1. Fuel storage racks	P	2	C,A	-	III-2	I	Y	(ww)
2. Defective fuel storage container	GE	2	C,A	-	Other	I	Y	
3. In-Vessel Rack	GE	2	C	-	Other	I	Y	
4. Control Rod Stor. Rack	P	2	C	-	III-3	I	Y	(zz7)
XVIII. Radwaste System								
1. Tanks, atmospheric	P	Other	W	D	API-650	NA	N	(q,r)(zz10)
2. Heat exchangers	P	Other	W	D	VIII	NA	N	(q, r)
3. Piping	P	Other	W	D	B31.1	NA	N	(q, r)
4. Pumps	P	Other	W	D	VIII	NA	N	(q, r, qq)
5. Valves	P	Other	W	D	B31.1	NA	N	(q, r)
6. Vessels	P	Other	W	D	VIII	NA	N	(q, r)
7. Tanks, 0 to 15 psig	P	Other	W	D	API-620	NA	N	(q, r, qq)
XIX. Reactor Water Cleanup System								
1. Vessels and appurtenances, filter/ demineralizer	GE	Other	C	C	III-3	NA	N	
2. Filter/demineralizer precoat subsystem	GE	Other	C	D	API-650 B31.1	NA	N	

TABLE 3.2-1: CLASSIFICATION OF SYSTEMS, COMPONENTS AND STRUCTURES (CONTINUED)

Principal Component(a)	Scope Supply (b)	Safety Class(c)	Location(d)	Quality Group Classification (e)	Principal Construction Code(f)	Seismic Category(g)	Q-List(h)	Comment
3. Heat exchangers, reactor water side and cooling water side	GE	Other	C	C	III-3 & TEMA C	NA	N	
4. Piping, between the RPV drain connection, recirculation system connections, and the outermost containment isolation valve associated with the RWCU guard pipe	P	1	C	A	III-1	I	Y	(i, xx)
5. Piping, beyond outermost isolation valves	P	Other	C	(hh)	(hh)	NA	N	(i)
6a. Pumps, cleanup recirculation and filter demineralizer holding	GE	Other	A,C	C	III-3	NA	N	
6b. Pump motors	GE	Other	A	-	Other	NA	N	

TABLE 3.2-1: CLASSIFICATION OF SYSTEMS, COMPONENTS AND STRUCTURES (CONTINUED)

Principal Component(a)		Scope Supply (b)	Safety Class(c)	Location(d)	Quality Group Classification (e)	Principal Construction Code(f)	Seismic Category(g)	Q-List(h)	Comment
7.	Valves, between the RPV drain connection, recirculation system connections, and the outermost containment isolation valve associated with the RWCU guard pipe	P	1	C	A	III-1	I	Y	(i)
8.	Valves, beyond outermost isolation valves	GE	Other	C	C	III-3	NA	N	(i)
		P	Other	C	C	III-3	NA	N	(i)
9.	Mechanical modules	GE	Other	C	-	Other	NA	N	(i)
10.	Piping and valves (other) forming part of containment boundary	P	2	C,A	B	III-2	I	Y	
11a.	Pumps, other	GE/P	Other	C	D	VIII	NA	N	
11b.	Pump, motors	GE/P	Other	C	-	Other	NA	N	
12.	Sample station	GE	Other	C	D	B31.1	NA	N	
XX.	Fuel Pool Cooling and Cleanup System (mm)								
1.	Vessels, filter/ demineralizers	GE	Other	A	C	III-3	NA	N	
2.	Drain tank	P	3	A	C	III-3	I	Y	
3.	Heat exchangers	P	3	A	C	III-3	I	Y	
4.	Piping and valves, cooling portion	P	3	A,C	C	III-3	I	Y	(i)
5a	Pumps, cooling portion	GE	3	A	C	III-3	I	Y	(i)
5b.	Pump motors, cooling portion	GE	3	A	-	Other	I	Y	

TABLE 3.2-1: CLASSIFICATION OF SYSTEMS, COMPONENTS AND STRUCTURES (CONTINUED)

Principal Component(a)	Scope Supply (b)	Safety Class(c)	Location(d)	Quality Group Classification (e)	Principal Construction Code(f)	Seismic Category(g)	Q-List(h)	Comment
6. Valves and piping, containment isolation	P	2	A,C	B	III-2	I	Y	(i)
7. Piping, cleanup portion	P	3	A	(hh)	(hh)	NA	N	
8. Tanks, other	GE	Other	A	C,D	III-3, VIII	NA	N	
9. Pumps, cleanup portion	GE	Other	A	C,D	III-3, VIII	NA	N	
10. Valves, cleanup portion	GE	Other	A	C,D	III-3, VIII	NA	N	
XXI. Control Room Panels								
1. Electrical modules, with safety function	GE	2	Z	-	Other	I	Y	
2. Cable, with safety function	P	2	Z	-	Other	NA	Y	(k)
XXII. Local Panels and Racks (Section 7.1) (ff)								
1. Electrical modules with safety function	GE	2	A,C	-	Other	I	Y	
2. Cable, with safety function	P	2	A,C	-	Other	NA	Y	(k)
XXIII. Offgas System								
1. Tanks	GE	Other	T, W	D	D100 or API-650 & SR(a)	NA	N	(ee)
2. Heat exchangers	GE	Other	T, W	D	VIII-1 & TEMA C	NA	N	(ee)
3. Piping	P	Other	T, W	D	B31.1	NA	N	(i), (ee)
4a. Pumps	GE	Other	T, W	D	Other	NA	N	(ee)
4b. Pump motors	GE	Other	T, W	NA	Other	NA	N	
5. Valves, flow control	GE,P	Other	T, W	D	B31.1	NA	N	(i), (ee)
6. Valves, other	P	Other	T, W	D	B31.1	NA	N	(i), (ee)
7. Mechanical modules, other	GE P	Other Other	T, W T,W	D D	Other Other	NA NA	N N	

TABLE 3.2-1: CLASSIFICATION OF SYSTEMS, COMPONENTS AND STRUCTURES (CONTINUED)

Principal Component(a)	Scope Supply (b)	Safety Class(c)	Location(d)	Quality Group Classification (e)	Principal Construction Code(f)	Seismic Category(g)	Q-List(h)	Comment
8. Pressure vessels	GE	Other	T, W	D	VIII-1	NA	N	(ee), (tt)
XXIV. Standby Service Water (Sections 9.2.1 and 9.2.5) (ff) (mm) System								
1. Piping	P	3	P,O,A,R,M	C	III-3	I	Y	
2. Pumps	P	3	P	C	III-3	I	Y	
3. Pump motors	P	3	P	-	Other	I	Y	
4. Fill tank	P	Other	A	D	API-650	NA	N	
5. Valves, other	P	3	P,A,R,M	C	III-3	I	Y	
6. Electrical modules, with safety function	P	3	P,A,R,M	-	Other	I	Y	
7. Cable, with safety function	P	3	P,O,R,M	-	Other	NA	Y	(k)
8. Cooling tower fans	P	3	O	C	Other	I	Y	
9. Cooling tower fan motors	P	3	O	-	Other	I	Y	
10. Valves, isolation	P	2	C,A	B	III-2	I	Y	
XXV. Component Cooling Water Systems (mm)								
1. Piping and valves forming part of containment boundary	P	2	A,C,B	III-2	I	Y		
2. Piping and valves associated with fuel pool heat exchangers	P	3	A	C	III-3	I	Y	
3. Heat exchangers	P	Other	A	D	VIII-1 TEMA C	NA	N	
4. Piping and valves, other	P	Other	A,C,D	B31.1	NA	N		
XXVI. Instrument, Plant, and Service Air Systems								

TABLE 3.2-1: CLASSIFICATION OF SYSTEMS, COMPONENTS AND STRUCTURES (CONTINUED)

Principal Component(a)		Scope Supply (b)	Safety Class(c)	Location(d)	Quality Group Classification (e)	Principal Construction Code(f)	Seismic Category(g)	Q-List(h)	Comment
1.	Piping and valves forming part of containment boundary	P	2	A,C,B	III-2	I	Y		
2.	Vessels and accumulators other than Item 1	P	Other	M,A,T	D	VIII-1	NA	NA	
3.	Piping and valves other than Items 1 and 2	P	Other	M,A,T	D	B31.1	NA	NA	
XXVII.	Diesel Generator Systems								
1.	Standby diesel generators								
	a. Diesel generators	P	2	R	-	Other	I	Y	
	b. Electrical modules with safety functions	P	3	R	-	Other	I	Y	
	c. Cable, with safety functions	P	3	R	-	Other	NA	Y	(k)
2.	HPCS diesel generator								
	a. Diesel generators	GE	2	R	-	Other	I	Y	
	b. Electrical modules with safety functions	GE	3	R	-	Other	I	Y	
	c. Cable, with safety functions	P	3	R	-	Other	NA	Y	(k)
XXVIII.	Combustible Gas Control System (mm)								

TABLE 3.2-1: CLASSIFICATION OF SYSTEMS, COMPONENTS AND STRUCTURES (CONTINUED)

Principal Component(a)		Scope Supply (b)	Safety Class(c)	Location(d)	Quality Group Classification (e)	Principal Construction Code(f)	Seismic Category(g)	Q-List(h)	Comment
1.	Piping and valves from primary containment through outer isolation valve	P	2	C	B	III-2	I	Y	
2.	Drywell purge compressor	P	2	C	B	III-2	I	Y	
3.	Drywell purge compressor motor	P	2	C	-	Other	I	Y	
4.	Deleted								
5.	Screens	P	C	-	Other	NA	N	Y	
6.	Containment purge comp.	P	Other	A	D	VIII-1	NA	N	
	Containment purge comp/motor	P		A	-	Other	NA	N	
7.	Elect. modules & cables	P	2	A	NA	NA	I	Y	
8.	Piping and valves from drywell purge compressor	P	2	C	B	III-2	I	Y	
9.	Piping and valves for vacuum relief system (includes drywell containment vacuum breakers)	P	2	C	B	III-2	I	Y	
10.	Piping and valves, containment purge system, other	P	Other	A	D	B31.1	NA	N	
11.	Hydrogen analyzer	P	2	A	-	Other	I	Y	

TABLE 3.2-1: CLASSIFICATION OF SYSTEMS, COMPONENTS AND STRUCTURES (CONTINUED)

Principal Component(a)	Scope Supply (b)	Safety Class(c)	Location(d)	Quality Group Classification (e)	Principal Construction Code(f)	Seismic Category(g)	Q-List(h)	Comment
12. Hydrogen igniter assemblies	P	2	C	B	III-2	I	Y	
XXIX. Standby Gas Treatment System (mm)								
1. Pressure components with safety function:								
a. Filter housing	P	2	A	-	Other	I	Y	
b. Fans and motors	P	2	A	-	Other	I	Y	
c. Ductwork	P	2	A	-	Other	I	Y	
d. Dampers and valves installed in ductwork	P	2	A	-	Other	I	Y	
e. Drain piping up to and including first isolation valve	P	2	A	B	III-2	I	Y	
2. Pressure components without safety function:								
a. Drain piping downstream of first normally closed isolation valve	P	Other	A	D	B31.1	NA	N	
3. Elec. modules with safety function	P	2	A	-	NA	I	Y	
4. Cable with safety function	P	2	A	-	NA	NA	Y	(k)

TABLE 3.2-1: CLASSIFICATION OF SYSTEMS, COMPONENTS AND STRUCTURES (CONTINUED)

Principal Component(a)		Scope Supply (b)	Safety Class(c)	Location(d)	Quality Group Classification (e)	Principal Construction Code(f)	Seismic Category(g)	Q-List(h)	Comment
5.	Charcoal filter media	P	2	A	-	Other	NA	Y	(yy)
XXX.	ESF Equipment Area HVAC Systems (mm)								
1.	Cooling coils, piping and valves with safety function	P	3	A	C	III-3	I	Y	
2.	Other components with safety function	P	3	A	C	Other	I	Y	
3.	Piping and valves without safety function	P	Other	A	D	B31.1	NA	N	
XXXI.	Power Conversion System								
1.	Main steam piping extending from the main steam shutoff valve up to the turbine stop valve (not including branch lines)	P	Other	T	D	B31.1	NA	N	(u)
2.	Steam piping and valves, other, including branch lines	P	Other	T	D	B31.1	NA	N	
3.	Reactor feedwater piping and valves, other	P	Other	T	D	B31.1	NA	N	ZZ9
4.	Main steam leads from turbine control valve to turbine casing	P	Other	T	D	B31.1	NA	N	(w)
XXXII.	Condensate Storage and Transfer System								

TABLE 3.2-1: CLASSIFICATION OF SYSTEMS, COMPONENTS AND STRUCTURES (CONTINUED)

Principal Component(a)		Scope Supply (b)	Safety Class(c)	Location(d)	Quality Group Classification (e)	Principal Construction Code(f)	Seismic Category(g)	Q-List(h)	Comment
1.	Condensate storage tank	P	Other	O	D	(x)	NA	N	
2.	Piping and valves								
	a. Forming part of containment boundary	P	2	A,C	B	III-2	I	Y	
	b. HPCS and RCIC suction piping	P	2	A,M	B	III-2	I	Y	
	c. Other	P	Other	O,T,M	D	B31.1	NA	N	
3.	Other components	P	Other	M	D	VIII-1	NA	N	
XXXIII.	Auxiliary ac Power System (mm)								
1.	Class 1E lower medium voltage system								
	a. Metal clad switch-gear 4.16 kV	P	2	Z	-	Other	I	Y	
	b. 9 kV power cable	P	2	A, R, Z, P, O	-	Other	I	Y	
	c. Large induction motors, 250 hp and larger	P	2	A, P	-	Other	I	Y	
2.	Class 1E low voltage system								
	a. Load center unit substations	P	2	A, P	-	Other	I	Y	
	b. Motor control centers	P	2	A, P, Z	-	Other	I	Y	
	c. 1000 and 600 volts power and control cable	P	2	C, A, P, Z, O, R	-	Other	I	Y	

TABLE 3.2-1: CLASSIFICATION OF SYSTEMS, COMPONENTS AND STRUCTURES (CONTINUED)

Principal Component(a)		Scope Supply (b)	Safety Class(c)	Location(d)	Quality Group Classification (e)	Principal Construction Code(f)	Seismic Category(g)	Q-List(h)	Comment
	d. Integral and functional hp induction motors	P	2	A, P, Z, T, R, O	-	Other	I	Y	
3.	Cable penetrations and protection								
	a. Containment cable penetration	P	2	C, A	-	Other	I	Y	
	b. Drywell cable penetration	P	2	C	-	Other	I	Y	
	c. Penetration protection cabinets	P	2	A	-	Other	I	Y	
4.	Conduit supports and tray supports (including safety-related ductbanks and manholes)	P	2	A, P, Z, O, R, C	-	Other	I	Y	
5.	Raceway installations	P	2	-	-	Other	I	Y	
6.	Load shedding and sequencing panels	P	2	Z	-	Other	I	Y	
7.	Transformers (only load centers and MCC transformers)	P	2	Z, A, P	-	Other	I	Y	
XXXIV.	125/250 Volt dc Power System (mm)								
1.	125 Volt Class 1E dc Power System								
	a. Batteries	P	2	Z	-	Other	I	Y	

TABLE 3.2-1: CLASSIFICATION OF SYSTEMS, COMPONENTS AND STRUCTURES (CONTINUED)

Principal Component(a)	Scope Supply (b)	Safety Class(c)	Location(d)	Quality Group Classification (e)	Principal Construction Code(f)	Seismic Category(g)	Q-List(h)	Comment
b. Battery chargers	P	2	Z	-	Other	I	Y	
c. dc distribution panels	P	2	Z,A	-	Other	I	Y	
XXXV. Other Components								
1. Containment polar crane	P	3	C	-	Other	I	Y	
2. Cask crane	P	3	A	-	Other	I	Y	
3. Control room HVAC system (mm)								
a. Fans and motors	P	3	2	-	Other	I	Y	
b. Cooling coils (DX)	P	3	2	-	NRI	I	Y	
c. Filters	P	3	2	-	Other	I	Y	
d. Humidifiers	P	Other	Z	-	Other	I	Y	
e. Condenser	P	3	Z	C	III-3	I	Y	
f. Charcoal filter housing	P	3	Z	-	Other	I	Y	
g. Ductwork and dampers excluding purge exhaust ductwork extending from the second containment isolation valve to the exhaust louver	P	3	Z	-	Other	I	Y	

TABLE 3.2-1: CLASSIFICATION OF SYSTEMS, COMPONENTS AND STRUCTURES (CONTINUED)

Principal Component(a)	Scope Supply (b)	Safety Class(c)	Location(d)	Quality Group Classification (e)	Principal Construction Code(f)	Seismic Category(g)	Q-List(h)	Comment
h. Valves with safety isolation function	P	3	Z	C	III-3	I	Y	
i. Utility exhaust fan	P	Other	Z	-	Other	NA	N	
j. Unit heaters	P	Other	Z	-	Other	NA	N	
k. Electrical modules with safety function	P	3	Z	-	Other	I	Y	
l. Cable with safety function	P	3	Z	-	Other	NA	Y	(k)
4. Diesel generator room ventilation (mm)								
a. Outside air fans and motors	P	3	R	-	Other	I	Y	
b. Ductwork and dampers	P	3	R	-	Other	I	Y	
c. Fan coil units including fans, motors, water coils, electrical coils, drain piping and valves without safety function	P	Other	R	-	Other	NA	N	

TABLE 3.2-1: CLASSIFICATION OF SYSTEMS, COMPONENTS AND STRUCTURES (CONTINUED)

Principal Component(a)	Scope Supply (b)	Safety Class(c)	Location(d)	Quality Group Classification (e)	Principal Construction Code(f)	Seismic Category(g)	Q-List(h)	Comment
5. Standby service water pump house ventilation system (mm)								
a. Outside air fans and motors	P	3	P	-	Other	I	Y	
b. Ductwork and dampers	P	3	P	-	Other	I	Y	
c. Unit heaters including fans, motors and electrical coils without safety function	P	Other	P	-	Other	NA	N	
6. Safeguard switchgear and battery rooms ventilation system (mm)								
a. Exhaust fans and motors	P	3	Z	-	Other	I	Y	
b. Supply air fans and motors	P	3	Z	-	Other	I	Y	
c. Ductwork and dampers	P	3	Z	-	Other	I	Y	
d. Electrical heaters	P	Other	Z	-	Other	I	N	

TABLE 3.2-1: CLASSIFICATION OF SYSTEMS, COMPONENTS AND STRUCTURES (CONTINUED)

Principal Component(a)		Scope Supply (b)	Safety Class(c)	Location(d)	Quality Group Classification (e)	Principal Construction Code(f)	Seismic Category(g)	Q-List(h)	Comment
	e. Remote shutdown panel room heat pump and associated ductwork and controls	P	Other	Z	-	Other	NA	N	
7.	Control building HVAC system								
	a. Elevator mach. rm. check dampers with safety function	P	3	Z	-	Other	I	Y	
	b. Other components without safety function	P	Other	Z	-	Other	NA	N	
XXXVI.	Drywell Chilled Water System								
1.	Piping and valves forming part of containment boundary	P	2	A,C	B	III-2	I	Y	
XXXVII.	Containment Cooling System								
1.	Piping and valves forming part of containment boundary	P	2	A, C	B	III-2	I	Y	
XXXVIII.	Makeup Water Treatment System								

TABLE 3.2-1: CLASSIFICATION OF SYSTEMS, COMPONENTS AND STRUCTURES (CONTINUED)

Principal Component(a)	Scope Supply (b)	Safety Class(c)	Location(d)	Quality Group Classification (e)	Principal Construction Code(f)	Seismic Category(g)	Q-List(h)	Comment
1. Piping and valves forming part of containment boundary	P	2	A,C	B	III-2	I	Y	
XXXIX. Auxiliary Building Isolation (mm)								
1. Piping and valves forming part of auxiliary building boundary	P	3	A,Other	C	III-3	I	Y	(y) (pp)
XL. Drywell Insulation								
1. Insulation on piping which is within the drywell	P	Other	C	-	Other	NA	N	(z)
XLI. Diesel Generator Auxiliary Systems (ii) (mm)								
1. Fuel Oil Subsystem								
a. Standby Diesel Generators								
1) Piping and valves-fuel oil storage tanks today tanks; fuel oil day tanks to engine; fuel oil bypass and return lines; day tank overflow line to storage tank	P	3	O,M,R	C	III-3	I	Y	

TABLE 3.2-1: CLASSIFICATION OF SYSTEMS, COMPONENTS AND STRUCTURES (CONTINUED)

Principal Component(a)	Scope Supply (b)	Safety Class(c)	Location(d)	Quality Group Classification (e)	Principal Construction Code(f)	Seismic Category(g)	Q-List(h)	Comment
2) Piping and valves-fuel oil drip return line	P	Other	R	D	B31.1(rr)	NA	N	(nn)
3) Fuel oil storage tanks and day tanks	P	3	O,R	C	III-3	I	Y	
4) Fuel oil pumps (except drip return pump)	P	3	M,R	C	III-3(t)	I	Y	
5) Pump motors (except dc motor-driven fuel oil pump motor)	P	3	M,R	-	Other	I	Y	
6) DC motor-driven fuel oil pump motor	P	Other	R	-	Other	NA	N	
7) Fuel oil drip return pump	P	Other	R	D	HI	NA	N	(nn)
8) Fuel oil drip return tank	P	Other	R	D	VIII-1	NA	N	(nn)
9) Fuel oil drip return pump motor	P	Other	R	-	Other	NA	N	
b. HPCS Diesel Generator								

TABLE 3.2-1: CLASSIFICATION OF SYSTEMS, COMPONENTS AND STRUCTURES (CONTINUED)

Principal Component(a)	Scope Supply (b)	Safety Class(c)	Location(d)	Quality Group Classification (e)	Principal Construction Code(f)	Seismic Category(g)	Q-List(h)	Comment
1) Piping and valves-fuel oil storage tank to day tank, fuel oil day tank to engine skid, day tank overflow to storage tank, fuel oil return lines from engine skid to day tank	P	3	O,M,R	C	III-3	I	Y	-
2) Piping and valves-engine skid	GE	3	R	D	B31.1(ZZ2)	I	Y	ZZ6
3) Fuel oil storage tanks and day tank	P	3	O,R	C	III-3	I	Y	-
4) Fuel oil transfer pump	P	3	O	C	III-3	I	Y	-
5) Fuel oil transfer pump motor	P	3	O	-	Other	I	Y	-
6) DC motor-driven fuel oil pump motor	GE	Other	R	-	Other	NA	Y	-
7) Fuel oil pumps								
i. Engine driven GE	GE	3	R	D	Other	I	Y	ZZ4
ii. DC motor	GE	3	R	D	Other	I	Y	ZZ4
2. Lube Oil Subsystem (oo)								

TABLE 3.2-1: CLASSIFICATION OF SYSTEMS, COMPONENTS AND STRUCTURES (CONTINUED)

Principal Component(a)	Scope Supply (b)	Safety Class(c)	Location(d)	Quality Group Classification (e)	Principal Construction Code(f)	Seismic Category(g)	Q-List(h)	Comment
a. Standby Diesel Generators								
1) Piping and valves	P	3	R	D(aa)	B31.1(rr)	I	Y	(cc)
2) Heat exchanger	P	3	R	D	VIII-1	I	Y	
3) Pumps	P	3	R	D	HI	I	Y	
4) Pressure vessels	P	3	R	D	VIII-1	I	Y	
5) Electric-driven Aux. lube oil pump motor (standby diesel generators)	P	3	R	-	Other	I	Y	
b. HPCS Diesel Generator								
1) Engine skid mounted piping and valves	GE	3	R	D	B31.1(ZZ2)	I	Y	ZZ6
2) Heat exchanger	GE	3	R	C	III-3	I	Y	
3) Pumps								
i. Engine Mounted	GE	3	R	D	Other	I	Y	ZZ4
ii. Skid Mounted	GE	3	R	D	Other	I	Y	ZZ4
3. Jacket Water Subsystem (00)								
a. Standby Diesel Generators								
1) Piping and valves	P	3	R	D(aa)	B31.1 (rr,ZZ13)	I	Y	ZZ14,ZZ15 (cc)
2) Heat exchanger	P	3	R	D	VIII-1	I	Y	

TABLE 3.2-1: CLASSIFICATION OF SYSTEMS, COMPONENTS AND STRUCTURES (CONTINUED)

Principal Component(a)	Scope Supply (b)	Safety Class(c)	Location(d)	Quality Group Classification (e)	Principal Construction Code(f)	Seismic Category(g)	Q-List(h)	Comment
3) Pressure vessel	PP	3	R	D	VIII-1	I	Y	
4) Pumps	P	3	R	D	HI	I	Y	ZZ16
5) Electric-driven jacket water pump motor (standby diesel generators)	P	3	R	-	Other	I	Y	
b. HPCS Diesel Generator								
i) Engine mounted	GE	3	R	D	Other	I	Y	ZZ3
ii) Engine skid mounted	GE	3	R	D	B31.1 (ZZ2)	I	Y	ZZ6
2) Heat exchanger	GE	3	R	C	III-3	I	Y	
3) Engine mounted pumps	GE	3	R	D	Other	I	Y	ZZ4
4. Combustion Air Intake and Exhaust Subsystem								
a. Standby Diesel Generators								
1) Piping and valves	P	Other	R	D	B31.1	I	N	(nn)
2) Silencers and filters	P	Other	R	D	Other	I	N	(nn)
b. HPCS Diesel Generator								

TABLE 3.2-1: CLASSIFICATION OF SYSTEMS, COMPONENTS AND STRUCTURES (CONTINUED)

Principal Component(a)	Scope Supply (b)	Safety Class(c)	Location(d)	Quality Group Classification (e)	Principal Construction Code(f)	Seismic Category(g)	Q-List(h)	Comment
1) Piping and valves flexible connections to air intake silencers to air intake filters, Flexible connections to exhaust silencers, from exhaust silencers to discharge	P	Other	R	D	B31.1	I	N	(nn)
2) Silencers and filters	GE	Other	R	D	Other	I	(nn)	
5. Starting Air Subsystem								
a. Standby Diesel Generators								
1) Piping and valves from starting air compressors up to (not including) inlet check valve at starting air storage tanks	P	Other	R	D	B31.1	NA	N	

TABLE 3.2-1: CLASSIFICATION OF SYSTEMS, COMPONENTS AND STRUCTURES (CONTINUED)

Principal Component(a)	Scope Supply (b)	Safety Class(c)	Location(d)	Quality Group Classification (e)	Principal Construction Code(f)	Seismic Category(g)	Q-List(h)	Comment
2) Piping and valves from and including inlet check valve at starting air storage tanks to engine air interface connections	P	3	R	D	III-3(SS)	I	Y	
3) Piping and valves:								
i. from the engine air interface connections up to and including the starting air solenoid valves.	P	3	R	D	(rr)	I	Y	-
ii. from starting air solenoid valves to engine.	P	3	R	C	III-3	I	Y	(ZZ2), (ZZ8)
4) Pressure vessels starting air storage tanks	P	3	R	D	VIII-1	I	Y	
5) Pressure vessels starting air dryer tanks	P	Other	R	D	VIII-1			
6) Compressors	P	Other	R	D	Other	NA	N	
b. HPCS Diesel Generator								

TABLE 3.2-1: CLASSIFICATION OF SYSTEMS, COMPONENTS AND STRUCTURES (CONTINUED)

Principal Component(a)	Scope Supply (b)	Safety Class(c)	Location(d)	Quality Group Classification (e)	Principal Construction Code(f)	Seismic Category(g)	Q-List(h)	Comment
1) Piping and valves-starting air compressors up to (not including) inlet check valve at starting air storage tanks	GE,P	Other	R	D	B31.1	NA	N	-
2) Piping and valves-from and including inlet check valve at tarting air storage tanks to flexible connections on outlet of starting air skid	GE	3	R	D	B31.1	I	Y	-
3) Piping and valves from the flexible connections on the engine skid to the engine air inlets	GE	3	R	D	Other	I	Y	-

TABLE 3.2-1: CLASSIFICATION OF SYSTEMS, COMPONENTS AND STRUCTURES (CONTINUED)

Principal Component(a)		Scope Supply (b)	Safety Class(c)	Location(d)	Quality Group Classification (e)	Principal Construction Code(f)	Seismic Category(g)	Q-List(h)	Comment
	4) Piping flexible connections on outlet of starting air skid up to and including the flexible connections on engine skid	P	3	R	C	III-3	I	Y	-
	5) Pressure vessels starting air storage tanks	GE	3	R	C	III-3	I	Y	-
	6) Pressure vessels starting air dryer tanks	GE	Other	R	D	VIII-1	NA	N	-
	7) Compressors	GE	Other	R	D	Other	NA	N	-
XLII.	Suppression Pool Makeup System (mm)								
1.	Piping and valves	P	2	C	B	III-2	I	Y	
XLIII.	Main Steam Isolation Valve Leakage Control System (mm)								
1.	Heaters and blowers	GE	2	A	-	Other	I	Y	(zz12)
2a.	Piping and valves up to and including first inboard system isolation valve	P	1	A	A	III-1	I	Y	
2b.	Other	P	2	A	B	III-2	I	Y	
3.	Cables with safety function	P	2	A,C,Z	-	Other	NA	Y	(k)

TABLE 3.2-1: CLASSIFICATION OF SYSTEMS, COMPONENTS AND STRUCTURES (CONTINUED)

Principal Component(a)		Scope Supply (b)	Safety Class(c)	Location(d)	Quality Group Classification (e)	Principal Construction Code(f)	Seismic Category(g)	Q-List(h)	Comment
4.	Electrical modules with safety function	GE	2	A,C,Z	-	Other	I	Y	
XLIV.	Feedwater Leakage Control System (mm)								
1.	Piping and valves of the inboard system up to and including the motor-operated isolation valve	P	2	A	B	III-2	I	Y	
2.	Piping and valves of the outboard system and the piping beyond the motor-operated isolation valve of the inboard system	P	2	A,C	B	III-2	I	Y	
XLV.	Suppression Pool Cleanup System								
1.	Piping and valves forming part of the containment boundary	P	2	A,C	B	III-2	I	Y	
XLVI.	Civil Structures (Section 6.2.1) (ff)								
1.	Containment	P	2	C	-	Other	I	Y	
2.	Auxiliary building	P	2	A	-	Other	I	Y	
3.	Diesel generator building	P	3	R	-	Other	I	Y	
4.	Standby service water cooling towers and storage basins	P	3	O	-	Other	I	Y	

TABLE 3.2-1: CLASSIFICATION OF SYSTEMS, COMPONENTS AND STRUCTURES (CONTINUED)

Principal Component(a)		Scope Supply (b)	Safety Class(c)	Location(d)	Quality Group Classification (e)	Principal Construction Code(f)	Seismic Category(g)	Q-List(h)	Comment
5.	Condensate storage tank and refueling water storage tank basin	P	Other	O	-	Other	NA	N	
6.	Control building	P	3	Z	-	Other	I	Y	
7.	Enclosure building	P	3	M	-	Other	I	Y	
8.	Radwaste building	P	Other	W	-	Other	(s)	N	
9.	Spent fuel pool	P	2	A	-	Other	I	Y	
10.	Missile barriers (containment, control, auxiliary, diesel generator buildings)	P	Other	A,C,R,Z	-	Other	I	Y	
11.	Seismic Category I electrical ductbanks and manholes	P	3	O	-	Other	I	Y	
12.	Reactor shield wall	P	Other	C	-	Other	I	Y	
13.	Drywell wall, roof, and head	P	3	C	-	Other	I	Y	
14.	Weir wall	3	C	-	Other	I	Y	Y	
15.	Drywell vents	P	3	C	-	Other	I	Y	
16.	Filter/demineralizer compartment	P	Other	C	-	Other	I	Y	
17.	Main steam tunnel	P	Other	A,C	-	Other	I	Y	
18.	RWCU pipe chase	P	Other	C	-	Other	I	Y	

TABLE 3.2-1: CLASSIFICATION OF SYSTEMS, COMPONENTS AND STRUCTURES (CONTINUED)

Principal Component(a)	Scope Supply (b)	Safety Class(c)	Location(d)	Quality Group Classification (e)	Principal Construction Code(f)	Seismic Category(g)	Q-List(h)	Comment
19. RWCU heat exchanger, holding pumps, and valve nest compartments	P	Other	C	-	Other	I	Y	
XLVII. Leak Detection System (mm)								
1. Electrical modules, instrumentation, with safety function	GE, P	2	A, C, Z	-	Other	I	Y	(dd) zz18
XLVIII. Post-Accident High-Range Radiation Monitoring System (mm)								
1. Radiation detectors	P	2	C	-	Other	I	Y	
2. Indicating trip units	P	2	Z	-	Other	I	Y	
3. Radiation recorder	P	2	Z	-	Other	I	Y	
XLIX. Containment Leak Rate Test System								
1. Piping and valves forming part of containment boundary	P	2	A,C	B	III-2	I	Y	
L. Plant Chilled Water System								
1. Piping and valves forming part of containment boundary	P	2	A,C	B	III-2	I	Y	
LI. Reactor Water Cleanup Filter Demineralizer System								
4. Piping and valves forming part of containment boundary	P	2	A,C	B	III-2	I	Y	

TABLE 3.2-1: CLASSIFICATION OF SYSTEMS, COMPONENTS AND STRUCTURES (CONTINUED)

Principal Component(a)	Scope Supply (b)	Safety Class(c)	Location(d)	Quality Group Classification (e)	Principal Construction Code(f)	Seismic Category(g)	Q-List(h)	Comment
LII. Floor and Equipment Drain System								
1. Piping and valves forming part of containment boundary	P	2	A,C	B	III-2	I	Y	
LIII. Containment and Drywell Instrumentation and Control System								
1. Piping and valves forming part of containment boundary	P	2	A,C	B	III-2	I	Y	
LIV. Plant Service Water System								
1. Piping and valves serving ADHRS, inside RHR C pump room up through room cooler isolation valves	P	3	A	C	III-3	I	Y	
2. Piping and valves serving as SSW isolation boundary	P	3	A	C	III-3	I	Y	

TABLE 3.2-1: CLASSIFICATION OF SYSTEMS, COMPONENTS AND STRUCTURES (CONTINUED)

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

(b)
GE = General Electric
P = Entergy Operations

(c) 1, 2, 3, Other = Safety classes defined in subsection 3.2.4

(d)
A = auxiliary building
C = part of, or within, containment
M = any other location
O = outdoors onsite
P = pump house
R = diesel generator building
T = turbine building
W = radwaste building
Z = control building

(e) A, B, C, D - NRC quality groups defined in Regulatory Guide 1.26. The equipment is constructed in accordance with the codes listed in Tables 3.2-2 and 3.2-3. N/A -Quality Group Classification not applicable to this equipment.

(f) Notations for principal construction codes are:

III-1, 2, 3, MC, NG = ASME Boiler and Pressure Vessel Code, Section III, Subsections NB, NC, ND, MC, NG, NF

VIII-1 = ASME Boiler and Pressure Vessel Code, Section VIII, Div. 1

API-620 = API620, Large, Welded, Low Pressure Storage Tanks

API-650 = API650, Welded Steel Tanks for Oil Storage

TEMP C = Tubular Exchanger Manufacturers Association, Class C

B31.1 = ANSI Std B31.1 Code for Power Piping

SR(a) = Nondestructive Tests Examination Requirements per ASME Section VIII, Div. 1

D100 = AWWA-100, Standard for Steel Tanks, Standpipes, Reservoirs, and Elevated Tanks for Water Storage

HI = Hydraulic Institute Standards

TABLE 3.2-1: CLASSIFICATION OF SYSTEMS, COMPONENTS AND STRUCTURES (CONTINUED)

Other = The principal code/standard/criteria for design, fabrication or construction of the component, system, or structure is "other" than those listed in this section. This includes civil and structural codes used in the design of safety-related structures identified in Sections 3.8, 3.9, or 9.1, and construction codes and guides for electrical equipment as identified in Section 8.1. Also included would be "no applicable code/standard/criteria."

(g) I - The equipment is constructed in accordance with the seismic requirements for the safe shutdown earthquake as described in Section 3.7, Seismic Design.
NA - The seismic requirements for the safe shutdown earthquake are not applicable to the equipment.

(h) Y - Require compliance with the requirements of 10 CFR 50, Appendix B.
N - Not within the scope of 10 CFR 50, Appendix B.

(i) 1. Lines 3/4 in. and smaller that are part of the reactor coolant pressure boundary are Safety Class 2.

2. All instrument lines that are connected to the reactor coolant pressure boundary and are utilized to actuate safety systems are designed and installed per Safety Class 2 from the outer isolation valve or the process shutoff valve (root valve) to the sensing instrumentation; however, N-stamping per NA-8000 of ASME Section III will not be performed beyond the root valve.

3. All instrument lines which are connected to the reactor coolant pressure boundary and are not utilized to actuate safety systems are Code Group D from the Outer isolation valve of the process shutoff valve (root valve) to the sensing instrumentation.

4. All other instrument lines:

- Through the root valve shall be of the same classification as the system to which they are attached
- Beyond the root valve, if used to actuate a safety system, shall be designed and installed per the same classification as the system to which they are attached; however, N-stamping per NA-8000 of ASME Section III will not be performed.

TABLE 3.2-1: CLASSIFICATION OF SYSTEMS, COMPONENTS AND STRUCTURES (CONTINUED)

- o Beyond the root valve, if not used to actuate a safety system, shall be Code Group D

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

(j) Safety/relief valve discharge piping is Quality Group C and seismic Category I from the relief valve to the anchor point; Quality Group D and nonseismic Category I from the anchor point to where the piping enters the weir annulus in the drywell for the non-ADS safety/relief valves, and Quality Group C and seismic Category I from the anchor point to where the piping enters the weir annulus in the drywell for the ADS safety/relief valves; and Quality Group C and seismic Category I from that point up to and including the quencher for both ADS and non-ADS safety/relief valves. Refer to Figure 5.2-8 for details.

(k) The trays and supports for safety-related cables are classified as seismic Category I.

(l) In addition to a swing check valve inside the drywell and a positive-acting check valve outside the drywell similar to an Atwood-Morrill boiler feed check valve as described in Catalog 63, Section I, a third valve with high leak-tight integrity will be provided in each line. The spring-loaded piston operator of the positive-acting check valve will be held open by air pressure during normal operation. Fail-open solenoid valves will be used to release air pressure to permit the check-valve piston operator to close. The positive-acting check valve and the high leak-tight integrity isolation valve will be remote manually operated from the control room using signals that indicate loss of control rod drive system return line flow.

(m) The control rod drive insert and withdraw lines from the drive flange up to and including the first valve on the hydraulic control unit is Safety Class 2.

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

TABLE 3.2-1: CLASSIFICATION OF SYSTEMS, COMPONENTS AND STRUCTURES (CONTINUED)

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

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

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

The pressure-containing portions (hydraulic, air, nitrogen) of the hydraulic control unit are designed, constructed, and inspected according to applicable codes.

The insert isolation valve (V-101), the withdraw isolation valve (V-102), and the upper scram discharge riser (V-114, V-112, and associated piping) meet all the requirements of ASME Boiler and Pressure Vessel Code, Section III, Class 2. The hydraulic accumulator and the nitrogen bottle meet the requirements of ASME Code, Section VIII. The remaining pressure integrity portions of the hydraulic control unit are designed to and meet all requirements of ANSI Std. B31.1. The two air pilot valves, the four directional control valves, the pressure gage, and the pressure switch have certain components that do not meet certain aspects ANSI STD B31.1 (i.e., wall thickness, material); they are, however, acceptable for code applications as defined in paragraph 104.7 of ANSI STD B31.1. These components are purchased from approved selected vendors for use in this pressure and temperature application. Also, we have extensive operating experience in the use of these components and they have proven reliability in our application.

TABLE 3.2-1: CLASSIFICATION OF SYSTEMS, COMPONENTS AND STRUCTURES (CONTINUED)

(o) The turbine does not fall within the applicable design codes. To assure that the turbine is fabricated to the standards commensurate with their safety and performance requirements, General Electric has established specific design requirements for this component which are as follows:

1. All welding was qualified in accordance with Section IX, ASME Boiler and Pressure Vessel Code.

2. All pressure-containing castings and fabrications were hydrotested in 1.5 X design pressure.

3. All high-pressure castings were radiographed according to:

   ASTM E-94
   E-142 Maximum feasible volume
   E-71, 186 or 280 Severity level 3

4. As-cast surfaces were magnetic particle or liquid penetrant tested according to ASME, Section III, Paragraph NB-2575, NC-2576, or NB-2576.

5. Wheel and shaft forgings were ultrasonically tested according to ASTM A-388.

6. Butt-welds were radiographed according to ASME, and magnetic particle or liquid penetrant tested according to ASME Boiler and Pressure Vessel Code. Acceptance standards were in accordance with ASME Boiler and Pressure Vessel Code Section III Paragraph NB-5340, NC-5340, NB-5350, NC-5350, respectively.

7. Notification is made on major repairs, and records maintained, thereof.

8. Record system and traceability according to ASME Section III, NA-4000

9. Control and identification according to ASME Section III, NA-4000,

10. Procedures conformed to ASME Section III, NB-5100, NC-5100,

11. Inspection personnel were qualified according to ASME Section III, NB-5500, NC-5500.

(p) The transfer tube, its support, and the containment penetration closure are designed to Seismic Category I requirements. The carrier, insert, upenders, valve, and other related hardware are non-Seismic Category I.

TABLE 3.2-1: CLASSIFICATION OF SYSTEMS, COMPONENTS AND STRUCTURES (CONTINUED)

The transfer tube, tube support, tube penetration sleeve, tube bellows assembly, and the containment penetration closure are designed, fabricated, tested, and inspected in accordance with the ASME Boiler and Pressure Vessel Code as follows:

Transfer tube, support, ASME Code Section III and penetration sleeve

Containment penetration ASME Code Section III, closure and transfer Class MC
tube bellows

The portion of the transfer tube and the parts immediately adjacent and attached to that portion used for containment isolation are Quality Group B. The remainder of the tube and the valves is Quality Group D. For those parts of the fuel transfer system, such as the sheave bore, fuel carrier up-ender, and similar parts for which no codes exist, quality group clarification did not apply.

(q) The liquid and solid radwaste systems were designed as seismic Category I and as Quality Group D plus additional design/material/testing requirements except for those portions of the system that can be postulated to fail, and for which a conservative analysis indicates doses or concentrations not in excess of 10 CFR Part 20 limits. The additions to Quality Group D requirements are as follows. These additions satisfy the requirements of Branch Technical Position ETSB 11-1, Revision 1, Section IV:

1.Materials

Materials for pressure-retaining components of process systems shall be selected from those covered by the material specifications listed in Section III of ASME Boiler and Pressure Vessel Code (or the equivalent ASTM or AWS specifications), except that cast, wrought, or malleable iron materials may not be used. The components meet all of the mandatory requirements of the material specifications with regard to manufacture, examination, repair, testing, identification, and certification. Plastic pipe is not used.

2.Construction and Testing

Pressure-retaining components of process systems utilize weld construction to the maximum practicable extent. Flanged joints are used only where serviceability requirements clearly indicate that such construction is preferable. Screwed connections in which threads provide the only seal are not permitted.

TABLE 3.2-1: CLASSIFICATION OF SYSTEMS, COMPONENTS AND STRUCTURES (CONTINUED)

Screwed connections backed up by seal welding or mechanical joints are permitted only on lines 3/4 inch nominal pipe size or larger but less than 2-1/2 inch nominal pipe size. In lines 2-1/2 inch nominal pipe size and above, pipe welds are of the butt joint type with backing rings not allowed in lines carrying sludges, resins, etc.

All welding constituting the pressure boundary of pressure-retaining components is performed by qualified welders employing qualified welding procedures. Qualifications are in accordance with Section IX of the ASME Boiler and Pressure Vessel Code. Completed process systems are pressure tested to the maximum practicable extent.

Piping systems other than sumps and floor drains are hydrostatically tested in their entirety, utilizing temporary plugs at atmospheric tank connections. Hydrostatic testing is performed at a pressure 1.5 times the design pressure, but in no case less than 75 psig. The test pressure is held for a minimum of 30 minutes, with no external leakage indicated. Pneumatic testing may be substituted for hydrostatic testing, in which case the test pressure is not less than the design pressure, nor should it exceed 110 percent of the design pressure.

(r) The process components of the radwaste system (liquid and solid) are under a program of quality control that is applied at the inspection, testing, construction, and operating levels. This is distinguished from the full quality assurance program (unless the equipment otherwise falls within the Q classification) in that full materials certification, etc., are not required or necessary for these systems.

(s) The design of the Radwaste Building satisfies the requirements of Branch Technical Position ETSB 11-1, Rev. 1, Nov. 24, 1975, as discussed in subsections 11.3 and 3.8.4.

(t) Vendor was unable to provide booster pumps to ASME Section III, Class 3. The pumps have been furnished with steel cases, material certification, certified test reports, witnessed hydrostatic, surface inspection and seismic calculations. A quality assurance program conforming to the requirements of 10 CFR 50, Appendix B has been applied to these pumps.

(u) A certification was obtained from the vendors of the turbine stop valves and turbine bypass valves that all cast pressure-retaining parts of a size and configuration for which volumetric examination methods are effective have been examined by radiographic methods by qualified personnel. Ultrasonic examination to equivalent standards was permitted as an alternate to radiographic methods.

TABLE 3.2-1: CLASSIFICATION OF SYSTEMS, COMPONENTS AND STRUCTURES (CONTINUED)

Examination procedures and acceptance standards were at least equivalent to those specified as supplementary types of examination in ANSI Std B31.1 Code, Paragraph 136.4.3.

All inspection records shall be maintained for the life of the plant. These records include data pertaining to qualification of inspection personnel, examination procedures, and examination results.

(v) In addition to a lift check valve inside the drywell and a positive-acting check valve outside the containment similar to an Atwood-Morrill boiler feed check valve as described in Catalog 63, Section I, a third valve with high leak-tight integrity will be provided in each line outside containment. The spring-loaded piston operator of the positive acting check valve will be held open by air pressure during normal operation. Fail-open solenoid valves will be used to release air pressure to permit the check-valve piston operator to close.

The classification of the feedwater lines from the reactor vessel to and including the second check valve is Safety Class 1; beyond the second check valve, the lines are Safety Class 2; beyond the remote manually operated isolation valve, the lines are Safety Class Other.

(w) The main steam leads from the turbine control valve to the turbine casing meet all of the requirements of Safety Class Other plus the following additional requirements:

1.All longitudinal and circumferential butt weld joints were radiographed (or ultrasonically tested to equivalent standards). Where size or configuration does not permit effective volumetric examination, magnetic particle, or liquid penetrant examination may be substituted. Examination procedures and acceptance standards were at least equivalent to those specified as supplementary types of examinations, Paragraph 136.4.3 in ANSI B31.1 Code.

2.All fillet and socket welds were examined by either magnetic particle or liquid penetrant method. All structural attachment welds to pressure-retaining materials were examined by either magnetic particle or liquid penetrant method. Examination procedures and acceptance standards were at least equivalent to those specified as supplementary types of examinations, Paragraph 136.4.3 in ANSI B31.1 Code.

3.All inspection records shall be maintained for the life of the plant. These records include data pertaining to qualification of inspection personnel, examination procedures, and examination results.

TABLE 3.2-1: CLASSIFICATION OF SYSTEMS, COMPONENTS AND STRUCTURES (CONTINUED)

A certification was obtained from the manufacturer of the main steam leads that the quality control program defined above has been accomplished.

(x) The condensate storage tank is designed, fabricated, and tested to meet the intent of API Std. 650. In addition, the specifications for this tank required (1) 100 percent surface examination of the side wall to bottom joint, and (2) 100 percent volumetric examination of the side wall weld joints.

(y) The auxiliary building is part of the boundary for the standby gas treatment system (SGTS). With the exception of the instrument air, fire protection, plant chilled water, and plant service water systems, isolation is provided on all lines larger than 2 inches not performing a safety function or supplying a source of makeup to the RPV such that auxiliary building integrity is maintained post accident. Lines 2 inches and smaller are small enough so that, if failure of the line would occur, the failure will not jeopardize SGTS operation. Postulated failures associated with the instrument air, fire protection, plant chilled water, and plant service water systems have also been shown to limit secondary containment in-leakage such that the function of the auxiliary building boundary is not compromised.

(z) The following insulation is installed on piping located in the drywell.

1.Main steam lines, feedwater lines, and reactor recirculation system - metal reflective insulation

2.Piping whose fluid temperature exceeds the maximum drywell ambient temperature (135 F) - metallic jacketed calcium silicate insulation

3.Piping whose fluid temperature in the range of 32-65 F - metallic jacketed fiberglass (anti-sweat) insulation

(aa) Liquid penetrant examination of welds in piping 2-in. diameter and larger

(bb) Deleted

(cc) In accordance with Entergy Operations' commitment to augment the Quality Group D design requirements, the assembly of the skid-mounted piping of these diesel generator auxiliary systems was performed by qualified welders under the provisions of ASME Section IX.

(dd) Only equipment associated with a safety action, eg., isolation, need conform to a safety function.

TABLE 3.2-1: CLASSIFICATION OF SYSTEMS, COMPONENTS AND STRUCTURES (CONTINUED)

(ee) Subsection 15.7.3 conservatively analyzes a postulated simultaneous failure of all the radwaste tanks in the radwaste building. The analysis assumes that 1 percent of the iodine is released to the atmosphere and, at the time of failure, all tanks are filled to capacity (this condition is not expected). The analysis evaluates the possible control room, site boundary, and LPZ exposures indicate, in light of the requirements of Regulatory Guides 1.26 and 1.29, that the radwaste system is properly classified and changes are not required.

Quality Group D classification of the offgas charcoal delay system is based upon Paragraph C.2.d of Regulatory Guide 1.26. The supporting justification is given in General Electric Topical Report, "A General Justification for Classification of Effluent Treatment System Equipment as Group D," NEDO 10734, February 1973, by C. W. Miller, et al. Offsite doses from component failure given in Table 3 of this report show that doses are far below 0.17 rem whole body (or its equivalent to parts of the body) at any representative site boundary. Results of additional calculations utilizing very conservative release rates are provided in Appendix 11A. Based on these calculations, the offgas system continues to meet the latest Regulatory requirements for classification of Group D.

The seismic classification of the offgas charcoal delay system is based upon Paragraph C.1.i, Regulatory Guide 1.29, which requires that such systems be Seismic Category I unless simultaneous failure would not result in conservatively calculated potential offsite exposure comparable to the guideline exposures of 10 CFR 100.

The GE Topical Report NEDO 10734, listed above, is based upon seismic failure to calculate release rates for the offgas charcoal delay system. Offsite doses from component failure given in Table 3 of this report show that doses are well below the 10 CFR 100 limits.

A summary of the offgas system design follows:

Item	Classification
Quality Group	
Piping and Valves	Quality Group D
Equipment Not Exposed to Process Gas	Quality Group D
Equipment Exposed to Process	Quality Group D plus Section VIII Material plus 100% radiography of pressure butt welds, and magnetic particle or liquid penetrant examination of other pressure boundary welds. Hydrotest and leak test. Welded nozzles except where maintenance access is required.
Seismic Classification	

TABLE 3.2-1: CLASSIFICATION OF SYSTEMS, COMPONENTS AND STRUCTURES (CONTINUED)

Piping, Valves, and Equipment	Non seismic
Building Upstream of Charcoal Tanks	Non seismic
Building Containing Charcoal Tanks	Non seismic
Safety Classification	Non essential

Justification:

Quality Group Classification.

Any postulated failure will not result in site boundary dose greater than 500 mRem whole body dose. To assure conservative design, additional quality control will be performed on the equipment handling and process gas.

Seismic Classification.

Based on conservative calculations, any postulated failure upstream of the charcoal adsorbers is below the guide level of Regulatory Guide 1.29. Failure of the charcoal adsorbers results in only marginal offsite doses when the NRC Staff's conservative criteria are considered.

Essential Component Classification.

Based on a reasonable engineering analysis, the offsite doses from equipment failure or uncontrolled release for considerably longer than 30 minutes, results in releases below 500 mRem whole body dose to one offsite person.

(ff) This is a safety-related structure, system, or component inside containment required for safe shutdown.

(gg) Owner's Certificate of Authorization (ASME III-NA3230)is not applicable in the state of Mississippi.

(hh) Piping has been fabricated, inspected, and installed in accordance with the ASME Code, Section III, Class 3 and has been designed in accordance with ANSI B31.1. “N” stamping, seismic Category I analysis, Q-listing, and third party inspection (Code Inspector) are not required. Hangers have been designed, fabricated, and installed per ANSI B31.1.

(ii) Quality Group Classifications (QGC) do not change at “By Bechtel-By

Seller" interfaces; QGC changes only at "Spec. Change" interfaces.

TABLE 3.2-1: CLASSIFICATION OF SYSTEMS, COMPONENTS AND STRUCTURES (CONTINUED)

(jj) Suppression pool strainers have been designed, fabricated, and inspected in accordance with ASME Code, Section III, Class 2 to the maximum extent possible. "N" stamping is not required.

(kk) Welding electrodes used on safety-related components are subject to appropriate quality assurance requirements.

(ll) Valve actuators for active safety-related valves are subject to appropriate quality assurance requirements.

(mm) The safety-related instrumentation described in Sections 7.2 through 7.6 are subject to pertinent requirements of the quality assurance program.

(nn) The following items are nonsafety-related. The appropriate portions of the operational QA program will apply to this list.

1. Masonry walls
2. Deleted
3. Spent fuel pool liner
4. Area radiation monitoring system
5. Normal operation and post-accident effluent airborne radioactivity monitoring systems
6. Normal operation and post-accident sampling system
7. Containment purge piping, valves, and filtration systems except for the portion associated with containment penetration
8. Plant-safety-parameter-display console
9. Plant shielding
10. Deleted
11. Fuel oil drip return line piping and valves
12. Fuel oil drip return pump
13. Fuel oil drip return tank
14. Combustion air intake and exhaust subsystem piping and valves
15. Combustion air intake and exhaust subsystem silencers and filters
16. Internal structures such as feedwater spargers, steam dryers, shroud head and steam separator assembly, incore guide tubes and stabilizers, and surveillance sample holders.

(oo) Justification for classifying these subsystems as Quality Group D are the guidelines specified in the NRC letter to MP&L dated July 11, 1975 and the codes and standards identified in Amendment 22 to the PSAR.

TABLE 3.2-1: CLASSIFICATION OF SYSTEMS, COMPONENTS AND STRUCTURES (CONTINUED)

(pp) Floor drain piping routed through and servicing the railroad bay area in the auxiliary building (refer to Figure 9.3-16) is installed as non-ASME Section III and non-Q-Listed piping; however, the piping has been furnished with material certifications, and all exposed welds have been surface inspected. The piping has also been analyzed and supported in accordance with seismic Category I requirements.

(qq) Tanks, 0 to 15 psig, may be designed to ASME Section VIII, without the Code Stamp. Pumps may also be designed to manufacturers' standards.

(rr) The classification is applicable for piping and valves beyond the engine interface connections to and including the solenoid valves. This piping and valves were originally supplied with all of the piping and valves between the engine interface connection and the engine and is termed as engine mounted piping and components. They include items such as headers and manifolds. The portion of piping between the solenoid valves and engine has been replaced. See Table XLI, 5, a, 3a. The design and fabrication of the remaining engine mounted piping and components is governed by the guidelines of Diesel Engine Manufacturers Associated (DEMA) standards and the manufacturer's own standard procedures.

A summary of their design is as follows:

1. All welding, including welder qualification, was in accordance with ASME Section IX;

2. The engine mounted components were conservatively designed in consideration of service pressure, temperature, flow and stress levels;

3. The jacket cooling water manifold was hydrostatically tested at 1.5 times the design pressure and the remaining components were tested at operating pressures and temperatures;

4. The diesel, including engine mounted piping and components, was designed and seismically tested for frequencies greater than 33 Hz. Additionally, the engine mounted starting air system piping was stress analyzed in accordance with ASME Code Section III, Class 3.

(ss) Starting air storage tank drain pipe and valve is per B31.1.

(tt) The gaseous radwaste system charcoal adsorber tanks have been designed to static seismic coefficients of 0.2g horizontal and 0g vertical. The stress levels in the charcoal adsorber support elements are less than 1.33 times the allowable stress levels permitted by the AISC Manual of Steel Construction, 7th Edition.

TABLE 3.2-1: CLASSIFICATION OF SYSTEMS, COMPONENTS AND STRUCTURES (CONTINUED)

(uu) Penetrations given in Table 6.2-44 which are not included in Table 3.2-1 are safety-related and controlled by the QA program.

(vv) The deep-bed charcoal filters of the Standby Gas Treatment System are safety-related and controlled by the QA program.

(ww) The Unit 1 spent fuel storage racks are fabricated in accordance with the materials, fabrication, installation and examination criteria of the 1980 ASME Code, including the 1981 Winter Addendum, Section III, Subsections NC and NF except: (1) where the weld stresses are less than 20% of yield, the vendor visual inspection criteria will apply in lieu of subsection NF visual inspection criteria, and (2) the spent fuel storage modules are not certified or stamped as Section III components.

(xx) The RWCU suction piping from valve Q1G33-F252 to and including valve Q1G33-F253 (see UFSAR Figure 5.4-21) was originally installed as ASME Section III, Class 2 piping. GGNS has since shown the equivalence of this piping to Class 1 requirements (ref: AECM-87/0231 and MAEC-87/0330). The piping from valve Q1G33-F252 to and including Q1G33-F253 has been added to the Class 1 portion of the ASME Section XI ISI program. An ASME Section III, Class 1 stress analysis has been performed for this section of 6"-DBB-140 piping.

(yy) Equipment originally supplied by GE for Unit 2 FPCCS, converted by Entergy Operations to present use.

(zz) For boundaries of ADHRS piping within the RHR system refer to Figure 5.4-17a.

(zz1) Seismic category and Q-list refer to pressure boundary components only.

(zz2) Based upon the design analysis, review of material specifications and records of fabrication, inspection and test, the engine skid mounted piping and valves in the lube oil, jacket water, starting air and fuel oil subsystems meet the intent of ANSI Standard B31.1.

(zz3) The design and fabrication of the engine mounted piping and components are governed by the guidelines of Diesel Engine Manufacturer's Association (DEMA) Standards and the Manufacturer's own standard procedures. A summary of their design is as follows:

1. All welding, including welder qualification, was done using the provisions of ASME Section IX as a guide.

2. The engine mounted components were conservatively designed in consideration of service pressure, temperature, flow and stress levels.

TABLE 3.2-1: CLASSIFICATION OF SYSTEMS, COMPONENTS AND STRUCTURES (CONTINUED)

3. The jacket cooling water manifolds for Grand Gulf Unit 1 were hydrostatically leak tested to 87 to 100 psig. Initial service leak testing at operating pressure and temperature is also performed for engine auxiliary piping.

4. The diesel including engine mounted piping and components was designed, and analyzed or tested, for seismic acceleration levels appropriate for the natural frequency of the component. Components having natural frequencies greater than 33 Hz were considered to be rigid and were qualified by static analysis.

(zz4) Per UFSAR Table 3.2-2 pumps operating below 150 psig and 212 F may be the engine manufacturer's standard pump for the intended service.

(zz5) The lube oil cooler was designed and fabricated by the Electro Motive Division of General Motors using their standard design procedures instead of TEMA C as specified in UFSAR Table 3.2-2. The EMD design requirements have been compared to those specified in TEMA C and found to be acceptable for this application.

(zz6) The engine skid mounted piping and valves in the lube oil, jacket water, starting air and fuel oil subsystems have been evaluated against the Quality Group D Augmented Requirements, as specified in MAEC 75/36, and it has been concluded that the design of the subject systems are in compliance with the intent of the Quality Group D Augmented requirements.

(zz7) The Unit 1 Control Rod Storage Rack is used to temporarily and/or permanently store control rods, control rod guide tubes, defective fuel storage canisters, and fuel sipping containers. ASME Section III, subsection NF, 1980 Edition including Winter 1981 Addendum, is used for guidance in the design, fabrication, inspection, and examination of this rack. This rack however, is not certified or stamped as an ASME Section III component.

(zz8) The starting air manifold assembly on the Division I/II Diesel Generator has been designed, fabricated and installed according to the guidelines of ASME BPVC Section III.

(zz9) Feedwater piping located in the turbine building from penetrations AP-19C and AP-20C to FW heaters 5A & 5B, heaters 6A & 6B to R.F.P pump C001A and C001B and to HP condenser shell N19B007A nozzles (Reference Isometric Dwg.: M-1323A & M-1328K) is seismically evaluated based on UBC (Uniform Building Code) seismic loading.

TABLE 3.2-1: CLASSIFICATION OF SYSTEMS, COMPONENTS AND STRUCTURES (CONTINUED)

(zz10) The Spent Resin Tank (G17A007) was exposed to an overpressure condition which resulted in this tank exceeding its maximum allowable design pressure and stresses. The tank was subsequently examined, evaluated, and tested to verify it is adequate for its intended Radwaste System function. Although this tank was originally designed, constructed and tested in accordance with ASME Code Section VIII, due to the overpressure event, the tank no longer meets requirements of ASME Code Section VIII, API-620, API-650, AWWA-D100, ANSI B96.1, or Branch Technical Position ETSB 11-1 (Revision 1).

(zz11) Humidifier has been abandoned in place.

(zz12) Inboard Blower and Heaters are not functional, but are required to remain in place for seismic considerations.

(zz13) The Jacket Water supply piping to the exhaust manifold water jacket and jacket water piping supply to the turbocharger have been designed and installed to ASME Section III, Class 3 Code. Testing was performed at operating pressure and temperature.

(zz14) The Jacket Water supply piping to the Cylinder Head Jacket Water has been designed and installed to ASME Section III, Class 3 Code. Testing was performed at operating pressure and temperature.

(zz15) For the Jacket Water supply piping to the exhaust manifold water jacket, turbocharger and cylinder head, the new piping and fittings will be purchased Safety Related to either the ASME Boiler and Pressure Vessel Code, Section III, Class 3 or the ANSI Piping Code, B31.1, meeting 10 CFR 50 Appendix B requirements. The flexible hoses and the flexible hose fittings were purchased safety related meeting the requirements of 10CFR50, Appendix B, the flex hoses and flex fittings meet the Society of Automotive Engineers standards SAE-J516 and SAE-J517, respectively, the hose fittings meet SAE-J514 and the testing of these parts meets the standards of SAE-J343.

(zz16) Information provided in this table is for the engine driven and motor driven jacket water pumps. Information for the electric jacket water heater circulation pump is the same with the following exceptions: Seismic Category is ’NA’ and Q-List is ’N’.

TABLE 3.2-1: CLASSIFICATION OF SYSTEMS, COMPONENTS AND STRUCTURES (CONTINUED)

(zz17) The SSW liquid radiation monitor skids are required to maintain their pressure retaining capabilities before, during and after an SSE in order to maintain the required 30 day SSW basin inventory. The skids were supplied non-safety related and non seismic. They were upgraded to safety related, seismic category I status by a limited scope in situ commercial grade dedication and a detailed seismic/pressure qualification analysis. The following codes and standards were used in this analysis: B31.1 (1967), ASME BPVC Section VIII Div 1 (1974), API-610 (8^{th} ed. 1995), AISC-SCM (7^{th} ed. 1969), ACI-318 (1971), EJMA Standard (6^{th} ed. 1983). The skids are considered to be non ASME code items.

(zz18) The systems for monitoring the RCPB in the drywell such as the Drywell Air Cooler Condensate Drain Line Instrumentation, etc. are not part of the plant protection or safety system. The air particulate and gaseous radiation monitors must remain functional through the Operating Basis Earthquake (ORE) with the Air Particulate Radiation monitor remaining functional through the Safe Shutdown Earthquake (SSE). The portion f the system that monitors the RCPB leakage beyond the second isolation valve and initiates an isolation function to maintain the integrity of the RCPB and containment boundary while limiting the loss of process fluid shall be Class 1E. All remaining monitoring, recording, and alarming of RCPB leakage are non-Class IE.

TABLE 3.2-2: CODE CLASSIFICATION GROUPS - INDUSTRY CODES AND STANDARDS FOR MECHANICAL COMPONENTS (A, B)

ASME Section III Code Applicable Subsections

Quality Group Classification	ASME Section III Code Classes	Pressure Vessels and Heat Exchangers	Pumps Valves, and Piping	Metal Containment Components	Storage Tanks (0-15 psig)	Storage Tanks Atmosphere
A	1	NA & NB TEMA C	NA & NB Note (c)	---	---	---
B	2 MC	NA & NC TEMA C	NA & NC Note (c)	NA & NE	NA & NC	NA & NC
C	3	NA & ND TEMA C	NA & NC Note (c)	---	NA & ND	NA & ND
D		ASME Sect. VIII Div.1	Piping and Valves B31.1	---	API -620 or equivalent	API-650 AWWA-D100 ANSI B96.1 or equivalent (e)
		TEMA C	Pumps Note (d)			

Notes:

(a) With options and additions as necessary for service conditions and environmental requirements.

(b) Components of the RCPB shall meet 10 CFR Part 50, Section 50.55a, Codes and Standards, and its amendments of the NRC regulations to meet the codes and addenda in effect based on issuance of the construction permit.

(c) For pumps classified A, B or C, applicable subsections NB, NC or ND, respectively, in ASME Section III Boiler and Pressure Vessel Code shall be used as a guide in calculating the thickness of pressure-retaining portions of the pump and in sizing cover bolting.

(d) For pumps classified in Group D and operating above 150 psi or 212 F, ASME Section VIII Div. 1 shall be used as a guide in calculating the wall thickness for pressure-retaining parts and in sizing the cover bolting. For pumps operating below 150 psi and 212 F manufacturer's standard pump for service intended may be used.

(e) The Spent Resin Tank (G17A007) was exposed to an overpressure condition which resulted in this tank exceeding its maximum allowable design pressure and stresses. The tank was subsequently examined, evaluated, and tested to verify it is adequate for its intended Radwaste System function. Although this tank was originally designed, constructed and tested in accordance with ASME Code Section VIII, due to the overpressure event, the tank no longer meets requirements of ASME Code Section VIII, API-620, API-650, AWWA-D100, ANSI B96.1, or Branch Technical Position ETSB 11-1 (Revision 1).

TABLE 3.2-3: SUMMARY OF SAFETY CLASS DESIGN REQUIREMENTS (MINIMUM)

Design Requirements	Safety Class			
	1[(d)]	2	3	Other
Quality Group Classification (a)(e)	A	B	C	C,D,N/A
Quality Assurance Requirement (b)	B	B	B	N/A
Seismic Category (c)(e)	I	I	I	N/A

Notes:

(a) The equipment is constructed based on the codes or standards indicated in Table 3.2-1. The requirements for mechanical equipment are in accordance with their Quality Group Classification as shown in Table 3.2-2.

(b) B - The equipment has been constructed in accordance with the quality assurance requirements of 10 CFR 50, Appendix B, as delineated in Chapter 17.

N/A - The equipment has been constructed in accordance with the quality assurance requirements consistent with accepted practice for steam power plants.

(c) I - The equipment of these safety classes has been constructed in accordance with the seismic requirements for the safe shutdown earthquake as described in Section 3.7. N/A - The seismic requirements for the safe shutdown earthquake are not applicable to the equipment of this classification.

(d) Application of code and addenda for the Reactor Coolant Pressure Boundary Class 1 Equipment is provided in Table 3.2-4.

(e) As indicated in Table 3.2-1, for an assigned Safety Class 1, 2, or 3, the Quality Group Classification or Seismic Category is not applicable in certain non-applicable or exceptional cases, e.g., reactor internal structures and general purpose grapple, respectively.(e)The Spent Resin Tank (G17A007) was exposed to an overpressure condition which resulted in this tank exceeding its maximum allowable design pressure and stresses. The tank was subsequently examined, evaluated, and tested to verify it is adequate for its intended Radwaste System function. Although this tank was originally designed, constructed and tested in accordance with ASME Code Section VIII, due to the overpressure event, the tank no longer meets requirements of ASME Code Section VIII, API-620, API-650, AWWA-D100, ANSI B96.1, or Branch Technical Position ETSB 11-1 (Revision 1).

TABLE 3.2-4: QUALITY GROUP A, B AND C COMPONENTS - APPLICABLE ASME CODE EDITION AND ADDENDA

Component Description	NSSS Components (4)(9) Master Parts List Number	Addenda
1. Recirculation system		
a. Gate valves	B33-F023	ASME Section III, 1974 Edition, with no addenda.
	B33-F067	ASME Section III, 1974 Edition, with no addenda.
b. Flow control valves	B33-F060	ASME Section III, 1971 Edition up to & including winter 1973 agenda
1) body		
2) topworks		Built under Section I of ASME Code to original Code requirements.
c. Pump	B33-COO1	ASME Section III, 1971 Edition up to & including summer 1973 addenda.
d. Piping	-	ASME Section III, 1974 Edition
2. Main steam system		
a. Main steam isolation valves	B21-F022 and B21-F028	ASME Section III, 1974 Edition, with no addenda.
b. Safety relief valves	B21-F041 B21-F047 B21-F051	ASME Section III, 1974 Edition, up to & including summer 1976 addenda

TABLE 3.2-4: QUALITY GROUP A, B AND C COMPONENTS - APPLICABLE ASME CODE EDITION AND ADDENDA (Continued)

c. Piping	-	ASME Section III, 1974 Edition, up to & including summer 1974 addenda.
3. High- pressure core spray system		
a. Valves	E22-F004	ASME Section III, 1971 Edition, up to & including winter 1973 addenda
4. Reactor Pressure Vessel	B13-D003	ASME Code, Section III 1971 Edition, up to & including winter 1972 addenda.

Non-NSSS Components (1)(2)(4)(9)

Piping-shop-fabricated 2 1/2 inches and larger	--	ASME Section III, 1971 Edition, through summer 1973 addenda.
Piping-field-fabricated and installed (all sizes)	--	ASME Section III, 1974 Edition, through summer 1974 addenda. (6)(S)
Piping-2 inches and smaller-Unit 1	--	ASME Section III, 1974 Edition, through summer 1974 addenda.
Flued heads	--	ASME Section III, 1971 Edition, through summer 1973 addenda.

Non-NSSS Components (1)(2)(4)(9)

Component Description	Master Parts List Number	Code and Addenda
Hangers and supports (excluding snubbers)	--	ASME Section III, 1974 Edition, no addenda.

TABLE 3.2-4: QUALITY GROUP A, B AND C COMPONENTS - APPLICABLE ASME CODE EDITION AND ADDENDA (Continued)

Snubbers	--	ASME Section III, 1974 Edition, through summer 1976 (3)
Fabrication of steel for pipe supports	--	ASME Section III, 1974 Edition, through summer 1974 addenda. (8)
Wye strainers	--	ASME Section III, 1974 Edition, through winter 1975 addenda.
Valves-2 1/2 inches and larger (globe and gate)	--	ASME Section III, 1971 Edition, through winter 1972 addenda.
Valves-2 1/2 inches and larger (power actuated check)	--	ASME Section III, 1974 Edition, through summer 1974 addenda and NB-2579(d) Summer 1975.
Valves-2 inches and smaller (all)	--	ASME Section III, 1974 Edition, through winter 1974 addenda. (Note 7)

NOTES:

1. The ASME Code Edition and Addenda in effect for field purchased components was the latest revision and addenda approved by the NRC per 10 CFR 50.55a at the time of purchase.

2. Piping design and stress analyses were performed in accordance with the ASME Code, Section III, 1974 Edition, through summer 1975 addenda, with the exception of Nuclear Class 2 and 3 flanged joints, for which pressure design and stress analyses were performed in accordance with the ASME Code, 1977 Edition, through Winter 1979 Addenda, paragraphs NC-3647, ND-3647, NC-3658, and ND-3658, as required by NC-3671.1 and ND-3671.1.

NOTES (Contd.)

3. Applies to Paragraph NF-1214.1, Fig. NF-1211.4-1; NF-3134.6 in lieu of NF-3274 of the 1974 Edition.

4. The edition and addenda of the codes listed in Table 3.2-4 are those used in the original construction of GGNS. Later edition and/or addenda that are adopted for use, as approved by 10CFR50.55a, are specified in the component's design specification as required by ASME Section III, NA-3250.

5. Owners Certificate of Authorization (ASME III-NA3230) is not applicable in the state of Mississippi.

6. Paragraphs NC-6221 and ND-6221 of the Winter 1975 addenda to ASME Section III shall be used for hydrostatic testing of piping systems.

7. The following materials may be used as an alternate to SA182 F6 for the backseat bushing of valves:

   i. SA182 F6a per the 1976 Summer Addenda of the Code

   ii. SA182 F6a Class II per the 1978 Winter Addenda of the Code

8. The following paragraphs of the 1974 ASME Code, Summer 1976 Addendum, shall be invoked:

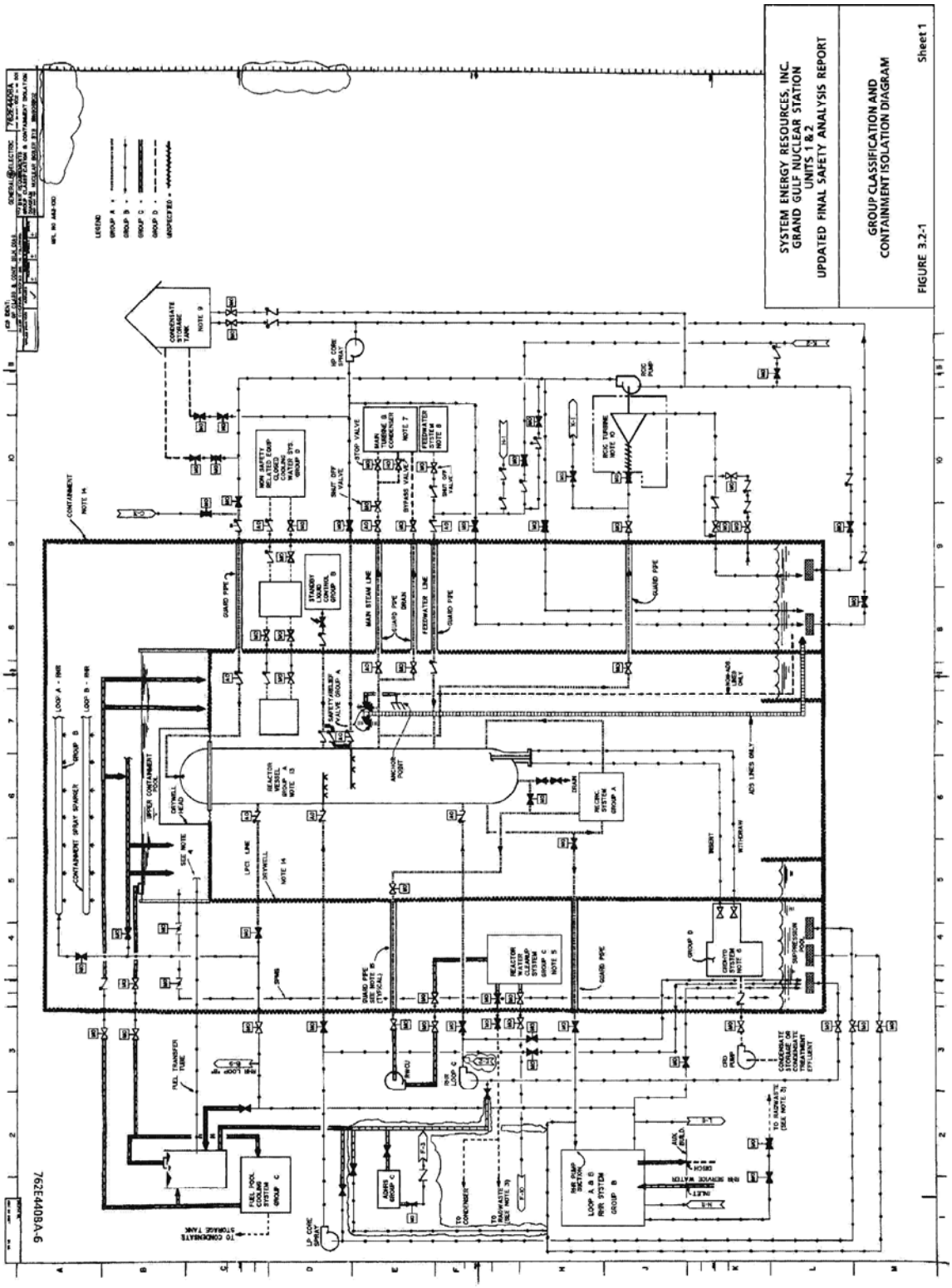
   i. NF-2610, for Class 2 and 3 pipe supports and all classes of component standard supports

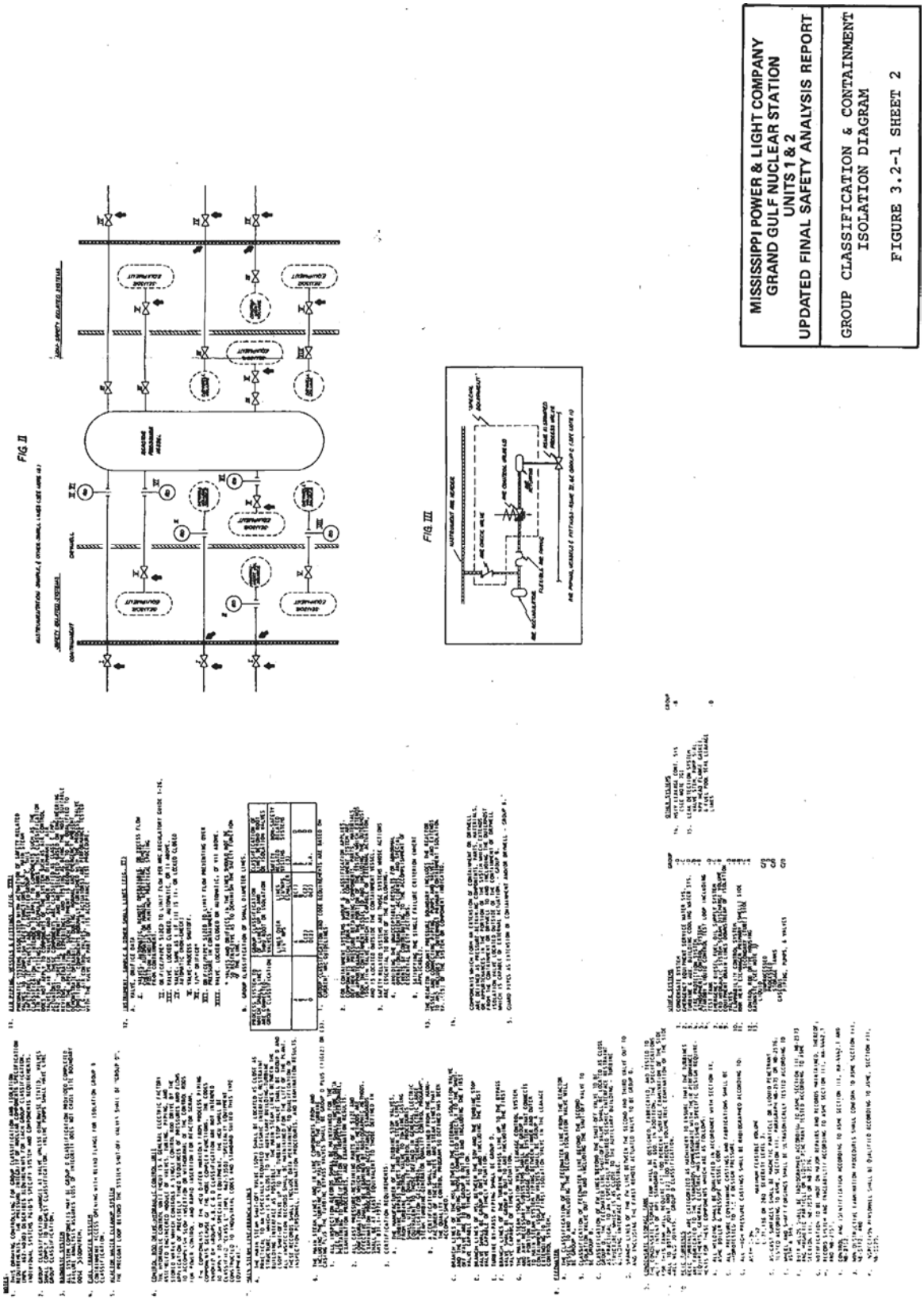
   ii. NC/ND-2610, for Class 2 and 3 welded attachments

9. ASME Section III, 1980 edition through Summer 1982 addenda has been adopted for the capacitive discharge method of welding (NX 4311.2 and NX 4311.3).

TABLE 3.2-5: ASME SECTION III COMPONENTS OR COMPONENT PARTS OBTAINED TO THE GUIDANCE OF GENERIC LETTER 89-09

Q1B21-RO-D101A (Reference DCP 93/0011-01)
Q1B21-RO-D101B (Reference DCP 93/0011-01)
Q1B21-RO-D101C (Reference DCP 93/0011-01)
Q1B21-RO-D101D (Reference DCP 93/0011-01)


SYSTEM ENERGY RESOURCES, INC.
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT
GROUP CLASSIFICATION AND
CONTAINMENT ISOLATION DIAGRAM
FIGURE 3.2-1
Sheet 1
LEGEND
GROUP A
GROUP B
GROUP C
GROUP D
MODIFIED
CONDENSATE STORAGE TANK NOTE 9
MAIN TURBINE & CONDENSER NOTE 7
FEEDWATER SYSTEM NOTE 8
RCIC TURBINE NOTE 10
RCIC PUMP
STOP VALVE
BYPASS VALVE
SHUT OFF VALVE
CONTAINMENT NOTE 14
GUARD PIPE
MAIN STEAM LINE
FEEDWATER LINE
STANDBY LIQUID CONTROL GROUP B
REACTOR VESSEL GROUP A NOTE 13
ANCHOR POST
RECIRC SYSTEM GROUP A
DRYWELL HEAD
SUPPRESSION POOL
CONTAINMENT SPRAY SPARGER
LOOP A - RHR
LOOP B - RHR
LPCI LINE
SKYWELL NOTE 14
REACTOR WATER CLEANUP SYSTEM GROUP C NOTE 5
CRD HYDRAULIC SYSTEM NOTE 6
CRD PUMP
FUEL TRANSFER TUBE
FUEL POOL COOLING SYSTEM GROUP C
RHR LOOP C
RHR PUMP SECTION LOOP A & B RHR SYSTEM GROUP B
ADHR GROUP C
TO CONDENSER
TO RADWASTE (SEE NOTE 3)
LP CORE SPRAY
TO CONDENSATE STORAGE TANK
RHR SERVICE WATER
DISCH
SUCT
CONDENSATE STORAGE OR RADWASTE TREATMENT EFFLUENT
TO RADWASTE (SEE NOTE 3)
7621-4108A-6



MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT

GROUP CLASSIFICATION & CONTAINMENT ISOLATION DIAGRAM

FIGURE 3.2-1 SHEET 2

3.3 WIND AND TORNADO LOADINGS

Seismic Category I structures are designed for the tornado and extreme wind phenomena which are defined in Section 2.1 of Bechtel Topical Report BC-TOP-3A (Ref. 1) and in ASCE Paper No. 3269, "Wind Forces on Structures" (Ref. 2). These reports furnish the criteria, data, formulae, and procedures for determining the maximum wind loadings on structures and parts of structures. Seismic Category I structures which are required for safe shutdown and structures which contain systems, equipment, and components required for safe shutdown after a tornado are designed for the tornado effects. These structures are identified in Table 3.3-1. The enclosure building and non-safety-related buildings adjacent to safety-related buildings will not undergo gross structural failure during a design basis tornado. Metal siding and other lightweight materials may blow off these buildings during the tornado.

3.3.1 Wind Loadings

3.3.1.1 Design Wind Velocity

The design wind velocity for all seismic Category I structures is 90 mph at 30 ft above ground for a 100-yr recurrence interval.

3.3.1.1.1 Basis for Wind Velocity Selection

The design wind velocity is based on Figure 5 of "New Distributions of Extreme Winds in the United States," by H.C.S. Thom (Ref. 3) (which is essentially in agreement with Figure 2 of ANSI Std. A 58.1 (Ref. 4) and is equal to the wind velocity shown for the Grand Gulf site locality. Interpretations of the U. S. Weather Bureau climatological data constitute the basis for the above figures.

3.3.1.1.2 Vertical Velocity Distribution and Gust Factors

The vertical velocity distribution used is in accordance with Table 1(b) of Reference 2. These criteria are compatible with the distribution found in exposure C (flat, open country; flat, open coastal belts; and grassland) of Section 6 of Reference 4. Appropriate gust factors must be determined and applied to the criteria found in Reference 2.

3.3.1.2 Determination of Applied Forces

The procedures used to convert the wind velocity into applied forces for structures are contained in Reference 2. For seismic Category I structures which are designed for tornado loading, the applied forces due to wind are calculated to determine if they are less severe than the applied forces due to tornado loadings. The applied tornado force magnitude and distribution are determined as discussed in subsection 3.3.2.2.

Appropriate stress levels and load factors discussed in Section 3.8 are to be considered in the determination of the governing loads.

The wind velocity on which the applied forces depend is given in subsection 3.3.1.1. The design pressures or design loads are obtained from equation 7 of Reference 2; drag coefficients for the appropriate building shapes are obtained from Table 2, Table 4, and Figure 6 of Reference 2. Since no Category I structures are slender and flexible, the above wind loading will be applied as a static load without need for the vortexshedding analysis required for flexible structures.

3.3.2 Tornado Loadings

For purposes of structural analysis, the effects of a tornado are described in Section 3.0 of Reference 1.

3.3.2.1 Applicable Design Parameters

Tornado resistant seismic Category I structures are analyzed for tornado loadings as outlined in Sections 3.3 and 3.4 of Reference 1. The loadings are calculated on the basis of a tornado with a maximum wind velocity of 360 mph which is the vector sum of a maximum peripheral rotational velocity of 290 mph and a translational velocity of 70 mph. The maximum design pressure drop is 3 psi with a maximum rate of change of 2 psi/sec. The radius from the center of the tornado at which the maximum wind velocity occurs is 150 ft. These parameters conform to those given in Regulatory Guide 1.76 for Region I. Tornado wind pressure loads, time intervals, and missile effects are combined in accordance with Reference 1. The design basis tornado missiles are discussed in subsection 3.5.1.4.

3.3.2.2 Determination of Forces on Structures

The methods employed to convert tornado loadings into forces and the distribution across the structures are outlined in Section 3.5 of Reference 1 and in Reference 2. The procedure for transforming the tornado-generated missile impact into an effective or equivalent static load on structures is delineated in subsection 3.5.3.1. Loading combinations and load factors used are discussed in Reference 1. The containment and control buildings are not vented structures. The exposed exterior walls of these structures are designed for the 3 psig pressure drop. The auxiliary building has blowout panels that are opened with 1.0 psig differential pressure during a tornado. A depressurization analysis is used in accordance with Reference 1 to determine the structural response of the building during the depressurization. The diesel generator building is designed as a partially vented building, but in lieu of a depressurization analysis, the interior walls are designed for the maximum pressure drop. The standby service water cooling towers and basin are completely vented buildings and no differential pressure is considered for the walls of these structures; however, for the case in which the basin is completely full, the basin roof is designed for the full differential pressure. The enclosure building is designed so that most of the siding becomes detached due to the high tornado winds, which completely vents the building; however, the main structural framing is designed to remain in place and to produce no missiles more critical than the ones listed in subsection 3.5.1.4. The diesel generator fuel oil tanks are underground and not subject to the pressure drop.

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

The design of all permanent structures, systems, and components not designed for tornado loadings is analytically checked to determine that during a tornado these structures, systems, and components will not provide missiles which may have more severe effects than those tornado-generated missiles listed in subsection 3.5.1.4. This is to assure that seismic Category I structures, systems, and components required for safe shutdown after a tornado will perform their intended functions.

All permanent structures, systems, and components not designed for tornado loadings are analyzed for their modes of failure to assure that their collapse will not affect the ability of the seismic Category I structures, systems, and components to perform their intended safe-shutdown functions.

3.3.3 References

1. Bechtel Topical Report BC-TOP-3A, Revision 3, August 1974, “Tornado and Extreme Wind Design Criteria for Nuclear Power Plants.”

2. ASCE Paper No. 3269, “Wind Forces on Structures,” Transactions of the American Society of Civil Engineers, Volume 126, Part II (1961).

3. “New Distributions of Extreme Winds in the United States,” HCSS Thom, Journal of the Structural Division Proceedings of the American Society of Civil Engineers, July 1968.

4. ANSI Standard A 58.1-1972, “Building Code Requirements for Minimum Design Loads in Buildings and Other Structures,” Committee A 58.1, American National Standards Institute.

TABLE 3.3-1: TORNADO-PROTECTED COMPONENTS AND TORNADO-RESISTANT ENCLOSURES

Protected Component	Tornado-Resistant Enclosure
Reactor coolant system and pressure boundary appurtenances	Containment structure
Reactor protection system and engineered safety features inside containment	Containment structure
Control room and auxiliary shutdown panel	Control building
Residual heat removal system and RCIC system	Auxiliary building
Safety/relief valves	Containment building
Spent fuel pool and cooling system	Structural components of the auxiliary building are designed so that tornado winds will not cause the component to fall into the fuel pool
Emergency diesel generators, including: a)Fuel oil storage and transfer system (except outdoor storage tank and pump) b)Cooling water system c)Starting air system d)Comubstion air intake and exhaust subsystem e)Lubrication system	Diesel generator building
Diesel fuel oil storage tanks and transfer pumps	Category I structural backfill
Main steam line isolation valves	Tornado- resistant enclosure (tunnel) out to the isolation valves; and containment structure
Standby service water pumps and piping	Standby service water pumphouse and cooling towers and basin

TABLE 3.3-1: TORNADO-PROTECTED COMPONENTS AND TORNADO-RESISTANT ENCLOSURES (Continued)

Protected Component	Tornado-Resistant Enclosure
External Class IE electrical cables	Seismic Category I manholes and ductbanks
Ultimate heat sink cooling fans	Fans designed for tornado wind loads, debris protection provided on fan outlets.

3.4 WATER LEVEL (FLOOD) DESIGN

Criteria for the design basis flood conform to the requirements of Regulatory Guide 1.59, "Design Basis Floods for Nuclear Power Plants," and Regulatory Guide 1.102, "Flood Protection for Nuclear Power Plants," as discussed in Appendix 3A of this FSAR. The probable maximum flood level is shown in Table 3.4-1.

3.4.1 Flood Protection

This section discusses the flood protection measures provided for Category I structures, systems, and components.

3.4.1.1 Safety-Related Systems

The safety-related systems and components for which flood protection is provided are the same as those identified in Paragraph C.1 of Regulatory Guide 1.29, with clarifications and exceptions as specified in Appendix 3A of this SAR. Floor protection of safety-related systems and components is provided for all postulated flood levels and conditions described in Section 2.4.

3.4.1.2 Description of Structures

Structures which house the safety-related equipment and which offer flood protection to this equipment are identified in Figure 3.4-1.

A description of these structures is provided in subsections 3.8.1, 3.8.4 and 3.8.5. Exterior or access openings and penetrations that are below the postulated design flood level are identified in Table 3.4-1.

The design groundwater level is tabulated for all seismic Category I structures in Table 3.4-1.

3.4.1.3 Flood Protection from External Sources

Category I structures that may be affected by design basis floods are designed to withstand the floods postulated in Section 2.4 using the "hardened" flood protection approach. These structures are identified in Figure 3.4-1. The hardened protection approach means structural provisions incorporated in the plant's design that will protect safety-related structures, systems, and components from the static and dynamic effects of a flood.

Safety-related systems and components are flood protected either because of their location above the postulated maximum flood level or because they are enclosed in reinforced concrete Category I structures which have

a. Wall thicknesses below flood level of not less than two feet

b. Waterstops provided in all construction joints below flood level

c. Watertight doors and equipment hatches installed below maximum flood level

d. Waterproof coating

e. Watertight piping and electrical penetrations below maximum flood level.

Waterproofing of foundations and walls of Category I structures below grade is accomplished principally by the use of waterstops at expansion and construction joints. The waterstop is natural or synthetic rubber. In addition to the waterstops, waterproofing is installed to protect exterior surfaces in areas of prolonged exposure to water. Typical details for waterstops and waterproofing installation are shown in Figure 3.8-86.

Additional specific provisions for flood protection include administrative procedures to assure that all watertight doors will be closed in the event of a flood warning. If local seepage occurs through the walls, it will be controlled by sumps and sump pumps.

In the occurrence of a flood, the flood levels would take a relatively long time to develop. This would allow ample lead time to perform necessary emergency actions for all accesses which need to be protected.

Piping penetration closures provide watertight seals for all pipe penetrations below the maximum flood level. The containment electrical penetrations are of two types: canister and modular. Both types are airtight by design (and thus watertight) and are monitored to insure against leaks. In areas other than the containment, electrical penetrations are closed with fire seals appropriate to the rating of the wall through which they pass.

These seals are qualified in accordance with the hose stream requirements of ASTM E 119-73 and have thus demonstrated their watertightness.

The safety-related components located below the design flood level inside a seismic Category I structure are identified in Table 3.4-2 with a summary of the protection provided. These components are shown in Figure 1.2-2., All safety-related components located below the design flood level are protected using the hardened protection approach.

The leaktight doors and wall penetrations in the areas listed in Table 3.4-2 as being below the maximum flood level will be inspected or tested periodically to demonstrate their leaktightness. The leaktight electrical and piping penetrations will be visually inspected to verify that they are installed in accordance with design requirements. Inspections are conducted in accordance with Fire Protection Program requirements that verify leak tight qualities are acceptable through the absence of cracks, holes, or otherwise degraded sealant.The leaktightness of the watertight doors will be verified by evaluating the door knife-edge to door seal seating efficiency. This will be done using powdered chalk or other suitable method. If potential leakage paths between the door and door seal are discovered, the door will be adjusted or reworked to eliminate leakage paths.

There are no safety-related systems or components which are capable of normal function while completely or partially flooded.

3.4.1.4 Flood Protection from Component Failures

Each postulated component failure for each watertight room is considered separately as a single postulated event occurring during normal plant operation. Each plant system identified in Tables 3.6A-16 and 3.6A-17 is reviewed to determine the failure which results in the most adverse flooding conditions. The criteria used in the evaluation of the most adverse failure of a fluid system piping run are discussed in Appendix 3C of this FSAR.

3.4.2 Analysis Procedure

The lateral hydrostatic pressure on the structures due to the probable maximum flood water level, as well as ground water and soil pressures, are determined as shown in Figure 2.5-91.

Structures, systems, and components are designed and analyzed for the maximum hydrostatic and hydrodynamic forces in accordance with the loads and load combinations indicated in subsections 3.8.1.3 and 3.8.6.2 using well established methods based on the general principles of engineering mechanics.

The minimum factor of safety against flotation for each seismic Category I building is listed in Table 3.8-1.

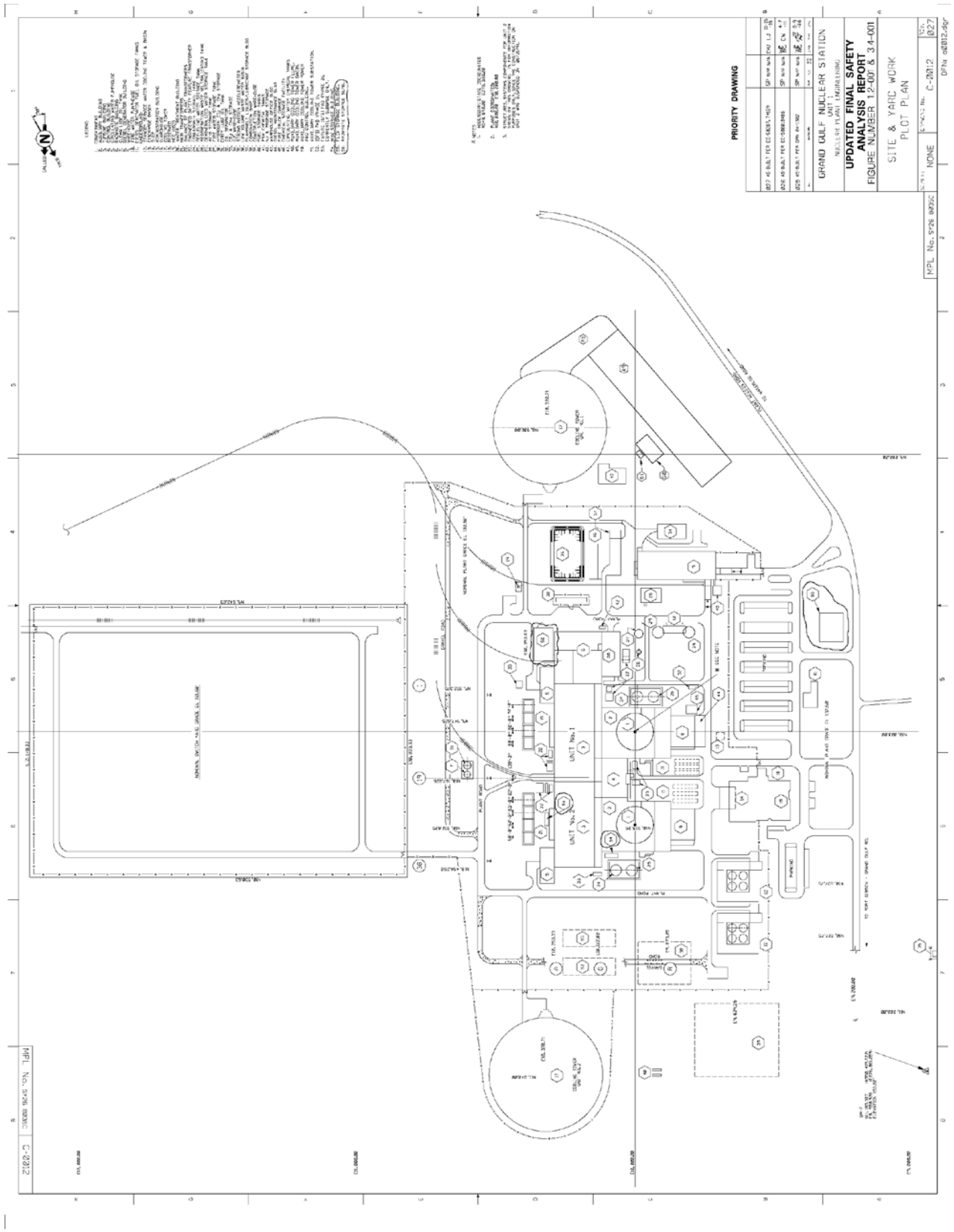
TABLE 3.4-1: STRUCTURES, PENETRATIONS, AND ACCESS OPENINGS DESIGNED FOR FLOOD PROTECTION

Structure	Containment Structure	Control Building	Standby Service Water Cooling Tower Basin	Auxiliary Building	Diesel Gen. Building	Condensate and Refueling Water Storage Tanks Basin
Probable maximum flood level ft-msl	103-0	103-0	103-0	103-0	103-0	103-0
Design water level ft-msl	114-6	114-6	114-6	114-6	114-6	114-6
Reference plant grade (ft)	132-6	132-6	132-6	132-6	132-6	132-6
Base slab ft-msl	93-0	93-0	82-6	93-0	133-0	117-6
Actual plant grade ft-msl	132-6	132-6	132-6	132-6	132-6	132-6
Access openings and penetrations below postulated design flood level	Suction and drain pipe penetrations at 96-0 abd 99-0 respectively	None	Piping penetrations at 90-6, 95-6 and 100-6	Pipe penetrations at approximately 95-0	None	None

TABLE 3.4-2: SEISMIC CATEGORY I COMPONENTS PROTECTED FROM ENVIRONMENTAL FLOODS

Component	Location	Elevation (ft-in.)	Max Flood Level (ft-in.)	Protection
1. RHR heat exchanger and pump (A)	Auxiliary building	93-0	103-0	Room isolation via water-tight doors and walls
2. RHR heat exchanger and pump (B)	Auxiliary building	93-0	103-0	Room isolation via water-tight doors and walls
3. RHR pump (C)	Auxiliary building	93-0	103-0	Room isolation via water-tight doors and walls
4. RCIC room cooler, pump, and turbine	Auxiliary building	93-0	103-0	Room isolation via water-tight doors and walls
5. HPCS room cooler and pump	Auxiliary building	93-0	103-0	Room isolation via water-tight doors and walls
6. LPCS room cooler and pump	Auxiliary building	93-0	103-0	Room isolation via water-tight doors and walls

TABLE 3.4-3: DELETED


PRIORITY DRAWING
GRAND GULF NUCLEAR STATION
UNIT 1
NUCLEAR PLANT ENGINEERING
UPDATED FINAL SAFETY ANALYSIS REPORT
FIGURE NUMBER 12-001 & 3.4-001
SITE & YARD WORK
PLOT PLAN
NONE
C-2012
027
MPL No. SY26 8005C
UNIT No. 1
UNIT No. 2
NORMAL PLANT SERVICE EL. 133'-6"
PLANT ROAD
PARKING

3.5 MISSILE PROTECTION

3.5.1 Missile Selection and Description

The following criteria have been adopted to assess the plant's capability to assure that, in the event of a generated missile of any type postulated in this section, there is:

a. No loss of containment function

b. Deleted

c. No loss of function to systems required to shutdown the reactor and maintain it in a safe shutdown condition, or mitigate the consequences of the missile damage assuming:

   1. No equipment is allowed to be damaged in one safety-related division, e.g., Division 1, from internally generated missiles originating from another safety-related division, e.g., Division 2.

   2. Missiles generated from nonsafety-related equipment shall not damage any safe shutdown equipment.

   3. Offsite power is not assumed to be in operation during the shutdown of the plant.

   The systems required to be protected are:

   1. Reactor Coolant Pressure Boundary

   2. RHR System (A & B only)

   3. Standby Service Water System

   4. Ultimate Heat Sink

   5. Automatic Depressurization System Relief Valves

   6. Standby Diesel Generator System

   7. CRD Hydraulic (Scram Section)

   8. RCIC System

   9. Fuel Pool Cooling System

10. High Pressure Core Spray System and Associated Diesel Generator System

11. Auxiliary Shutdown Panel

12. Reactor Protection System

13. All Containment Isolation Valves

14. HVAC Systems Required During the Operation of the Above

15. Electrical Systems and Control Systems and Wiring Required for Operation of Items 1 through 13 above

16. MSIV LC System

17. FW LC System

d. No offsite exposure exceeding the guidelines of 10 CFR 100

e. No loss of integrity of the spent fuel pool

f. Although Class IE sensors to the reactor protection system (RPS) are located on the turbine control valve and stop valve, failure of these sensors will not prevent the reactor from being safely shut down since other RPS sensors (high pressure scram or high flux scram) located in safety-related buildings provide sufficient backup. Therefore, these sensors are not protected from missile hazards inside the turbine building.

g. Although some Class IE sensor cabling to the RPS and Leak Detection systems may be affected by potential missiles within the RWCU heat exchanger room (reference Table 3.5-4), failure of these circuits will not prevent mitigation of the missile effects or the reactor from being safely shut down. A detailed review of the effects for each missile has been performed. This review showed that other channels and sensors are available to perform the functions disabled by the missile.

Protection of essential structures, systems, and components to meet the above criteria is afforded by one or more of the following practices:

a. To locate the system or component in an individual missile-proof structure

b. To physically separate redundant systems or components of the system away from the missile trajectory path or calculated range

c. To provide localized protective shields or barriers for systems and components

d. To design the particular structure or local protective shield/barrier to withstand the impact of the most damaging missile

e. To provide design features on the potential missile source to prevent missile generation

f. To orient the potential missile source in such a manner as to prevent unacceptable consequences due to missile generation

3.5.1.1 Internally Generated Missiles (Outside Containment)

Missile protection is provided within the plant structures important to safety outside the containment for the two general sources of postulated missiles:

a. Rotating component failure

b. Pressurized component failure

A tabulation of all safety-related structures, systems, and components outside the containment, their location, seismic category, quality group classification, and the applicable FSAR sections which include system piping and instrumentation drawings describing safety design features is given in Table 3.2-1. General arrangement and section detail drawings are located in Section 1.2.

3.5.1.1.1 Rotating Component Failure Missiles

A tabulation of missiles generated by postulated failures of rotating components in systems normally operating during power plant operation is given in Table 3.5-1.

Missile selection is based on the following conditions:

a. All rotating components not having synchronous motors which operate during normal operating plant conditions are considered capable of becoming missiles.

b. All rotating components having synchronous motors that operate during normal plant conditions have been evaluated to determine if their couplings can become missiles. The motors will not become missiles because the rotation speed is limited to within design speed should the coupling suffer instantaneous failure. The pump impeller or fan blades will not become missiles during coupling failures because braking forces applied by the process fluids will limit the rotational speed to less than the normal operating speed should a coupling failure occur. The coupling will remain inside the pump coupling guard or fan housing should the coupling fail.

The following general categories of system rotating components were reviewed:

a. Pumps

b. Fans

c. Compressors

d. Turbines

3.5.1.1.2 Pressurized Component Failure Missiles

A tabulation of missiles generated by postulated failures of pressurized components in systems normally operating during power plant operation, is given in Table 3.5-2.

The bases for selection are:

a. Pressurized components in systems whose service temperature exceeds 200 F or whose design pressure exceeds 275 psig are evaluated as to their potential for becoming a missile.

   Piping which exceeded 200 F or 275 psig for 2 percent or less of the time the system is in operation (or if the system was exposed to pressures or temperatures higher than the above limits for less than 1 percent of plant operation) was excluded from missile evaluation.

b. Thermometers or other detectors installed on piping or in wells are evaluated as potential missiles if the failure of a single circumferential weld would cause their ejection.

c. Unrestrained sections of piping such as vents, drains, and test connections are evaluated as potential missiles if the failure of a single circumferential weld could cause their ejection.

d. Dead end flanges are evaluated as potential missiles if the failure of a single circumferential weld could cause their ejection.

e. Valves of ANSI 600 psig rating and above, constructed in accordance with the ASME Code, Section III, are pressure seal bonnet-type valves. For pressure seal bonnet valves, valve bonnets are prevented from becoming missiles by the retaining ring, which would have to fail in these, and by the yoke, which would capture the bonnet or reduce bonnet energy.

   Because of the highly conservative design of the retaining ring of these valves, bonnet ejection is highly improbable. Hence, bonnets are not considered credible missiles.

f. Most valves of ANSI 600 psig rating and below are valves with bolted bonnets. Valve bonnets are prevented from becoming missiles by limiting the stresses in the bonnet-to-body bolting material by the rules set forth in the ASME Code, Section III, and by designing the flanges in accordance with the applicable code requirements. Even if bolt failure were to occur, the likelihood of all bolts experiencing a simultaneous complete severance failure is very remote. The wide-spread use of valves with bolted bonnets and the low historical incidence of complete severance failure of the valve bonnets, confirm that bolted valve bonnets need not be considered as credible missiles.

g. Valve stems are not considered as potential missiles if at least one feature in addition to the stem threads is included in their design to prevent ejection. Valves with backseats are prevented from becoming missiles by this feature. In addition, air or motor operated valve stems

are effectively restrained by the valve operators. No credible valve stem missiles were identified at Grand Gulf.

h. Nuts, bolts, nut and bolt combinations, and nut and stud combinations have only a small amount of stored energy and thus are of no concern as potential missiles.

i. In addition to the pressurized components noted in Item a. above, pressure vessels and pressurized bottles containing noncondensible gases (air, nitrogen, carbon dioxide, etc.) with an operating pressure in excess of 275 psig have been evaluated as potential sources of missiles. Fire extinguishers need not be considered potential sources of missiles based on the following:

   1. Fire extinguishers do not experience the operating transients or cyclic bending stresses which usually exist in piping systems.

   2. Fire extinguishers are required to be equipped with pressure relief devices per UL standards.

   3. UL standards require fire extinguishers to undergo drop tests, vibration tests, and pressure tests.

   4. Per Plant Staff procedures, extinguishers are periodically hydrostatically tested in compliance with NFPA 10.

   5. Plant Staff procedures require a monthly visual and a periodic detailed inspection of all fire extinguishers.

3.5.1.1.3 Evaluation of Potential Missiles

[redacted]

3.5.1.2 Internally Generated Missiles (Inside Containment)

Missile protection is provided within the containment structure for the two general sources of postulated missiles below:

a. Rotating component failures

b. Pressurized component failures

A tabulation of all safety-related structures, systems, and components inside the containment, as well as their location, seismic category, and quality group classification, which include safety design features, is given in Table 3.2-1. General arrangement and section detail drawings are located in Section 1.2.

3.5.1.2.1 Rotating Component Failure Missiles

A tabulation of missiles generated by postulated failures of rotating components in systems normally operating during power plant operation is given in Table 3.5-3.

Missile selection is based on the following conditions:

a. All rotating components, not having synchronous motors which operate during normal operating plant conditions, are considered capable of becoming missiles.

b. All rotating components having synchronous motors that operate during normal plant conditions have been evaluated to determine if their couplings can become missiles. The motors will not become missiles because the rotation speed is limited to within design speed should the coupling suffer instantaneous failure. The pump impeller or fan blades will not become missiles during coupling failures because braking forces applied by the process fluids will limit the rotational speed to less than the normal operating speed should a coupling failure occur. The coupling will remain inside the pump coupling guard or fan housing should the coupling fail.

The following general categories of system rotating components were reviewed:

a. Pumps

b. Fans

c. Compressors

d. Turbines

3.5.1.2.2 Pressurized Component Failure Missiles

A tabulation of missiles generated by postulated failures of pressurized components in systems normally operating during power plant operation, is given in Table 3.5-4.

The bases for selection are:

a. Pressurized components in systems whose service temperature exceeds 200 F or whose design pressure exceeds 275 psig are evaluated as to their potential for becoming a missile.

b. Thermometers or other detectors installed on piping or in wells are evaluated as potential missiles if failure of a single circumferential weld would cause their ejection.

c. Unrestrained sections of piping such as vents, drains, and test connections are evaluated as potential missiles if the failure of a single circumferential weld could cause their ejection.

d. Dead end flanges are evaluated as potential missiles if the failure of a single circumferential weld could cause their ejection.

e. Valves of ANSI 900 psig rating and above, constructed in accordance with the ASME Code, Section III, are pressure seal bonnet-type valves. For pressure seal bonnet valves, valve bonnets are prevented from becoming missiles by the retaining ring, which would have to fail in shear, and by the yoke, which would capture the bonnet or reduce bonnet energy.

   Because of the highly conservative design of the retaining ring of these valves, bonnet ejection is highly improbable. Hence, bonnets are not considered credible missiles.

f. Most valves of ANSI rating 600 psig and below are valves with bolted bonnets. Valve bonnets are prevented from becoming missiles by limiting the stresses in the bonnet-to-body bolting material by the rules set forth in the ASME Code, Section III, and by designing the flanges in accordance with the applicable code requirements. Even if bolt failure were to occur, the likelihood of all bolts experiencing a simultaneous complete severance failure is very remote. The wide-spread use of valves with bolted bonnets and the low historical incidence of complete severance failure of the valve bonnets, confirm that bolted valve bonnets need not be considered as credible missiles.

g. Valve stems are not considered as potential missiles if at least one feature in addition to the stem threads is included in their design to prevent ejection. Valves with backseats are prevented from becoming missiles by this feature. In addition, air or motor operated valve stems are effectively restrained by the valve operators. No credible valve stem missiles were identified at Grand Gulf.

h. A failure of the safety valve header is not considered a credible event for the Grand Gulf design because of the following factors:

   1. The cause of safety valve header failures as described in NUREG-0307 has been identified and corrected in subsequent plant designs. In fact, as demonstrated in Section 3.6B, the safety/ relief valve piping and flanges for Grand Gulf do not even have sufficiently high stress levels to qualify as potential pipe break locations due to the conservative design.

   2. The Grand Gulf design is in accordance with a more recent version of ASME Section III which reflects the experience gained from the two failures.

   3. [HISTORICAL INFORMATION] [Quality assurance during construction, preoperational testing, and in-service inspection during operation all provide additional defense-in-depth assurance that such a failure will not occur.]

i. Nuts, bolts, nut and bolt combinations, and nut and stud combinations have only a small amount of stored energy and thus are of no concern as potential missiles.

j. In addition to the pressurized components noted in item a. above, pressure vessels and pressurized bottles containing noncondensible gases (air, nitrogen, carbon dioxide, etc.) with an operating pressure in excess of 275 psig have been evaluated as potential sources of missiles.

   Fire extinguishers need not be considered potential sources of missiles based on the following:

1. Fire extinguishers do not experience the operating transients or cyclic bending stresses which usually exist in piping systems.

2. Fire extinguishers are required to be equipped with pressure relief devices per UL standards.

3. UL standards require fire extinguishers to undergo drop tests, vibration tests, and pressure tests.

4. Per Plant Staff procedures, fire extinguishers are periodically hydrostatically tested in compliance with NFPA 10.

5. Plant Staff procedures require a monthly visual and a periodic detailed inspection of all fire extinguishers.

The nitrogen bottles for the CRD system are housed in the HCU modules. The bottles are seismically supported and therefore do not present a hazard as a falling object. The relief valve on the top of the bottle is the only source available for postulated internally generated missiles. Should the valve become a missile, the orientation of the valve within the module is such that any damage caused by this postulated missile would be confined to that module.

The safety relief valve air accumulators and air receivers are located inside the drywell and have operating pressures below 275 psig. The accumulators, accumulator drain valves, and the accumulator and receiver relief valves have been previously evaluated as potential sources of internally generated missiles. Should a pipe break occur at the inlet or outlet lines, the thrust forces will not cause the accumulator to become a missile. Also, due to inherent design such as system separation and redundancy and due to the relatively low energy associated with the accumulator/receiver relief and drain valves, no damage is expected if a postulated missile was to occur. All other accumulator parts, specifically handholes, are not considered sources of missiles due to the operating pressure being below 275 psig.

k. Secondary missiles are missiles generated by primary missiles that come in contact with certain components, hardware and equipment. Where appropriate, secondary

missiles were identified and evaluated as listed in Table 3.5-9. The secondary missile is a smaller, less substantial object than the initiating primary missile. The secondary missile is assumed to remain in the same zone of influence as the primary missile. Since the secondary missile is less substantial than the primary missile and the target area is enveloped by the primary missile, the secondary missile effects are less severe and are bounded by the primary missile. Therefore, further evaluations of secondary missiles are not considered necessary after issuance of the full power operating license.

3.5.1.2.3 Evaluation of Potential Missiles

Refer to subsection 3.5.1.1.3 for the evaluation of potential missiles.

3.5.1.2.4 Evaluation of Potential Gravitational Missiles Inside Containment

Seismic Category I systems, components, and structures are not potential missile sources.

Nonseismic items and systems inside the containment are classified as follows:

a. General

   All nonsafety-related items such as piping, non-Class IE conduit, instrument tubing trays, structures, and HVAC ducting are supported to prevent collapse during an SSE whether or not a hazard potential is evident.

b. Cable Tray

   All cable trays for both Class IE and non-Class IE circuits are seismically supported whether or not a hazard potential is evident.

c. Deleted

d. Equipment for Maintenance

All other equipment, such as hoists, that is required during maintenance will either be removed during operation or is restrained to prevent it from becoming a missile.

3.5.1.3 Turbine Missiles

3.5.1.3.1 Turbine Placement and Orientation

The plan and elevation views showing the turbine placement and orientation are shown in Figures 3.5-1 and 3.5-2. A discussion of turbine missile characteristics and probabilities is contained in the manufacturer's Engineering Report ER-503, Turbine Missile Analysis. Additional details of the turbine disc integrity are provided in Section 10.2.3. A discussion of turbine missile characteristics for the upgraded low pressure (LP) turbines is contained in Siemens (manufacturer) Design Report FS 4/1018/1995, which is listed as Reference 8 of Section 10.2.3.8.

Figure 3.5-1 shows the orientation of the low pressure turbine discs in relation to the structures to be protected from externally generated missiles as noted in subsection 3.5.2.3.

Figure 3.5-2 illustrates the angles of departure, the relative elevations of adjacent structures, and the location of those structures relative to the turbine shaft.

Low trajectory missiles are contained after ejection from low pressure turbine discs within the range of angles of ±25˚. Low trajectory missiles which are ejected toward essential systems will be confined by shield walls which are 4'-6" thick and are located adjacent to the turbines. These shield walls are shown in Figures 3.5-1 and 3.5-2 and are designed using a Time-History Analysis technique developed from experimental data reported in References 2 and 3.

3.5.1.3.2 Missile Identification and Characteristics

The turbine-generator for the Grand Gulf Nuclear Station was manufactured by Allis-Chalmers Power Systems and upgraded by Siemens Power Corporation. The potential turbine missiles and the mathematical model used in identifying turbine missile characteristics and probabilities for the original LP turbines are contained in the manufacturer's engineering report ER-503, Turbine Missile Analysis, which is Reference 3 of Section 10.2.3.8.

A listing of turbine missile characteristics resulting from the upgraded LP turbine discs is contained in Siemens Design Report FS 4/1018/1995, which is listed in Reference 8 of Section 10.2.3.8. Also, Siemens has provided supplement to this missile analysis report, which is listed in Reference 11 of Section 10.2.3.8.

3.5.1.3.3 Probability Analysis for High Trajectory Missiles

The probability of significant damage (P) to Category I structures, systems, or components in the plant due to turbine failure is assessed by first determining the separate probabilities of turbine failure missile ejection (P_1), the missile striking a target structure, system, or component (P_2), and the significant damage to the target (P_3). The overall annual probability (P) of a turbine missile striking and significantly damaging a seismic Category I structure, system, or component, is then equal to $P_1 \times P_2 \times P_3$.

A constant value for P_1 of 1×10^{-4}/yr, without regard to historical operating data, has been used for this probability analysis.

The probability of a turbine missile striking a seismic category I structure with sufficient velocity to penetrate one-half the wall (P_2) is calculated in accordance with the procedure discussed in Appendix 3B. For high trajectory and low trajectory missiles, P_2 is calculated to be 1.2654×10^{-4}/yr.

The probability of significantly damaging a target (P_3) is assumed equal to one; that is, each time a high trajectory turbine missile strikes a Category I structure with sufficient velocity to penetrate one-half the wall, it causes significant damage to safety equipment. The modified Petry formula is used to determine if a high trajectory missile will penetrate more than one-half the target barrier thickness. A missile is assumed to do damage if it penetrates more than one-half the target barrier thickness. All Category I structures are considered subject to significant damage.

The probability, P, of significant damage to Category I structures, systems, or components is calculated to be 0.127×10^{-7}/yr. This probability is well within the value of 1×10^{-7}/yr

which is the upper limit on annual probability for a postulated missile causing unacceptable damage, based on ANSI Std N177, (draft standard) "Plant Design Against Missiles." Therefore, no specific protective measures against high trajectory missiles are required. The turbine missile damage probability (P) per target for missiles is provided in Table 3.5-7.

3.5.1.3.4 Turbine Overspeed Protection

A description of the turbine overspeed protection systems is described in Section 10.2. The turbine disc integrity is described in subsection 10.2.3. The component reliability and testing procedures are described in Section 10.2.

3.5.1.3.5 Turbine Valve Testing

The turbine valve testing and test frequency are described in Section 10.2.

3.5.1.3.6 Turbine Characteristics

The turbine characteristics are as described in Section 10.2, Turbine Generator. Operation of the turbine-generator system under normal and transient conditions is as described in Section 10.2.

3.5.1.4 Missiles Generated by Natural Phenomena

[redacted]

Vertical velocities of 70 percent of the indicated horizontal velocities are considered for all missiles except the 1-in. diameter steel rod which is critical for penetration and is assumed to have a vertical velocity equal to the horizontal velocity. These design basis missiles are in accordance with Standard Review Plan 3.5.1.4, Revision 1 (Draft).

3.5.1.5 Site Proximity Missiles (Except Aircraft)

As discussed in Section 2.2. "Nearby Industrial, Transportation, and Military Facilities," there is no credible basis for anticipating site proximity missiles.

3.5.1.6 Aircraft Hazards

As discussed in Section 2.2, "Nearby Industrial, Transportation, and Military Facilities," there is no credible basis for anticipating aircraft hazards.

3.5.2 Structures, Systems, and Components to be Protected from Missiles

3.5.2.1 General

[redacted]

3.5.2.2 Missile Barriers Within Containment

[redacted]

3.5.2.3 Barriers for Missiles Generated Outside of Plant Structures

[redacted]

3.5.2.4 Missile Barriers Within Plant Structures Other Than Containment

[redacted]

3.5.2.5 Missile Barriers for Outdoor Equipment

[redacted]

3.5.3 Barrier Design Procedures

[redacted]

3.5.3.1 Tornado Missile Barrier Design Procedures

[redacted]

3.5.3.2 Barrier Design Procedures for Internally Generated Missiles

In general, protection from internal missiles is provided by barriers. The procedures and calculations employed in design of missile-resistant barriers for turbine missiles and other internally generated missiles are described in Bechtel Topical Report, "Design of Structures for Missile Impact," BC-TOP-9A (Ref. 4).

3.5.3.3 Tornado Missile Risk Evaluator

The Nuclear Energy Institute (NEI) developed the Tornado Missile Risk Evaluator (TMRE) risk-informed methodology for identifying and evaluating the safety significance associated with structures, systems, and components (SSCs) that are exposed to potential tornado-generated missiles. TMRE is an alternative methodology for determining whether protection from tornado-generated missiles is required. The methodology can only be applied to discovered conditions where tornado missile protection was not provided, and cannot be used to avoid providing tornado missile protection in the plant modification process. The TMRE methodology was transmitted to the NRC by NEI as NEI 17-02, Revision 1, on September 21, 2017 and is hereby incorporated by reference into this UFSAR.

3.5.4 References

1. NEI 17-02, Revision 1, “Tornado Missile Risk Evaluator (TMRE) Industry Guidance Document,” September 2017

2. [HISTORICAL INFORMATION] [“Missile Impact Testing of Reinforced Concrete Panels,” Calspan Report No. HC-56-9-D-1, Calspan Corporation, Buffalo, New York, January 1975.

3. Stephenson, A. E., “Tornado Vulnerability Nuclear Production Facilities,” Sandia Laboratories, April 1975.

4. “Design of Structures for Missile Impact,” BC-TOP-9A, Revision 2, Bechtel Power Corporation, San Francisco, California, September 1974.

5. Stephenson, A. E., “Full Scale Tornado Missile Impact Tests,” EPRI Report No. NF-440, Sandia Laboratories, July 1977.]

TABLE 3.5-1: POTENTIAL MISSILES (ROTATING) OUTSIDE CONTAINMENT

Missile Identification	Source of Missile
[redacted]	

TABLE 3.5-1: POTENTIAL MISSILES (ROTATING) OUTSIDE CONTAINMENT (CONTINUED)

	Missile Identification	Source of Missile
	[redacted]	[redacted]
II.	**Diesel Generator Building**	
	[redacted]	[redacted]
III.	**Control Building**	
	[redacted]	[redacted]

TABLE 3.5-2: POTENTIAL MISSILES (PRESSURIZED) OUTSIDE CONTAINMENT

Missile Identification

System	Item P&ID Number	Item Identification	Evaluation Codes
[redacted]			

TABLE 3.5-2: POTENTIAL MISSILES (PRESSURIZED) OUTSIDE CONTAINMENT
(Continued)

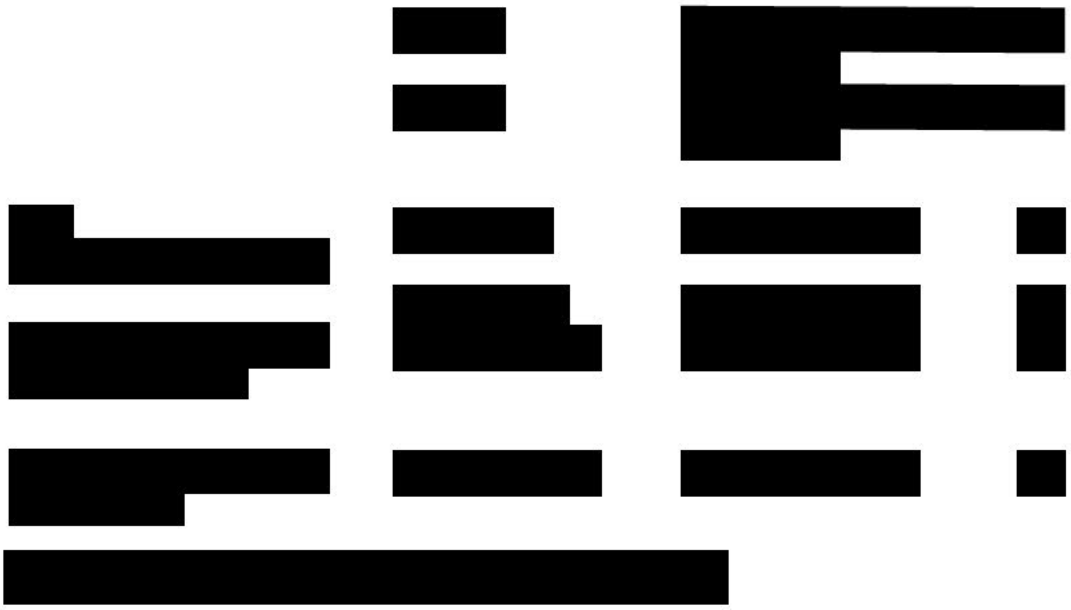


TABLE 3.5-3: POTENTIAL MISSILES (ROTATING) INSIDE CONTAINMENT

Missile Identification	Source of Missile
[redacted]	[redacted]
[redacted]	[redacted]
[redacted]	[redacted]
[redacted]	[redacted]
[redacted]	[redacted]
[redacted]	[redacted]
[redacted]	[redacted]
[redacted]	[redacted]
[redacted]	[redacted]
[redacted]	[redacted]

TABLE 3.5-4: POTENTIAL MISSILES (PRESSURIZED) INSIDE CONTAINMENT

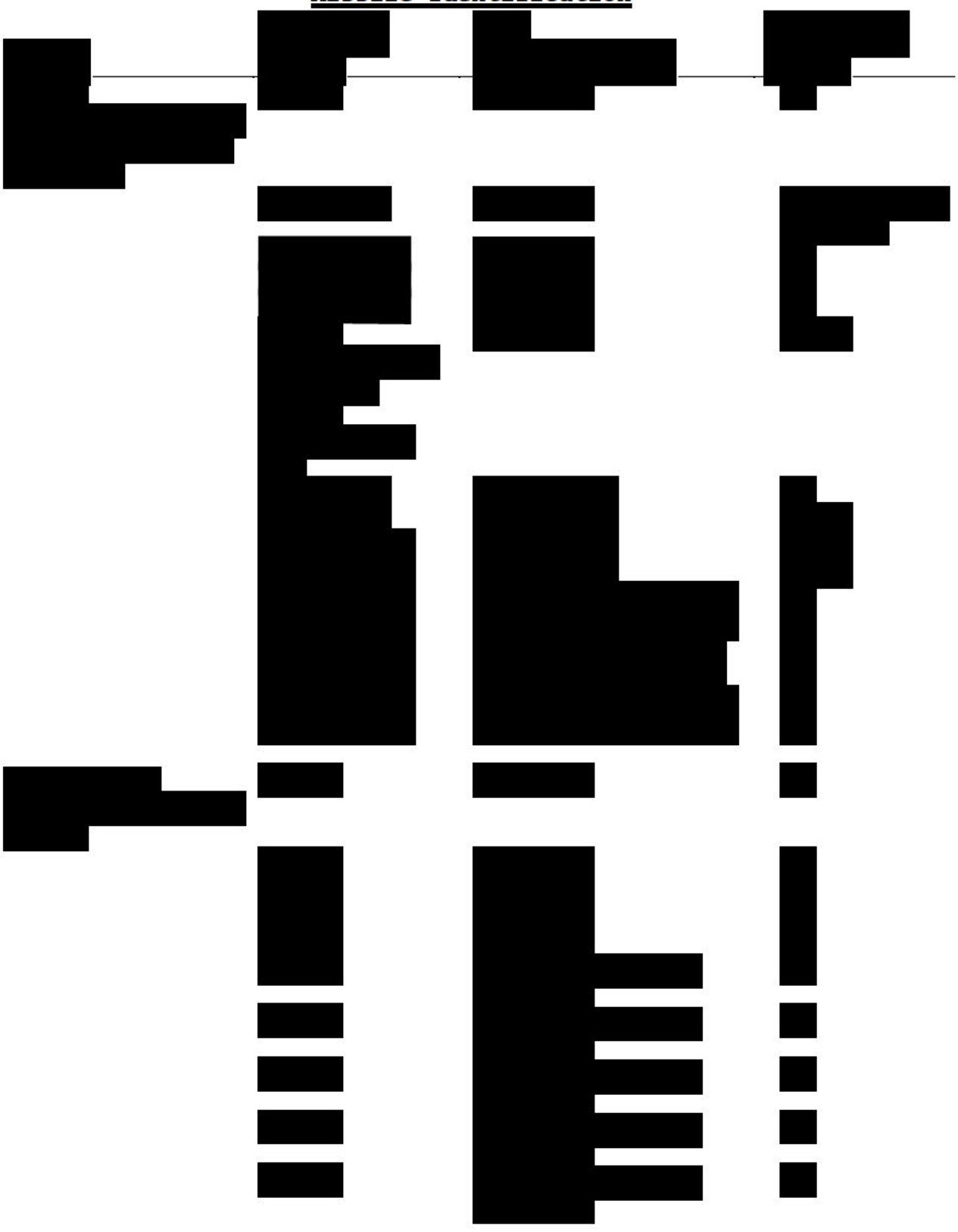


TABLE 3.5-4: POTENTIAL MISSILES (PRESSURIZED) INSIDE CONTAINMENT
Missile Identification (CONTINUED)

System	Item P&ID Number	Item Identification	Evaluation Codes*
[redacted]			

TABLE 3.5-4: POTENTIAL MISSILES (PRESSURIZED) INSIDE CONTAINMENT
Missile Identification (CONTINUED)

Table 3.5-4a: Safety Related Structures, Systems and Components That Do Not Require Protection from Tornado Generated Missiles Based on Tornado Missile Risk Evaluator Methodology: Non-Conforming (Safety-Related) SSC Vulnerabilities

Item	System ID	Vulnerability Description	General Location
1	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]

TABLE 3.5-5: MISSILE CHARACTERISTICS OF THE SELECTED GENERIC POTENTIAL PRESSURIZED MISSILES

Missile	Velocity (Ft/s)	Area (Ft^2)	Weight (lbs)
Instrument wells and/or connections	80	0.0055	3
Instruments (e.g., RTD)	220	0.072	8
4" blank flange assembly	260	0.72	105
8" blank flange assembly	290	1.866	384

Note: The nonflashing subcooled water imparts the highest initial velocity on the missile.

TABLE 3.5-6: MISSILE BARRIERS FOR NATURAL PHENOMENA AND INTERNALLY GENERATED MISSILES

Protected Components	Missile Barrier
[redacted]	[redacted]

TABLE 3.5-7: CUMULATIVE PROBABILITY OF HYPOTHETICAL TURBINE MISSILES DAMAGING SAFETY-RELATED STRUCTURES, SYSTEMS, AND COMPONENTS - P_4

TARGET	P_4
[redacted]	[redacted]
[redacted]	[redacted]
[redacted]	[redacted]
[redacted]	[redacted]
[redacted]	[redacted]
[redacted]	[redacted]
[redacted]	[redacted]

TABLE 3.5-8: SAFETY-RELATED COMPONENTS LOCATED OUTDOORS

Component	Protection Against Tornado Generated Missiles, Turbine Missiles*,or a Seismic Event
[redacted]	[redacted]
[redacted]	[redacted]
[redacted]	[redacted]
[redacted]	[redacted]
[redacted]	[redacted]
[redacted]	[redacted]
[redacted]	[redacted]
[redacted]	[redacted]
[redacted]	[redacted]

TABLE 3.5-8: SAFETY-RELATED COMPONENTS LOCATED OUTDOORS (Continued)

Component	Protection Against Tornado Generated Missiles, Turbine Missiles*,or a Seismic Event
[redacted]	[redacted]
[redacted]	[redacted]
[redacted]	[redacted]
[redacted]	[redacted]
[redacted]	[redacted]
[redacted]	[redacted]
[redacted]	[redacted]
[redacted]	[redacted]

TABLE 3.5-8: SAFETY-RELATED COMPONENTS LOCATED OUTDOORS (Continued)

Component	Protection Against Tornado Generated Missiles, Turbine Missiles*,or a Seismic Event
[redacted]	
[redacted]	[redacted]
[redacted]	[redacted]
[redacted]	[redacted]
[redacted]	[redacted]
[redacted]	[redacted]
[redacted]	[redacted]

TABLE 3.5-8: SAFETY-RELATED COMPONENTS LOCATED OUTDOORS (Continued)

Component	Protection Against Tornado Generated Missiles, Turbine Missiles*,or a Seismic Event
[redacted]	[redacted]
[redacted]	[redacted]
[redacted]	[redacted]
[redacted]	[redacted]
[redacted]	[redacted]
[redacted]	[redacted]
[redacted]	[redacted]
[redacted]	

TABLE 3.5-9: EVALUATION OF SECONDARY MISSILES INSIDE CONTAINMENT

All missiles identified below are in the Reactor Recirculation System (B33).

Primary Missile	Secondary Missile (Note 2)	Evaluation
B33-TE-N028A	Electrical Pullbox PBD412	See Note 1
B33-TE-NO21	Electrical Pullbox PBB417, Conduits lCBRW432 & lCBRW433	See Note 1
Decontamination Drain, 4" Blank Flange:		
Reactor Recirculation Pump -		
B Suction	Electrical Pullbox PBE409, Conduits lCERM414,lCERM413	See Note 1
A Discharge	Electrical Pullbox PBD411, Conduits lCDRM410, lCDRM423, lCDRM436, lCDRM424, lCDRM425 & lCDRM427	See Note 1
B Discharge	4" Piping HBD-942	See Note 1

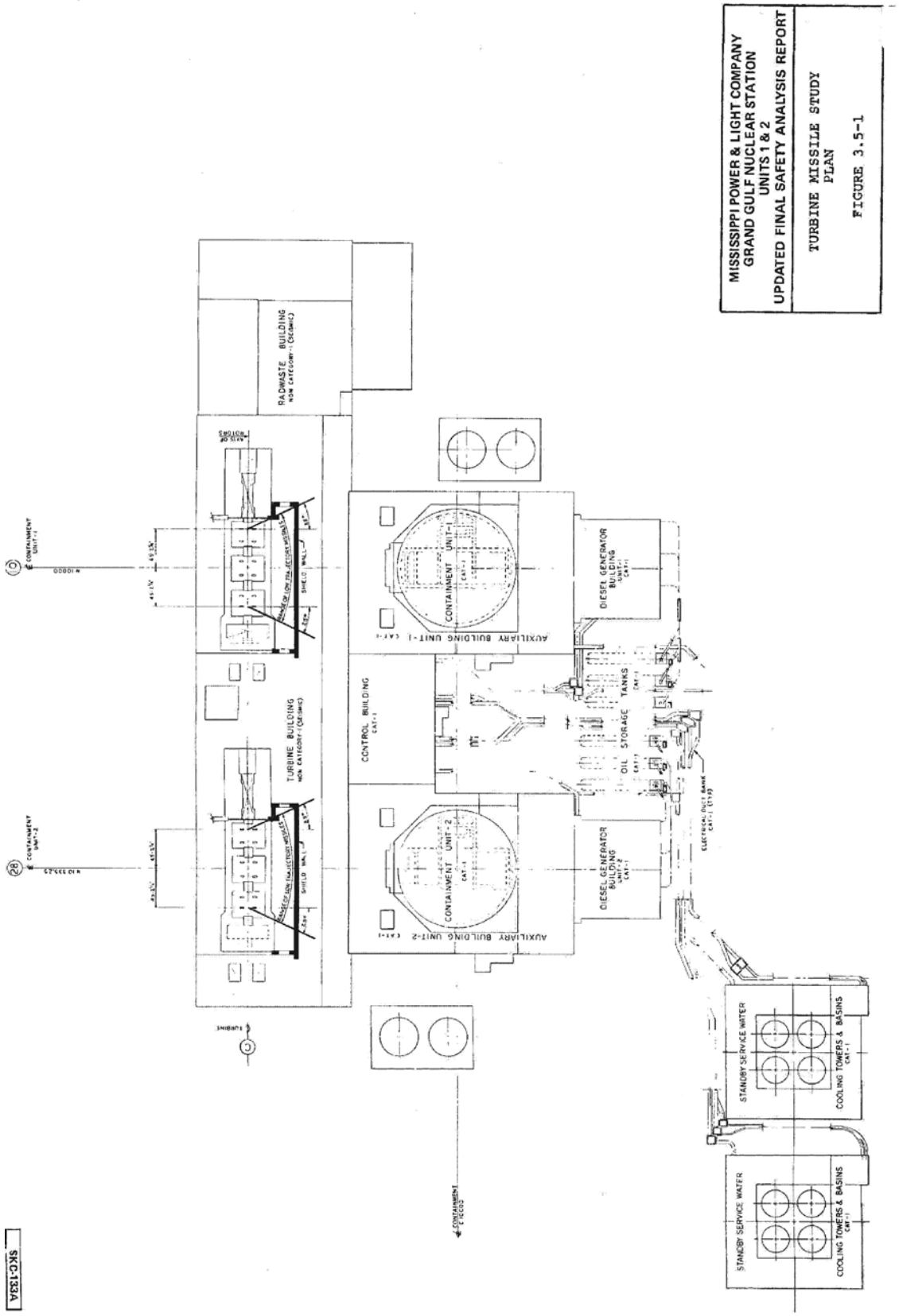
Notes:

1. Both the potential primary and the potential secondary missiles will come in contact with the RPV pedestal or the weir wall, as applicable. All debris associated with the missiles will fall harmlessly to the drywell floor. Safety-related systems and components will not be affected.

2. The secondary missiles listed below were identified and evaluated prior to the operating license. As discussed in subsection 3.5.1.2.2.k, further evaluations of secondary missiles are not considered necessary.

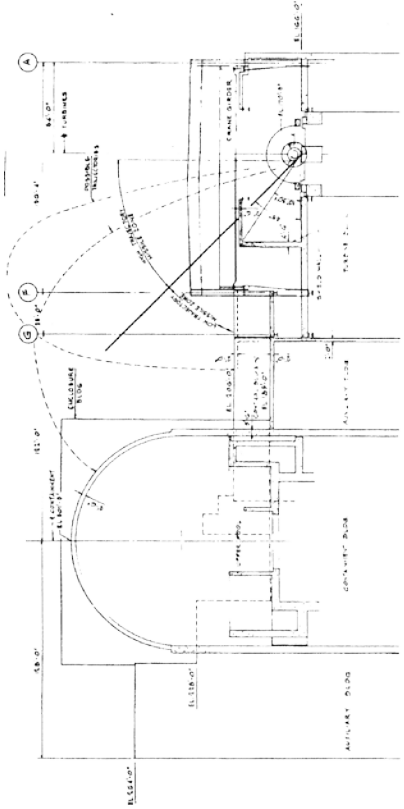
TABLE 3.5-10: TORNADO MISSILE PROTECTION PARAMETERS (SUBSECTIONS REFERENCED)

Seismic Category I Structure	Barrier Thickness	Concrete Barrier Strength (psi)	Curing Time On Which Concrete Strength is Based
Containment	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]

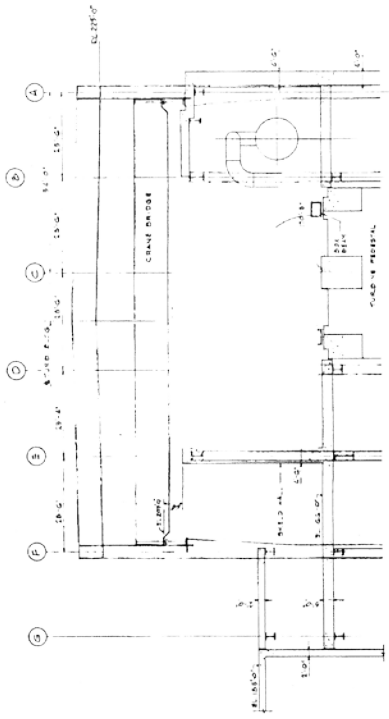

MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT
TURBINE MISSILE STUDY
PLAN
FIGURE 3.5-1
RADWASTE BUILDING
TURBINE BUILDING
CONTROL BUILDING
CONTAINMENT UNIT-1
CONTAINMENT UNIT-2
AUXILIARY BUILDING UNIT-1
AUXILIARY BUILDING UNIT-2
DIESEL GENERATOR BUILDING
OIL STORAGE TANKS
STANDBY SERVICE WATER
COOLING TOWERS & BASINS
SKC-1324

MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT

TURBINE MISSILE STUDY
FIGURE 3.5-2



PLANT SECTION THRU AUXILIARY BUILDING



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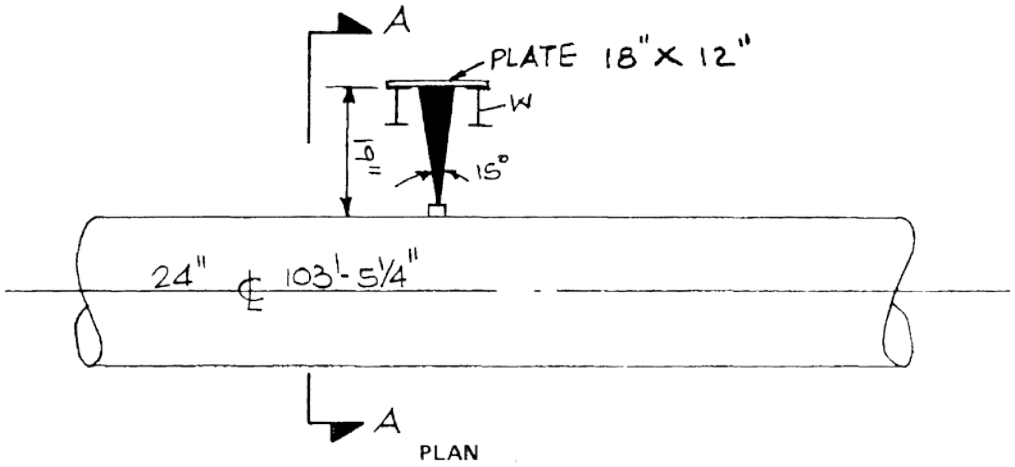
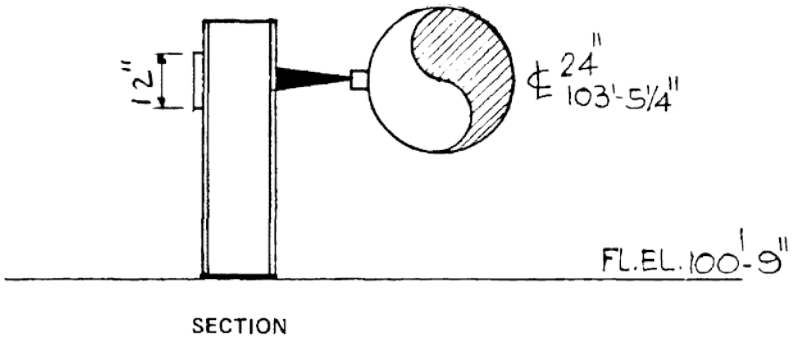
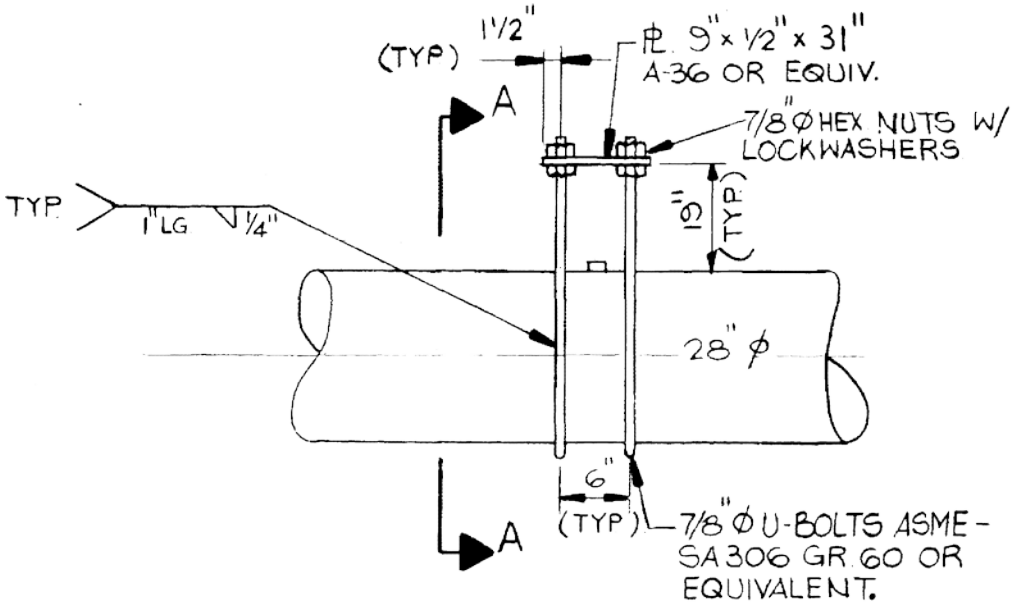




MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT
TYPICAL FLOOR-MOUNTED MISSILE BARRIER FIGURE 3.5-3



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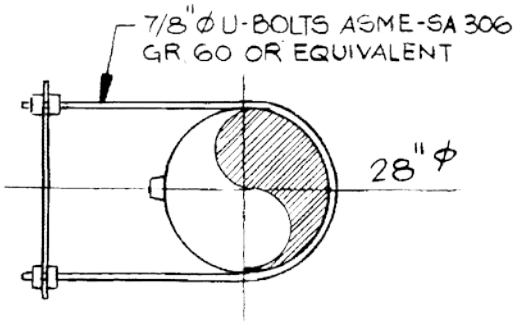
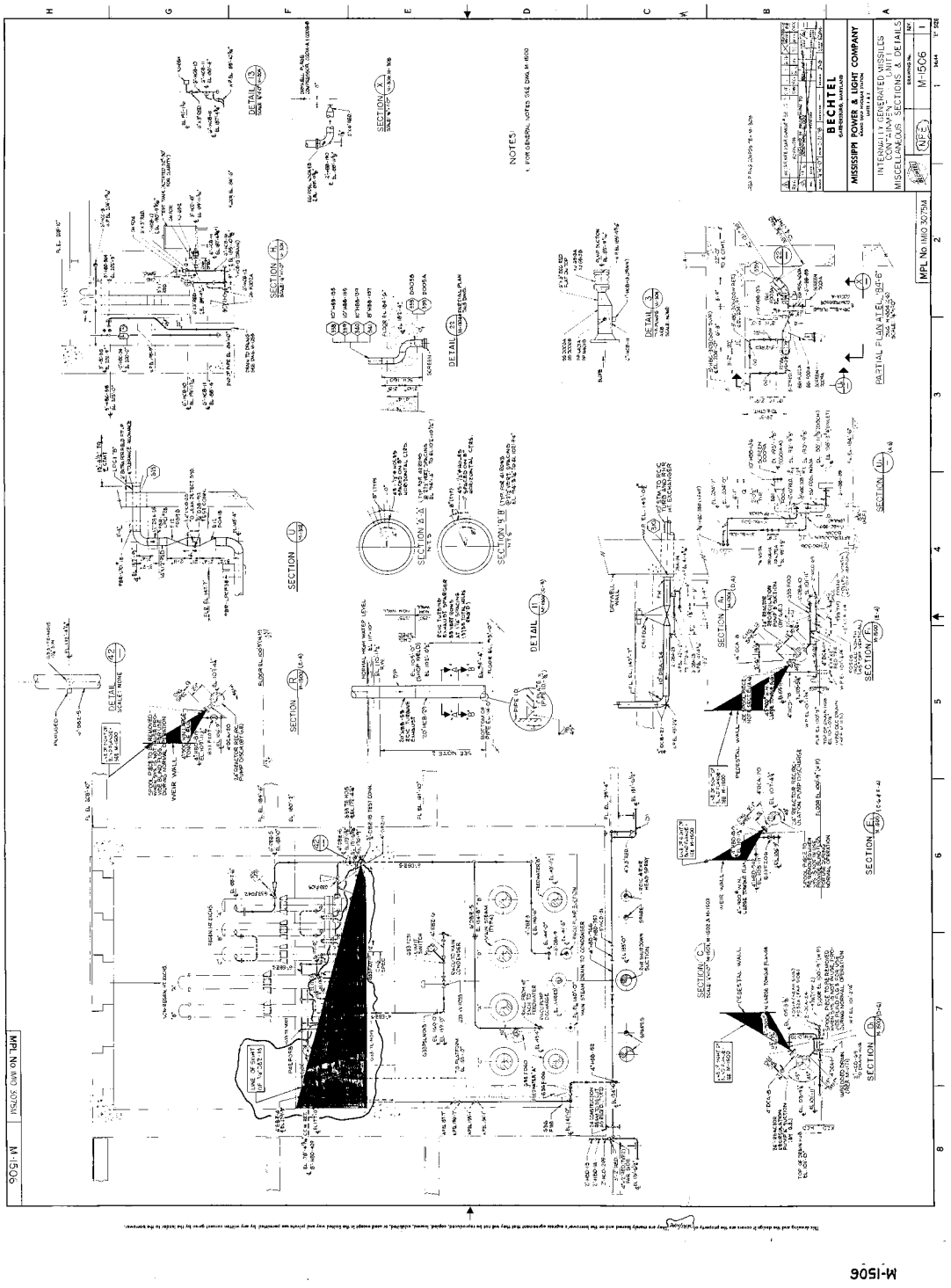
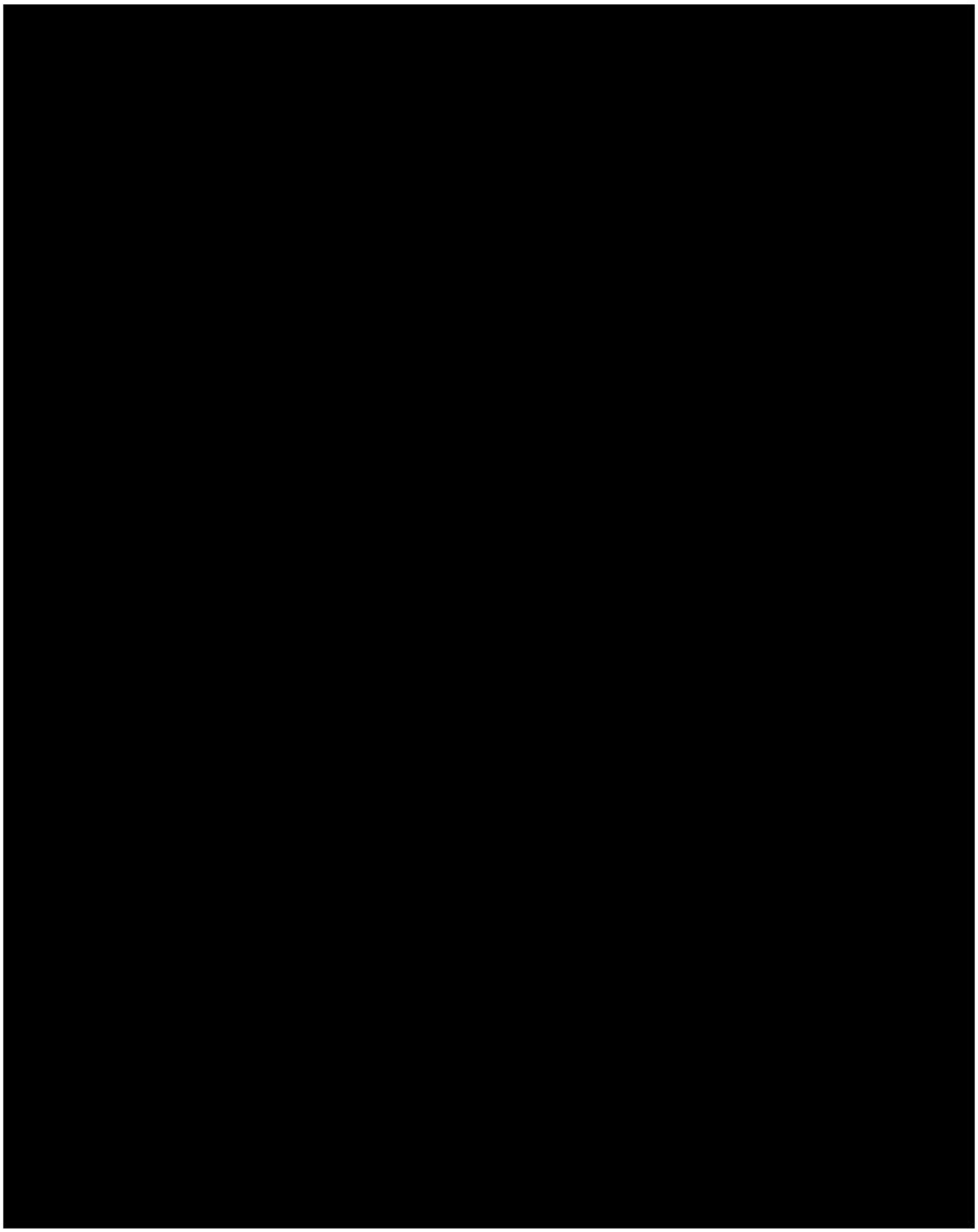
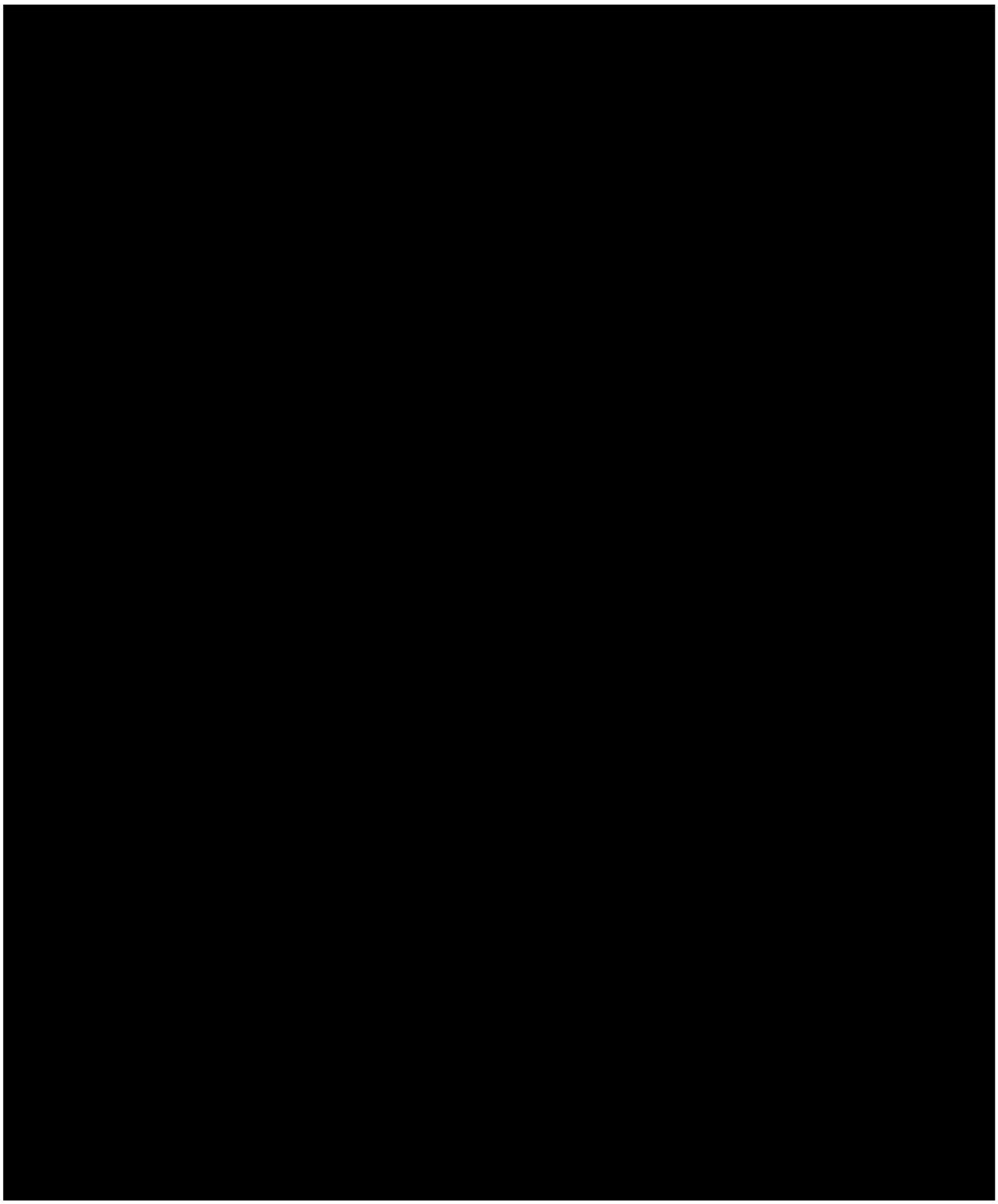


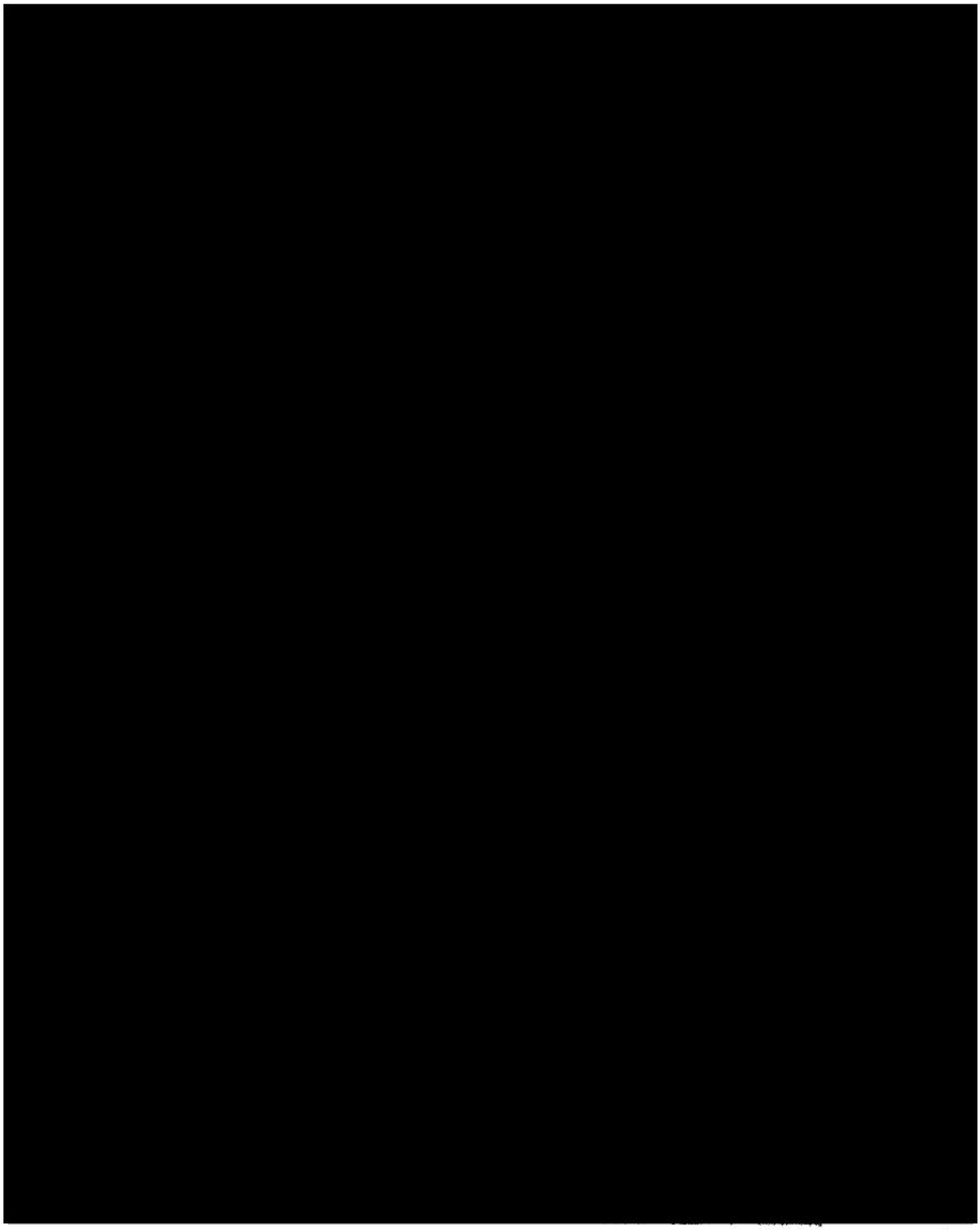
SECTION "A"

MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT
TYPICAL PIPE-MOUNTED MISSILE BARRIER FIGURE 3.5-4


MISSISSIPPI POWER & LIGHT COMPANY
BECHTEL
INTERMEDIATE GENERATED MISSILES CONTAINMENT UNIT
MISCELLANEOUS SECTIONS & DETAILS
M-1506
NOTES:
1. FOR GENERAL NOTES SEE DWG. M-1500
PARTIAL PLAN AT EL. 134'-6"
M-1506







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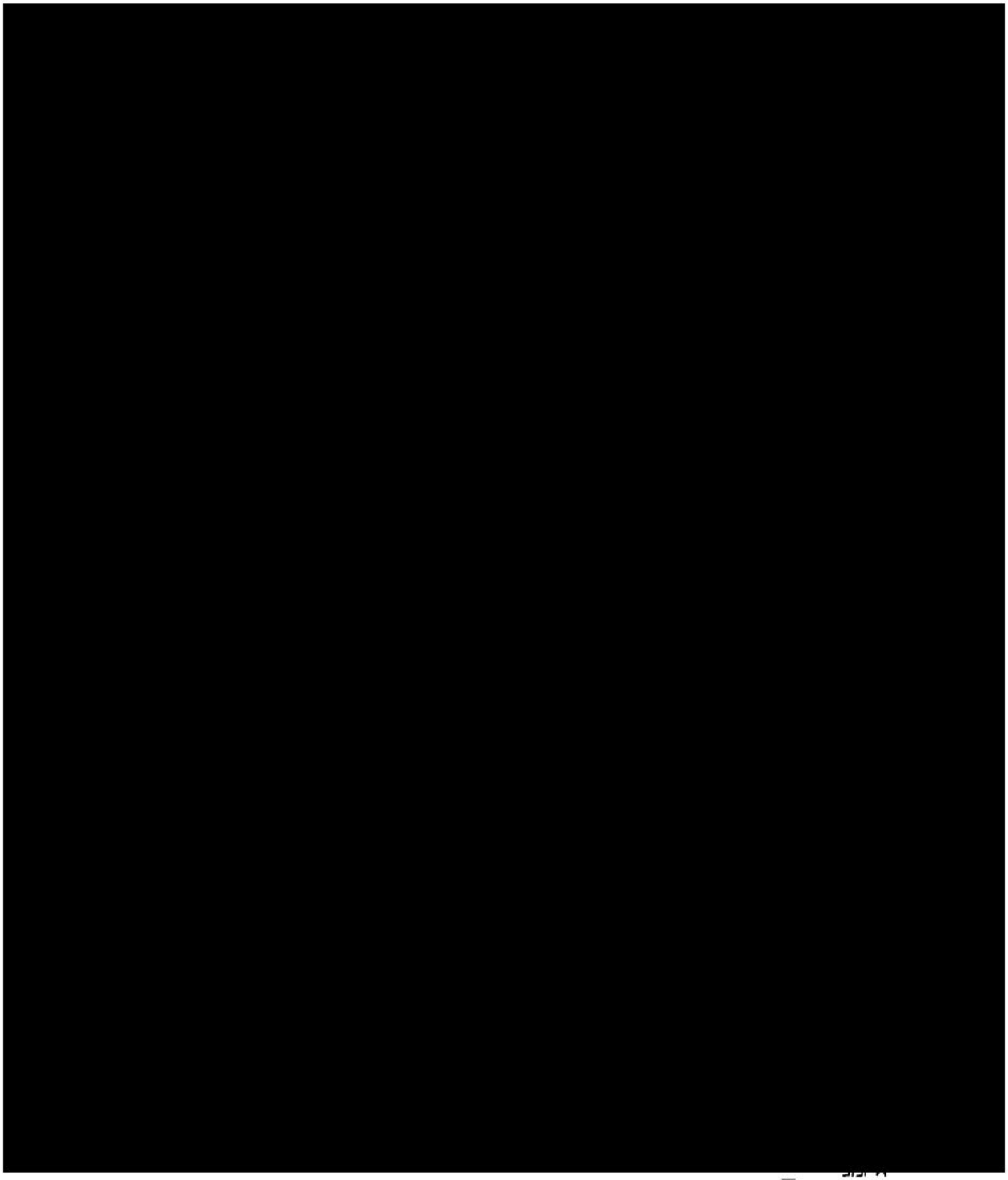

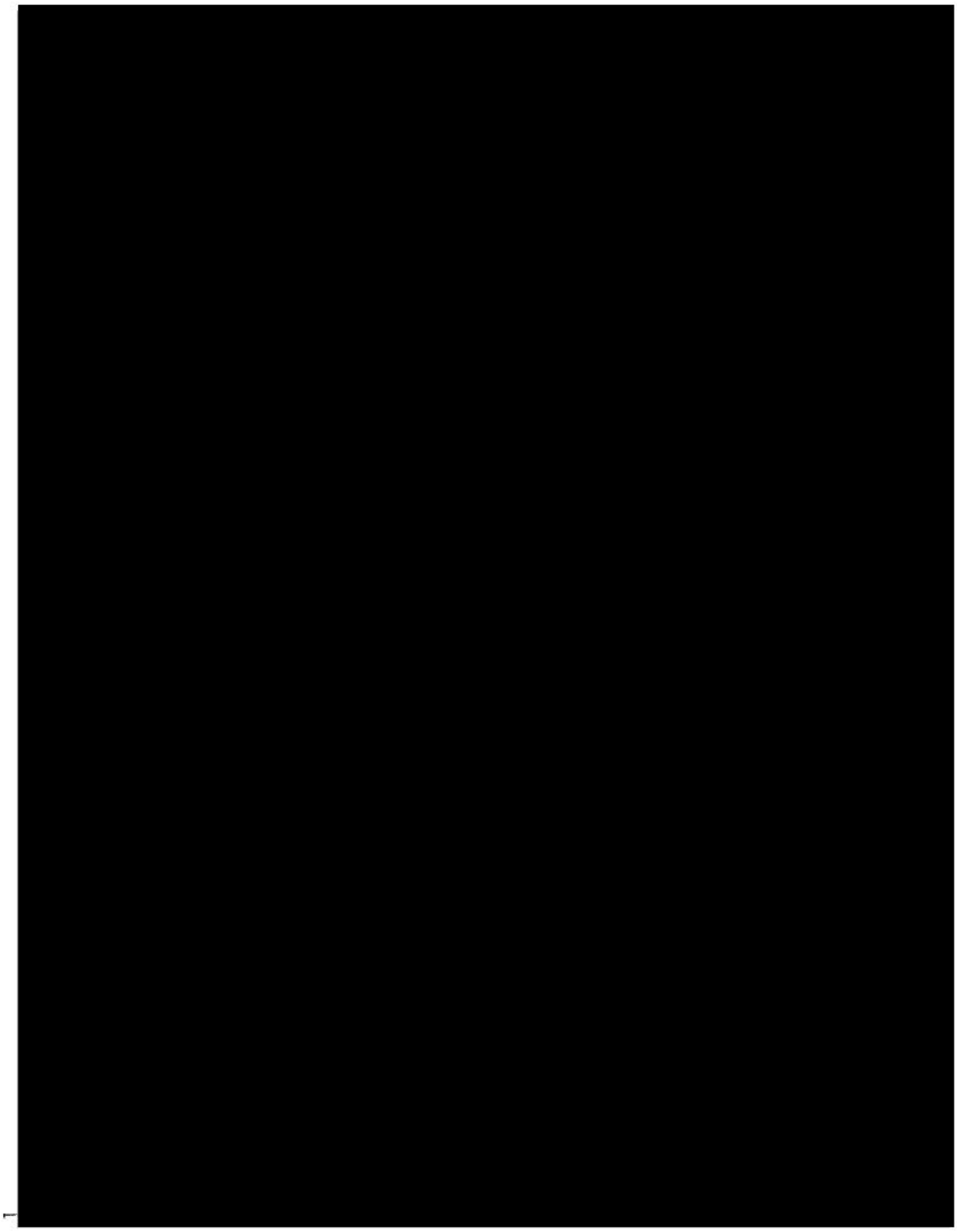

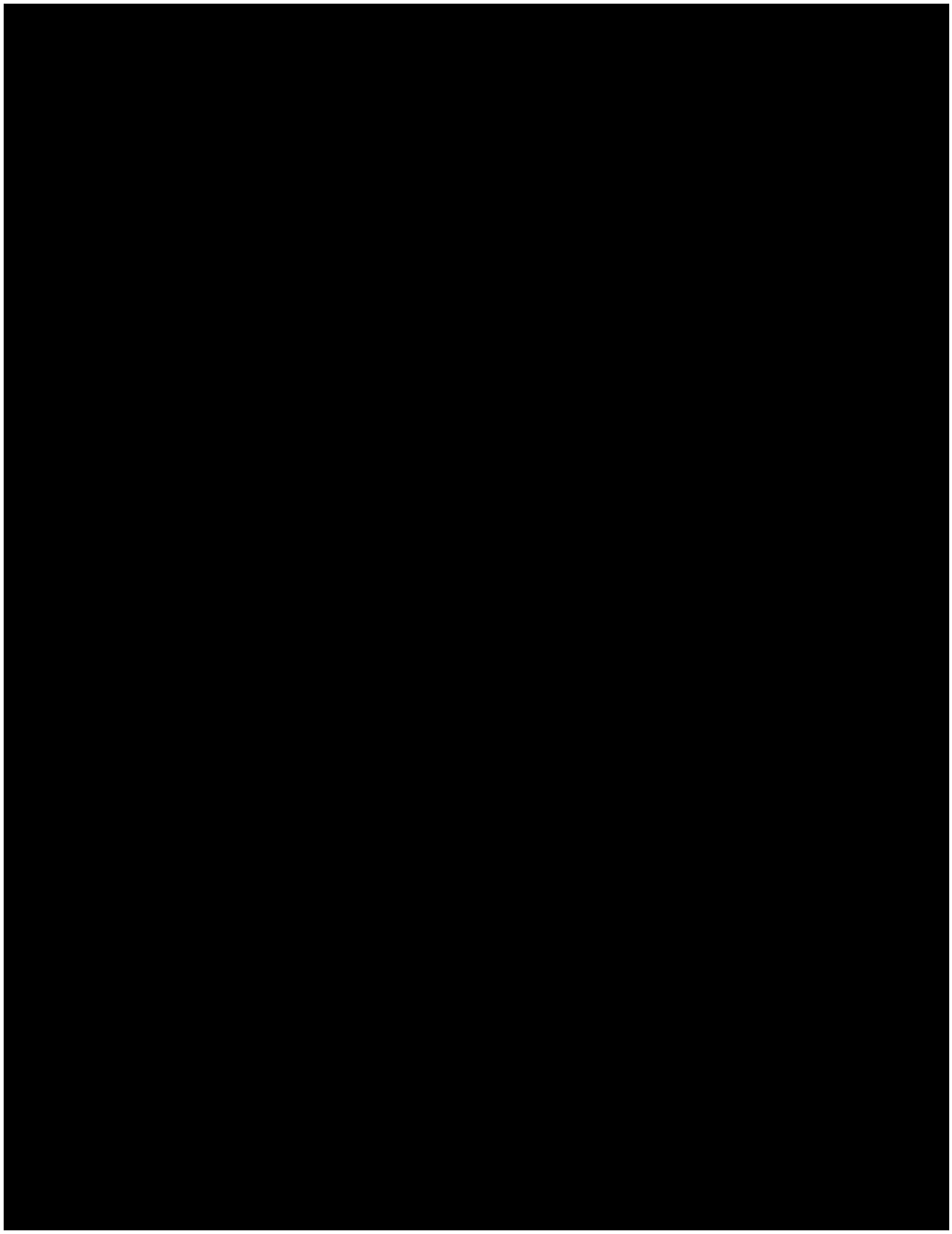

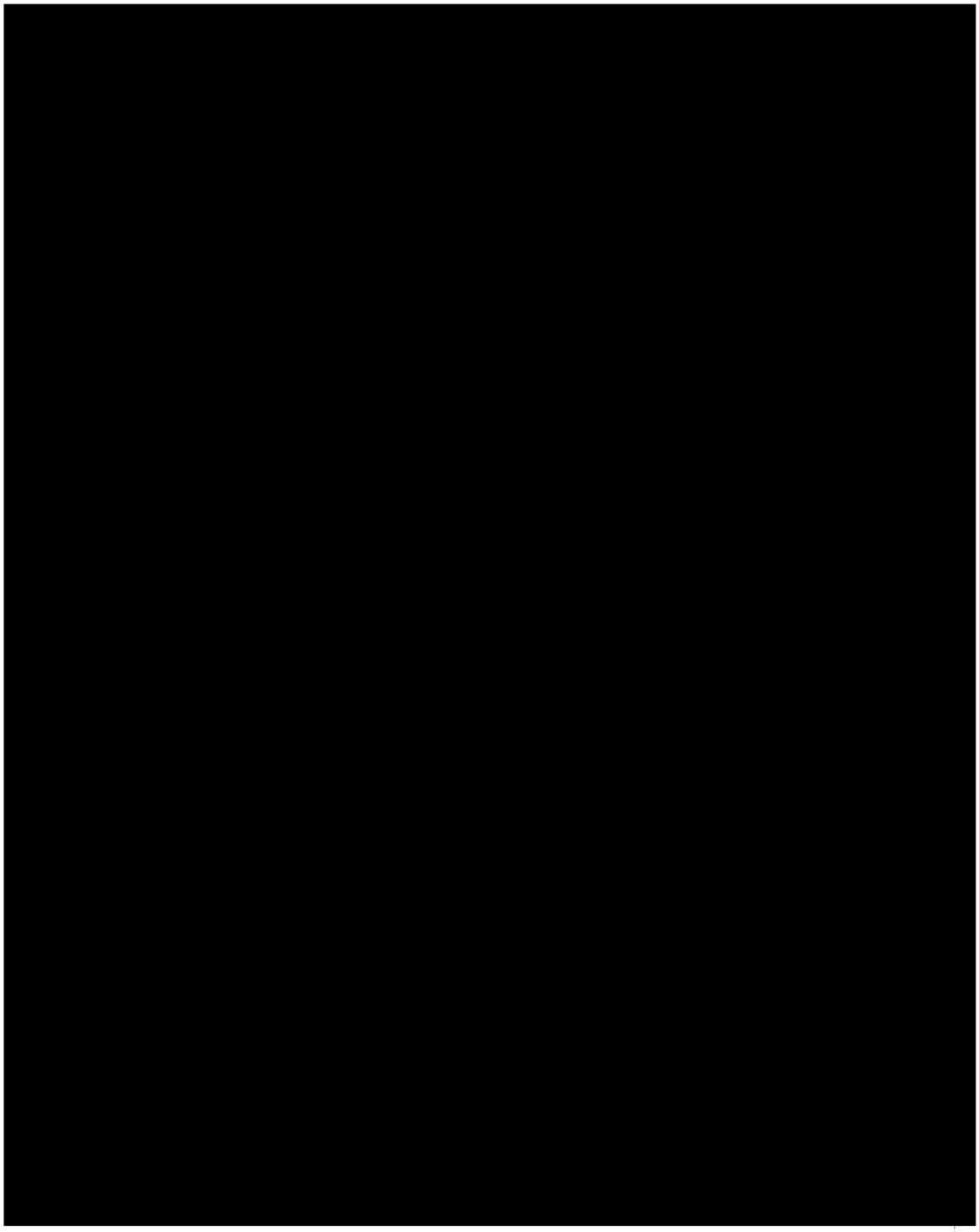

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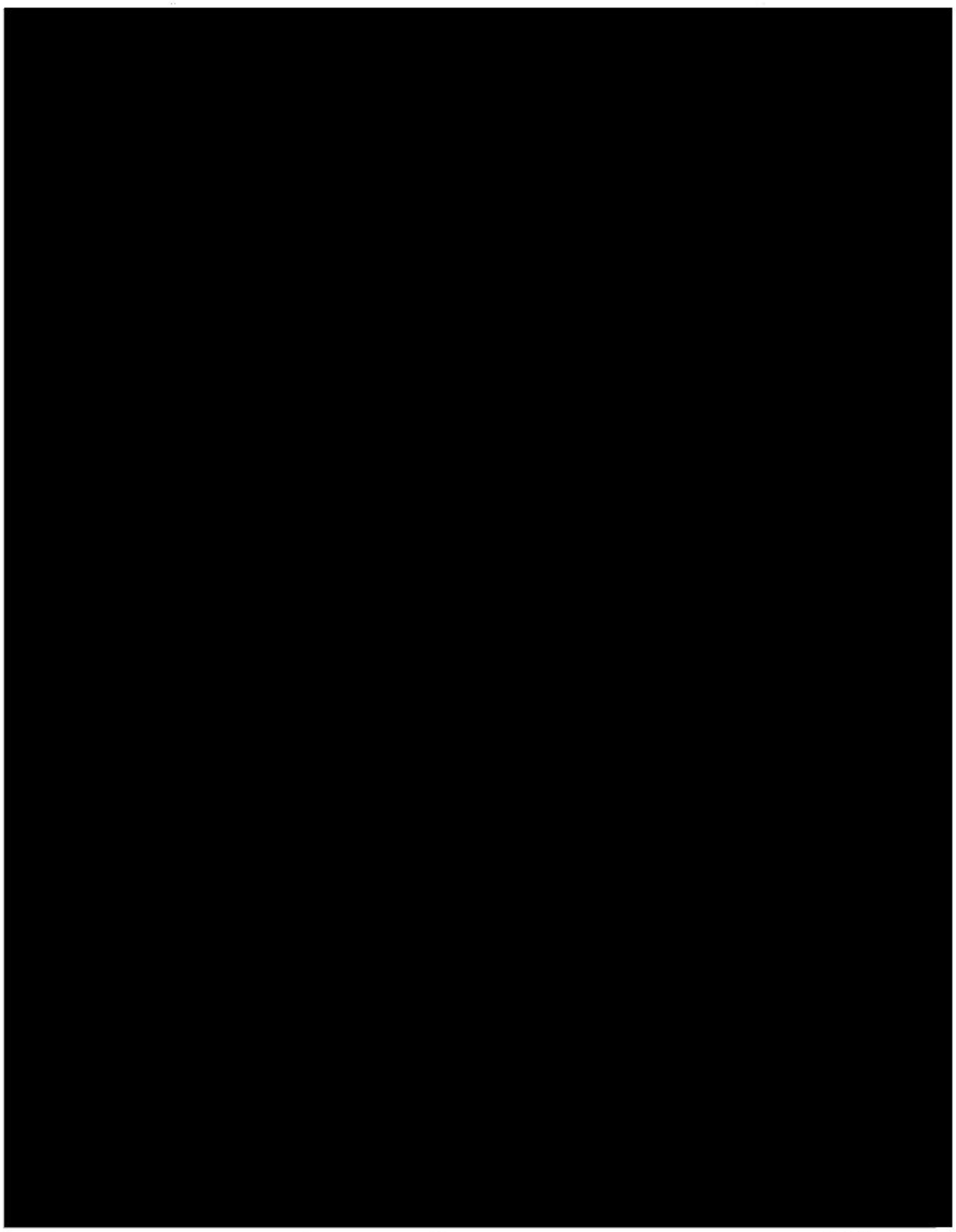
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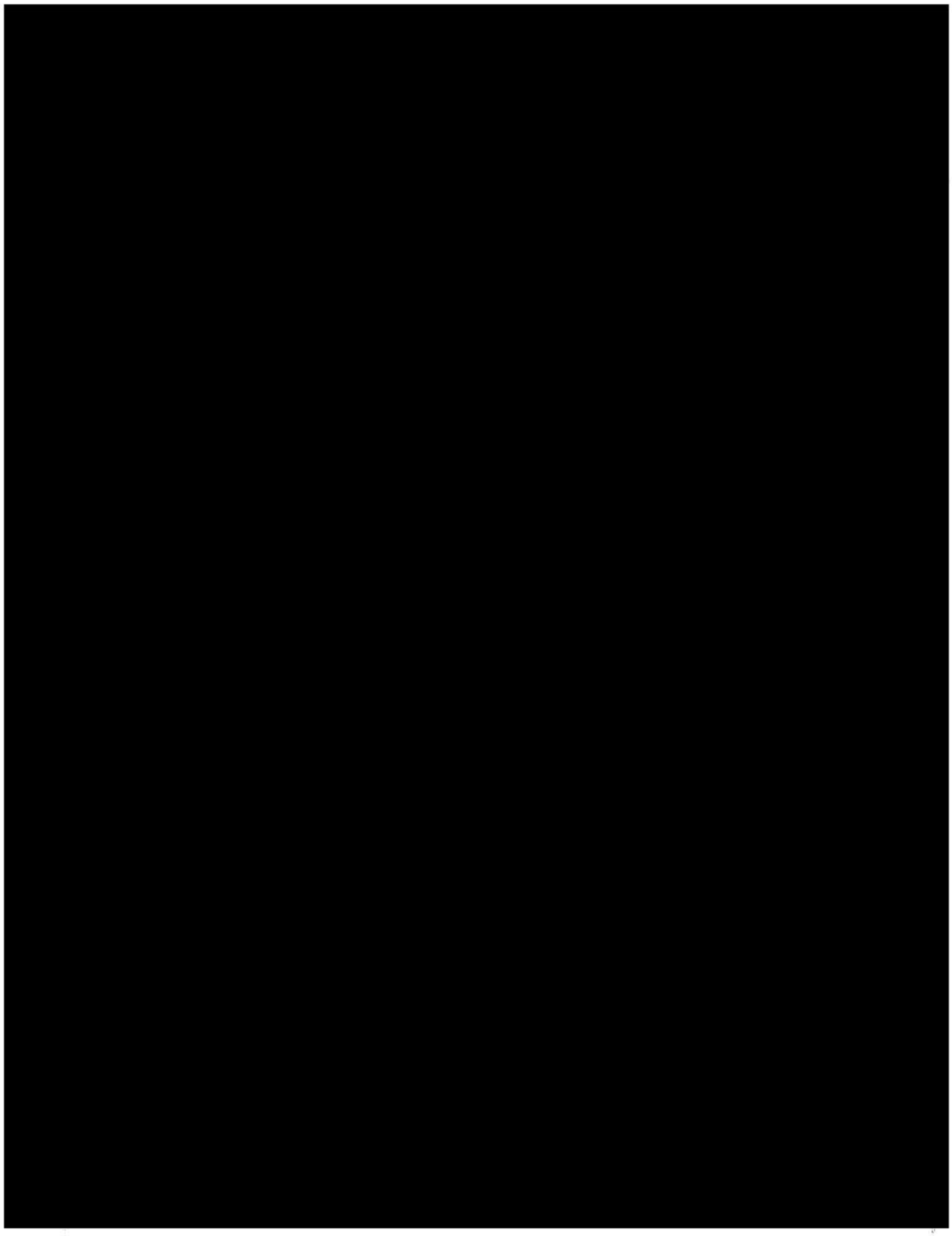

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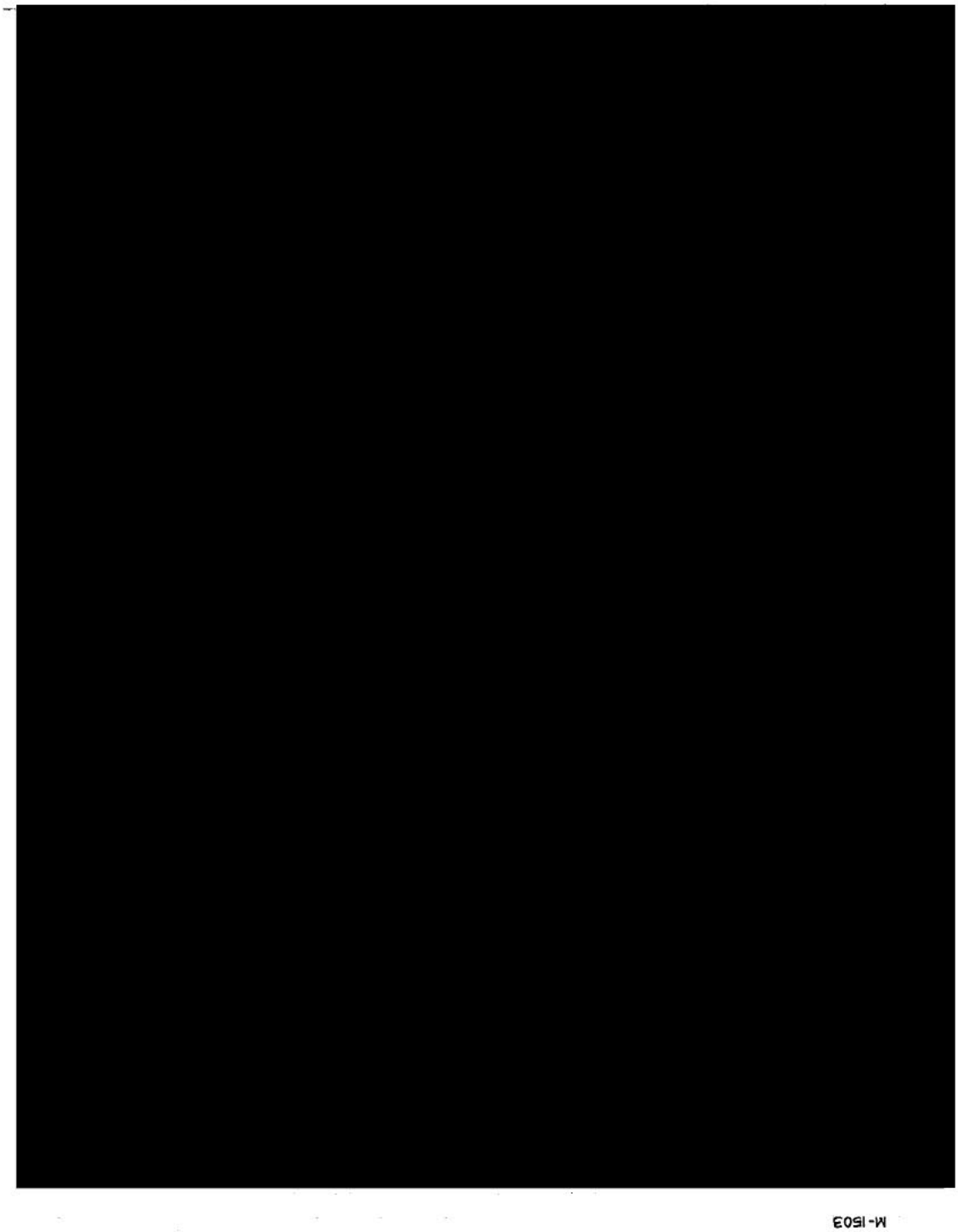

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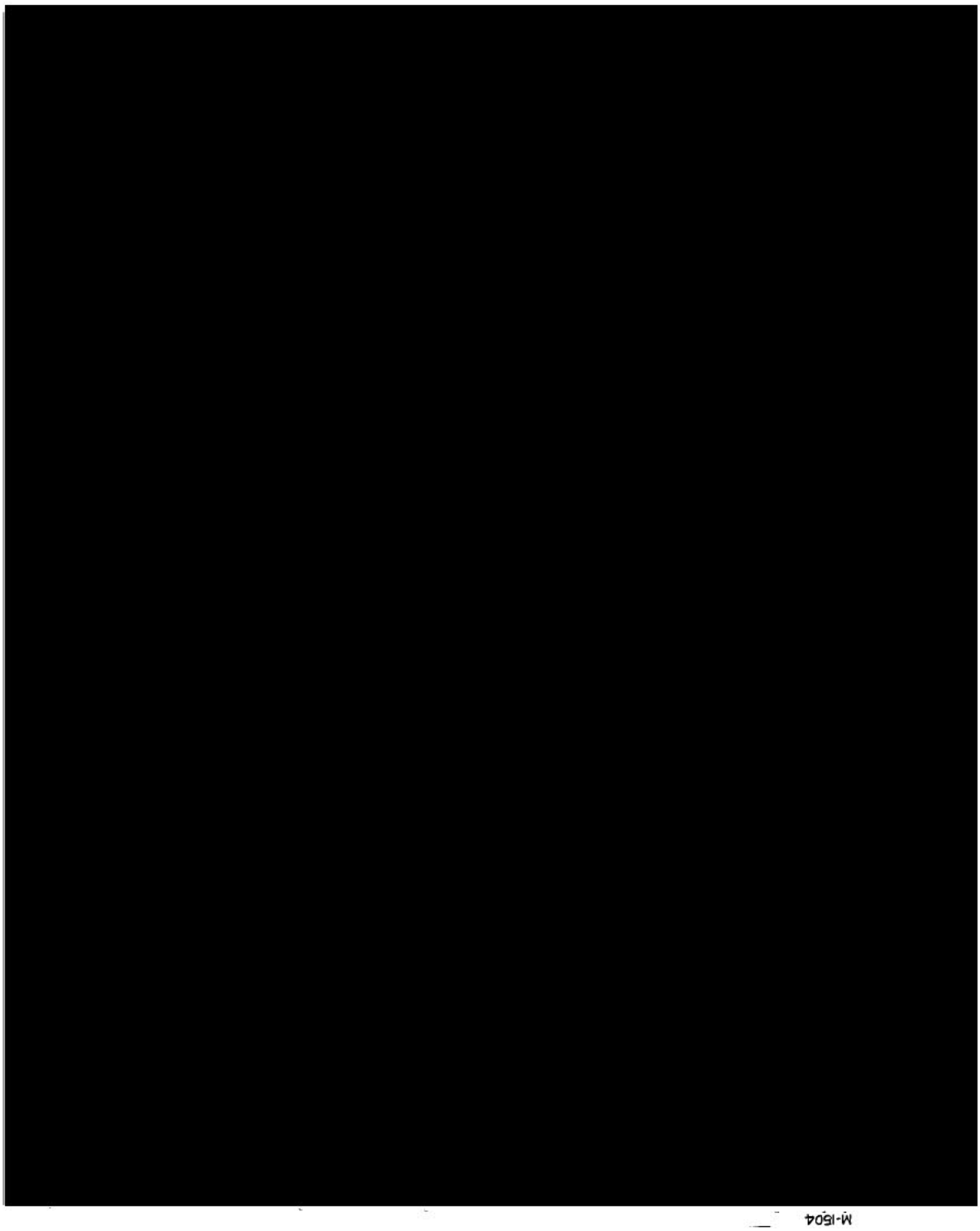

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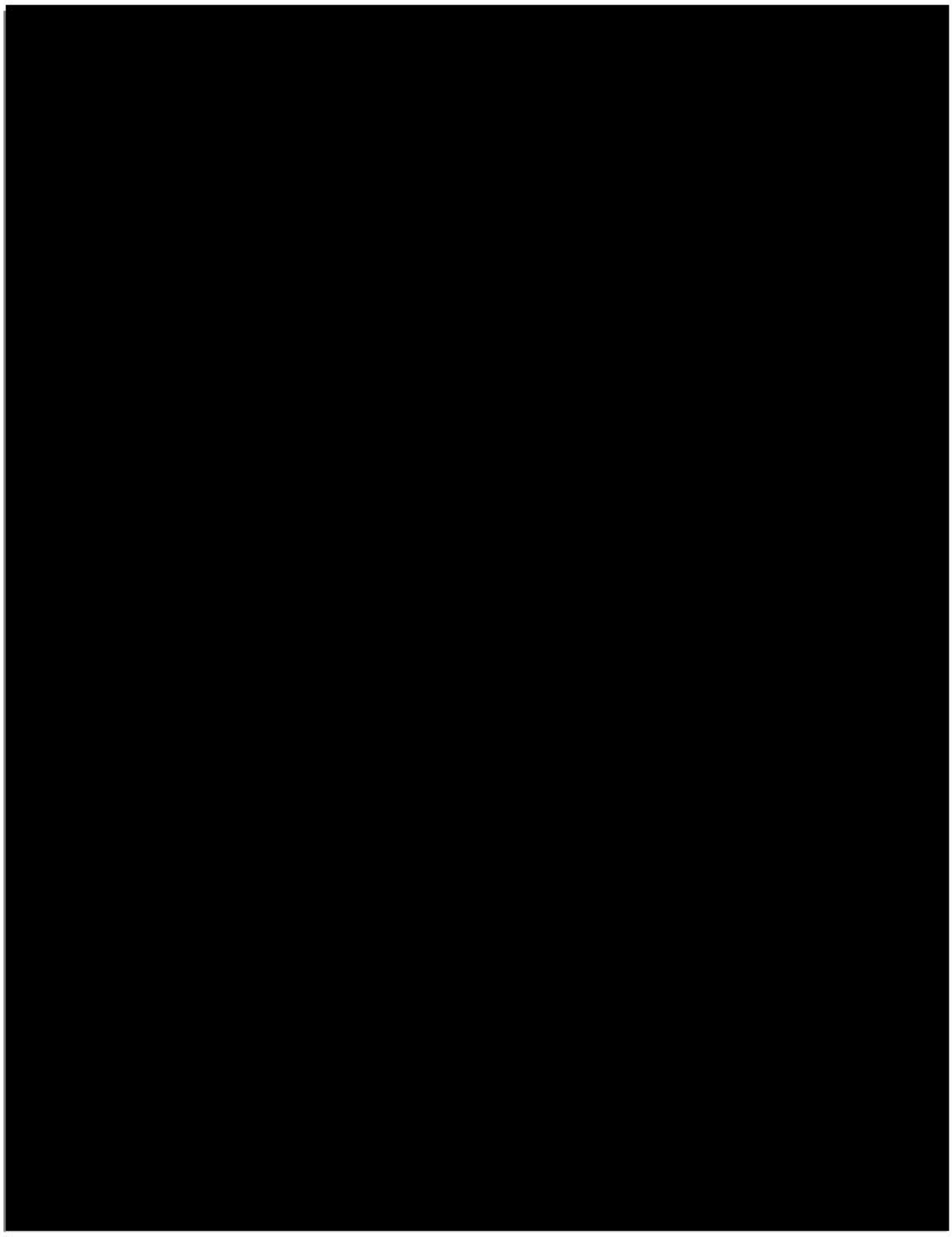

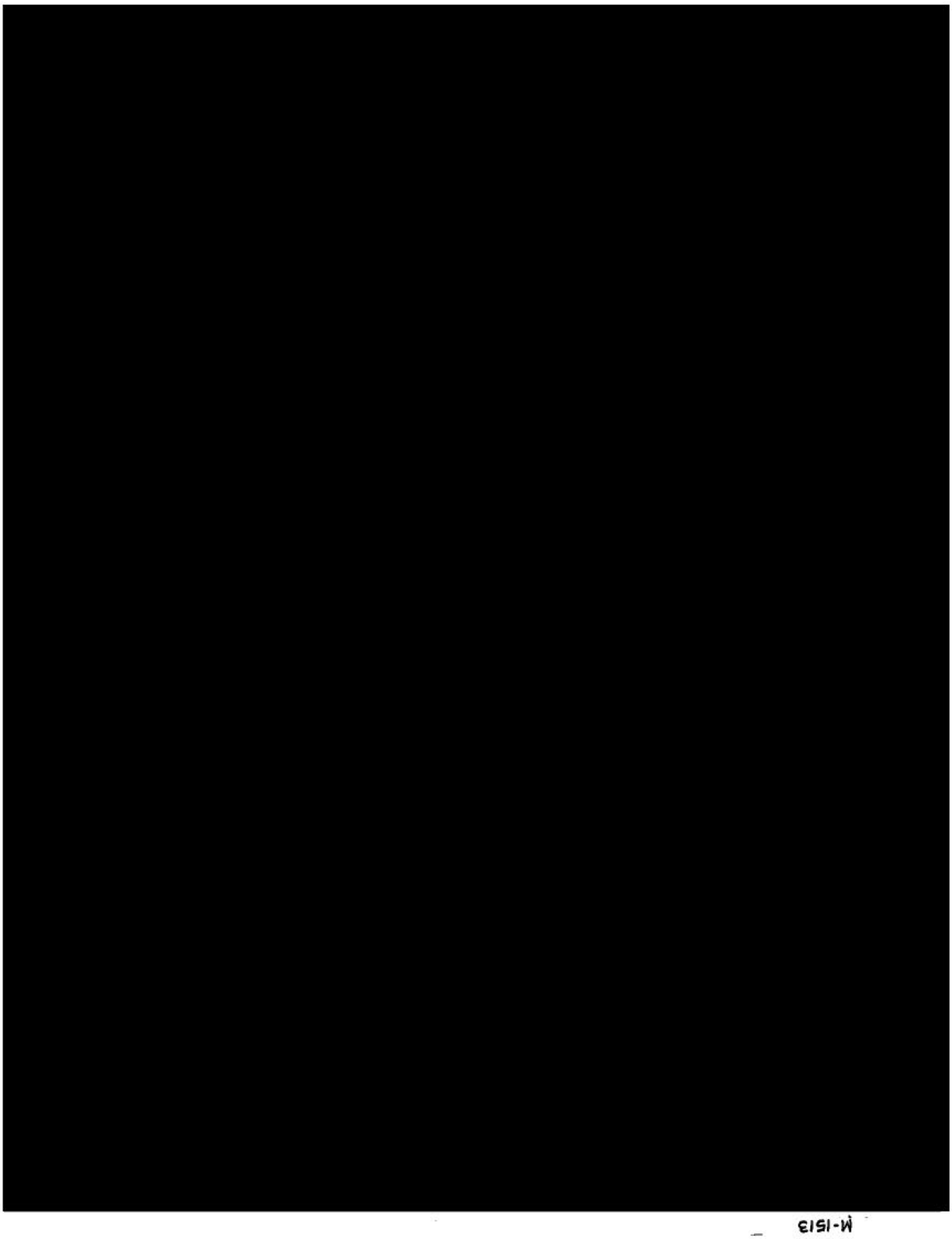

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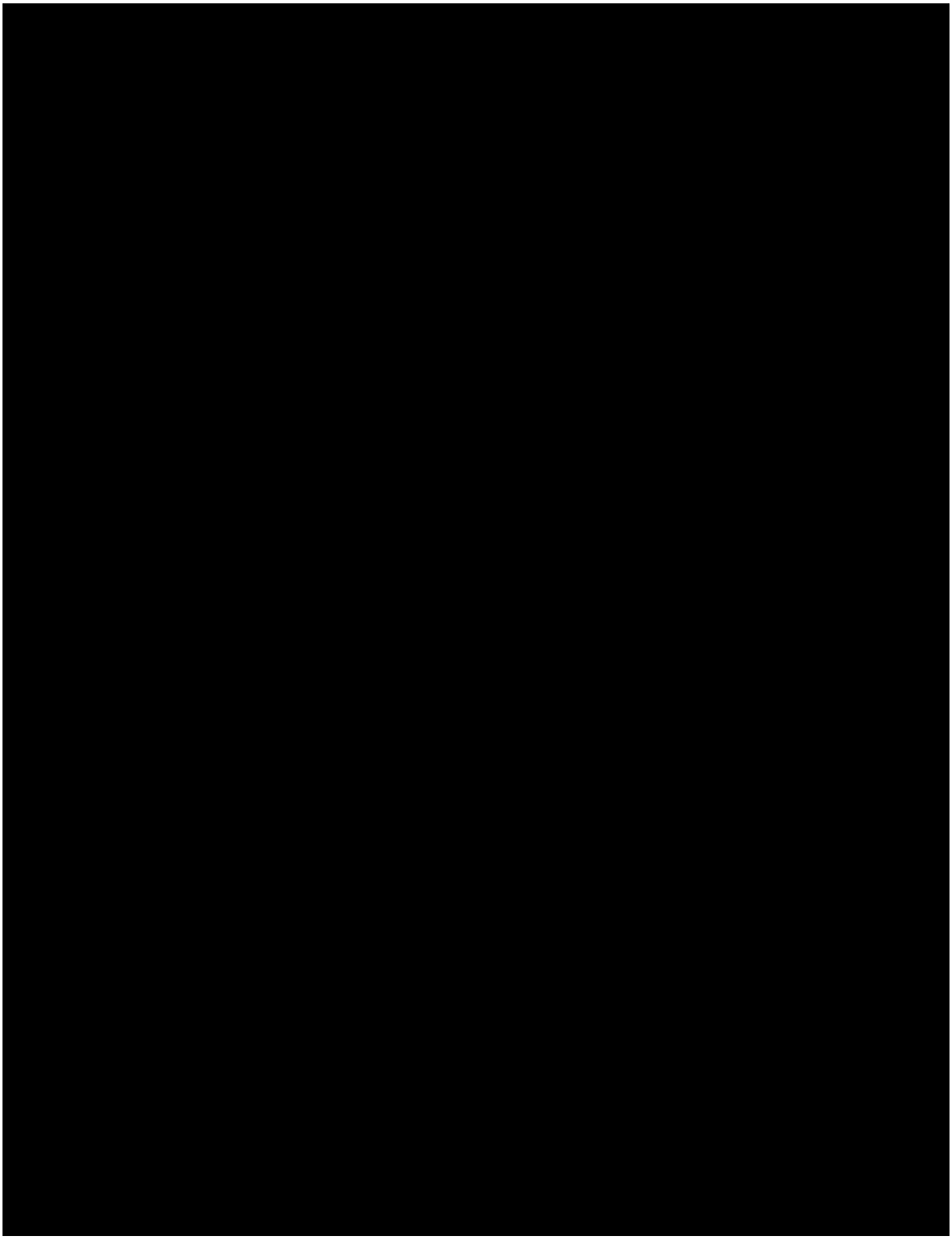
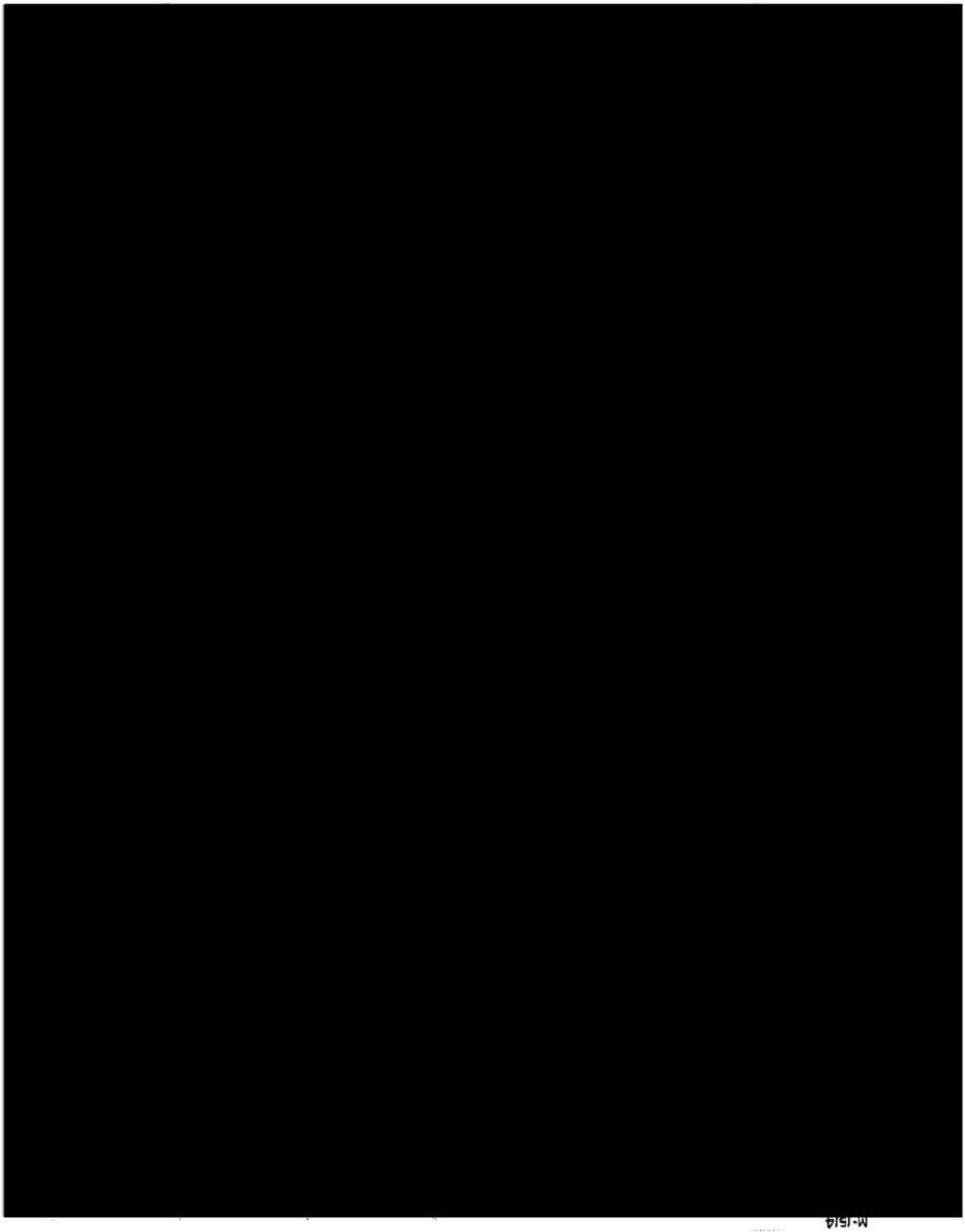
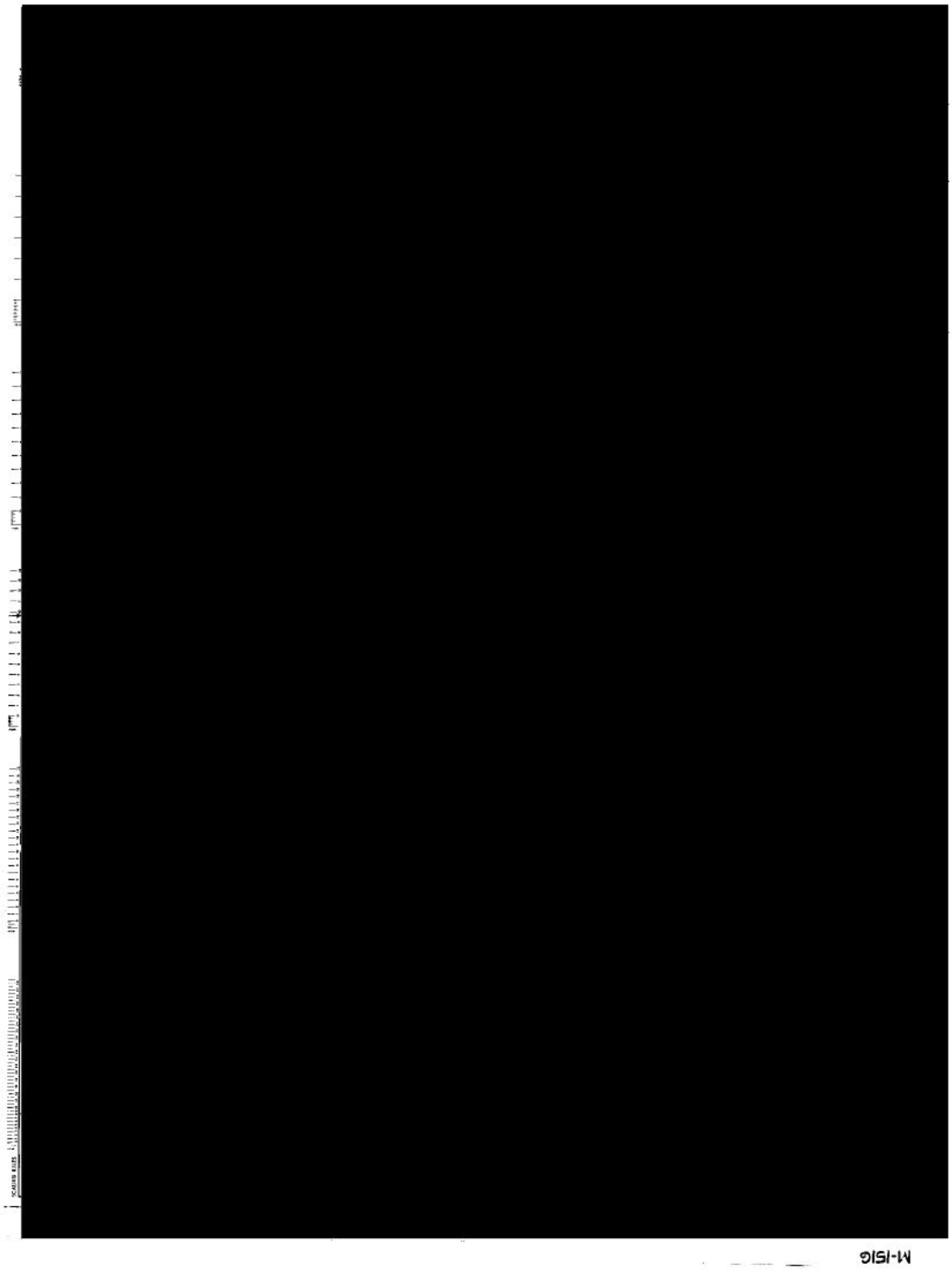






3.6 PROTECTION AGAINST THE DYNAMIC EFFECTS ASSOCIATED WITH THE POSTULATED RUPTURE OF PIPING

This section is divided into two parts; Section 3.6A is applicable to all piping except the piping covered in 3.6B.

Section 3.6B is applicable to the recirculation piping and the in-containment portions of main steam piping.

3.6A PROTECTION AGAINST DYNAMIC EFFECTS ASSOCIATED WITH THE POSTULATED RUPTURE OF PIPING (EXCEPT THE PIPING COVERED IN SECTION 3.6B)

This section describes the design bases and protective measures which ensure that the containment, essential systems, components and equipment, and other essential structures are adequately protected from the effects associated with the postulated rupture of high-energy piping and cracks of moderate-energy piping both inside and outside the containment.

3.6A.1 Postulated Piping Failures in Fluid Systems Inside and Outside the Containment

This section sets forth the design bases, description, and safety evaluation for determining the effects of postulated piping failures in fluid systems both inside and outside the containment, including protective measures.

3.6A.1.1 Design Bases

a. Criteria

The pipe failure protection conforms to Appendix A of 10 CFR 50, General Design Criterion 4, Environmental and Missile Design Bases. The overall design for this protection is in compliance with USNRC Regulatory Guide 1.46 and NRC Branch Technical Positions (BTP) APCSB 3-1 and MEB 3-1, (Reference 7) the implementation of which is discussed herein.

b. Objectives

Protection against pipe failure effects was provided to fulfill the following objectives:

1. Assure that the reactor can be shut down safely and maintained in a safe cold shutdown condition or mitigate the consequences of a loss-of-coolant accident (LOCA).

2. Assure that containment integrity is maintained.

3. A pipe break which is not a loss of reactor coolant must not cause a loss of reactor coolant beyond makeup capabilities.

4. Assure that the radiological doses of a postulated piping failure remain below the limits of 10 CFR 100, 10 CFR 50.67 and 10 CFR 50 GDC 19.

c. Assumptions

The following assumptions were used to determine the protection requirements:

1. Pipe breaks or cracks were postulated to occur during normal plant operation (i.e., reactor startup, operation at power, hot standby or reactor cooldown to a cold shutdown).

2. Only high-energy piping as defined in subsection 3.6A.2.1a was considered as a pipe break initiator. Moderate-energy piping as defined in subsection 3.6A.2.1b was capable of producing only cracks.

3. Pipe breaks were evaluated for the effects of pipe whip, jet impingement, flooding, room pressurization, and other environmental effects such as temperature.

4. Pipe cracks were evaluated for wetting from spray, flooding, and other environmental effects.

5. Each longitudinal or circumferential break in high-energy fluid system piping, or leakage crack in moderate-energy fluid system piping, was considered separately as a single postulated initial event occurring during normal plant conditions (applicable to seismic and nonseismic piping).

6. Pipe failures (breaks or cracks) inside the containment were not evaluated concurrently with failures outside the containment.

7. Offsite power was assumed to be unavailable if a trip of the turbine-generator system or reactor protection system was a direct consequence of the postulated piping failure, unless it was more conservative to assume that offsite power was available (e.g., a feedwater line break with offsite power available leads to a larger inventory of water for flooding considerations).

8. A single active component failure was assumed in systems used to mitigate consequences of the postulated piping failure and to safely shut down the reactor, except as noted in paragraph 9 below. The single active component failure was assumed to occur in addition to the postulated piping failure, such as unit trip and loss of offsite power.

9. Where the postulated piping failure was 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 were not assumed provided the system was designed to the following criteria: 1) seismic Category I standards, 2) was powered from both offsite and onsite sources, and 3) was constructed, operated, and inspected to quality assurance, testing, and inservice inspection standards appropriate for nuclear safety systems. Examples of systems that qualify as moderate-energy, dual-purpose essential systems are standby service water systems and residual heat removal systems.

10. All available systems, including those actuated by operator actions, were employed to mitigate the consequences of a postulated piping failure to the extent clarified in the following paragraphs:

(a) In judging the availability of systems, account was taken of the postulated failure and its direct consequences such as unit trip and loss of offsite power, and of the assumed single active component failure and its direct consequences. The feasibility of carrying out operator actions was judged on the basis of ample time and adequate access to equipment being available for the proposed actions. Although a postulated high/moderate-energy line failure outside the containment may require a cold shutdown, up to 8 hours in hot standby was allowed in order for plant personnel to assess the situation and make repairs.

(b) The use of nonseismic piping outside the containment is clarified in the following paragraphs:

(1) For non-Category I piping, it was assumed that an earthquake will be the cause of the pipe failure; thus, only seismic Category I equipment could be used to mitigate the consequences of the failure and bring the plant to a safe shutdown.

(2) For seismic Category I piping, it was assumed that the failure was caused by some mechanism other than an earthquake; thus, non-Category I equipment could be used to bring the plant to a safe shutdown, subject to the power available to operate such equipment as discussed in paragraph 10.(a) above. Grand Gulf has not taken credit for the use of nonseismic equipment.

(c) The use of nonseismic piping inside containment was permitted only on a case-by-case basis with the necessary justification. The piping is analyzed for normal as well as faulted loads including “New Loads.” The computer code (ME-101) and loading combinations used in the stress analysis are the same as those used for Class 1, 2 and 3 piping. The stress levels in the piping

systems are kept below the faulted Code allowable limits. Grand Gulf has not taken credit for the use of nonseismic piping.

11. A whipping pipe was not considered capable of rupturing impacted pipes of equal or greater nominal pipe diameter and equal or greater thickness.

12. Pipe whip was assumed to occur in the plane defined by the piping geometry and to cause movement in the direction of the jet reaction.

    If unrestrained, a whipping pipe having a constant energy source was considered to form a plastic hinge and rotate about the nearest rigid restraint, anchor, or wall penetration. If unrestrained, a whipping pipe without a constant energy source (i.e., a break at a closed valve with only one side subject to pressure) was not considered capable of forming a plastic hinge and rotating provided its movement could be defined and evaluated.

13. The fluid internal energy associated with the pipe break reaction could take into account any line restrictions (e.g., flow limiter) between the pressure source and break location and absence of energy reservoirs, as applicable.

14. Protection of the reactor pressure vessel from the surface impact effects of pipe rupture was not considered due to its relative thickness.

15. Initial pipe break events were not assumed to occur in pump and valve bodies because of their greater wall thicknesses and their usual location in the low stress portions of the piping systems.

d. Approach

To comply with the objectives previously defined, systems, components, and equipment required to safely shut down the plant and mitigate the consequences of postulated piping failures (hereinafter called essential) were reviewed to determine their susceptibility to the failure effects. The

break and crack locations were determined in accordance with subsection 3.6A.2. Figures 3.6A-1A through 3.6A-13I show the high-energy pipe break locations.

Those essential items which were subject to the consequences of pipe failures are summarized in Tables 3.6A-18 through 3.6A-27. The type of hazard (i.e., whipping, jet impingement, spraying, and flooding) and the type of protection provided are also shown. This summary was based on the detailed failure mode analysis discussed in subsection 3.6A.1.3 and Appendix 3C.

3.6A.1.2 Description

Essential systems, components, and equipment, or portions thereof are provided on Table 3.6A-16 for breaks/cracks inside the containment and Table 3.6A-17 for breaks/cracks outside the containment.

A list of lines considered high energy per subsection 3.6A.2.1a are given in Tables 3.6A-14 and 3.6A-15 for inside and outside the containment, respectively.

Moderate-energy piping defined in subsection 3.6A.2.1b is not specifically listed.

Composite drawings, Figures 3.6A-14 through 3.6A-30, show the routing of high-energy piping in relation to compartments inside the containment and auxiliary building. Nearby essential items are discussed in Appendix 3C.

Pressure response analyses were performed for the subcompartments containing high-energy piping. For a detailed discussion of the line breaks selected, vent paths, room volumes, analytical methods, pressure results, etc, refer to subsection 6.2.1.2 for containment subcompartments and Appendix 3E for subcompartments located outside the containment.

The effects of pipe whip, jet impingement, spraying, and flooding on essential systems, components, and equipment are discussed in Appendix 3C, Failure Mode Analysis.

In particular, there are no high-energy lines anywhere near the control room. As such, there are no effects upon the habitability of the control room by pipe break either from pipe whip, jet impingement, or transport of steam. Further discussion on control room habitability systems is provided in Section 6.4.

3.6A.1.3 Safety Evaluation

3.6A.1.3.1 General

An analysis of pipe failures postulated in the design was performed to identify those safety-related systems, components, and equipment that provide protective actions required to mitigate, to acceptable limits, the consequences of the postulated pipe failure.

By means of the design features such as separation, barriers, and pipe whip restraints, all of which are discussed below, the effects of breaks and cracks will not damage essential items to an extent that would impair their design function or necessary component operability.

Specific design features used for protecting the essential systems, components, and equipment are discussed in detail in Appendix 3C. The ability of specific safety-related systems to withstand a single active failure concurrent with a postulated event is discussed as applicable.

If a review of the pipe layout and plant arrangement drawings showed that the effects of the postulated breaks/cracks, on a reasonable basis, are isolated, physically remote, or restrained by plant design features from essential systems or components, no further evaluation was performed.

3.6A.1.3.2 Protection Methods

a. General

   The effects associated with a particular break/crack must be mechanistically consistent with the failure. Thus, actual pipe dimensions, piping layouts, material properties, and equipment arrangements were considered in defining the specific measures for protection against actual pipe movement and other associated consequences of postulated failures.

1. Protection against the dynamic effects of pipe failures was provided in the form of pipe whip restraints, equipment shields, and physical separation of piping, equipment, and instrumentation.

2. The precise method chosen depended largely upon limitations placed on the designer such as accessibility, maintenance, and proximity to other pipes.

b. Separation

The plant arrangement provided separation to the extent practicable between redundant safety systems (including their auxiliaries) in order to prevent loss of safety function as a result of hazards different from those for which the system is required to function, as well as for the specific event for which the system is required to be functional. Separation between redundant safety systems with their related auxiliary supporting features, therefore, was the basic protective measure incorporated in the design to protect against the dynamic effects of postulated pipe failures.

In general, layout of the facility followed a multistep process to ensure adequate separation.

1. Safety-related systems were located away from most high-energy piping.

2. Redundant (e.g., A and B trains) safety subsystems were located in separate compartments.

3. As necessary, specific components were enclosed to retain the redundancy required for those systems that must function as a consequence of specific piping failure.

4. Drainage systems were reviewed to assure their adequacy for flooding prevention.

c. Barriers, Shields, and Enclosures

Protection requirements were met through the protection afforded by the walls, floors, columns, abutments, and foundations in many cases. Where adequate protection did

not already exist due to separation, additional barriers, deflectors, or shields were provided as necessary to meet the functional protection requirements.

d. Piping Restraint Protection

Measures for protection against pipe whipping as a result of high-energy pipe breaks were not provided where any one of the following applied:

1. The piping was physically separated (or isolated) from any essential safety-related structure, system, or component required to place the plant in a safe shutdown condition following the postulated rupture, or was restrained from whipping by plant design features such as concrete encasement.

2. Following a single break, the unrestrained pipe movement of either end of the ruptured pipe could not damage, to an unacceptable level, any structure, system, or component required to place the plant in a safe shutdown condition following the postulated rupture.

3. The energy associated with the whipping pipe was demonstrated to be insufficient to impair, to an unacceptable level, the safety function of any structure, system, or component required to place the plant in a safe shutdown condition following the postulated rupture.

The design criteria for restraints is given in subsection 3.6A.2.3. No exemptions from pipe whip protection requirements, as defined in paragraph 3.6B.2.1.1.4, have been claimed.

3.6A.1.3.3 Specific Protection Measures

a. Nonessential systems and system components are not required for the safe shutdown of the reactor, nor are they required for the limitation of the offsite release in the event of a pipe rupture. However, while none of this equipment is needed during or following a pipe break event, pipe whip protection was considered where a nonessential system or component failure could initiate or

escalate a pipe break event in an essential system or component, or another nonessential system whose failure could affect an essential system.

b. For piping systems which are high-energy through the containment and drywell penetrations, isolation valves were located as close to the containment as possible. This was done to facilitate design of moment guides to maintain operability of isolation valves.

c. The pressure, water level, and flow sensor instrumentation for those essential systems which are required to function following a pipe rupture were protected.

d. To minimize the potential for damage to both of the independent high-pressure automatic depressurization systems (ADS) in the event of a partial break in the pressurized portion of the HPCS, no portion of the normally pressurized HPCS piping was located within jet impingement damage distance of any component considered essential to the operation of the ADS.

e. High-energy fluid system piping restraints and protective measures were designed such that a postulated break in one pipe could not, in turn, lead to a rupture of other nearby pipes or components, if the secondary rupture could result in consequences that would be considered unacceptable for the initial postulated break.

f. Deleted

g. For any postulated pipe rupture, the structural integrity of the containment structure was maintained. In addition, for those postulated ruptures classified as a loss of reactor coolant, the design leaktightness of the containment fission product barrier was maintained.

h. Safety/relief valves and the RCIC steam line were located and restrained such that a pipe failure would not prevent depressurization.

i. Separation was provided to preserve the independence of the low pressure core spray (LPCS) and low pressure core injection (LPCI) systems.

j. High-energy piping which penetrated both the drywell and the containment was provided with guard pipes in accordance with subsection 3.6A.2.4. The encapsulated piping was designed in accordance with the criteria of subsection 3.6A.2.1(c)(2).

   Protective measures utilized to meet these requirements considered access requirements for conducting the inservice examinations specified in the ASME Boiler and Pressure Vessel Code, Section XI, "Inservice Inspection of Nuclear Power Plant Components."

k. To maintain the ability to insert the control rods in the event of a pipe break, the control rod drive (CRD) insert and withdraw lines were not allowed to be locally crimped to less than 35 percent of the original flow area.

l. The escape of steam, water, combustible or corrosive fluids, gases, and heat in the event of a pipe rupture did not preclude:

   1. Accessibility to any areas as required to cope with the postulated pipe rupture

   2. Habitability of the control room

   3. The ability of essential instrumentation, electric power supplies, components, and controls to perform their safety function.

3.6A.2 Determination of Break Locations and Dynamic Effects Associated with the Postulated Rupture of Piping

This section describes the design bases for locating postulated breaks/cracks in high-energy/moderate-energy piping inside and outside of the containment; the procedures used to define the jet thrust reaction at the break location; and the procedures used to define the jet impingement loading on adjacent essential structures, systems, or components; restraint design; and guard pipe assembly design.

3.6A.2.1 Criteria Used to Define High/Moderate Energy and Break/Crack Location and Configuration

a. Definition of High-Energy Piping

High-energy piping included those systems or portions of systems in which the maximum operating temperature exceeded 200 F or the maximum operating pressure exceeded 275 psig during normal plant conditions.

b. Definition of Moderate-Energy Piping

1. Piping systems or portions of systems pressurized above atmospheric pressure during normal plant conditions and not identified as high-energy piping were considered moderate-energy piping.

2. Piping which exceeded 200 F or 275 psig for 2 percent or less of the time the system is in operation (or if the system was exposed to pressures or temperatures higher than the above limits for less than 1 percent of plant operation) was considered as moderate-energy piping.

c. Postulation of Pipe Breaks

NRC Regulatory Guide 1.46 and Branch Technical Position MEB 3-1 of NRC Standard Review Plan, subsection 3.6.2, (Reference 6) were used as criteria for postulation of pipe breaks. The stress limits used, however, were those of Branch Technical Position MEB 3-1. This concept was presented to the NRC on January 7, 1976 (Ref. 1). All ASME Section III piping is also seismic Category I.

1. With the exception of those portions of piping identified in subsection 3.6A.2.1(c)(2), breaks were postulated only in high-energy piping at the following locations:

(a) ASME Section III - Class 1 Piping

(1) Terminal ends, including:

- Piping pressure vessel or equipment nozzle intersection
- High-energy moderate-energy boundary
- A branch intersection point was considered a terminal end for the branch line unless the branch and the main

piping systems were modeled in the same static, dynamic, and thermal analyses; the branch run is at least one-half the nominal pipe size of the main run unless smaller branch run sizes are specifically justified; and the intersection is not constrained to the building structure. This is consistent with Reference 1.

(2) At intermediate locations where the maximum stress range between any two load sets (including zero load set), according to ASME Code Section III, Subarticle NB-3600, for upset plant conditions and an independent OBE event transient, exceeded the following:

- o If the stress range calculated using equation (10) of the Code exceeded 2.4 Sm but was not greater than 3 Sm, no breaks were postulated unless the cumulative usage factor exceeded 0.1.

- o The stress ranges, as calculated by equations (12) or (13) of the Code, exceeded 2.4 Sm or if the cumulative usage factor exceeded 0.1 when equation (10) exceeded 3 Sm.

- o In piping systems where the stresses and cumulative usage factors were lower than the above limits, a minimum of two intermediate break locations were postulated on the basis of stress and fatigue levels as well as on the basis of providing adequate pipe break protection in the entire system. Additionally, where the piping consisted of a straight run (shorter than 10 pipe diameters in length) without fittings, welded attachments, or valves, a minimum of one location was chosen based on the highest stress.

(b) ASME Section III - Class 2 and 3 piping within protective structures

(1) Terminal ends, including:

- o Piping pressure vessel or equipment nozzle intersection
- o High-energy moderate-energy boundary
- o A branch intersection point was considered a terminal end for the branch line unless the branch and the main piping systems were modeled in the same static, dynamic, and thermal analyses; the branch run is at least one-half the nominal pipe size of the main run unless smaller branch run sizes are specifically justified; and the intersection is not constrained to the building structure. This is consistent with Reference 1.

(2) At intermediate locations between terminal ends where the maximum stress ranges as calculated by the sum of equations (9) and (10) in Subarticle NC-3652, ASME Code, Section III, considering normal and upset plant conditions (i.e., sustained loads, occasional loads, and thermal expansion) and an OBE event exceeded 0.8 (1.2Sh + SA).

(3) In piping systems where the stresses were lower than the above limits, a minimum of two intermediate break locations were postulated on the basis of stress levels as well as on the basis of providing adequate pipe break protection in the entire system. Additionally, where the piping consisted of a straight run (shorter than 10 pipe diameters in length)

without fittings, welded attachments, or valves, a minimum of one location was chosen based on the highest stress.

(c) ASME Section III - Class 2 and 3 piping not enclosed within protective structures

Breaks were postulated as stated in paragraph (b) above in each piping and branch run outside but routed adjacent to a protective structure or compartment containing essential systems and components required for safe shutdown. (Such piping was considered as located adjacent to a protective structure if the distance between the piping and structure was insufficient to preclude impairment of the integrity of the structure from the effects of a postulated piping failure, assuming the piping is unrestrained.)

(d) Non-Nuclear Piping (Both Seismic and Nonseismic)

Breaks in non-nuclear piping were postulated in each run or branch run at worst-case locations based on location of essential equipment. The selection of worst-case locations was made by performing room-by-room examination of essential equipment, evaluating the effects from the worst break/crack location.

2. Breaks were not postulated in portions of Class 1 and 2 piping between containment isolation valves up to and including the valves and adjacent pipe whip restraints (when required), provided subject piping met the following provisions:

(a) Stresses did not exceed those specified in subsection 3.6A.2.1c.1.

(b) Cumulative usage factors did not exceed those specified in subsection 3.6A.2.1c.1 (applicable to Class 1 piping only).

(c) The maximum stress in the "no break zone" shall not exceed the following limits when subjected to the combined loadings of internal pressure, deadweight, and pipe break beyond the no break zone:

(1) For Class 1 piping, the stress per equation (9), subarticle NB-3652 of ASME Section III shall not exceed 2.25 Sm.

(2) For Class 2 piping, the stress per equation (9), subarticle NC-3652 of ASME Section III shall not exceed 1.8 Sh.

(d) The number of circumferential and longitudinal piping welds and branch connections were minimized. (Where guard pipes are used, the enclosed portion of fluid system piping is seamless. Access provisions were made to permit inservice volumetric examination of the circumferential weld in the process piping.) Refer to subsection 3.6A.2.4 for the design requirements of guard pipe.

(e) When required for isolation valve operability, structural integrity, or the containment integrity, whip restraints capable of resisting torsional and bending moments produced by a postulated pipe break either upstream or downstream of the containment isolation valves were located reasonably close to the isolation valves.

However, the restraints on the outside of the containment were not located so close as to reduce the optimum design demonstrated to the NRC in late 1975 in a series of meetings for the auxiliary building steam tunnel design (Reference 2).

In summary, it was shown that the length of the tunnel, location of moment guides in the east wall of the auxiliary building, and whip restraints just outside the tunnel wall and

floor penetrations for connecting piping routed to other areas of the auxiliary building, resulted in low stresses when considering:

(1) SSE differential building movements

(2) Space requirements for isolation valves, guard pipes, flued heads, and other piping and components

(3) Minimum space for maintenance

(4) Maximum accessibility for inservice inspections performed every inspection interval

(f) Breaks were postulated just beyond the break exclusion boundaries.

(g) Inservice examinations

Preservice and subsequent inservice examinations are performed in accordance with the requirements specified in ASME Code Section XI. In addition, during each inspection interval, volumetric examination of nonexempt ASME Code Section XI, Class 1 and 2 circumferential butt welds greater than 4 inch NPS will be performed on those portions of high-energy fluid system piping between the first moment-limiting restraint outside containment and the first moment-limiting restraint inside containment in accordance with the risk-informed methodology described in EPRI Report 1006937 incorporated through GGNS-PS-06-0002. This area of interest is referred to as the break exclusion region. Examinations shall be performed in accordance with ASME Section XI, except where specific written relief has been granted by the Commission pursuant to 10 CFR 50, Section 50.55a(g)(6)(i).

(h) The restraints do not prevent the access required to conduct inservice inspection examinations specified in Section XI of the ASME Code.

(i) Welded attachments, for pipe supports or other purposes, to these portions of piping were avoided except where detailed stress analyses could be performed to demonstrate compliance with the limits of subsection 3.6A.2.lc.1.

3. The following types of breaks were postulated to occur at the locations determined in accordance with subsections 3.6A.2.lc.1 and 3.6A.2.lc.2:

(a) ASME Section III piping

(1) Breaks were not postulated in CRD piping** 1-1/4-inch NPS and smaller and all other piping 1-inch NPS and smaller.

(2) At terminal ends, only circumferential breaks were postulated.

(3) At intermediate locations where both the stress and usage factor were less than the limits of subsection 3.6A.2.lc.1, only circumferential breaks were postulated.

(4) At intermediate locations where the stress and/or usage factor exceeded the limits of subsection 3.6A.2.lc.1, the following type of breaks were postulated:

- Circumferential breaks for pipe sizes less than 4-inch NPS, but greater than the size exemption stated in (1) above.

- Circumferential and longitudinal for pipe sizes 4-inch NPS and larger.

(b) For non-nuclear piping the following combination of breaks were evaluated:

(1) Circumferential breaks in piping larger than 1-inch NPS.

**The NRC has granted exclusion of pipe whip considerations, specifically to the CRD insert and withdrawal (1 and 1-1/4 inch) lines. For reference, see pages 3 and 4 of the Safety Evaluation Report for the design of GESSAR-238, NSSS (Docket No. STN 50-550).

(2) Longitudinal breaks in piping larger than 4-inches NPS.

(c) Instrument lines 1 inch and less nominal pipe or tubing size meet the provisions of Regulatory Guide 1.11.

4. Break Configuration

(a) The ends of a circumferentially ruptured pipe were assumed to be displaced laterally by a distance equal to one pipe diameter relative to each other unless both ends were restrained by close gap restraints limiting movement in the lateral direction so that one broken end partially blocked flow from the other broken end, as permitted by paragraph B.3.a(4) of MEB 3-1. In the restrained end case, advantage could be taken of the smaller flow area in calculation of the jet force. For the unrestrained case, the effective cross-sectional (inside diameter) flow area of the pipe may be used in the jet discharge evaluation. Movement was assumed to be in the direction of the jet reaction initially, and total path controlled by the piping geometry.

(b) The orientation of a longitudinal break, except when otherwise justified by a detailed stress analysis, was considered to cause piping movement normal to the plane of the piping system. The flow area of such a break was equal to the cross-sectional flow area of the pipe.

(c) Longitudinal and circumferential breaks were not postulated concurrently.

(d) Postulation of Cracks

(1) Through-wall leakage cracks were postulated at worst-case locations (the selection of worst-case locations was made by performing room-by-room examination of essential equipment, evaluating the effects from the worst break/crack location) in moderate-energy piping larger than 1-inch NPS located within, or outside and adjacent to, protective structures except as noted in the following:

- Where the maximum stress range in seismic Category I, Class 2 or 3 piping (ASME Code, Section III), or nonnuclear piping was less than 0.4 $(1.2S_h + S_A)$.*
- In those portions of seismic Category I piping between containment isolation valves provided they met the requirements of ASME Code, Section III, Subarticle NE-1120, and were designed such that the maximum stress range did not exceed 0.4 (1.2Sh + SA)* for ASME Code, Section III Class 2 piping.
- Cracks were not postulated in moderate-energy fluid system piping located in an area in which a break in high-energy fluid system was postulated, provided such cracks would not result in more limiting environmental conditions than the high-energy piping break.

*Where Sh is the basic material allowable stress at design temperature, and SA is the allowable stress range for expansion stresses. The allowable stress range for expansion stresses is the maximum absolute difference between stresses calculated due to different loading combinations. SA is calculated in accordance with ASME Section III, paragraph NC-3611.2.

Cracks were not postulated when a review of the piping layout and plant arrangement drawings showed that the effects of through-wall leakage cracks at any location in piping designed to seismic or nonseismic standards were isolated or physically remote from structures, systems, and components required for safe shutdown.

(2) The cracks were postulated to occur individually at locations that resulted in the maximum effects from fluid spraying and flooding, with the consequent hazards or environmental conditions. Flooding effects were determined on the basis of a conservatively estimated time period required to effect corrective actions.

(3) The crack opening was assumed to be a circular orifice of cross-sectional flow area equal to one-half the pipe inside diameter times one-half the pipe wall thickness.

(4) Instrument lines 1 inch and less nominal pipe or tubing size shall meet the provisions of Regulatory Guide 1.11.

(5) Plant conditions for postulation of leakage cracks are identical to those used for postulation of circumferential and longitudinal breaks. In addition, leakage cracks were postulated in accordance with paragraphs (1) through (4) above in systems which occasionally operate at high-energy conditions, but are classified as moderate-energy systems, as stated in subsection 3.6A.2.1.b.2 and in footnote 6 of paragraph B.2.e of MEB 3-1.

(6) In moderate-energy Class 1 piping (ASME Code Section III), leakage cracks were postulated at the break locations

determined in accordance with subsection 3.6A.2.1.c.1(a). No crack exclusion options were used.

3.6A.2.2 Analytical Methods

The methods used to define the forcing functions for the pipe break whip dynamic analyses were in accordance with Bechtel Topical Report BN-TOP-2, Revision 2, approved for use by the NRC in a letter dated June 17, 1974. This report describes the direction, thrust coefficients, rise time, magnitude, duration, and rebound effects. The pressures and temperatures used were obtained from the normal operating conditions of the particular systems being evaluated.

The jet impingement effects were calculated using the methods of Section 2.3 of BN-TOP-2, Revision 2, and ANSI 58.2. If the ability of the target component to perform an essential function could not be assured as a result of the location (including separation), either the ruptured pipe was restrained to redirect the jet in a harmless direction, or a suitable barrier was installed to safely deflect the jet. The effect of wetting the target (as opposed to the dynamic effect of the jet) was treated as part of the flooding evaluation, as discussed in Section 3C.4 of Appendix 3C.

3.6A.2.3 Location, Design, and Analysis of Restraints

The analytical methods of BN-TOP-2, Revision 2, and ANSI 58.2 were used to determine the jet impingement forces resulting from the circumferential and longitudinal breaks postulated in subsection 3.6A.2.1. This information was then used in the protection evaluation described in subsection 3.6A.1.3. Refer to subsection 3.9.3.1.2 for a more detailed discussion of piping system evaluation exposed to high energy pipe breaks.

This section describes the design of restraints used to protect the essential systems, components, and equipment from the effects of pipe whip.

3.6A.2.3.1 Location of Restraints

Restraints were located as close to the postulated break as practical. Restraints located closer than a lower-bound distance (defined below) required no further evaluation, as the pipe whip is limited by the rigidity of the piping, with stresses below yield. Restraints located at a distance between that lower-bound

value and an upper-bound value (also defined below) would result in formation of a plastic hinge at the restraint, with bending stresses at the hinge exceeding the yield stress of the piping; piping thus restrained required evaluation to assure that any essential equipment located within the area swept by the whipping pipe was adequately protected from unacceptable damage.
Restraints located beyond this upper-bound distance from the break location would permit the pipe to collapse at the restraint and form a 180˚ hinge unless otherwise prevented.

The lower-bound distance is based on an elastic-perfectly plastic material property, using minimum yield stress as given in the ASME Code. This distance is determined from the equation:

$$L_1 = \frac{Z_p S_y}{\text{Jet Force}}$$

where

L_1 = Lower-bound distance, inches

Z_p = Plastic section modulus,

inches S_y = Yield stress, psi

Jet Force = Operating pressure (psi) x flow area (inches^2)
x thrust coefficient (dimensionless)

The upper-bound distance is based on the collapse moment of the pipe, including nonlinear work hardening and ovalization of the pipe cross-section (Ref. 4), and on the ultimate strength of the material as given in the ASME Code. This distance is determined from the equation:

$$L_2 = \frac{1.07 \times S_u}{\text{Jet Force}} \quad \frac{R^{3.14} - r^{3.14}}{R^{0.14}} \qquad \text{for carbon steel}$$

$$L_2 = \frac{1.0 \times S_u}{\text{Jet Force}} \quad \frac{R^{3.18} - r^{3.18}}{R^{0.18}} \qquad \text{for stainless steel}$$

where

L_2 = Upper-bound distance, inches

S_u = Ultimate stress, psi

R = Pipe outside diameter, inches

r = Pipe inside diameter, inches

Jet Force as for L_1, above

3.6A.2.3.2 Types of Restraints

To satisfy varying requirements of available space, permissible pipe deflection, equipment operability, etc., four general designs of restraints have been used. These include two varieties of close-gap restraints and two of large-gap restraints.

3.6A.2.3.2.1 Close-Gap Restraints

These restraints were installed with small gaps (clearances) between the pipe and the restraint. One variety, as typified by Figure 3.6A-34, is a moment guide, which limits the pipe movement in the transverse direction, for some applications in the axial direction, and rotation about any axis (i.e., bending moments or torsional movements) at the restraint. The other variety of close-gap restraint, typified by Figure 3.6A-35, is simpler; it would normally limit movement in the transverse direction. These close-gap restraints were used when it was necessary to minimize deflections and stresses which could impair operability of valves, endanger the integrity of containment penetrations, or otherwise justify their detrimental effect on accessibility.

3.6A.2.3.2.2 Large-Gap Restraints

One type of large-gap restraints is designed by General Electric-Hitachi (GEH); the other uses energy-absorbing honeycomb material.

The GEH U-bolt type of restraint was used where the piping was quite flexible, i.e., large piping deflection could be tolerated without causing rupture in the piping elsewhere. This restraint let the pipe move 9 to 12 inches if the jet force was along the axis of the restraint, and 25 to 36 inches if the jet force was transverse to the axis of the restraint (thus plastically deforming the restraint's structural member, absorbing and dissipating energy). Both of these movements were based on

roughly a 4-inch initial clearance. An obvious disadvantage to the use of this type of restraint is when the jet force is in the transverse direction to the restraint axis. The advantages were noninterference in the normal operation of the plant, and easier accessibility for inservice inspection (as the U-bolts can be removed). See Figures 3.6A-38, 39, 40 and 41.

The energy dissipating type of restraint was used when the GE restraints could not be accommodated for clearance reasons or could not be oriented such that the jet force was directed along the axis of the restraint. Basically, this type of restraint consisted of substructures which remained elastic during an impact, with an energy dissipating material (stainless steel honeycomb) in between the pipe and this substructure. See Figures 3.6A-36 and -37.

3.6A.2.3.3 Analytical Methods

Any of four methods was used to determine the response of the piping/restraint/supporting structure to the jet thrust developed by the postulated piping failure. These methods were:

a. Collapse moment of ruptured pipe

b. Energy balance

c. Dynamic (time history)

d. Jet thrust with dynamic load factor of 2

Choice of method used was based on the required accuracy of the results, all with adequate conservatism. All methods addressed, to the degree appropriate, the following effects:

a. Stiffness characteristics of the piping system, restraint system, major components, and supporting walls and structures

b. Transient forcing functions acting on the piping system, and jet thrusts on structures

c. Elastic and inelastic deformation of piping and/or restraints

d. Insulation thickness (applicable to large-gap restraints only)

e. Seismic and thermal movements (for determination of clearance values)

Energy balance or dynamic analyses were not performed on an end of failed piping which does not connect to a fluid energy reservoir having the capacity of developing a jet stream. Such ends were statically analyzed to determine their response to the impact of the jet produced by the high-energy side of the pipe break.

3.6A.2.3.3.6 Collapse Moment Method

This is a simple but very conservative method, based on the premise that the forces and moments which the restraint must accept can be no greater than the pipe can apply. The jet force is applied at the pipe break; multiplied by the distance between the break and the restraint; and the bending moment at the restraint is found. This moment cannot be greater than that which would produce bending stresses equal to the ultimate stress of the pipe material. The restraint is then designed, with appropriate safety factors, to accept this collapse moment. (Refer also to the equation for L in subsection 3.6A.2.3.2.) This method was used primarily for the moment guide type of close-gap restraint.

3.6A.2.3.3.7 Energy Balance Method

This method of analysis, while less accurate than the dynamic analysis method, is far less complicated and time-consuming. The results are very conservative; it is used for applications where space and other limitations permit the greater conservatism and the considerably higher cost of the dynamic analysis is not justified. Specifically, it was used for the design and evaluation of the large-gap restraints. The method is described below:

a. Forcing Function - obtained per BN-TOP-2, Revision 2.

b. Resistance Response of Piping System - the resistance of piping system (load-deflection response) was determined by piping flexibility program ME101 (described in FSAR Section 3.7) or by hand calculation. This was achieved by a static analysis (by inputting the force at the postulated pipe break locations). The displacement obtained for a corresponding force gave the force-deflection response of the piping system in the elastic range. A perfectly plastic response for the piping system was assumed when the intensified stress (due to the stress

intensification factor of the fitting) at the first elbow beyond the pipe whip restraint reached yield stress of the material.

c. Restraint Response:

GEH Restraints - the analytical procedures used to obtain the response (load-deflection) of the U-bolt GEH restraint is outlined in Reference 5. The load-deflection response was calculated on the basis of one turn of the U-bolt. If o was the angle the U-bolt and pipe arc of contact subtended at the center of pipe, then the effective clearance, load-deflection relation was given by equations (1) and (2), respectively (Figure 3.6A-43).

$$C = C_1 + C_2$$

$$= \left(L_1 + \frac{\pi}{2} R_1 - R_2 \varnothing_o\right) \sin\varnothing_o - R_2 \cos\varnothing_o - L_1 \qquad (1)$$

$$Y - Y_g = \left(\frac{R_r}{2AK}\right)^{\frac{1}{n}} \frac{\left[L_2 + \frac{\pi}{2} R_1 - R_2\varnothing_o + R_2\left(\frac{n}{f}\right)\left(1 - e^{\frac{-f\varnothing_o}{n}}\right)\right]}{\sin\varnothing_o\left(\frac{1-n}{n}\right)} \qquad (2)$$

where

C = Total effective clearance, inches

C_1 = Initial clearance, inches

C_2 = Additional clearance, inches

Y = Pipe deflection, inches

Y_g = Gap, inches

R_r = Load, lbs

A = Area of cross-section of the U-rod, square inches

R_1 = Radius from the center of pipe to inside of the U-bolt, inches

R_2 = Outer radius of the pipe, inches

$\varnothing_o$ = Half of the contact angle, radians

f = Coefficient of friction

L_2 = Rod length from end of clevis to centerline of the pipe, inches

n = Nonlinear work hardening index

K = Coefficient in nonlinear stress-strain relation

The nonlinear portion of the restraint response was approximated by a linear curve as shown in the Figure 3.6A-43.

d. Energy Balance and Load Transmitted to the Structure

1. GEH Restraint - as shown in Figure 3.6A-43 and in accordance with BN-TOP-2, the area under the forcing function curve was equated to the areas under the restraint resistance and piping resistance curves, and an inelastic collision was assumed to get the following equations (Refer to Figure 3.6A-43):

$$\left(\frac{M_p}{M_p + M_r}\right) F Y_g + F(Y - Y_g) = \left(\frac{M_p}{M_p + M_r}\right) R_p \left(Y_g - \frac{Y_p}{2}\right)$$

$$+ R_p (Y - Y_g) + \left\{R_{or}(Y - Y_g) - \frac{R_{or}}{2}(Y_{or} - Y_g) + (R_r - R_{or})\left(\frac{Y - Y_{or}}{2}\right)\right\} N$$

or

$$N = \frac{\left(\frac{M_p}{M_p + M_r}\right)\left\{F Y_g - R_p\left(Y_g - \frac{Y_p}{2}\right)\right\} + F(Y - Y_g) - R_p(Y - Y_g)}{R_{or}\left(\frac{Y - Y_g}{2}\right) + R_r\left(\frac{Y - Y_{or}}{2}\right)}$$

where

M_p = mass of the pipe associated with impact

M_r = mass of the restraint

N = number of U-loops of GE Restraint

F = forcing function, lb

Y = pipe displacement, total, inches

Y_g = pipe displacement, rest-to-restraint, inches

Y_{or} = pipe displacement, at end of linear range, inches

R_p = pipe resistance, lb.

R_{or} = pipe resistance at Y = Y_{or}, lb

The variables Y and R_r were iterated to obtain an integer for N. As a result, the load transmitted to the structure was equal to K x N x R_r lbs, when K represented the amplification factor to include any possible rebound effects on the structure. The appropriate values of K were selected from BN-TOP-2. The design procedure used limited the maximum strain in the GE U-bolt to 0.25 of the ultimate uniform strain.

2. Energy Dissipating Restraint (Non-GE) - this was basically an energy dissipating material which was supported by a substructure. This substructure was designed to remain elastic during the impact. All the energy is absorbed and dissipated by the collapse of this crushable honeycomb core. The properties as a function of cell size and web size of the honeycomb core were obtained from the manufacturer of the specific material used.

   The restraint resistance R_r was determined from equation (1) below. As shown in Figures 3.6A-36 and 3.6A-37, let L_a, L_b, and t be the length in the axial direction of the pipe, the length along the transverse direction of the pipe (perpendicular to jet force), and the thickness in the direction of the jet force and P_c is the crushing stress of the material. Referring to Figure 3.6A-43:

$$FY = R_r(Y - Y_g) + \left(R_p \frac{Y_p}{2}\right) + R_p(Y - Y_p)$$

$$R_r = \frac{FY - R_p\left(Y - \frac{Y_p}{2}\right)}{Y - Y_g}$$

$$L_a L_b P_c \geq R_r \quad \text{and} \quad et \geq (Y - Y_g)$$

where e is the maximum strain allowed.

For a suitable value of P_c, L_aL_b was determined from equation (2). The exact dimension of L_a or L_b was determined depending on spatial considerations. Where crushable honeycomb energy absorbing material is used, the material will not experience a deflection in excess of that which is defined by the horizontal portion of its load deflection curve, under designed loads.

3.6A.2.3.3.5 Dynamic Analysis Method

BN-TOP-2, Revision 2 (Ref. 3), describes the time history dynamic analysis. This analysis consists of a realistic forcing function (jet force history) and geometrical (gaps) as well as material (inelastic deformations) nonlinearities. This is the most accurate, but also the most time-consuming, of the four methods used. It was used primarily when it was necessary to prove operability of valves or other essential items, and when the extreme conservatism of the simpler methods of analysis were impractical. Computer programs ANSYS or PIPERUP (CDC) which are available to the public were used to determine the adequacy of the supporting structure.

3.6A.2.3.3.6 Jet Thrust with Load Factor

For some close-gap restraints, this simplified method was used. Briefly, the force on the restraint was taken as equal to the jet thrust (pressure x area x thrust coefficient) multiplied by a dynamic load factor. This load factor was conservatively assumed to be 2, the largest possible for a restraint which was virtually in contact with the pipe. (If the clearance between pipe and

restraint was large enough to permit the whipping pipe to attain significant velocity before contacting the restraint, thus causing impact effects, other analytical methods were used.)

3.6A.2.4 Guard Pipe Assembly Design

3.6A.2.4.1 General

Piping which normally carries high-energy fluid and passes through both the drywell and the containment was provided with guard pipes in accordance with NRC Branch Technical Positions APCSB 3-1, paragraph B.2c(2) and MEB 3-1 paragraph B.1.b(6).

Although these lines were designed to comply with the "break exclusion requirements" described in subsection 3.6A.2.1(c)(2), through-wall leakage cracks were postulated. The crack requirement resulted from meetings with the ACRS and NRC (AEC at that time) prior to issuance of the construction permit.

The guard pipes would protect the containment from being overpressurized by such a crack and vent the fluid discharge back into the drywell, designed for such an event.

The following lines were provided with guard pipes:

4 - 28-inch diameter main steam lines

2 - 24-inch diameter feedwater lines

1 - 20-inch diameter RHR shutdown-suction line

1 - 10-inch diameter steam supply line to RCIC turbine

1 - 6-inch diameter RHR to head spray line

1 - 6-inch diameter reactor water cleanup pump suction

1 - 3-inch diameter main steam drain line to condenser

The 20-inch RHR shutdown suction line was classified as high-energy at the design stage; since then it has been reclassified as moderate-energy, as the system experiences the high-energy condition less than two percent of the operating life. A typical guard pipe assembly consisted of a flued-head, embedded sleeves

at containment and drywell walls, a bellows assembly unit at the drywell wall, guides, and large diameter interconnecting guard pipes assembled as shown in Figure 3.6A-33.

3.6A.2.4.2 Design Criteria

ASME Code Section III requirements, including the design criteria for the process pipe, flued-head, guard pipe, and bellows assembly unit are shown in Table 3.6A-28.

Code paragraph references in Table 3.6A-28 are in accordance with the code in effect at time of contract date as follows:

Component	Applicable Code*
Flued-head design	1974 Edition; including addenda through summer 1974 (For GEH supplied)
	1971 Edition; including addenda through summer 1973 (For Bechtel supplied)
Bellows design	1974 Edition
Process pipe and guard pipe design	1974 Edition

* Any code cases used are listed in FSAR Table 3.9-28.

Field fabrication/ installation	1974 Edition
Piping stress analysis for guard pipe and process pipe	1974 Edition, including summer 1975

3.6A.2.4.3 Design Considerations

a. Bellows Assembly

The bellows assembly consisted of two 2-ply bellows connected by a spool piece that provided adequate flexibility for lateral movements.

The bellows assembly was designed to absorb the following movements:

1. The relative seismic movements between the containment penetration and the drywell penetration

2. The relative thermal movements between the containment penetration and the drywell penetration under the worst condition which is a LOCA inside drywell and the coldest temperature of the containment wall

3. The lateral and axial thermal growth of the guard pipe at a temperature equal to the design temperature of the process pipe

4. The lateral seismic displacement of the guard pipe

5. The safety margin to account for tolerances required during fabrication and installation

b. Inservice Inspection (ISI) Ports

The circumferential weld (joining the flued-head and process pipe) encapsulated by the portion of the guard pipe which protrudes beyond the containment will be volumetrically examined once every 10-year interval in accordance with ASME Section XI (Refer to subsection 5.2.4 for details on inservice inspection requirements).

Two 4 x 6-inch elliptical access ports spaced 180 degrees apart shall be provided for this examination. The inner cover of the access port seats against the guard pipe when the guard pipe is pressurized. A weld is located between the inner cover and the guard pipe to provide part of the Containment boundary.

c. Guides

Internal guides were provided for the guard pipes as follows:

1. Guide No. 1, shown in Figure 3.6A-33, will function as a jet deflector and seismic restraint limiting movement in lateral directions. This guide was

located between the drywell sleeve and the guard pipe. The open bypass area did not exceed one square foot. This limitation on the open bypass area will protect the containment against overpressurization in the event of a postulated bellows failure.

2. Guide No. 2, shown in Figure 3.6A-33, will function as a seismic restraint limiting movement in lateral directions as required by analysis.

3. Guide No. 3 will be provided as required by stress analysis to act solely as a seismic restraint which limits lateral movements.

4. Restraints were provided outside the no break zone as required by pipe break analysis.

d. Pipe Break Effects

All process pipes encapsulated by guard pipe were fabricated from seamless carbon steel, SA-106 Grade B or C material, to eliminate longitudinal welds.

The through-wall crack loads for guard pipe design were calculated considering a postulated non-mechanistic break area equal to the cross-sectional area of the process pipe. These loads were added in the appropriate loading condition shown in Table 3.6A-28. Jet impingement and whip loads were not applicable.

e. Installation and Testing of the Guard Pipe Assembly

1. The guard pipe components were installed in accordance with the ASME Code subsections listed in Table 3.6A-28.

2. After installation the guard pipe assembly was hydrostatically tested to a minimum pressure equal to the process pipe operating pressure.

3. The portion of the guard pipe between the drywell wall and the containment was provided with a radial ring with a projection. The penetration bellows assembly was welded in the field to this ring.

4. In order to facilitate erection, the guard pipe between the containment and the drywell was shipped in two pieces except for main steam guard pipe which was shipped in three pieces.

5. The penetration bellows assembly was shipped with butt weld end preparation for field welds as shown on Figure 3.6A-33. The penetration bellows assembly was shop tested as per ASME Code Section III requirements.

6. After erection, the penetration bellows assembly and drywell wall s1eeve were tested with the drywell structure.

7. The process pipe inside the guard pipe assembly was hydrostatically tested with its system hydro test.

f. Independent Design Evaluation

[HISTORICAL INFORMATION] [The design of the guard pipe assembly is the responsibility of Bechtel Power Corporation's Gaithersburg Power Division.] An independent design evaluation was performed by a separate power division of Bechtel Power Corporation. The design was found to be acceptable and in accordance with the licensing commitments.

g. Guard Pipe Design Pressure

The guard pipe design pressure equaled the maximum operating pressure of the process pipe.

3.6A.2.5 Material to be Submitted for the Operating License Review

Pipe break and crack locations were obtained in accordance with the criteria of subsection 3.6A.2.1. High-energy piping with break locations identified are provided in isometric drawings, Figures 3.6A-1A through 3.6A-13I. Break types are also shown (i.e. circumferential or longitudinal). High-energy piping composites, Figures 3.6A-14 through 30, have been provided to graphically show the pipes in relation to the rooms, equipment, and other piping. Documents which contain the stress results and stress nodes are referenced on the appropriate UFSAR Figure. The referenced documents are maintained in accordance with

engineering administrative procedures. The location of cracks is adequately defined in the failure effects evaluation, Appendix 3C.

The augmented inservice inspection plan is discussed in subsection 6.6. Pipe whip restraints were designed as discussed in subsection 3.6A.2.3. The restraint locations and orientation are shown on Figures 3.6A-1A through 3.6A-13I. (The same figures depicting the break locations.) Jet thrust and impingement forces were determined in accordance with subsections 3.6A.2.2 and 3.6A.2.3.

The effects of breaks and cracks are discussed in detail in Appendix 3C, Failure Mode Analysis. The results of this appendix were based on the protection evaluation criteria of subsection 3.6A.1. Any protective measures to assure a safe shutdown (i.e., barriers, separation, and restraints) are also discussed.

The guard pipe final design is given in subsection 3.6A.2.4 and associated Table 3.6A-28 and Figure 3.6A-33.

3.6A.2.6 References

1. Ashley, R.L., "Meeting on MEB-1 Stress Limits, January 7, 1976," Bechtel letter to R.R. Maccary of the NRC, February 18, 1976.

2. Ashley, R.L., "Grand Gulf Auxiliary Steam Tunnel Design" Meeting March 17, 1976, Bechtel letter to R.R. Maccary of the NRC, April 6, 1976.

3. Bechtel Topical Report BN-TOP-2, Revision 2.

4. Gerber, T.L., "Plastic Deformation of Piping Due to Pipe-Whip Loading", ASME Publication 74-NE-1.

5. Esswein, Gene A., "Development of a Plastic Strain Energy Absorbing Pipe Whip Restraint Design", Paper presented in Structural Mechanics Conference, Chicago, December 1973.

6. NRC Standard Review Plans 3.6.2 (Nov. 1975).

7. NRC Branch topical positions MEB 3-1 (Nov. 1975) and APCSB 3-1 (Nov. 1975).

TABLE 3.6A-1: DELETED

TABLE 3.6A-2: DELETED

TABLE 3.6A-3: DELETED

TABLE 3.6A-4: DELETED

TABLE 3.6A-5: DELETED

TABLE 3.6A-6: DELETED

TABLE 3.6A-7: DELETED

TABLE 3.6A-8: DELETED

TABLE 3.6A-9: DELETED

TABLE 3.6A-10: DELETED

TABLE 3.6A-11: DELETED

TABLE 3.6A-12: DELETED

TABLE 3.6A-13A: DELETED

TABLE 3.6A-13B: DELETED

TABLE 3.6A-14: HIGH-ENERGY PIPING INSIDE CONTAINMENT*

Piping Description
Main Steam
Main Steam drains
Steam Supply to RCIC and RHR
Feedwater
Recirculation
Reactor Water Cleanup
HPCS (RPV to first check valve)
LPCS (RPV to first check valve)
CRD (Drive water return only)**
RHR (Suction from recirc "B" to first normally closed gate valve)
Standby Liquid Control (RPV to first check valve)
Reactor Drains (RPV to first closed valves)
Miscellaneous 3-inch and smaller piping***

* Refer to Figures 3.6A-1A through 3.6A-13I and Appendix 3C for further definition of boundaries.

** The 3-inch drive water return connecting to the RPV has been deleted and the RPV nozzle capped.

*** Refer to subsection 3C.2.9 of Appendix 3C for specific piping and their analyses.

TABLE 3.6A-15: HIGH ENERGY PIPING OUTSIDE CONTAINMENT*

Piping Description
Main Steam
Main Steam Drains
Main Steam Isolation Valve Leakage Control
Steam supply to RCIC Turbine and RHR (up to the blind fitting)
Feedwater
Feedwater Isolation Valve
Leakage Control
CRD (Drive water return only)
RHR (injection to feedwater from nearest check valves in the RHR lines)
Reactor Water Cleanup to Feedwater via RHR
Reactor Water Cleanup (pumps suction and discharge)

* Refer to Figures 3.6A-1A thru 3.6A-13I for further definition of boundaries.

TABLE 3.6A-16: ESSENTIAL SYSTEMS/COMPONENTS/EQUIPMENT *** EVALUATED FOR PIPE FAILURES INSIDE CONTAINMENT

A. Reactor Coolant Pressure Boundary (up to and including the outboard isolation valves)

B. Containment Isolation System and Containment Boundary (including liner plate)

C. Reactor Protection System (SCRAM SIGNALS)

D. Emergency Core Cooling Systems** (For LOCA only)

One of the following combinations is available:

1) HPCS + LPCS + LPCI A

2) HPCS + LPCI B + LPCI C

3) LPCS + LPCI A + LPCI B + LPCI C + ADS****

E. Core Cooling Systems (other than LOCA)

1) HPCS or RCIC

2) One LPCI or LPCS

3) RHR Shutdown cooling mode (one loop)

4) RHR Suppression Pool Cooling Mode (one loop)

F. Control Rod Drive (scram/rod insertion portion only)

G. Flow Restrictors (passive)

H. Combustible Gas Control (for LOCA only)

I. Suppression Pool Makeup (POST LOCA)

J. Standby Gas Treatment* (for LOCA only)

K. Control Room Environmental*

L. Spent Fuel Pool Cooling*

M. The following equipment/systems, or portions thereof, required to assure the proper operation of those essential items listed in A thru K.

1) Class IE electrical systems, ac and dc (including diesel generator system*, 4160 and 600v. emergency buses*, motor control centers*, switchgear*, batteries* and distribution systems.)

TABLE 3.6A-16: ESSENTIAL SYSTEMS/COMPONENTS/EQUIPMENT * EVALUATED FOR PIPE FAILURES INSIDE CONTAINMENT (CONTINUED)**

2) Standby Service Water* to the following:

a) Room Coolers

b) Pumps Coolers (Motors and seals)

c) Diesel Generator Jacket Coolers

d) Electrical Switchgear Coolers

3) Environmental Systems* (HVAC)

4) Instrumentation (including Post LOCA monitoring)

N. Standby Liquid Control (ATWS/SEISMIC event and DBA LOCA)

*Located outside containment but listed for completeness of essential shutdown requirements.

**Reference Section 6.3 for detailed discussion of emergency core cooling capabilities.

***The essential items listed in this table are protected in accordance with subsection 3.6A.1 consistent with the particular pipe break evaluated.

****ADS is only required for breaks with less than 0.7 ft^2 of flow area.

TABLE 3.6A-17: ESSENTIAL SYSTEMS/COMPONENTS/EQUIPMENT * EVALUATED FOR PIPE FAILURES OUTSIDE CONTAINMENT

A. Containment Isolation System and Containment boundary

B. Reactor Protection System (SCRAM signals)

C. Core Cooling Systems

1) HPCS or RCIC

2) one LPCI or LPCS

3) RHR shutdown cooling mode (one loop)

4) RHR Suppression Pool Cooling Mode (one loop)

D. Flow Restrictors

E. Control Room Habitability

F. Spent Fuel Pool Cooling

G. Standby Gas Treatment

H. The following equipment/systems, or portions thereof, required to assure the proper operation of those essential items listed in A through G.

1. Class IE electrical systems, ac and dc (including diesel generator system, 4160 and 600V. emergency buses, motor control centers, switchgear, batteries, auxiliary shutdown control panel, and distribution systems)

2. Standby Service Water to the following:

a) Room Coolers

b) Pump Coolers (motors and seals)

c) Diesel Generator Jacket Coolers

d) Electrical Switchgear Coolers

3. HVAC

4. Instrumentation (including post accident monitoring.)

*The essential items listed in this table are protected in accordance with subsection 3.6A.1 consistent with the particular pipe break evaluated.

TABLE 3.6A-18: SUMMARY OF HIGH-ENERGY PIPING FAILURE ANALYSIS

Main Steam Piping
(For explanatory notes, refer to Table 3.6A-27A)

Pipe Function	Size, Inches	Failure Location	Type of Failure	Essential[9] Equipment	Protection	Remarks	Hazard[1]	Appendix 3C Consequence
Main steam inside drywell (Reference Figures 3.6-1A and B)	28	RPV nozzles 1A, 1B, 1C, 1D	Circum	None	Note 2 Note 3 Note 4 Whip restraints	Ref. Figs. 3.6A-1A 3.6A-1B	A B C D	3C.2.1 3C.2.1 Note 5 3C.2.1
	28	Elbows 5B, 5C, 6A, 6D	Circum	None	Note 2 Note 3 Note 4 Whip restraints		A B C D	3C.2.1 3C.2.1 Note 5 3C.2.1
	28	Elbows 3A, 3D	Circum	12-inch feedwater piping 18-inch feedwater piping CRD piping	Note 2 Note 3 Note 4 Whip restraints		A B C D	3C.2.1 3C.2.1 Note 5 3C.2.1
	28	Elbows 2B, 2C	Circum	14-inch LPCS piping 14-inch HPCS piping CRD piping	Note 2 Note 3 Note 4 Whip restraints and pipe stiffness		A B C D	3C.2.1 3C.2.1 Note 5 3C.2.1
	28	Elbows 4A, 3B, 3C, 4D	Circum	14-inch LPCS piping 14-inch HPCS piping 12-inch feedwater piping 28-inch MS piping MSIV	Note 2 Note 3 Note 4 Whip restraints		A B C D	3C.2.1 3C.2.1 Note 5 3C.2.1

TABLE 3.6A-18: SUMMARY OF HIGH-ENERGY PIPING FAILURE ANALYSIS (CONTINUED)

Pipe Function	Size, Inches	Failure Location	Type of Failure	Essential[9] Equipment	Protection	Remarks	Hazard[1]	Appendix 3C Consequence
	28	Inlet to MSIV 4B, 4C, 5A, 5D	Circum	12-inch LPCI piping 28-inch MS piping MSIV	Note 2 Note 3 Note 8 Whip restraints		A B C D	3C.2.1 3C.2.1 Note 8 3C.2.1
Main steam inside turbine bldg (Reference Figure 3.6A-2)	28	Inside turbine bldg	Circum	East wall of auxiliary bldg	Not required Not required Turbine bldg wall Whip restraints	Ref. Fig. 3.6A-2	A B C D	Note 6 Note 6 Note 6 Note 6

TABLE 3.6A-19: SUMMARY OF HIGH ENERGY PIPING FAILURE ANALYSIS

Feedwater Piping (Including RHR and RWCU Injection Piping)
(For explanatory notes, refer to Table 3.6A-27A)

Pipe, Function	Size Inches	Failure Location	Type of Failure	Essential[9] Equipment	Protection	Remarks	Hazard[1]	Appendix 3C Consequence
Feedwater in drywell (Reference Figures 3.6A-5A and B)	24 and 12	Each tee, 24" x 24" x 12" FW-1	Circum	MSRV disch. piping 10" RCIC piping 12" LPCI piping, 6" RWCU piping, CRD piping	Note 2 Note 3 Note 4 Whip restraints	Breaks chosen based on usage factors	A B C D	3C.2.2 3C.2.2 Note 5 3C.2.2
	24 and 12	Each tee, 24" x 24" x 12" FW-3	Circum	10" RCIC Piping 12" LPCI Piping 6" RWCU Piping MSRV disch. piping CRD piping	Note 2 Note 3 Note 4 Whip restraints	Breaks chosen based on usage factors	A B C D	3C.2.2 3C.2.2 Note 5 3C.2.2
	24 and 12	Each tee, 24" x 24" x 12" FW-7	Circum	14" and 12" LPCI MSRV disch. piping	Note 2 Note 3 Note 4 Whip restraints	Breaks chosen based on usage factors	A B C D	3C.2.2 3C.2.2 Note 5 3C.2.2
	18 and 12	Each tee, 18" x 18" x 12" FW-5	Long	MSRV disch. piping	Note 2 Note 3 Note 4 Whip restraints	Breaks chosen based on usage factors	A B C D	3C.2.2 3C.2.2 Note 5 3C.2.2
	12	RPV Nozzles FW-9, 10, 11	Circum	MSRVs 10" RCIC piping 12" LPCI piping	Note 2 Note 3 Note 4 Whip restraints	Terminal ends	A B C D	3C.2.2 3C.2.2 Note 5 3C.2.2
	18 and 12	Tee 18" x 18" x 12" FW-4, 6, 8	Circum	14" LPCI piping 12" LPCI Piping CRD Piping 12" Recirc. piping	Note 2 Note 3 Note 4 Whip restraints	Breaks chosen based on usage factors	A B C D	3C.2.2 3C.2.2 Note 5 3C.2.2

TABLE 3.6A-19: SUMMARY OF HIGH ENERGY PIPING FAILURE ANALYSIS (Continued)

Feedwater Piping (Including RHR and RWCU Injection Piping)
(For explanatory notes, refer to Table 3.6A-27A)

Pipe, Function	Size Inches	Failure Location	Type of Failure	Essential[9] Equipment	Protection	Remarks	Hazard[1]	Appendix 3C Consequence
RHR injection in aux bldg (Reference Figure 3.6A-6)	12	Pipe at end of no- break zone, immediately below tunnel floor FW-5 & 10	Circum	Integrity of no-break zone	Note 2 Note 3 Note 4 Whip restraints	End of no-break zone	A B C D	3C.2.2 3C.2.2 Note 7 3C.2.2
	12	End of check valve FW-1 & 6	Circum	RHR valves and operators	Note 2 Note 3 Note 4 Separation	Terminal end	A B C D	3C.2.2 3C.2.2 Note 7 3C.2.2
	12	Elbows FW-2 & 3 FW-7 & 8	Circum	RHR valves and operators	Note 2 Note 3 Note 4 Whip restraints and separation	Intermediate break location	A B C D	3C.2.2 3C.2.2 Note 7 3.C.2.2
	12	At anchor point FW-4 & 9	Circum	RHR valves and operators	Note 2 Note 3 Note 4 Whip restraints and separation	Terminal end	A B C D	3C.2.2 3C.2.2 Note 7 3C.2.2
	6	Normally closed check valve FW-11	Circum	RHR valves and operators	Note 2 Note 3 Note 4 Whip restraints and separation	Terminal end	A B C D	3C.2.2 3C.2.2 Note 7 Note 7

TABLE 3.6A-20: SUMMARY OF HIGH-ENERGY PIPING FAILURE ANALYSIS

Reactor Recirculation System and RHR Suction
(For explanatory notes, refer to Table 3.6A-27A)

Pipe, Function	Size, Inches	Failure Location	Type of Failure	Essential[9] Equipment	Protection	Remarks	Hazard[1]	Appendix 3C Consequence
Recirculation System (Reference Figure 3.6B-4a)								
Risers	12	RPV nozzles RD-1 through RD-6	Circum	None	Note 2 Note 3 Note 4 Separation and restraints	See GE Section 3.6.B	A B C D	3C.2.3 3C.2.3 Note 5 3C.2.3
Header	15	Tees to 12" Risers RD-7 through RD-12	Circum	CRD piping 12" feedwater 28" main steam 14" LPCS 14" LPCS	Note 2 Note 3 Note 4 Separation and restraints	See GE Section 3.6.B	A B C D	3C.2.3 3C.2.3 Note 5 3C.2.3
Pump suction	24	RPV nozzle RS-1	Circum	None	Note 2 Note 3 Note 4 Separation and restraints	See GE Section 3.6.B	A B C D	3C.2.3 3C.2.3 Note 5 3C.2.3
RHR suction (Reference Figure 3.6A-9)	20	Tee to recirculation system RS-3, SS-1	Circum and long	None	Note 2 Note 3 Note 4 Separation and whip restraints	Terminal end of run	A B C D	3C.2.3 3C.2.3 Note 5 3C.2.3
	20	Inlet to valve F009 SS-4	Circum	None	Note 2 Note 3 Note 4 Separation and whip restraints	Terminal end of run	A B C D	3C.2.3 3C.2.3 Note 5 3C.2.3

TABLE 3.6A-20: SUMMARY OF HIGH-ENERGY PIPING FAILURE ANALYSIS (Continued)

Reactor Recirculation System and RHR Suction
(For explanatory notes, refer to Table 3.6A-27A)

Pipe, Function	Size, Inches	Failure Location	Type of Failure	Essential[9] Equipment	Protection	Remarks	Hazard[1]	Appendix 3C Consequence
	20	Both ends of elbow SS-2 & 3	Circum	None	Note 2 Note 3 Note 4 Separation and whip restraints	Intermediate break location	A B C D	3C.2.3 3C.2.3 Note 5 3C.2.3
	20	Elbow SS-5	Long	None	Note 2 Note 3 Note 4 Separation and whip restraints	Intermediate break location	A B C D	3C.2.3 3C.2.3 Note 5 3C.2.3

TABLE 3.6A-21: SUMMARY OF HIGH-ENERGY PIPING FAILURE ANALYSIS

Steam to RCIC Turbine
(For explanatory notes, refer to Table 3.6A-27A)

Pipe, Function	Size, Inches	Failure Location	Type of Failure	Essential[a] Equipment	Protection	Remarks	Hazard	Appendix 3C Consequence
From main steam line "A" to drywell penetration (Reference Figure 3.6A-3)	10	Tee to main steam line "A" RCIC-1	Circum	Main steam safety/relief/ADS valves and associated control system (Note 10)	Note 2 Note 3 Note 4 Whip restraints	Terminal end of branch run	A B C D	3C.2.4 3C.2.4 Note 5 3C.2.4
		Elbow at bottom of vertical leg RCIC-5 & 7	Circum	Recirc suction	Note 2 Note 3 Note 4 Moment guide	Intermediate location	A B C D	3C.2.4 3C.2.4 Note 5 3C.2.4
		Next to moment guide near inlet to valve RCIC-6	Circum	None	Note 2 Note 3 Note 4 Moment guide	Start of no-break zone	A B C D	3C.2.4 3C.2.4 Note 5 3C.2.4
		Elbow at top of vertical leg RCIC-3 & 8	Circum	None	Note 2 Note 3 Note 4 Whip restraints	Intermediate location	A B C D	3C.2.4 3C.2.4 3C.2.4 3C.2.4
Inside auxiliary bldg (Reference Figure 3.6A-4)	6	In pipe chase at end of no-break zone RCIC-11A	Circum	Isolation valve	Note 2 Note 3 Note 4 Separation, enclosure, whip restraint	End of no-break zone	A B C C	3C.2.4 3C.2.4 3C.2.4 3C.2.4
		Inlet to elbow at bottom of vertical leg RCIC-11B	Circum	Isolation valve	Note 2 Note 3 Note 4 Separation Whip restraint	Intermediate location	A B C D	3C.2.4 3C.2.4 3C.2.4 3C.2.4
		2nd Tee, El. - 96' RCIC-11C	Circum	6" RCIC Suction	Note 2 Note 3 Note 4 Separation	Intermediate location	A B C D	3C.2.4 3C.2.4 3C.2.4 3C.2.4

TABLE 3.6A-21: SUMMARY OF HIGH-ENERGY PIPING FAILURE ANALYSIS (CONTINUED)

Steam to RCIC Turbine
(For explanatory notes, refer to Table 3.6A-27A)

Pipe, Function	Size, Inches	Failure Location	Type of Failure	Essential[a] Equipment	Protection	Remarks	Hazard	Appendix 3C Consequence
		Inlet to normally closed valve F045 RCIC-12	Circum	6" RCIC Suction	Note 2 Note 3 Note 4 Separation	Terminal end of high energy piping	A B C D	3C.2.4 3C.2.4 3C.2.4 3C.2.4

TABLE 3.6A-22: DELETED

TABLE 3.6A-23: SUMMARY OF HIGH-ENERGY PIPING FAILURE ANALYSIS

RWCU Piping
(For explanatory notes, refer to Table 3.6A-27A)

Pipe, Function	Size, Inches	Failure Location	Type of Failure	Essential[9] Equipment	Protection	Remarks	Hazard[1]	Appendix 3C Consequence
RPV to drywell penetration (Reference Figure 3.6A-7)	4	Connection (Tee) to 2" RPV Drains -2 RWCU-12B, 17B	Circum	CRD hydraulic	Note 2 Note 3 Note 4 Whip restraints	Below RPV, inside pedestal terminal ends	A B C D	3C.2.6 3C.2.6 Note 6 3C.2.6
	4	Connection to recirc piping (2) RWCU-1B, 11B	Circum	None	Note 2 Note 3 Note 4 Whip restraints	Terminal ends	A B C D	3C.2.6 3C.2.6 Note 6 3C.2.6
	4	Dissimilar metal weld RWCU-37	Circum	None	Note 2 Note 3 Note 4 Whip restraints	Intermediate based on dissimilar stresses	A B C D	3C.2.6 3C.2.6 Note 6 3C.2.6
	4,6	RWCU-6B, 7B, 9B	Circum long	None	Note 2 Note 3 Note 4 Whip restraints	Intermediates, based on usage factor	A B C D	3C.2.6 3C.2.6 Note 6 3C.2.6
	6	Tee RWCU-8B, 32	Circum	Isolation valve penetration	Note 2 Note 3 Note 4 Whip restraints	Boundary of no-break zone	A B C D	3C.2.6 3C.2.6 Note 6 3C.2.6
RWCU pump rooms in aux building (Reference Figure 3.6A-8)	4	All fittings, valves, and boundary between nuclear/nonnuclear piping	Circum	None	Note 2 Note 3 Note 4 Whip restraints	Nonnuclear piping; terminal end	A B C D	3C.2.6 3C.2.6 3C.2.6 3C.2.6
Heat exchanger room	4,6	See Note Below	Circum	None	Note 2 Note 3 Note 4 Separation	Nonnuclear piping	A B C D	3C.2.6 3C.2.6 3C.2.6 3C.2.6

TABLE 3.6A-23: SUMMARY OF HIGH-ENERGY PIPING FAILURE ANALYSIS (Continued)

RWCU Piping
(For explanatory notes, refer to Table 3.6A-27A)

Pipe, Function	Size, Inches	Failure Location	Type of Failure	Essential[9] Equipment	Protection	Remarks	Hazard[1]	Appendix 3C Consequence
Filter/ demineralizer room	3,4	All fittings, valves	Circum	None	Note 2 Note 3 Note 4 Separation	Nonnuclear piping	A B C D	3C.2.6 3C.2.6 3C.2.6 3C.2.6
Containment steam tunnel	4	All fittings, valves	Circum	None	Note 2 Note 3 Note 4 Separation	Nonnuclear piping	A B C D	3C.2.6 3C.2.6 3C.2.6 3C.2.6

Note: Circumferential breaks were postulated at all fittings, valves, etc., per FSAR Subsection 3.6A.2.1, except for the region in the vicinity of isolation valve F251. In this region piping is seismically analyzed and the breaks were postulated based on stress analysis data. Refer to Figure 3.6A-7 for additional information.

TABLE 3.6A-24: SUMMARY OF HIGH-ENERGY PIPING FAILURE ANALYSIS

Low-Pressure Core Spray System
(For explanatory notes, refer to Table 3.6A-27A)

Pipe, Function	Size, Inches	Failure Location	Type of Failure	Essential[9] Equipment	Protection	Remarks	Hazard[1]	Appendix 3C Consequence
LPCS in containment (Reference Figure 3.6A-11)	12	RPV nozzle LPCS-4	Circum	None	Note 2 Note 3 Note 4 Whip restraints	Terminal end	A B C D	3C.2.7 3C.2.7 Note 5 3C.2.7
	14	Top end of valve F007 LPCS-6	Circum	12" LPCI Piping CRD Piping 12" Feedwater Piping	Note 2 Note 3 Note 4 Whip restraints	Intermediate break	A B C D	3C.2.7 3C.2.7 Note 5 3C.2.7
	14	Bottom end of valve F007 LPCS-7	Circum	12" LPCI Piping CRD Piping 12" Feedwater Piping	Note 2 Note 3 Note 4 Whip restraints	Intermediate break	A B C D	3C.2.7 3C.2.7 Note 5 3C.2.7
	14	Outlet of valve F006 LPCS-1	Circum	12" LPCI Piping CRD Piping 12" Feedwater Piping	Note 2 Note 3 Note 4 Whip restraints	Terminal end	A B C D	3C.2.7 3C.2.7 Note 5 3C.2.7

TABLE 3.6A-25: SUMMARY OF HIGH-ENERGY PIPING FAILURE ANALYSIS

High-Pressure Core Spray System
(For explanatory notes, refer to Table 3.6A-27A)

Pipe, Function	Size, Inches	Failure Location	Type of Failure	Essential[9] Equipment	Protection	Remarks	Hazard[1]	Appendix 3C Consequence
HPCS in containment (Reference Figure 3.6A-10)	12	RPV nozzle HPCS-1	Circum	CRD Piping	Note 2 Note 3 Note 4 Whip restraints	Terminal end	A B C D	3C.2.7 3C.2.7 Note 5 3C.2.7
	14	Top of valve F036 HPCS-5	Circum	14" LPCI Piping CRD Piping 12" Feedwater Piping	Note 2 Note 3 Note 4 Whip restraints	Intermediate break	A B C D	3C.2.7 3C.2.7 Note 5 3C.2.7
	14	Bottom of valve F036 HPCS-6	Circum	14" LPCI Piping CRD Piping 12" Feedwater Piping	Note 2 Note 3 Note 4 Whip restraints	Intermediate break	A B C D	3C.2.7 3C.2.7 Note 5 3C.2.7
	14	Top of valve F005	Circum	14" LPCI Piping CRD Piping 12" Feedwater Piping	Note 2 Note 3 Note 4 Whip restraints	Terminal end	A B C D	3C.2.7 3C.2.7 Note 5 3C.2.7
	14	Top of valve F036 HPCS-5	Long	12" Feedwater Piping 14"LPCI Piping	Note 2 Note 3 Note 4 Whip restraints		A B C D	3C.2.7 3C.2.7 Note 5 3C.2.7
	14	Bottom of valve F036 HPCS-6	Long	12" Feedwater Piping 14" LPCI Piping	Note 2 Note 3 Note 4 Whip restraints		A B C D	3C.2.7 3C.2.7 Note 5 3C.2.7

TABLE 3.6A-26: SUMMARY OF HIGH-ENERGY PIPING FAILURE ANALYSIS

LPCI Mode of RHR
(For explanatory notes, refer to Table 3.6A-27A)

Pipe, Function	Size, Inches	Failure Location	Type of Failure	Essential[9] Equipment	Protection	Remarks	Hazard[1]	Appendix 3C Consequence
LPCI "A" in containment (Reference Figure 3.6A-12)	12	RPV nozzle LPCI-1A	Circum	None	Note 2 Note 3 Note 4 Whip restraint	Terminal end	A B C D	3C.2.8 3C.2.8 Note 5 3C.2.8
	12	RPV end of upper elbow LPCI-2A	Circum	MSRV	Note 2 Note 3 Note 4 Whip restraint	Intermediate location	A B C D	3C.2.8 3C.2.8 Note 5 3C.2.8
	14	Valve end of lower elbow LPCI-3A	Circum	MSRV	Note 2 Note 3 Note 4 Energy absorber	Intermediate location	A B C D	3C.2.8 3C.2.8 Note 5 3C.2.8
	14	Outlet valve F041A LPCI-4A	Circum	MSRV	Note 2 Note 3 Note 4 Energy absorber	Terminal end	A B C D	3C.2.8 3C.2.8 Note 5 3C.2.8
LPCI "B" in containment (Reference Figure 3.6A-12)	12	RPV nozzle LPCI-1B	Circum	None	Note 2 Note 3 Note 4 Whip restraints	Terminal end	A B C D	3C.2.8 3C.2.8 Note 5 3C.2.8
	12	RPV end of elbow LPCI-2B	Circum	None	Note 2 Note 3 Note 4 Whip restraints	Intermediate location	A B C D	3C.2.8 3C.2.8 Note 5 3C.2.8
	12	Other end of elbow LPCI-3B	Circum	12" Feedwater Piping	Note 2 Note 3 Note 4 Whip restraints	Intermediate location	A B C D	3C.2.8 3C.2.8 Note 5 3C.2.8

TABLE 3.6A-26: SUMMARY OF HIGH-ENERGY PIPING FAILURE ANALYSIS (Continued)

LPCI Mode of RHR
(For explanatory notes, refer to Table 3.6A-27A)

Pipe, Function	Size, Inches	Failure Location	Type of Failure	Essential[9] Equipment	Protection	Remarks	Hazard[1]	Appendix 3C Consequence
	14	Outlet of valve F041B LPCI-4B	Circum	16" & 12" Recirculation Piping 12" Feedwater Piping	Note 2 Note 3 Note 4 Whip restraints	Terminal end	A B C D	3C.2.8 3C.2.8 3C.2.8 3C.2.8
LPCI "C" in containment (Reference Figure 3.6A-12)	12	RPV nozzle LPCI-1C	Circum	None	Note 2 Note 3 Note 4 Whip restraints	Terminal end	A B C D	3C.2.8 3C.2.8 Note 5 3C.2.8
	12	RPV end of elbow LPCI-2C	Circum	12” Feedwater Piping	Note 2 Note 3 Note 4 Whip restraints	Intermediate location	A B C D	3C.2.8 3C.2.8 Note 5 3C.2.8
	12	Other end of elbow LPCI-3C	Circum	12” Feedwater Piping	Note 2 Note 3 Note 4 Whip restraints	Intermediate location	A B C D	3C.2.8 3C.2.8 Note 5 3C.2.8
	12	Outlet of valve F241 LPCI-4C	Circum	12” Feedwater Piping	Note 2 Note 3 Note 4 Whip restraints	Terminal end	A B C D	3C.2.8 3C.2.8 Note 5 3C.2.8

TABLE 3.6A-27: SUMMARY OF HIGH-ENERGY PIPING FAILURE ANALYSIS

Refer to Appendix 3C, section 3C.2.9, for a discussion of 3" and smaller postulated high-energy piping failures.

TABLE 3.6A-27A: EXPLANATORY NOTES FOR TABLES 3.6A-18 THROUGH 3.6A-27

Note 1 HAZARD:

A - Flooding
B - Environment
C - Jet impingement/asymmetric pressure loading/spraying
D - Pipe whip

Note 2 Refer to Appendix 3C.4 for compartment flooding discussion. Protective measures consist of enclosures, seals, leak detection, and drainage systems.

Note 3 Essential components are designed to function both during and following the environment associated with the loss-of-coolant accident including local water and steam environments. Refer to Sections 3.11 and 6.2.

Note 4 Although separation minimizes the potential for jet impingement on redundant components, both jet impingement and asymmetric pressure loads on components are evaluated by either analyzing the affected component for the pressure loading or determining that the pressure load is insignificant for the component. An alternate approach is to assume that the component has failed while demonstrating safe shutdown capability with a concurrent loss of offsite power and the most limiting single failure of another component. If structural analysis determines that an essential component is overstressed, then protective measures such as deflectors or restraints are added. Vents are employed where pressures are excessive; spray shields are utilized where the jet is reduced to spray or where secondary spraying is of any consequence.

TABLE 3.6A-27A: EXPLANATORY NOTES FOR TABLES 3.6A-18 THROUGH 3.6A-27 (CONTINUED)

Note 5 Except for the CRD insert and withdrawal lines, piping smaller than 1-1/2 inches in diameter and electrical conduit are field routed. In the drywell for major line breaks most small bore piping is not required for safe shutdown. The components they serve are either fail-safe, such as ADS (Note 10) and MSIV, or not required, such as jet pump instrumentation. The instrumentation lines which sense the LOCA and instigate a scram are open ended and routed outside the drywell. They are redundant and not affected by a single failure. Post-accident monitoring instrument lines are protected when a single failure in combination with the pipe break and loss of off-site power could render the system inoperable. Because most of the small bore lines in the drywell are connected to the reactor coolant pressure boundary and are not capable of being isolated after their rupture, they may contribute to the LOCA, but their small cross-sectional area and separated routing prevent a significant increase in loss of coolant. However, the field routing is reviewed to ensure that the consequence of their failure when combined with the original rupture does not exceed the blowdown rate of the design basis accident. Because of their energy level, the failure of these components is assumed not to result in the secondary failure of an essential component. In the event analysis should prove different, protection required would also cause a minimum of construction changes or additions in present design. Conduit to certain hydrogen ignitors is protected to assure their availability following a LOCA inside the drywell.

Note 6 As discussed in Appendix 3C, a rupture of either the main steam piping or feedwater piping inside the turbine building does not affect safety-related components because there are no components necessary for safe shutdown located within the turbine building and the auxiliary building east wall is protected from pipe whip by energy absorbers located on the main steam and feedwater piping in the turbine building. The main steam isolation valves will close upon an indication of high steam line flow from the instrument lines inside the drywell and the feedwater isolation valves will prevent reverse flow. The response to IEB 79-22 addresses pipe breaks in the turbine building.

TABLE 3.6A-27A: EXPLANATORY NOTES FOR TABLES 3.6A-18 THROUGH 3.6A-27 (CONTINUED)

Note 7 Further damage of the RHR train with the postulated rupture does not prevent safe shutdown. As specified in subsection 3.6A.1.1.C.9, a single failure of an active component in one train of the RHR need not be postulated following a piping failure in the other. Also, a failure of the RHR suction line valves to open will not prevent cold shutdown because the ADS may be used to depressurize to the suppression pool.

Note 8 Jet impingement evaluation was not considered necessary at main steam terminal end breaks 4B, 4C, 5A, and 5D for the following reasons:

1) These breaks are located at the no break zone adjacent to the torsional guide to evaluate the no break zone components for the effect of pipe whip.

2) The stresses and usage factor of subsection 3.6A.2.1c.1 are not exceeded.

3) These breaks are not “true” terminal ends by definition since the torsional guides do not rigidly constrain normal pipe movement.

Note 9 Only major equipment and piping are listed. Field-routed items such as conduit and small bore piping are identified during field walkdowns. Refer to Note 5.

Note 10 The Automatic Depressurization System (ADS) function is only required for postulated LOCA breaks with flow area less than 0.7 ft2. Therefore, only high-energy piping 10 inches in diameter and smaller need be reviewed for effects on the ADS system. Only three of the seven ADS relief valves are required to rapidly depressurize the RPV assuming no blowdown through the break. Consideration of the break opening further reduces the required number of ADS valves.

TABLE 3.6A-28: SUMMARY OF DESIGN REQUIREMENTS FOR GUARD PIPE ASSEMBLY

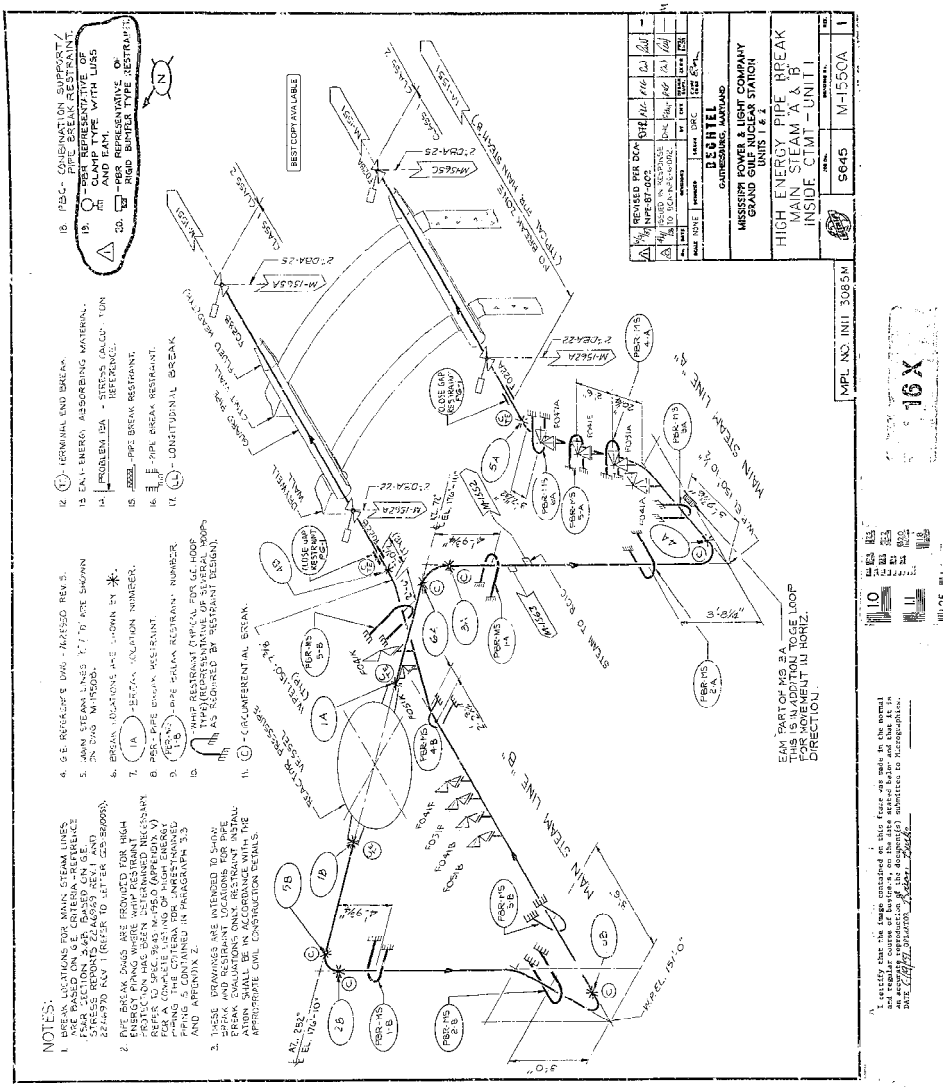
ASME-Section III Code Paragraph Reference:	Component				
	Process Pipe	Flued Head	Guard Pipe	Bellows Assembly Unit (Note 7)	
Design Consideration and analysis Criteria	NB-3100 NB-3620,3630,3650	NB-3100 NB-3620,3630,3650	NC-3100 and 3600	Bellows: Piping:	NE-3365, 3365.2 NE-3100 and NC-3600
Fabrication and Installation	NB-4100,4220,4230, 4310,4320,4330, 4360,4400,4600,	NB-4100,4220,4230, 4310,4320,4330, 4360,4400,4600	NC-4000 per NE-4800	Bellows: Piping:	NE-4810 NE-4000
Examination	NB-5100,5200,5300, 5500	NB-5100,5200,5300, 5500	Section NC per NE-5700	Bellows: Piping:	NE-5710 NE-5700
Testing	NB-6210,6220	NB-6210,6220, and NE-6310,6320	NC-6200 except hydrostatic test pressure equal to design pressure	Bellows: Piping:	NE-6510 NE-6510
Stamping	NA-8000	NA-8000	NOT APPLICABLE	NOT APPLICABLE	
Stress Reports	NA-3260,3350	NA-3260,3350	NOT REQUIRED	NOT REQUIRED	
Wall Thickness Sizing Calculations	NB-3641.1	NB-3641.1	NC-3641.1	Bellows: Piping:	See Note 2 NC-3641
Loading Conditions	NB-3111	NB-3111	NC-3111	NE-3111,3365.2 See Note 1	

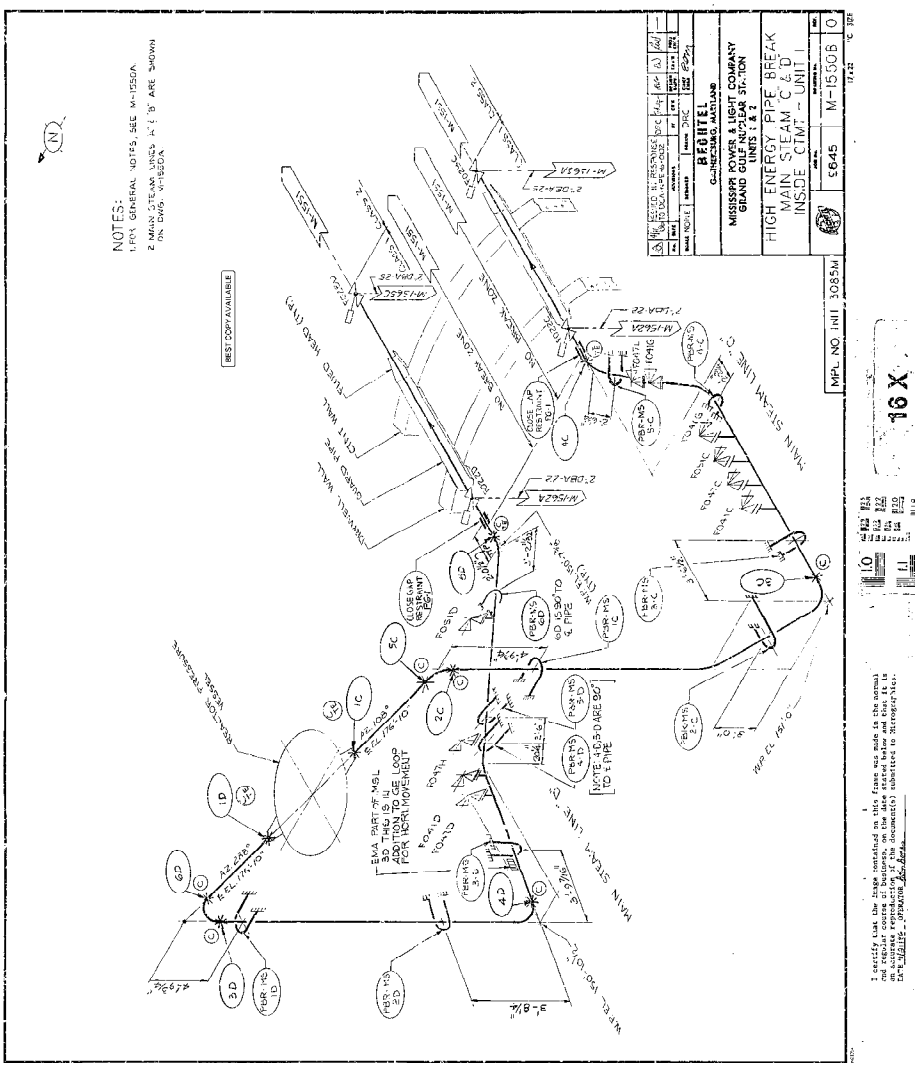
TABLE 3.6-28: SUMMARY OF DESIGN REQUIREMENTS FOR GUARD PIPE ASSEMBLY (CONTINUED)

Stress Limits	NB-3220,3611.1 See Note 5	NB-3220,3611.1	NC-3600 See Note 6	NE-3131,3365.2 See Note 1
Material	NB-2000 Seamless Carbon Steel SA-106 Grade B or C	NB-2000 Forged Carbon Steel SA-508 Gr. 1	NC-2000 per NE-2121.4 Welded Carbon Steel SA-155 Gr. KCF70- Class 1 or Seamless Carbon Steel SA-106 Gr. C (Note 3) Gr. B (Note 4)	Bellows: NE-3365(g) Stainless Steel SA-240 Type 304 or 316 Piping: NE-2000 carbon steel ASME SA 516 Gr. 70 or SA 106 GR. B

NOTES:

1. Requirements of NRC Regulatory Guide 1.57 were applied except jet/force impingement loads were not applicable.
2. The bellows were sized per the Expansion Joint Manufacturers Association (EJMA) Standards.
3. For 24" diameter pipe and larger, included impact testing.
4. For pipe diameters less than 24", included impact testing.
5. During normal condition, the stress was such that the Code usage factor was less than 0.5.
6. Stresses also met NE-3131(c) using loadings associated with design pressure and temperature in combination with the SSE (safe shutdown earthquake) per Regulatory Guide 1.57. Allowable stress values per ASME Section III, Appendix I -Table I-7.1 in lieu of Table I-10.0.
7. Bellows assembly included interconnecting piping between bellows.


NOTES:
MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
HIGH ENERGY PIPE BREAK
MAIN STEAM 'A' & 'B'
INSIDE CTMT - UNIT 1
GILBERT/COMMONWEALTH
DESIGNED
DRAWN
CHECKED
QA
DESIGN ENGINEER
DRAFTING SUPERVISOR
PROJECT ENGINEER
PROJECT MANAGER
DESIGN ENGINEERING MANAGER
DRAWING NO.
M-1550A
9645
MPL NO. INI1 3085M
REVISED PER DCA NPE-87-0012
REVISED PER DCA NPE-87-0012
REVISED PER DCA NPE-87-0012



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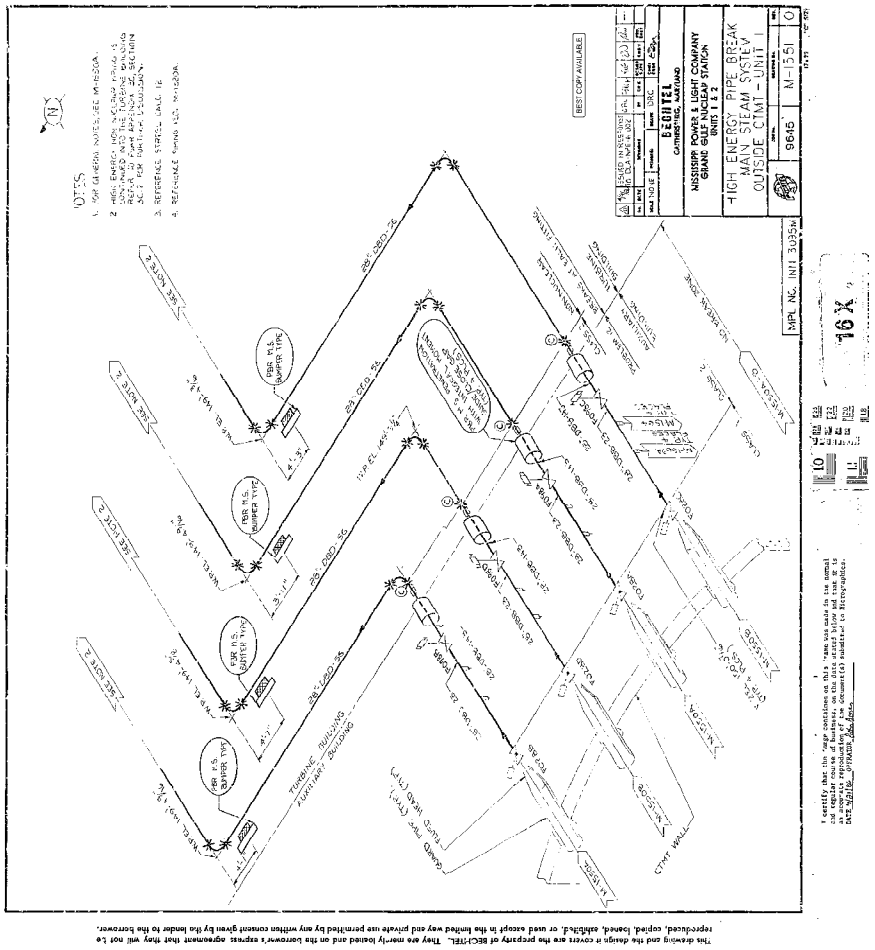
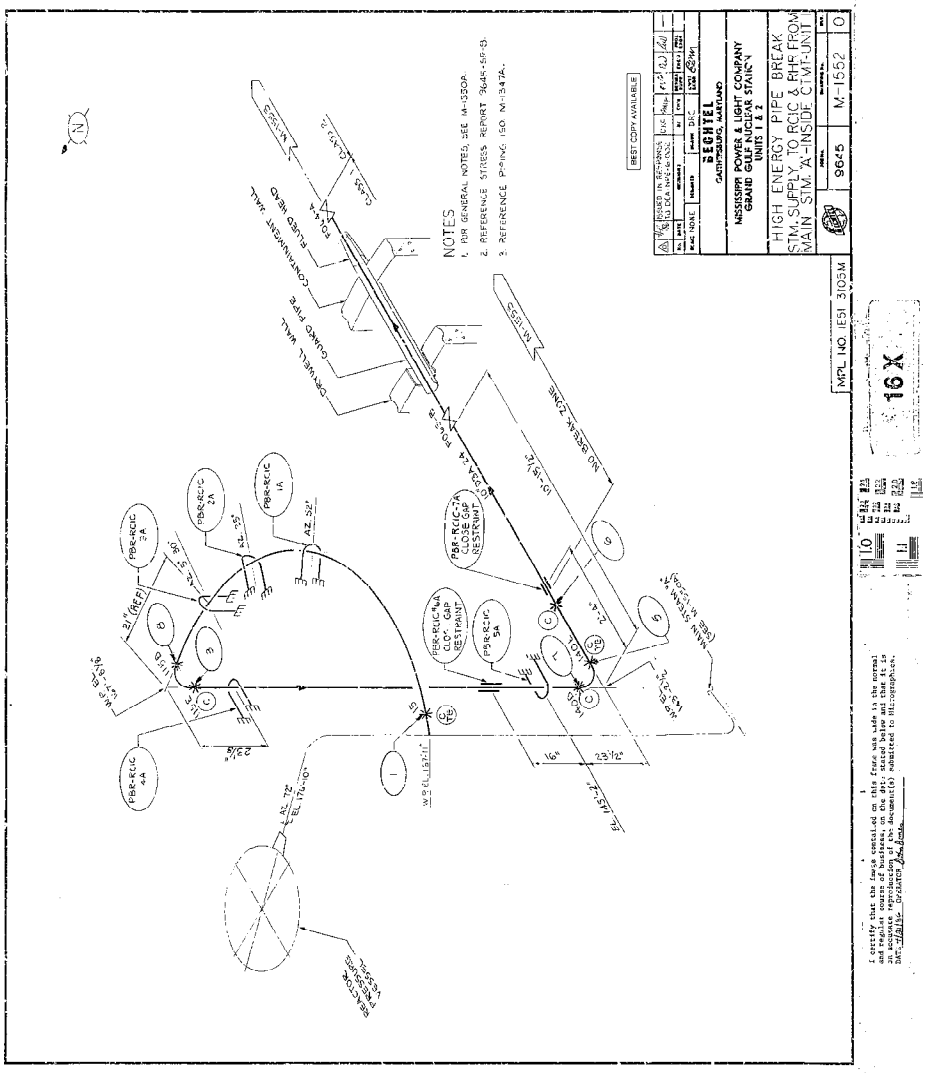

BECHTEL
MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
HIGH ENERGY PIPE BREAK
MAIN STEAM SYSTEM
OUTSIDE CTMT - UNIT 1
9645
M-1551
TURBINE BUILDING
AUXILIARY BUILDING
CONTROL BUILDING
REACTOR BUILDING
CTMT WALL
PSR GRID 25
PSR GRID 26
PSR GRID 27
PSR GRID 28
PSR GRID 29
PSR GRID 30
PSR GRID 31
PSR GRID 32
PSR GRID 33
PSR GRID 34
PSR GRID 35
PSR GRID 36
PSR GRID 37
PSR GRID



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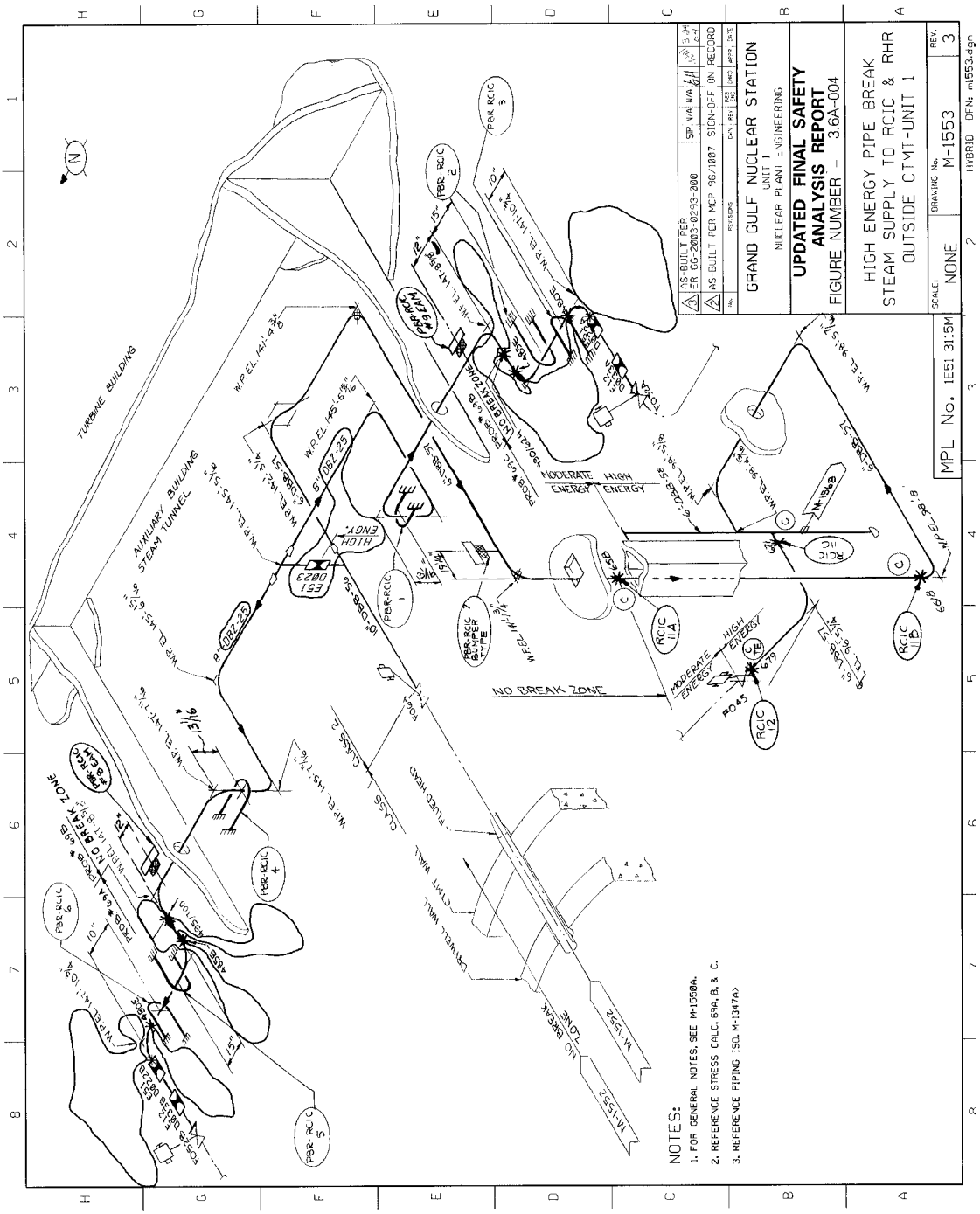
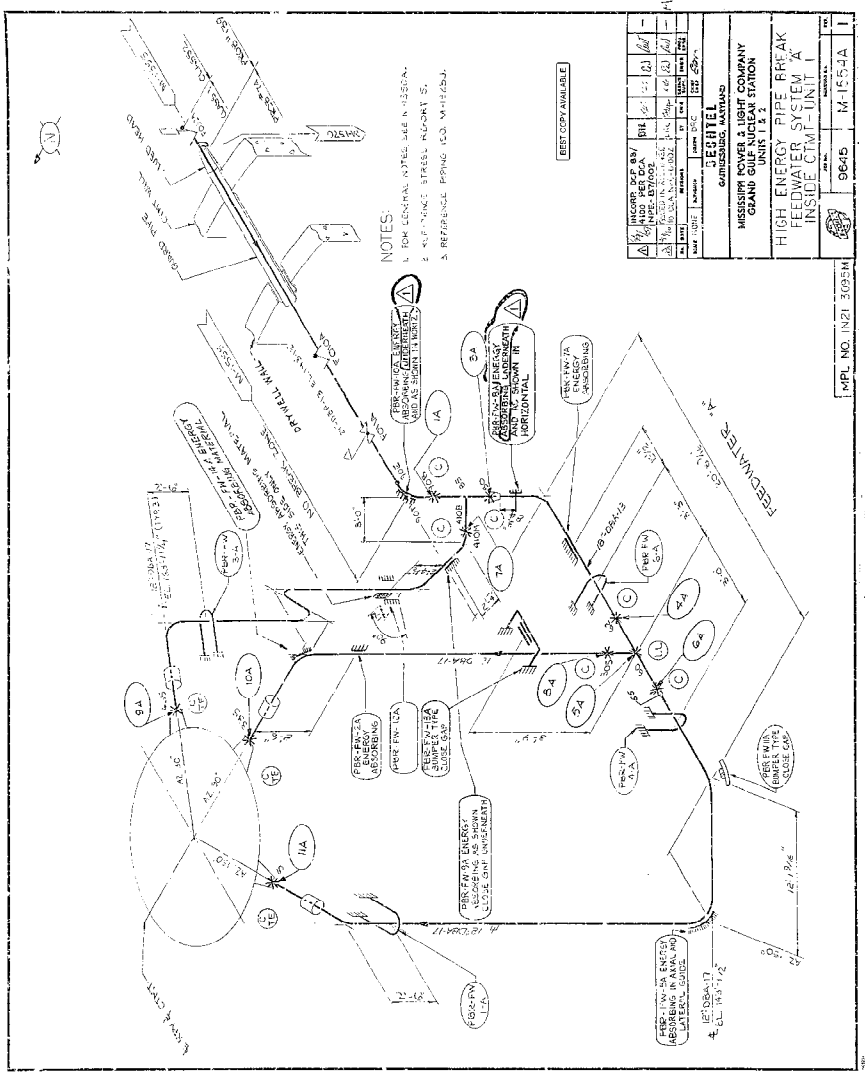

GRAND GULF NUCLEAR STATION
UNIT 1
NUCLEAR PLANT ENGINEERING
UPDATED FINAL SAFETY ANALYSIS REPORT
FIGURE NUMBER – 3.6A-004
HIGH ENERGY PIPE BREAK
STEAM SUPPLY TO RCIC & RHR
OUTSIDE CTMT-UNIT 1
DRAWING No. M-1553
REV. 3
SCALE: NONE
MPL No. 1E51 3115M
HYBRID DFN: m1553.dgn
AS-BUILT PER ER GG-2003-0293-000
AS-BUILT PER MCP 96/1087 SIGN-OFF ON RECORD
TURBINE BUILDING
AUXILIARY BUILDING STEAM TUNNEL
MODERATE ENERGY
HIGH ENERGY
NO BREAK ZONE
NOTES:
1. FOR GENERAL NOTES, SEE M-1550A.
2. REFERENCE STRESS CALC. 69A, B, & C.
3. REFERENCE PIPING ISO. M-1347A)

Figure 3.6A-4A

DELETED



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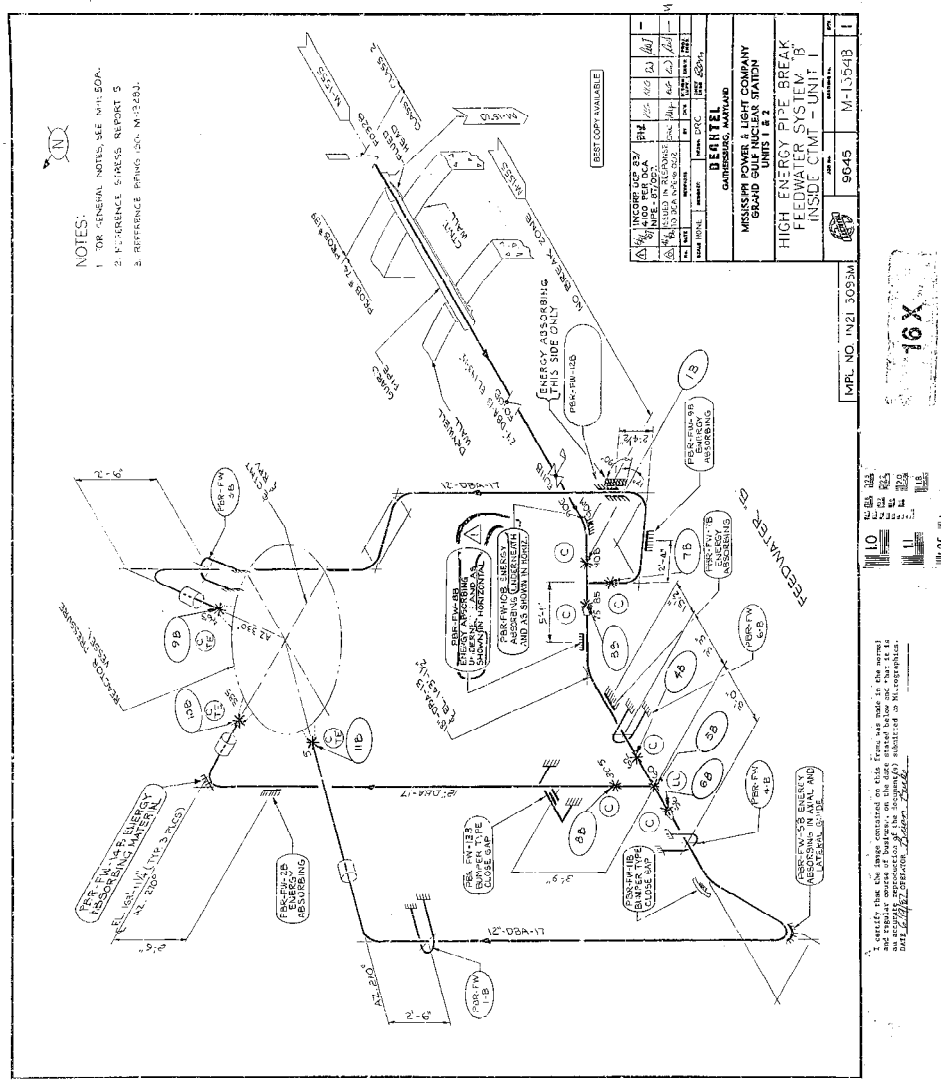

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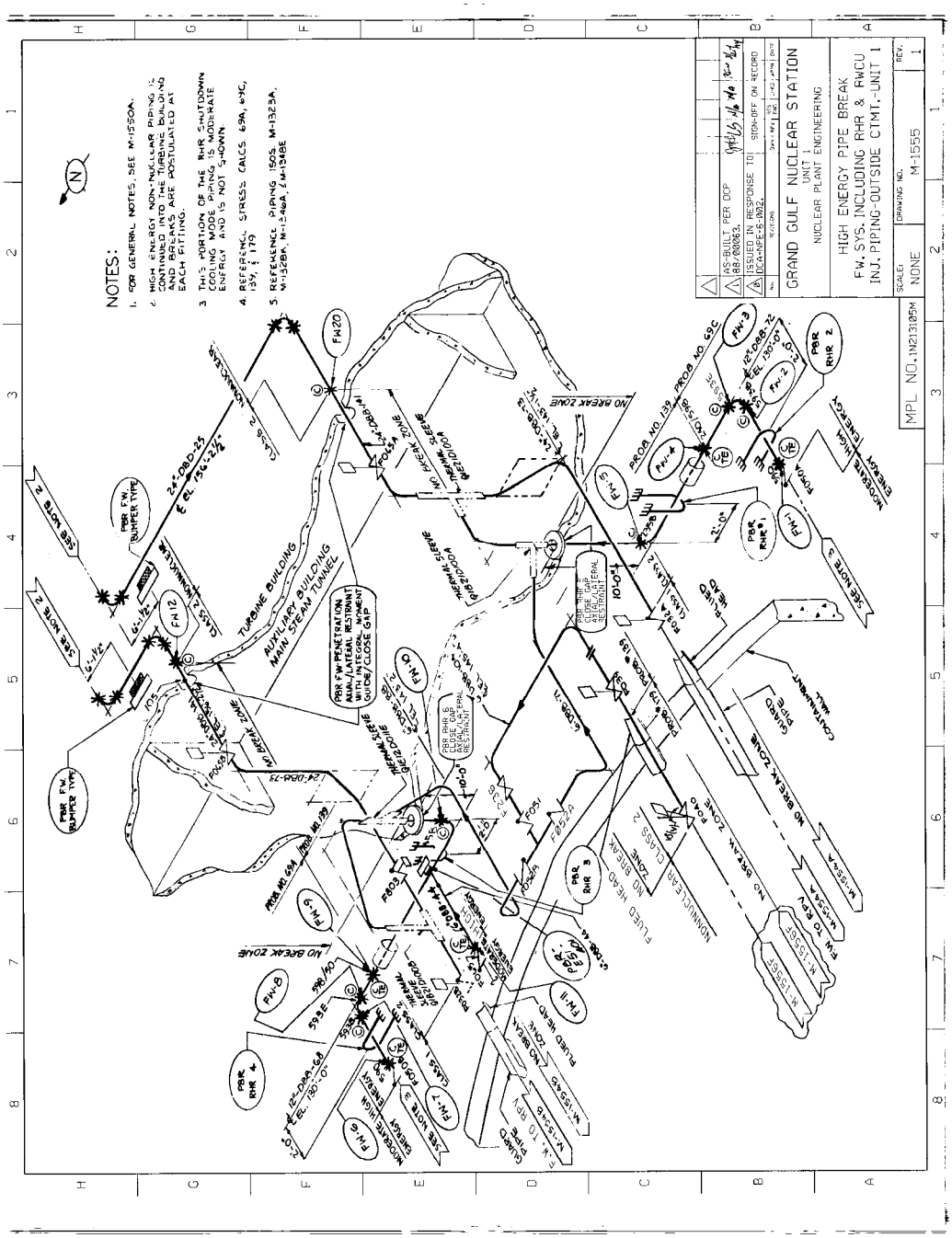
NOTES:
1. FOR GENERAL NOTES, SEE M-1550A.
2. HIGH ENERGY NON-NUCLEAR PIPING IS CONTINUED INTO THE TURBINE BUILDING AND BREAKS ARE POSTULATED AT EACH FITTING.
3. THIS PORTION OF THE RHR SHUTDOWN COOLING MODE PIPING IS MODERATE ENERGY AND IS NOT SHOWN.
4. REFERENCE STRESS CALCS 69A, 69C, 139, & 179
5. REFERENCE PIPING ISOS. M-1323A, M-1328A, M-1348A, & M-1348
GRAND GULF NUCLEAR STATION
UNIT 1
NUCLEAR PLANT ENGINEERING
HIGH ENERGY PIPE BREAK
FW, SYS. INCLUDING RHR & RWCU
INJ. PIPING-OUTSIDE CTMT.-UNIT 1
MPL NO. 1N213105M
SCALE: NONE
DRAWING NO. M-1555
REV. 1
AS-BUILT PER OCP 88/00083.
ISSUED IN RESPONSE TO DCP#/CE-8-00162.
SIGN-OFF ON RECORD
TURBINE BUILDING
AUXILIARY BUILDING
MAIN STEAM TUNNEL
NO BREAK ZONE
FLUED HEAD ZONE
NONNUCLEAR ZONE
CONTAINMENT
PBR FW PENETRATION MAIN/LATERAL RESTRAINT WITH INTEGRAL GUIDE/CLOSE GAP
PBR RHR-B MAIN/LATERAL RESTRAINT WITH INTEGRAL GUIDE/CLOSE GAP
PBR FW BUMPER TYPE
PBR RHR 1
PBR RHR 2
PBR RHR 3
PBR RHR 4
MODERATE ENERGY BREAK
SEE NOTE 2
SEE NOTE 3
FW-1
FW-2
FW-3
FW-4
FW-5
FW-6
FW-7
FW-8
FW-9
FW-10
FW-11
FW-12
FW-13
FW-14
FW-15
FW-16
FW-17
FW-18
FW-19
FW-20
24"-DBA-75
12"-DBA-68
6"-DBA-73
2"-DBA-72
M-1555A
M-1555B
M-1555C
M-1555D
M-1555E
M-1555F
M-1555G
M-1555H
M-1555J
M-1555K
M-1555L
M-1555M
M-1555N
M-1555P
M-1555Q
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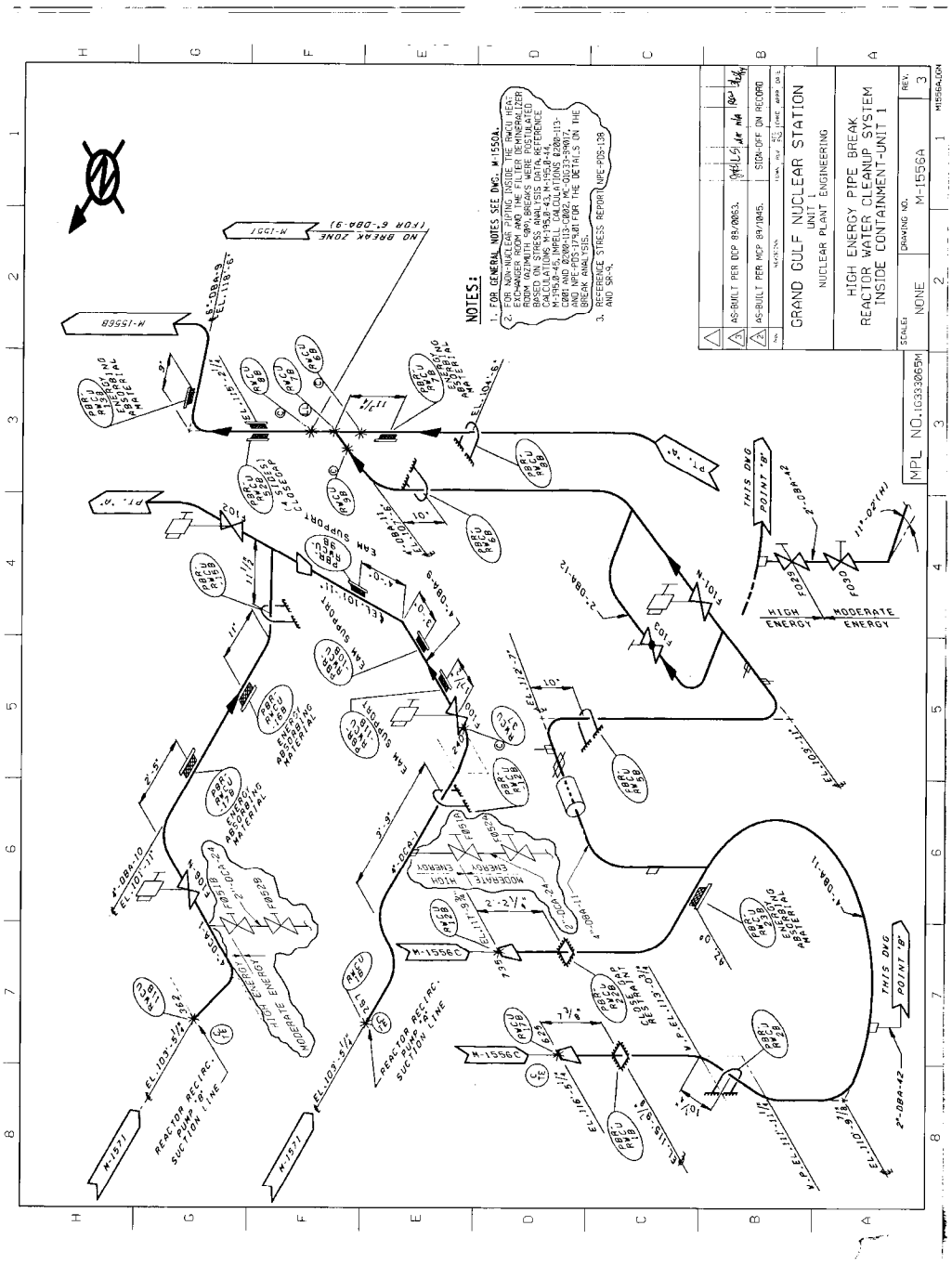
M-1555Z


NOTES:
1. FOR GENERAL NOTES SEE DWG. M-1550A.
2. (5) NON-NUCLEAR PIPING INSIDE THE SHIELDED EXCHANGER ROOM AND THE FILLER DEMINERALIZER ROOM (AZIMUTH 180°) BREAKS WERE POSTULATED BASED ON STRESS ANALYSIS DATA REFERENCE ANCO ANALYSIS REPORTS NO. 4, 5, 6, 7, 8, 9, M-1350-45, NMPELL CALCULATIONS 8200-113-0001 AND 8200-113-0002 MC-01033-39017, MC-01033-39018 AND MC-01033-39019 FOR THE BREAK AREAS SIZES.
3. REFERENCE STRESS REPORT NPE-905-138 AND SR-9.
GRAND GULF NUCLEAR STATION
UNIT 1
NUCLEAR PLANT ENGINEERING
HIGH ENERGY PIPE BREAK
REACTOR WATER CLEANUP SYSTEM
INSIDE CONTAINMENT-UNIT 1
SCALE: NONE
DRAWING NO. M-1556A
REV. 3
MPL NO. 1G33306SM
M1556A.DGN
AS-BUILT PER DCP 89/0063.
AS-BUILT PER MCP 89/1045.
SIDM-OFF ON RECORD
HIGH ENERGY
MODERATE ENERGY
THIS DWG POINT "B"
THIS DWG POINT "A"
REACTOR RECIRC. PUMP 'B' SUCTION LINE
REACTOR RECIRC. PUMP 'A' SUCTION LINE
M-1556B
M-1556C
M-1571
M-1557
(FROM 6"-DBA-9)
NO BREAK ZONE
EL. 118'-6"
EL. 101'-6"
EL. 103'-5 1/2"
EL. 110'-9 1/8"
CAM SUPPORT
ENERGY ABSORBING MATERIAL
RBFJ R1B
RBFU R1B
RBCU R1B
RBCL R1B
FW1B
FW2B
FW3B
FW4B
FW5B
FW6B
FW7B
FW8B
FW9B
FW10B
FW11B
FW12B
FW13B
FW14B
FW15B
FW16B
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FW310B
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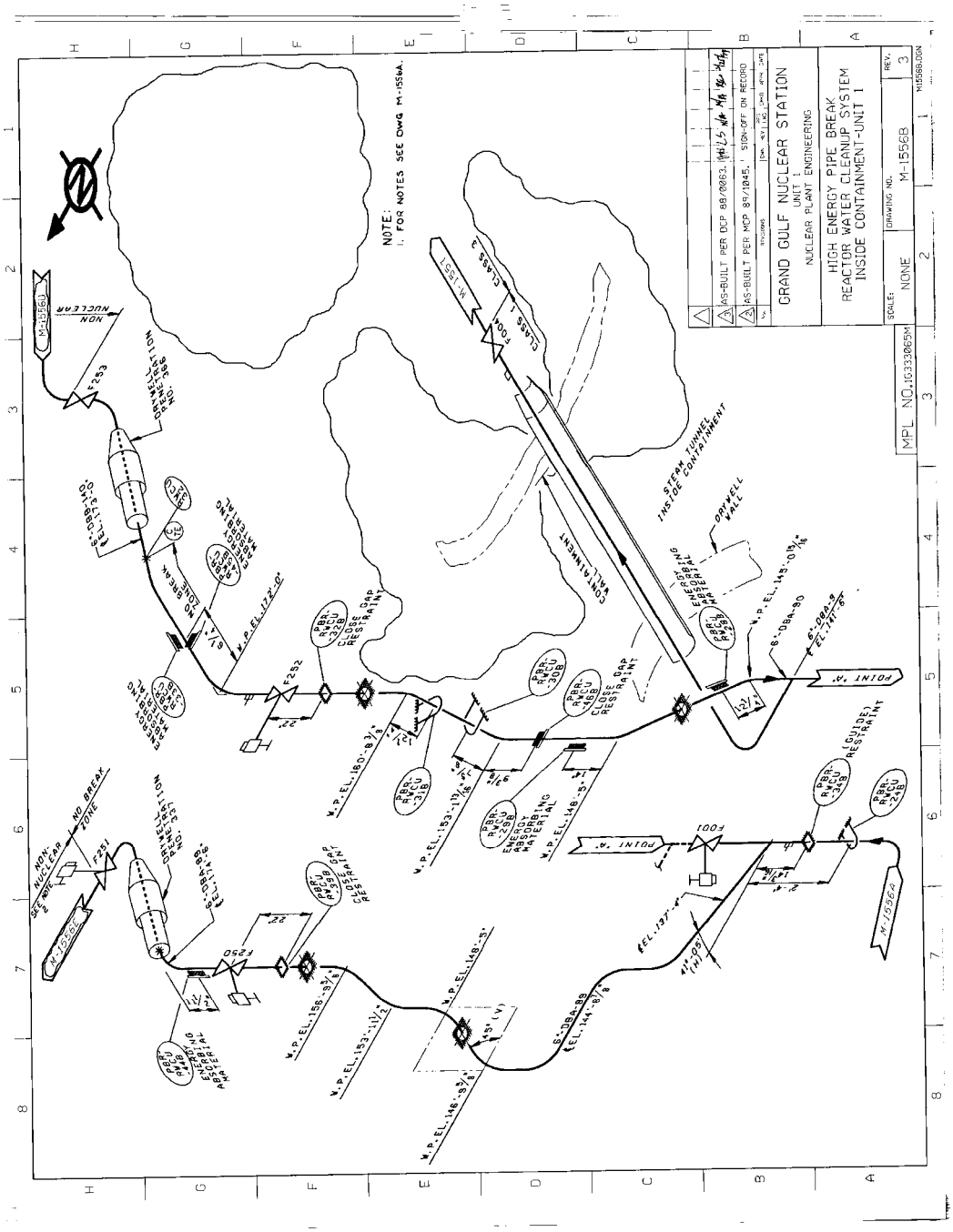
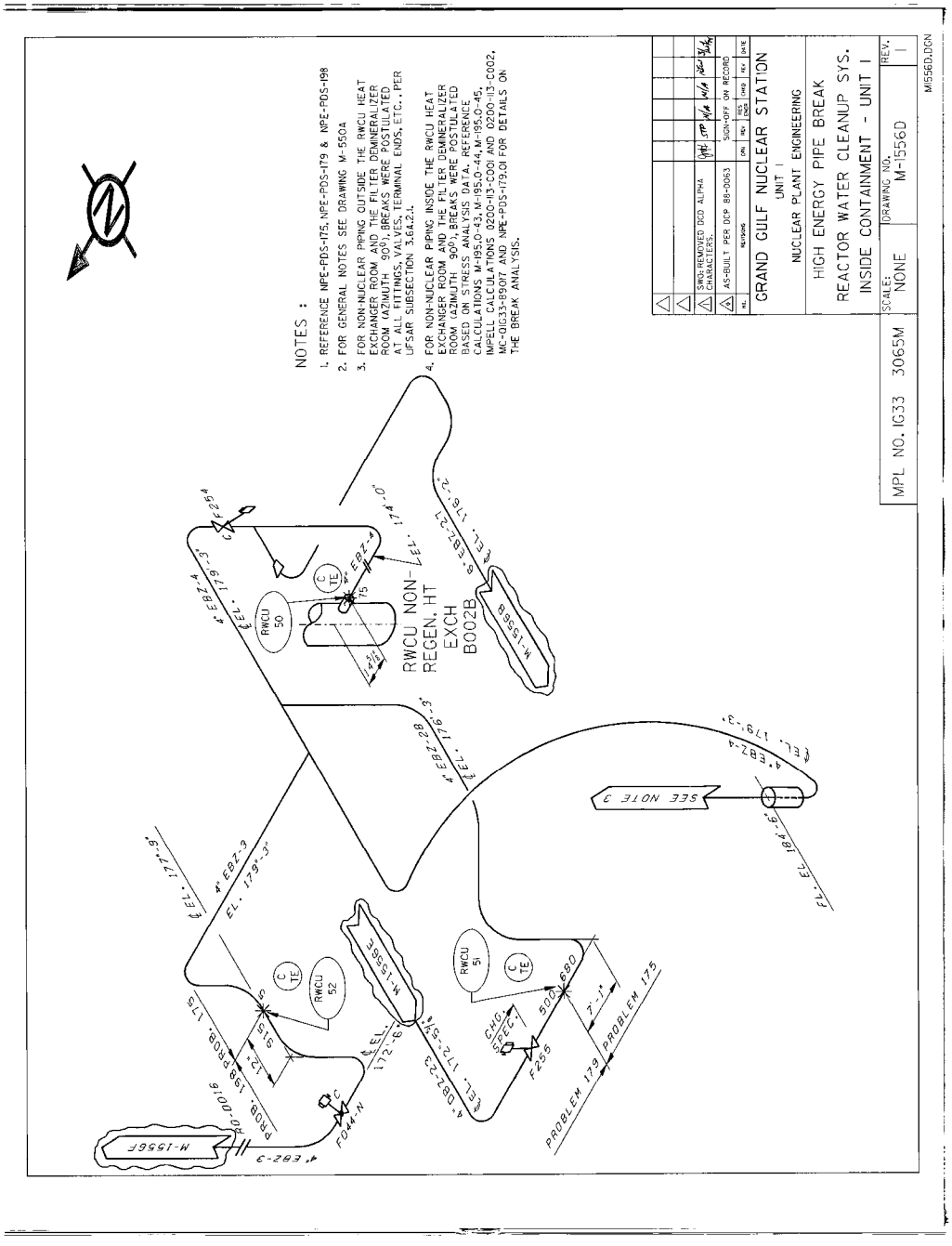
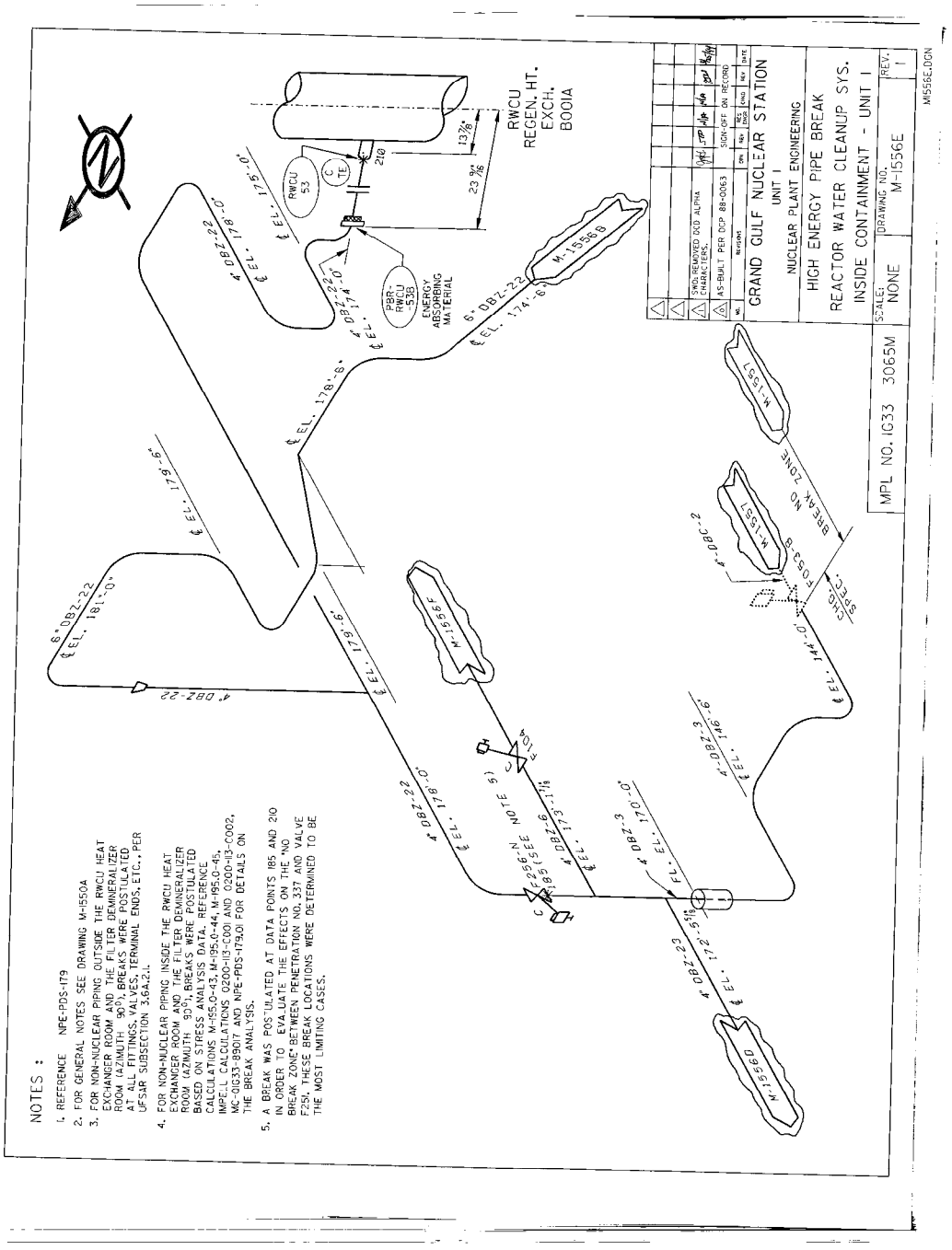
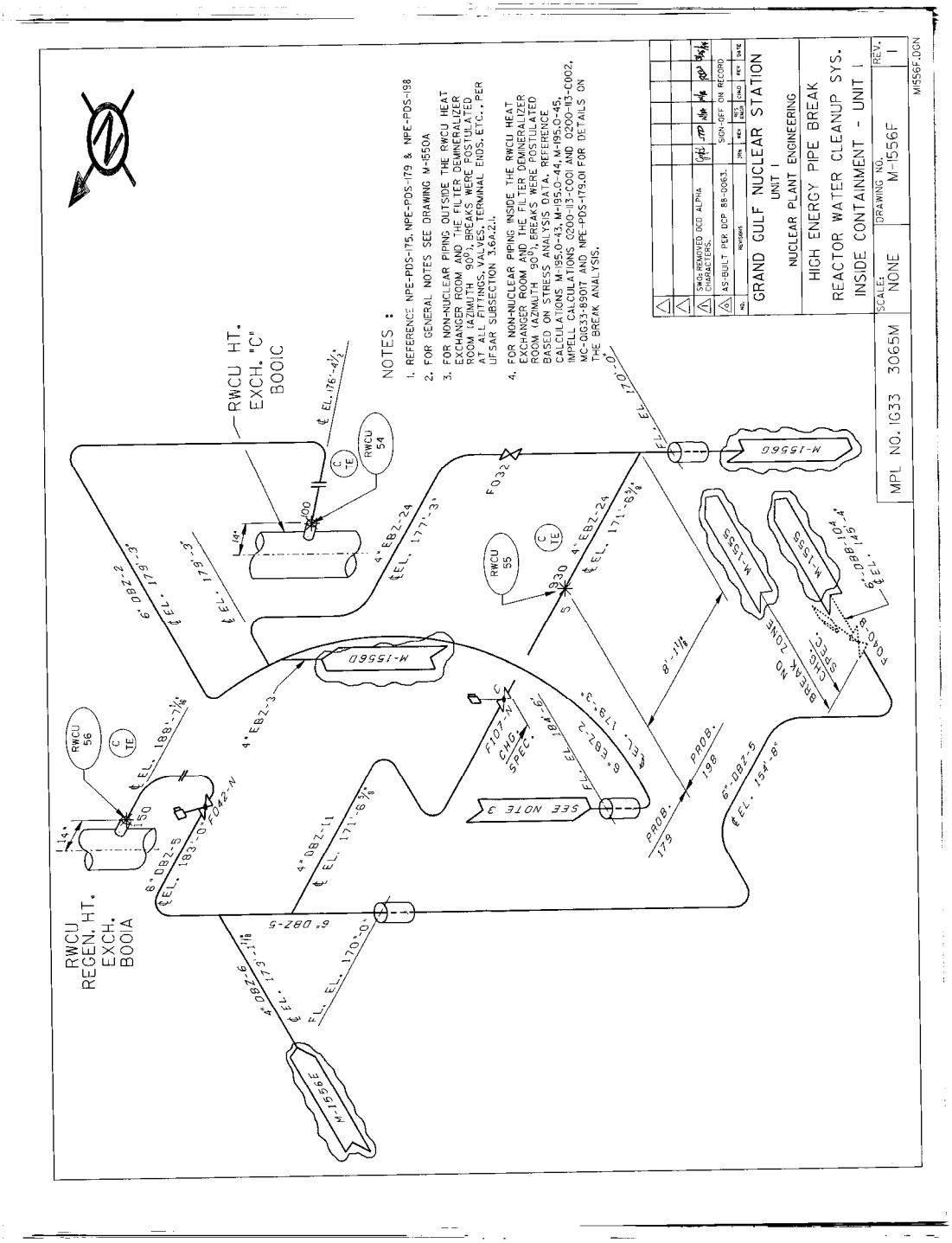
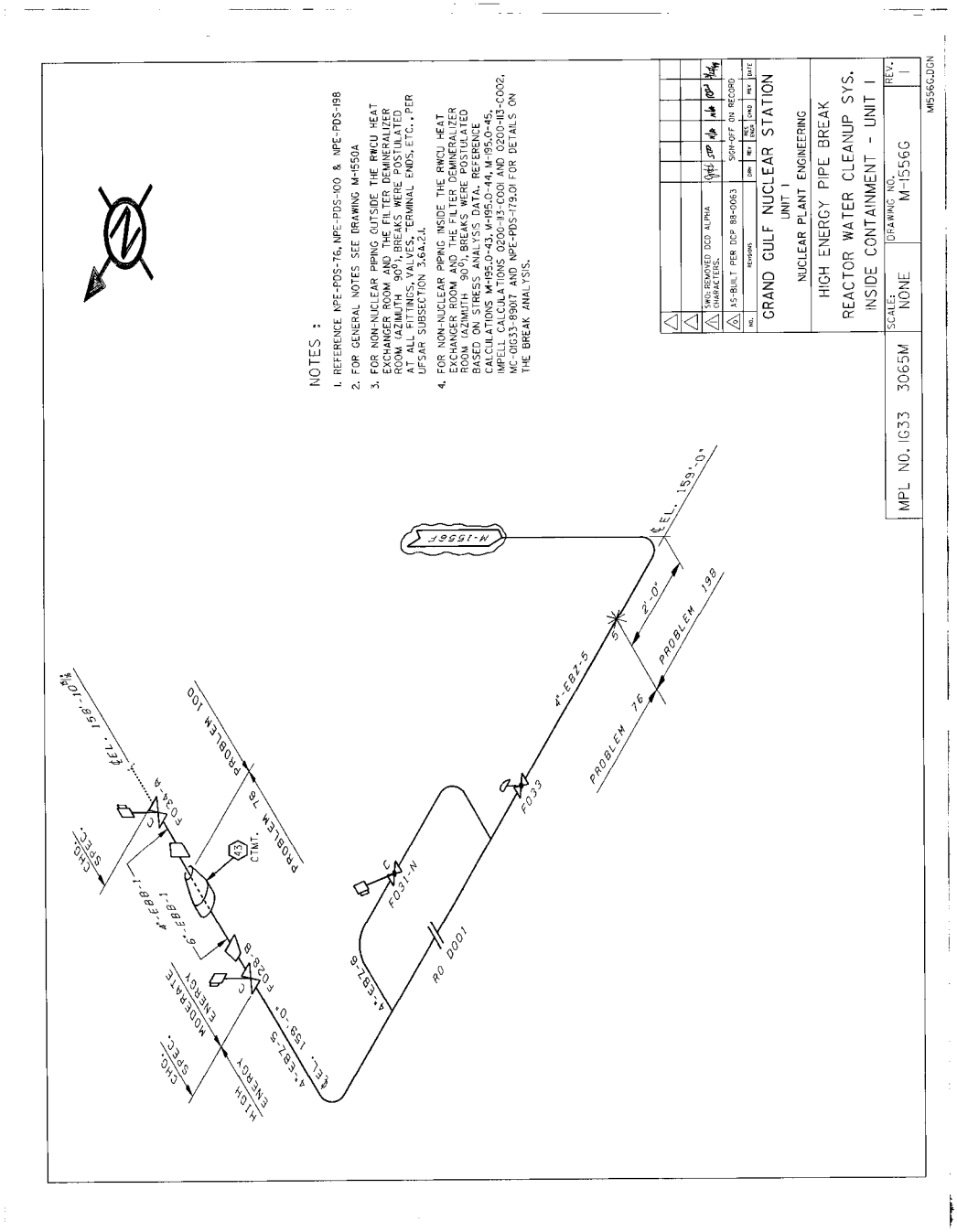
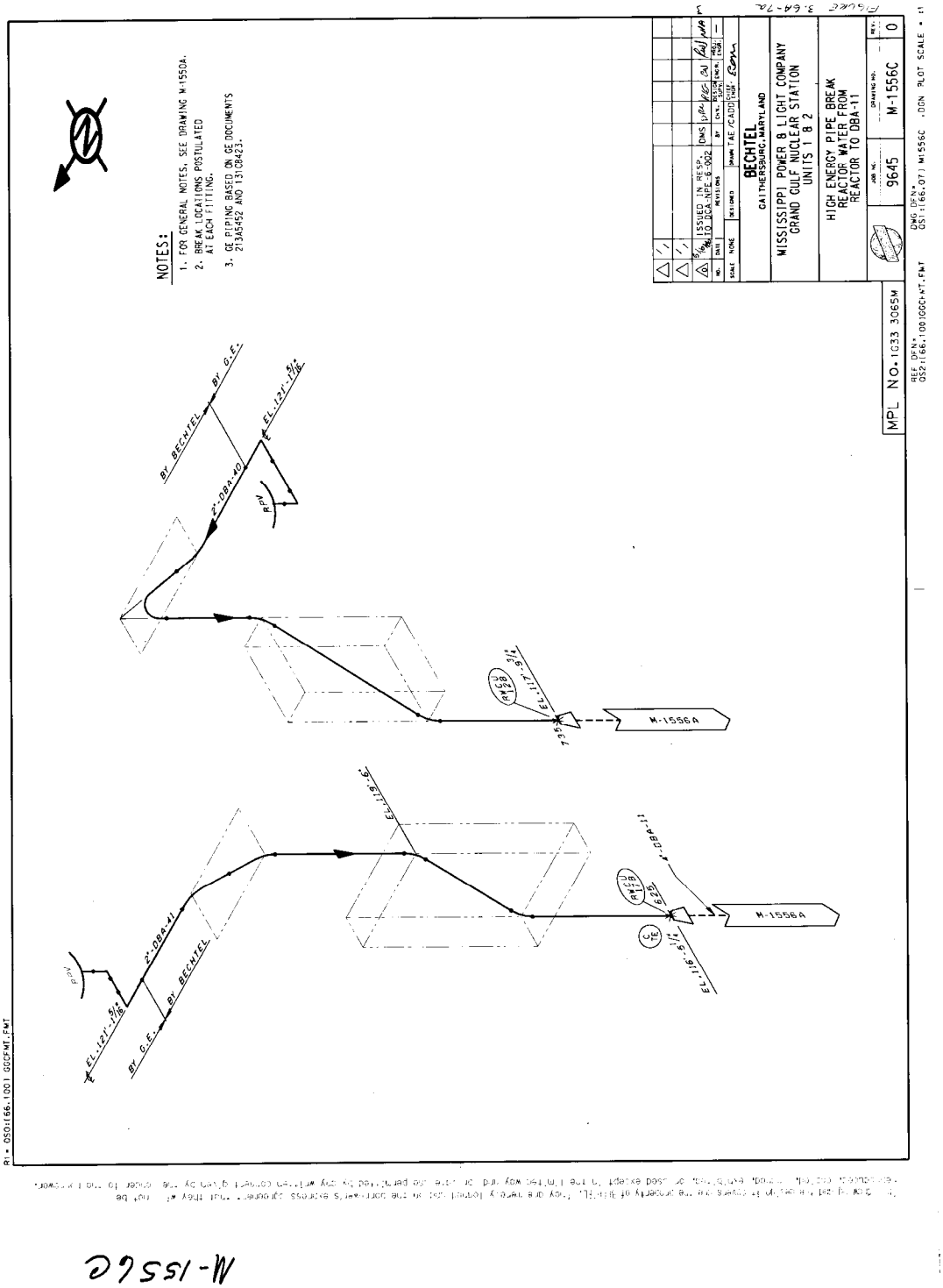

GRAND GULF NUCLEAR STATION
UNIT 1
NUCLEAR PLANT ENGINEERING
HIGH ENERGY PIPE BREAK
REACTOR WATER CLEANUP SYSTEM
INSIDE CONTAINMENT-UNIT 1
DRAWING NO. M-1556B
SCALE: NONE
REV. 3
NOTE:
1. FOR NOTES SEE DWG M-1556A.
NON NUCLEAR
NO BREAK ZONE
POINT 'A'
STEAM TUNNEL INSIDE CONTAINMENT
DRYWELL WALL
CONTAINMENT WALL
CLASS 1
CLASS 2
POOL
M-1556A
M-1556B
M-1556C


NOTES :
1. REFERENCE NPE-PDS-175, NPE-PDS-179 & NPE-PDS-198
2. FOR GENERAL NOTES SEE DRAWING M-550A
3. FOR NON-NUCLEAR PIPING OUTSIDE THE RWCU HEAT EXCHANGER ROOM AND THE FILTER DEMINERALIZER ROOM (AZIMUTH 90°), BREAKS WERE POSTULATED AT ALL FITTINGS, VALVES, TERMINAL ENDS, ETC., PER UFSAR SUBSECTION 3.6A.2.4.
4. FOR NON-NUCLEAR PIPING INSIDE THE RWCU HEAT EXCHANGER ROOM AND THE FILTER DEMINERALIZER ROOM (AZIMUTH 90°), BREAKS WERE POSTULATED BASED ON STRESS ANALYSIS DATA. REFERENCE CALCULATIONS M-195.0-43, M-195.0-44, M-195.0-45, IMPELL CALCULATIONS 0200-113-C001 AND 0200-113-C002, MC-0103-89017 AND NPE-PDS-179.0 FOR DETAILS ON THE BREAK ANALYSIS.
SWOS REMOVED DCD ALPHA CHARACTERS.
AS-BUILT PER DCP 88-0063
REVISIONS
SIGN-OFF ON RECORD
GRAND GULF NUCLEAR STATION
UNIT 1
NUCLEAR PLANT ENGINEERING
HIGH ENERGY PIPE BREAK
REACTOR WATER CLEANUP SYS.
INSIDE CONTAINMENT - UNIT 1
MPL NO. IG33 3065M
SCALE: NONE
DRAWING NO. M-1556D
REV. 1
M1556D.DGN
RWCU NON-REGEN. HT EXCH B002B
RWCU 50
RWCU 52
RWCU 51
M-1556B
M-1556E
M-1556F
4"-EBZ-4
4"-EBZ-3
4"-EBZ-28
4"-EBZ-23
4"-EBZ-2
EL. 171'-9"
EL. 179'-3"
EL. 176'-3"
EL. 172'-5½"
EL. 179'-3"
EL. 181'-6"
SEE NOTE 3
PROB. 180 PROB. 175
PROBLEM 179
PROBLEM 175
CHG. SPEC.
F235
F044-N
5"x3"
6"x4"
7'-1"
9'-11"
12'-2"

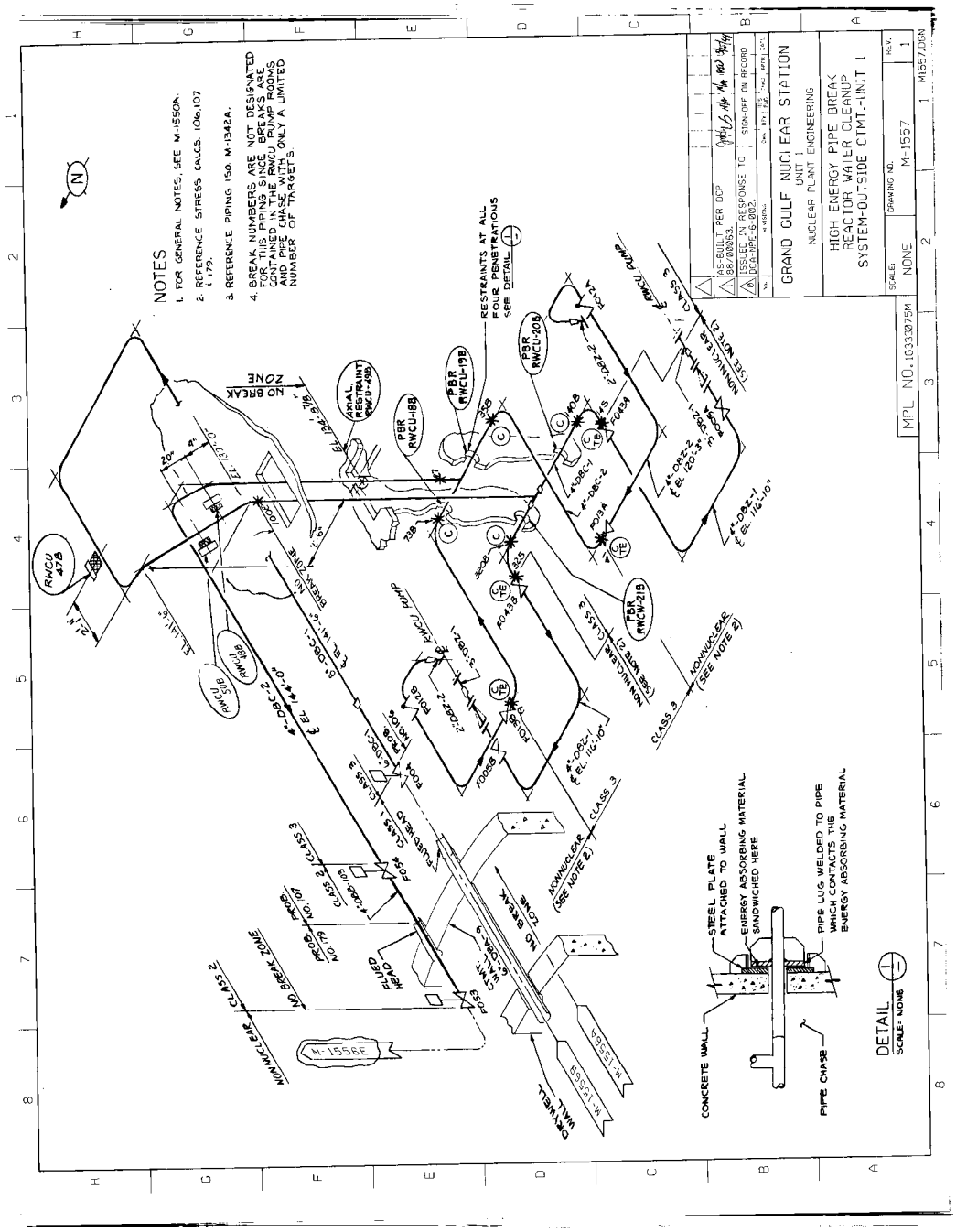

NOTES :
1. REFERENCE NPE-PDS-179
2. FOR GENERAL NOTES SEE DRAWING M-1550A
3. FOR NON-NUCLEAR PIPING OUTSIDE THE RWCU HEAT EXCHANGER ROOM AND THE FILTER DEMINERALIZER ROOM (AZIMUTH 90°), BREAKS WERE POSTULATED AT ALL FITTINGS, VALVES, TERMINAL ENDS, ETC., PER UFSAR SUBSECTION 3.6A.2.1.
4. FOR NON-NUCLEAR PIPING INSIDE THE RWCU HEAT EXCHANGER ROOM AND THE FILTER DEMINERALIZER ROOM (AZIMUTH 90°), BREAKS WERE POSTULATED BASED ON STRESS ANALYSIS DATA, REFERENCE CALCULATIONS M-155.0-43, M-195.0-44, M-195.0-45, IMPELL CALCULATIONS 0200-113-C00 AND 0200-113-C002, MC-0G33-89017 AND NPE-PDS-179.01 FOR DETAILS ON THE BREAK ANALYSIS.
5. A BREAK WAS POSTULATED AT DATA POINTS 185 AND 210 IN ORDER TO EVALUATE THE EFFECTS ON THE "NO BREAK ZONE" BETWEEN PENETRATION NO. 337 AND VALVE F251. THESE BREAK LOCATIONS WERE DETERMINED TO BE THE MOST LIMITING CASES.
RWCU REGEN. HT. EXCH. B001A
RWCU 53
PBR-RWCU 538
ENERGY ABSORBING MATERIAL
M-1556B
M-1556T
M-1556F
M-1556D
BREAK NO ZONE
SEE NOTE 5
GRAND GULF NUCLEAR STATION
UNIT 1
NUCLEAR PLANT ENGINEERING
HIGH ENERGY PIPE BREAK
REACTOR WATER CLEANUP SYS.
INSIDE CONTAINMENT - UNIT 1
MPL NO. 1G33 3065M
SCALE: NONE
DRAWING NO. M-1556E
REV. 1
AS-BUILT PER DCP 88-0063
SIGN-OFF ON RECORD
M1556E.DGN

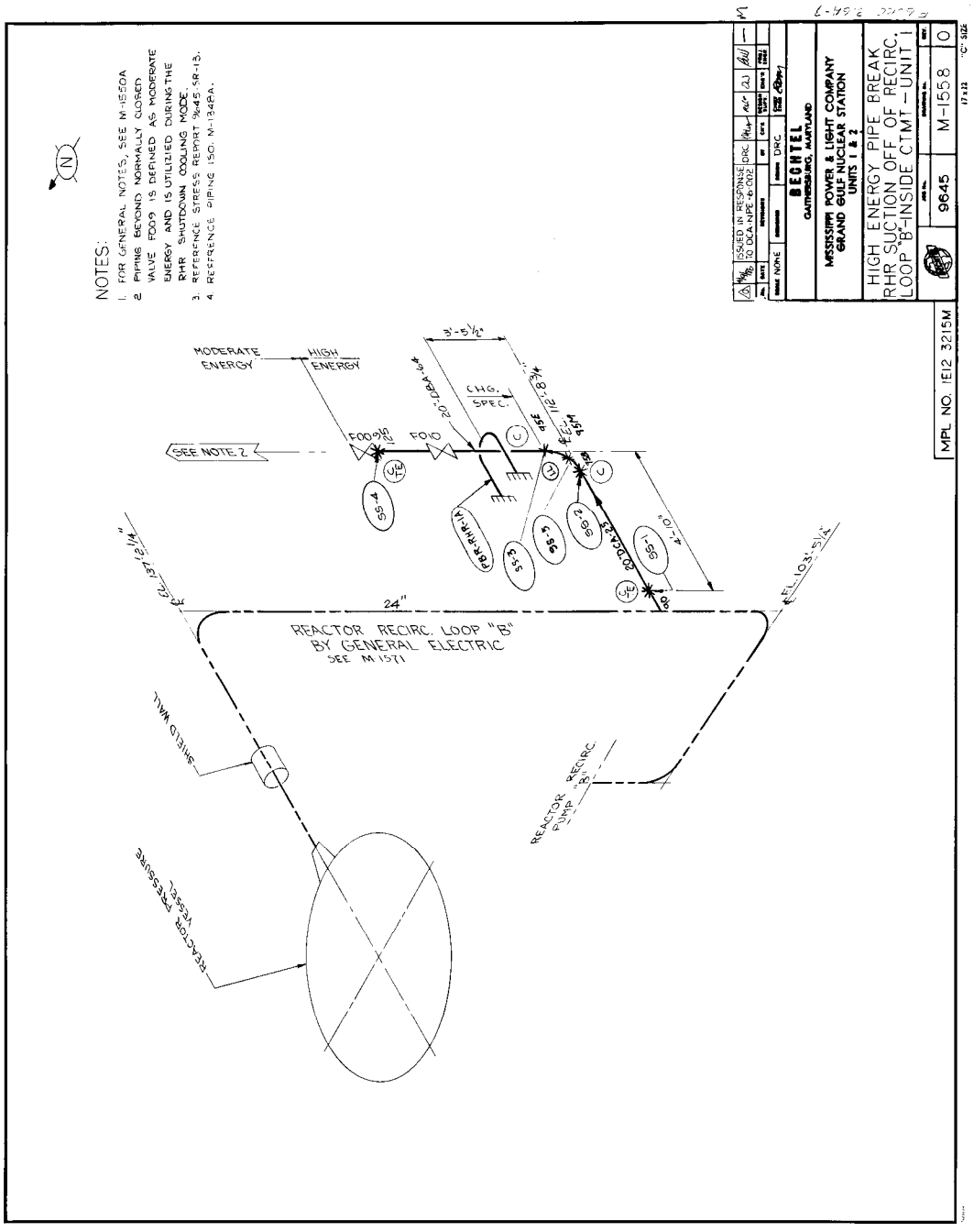

NOTES :
1. REFERENCE NPE-PDS-175, NPE-PDS-179 & NPE-PDS-198
2. FOR GENERAL NOTES SEE DRAWING M-1550A
3. FOR NON-NUCLEAR PIPING OUTSIDE THE RWCU HEAT EXCHANGER ROOM AND THE FILTER DEMINERALIZER ROOM (AZIMUTH 90°), BREAKS WERE POSTULATED AT ALL FITTINGS, VALVES, TERMINAL ENDS, ETC., PER UFSAR SUBSECTION 3.6A.2.1.
4. FOR NON-NUCLEAR PIPING INSIDE THE RWCU HEAT EXCHANGER ROOM AND THE FILTER DEMINERALIZER ROOM (AZIMUTH 90°), BREAKS WERE POSTULATED BASED ON STRESS ANALYSIS DATA. REFERENCE CALCULATIONS M-195.0-43, M-195.0-44, M-195.0-45, IMPELL CALCULATIONS 0200-113-C001 AND 0200-113-C002, MC-0IG33-B901T AND NPE-PDS-179.01 FOR DETAILS ON THE BREAK ANALYSIS.
RWCU HT. EXCH. "C" B001C
RWCU REGEN. HT. EXCH. B001A
RWCU 54
RWCU 55
RWCU 56
M-1556G
M-1556D
M-1555
BREAK NO. 2 ZONE
CHG. SPEC.
SEE NOTE 3
PROB. 179
PROB. 198
GRAND GULF NUCLEAR STATION
UNIT 1
NUCLEAR PLANT ENGINEERING
HIGH ENERGY PIPE BREAK
REACTOR WATER CLEANUP SYS.
INSIDE CONTAINMENT - UNIT 1
SCALE: NONE
DRAWING NO. M-1556F
REV. 1
MPL NO. 1G33 3065M
M556F.DGN


NOTES :
1. REFERENCE NPE-PDS-76, NPE-PDS-100 & NPE-PDS-198
2. FOR GENERAL NOTES SEE DRAWING M-1550A
3. FOR NON-NUCLEAR PIPING OUTSIDE THE RWCU HEAT EXCHANGER ROOM AND THE FILTER DEMINERALIZER ROOM (AZIMUTH 90°), BREAKS WERE POSTULATED AT ALL FITTINGS, VALVES, TERMINAL ENDS, ETC., PER UFSAR SUBSECTION 3.6A.2.1.
4. FOR NON-NUCLEAR PIPING INSIDE THE RWCU HEAT EXCHANGER ROOM AND THE FILTER DEMINERALIZER ROOM (AZIMUTH 90°), BREAKS WERE POSTULATED BASED ON STRESS ANALYSIS DATA. REFERENCE CALCULATIONS M-195.0-43, M-195.0-44, M-95.0-45, IMPELL CALCULATIONS 0200-113-C001 AND 0200-113-C002, MC-0133-8907 AND NPE-PDS-179.01 FOR DETAILS ON THE BREAK ANALYSIS.
SWG REMOVED DCD ALPHA CHARACTERS.
AS-BUILT PER DCP 88-0063
SIGN-OFF ON RECORD
REVISIONS
GRAND GULF NUCLEAR STATION
UNIT 1
NUCLEAR PLANT ENGINEERING
HIGH ENERGY PIPE BREAK
REACTOR WATER CLEANUP SYS.
INSIDE CONTAINMENT - UNIT 1
MPL NO. 1G33 3065M
SCALE: NONE
DRAWING NO. M-1556G
REV. 1
M1556G.DGN
N-1556F
EL. 159'-0"
2'-0"
PROBLEM 198
PROBLEM 76
4"-EBZ-5
FD33
FD31-N
RD DD01
4"-EBZ-5
EL. 159'-0"
F028
6"-EBB-1
F034
EL. 158'-10 1/2"
PROBLEM 100
CTMT.
SHOCK SPEC.
MODERATE ENERGY
HIGH ENERGY
CMD. SPEC.


NOTES:
1. FOR GENERAL NOTES, SEE DRAWING M-1550A.
2. BREAK LOCATIONS POSTULATED AT EACH FITTING.
3. GE PIPING BASED ON GE DOCUMENTS 21345452 AND 131C8423.
BECHTEL
GAITHERSBURG, MARYLAND
MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
HIGH ENERGY PIPE BREAK
REACTOR WATER FROM
REACTOR TO DBA-11
DRAWING NO. M-1556C
REV. 0
JOB NO. 9645
MPL NO. 1033 3065M
BY BECHTEL
BY G.E.
EL. 121'-5 1/8"
2"-DBA-40
RPV
M-1556A
EL. 119'-6"
EL. 119'-8 1/4"
2"-DBA-41

M-1556C


NOTES
1. FOR GENERAL NOTES, SEE M-1550A.
2. REFERENCE STRESS CALCS. 106.107 & 179.
3. REFERENCE PIPING ISO. M-1342A.
4. BREAK NUMBERS ARE NOT DESIGNATED FOR THIS PIPING SINCE BREAKS ARE ANTICIPATED IN THE RWCU PUMP ROOMS AND PIPE WHIP RESTRAINTS ONLY A LIMITED NUMBER OF TARGETS
RESTRAINTS AT ALL FOUR PENETRATIONS SEE DETAIL
NO BREAK ZONE
RWCU 47B
RWCU 48B
PBR RWCU-49S
PBR RWCU-19B
PBR RWCU-20S
PBR RWCU-21B
NONNUCLEAR CLASS 3
NONNUCLEAR (SEE NOTE 2)
CLASS 3
CLASS 2
CLASS 1
M-1556E
M-1556B
M-1556A
OVERHEAD WALL
DETAIL
SCALE: NONE
STEEL PLATE ATTACHED TO WALL
ENERGY ABSORBING MATERIAL SANDWICHED HERE
PIPE LUG WELDED TO PIPE WHICH CONTACTS THE ENERGY ABSORBING MATERIAL
CONCRETE WALL
PIPE CHASE
AS-BUILT PER DCP 88-2005
ISSUED IN RESPONSE TO DCA-HPE-G-0002
STOP-OFF ON RECORD
GRAND GULF NUCLEAR STATION
UNIT 1
NUCLEAR PLANT ENGINEERING
HIGH ENERGY PIPE BREAK
REACTOR WATER CLEANUP
SYSTEM-OUTSIDE CTMT.-UNIT 1
SCALE: NONE
DRAWING NO. M-1557
REV. 1
MPL NO. 103330750
M1557.DGN



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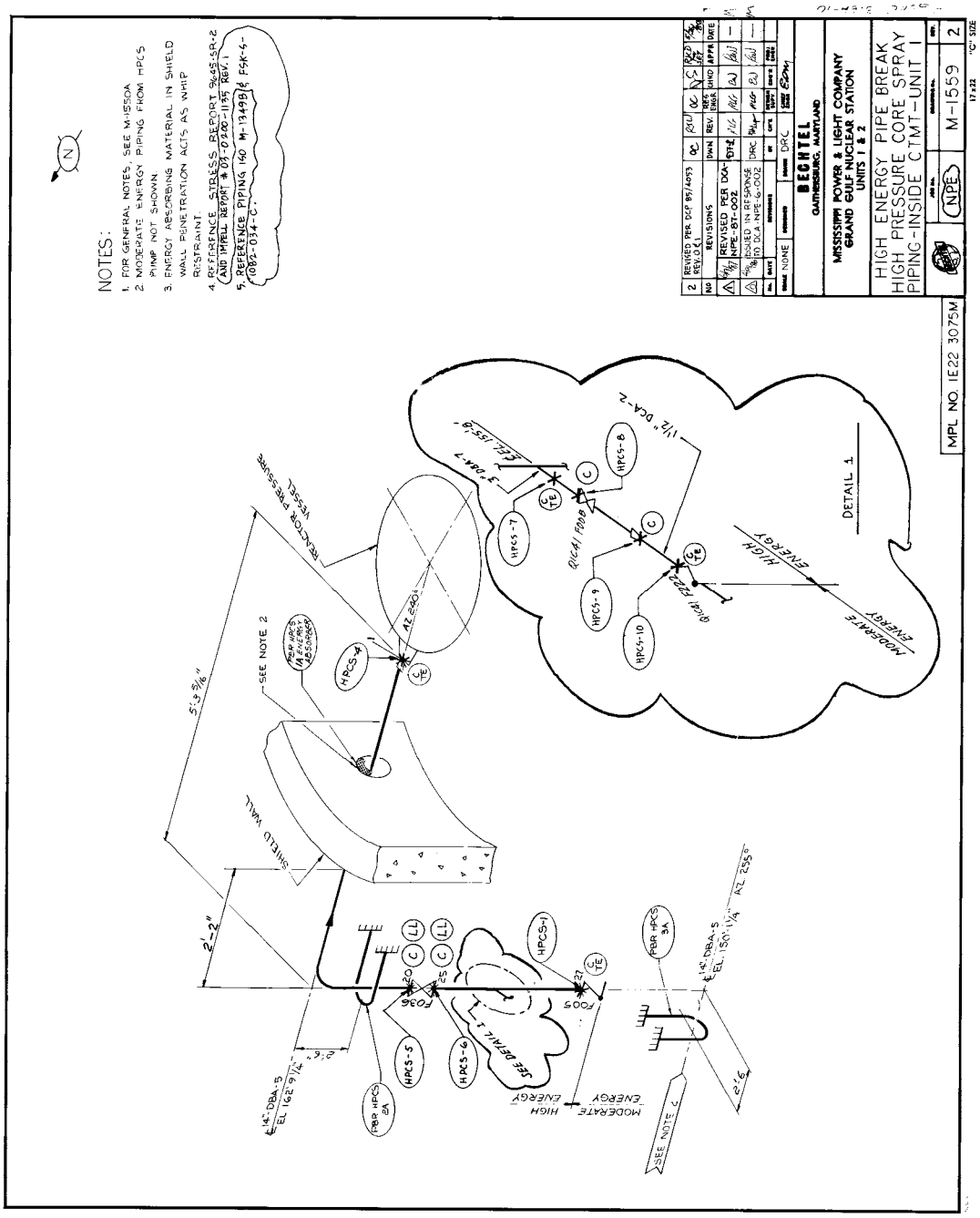
M-1558



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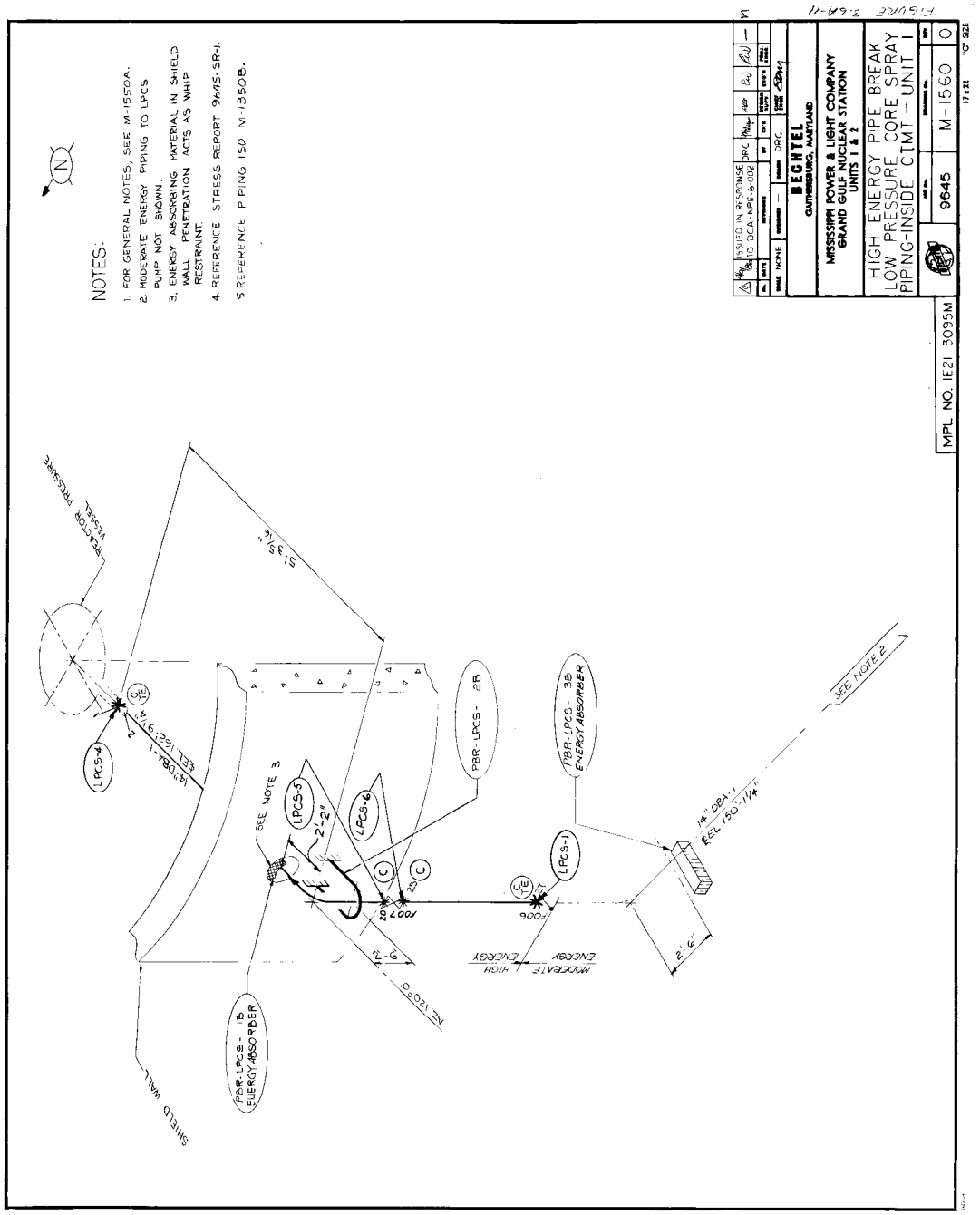
M-1559



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M-1560

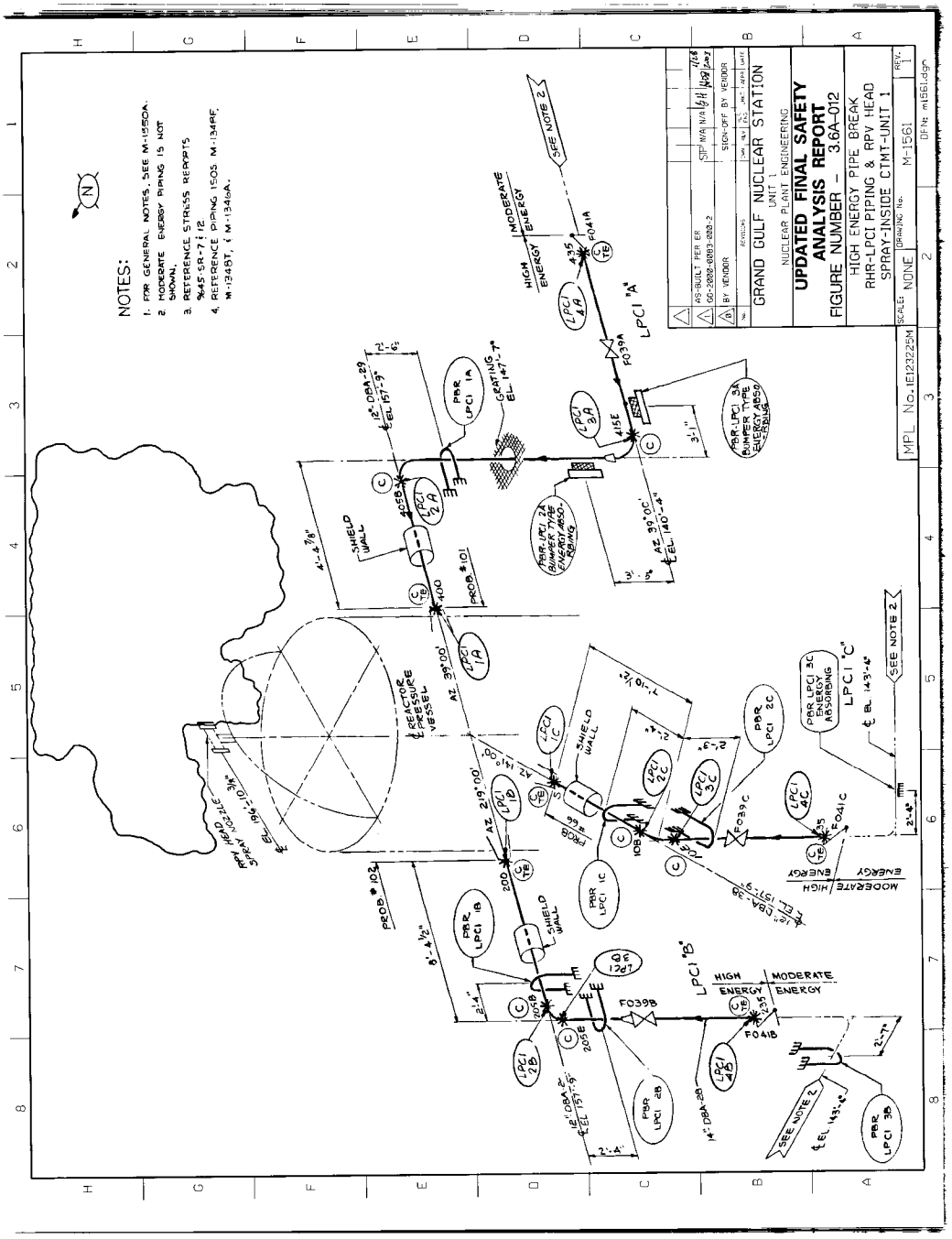
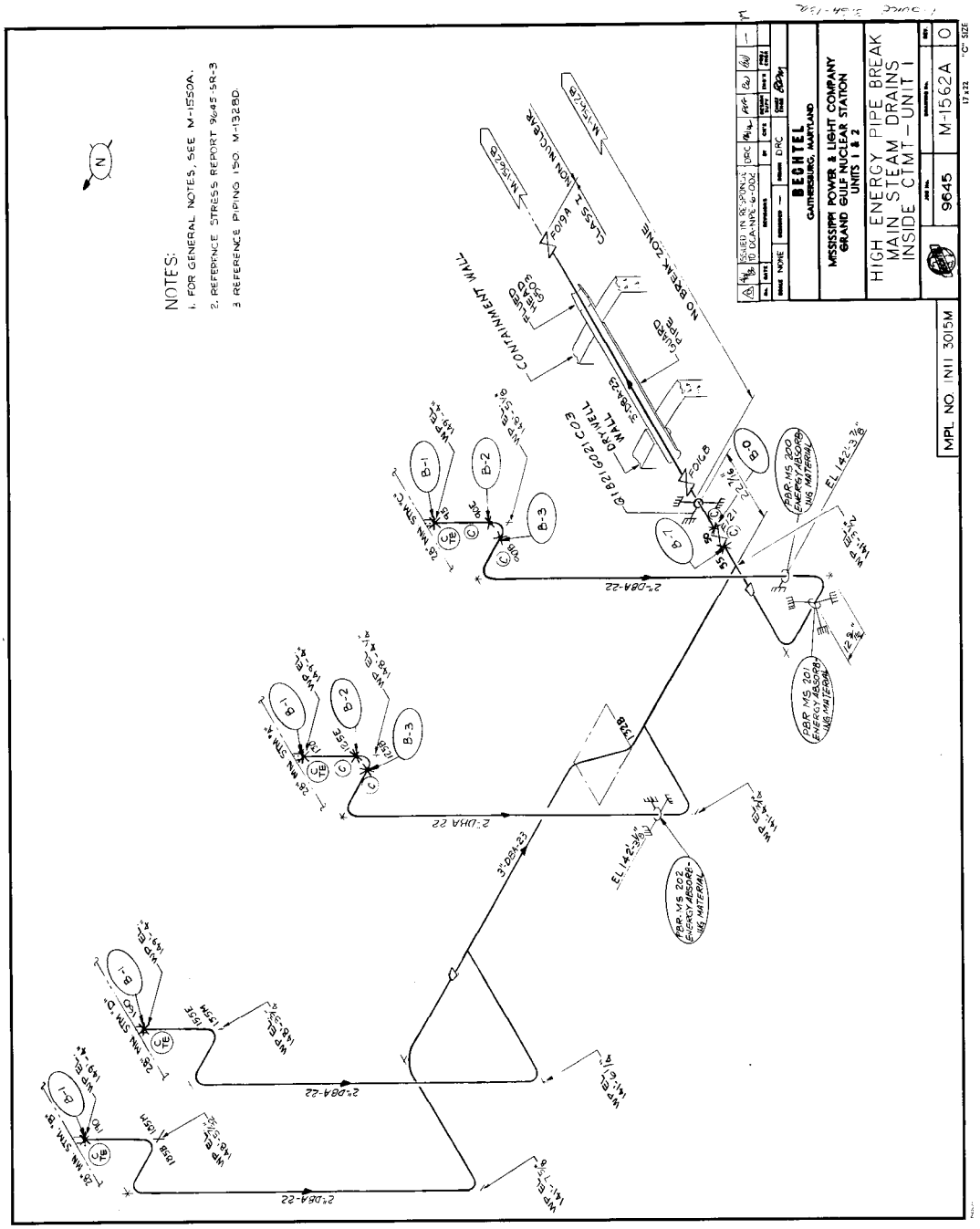

NOTES:
1. FOR GENERAL NOTES, SEE M-1150A.
2. MODERATE ENERGY PIPING IS NOT SHOWN.
3. REFERENCE STRESS REPORTS 945.58-7 & 12
4. REFERENCE PIPING ISOS M-1346F, M-1346T, & M-1346A.
UPDATED FINAL SAFETY ANALYSIS REPORT
FIGURE NUMBER – 3.6A-012
HIGH ENERGY PIPE BREAK
RHR-LPCI PIPING & RPV HEAD SPRAY-INSIDE CTMT-UNIT 1
GRAND GULF NUCLEAR STATION
UNIT 1
NUCLEAR PLANT ENGINEERING
LPCI "A"
LPCI "B"
LPCI "C"
REACTOR PRESSURE VESSEL
SHIELD WALL
PROB #101
PROB #102
HIGH ENERGY
MODERATE ENERGY
SEE NOTE 2
GRATING EL. 147'-7"
PBR LPCI 1A
PBR LPCI 1B
PBR LPCI 1C
PBR LPCI 2A
PBR LPCI 2B
PBR LPCI 2C
PBR LPCI 3A
PBR LPCI 3B
PBR LPCI 3C
PBR LPCI 3A BUMPER TYPE ENERGY ABSORBING
PBR LPCI 3C ENERGY ABSORBING
F039A
F039B
F039C
F041A
F041B
F041C
MPL No. IE12325M
DRAWING No. M-1561
SCALE: NONE
DFN: m1561.dgn
REV. 1

Figure 3.6A-13

DELETED



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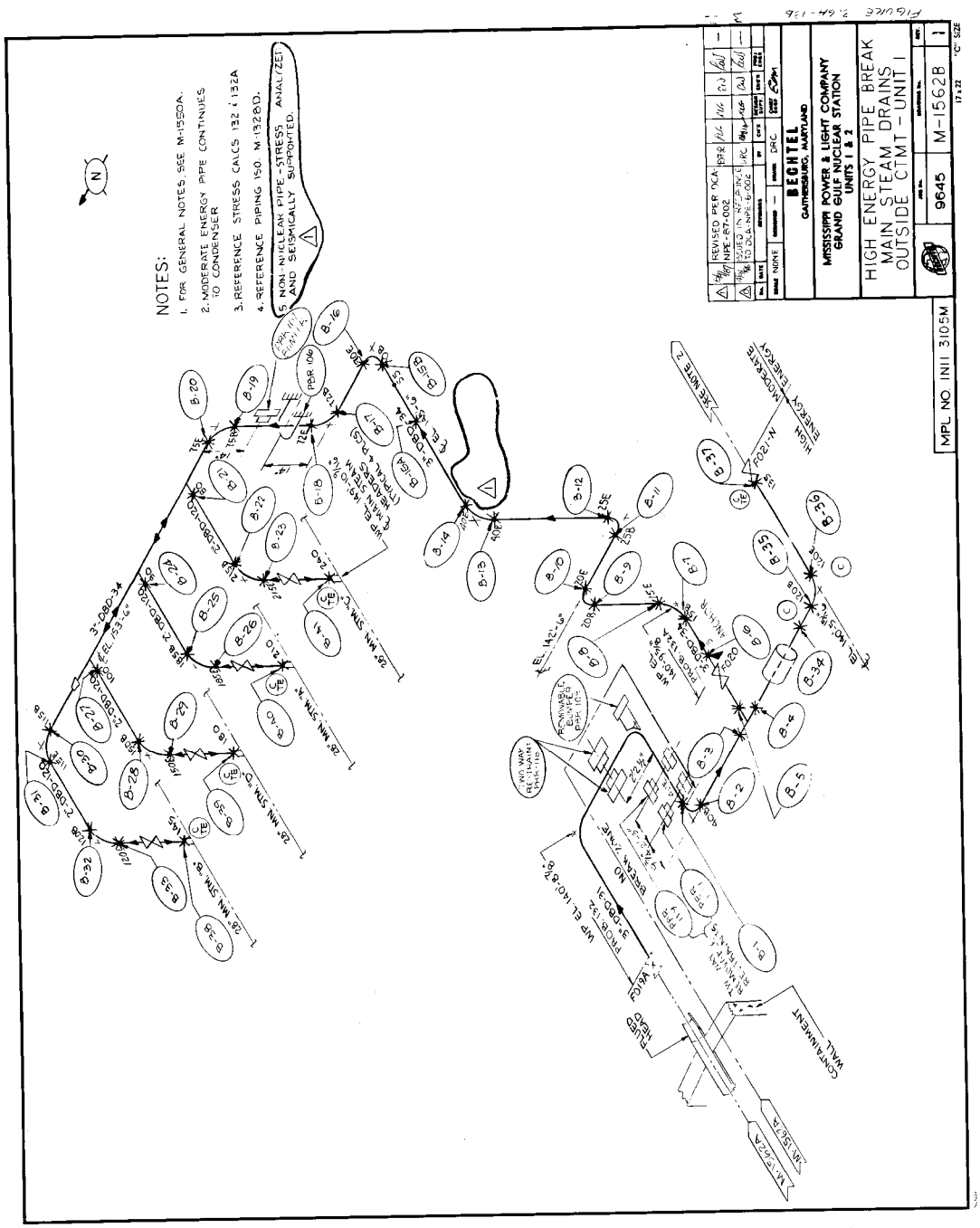
M-1562A


NOTES:
1. FOR GENERAL NOTES, SEE M-1150A.
2. MODERATE ENERGY PIPE CONTINUES TO CONDENSER
3. REFERENCE STRESS CALCS 132 & 132A
4. REFERENCE PIPING ISO. M-132BD.
5. NON-NUCLEAR PIPE-STRESS ANAL/ZED AND SEISMICALLY SUPPORTED.
BECHTEL
GAITHERSBURG, MARYLAND
MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
HIGH ENERGY PIPE BREAK
MAIN STEAM DRAINS
OUTSIDE CTMT - UNIT 1
9645
M-1562B
MPL NO. INII 3105M
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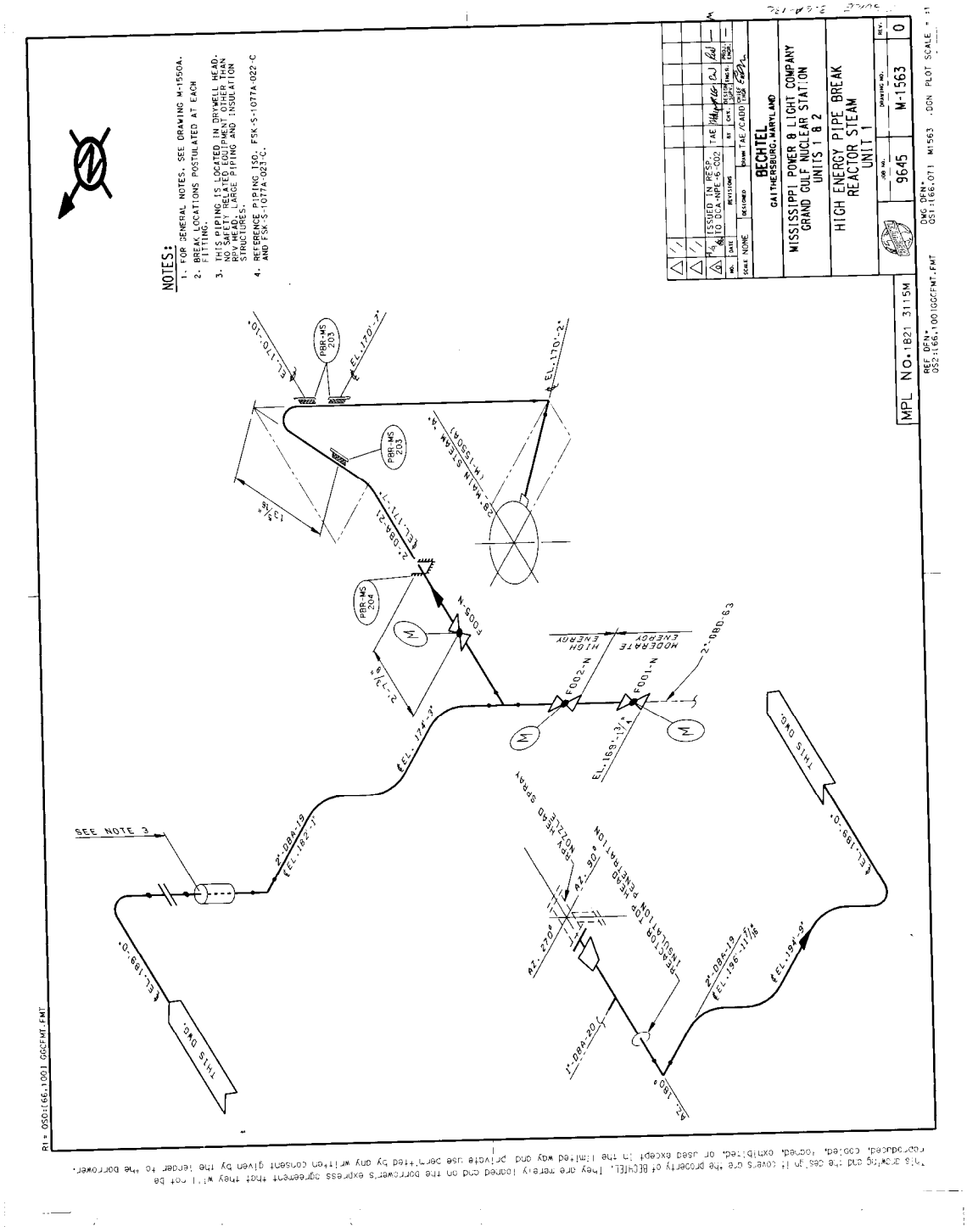
M-1562B



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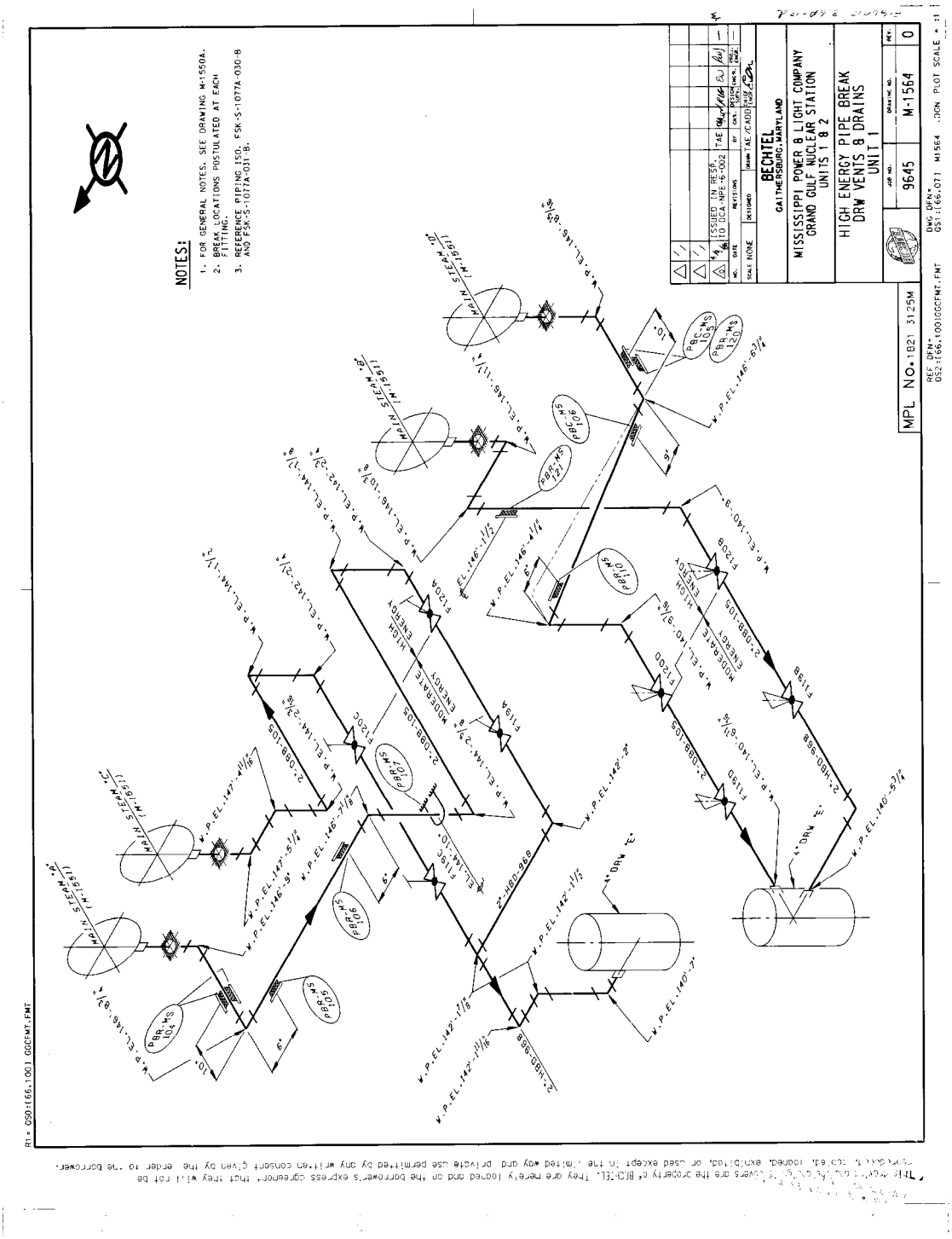
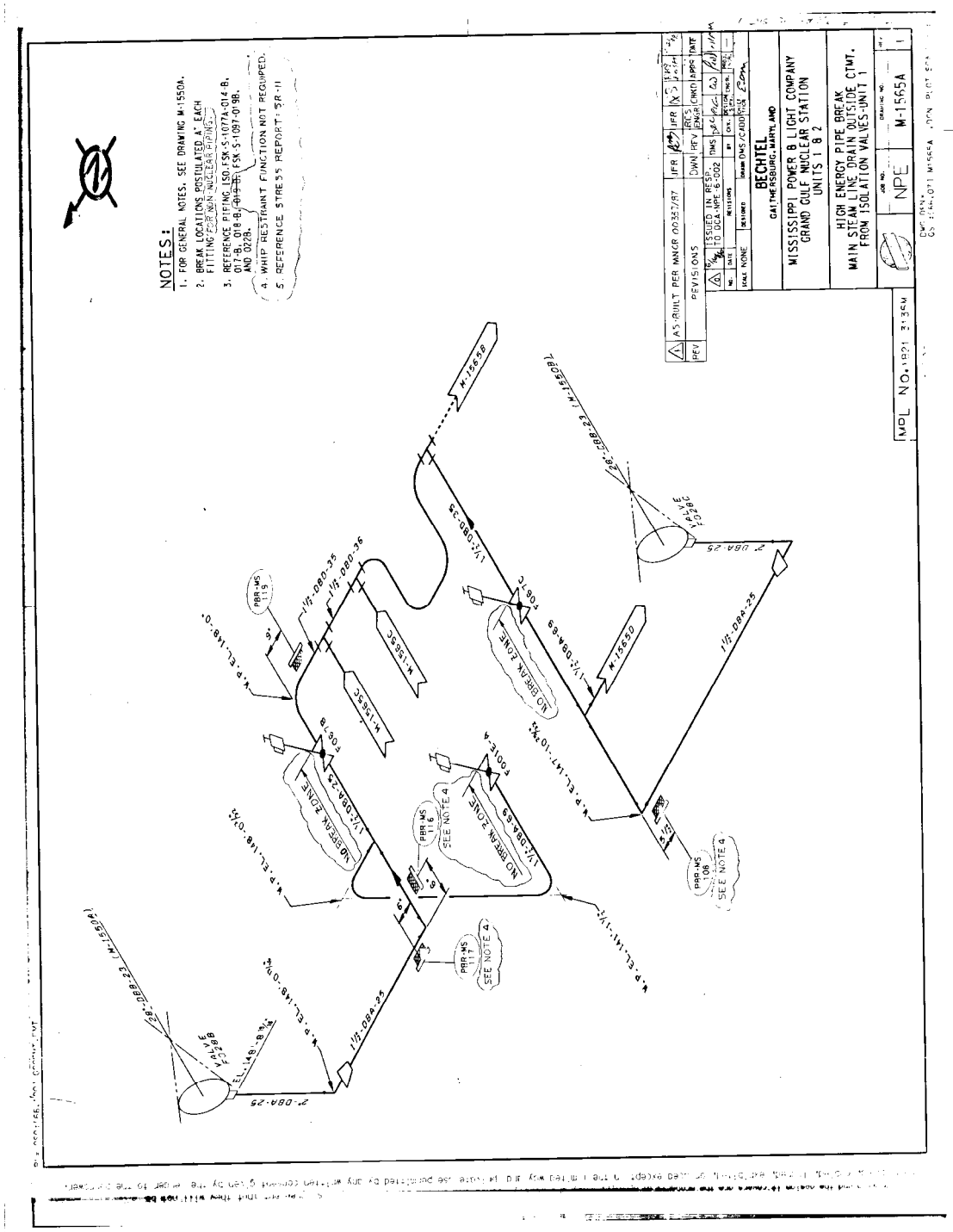
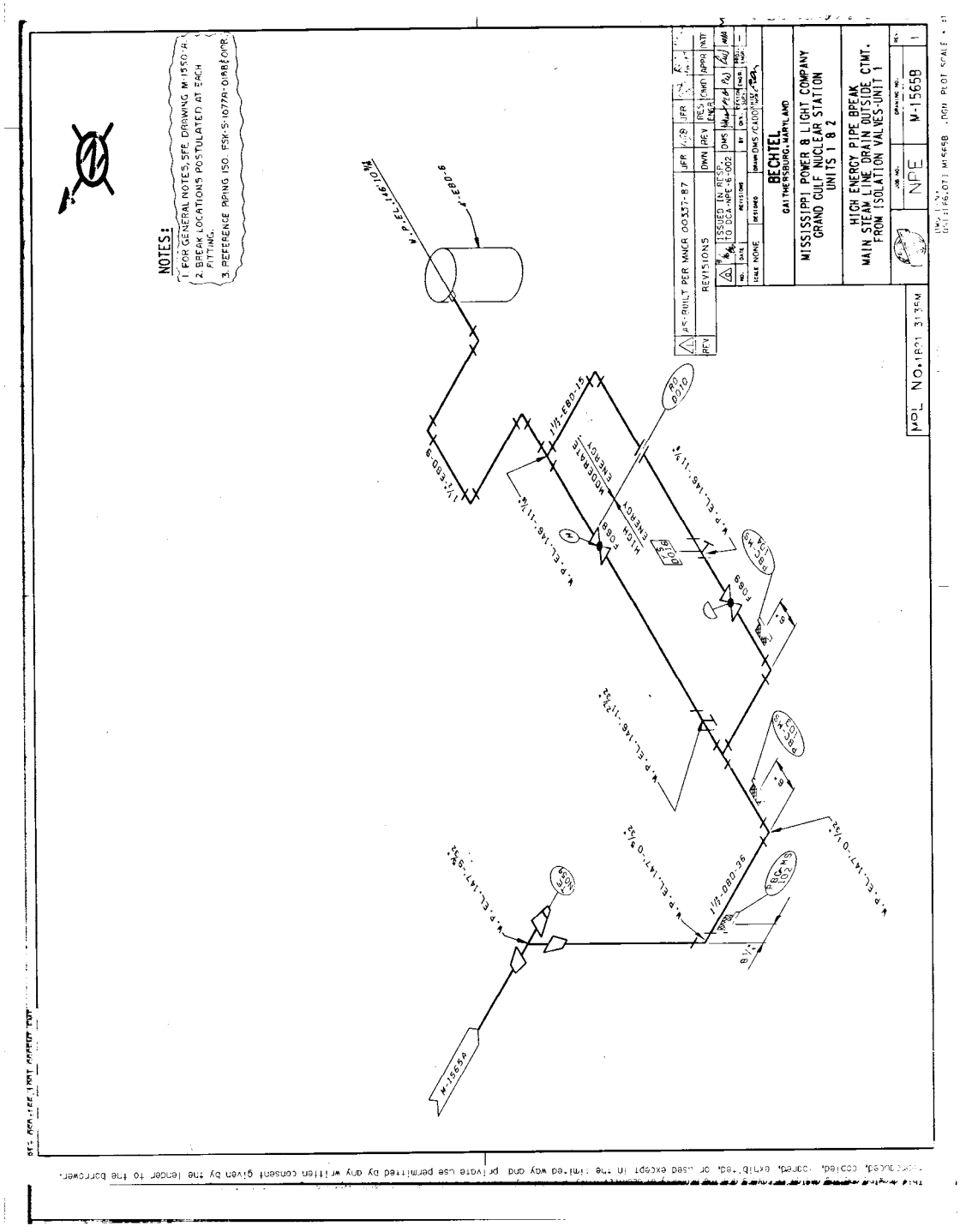
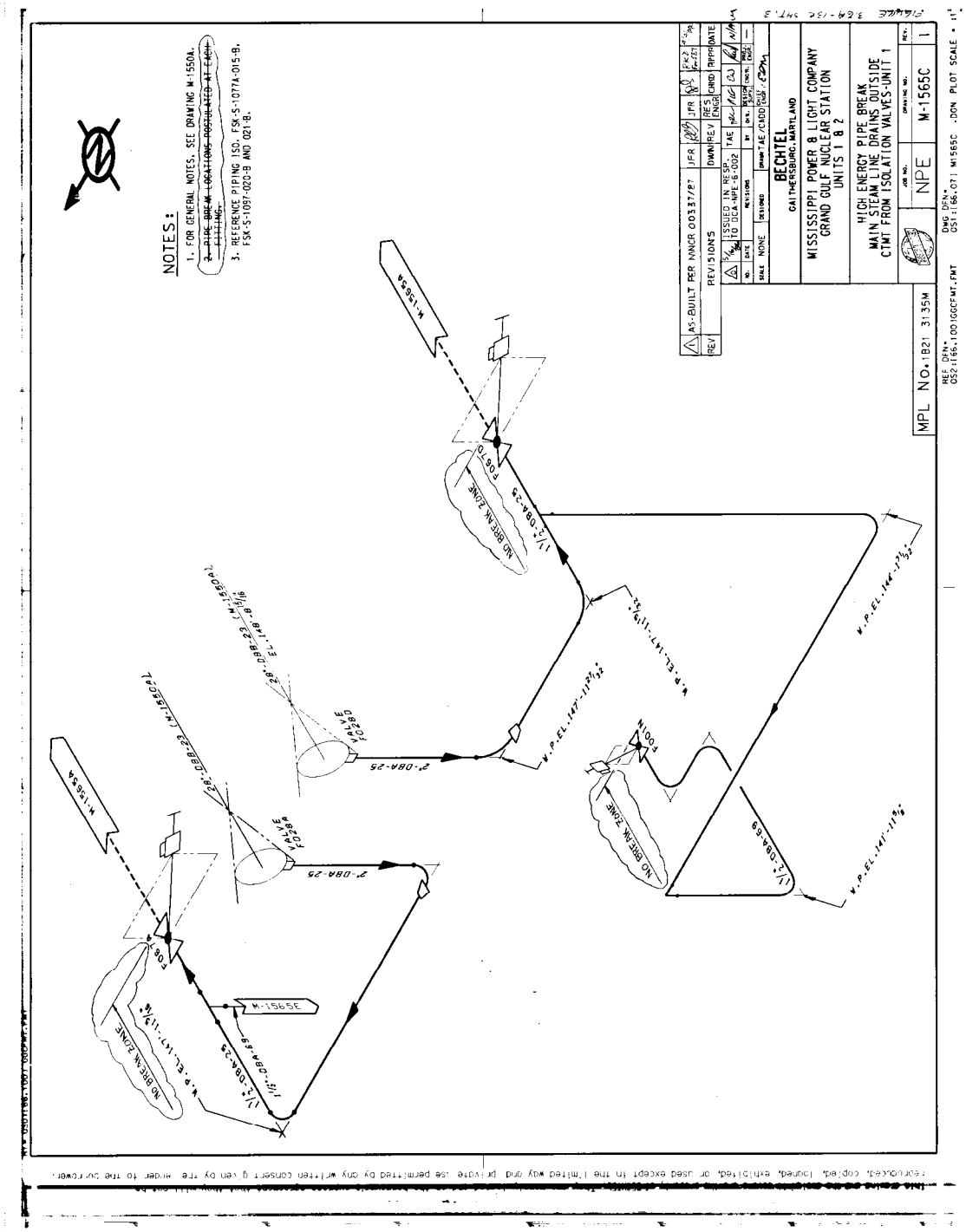
M-1563

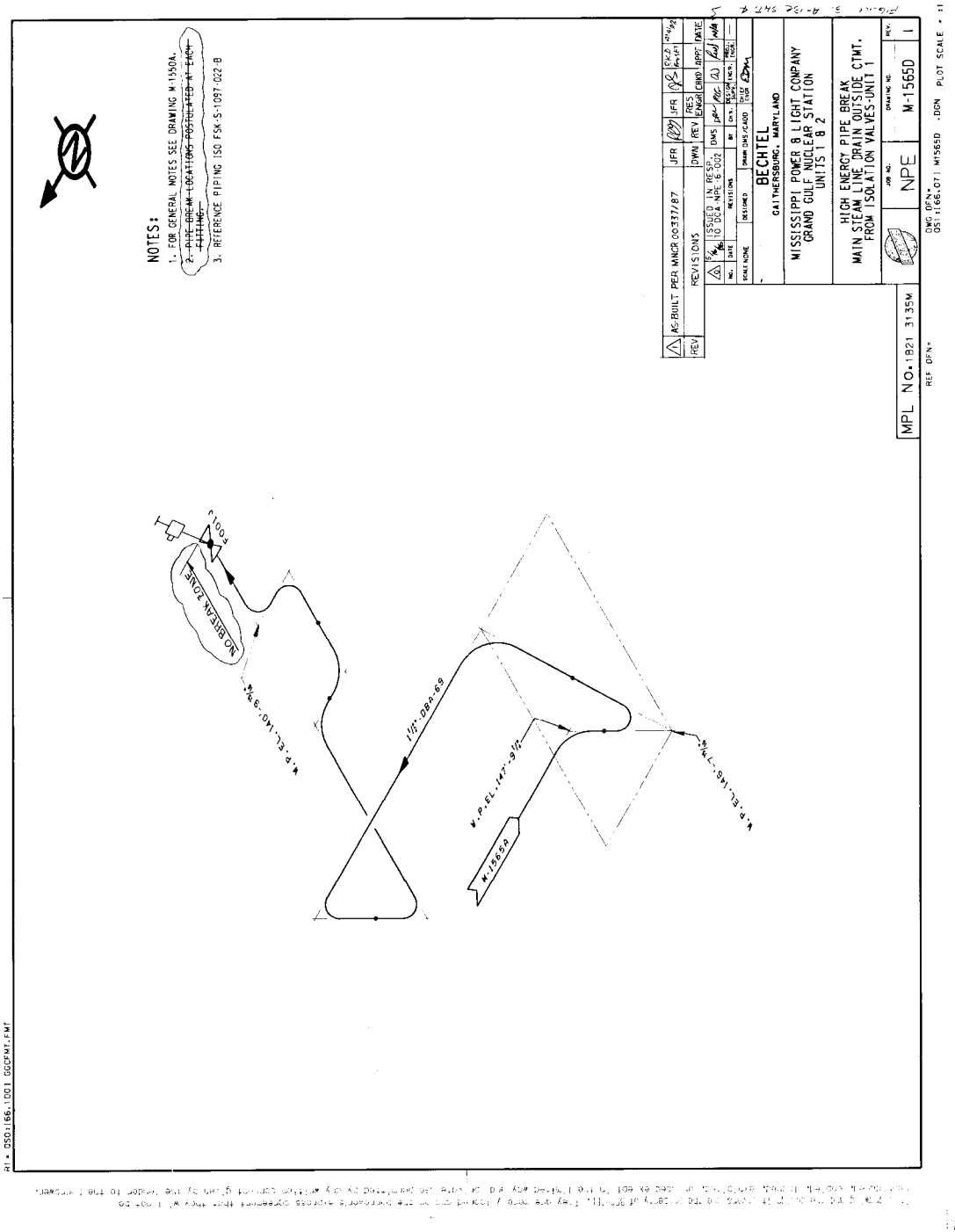

NOTES:
1. FOR GENERAL NOTES, SEE DRAWING M-1550A.
2. BREAK LOCATIONS POSTULATED AT EACH FITTING.
3. REFERENCE PIPING ISO. FSK-S-1077A-030-B AND FSK-S-1077A-031-B.
MAIN STEAM 'A'
MAIN STEAM 'B'
MAIN STEAM 'C'
MAIN STEAM 'D'
HIGH ENERGY
MODERATE ENERGY
MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
HIGH ENERGY PIPE BREAK
DRW VENTS & DRAINS
UNIT 1
BECHTEL
GAITHERSBURG, MARYLAND
M-1564
9645
MPL No. 1B21 3125M

M-1507

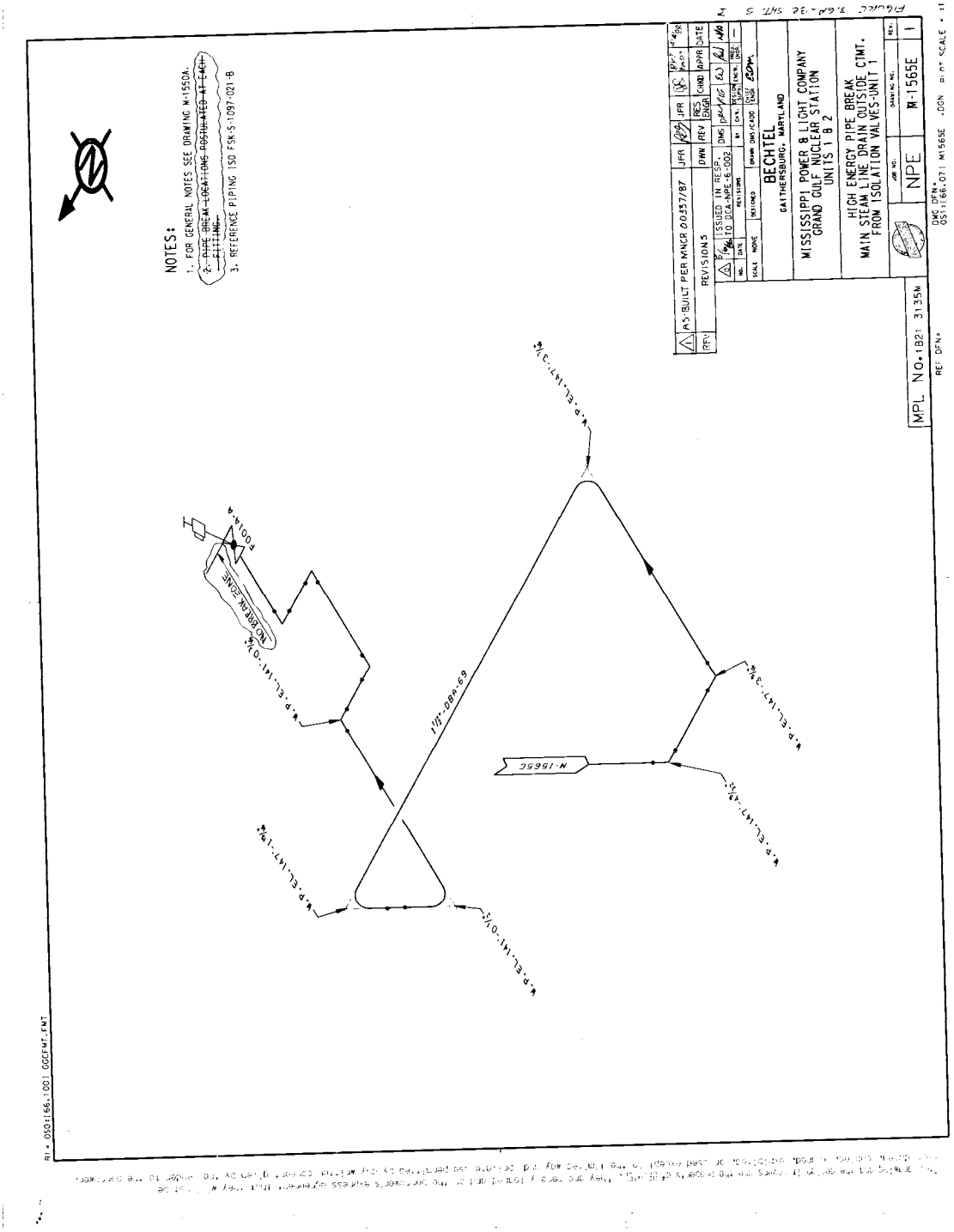

NOTES:
1. FOR GENERAL NOTES, SEE DRAWING M-1550A.
2. BREAK LOCATIONS POSTULATED AT EACH FITTING FOR NON-NUCLEAR PIPING.
3. REFERENCE PIPING ISO FSK-S-1077A-014-B, 015-B, 016-B, 017-B, FSK-S-1097-010-B, AND 0228.
4. WHIP RESTRAINT FUNCTION NOT REQUIRED.
5. REFERENCE STRESS REPORT FSR-11
AS-BUILT PER MNGR 00337/87
REVISIONS
ISSUED IN RESP TO DCA-NPE-86-002
NONE
BECHTEL
GAITHERSBURG, MARYLAND
MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
HIGH ENERGY PIPE BREAK
MAIN STEAM LINE DRAIN OUTSIDE CTMT.
FROM ISOLATION VALVES-UNIT 1
NPE
M-1565A
MPL NO. 11821.21.35M
M-1563B
M-1563D
M-1563C
M-1563E
1½"-DBA-35
1½"-DBA-36
1½"-DBA-33
1½"-DBA-69
1½"-DBA-25
2"-DBA-25
2"-DBB-27
VALVE F028B
VALVE F028C
NO BREAK ZONE
PBR-MS 119
PBR-MS 116
PBR-MS 117
PBR-MS 118
SEE NOTE 4
FOOTE-A
FOOTE-B
FOOTE-C
FOOTE-D
9"
6"
8"
3"
4"-EL-148'-0"
4"-EL-147'-10½"
4"-EL-141'-11½"
2"-EL-148'-8½"


NOTES:
1. FOR GENERAL NOTES, SEE DRAWING M-1550-A.
2. BREAK LOCATIONS POSTULATED AT EACH FITTING.
3. REFERENCE APING ISO. FSK-S-1077A-01B&C/OPR.
AS-BUILT PER NNCR 00337-87
REVISIONS
ISSUED IN RESP. TO DCA-NPE-8-002
BECHTEL
GAITHERSBURG, MARYLAND
MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
HIGH ENERGY PIPE BREAK
MAIN STEAM LINE DRAIN OUTSIDE CTMT.
FROM ISOLATION VALVES-UNIT 1
NPE
M-1565B
MODERATE ENERGY
HIGH ENERGY
M-1565A

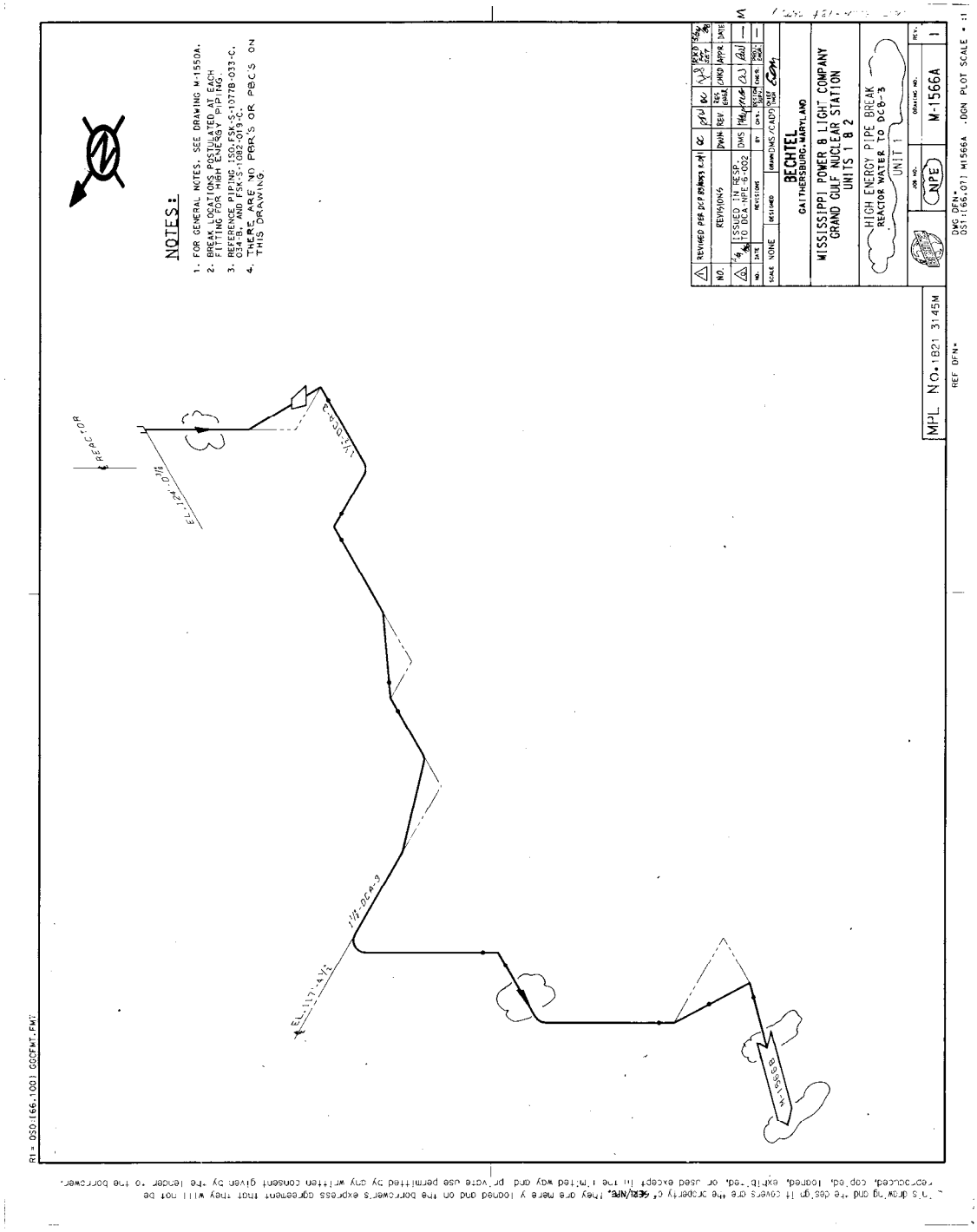

NOTES:
1. FOR GENERAL NOTES, SEE DRAWING M-1550A.
2. PIPE BREAK LOCATIONS POSTULATED AT EACH FITTING.
3. REFERENCE PIPING ISO. FSK-S-1077A-015-B, FSK-S-1097-020-B AND 021-B.
M-1565A
M-1565E
M-1565C
NO BREAK ZONE
W.P.EL.147'-11½"
2"-DBA-25
2"-DBA-26
1½"-DBA-69
MPL NO. 1B21 3135W
BECHTEL
GAITHERSBURG, MARYLAND
MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
HIGH ENERGY PIPE BREAK
MAIN STEAM LINE DRAINS OUTSIDE
CTMT FROM ISOLATION VALVES-UNIT 1
NPE
M-1565C
AS-BUILT PER NNCR 00337/87
ISSUED IN RESP TO ECN NPE-96-0002


NOTES:
1. FOR GENERAL NOTES SEE DRAWING M-1350A.
2. PIPE BREAK LOCATIONS POSTULATED AT EACH FITTING.
3. REFERENCE PIPING ISO FSK-S-1097-022-B
AS-BUILT PER MNCR 00337/87
REVISIONS
ISSUED IN RESP. TO DCA NPE 6-002
BECHTEL
GAITHERSBURG, MARYLAND
MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
HIGH ENERGY PIPE BREAK
MAIN STEAM LINE DRAIN OUTSIDE CTMT.
FROM ISOLATION VALVES-UNIT 1
JOB NO. NPE
DRAWING NO. M-1565D
REV. 1
MPL No. 1821 3135W
NO BREAK ZONE
1½"-DBA-SS
M-1565A

M-1565D


NOTES:
1. FOR GENERAL NOTES SEE DRAWING M-1550A.
2. PIPE BREAK LOCATIONS ASSUMED AT EACH FITTING.
3. REFERENCE PIPING ISO FSK-S-1097-021-B
AS-BUILT PER MNCR 00357/87
REVISIONS
ISSUED IN RESP. TO DCA-NPE-8-0092
BECHTEL
GAITHERSBURG, MARYLAND
MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
HIGH ENERGY PIPE BREAK
MAIN STEAM LINE DRAIN OUTSIDE CIVIT.
FROM ISOLATION VALVES-UNIT 1
NPE
M-1565E
MPL NO. 1921 3135M
N-1565E
1½"-DBA-69
NO BREAK ZONE
F001A-A

M-1565E



M-1566A

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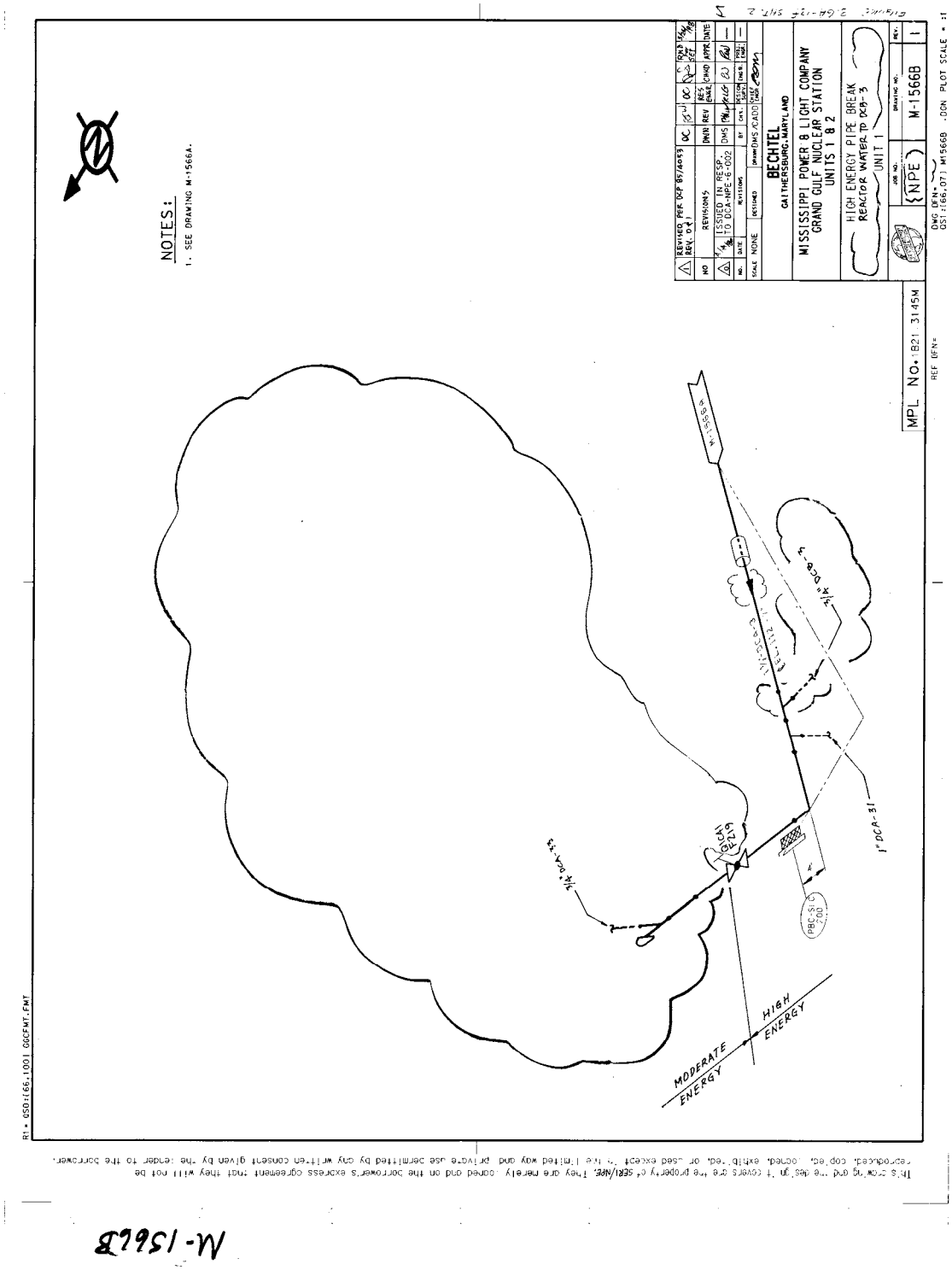
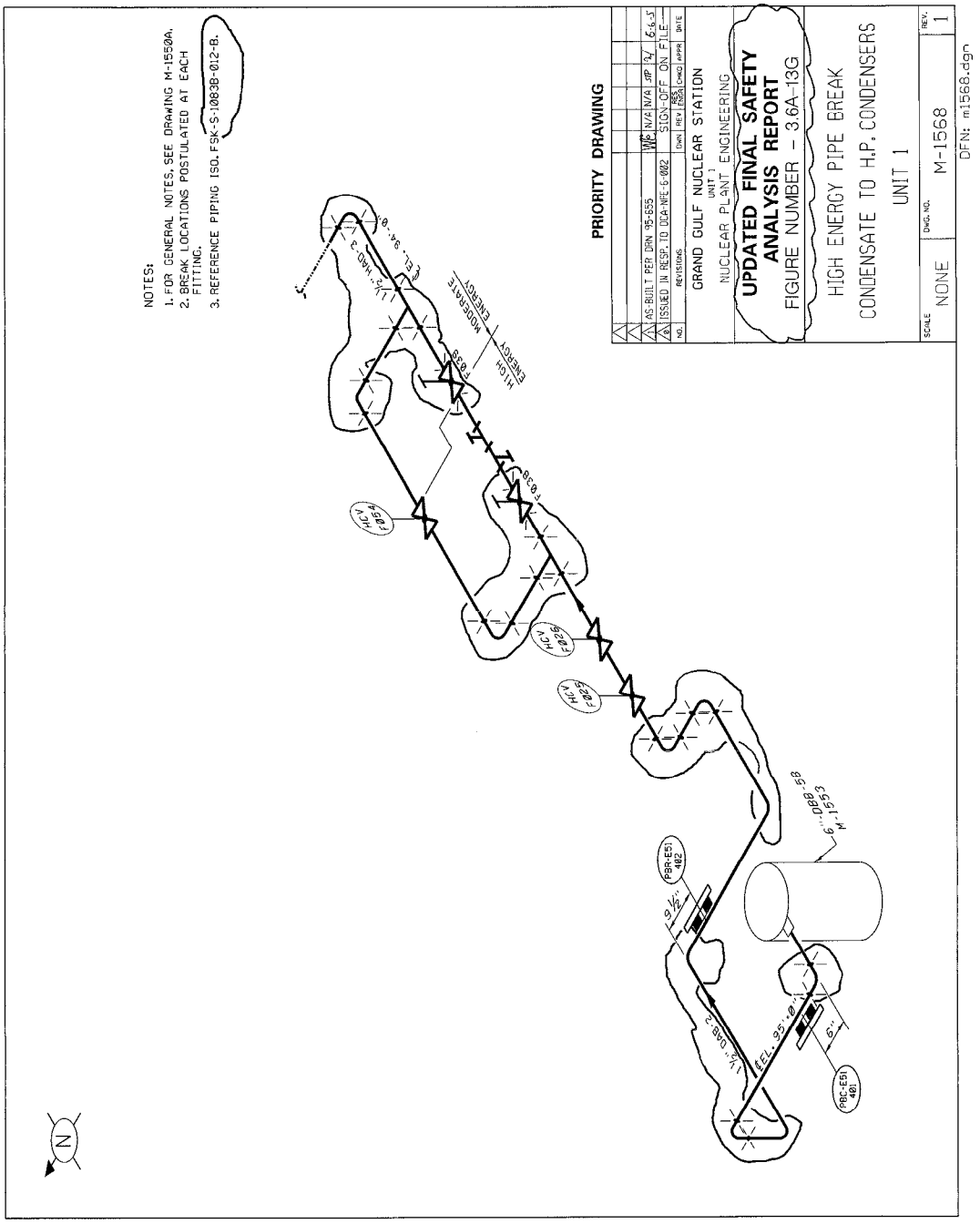
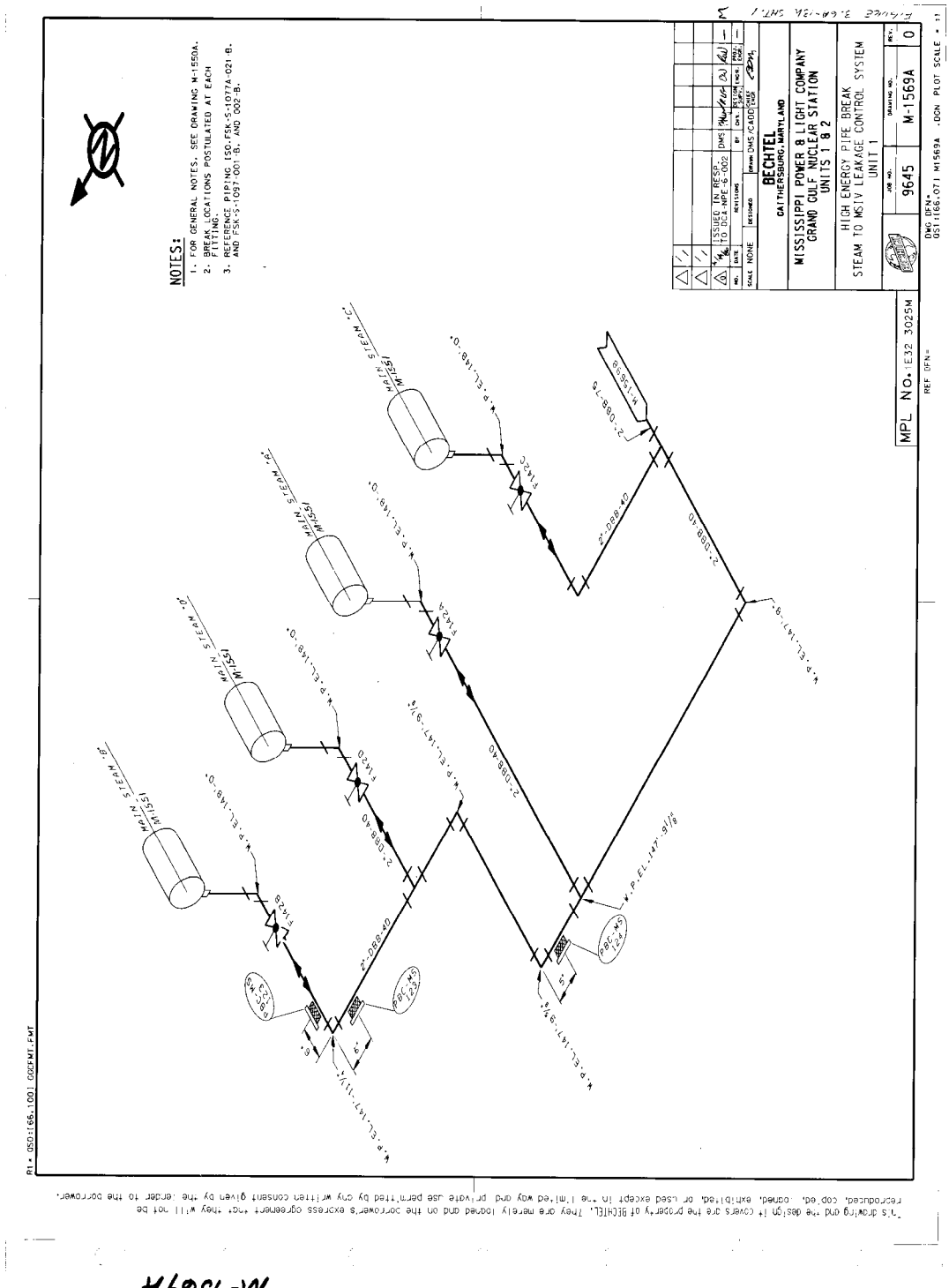

NOTES:
1. SEE DRAWING M-1566A.
MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
HIGH ENERGY PIPE BREAK
REACTOR WATER TP DCA-3
UNIT 1
BECHTEL
GAITHERSBURG, MARYLAND
M-1566B
MPL NO. 1821 3145M
MODERATE ENERGY
HIGH ENERGY


NOTES:
1. FOR GENERAL NOTES, SEE DRAWING M-1550A.
2. BREAK LOCATIONS POSTULATED AT EACH FITTING.
3. REFERENCE PIPING ISO. FSK-S-10838-012-B.
PRIORITY DRAWING
AS-BUILT PER DRN 95-655
ISSUED IN RESP. TO DCA-NEE-G-0002
SIGN-OFF ON FILE
GRAND GULF NUCLEAR STATION
UNIT 1
NUCLEAR PLANT ENGINEERING
UPDATED FINAL SAFETY ANALYSIS REPORT
FIGURE NUMBER - 3.6A-13G
HIGH ENERGY PIPE BREAK
CONDENSATE TO H.P. CONDENSERS
UNIT 1
SCALE NONE
DWG. NO. M-1568
REV. 1
DFN: m1568.dgn
MODERATE ENERGY
HIGH ENERGY
EL. 95'-0"
N



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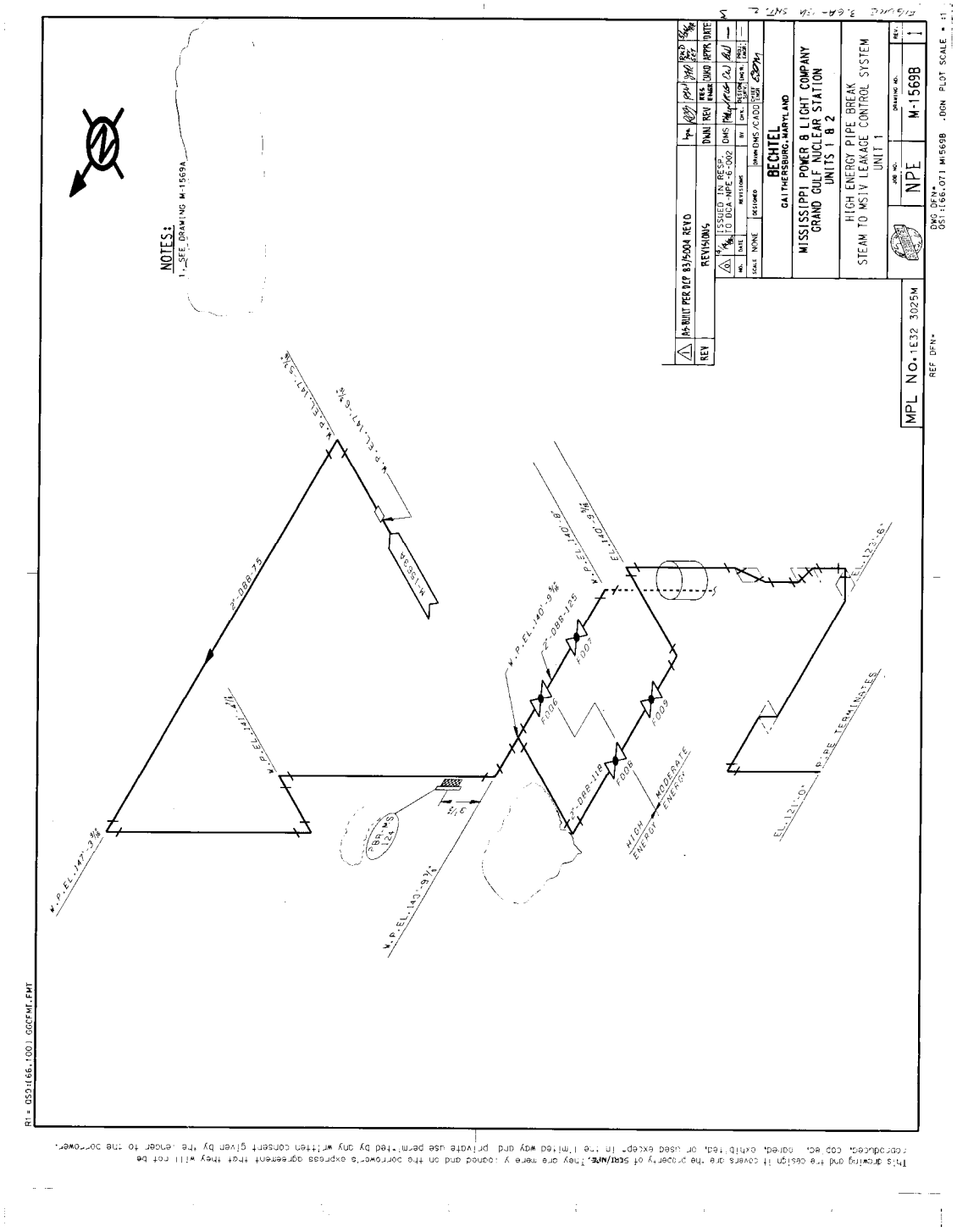
M-1569A



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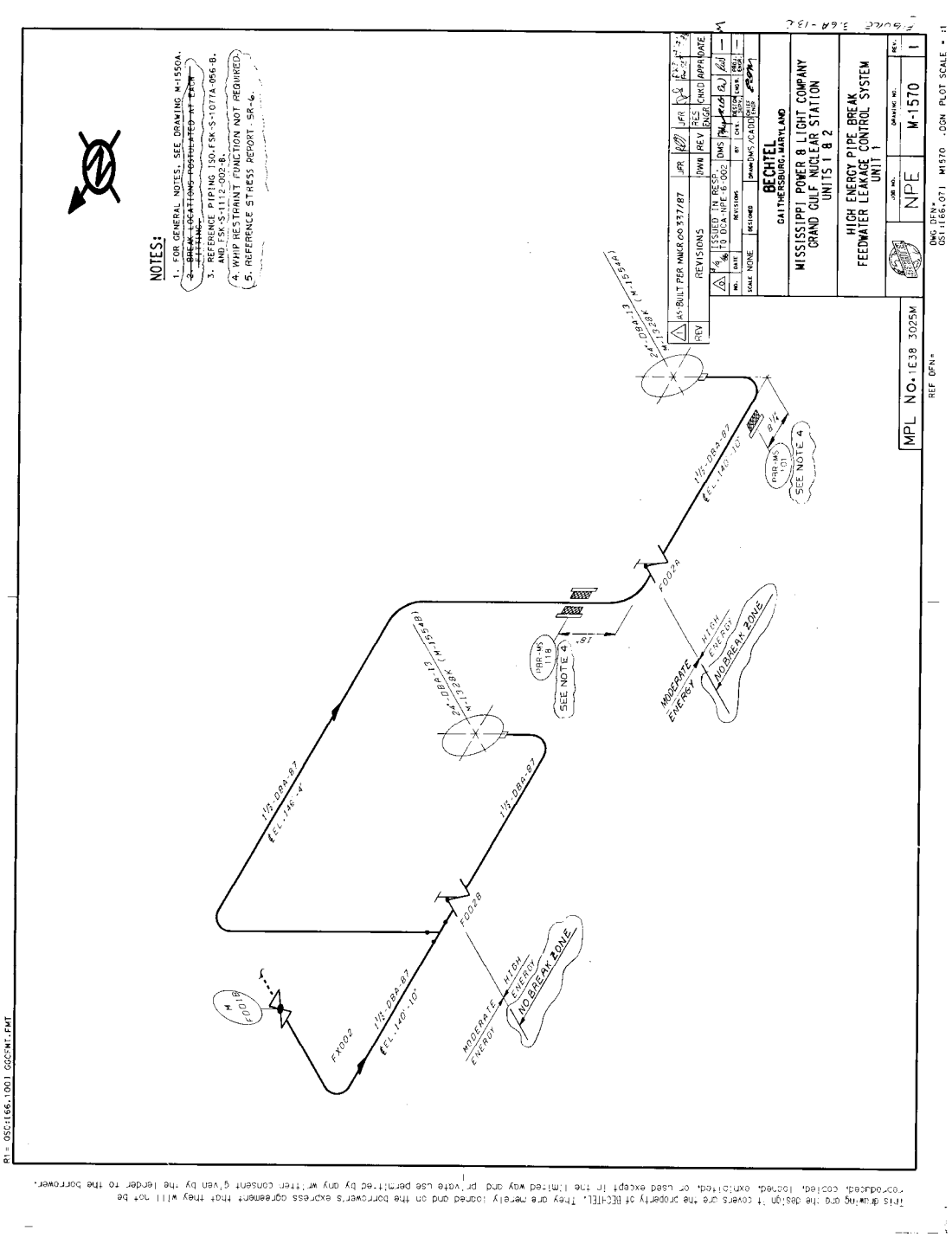
NOTES:
1. FOR GENERAL NOTES, SEE DRAWING M-1550A.
2. BREAK LOCATIONS POSITIONED AT EACH FITTING.
3. REFERENCE PIPING ISO. FSK-S-1077A-056-B, AND FSK-S-1112-002-B.
4. WHIP RESTRAINT FUNCTION NOT REQUIRED.
5. REFERENCE STRESS REPORT SR-6.
AS-BUILT PER NUCR 00 537/87
REVISIONS
ISSUED IN RESP TO DCA NPE-87-0002
BECHTEL
GAITHERSBURG, MARYLAND
MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
HIGH ENERGY PIPE BREAK
FEEDWATER LEAKAGE CONTROL SYSTEM
UNIT 1
NPE
M-1570
MPL NO. 1E38 3025M
REF DFN-
DWG DFN- OSC:1166.071 M1570 .DGN PLOT SCALE = 1
1 1/2" DBA-87
EL. 140'-10"
EL. 146'-4"
EL. 128'-6" (M.152 gal)
EL. 128'-6" (M.152 gal)
FX002
FX002A
FX002B
FX018
HRR-MS 118
HRR-MS 101
SEE NOTE 4
MODERATE ENERGY
HIGH ENERGY
NO BREAK ZONE
R1 = OSC:1166.1001 GGCSNT.FMT
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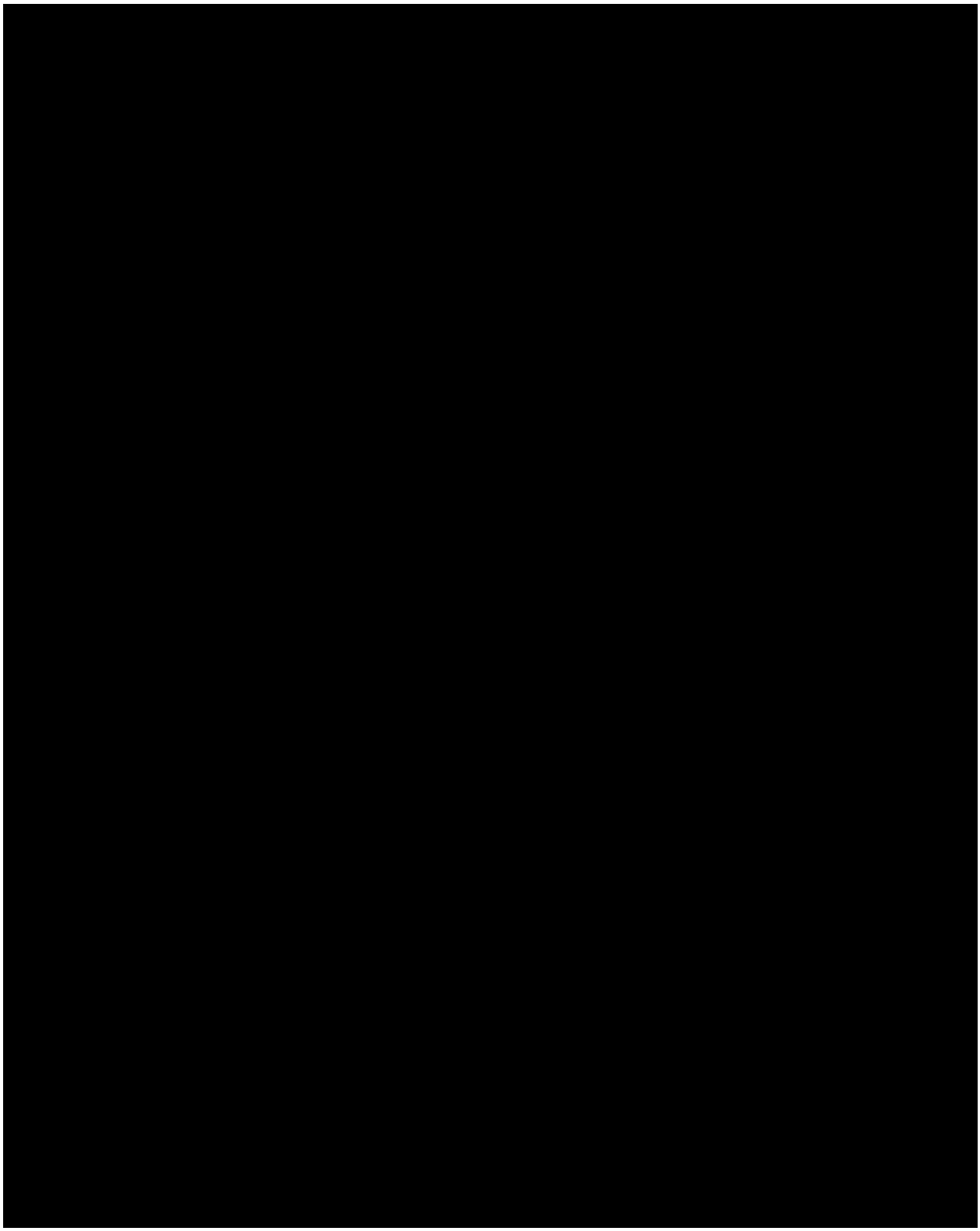
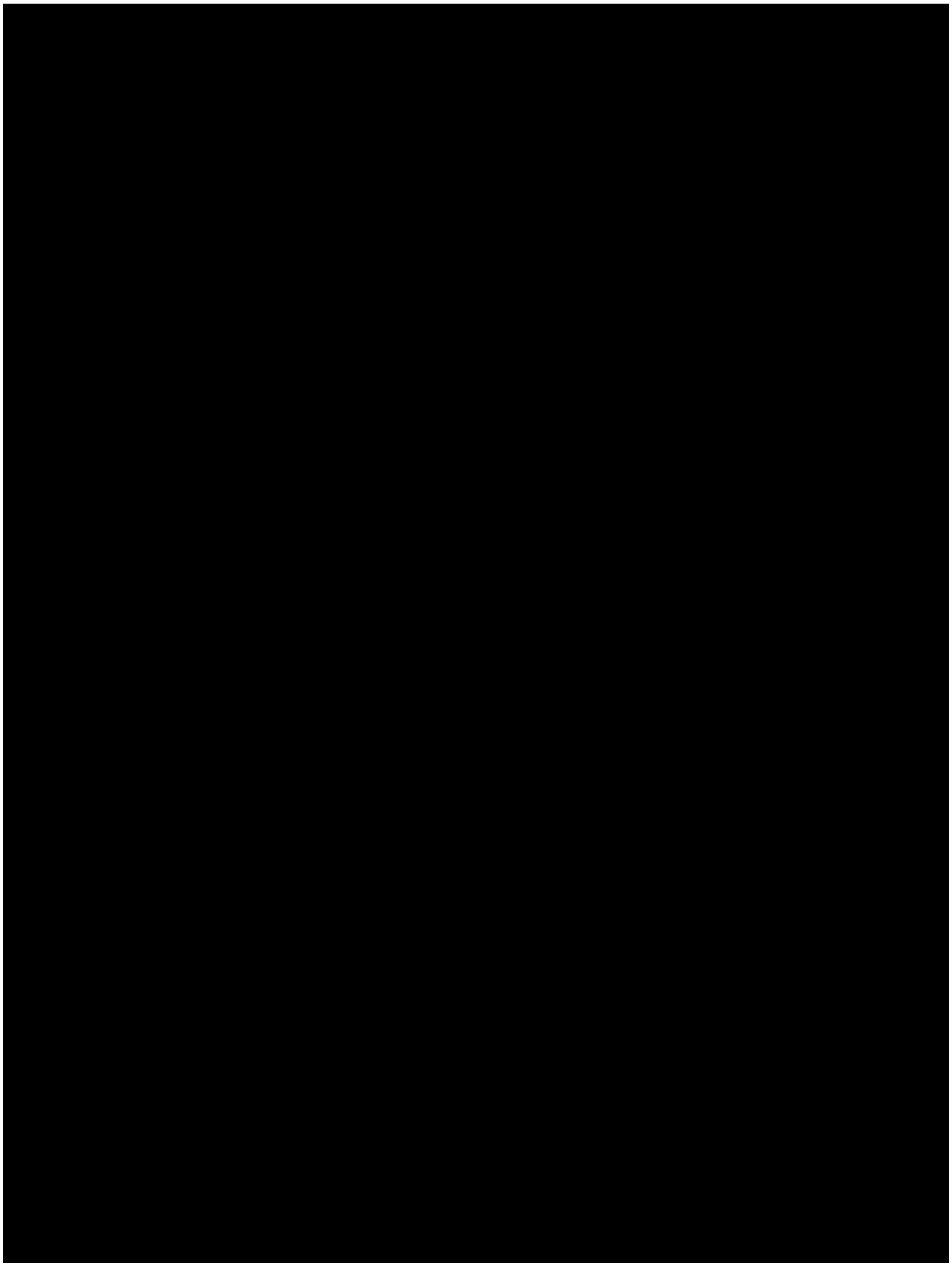
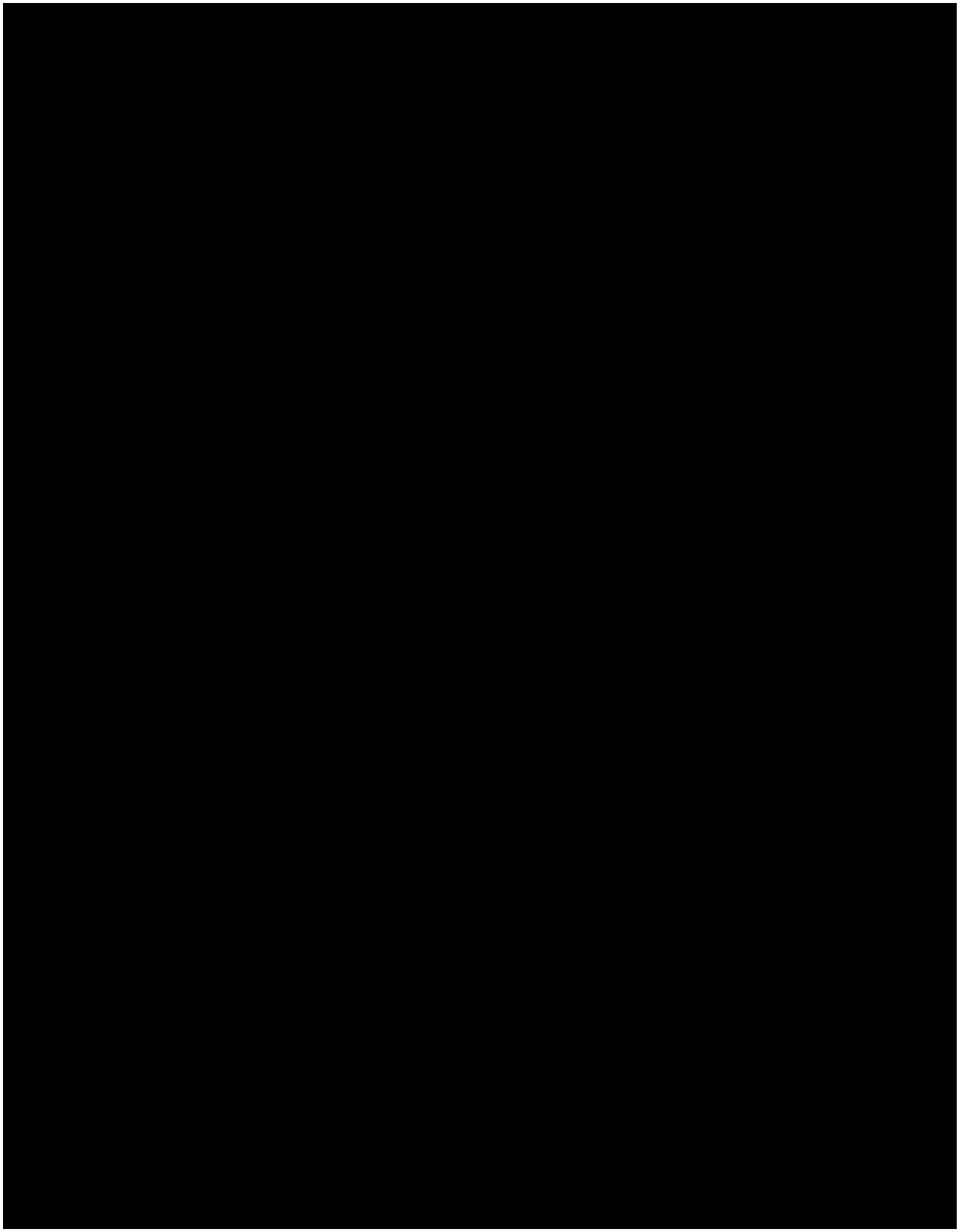
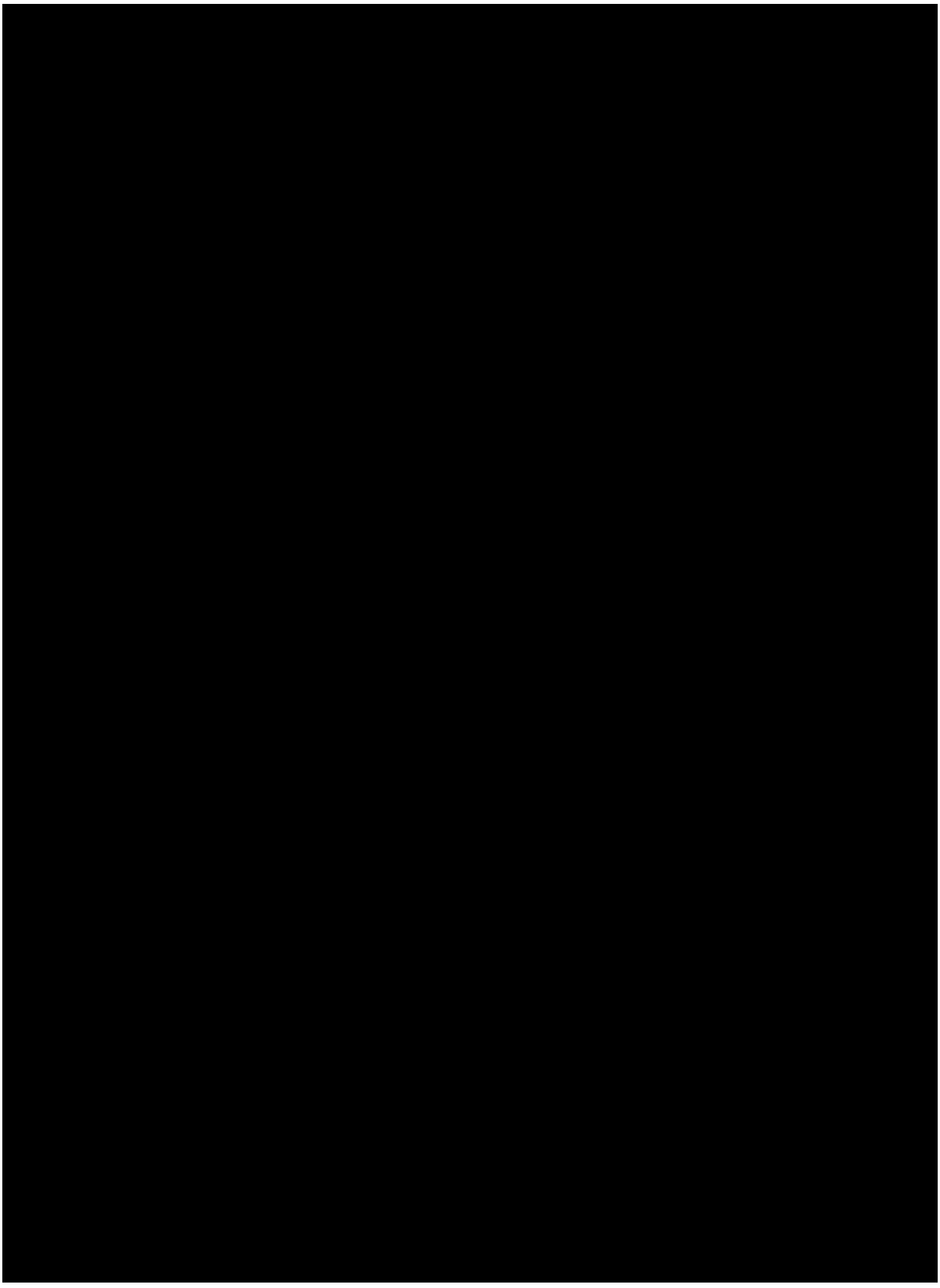
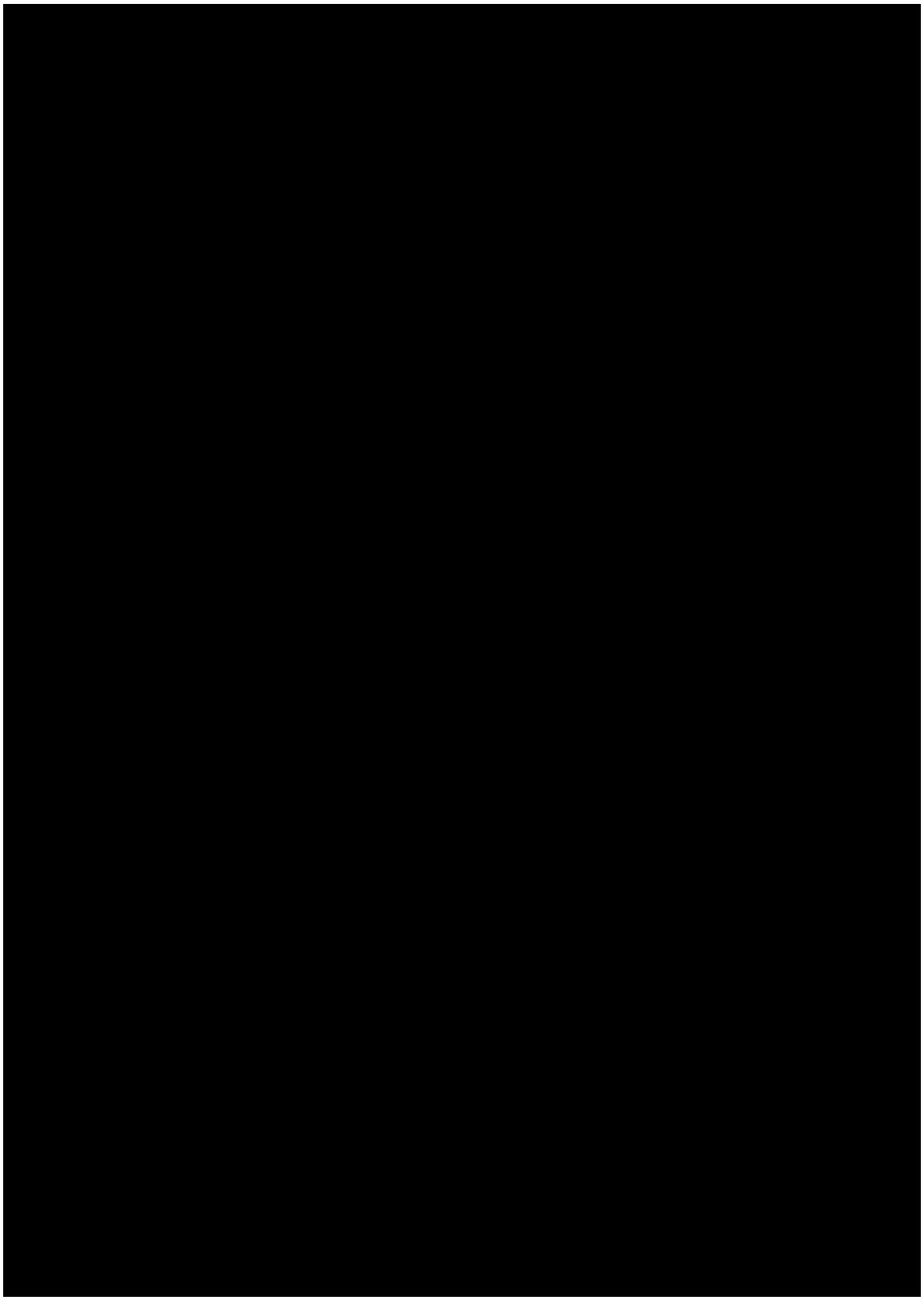
M-15 10

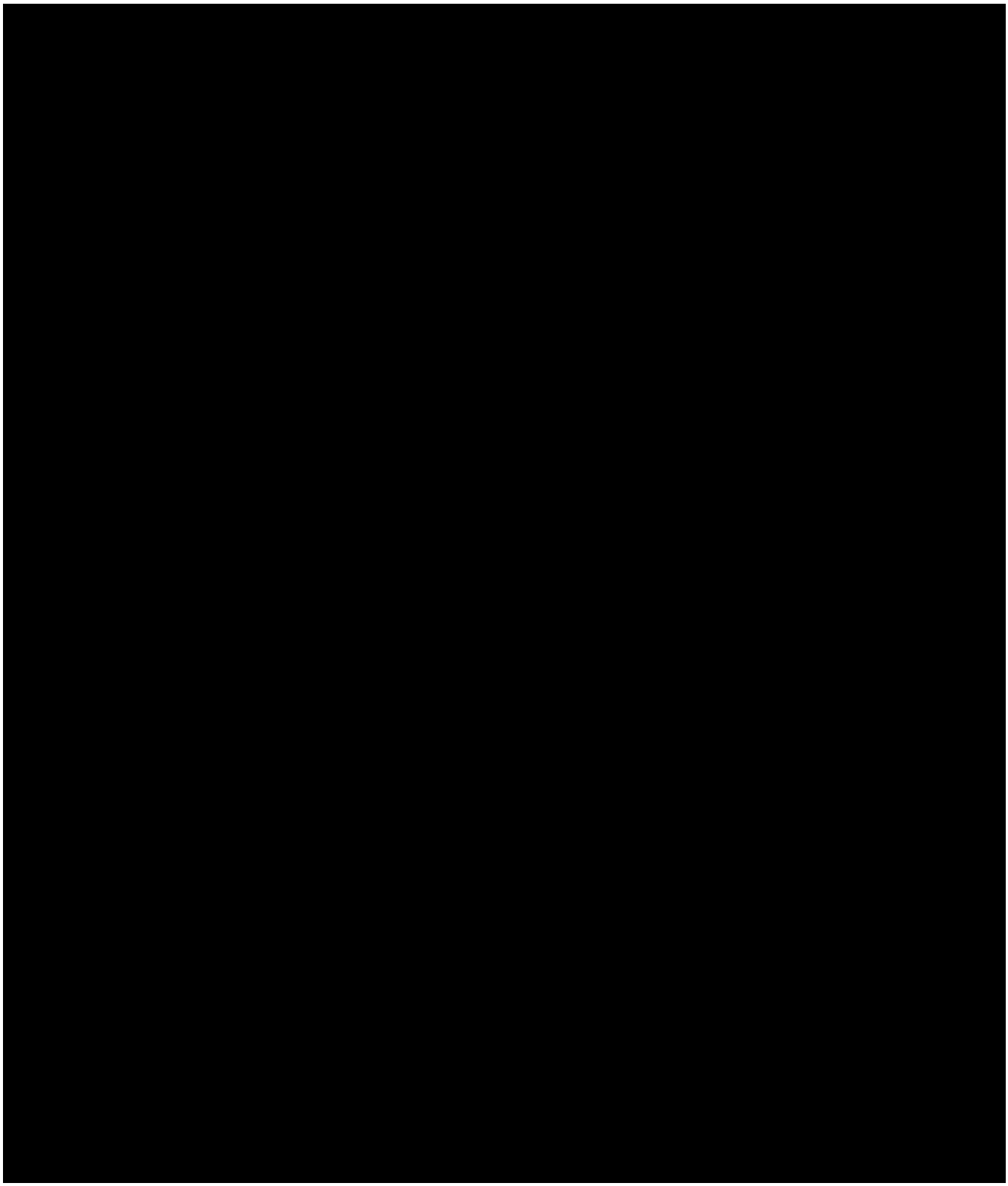


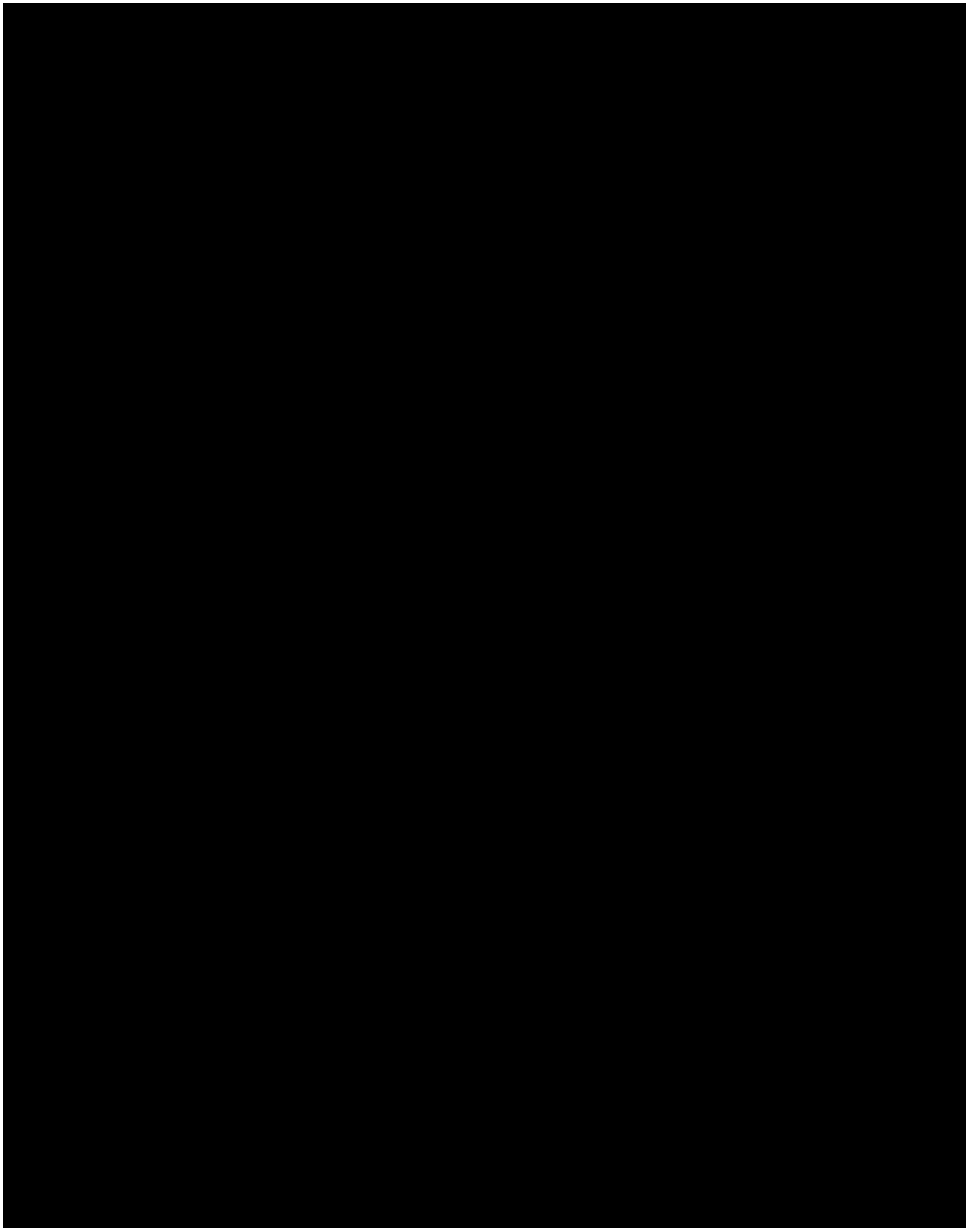


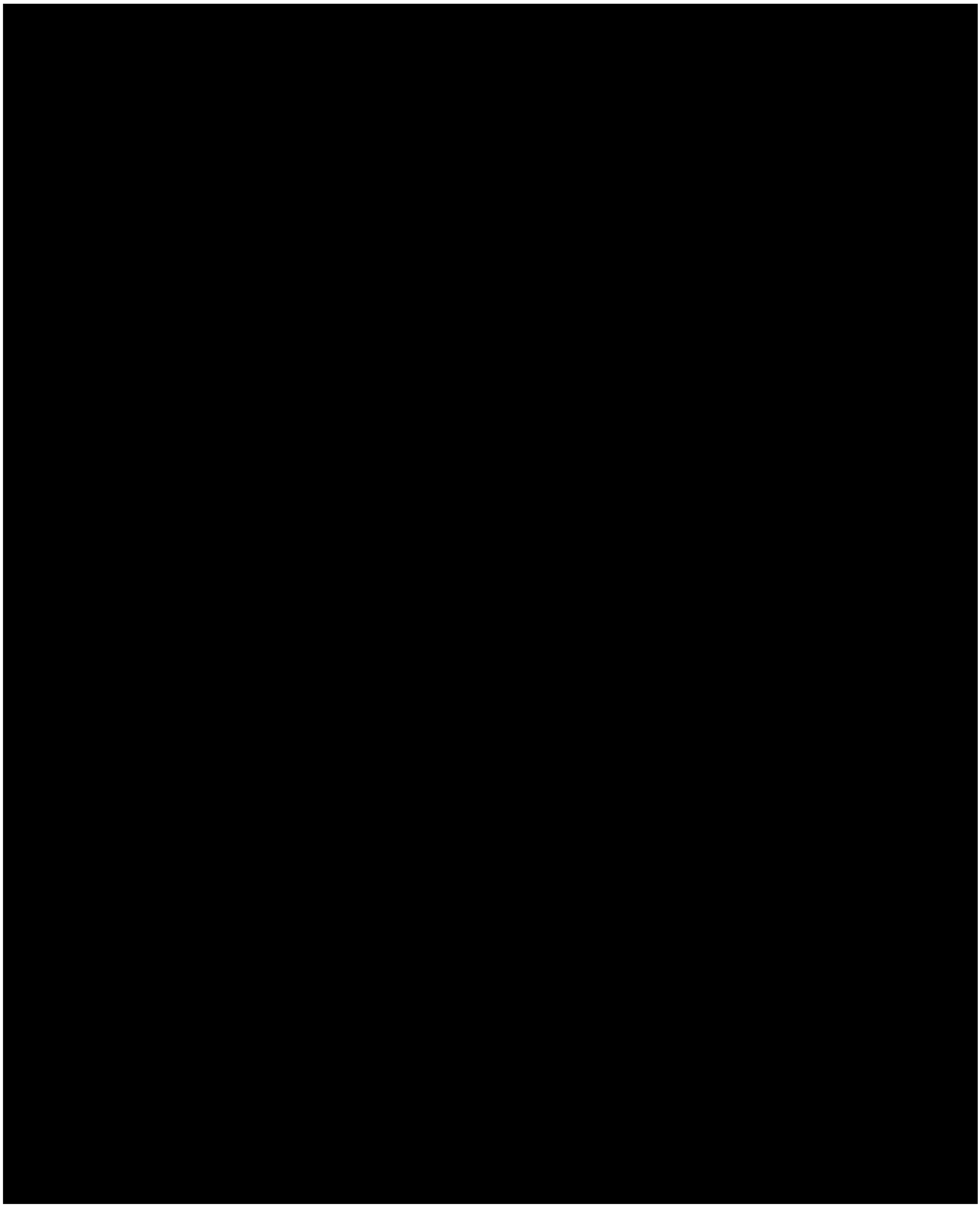


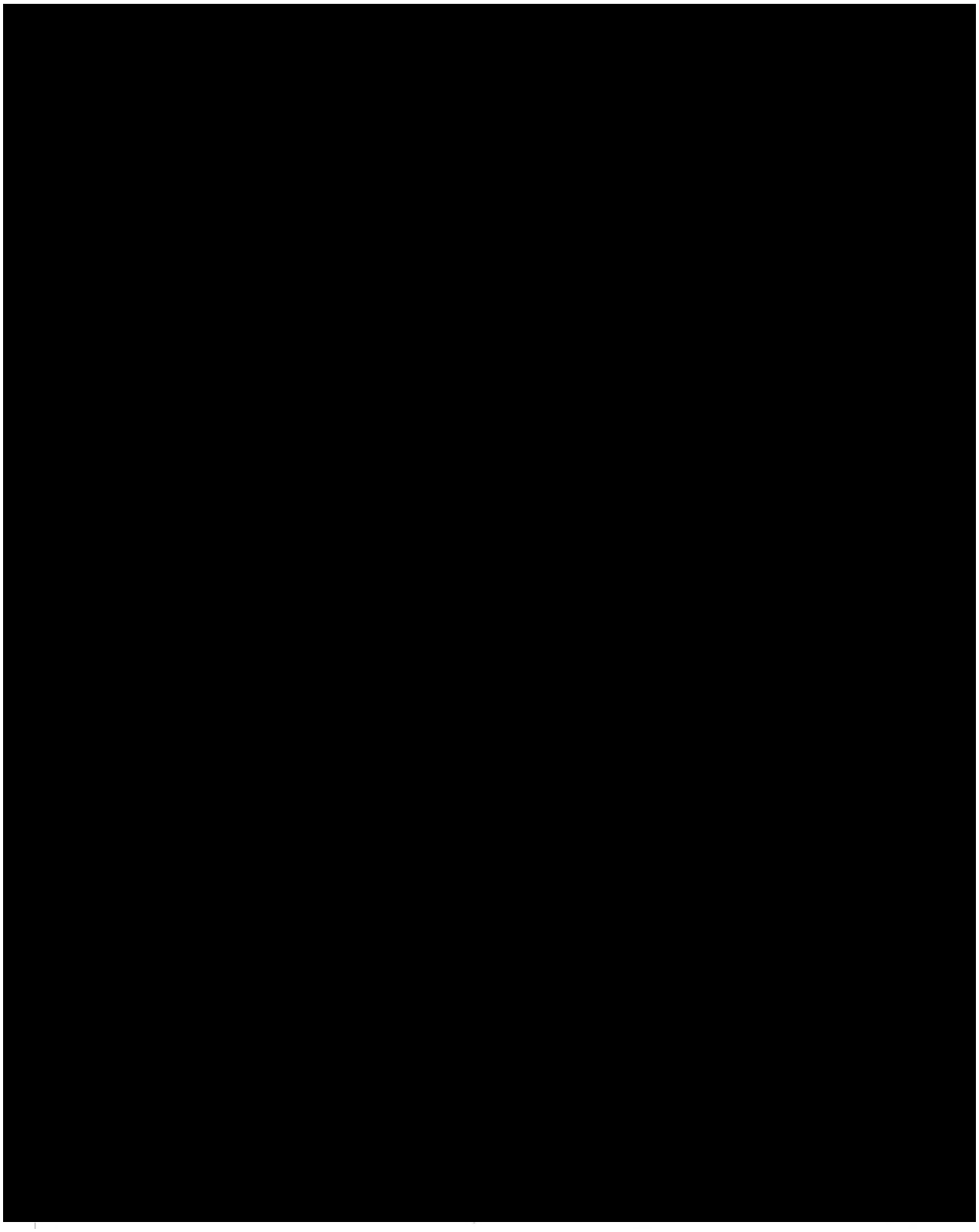


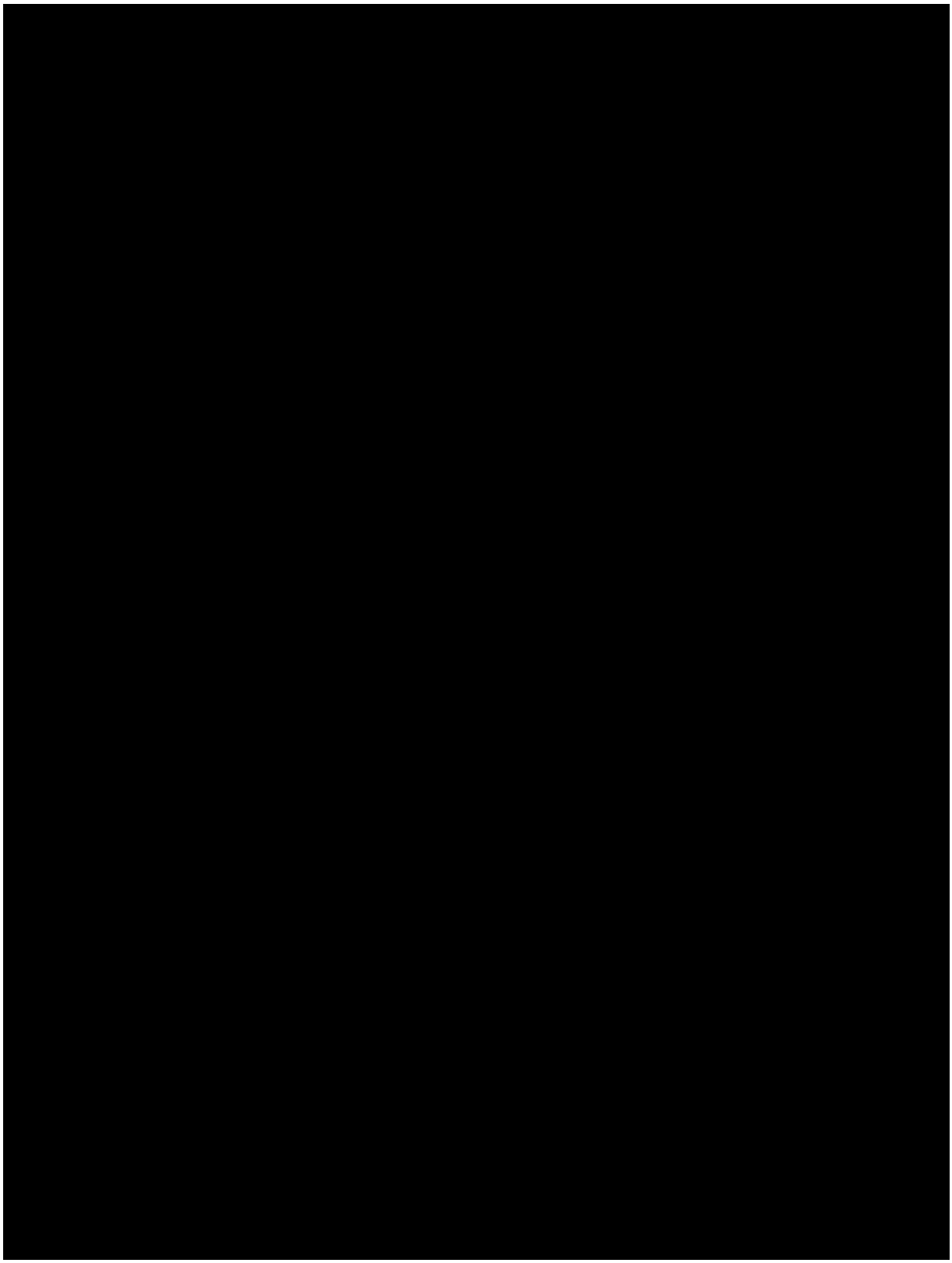


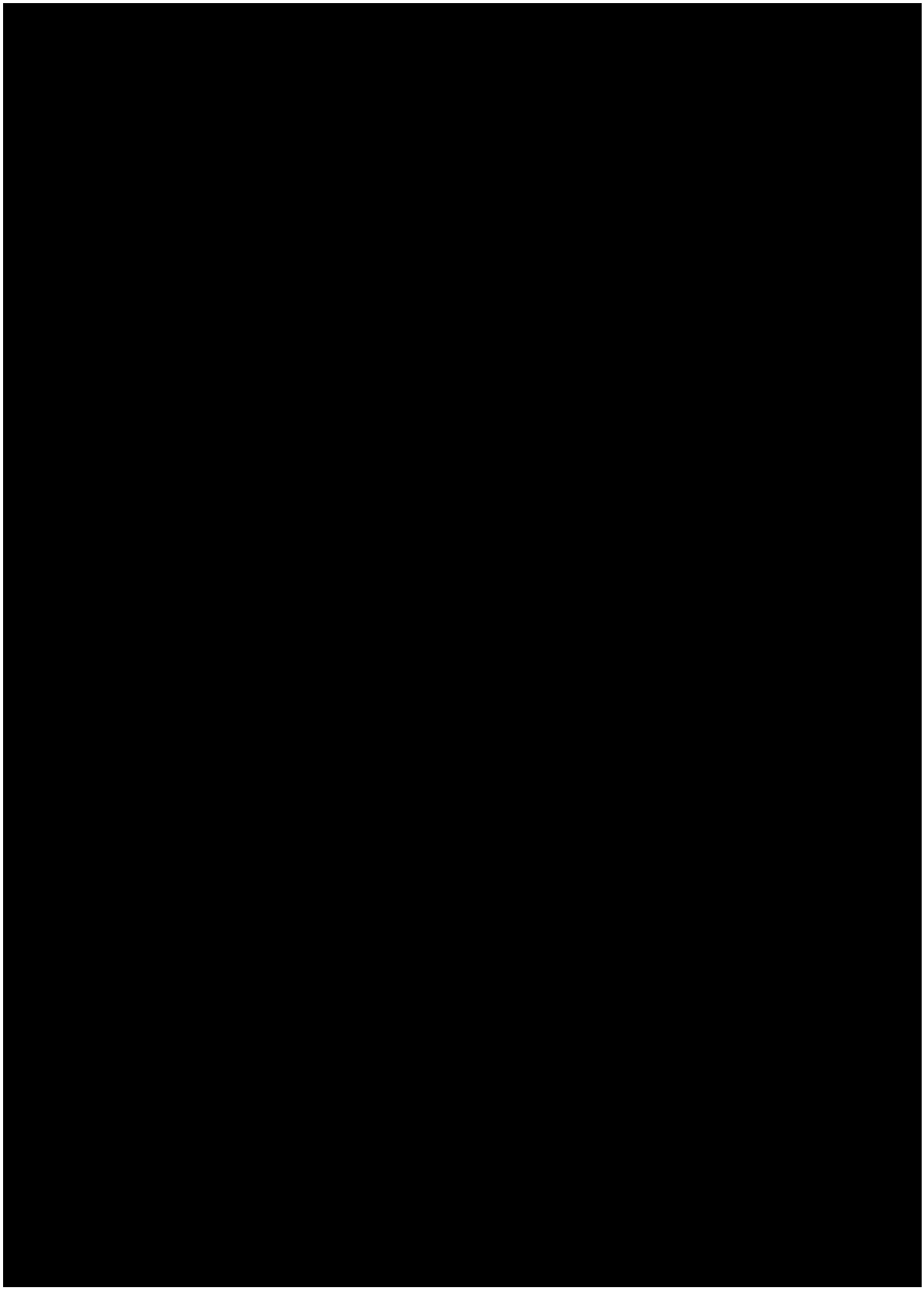


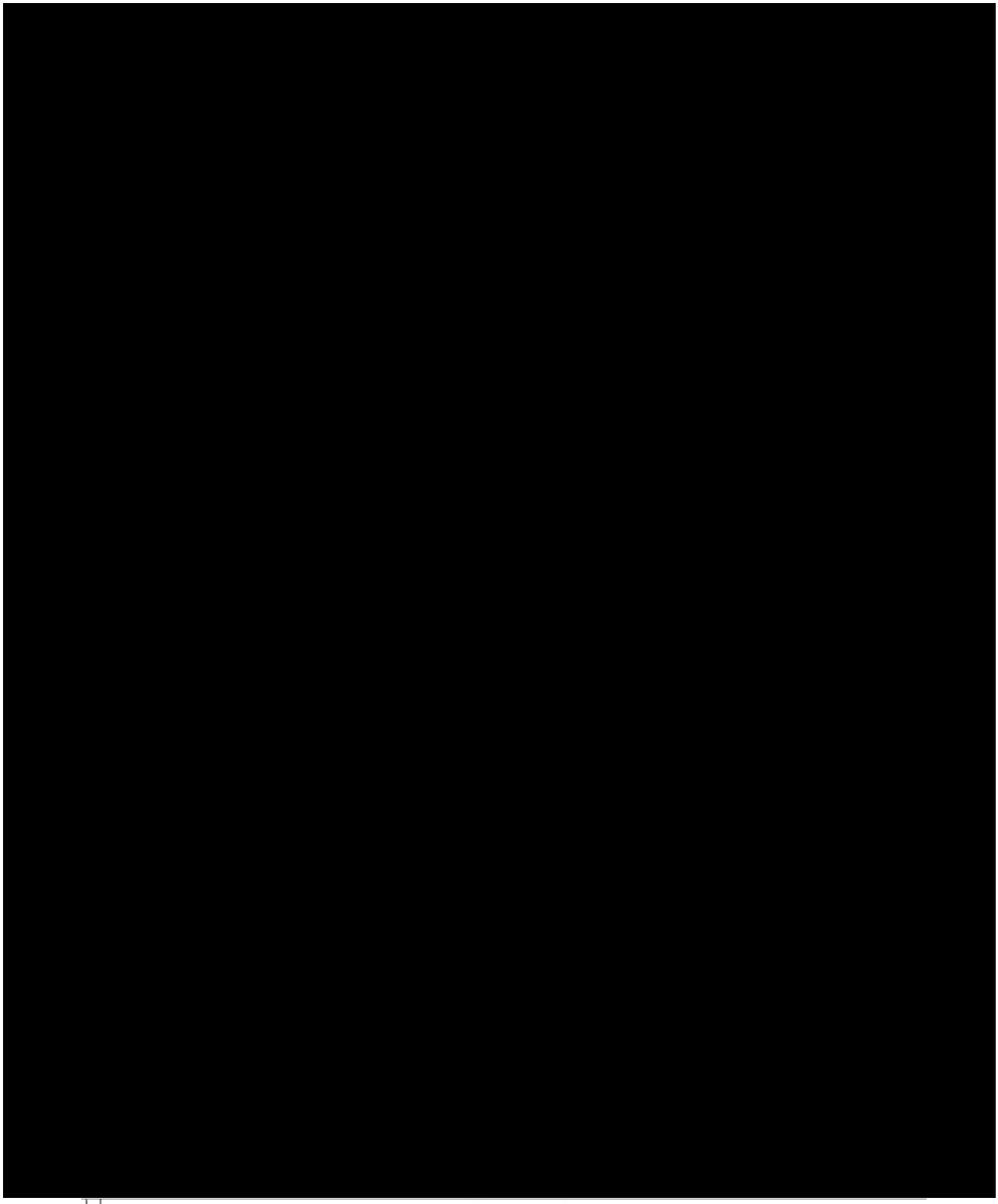


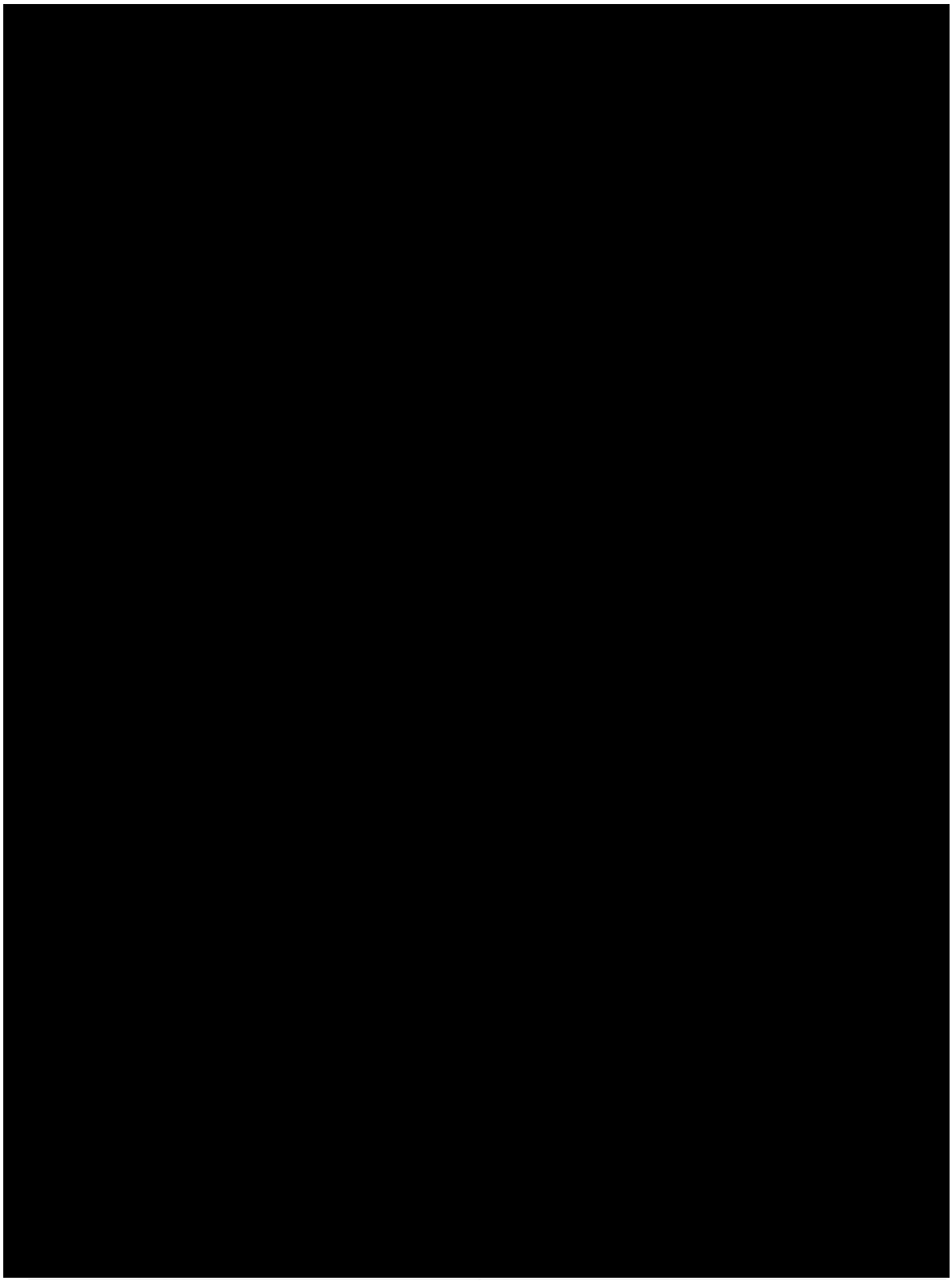


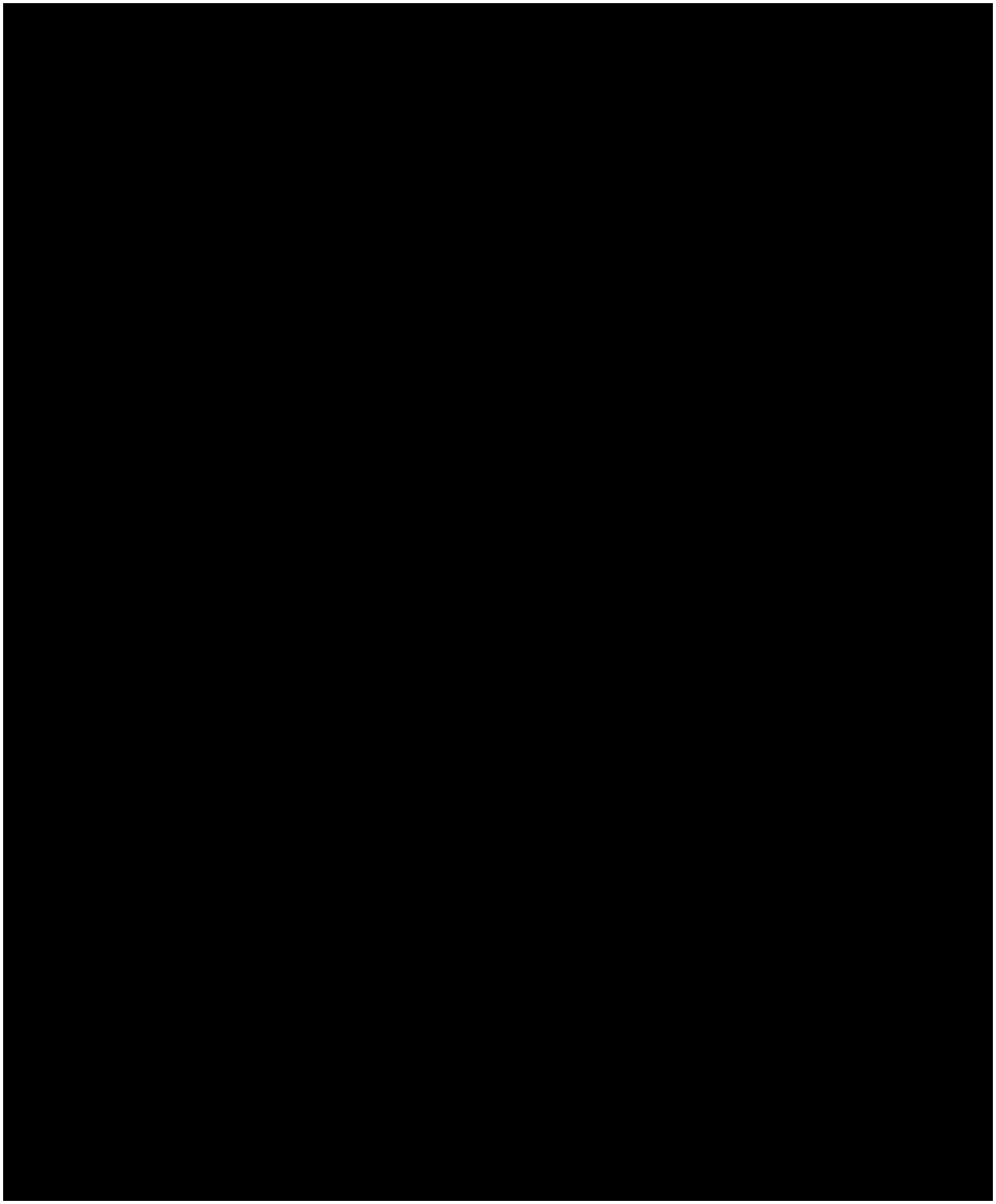


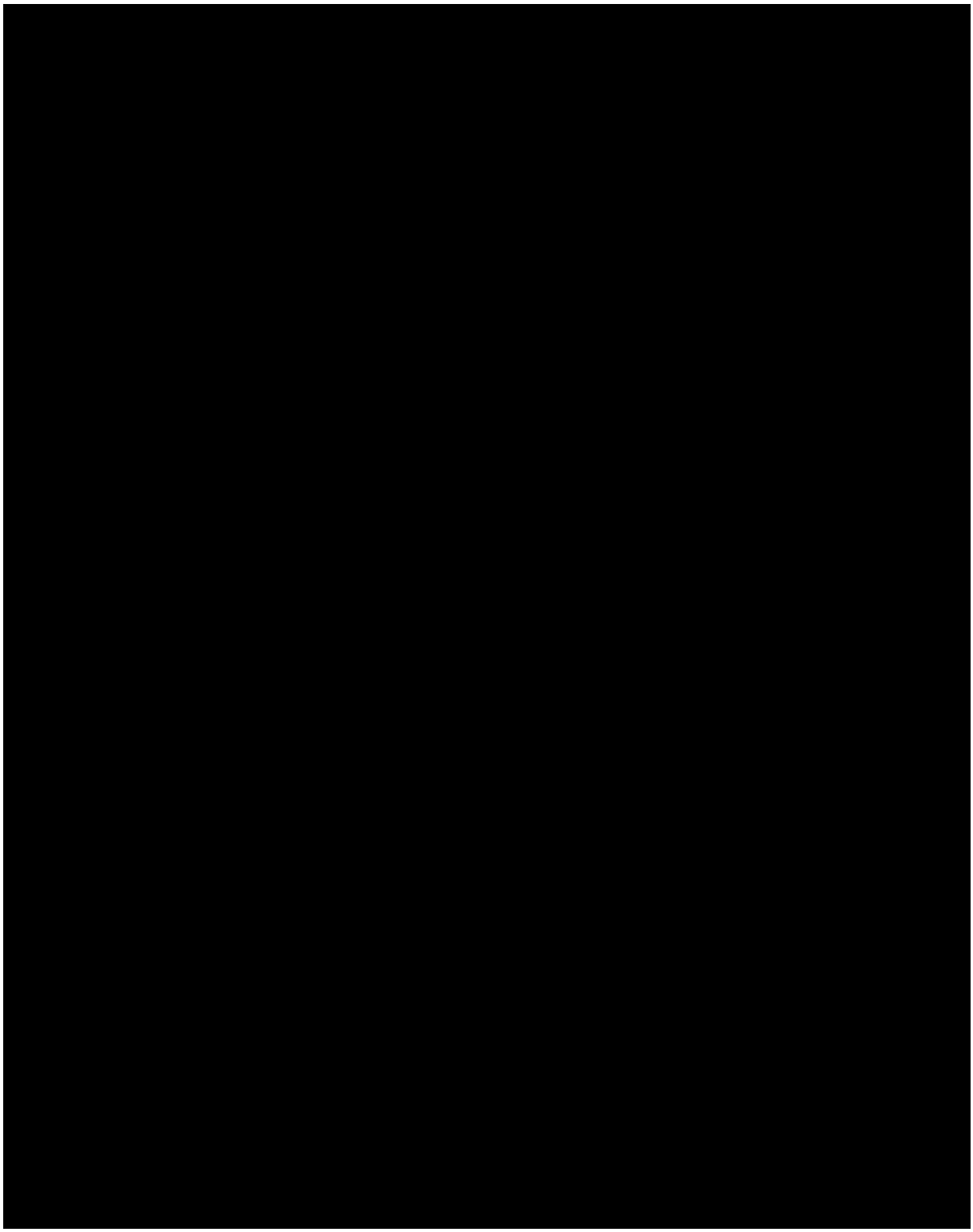


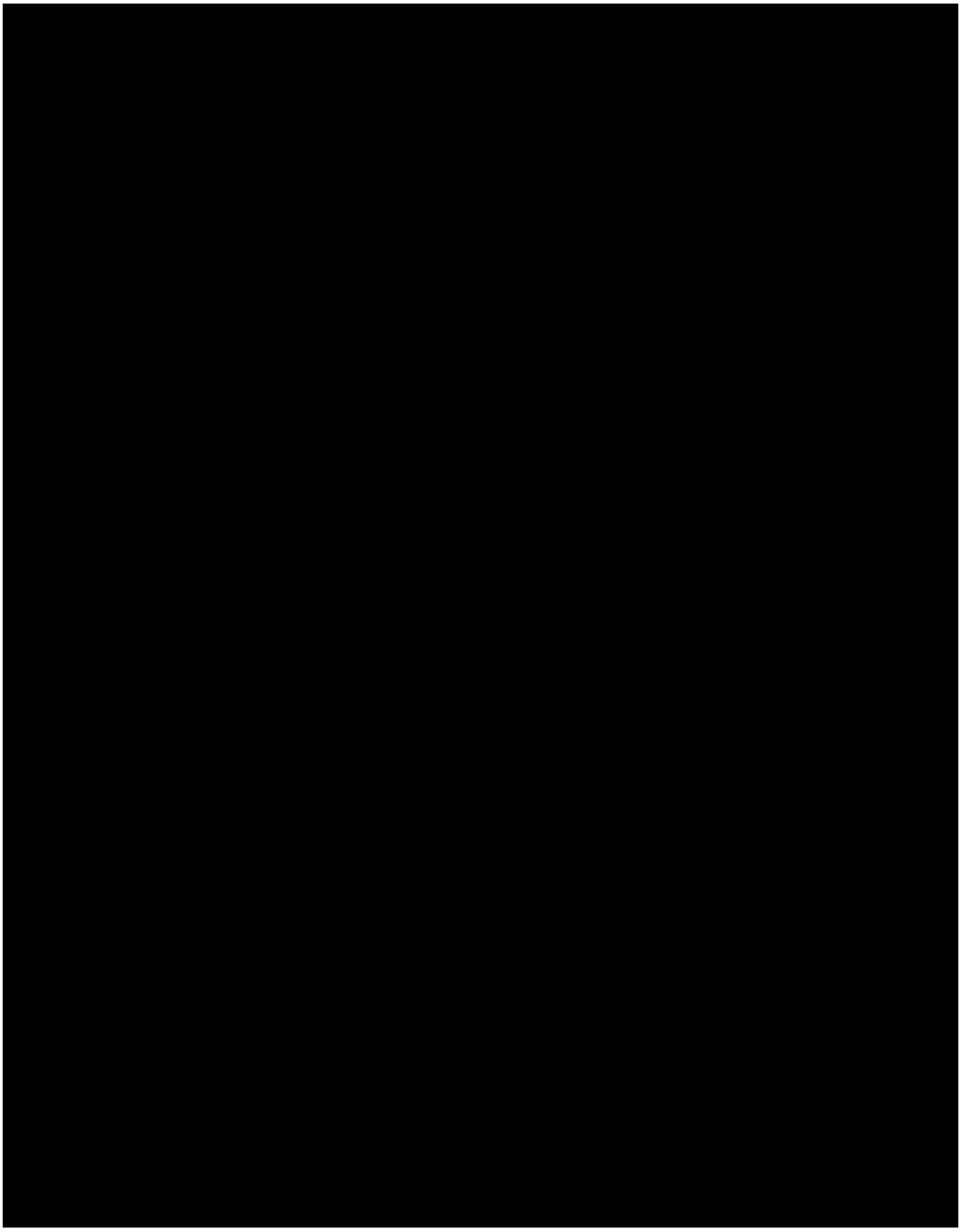


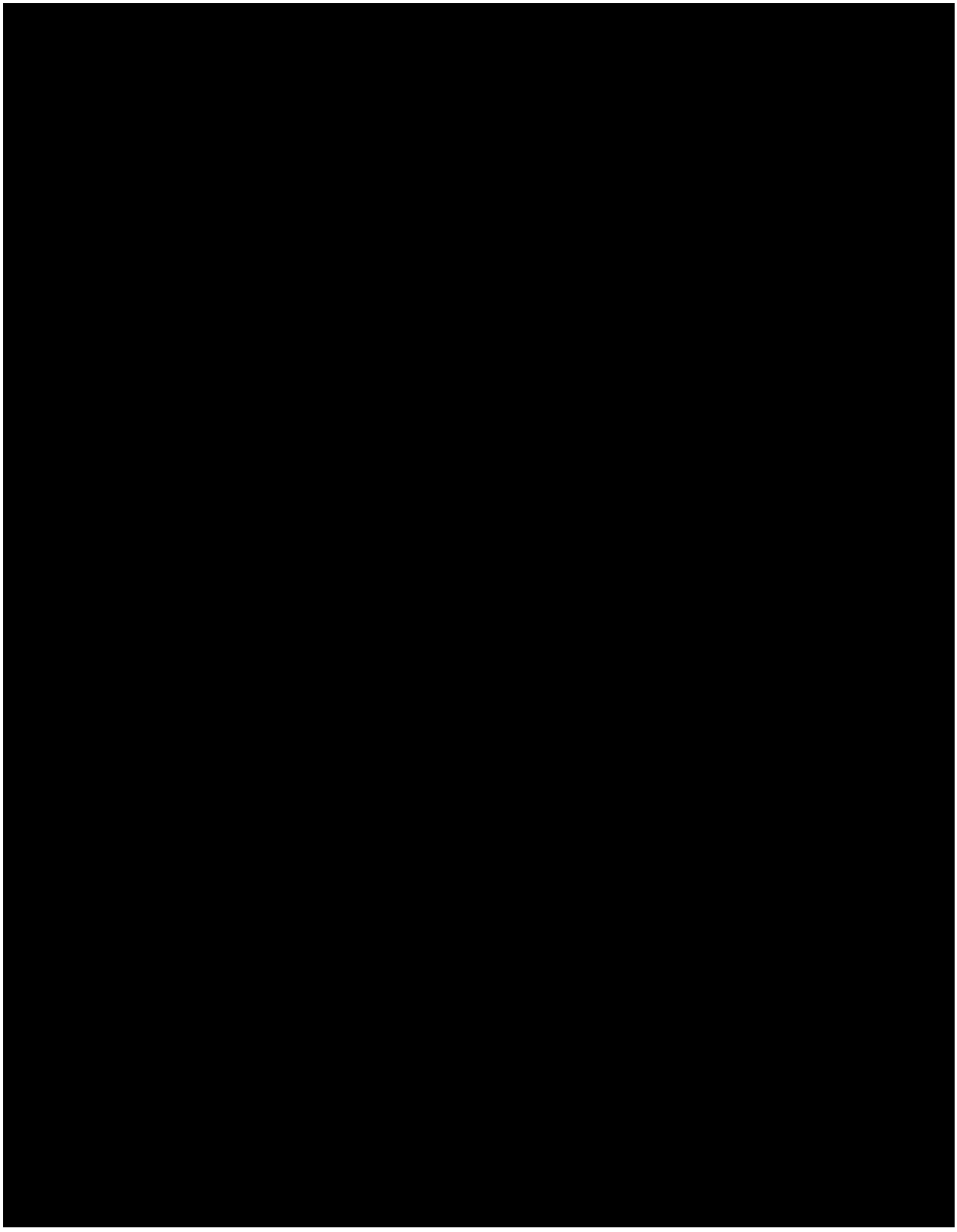


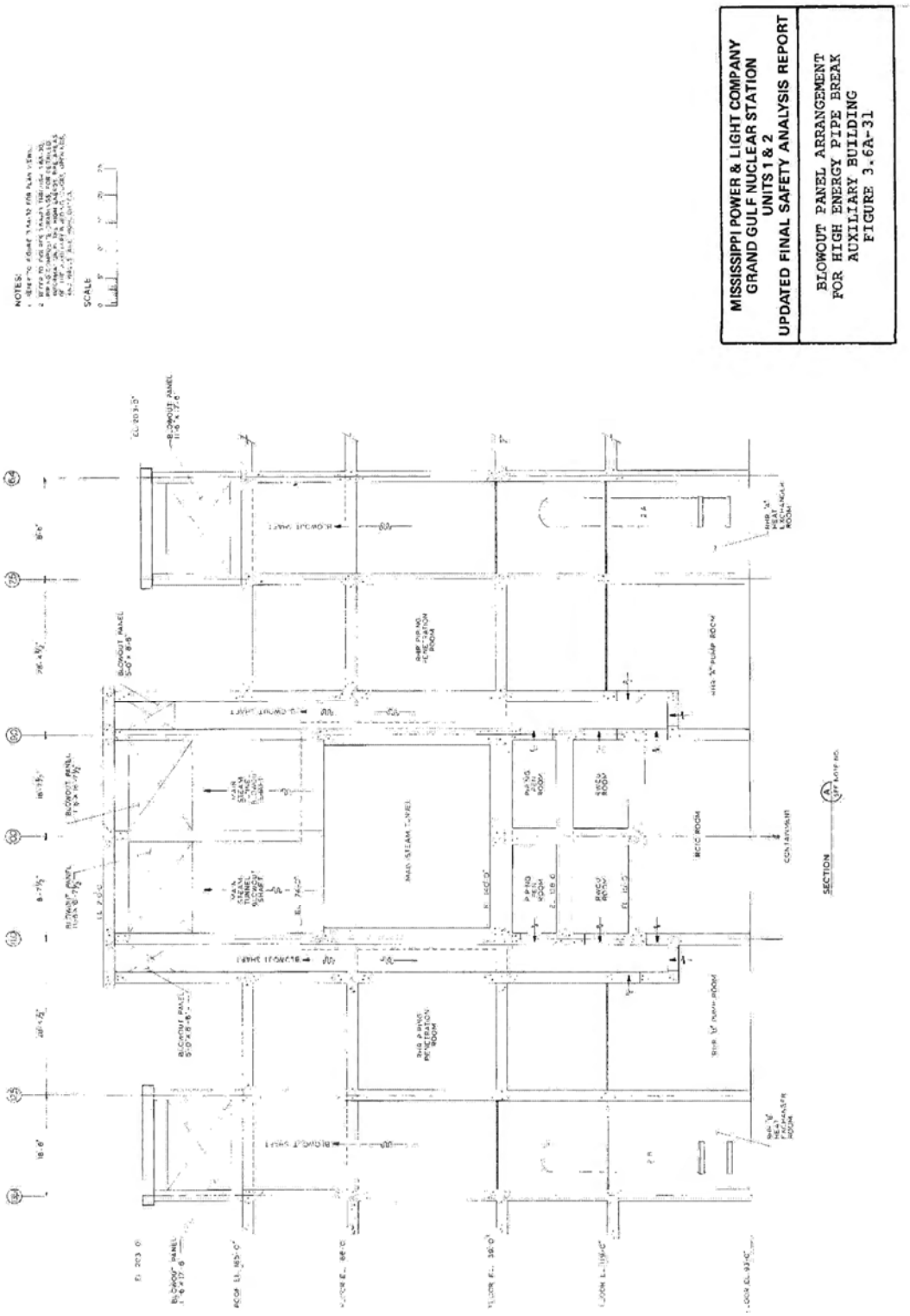


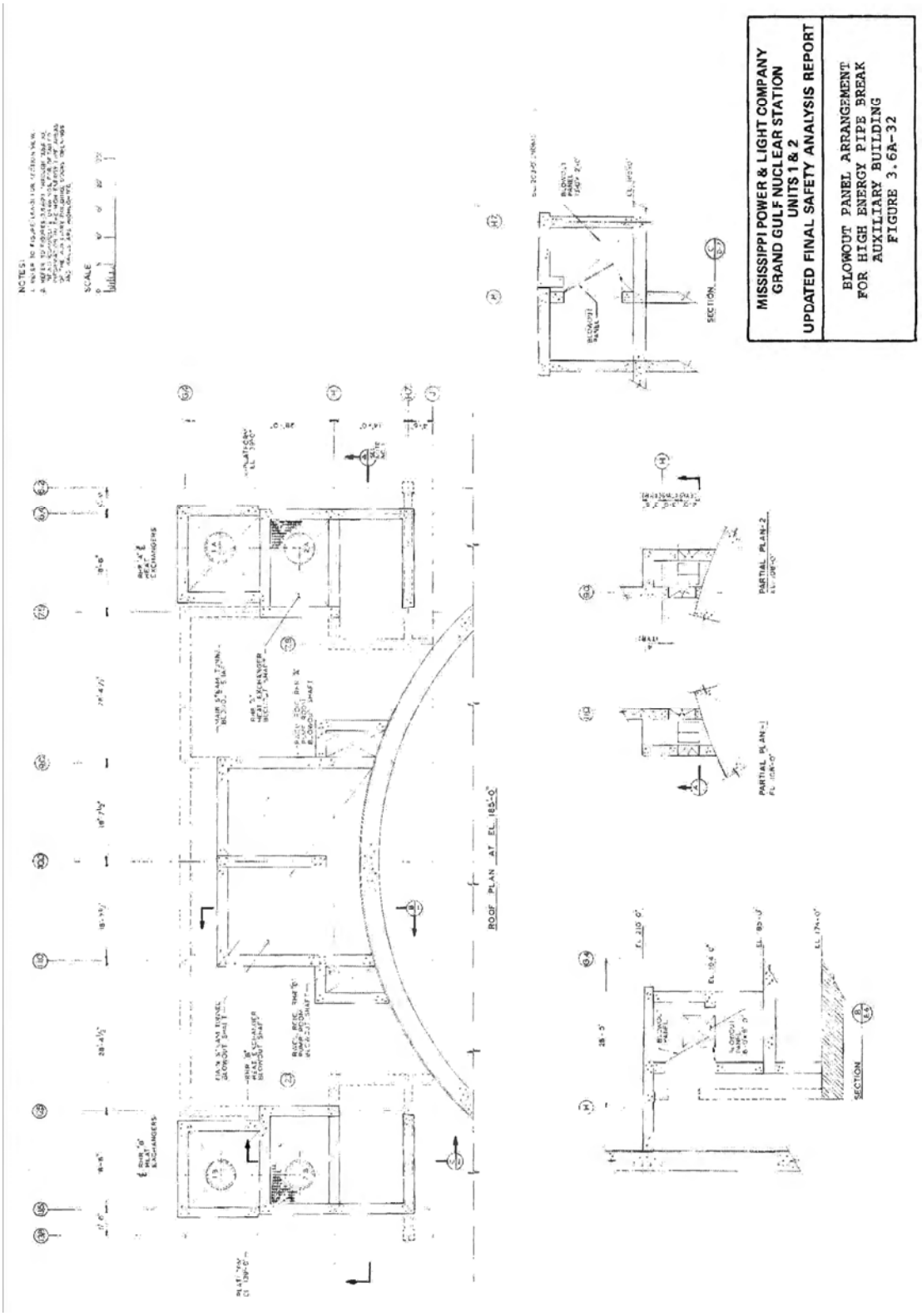


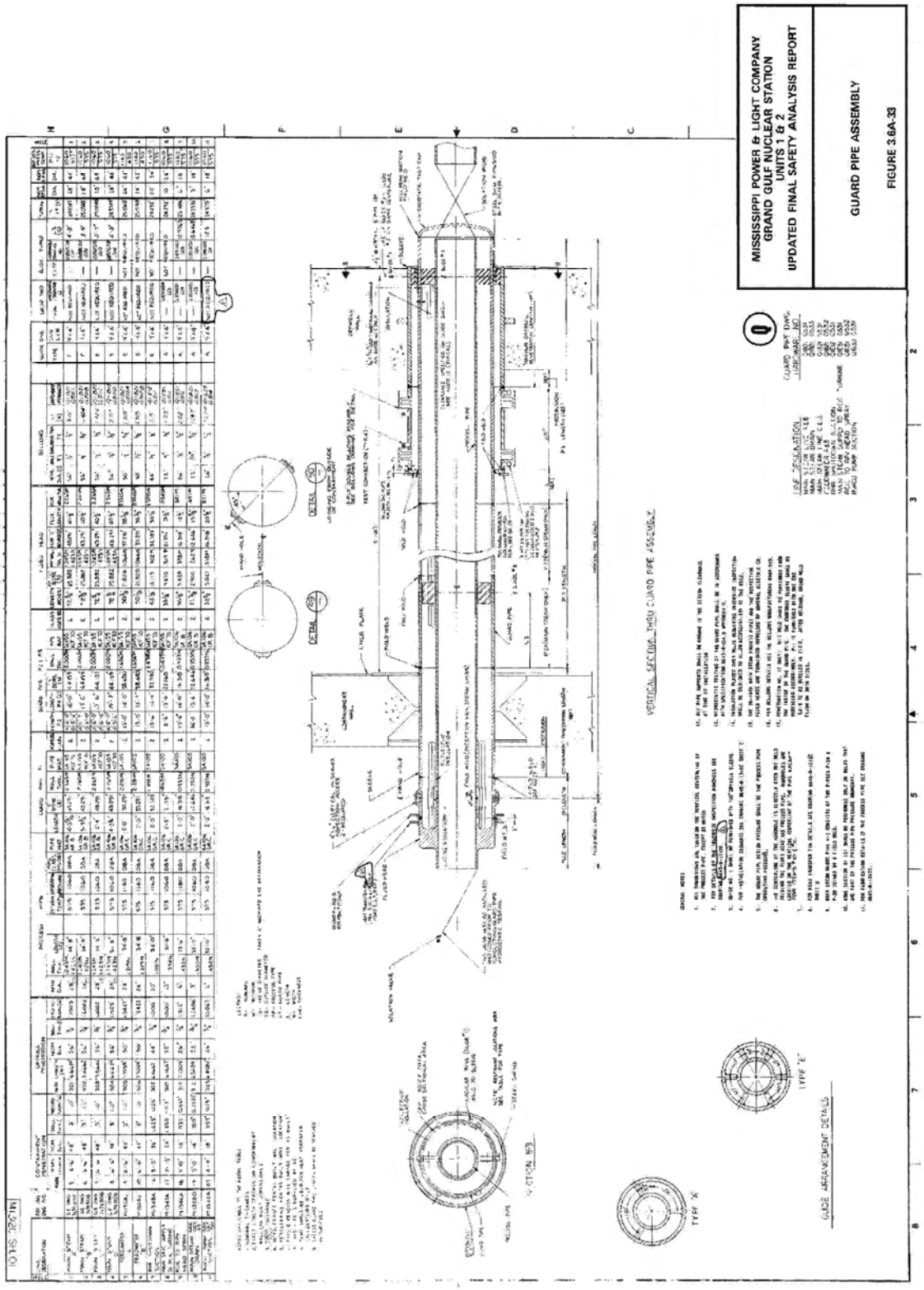


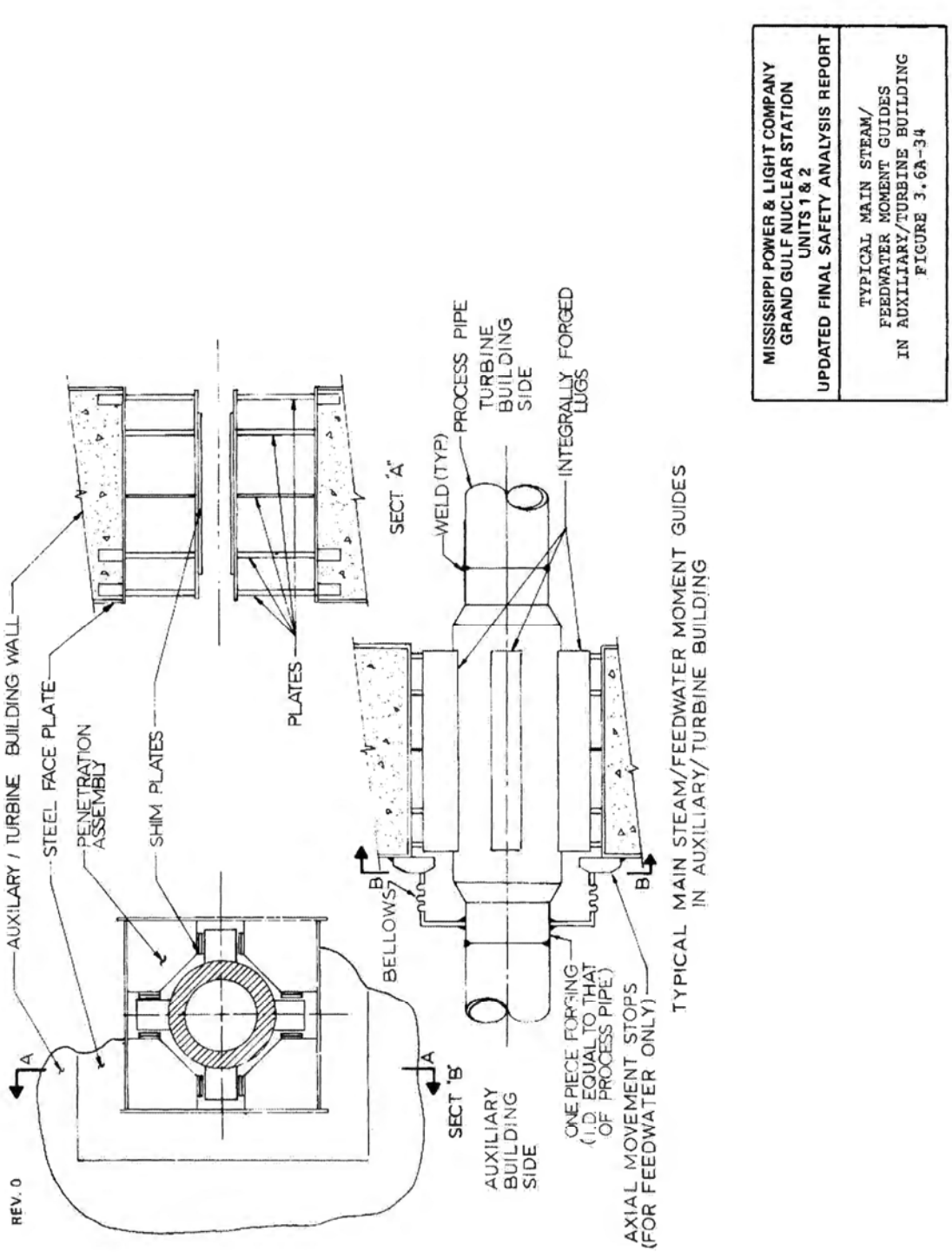


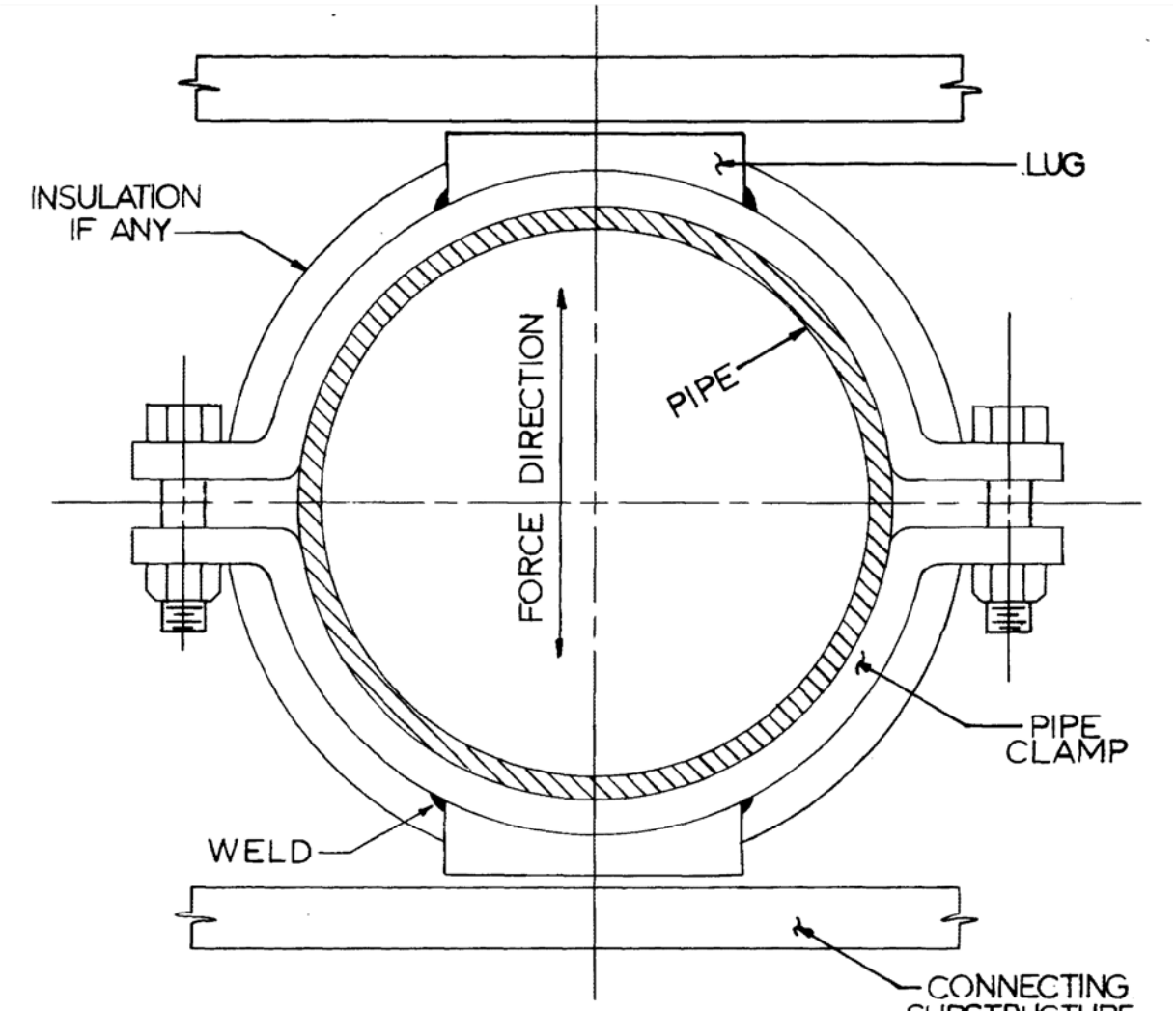



MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT
BLOWOUT PANEL ARRANGEMENT
FOR HIGH ENERGY PIPE BREAK
AUXILIARY BUILDING
FIGURE 3.6A-31
NOTES:
SCALE:
SECTION


MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT
BLOWOUT PANEL ARRANGEMENT
FOR HIGH ENERGY PIPE BREAK
AUXILIARY BUILDING
FIGURE 3.6A-32
ROOF PLAN AT EL. 185'-0"
SECTION
PARTIAL PLAN-2
PARTIAL PLAN-1
SECTION

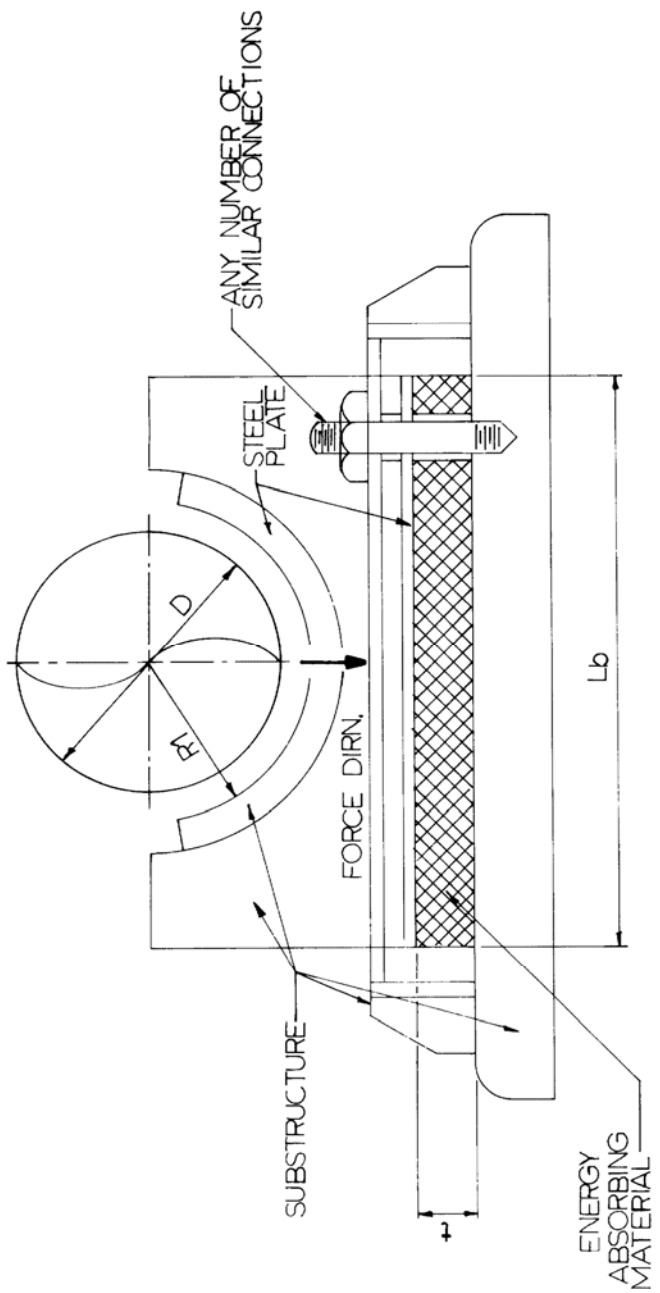

MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT
GUARD PIPE ASSEMBLY
FIGURE 3.6A-33


AUXILIARY / TURBINE BUILDING WALL
STEEL FACE PLATE
PENETRATION ASSEMBLY
SHIM PLATES
PLATES
SECT 'A'
WELD (TYP)
PROCESS PIPE
TURBINE BUILDING SIDE
INTEGRALLY FORGED LUGS
BELLOWS
B
B
A
A
SECT 'B'
AUXILIARY BUILDING SIDE
ONE PIECE FORGING (I.D. EQUAL TO THAT OF PROCESS PIPE)
AXIAL MOVEMENT STOPS (FOR FEEDWATER ONLY)
REV. 0
TYPICAL MAIN STEAM/FEEDWATER MOMENT GUIDES IN AUXILIARY/TURBINE BUILDING
MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT
TYPICAL MAIN STEAM/
FEEDWATER MOMENT GUIDES
IN AUXILIARY/TURBINE BUILDING
FIGURE 3.6A-34



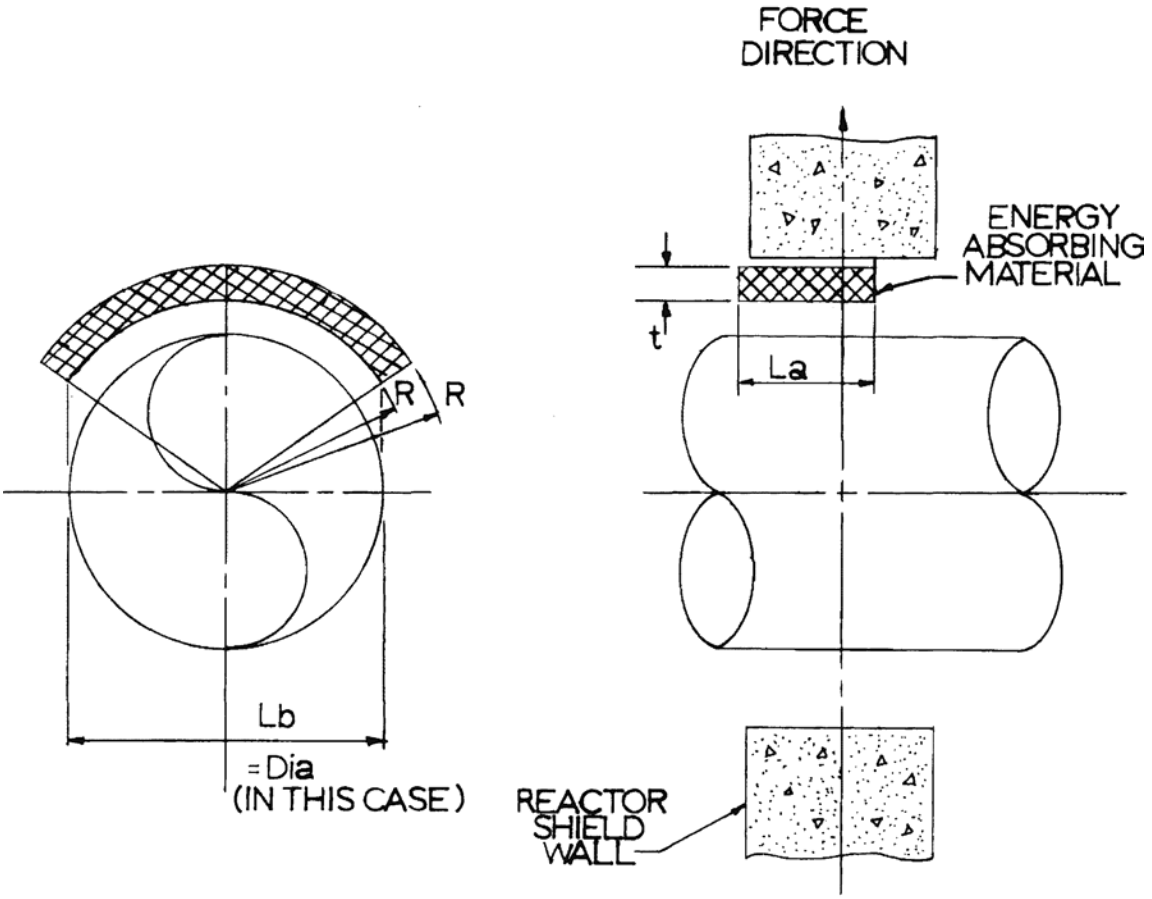
TYPICAL CLOSE-GAP RESTRAINT

MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT
TYPICAL CLOSE-GAP RESTRAINT FIGURE 3.6A-35

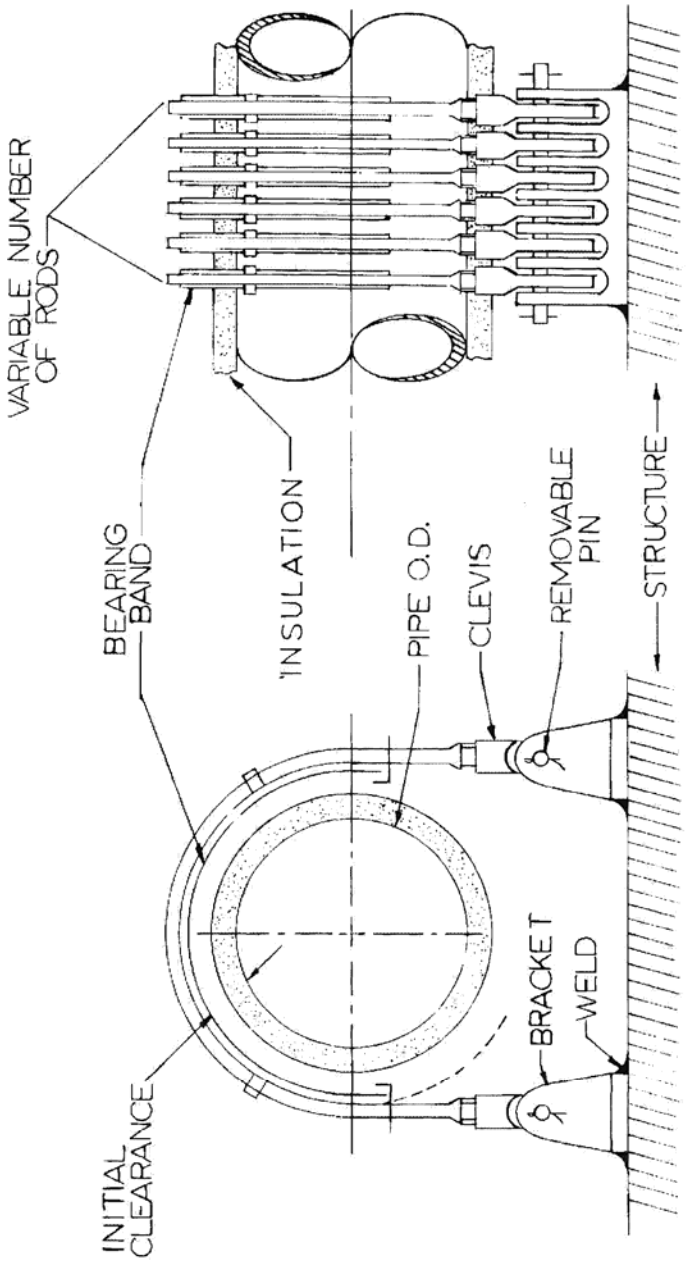


TYPICAL ENERGY DISSIPATING RESTRAINT

MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT	TYPICAL ENERGY DISSIPATING RESTRAINT FIGURE 3.6A-36

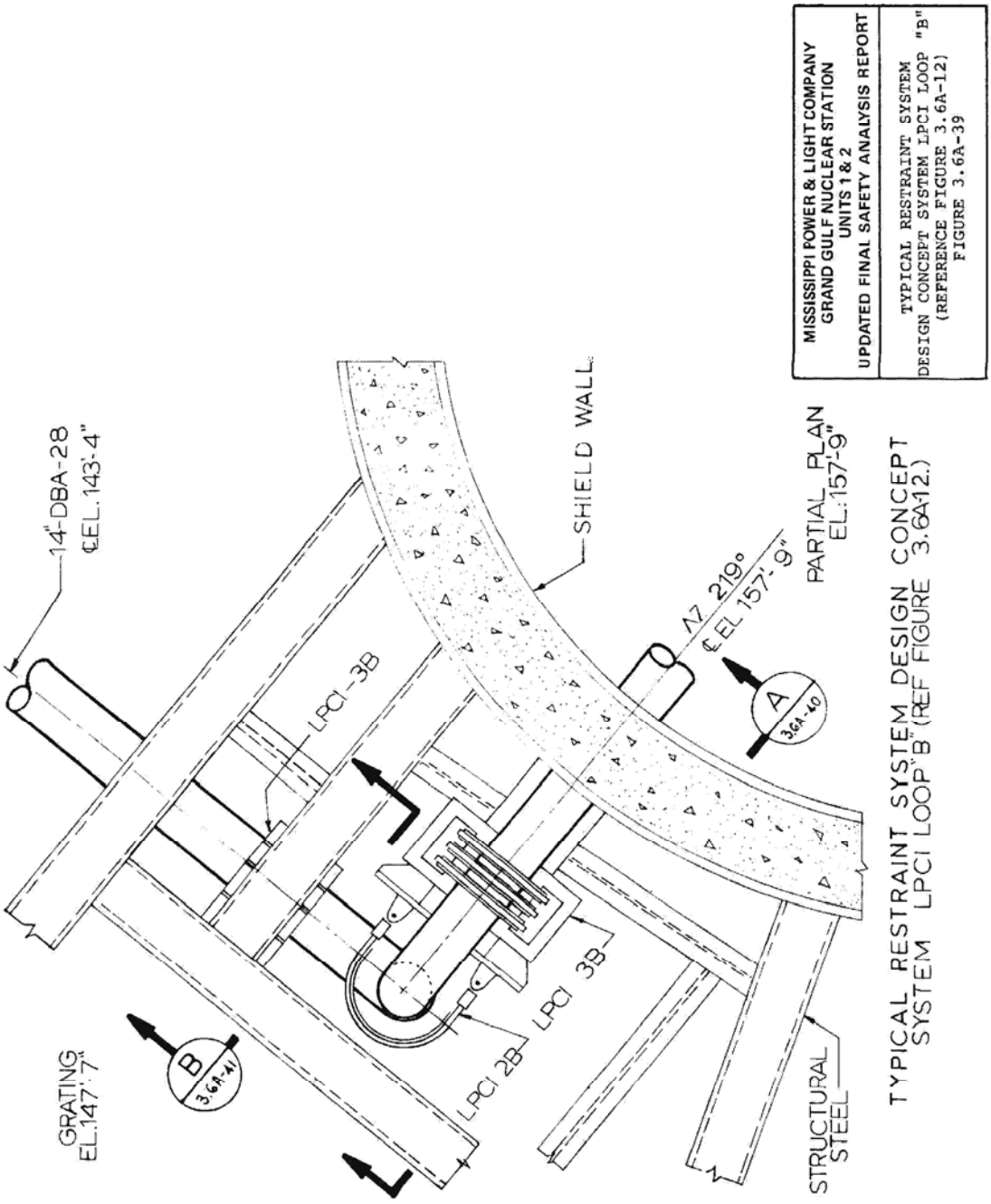


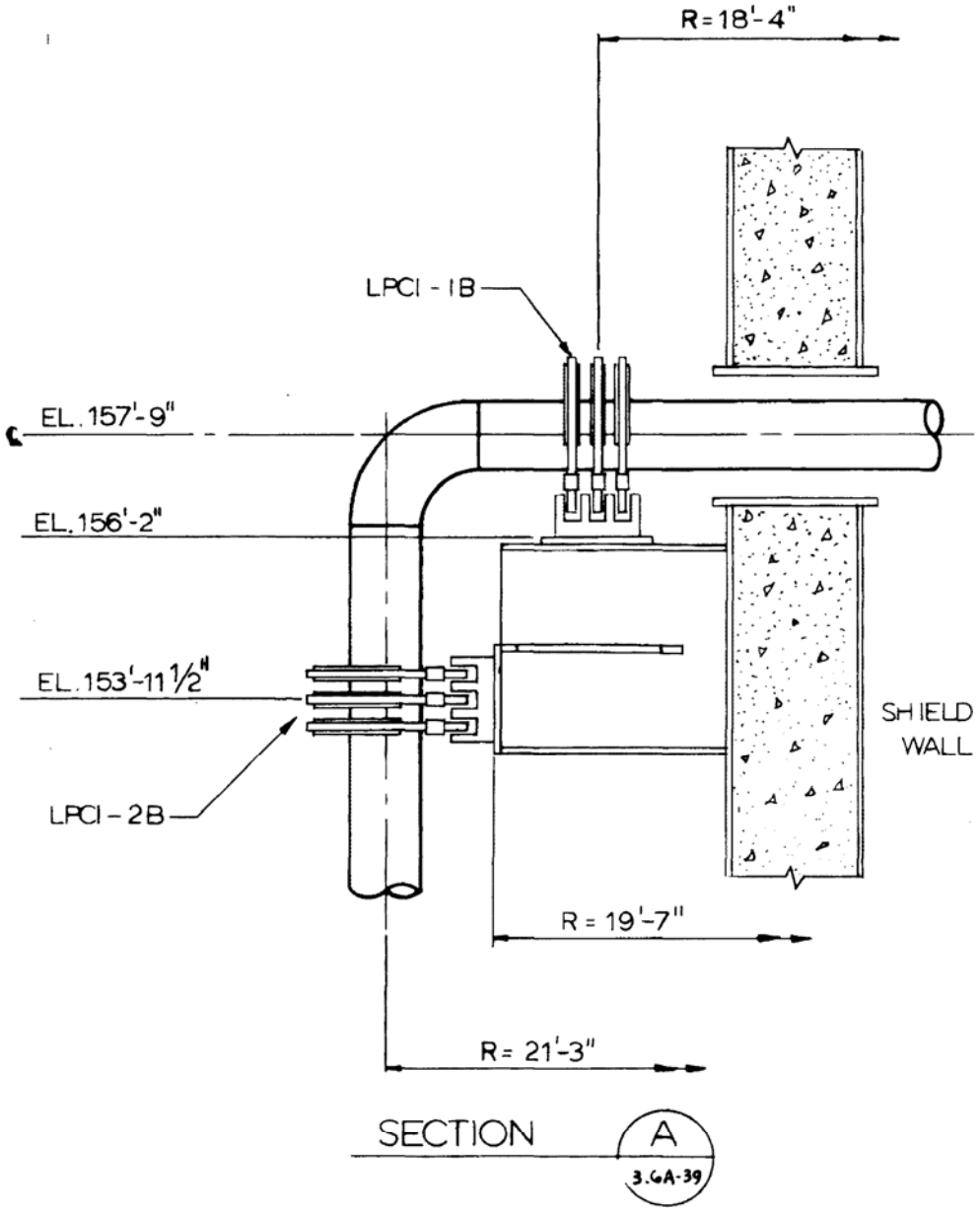
MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 **UPDATED FINAL SAFETY ANALYSIS REPORT**
TYPICAL ENERGY DISSIPATING RESTRAINT FIGURE 3.6A-37



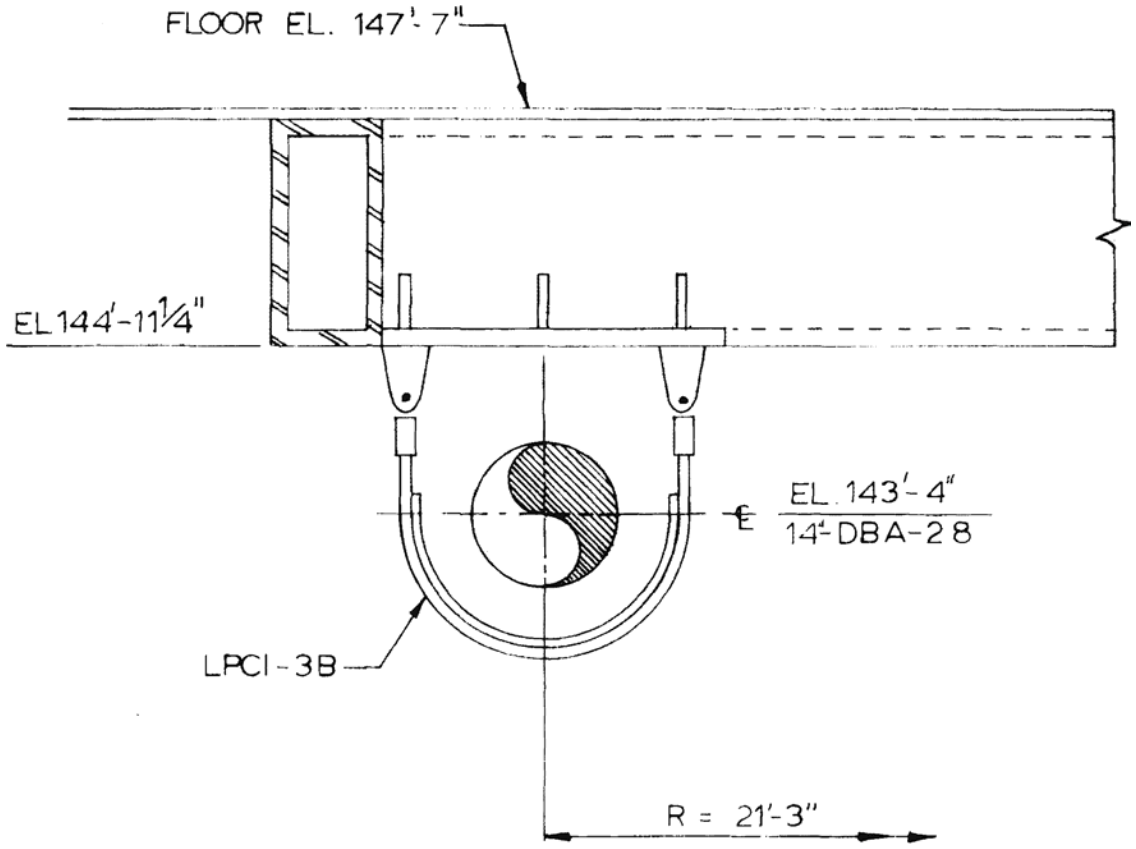
TYPICAL GEH PIPE WHIP RESTRAINT CONFIGURATION

MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT	TYPICAL GEH PIPE WHIP RESTRAINT CONFIGURATION FIGURE 3.6A-38


MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT
TYPICAL RESTRAINT SYSTEM
DESIGN CONCEPT SYSTEM LPCI LOOP "B"
(REFERENCE FIGURE 3.6A-12)
FIGURE 3.6A-39
14"-DBA-28
℄EL. 143'-4"
SHIELD WALL
LPCI-3B
LPCI 2B
LPCI 3B
GRATING
EL. 147'-7"
B
3.6A-41
STRUCTURAL STEEL
A7 219°
℄EL 157'-9"
PARTIAL PLAN
EL: 157'-9"
A
3.6A-40
TYPICAL RESTRAINT SYSTEM DESIGN CONCEPT
SYSTEM LPCI LOOP 'B' (REF FIGURE 3.6A12.)

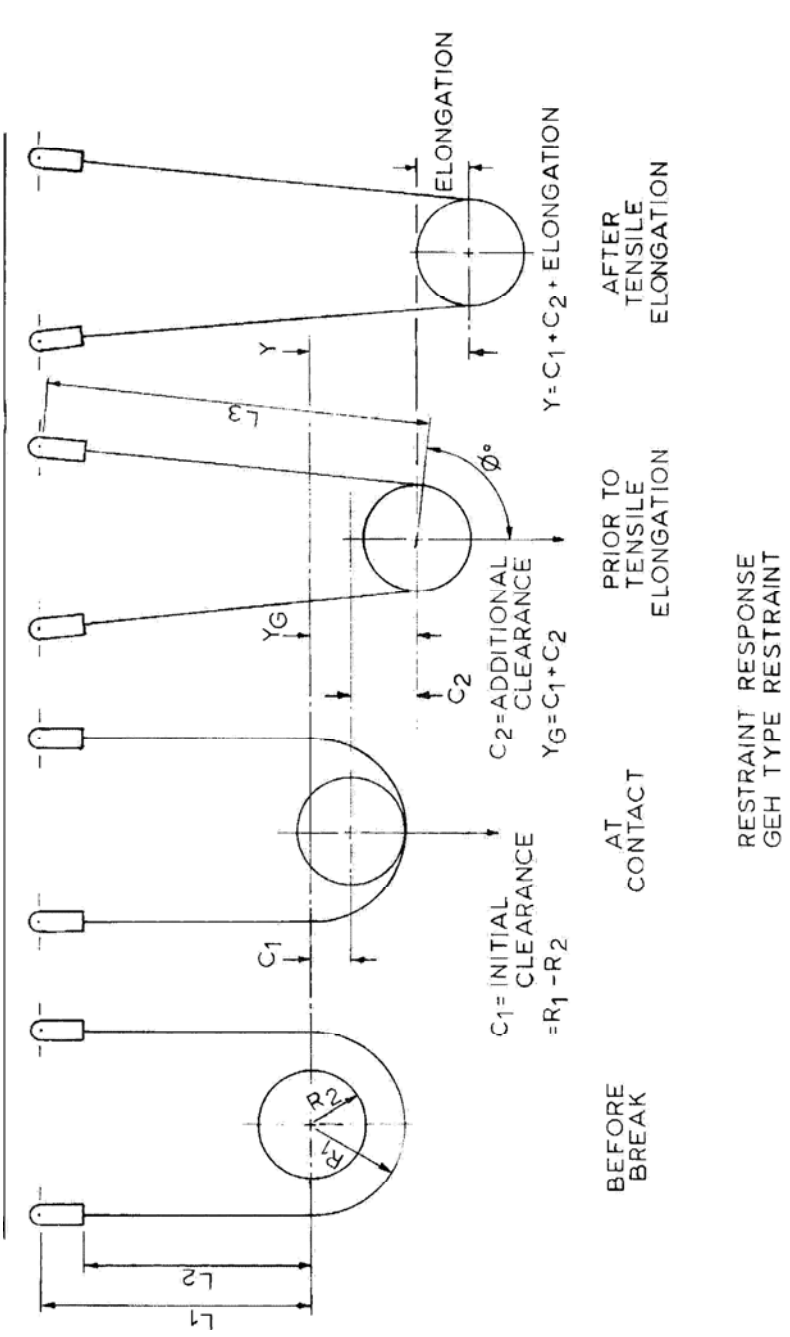


MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT
SECTION A FIGURE 3.6A-40



SECTION B
3.6A-39

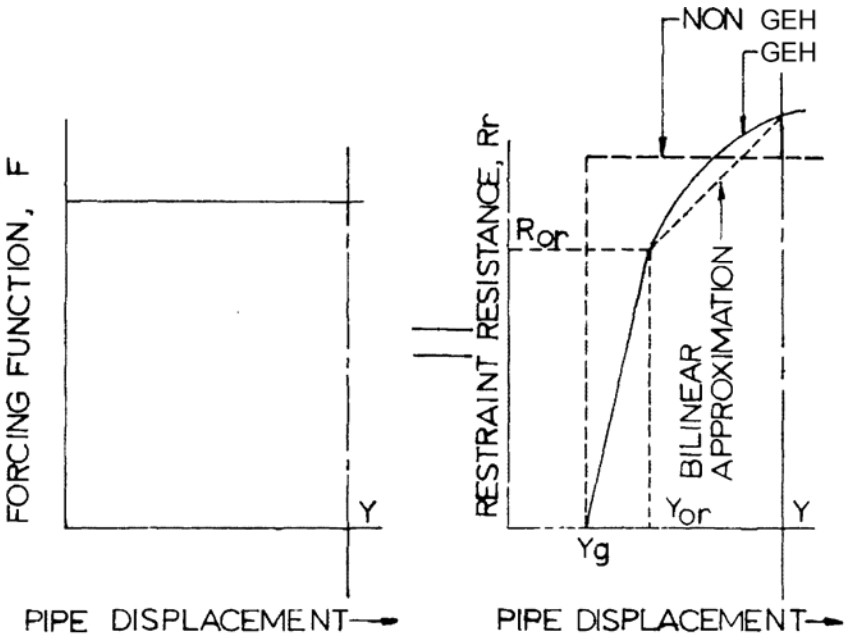
MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT
SECTION B FIGURE 3.6A-41

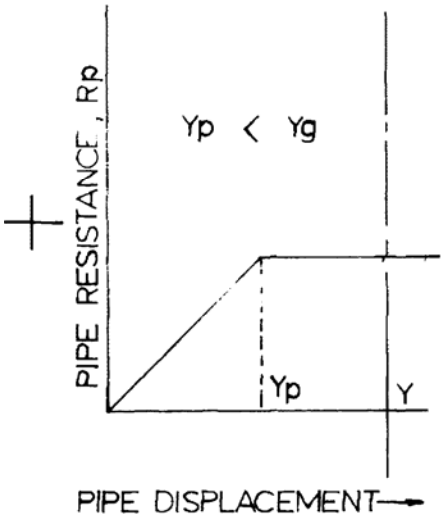


MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT

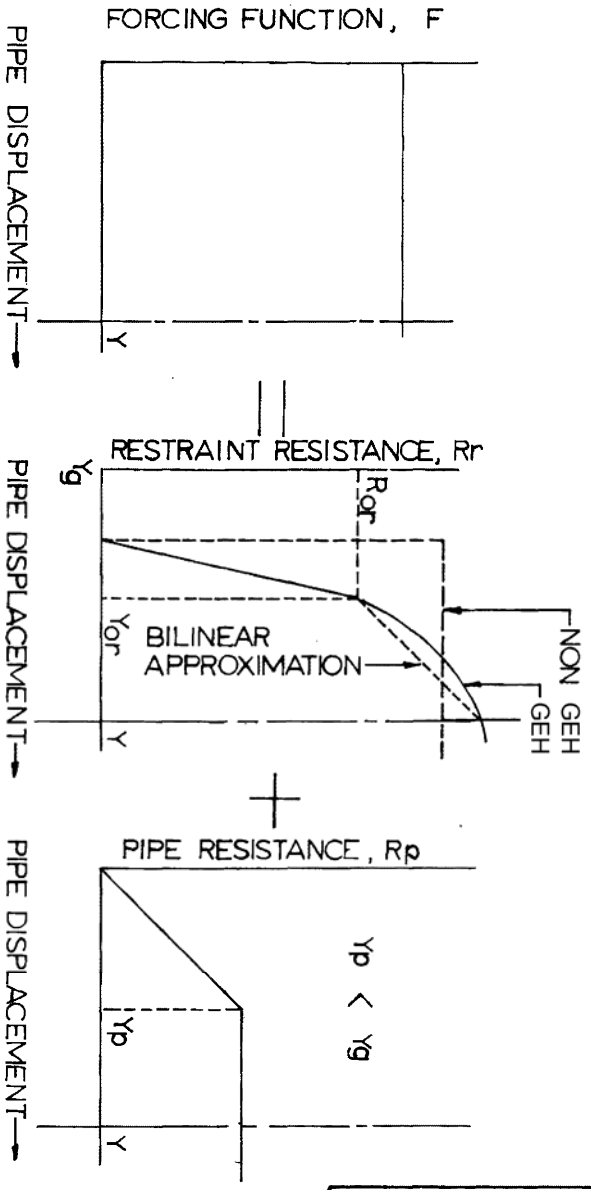
RESTRAINT RESPONSE
GEH TYPE RESTRAINT

FIGURE 3.6A-42





MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT
ENERGY BALANCE METHOD RESTRAINT/PIPE RESPONSE FIGURE 3.6A-43



MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT
ENERGY BALANCE METHOD RESTRAINT/PIPE RESPONSE FIGURE 3.6A-44

3.6B PROTECTION AGAINST DYNAMIC EFFECTS ASSOCIATED WITH POSTULATED RUPTURE OF PIPING (IN THE RECIRCULATION SYSTEM AND IN-CONTAINMENT MAIN STEAM)

3.6B.1 Postulated Piping Failures in Fluid Systems Inside and Outside of Containment

Refer to subsection 3.6A.1.

3.6B.2 Determination of Break Locations and Dynamic Effects Associated with the Postulated Rupture of Piping

Information concerning break and crack location criteria and methods of analysis is presented in this portion of the Safety Analysis Report. The location criteria and methods of analysis are needed to evaluate the dynamic effects associated with postulated breaks and cracks in high- and moderate-energy fluid system piping inside and outside of primary containment. This information confirms that the requirements for the protection of structures, systems, and components relied upon for safe reactor shutdown or to mitigate the consequences of a postulated pipe break have been met.

3.6B.2.1 Criteria Used to Define Break and Crack Location and Configuration

The following section establishes the criteria for the location and configuration of postulated breaks and cracks.

3.6B.2.1.1 Criteria for Main Steam Piping System - Inside Containment

3.6B.2.1.1.1 Definition of High-Energy Fluid System

High-energy fluid systems are defined to be those systems or portions of systems, that during normal plant conditions are either in operation or are maintained pressurized under conditions where either or both of the following are met:

Maximum temperature exceeds 200 F, or
Maximum pressure exceeds 275 psig.

“Normal plant conditions” are defined as the plant operating conditions during reactor startup, power plant operation, and reactor cold shutdown, but excluding test modes.

3.6B.2.1.1.2 Definition of Moderate-Energy Fluid System

Moderate-energy fluid systems are defined to be those systems, or portions of systems, that during normal plant conditions are either in operation or are maintained pressurized under conditions where both of the following are met:

Maximum temperature is 200 F or less, and
Maximum pressure is 275 psig or less.

Piping systems are classified as moderate-energy systems, when they operate as high-energy piping for only short operational periods in performing their system function but, for the major operational period qualify as moderate-energy fluid systems. An operational period is considered "short" if the total fraction of time that the system operates within the pressure-temperature conditions specified for a high-energy fluid system is less than two percent of the total operating time for which the system is designed.

3.6B.2.1.1.3 Postulated Pipe Breaks and Cracks

A postulated pipe break is defined as a sudden, gross failure of the pressure boundary either in the form of a complete circumferential severance (guillotine break) or as development of a sudden longitudinal, uncontrolled crack (longitudinal split) and is postulated for high-energy fluid systems only. For moderate-energy fluid systems, pipe breaks are confined to postulation of controlled cracks in piping and branch runs. These cracks affect the surrounding environmental conditions only, and do not result in whipping of the cracked pipe.

The following high-energy piping systems (or portions of systems) are considered as potential initiators of a postulated pipe break during normal plant conditions and are analyzed for potential damage dynamic effects:

a. All piping which is part of the reactor coolant pressure boundary and subject to reactor pressure continuously during station operation

b. All piping which is beyond the second isolation valve but which is subject to reactor pressure continuously during station operation

c. In addition to piping under a. and b. above, all other piping systems or portions of piping systems considered high-energy systems.

Portions of piping systems that are isolated from the source of the high-energy fluid during normal plant conditions are exempted from consideration of postulated pipe breaks. This would include portions of piping systems beyond a normally closed valve. Pump and valve bodies are also exempted from consideration of pipe break because of their greater wall thickness.

A high-energy piping system break is not postulated simultaneously with a moderate-energy piping system crack nor is any pipe break or crack outside the containment postulated concurrently with a postulated pipe break inside the containment.

3.6B.2.1.1.4 Exemptions from Pipe Whip Protection Requirements

Protection from pipe whip need not be provided if any one of the following conditions exists:

a. Piping which is classified as moderate-energy piping

b. Following a single postulated pipe break, piping for which the unrestrained movement of either end of the ruptured pipe in any feasible direction about a plastic hinge, formed within the piping, cannot impact any structure, system, or component important to safety.

c. Piping for which the internal energy level associated with whipping is insufficient to impair the safety function of any structure, system, or component to an unacceptable level. Any line restrictions (e.g., flow limiters) between the pressure source and break location, and the effects of either a single-ended or double-ended flow condition are accounted for, in the determination of the internal fluid energy level associated with the postulated pipe break reaction. The energy level in a whipping pipe is considered as insufficient to rupture an impacted pipe of equal or greater nominal pipe size and equal or heavier wall thickness.

No exemptions from pipe whip protection requirements have been claimed.

3.6B.2.1.1.5 Location for Postulated Pipe Breaks

Postulated pipe break locations are selected in accordance with the intent of Regulatory Guide 1.46, the U.S. Nuclear Regulatory Commission (NRC) Branch Technical Position APCSB 3-1, Appendix B, and as expanded in NRC Branch Technical Position MEB 3-1. For ASME Code Section III, Class 1 piping systems which are classified as high energy, the postulated break locations are:

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

b. At intermediate locations between the terminal ends where the maximum stress range between any two load sets (including zero load set) according to subarticle

NB-3600, ASME Code, Section III, for upset plant conditions and an independent OBE event transient, exceeds the following:

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

2. The stress ranges, as calculated by equations (12) or (13) of the Code, exceed 2.4 S_m or if the cumulative usage factor exceeds 0.1 when equation (10) exceeds 3 S_m.

c. In the event that two or more intermediate locations cannot be determined by stress or usage factor limits, a total of two intermediate locations were identified on a reasonable basis (see Note) for each piping run or branch run.

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*Terminal ends are extremities of piping runs that connect to structures, equipment, or pipe anchors that are assumed to act as rigid constraints to free thermal expansion of piping. A branch connection to a main piping run is a terminal end for a branch run, except when the branch and main run are modeled as a common piping system during the piping stress analysis.

Note: Reasonable bases* are one or more of the following:

1. Fitting locations

2. Highest stress or usage factor location

For ASME Code, Section III, Class 1 piping systems which contain moderate-energy fluids, through wall leakage cracks were postulated at locations that:

a. Demonstrate the adequacy of separation, or other means of protection, from required structures, systems, and components.

b. Through-wall leakage cracks were postulated in moderate-energy fluid system piping located within structures and compartments containing required systems and components. The cracks were postulated to occur individually at locations appropriate to form the basis for providing required protection from the hazards of fluid spraying, flooding, pressurization and other environmental conditions.

c. Moderate-energy fluid system piping or portions thereof that are located within a compartment or Confined area containing a postulated break in high-energy fluid system piping were acceptable without postulation of through-wall leakage cracks except where a postulated leakage crack in the moderate-energy fluid system piping results in more severe environmental conditions than the break in the proximate high-energy fluid piping system, in which case the provisions of paragraph b. above were applied.

Criteria for break locations in ASME Code, Section III, Class 1 piping systems in the area of the containment isolation valves are provided in subsection 3.6B.2.1.3.

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*Where more than two such intermediate locations are possible using the application of the above reasonable basis, those two locations possessing the greatest damage potential will be used. A break at each end of a fitting may be classified as two discrete break locations where the stress analysis is sufficiently detailed to differentiate stresses at each postulated break.

3.6B.2.1.1.6 Types of Breaks to be Postulated in Fluid System Piping

The following types of breaks were postulated in high-energy fluid system piping:

a. No breaks need be postulated in piping having a nominal diameter less than or equal to one inch.

b. Circumferential breaks were postulated only in piping exceeding a one-inch nominal pipe diameter.

c. Longitudinal splits were postulated only in piping having a nominal diameter equal to or greater than four inches.

d. Circumferential breaks were assumed at all terminal ends and at intermediate locations identified by the criteria in subsection 3.6B.2.1.1.5 for Class 1 piping systems. At each of the intermediate postulated break locations identified to exceed the stress and usage factor limits of the criteria in subsection 3.6B.2.1.1.5 for Class 1 piping systems, either a circumferential or a longitudinal break, or both, was postulated per the following:

    1. Circumferential breaks were postulated at fitting joints.

    2. Longitudinal breaks were postulated in the center of the fitting at two diametrically opposed points (but not concurrently) located so that the reaction force is perpendicular to the plane of the piping and produces out-of-plane bending.

    3. Consideration was given to the occurrence of either a longitudinal or circumferential break. Examination of the state of stress in the vicinity of the postulated break location may be used to identify the most probable type of break. If the maximum stress range in the longitudinal direction is greater than 1.5 times the maximum stress range in the circumferential direction, only the circumferential break may be postulated, and conversely if maximum stress range in the circumferential direction is greater than 1.5 times the stress range in the longitudinal direction, only the longitudinal break may be postulated. If no significant difference between the circumferential

and longitudinal stresses is determined, then both types of breaks were considered. The consideration of maximum stress range is not used to exempt circumferential breaks postulated on NSSS piping.

4. At intermediate locations chosen to satisfy the minimum break location criteria, only circumferential breaks were postulated.

e. For design purposes, a longitudinal break area was assumed to be the equivalent of one circumferential pipe area for all the longitudinal pipe breaks postulated on piping.

f. For both longitudinal and circumferential breaks, after assessing the contribution of upstream piping flexibilities, pipe whipping was assumed to occur in the plane defined by the piping geometry and configuration for circumferential breaks and out-of-plane for longitudinal breaks, and to cause pipe movement in the direction of the jet reaction.

g. For a circumferential break, the dynamic force of the jet discharge at the break location was based upon the effective cross-sectional flow area of the pipe and on a calculated fluid pressure as modified by an analytically or experimentally determined thrust coefficient. Justifiable line restrictions, flow limiters, and the absence of energy reservoirs were used, as applicable, in the reduction of the jet discharge.

The following through-wall leakage cracks are postulated in moderate-energy fluid systems (or portions of systems):

a. Cracks were postulated in moderate-energy fluid system piping and branch runs exceeding a nominal pipe size of 1 inch.

b. Crack openings were assumed as circular orifices of cross-sectional piping flow area equal to that of a rectangle whose dimensions are one-half pipe diameter in the length and one-half pipe wall thickness in width.

c. The flow from the crack opening was assumed to result in an environment that wets all unprotected components within the structure or compartment, with consequent flooding in

the compartment and communicating compartments based on a conservatively estimated timeflow period to effect corrective actions.

d. Through-wall leakage cracks instead of breaks may be postulated in the piping of those fluid systems that qualify as high-energy fluid systems for only short operational periods as defined in subsection 3.6B.2.1.1.2.

3.6B.2.1.2 Criteria for Recirculation Piping System - Inside Containment

The criteria employed for evaluation of the recirculation system piping, inside containment, are provided by subsection 3.6B.2.1.1.

3.6B.2.1.3 Criteria for Main Steam Piping System in Area of Containment Isolation Valves

3.6B.2.1.3.1 Locations for Postulated Pipe Breaks

No pipe break is postulated in the portion of piping between primary containment isolation valves provided that:

a. The stresses meet the requirements of subsection 3.6B.2.1.1.5.

b. The design requirements of subsections 3.6B.2.1.3.2, 3.6B.2.1.3.3, and 3.6B.2.1.3.4 are met.

3.6B.2.1.3.2 Welded Attachments to the Process Pipe

Welded attachments, for pipe supports or other purposes, to these portions of piping were avoided except where detailed stress analyses or tests are performed to demonstrate compliance with the limits of subsection 3.6B.2.1.1.5. In addition, the number of circumferential and longitudinal piping welds and branch connections were minimized. Where guard pipes are used, the enclosed portion of fluid system piping is of seamless construction unless specific provisions for access were made to permit inservice volumetric examination of the longitudinal weld. The length of these portions of piping was reduced to the minimum length practical. Inservice examinations will be in accordance with subsection 3.6A.2.1.c.2(g).

3.6B.2.1.3.3 Design of Pipe Anchors or Restraints

Pipe anchors or restraints (e.g., connections of containment penetrations and pipe whip restraints) were designed such that they are not welded directly to the outer surface of the piping except where such welds are 100 percent volumetrically examinable inservice and a detailed stress analysis was performed to demonstrate compliance with the limits of subsection 3.6B.2.1.1.5.

3.6B.2.1.3.4 Guard Pipe Design

The design for the guard pipe assembly is in accordance with NRC BTP MEB 3-1, Paragraph B.1.b(6), as discussed in subsection 3.6A.2.4.

3.6B.2.2 Analytical Methods to Define Blowdown Forcing Functions and Response Models

3.6B.2.2.1 Analytical Methods to Define Blowdown Forcing Functions

The rupture of a pressurized pipe causes the flow characteristics of the system to change, creating reaction forces which can dynamically excite the piping system. The reaction forces are a function of time and space and depend upon fluid state within the pipe prior to rupture, break flow area, frictional losses, plant system characteristics, piping system, and other factors. The methods used to calculate the reaction forces for various piping systems are presented in the following subsections.

3.6B.2.2.1.1 Main Steam Piping System - Inside Containment

This subsection is applicable only to the loads calculated for the design of moment guides G101 A through D. The blowdown forcing functions are as described in FSAR subsection 3.6A.2 for all other main steam break loads.

The criteria that were used for calculation of fluid blowdown forcing functions include:

a. Circumferential breaks were assumed to result in pipe severance and separation amounting to at least 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 the inelastic pipe whip analysis (subsection 3.6B.2.2.2).

b. The dynamic force of the jet discharge at the break location was based on the effective cross-sectional flow area of the pipe and on a calculated fluid pressure as modified by an analytically or experimentally determined thrust coefficient. Limited pipe displacement at the break location, line restrictions, flow limiters, positive pump-controlled flow, and the absence of energy reservoirs may be taken into account, as applicable, in the reduction of jet discharge.

c. Breaks are postulated to occur instantaneously.

Blowdown forcing functions are determined by either of two methods given in a. and b. below:

a. The predicted blowdown forces on pipes fed by a pressure vessel can be described by transient and steady-state forcing functions. The forcing functions used are based on methods described in Reference 1. These are simply described as follows:

    1. The transient forcing functions at points along the pipe result from the propagation of waves (wave thrust) along the pipe and from the reaction force due to the momentum of the fluid leaving the end of the pipe (blowdown thrust).

    2. The waves cause various sections of the pipe to be loaded with time-dependent forces. It was assumed that the pipe is one-dimensional, in that there is no attenuation or reflection of the pressure waves at bends, elbows, and the like. Following the rupture, a decompression wave was assumed to travel from the break at a speed equal to the local speed of sound within the fluid. Wave reflections were assumed to occur at the break end, changes in direction of piping, and the pressure vessel until a steady flow condition is established. Vessel and free space conditions were used as boundary conditions. The blowdown thrust causes a reaction force perpendicular to the pipe break.

3. The initial blowdown force on the pipe is taken as the sum of the wave and blowdown thrusts and is equal to the vessel pressure (P_o) times the break area (A). After the initial decompression period (i.e., the time it takes for a wave to reach the first change in direction), the force is assumed to drop off to the value of the blowdown thrust (i.e., 0.7 P_oA).

4. Time histories of transient pressure, flow rate, and other thermodynamic properties of the fluid can be used to calculate the blowdown force on the pipe using the following equation:

$$F = (P - P_a) + \frac{\rho u^2}{g} A$$

F = Blowdown force
P = Pressure at exit plane P
P_a = Ambient pressure
u = Velocity at exit plane
ρ = Density at exit plane
A = Area of break
g = Gravitational constant

5. Following the transient period a steady-state period is assumed to exist. Steady-state blowdown forces are calculated including frictional effects. For saturated steam, these effects reduce the blowdown forces from the theoretical maximum of 1.26 P_oA. The method of accounting for these effects is presented in Reference 1. For subcooled water, a reduction from the theoretical maximum of 2.0 P_oA is found through the use of Bernoulli's and standard equations such as Darcy's equation, which account for friction.

b. The following is an alternate method for calculating blowdown forcing functions:

The computer code RELAP3 (Ref. 2) is used to obtain exit plane thermodynamic states for postulated ruptures. Specifically, RELAP3 supplies exit pressure, specific volume, and mass rate. From these data, the blowdown reaction load is calculated using the following relation:

$$\frac{T}{AE} = P_E - P_{OO} + \frac{G_E^{\,2}\,\overline{v}_E}{g_c}$$

$$R = -\frac{T}{AE} \times AE$$

$\frac{T}{AE}$ = thrust per unit break area - lb_f/ft^2

P_E = exit pressure - lb_f/ft^2

P_{OO} = receiver pressure - lb_f/ft^2

G_E = exit mass flux - $\frac{lbm}{sec-ft^2}$

$\overline{v}_E$ = exit specific volume - ft^3/lb_m

g_c = gravitational constant - $\frac{32.174\ ft-lb_m}{lb_f-sec^2}$

R = Reactor force on the pipe - lb_f

3.6B.2.2.1.2 Recirculation Piping System - Inside Containment

The criteria and methods used to calculate fluid blowdown forcing functions for postulated ruptures of the recirculation piping system, inside containment, are also provided by subsection 3.6B.2.2.1.1.

3.6B.2.2.2 Pipe Whip Dynamic Response Analyses

The prediction of time-dependent and steady-thrust reaction loads caused by blowdown of sub-cooled, saturated, and two-phase fluid from a ruptured pipe is used in design and evaluation of dynamic effects of pipe breaks. A detailed discussion of the analytical methods employed to compute these blowdown loads is given in subsection 3.6B.2.2.1. A detailed discussion of analytical methods used to account for this loading is discussed below.

3.6B.2.2.2.1 Main Steam Piping System - Inside Containment

This section is applicable only to the loads calculated for the design of moment guides G101 A through D. The blowdown forcing functions are as described in FSAR subsection 3.6A.2 for all other main steam break loads.

The criteria used for performing the pipe whip dynamic response analyses include:

a. A pipe whip analysis was performed for each postulated pipe break. However, a given analysis could be used for more than one postulated break location if the blowdown forcing function, piping and restraint system geometry, and piping and restraint system properties are conservative for other break locations.

b. The analysis includes the dynamic response of the pipe in question and the pipe whip restraints which transmit loading to the structures.

c. The analytical model adequately represents the mass/ inertia and stiffness properties of the system.

d. Pipe whipping was assumed to occur in the plane defined by the piping geometry and configuration and to cause pipe movement in the direction of the jet reaction.

e. Piping within the broken loop was no longer considered part of the RCPB. Plastic deformation in the pipe was considered as a potential energy absorber. Limits of strain were imposed which were similar to strain levels allowed in restraint plastic members. Piping systems were designed so that plastic instability did not occur in the pipe at the design dynamic and static loads unless damage studies were performed which showed the consequences did not result in direct damage to any essential system or component.

f. Components such as vessel safe ends and valves which were attached to the broken piping system and did not serve a safety function or whose failure did not further escalate the consequences of the accident were not designed to meet ASME Code imposed limits for essential components under faulted loading. However, if these components were required for safe shutdown, or serve a safety function to

protect the structural integrity of an essential component, limits to meet the Code requirements for faulted conditions and limits to ensure operability if required were met.

The pipe whip analysis was performed using the PDA computer program (Ref. 3). PDA is a computer program used to determine the response of a pipe subjected to the thrust force occurring after a pipe break. The program treats the situation in terms of generic pipe break configuration, which involves a straight, uniform pipe fixed at one end and subjected to a time-dependent thrust-force at the other end. A typical restraint used to reduce the resulting deformation is also included at a location between the two ends. Nonlinear and time-independent stress-strain relations are used for the pipe and the restraint. Similar to the popular plastic-hinge concept, bending of the pipe is assumed to occur only at the fixed end and at the location supported by the restraint. Shear deformation is also neglected. The pipe bending moment deflection (or rotation) relation used for these locations was obtained from a static nonlinear cantilever beam analysis. Using the moment-rotation relation, nonlinear equations of motion of the pipe were formulated using an energy consideration and the equations were numerically integrated in small time steps to yield time-history information of the deformed pipe.

A comprehensive verification program has been performed to demonstrate the conservatisms inherent in the PDA pipe whip computer program and the analytical methods utilized. Part of this verification program included an independent analysis by Nuclear Services Corporation, under contract to the General Electric Company, of the recirculation piping system for the 1969 Standard Plant Design. The recirculation piping system was chosen for study due to its complex piping arrangement and assorted pipe sizes. The NSC analysis included elastic-plastic pipe properties, elastic-plastic restraint properties, and gaps between the restraint and pipe and is documented in Reference 4. The piping/restraint system geometry and properties and fluid blowdown forces were the same in both analyses. However, a linear approximation was made by NSC for the restraint load - deflection curve supplied by GEH. This approximation is demonstrated in Figure 3.6B-1. The effect of this approximation is to give lower energy absorption of a given restraint deflection. Typically, this yields higher restraint deflections and lower restraint to structure loads than the GEH analysis. The deflection limit used by NSC is the design deflection at one-half of the ultimate

uniform strain for the GEH restraint design. The restraint properties used for both analyses are provided in Table 3.6B-1 and Figure 3.6B-2.

A comparison of the NSC analysis with the PDA analysis, as presented in Table 3.6B-2, shows that PDA predicts higher loads in 15 of the 18 restraints analyzed. This is due to the NSC model including energy absorbing effects in secondary pipe elements and structural members. However, PDA predicts higher restraint deflections in 50 percent of the restraints. The higher deflections predicted by NSC for the lower loads are caused by the linear approximation used for the force - deflection curve rather than by differences in computer techniques. This comparison demonstrates that the simplified modeling system used in PDA is adequate for pipe rupture loading, restraint performance, and pipe movement predictions within the meaningful design requirements for these low probability postulated accidents.

3.6B.2.2.2.2 Recirculation Piping System - Inside Containment

The criteria and methods used to determine pipe whip dynamic response for postulated ruptures of the recirculation piping system, inside containment, are provided by subsection 3.6B.2.2.2.1.

3.6B.2.3 Dynamic Analysis Methods to Verify Integrity Operability

3.6B.2.3.1 Jet Impingement Analyses and Effects on Safety-Related Components

The methods used to evaluate the jet effects resulting from the postulated breaks of high-energy piping are provided in subsection 3.6A.2.

3.6B.2.3.2 Pipe Whip Effects on Safety-Related Components

Subsection 3.6A.2 provides the criteria and methods used to evaluate the effects of pipe displacements on safety-related structures, systems, and components following a postulated pipe rupture.

3.6B.2.3.3 Loading Combinations and Design Criteria for Pipe Whip Restraints

Pipe whip restraints, as differentiated from piping supports, are designed to function and carry load for an extremely low probability gross failure in a piping system carrying high-energy fluid. The piping integrity does not usually depend on the pipe whip restraints for any loading combination. When the piping integrity is lost because of a postulated break, the pipe whip restraint acts to limit the movement of the broken pipe to an acceptable distance. The pipe whip restraints (i.e., those devices which serve only to control the movement of a ruptured pipe following gross failure) will be subjected to once in a lifetime loading. For the purpose of design, the pipe break event is considered to be a faulted plant condition and the pipe, its restraints, and structure to which the restraint is attached, are analyzed and designed accordingly.

3.6B.2.3.3.1 Main Steam Pipe Whip Restraints - Inside Containment

Refer to Section 3.6A.

3.6B.2.3.3.2 Recirculation Piping System Pipe Whip Restraints

The pipe whip restraints designed, tested, fabricated, and installed by GEH for the recirculation loop piping, utilize energy absorbing U-rods to attenuate the kinetic energy of a ruptured pipe. A typical pipe whip restraint is shown in Figure 3.6B-3. A principal feature of these restraints is that they are installed with several inches of annular clearance between them and the process pipe. This allows for installation of normal piping insulation and unrestricted pipe thermal movements. Select critical locations inside primary containment are also monitored during hotfunctional testing to provide verification of adequate clearances prior to plant operation.

The specific design objectives for the restraints are:

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

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

c. The restraints provide minimum hindrance to inservice inspection of the process piping.

For the purposes of design, the pipe whip restraints are designed for the following dynamic loads:

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

b. Dynamic inertia loads of the moving pipe section which is accelerated by the blowdown thrust, and subsequent impact on the restraint.

c. Design characteristics of the pipe whip restraints are included and verified by the pipe whip dynamic analysis described in subsection 3.6B.2.2.2.2.

d. Since the pipe whip restraints are not contacted during normal plant operation, the postulated pipe rupture event is the only design loading condition.

The recirculation loop pipe whip restraints are composed of several components, each of which perform a different function. These components are categorized as Types I, II, III, and IV, as described below:

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

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

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

Type IV - Structural and civil components - steel and concrete structures which ultimately must carry the restraint load (e.g., RPV shield wall, trusses).

Each of these components is typically constructed of a different material, with a different design objective in order to perform the overall design function. Therefore, the material and inspection requirements and design limits for each are somewhat different. These requirements for each component are as given below:

a. Type I Restraint Material (e.g., U-rods)

1. Materials - All materials which are used to absorb energy through significant plastic deformation conform to:

(a) ASME - Section III, subsection NB, B&PV Code for Class I Components; or

(b) ASTM Specifications with consideration for brittle fracture control; or

(c) ASME - Section III, subsection NF, B&PV Code if applicable

(d) GEH Material Specifications

2. Inspection - Inspection and identification of material shall conform to:

(a) ASME - Section III, subsection NB, B&PV Code for Class I Components (Section V Nondestructive Examination Methods); or

(b) ASTM Specifications procedures including volumetric and surface inspection; or

(c) ASME - Section III, subsection NF, B&PV Code if applicable

(d) GEH Methods and Acceptance Standards

3. Design limits

(a) Design local strain - The permanent strain in metallic ductile materials is limited to:

- o 50 percent of the minimum actual ultimate uniform strain (at the maximum stress on an engineering stress-strain curve) based on restraint material tests; or
- o 1/2 of minimum percent elongation as specified in the applicable ASME Section III, B&PV Code, or ASTM Specifications, when demonstrated to be as or more conservative than the above

(b) Design steady-state load - The maximum restraint load is limited to:

- o 80 percent of the minimum calculated static ultimate restraint strength at the drywell design temperature. This strain is less than 50 percent of the ultimate uniform strain for all materials which will be used for Type I components.

(c) Dynamic material mechanical properties - The material selected must exhibit tensile impact properties which are not less than:

- o 70 percent of the static percent elongation; or
- o 80 percent of the statically determined minimum total energy absorption

b. Type II restraint material (e.g., clevises, brackets, pins)

1. Materials - Material selection conforms to:

(a) ASTM Specifications including consideration for brittle fracture control; or

(b) ASME - Section III, subsection NF, B&PV Code, if applicable

(c) GEH Material Specifications |

2. Inspection - Inspection conforms to:

(a) ASME/ASTM requirements or process qualification and finished part surface inspection per ASTM methods; or

(b) ASME - Section III, subsection NF, B&PV Code, if applicable

(c) GEH Methods and Acceptance Standards |

3. Design limits - Design limits are based on the following stress limits:

(a) Primary stresses (in accordance with definitions in ASME Section III) shall be limited to the higher of:

70 percent of S_u where as S_u = minimum ultimate strength by tests or ASTM specification

For the materials used in Type II and III components $0.7S_u$ corresponds to a strain of less than 0.5 of the ultimate uniform strain.

$S_y + 1/3\ (S_u - S_y)$ where S_y = minimum yield strength by test or ASTM specification; or

(b) Recommended stress limits per ASME Section III, subsection NF for faulted conditions, if applicable

c. Type III restraint material (fasteners)

1. Materials - Fastener material conforms to ASTM, ASME, or MIL requirements

2. Inspection - All fasteners are inspected or certified per applicable ASTM, ASME, or MIL specifications

3. Design limits - Same as Type II

d. Type III restraint material (welds)

1. Materials - Weld materials for attachment to carbon steel structures are limited to low hydrogen type

2. Inspection - Liquid penetrant surface inspection is performed per:

   (a) ASTM Specification E165; or

   (b) AWS Structural Welding Codes, AWS-D1.1

3. Design limits - Design limits are based on the following stress limits:

   (a) The maximum primary weld stress intensity (two times maximum shear stress) is limited to three times AWS or AISC building allowable weld shear stress.

4. Procedures - Procedures and welders are qualified per Revision 2 of 1975 edition of AWS D1.1 for welding in building structures.*

e. Type IV restraint material (structural and civil components)

   Material, inspection, and design requirements for the structural and civil components are provided by industry standards such as AISC, ACI, and ASME Section III, Division II, along with appropriate requirements imposed for similar loading events. These components are also designed for other operational and accident loadings, seismic loadings, wind loadings, and tornado loadings.

The design basis approach of categorizing components is consistent in allowing less stringent inspection requirements for those components subject to lower stresses. Considerable strength margins exist in Type II through IV components event to the limit of load capacity (fracture) of a Type I component. Impact properties in all components are considered since brittle type failures could reduce the restraint system effectiveness.

*Later editions that are adopted for use are specified in the component's/structures' specification or standard

In addition to the design considerations discussed above, strain rate effects and other material property variations have been considered in the design of the pipe whip restraints. The material properties utilized in the design have included one or more of the following methods:

a. Code minimum or specification yield and ultimate strength values for the affected components and structures are used for both the dynamic and steady state events, or

b. Not more than a 10 percent increase in code or specification values is used when designing components or structures for the dynamic event. Code minimum or specification yield and ultimate strength values are used for the steady state loads, or

c. Representative or actual test data values are used in the design of components and structures, including justifiably elevated strain rate affected stress limits in excess of 10 percent, or

d. Representative or actual test data are used for any affected component(s) and the minimum code or specification values for the structures for the dynamic and the steady state events.

3.6B.2.4 Guard Pipe Assembly Design Criteria

This is discussed in subsection 3.6A.2.4.

3.6B.2.5 Material to be Submitted for the Operating License Review

3.6B.2.5.1 Implementation of Criteria for Pipe Break and Crack Location and Orientation

3.6B.2.5.1.1 Postulated Pipe Breaks in Main Steam Piping System Inside Containment

The criteria for selection of postulated pipe breaks in the main steam piping system, inside containment, are provided in subsection 3.6B.2.1.1. The postulated pipe break locations and types selected in accordance with these criteria for main steam lines A - D, are shown on Figure 3.6B-5a. Conformance with these criteria is demonstrated in the documents referenced on Figure 3.6A-1A and the associated model diagram of the piping as shown on

Figures 3.6B-6a and 3.6B-6b. The referenced documents are maintained in accordance with engineering administrative procedures.

For each line, no breaks have been postulated in that portion of the main steam piping near the containment isolation valves in accordance with the criteria of subsection 3.6B.2.1.3. Conformance with these criteria is demonstrated in Table 3.6B-4.

3.6B.2.5.1.2 Postulated Pipe Breaks in Recirculation Piping System - Inside Containment

The criteria for selection of postulated pipe breaks in the recirculation piping system, inside containment, are provided in subsection 3.6B.2.1.2. The postulated pipe break locations and types selected in accordance with these criteria are shown in Figure 3.6B-4a. Conformance with these criteria is demonstrated in the documents referenced on the figure. The referenced documents are maintained in accordance with engineering administrative procedures.

3.6B.2.5.2 Implementation of Special Protection Criteria

3.6B.2.5.2.1 Pipe Whip Restraints for Main Steam Piping System Inside Containment

Refer to subsection 3.6A.2.

3.6B.2.5.2.2 Pipe Whip Restraints for Recirculation Piping System - Inside Containment

The pipe whip restraints provided for this recirculation piping system are also shown in Figures 3.6B-4a and 3.6B-3. This system of restraints has also been found to prevent unrestrained pipe whip resulting from a postulated rupture at any of the identified break locations.

3.6B.2.5.3 Summary of Jet Effects Analysis Results

Refer to subsection 3.6A.2.5.3.

3.6B.2.5.4 Implementation of Guard Pipe Criteria

3.6B.2.5.4.1 Main Steam Piping System Near Containment Penetration Areas

The stress levels in the main steam piping near the containment isolation valves, and comparisons to the criteria allowable of subsection 3.6B.2.1.6, are provided in Table 3.6B-4. Details of the guard pipe design are provided in subsection 3.6A.2.4.

3.6B.3 References

1. GE Spec. No. 22A2625 - "System Criteria and Applications for Protection Against the Dynamic Effects of Pipe Break."

2. RELAP 3 - A Computer Program for Reactor Blowdown Analysis IN-1321, issued June 1970, Reactor Technology TID-4500.

3. GE Report NEDE-10313 - "PDA - Pipe Dynamic Analysis Program for Pipe Rupture Movement." (Proprietary Filing).

4. Nuclear Services Corporation Report No. GEN-02-02, "Final Report Pipe Rupture Analysis of Recirculation System for 1969 Standard Plant Design."

TABLE 3.6B-1: RESTRAINT DATA

General Restraint Data for 1 Bar of a Restraint

$F = C_2 (\text{restraint})^n$

Where restraint = pipe - total clearance

Pipe Size (in.)	Rest Load Direction	C_2	n	Limit Restraint	Initial Clearance	Effective Clearance	Total Clearance
12	0°	27,733	.24	6.129	4	1.941	5.941
12	90°	14,795	.401	9.063	4	12.247	16.247
16	0°	109,265	.24	6.278	4	1.934	5.934
16	90°	62,599	.377	8.978	4	12.187	16.187
24	0°	102,228	.24	8.222	4	1.984	5.984
24	90°	55,531	.375	11.972	4	13.685	17.685
24	38°*	109,888	.24	5.588	4	5.698	9.698
24	52°*	109,835	.24	5.473	4	8.462	12.462

* Applies to Restraint RCR 3 only.

TABLE 3.6B-2: COMPARISON OF PDA AND NSC CODE

Break Indent (Figure 3.6-2)	Restraint Indent (Figure 3.6-2)	No. of Bars		Load (kips)		Restraint Deflection (in.)		% of Design Restraint Deflection		Pipe Deflection (in.)	
		PDA	NSC	PDA	NSC	PDA	NSC	PDA	NSC	PDA	NSC
$RC1_J$	RCR1	5	5	803.2	788.3	6.57	7.926	79.93	96.4	17.72	15.58
$RC2_{LL}$	RCR1	5	5	766.4	458.4	14.99	7.495	125.0	62.6	35.83	24.52
$RC3_{LL}$	RCR2	6	6	747.0	639.7	2.27	3.73	27.65	45.35	17.16	20.11
$RC3_{LL}$	RCR2	6	6	796.6	780.3	10.22	10.54	57.8	59.6	41.48	43.0
$RC4_{LL}$	RCR3	5	5	846.0	838.4	7.64	8.05	92.95	97.98	18.87	16.43
$RC4_{LL}$	RCR3	8	8	1319.0	1073.9	5.43	4.62	99.23	76.85	23.38	17.25
$RC4C_V$	RCR3	8	8	1260.7	1275.0	4.49	5.58	80.37	99.89	22.56	18.73
$RC6A_V$	RCR3	8	8	928.5	722.5	1.22	1.77	22.46	31.7	23.68	95.39
$RC7_J$	RCR7	6	6	953.3	80.61	6.28	5.76	76.4	70.12	16.46	21.63
$RC8_{LL}$	RCR6	4	4	599.0	0	8.28	0	112.46	0	26.76	8.39
	RCR7	6	6	895.0	0	8.16	0	110.76	0	29.316	
$RC9C_V$	RCR6	4	4	575.8	520.16	4.16	5.53	50.63	67.33	13.2	14.56
$RC9_{LL}$	RCR8	6	6	830.2	546.8	11.408	6.815	95.29	56.9	36.612	26.24
RC11A	RCR8	6	6	818.3	493.6	10.98	5.99	91.72	50.07	31.404	23.71
RC13	RCR10	4	4	668.4	478.0	5.87	3.66	93.5	58.39	13.37	10.44
RC16	RCR11	4	8	687.4	518.4	6.59	4.38	105.0	69.86	15.37	10.22
$RC14C_V$	RCR20	8	8	285.0	309.6	2.83	5.88	46.3	95.92	15.45	13.96
$RC14_{LL}$	RCR20	8	8	116.3	129.9	0.96	3.36	10.5	37.1	22.13	23.56

TABLE 3.6B-3: DELETED

TABLE 3.6B-3A: DELETED

TABLE 3.6B-3B: DELETED

TABLE 3.6B-4: SUMMARY OF PIPE STRESSES IN PORTION OF MAIN STEAM PIPING BETWEEN PRIMARY CONTAINMENT ISOLATION VALVE

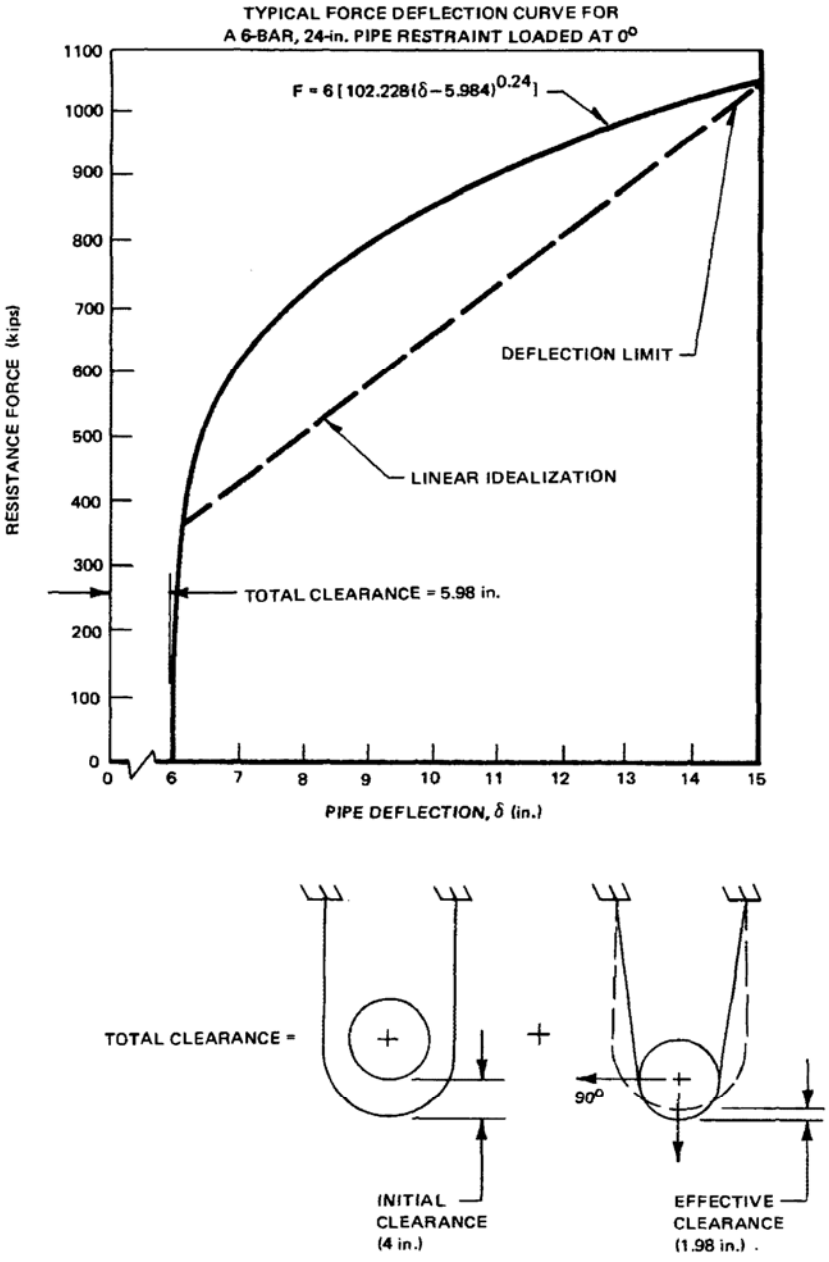
Main Steam Lines - A and D

Node[1]	Eq 10 Stress[2] (psi)	Eq 12 Stress[2] (psi)	Eq 13 Stress[2] (psi)	Cumulative[2,4] Usage Factor	2.4 S_m[3] (psi)	3.0 S_m[3] (psi)
26	22223	3830	15882	0.000	42480	53100
27	19889	1537	15785	0.000	42480	53100
28	34409	635	29772	0.006	42480	53100

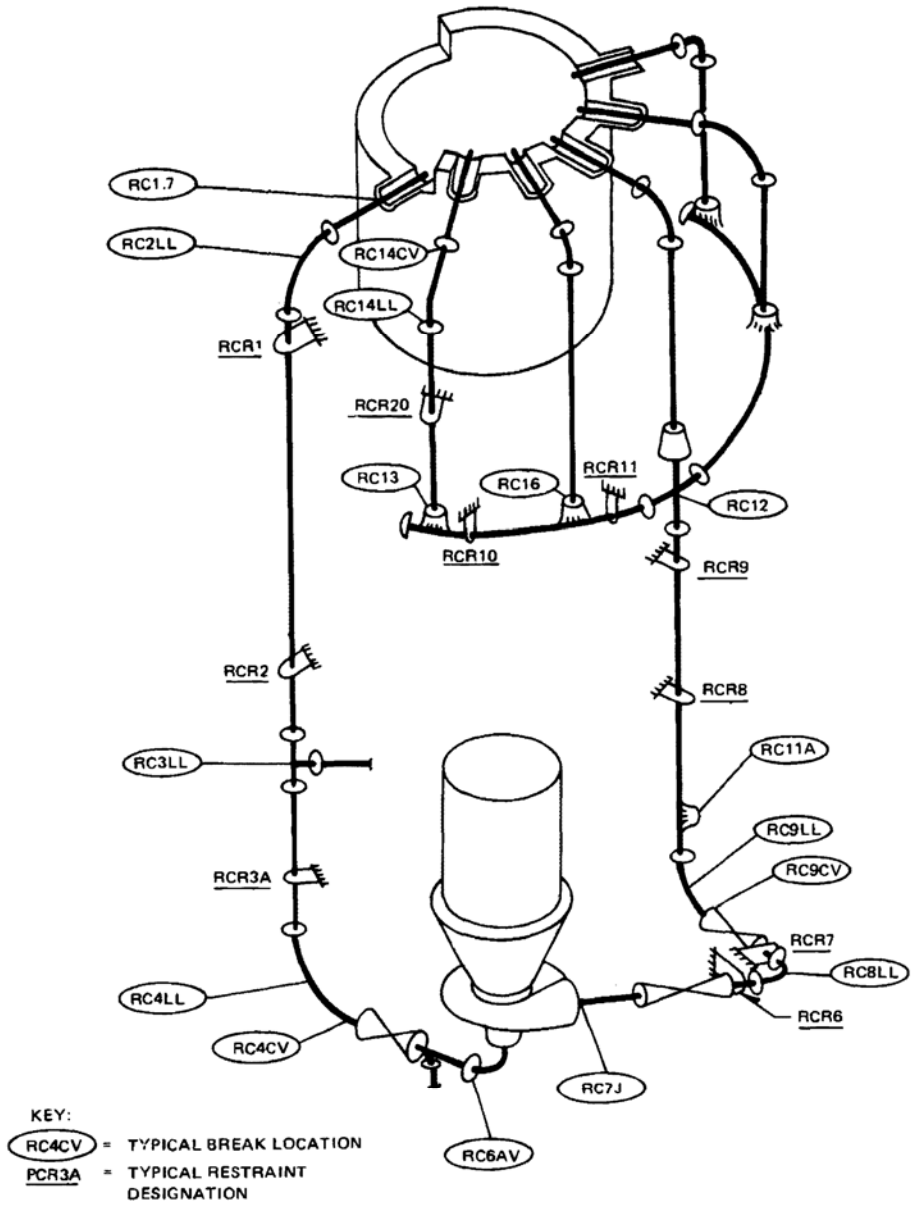
Main Steam Lines - B and C

Node[1]	Eq 10 Stress[2] (psi)	Eq 12 Stress[2] (psi)	Eq 13 Stress[2] (psi)	Cumulative[2,4] Usage Factor	2.4 S_m[3] (psi)	3.0 S_m[3] (psi)
28	21457	3721	16073	0.000	42480	53100
29	18956	1449	15793	0.000	42480	53100
30	34030	416	29706	0.006	42480	53100

Notes:
1. See Figures 3.6B-6a and 3.6B-6b for pipe node locations.
2. Equation (10), (12), and (13) stresses, and cumulative usage factors calculated per ASME Section III, Subarticle NB-3653.
3. Design stress intensity values, Sm, selected per ASME Code, Section III, Appendix I.
4. The fatigue usage factor shown in this table is the design fatigue usage factor. For operating fatigue usage factor see Mechanical Standard MS-40.



MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT	TYPICAL RESTRAINT FORCE-DEFLECTION CURVE FIGURE 3.6B-1



MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT	BREAK LOCATIONS AND RESTRAINTS ANALYZED PDA VERIFICATION PROGRAM FIGURE 3.6B-2

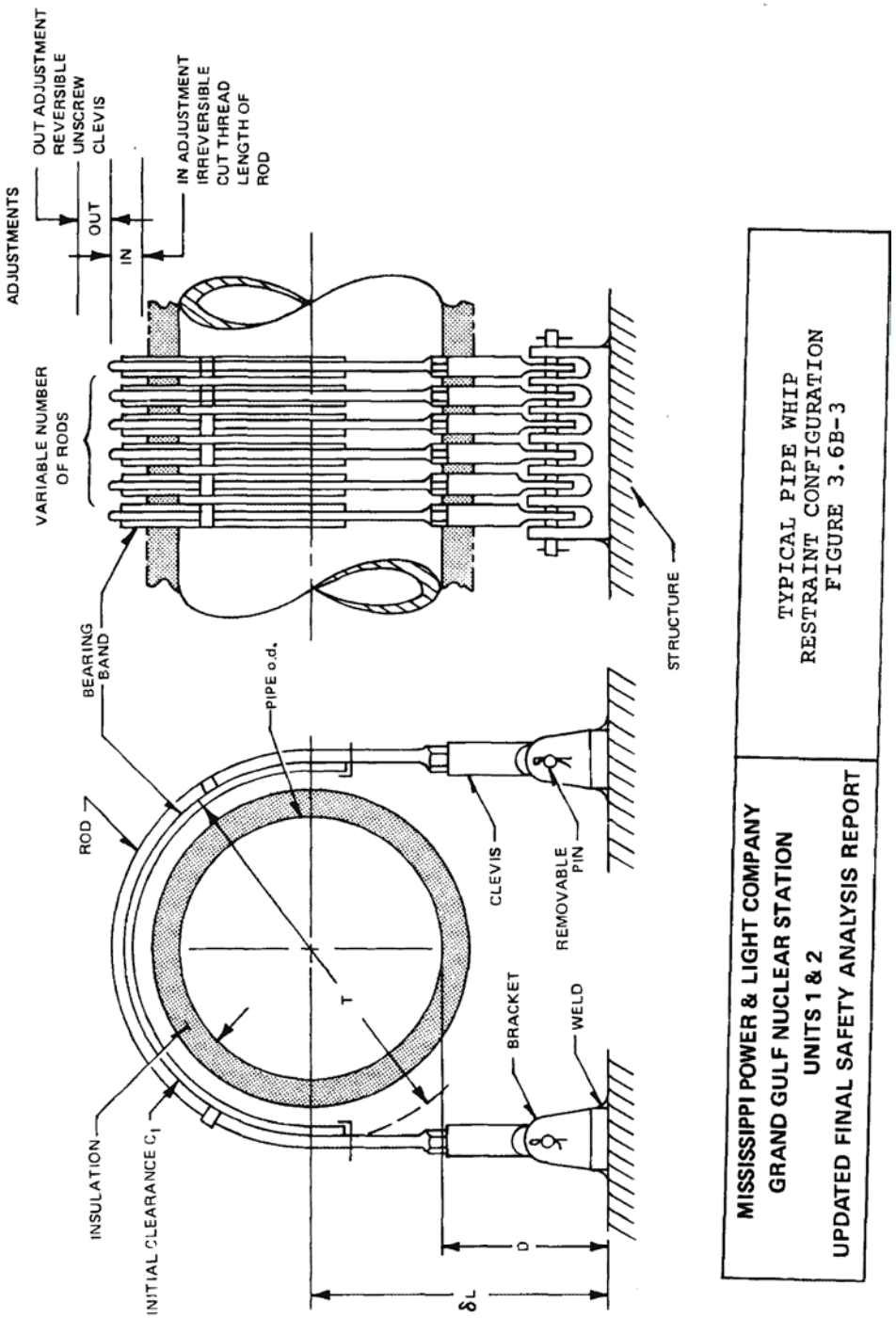

ADJUSTMENTS
OUT ADJUSTMENT REVERSIBLE UNSCREW CLEVIS
IN ADJUSTMENT IRREVERSIBLE CUT THREAD LENGTH OF ROD
OUT
IN
VARIABLE NUMBER OF RODS
BEARING BAND
PIPE o.d.
ROD
INSULATION
INITIAL CLEARANCE C_1
T
CLEVIS
REMOVABLE PIN
BRACKET
WELD
STRUCTURE
D
δL
MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT
TYPICAL PIPE WHIP
RESTRAINT CONFIGURATION
FIGURE 3.6B-3

Figure 3.6B-4

Deleted

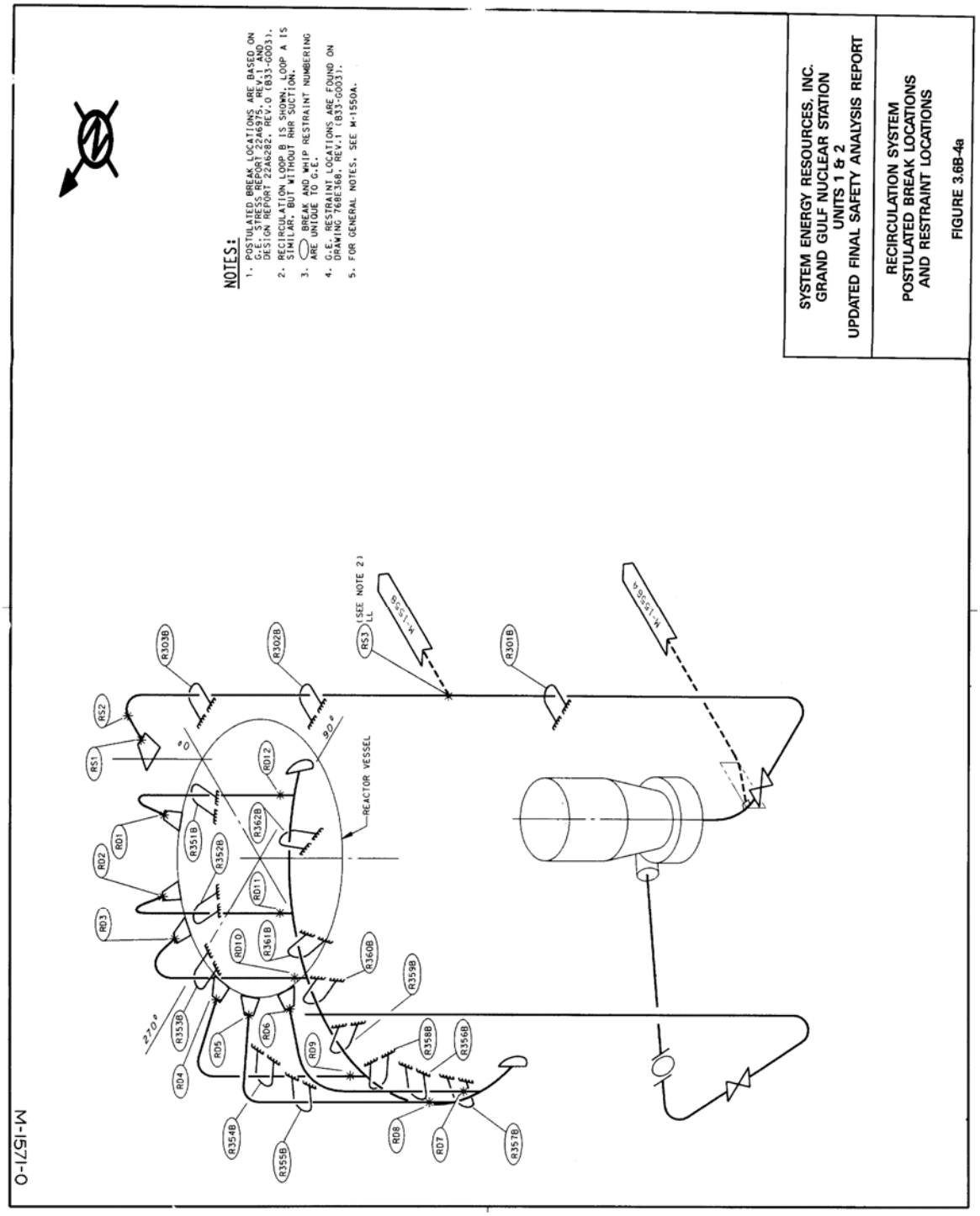

NOTES:
1. POSTULATED BREAK LOCATIONS ARE BASED ON G.E. STRESS REPORT 22A6975, REV.1 AND DESIGN REPORT 22A6282, REV.0 (B33-G003).
2. RECIRCULATION LOOP B IS SHOWN, LOOP A IS SIMILAR, BUT WITHOUT RHR SUCTION.
3. BREAK AND WHIP RESTRAINT NUMBERING ARE UNIQUE TO G.E.
4. G.E. RESTRAINT LOCATIONS ARE FOUND ON DRAWING 76E1368, REV.1 (B33-G003).
5. FOR GENERAL NOTES, SEE M-1550A.
SYSTEM ENERGY RESOURCES, INC.
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT
RECIRCULATION SYSTEM
POSTULATED BREAK LOCATIONS
AND RESTRAINT LOCATIONS
FIGURE 3.6B-4a
M-1571-0
M-1550A
M-1551B
(SEE NOTE 2)
REACTOR VESSEL
0°
90°
270°
R303B
R302B
RS3
RS2
RS1
R301B
RD12
R362B
R351B
R352B
RD1
RD2
RD11
RD3
RD10
R361B
R360B
R359B
R353B
RD5
RD6
RD9
R358B
R356B
R354B
R355B
RD8
RD7
R357B

Figure 3.6B-4B

Deleted

Figure 3.6B-5

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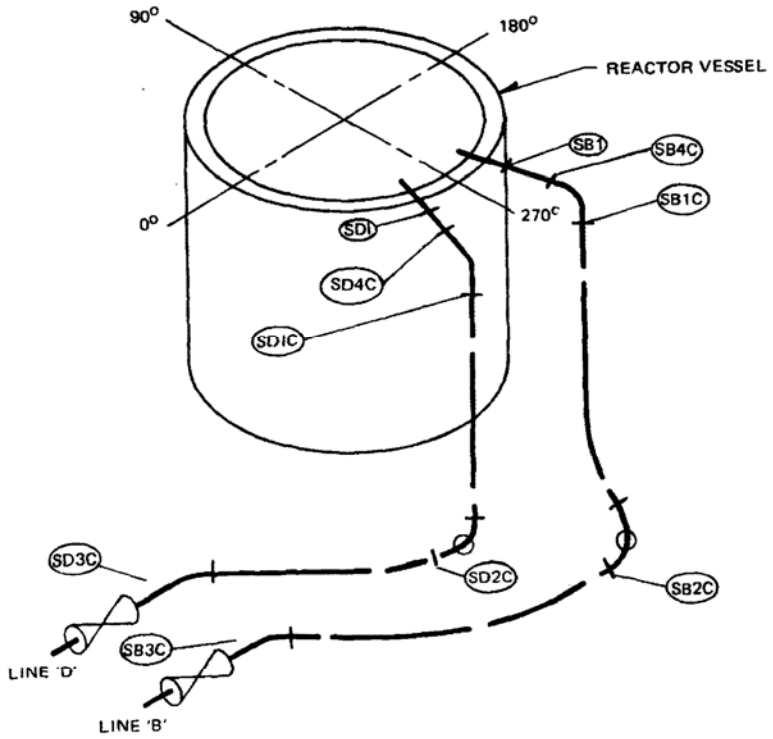
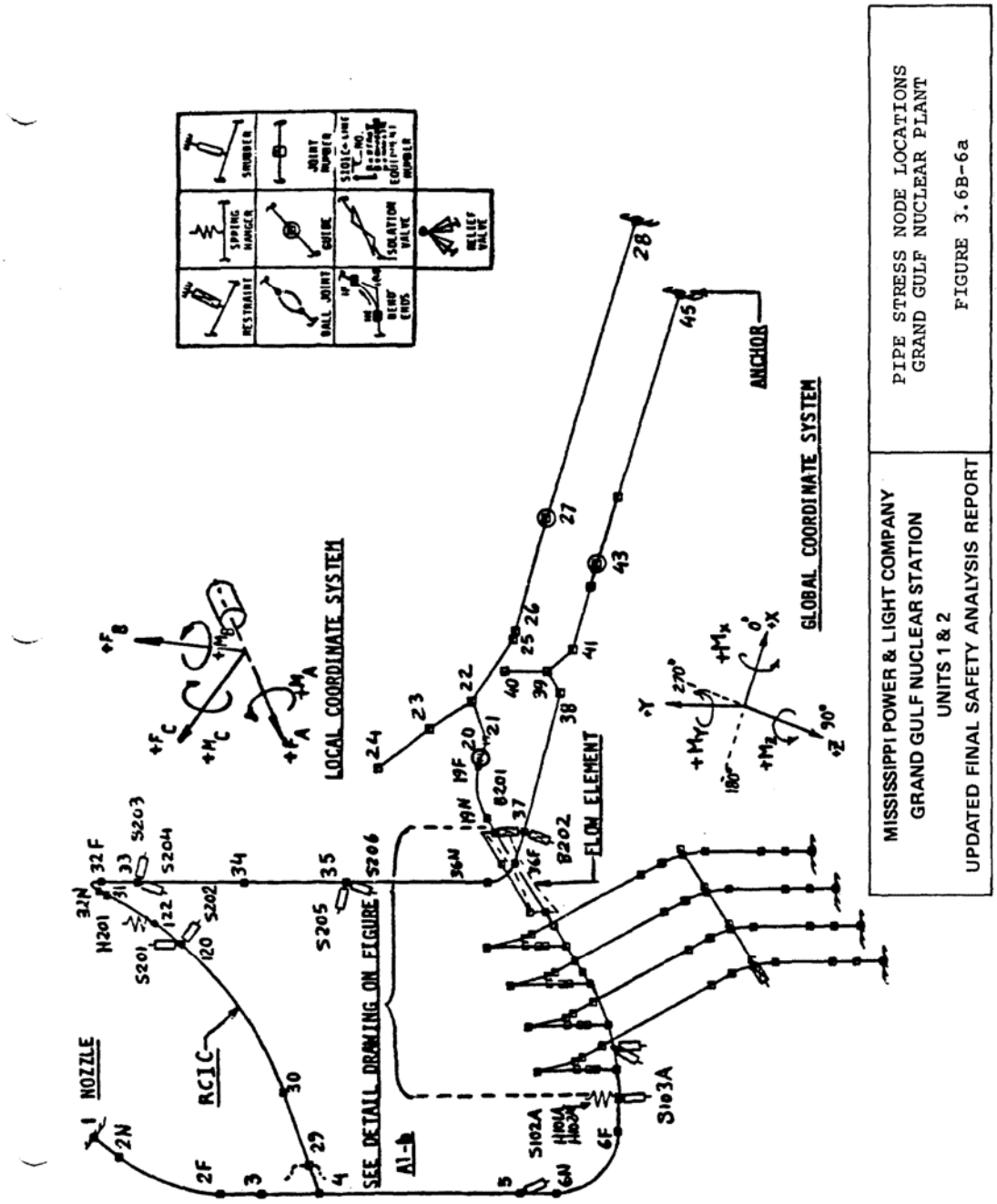


STEAM LINES 'B' AND 'D' SHOWN
(LINE 'A' IS A MIRROR IMAGE OF LINE 'D' EXCEPT FOR RCIC CONNECTION)
(LINE 'C' IS A MIRROR IMAGE OF LINE 'B')
(SB1) = CIRCUMFERENTIAL BREAK
REFER TO FUGURE 3.6A–1A AND 1B FOR RESTRAINT LOCATIONS

MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT	MAIN STEAM LINE POSTULATED BREAK LOCATIONS AND RESTRAINT LOCATIONS FIGURE 3.6B-5a

Figure 3.6B-5B

Deleted


NOZZLE
RCIC
SEE DETAIL DRAWING ON FIGURE A1-b
LOCAL COORDINATE SYSTEM
FLOW ELEMENT
GLOBAL COORDINATE SYSTEM
ANCHOR
RESTRAINT
SPRING HANGER
BALL JOINT
GUIDE
SNUBBER
ISOLATION VALVE
RELIEF VALVE
MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT
PIPE STRESS NODE LOCATIONS
GRAND GULF NUCLEAR PLANT
FIGURE 3.6B-6a

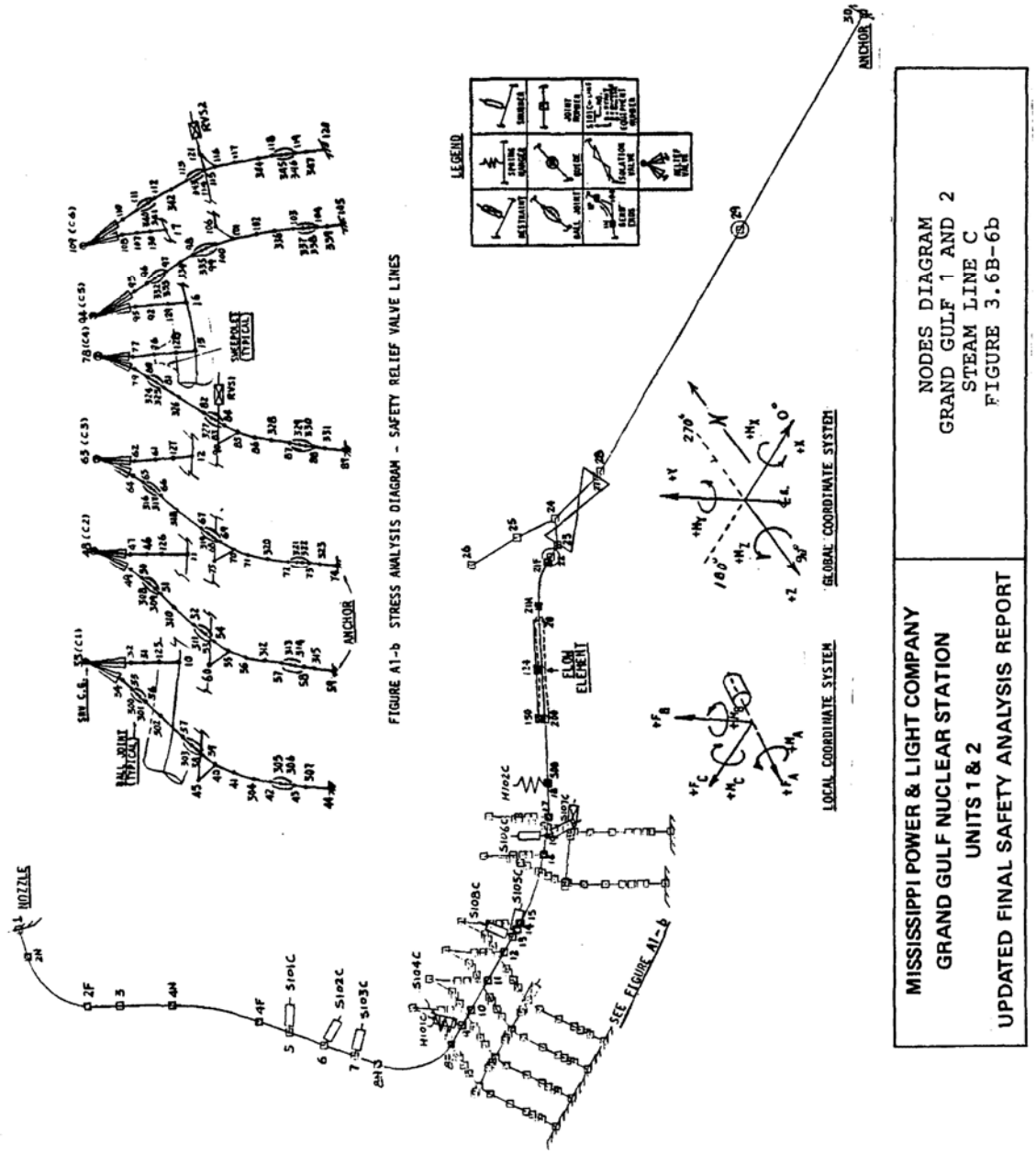


FIGURE A1-b STRESS ANALYSIS DIAGRAM - SAFETY RELIEF VALVE LINES
LEGEND
NOZZLE
ANCHOR
SEE FIGURE A1-b
LOCAL COORDINATE SYSTEM
GLOBAL COORDINATE SYSTEM
MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT
NODES DIAGRAM
GRAND GULF 1 AND 2
STEAM LINE C
FIGURE 3.6B-6b

3.7 SEISMIC DESIGN

All plant structures, systems, and components are separated into the following seismic classifications:

a. Plant structures

   1. Category I structures

   2. Non-Category I structures

   3. Non-Category I structures which have been seismically analyzed to prevent gross collapse during an SSE.

b. Plant systems and components

   1. Seismic Category I systems and components

   2. Seismically analyzed systems and components

   3. Nonseismic systems and components

Seismic classifications of the structures, systems, and components are included in subsection 3.2.1.

Since all structures, systems, and components related to plant safety are required to have the capability to withstand potential earthquakes, each of these structures or systems have been designed to seismic Category I. Non-safety-related systems, and components whose continued function is not required to shut down the plant or mitigate the consequences of a LOCA, but whose failure during an SSE could affect safety-related equipment, were seismically analyzed and supported, as necessary, to prevent such collapse on safety-related equipment.

For each Category I structure, a complete seismic analysis was performed using either a lumped mass or finite element approach. Once the structural mathematical models are made by taking all important dynamic degrees of freedom into consideration and with the equivalent soil spring constants evaluated, the dynamic properties of the soil-structure interactions were computed. By the use of ground movement data fed into the soil-structure interacting dynamic system, responses at various points in the structure were determined.

To ensure the structural adequacy and overall structural stability of the soil-structure interacted vibrating system, square-root-of-the-sum-of-the-squares (SRSS) response values of inertia forces, member forces, and soil spring restoring forces were evaluated by the response spectra method.

For structures whose floors support or hang seismic Category I systems and components, time-history methods were used to provide the appropriate floor response spectra for those secondary dynamic systems.

Generally seismic Category I systems and components are designed or test qualified to withstand events of five OBE followed by one SSE. To ensure that the system and components were designed for these seismic events without compromise to their structural integrities and functional capabilities, analysis or test was performed.

All non-Category I structures have been designed using the Uniform Building Code, 1970 edition (Reference 7). Modified or new non-Category I structures can be designed based on the latest edition of Uniform Building Code. However, non-Category I structures whose failure could endanger Category I structures were designed to withstand an SSE.

3.7.1 Seismic Input

Geologic and seismologic surveys of the site were conducted to establish two design earthquakes with different intensities of ground motion. These are the operating basis earthquake (OBE) and the safe shutdown earthquake (SSE). The OBE is postulated to be the earthquake which could reasonably be expected to affect the plant site during the operating life of the plant. It is that earthquake which produces the vibratory ground motion for which the seismic Category I structures, systems, and components, and the seismically analyzed systems and components of the plant, which are necessary for continued operation without undue risk to the health and safety of the public, are designed to remain functional.

The SSE represents the strongest vibratory ground motion that could occur at the site. It is that earthquake for which these features (as mentioned for OBE) are designed to remain functional.

These seismic Category I structures, systems, and components, and the seismically analyzed systems and components of the plant are necessary to assure: (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 offsite exposures comparable to the guideline exposure of 10 CFR 100, 10 CFR 50.67 and 10 CFR 50 GDC 19.

The design earthquakes, OBE and SSE, for the plant are specified by OBE and SSE design response spectra. These criteria are based upon the plant site geologic investigations and seismologic recommendations as described in Section 2.5.

From the history of seismic activity adjacent to the plant site, a statistical analysis was employed to define the seismic environment that is postulated to occur for both the OBE and SSE as specified in subsections 3.7.1.1 and 3.7.1.2.

3.7.1.1 Design Responsen Spectra

3.7.1.1.1 Safe Shutdown Earthquake (SSE) Design Response Spectra (Other than NSS Systems)

The SSE design response spectra define the free field vibratory motions of the safe shutdown earthquake for the Grand Gulf site.

The peak ground acceleration of the SSE has been established to be 0.15g based on the vibratory ground motion studies as described in subsection 2.5.2.

The design spectra which were used for plant seismic design were obtained by modifying Newmark's curves (Newmark, 1967, Ref. 2). In developing these spectra, variations in site conditions, foundation properties, and effects of focal and epicentral distance from the site were considered. Figure 3.7-1 shows the design spectra for damping values ranging from 0 to 10 percent of critical for a peak ground acceleration of 0.15g (SSE). Figure 3.7-1 also shows that for 2 percent damping, the acceleration amplification ratio is greater than 1.0 in the frequency range from 6.7 to 33 Hz. For the frequency range of 2.0 to 6.7 Hz, the acceleration amplification ratios are constant for each damping. Table 3.7-1 shows the ratios of these constant spectral accelerations to the ground acceleration for five damping values.

The amplification factors for the Grand Gulf design spectra conform to those stated in Table 2-1 of BC-TOP-4, Rev. 1, including Addendum 1 (Ref. 31).

The design spectra for SSE (Figure 3.7-1) are compared to the design spectra representing the "mean" and "mean-plus-one-standard-deviation" of the amplification factors used in developing Regulatory Guide 1.60 (Reference 14). Figures 3.7-59 through 3.7-64 illustrate the comparison of the response spectra for SSE with the response spectra used in developing Regulatory Guide 1.60 that represent a probability level of 50 percent (mean). The values of the mean of the amplification factors which were used to develop Figures 3.7-59 through 3.7-64 were taken from Table 3 and Table 4 of Reference 1 of Regulatory Guide 1.60. Figures 3.7-65 through 3.7-70 illustrate the comparison of the design spectra for SSE with the design spectra used in developing Regulatory Guide 1.60 that represent a probability level of 84.1 percent (mean-plus-one-standard-deviation). The values of the mean-plus-one-standard-deviation of the amplification factors which were used to develop Figures 3.7-65 through 3.7-70 were taken from Table 6 of Reference 1 of Regulatory Guide 1.60. In addition to the above figures, Table 3.7-16 compares the SSE acceleration amplification factors with the mean and the mean-plus-one-standard-deviation of the amplification factors used in developing Regulatory Guide 1.60.

The design response spectra of Regulatory Guide 1.60 and the damping values of Regulatory Guide 1.61 have been used to generate SSE response loads for the lateral and vertical directions. Figures 3.7-71 through 3.7-81 contain these results for all Category I structures. These response loads can be compared with the Grand Gulf SSE design basis response loads shown in Figures 3.7-28 through 3.7-54, based on the Grand Gulf design spectra.

The frequency range of the primary response modes for seismic Category I structures including percent of total response is shown in Figures 2.5-59b through 2.5-59f and Figures 3.7-68a through 3.7-68e.

3.7.1.1.2 Operating Basis Earthquake (OBE) Design Response Spectra (Other than NSS Systems)

The OBE design response spectra define the free field vibratory motions of the operating basis earthquake for the Grand Gulf site.

The amplification characteristics of the OBE are identical to those for the SSE.

Seismic Category I structures, systems, and components have also been designed to withstand the effects of vibratory motion of at least the operating basis earthquake in combination with other appropriate loads within allowable stress limits of applicable codes. Figure 3.7-2 shows the spectra at various damping ratios for the OBE having a ground peak acceleration of 0.075g. These spectra have also been obtained by modifying Newmark's curves as described in subsection 3.7.1.1.1.

For the vertical direction of seismic motion, the corresponding design earthquake was taken to be two-thirds of that established for the horizontal direction at the site.

Thus, for the plant horizontal design response specified by Figures 3.7-1 and 3.7-2, the acceleration amplification ratios, as shown in Table 3.7-1 are in agreement with those shown in BC-TOP-4, Rev. 1, including Addendum 1.

3.7.1.1.3 Design Response Spectra for NSS Systems

For equipment within the General Electric scope of supply, see Appendix 3A, Regulatory Guide 1.60.

3.7.1.2 Design Time-History

The response spectra generated from the free field time-history are required to envelope the design spectra for a sufficient number of frequencies within the frequency range of interest, (0.2 to 33.0 Hz). Since no recorded earthquake motions could fulfill this requirement, a synthetic time-history motion was generated by modifying an actual earthquake record according to the techniques proposed by Tsai (Ref. 15). The plant design synthesized time-history has been obtained through modification of the 1940 (north-south component) El Centro earthquake for a total duration of 24 seconds with a uniform time increment of 0.01 second. The modification consists of an iterative process whereby the actual earthquake time-history is manipulated to raise or suppress its response spectrum until the design spectrum is satisfactorily enveloped for a specified value of damping. The above technique is described in Section 2.5 of BC-TOP-4, Rev. 1.

The OBE synthetic time-history has been obtained so that this time-history, normalized to 0.075g, generates a 2 percent response spectrum fully enveloping the 2 percent design OBE response spectrum (See Figure 3.7-12). To obtain the SSE synthetic time-history, a scale factor of two was applied to the OBE synthetic time-history. The same accelerogram has been used for vertical motion except that the maximum acceleration has been normalized to 0.10g and 0.05g for SSE and OBE, respectively. The synthesized SSE (normalized to 0.15g) time-history accelerogram is shown in Figure 3.7-3. The 2 percent damping OBE design spectrum was chosen to develop the OBE synthesized time-history since this is the lowest structural (material) damping ratio used for analyses of seismic Category I structures. As a result of soil-structure interaction, the composite modal damping values of the important modes (those modes with relatively high participation factors) of all seismic Category I structures increase to values much higher than 2 percent. Therefore, if the response spectrum of the synthetic time-history falls below the design spectrum for damping values of 1 percent or less, there will be no effect on the design conservatism of Category I structures.

Figures 3.7-4 through 3.7-15 show the comparison of the response spectra generated from the synthetic time-history and the design spectra for 0, 0.5, 1, 2, 3, and 5 percent critical damping, respectively. These spectra are computed at 71 frequencies as shown in Table 3.7-2. These frequencies have been chosen so that most of the increments do not exceed 5 percent within the frequency range of 1 to 5 Hz. The design time-history generated for the plant was utilized as an input to each Category I structural vibration system and represented the free field ground motion.

3.7.1.3 Critical Damping Values

3.7.1.3.1 Seismic Category I Structures, Systems, and Components (Other than NSSS)

Table 3.7-3 summarizes, for seismic Category I structures, systems, and components, the critical damping ratios typically utilized for both OBE and SSE seismic analyses. These values are equal to or less than the damping values in NRC Regulatory Guide 1.61. Use of damping ratios higher than those specified in Regulatory Guide 1.61 for system and component seismic response is made only when justified by special studies or tests.

Subsection 3.7.2.15 describes the manner in which these damping ratios have been employed to analytically evaluate the composite modal damping for the system.

3.7.1.3.2 Soil

For a soil-structure interaction, dynamic system, two types of soil damping are considered: material and geometrical damping. The energy dissipation mechanism for material damping is analogous to structural damping, whereas the geometrical damping results from the radiation of wave energy away from the structure into the earth. Geometrical or radiation damping is usually the dominant type, therefore (for simplicity in analysis), it is sufficient to consider only soil geometrical damping. For a typical soil-structure interacted, lumped parameter system as shown on Figures 3.7-16 and 3.7-17, the equivalent damping coefficients C_x, C_Ψ, C_z, and C_t are formulated according to the equations given in Tables 3.7-4 and 3.7-5 which consider geometrical damping only. The structural element damping ratios along with the soil damping characteristics are utilized to obtain the composite modal damping of the soil-structure interacted system.

3.7.1.3.3 Critical Damping Values For NSS Systems

The damping factors indicated in Table 3.7-6 were used in the response analysis of various structures and systems, and in preparation of floor response spectra used as forcing inputs for piping and equipment analysis or testing. These values are consistent with those given in NRC Regulatory Guide 1.61.

3.7.1.4 Supporting Media for Category I Structures

Except for the diesel generator building, all Category I structures are directly underlain by the Catahoula Formation. This formation extends from approximately El. 93 to minus El. 225 at the site. The material of the Catahoula Formation is an over-consolidated sandy clay which is approximately 320 feet thick. Beneath the Catahoula Formation are various systems of interbedded, over-consolidated sandy clays, shales, fossiliferous limestones, and marl. These materials extend for several thousand feet beneath the site.

A detailed discussion of the plant site geologic and soil investigations and seismologic recommendations is presented in Section 2.5.

The Catahoula Formation is capable of supporting heavy loads without excessive settlement and is the bearing stratum for the major plant structures.

For the seismic analysis of Category I structures, except the diesel generator building, fuel tanks, pipes, and duct bank systems, the building foundations are directly supported by the Catahoula Formation which in turn has been assumed to be a half space.

In the case of the diesel generator building, its foundation is supported by approximately 50 feet of compacted backfill beneath which is the Catahoula Formation. This foundation system has been represented by one soil layer over a half space. Table 3.7-7 lists the physical properties of the foundation materials, while Table 3.7-8 gives those ratios of the embedded depth to the structural dimension. The appropriate modeling of the supporting media are based upon these ratios.

Refer to subsections 2.5.1 and 2.5.4 for details on the properties of the foundation soil.

3.7.2 Seismic System Analysis

This subsection applies only to the design of seismic Category I structures and the reactor pressure vessel. Subsection 3.7.3 applies to all seismic Category I piping systems.

3.7.2.1 Seismic Analysis Methods

3.7.2.1.1 Seismic Analysis Methods (Other than NSSS)

The seismic analysis of Category I structures, systems, and components utilized both the response spectrum and time-history methods of analysis, depending upon the desired design data. These analytical methods apply the modal superposition technique whereby the response to appropriate mode shapes is combined to obtain the total system response. A summary listing of methods used for seismic analysis of Category I structures is given in Table 3.7-9.

Descriptions of the methods used for seismic analysis are given in subsections 3.7.2.1.1 and 3.7.2.1.2. A summary listing of various methods used for seismic analysis of non-Category I structures is given in Table 3.7-10.

3.7.2.1.1.1 Analytical Techniques

The analytical techniques used to compute structural responses are described in Sections 4.0 through 4.2 of Bechtel Topical Report, BC-TOP-4, Rev. 1, including Addendum 1.

3.7.2.1.1.2 Analytical Considerations

3.7.2.1.1.2.1 Rocking and Torsional Effects

Rocking effect due to horizontal ground movement is considered in the soil-structure interaction analysis as described in Appendix D of BC-TOP-4, Rev. 1, including Addendum 1. The torsional effect is considered as described in subsection 3.7.2.11.

3.7.2.1.1.2.2 Hydrodynamic Effects

For a dynamic system in which a liquid such as water is involved, the necessary hydrodynamic effects are taken into consideration in the following manner:

a. For the case of a vibrating system containing liquid, the liquid mass is idealized into two volumes: the upper liquid which, under oscillatory motion, exerts convective force on the containing wall and the lower liquid which produces only an impulsive force on the wall without moving in an oscillatory manner. The mathematical model based on this consideration is described in detail in Reference 29.

b. For the case of a vibrating system containing liquid in which other internal vibrating subcomponents are submerged, the mass of the outer container and the subcomponent are coupled by the liquid in between so that the mass matrix will have an off-diagonal negative coupling term called apparent mass. The necessary formulas used for computing these terms can be found in Reference 30. For the containment building design, the situation of the flooded building inside to El. 208-10 of the refueling floor was subjected to the SSE event. This situation

arises from post-LOCA core recovery considerations. However, water level at this elevation is extremely conservative.

3.7.2.1.1.3 Description of Mathematical Models

The mathematical models used for seismic analysis of the major Category I structures are described in sequence below. The models developed for each Category I structure, as shown on Figures 3.7-18 through 3.7-20, and 3.7-22 through 3.7-24, which yielded the seismic design loads of Figures 3.7-28 through 3.7-54, are based on three-dimensional geometry. The properties of the model corresponding to the North-South, East-West, and Vertical directions are used in the analyses for the three orthogonal axes. In this way, the structural response to each of the three orthogonal components of earthquake input motion is analyzed. These individual component results are then combined according to the methods described in subsection 3.7.2.6.

3.7.2.1.1.3.1 Model for Containment, Interior Structures, and Enclosure Building

Figure 3.7-18 shows the mathematical model for the containment structure. The structure is axisymmetric except the portion of internal structure above the drywell and enclosure building. The seismic analysis has been made in vertical and plant N-S and E-W directions for SSE and OBE by considering soil-structure interaction.

The containment building has been considered as a planar structure with 1 dynamic degree of freedom at each floor level and 2 dynamic degrees of freedom at the foundation level for rocking and translation.

In order to take into account the effect of the enclosure building on the seismic analysis of the containment, the weight of the enclosure building is distributed to mass points 7, 9, and 11 by structural springs. By use of this mathematical model, response spectra at mass points 7, 9, and 11 were generated and used as input to a three-dimensional mathematical model of the enclosure building for final enclosure building seismic response analysis.

The mass of the containment upper pool is modeled in lump mass numbers 16 and 17 as shown on Figure 3.7-18 with the effects of pool sloshing accounted for by mass number 17 and spring K4. This pool is full during normal operation.

The upper pool is separated physically from the containment shell by an air gap, thus permitting relative deformation between the walls. To assure that no building-to-building interaction occurs, a separation of 2 inches along the boundary of the containment building with the auxiliary building is provided, except at nine (9) locations at E1.93 where the auxiliary building walls cantilever over the top of the containment building foundation. At these locations no gap exists, only a joint where the walls are in contact with the top of the slab (as shown on Figures 3.8-77 and 3.8-85) due to the inadvertent omission of the 2-inch gap. An evaluation of the seismic effects of the resulting structure-to-structure interaction shows negligible effects on the seismic design based on the model shown on Figure 3.7-18. Any interaction through the foundation media is not considered.

3.7.2.1.1.3.2 Auxiliary Building

Figures 3.7-19 and 3.7-20 show the mathematical models for the auxiliary building. Also shown on Figure 3.7-21 is a three-dimensional mathematical model for the auxiliary and the control buildings which is used to study the effect of structure-to-structure interaction between the two buildings.

In considering the location of the center of gravity of the three-dimensional model in the E-W direction, torsional effects, if any, due to N-S ground movement input, are studied. Shown also in the mathematical model are springs TR1 to TR6 representing the air gaps between the auxiliary and control buildings which are filled with a compressible material.

Comparison of the floor response spectra, based on this model and Figure 3.7-19 and Figure 3.7-20, indicates that the torsional effect as well as structure-to-structure interaction due to the compressible material are not significant enough to alter the seismic design based on Figures 3.7-19 and 3.7-20.

Since the control building was shown to have an insignificant impact on the response of the auxiliary building, a three-dimensional model of the auxiliary building alone was developed as shown in Figure 3.7-21a to further evaluate the effects of building eccentricities on total response. This evaluation is further discussed in subsection 3.7.2.11.

As noted in subsection 3.7.2.1.1.3.1, nine (9) of the 15 walls which cantilever over of the containment foundation were inadvertently placed without the 2-inch seismic gap between the

wall and the top of the foundation. The effects of this contact were evaluated, and the structure-to-structure interaction due to this contact are negligible and do not alter the seismic design based on Figures 3.7-19 and 3.7-20.

3.7.2.1.1.3.3 Control Building

Figure 3.7-22 is the mathematical model for the control building. Seismic design of the building and the generation of floor response spectra are based on this mathematical model.

3.7.2.1.1.3.4 Diesel Generator Building

Based on the mathematical model as shown in Figure 3.7-23, response spectra for the diesel generator building are generated for mass points at the foundation slab and the ventilation equipment floor above for the SSE and OBE.

Soil structure interaction has been considered by evaluating frequency dependent soil impedance functions for a layer of compact soil and the remaining half space underlying the Catahoula Formation.

3.7.2.1.1.3.5 Standby Service Water Cooling Tower Basins

The entire basin is embedded in the soil, and the embedment ratio defined by the embedded depth to the least plan dimension of the structure, is approximately 0.3. The basin has been analyzed by finite element and lumped mass elastic half-space methods.

A discussion of the soil-structure analysis using finite element models is included in subsection 3.7.2.4 (see Figures 3.7-25 and 3.7-26). The lumped mass, elastic half-space mathematical model is shown on Figure 3.7-24. The response from these two methods are enveloped to obtain seismic structural design loadings and floor response spectra: Modal responses were combined using the square-root-of-the-sum-of-the-squares approach. The structural design loads from the lumped mass elastic half-space analysis enveloped those obtained using the peak responses from the finite element approach.

3.7.2.1.1.4 Determination of Composite Modal Damping

Since the damping matrix, in general, and the soil-structure interaction vibration system contain non-zero off-diagonal terms, the exact solution of the response of the system can be obtained

only by directly integrating the coupled equations of motion shown in Equation D-8 in Appendix D of BC-TOP-4, Rev. 1, including Addendum 1. However, by properly determining the composite modal damping for decoupled equations of motion, a method of modal analysis can still be used with good approximation. The detailed technique of evaluating these composite modal damping values is given in Appendix D of BC-TOP-4, Rev. 1, including Addendum 1.

The composite modal damping computations are performed for all Category I structures by means of computer code CE-931 or CE-251 and are used in the final response calculations.

3.7.2.1.2 Seismic Analysis Methods for NSS Systems

Analysis of seismic Category I NSS systems and components is accomplished, where applicable, using the response spectrum or time-history approach. Either approach utilizes the natural period, mode shapes, and appropriate damping factors of the particular system. Certain pieces of equipment having very high natural frequencies may be analyzed statically. In some cases, dynamic testing of equipment may be used for seismic qualification.

The time-history analyses involve the solution of the equations of the dynamic equilibrium (subsection 3.7.2.1.2.1) by means of the methods discussed in subsection 3.7.2.1.2.2. In this case, the duration of motion is of sufficient length to ensure that the maximum values of response have been obtained.

A response spectrum analysis involves the solution of the equations of motion (subsection 3.7.2.1.2.1) by the method discussed in subsection 3.7.2.1.2.3.

3.7.2.1.2.1 The Equations of Dynamic Equilibrium

Assuming velocity proportional damping, the dynamic equilibrium equations for a lumped mass, distributed stiffness system are expressed in matrix form as:

$$[M]\{\ddot{u}(t)\} + [C]\{\dot{u}(t)\} + [K]\{u(t)\} = \{P(t)\} \qquad (3.7\text{-}1)$$

where:

$u(t)$ = time dependent displacement of nonsupport points relative to the supports

$\dot{u}(t)$ = time dependent velocity of nonsupport points relative to the supports

$\ddot{u}(t)$ = time dependent acceleration of nonsupport points relative to the supports

[M] = diagonal matrix of lumped masses

[C] = damping matrix

[K] = stiffness matrix

P(t) = time dependent inertial forces acting at non-support points

3.7.2.1.2.2 Solution of the Equations of Motion by Mode Superposition

The first technique used for the solution of the equations of motion is the method of Mode Superposition.

The set of homogenous equations represented by the undamped free vibration of the system is

$$[M]\{\ddot{u}(t)\} + [K]\{u(t)\} = 0 \qquad (3.7\text{-}2)$$

Since the free oscillations are assumed to be harmonic, the displacements can be written as

$$\{u(t)\} = \{\varnothing\} e^{iwt} \qquad (3.7\text{-}3)$$

where:

$\{\varnothing\}$ = column matrix of the amplitude of displacements, u

w = circular frequency of oscillation

t = time

Substituting equation 3.7-3 and its derivatives in equation 3.7-2 and noting that e^{iwt} is not necessarily zero for all values of wt yields

$$-w^2[M] + [K]\{\varnothing\} = \{0\} \qquad (3.7\text{-}4)$$

Equation 3.7-4 is the classical algebraic eigenvalue problem wherein the eigenvalues are the frequencies of vibrations w_i and the eigenvectors are the mode shapes, Ø i.

3.7.2.1.2.3 Analysis by Response Spectrum

As an alternative to the step-by-step mode superposition method described in subsection 3.7.2.1.2.2, the response spectrum method may be used. The response spectrum method is based on the fact that the modal responses can be expressed as a set of integral equations, rather than a set of differential equations. The advantage of this form of solution is that for a given ground motion the only variables under the integral are the damping factor and the frequency. Thus, for a specified damping factor, it is possible to construct a curve which gives a maximum value of the integral as a function of frequency. This curve is called a response spectrum for the particular input motion and the specified damping factor. The integral has units of velocity, consequently the maximum of the integral is called the spectral velocity.

Using the calculated natural frequencies of vibration of the system, the maximum values of the modal responses are determined directly from the appropriate response spectrum. The modal maxima are then combined as discussed in subsection 3.7.3.7.

3.7.2.1.2.4 Support Displacements in Multi-Supported Structures

The preceding sections have discussed analysis procedures for forces and displacements induced by time-dependent support accelerations. In a multi-supported structure, there are, in addition, time-dependent support displacements which produce additional displacements at nonsupport points and pseudo-static forces at both support and nonsupport points.

The governing equation of motion of a structural system which is supported at more than one point and has different excitations applied at each may be expressed in the following concise matrix form:

$$\begin{bmatrix} M_a & 0 \\ 0 & M_s \end{bmatrix} \begin{Bmatrix} \ddot{U}_a \\ \ddot{U}_s \end{Bmatrix} + \begin{bmatrix} C_{aa} & C_{as} \\ C_{as} & C_{ss} \end{bmatrix} \begin{Bmatrix} \dot{U}_a \\ \dot{U}_s \end{Bmatrix} + \begin{bmatrix} K_{aa} & K_{as} \\ K_{as} & K_{ss} \end{bmatrix} \begin{Bmatrix} U_a \\ U_s \end{Bmatrix} = \begin{Bmatrix} F_a \\ F_s \end{Bmatrix} \quad (3.7\text{-}5)$$

where U_a represents the displacement of the active (unsupported) degrees of freedom, $\bar{U}_s$ denotes the specified displacements of support points, M_a and M_s are the lumped diagonal mass matrices associated with the active degrees of freedom, and the support points, respectively, C_{aa} and K_{aa}, are the damping matrix and elastic stiffness matrix expressing the forces developed in the active degrees of freedom due to the motion of the active degrees of freedom, C_{aa} and K_{aa} are the support forces due to unit velocities and displacement of the supports. C_{as} and K_{as} are the damping and stiffness matrices denoting the coupling forces developed in the active degrees of freedom by the motion of the supports, and vice versa, F_a is the prescribed external time-dependent force applied on the active degrees of freedom, and F_s is the reaction force at the system support points. Total differentiation with respect to time is denoted by ($\dot{}$). Also, the contributions of the fixed degrees of freedom have been removed in the above equation. The procedure utilized to construct the damping matrix is discussed in subsection 3.7.2.15. The mass matrix and elastic stiffness matrix are formulated by using standard procedures.
Equation 3.7-5 can be separated into two sets of equations. The first set of equations can be written as:

$$[M_a]\{\ddot{U}_a\} + [C_{aa}]\{\dot{U}_a\} + [K_{aa}]\{U_a\} + [C_{as}]\{\dot{\bar{U}}_s\} + [K_{as}]\{\bar{U}_s\} = \{F_a\}$$

and the second set as

$$[M_s]\{\ddot{\bar{U}}_s\} + [C_{ss}]\{\dot{\bar{U}}_s\} + [K_{ss}]\{\bar{U}_s\} + [C_{as}]\{\dot{U}_a\} + [K_{as}]\{U_a\} = \{F_s\}$$

The time-wise solution of equation 3.7-6a can be obtained easily by using the standard normal mode solution technique. After obtaining the displacement response of the active degrees of freedom, U_a, equation 3.7-6b then can be used to solve the support point reaction forces, F_s.

Modal damping is used to determine the solutions of the uncoupled form of equation 3.7-6a. The procedure is identical to that described in subsection 3.7.2.15. C_{aa} and K_{aa} are the damping matrix and elastic stiffness matrix, respectively, expressing the forces developed in the active degrees of freedom.

3.7.2.1.2.5 Dynamic Analysis of Category I Structures, Systems, and Components

Time-history techniques and the response spectrum technique are used as applicable for the dynamic analysis of Category I structures, systems, and components which are sensitive to dynamic seismic events.

3.7.2.1.2.5.1 Dynamic Analysis of Piping Systems

Each pipe line is idealized as a mathematical model consisting of lumped masses connected by elastic members. The stiffness matrix for the piping system is determined using the elastic properties of the pipe. This includes the effects of torsional, bending, shear, and axial deformations as well as change in stiffness due to curved members. Next, the mode shapes and the undamped natural frequencies are obtained. The dynamic response of the system is calculated by using the response spectrum method of analysis. When the piping system is being anchored and supported at points with different excitations, the response spectrum analysis is performed using the envelope response spectrum of all attachment points. Alternately, the multiple excitation analyses methods may be used where acceleration time-histories or response spectra are applied to all piping system attachment points.

The relative displacement between anchors is determined from the dynamic analysis of the structures. The results of the relative anchor point displacements are used for a static analysis to determine the additional stresses due to relative anchor point displacements.

3.7.2.1.2.5.2 Dynamic Analysis of Equipment

Equipment is idealized as a mathematical model consisting of lumped masses connected by elastic members or springs. Analytical results for some selected large seismic Category I equipment are given in Table 3.9-2.

When the equipment is anchored and supported at points with different excitation, the response spectrum analysis is performed using the envelope response spectrum of all attached points. This is consistent with the SRP. Alternately, the multiple excitation analysis methods may be used where acceleration time-histories or response spectra are applied to all equipment attachment points. The time-histories method provides realistic results and is accepted by the SRP. The response spectrum envelope method

provides the most conservative results. The multiple excitation response spectrum method provides results bounded by the envelope response spectrum method on the high side and by the time-histories method on the low side. This justifies the conservatism of the multiple excitation response spectrum method.

Subsection 3.7.2.1.2.5.2.1 provides details of how the relative displacements between supports are generated and used in the static analysis of systems with differential support motion. The support displacements are calculated on a structural modal basis. The mode of displacements are scaled up to account for the peak broadening effects, i.e., they are scaled up to account for any potential variations in structural frequencies due to the uncertainties associated with the soil and structure mechanical properties and the uncertainties introduced by simplifying assumptions inherent to modeling. The scaled modal displacements then form the basis for the static secondary stress calculations, as explained in subsection 3.7.2.1.2.5.2.1.

The relative displacement between supports is determined from the dynamic analysis of the structure. The relative support point displacements are used for a static analysis to determine the additional stresses due to support displacements. Further details are given in subsection 3.7.2.1.2.5.2.1.

3.7.2.1.2.5.2.1 Differential Seismic Movement of Interconnected Components

The procedure for considering differential displacements for equipment anchored and supported at points with different displacement excitation is as follows.

The relative displacements between the supporting points induce additional stresses in the equipment supported at these points. These stresses can be evaluated by performing a static analysis where each of the supporting points is displaced a prescribed amount. From the dynamic analysis of the complete structure, the time history of displacement at each supporting point is available. These displacements are used to calculate stresses by determining the peak modal responses. The stresses thus obtained for each natural mode are then superposed for all modal displacements of the structure by the SRSS method.

In the static calculation of the stresses due to relative displacements in the response spectrum method, the maximum value of the modal displacement is used. Therefore, the mathematical

model of the equipment is subjected to a maximum displacement at its supporting points obtained from the modal displacements. This procedure is repeated for the significant modes (modes contributing most to the total displacement response at the supporting point) of the structure. The total stresses due to relative displacement is obtained by combining the modal results using the square-root-of-the-sum-of-the-squares (SRSS) method. Since the maximum displacement for different modes does not occur at the same time, the SRSS method is a realistic and practical method.

When a component is covered by the ASME Boiler and Pressure Vessel Code, the stresses due to relative displacement as obtained above are treated as secondary stresses.

3.7.2.1.2.5.3 Dynamic Analysis of Structures

The analysis techniques used for Category I structures are described in subsection 3.7.2.1.

3.7.2.1.2.6 Seismic Qualification by Testing

For certain seismic Category I equipment and components where dynamic testing is necessary to ensure functional integrity, test performance data and results will reflect the following:

a. Performance data of equipment which, under the specified conditions, has been subjected to dynamic loads equal to or greater than those to be experienced under the specified seismic conditions.

b. Test data from previously tested comparable equipment which, under similar conditions, has been subjected to dynamic loads equal to or greater than those specified.

Actual testing of equipment is in accordance with one of the methods described in Sections 3.9 and 3.10.

Alternate test procedures that satisfy the requirements of these criteria will be allowed, subject to review by the responsible engineer.

3.7.2.2 Natural Frequencies and Response Loads

The first few significant natural frequencies for all Category I structures are presented in Table 3.7-11. In addition, the response spectra at various critical locations within the Category I structures are generated as shown in the above table to define the seismic environment at the equipment location.

Response loads (for Category I structures) which were determined by seismic system analyses are shown on Figures 3.7-28 through 3.7-54.

The response spectra are generated for all Category I structures to define the seismic environment for the subsystem analyses.

3.7.2.3 Procedure Used for Modeling

3.7.2.3.1 Procedure Used for Modeling Systems Other than the Reactor Vessel

Two basic procedures have been used to model the seismic Category I dynamic system:

a. In the first procedure, the system has been represented by an equivalent beam and a system of lumped masses located at the elevation of mass concentrations, such as floors. For a structure such as the containment shell having continuous mass distributions, a sufficient number of mass points are chosen so that the vibration modes of interest can be adequately defined. In general, to adequately define the frequency of the highest mode to be used in the analysis, the minimum number of lumped masses is twice this mode number. In the first procedure, soil is represented by frequency independent foundation impedances.

b. In the second procedure, the structure or system is represented as a two- or three-dimensional mathematical finite element model using any combinations of beams, plates, and solid elements. The extent of details in the

mathematical models is based upon the complexity of the actual structures and the information required from the analysis.

In both procedures, the mass, stiffness, and damping matrices are formulated, and the system response to seismic excitation is obtained through the use of appropriate computer programs.

In applying either modeling procedure, primary consideration was given to adequately predict the dynamic response of the system. This can only be accomplished by ensuring that the type and number of lumped parameters used for idealization will adequately simulate the dynamic characteristics of the overall system. Other considerations made for proper idealization include how to obtain internal member forces and stresses and how to determine relative deflections for clearance purposes between major internal components.

Typical mathematical models at a soil-structure lumped mass system for horizontal and vertical input motions are shown on Figures 3.7-16 and 3.7-17.

The equipment, component, or system has been lumped into the supporting structure mass if its estimated weight is less than one-tenth that of the supporting mass. This equipment was subsequently analyzed by applying the response spectrum generated at the support point.

Other considerations applied to modeling techniques are discussed in subsection 3.7.2.1.

3.7.2.3.2 Modeling of Reactor Pressure Vessel and Internals

The seismic loads on the reactor pressure vessel (RPV) and internals are based on a dynamic analysis of an entire RPV-building complex with the appropriate forcing function supplied at ground level. For this analysis, the model shown on Figure 3.7-27 and the mathematical model of the building are coupled together.

This mathematical model consists of lumped masses connected by elastic (linear) members. Using the elastic properties of the structural components, the stiffness properties of the model are determined. This includes the effects of both bending and shear.

In order to facilitate hydrodynamic mass calculations, several mass points (fuel, shroud, vessel) are selected at the same elevation. The various lengths of control rod drive housings are grouped into the two representative lengths shown. These lengths represent the longest and shortest housings in order to adequately represent the full range of frequency response of the housings. The high fundamental natural frequencies of the CRD housings result in very small seismic loads. Furthermore, the small frequency differences between the various housings due to the length differences result in negligible differences in dynamic response. Hence, the modeling of intermediate length members becomes unnecessary. Not included in the mathematical model are light components such as jet pumps, in-core guide tubes and housings, sparger, and their supply headers. This is done to reduce the complexity of the dynamic model. If the seismic responses of these components are needed, they can be determined after the system response has been found.

The presence of a fluid and other structural components (e.g., fuel within the RPV) introduces a dynamic coupling effect. Dynamic effects of water enclosed by the RPV are accounted for by introduction of a hydrodynamic mass matrix, which serves to link the acceleration terms of the equations of motion of

points at the same elevation in concentric cylinders with a fluid entrapped in the annulus. The details of the hydrodynamic mass derivation are given in Reference 6. The seismic model of the RPV and internals has two horizontal coordinates for each mass point considered in the analysis. The remaining translational coordinate (vertical) is excluded because the vertical frequencies of the RPV and internals are well above the significant horizontal frequencies. Furthermore, all support structures, building and containment walls have a common centerline, and hence, the coupling effects are negligible. A separate vertical analysis is performed. Dynamic loads due to vertical motion are added to or subtracted from the static weight of components, whichever is the more conservative. The two rotational coordinates about each node point are excluded because the moment contribution of rotary inertia from surrounding nodes is negligible. Since all deflections are assumed to be within the elastic range, the rigidity of some components may be accounted for by equivalent linear springs.

The shroud support plate is loaded in its own plane during a seismic event and hence is extremely stiff and therefore may be modeled as a rigid link in the translational direction. The shroud support legs and the local flexibilities of the vessel and shroud contribute to the rotational flexibilities and are modeled as an equivalent torsional spring.

The comparison between the calculated maximum seismic loads and the allowable loads in the RPV and internals is given in Table 3.9-2b.

3.7.2.4 Soil-Structure Interaction

For structures that are founded on the ground surface or have shallow embedment (defined as embedded depth less than 15 percent of the least base width of a structure), the methodology of the lumped mass representation and lumped parameter approach is used to represent the soil-structure interaction (Ref. 9). This method is discussed below and in a greater detail in Appendix D of BC-TOP-4, Rev. 1, including Addendum 1 (Ref. 31).

The lumped representation is derived from analyzing the model composed of a rigid mat resting on the surface of an elastic half-space. The resultant foundation compliance is frequency dependent, but can be approximated by a constant compliance for engineering applications (Refs. 9 and 16). The foundation compliance is a function of the mass and dimensions of the foundation mat and the properties of the foundation medium. As a mechanical analog, this compliance function can be represented by springs and dampers; the latter represents the radiation effect of the wave energy and usually predominates over the material damping of the foundation medium. Figure 3.7-6 shows schematically a lumped mass model of the structure and foundation, with foundation springs k_x and k_ψ, and rotational dampers c_x and c_ψ, representing the foundation compliance subjected to horizontal seismic excitation. The foundation is represented by k_z and c_z for vertical motion, and k_t and c_t for torsion.

Tables 3.7-4 and 3.7-5 show the expressions for equivalent spring constants and damping coefficients for both circular and rectangular base mats. Both the equivalent spring constants and damping should be used simultaneously in conjunction with the mass of the base mat. Also, the material damping of soils is neglected in the lumped representation.

Also shown in Table 3.7-8 are the building embedment ratios based on different modeling techniques described in subsection 3.7.2.3.

With the foundation compliance specified, the structure-foundation system is formulated by coupling the fixed base structure with the foundation through the base mat. The method is described in Appendix D of BC-TOP-4, Rev. 1, including Addendum 1, with the structure represented by its fixed base normal modes. The interaction system becomes a coupled system due to the presence of the damping matrix. However, it is usually sufficient to represent this coupled system by normal modes. Appendix D of BC-TOP-4, Rev. 1, including Addendum 1 (Ref. 31) shows one technique to determine the composite modal damping of the interaction system. This is accomplished by requiring that at certain locations of the structure, the dynamic amplification functions of both the coupled and the uncoupled systems match each other at the natural frequencies. However, to be conservative for the analysis, any of the computed composite dampings which exceed 10 percent of critical will be replaced by a maximum of 10 percent of critical except for those modes that are clearly associated with rigid body translation or rotation of the structure (Refs. 17 and 18).

The seismic Category I structures that have shallow embedment are:

Containment
Auxiliary
Control
Diesel generator

For structures that are deeply embedded, the finite element method of analysis is used to represent the soil-structure interaction.

The Category I structures that are treated as either deeply embedded or completely embedded are:

a. Standby service water cooling towers and basins (deeply embedded)

b. Diesel generator fuel tanks (completely embedded)

For structures with considerable embedment, the soil-structure interaction, as in the standby service water cooling towers and basins, is studied by a dynamic analysis using the finite element

method which simulates the nonlinear behavior of soils subjected to earthquake loading. The method, which is known as the equivalent linear method, involves a linear dynamic analysis combined with an iterative technique which ensures that the stiffness and damping characteristics of the subsoil are compatible, at all points, with the strain level to which the soil is excited. The validity of the equivalent linear method has been confirmed by successful application of one-dimensional earthquake amplification problems as discussed in References 10, 11, and 12.

The soil-structure interaction effect was initially evaluated using the LUSH/SHAKE computer programs and was later reevaluated using the FLUSH computer program (an improved version of the LUSH program) to incorporate certain refinements in the finite element model.

In the FLUSH model, the soil-structure system is approximated by a plane-strain finite element model which extends to a depth of 200 feet below grade. The soil is represented by 40 layers of elements. A typical model contains 242 elements. These elements are generally wider than they are deep since response of the model is dominated by vertically propagating waves. The grid size chosen is small enough to accommodate, as far as practical, components of motion which have frequencies lower than 33 Hz. Parametric studies of soil profiles show that further refinement of the finite element grid sizes to accommodate higher frequencies does not have an appreciable effect on response of the structure. The FLUSH finite element model used for the study of soil-structure interaction is shown on Figures 3.7-25 and 3.7-26.

The base of the finite element model, which is 200 feet below grade, is assumed rigid and is given a specified earthquake input motion. This input motion is determined as follows:

Using the soil profile at the edge of the finite element model to define the free field, an infinite layered system is developed. A motion applied at grade within the layered system is deconvoluted to the base of the system. The element strain dependent properties are revised and the process repeated until strain compatible soil properties are obtained. The resultant motion at the base of the layered system is then applied to the base of the finite element model by the FLUSH program.

The input to the finite element model shown on Figure 3.7-26 consists of horizontal acceleration histories. However, similar computations are made using vertical input motions which produce free field amplitudes 2/3 of those of the horizontal control motions. The horizontal and vertical motions are assumed to act separately. The results can then be combined in any appropriate fashion.

The boundary conditions on the vertical boundaries of the model depend on the input motion. When the input motion is horizontal, the nodal points on the vertical faces are restrained from moving vertically. This is justified since, in the free field, all motions are indeed horizontal because they consist of vertically propagating shear waves. Similarly, in the case of vertical input motion on the base, the vertical sides of the model are restrained from moving horizontally.

A comparison of response calculations from a finite element analysis and an elastic half-space lumped-spring analysis for the deeply embedded standby service water cooling basin has been made considering the current position of the Regulatory Staff.

The elastic half-space lumped-spring analysis response loads (see Figures 3.7-47 and 3.7-48, SRSS Effective Forces) and the corresponding SRSS accelerations have been compared to the peak accelerations from a finite element analysis for horizontal and vertical OBE and SSE seismic excitations. The response loads from the elastic half-space lumped-spring analysis provide a conservative design basis for the structure when compared to the finite element analysis results and have been used for the structural design of this building.

The floor acceleration response spectra from the elastic half-space lumped-spring analysis and the FLUSH finite element analysis for horizontal and vertical OBE and SSE excitations have been enveloped to generate the design basis floor response spectra. As an added conservatism, the original FLUSH finite element floor response spectra are also included in the above design basis enveloped floor response spectra.

3.7.2.5 Development of Floor Response Spectra

In order to predict the seismic environment for equipment located at various elevations within a structure, floor response spectra have been developed using a time-history analysis technique.

The procedure for a lumped mass elastic half-space analysis entails first developing the mathematical model assuming a linear system, and then obtaining its natural frequencies and mode shapes. The dynamic response at the mass points is subsequently obtained by applying the synthesized earthquake time-history to the structural base. This input time-history is synthesized so that the computed spectral values are equal to or greater than those of the design spectrum.

In the finite element analyses, the above synthesized earthquake time-history is applied at grade in the free field and deconvoluted to the base of the finite element model as described in subsection 3.7.2.4. The dynamic response of the model is then obtained by applying this motion at the base of the finite element model and computing acceleration time-histories at selected mass points.

Using the absolute acceleration time-history response of a particular mass point, a spectrum response curve is developed and enveloped into a design acceleration spectrum to be utilized for the seismic analysis of equipment located at the mass point. Horizontal and vertical response spectra are computed using various damping values for both SSE and OBE. The seismic response to the time-history is obtained through the use of the CE-920 computer program. These time-histories are subsequently utilized as data to compute and plot the in-structure response spectra using the CE-921 or CE-802 computer program. The response spectra values are computed for the system's natural frequencies, in addition to the 49 frequencies tabulated in Table 5-1 of Bechtel Topical Report BC-TOP-4, Rev. 1, including Addendum 1. The computer-generated floor response spectra are smoothed so that they envelope all points for which the response is calculated. The initially computed floor response spectra peaks associated with the structural frequencies are widened by ±15 percent, and the shape of the initial spectra is maintained. The conservativeness of the enveloping technique and the percent of widening are discussed in Section 5.0 of BC-TOP-4, Rev. 1, including Addendum 1. The methods by which the effects of parameter variations (such as material properties and seismic environment uncertainties) are accounted for in the floor response spectra are discussed in subsection 3.7.2.9. The floor response spectra of the seismic Category I structures have been provided for review separately.

3.7.2.6 Three Components of Earthquake Motion (For Other Than NSS Systems)

When applying the response spectrum method, the effects of both horizontal and vertical components of earthquake motion are considered regardless of whether the structure is idealized either as a lumped parameter or finite element system. In order to properly account for the responses of systems subjected to multidirectional excitation, the manner in which their additive effects are combined must be considered since phasing of the responses has not been realistically simulated. Consequently, a statistical combination is used to obtain the net response according to the criterion of the square-root-of-the-sum-of-the-squares which accounts for the randomness of magnitude and direction of earthquake motion. This SRSS criterion, considering three components of ground motion, is used for final seismic stress computation for steel structural design and used for resultant seismic member force computations and reinforced concrete structural design.

3.7.2.7 Combination of Modal Responses

3.7.2.7.1 For Other Than NSS Systems

In utilizing the response spectrum analysis to predict seismic design loads, structural responses are calculated independently for each of the three orthogonal directions of assumed seismic excitation. When the modes of vibration for a structure are not closely spaced, the representative maximum response value of interest is obtained by the criterion of the square-root-of-the-sum-of-the-squares. Closely spaced modes are those where the frequency is less than 10 percent that of the adjacent lower mode. If the modes are closely spaced, the responses of these modes are combined in an absolute manner; the resulting total is treated as a pseudo-mode and then combined with the remainder of the modal responses in an SRSS fashion.

3.7.2.7.2 For NSS Systems

In a response spectrum method of modal dynamic analysis, all modes except the closely spaced modes (i.e., the difference between any two natural frequencies is equal to or less than 10 percent of the lower frequency) are combined by the square-root-of-the-sum-of-the-squares (SRSS) method as described in subsection 3.7.2.7.2.1. The treatment of closely spaced modes is described in subsection 3.7.2.7.2.2.

3.7.2.7.2.1 Square Root of the Sum of the Squares

The square-root-of-the-sum-of-the-squares (SRSS) combination of modal responses is defined mathematically as:

$$R = \left(\sum_{i=1}^{n} (R_i)^2 \right)^{1/2}$$

where

R = Combined Response

R_i = Response in the i^{th} mode

n = Number of modes considered in the analysis

3.7.2.7.2.2 Procedure of Combining Closely Spaced Modal Response

Both time-history and response spectrum methods of analysis are used for seismic analysis. When the time-history method is used, the vector sum at every time step will be used to calculate the maximum response. The use of the time-history analysis method precludes the need to consider closely spaced modes. This method is defined mathematically as:

$$R = \left(\sum_{k=1}^{N} \sum_{s=1}^{N} |R_k \quad R_s| \varepsilon_{ks} \right)^{1/2}$$

where

R = Representative maximum value of a particular response of a given element to a given component of excitation

R_k = Peak value of the response of the element due to the k^{th} mode

N = Number of significant modes considered in the modal response combination

R_s = Peak value of the response of the element attributed to s^{th} mode

Also,

$$\varepsilon_{ks} = \left\{1 + \left[\frac{(\omega_k' - \omega_s')}{\beta_k' \omega_k + \beta_s' \omega_s}\right]^2\right\}^{-1}$$

in which:

$$\omega_k' = \omega_k (1 - \beta_k^2)^{1/2} \quad , \quad \beta_k' = \beta_k + \frac{2}{t_d \omega_k}$$

where

ω_k = Modal frequency in the k^{th} mode

β_k = Damping ratio in the k^{th} mode

t_d = Duration of the earthquake

3.7.2.8 Interaction of Non-Category I Structures with Category I Structures

The manner in which the seismic analysis of non-Category I structures is treated depends upon their location with adjacent Category I structures. Any isolated non-Category I structure, whose failure due to a seismic event would not endanger a Category I structure, is designed in accordance with codes and standards applicable to non-Category I structures. A non-Category I structure whose failure could endanger a Category I structure is designed to assure that interaction during a seismic event does not occur. To predict the response, a dynamic analysis is used in lieu of Uniform Building Code design methods.

For the Grand Gulf Nuclear Station, the only non-Category I structure immediately adjacent to a seismic Category I structure is the turbine building. This structure is analyzed and designed in the same manner as Category I structures, thereby assuring that the integrity of adjacent Category I structures (auxiliary and control buildings) is maintained.

The turbine pedestal is an independent structure within the turbine building, separated by a minimum gap of 1-½ inches from the turbine building and designed on the basis of a UBC earthquake for Zone 1. Nevertheless the gap is sufficient to prevent

hammering considering an SRSS combination of the SSE displacements for the pedestal and turbine building structures. Consequently, the overall structural behavior of the two structures during an SSE event will not impair the function of any Category I structure or systems.

3.7.2.9 Effects of Parameter Variations on Floor Response Spectra

The in-structure response spectra which are computed from the absolute acceleration time-history response generally reflect two parameters: a) the amplification of the free field input produced by the soil-structure system, and b) the frequency content associated with these amplification regions.

To account for the effect of structural frequency variation due both to the possible variations in the material properties of the structure and soil and to the uncertainties in the techniques of seismic analysis, the initially computed floor response spectra are smoothed, and peaks associated with the structural frequencies are widened. The amount of peak widening, e.g., $\pm \Delta f_j$, associated with the structural frequency f_j is determined according to the method described below.

Let f be the structural frequency which is determined by using the most probable material and section properties in formulating the structural model. The variation in the structural frequency is determined by evaluating the individual frequency variation due to the most probable variation in each parameter that is of significant effect, such as the soil modulus, material density, etc. The total frequency variation, $\pm f_j$, is then determined by taking the square-root-of-the-sum-of-the-squares of a minimum variation of 0.05 f_j and the individual frequency variation $(\Delta f_j)_n$, that is:

$$\Delta f_j = \sqrt{(0.05 f_j)^2 + (\Delta f_j)_n^2} \qquad (3.7\text{-}7)$$

A value of 0.10 f_j is used if the actual computed value of Δf_j is less than 0.10 f_j. If the above procedure is not used, Δf_j is taken as $0.15 f_j$.

When a two- or three-dimensional structural analysis is performed, the structural motion at a given location and in a given direction would have contributions from both the vertical and the two horizontal analyses. The contribution from each individual analysis will generate a floor response spectrum at the same location and for the same direction. These individual floor response spectra are then combined according to the sum of the absolute values criteria to predict the total floor response spectrum at the given location and for the given direction.

3.7.2.10 Use of Constant Vertical Static Factors (For Other Than NSS Systems)

No constant vertical static factors are utilized for the design of Category I structures. The methods of analysis, as described in subsection 3.7.2.1.1, are applied for both horizontal and vertical directions of seismic motion. The magnitude of the vertical excitation is 2/3 that in the horizontal direction.

3.7.2.11 Method Used to Account for Torsional Effects (For Other Than NSS Systems)

The dynamic analysis of structures is covered in subsection 3.7.2.1. The effects of torsion are considered in the analyses as described in the following paragraphs.

When a horizontal load is applied to an asymmetric structure (where the center of mass does not coincide with the center of resistance), coupling between the two horizontal motions occurs. The degree of coupling depends on the amount of eccentricity and the ratio of the uncoupled torsional frequency to the uncoupled lateral frequency. If the uncoupled torsional frequency is high with respect to the uncoupled lateral frequency, and if the eccentricities are small, the coupling between translation and torsion can be neglected and the structure can be treated as a symmetric structure.

In the three-dimensional finite element modeling of structures, the geometry of the structures is preserved, thus, the torsion effects are automatically taken into consideration.

For three-dimensional lumped-mass models which have translation-torsion coupling, the locations of the centers of mass (CM) and centers of rigidity (CR) are determined. Each idealized element is considered to run through the CR and is connected at its ends

to the CM by a rigid bar. However, the couplings between vertical and horizontal responses are very small and can be neglected. Therefore, the vertical and horizontal model can be separated.

When torsional effects are taken into account for the dynamic analysis through the use of three-dimensional mathematical models, the need for static factors and accidental torsion is eliminated.

To assure that torsional effects of all Category I buildings have been adequately considered, a three-dimensional model of the auxiliary building, shown on Figure 3.7-21a, was developed. The auxiliary building was chosen because it is the most non-symmetrical Category I building. This three-dimensional lumped mass model accounts for the actual locations of the centers of mass and centers of rigidity at each elevation.

The response spectra generated using a three-dimensional model were then compared with those based on the mathematical models used for seismic design, shown on Figures 3.7-19 and 3.7-20. This comparison showed that the effects of torsion had a negligible effect on the floor response spectra used as the basis for piping and equipment analysis in the auxiliary building.

The structural response accelerations calculated from the CE-800 response spectrum analysis were compared with those calculated using the mathematical models shown on Figures 3.7-19 and 3.7-20. This comparison also showed that the effects of torsion had a negligible effect on the response loads shown on Figures 3.7-38 through 3.7-43, which were used in the structural design of the auxiliary building.

An additional consideration related to the torsional response of symmetrical structures has not been considered in seismic design; however, this subject will be examined and appropriate measures will be taken, if found necessary.

3.7.2.12 Comparison of Responses (For Other Than NSS Systems)

When both modal response spectrum and time-history methods are applied, the responses obtained from both methods at selected points are tabulated in Table 3.7-12 for a check and comparison.

3.7.2.13 Methods for Seismic Analysis of Dams

Since no dams are utilized directly or indirectly to provide water for the cooling system, this section is not applicable to this power plant.

3.7.2.14 Determination of Category I Structure Overturning Moments

When the combined effect of earthquake ground motion and structural response is strong enough, the structure will undergo a rocking motion, pivoting about either edge of the base. When the amplitude of rocking motion becomes so large that the center of structural mass reaches a position right above either edge of the base, the structure becomes unstable and may tip over (see structural position (b) as indicated by dotted lines on Figure 3.7-56). The mechanism of such rocking motion is that of an inverted pendulum, and its natural period is very long compared with that of the linear, elastic structural response. Hence, so far as overturning evaluation is concerned, the structure can be treated as a rigid body.

The maximum kinetic energy is conservatively estimated to be:

$$E_s = \frac{1}{2} \sum m_i \left[(v_H)_i^2 + (v_V)_i^2 \right] \qquad (3.7\text{-}8)$$

in which $(v_H)_i$ and $(v_V)_i$ are the maximum values of the total lateral velocity and total vertical velocity, respectively, of mass mi. $(v_H)_i$ and $(v_V)_i$ may be computed as follows:

$$(v_H)_i^2 = (v_x)_i^2 + (v_H)_g^2$$

$$(v_V)_i^2 = (v_z)_i^2 + (v_v)_g^2 \qquad (3.7\text{-}9)$$

in which $(v_H)_g$ and $(u_V)_g$ are the peak horizontal and vertical ground velocity, respectively, and $(v_x)_i$ and $(v_z)_i$ are the maximum values of the relative lateral and vertical velocity of mass m_i.

Letting m_o be the total mass of the structure and base mat, the energy required to overturn the structure is equal to:

$$E_o = m_o g \Delta h$$

where h is the height to which the center of mass of the structure must be lifted to reach the overturning position, i.e., position (b) on Figure 3.7-56. Because the structure may not be a symmetrical one, the value of h is computed with respect to the edge that is nearer to the center of mass. The structure is defined stable against overturning when the ratio E_o/E_s exceeds 1.5.

The above calculations assume the structure rests on the ground surface and, hence, are very conservative if the structure is embedded to a considerable depth. Embedded effect is considered only when the above criterion is exceeded.

3.7.2.15 Analysis Procedure for Damping

3.7.2.15.1 Category I Structures, Systems, and Components Other Than NSS System

For structures composed of major subsystems that are made of different materials, the composite modal damping is computed by first constructing the following matrix:

$$B = \varnothing^T \; -\beta_m - \; M \; \varnothing$$

Where Ø is the modal matrix, and $[-\beta_m-]$ is a diagonal matrix made up of the damping values specified for the subsystems. The composite modal damping is then obtained from B by using the diagonal terms after they are divided by the generalized mass of the corresponding mode, where the generalized mass is defined by m_j as follows:

$$-m_j - \; = \varnothing^T \; M \; \varnothing \qquad (3.7\text{-}11)$$

To account for different damping characteristics of the superstructure and the foundation soil during soil-structure interaction, refer to subsection 3.7.2.1.1.4.

3.7.2.15.2 Analysis Procedure for Damping (NSS Systems)

In a linear dynamic analysis the procedure to be utilized to properly account for damping in different elements of a coupled system model is as follows:

a. The structural damping of the various structural elements of the model are first specified. Each value is referred to as the damping ratio (B_j) of a particular component which contributes to the complete stiffness of the system.

b. Perform a modal analysis of the linear system model. This results in a modal matrix (Ø) normalized such that $Ø^T_i \, K Ø_i = W^2_i$ where K is the stiffness matrix, W_i the circular natural frequency of mode i, and $Ø^T_i$ is the transpose of Ø, which is a column vector of Ø corresponding to the mode shape of mode i. Matrix Ø contains all translational and rotational coordinates.

c. Using the strain energy of the individual components as a weighing function, the following equation can be derived to obtain a suitable damping ratio (B_i) for the ith mode.

$$B_i = \frac{\sum_{k=1}^{N} Ø_i^T \, B_j \, K_j \, Ø_i}{w_i^2}$$

Where

N = Total number of structural elements

$Ø_i$ = Mode shape for mode i ($Ø^T_i$ as transpose)

B_j = Percent damping associated with element j

K_j = Stiffness contribution of element j

w_i = Circular natural frequency of mode i

3.7.3 Seismic Subsystem Analysis

In addition to the loads imposed on the system under normal operating conditions, the design of equipment supports requires that consideration also be given to abnormal loading conditions such as an earthquake. Two types of seismic loadings are considered: operational basis earthquake (OBE) and safe shutdown earthquake (SSE).

For the operational basis earthquake loading condition, the nuclear steam supply system is designed to be capable of continued safe operation. Therefore, for this loading condition, critical structures and equipment are required to operate within design limits. The seismic design for the safe shutdown earthquake is intended to provide a margin in design that assures capability to shut down and maintain the nuclear facility in a safe condition. In this case, it is only necessary to ensure that required critical systems and components do not lose their capability to perform their safety-related function. This has come to be referred to as the "no loss of function" criteria and the loading condition as the safe shutdown earthquake loading condition.

Not all critical components have the same functional requirements for safety. For example, the reactor containment must retain capability to restrict leakage to an acceptable level. Therefore, based on present practice, general elastic behavior of this structure under the SSE loading condition must be ensured. On the other hand, many components can experience significant permanent deformation without loss of function. Piping and vessels are examples of the latter where the principal requirement is that they retain their contents and allow fluid flow.

A summary of the seismic requirements for ASME Code, Section III, Class I components is given in Table 3.2-1.

3.7.3.1 Seismic Analysis Methods

This section discusses the methods by which seismic Category I subsystems and components are qualified to ensure the functional integrity of the specific operating requirements which characterize their seismic Category I designations. In order to

demonstrate their capability to withstand seismic forces without loss of function, a detailed investigation is necessary. Specific requirements and procedures for qualifying seismic Category I equipment are described in Section 3.10.

In general, the equipment manufacturer has been given the choice of three methods by which the equipment can be seismically qualified: a) by using either a dynamic analysis, b) testing procedures, or c) a combination of a) and b). Appropriate design response spectra (OBE and SSE) are furnished to the manufacturer for seismic qualification purposes. Additional information such as input time-history is supplied only when necessary.

When analysis is used to qualify seismic Category I systems and components, the analytical techniques have to conservatively account for the dynamic nature of the systems or components. Both the SSE and OBE, with their different damping values, are considered when the dynamic analysis is performed.

The general approach employed in the dynamic analysis of seismic Category I equipment and component design is based on the response spectrum technique. The time-history analysis of Category I structures, as described in subsection 3.7.2.1.1 generates in-structure response spectrum curves and time-histories at various support elevations for use in the analysis of systems and equipment.

At each level of the structure where vital items are located, horizontal spectra for each of the two major axes of the structure and a vertical response spectrum are developed, as described in subsection 3.7.2.5.

When items are supported at two or more elevations, the response spectra of each elevation are superimposed on each other and the resulting spectrum is the upper bound envelope of all the individual spectrum curves considered.

For vibrating systems and their supports, multi-mass models are used in accordance with the lumped parameter modeling techniques and normal mode theory described in subsection 3.7.2.1.1 and the references listed in subsection 3.7.5. Piping analysis is described in subsection 3.7.3.3.

When testing is used to qualify seismic Category I systems and components, all the loads normally acting on the equipment are simulated to act on the equipment during the test. The actual

mounting of the equipment is also simulated or duplicated. Tests are performed by supplying random input to the shaker table to such an extent that tested response spectra are equal to or greater than the required response spectra.

For certain seismic Category I equipment and components, where dynamic testing is necessary to ensure functional integrity, test performance data and results reflect the following:

a. Performance data of equipment which has been subjected under the specified conditions to dynamic loads equal to or greater than those experienced under the specified seismic conditions.

b. Test data from previously tested comparable equipment which has been subjected under similar conditions to dynamic loads equal to or greater than those specified.

c. Actual testing of equipment in accordance with one of the methods described in Sections 3.9 and 3.10.

3.7.3.2 Determination of Number of Earthquake Cycles

To evaluate the number of cycles which exist within a given earthquake, a typical boiling water reactor building dynamic model was excited by three different recorded time-histories: May 18, 1940, El Centro NS component, 29.4 sec; 1952, Taft N 69° W component, 30 sec; and March 1957, Golden Gate S 80° E component, 13.2 sec. The modal response was truncated so that the response of three different frequency bandwidths could be studied: 0+-10, 10-20, and 20-50 Hz. This was done to give a good approximation of the cyclic behavior expected from structures with different frequency content.

Enveloping the results from the three earthquakes and averaging the results from several different points of the dynamic model, the cyclic behavior given in Table 3.7-13 was formed.

Independent of earthquake or component frequency, 99.5 percent of the stress reversals occur below 75 percent of the maximum stress level and 95 percent of the reversals lie below 50 percent of the maximum stress level. This relationship is graphically shown on Figure 3.7-57.

In summary, the cyclic behavior number of fatigue cycles of a component during an earthquake is found in the following manner:

a. The fundamental frequency and peak seismic loads are found by a standard seismic analysis.

b. The number of cycles which the component experiences is found in Table 3.7-13 according to the frequency range within which the fundamental frequency lies.

c. For fatigue evaluation, one-half percent (0.005) of these cycles is conservatively assumed to be at the peak load and 4.5 percent (0.045) at three-quarters of the peak. The remainder of the cycles have negligible contribution to fatigue usage.

The safe shutdown earthquake has the highest level of response. However, the encounter probability of the SSE is so small that it is not necessary to postulate the possibility of more than one SSE during the 40-year life of a plant. Fatigue evaluation due to the SSE is not necessary since it is a faulted condition and thus not required by the ASME Code, Section III.

The OBE is an upset condition and must be included in fatigue evaluations according to the ASME Code, Section III. Investigation of seismic histories for many plants shows that during a 40-year life it is probable that five earthquakes with intensities one-tenth of the SSE intensity, and one earthquake approximately 20 percent of the proposed SSE intensity, will occur. Therefore, the probability of even an OBE is extremely low. The probability of a lesser intensity earthquake is more realistic to consider. However, a sample study has shown that a total of 20 one-quarter SSE events (25% intensity) would be required to produce the stress levels experienced during one OBE event. Further study of strong motion in earthquakes shows that the use of 10 cycles is conservative. To cover the combined effects of these earthquakes, one OBE intensity earthquake with 10 peak stress cycles is postulated for fatigue evaluation.

Subsection 3.9.1.1 presents the number of fatigue cycles used in the design of GE-supplied subsystems.

When a subsystem is qualified by testing, the subsystem is first subjected to one OBE environment followed by one SSE. During these durations, if no malfunctions of the equipment or structural failures have been observed, the equipment passes the fatigue criteria. No fatigue checks are performed when the analysis method is used in a subsystem qualification.

3.7.3.3 Procedure Used for Modeling

3.7.3.3.1 Procedures Used to Model Subsystems Other Than NSSS

3.7.3.3.1.1 Summary

If a dynamic analysis is used to predict the actual response of a piping system to the specified forcing function, the dynamic model must adequately represent all significant modes, selection of the proper response spectrum curves, and proper location of anchors to separate seismic Category I from nonseismic piping systems.

3.7.3.3.1.2 Selection of Mass Points

When performing a dynamic analysis, a piping system is idealized as a mathematical model consisting of lumped masses connected by elastic members. The elastic members are given the properties of the piping system being analyzed. The lumped masses are carefully located to adequately represent the dynamic and elastic properties of the piping system. A lumped mass is located at the beginning and end of every elbow or valve, at the extended valve operator, and at the intersection of every tee. On straight runs, lumped masses are located at spacings no greater than the span length corresponding to 33 Hz. A mass point is located at every extended mass to account for torsional effects on the piping system. (The valve purchase specifications require the extended top works to have a fundamental frequency on the rigid side of the response spectrum curve.) In addition, the increased stiffness and mass of valves are considered in the modeling of a piping system.

3.7.3.3.1.3 Selection of Spectrum Curves

In selecting the spectrum curve to be used for dynamic analysis of a particular piping system, a curve is chosen which will most closely describe the accelerations existing at the end points and restraints of the system. For a piping system spanning a large elevation difference within one structure, the worst single floor response spectrum should be used as input to all supports. In cases of piping systems which run between different structures, a single response spectrum enveloping all spectral acceleration values was used.

3.7.3.3.1.4 Interface of Seismic Category I and Other Piping Systems

In certain instances, seismic Category I piping may be connected to nonseismic piping at locations (other than a piece of equipment) which, for purposes of analysis, could be considered an anchor. These transition points typically occur at seismic Category I valves. Since a dynamic analysis must be modeled from anchor point to anchor point, two options exist:

a. Specify a structural anchor at the seismic Category I valve and analyze the seismic Category I system; or, if impractical, to design an anchor.

b. Analyze the system from the anchor point in the seismic Category I system through the valve and to the first anchor point in the nonseismic system.

Where small, nonseismic piping is directly attached to seismic Category I piping, its effect on the seismic Category I piping is accounted for by lumping a portion of its mass with the seismic Category I piping at the point of attachment.

3.7.3.3.1.5 Effect of Differential Building Movements

In most cases, piping systems are anchored and restrained to floors and walls of buildings that may have differential movements during a seismic event. These may range from insignificant differential displacements between rigid walls of a common building at low elevations to relatively large displacements between separate buildings at a high-seismicity site.

Differential end-point or restraint deflections cause forces and moments to be induced into the piping system. The stress thus produced is a secondary stress. It is justifiable to place this stress, which results from restraint of free end displacement of the piping system, in the secondary stress category because the stresses are self-limiting and, when the stresses exceed yield strength, minor distortions or deformations within the piping system satisfy the condition which caused the stress to occur.

This contribution of the earthquake produces a stress-exhibiting property much like a thermal expansion stress, and a static analysis may be used to obtain actual stresses. The differential displacements are obtained from the dynamic analysis of the

building. These displacements are applied to the piping anchors and restraints corresponding to the maximum differential displacements which could occur. The static analysis is made three times; once for one of the horizontal differential displacements, once for the other horizontal differential displacements, and once for the vertical.

3.7.3.3.2 Procedure Used for Modeling NSS Systems

3.7.3.3.2.1 Modeling of Piping Systems

The continuous piping system is modeled as an assemblage of the beams. The mass of each beam is lumped at the nodes connected by weightless elastic members, representing the physical properties of each segment. The pipe lengths between mass points are no greater than the length which would have a natural frequency of 33 Hz when calculated as a simply supported beam. All concentrated weights on the piping system such as main valves, relief valves, pumps, and motors are modeled as lumped masses. The torsional effects of the valve operators and other equipment with offset center of gravity with respect to center line of the pipe is included in the analytical model. If the torsional effect is expected to cause pipe stresses less than 500 psi, this effect may be neglected.

3.7.3.3.2.2 Modeling of Equipment

For dynamic analysis, seismic Category I equipment is represented by lumped mass systems which consist of discrete masses connected by weightless springs. The criteria used to lump masses are:

a. The number of modes of a dynamic system is controlled by the number of masses used. Therefore, the number of masses is chosen so that all significant modes are included. The modes are considered significant if the corresponding natural frequencies are less than 33 Hz and the stress calculated from these modes is greater than 10 percent of the total stresses obtained from lower modes.

b. Mass is lumped at any point where a significant concentrated weight is located. Examples are: the motor in the analysis of pump motor stand, the impeller in the analysis of pump shaft, etc.

c. If the equipment has a free-end overhang span whose flexibility is significant compared to the center span, a mass is lumped at the overhang span.

d. When a mass is lumped between two supports, it is located at a point where the maximum displacement is expected to occur. This tends to conservatively lower the natural frequencies of the equipment. Similarly, in the case of live loads (mobile) and a variable support stiffness, the location of the load and the magnitude of support stiffness are chosen so as to yield the lowest frequency content for the system. This is to ensure conservative dynamic loads since equipment frequencies are such that the floor spectra peak is in the lower frequency range. If such is not the case, the model is adjusted to give more conservative results.

3.7.3.3.3 Field Location of Supports and Restraints

The field location of seismic supports and restraints for seismic Category I piping and piping system components is selected to satisfy the following two conditions:

a. The location selected must furnish the required response to control strain within allowable limits.

b. Adequate building strength for attachment of the components must be available.

The final location of seismic supports and restraints for seismic Category I piping, piping system components, and equipment, including the placement of snubbers, is checked against the drawings and instructions issued by the engineer. An additional examination of these supports and restraining devices is made to assure that the location and characteristics of these supports and restraining devices are consistent with the dynamic and static analyses of the systems.

3.7.3.4 Basis of Selection of Frequencies

All modal frequencies in the range of 0.25 to 33 Hz are considered in the seismic analysis and testing of systems and components. These frequencies cover the natural frequencies of most of the components and structures under consideration. If the fundamental frequency of a component is greater than or equal to 33 Hz, it is

treated as rigid and analyzed accordingly. Frequencies less than 0.25 Hz are not considered as they represent very flexible structures and are not encountered in this plant.

The frequency range between 0.25 Hz and 33 Hz covers the range of response amplification for the broad band seismic response spectrum used in the design. Modal frequencies of systems and components are determined by analysis or test. Testing for modal frequencies is performed in accordance with the provisions of IEEE-344-71 or IEEE-344-75. Analysis for modal frequencies consists of the development of a mathematical model of the system/ component for use in modal analysis.

Where practical, the systems and components are designed such that their first primary mode of vibration is above 33 Hz. Where this is not possible or where primary modal frequencies correspond to the dominant structural frequencies, the actual modal frequencies of the system or component below 33 Hz are included in the analysis (or test) considering response amplification at that modal frequency from the applicable required response spectrum. The equipment is then designed for the loads resulting from the analysis or test. This ensures that the final design of systems and components accounts for all of the expected seismic loads in accordance with the requirements contained in SRP section 3.7.3.

3.7.3.5 Use of the Equivalent Static Load Method of Analysis

3.7.3.5.1 Equivalent Dynamic Analysis

As differentiated from the dynamic analysis described in subsection 3.7.3.8.1.2.4 which produces a unique analysis for each piping system, this approach results in charts and tables for each site showing span lengths and restraint forces for pipes at various building elevations.

Since the technique uses a modified response spectrum curve, the method uses the same title.

3.7.3.5.2 Modified Spectrum Method

A piping system may be considered seismically acceptable in accordance with ASME Code, Section III, if it can be divided into a series of spans. These spans are limited by guides: two mutually perpendicular restraints normal to the pipe at each change of direction, at all concentrated masses (e.g., valves), at all

extended masses, at each tee, and at a maximum spacing on straight runs of piping determined by dynamic calculations based on a modified spectrum curve.

The spectrum curve for a particular building elevation is modified so that the flexible side of the peak of the curve remains constant at the peak spectral acceleration for decreasing frequencies as shown in Figure 3.7-55.

If a dynamic analysis were performed using the above spectrum, the results would, by inspection, be conservative. The fundamental frequency of the piping system, supported as stated above, is greater than or equal to the fundamental frequency of a simply supported beam of maximum seismic span which is calculated as follows:

$$f = \frac{\pi}{2}\sqrt{\frac{EI}{mL^4}} \qquad \text{Equation 3.7-13}$$

where:

f = fundamental frequency

E = modulus of elasticity

I = moment of inertia

m = mass per unit length

L = maximum seismic span (maximum distance between two seismic guides)

The following is a description of the development of the Modified Spectrum Method. The circular frequency of a simply supported beam is calculated:

$$W_n = (n\,\pi)^2 \sqrt{\frac{EI}{mL^4}} \qquad \text{Equation 3.7-14}$$

where:

W_n = natural circular frequency of the nth mode

The response spectrum method is then used to find the maximum response of each mode

$$V_{2n-1} = \frac{4mSa_{2n-1}}{(2n-1)W^2_{2n-1}} \quad , V_{2n} = 0 \qquad \text{Equation 3.7-15}$$

where

Sa_n = spectral acceleration value of the modified spectrum curve for the nth mode

V_n = maximum displacement (at midspan) due to nth mode

n = 1,2,3,... etc.

The maximum internal moment (at mid-span) and the maximum restraint force (at the support) are determined from the equations:

$$M_{2n-1} = \frac{4mL^2Sa_{2n-1}}{\pi^3(2n-1)^3} \quad , M_{2n} = 0 \qquad \text{Equation 3.7-16}$$

$$R_{2n-1} = \frac{4mL^2Sa_{2n-1}}{\pi^2(2n-1)^2} \quad , R_{2n} = 0 \qquad \text{Equation 3.7-17}$$

where:

M_n = maximum internal moment (at mid-span) due to the nth mode

R_n = maximum restraint force (at the support) due to the nth mode

The modal displacements, internal moments, and restraint forces are combined by the square-root-of-the-sum-of-the-squares (SRSS) method.

From the nature of the modified spectrum curve, the spectral acceleration for the first mode is always the largest value of the spectral accelerations of any mode. The first mode frequency for a given span is calculated and the resultant spectral acceleration is obtained. This maximum acceleration is then applied to all higher modes giving conservative results. The variables in equations 3.7-15 through 3.7-17 can then be eliminated and the equations reduced to:

$$V = \frac{.0131mL^4Sa}{EI} \qquad \text{Equation 3.7-18}$$

$$M = .1291\ mL^2Sa \qquad \text{Equation 3.7-19}$$

$$R = .8164\ mLSa \qquad \text{Equation 3.7-20}$$

where:

Sa = spectral acceleration of modified spectrum curve corresponds to the fundamental frequency of the maximum seismic span

V = maximum displacement

M = maximum internal moment R = maximum restraint force

The analysis is made for two horizontal and one vertical excitation. The horizontal and vertical responses are then combined on the SRSS basis.

3.7.3.6 Three Components of Earthquake Motion

3.7.3.6.1 For Subsystems Other Than NSSS

Responses from two horizontal and one vertical earthquake motions are found by using the SRSS method to get codirectional stress or member forces for analysis performed by the response spectra method.

3.7.3.6.2 For NSS Systems

The total seismic response is predicted by combining the response calculated from the two horizontal and the vertical analyses. When the response spectrum method is used, the methods for combining the loads from the three analyses will be based on the following three basic principles:

a. The peak responses of the different modes for the same earthquake excitation do not occur at the same time.

b. The peak responses of a particular mode due to earthquake excitations from different directions do not occur at the same time.

c. The peak stress due to different modes and due to different excitations may not occur at the same location nor in the same direction.

To implement the above principles, the three translational components of earthquake excitation are statistically combined by the method of the square-root-of-the-sum-of-the-squares (SRSS) of all three of the responses of interest (e.g., strain, displacement, stress, moment, shear).

The seismic responses are combined in the following manner:

$$\bar{\sigma}_{ij} = \left(\sum_{k=1}^{3} (\sigma_{ijk})^2 \right)^{1/2}$$

3.7.3.7 Combination of Modal Responses

3.7.3.7.1 For Subsystems Other Than NSSS

The basis for combining the modal responses (e.g., displacements, effective inertia forces and accelerations, internal forces and moments, support reactions, and stresses) as described in subsection 3.7.3.5 for all piping systems except safety-related piping systems is the square-root-of-the-sum-of-the-squares (SRSS). In order to obtain the most conservative results, the three directional (vertical and two horizontal) responses obtained by the modal combinations of each direction are then combined by the SRSS method.

For all safety-related piping systems required to safely shut down the plant during an earthquake, the modal responses are combined as required by Regulatory Guide 1.92, Revision 1, using the grouping method.

After the total internal moments, support reactions, and stresses are obtained by combining the modal and directional internal moments, support reactions and stresses, they are then combined with other loadings (thermal, weight, and pressure) in accordance with ASME Code, Section III.

3.7.3.7.2 For NSS Systems

In a response spectrum method of modal dynamic analysis, all modes except the closely spaced modes (i.e., the difference between any two natural frequencies is equal to or less than 10 percent of the lower frequency) are combined by the square-root-of-the-sum-of-the-squares (SRSS) method as described in subsection 3.7.3.7.2.1. The treatment of closely spaced modes is described in subsection 3.7.3.7.2.2.

3.7.3.7.2.1 Square-Root-of-the-Sum-of-the-Squares

The square-root-of-the-sum-of-the-squares (SRSS) combination of modal responses is defined mathematically as:

$$R = \left(\sum_{i=1}^{n} (R_i)^2 \right)^{1/2}$$

where

R = Combined response

R_i = Response in the i^{th} mode

n = Number of modes considered in the analysis

3.7.3.7.2.2 Procedure of Combining Closely Spaced Modal Response

Both time-history and response spectrum methods of analysis are used for seismic analysis. When the time-history method is used, the vector sum at every time step will be used to calculate the maximum response. The use of the time-history analysis method precludes the need to consider closely spaced modes. This method is defined mathematically as:

$$R = \left(\sum_{k=1}^{N} \sum_{s=1}^{N} |R_k R_s| \varepsilon_{ks} \right)^{1/2}$$

where

R = Representative maximum value of a particular response of a given element to a given component of excitation

R_k = Peak value of the response of the element due to the k^{th} mode

N = Number of significant modes considered in the modal response combination

R_s = Peak value of the response of the element attributed to s^{th} mode

Also,

$$\varepsilon_{ks} = 1 + \frac{(\omega_k' - \omega_s')}{\beta_k' \omega_k + \beta_s' \omega_s}^{2}{}^{-1}$$

in which:

$$\omega_k' = \omega_k \left(1 - \beta_k^2\right)^{1/2}, \quad \beta_k' = \beta_k + \frac{2}{t_d \omega_k}$$

where

ω_k = Modal frequency in the k^{th} mode

β_k = Damping ratio in the k^{th} mode

t_d = Duration of the earthquake

3.7.3.8 Analytical Procedures for Piping

3.7.3.8.1 For Subsystems Other Than NSSS

3.7.3.8.1.1 Introduction

In the event of a seismic occurrence near the Grand Gulf Power Plant, the integrity of certain piping systems within the plant must be ensured. To that end, specific systems are classified as seismic Category I and are analyzed to withstand the earthquake specified for the plant site.

Piping classified as seismic Category I is designed to withstand levels of loading imposed by two hypothetical earthquakes: the safe shutdown earthquake (SSE) and the operating basis earthquake (OBE). The criteria for each are delineated as follows.

Seismic Category I piping systems are those required to retain their integrity in the event of an SSE. These systems are necessary to:

a. Ensure the integrity of the reactor coolant pressure boundary

b. Shut down the reactor and maintain it in a safe shutdown condition

c. Prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the guideline exposures of 10 CFR 100, 10 CFR 50.67 and 10 CFR 50 GDC 19.

Analysis of buried piping is described in Section 6 of Bechtel Topical Report BC-TOP-4, Rev. 1, including Addendum 1, and subsection 3.7.3.12.

3.7.3.8.1.2 Analytical Techniques

3.7.3.8.1.2.1 Summary

Piping may be generally classified according to the dynamic response of the system. Systems are considered rigid if they are supported and restrained so as to cause the first mode of vibration to occur in the rigid range of the response spectrum

curve (see subsection 3.7.3.8.1.2.2 for definition). All other piping is considered flexible. General guidelines can be stated which relate pipe diameter to analytical technique.

3.7.3.8.1.2.2 Rigid Piping

The rigid range of the response spectrum curve is loosely defined as that portion in which there is no significant change in spectral acceleration with increasing frequencies (see point A on Figure 3.7-55.) If piping is supported and restrained so that the first mode of vibration occurs in this range, it is classified as rigid.

Rigid piping systems are analyzed with static equivalent loads corresponding to the acceleration in the rigid range of the response spectrum curves for the applicable floor elevations.

Both horizontal and vertical static equivalent loads are applied to the rigid piping systems. The responses of the component for two horizontal and one vertical direction are combined on a square-root-of-the-sum-of-the-squares (SRSS) basis. The stresses are then computed in accordance with the ASME Code, Section III. The rigid range is dependent on building response and is determined on a case basis. The rigid range floor spectrum typically begins between 20 and 33 Hz.

Classification of a specific piping system may be made in either of the following ways:

a. Restraints may be located so that no span between rigid restraints exceeds the length of a simply supported beam with a rigid range frequency. In addition, restraints are located at changes in direction, concentrated masses, and extended masses.

b. A dynamic analysis may be run to obtain the mode shapes of the piping system. If the first mode frequency is found to be in the rigid range, the system can be assumed rigid.

3.7.3.8.1.2.3 Flexible Piping

Piping that cannot be classified as rigid by the method defined above is assumed to be flexible and the analytical technique incorporates consideration of pipe natural frequencies in addition to the fundamental frequency.

3.7.3.8.1.2.4 Dynamic Analysis

The dynamic analysis of flexible piping systems is performed using the response spectrum method. A flexible piping system is idealized as a mathematical model consisting of lumped masses connected by massless elastic members. The lumped masses are carefully located so as to adequately represent the dynamic and elastic properties of the piping system (see subsection 3.7.3.3.2). The three-dimensional stiffness matrix of the mathematical model is determined by the direct stiffness method. Axial, shear, flexural, and torsional deformations of each member are included. For curved members, a decreased stiffness is used in accordance with ASME Code, Section III. The mass matrix is also calculated.

After the stiffness and mass matrices of the mathematical model are calculated, the natural frequencies of piping systems and corresponding mode shapes are determined using the following equation:

$$(\underline{K} - W_n^2 \underline{M})\underline{\varnothing}_n = \underline{0} \qquad \text{Equation 3.7-21}$$

where:

$\underline{K}$ = stiffness matrix

W_n = natural circular frequency for the nth mode

$\underline{M}$ = mass matrix

$\underline{\varnothing}_n$ = mode shape matrix for the nth mode

$\underline{0}$ = zero matrix

The Givens or Jacobi method is used in the solution of the above equation. The mode shapes are normalized as follows:

$$\underline{\varnothing}_n^t \underline{M} \underline{\varnothing}_n = 1 \qquad \text{Equation 3.7-22}$$

A generalized mass matrix is calculated, and should correspond to:

$$\underline{\varnothing}^t \underline{M} \underline{\varnothing} = \underline{I}$$

where,

$\underline{\emptyset}$ = matrix of mode shapes

$\underline{\emptyset}^t$ = transpose of $\underline{\emptyset}$

$\underline{I}$ = identity matrix

If any one of the off-diagonal terms in the generation of the left-hand side of equation 3.7-22 is greater than 1×10^{-4}, the problem is aborted. This occurs when poor or improper modeling of the piping system exists.

The response spectrum method is then used to find the maximum response of each mode:

$$Y_n(t)_{max} = \frac{\underline{\emptyset}_n^t \underline{M}\underline{D}\, Sa_n}{W_n^2 M_n} \qquad \text{Equation 3.7-23}$$

where,

Sa_n = spectral acceleration value for the nth mode

$\underline{D}$ = earthquake vector matrix, used to introduce earthquake direction to the response analysis

$\underline{\emptyset}^t_n$ = transpose of the nth mode shape

M_n = generalized mass of the nth mode; equals one by equation 3.7-22

Y_n = generalized coordinate for the nth mode

Using the maximum generalized coordinate for each mode, the maximum displacements associated with each mode are calculated:

$$\underline{V}_n = \underline{\emptyset}_n\, Y_n(t)_{max} \qquad \text{Equation 3.7-24}$$

Once the appropriate maximum modal displacements have been determined for each mass point, the effective inertia forces for each mode are computed;

$$\underline{Q}_n = \underline{k}\ \underline{V}_n$$

where,

$\underline{Q}_n$ = effective inertia force matrix due to nth mode

$\underline{V}_n$ = displacement matrix due to nth mode

The effective acceleration for each mode is calculated:

$$\underline{A}_n = \underline{M}^{-1} \underline{Q}_n$$

where,

$\underline{A}_n$ = effective acceleration matrix due to nth mode

$\underline{M}^{-1}$ = the inverse of mass matrix

After the effective inertia forces have been determined, the internal forces and moments for each mode are also calculated:

$$\underline{S}_n = \underline{b}\ \underline{Q}_n$$

where,

$\underline{S}_n$ = internal force and moment matrix due to the nth mode

$\underline{b}$ = force transformation matrix

The modal stresses are then calculated from the modal internal forces and moments in accordance with ASME Code, Section III. The analysis is made three times: once for the vertical direction and once for each of the two principal horizontal directions of the building. The method of combining the modal responses (i.e., displacements, effective inertia forces, effective accelerations, internal forces and moments, support reactions, and stresses) and the responses due to three (one vertical and two horizontal) directions is described in subsections 3.7.3.8.1.2.5 and 3.7.3.8.1.2.6.

3.7.3.8.1.2.5 Modal and Directional Response Combination

The basis for combining the modal responses (e.g., displacements, effective inertia forces and accelerations, internal forces and moments, support reactions, and stresses) as described in subsection 3.7.3.5 is the square-root-of-the-sum-of-the-squares (SRSS). In order to obtain the most conservative results, the

three directional (vertical and two horizontal) responses obtained by the modal combinations of each direction are then combined by the SRSS method.

After the total internal moments, support reactions, and stresses are obtained by combining the modal and directional internal moments, support reactions and stresses, they are then combined with other loadings (thermal, weight, and pressure) in accordance with ASME Code, Section III.

All safety-related piping systems required to safely shut down the plant during an earthquake are analyzed by the grouping method.

3.7.3.8.1.2.6 Cyclic Criteria

As noted in Paragraph 3.7.3.2, 10 peak OBE stress cycles will be used in the fatigue evaluation of ASME Code, Section III, Nuclear Class 1 piping.

3.7.3.8.1.3 Computer Codes for Seismic Analysis

3.7.3.8.1.3.1 Seismic Analysis of Piping Systems - ME632

The ME632 computer program used in the seismic analysis of piping systems was developed by Bechtel International Corporation in San Francisco, California. [HISTORICAL INFORMATION] [Verification of computer program ME632 was performed after significant program changes. In the past, the verification was done by comparing results with an independent piping program used at EDS Nuclear, Incorporated. EDS Nuclear has verified computer program PISOLIA V through benchmarks against STRESS (MIT, IBM) for stiffness computations, SAP (US Berkeley), and STARDYNE (CDC) through dynamic analyses and extensive hand calculations. Recently, verification was performed by using the ASME Pressure Vessel and Piping 1974 Computer Program benchmarks.]

3.7.3.8.1.3.2 Static and Seismic Analysis of Piping Systems - ME101

ME101 is a finite element computer program which performs linear elastic analysis of piping systems using standard beam theory techniques. The input data format is specifically designed for pipe stress engineering, and the English system of units is used.

A thorough checking of the input has been coordinated in the program. In addition, modifications aimed at achieving an improved model are performed automatically.

The output may be used directly for piping design and for conformation to code and other ANSI Standard regulatory requirements. Two piping codes, ASME Code, 1974 and ANSI Standard B31.1, Summer 1973 addenda, are incorporated in the program to the extent of computing flexibility factors, stress intensification factors, and stresses.

ME101 may be used for the static and seismic analysis of piping systems. Static analysis considers one or more of the following: thermal expansion, dead weight, uniformly distributed loads, and externally applied forces, moments, displacements, and rotations.

Seismic analysis is based on standard normal mode techniques and uses response spectrum data. Two methods of eigenvalue solution are available. Determinant Search of Subspace Iteration considers all data points as mass points. Kinematic Reduction and Householder QR consider masses only at specified data points in designated directions.

ME101 generates isometric plots of the piping configuration with optional node numbering. The plots are obtained by either the ZETA or CALCOMP 1036 plotter.

The program uses out-of-core solution techniques for both static and dynamic analysis and has no limitations on the problem size. However, very large systems may become prohibitive due to cost of computation.

3.7.3.8.1.3.3 Verification Procedure

ME101 results have been compared against the following:

a. ME632, Computer Program, Seismic Analysis of Piping Systems, VERB MOD8, 1976 Bechtel International Corporation, San Francisco, California.

b. Pressure Vessel and Piping 1972 Computer Programs Verification, American Society of Mechanical Engineers.

c. Hand calculations.

3.7.3.8.1.4 Damping Ratio

The damping ratio percentage of critical damping of piping systems is that delineated in Regulatory Guide 1.61. The damping ratio is assumed to be the same for all the modes and is as follows:

	Damping Ratio (Percentage)	
Pipe size	OBE	SSE
Pipe diameter greater than 12 inches	2	3
Pipe diameter less than or equal to 12 inches	1	2

3.7.3.8.2 For NSS Systems

The analytical procedures for piping analysis has been described in subsection 3.7.2.1.2.5.1. Methods to include differential piping support movements at different support points has been described in subsection 3.7.2.1.2.4.

3.7.3.9 Multiple Supported Equipment Components with Distinct Inputs

The procedure and criteria for analysis has been described in subsection 3.7.2.1.2.5.2.

3.7.3.10 Use of Constant Vertical Static Factors

Constant vertical static factors were not used.

3.7.3.11 Torsional Effects of Eccentric Masses

The significant torsional effects of valves and other eccentric masses (including valve operators) are taken into account in the seismic piping analysis by using a cantilevered member with a lumped mass at the free end to model the eccentric weight of the valve and motor operator. The method applied is discussed in detail under subsection 3.7.3.3.2.

3.7.3.12 Buried Seismic Category I Piping Systems and Tunnels

The seismic Category I standby service water and HPCS service water yard piping is buried entirely within the Category I structural backfill that is used for backfilling the power block excavation.

All the support interfaces between seismic Category I piping buried in Category I structural backfill and the building structures are between El. 95.0 feet and 133.0 feet. The pipe is considered as a semi-infinite beam on an elastic soil foundation with built-in ends at the supports to the structure. The reactions at these supports are due to small relative movements of these supports and the buried pipe. The maximum expected seismic displacement of the structure at the elevation of these supports, the modulus of subgrade reaction of the soil foundation, and the assumption that the pipe moves with the surrounding structural backfill under the propagation of seismic compressional and shear waves, is used to calculate the resultant pipe stresses due to seismic loading. These stress components due to seismic loading from three orthogonal directions of seismic motion for OBE and SSE are computed in accordance with the methodologies described in Section 6 of Bechtel Topical Report BC-TOP-4, including Addendum 1. The resultant seismic stress components are determined by SRSS of these components from each orthogonal direction of seismic motion. The seismic stress components are combined with the stress components from dead load, live load, thermal load, differential settlement, etc., to determine total resultant stresses.

3.7.3.13 Interaction of Other Piping with Seismic Category I Piping

3.7.3.13.1 For Subsystems Other Than NSSS

In certain instances, seismic Category I piping may be connected to non-seismic Category I piping at locations other than a piece of equipment, which, for purposes of analysis, could be considered an anchor. These transition points typically occur at seismic Category I valves. Since a dynamic analysis must be modeled from pipe anchor point to anchor point, two options exist:

a. Specify a structural anchor at the seismic Category I valve and analyze the Category I system; or if impractical to design an anchor,

b. Analyze the system from the anchor point in the seismic Category I system through the valve and to the first anchor point in the non-seismic system.

Where small, non-seismic Category I piping is directly attached to seismic Category I piping, its effect on the seismic Category I piping is accounted for by lumping a portion of its mass with the seismic Category I piping at the point of attachment.

3.7.3.13.2 For NSS Systems

When non-seismic piping is attached to seismic Category I piping, the other piping is analytically simulated in a manner that does not significantly degrade the accuracy of the analysis of the seismic Category I piping. Furthermore, the other piping is designed to withstand the SSE without failing in a manner that would cause the seismic Category I piping to fail.

3.7.3.14 Seismic Analysis for Reactor Internals

The modeling of RPV internals has been discussed in subsection 3.7.2.3.2. The damping values are given in Table 3.7-6. A comparison of responses is shown in Table 3.9-2b.

3.7.3.15 Analysis Procedure for Damping

The equation described in subsection 3.7.2.15 is used to obtain composite damping for structures composed of major subsystems which are made of different materials. For composite damping values due to structure-soil interaction, refer to Appendix D of BC-TOP-4, Rev. 1, including Addendum 1.

3.7.4 Seismic Instrumentation Program

3.7.4.1 Comparison with NRC Regulatory Guide 1.12, Rev. 1 (April 1974)

The project position on the above guide is discussed in Appendix 3A.

3.7.4.2 Location and Description of Instrumentation

The following instrumentation and associated equipment are used to measure plant response to earthquake motion (see also Table 3.7-17 and Figure 3.7-82):

a. Six triaxial time-history strong-motion accelerometers (SMA)

b. Seven triaxial peak-recording accelerographs (PRA)

c. Deleted

d. Deleted

e. Deleted

f. Deleted

g. Central and local recording and playback equipment

Instrument characteristics and instrumentation station accuracy meet the requirements of ANSI Standard N18.5, Sections 5 and 6 except as discussed in Appendix 3A/1.12-1, paragraph 3.

3.7.4.2.1 Triaxial Strong-Motion Accelerometers

Triaxial strong-motion accelerometers (SMA) produce a record of time-varying acceleration at the sensor location. This data is used directly for analysis and comparison with reference information and may be converted to response spectra form for spectral comparisons with design parameters.

Each sensor unit contains a solid state accelerometer which measure acceleration in three mutually orthogonal directions. All accelerometers have their principal axes oriented identically, with one horizontal axis parallel to the major horizontal axis assumed in the seismic analysis.

One SMA is located on the Unit 1 containment base slab such that it measures the input vibratory motion on the base slab. Because of the suppression pool inside the Mark III containment structure this SMA will be located outside the containment in the auxiliary building. A second SMA is located in Unit l containment attached to the drywell wall at El. 150'-6" on the same containment azimuth as the base slab SMA. These SMAs are oriented with their axes aligned parallel with the major axes of the building used in the

construction of the mathematical model. These will provide data on the frequency, amplitude, and phase relationship of the seismic response of the containment structure. A third SMA is installed in Unit 1 containment attached to a reactor piping support. A fourth SMA is located in the Unit 1 auxiliary building attached to one of the standby gas treatment system filter train supports which is seismic Category I equipment. A fifth SMA is located in the standby service water pump house A, which is an independent Category I structure. A sixth SMA is located in the free field approximately 250 feet from any station structure, with axes oriented in the same direction as the containment building accelerometers.

All SMA units will trigger recording of events when the 0.01g threshold is exceeded. A record of the event is recorded by each individual Motion Recorder (MR) in which an accelerometer is internally mounted. The data is also transmitted to and recorded by the Network Control Center (NCC). The NCC has an LED display which signals which MR exceeded the system activation threshold and triggered the recording process and that the NCC is recording. LEDs display when peak acceleration at any MR exceeds OBE. A single LED displays when peak acceleration at one or more MR exceeds SSE.

The recording and playback systems have a special cabinet furnished for those instruments and devices necessary for system testing, annunciating, calibration, and control. This cabinet is located in the control room.

3.7.4.2.2 Triaxial Peak-Recording Accelerographs

Each sensor unit contains three peak-recording accelerographs (PRA) mounted in a mutually orthogonal array. PRAs that are mounted directly on equipment have one axis coincident with the principal equipment axis. All other PRAs have their principal axes oriented identically, with one horizontal axis parallel to the major horizontal axis assumed in the seismic analysis.

One PRA is located on the auxiliary building base of Unit 1 and another on the control building base. These PRAs measure the peak input vibratory motion on the base slabs. A third PRA is located in the control room to measure the peak output vibratory motion to the main control boards. Another PRA is located on the outside of

the containment structure of Unit 1 near its top to measure the peak amplification of the structure. An additional PRA is located on one of the standby diesel generators of Unit 1.

PRAs are also used to verify the continued availability of seismic Category I systems and equipment within the drywell structures of each unit. One PRA is mounted on the reactor vessel support. Data from PRAs must be manually retrieved following an earthquake and are used in the detailed investigations for particular structures, systems, and equipment.

3.7.4.2.3 Deleted

3.7.4.2.4 Deleted

3.7.4.2.5 Deleted

3.7.4.2.6 Recording and Playback Equipment

A cabinet located in the Control Room houses the NCC, power supplies for the NCC and Mrs, Human Machine Interface (HMI) system computer and dedicated printer. The NCC is capable of calculating peak acceleration from the three accelerometer axes as well as recording and displaying the time-history record of the event. Data can be downloaded directly from each MR to a laptop for analysis should be data be unavailable at the NCC. The NCC all Rs have internal battery backup that will allow recording and playbacks of data for a minimum of 24 hours should the units lose power. The system is capable of calculating response spectra in the frequency domain and Cumulative Average Velocity (CAV) which is available immediately after an earthquake occurrence. Response spectra and CAV are suitable for making timely operating decisions by comparing them with the design spectra. A report of the data analysis is automatically generated and available for display in the Control Room. The report is also available to be printed from a dedicated printer.

The cabinet also contains audible and visual annunciators wired to relays in the NCC that display initiation of NCC recording, OBE exceedance at the containment basemat MR, and OBE exceedance at the MR mounted on the drywell wall.

3.7.4.3 Control Room Operator Notification

Activation of the Motion recorders causes an audible and visual annunciation in the Control Room to alert the plant operator that an earthquake has occurred. This annunciation is set to occur at 0.01g vertical and / or horizontal acceleration. System activation causes initiation of the SMA recording system.

Audible and visual annunciators are provided in the Control Room to indicate if OBE floor accelerations have been exceeded for any of the l accelerometer locations.

The peak acceleration level experienced on the containment base slab is available immediately following the earthquake. This can be obtained from the NCC HMI screen and by printing out the report of the event.

Significant response spectra from the containment base slab is available in the Control Room immediately following an earthquake for comparing the measured response spectra with the OBE and SSE response spectra. The NCC has the capability to provide indication to the control room if the design response spectra over the range of frequencies of interest have been exceeded. The report also contains CAV for additional analysis of the earthquake impact.

3.7.4.4 Comparison of Measured and Predicted Responses

Initial determination of the earthquake level is performed immediately after the earthquake by comparing the measured response spectra from the containment base slab with the OBE and SSE response spectra for the corresponding location. If the measured spectra exceed the OBE response spectra, the plant will be shut down and a detailed analysis of the earthquake motion will be undertaken.

After all earthquakes, the data from all seismic recorders and recording instruments is retrieved. When the OBE has been exceeded, the data from these instruments is analyzed to obtain the seismic accelerations experienced at the location of major Category I structures and equipment. The measured responses from the strong-motion accelerometers, the peak-recording accelerographs, and the response spectrum analyzer are used to determine the response spectra at the location of each seismic Category I structure and system. These spectra are compared with those used in the design to determine whether the structure or

system is still adequate for future use. Equipment-mounted, peak-recording accelerographs are used to determine if the design limitation of the specific equipment to which they are fastened has been exceeded.

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TABLE 3.7-1: AMPLIFICATION RATIOS FOR DESIGN SPECTRA FOR 2.0 TO 6.7 HZ FREQUENCY RANGE

Damping (% of critical)	Amplification Ratio
0.5	4.8
2.0	3.5
5.0	2.6
7.0	1.9
10.0	1.5

TABLE 3.7-2: 71 FREQUENCIES (IN CYCLES PER SECOND) FOR WHICH RESPONSE SPECTRA ARE COMPUTED

0.2,...(increment = 0.1 cps)... 3.0, 3.15, 3.3,
3.45, 3.6, 3.8, 4.0, 4.2, 4.4, 4.7, 5.0, 5.25,
5.5, 5.75, 6.0, 6.25, 6.5, 6.75, 7.0, 7.3, 7.6,
8.0, 8.5, 9.0, 9.5, 10, 10.5, 11, 11.5, 12, 12.5,
13, 13.5, 14, 14.5, 15, 16.5, 18, 20, 22, 25, 28, 33.

TABLE 3.7-3: CRITICAL DAMPING VALUES PER MODE FOR FIXED BASE STRUCTURES AND COMPONENTS

Structure, System, or Components	Critical Damping for OBE (%)	Critical Damping for SSE (%)	Applications
Reinforced Concrete Structures	2	5	All Category I Structures (fixed base)
Bolted Steel Structures	2	3	Building Frames
Equipment and Pipe (diameter in excess of 12 inches)	2	3	Category I Piping
Small Diameter Piping System (diameter less than 12 inches)	1	2	Category I Piping
Bolted Steel Systems	(Note 1)	(Note 1)	Cable Tray Systems Only
Bolted Steel Systems	7	7	Conduit Systems
Bolted Steel Systems	4	7	HVAC Duct Systems (Note 2)
Welded Steel Systems	2	4	HVAC Duct Systems (Note 3)
Steel and Concrete Composite	N/A (Note 4)	7	Turbine Building
Mechanical Equipment	$\leq$2	$\leq$3	Pumps, fans and cranes

Note 1: The damping ratio, the analysis method, and much of the design criteria pertaining to electrical raceway systems were verified by the Cable Tray and Conduit Raceway Test Program. During that program, some 2,000 dynamic tests were performed on several hundred varied cable tray and conduit support

TABLE 3.7-3: CRITICAL DAMPING VALUES PER MODE FOR FIXED BASE STRUCTURES AND COMPONENTS (CONTINUED)

systems. The effects of numerous parameters that could possibly influence system dynamics were investigated. Also, several different types of tray, conduit, and supports from various manufacturers were tested. As a result of this extensive test program and related activities, a conservative design basis for Class IE cable tray and conduit systems has been developed. This information is summarized in a report entitled, "Technical Report-Cable Tray and Conduit Raceway Test Program." Several copies of this report have been submitted to the NRC by letter, from R. Kosiba, Bechtel, to Dr. F. Schauer, NRC, dated January 21, 1980.

For the Category I cable tray systems, the level of system damping is a function of the level of seismic input motion; the following table is used to determine the maximum allowable damping value for cable tray systems:

Instructure ZPA	% Critical Damping (Maximum)
less than 0.1g	7%
0.1g to 0.35g	7% to 20%*
greater than 0.35g	20%

*damping varies linearly as a function of ZPA.

Note 2: HVAC ducts with bolted transverse joints.

Note 3: HVAC ducts with welded transverse joints.

Note 4: The turbine building is designed for SSE only to ensure that it will not collapse onto an adjacent seismic Category I structure. Since the SSE is the most severe seismic event, an analysis for OBE was not performed.

TABLE 3.7-4: LUMPED REPRESENTATION OF STRUCTURE-FOUNDATION INTERACTION FOR CIRCULAR BASE

Motion	Equivalent Spring Constant	Equivalent Damping Coefficient
Horizontal	$k_x = \frac{32(1-\upsilon)GR}{7-8\upsilon}$	$c_x = 0.576 k_x R\sqrt{\rho/G}$
Rocking	$k_\psi = \frac{8GR^3}{3(1-\upsilon)}$	$c_\psi = \frac{0.30}{1+B_\psi} k_\psi R\sqrt{\rho/G}$
Vertical	$k_z = \frac{4GR}{1-\upsilon}$	$c_z = 0.85 k_z R\sqrt{\rho G}$
Torsion	$k_t = 16\ GR^3/3$	$c_t = \frac{\sqrt{k_t I_t}}{1+2I_t/\rho R^5}$

in which

υ = POISSON'S ratio of foundation medium
G = shear modulus of foundation medium
G = shear modulus of foundation medium
ρ = density of foundation medium

$$B_\psi = \frac{3(1-\upsilon)I_o}{8\rho R^5}$$

I_o = total mass moment of inertia of structure and base mat about the rocking axis at the base
I_t = polar mass moment of inertia of structure and base mat

TABLE 3.7-5: LUMPED REPRESENTATION OF STRUCTURE-FOUNDATION INTERACTION FOR RECTANGULAR BASE

Motion	Equivalent Spring Constant	Equivalent Damping Coefficient
Horizontal	$k_x = 2(1+\upsilon)G\,\beta_x\sqrt{BL}$	Use the results for circular base with the following equivalent radius R:
Rocking	$k_\psi = \frac{G}{1-\upsilon}\beta_\psi BL^2$	(1) $R = \sqrt{BL/\pi}$ for translation
Vertical	$k_z = \frac{G}{1-\upsilon}\beta_z\sqrt{BL}$	(2) $R = \sqrt[4]{BL^3}\,/(3\pi)$ for rocking
Torsion	Use Table 3.7-4 for	$R = \sqrt[4]{BL(B^2+L^2)}/(6\pi)$

in which υ and G are as defined previously, and

B = width of the base mat perpendicular to the direction of horizontal excitation

L = length of the base mat in the direction of horizontal excitation

$\beta_x, \beta_\psi \beta_z$ = constants that are functions of the dimensional ratio, L/B. See Fig. A

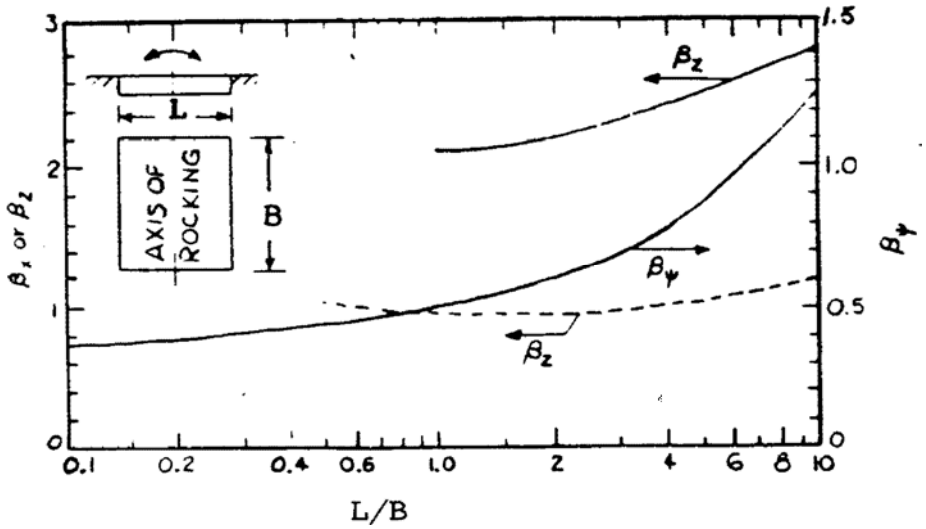


Fig. A Constants β_x, β_ψ and β_z for Rectangular Bases

TABLE 3.7-6: CRITICAL DAMPING RATIOS FOR DIFFERENT MATERIALS FOR NSS SYSTEM

Item		Percent Critical Damping OBE Condition	Percent Critical Damping SSE Condition
Welded structural assemblies (equipment and supports)		2.0	3.0
Vital piping systems	- Diameter greater than 12 in.	2.0	3.0
	- Diameter less than or equal to 12 in.	1.0	2.0
Reactor pressure vessel, support skirt, shroud head, separator and guide tubes		2.0	4.0
Fuel		6.0	6.0

Other values may be used if they are indicated to be reliable by experiment or study.

TABLE 3.7-7: PHYSICAL PROPERTIES OF SOIL MEDIA SUPPORTING CATEGORY I STRUCTURE

Physical Properties	Structural Backfill	Catahoula Formation
Density γ	125 pcf	120 pcf
Poisson's ratio	0.4	0.47
Shear wave velocity	690 fps	1500 ~ 1600 ft/sec
Shear modulus	1850 ± 550 ksf	8400 ~ 9500 ksf
Young's modulus	4440 ± 1320 ksf	24650 ~ 30384 ksf
Damping ratio	0.08 ± 0.02	0.02

TABLE 3.7-8: SOIL-STRUCTURE INTERACTION PARAMETERS TAKEN INTO CONSIDERATION IN SEISMIC MODELING

Category 1 Building	Underlain Layer	Embedded Depth from Top Layer	Structural Foundation Dimension (Ft)	Depth Struct. Fdn. Ratio	Soil Modeling Technique Used
Containment	Catahoula half space	0	134 (diam.)	0	Half space represented by equivalent soil springs and radiation type damping
Auxiliary building	Catahoula half space	45.5 ft. 3 sides	249 x 179	45.5/249 = 0.183 <0.3	Same as above
Control building	Catahoula half space	46.5 ft 1 side	142 x 96	46.5/142 = 0.327 for one side	Same as above
Diesel generator building	60 ft. of compacted backfill & Catahoula	4.5 ft 3 sides	121 x 94	4.5/121 = 0.037 <<0.3	Two layers equivalent soil springs constant
Standby service water cooling towers and storage basins	Loess clay sand on Catahoula	54.5 ft 4 sides	150 x 150	54.5/150 = 0.363	Two dimensional finite element representation of soil with material damping

TABLE 3.7-9: METHODS USED FOR SEISMIC ANALYSIS OF CATEGORY I STRUCTURES

Category I Building	Modeling Techniques for Structures and Foundation	Damping	Method of Analysis
Containment	Lumped masses & half space equivalent soil spring constant	Soil-structure interacted composite modal damping	Time-history analysis and response spectrum method both by modal superposition
Auxiliary building	Same as above	Same as above	Same as above
Control building	Same as above	Same as above	Same as above
Diesel generator building	Lumped masses and 2 layered half space for equivalent soil spring constant	Same as above	Same as above
Standby service water cooling towers and storage basins	Lumped masses and half-space with equivalent soil spring constant	Soil-structure interacted composite modal damping	Time-history analysis and response spectrum method both by modal superposition
	Finite element for structure and various soil layers	By the use of complex soil moduli	Response analysis through frequency domain by "SHAKE" and "LUSH", and "FLUSH" programs
Enclosure building	Finite element and lumped masses for nodal pts.	N/A	No response computation is made after eigenvalue extraction

TABLE 3.7-10: METHODS USED FOR SEISMIC ANALYSIS OF NON-CATEGORY I STRUCTURES

Non-Category I Structures	Method of Seismic Design
Turbine building	Response spectra method for SSE
Turbine pedestal	Response spectra method for SSE
Radwaste building	Response spectra method for SSE
Circulating water pumphouse	UBC 1970 code, See note below
Natural Draft Cooling tower	UBC 1970 code, See note below
Auxiliary Cooling Tower	UBC 1997 code, See note below
Auxiliary Cooling Tower Power and Control Building	UBC 1997 code, See note below
Water treatment building	UBC 1970 code, See note below
Fire water pumphouse	UBC 1970 code, See note below
Condensate storage tank and refueling water storage tank basin	Response spectra method for SSE

Note: Modified or new non-Category I structures can be designed based on the latest edition of UBC code.

TABLE 3.7-11: SIGNIFICANT NATURAL FREQUENCIES FOR CATEGORY I STRUCTURES

Structures	Direction	Significant Natural Frequencies (cps)				
(Operating condition) Containment building	E-W	2.53	3.08	5.38	6.17	6.94
		7.38	8.82	10.13	16.03	18.41
		24.21	25.74	29.36	29.75	30.3
	N-S	2.53	3.08	5.37	6.17	6.94
		7.40	8.82	10.13	16.26	18.64
		23.75	25.01	29.36	29.67	30.18
	Vertical	5.08	8.99	15.59	23.31	28.85
		31.52	43.27			
(Flooded to El. 208.8) Containment building	E-W	1.64	2.07	3.41	4.69	5.56
		6.33	7.10	8.87	9.84	13.97
		14.81	16.80	19.22	20.33	23.4
	Vertical	4.2	8.96	15.48	20.29	
		27.03	31.52			
Auxiliary building	E-W	4.05	8.69	12.25	17.20	
		26.05	33.02			
	N-S	3.7	8.91	14.47	18.34	
		25.58	35.65			
	Vertical	6.09	25.5	39.2		
Control building	E-W	3.65	9.48	18.01	22.85	36.1
	N-S	4.18	9.78	17.15	24.75	37.8
	Vertical	6.82	27.06			
Diesel generator building	N-S	5.29	8.46	31.62		
	E-W	5.39	10.35	38.41		
	Vertical	7.87	69.57			

TABLE 3.7-11: SIGNIFICANT NATURAL FREQUENCIES FOR CATEGORY I STRUCTURES (Continued)

Standby service water cooling tower and makeup basin	Horizon-tal	0.19* 22.29	4.85 35.97	11.4	19.98
	Vertical	7.64	33.9		

* Water Sloshing Frequency

TABLE 3.7-12: COMPARISON OF ABSOLUTE ACCELERATION RESPONSES (N-S, SSE)

Locations in Category I Struct.	By Response Spectra Method	By Time-History Method
Containment, dome, El. 290.4	0.592 g	0.802 g
Containment, drywell El. 208.8	0.394 g	0.507 g
Containment, shield, El. 169.3	1.056 g	1.123 g
Containment, reactor top	1.101 g	1.273 g
Containment, foundation mat	0.147 g	0.191 g
Auxiliary building, El. 264.0	0.357 g	0.576 g
Auxiliary building, foundation slab	0.118 g	0.225 g
Control buildings El. 189	0.239 g	0.501 g
Control building foundation slab	0.125 g	0.206 g
Diesel generator building El. 170	0.187 g	0.224 g
Diesel generator building El. 134	0.155 g	0.209 g

TABLE 3.7-13: NUMBER OF DYNAMIC RESPONSE CYCLES EXPECTED DURING A SEISMIC EVENT

Frequency Band (H_2)z	0^+ - 10	10 - 20	20 - 50
Total number of seismic cycles	168	359	643
Seismic cycles 0.5% of peak loads to 75% of peak loads	0.8	1.8	3.2
Seismic cycles 4.5% of peak loads to 75% of peak loads	7.5	16.2	28.9

TABLE 3.7-14: DELETED

TABLE 3.7-15: DELETED

TABLE 3.7-16: COMPARISON OF AMPLIFICATION RESPONSE SPECTRA (SSE WITH REGULATORY GUIDE 1.60)

AMPLIFICATION FACTORS FOR VERTICAL AND
HORIZONTAL DESIGN SPECTRA
(SSE AND OBE)

Damping	Displacement Amplification	Acceleration Amplification		
(% Critical)	.25 cps	3.5 cps	9 cps	33 cps
0.5	2.11	4.67	3.80	1.37
1	1.85	4.40	3.53	1.23
2	1.67	3.50	2.80	1.00
3	1.63	3.00	2.37	<1.0
5	1.52	2.67	2.08	<1.0
7	1.14	2.27	1.80	<1.0
10	0.81	1.53	1.20	<1.0

TABLE 3.7-16: COMPARISON OF AMPLIFICATION RESPONSE SPECTRA (SSE WITH REGULATORY GUIDE 1.60)(CONTINUED)

AMPLIFICATION FACTORS FOR
HORIZONTAL DESIGN SPECTRA

Damping (% Critial)	Displacement Amplification Point D .25 cps	Acceleration Amplification Point A Period (sec)	Point A Ampl. Factor	Point B Period (sec)	Point B Ampl. Factor	Point C Period (sec)	Point C Ampl. Factor
0.5	1.97	0.030	1.0	0.12	3.2	0.35	4.0
1.0	1.86 (interpolated)	0.032	1.0	0.12	2.8	0.35	3.5
2.0	1.68	0.034	1.0	0.12	2.5	0.35	2.9
5.0	1.40	0.036	1.0	0.12	2.0	0.35	2.3
7.0	1.26 (interpolated)	0.038	1.0	0.12	1.85	0.35	2.0
10.0	1.15	0.040	1.0	0.12	1.7	0.35	1.75

Note: Points A, B, C, taken from Table 3, Reference 1 of Regulatory Guide 1.60.
Point D taken from Table 4, Reference 1 of Regulatory Guide 1.60.

TABLE 3.7-16: COMPARISON OF AMPLIFICATION RESPONSE SPECTRA (SSE WITH REGULATORY GUIDE 1.60)(CONTINUED)

AMPLIFICATION FACTORS FOR
VERTICAL DESIGN SPECTRA

Damping (% Critical)	Displacement Amplification Point D .25 cps	Acceleration Amplification Point A Period (sec)	Point A Ampl. Factor	Point B Period (sec)	Point B Ampl. Factor	Point C Period (sec)	Point C Ampl. Factor
0.5	1.31	0.030	1.0	0.12	3.2	0.35	2.92
1.0	1.24 (interpolated)	0.032	1.0	0.12	2.8	0.35	2.59
2.0	1.12	0.034	1.0	0.12	2.5	0.35	2.21
5.0	.93	0.036	1.0	0.12	2.0	0.35	1.75
7.0	.84 (interpolated)	0.038	1.0	0.12	1.85	0.35	1.52
10.0	.77	0.040	1.0	0.12	1.7	0.35	1.35

Note: Point C scaled by 2/3 in accordance with Reference 1 of Regulatory Guide 1.60.
Point C scaled in accordance with Regulatory Guide 1.60.
Points A, B taken from Table 3, Reference 1 of Regulatory Guide 1.60
Point D taken from Table 3, Reference 1 of Regulatory Guide 1.60.

TABLE 3.7-16: COMPARISON OF AMPLIFICATION RESPONSE SPECTRA (SSE WITH REGULATORY GUIDE 1.60)(CONTINUED)

AMPLIFICATION FACTORS FOR
HORIZONTAL DESIGN SPECTRA

Damping (% Critical)	Displacement Amplification† 0.25 cps	Acceleration Amplification 2.5 cps	9 cps	33 cps
0.5	3.20	5.95	4.96	1.0
1	2.83	5.05	4.25	1.0
2	2.50	4.25	3.54	1.0
5	2.05	3.13	2.61	1.0
7	1.88	2.72	2.27	1.0
10	1.70	2.28	1.90	1.0

†Note: Maximum horizontal ground displacement is specified as 36 inches for lg peak ground acceleration.

TABLE 3.7-16: COMPARISON OF AMPLIFICATION RESPONSE SPECTRA (SSE WITH REGULATORY GUIDE 1.60)(CONTINUED)

AMPLIFICATION FACTORS FOR
VERTICAL DESIGN SPECTRA

Damping	Displacement Amplification†	Acceleration Amplification		
(% Critical)	0.25 cps	2.5 cps	9 cps	33 cps
0.5	2.13	5.67	4.96	1.0
1	1.89	4.95	4.25	1.0
2	1.67	4.05	3.54	1.0
5	1.37	2.98	2.61	1.0
7	1.25	2.59	2.27	1.0
10	1.13	2.17	1.90	1.0

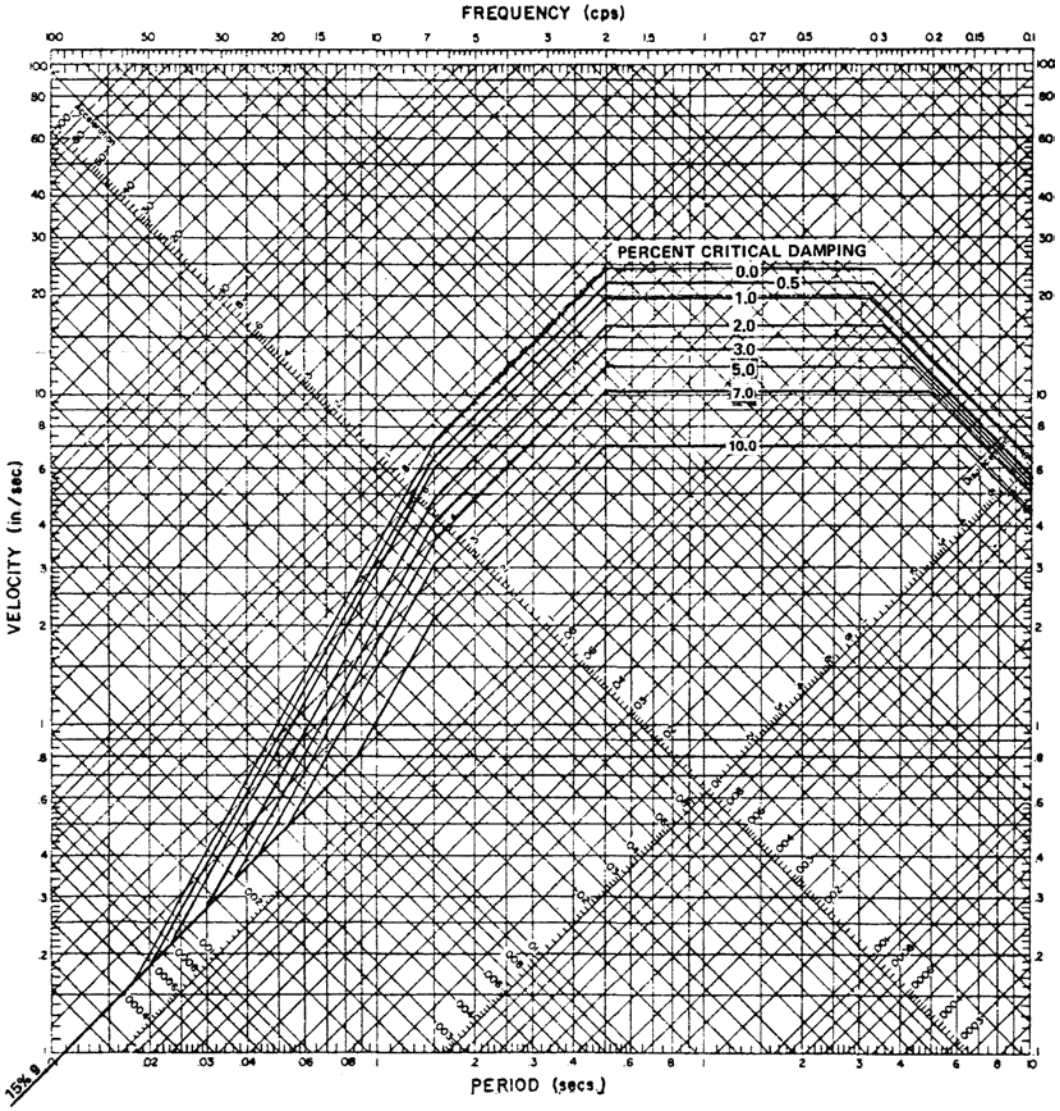
†Note: Maximum vertical ground displacement is specified as 36 inches for 1g peak vertical ground acceleration.

TABLE 3.7-17: SEISMIC MONITORING INSTRUMENTATION

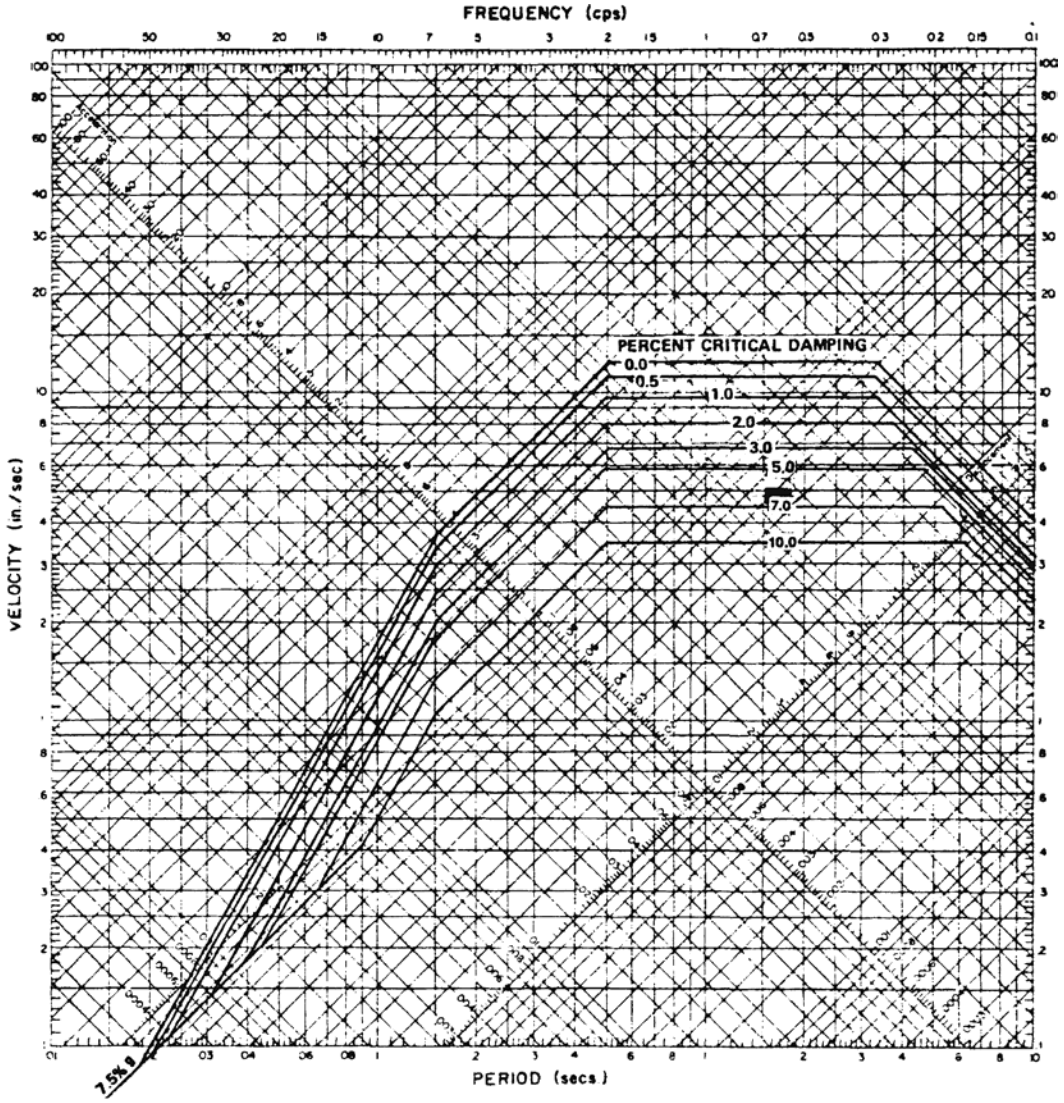
INSTRUMENTATION	TRIAXIAL STRONG MOTION ACCELEROMETER (SMA)	TRIAXIAL PEAK RECORDING ACCELEROGRAPH (PRA)
RANGE	**0.001 to 2.0g**	**0.01 to 2.0g**
	UNIT 1	**UNIT 1**
Location		
I. Free Field	1	
II. Containment Base Mat	1*	
Drywell Wall@ Elevation 150'-6" and Reactor Piping Supports	1*	
Reactor Vessel Supports	1	
III. Outside Containment		1
Category I Structure (SSW Pump House A)	1	
(SSW Pump House B)		1**
Category I Equipment Supports (SGTS Filter Train)	1	
Category I Equipment (Standby Diesel Generator)		1
Auxiliary Building Base Mat		1
Control Building Base		1***
Control Room		1
Near Containment Top		1

* Control Room Annunciation

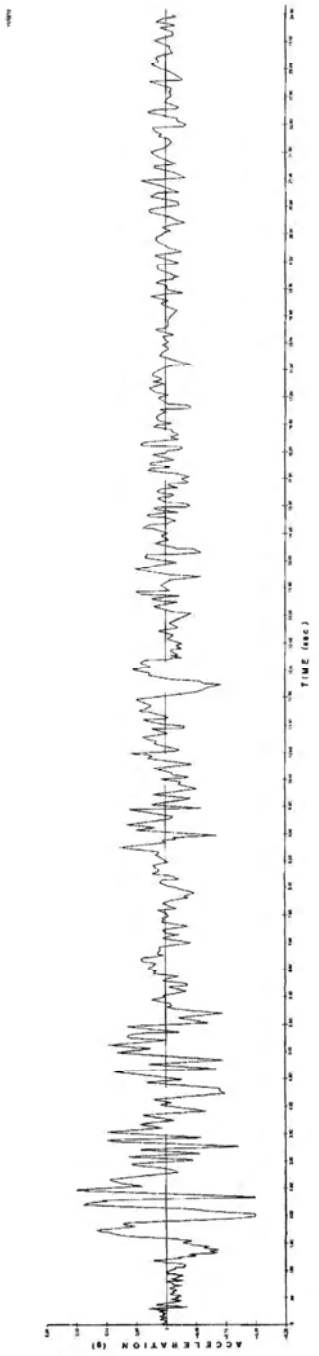
*** Range of Accelerograph is 0.013 to 5 g Control Room Annunciation



MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT	DESIGN SPECTRA FOR SSE FIGURE 3.7-1



MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT	DESIGN SPECTRA FOR OBE FIGURE 3.7-2



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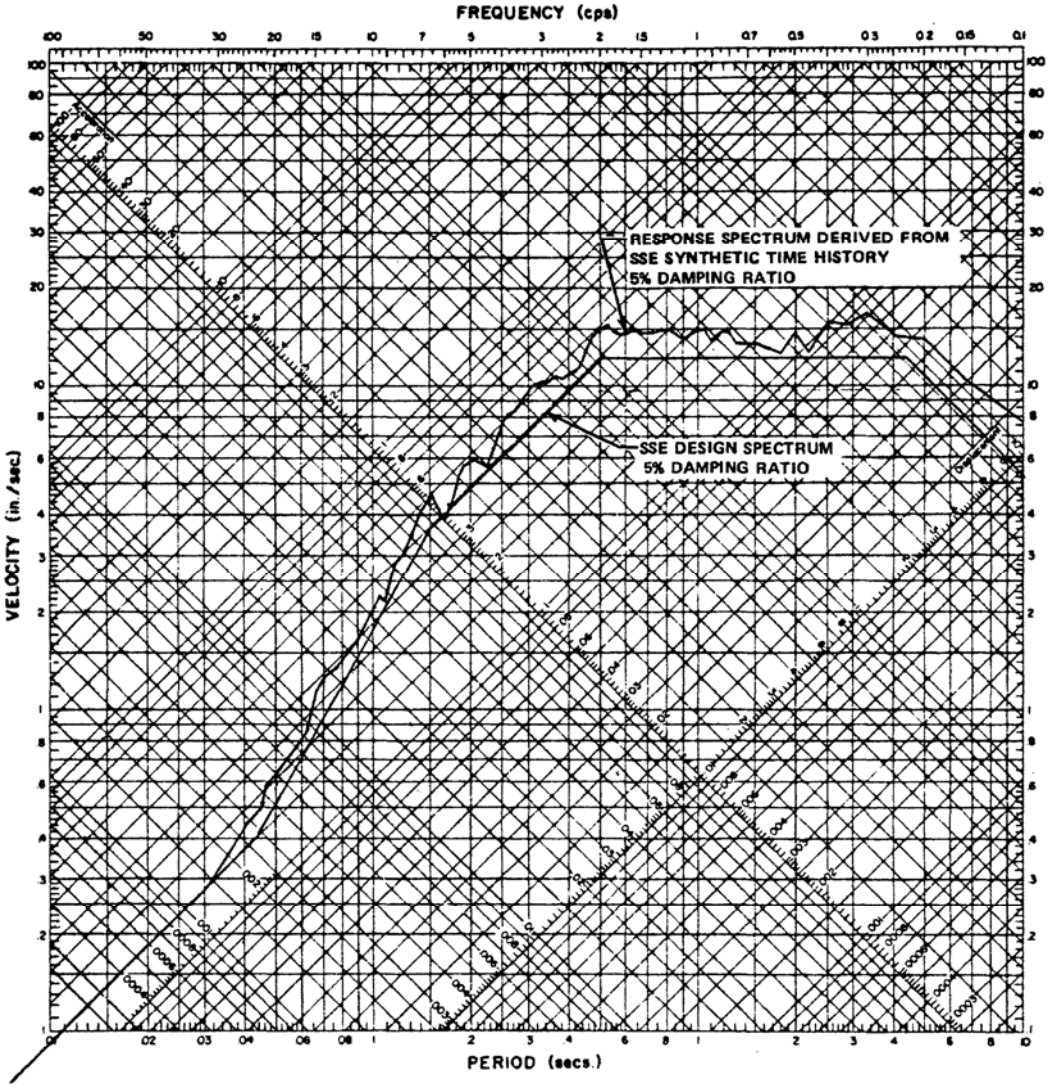
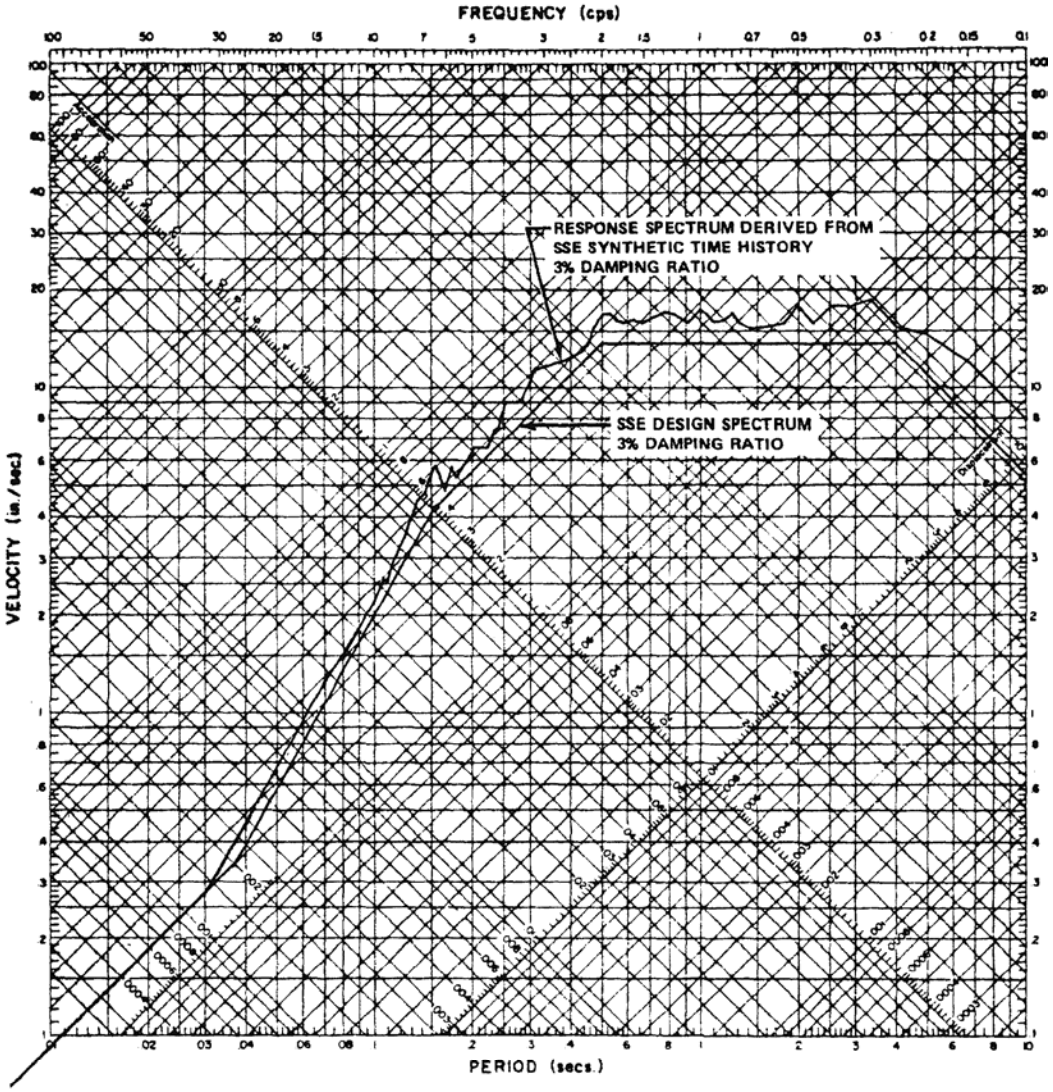
SYNTHETIC TIME HISTORY FOR SSE

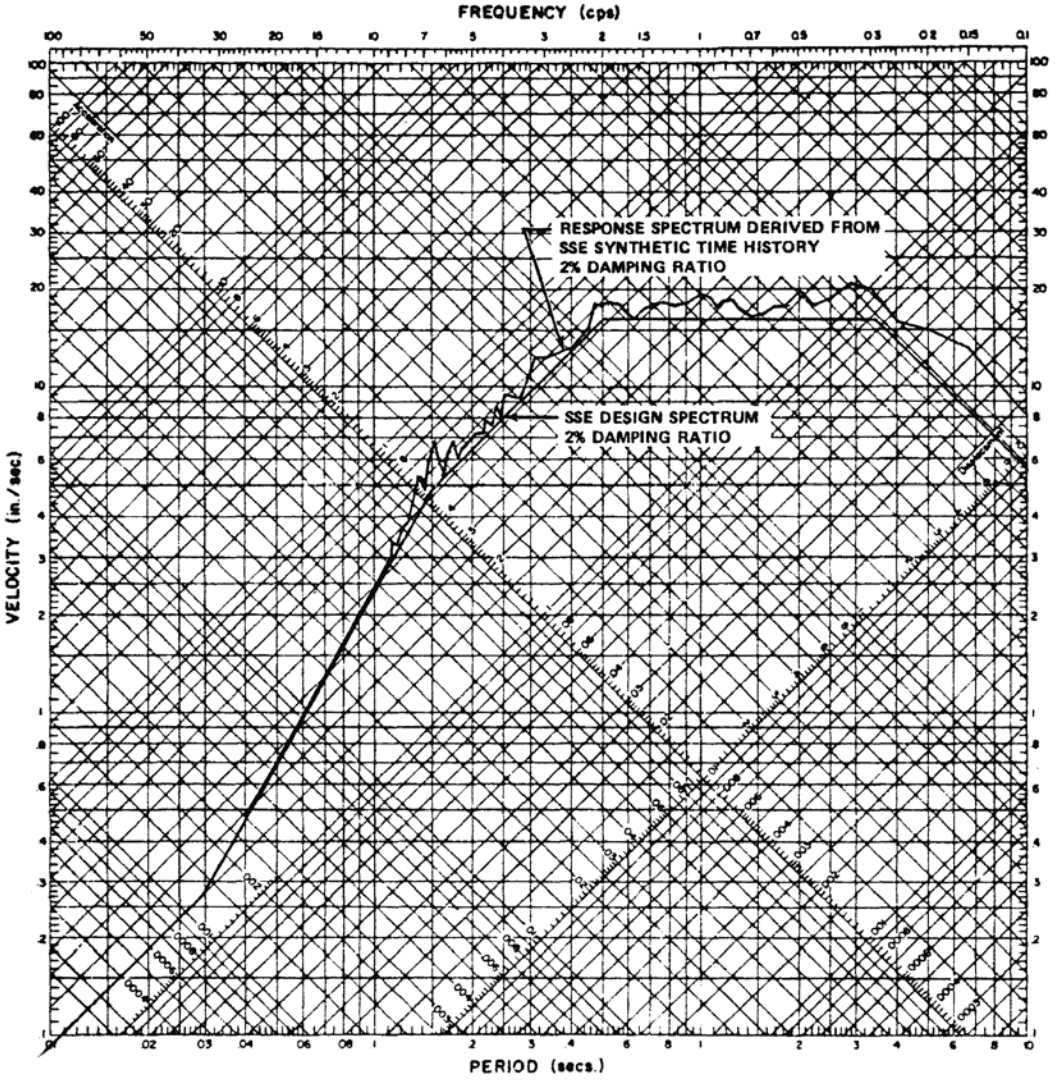
FIGURE 3.7-3



MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT	SSE DESIGN SPECTRUM AND RESPONSE SPECTRUM DERIVED FROM SSE SYNTHETIC TIME HISTORY BASED ON 5% DAMPING RATIO FIGURE 3.7-4



MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT	SSE DESIGN SPECTRUM AND RESPONSE SPECTRUM DERIVED FROM SSE SYNTHETIC TIME HISTORY BASED ON 3% DAMPING RATIO FIGURE 3.7-5



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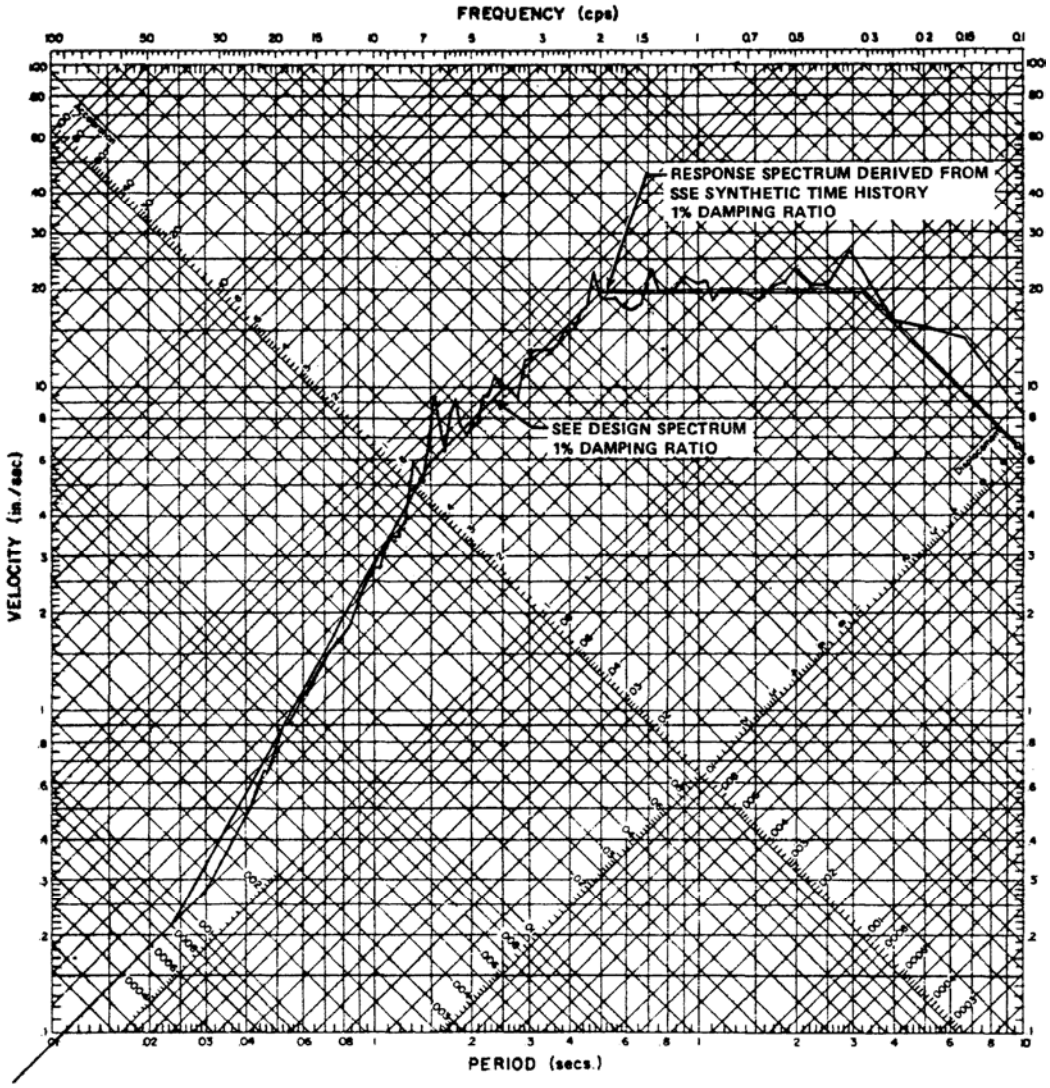
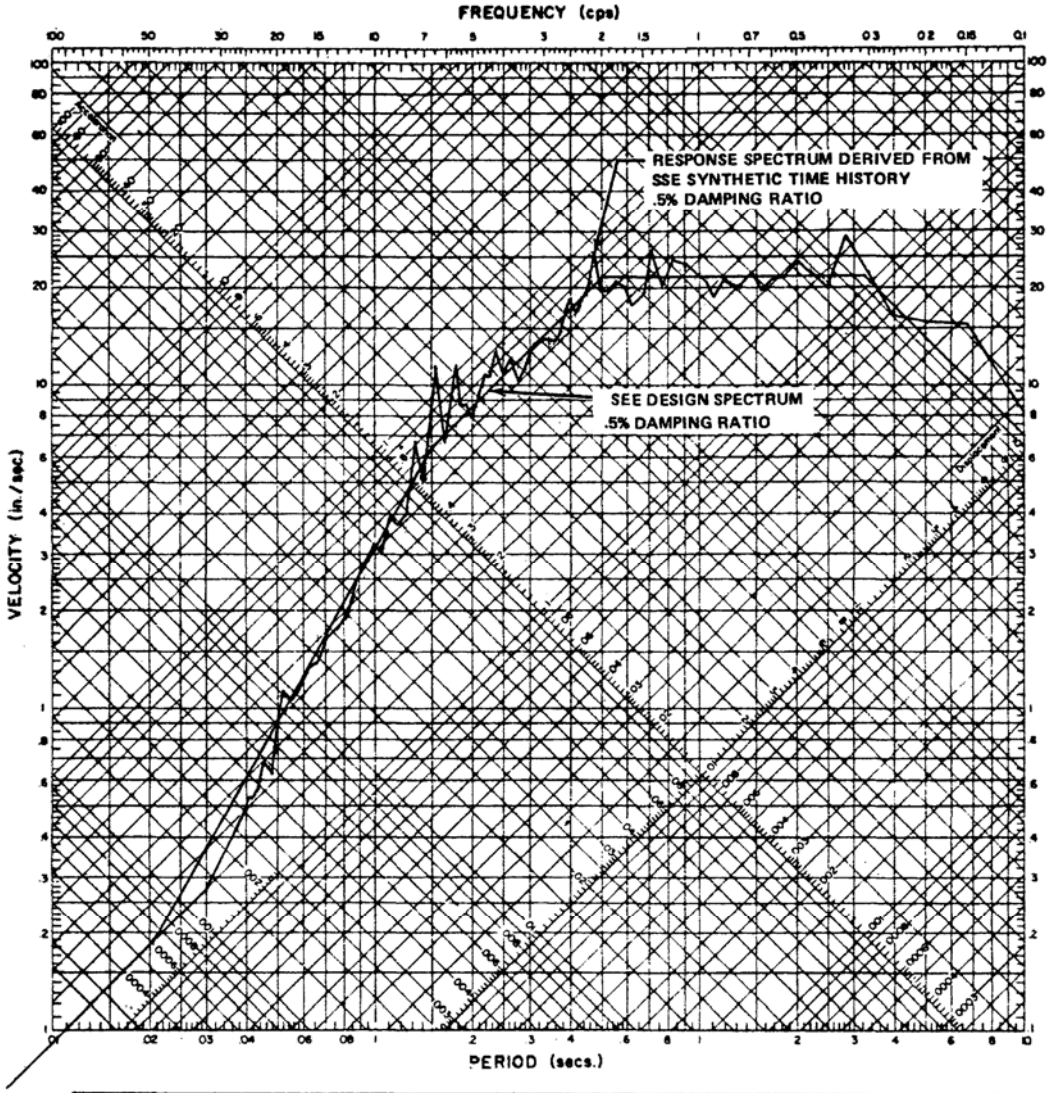
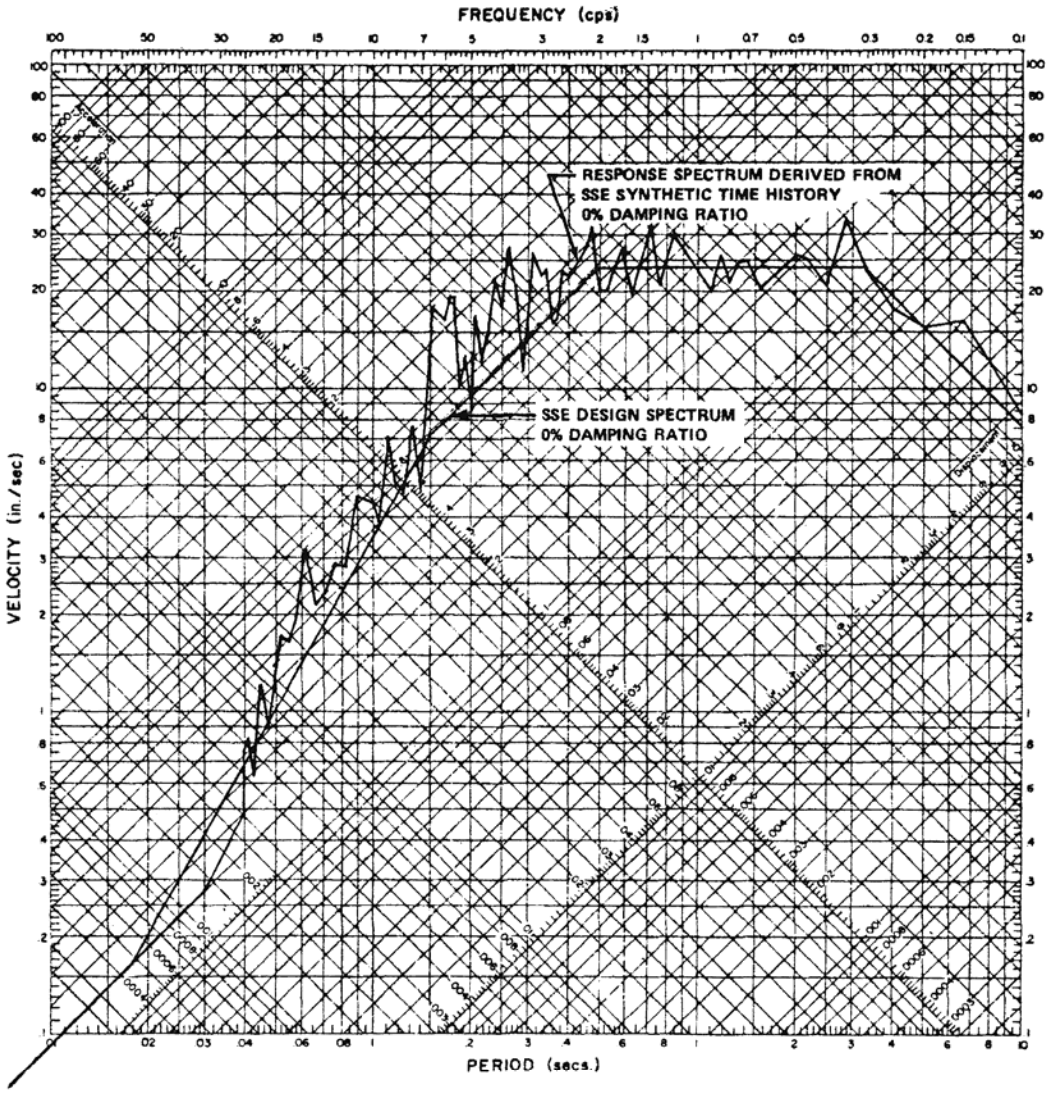


MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT	SSE DESIGN SPECTRUM AND RESPONSE SPECTRUM DERIVED FROM SSE SYNTHETIC TIME HISTORY BASED ON 1% DAMPING RATIO FIGURE 3.7-7



MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT	SSE DESIGN SPECTRUM AND RESPONSE SPECTRUM DERIVED FROM SSE SYNTHETIC TIME HISTORY BASED ON .5% DAMPING RATION FIGURE 3.7-8



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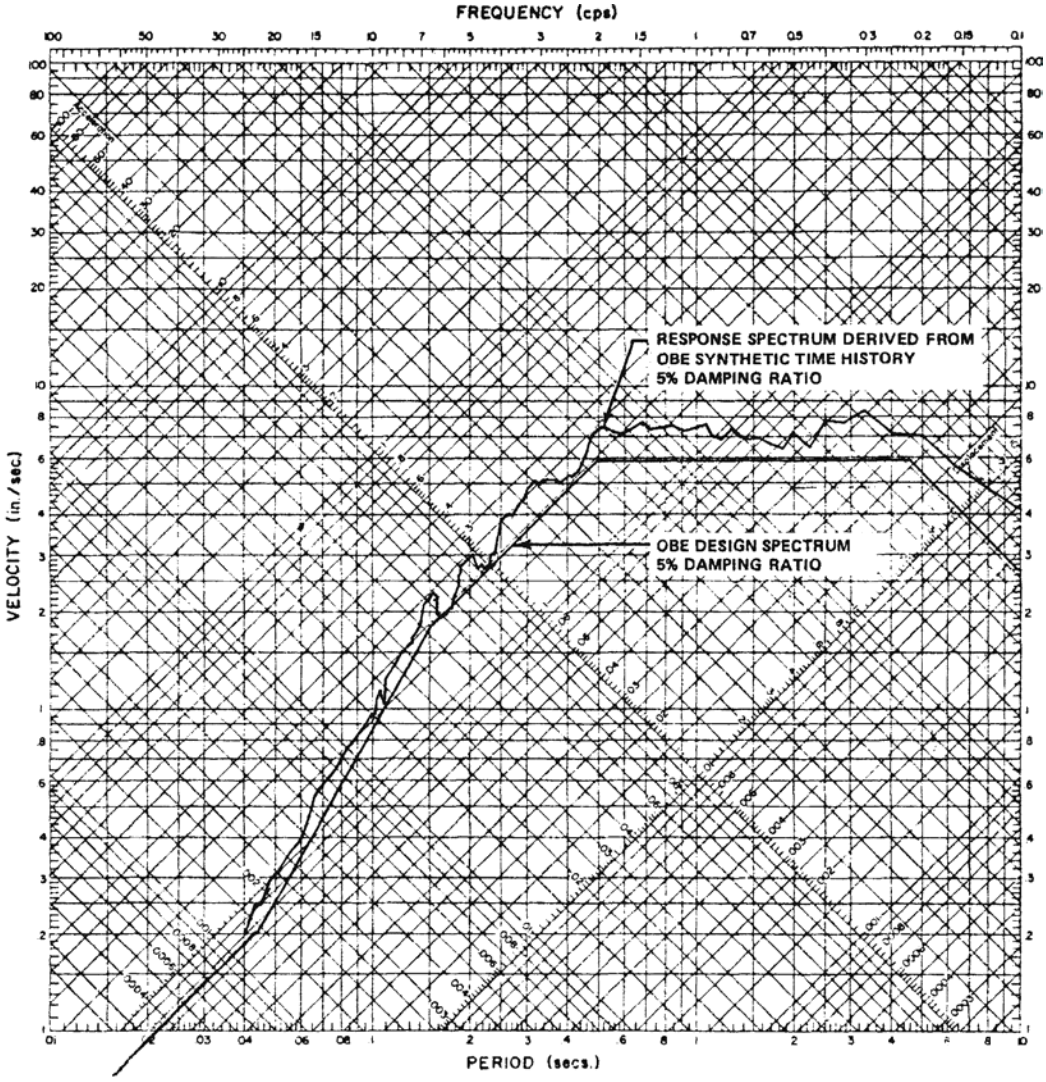

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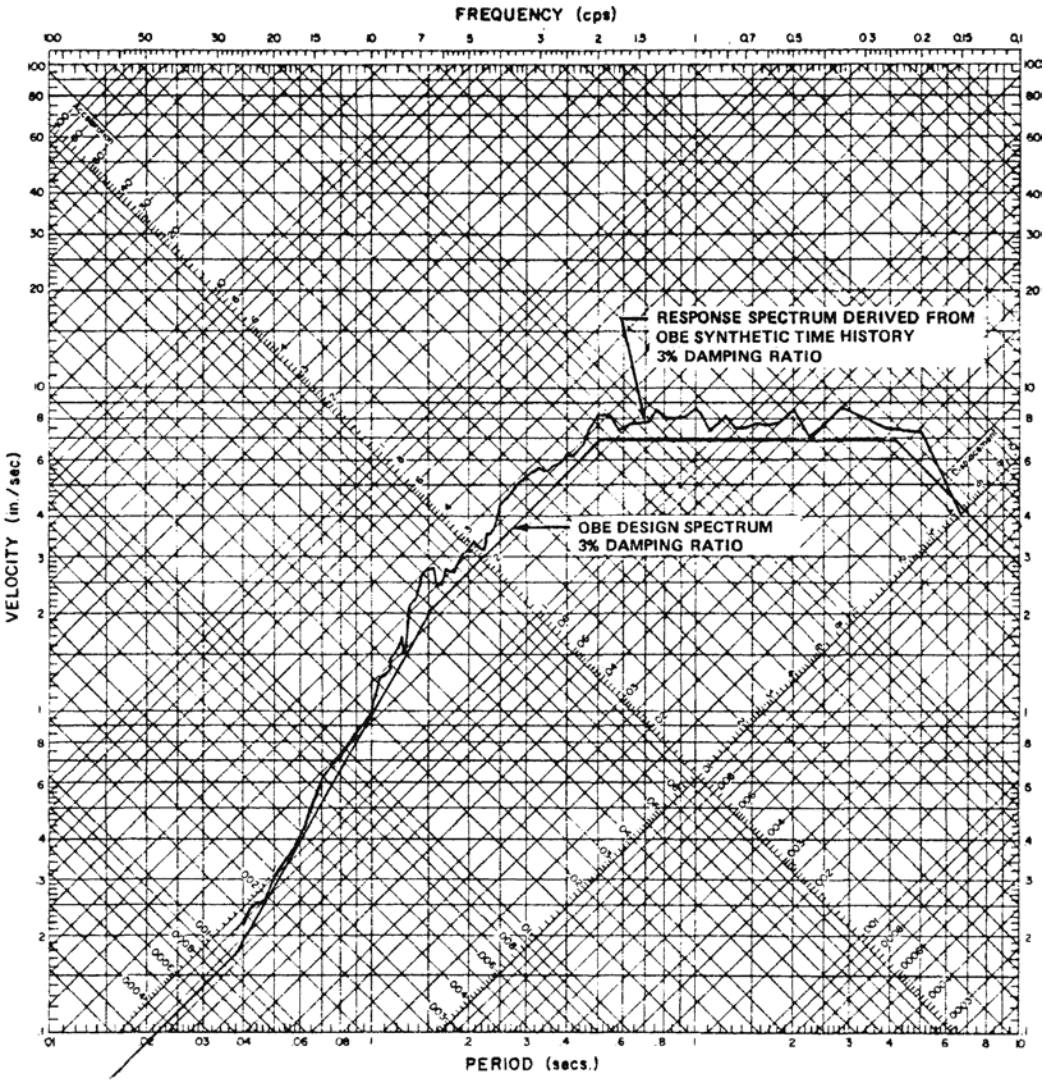
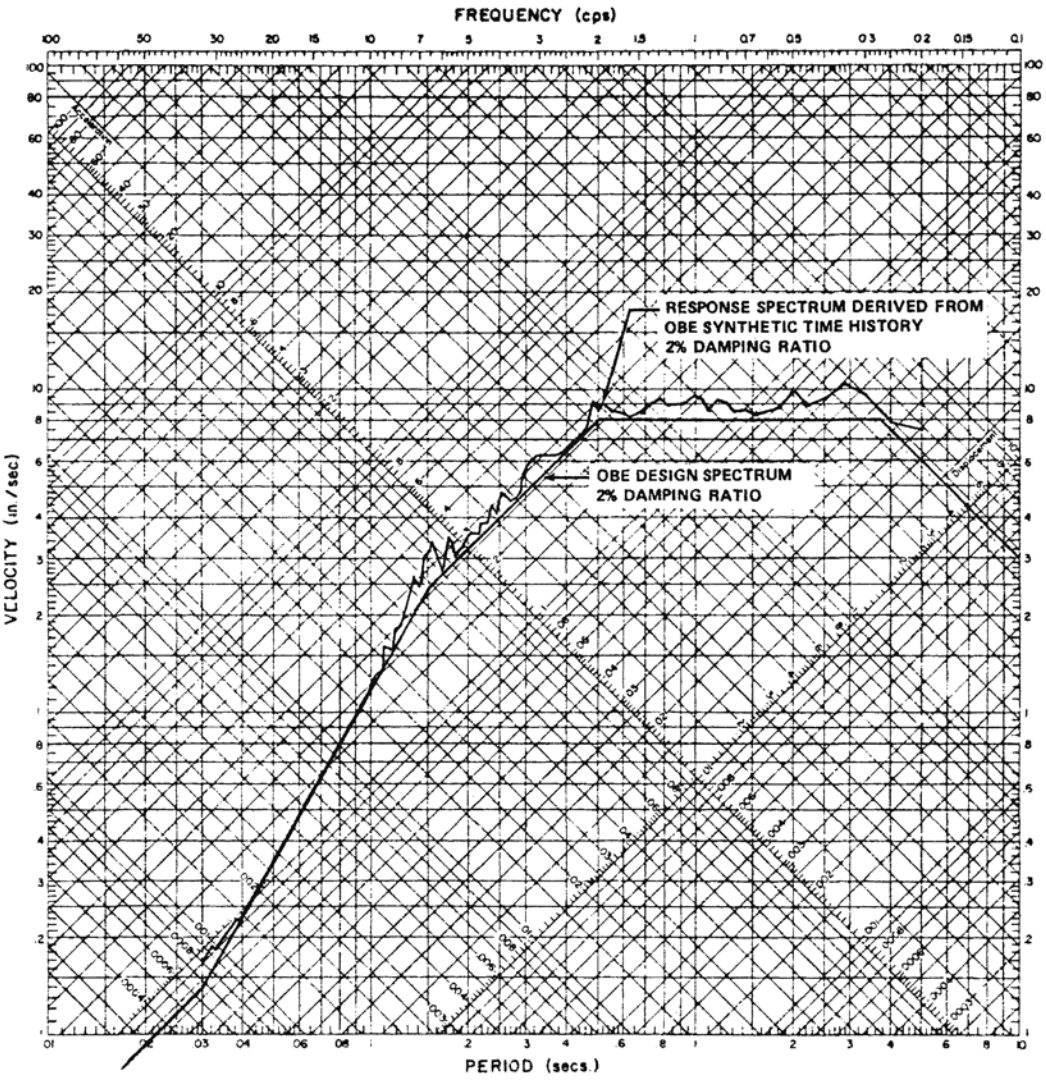


MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT	OBE DESIGN SPECTRUM AND RESPONSE SPECTRUM DERIVED FROM OBE SYNTHETIC TIME HISTORY BASED ON 3% DAMPING RATIO FIGURE 3.7-11



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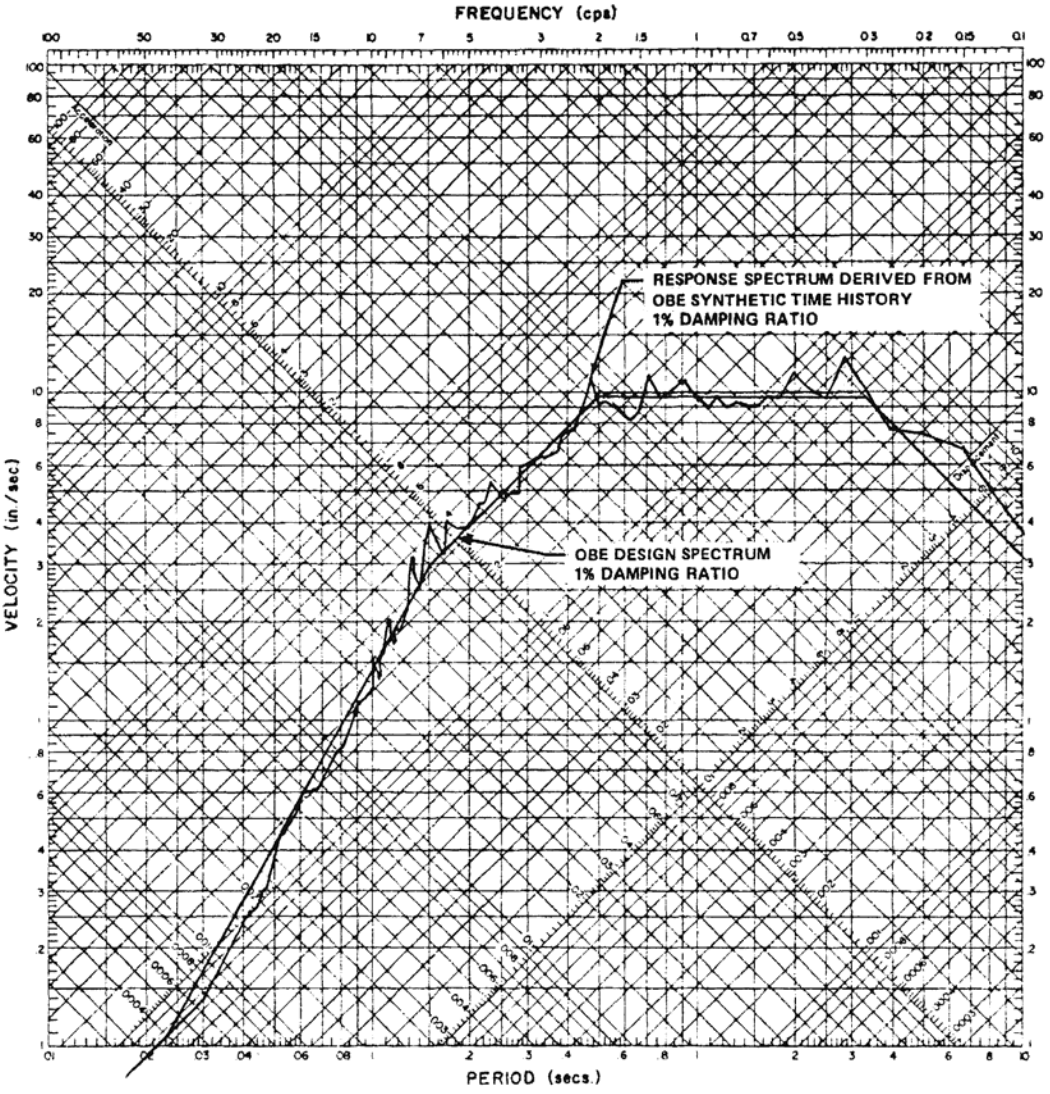

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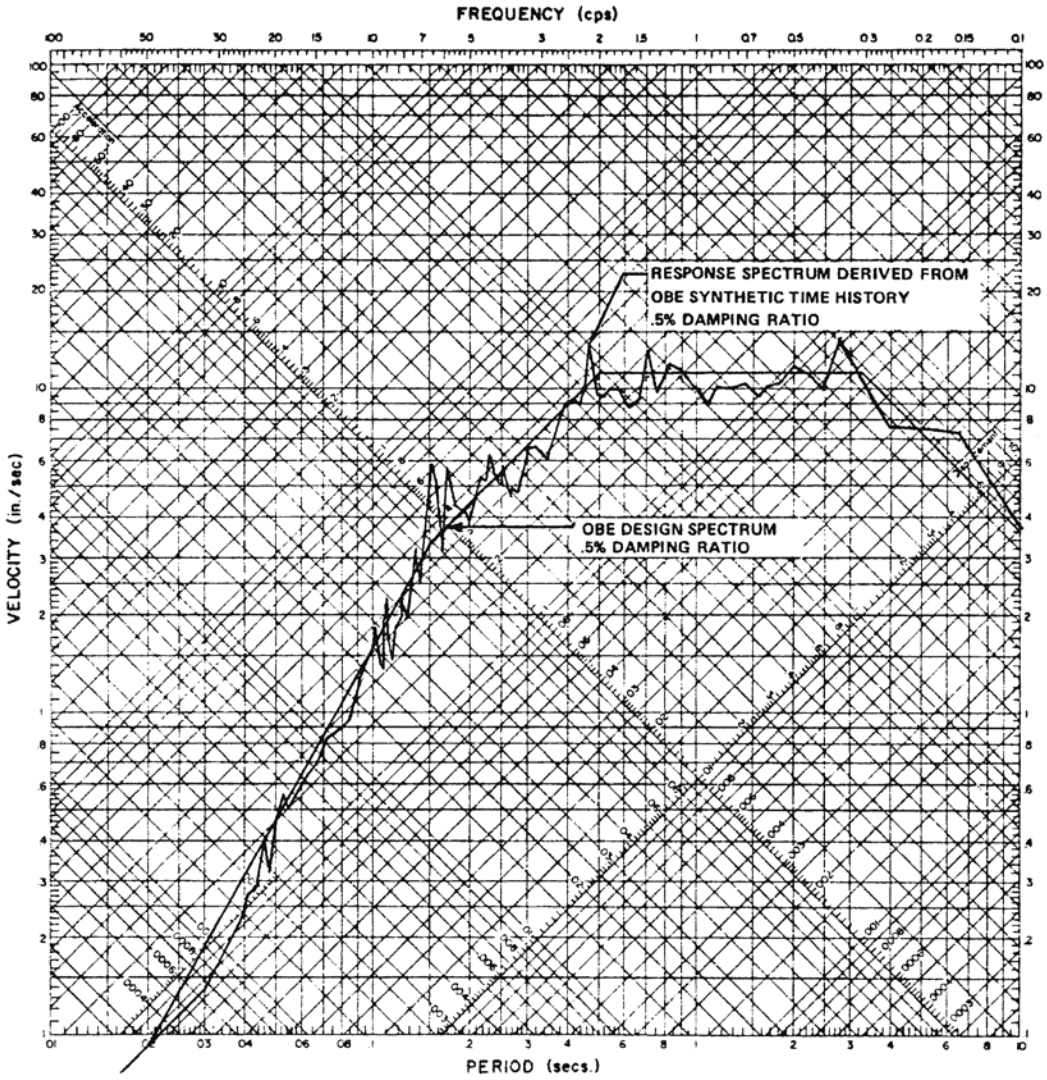

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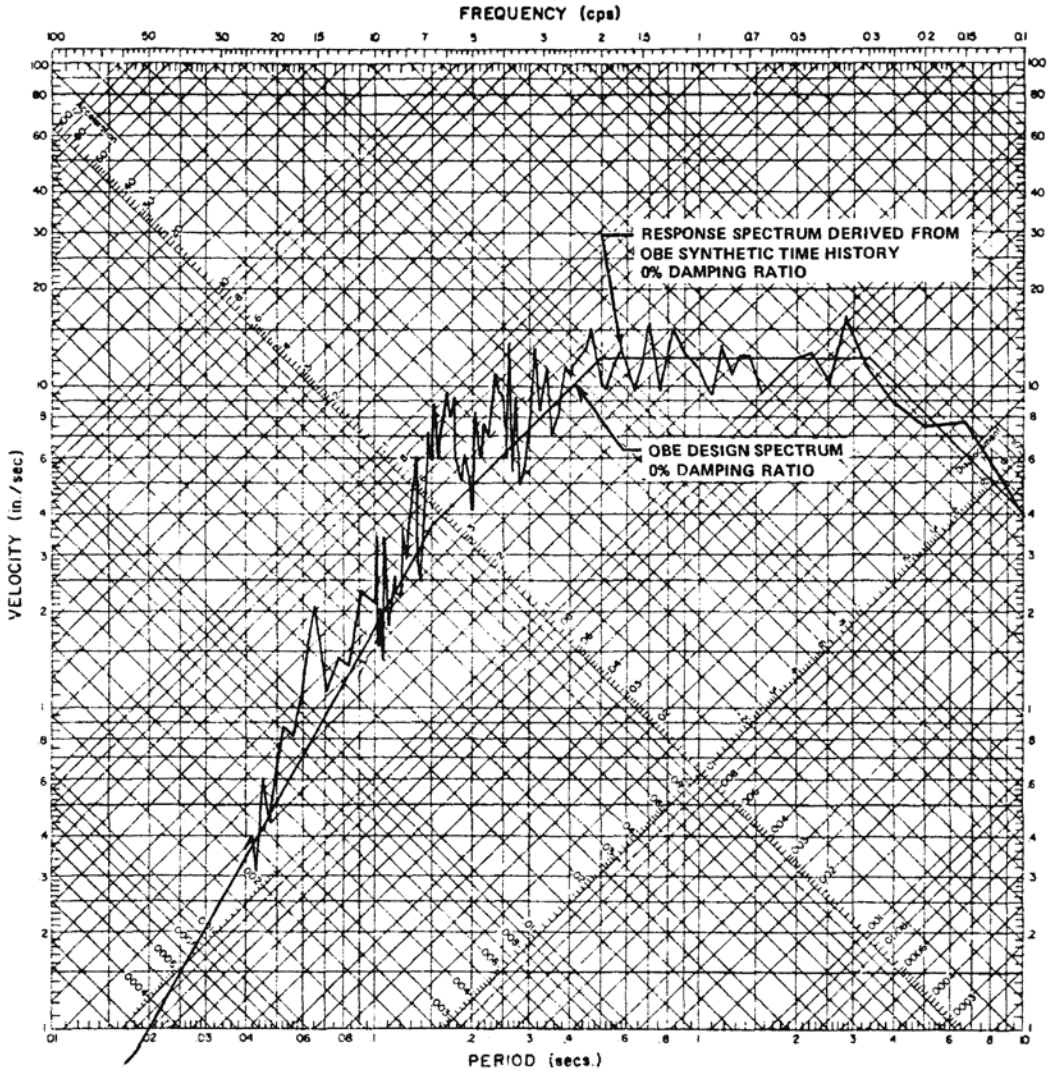

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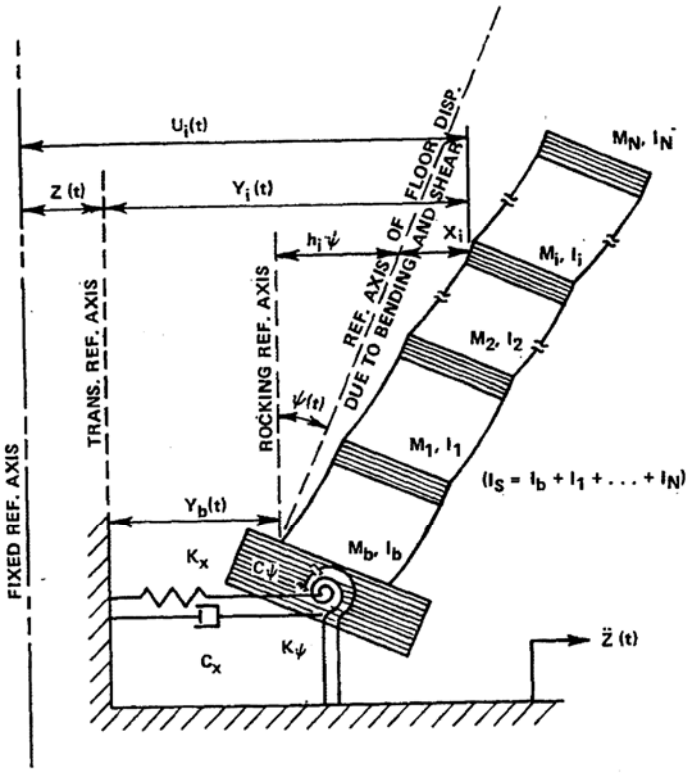

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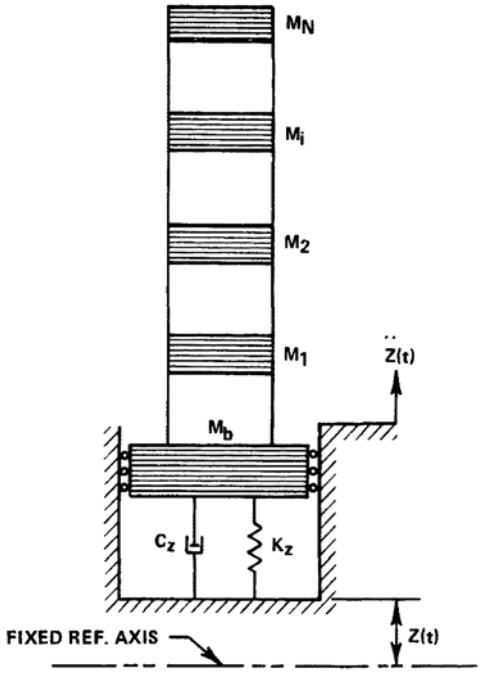
MATHEMATICAL MODEL
OF SOIL-STRUCTURE SYSTEM
FOR HORIZONTAL INPUT MOTION

FIGURE 3.7-16



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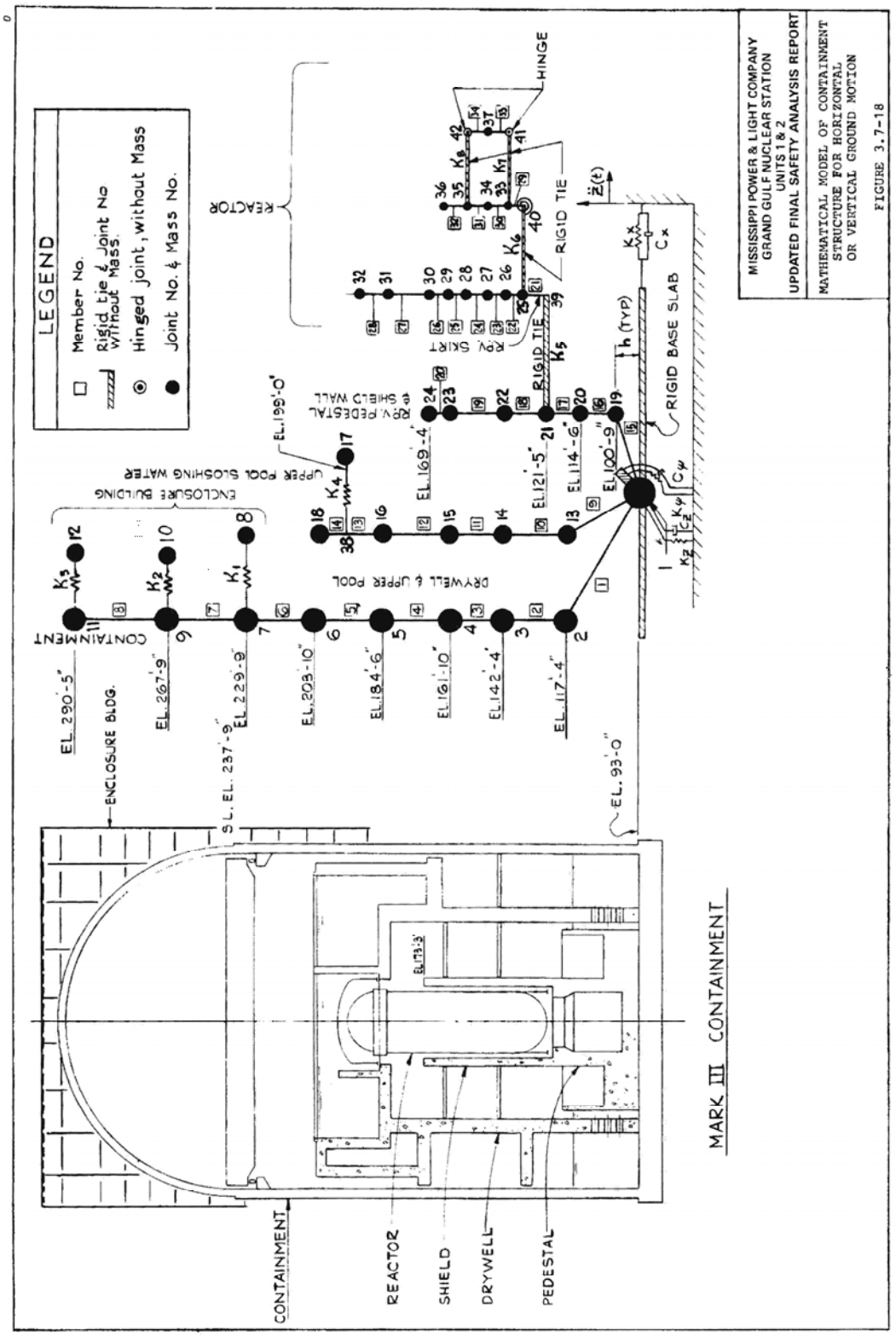
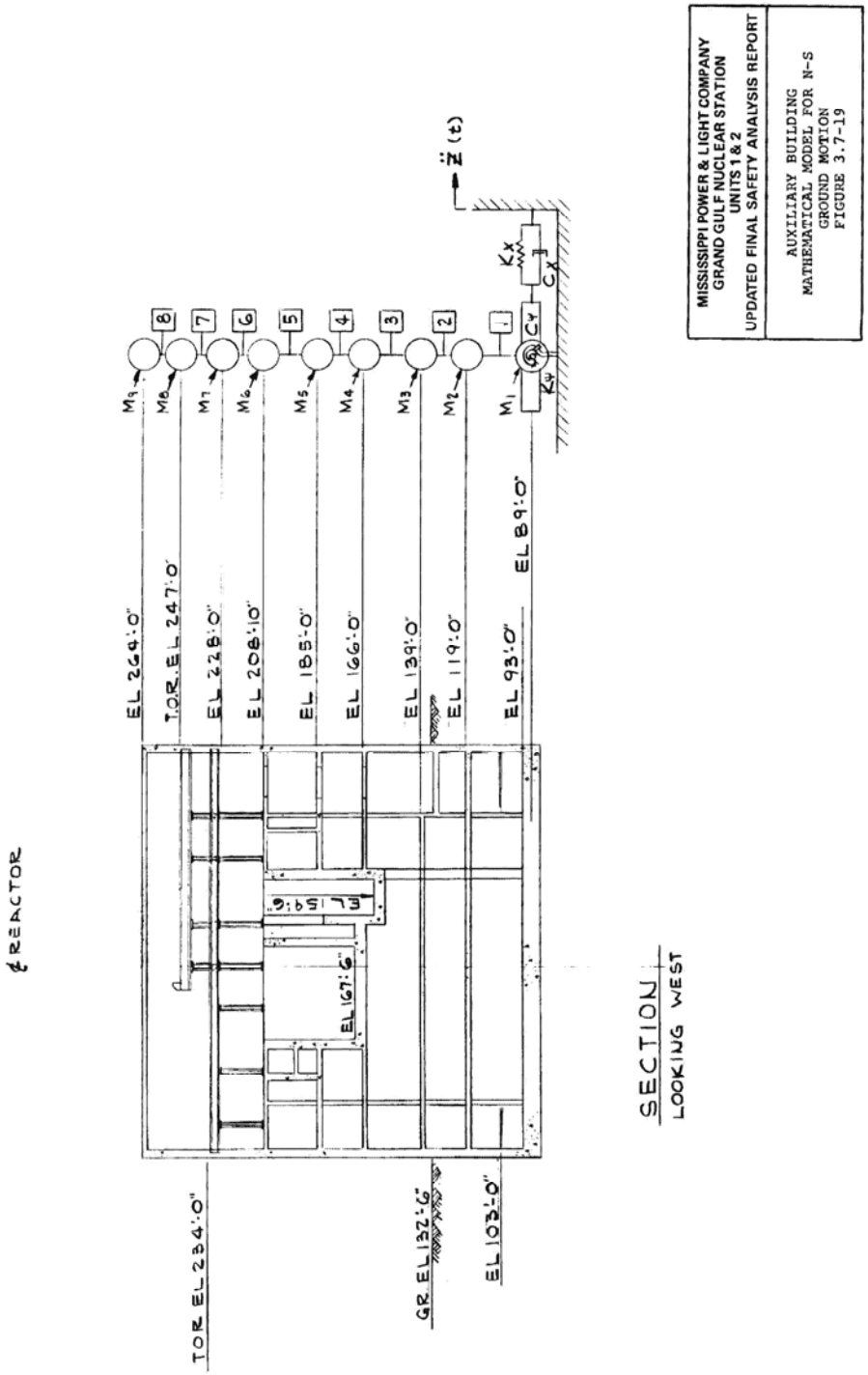
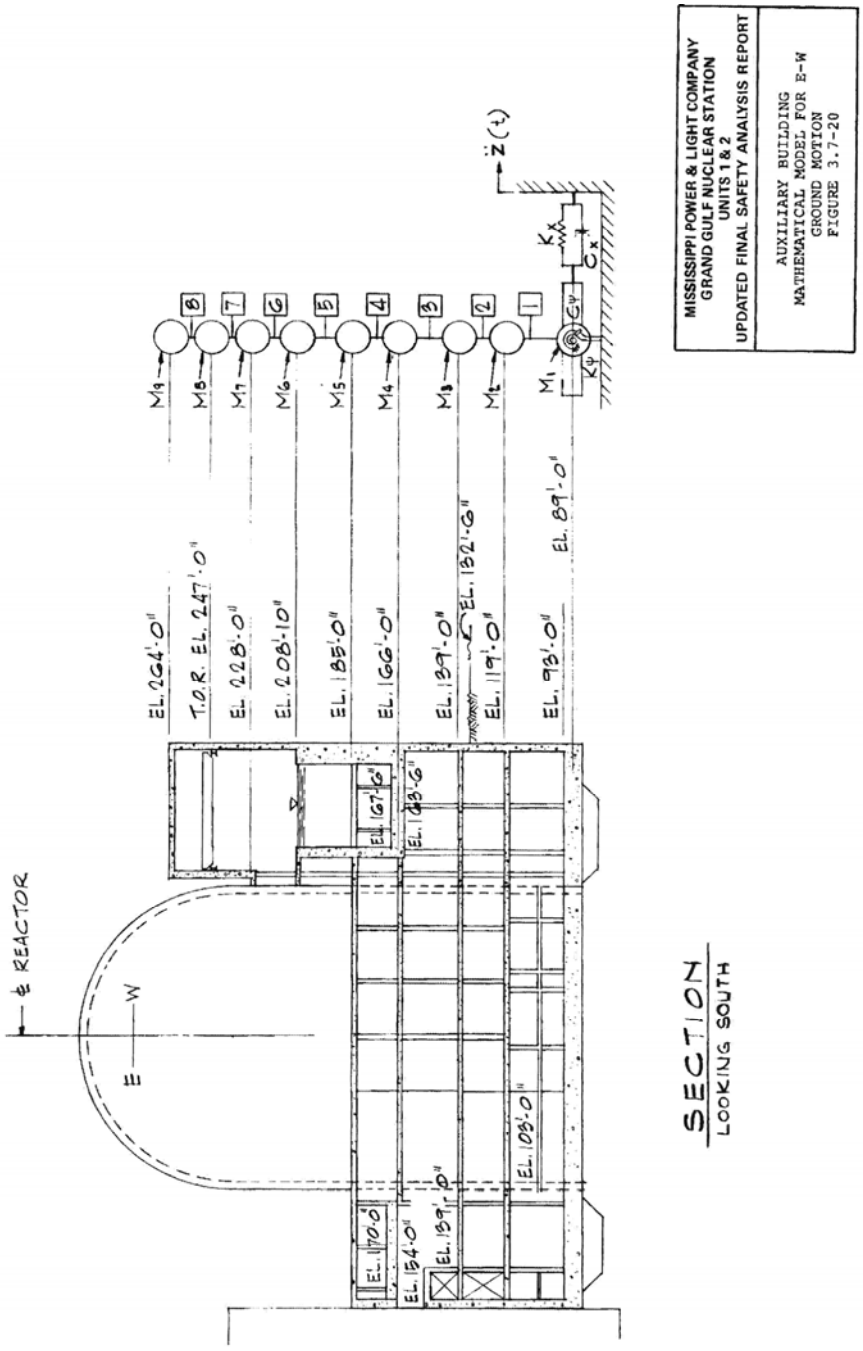
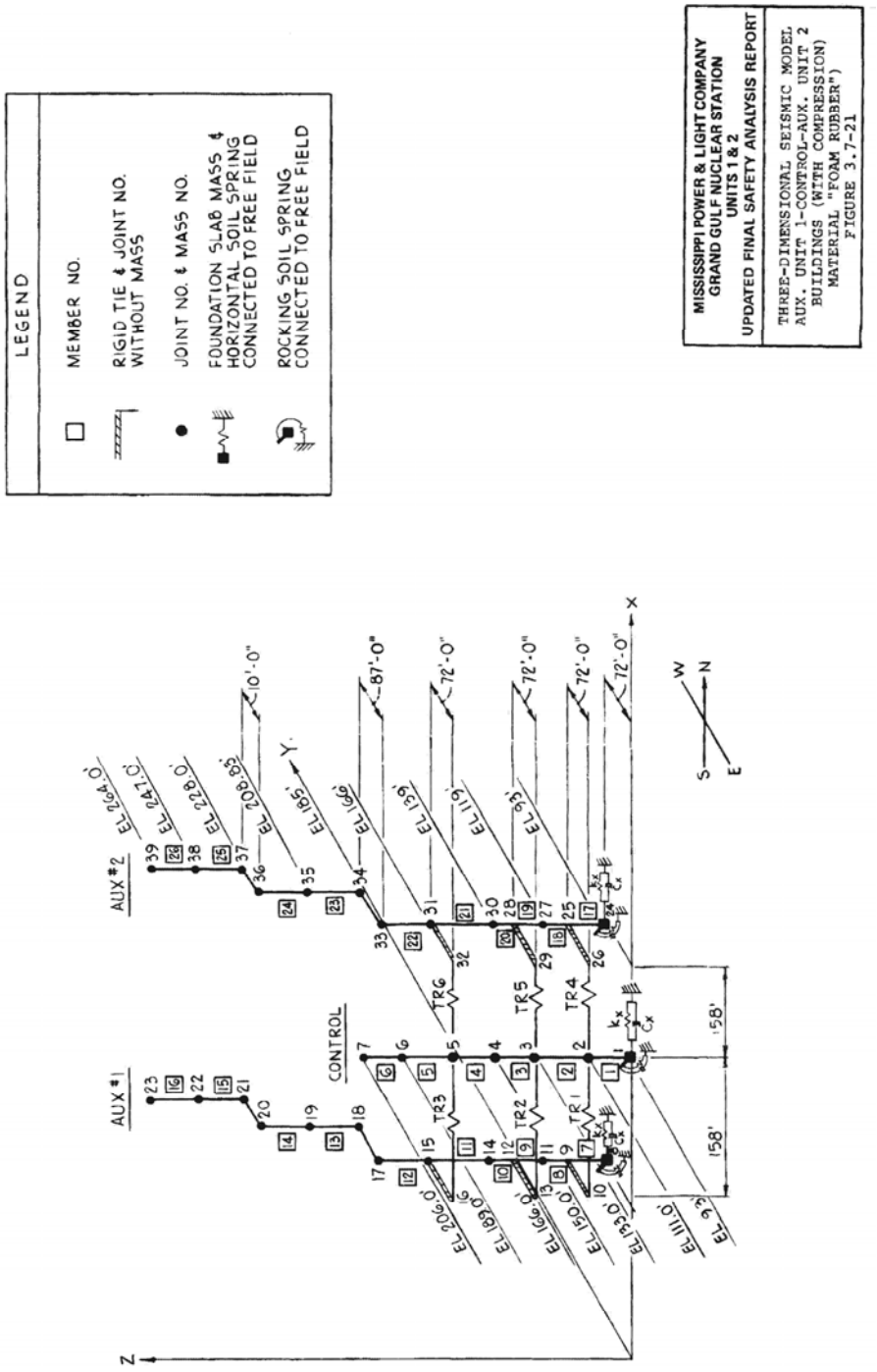
MATHEMATICAL MODEL
SOIL-STRUCTURE LUMPED MASS SYSTEM
FOR VERTICAL INPUT MOTION

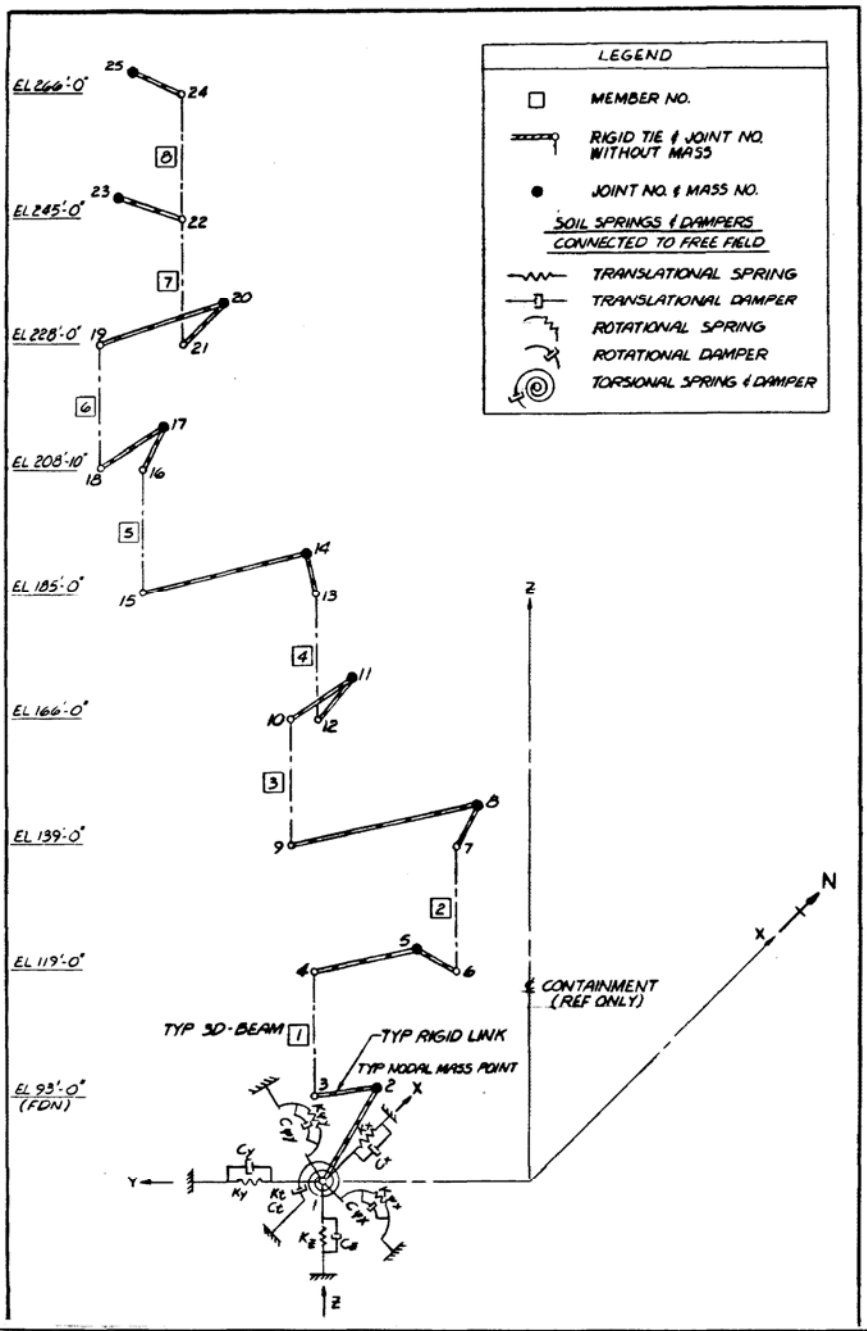
FIGURE 3.7-17


LEGEND
Member No.
Rigid tie & Joint No without Mass.
Hinged joint, without Mass
Joint No. & Mass No.
REACTOR
HINGE
RIGID TIE
RIGID BASE SLAB
RPV PEDESTAL & SHIELD WALL
UPPER POOL SLOSHING WATER
ENCLOSURE BUILDING
DRYWELL & UPPER POOL
CONTAINMENT
ENCLOSURE BLDG.
SL. EL. 237'-9"
EL. 290'-5"
EL. 267'-9"
EL. 229'-9"
EL. 209'-10"
EL. 184'-6"
EL. 161'-10"
EL. 142'-4"
EL. 117'-4"
EL. 93'-0"
EL. 196'-0"
EL. 169'-4"
EL. 121'-5"
EL. 114'-6"
EL. 100'-9"
MARK III CONTAINMENT
CONTAINMENT
REACTOR
SHIELD
DRYWELL
PEDESTAL
MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT
MATHEMATICAL MODEL OF CONTAINMENT STRUCTURE FOR HORIZONTAL OR VERTICAL GROUND MOTION
FIGURE 3.7-18

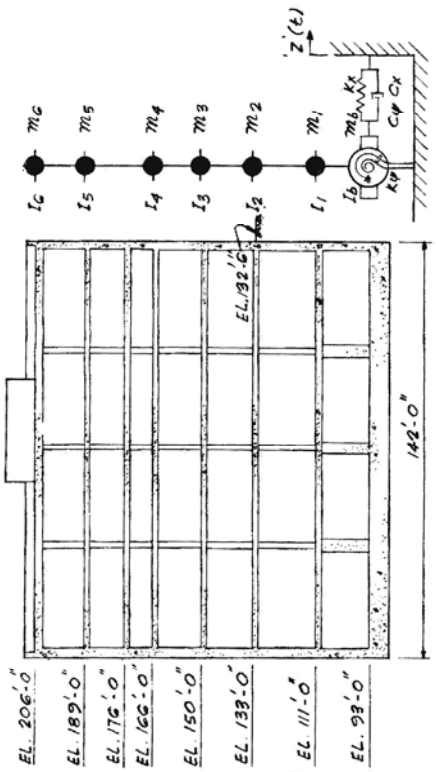

MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT
AUXILIARY BUILDING
MATHEMATICAL MODEL FOR N-S
GROUND MOTION
FIGURE 3.7-19
z̈ (t)
Kx
Cx
Cy
Ky
M9
M8
M7
M6
M5
M4
M3
M2
M1
EL 264'-0"
TOR EL 247'-0"
EL 228'-0"
EL 208'-10"
EL 185'-0"
EL 166'-0"
EL 139'-0"
EL 119'-0"
EL 93'-0"
EL 89'-0"
EL 67'-6"
TOR EL 234'-0"
GR EL 132'-6"
EL 103'-0"
℄ REACTOR
SECTION
LOOKING WEST


MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT
AUXILIARY BUILDING
MATHEMATICAL MODEL FOR E-W
GROUND MOTION
FIGURE 3.7-20
SECTION
LOOKING SOUTH
℄ REACTOR
E
W
EL. 264'-0"
T.O.R. EL. 247'-0"
EL. 228'-0"
EL. 208'-10"
EL. 185'-0"
EL. 166'-0"
EL. 139'-0"
EL. 119'-0"
EL. 95'-0"
EL. 89'-0"
EL. 152'-6"
EL. 167'-6"
EL. 143'-6"
EL. 108'-0"
EL. 170'-0"
EL. 154'-0"
EL. 139'-0"


LEGEND
MEMBER NO.
RIGID TIE & JOINT NO. WITHOUT MASS
JOINT NO. & MASS NO.
FOUNDATION SLAB MASS & HORIZONTAL SOIL SPRING CONNECTED TO FREE FIELD
ROCKING SOIL SPRING CONNECTED TO FREE FIELD
MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT
THREE-DIMENSIONAL SEISMIC MODEL AUX. UNIT 1-CONTROL-AUX. UNIT 2 BUILDINGS (WITH COMPRESSION) MATERIAL "FOAM RUBBER")
FIGURE 3.7-21
AUX #2
AUX #1
CONTROL
TR1
TR2
TR3
TR4
TR5
TR6
EL 224.0'
EL 217.0'
EL 208.83'
EL 185'
EL 160'
EL 119'
EL 93'
EL 206.0'
EL 189.0'
EL 169.0'
EL 151.0'
EL 133.0'
EL 111.0'
EL 93'
10'-0"
87'-0"
72'-0"
72'-0"
72'-0"
72'-0"
158'
158'
X
Y
Z
W
N
S
E



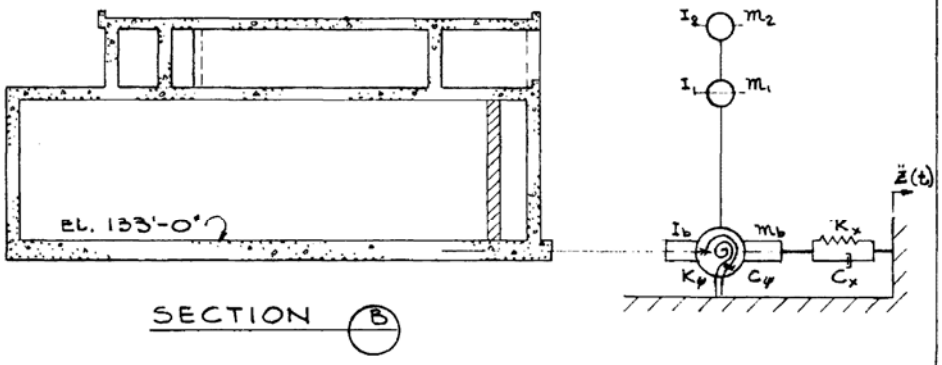
MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT	AUXILIARY BUILDING THREE DIMENSIONAL SEISMIC MODEL FIGURE 3.7-21a



MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT

CONTROL BUILDING
MATHEMATICAL MODEL

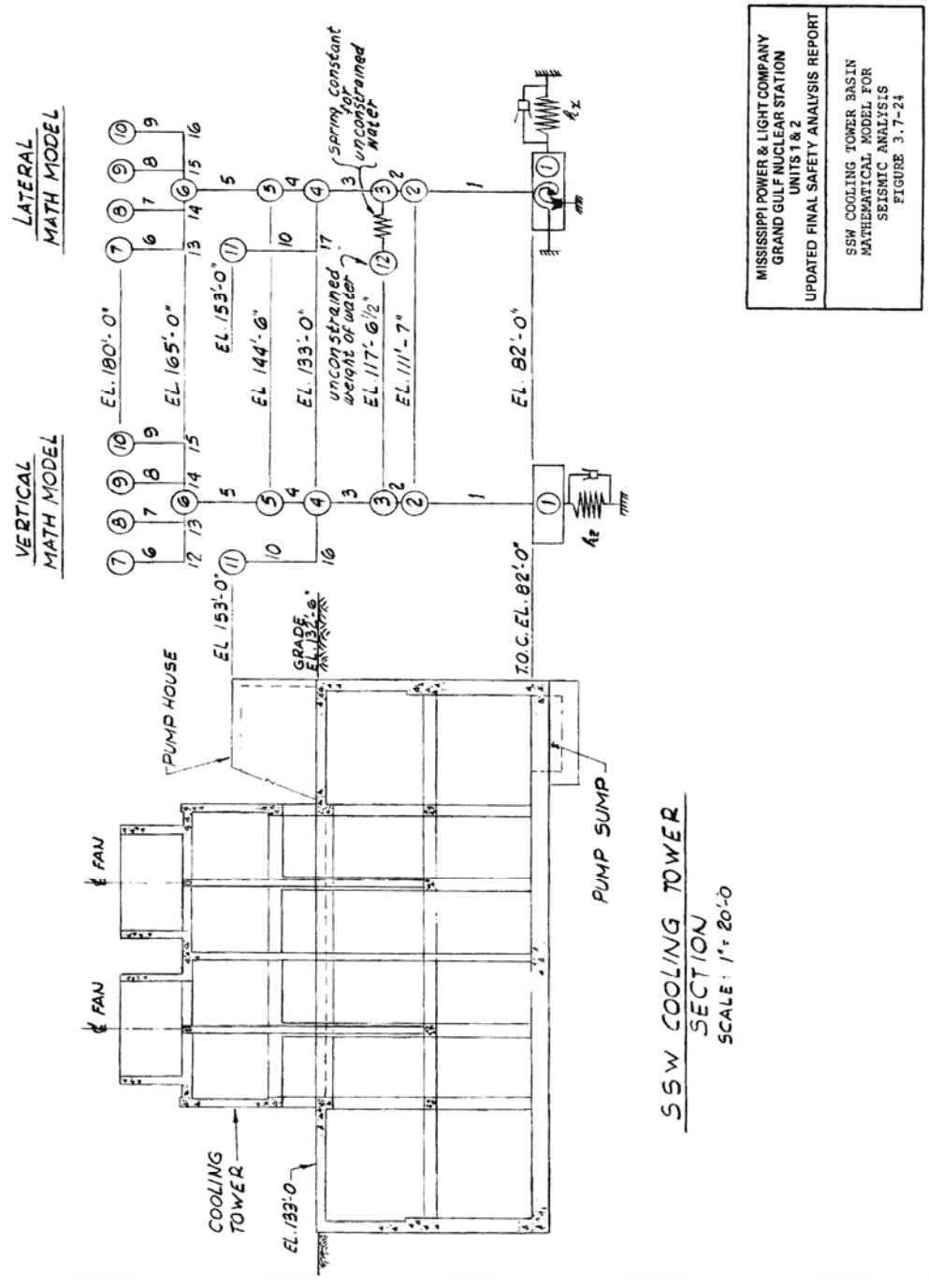
FIGURE 3.7-22

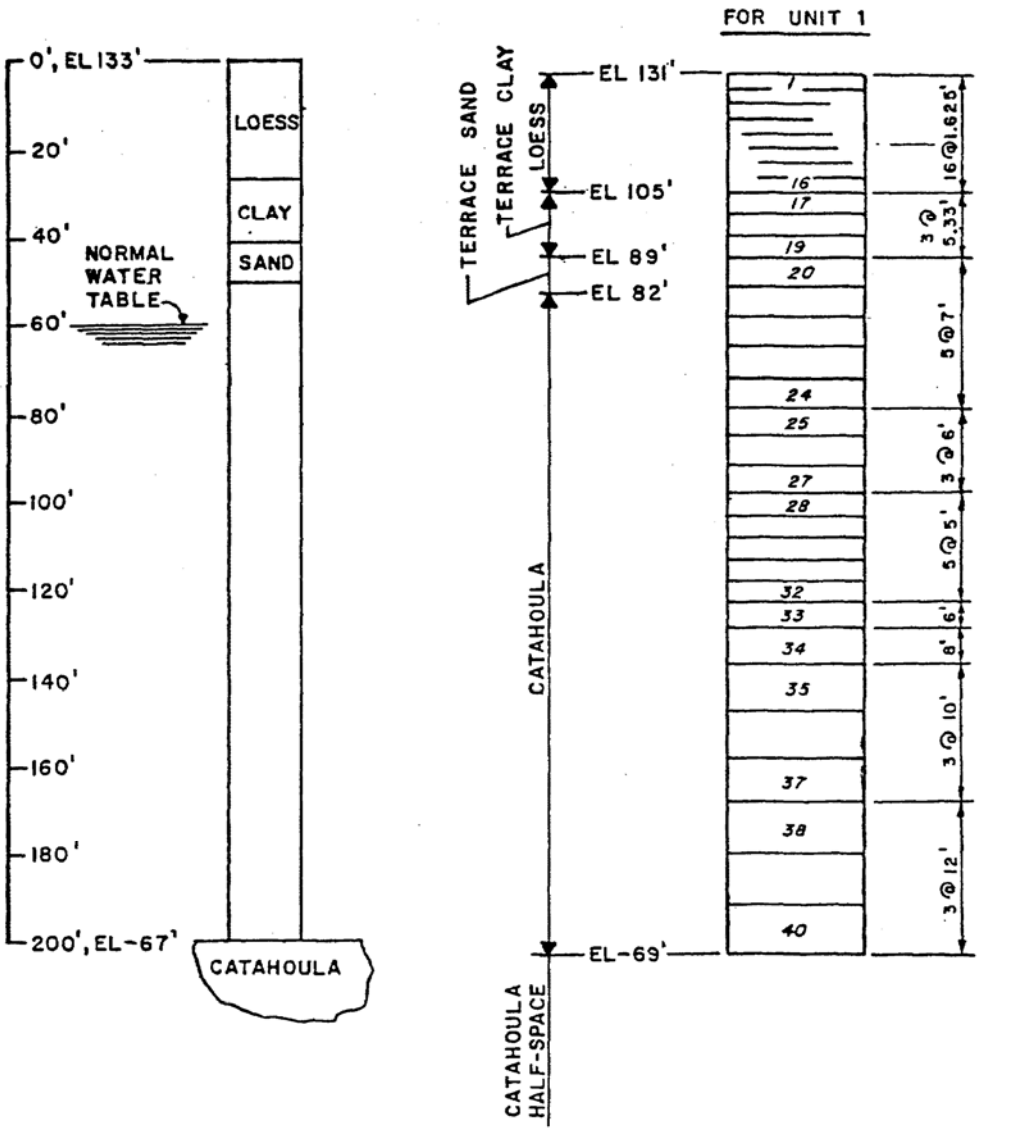


MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT

DIESEL GENERATOR BUILDING
MATHEMATICAL MODEL

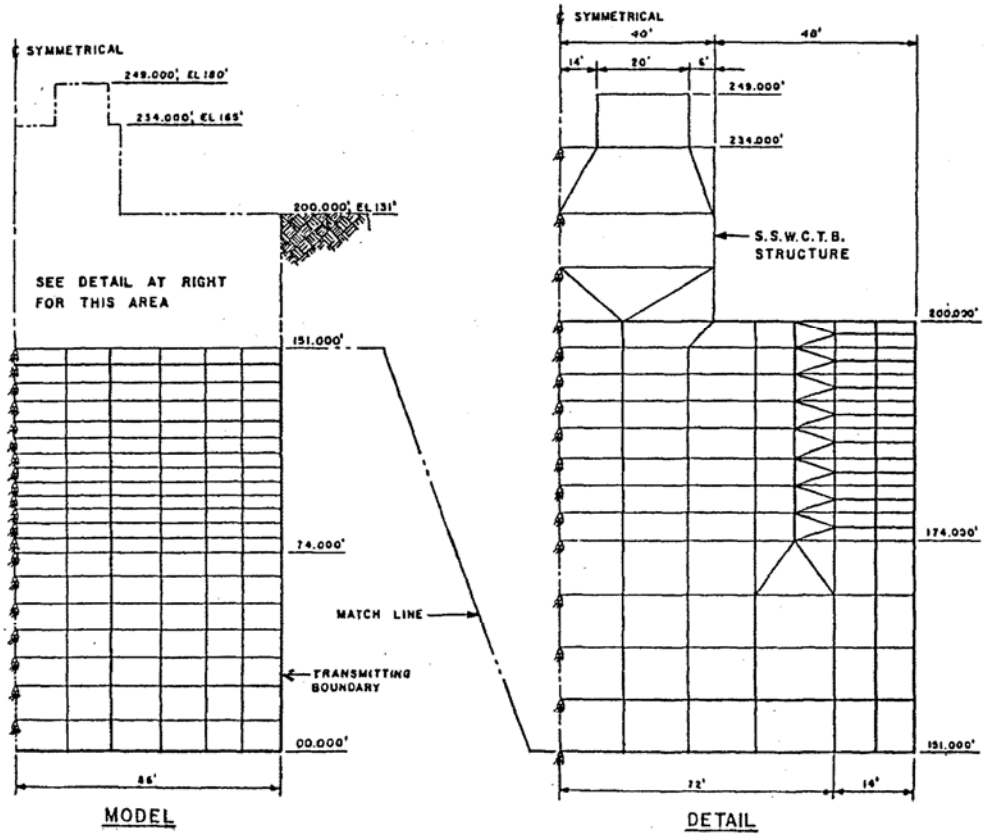
FIGURE 3.7-23


LATERAL MATH MODEL
VERTICAL MATH MODEL
EL. 180'-0"
EL. 165'-0"
EL. 153'-0"
EL. 144'-6"
EL. 133'-0"
EL. 117'-6 1/2"
EL. 111'-7"
EL. 82'-0"
spring constant for unconstrained water
unconstrained weight of water
EL. 153'-0"
GRADE EL. 136'-6"
T.O.C. EL. 82'-0"
PUMP HOUSE
PUMP SUMP
FAN
FAN
COOLING TOWER
EL. 133'-0
SSW COOLING TOWER
SECTION
SCALE: 1" = 20'-0
MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT
SSW COOLING TOWER BASIN MATHEMATICAL MODEL FOR SEISMIC ANALYSIS
FIGURE 3.7-24



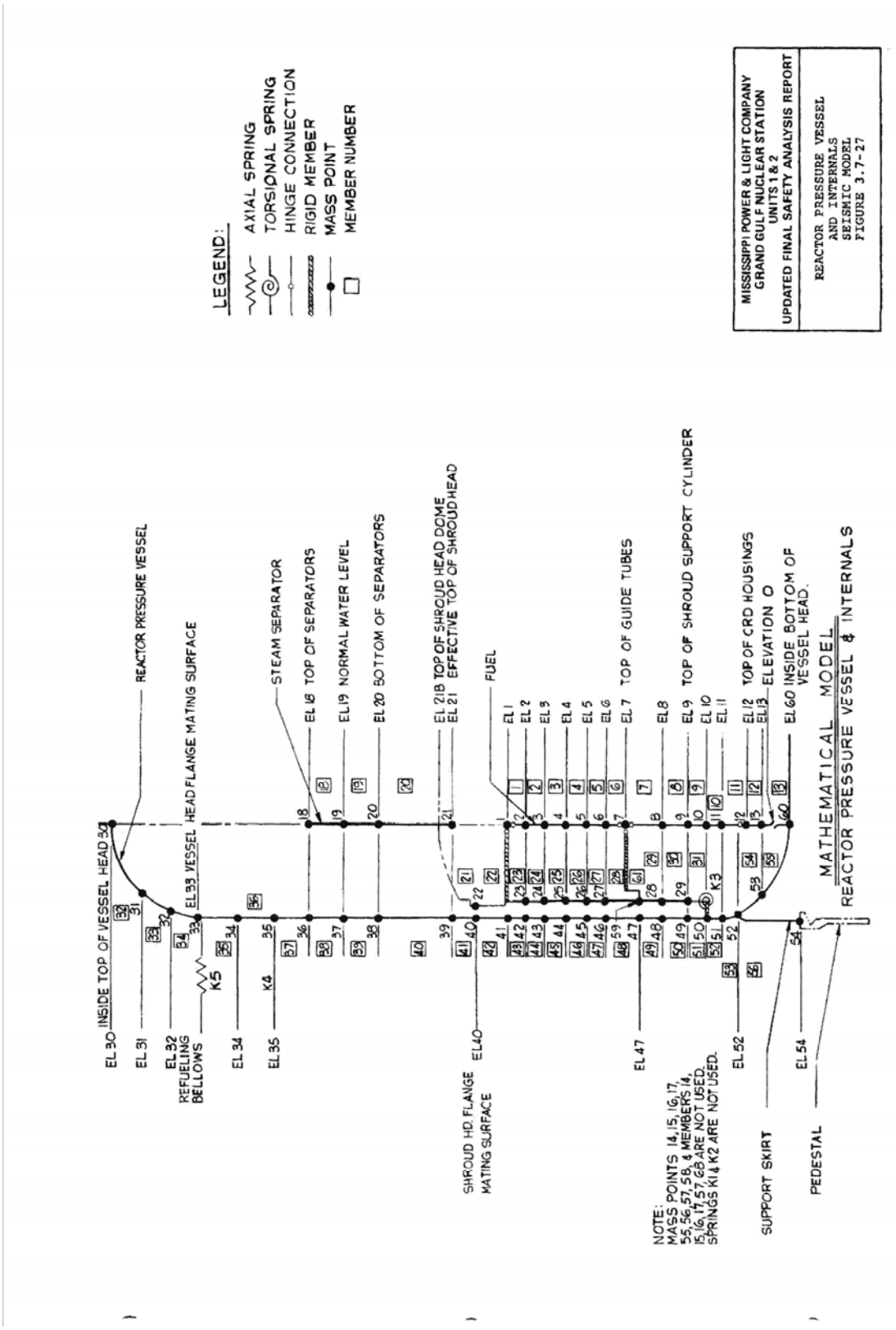
SYSTEM ENERGY RESOURCES, INC.
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT

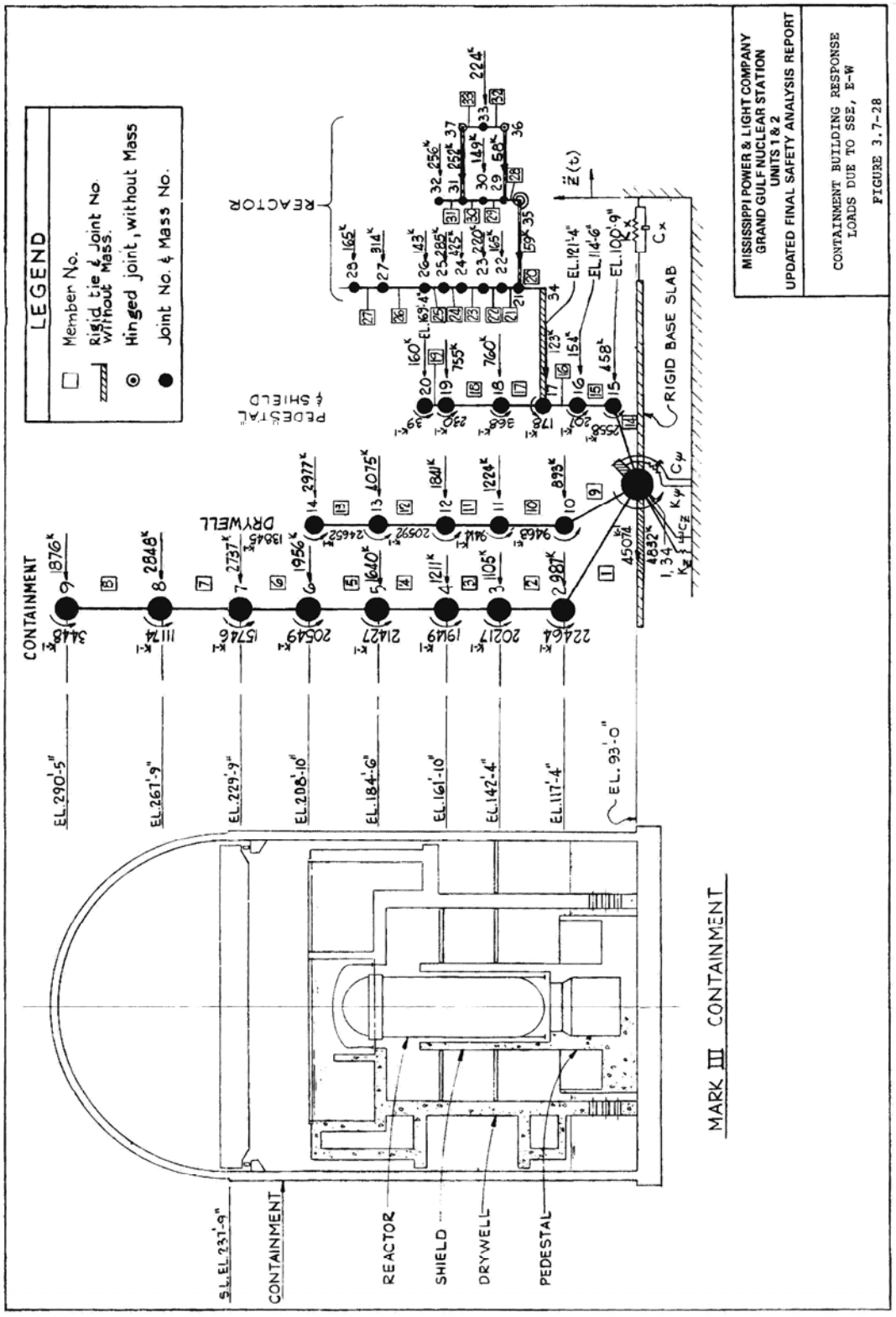
MATHEMATICAL MODEL OF SOIL COLUMN FOR GENERATION OF BASE ROCK MOTION AT 151 FT BELOW TOP OF CATAHOULA
FIGURE 3.7-25

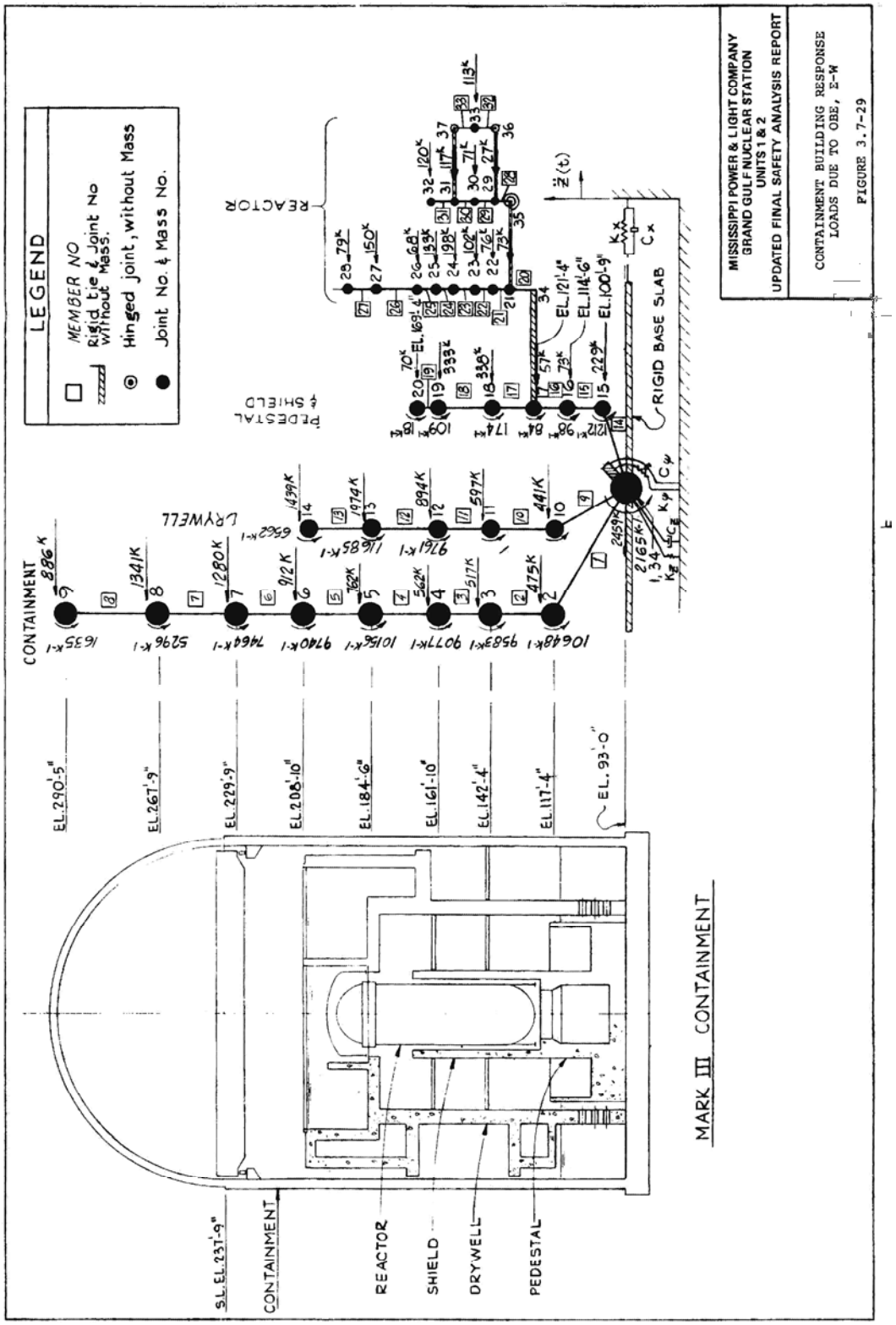


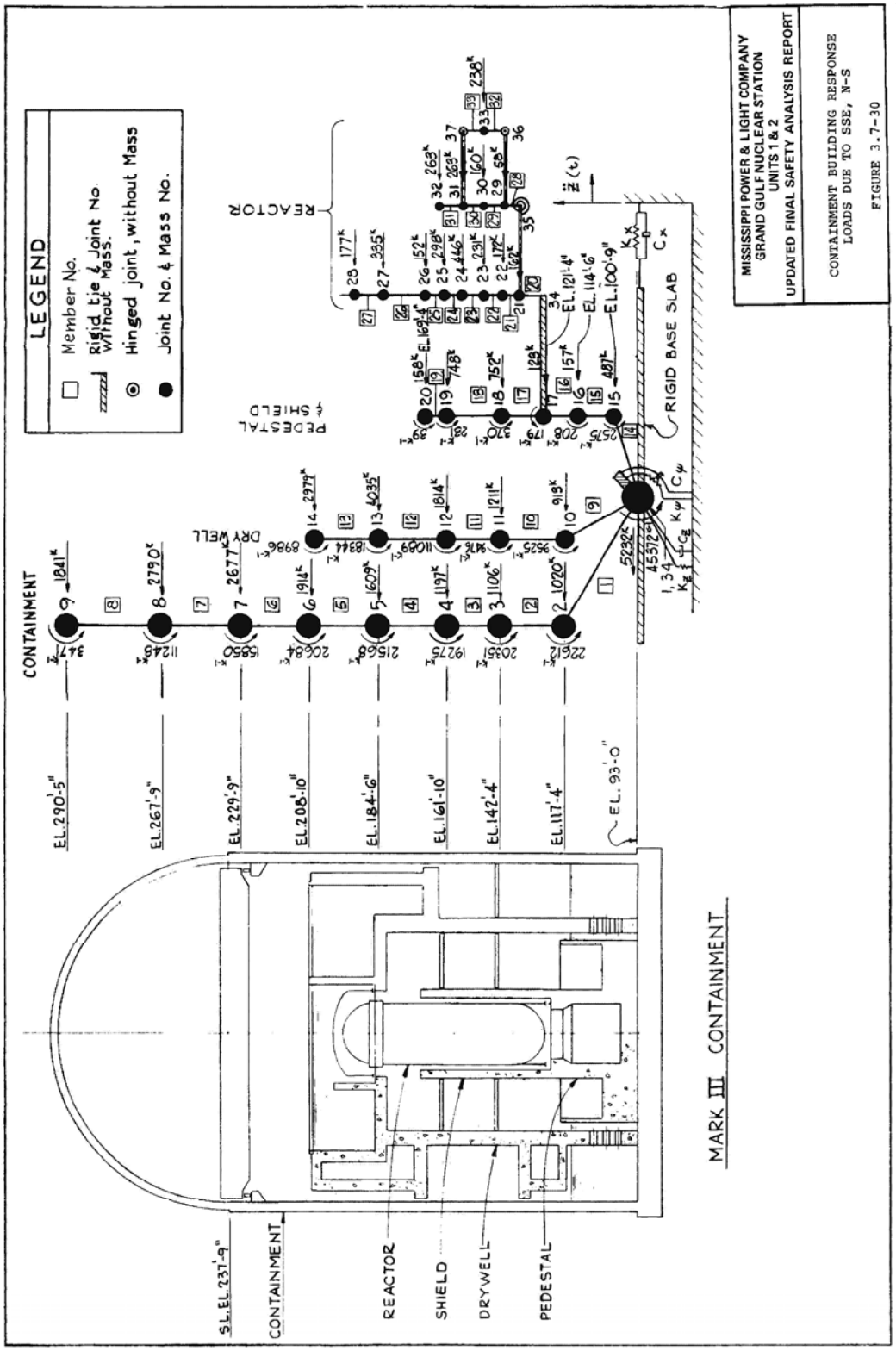
SYSTEM ENERGY RESOURCES, INC.
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT

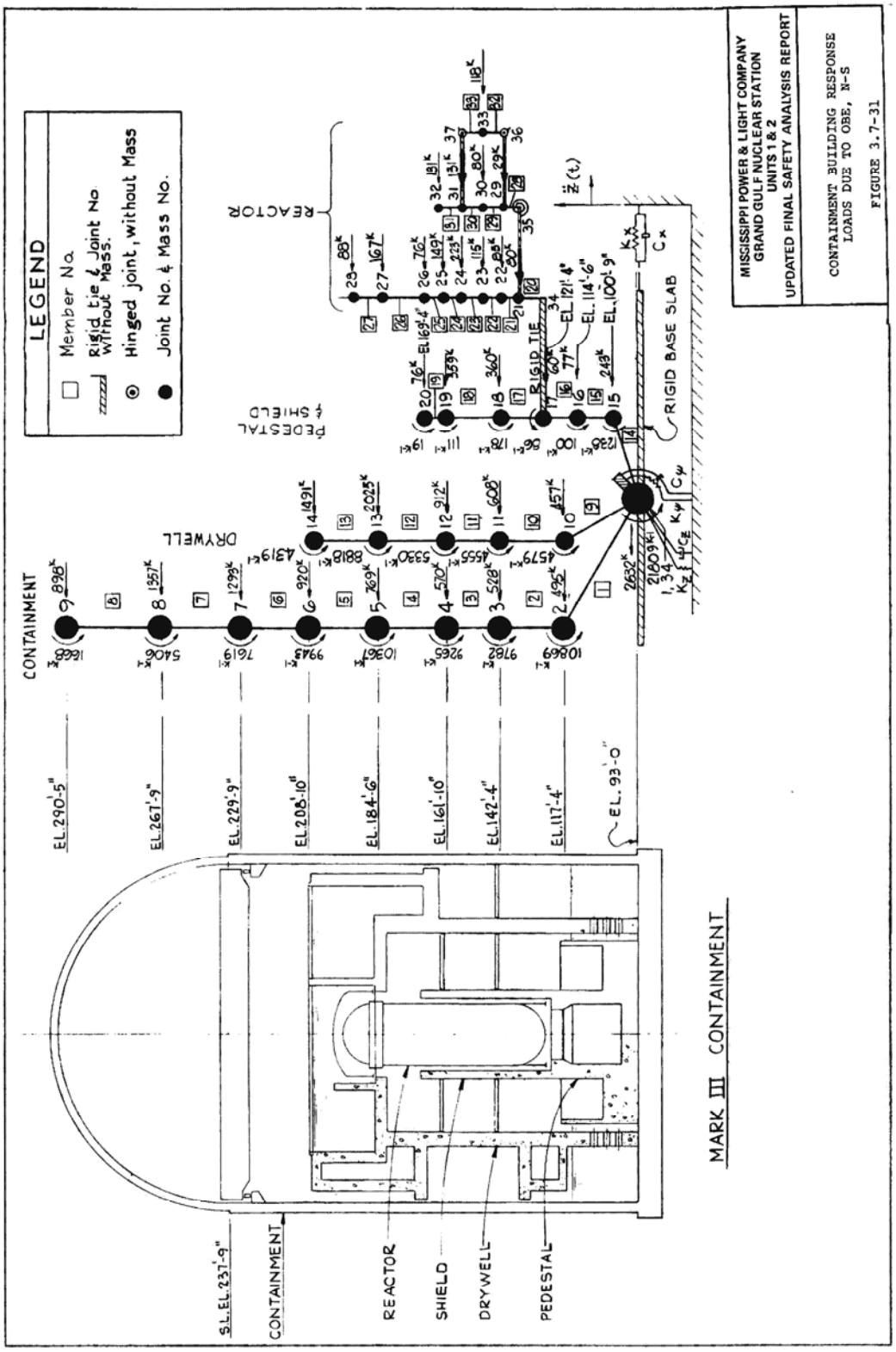
"FLUSH" MATHEMATICAL MODEL FOR SEISMIC ANALYSIS SSW COOLING TOWER & BASIN UNDERLAIN BY MULTILAYERED SOIL (FOR SUBSYSTEM ANALYSIS PURPOSE)
FIGURE 3.7-26

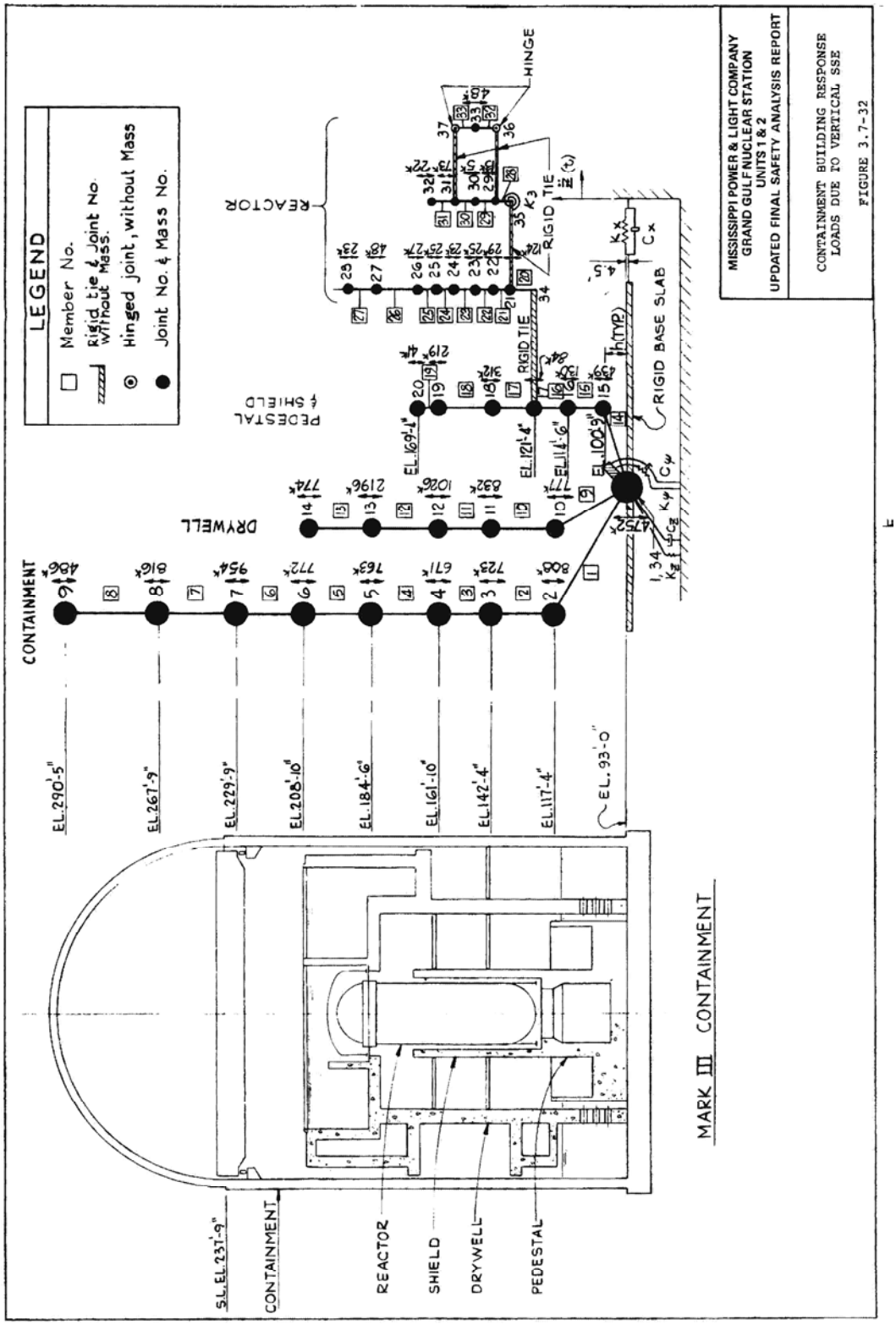

LEGEND:
AXIAL SPRING
TORSIONAL SPRING
HINGE CONNECTION
RIGID MEMBER
MASS POINT
MEMBER NUMBER
MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT
REACTOR PRESSURE VESSEL AND INTERNALS SEISMIC MODEL
FIGURE 3.7-27
MATHEMATICAL MODEL
REACTOR PRESSURE VESSEL & INTERNALS
INSIDE TOP OF VESSEL HEAD
REACTOR PRESSURE VESSEL
VESSEL HEAD FLANGE MATING SURFACE
STEAM SEPARATOR
TOP OF SEPARATORS
NORMAL WATER LEVEL
BOTTOM OF SEPARATORS
TOP OF SHROUD HEAD DOME
EFFECTIVE TOP OF SHROUD HEAD
FUEL
TOP OF GUIDE TUBES
TOP OF SHROUD SUPPORT CYLINDER
TOP OF CRD HOUSINGS
ELEVATION O
INSIDE BOTTOM OF VESSEL HEAD.
REFUELING BELLOWS
SHROUD HD FLANGE MATING SURFACE
SUPPORT SKIRT
PEDESTAL
NOTE:
MASS POINTS 14, 15, 16, 17, 55, 56, 57, 58, & MEMBERS 14, 15, 16, 17, 57, 68 ARE NOT USED.
SPRINGS K1 & K2 ARE NOT USED.

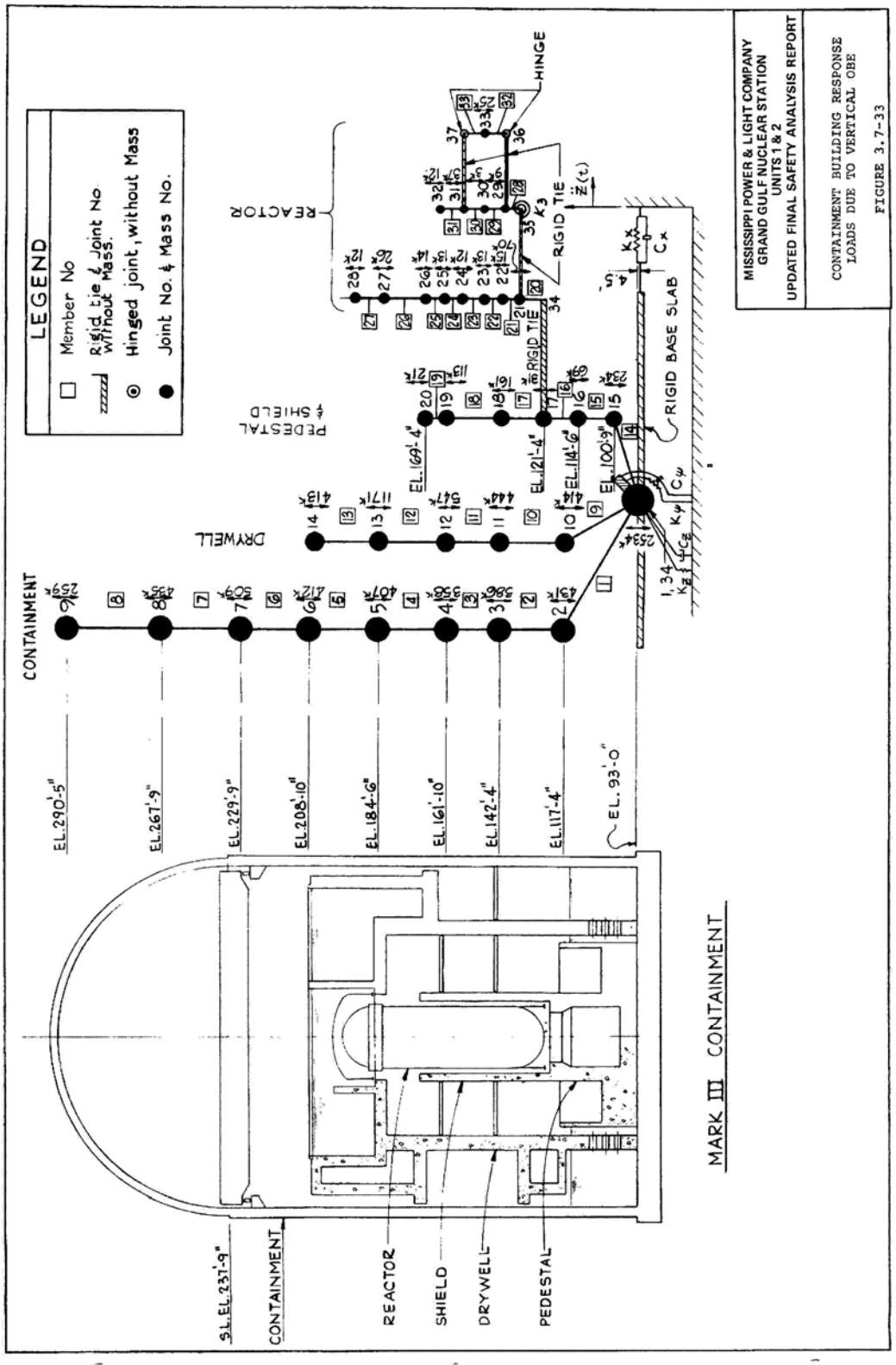

LEGEND
Member No.
Rigid tie & Joint No. without Mass.
Hinged joint, without Mass
Joint No. & Mass No.
MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT
CONTAINMENT BUILDING RESPONSE
LOADS DUE TO SSE, E-W
FIGURE 3.7-28
REACTOR
PEDESTAL & SHIELD
DRYWELL
CONTAINMENT
RIGID BASE SLAB
EL. 290'-5"
EL. 261'-9"
EL. 229'-9"
EL. 208'-10"
EL. 184'-6"
EL. 161'-10"
EL. 142'-4"
EL. 117'-4"
EL. 93'-0"
MARK III CONTAINMENT
REACTOR
SHIELD
DRYWELL
PEDESTAL
CONTAINMENT
SL. EL. 231'-9"

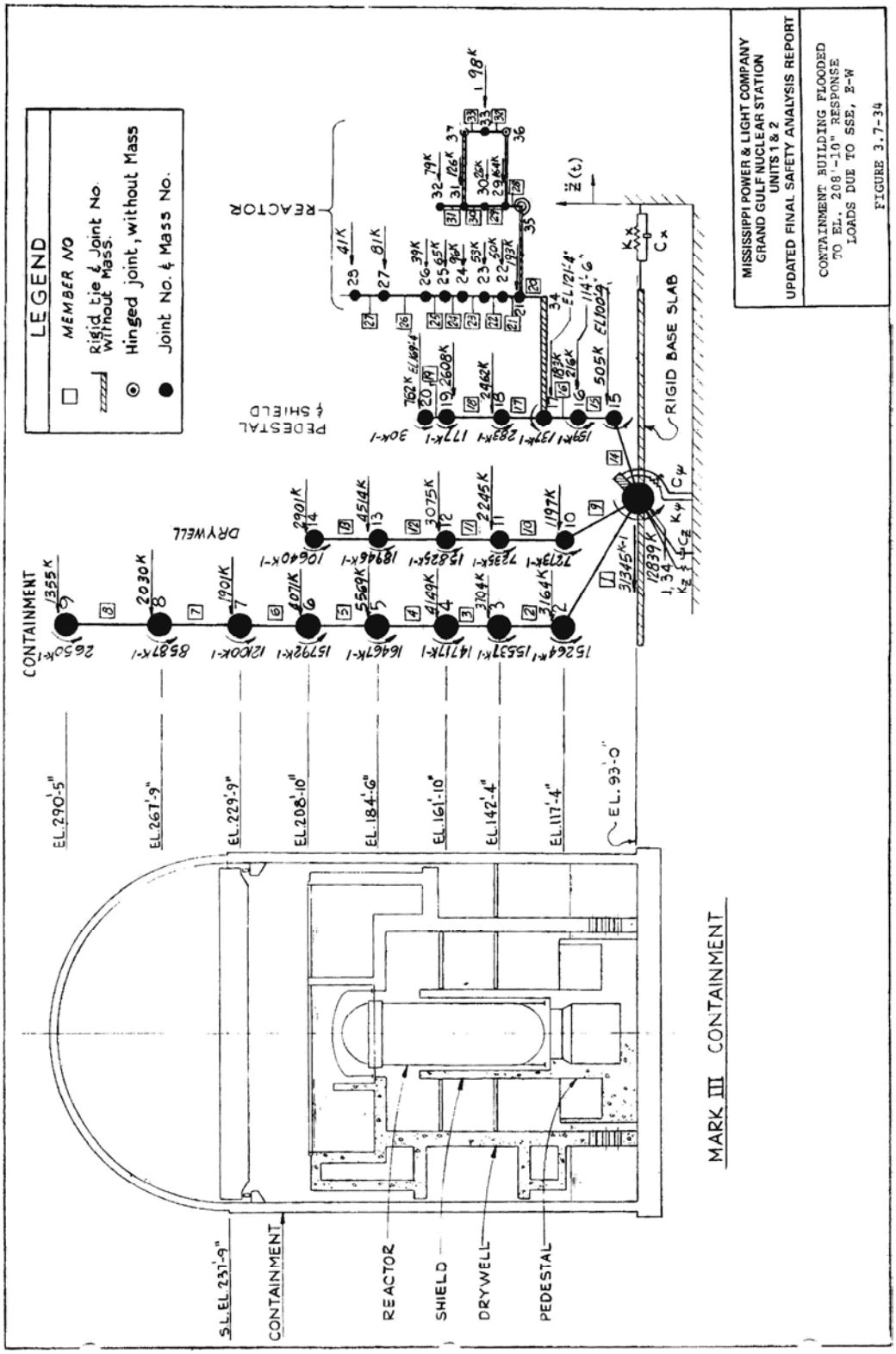

LEGEND
MEMBER NO
Rigid tie & Joint No without Mass.
Hinged joint, without Mass
Joint No. & Mass No.
REACTOR
PEDESTAL & SHIELD
DRYWELL
RIGID BASE SLAB
CONTAINMENT
EL. 290'-5"
EL. 267'-9"
EL. 229'-9"
EL. 208'-10"
EL. 184'-6"
EL. 161'-10"
EL. 142'-4"
EL. 117'-4"
EL. 93'-0"
SL. EL. 237'-9"
CONTAINMENT
REACTOR
SHIELD
DRYWELL
PEDESTAL
MARK III CONTAINMENT
MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT
CONTAINMENT BUILDING RESPONSE
LOADS DUE TO OBE, E-W
FIGURE 3.7-29

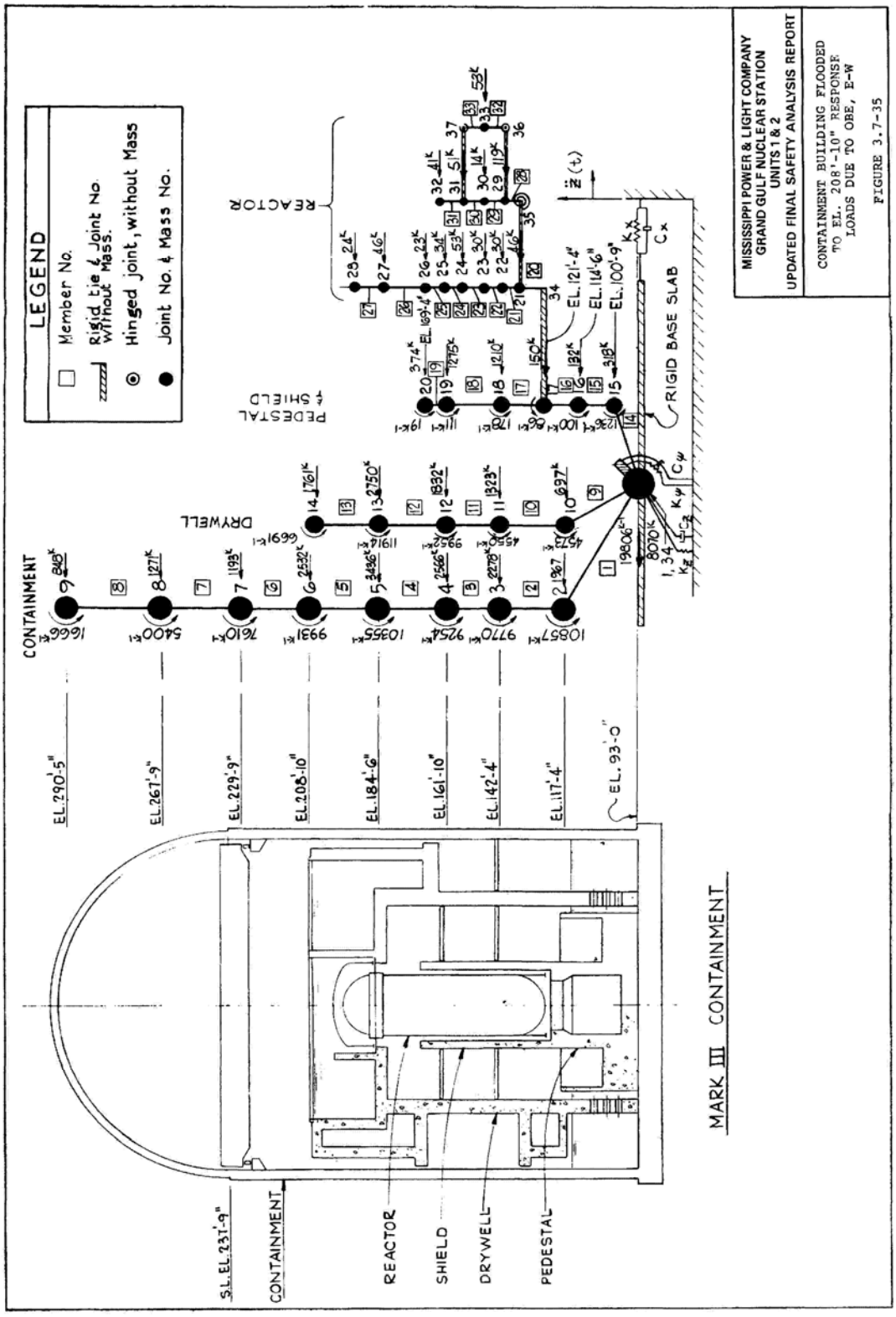

LEGEND
Member No.
Rigid tie & Joint No. without Mass.
Hinged joint, without Mass
Joint No. & Mass No.
MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT
CONTAINMENT BUILDING RESPONSE LOADS DUE TO SSE, N-S
FIGURE 3.7-30
REACTOR
PEDESTAL & SHIELD
DRY WELL
CONTAINMENT
RIGID BASE SLAB
EL. 290'-5"
EL. 267'-9"
EL. 229'-9"
EL. 208'-10"
EL. 184'-6"
EL. 161'-10"
EL. 142'-4"
EL. 117'-4"
EL. 93'-0"
SL EL 231'-9"
MARK III CONTAINMENT
REACTOR
SHIELD
DRYWELL
PEDESTAL

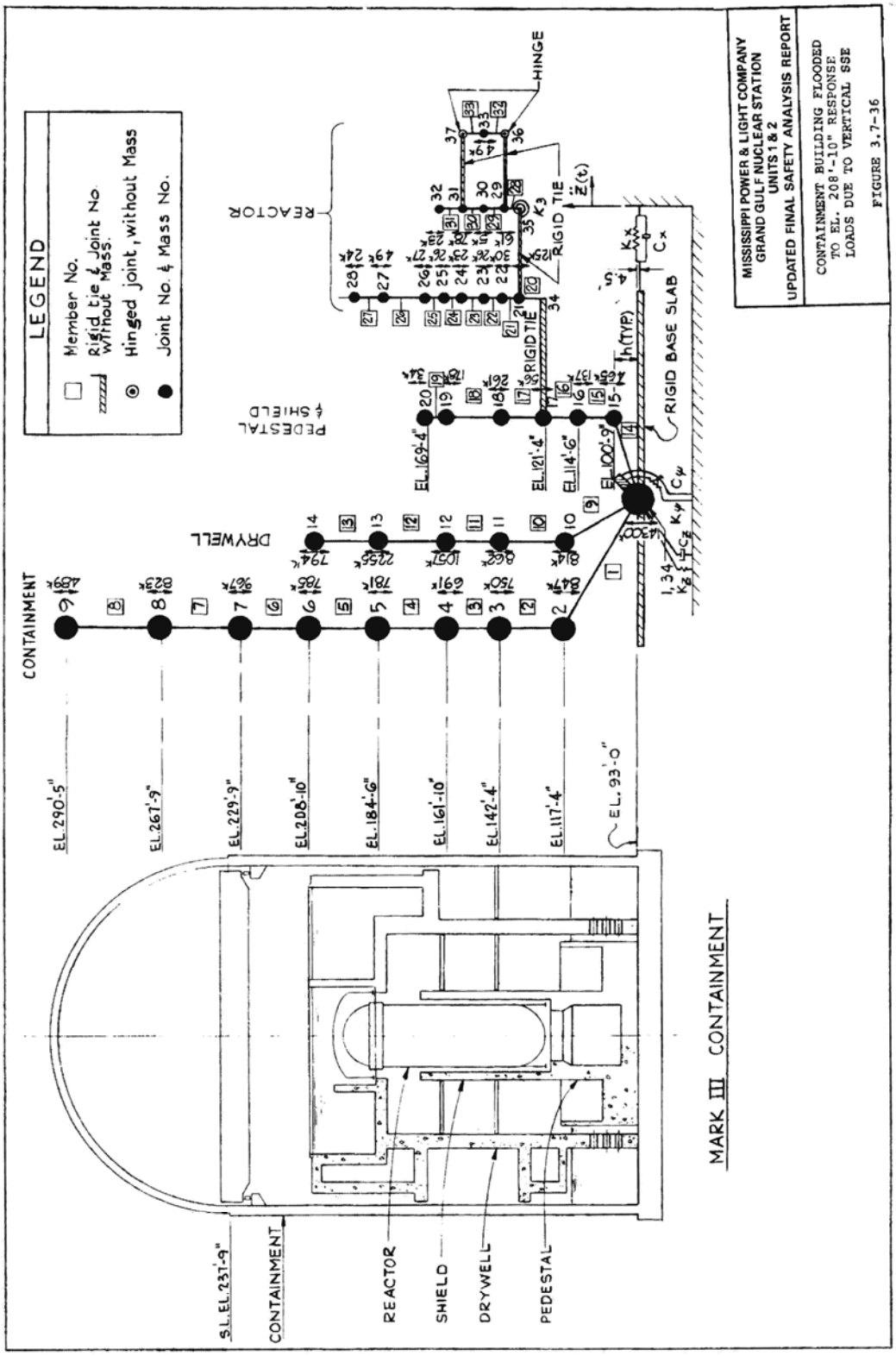

LEGEND
Member No.
Rigid tie & Joint No. Without Mass.
Hinged joint, without Mass
Joint No. & Mass No.
MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT
CONTAINMENT BUILDING RESPONSE LOADS DUE TO OBE, N-S
FIGURE 3.7-31
REACTOR
PEDESTAL & SHIELD
RIGID TIE
RIGID BASE SLAB
DRYWELL
CONTAINMENT
EL. 290'-5"
EL. 267'-9"
EL. 229'-9"
EL. 205'-10"
EL. 184'-6"
EL. 161'-10"
EL. 142'-4"
EL. 117'-4"
EL. 93'-0"
EL. 121'-4"
EL. 114'-6"
EL. 100'-9"
SL. EL. 237'-9"
MARK III CONTAINMENT

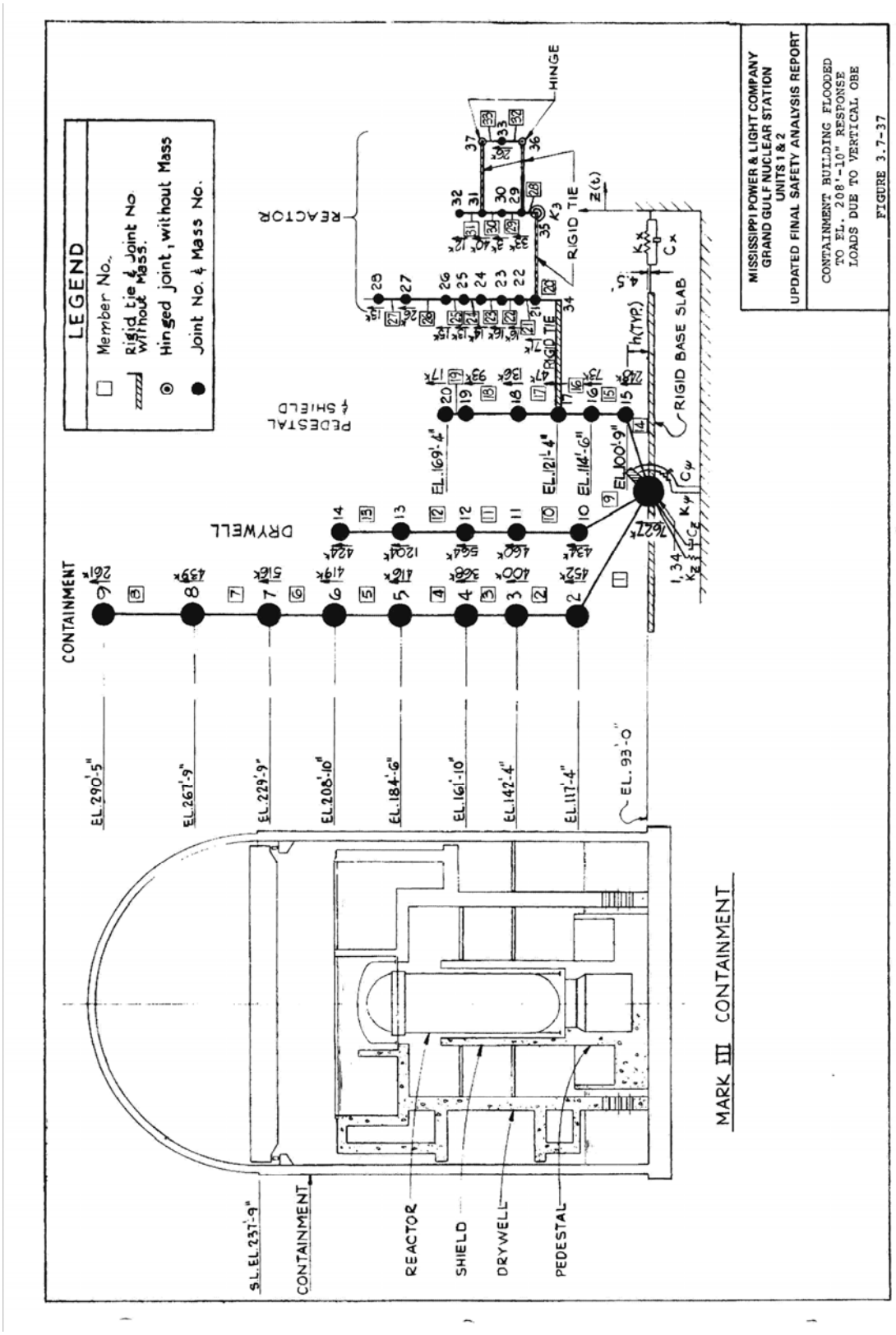

LEGEND
Member No.
Rigid tie & Joint No. without Mass.
Hinged joint, without Mass
Joint No. & Mass No.
MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT
CONTAINMENT BUILDING RESPONSE LOADS DUE TO VERTICAL SSE
FIGURE 3.7-32
REACTOR
HINGE
RIGID TIE
PEDESTAL & SHIELD
DRYWELL
RIGID BASE SLAB
CONTAINMENT
EL. 290'-5"
EL. 267'-9"
EL. 229'-9"
EL. 205'-10"
EL. 184'-6"
EL. 161'-10"
EL. 142'-4"
EL. 117'-4"
EL. 93'-0"
SL. EL. 237'-9"
MARK III CONTAINMENT
REACTOR
SHIELD
DRYWELL
PEDESTAL

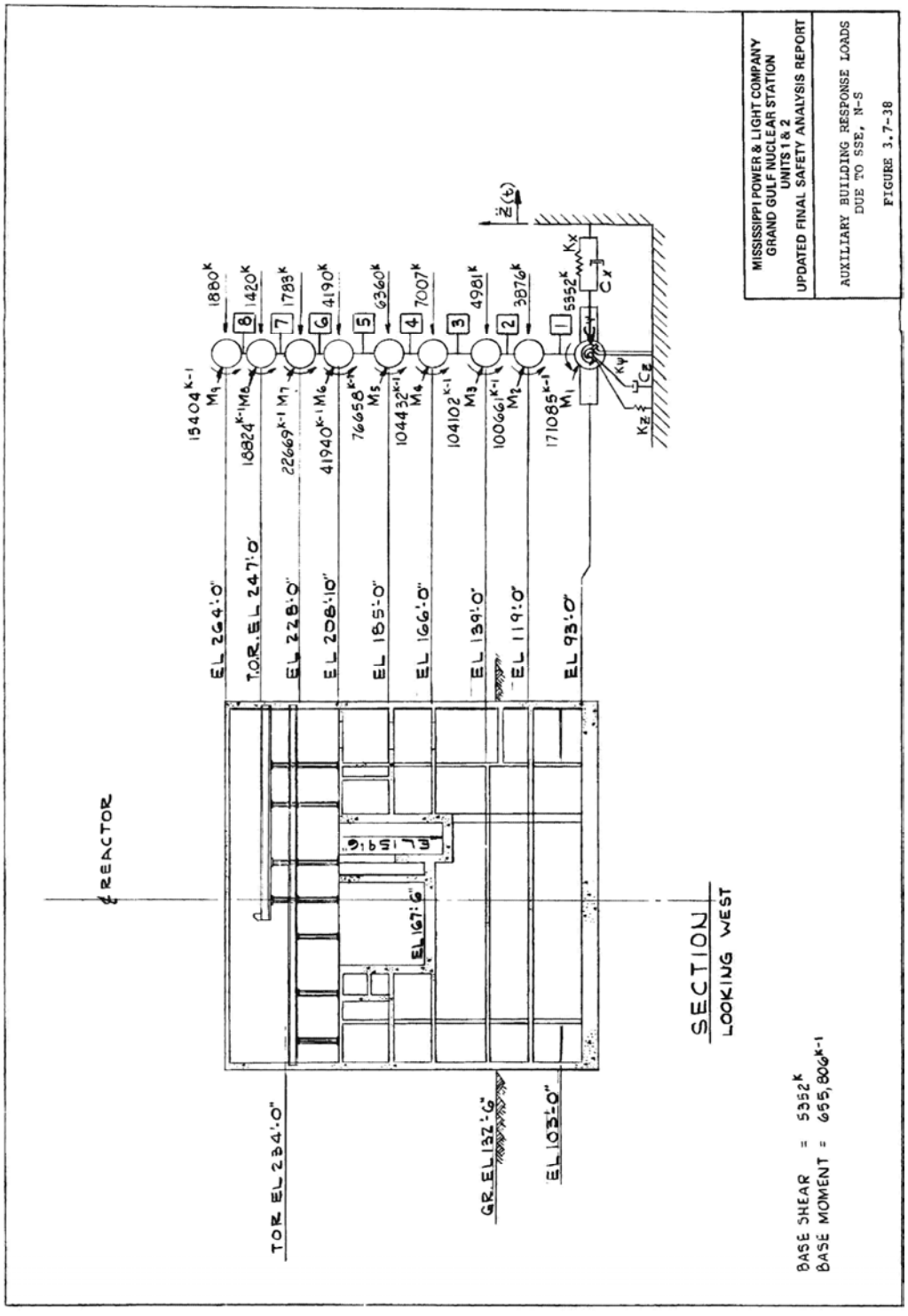

LEGEND
Member No
Rigid tie & Joint No. without Mass.
Hinged joint, without Mass
Joint No. & Mass No.
MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT
CONTAINMENT BUILDING RESPONSE
LOADS DUE TO VERTICAL OBE
FIGURE 3.7-33
REACTOR
HINGE
RIGID TIE
PEDESTAL & SHIELD
RIGID BASE SLAB
DRYWELL
CONTAINMENT
EL. 290'-5"
EL. 267'-9"
EL. 229'-9"
EL. 208'-10"
EL. 184'-6"
EL. 161'-10"
EL. 142'-4"
EL. 117'-4"
EL. 93'-0"
EL. 169'-4"
EL. 121'-4"
EL. 114'-6"
EL. 100'-9"
SL. EL. 237'-9"
CONTAINMENT
REACTOR
SHIELD
DRYWELL
PEDESTAL
MARK III CONTAINMENT

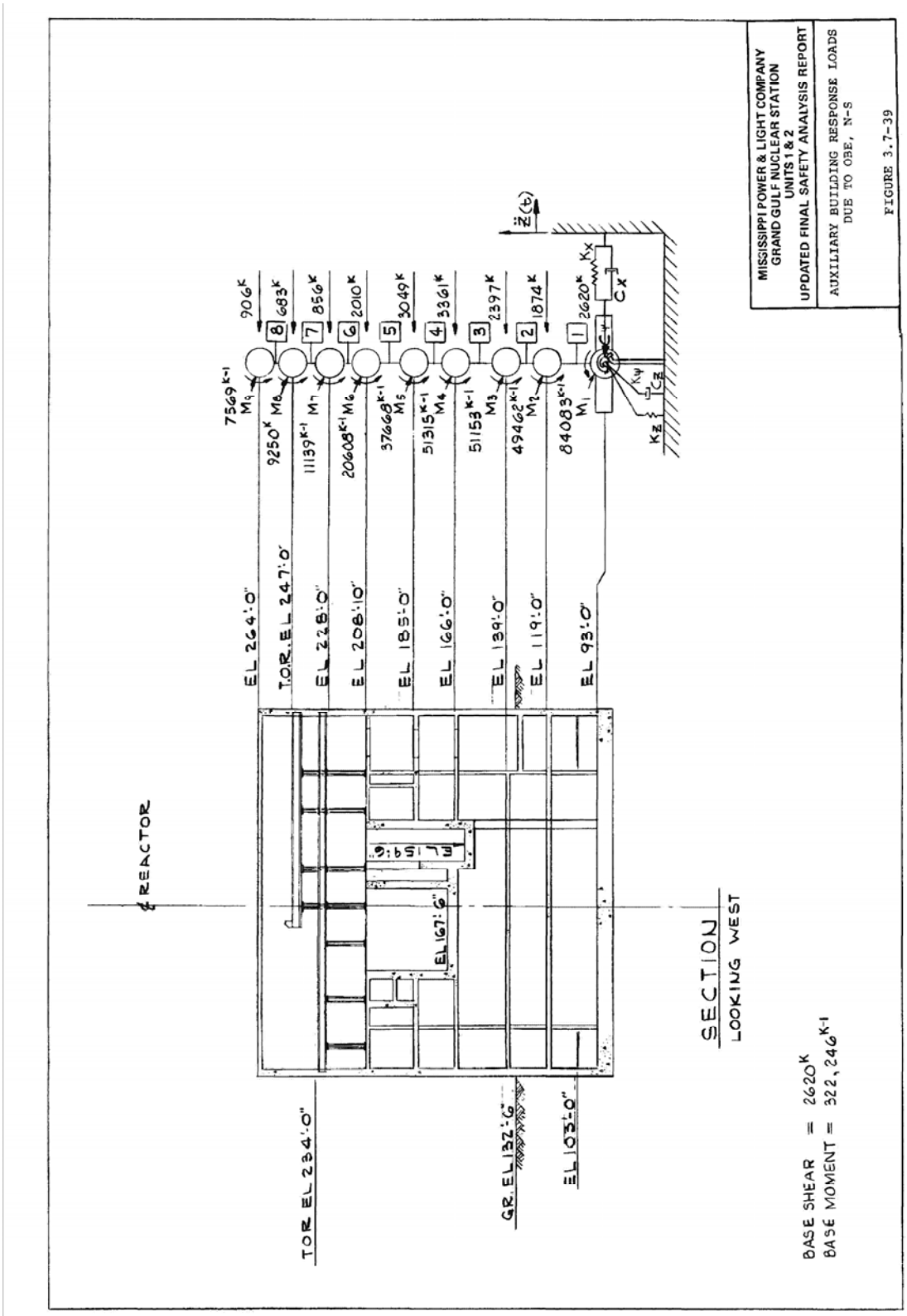

LEGEND
MEMBER NO
Rigid tie & Joint No. without Mass.
Hinged joint, without Mass
Joint No. & Mass No.
MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT
CONTAINMENT BUILDING FLOODED TO EL. 208'-10" RESPONSE LOADS DUE TO SSE, E-W
FIGURE 3.7-34
REACTOR
PEDESTAL & SHIELD
DRYWELL
CONTAINMENT
RIGID BASE SLAB
EL. 290'-5"
EL. 267'-9"
EL. 229'-9"
EL. 208'-10"
EL. 184'-6"
EL. 161'-10"
EL. 142'-4"
EL. 117'-4"
EL. 93'-0"
SL EL 231'-9"
CONTAINMENT
REACTOR
SHIELD
DRYWELL
PEDESTAL
MARK III CONTAINMENT

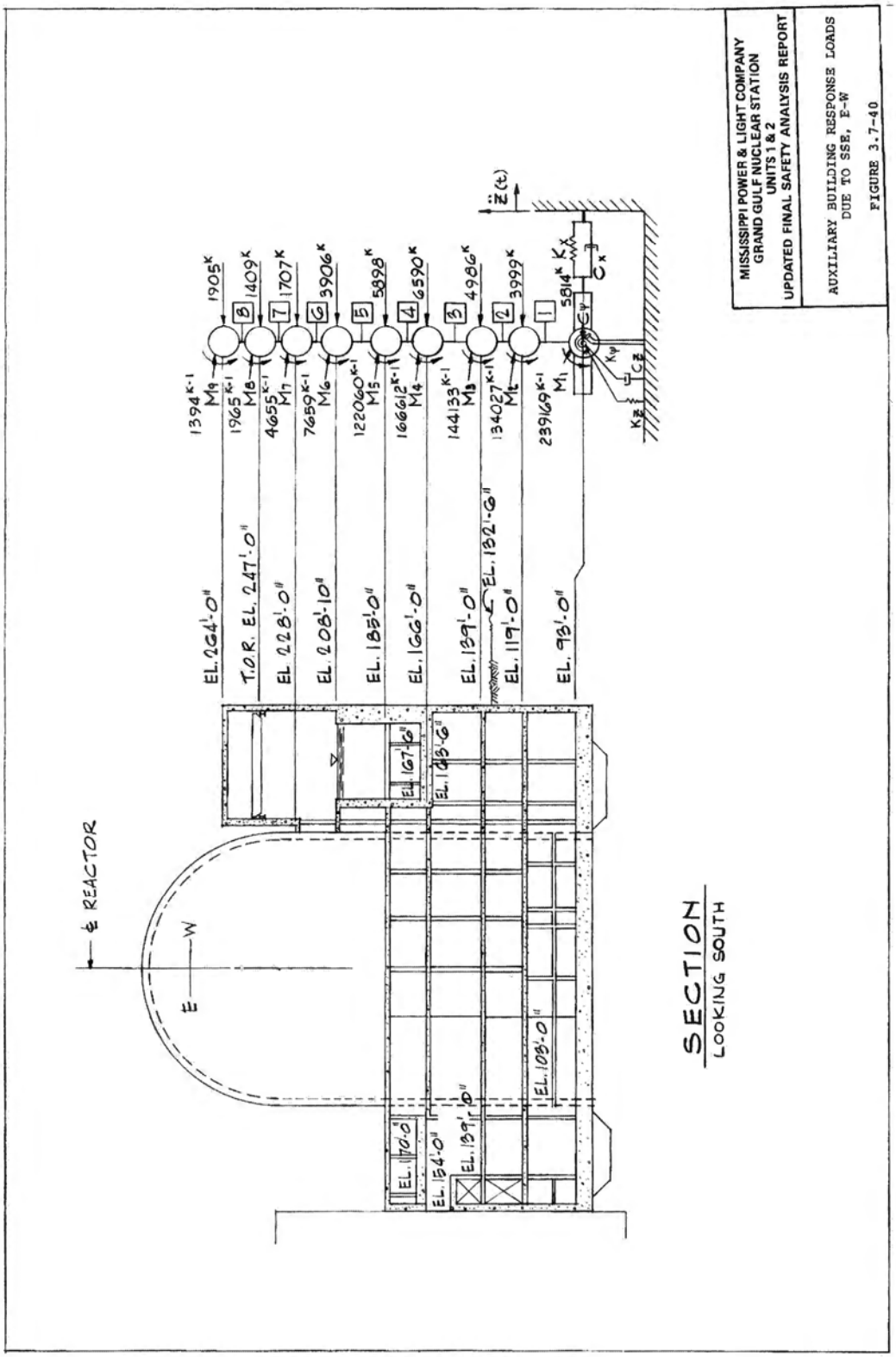

LEGEND
Member No.
Rigid tie & Joint No. without Mass.
Hinged joint, without Mass
Joint No. & Mass No.
MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT
CONTAINMENT BUILDING FLOODED TO EL. 208'-10" RESPONSE LOADS DUE TO OBE, E-W
FIGURE 3.7-35
REACTOR
PEDESTAL & SHIELD
DRYWELL
CONTAINMENT
RIGID BASE SLAB
EL. 290'-5"
EL. 267'-9"
EL. 229'-9"
EL. 208'-10"
EL. 184'-6"
EL. 161'-10"
EL. 142'-4"
EL. 117'-4"
EL. 93'-0"
EL. 121'-4"
EL. 114'-6"
EL. 100'-9"
MARK III CONTAINMENT
S.L. EL. 231'-9"
CONTAINMENT
REACTOR
SHIELD
DRYWELL
PEDESTAL

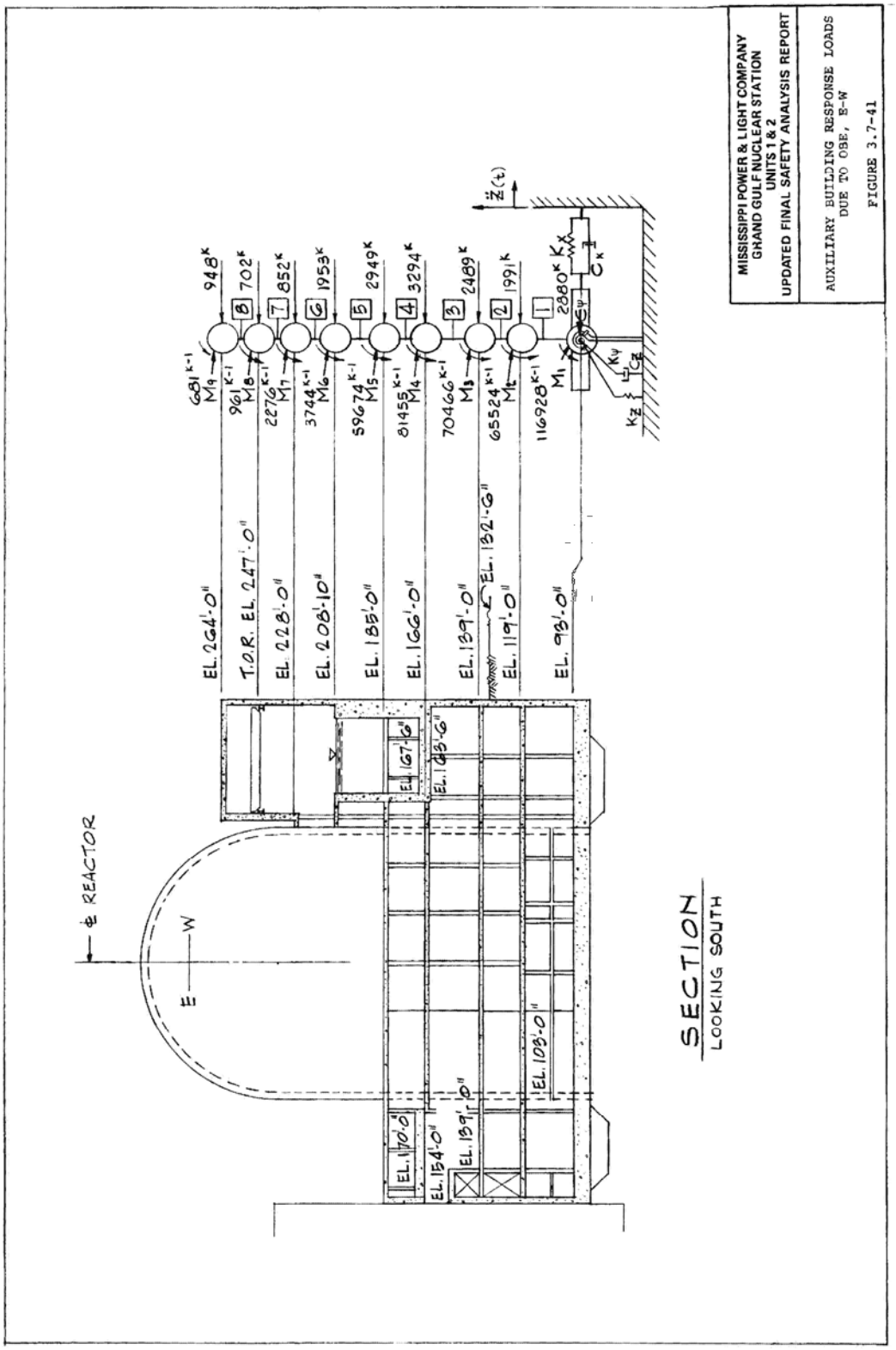

LEGEND
Member No.
Rigid tie & Joint No. without Mass.
Hinged joint, without Mass
Joint No. & Mass No.
MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT
CONTAINMENT BUILDING FLOODED TO EL. 208'-10" RESPONSE LOADS DUE TO VERTICAL SSE
FIGURE 3.7-36
HINGE
REACTOR
RIGID TIE
PEDESTAL & SHIELD
RIGID BASE SLAB
DRYWELL
CONTAINMENT
EL. 290'-5"
EL. 261'-9"
EL. 229'-9"
EL. 208'-10"
EL. 184'-6"
EL. 161'-10"
EL. 142'-4"
EL. 117'-4"
EL. 93'-0"
S.L. EL. 231'-9"
REACTOR
SHIELD
DRYWELL
PEDESTAL
MARK III CONTAINMENT

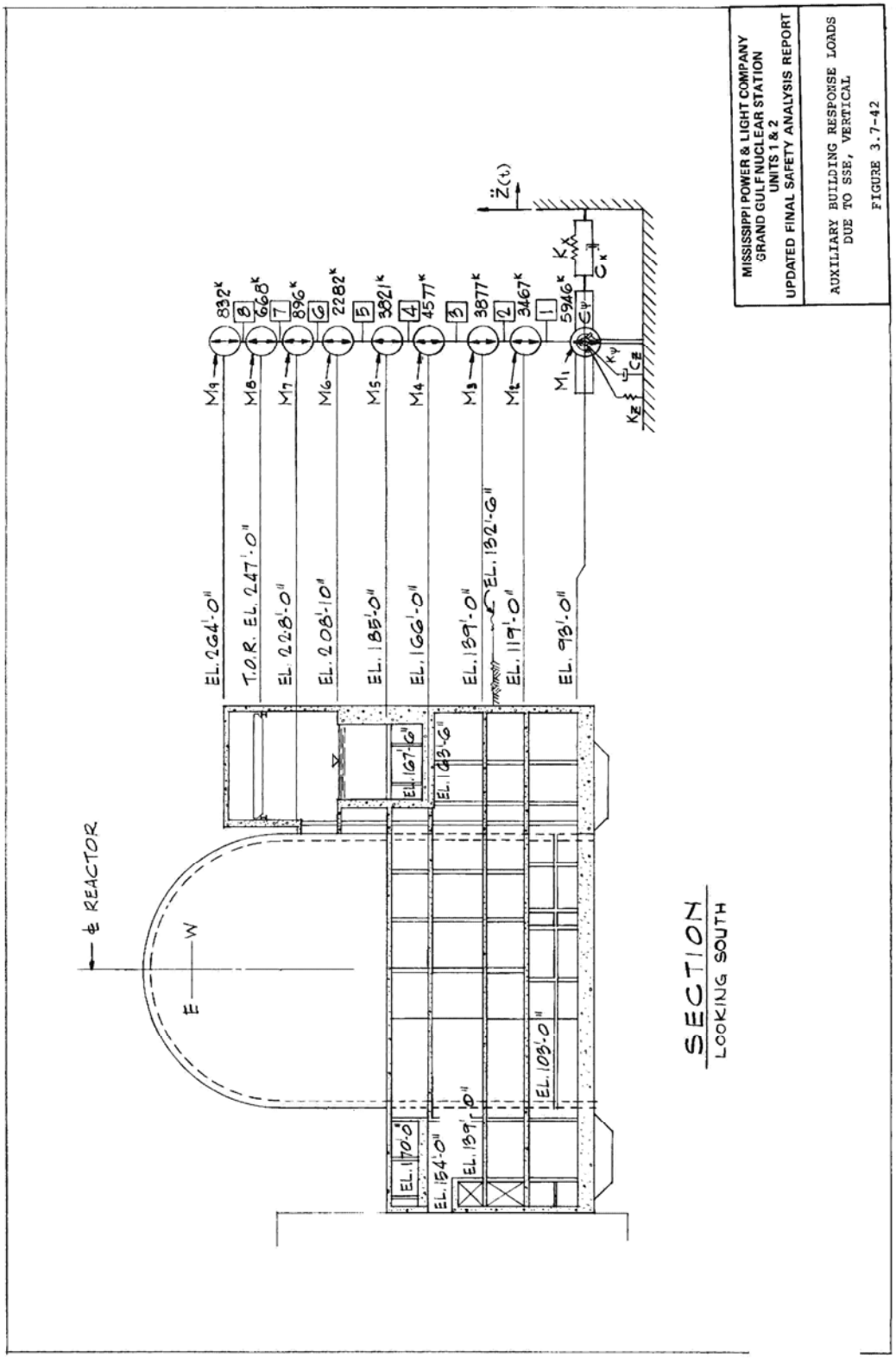

LEGEND
Member No.
Rigid tie & Joint No without Mass.
Hinged joint, without Mass
Joint No. & Mass No.
MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT
CONTAINMENT BUILDING FLOODED TO EL. 208'-10" RESPONSE LOADS DUE TO VERTICAL OBE
FIGURE 3.7-37
REACTOR
HINGE
RIGID TIE
RIGID BASE SLAB
PEDESTAL & SHIELD
DRYWELL
CONTAINMENT
EL. 290'-5"
EL. 267'-9"
EL. 229'-9"
EL. 208'-10"
EL. 184'-6"
EL. 161'-10"
EL. 142'-4"
EL. 117'-4"
EL. 93'-0"
EL. 169'-4"
EL. 121'-4"
EL. 114'-6"
EL. 100'-9"
SL EL 237'-9"
CONTAINMENT
REACTOR
SHIELD
DRYWELL
PEDESTAL
MARK III CONTAINMENT

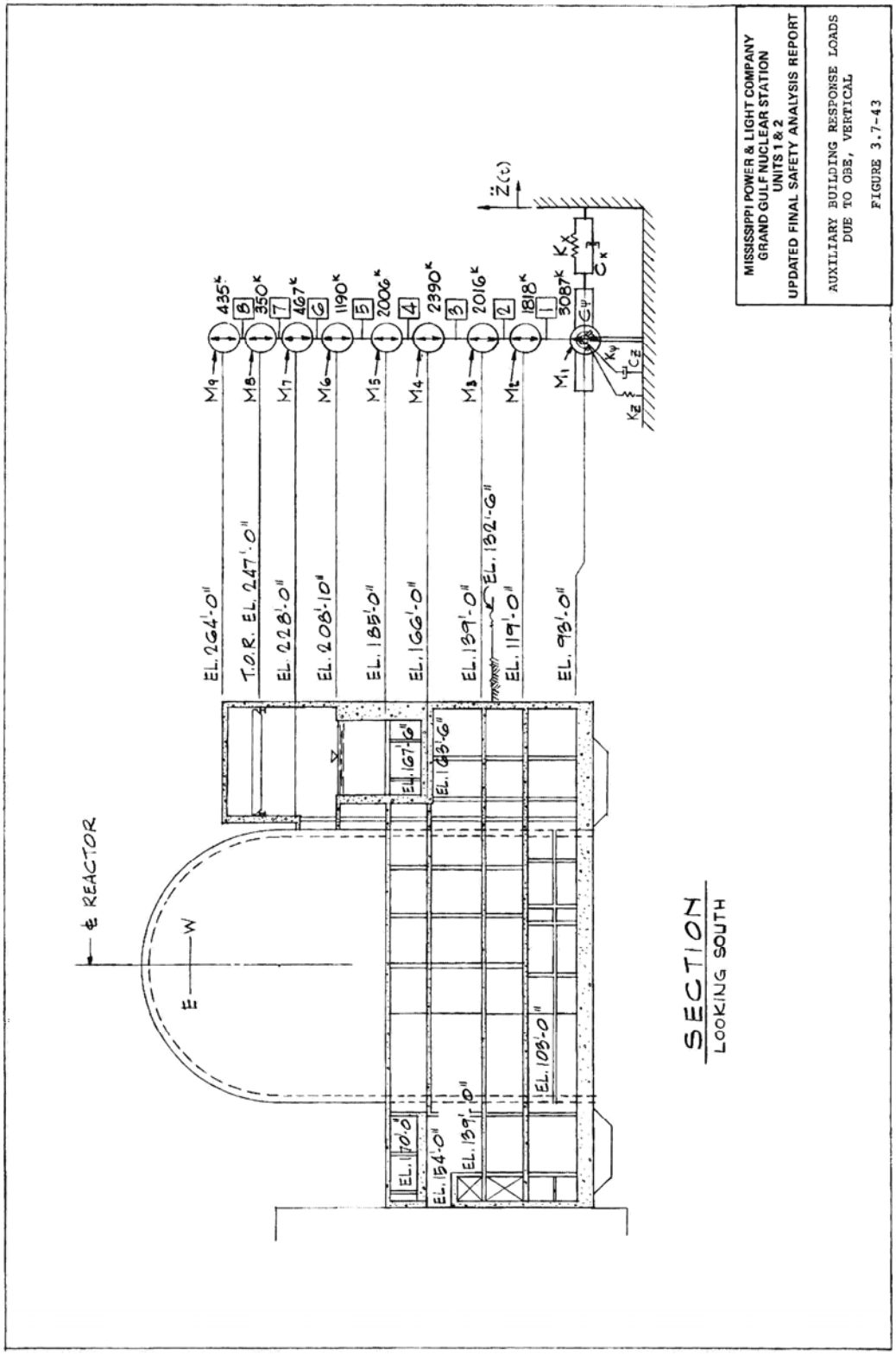

MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT
AUXILIARY BUILDING RESPONSE LOADS
DUE TO SSE, N-S
FIGURE 3.7-38
SECTION
LOOKING WEST
REACTOR
EL 264'-0"
T.O.R. EL 247'-0"
EL 228'-0"
EL 208'-10"
EL 185'-0"
EL 166'-0"
EL 139'-0"
EL 119'-0"
EL 93'-0"
EL 167'-6"
TOR EL 234'-0"
GR. EL 132'-6"
EL 103'-0"
BASE SHEAR = 5352K
BASE MOMENT = 655,806K-1

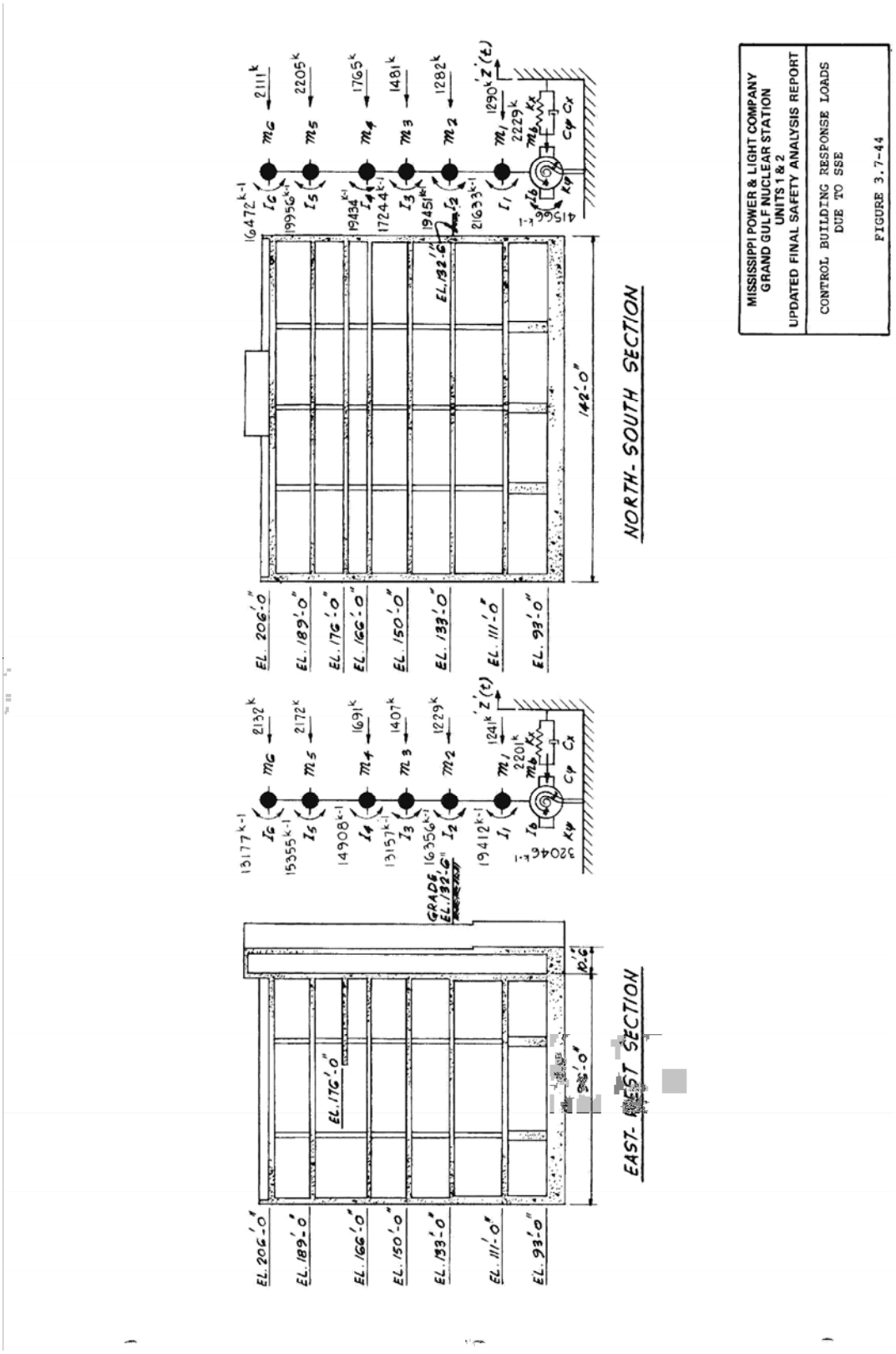

MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT
AUXILIARY BUILDING RESPONSE LOADS
DUE TO OBE, N-S
FIGURE 3.7-39
SECTION
LOOKING WEST
℄ REACTOR
TOR. EL 234'-0"
GR. EL 132'-6"
EL 103'-0"
EL 167'-6"
EL 264'-0"
TOR. EL 247'-0"
EL 228'-0"
EL 208'-10"
EL 185'-0"
EL 166'-0"
EL 139'-0"
EL 119'-0"
EL 93'-0"
906K
683K
856K
2010K
3049K
3361K
2397K
1874K
2620K
7569K-I
9250K
11139K-I
20608K-I
37669K-I
51315K-I
51153K-I
49462K-I
84083K-I
BASE SHEAR = 2620K
BASE MOMENT = 322,246K-I

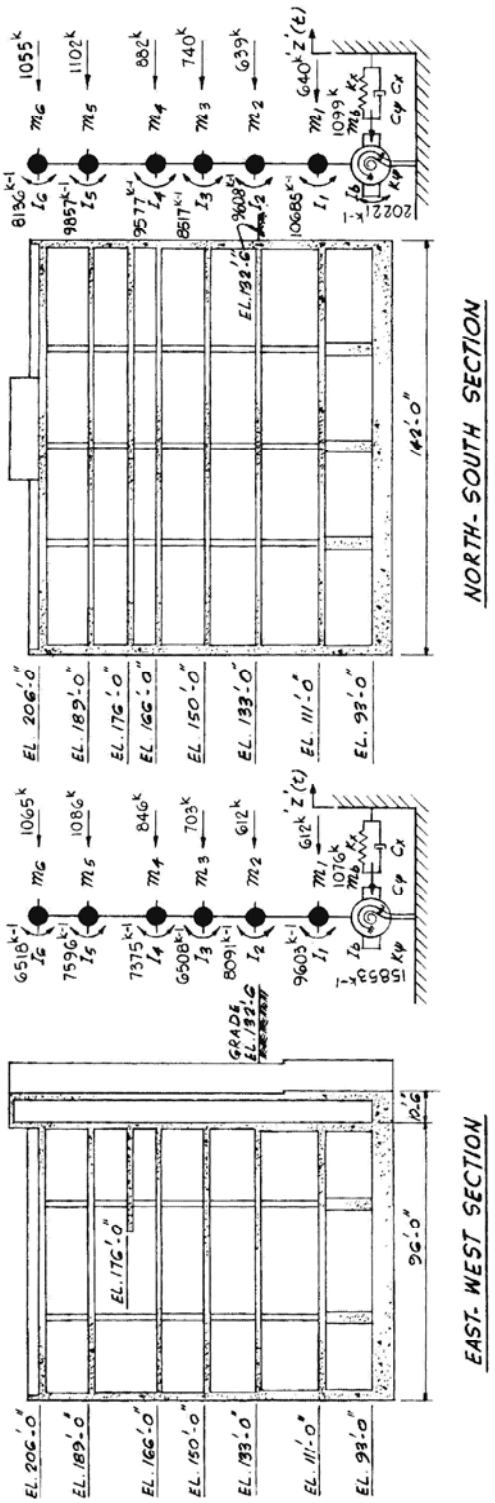

MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT
AUXILIARY BUILDING RESPONSE LOADS
DUE TO SSE, E-W
FIGURE 3.7-40
SECTION
LOOKING SOUTH
℄ REACTOR
E
W
T.O.R. EL. 247'-0"
EL. 228'-0"
EL. 208'-10"
EL. 185'-0"
EL. 166'-0"
EL. 139'-0"
EL. 132'-6"
EL. 119'-0"
EL. 93'-0"
EL. 167'-10"
EL. 145'-6"
EL. 108'-0"
EL. 70'-0"
EL. 154'-0"
EL. 191'-10"
1905K
1409K
1707K
3906K
5898K
6590K
4986K
3999K
5814K
1394K-I
1965K-I
4655K-I
7659K-I
122060K-I
166612K-I
144133K-I
134027K-I
239169K-I
M9
M8
M7
M6
M5
M4
M3
M2
M1
B
7
6
5
4
3
2
1
Kx
Cx
Kψ
Cψ
Kz
Cz
z̈(t)


MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT
AUXILIARY BUILDING RESPONSE LOADS
DUE TO OBE, E-W
FIGURE 3.7-41
SECTION
LOOKING SOUTH
℄ REACTOR
E
W
EL. 264'-0"
T.O.R. EL. 247'-0"
EL. 228'-0"
EL. 208'-10"
EL. 185'-0"
EL. 166'-0"
EL. 139'-0"
EL. 119'-0"
EL. 93'-0"
EL. 182'-6"
EL. 167'-0"
EL. 148'-6"
EL. 108'-0"
EL. 170'-0"
EL. 154'-0"
EL. 139'-0"
948K
702K
852K
1953K
2949K
3294K
2489K
1991K
2880K
681 K-I
961 K-I
2276 K-I
3744 K-I
59674 K-I
81455 K-I
70466 K-I
65524 K-I
116928 K-I
Z(t)
Kx
Cx
Kψ
Cψ
Kz
Cz


MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT
AUXILIARY BUILDING RESPONSE LOADS DUE TO SSE, VERTICAL
FIGURE 3.7-42
SECTION
LOOKING SOUTH
℄ REACTOR
E
W
EL. 224'-0"
T.O.R. EL. 247'-0"
EL. 228'-0"
EL. 208'-10"
EL. 185'-0"
EL. 166'-0"
EL. 139'-0"
EL. 152'-6"
EL. 119'-0"
EL. 98'-0"
EL. 167'-0"
EL. 143'-6"
EL. 170'-0"
EL. 154'-0"
EL. 139'-10"
EL. 109'-0"
832K
668K
896K
2282K
3821K
4577K
3877K
3467K
5946K
M9
M8
M7
M6
M5
M4
M3
M2
M1
8
7
6
5
4
3
2
1
Z̈(t)
Kx
Cx
Kψ
Cψ
Kz
Cz


MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT
AUXILIARY BUILDING RESPONSE LOADS
DUE TO OBE, VERTICAL
FIGURE 3.7-43
SECTION
LOOKING SOUTH
℄ REACTOR
W
E
EL. 264'-0"
T.O.R. EL. 247'-0"
EL. 228'-0"
EL. 205'-10"
EL. 185'-0"
EL. 166'-0"
EL. 139'-0"
EL. 119'-0"
EL. 93'-0"
EL. 182'-6"
EL. 167'-6"
EL. 153'-6"
EL. 105'-0"
EL. 70'-0"
EL. 154'-0"
EL. 139'-10"
435K
350K
467K
1190K
2006K
2390K
2016K
1818K
3087K
Z(t)

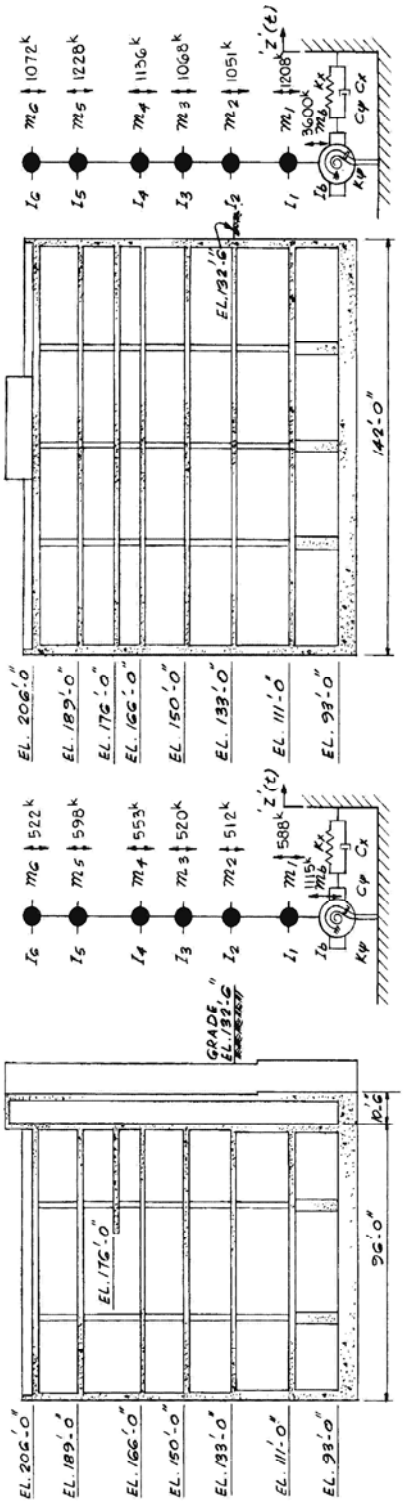

NORTH-SOUTH SECTION
EAST-WEST SECTION
EL. 206'-0"
EL. 189'-0"
EL. 176'-0"
EL. 166'-0"
EL. 150'-0"
EL. 133'-0"
EL. 111'-0"
EL. 93'-0"
142'-0"
GRADE EL. 132'-6"
MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT
CONTROL BUILDING RESPONSE LOADS DUE TO SSE
FIGURE 3.7-44



MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT

CONTROL BUILDING RESPONSE LOADS DUE TO OBE

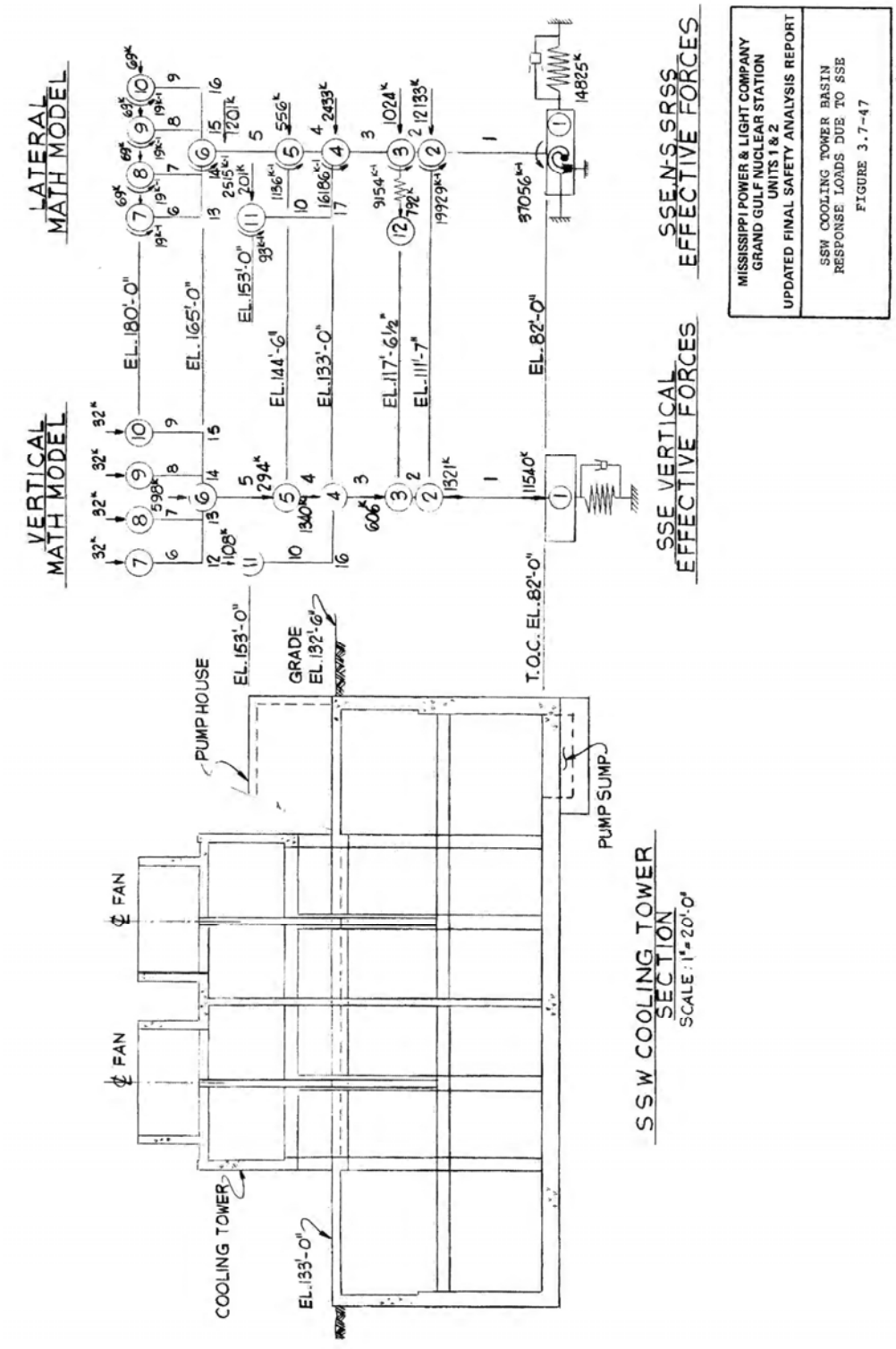
FIGURE 3.7-45

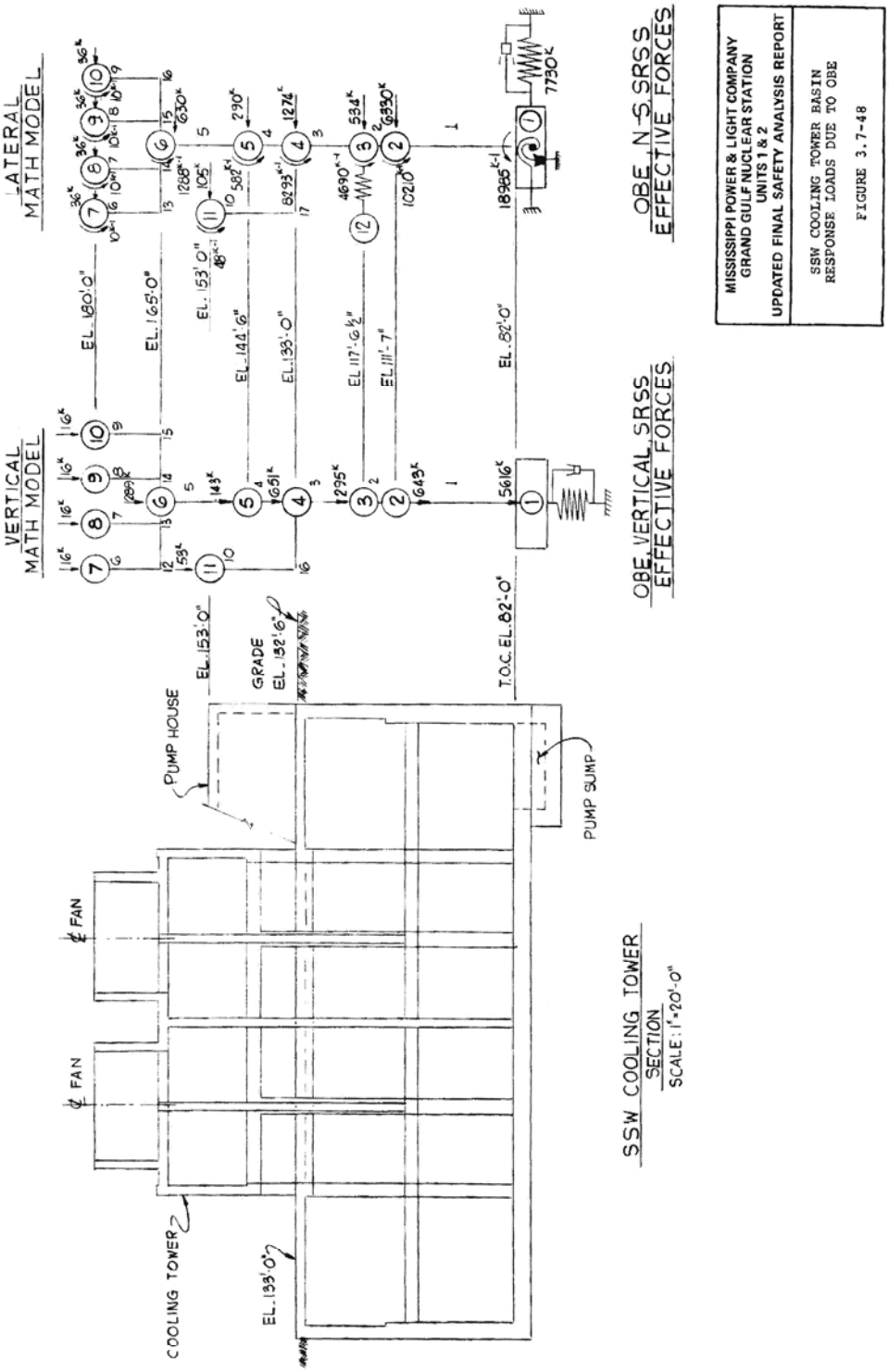


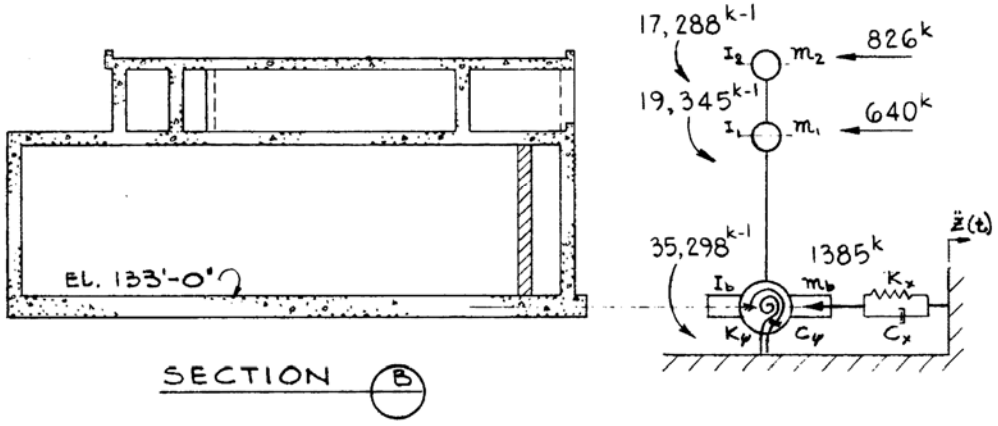
MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT

CONTROL BUILDING VERTICAL RESPONSE LOADS DUE TO SSE AND OBE

FIGURE 3.7-46


LATERAL MATH MODEL
VERTICAL MATH MODEL
SSE N-S SRSS EFFECTIVE FORCES
SSE VERTICAL EFFECTIVE FORCES
EL. 180'-0"
EL. 165'-0"
EL. 153'-0"
EL. 144'-6"
EL. 133'-0"
EL. 117'-6½"
EL. 111'-7"
EL. 82'-0"
T.O.C. EL. 82'-0"
PUMP HOUSE
EL. 153'-0"
GRADE EL. 132'-6"
PUMP SUMP
℄ FAN
℄ FAN
COOLING TOWER
EL. 133'-0"
S S W COOLING TOWER SECTION
SCALE: 1"=20'-0"
MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT
SSW COOLING TOWER BASIN
RESPONSE LOADS DUE TO SSE
FIGURE 3.7-47

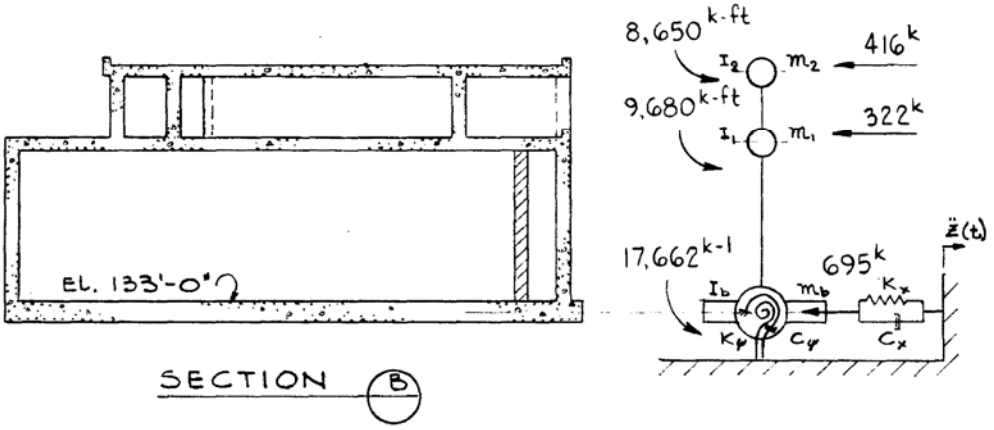

LATERAL MATH MODEL
VERTICAL MATH MODEL
EL. 180'-0"
EL. 165'-0"
EL. 153'-0"
EL. 144'-6"
EL. 133'-0"
EL. 117'-6½"
EL. 111'-7"
EL. 82'-0"
OBE N-S SRSS EFFECTIVE FORCES
OBE VERTICAL SRSS EFFECTIVE FORCES
7730K
18965K-I
10222K-I
8293K-I
10582K-I
4690K-I
5616K
6430K
6643K
6295K
6651K
5931K
EL. 153'-0"
GRADE EL. 132'-6"
T.O.C. EL. 82'-0"
PUMP HOUSE
PUMP SUMP
FAN
FAN
COOLING TOWER
EL. 133'-0"
SSW COOLING TOWER
SECTION
SCALE: 1"=20'-0"
MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT
SSW COOLING TOWER BASIN
RESPONSE LOADS DUE TO OBE
FIGURE 3.7-48



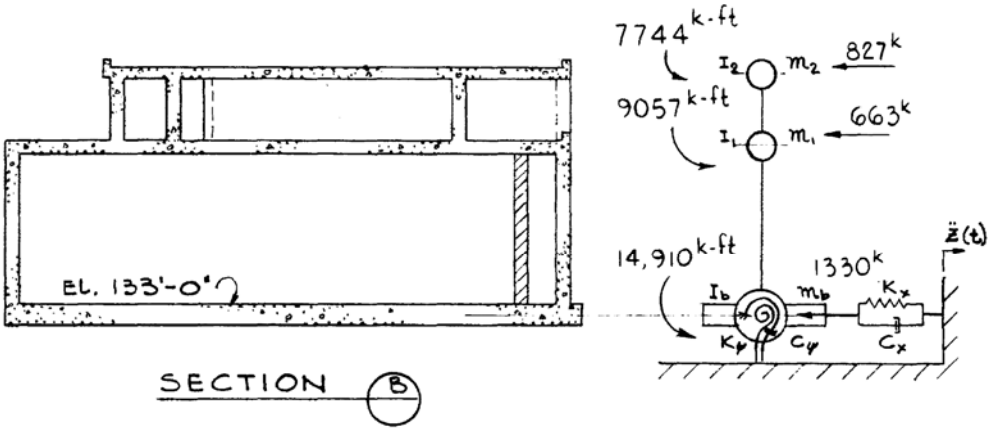
MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT

DIESEL GENERATOR BUILDING
RESPONSE LOADS
DUE TO N-S, SSE

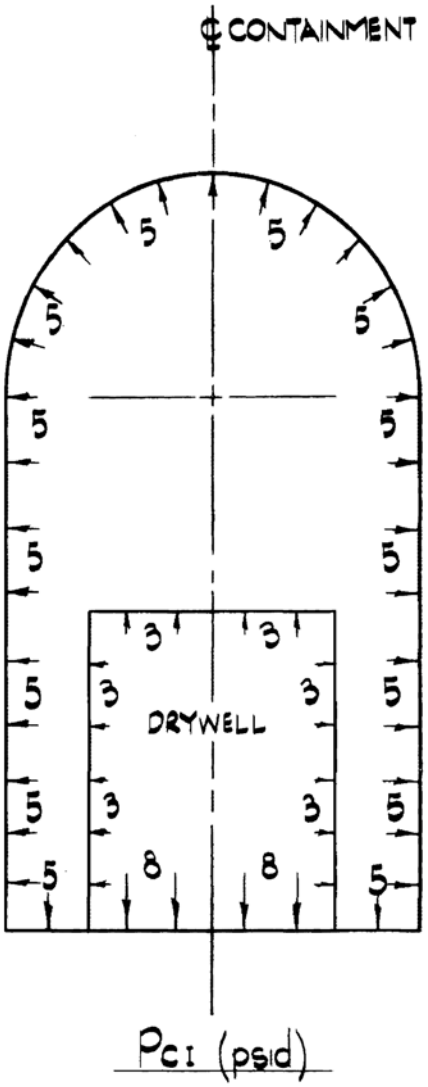
FIGURE 3.7-49



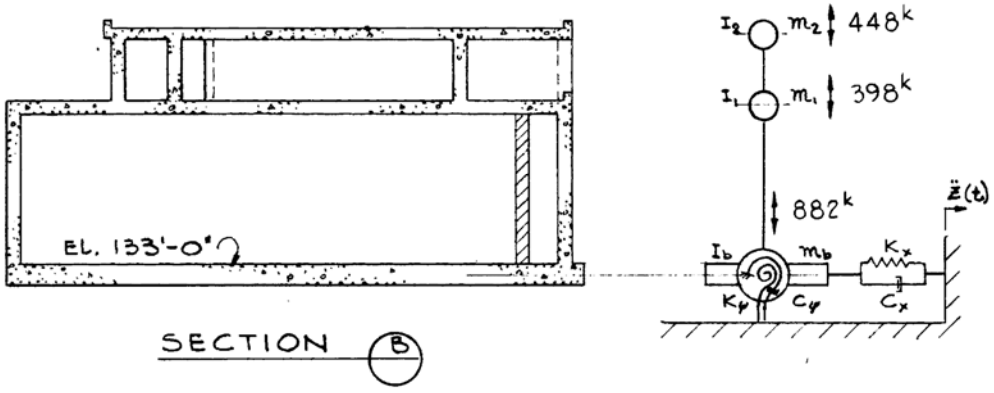
MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT
DIESEL GENERATOR BUILDING RESPONSE LOADS DUE TO N-S, OBE FIGURE 3.7-50



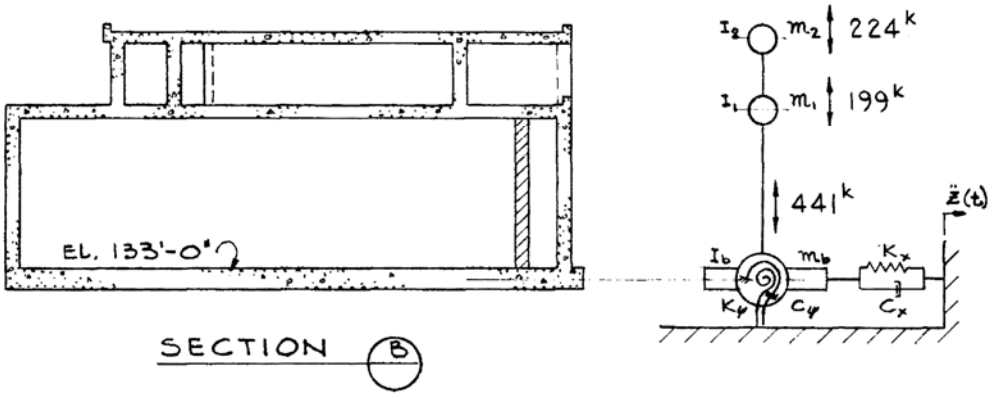
MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT
DIESEL GENERATOR BUILDING RESPONSE LOADS DUE TO E-W, SSE FIGURE 3.7-51



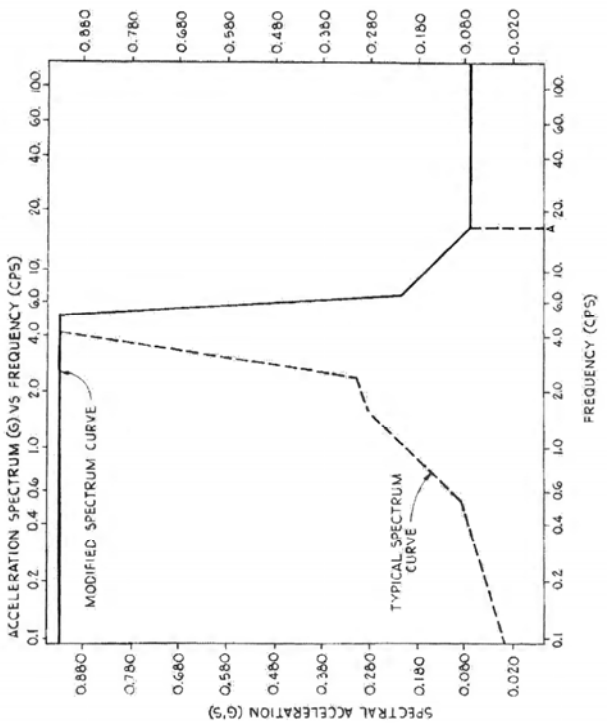
MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT
CONTAINMENT AND DRYWELL DESIGN PRESSURE (P_{CI}) INTERMEDIATE BREAK ACCIDENT FIGURE 3.8-52



MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT
DIESEL GENERATOR BUILDING RESPONSE LOADS DUE TO VERTICAL SSE FIGURE 3.7-53



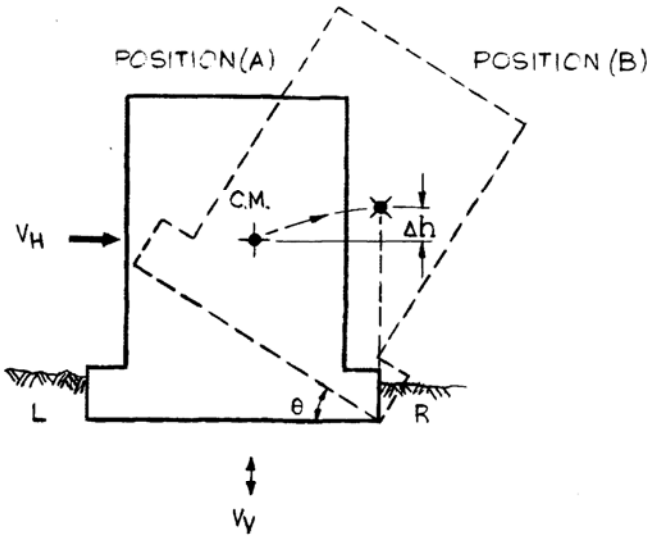
MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT
DIESEL GENERATOR BUILDING RESPONSE LOADS DUE TO VERTICAL OBE FIGURE 3.7-54



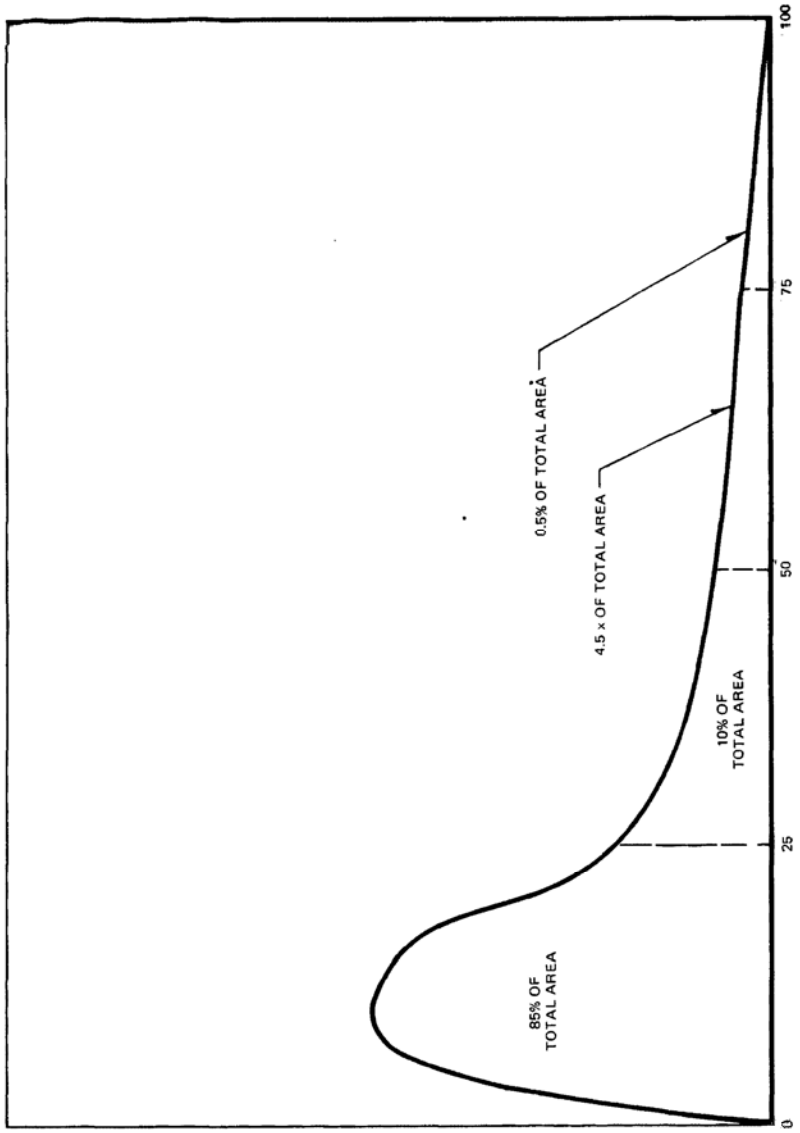
MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT

MODIFIED RESPONSE SPECTRUM CURVE

FIGURE 3.7-55

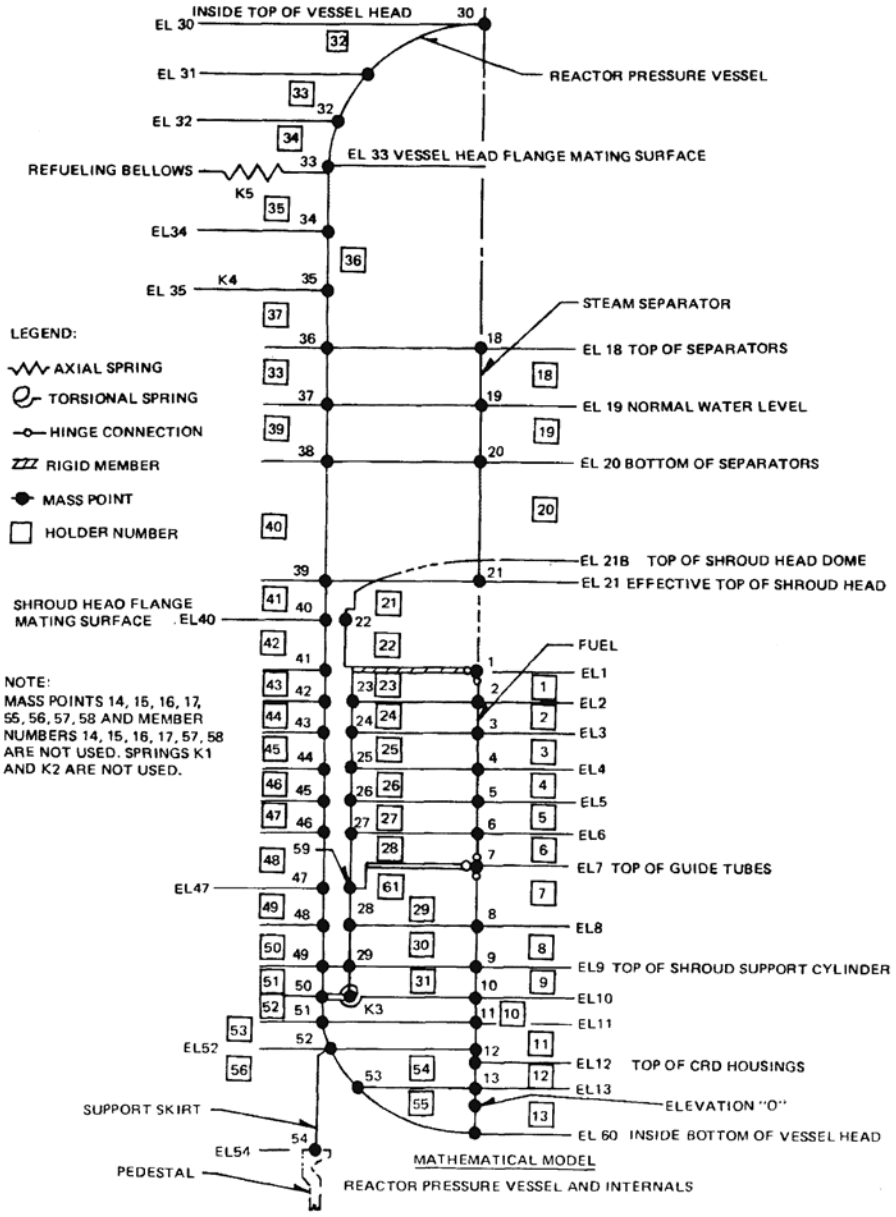


MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT
STRUCTURAL POSITION FOR EVALUATING OVERTURNING DURING SEISMIC EVENT FIGURE 3.7-56

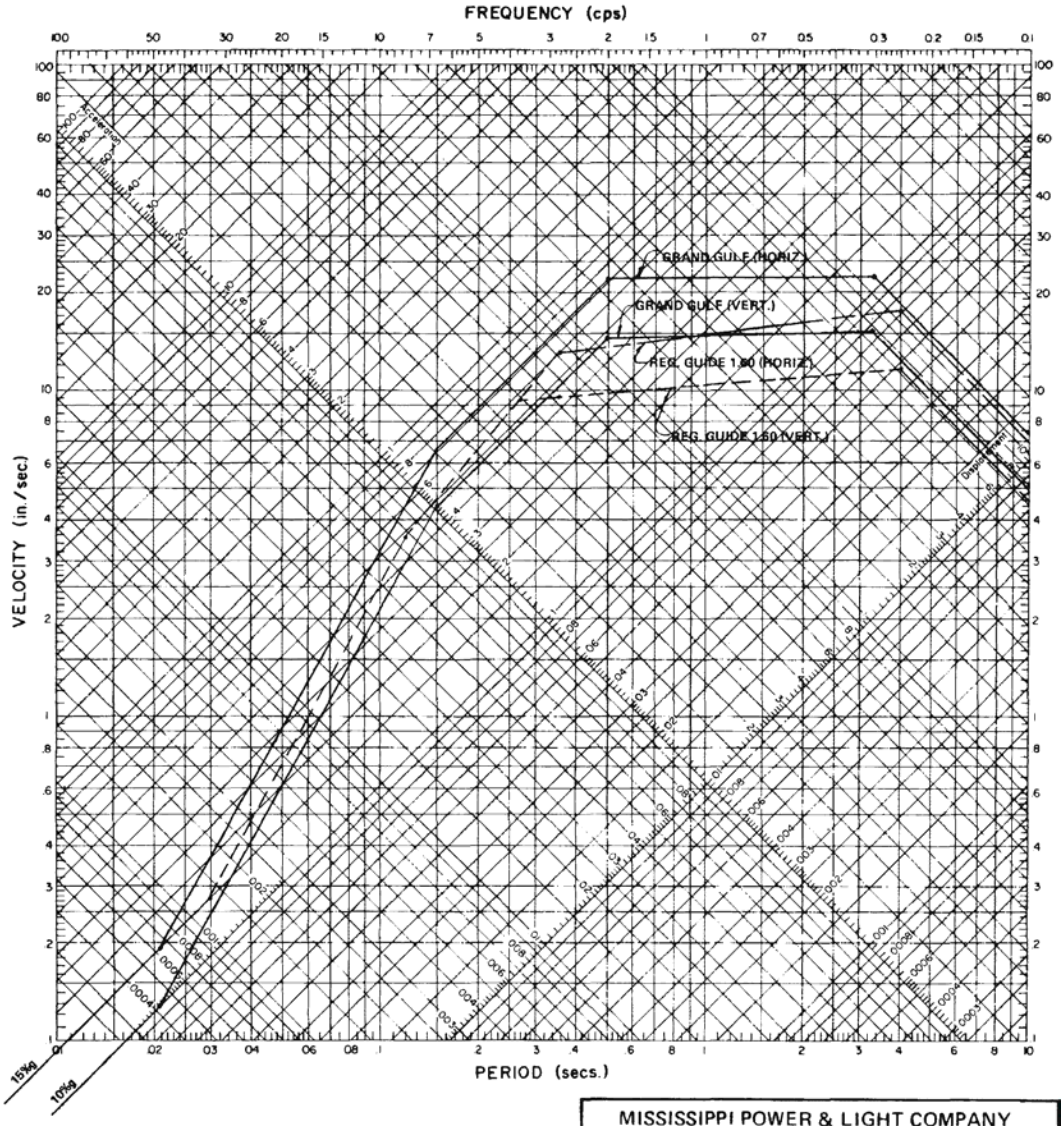


MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT

DENSITY OF STRESS REVERSALS
FIGURE 3.7-57



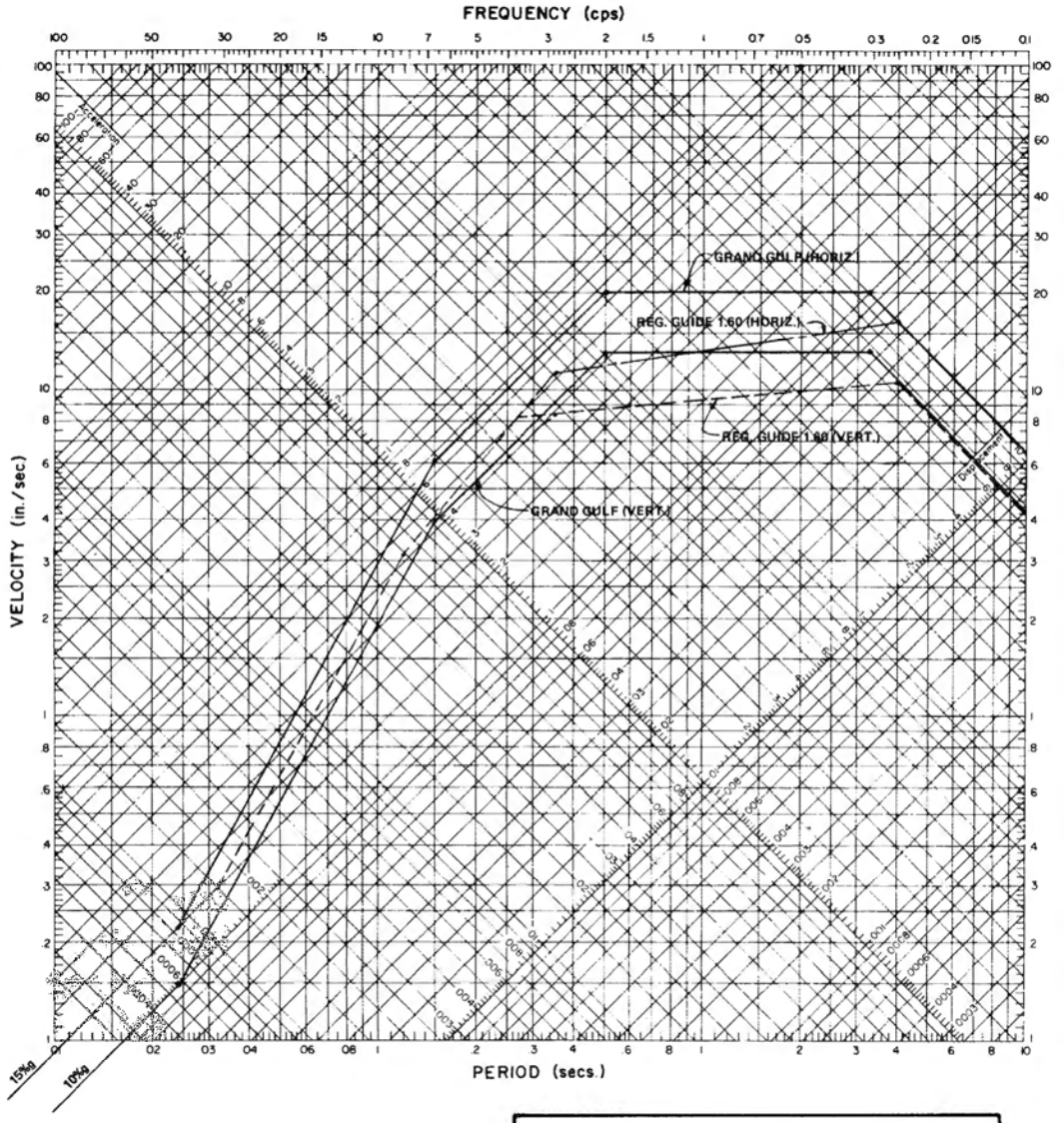
MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT	REACTOR PRESSURE VESSEL AND INTERNAL SEISMIC MODEL (GRAND GULF) FIGURE 3.7-58



MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT

COMPARISON OF SSE DESIGN SPECTRA
0.5% DAMPING
MEAN

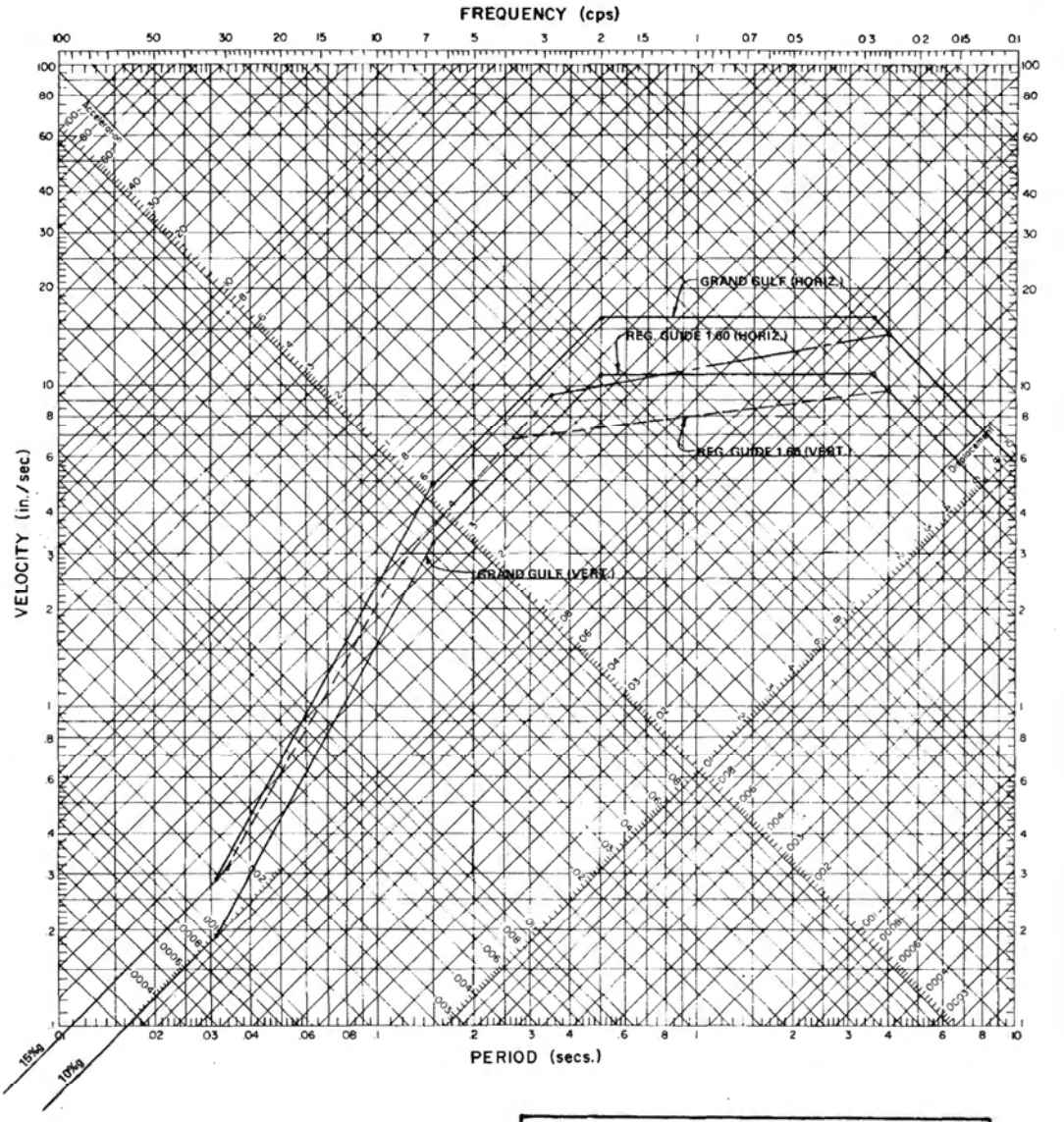
FIGURE 3.7-59



MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT

COMPARISON OF SSE DESIGN SPECTRA
1% DAMPING
MEAN

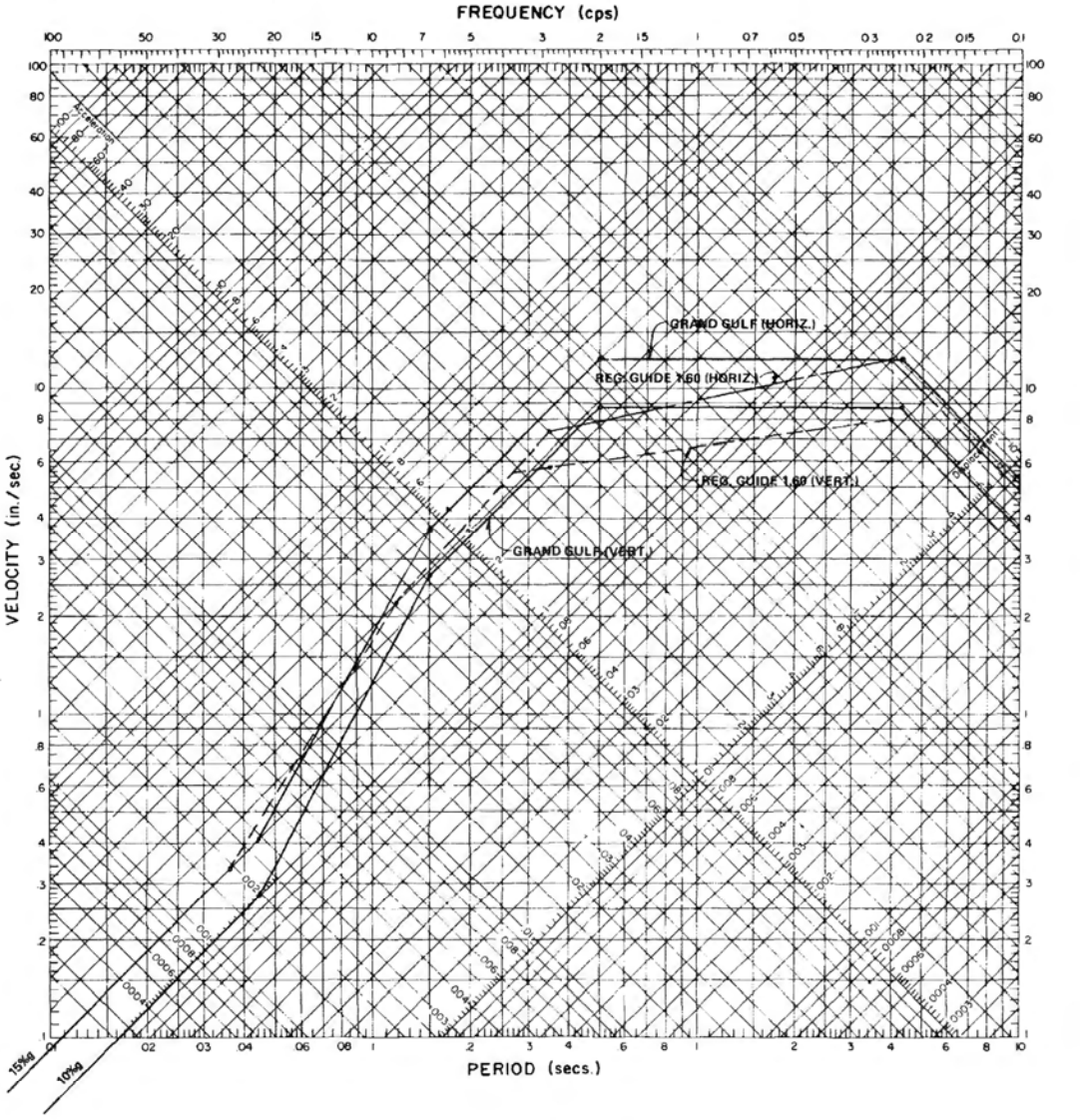
FIGURE 3.7-60



MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT

COMPARISON OF SSE DESIGN SPECTRA
2% DAMPING
MEAN

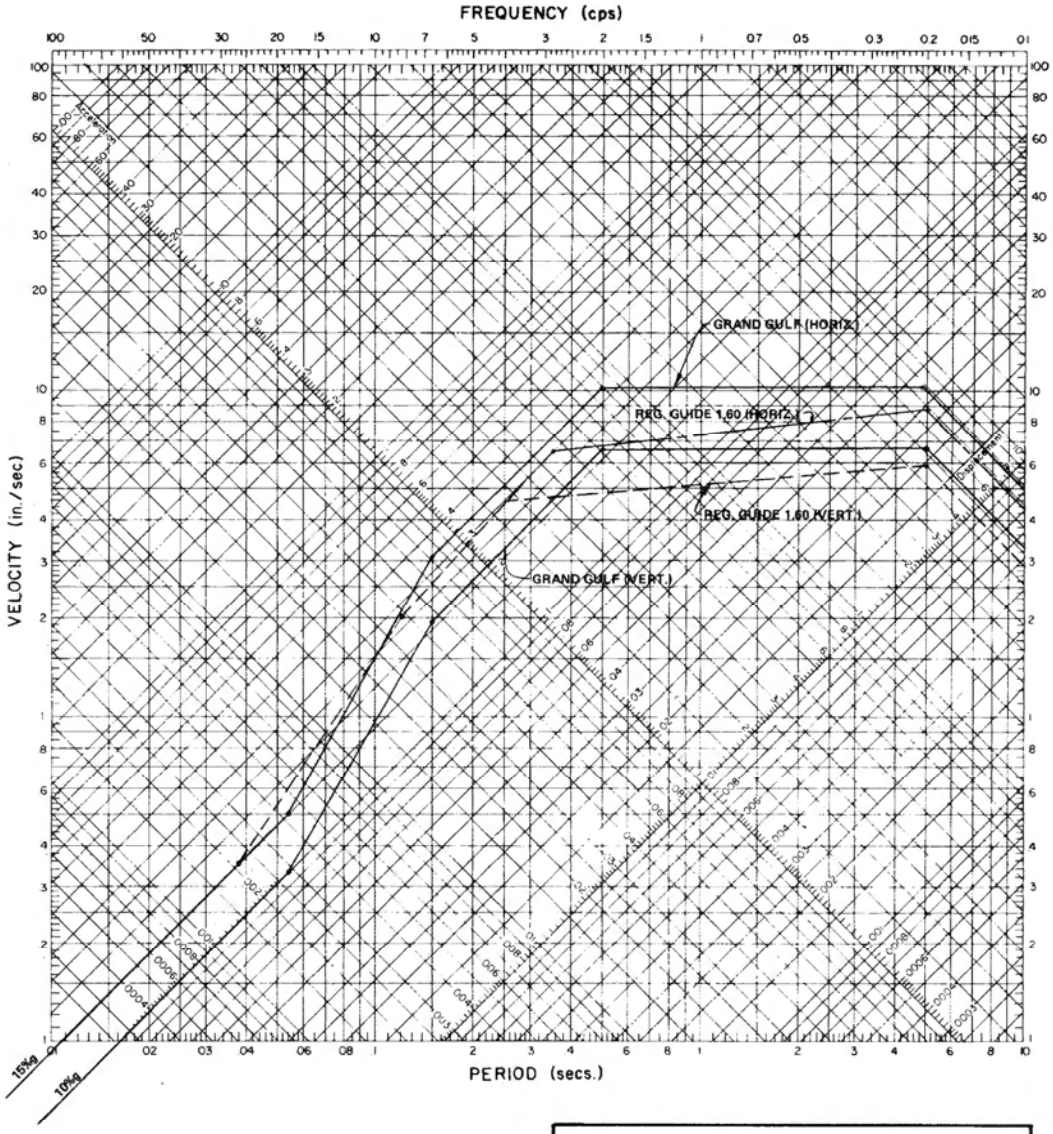
FIGURE 3.7-61



MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT

COMPARISON OF SSE DESIGN SPECTRA
5% DAMPING
MEAN

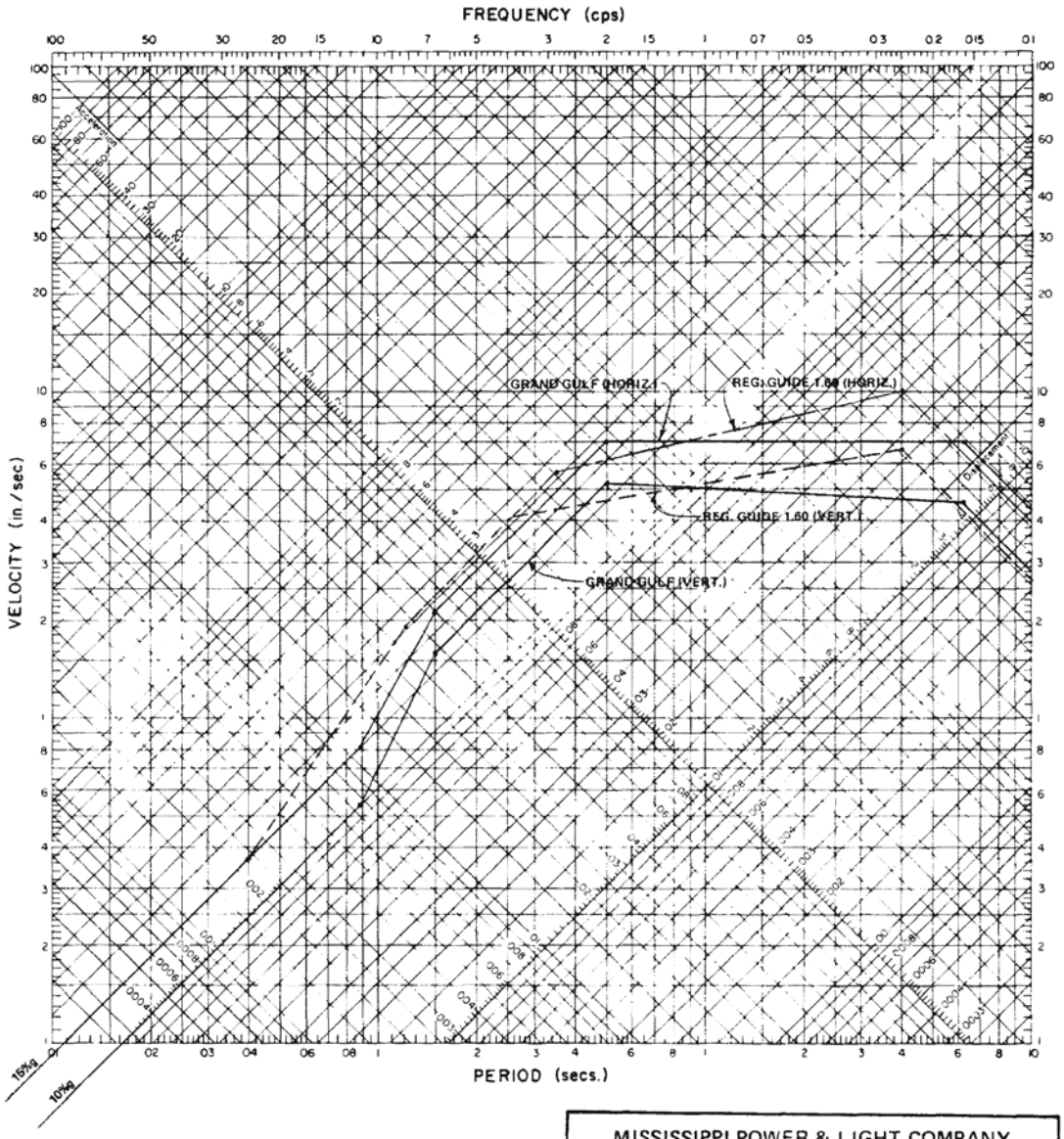
FIGURE 3.7-62



MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT

COMPARISON OF SSE DESIGN SPECTRA
7% DAMPING
MEAN

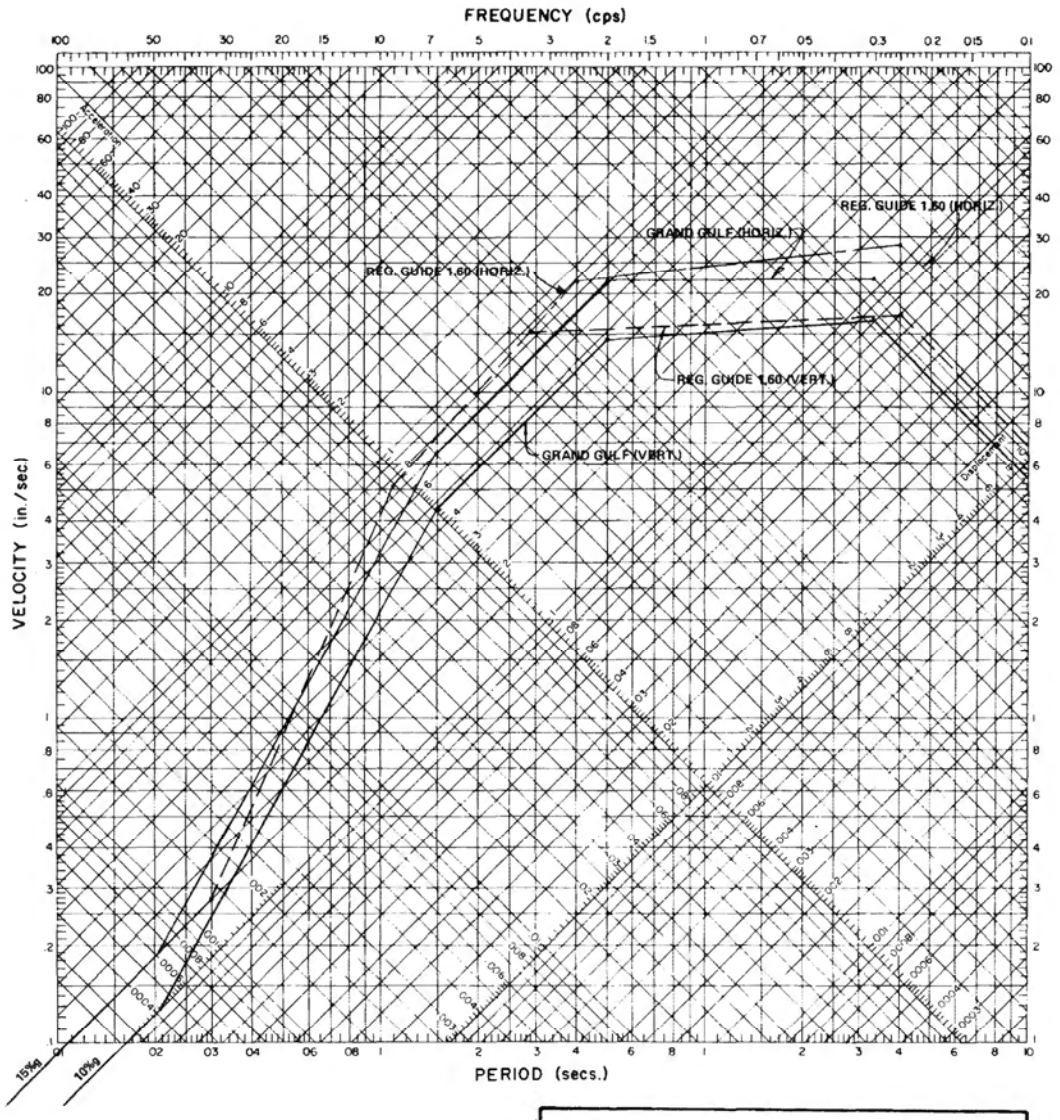
FIGURE 3.7-63



MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT

COMPARISON OF SSE DESIGN SPECTRA
10% DAMPING
MEAN

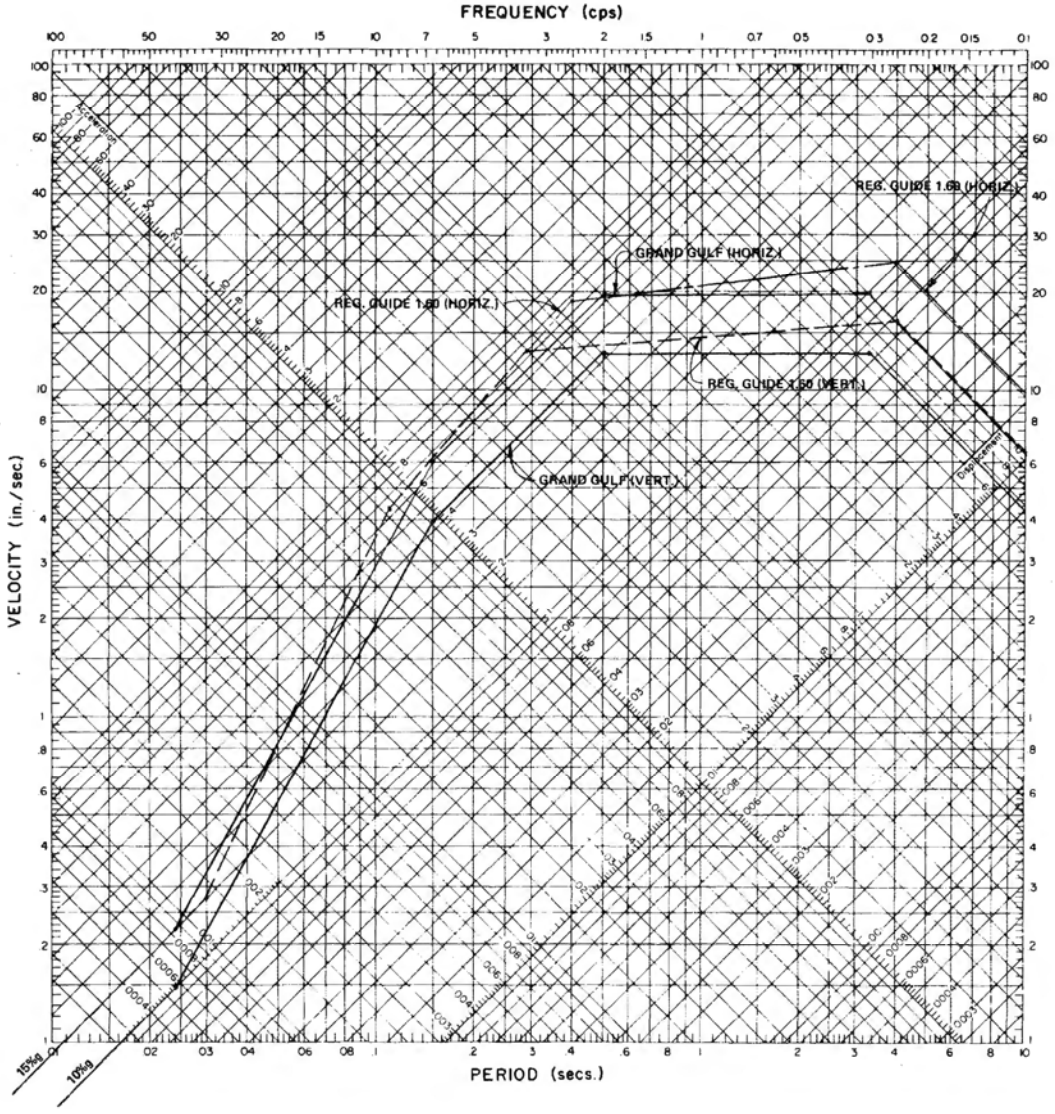
FIGURE 3.7-64



MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT

COMPARISON OF SSE DESIGN SPECTRA
0.5% DAMPING
MEAN PLUS ONE STANDARD DEVIATION

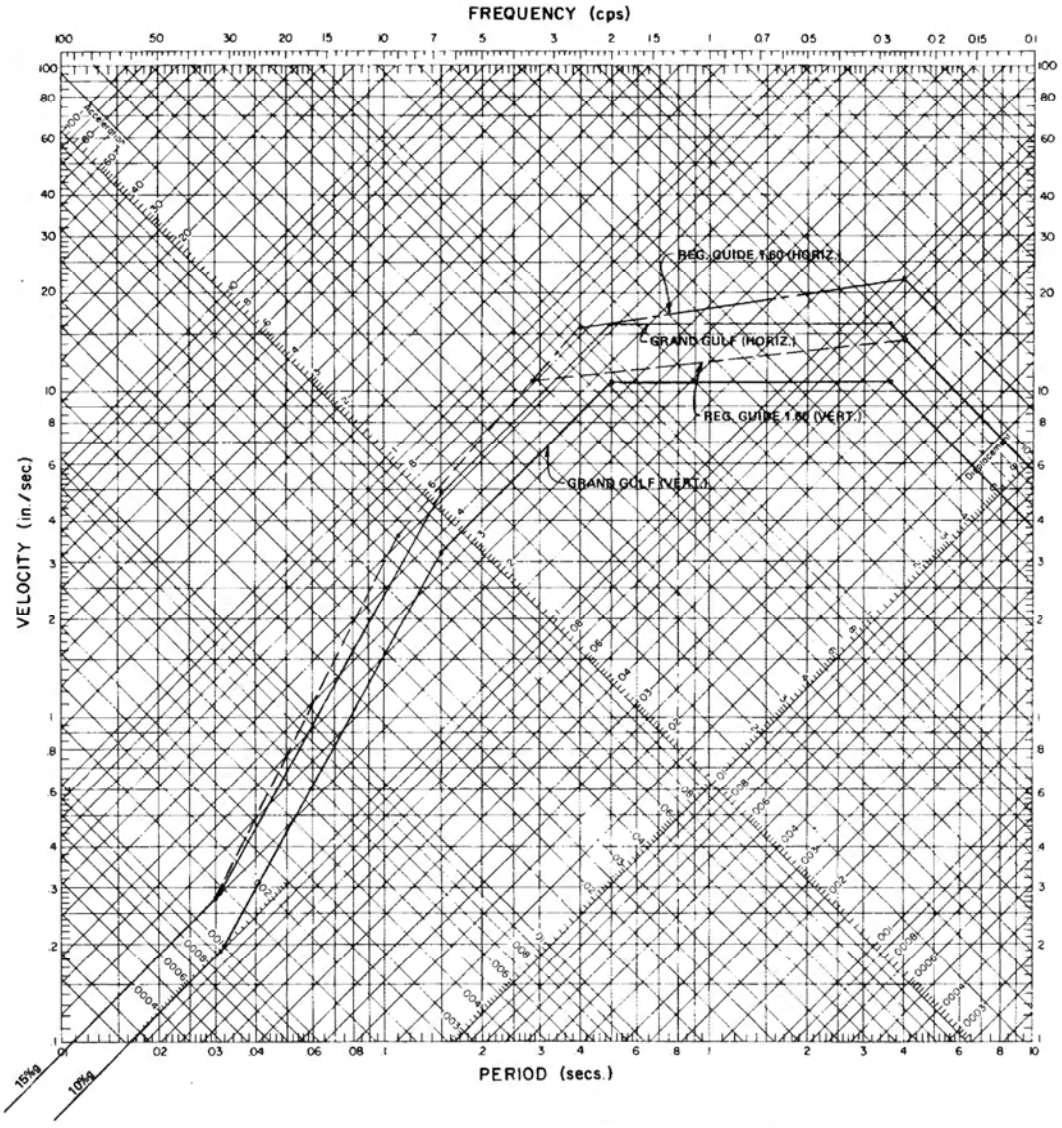
FIGURE 3.7-65



MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT

COMPARISON OF SSE DESIGN SPECTRA
1% DAMPING
MEAN PLUS ONE STANDARD DEVIATION

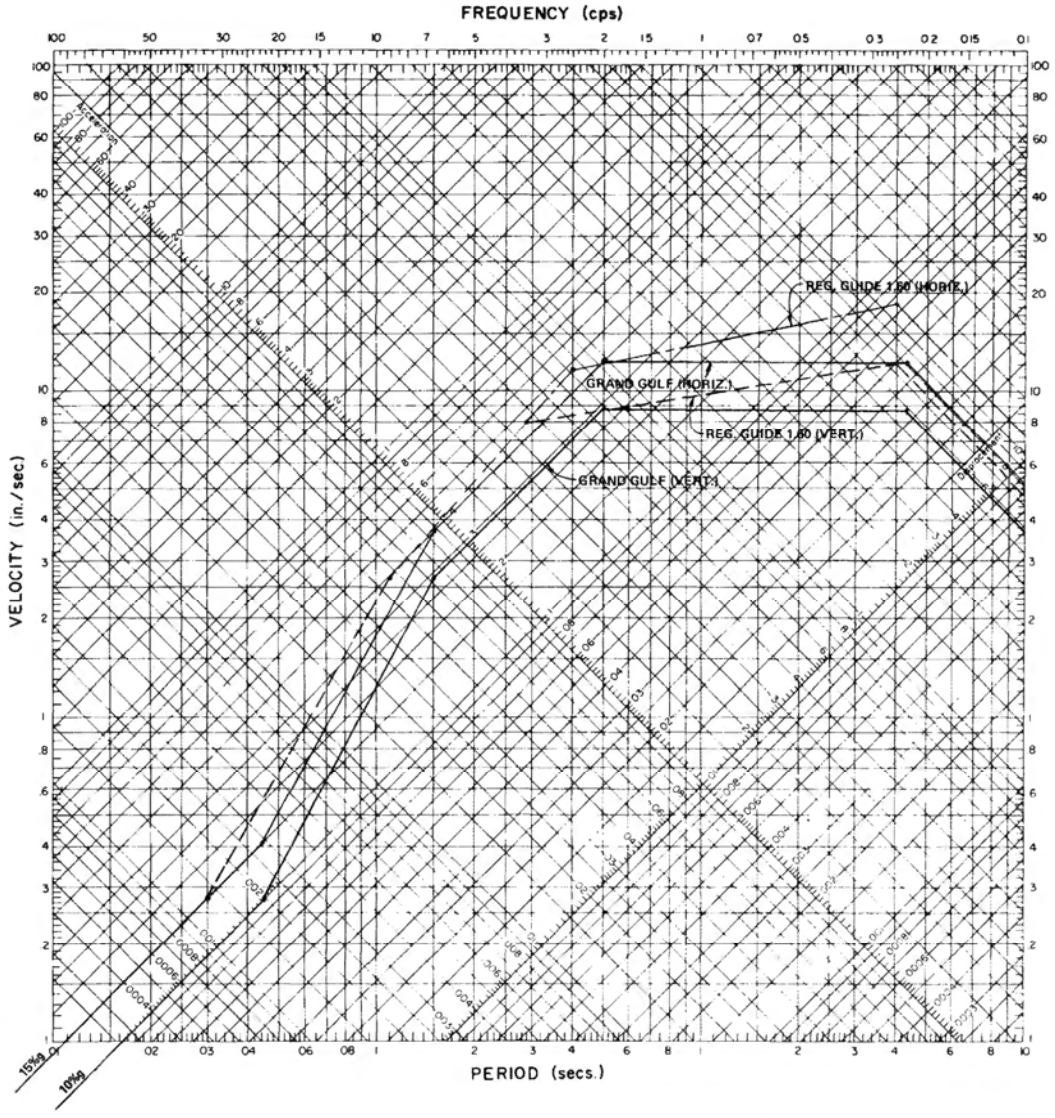
FIGURE 3.7-66



MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT

COMPARISON OF SSE DESIGN SPECTRA
2% DAMPING
MEAN PLUS ONE STANDARD DEVIATION

FIGURE 3.7-67



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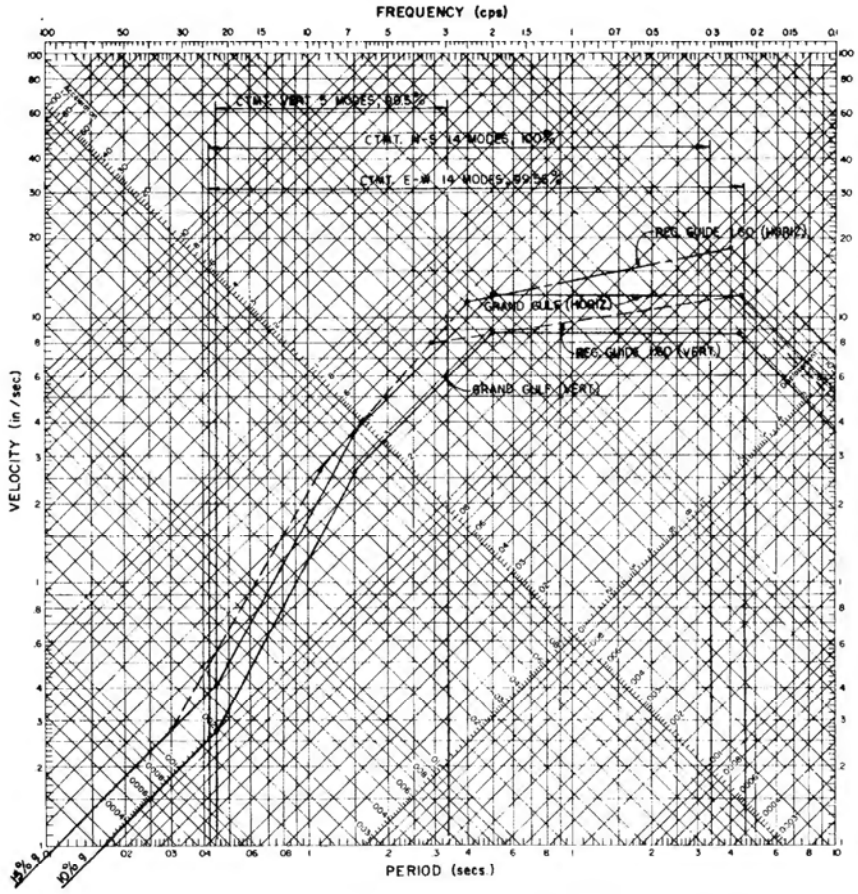
COMPARISON OF SSE DESIGN SPECTRA
5% DAMPING
MEAN PLUS ONE STANDARD DEVIATION

FIGURE 3.7-68



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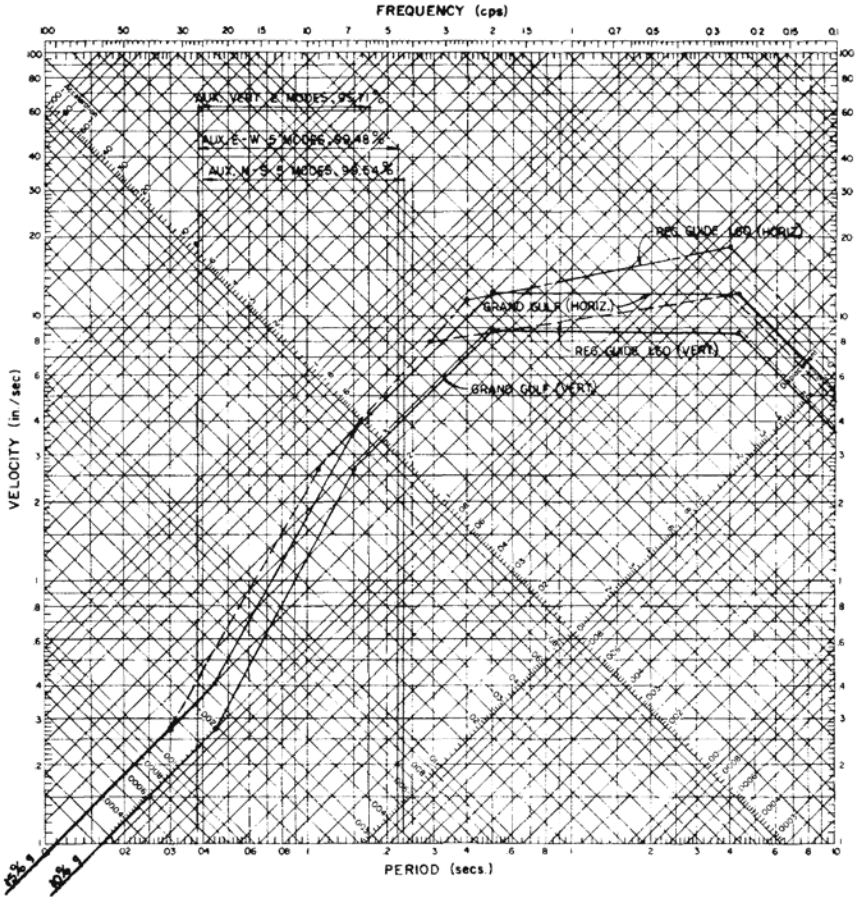
CONTAINMENT BUILDING
PRIMARY RESPONSE MODES
FIGURE 3.7-68a



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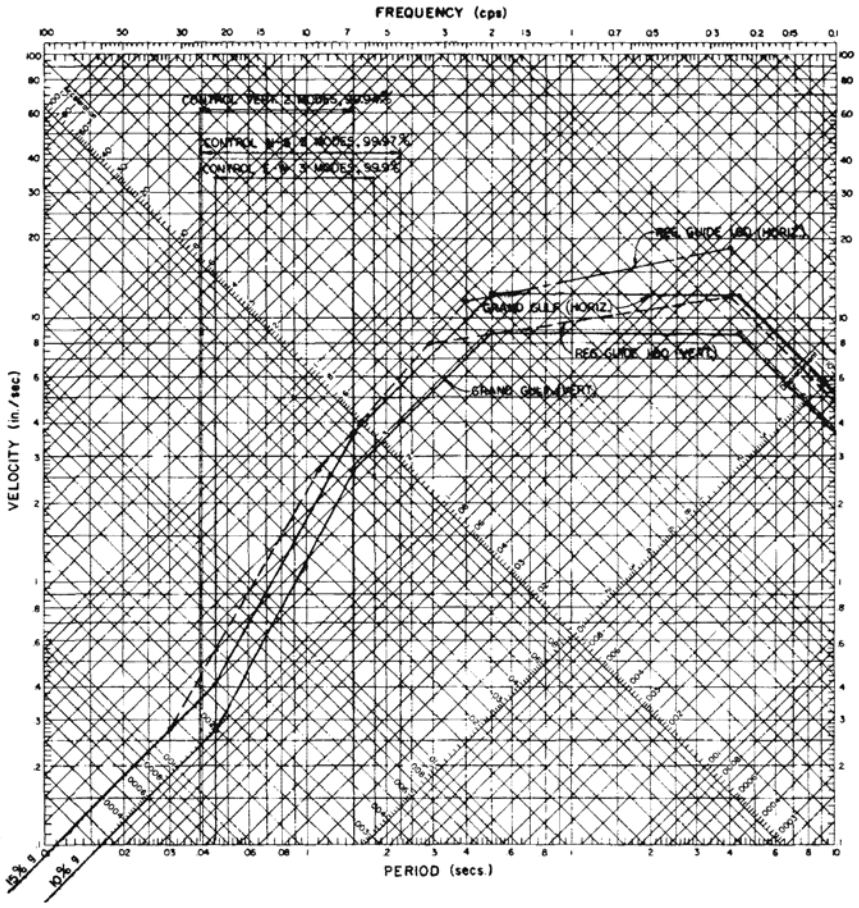
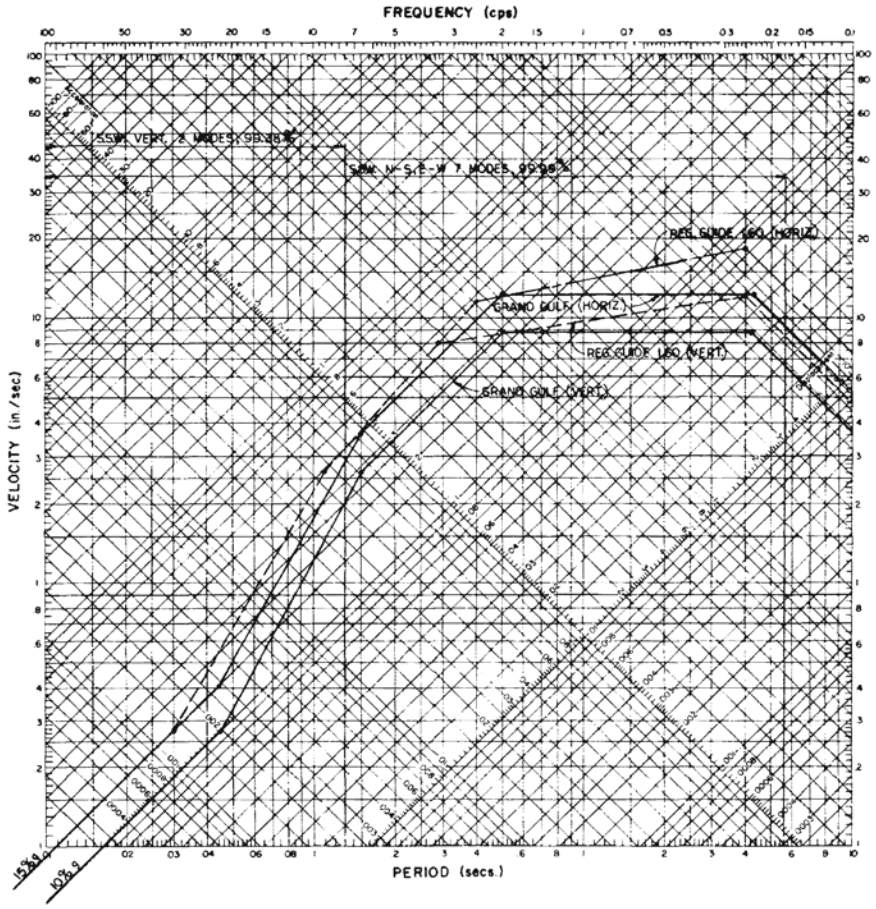
AUXILIARY BUILDING
PRIMARY RESPONSE MODES
FIGURE 3.7-68b



MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT
CONTROL BUILDING PRIMARY RESPONSE MODES FIGURE 3.7-68c



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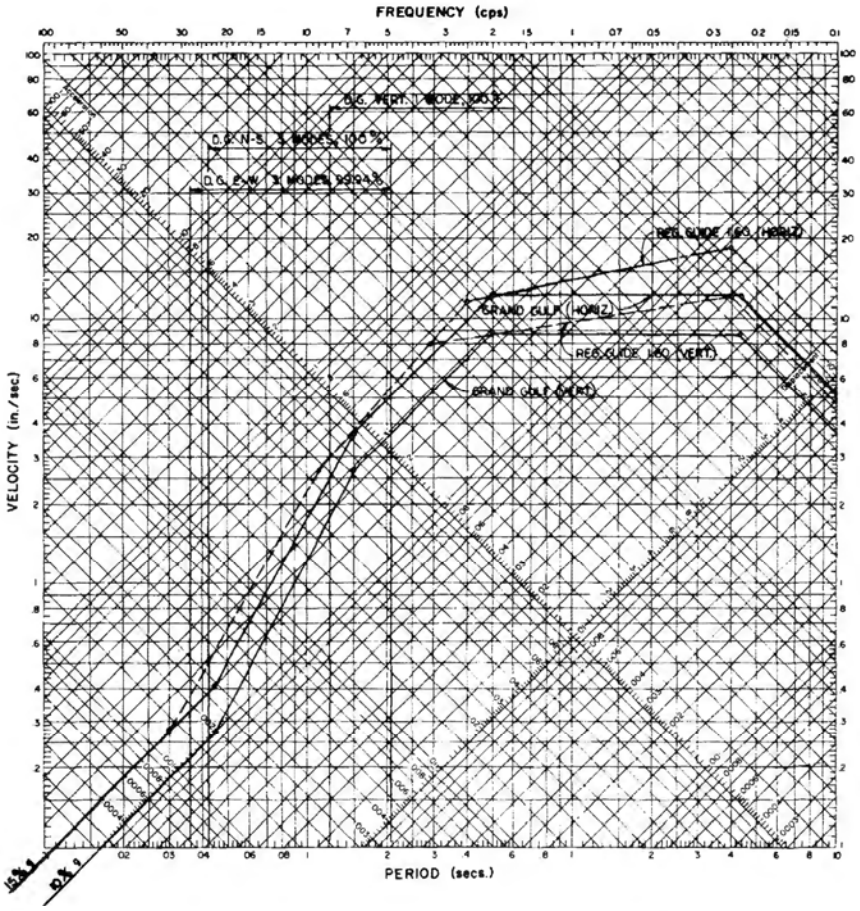
STANDBY SERVICE WATER
COOLING TOWER BASIN
PRIMARY RESPONSE MODES

FIGURE 3.7-68d



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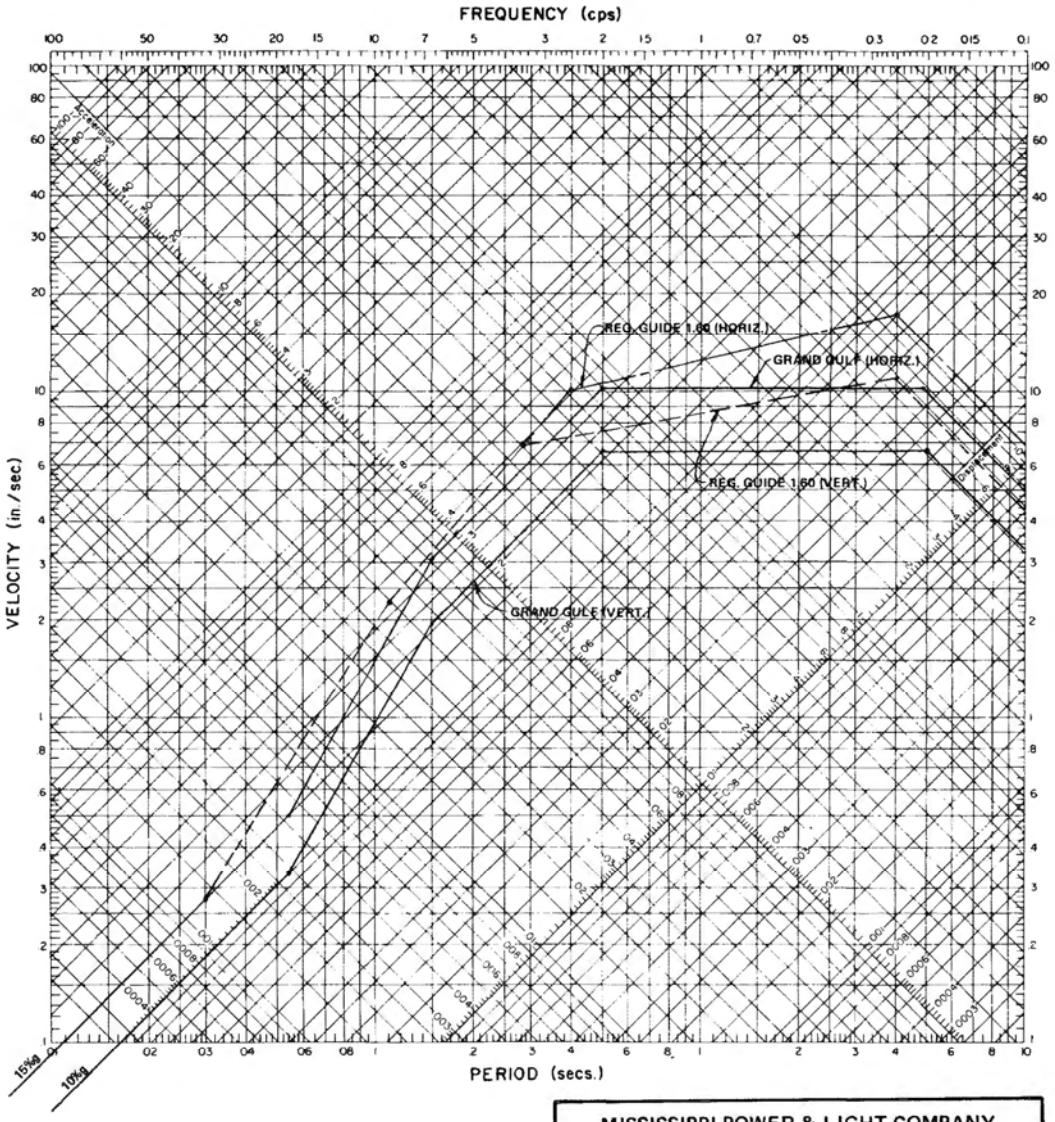
DIESEL GENERATOR BUILDING
PRIMARY RESPONSE MODES
FIGURE 3.7-68e



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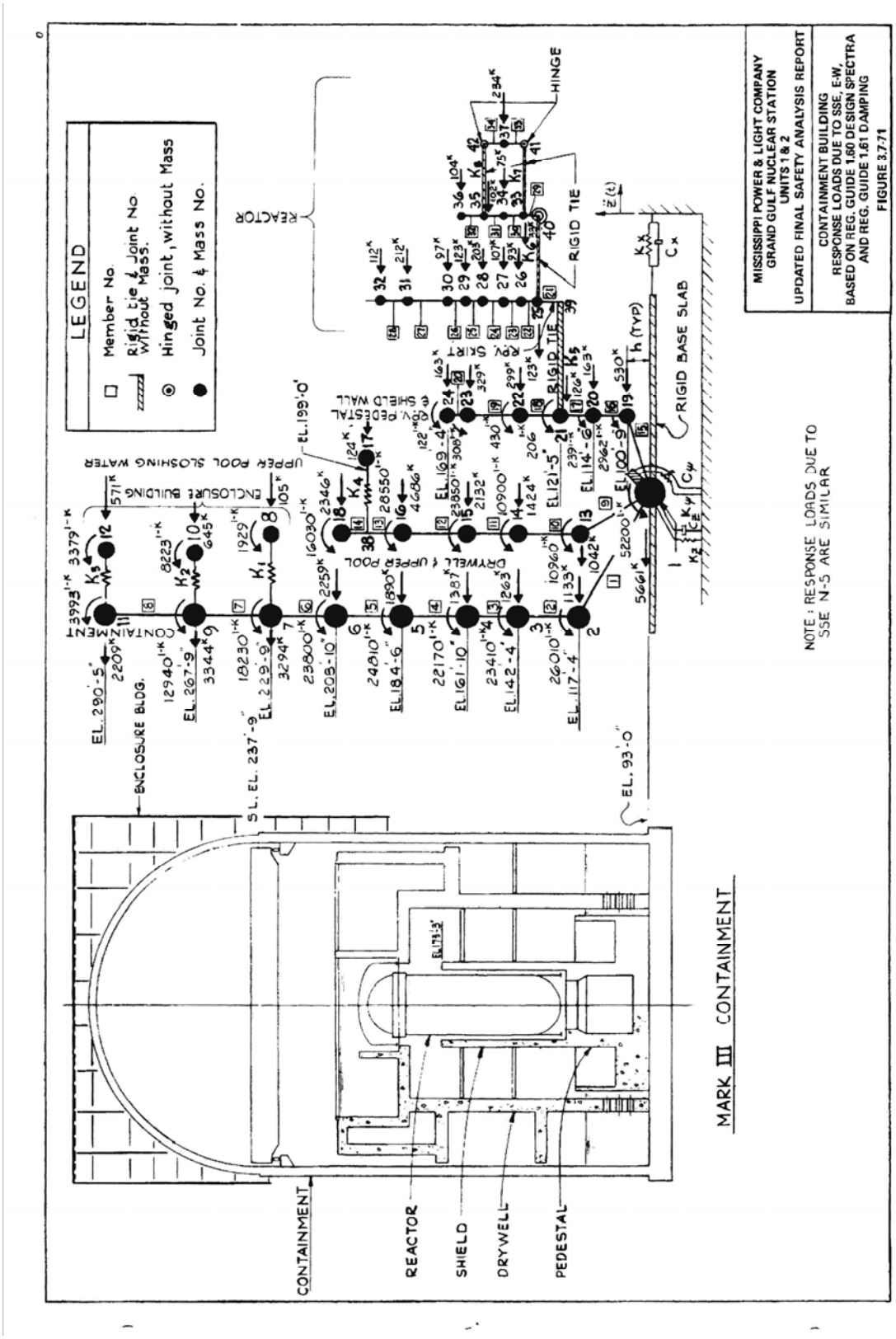
COMPARISON OF SSE DESIGN SPECTRA
7% DAMPING
MEAN PLUS ONE STANDARD DEVIATION

FIGURE 3.7-69


LEGEND
Member No.
Rigid tie & Joint No without Mass.
Hinged joint, without Mass
Joint No. & Mass No.
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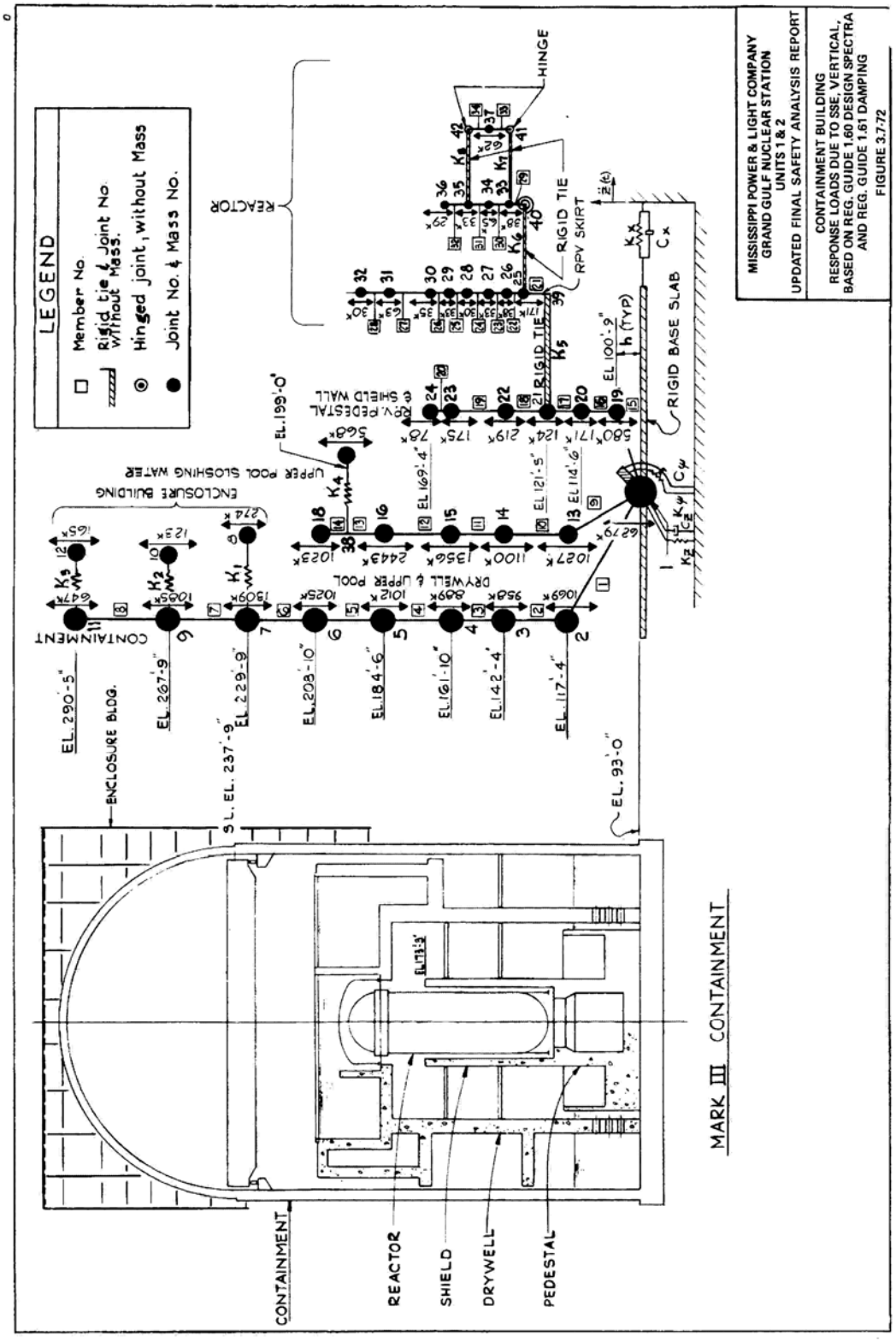
CONTAINMENT BUILDING
RESPONSE LOADS DUE TO SSE, E-W,
BASED ON REG. GUIDE 1.60 DESIGN SPECTRA
AND REG. GUIDE 1.61 DAMPING
FIGURE 3.7-71
REACTOR
HINGE
RIGID TIE
RIGID BASE SLAB
ENCLOSURE BUILDING
UPPER POOL SLOSHING WATER
CONTAINMENT
DRYWELL & UPPER POOL
EL. 290'-5"
EL. 267'-9"
EL. 237'-9"
EL. 206'-10"
EL. 184'-6"
EL. 161'-10"
EL. 142'-4"
EL. 117'-4"
EL. 93'-0"
EL. 199'-0"
EL. 169'-4"
EL. 121'-5"
EL. 100'-9"
NOTE: RESPONSE LOADS DUE TO SSE N-S ARE SIMILAR
MARK III CONTAINMENT
CONTAINMENT
REACTOR
SHIELD
DRYWELL
PEDESTAL


LEGEND
Member No.
Rigid tie & Joint No. without Mass.
Hinged joint, without Mass
Joint No. & Mass No.
MISSISSIPPI POWER & LIGHT COMPANY
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UNITS 1 & 2
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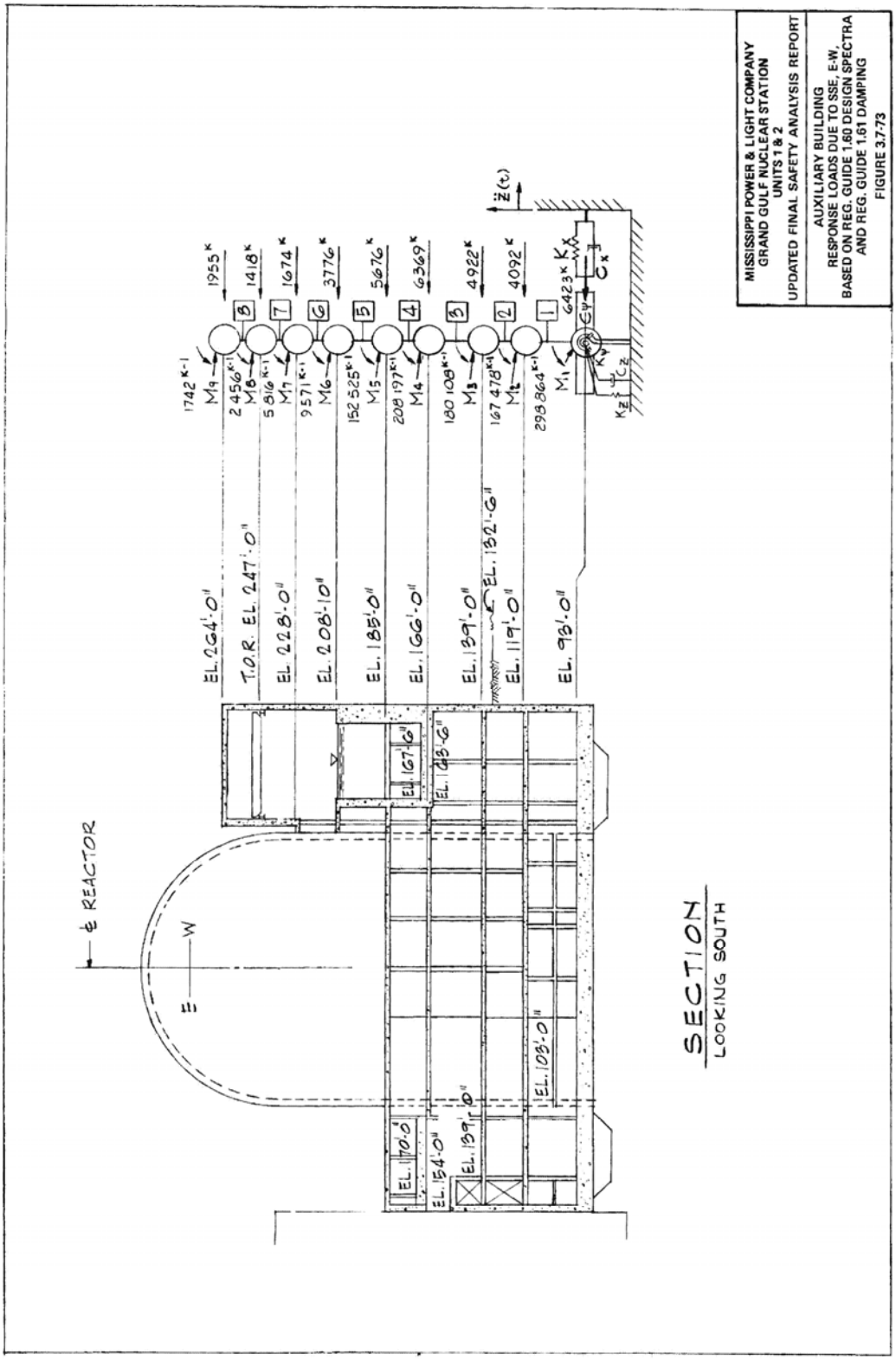
CONTAINMENT BUILDING
RESPONSE LOADS DUE TO SSE, VERTICAL, BASED ON REG. GUIDE 1.60 DESIGN SPECTRA AND REG. GUIDE 1.61 DAMPING
FIGURE 3.7-72
REACTOR
HINGE
RIGID TIE
RPV SKIRT
RIGID BASE SLAB
EL. 199'-0"
EL. 100'-9"
h (TYP)
RPV PEDESTAL
SHIELD WALL
ENCLOSURE BUILDING
UPPER POOL SLOSHING WATER
DRYWELL & UPPER POOL
CONTAINMENT
EL. 290'-5"
EL. 267'-9"
EL. 237'-9"
EL. 229'-9"
EL. 208'-10"
EL. 184'-6"
EL. 161'-10"
EL. 142'-4"
EL. 117'-4"
EL. 93'-0"
ENCLOSURE BLDG.
S.L.
MARK III CONTAINMENT
CONTAINMENT
REACTOR
SHIELD
DRYWELL
PEDESTAL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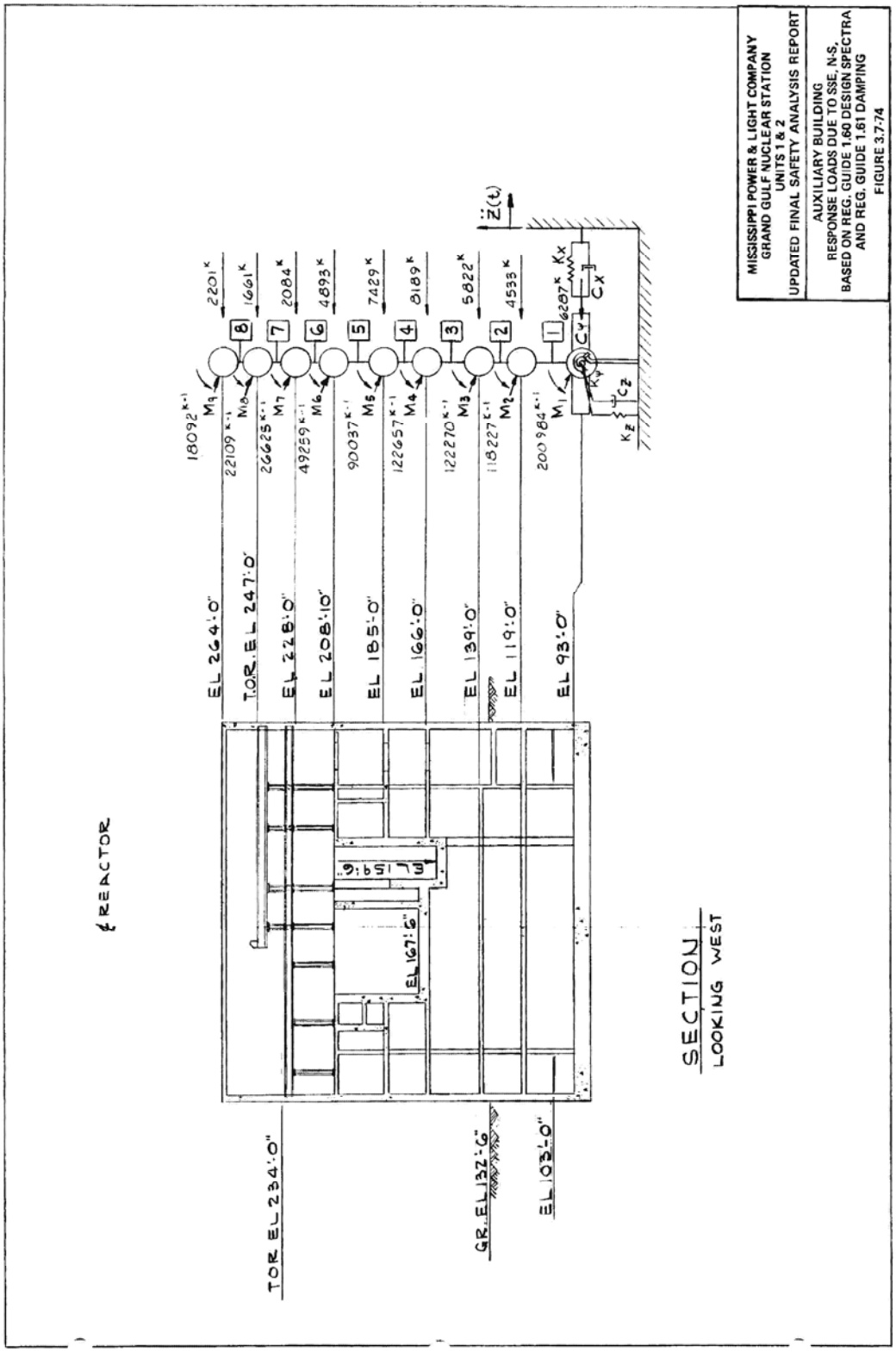
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT
AUXILIARY BUILDING
RESPONSE LOADS DUE TO SSE E-W,
BASED ON REG. GUIDE 1.60 DESIGN SPECTRA
AND REG. GUIDE 1.61 DAMPING
FIGURE 3.7-73
SECTION
LOOKING SOUTH
℄ REACTOR
W
E
T.O.R. EL. 247'-0"
EL. 264'-0"
EL. 228'-0"
EL. 208'-10"
EL. 185'-0"
EL. 160'-0"
EL. 139'-0"
EL. 119'-0"
EL. 98'-0"
EL. 182'-6"
EL. 167'-10"
EL. 148'-6"
EL. 105'-0"
EL. 70'-0"
EL. 154'-0"
EL. 139'-0"
1955K
1418K
1674K
3776K
5676K
9369K
4922K
4092K
6423K
1742K-1
2456K-1
5816K-1
9571K-1
152 525K-1
208 197K-1
180 108K-1
167 478K-1
293 864K-1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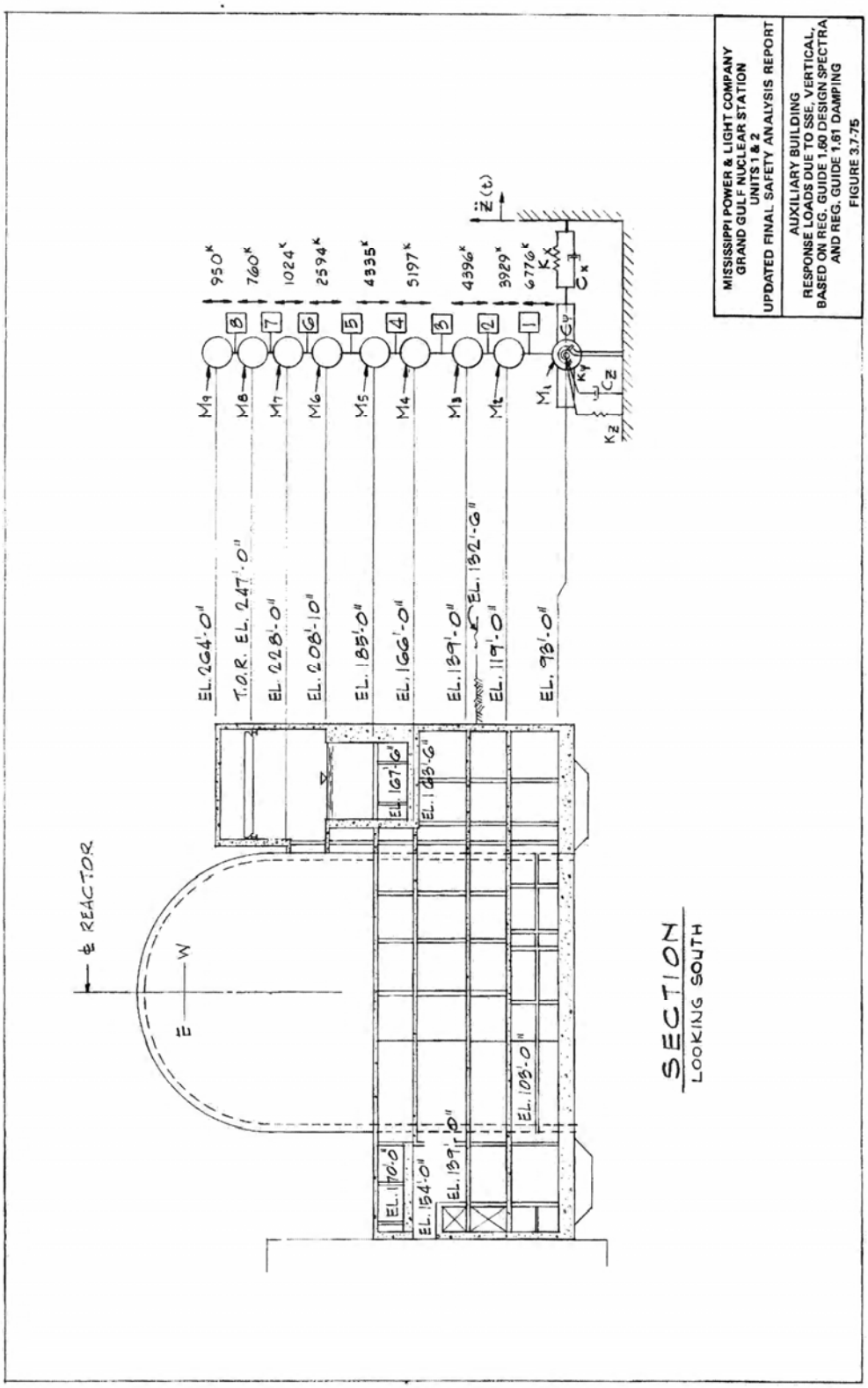
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT
AUXILIARY BUILDING
RESPONSE LOADS DUE TO SSE, N-S,
BASED ON REG. GUIDE 1.60 DESIGN SPECTRA
AND REG. GUIDE 1.61 DAMPING
FIGURE 3.7-74
SECTION
LOOKING WEST
EL 264'-0"
T.O.R. EL 247'-0"
EL 228'-0"
EL 208'-10"
EL 185'-0"
EL 166'-0"
EL 139'-0"
EL 119'-0"
EL 93'-0"
EL 167'-6"
TOR EL 234'-0"
GR. EL 132'-6"
EL 103'-0"
REACTOR
Z̈(t)
Kx
Cx
Cy
Kz
Cz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UNITS 1 & 2
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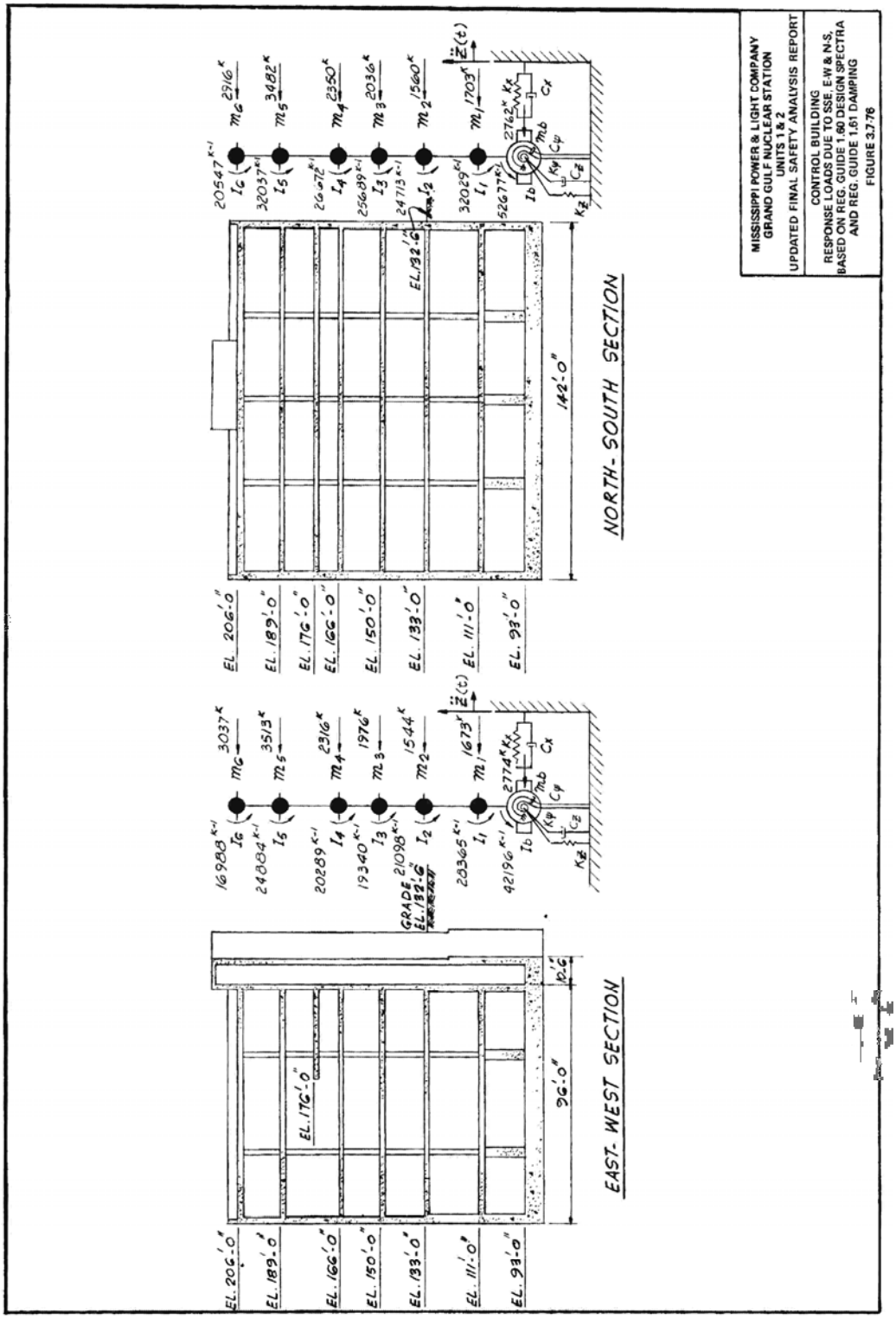
AUXILIARY BUILDING
RESPONSE LOADS DUE TO SSE, VERTICAL, BASED ON REG. GUIDE 1.60 DESIGN SPECTRA AND REG. GUIDE 1.61 DAMPING
FIGURE 3.7-75
950K
760K
1024K
2594K
4335K
5197K
4396K
3929K
4776K
EL. 264'-0"
T.O.R. EL. 247'-0"
EL. 228'-0"
EL. 208'-10"
EL. 185'-0"
EL. 166'-0"
EL. 139'-0"
EL. 119'-0"
EL. 98'-0"
EL. 152'-6"
EL. 167'-0"
EL. 148'-6"
EL. 105'-0"
EL. 70'-0"
EL. 154'-0"
EL. 139'-0"
℄ REACTOR
W
E
SECTION
LOOKING SOUTH


MISSISSIPPI POWER & LIGHT COMPANY
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UNITS 1 & 2
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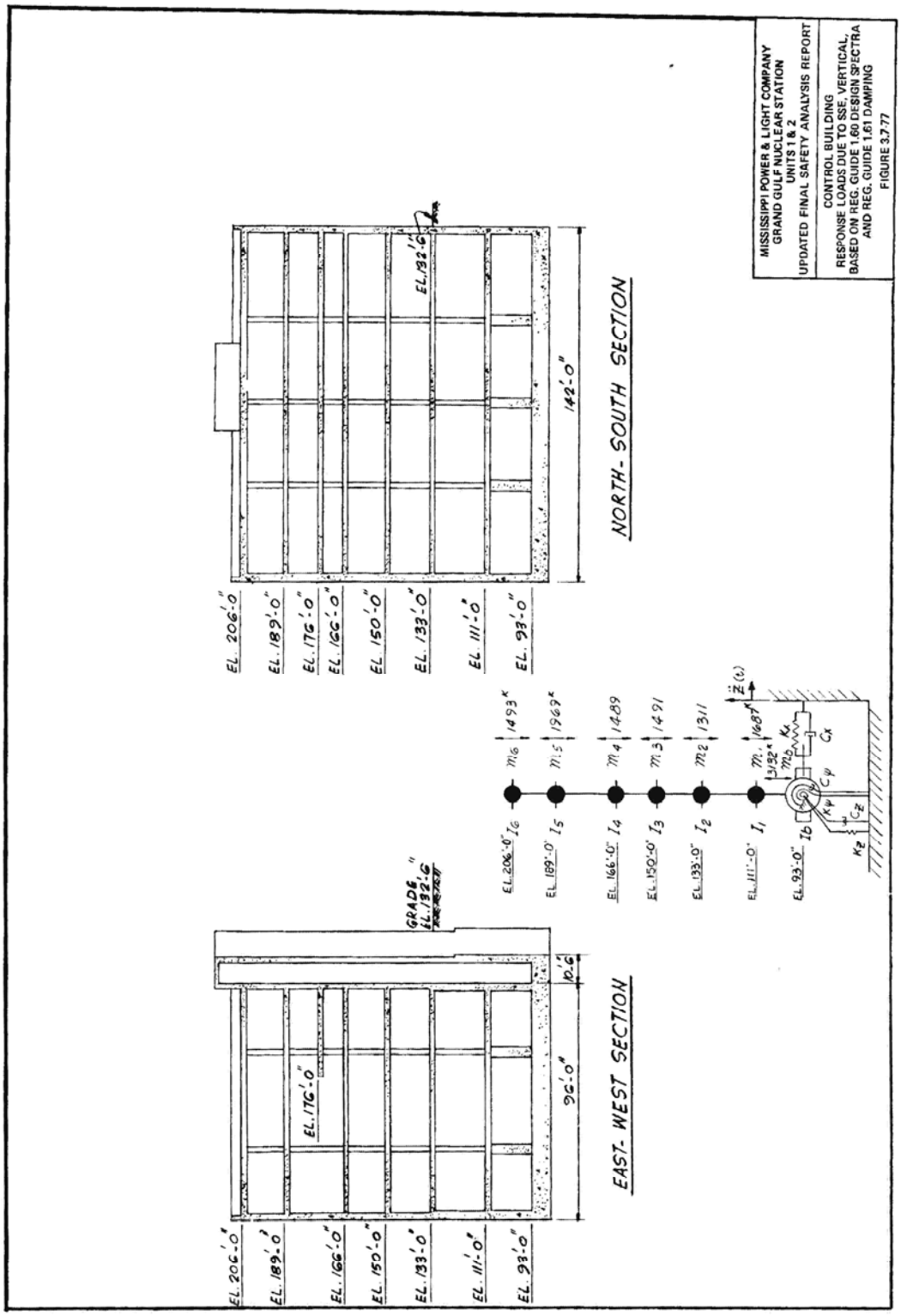
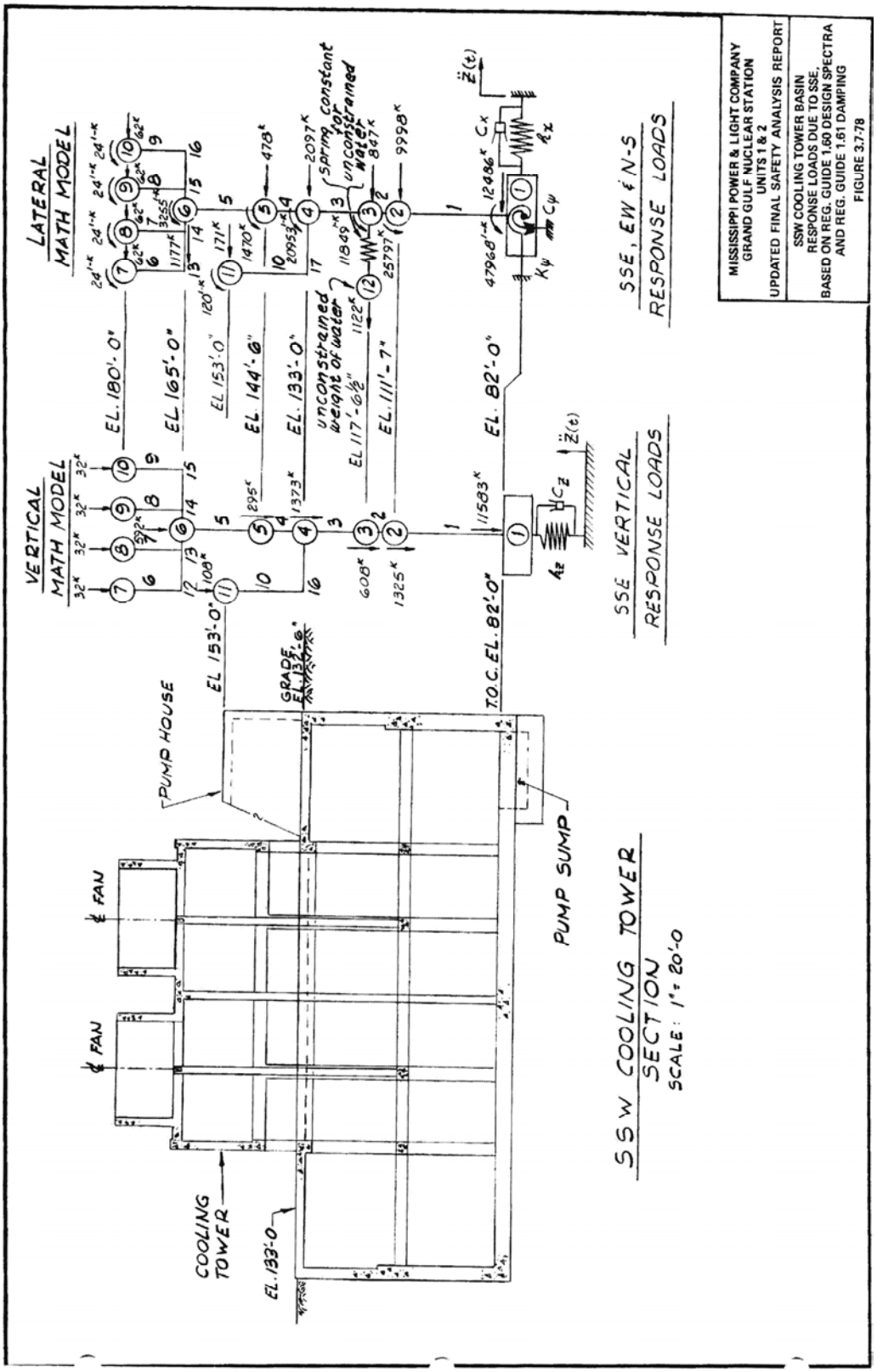
CONTROL BUILDING
RESPONSE LOADS DUE TO SSE, E-W & N-S, BASED ON REG. GUIDE 1.60 DESIGN SPECTRA AND REG. GUIDE 1.61 DAMPING
FIGURE 3.7-76
NORTH-SOUTH SECTION
EAST-WEST SECTION
EL. 206'-0"
EL. 189'-0"
EL. 176'-0"
EL. 166'-0"
EL. 150'-0"
EL. 133'-0"
EL. 111'-0"
EL. 93'-0"
142'-0"
96'-0"
GRADE EL. 132'-6"


MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT
CONTROL BUILDING
RESPONSE LOADS DUE TO SSE, VERTICAL,
BASED ON REG. GUIDE 1.60 DESIGN SPECTRA
AND REG. GUIDE 1.61 DAMPING
FIGURE 3.7-77
NORTH-SOUTH SECTION
EAST-WEST SECTION
EL. 206'-0"
EL. 189'-0"
EL. 176'-0"
EL. 166'-0"
EL. 150'-0"
EL. 133'-0"
EL. 111'-0"
EL. 93'-0"
142'-0"
92'-0"
GRADE EL. 132'-6"
EL. 132'-6"
m6 1493K
m5 1968K
m4 1489
m3 1491
m2 1311
m1 1687K
I6
I5
I4
I3
I2
I1
I0
Z(t)
Kx
Cx
Cψ
Kψ
Cz
Kz


LATERAL MATH MODEL
VERTICAL MATH MODEL
EL. 180'-0"
EL 165'-0"
EL 153'-0"
EL 144'-6"
EL. 133'-0"
EL 117'-6½"
EL. 111'-7"
EL. 82'-0"
EL 153'-0"
GRADE EL 137'-6"
T.O.C. EL. 82'-0"
EL. 133'-0"
PUMP HOUSE
PUMP SUMP
COOLING TOWER
FAN
FAN
unconstrained weight of water
spring constant
uncoupled
SSW COOLING TOWER
SECTION
SCALE: 1" = 20'-0
SSE VERTICAL
RESPONSE LOADS
SSE, EW & NS
RESPONSE LOADS
MISSISSIPPI POWER & LIGHT COMPANY
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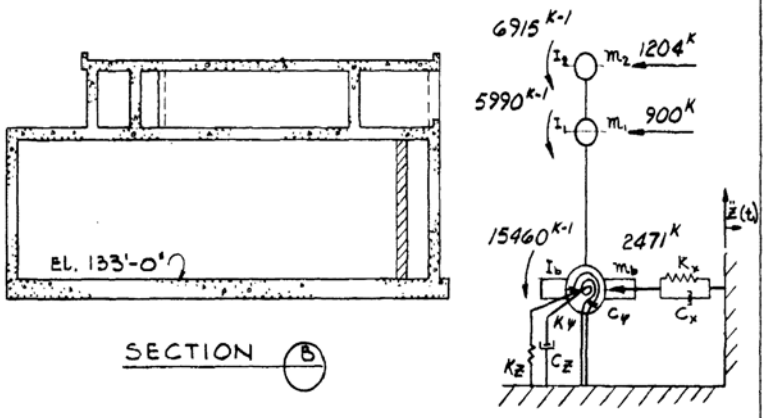
SSW COOLING TOWER BASIN
RESPONSE LOADS DUE TO SSE,
BASED ON REG. GUIDE 1.60 DESIGN SPECTRA
AND REG. GUIDE 1.61 DAMPING
FIGURE 3.7-78



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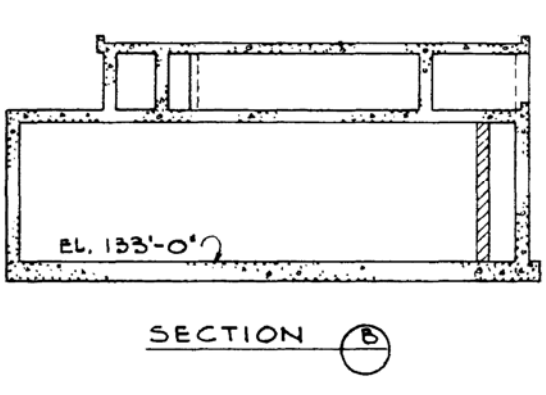
DIESEL GENERATOR BUILDING
RESPONSE LOADS DUE TO SSE, E-W, BASED
ON REG. GUIDE 1.60 DESIGN SPECTRA
AND REG. GUIDE 1.61 DAMPING
FIGURE 3.7 - 79



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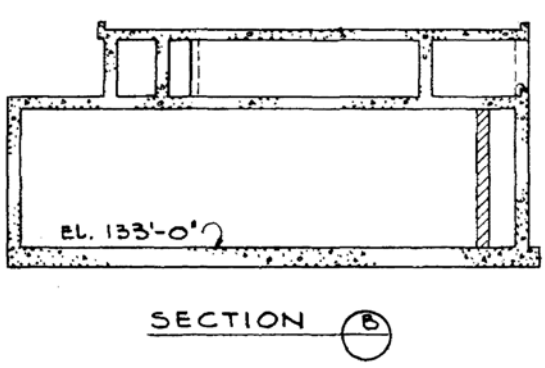
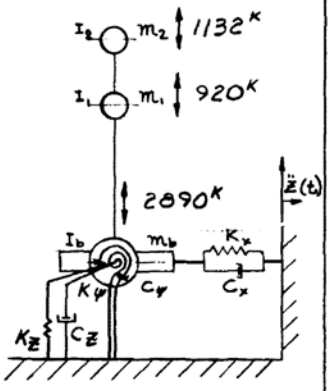
DIESEL GENERATOR BUILDING
RESPONSE LOADS DUE TO SSE, N-S, BASED
ON REG. GUIDE 1.60 DESIGN SPECTRA
AND REG. GUIDE 1.61 DAMPING

FIGURE 3.7-80





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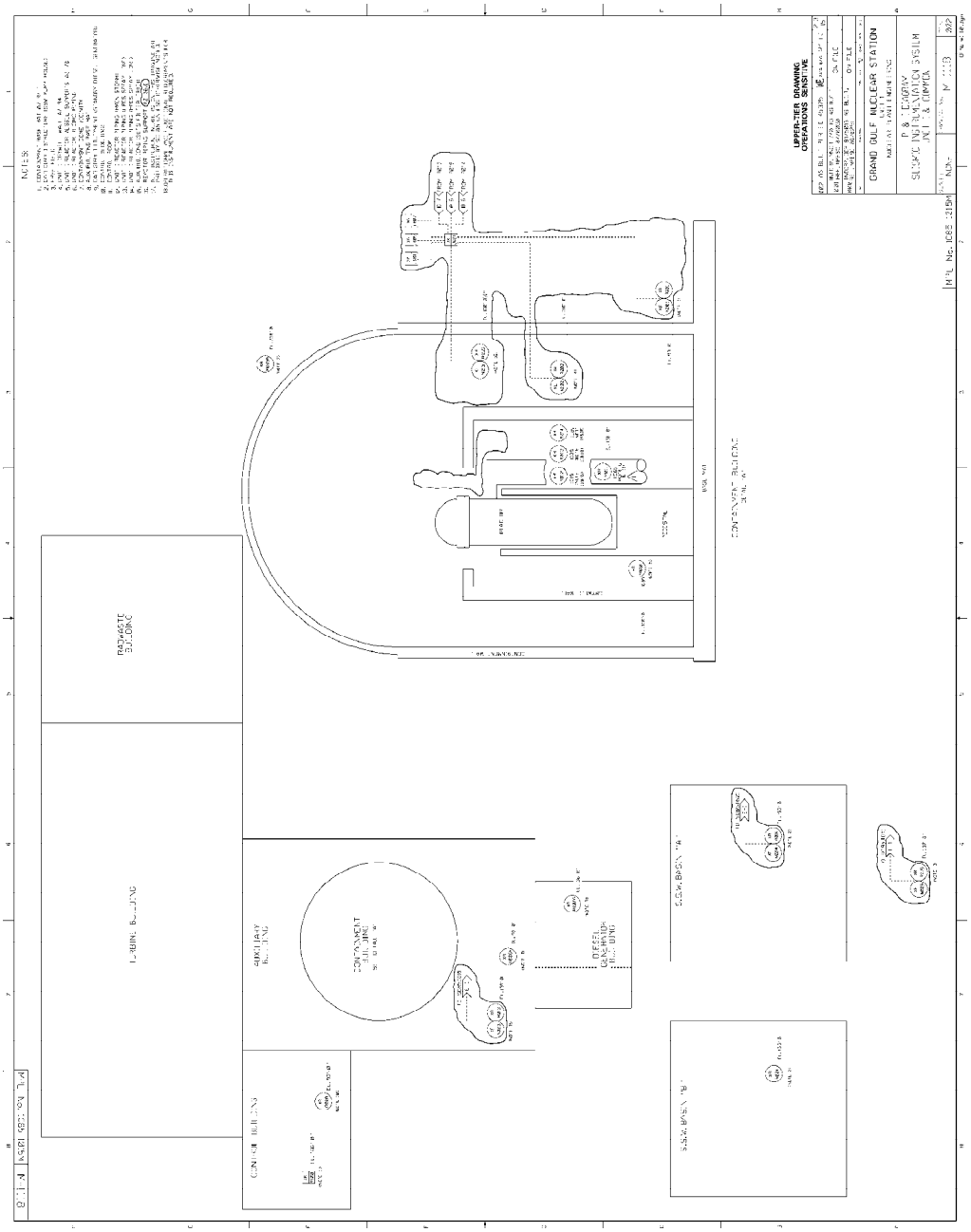
DIESEL GENERATOR BUILDING
RESPONSE LOADS DUE TO SSE, VERTICAL,
BASED ON REG. GUIDE 1.60 DESIGN SPECTRA
AND REG. GUIDE 1.61 DAMPING

FIGURE 3.7-81



3.8 DESIGN OF SAFETY-RELATED STRUCTURES

3.8.1 Concrete Containment

The containment structure is designed to house the primary nuclear system and is part of the containment system whose functional requirement is the control of the release of radio-activity from a primary nuclear system. This section describes the structural design considerations for the containment.

3.8.1.1 Description of the Containment

3.8.1.1.1 Concrete Containment

The containment is shown in Figure 3.8-1.

It consists of three basic parts:

Flat circular foundation mat

Right circular cylinder

Hemispherical dome

The containment cylindrical wall, dome, and foundation mat are constructed of cast-in-place, conventionally reinforced concrete.

The containment wall, dome, and associated internal structures are founded on a circular mat which is 9'-6" thick and has an outside diameter of 134'-0".

The main foundation mat reinforcement consists of a top layer of reinforcement, bottom layer of reinforcement, and vertical, single leg, shear reinforcement. The reinforcement layers for the bottom reinforcement are arranged in a rectangular grid. The reinforcement layers for the top cage are arranged in a rectangular grid at the center of the slab and then radiate outward in a polar pattern in order to mate with the embedded portions of interior structures as well as the embedded wall reinforcement. For further details of mat reinforcement, see Figures 3.8-2 through 3.8-4.

The containment wall is a right, circular cylinder, 3'-6" thick, with an inside radius of 62'-0" and has a height of 144'-9" from the top of the foundation mat to the springline. The wall is thickened to 4'-0" only in the local area of the main steam and feedwater penetrations in order to accommodate added

reinforcement to resist pipe rupture forces from high energy lines. This thickened wall area extends over a rectangular boundary of 34'-11" in plan and 41'-8" in elevation. The 3'-6" thick cylindrical wall extends 5'-9" above the springline intersection of the wall and hemispherical dome to accommodate added reinforcement which resists loadings from polar crane brackets embedded in the shell.

The main reinforcement in the wall consists of inside and outside layers of hoop reinforcement, outside vertical reinforcement, and diagonal reinforcement placed in two directions to form a helix with an angle of approximately 45 degrees from the vertical axis of the shell.

Adequate inside face vertical reinforcement is placed at major discontinuities in the wall including the intersection of the wall and foundation mat, the vicinity of the springline, and the area around major penetrations (i.e., equipment hatch and personnel locks), pipe penetrations, steel embeds anchoring the enclosure building to the exterior face of the wall, floor brackets, and polar crane bracket embeds. In addition, inside face vertical reinforcement is placed at all other locations in the wall to provide a minimum area of 0.0017 times the gross cross-sectional area of the section. This reinforcement maintains structural integrity after a shutdown of the reactor with an ensuing rapid fall of the liner plate temperature. Horizontal, single leg shear ties are placed in the wall where required by design for radial shear. Main reinforcement is bent around major penetrations, including the personnel locks, except in the vicinity of the equipment hatch. All hoop and vertical reinforcement around the equipment hatch is connected to a stiffener ring, fabricated of steel conforming to ASTM SA-516, Grade 70, by the use of B-series Cadweld mechanical splices which are welded to the ring. All diagonal reinforcement is terminated inside the flanges of the ring and anchored by the use of special mechanical anchorages.

Main reinforcement in the wall, which terminates in a location where membrane tension or membrane plus flexural tension is predicted in a direction perpendicular to the bars, is anchored by the use of special mechanical anchorage details. Reinforcement, which is spliced in a region where membrane tension or membrane plus flexural tension is predicted in a direction perpendicular to the bars to be spliced, is spliced with a full positive mechanical Cadweld splice.

Details of the main wall reinforcement are shown in Figures 3.8-5 through 3.8-8.

The containment wall and foundation mat are separated by a 2-inch gap from the auxiliary building except as described in subsection 3.7.2.1.1.3.1 and noted on Figure 3.8-77 to preclude significant interaction of these Category I structures during seismic disturbances. The 2-inch gap between the structures is filled with a compressible joint filler material.

Vertical waterstops are provided on the outside of the wall around the containment perimeter and mate with the auxiliary building walls so as to contain radioactive or nonradioactive water from severed piping in the respective auxiliary building rooms, thereby preventing flooding of adjacent auxiliary building rooms. These waterstops extend from 3 feet below the top of the foundation mat to 25 feet above the top of the mat. The vertical waterstops are connected to a horizontal waterstop which is 3 feet below the top of the mat and which completely encircles the slab and mates with the auxiliary building foundation mat. The use of the horizontal waterstop will ensure, in the unlikely event of a severed pipe in the auxiliary building, that radioactive water will not penetrate into the soil strata.

The containment dome is a hemispherical shell, 2'-6" thick, with an inside radius of 62'-0". The main reinforcement in the dome consists of two outside face families of reinforcement in the form of an inverted U-shape, which are perpendicular to each other and are a continuation of the outside face vertical reinforcement in the shell wall.

Additional main reinforcement includes inside and outside face circumferential (hoop) reinforcement placed in the dome to an angle of approximately 45 degrees above the springline. Diagonal reinforcement, continuous with diagonal wall reinforcement, is placed in two directions to form a helix and extends into the dome to an angle of approximately 40 degrees above the springline. Inside face vertical reinforcement is placed at the springline discontinuity and is continuous with the containment wall inside face vertical reinforcement. In addition, inside face reinforcement is placed at all other locations in the dome to provide a minimum area in each direction of 0.0025 times the gross cross-sectional area of the section. This reinforcement will maintain structural integrity after a shutdown of the reactor for reasons previously stated for the wall.

Single leg shear ties, placed perpendicular to the liner plate, are provided in the dome where required by design for radial shear. Termination and splicing of dome reinforcement in a region where membrane tension or membrane plus flexural tension is predicted in a direction perpendicular to the bars are treated in an identical manner as wall reinforcement, as stated above.

For further details of dome reinforcement, See Figure 3.8-9.

The enclosure building is founded on the dome and wall of the containment and is anchored to steel embedded plates and concrete piers.

3.8.1.1.2 Containment Liner Plate

The internal surface of the containment is completely lined with welded steel plate. The purpose of this liner plate is to form a leak-tight barrier that minimizes radioactive leakage to the environment.

The liner plate consists of plates chosen on the basis that they have sufficient strength and ductility to resist the calculated strains due to the design load combinations and remain leaktight. The plate is stiffened by use of structural T-sections, channels, and angles to carry the design loads and to anchor the liner plate to the concrete shell. In addition, the liner plate has been thickened around major penetrations.

A leak chase channel system has been provided at all seam welds that are inaccessible for inspection during plant operation.

The suppression pool area of the containment liner has been fabricated out of stainless steel which will serve as the fission product barrier. The stainless steel liner plate has also been designed to withstand the negative pressure loads due to quencher discharge bubble pressures. The following loads were added vectorially to evaluate the state of stress in the liner plate: hydrostatic pressure, negative bubble pressure, temperature effects, and containment pressure.

The 1/4-inch carbon steel liner plate for the containment shell above the suppression pool and dome was fabricated from ASTM A 285 material. All thickened carbon steel liner plate for the drywell was fabricated from ASTM A 516 Grade 70 plate. The drywell liner plate, which forms the suppression pool walls, was fabricated using a stainless steel clad carbon steel plate having

a base of ASTM A 537 and clad with ASTM A 240 type 304. The containment base mat liner has been fabricated from ASTM A 240 type 304 stainless steel plates.

There are three structures that penetrate the stainless steel liner plate covering the base slab. They are the drywell wall embed and the tension ties and shear key for the RPV support pedestal. In each case, the 1/4-inch liner plate butts against the penetrating structure and is seal welded to it.

All of the penetrating structures were designed and fabricated from the same material as the stainless steel floor liner plates so that they can perform the same function as the liner plate they replace.

The erection of the liner plate was performed using standard construction procedures.

3.8.1.1.3 Polar Crane Brackets

The polar crane bracket consists of vertical and horizontal plates which are welded together in the shape of a tapered box section. The tapered box section is cantilevered from the containment cylindrical concrete wall. A vertical, embedded bearing plate is welded to the end of and forms a tee section with the top flange plate which penetrates the thickened liner plate and anchors into the concrete wall. The tapered box section is welded to the thickened liner plate along the perimeter of the box section.

Vertical shear is transferred from the web of the bracket box section to the thickened liner plate and ribs which transfer the shear to the tee section and into the concrete. The torsional moment is transferred to the concrete wall by the ribs bearing against the concrete. For details of the polar crane bracket, see Figure 3.8-10.

All steel plate material is in accordance with ASTM A 516 Grade 70 and is ultrasonically tested before it is used in the fabrication of the bracket.

All welding is in accordance with AWS Standard D1.1 Code. Corrosion protection is provided by using Carbozinc 11 applied to a blast-cleaned surface conforming to the requirement of SSPC-SP-10 (near white).

3.8.1.2 Applicable Codes, Standards, and Specifications

The design of the concrete containment, liner plate, containment penetrations, floor brackets, and polar crane brackets conform to the applicable codes, standards, and specifications listed below, except where specifically stated otherwise. Where contradictions occur between codes, the more stringent shall apply.

3.8.1.2.1 Regulations

a. Code of Federal Regulations, Title 10, Energy, Part 50, "Licensing of Production and Utilization Facilities."

b. Code of Federal Regulations (CFR), Title 10 - Energy, Part 100, Reactor Site Criteria, (10 CFR 100), including Appendix A thereto, "Seismic and Geologic Siting Criteria for Nuclear Power Plants."

c. Occupational Safety and Health Administration (OSHA), Department of Labor, Occupational Safety and Health Standards, Title 29: Labor, Parts 15 and 19.

3.8.1.2.2 Codes and Standards

3.8.1.2.2.1 Design Codes

a. American Institute of Steel Construction (AISC), Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings, 1969 edition.

   An exception to this code is as follows:

   Table 1.17.5 - Minimum Size of Fillet Welds

   The requirements of this table shall not apply where shown by stress analysis that a smaller weld size may be used.

b. American Concrete Institute (ACI), Building Code Requirement for Reinforced Concrete, ACI 318-71.

c. American Welding Society (AWS), Code for Welding in Building Construction, AWS D1.1-72. Later editions that are adopted for use are specified in the component's/ structures' specification or standard.

   Exceptions taken to this code are as follows:

Paragraph 1.2 - Base Metal and Paragraph 8.2 - Base Metal

Paragraph 8.2.1 of AWS D1.1 contains a list of ASTM materials which may be used with prequalified welding procedure specifications. This code requires that welding procedure specifications be qualified for each material used for structural applications which are not included in the list. This AWS code requirement is not implemented for the following carbon steels used for structural applications: ASTM A 105; ASTM A 106, Gr. A, B, and C; ASTM A 366; ASTM A 446, Gr. A; ASTM A 516, Gr. 70; ASTM A 526; ASTM A 569; ASTM A 570, Gr. B, C, and D. The above listed materials have chemical compositions and physical properties similar to those carbon steels listed in paragraph 8.2.1 and have welding parameters that would be no different from those welding parameters contained in prequalified welding procedures specifications for the carbon steels listed in paragraph 8.2.1. Stainless steels (including ASTM A 167, Gr. 304; ASTM A 240, TP 304; and ASTM A 276, TP 304) are welded using welding procedure specifications meeting the requirements of the ASME Code, Section IX.

Paragraph 1.3 - Welding Processes

The gas-tungsten arc welding process, as used in the installation of certain miscellaneous structural steel, particularly stainless steel, use welding procedure specifications meeting the requirements of ASME Code, Section IX for both carbon and stainless steels.

Section 2 - Design of Welded Connections

Full penetration groove welds which cannot be back gouged or which cannot use backing strips due to joint configuration or accessibility are welded using Type E6010 electrodes for the root pass, accompanied by welding out the remainder of the weld with Type E7018 electrodes.

Table 2.7 - Minimum Fillet Weld Size for Prequalified Joints

The requirements of this table shall not apply where shown by stress analysis that a smaller weld size may be used.

Paragraph 3.6.4 - Weld Profiles

Prior to June 1, 1987, the requirement is changed to: "For buildings and tubular structures, undercut shall be no more than 1/32-inch (0.8 mm) deep in any direction for material 3/16 inch and greater in thickness."

As of June 1, 1987 the following requirements shall apply:

Paragraph 3.6 - Weld Profiles

Weld profiles acceptance criteria shall be as required by NCIG-01, Revision 2. NCIG-01 is invoked by a modification to AWS D1.1 paragraph 8.15 as discussed below.

Paragraph 4.9.2 - Electrode Storage And Issue

The requirement for E7018 (low hydrogen) electrodes is changed to: "E7018 electrodes that are not used within 14 hours...shall be redried before use."

Section 5 - Qualification and Paragraph 2.6 - Joint Qualification

Welding procedure specifications and welders qualified to ASME Code, Section IX requirements are permitted for structural steel applications in lieu of welding procedure specifications and welders meeting the requirements of AWS D1.1. Joint designs not prequalified to the requirements of AWS D1.1 are permitted using welding procedure specifications and welders qualified to ASME Code, Section IX requirements.

Paragraph 8.15 - Quality of Welds

All welds for Category I structural steel are visually inspected, and the inspection results are documented. Welds for Category I miscellaneous and embedded steel are visually inspected on a surveillance basis, and the surveillance inspection results are documented.

As of June 1, 1987 the following requirements shall apply:

Acceptance criteria for visual inspection of welds shall be per NCIG-01, Revision 2, paragraph 2.0. Inspectors shall be trained in accordance with the elements of NCIG-03, Revision H.

d. American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Division I, Section III, Subsection NE:

1971 Edition with Summer of 1973 Addendum	Fabrication of the Containment Building Liner Plate
1971 Edition with Summer of 1973 Addendum	Fabrication of Penetrations and Other Elements Serving Pressure Vessel Functions (with the exceptions noted below)
1974 Edition	Design and Fabrication of Containment Personnel Air-locks and Equipment Hatches

Exceptions to this code are as follows:

For penetration closure plates only - impact testing requirements and exemptions per subsection NE 1980 edition with winter 1980 and winter 1982 addenda, and ASME subsection NC, 1980 edition and winter 1980 addendum, as applicable.

e. American Society of Civil Engineers (ASCE), Paper 3269, Wind Forces on Structures, governs certain wind design requirements.

f. American National Standards Institute (ANSI) B.31.1, Power Piping.

g. American Concrete Institute, Criteria for Reinforced Concrete Nuclear Power Containment Structures, ACI Committee 349, Title No. 69-2, Journal of the American Concrete Institute, Jan. 1972

Exceptions and interpretations taken to this code are as follows:

1. Chapter 2, Section 2.3.1, Definition of T_a

The definition of T_a is changed to: Added thermal effects (over and above operating thermal effects) which may occur during a design accident and which correspond to the unfactored design pressure, P. This temperature transient is also applied in conjunction with factored pressure loads, i.e., 1.25P and 1.5P.

2. Chapter 2, Section 2.3.3, Capacity Reduction Factor

   This section is replaced by: The capacity reduction factor for reinforcing in the containment shell elements and foundation mat shall be = 0.9 for axial compression as well as axial tension and flexure. The 0 factors for other elements shall be in accordance with ACI 318-71.

3. Chapter 2, Section 2.2.1, Stress and Strain Limitations

   The first two sentences are replaced by: Under all U-loading combinations as defined in subsection 3.8.1.3 of this SAR, the reinforcement comprising the primary tensile reinforcement of a section shall have a maximum predicted stress of not more than $0f_y$.

4. Chapter 2, Section 2.5.2, Reinforcement Anchorage

   In this section, replace the word "tension" with the phrase, "membrane tension or membrane plus flexural tension."

5. Chapter 2, Section 2.5.3, Splices in Reinforcement

   In this section, replace the word "tension" with the phrase "membrane tension or membrane plus flexural tension."

6. Chapter 2, Section 2.4.1, Shells

   The third paragraph is deleted.

h. For ductile concrete expansion anchors use Appendix B, Steel Embedments, to the American Concrete Institute, Code Requirements for Nuclear Safety-Related Concrete Structures, ACI 349-80 with the 1984 Supplement. The following sections of Appendix B relating specifically to

expansion anchors will be applicable: Sections B.7.1 through B.7.5; but the requirements of B.7.2 will be modified by the applicable requirements of NRC IE Bulletin 79-02.

i. Nuclear Construction Issues Group (NCIG), Visual Weld Acceptance Criteria for Structural Welding at Nuclear Plants; NCIG-01, Revision 2 (Reference 20).

j. Nuclear Construction Issues Group (NCIG), Training Manual for Inspectors of Structural Welds at Nuclear Power Plants Using the Acceptance Criteria of NCIG-01; NCIG-03, Revision H (Reference 21).

3.8.1.2.2.2 Construction Codes of Practice

The following codes of practice, with indicated exceptions, established the standards of construction procedure:

a. ACI 301-72, Specifications for Structural Concrete for Buildings

   Exceptions:

   Chapter 1, General

   Section 1.4, ASTM Standards: later issues of the Standards cited may be used, when appropriate.

   Section 1.5, Other Reference Standards and Recommended Practices: later issues than those cited may be used, when appropriate.

   Chapter 4, Formwork

   Section 4.3, Tolerances: The exceptions to the tolerances of Table 4.3.1 are itemized in Table 3.8-25.

   Section 4.5, Removal of Forms: The following requirements shall apply instead of the requirements of subsection 4.5.4:

4.5.4 Forms for columns, walls, sides of beams, slabs and girders, and other parts not supporting the weight of the concrete shall be removed as soon as practicable in order to avoid delay in curing and repairing surface imperfections. Wood forms or insulated steel forms for members three feet or greater in thickness shall be stripped or loosened within 24 hours. Forms shall be left in place for seven days when removal of forms after 24 hours exposes surfaces of mass concrete to an ambient temperature below 50 F.

Chapter 5, Reinforcement

Section 5.4, Fabricating and Placing Tolerances: The following requirements shall apply instead of the requirements specified in subsection 5.4.2 and its subsections:

5.4.2 Center to center bar spacing: plus or minus two inches, except that the minimum clear distance specified in ACI 318-71, Chapter 7, Section 7.4 and the requirements of Section 7.3, ACI 318-71 shall be maintained (see Table 3.8-25 for exceptions to Section 7.3) and the total number of bars shall be maintained within the tolerances given below:

Number of Bars	Tolerance (Percent)
Greater than required	10
Less than required	5

Chapter 8, Placing

Section 8.2, Conveying: The following requirements shall apply in place of the requirements specified in subsection 8.2.2.4:

8.2.2.4 Pumping or pneumatic conveying equipment shall be of a suitable kind with adequate pumping capacity. Pneumatic placement shall be controlled so that segregation is not apparent in discharged concrete.

Concrete shall not be conveyed through pipe made of aluminum or aluminum alloy.

Chapter 14, Massive Concrete

Temperature shall be controlled as specified below for all concrete to be placed when the least dimension is greater than 3 feet.

Section 14.4, Placing: The following requirements shall apply instead of subsections 14.4.1, 14.4.2, and 14.4.3:

14.4.1 Slump: The slump working limit of the structural concrete shall be as found in the project specifications, with an inadvertency margin (tolerance) of plus two inches for individual batches. The inadvertency margin as defined in the project specifications pertains to occasional batches only.

14.4.2 Placing Temperature: The maximum temperature of concrete when deposited shall be 75 F in place of 70 F. Concrete greater than 6 feet in the least dimension shall have a placing temperature as near 65 F as can be obtained.

14.4.3 Depth of Layers: Concrete shall be placed in layers approximately 18 inches thick and not more than 24 inches thick. Vibrator heads shall extend into the previously placed layer.

Section 14.5, Curing and Protection: The following requirements shall apply in place of subsections 14.5.1, 14.5.2 and 14.5.3:

14.5.1 Minimum Curing Period: In place of a minimum curing period of two weeks, the minimum curing period shall be seven days for heavily reinforced massive sections, as specified in ACI 308-71, Section 3.3.3 provided the temperature of the surrounding air is above 40 F and is expected to remain above this temperature for the entire curing period.

14.5.2 When the temperature of the surrounding air is 40 F or below or is expected to drop below 40 F in the next 12 hours, adequate means shall be provided to maintain the temperature of the surrounding air between 50 and 70 F for the remainder of the seven-day curing period.

14.5.3 Moist Curing: As an alternative to keeping the forms and exposed concrete continuously wet for at least the first 48 hours after placing, and also whenever the surrounding air temperature is above 90 F, the provisions of Section 12.2 - Preservation of Moisture, ACI 301-72, shall apply.

b. ACI 306-66, Recommended Practice for Cold Weather Concreting Addition:

Requirements for winter concreting shall conform to ACI Standard 306-66 with the added requirement that curing periods shall be as herein stated under “exceptions” for Section 14.5 (above) of ACI Standard 301-72.

c. ACI 318-71, Building Code Requirements for Reinforced Concrete

Exceptions:

Chapter 4, Concrete Quality

Subsection 4.2.5 and Table 4.2.5: Concrete air content shall be within 2 to 6 percent by volume for all sizes of coarse aggregates. However, when air content falls below 2 percent, the frequency of correlation testing shall be increased to each 100 cubic yards until test results fall within 2 to 6 percent by volume. If air content is not brought within limits, as determined by the Contractor, the concrete placement shall cease until the problem is corrected.

Subsection 4.3.1: Where applicable, the following requirement shall be used instead of the last sentence: Each strength test result shall be the average of two cylinders from the same sample tested at 28 or 91 days.

Chapter 5, Mixing and Placing Concrete

Section 5.4, Depositing: The following requirement shall apply in place of subsection 5.4.4:

5.4.4 On the bottom of formed beams and slabs where conditions make consolidation difficult or where reinforcement is congested, batches of grout containing the same proportions of cement and sand as those used in the concrete shall first be deposited in the forms to a depth of approximately 1/4 inch.

Chapter 6, Formwork, Embedded Pipes, and Construction Joints

Section 6.3, Conduits and Pipes Embedded in Concrete: The following requirements shall apply in place of subsections 6.3.2.2, 6.3.2.3, and 6.3.2.4:

6.3.2.2 and 6.3.2.3 All pipes carrying liquid, gas, or vapor at temperatures exceeding 150 F or pressures exceeding 200 psia, or both, shall not be embedded in structural concrete. If an embedment is necessary, the pipes shall be provided with suitably designed sleeves.

6.3.2.4 All embedded piping and fittings shall be tested to the specific provisions of ASME Code, Section III, USAS B31.7, ANSI B31.1, and/or local codes, as may be applicable, that govern the system of which the embedded piping is a part. Embedded gravity drainage piping, which is not covered by any of the above codes, shall be hydrostatically tested to 15 psig in accordance with ANSI B31.1. Floor and wall penetrations are not considered embedded piping and therefore are not hydrostatically tested. Where no other code governs a particular pipe embedded in structural concrete, the requirements of ACI 318-71, subsection 6.3.2.4, shall govern.

Subsection 6.4.1: The following requirements shall apply instead of subsection 6.4.1: Joints not indicated on the plans shall be so made and located as not to impair significantly the strength of the structure. Both horizontal and vertical construction joints shall be thoroughly cleaned of all coatings, debris, and other foreign materials. All concrete surfaces to receive new concrete shall be free of surface water, but shall be thoroughly dampened to prevent absorption of water from the new concrete. All horizontal surfaces shall be covered, immediately before the concrete is placed, with a nominal 1/4 inch of grout thoroughly broomed into the surface. For congested areas where brooming is impossible, the grout shall be forced ahead of the concrete. Grout used in construction joints shall have the same sand-cement ratio as the concrete with which it is to be used.

Chapter 7, Details of Reinforcement, Section 7.3, Placing Reinforcement: The following requirements that shall apply instead of the tolerances of paragraph 7.3.2 are tabulated in Table 3.8-25.

Chapter 11, Shear and Torsion

Section 11.16, This section is not applicable for tangential shear design.

Section 11.2.2, This section is not applicable for radial shear design.

Chapter 19, Shells and Folded Plate Members

Subsections 19.2.2, 19.3.2, 19.5.1, 19.5.2, 19.5.4, and the last sentence of 19.5.3 are not applicable for containment design.

d. ACI 347-68, Recommended Practice for Concrete Formwork

e. ASTM C 94-72, Ready-Mixed Concrete

f. ACI 305-72, Recommended Practice for Hot Weather Concreting

Exceptions:

Chapter 2, subsection 2.2.1: For massive concrete, the placing temperature shall be as near 65 F as jobsite conditions permit, but the maximum placing temperature shall be 75 F.

Chapter 4: Curing shall be in accordance with the provisions of ACI 301-72, Chapters 12 and 14, and Section 14.5, with the exceptions to Section 14.5 listed above.

g. ACI 211.1-70, Recommended Practice for Selecting Proportions for Concrete

h. ACI 304-73, Recommended Practice for Measuring, Mixing, Transporting, and Placing Concrete

i. ACI 315-65, Manual of Standard Practice for Detailing Reinforced Concrete Structures

j. ASME Boiler and Pressure Vessel Code, Section VIII

Subsection B, Requirements Pertaining to Methods of Fabrication of Pressure Vessels

Part UW, Requirements for Pressure Vessels Fabricated by Welding.

k. AWS D1.1, Code for Welding in Building Construction

In every instance, the construction procedure will equal or exceed the recommendations set forth in the foregoing publications except as noted above. The extent to which each detailed process will exceed standard requirements cannot be abstracted without incorporating all applicable job specifications and anticipating hypothetical construction problems and conditions.

3.8.1.2.3 General Design Criteria, Regulatory Guides, Industry Standards, and Topical Reports

a. 10 CFR 50, Appendix A, "General Design Criteria for Nuclear Power Plants." Conformance is discussed in Section 3.1.

   Criteria Nos. 2, 16, 51, and 53.

b. NRC Regulatory Guides (exceptions are discussed in Appendix 3A).

   Regulatory Guide 1.10, Mechanical (Cadweld) Splices in Reinforcing Bars of Seismic Category I Concrete Structures

   Regulatory Guide 1.15, Testing Reinforcing Bars for Seismic Category I Concrete Structures

   Regulatory Guide 1.18, Structural Acceptance Test for Concrete Primary Reactor Containments

   Regulatory Guide 1.19, Nondestructive Examination of Primary Containment Liner Welds

   Regulatory Guide 1.60, Design Response Spectra for Seismic Design of Nuclear Power Plants

   Regulatory Guide 1.61, Damping Values for Seismic Design of Nuclear Power Plants

   Regulatory Guide 1.76, Design Basis Tornado for Nuclear Power Plants

   Regulatory Guide 1.102, Flood Protection for Nuclear Power Plants

   Regulatory Guide 1.115, Protection Against Low-Trajectory Turbine Missiles

Regulatory Guide 1.122, Development of Floor Design Response Spectra for Seismic Design of Floor Supported Equipment or Components

c. Industry Standards

Nationally recognized industry standards, such as those published by the American Society for Testing and Materials (ASTM) and the American National Standards Institute (ANSI), are used to describe the material properties, testing procedures, fabrication methods, and construction methods.

d. Bechtel Power Corporation Topical Reports

BN-TOP-1, Rev. 1, Testing Criteria for Integrated Leak Rate Testing of Primary Containment Structures for Nuclear Power Plants

BN-TOP-2, Rev. 2, Design for Pipe Rupture Effects

BC-TOP-1, Rev. 1, Containment Building Liner Plate Design

BC-TOP-3A, Rev. 3, Tornado and Extreme Wind Design Criteria for Nuclear Power Plants

BC-TOP-4, Rev. 1, Seismic Analysis of Structures and Equipment for Nuclear Power Plants, including Addendum 1 (Ref. 19)

BC-TOP-5A, Rev. 3, Prestressed Concrete Nuclear Reactor Containment Structures (for containment liner plate only)

3.8.1.2.4 Specifications

[HISTORICAL INFORMATION] [Specifications are prepared to cover the areas related to the design and construction of the containment. These specifications are prepared by Bechtel Power Corporation specifically for this containment. The specifications emphasize important points of the industry standards for these structures and reduce options such as would otherwise be permitted by industry standards. Unless specifically noted otherwise, these specifications do not deviate from the applicable industry standards and, therefore, are not included in the SAR. These specifications cover the following areas:

a. Furnishing and delivery of concrete

b. Purchasing, forming, placing, and curing of concrete

c. Furnishing, detailing, fabricating, delivery, and placing of reinforcing steel

d. Furnishing, delivery, and installation of exothermic splicing

e. Furnishing, delivery, and erection of structural steel

f. Furnishing, delivery, and erection of steel liner plate

g. Furnishing, delivery, and installation of containment airlocks and penetrations]

3.8.1.3 Loads and Loading Combinations

The containment is analyzed and designed for all credible conditions of loading including normal loads, preoperational testing loads, loads during severe environmental conditions, loads during extreme environmental conditions, loads during abnormal plant conditions, loads during post-abnormal plant conditions, and site-related loads as delineated in the following text.

The containment was initially designed in accordance with the loads defined in GE Topical Report NEDO 11314-08 (GESSAR Appendix 3B). Additional loads initially defined in GE document 22A4365, Interim Containment Loads Report (ICLR), Rev. 2, and later defined by GE document 22A7000, Rev. 2 (GESSAR II, Appendix 3B; Grand Gulf FSAR, Appendix 6D), have been considered for the final design verification of the containment. The loads within the loading combinations are combined using the absolute sum technique. (Those loads which are defined as reversible in algebraic sign are combined in such a way as to produce the maximum resultant stresses in the structure. All other loads are combined in accordance with their direction of application to the structure.)

3.8.1.3.1 Normal Loads

a. D - Dead load of the structure and equipment plus any other permanent loads, including vertical and lateral pressures of liquids.

b. L - Live loads, including any movable equipment loads and other loads which vary in intensity and occurrence, such as forces exerted by the lateral pressure of soil.

c. T_o - Thermal effects during normal operating, startup, or shutdown conditions, including liner plate expansion, equipment and pipe reactions, and thermal gradients based on the most critical transient or steady state thermal gradient. Operating thermal gradients for the containment dome, wall, and foundation mat are shown in Figures 3.8-11 through 3.8-15.

d. Z - Pressure difference between the interior and exterior of the containment, considering both interior pressure changes because of heating or cooling and exterior atmospheric pressure variations. An external pressure loading with a differential of 3 psig from outside to inside is considered.

e. Construction Loads: Construction loads are loads which are applied to the containment from start to completion of construction. The definitions for D, L, and T_o given above are applicable, but are based on actual construction methods and/or conditions.

f. Safety Relief Valve Loads - The containment experiences oscillatory dynamic pressure loadings resulting from discharge of safety relief valves into the suppression pool. The development of these loads is in accordance with the methods described in Appendices 6A and 6D of this FSAR. The number and combinations of valves that will open during a reactor vessel pressure transient are as follows:

   1. C_1 - Design pressure load on the suppression pool boundary resulting from discharge of one safety relief valve into the suppression pool (positive or negative). Positive pressures for the first actuation and the subsequent actuation of this loading to be sustained by the containment are shown in Figures 3.8-16a through 3.8-25b.

   2. C_2 - Design pressure load on the suppression pool boundary resulting from discharge of two adjacent safety relief valves into the suppression pool

(positive or negative). Positive pressures for this loading to be sustained by the containment are shown in Figures 3.8-26a through 3.8-30a.

3. C_{11} - Design pressure load on the suppression pool boundary resulting from discharge of eleven safety relief valves into the suppression pool (positive or negative). Positive pressures for this loading to be sustained by the containment are shown in Figures 3.8-31a through 3.8-35b.

4. C_A - Design pressure load on the suppression pool boundary resulting from automatic depressurization system (ADS) discharge of eight safety relief valves into the suppression pool (positive or negative). Positive pressures for this loading to be sustained by the containment are shown in Figures 3.8-36a through 3.8-40b.

5. C' - Design pressure load on the suppression pool boundary resulting from discharge of all twenty safety relief valves into the suppression pool (positive or negative). Positive pressures for this loading to be sustained by the containment are shown in Figures 3.8-41a through 3.8-45b.

g. I - Impact load due to crane operation.

3.8.1.3.2 Preoperational Testing Loads

a. [HISTORICAL INFORMATION] [Test loads are loads which are applied during the structural integrity test. The containment test pressure is 1.15 x 15 psig = 17.25 psig.

b. T_t - Thermal effects during the structural integrity test.]

3.8.1.3.3 Severe Environmental Loads

a. E - Operating Basis Earthquake (OBE) as defined in Section 3.7 of this safety analysis report.

b. W - Loads generated by the design wind specified for the plant site as defined in Section 3.3 of this safety analysis report.

3.8.1.3.4 Extreme Environmental Loads

a. E' - Safe Shutdown Earthquake (SSE) as defined in Section 3.7 of this safety analysis report.

b. W' - Loads generated by the tornado specified for the plant site as defined in Section 3.3 of this safety analysis report.

3.8.1.3.5 Abnormal Plant Loads

a. R - Local force or pressure on containment structure or penetration caused by rupture of any one pipe as defined in Section 3.6 of this safety analysis report.

   1. Local compartmental pressures on the containment wall from ruptured pipes in compartments outside the wall within the auxiliary building are as defined in Appendix 3E of this SAR.

   2. Local compartmental pressures inside the containment include those from the main steam pipe tunnel and are as defined in subsection 6.2.1.2 of this SAR.

b. Design Basis Accident (DBA) - The containment is designed to sustain the following loadings as a result of a loss-of-coolant accident (LOCA).

   1. P_{CD} - Containment design loss-of-coolant accident pressure load during a design basis accident as presented in subsection 6.2.1 of this safety analysis report. The containment design pressure is 15 psig.

   2. T_a - Added thermal effects (over and above operating thermal effects) which may occur during a design accident and which correspond to the unfactored design pressure P_{CD}. Accident thermal gradients for the containment dome, wall, and foundation mat are shown in Figures 3.8-11 through 3.8-15.

   3. P_{DD} - Peak drywell design pressure load during a design basis accident. The drywell design pressure is 30 psig.

4. P_{ND} - Peak drywell design negative pressure load during a design basis accident. Pressure diagrams for this load are shown in Figure 3.8-46.

5. P_{CDC} - Containment and drywell design pressure, during a design basis accident (DBA), concurrent with chugging (this loading is postulated in the interval from t = 30 sec. to t = 100 sec. during the DBA). (See Figure 3.8-48a.)

6. Design basis accident (DBA) suppression pool loads - Immediately following a design basis accident, the containment wall, foundation mat, drywell wall, weir wall, and structures located above the suppression pool will experience various pressure and dynamic loadings as a result of the vent DBA clearing process into the suppression pool. The development of these loads is in accordance with the methods described in Appendices 6A and 6D. The following are the loadings resulting from the DBA phenomena.

   R_B - Pool swell bubble pressure on containment (axisymmetrical). The pressure diagram for this load is shown in Figure 3.8-47 and Appendix 6D.

   R_{BA} - Nonaxisymmetrical pool swell bubble pressure on containment. This load has the same pressure ordinate as R_B above and is applied to the containment from 0 to 180 degrees in plan.

   R_D - Containment wetwell differential pressure axisymmetrical due to flow restriction of the hydraulic control unit (HCU) floor to pool swell concurrent with a differential drywell pressure. The pressure diagrams for this load are shown in Figure 3.8-48 and Appendix 6D.

   R_P - Impact and drag loads on structures located above the pool suppression due to upward pool swell. Loads on these structures are shown in Appendix 6D.

R_F - Fallback loads of cascading water onto structures above the suppression pool immediately following pool swell. Loads on these structures are shown in Appendix 6D.

R_C - An oscillatory dynamic loading (chugging) in the top vent and weir annulus or on the suppression pool boundary due to steam condensation inside the top vent or at the top vent exit during periods of low steam mass flow in the top vent only following a LOCA. Pressure distributions for this loading are shown in Appendix 6D.

R_{CO} - An oscillatory dynamic loading (condensation oscillation), on the suppression pool boundary due to steam condensation at the vent exits during the period of high steam mass flow through the vents following a LOCA. Pressure distributions for this loading are shown in Appendix 6D.

R_W - Inward load through the vents applied to the weir wall, due to negative drywell pressure differential. Pressure distributions for this loading are shown in Appendix 6D.

c. Small Break Accident (SBA) - The small break accident is deemed to be an accident during which a pipe rupture is not large enough to automatically depressurize the reactor, as presented in subsection 6.2.1 of this SAR. The containment is designed to sustain the following loads during an SBA.

1. P_{DS} - Containment and drywell design pressure load during the SBA. Pressure diagrams for this load are shown in Figure 3.8-49.

2. P_{DN} - Drywell design negative pressure load associated with chugging (water oscillation back and forth in the top row of drywell vents) during an SBA. Pressure diagrams for this load are shown in Figure 3.8-50.

3. T_a - Added thermal effects (over and above operating thermal effects) which may occur during a small break accident and which correspond to the unfactored design pressure, P_{DS}. Thermal gradients for the foundation mat are shown in Figure 3.8-51.

4. R_C - See definition in subsection 3.8.1.3.5.b above.

d. Intermediate Break Accident (IBA) - The intermediate break accident is deemed to be an accident during which a pipe rupture, albeit smaller than a DBA, is of sufficient size to automatically depressurize the primary system due to loss of fluid and/or automatic initiation of the ECCS systems, as presented in subsection 6.2.1 of this SAR. The containment is designed to sustain the following loadings during an IBA.

1. P_{CI} - Containment and drywell design pressure load during an IBA. Pressure diagrams for this load are shown in Figure 3.8-52.

2. T_a - Added thermal effects (over and above operating thermal effects) which correspond to the unfactored design pressure, P_{CI}.

3. R_C - See definition in subsection 3.8.1.3.5.b above.

4. R_{CO} - See definition in subsection 3.8.1.3.5.b above.

3.8.1.3.6 Post-Abnormal Plant Loads

For fuel recovery subsequent to a design basis accident, the containment is conservatively designed for flooding to a datum level of 55'-6" above the top of the active fuel in the reactor core.

B - Hydrostatic load due to post-LOCA flooding of the containment to a level 115'-10" above the top of the foundation mat.

3.8.1.3.7 Site Related Loads

A - Hydrostatic load due to probable maximum flood as defined in Section 3.4 of this SAR. Other loads associated with activities near the site are discussed in Section 2.2 and loads from postulated missiles are discussed in Section 3.5, both of this SAR.

3.8.1.3.8 Loading Combinations for the Containment Structure and Liner Plate

Notations used in this section are as follows:

S - required capacity of the section such that the stress limitations of Section 8.10, ACI 318-71, are satisfied, with the exception of the permissible tensile stress for reinforcement, which shall be a maximum of 0.5 Fy and not more than 30,000 psi.

U - required ultimate load capacity as defined in ACI 318-71.

3.8.1.3.8.1 Normal Operating Conditions

$S = 1.0D + 1.0L + 1.0T_o + 1.0Z$

$U = 1.4D + 1.7L + 1.0T_o + 1.0Z$

$S = 1.0D + 1.0L + 1.0T_o + 1.0C'(\pm)$

$S = 1.0D + 1.0L + 1.0T_o + 1.0C_{11}\ (\pm)$

$S = 1.0D + 1.0L + 1.0T_o + 1.0C_A\ (\pm)$

$S = 1.0D + 1.0L + 1.0T_o + 1.0C_2\ (\pm)$

$S = 1.0D + 1.0L + 1.0T_o + 1.0C_1\ (\pm)$

$U = 1.4D + 1.7L + 1.0T_o + 1.7C'\ (\pm)$

$U = 1.4D + 1.7L + 1.0T_o + 1.7C_{11}\ (\pm)$

$U = 1.4D + 1.7L + 1.0T_o + 1.7C_A\ (\pm)$

$U = 1.4D + 1.7L + 1.0T_o + 1.7C_2\ (\pm)$

$U = 1.4D + 1.7L + 1.0T_o + 1.7C_1$ (±)

3.8.1.3.8.2 Preoperational Testing Conditions

$S = 1.0D + 1.0L + 1.15P_{CD} + T_t$

3.8.1.3.8.3 Normal Operating and Severe Environmental Conditions

$U = 1.0D + 1.0L + 1.0T_o + 1.0C'$ (±) $+ 1.25E$

$U = 1.0D + 1.0L + 1.0T_o + 1.0C_{11}$ (±) $+ 1.25E$

$U = 1.0D + 1.0L + 1.0T_o + 1.0C_A$ (±) $+ 1.25E$

$U = 1.0D + 1.0L + 1.0T_o + 1.0C_2$ (±) $+ 1.25E$

$U = 1.0D + 1.0L + 1.0T_o + 1.0C_1$ (±) $+ 1.25E$

$U = 1.0D + 1.25L + 1.25$(E or W) $+ 1.0T_o + 1.0Z$

3.8.1.3.8.4 Normal Operating and Extreme Environmental Conditions

$U = 1.0D + 1.0L + 1.0T_o + 1.0Z + 1.0W'$

$U = 1.0D + 1.0L + 1.0T_o + 1.0C'$ (±) $+ 1.0E'$

$U = 1.0D + 1.0L + 1.0T_o + 1.0C_{11}$ (±) $+ 1.0E'$

$U = 1.0D + 1.0L + 1.0T_o + 1.0C_A$ (±) $+ 1.0E'$

$U = 1.0D + 1.0L + 1.0T_o + 1.0C_2$ (±) $+ 1.0E'$

$U = 1.0D + 1.0L + 1.0T_o + 1.0C_1$ (±) $+ 1.0E'$

3.8.1.3.8.5 Normal Operating and Abnormal Plant Conditions

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.50P_{CD} + 1.0R$

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.50$(R$_B$ or R$_{BA}$)$+ 1.50P_{DD} + 1.25C_1$ (±) $+ 1.0R$

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.50P_{ND} + 1.0R$

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.50(R_D + R_P + R_F + R_{CO}) + 1.25C_1 (\pm) + 1.0R$

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.50\ R_C + 1.50P_{CDC} + 1.0R$

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.25C_A (\pm) + 1.50R_C + 1.50P_{CI}$

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.25C_1 (\pm) + 1.50R_{CO} + 1.50P_{CI}$

$U = 1.OD + 1.0L + 1.0(T_o + T_a) + 1.25C_2 (\pm) + 1.50R_{CO} + 1.50P_{CI}$

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.25C_1 (\pm) + 1.50R_C + 1.50(P_{DS} \text{ or } P_{DN})$

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.25C_2 (\pm) + 1.50R_C + 1.50(P_{DS} \text{ or } P_{DN})$

$U = 1.OD + 1.0L + 1.0(T_o + T_a) + 1.25C_A (\pm) + 1.50R_C + 1.50(P_{DS} \text{ or } P_{DN})$

3.8.1.3.8.6 Normal Operating, Abnormal Plant, and Severe Environmental Conditions

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.25P_{CD} + 1.25\ (E \text{ or } W) + 1.0R$

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.25(R_B \text{ or } R_{BA}) + 1.25P_{DD} + 1.0C_1(\pm) + 1.25E + 1.0R$

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.25P_{ND} + 1.25E + 1.0R$

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.25(R_D + R_P + R_F + R_{CO}) + 1.25E + 1.0C_1 (\pm)$

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.25R_C + 1.25P_{CDC} + 1.25\ (E \text{ or } W) + 1.0R$

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.0C_A (\pm) + 1.25R_C + 1.25P_{CI} + 1.25E$

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.0C_2 (\pm) + 1.25R_{CO} + 1.25P_{CI} + 1.25E$

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.0C_1\ (\pm) + 1.25R_{CO} + 1.25P_{CI} + 1.25E$

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.0C_1\ (\pm) + 1.25R_C + 1.25(P_{DS}$ or $P_{DN}) + 1.25E$

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.0C_2\ (\pm)\ 1.25R_C + 1.25(P_{DS}$ or $P_{DN}) + 1.25E$

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.0C_A\ (\pm) + 1.25R_C + 1.25(P_{DS}$ or $P_{DN}) + 1.25E$

3.8.1.3.8.7 Normal Operating, Abnormal Plant, and Extreme Environmental Conditions

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.0P_{CD} + 1.0E' + 1.0R$

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.0(R_B$ or $R_{BA}) + 1.0P_{DD} + 1.0C_1\ (\pm) + 1.0E' + 1.0R$

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.0P_{ND} + 1.0E' + 1.0R$

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.0(R_D + R_P + R_F + R_{CO}) + 1.0C_1\ (\pm) + 1.0E'$

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.0R_C + 1.0P_{CDC} + 1.0E' + 1.0R$

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.0C_A\ (\pm) + 1.0R_C + 1.0P_{CI} + 1.0E'$

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.0C_1\ (\pm) + 1.0R_{CO} + 1.0P_{CI} + 1.0E'$

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.0C_2\ (\pm) + 1.0R_{CO} + 1.0P_{CI} + 1.0E'$

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.0C_1\ (\pm) + 1.0R_C + 1.0(P_{DS}$ or $P_{DN}) + 1.0E'$

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.0C_2\ (\pm) + 1.0R_C + 1.0(P_{DS}$ or $P_{DN}) + 1.0E'$

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.0C_A\ (\pm) + 1.0R_C + 1.0(P_{DS}$ or $P_{DN}) + 1.0E'$

3.8.1.3.8.8 Normal Operating, Post-Abnormal Plant, and Severe Environmental Conditions

$U = 1.0D + 1.0L + 1.0B + 1.0(E \text{ or } W)$

3.8.1.3.8.9 Normal Operating and Site Related Conditions

$U = 1.0D + 1.0L + 1.0T_o + 1.0A + 1.0Z$

3.8.1.3.9 Loading Combinations for Polar Crane Brackets

Additional notation:

F_s = Allowable elastic stress as defined in AISC "Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings," 1969 edition. In no case shall the allowable stress exceed 90 percent of the yield stress in bending, 85 percent of the yield stress in axial tension, and 50 percent of the yield stress in shear.

3.8.1.3.9.1 Normal Operating Conditions

$F_s = D + L + I$

3.8.1.3.9.2 Normal Operating and Severe Environmental Conditions

$1.25F_s = D + L + I \pm E$

3.8.1.3.9.3 Normal Operating and Extreme Environmental Conditions

$1.25F_s = D + L + I \pm E'$

3.8.1.4 Design and Analysis Procedures

This subsection describes the analytical and design procedures used in designing the containment.

3.8.1.4.1 Containment Cylindrical Wall, Dome, and Foundation Mat

3.8.1.4.1.1 Analytical Methods

The analytical methods used for nonaxisymmetrical and axisymmetrical loadings imposed on the containment shell and foundation mat are as follows.

3.8.1.4.1.1.1 Nonaxisymmetrical Loads

Nonaxisymmetrical loads imposed on the containment structure include the following, and are as defined in subsection 3.8.1.3 of this safety analysis report.

a. Tornado wind

b. Design wind

c. Safe shutdown earthquake

d. Operating basis earthquake

e. Local pipe rupture forces, including local compartmental pressures on the shell from ruptured pipes in compartments inside or outside the containment

f. Pool swell bubble pressure

g. Safety relief valve actuation in suppression pool

h. Loadings from embedded steel brackets in the wall and dome, including floor brackets, polar crane brackets, and enclosure building brackets/concrete piers

The containment structure is analyzed for the safe shutdown earthquake (SSE), operating basis earthquake (OBE), nonaxisymmetrical design basis accident (DBA), pool swell bubble pressure, and nonaxisymmetrical pressures from safety relief valve (SRV) discharge by the use of ASHSD. ASHSD is a finite element computer program which performs a linear elastic analysis of an axisymmetrical structure for axisymmetric or nonaxisymmetric, static or dynamic loadings. Response from a static analysis includes deflections and stresses. Response from a dynamic analysis includes deflections, accelerations, and stresses.

For the OBE and SSE ASHSD analysis, computer input data consists of inertial force (SRSS effective forces) output data from the computer programs described in Section 3.7 of this safety analysis report.

An equivalent static analysis was performed for the nonaxisymmetric pool swell bubble pressure loading using the peak pressures for this loading.

The containment structure is idealized in ASHSD as an assemblage of constant strain solid elements of triangular or quadrilateral shape and isoparametric thin shell elements, which are expressed as a series of conical frustrums.

The finite element model used for these loadings, showing the assumed boundary conditions, demarcation of soil stratum, distribution of shell and solid elements, and modeling of associated internal structures, is shown in Figures 3.8-54 and 3.8-55. This model accounts for the incompatibility at the junction of shell and solid elements. This incompatibility with regard to degrees of freedom associated with the types of elements involved (four degrees of freedom per node for shell elements and three for solid elements) is resolved by making a transition from the shell to the solid. The technique for the transition involves extending a shell element into a layer of solid elements and adjusting the thickness of each element type to maintain the proper stiffness through the transition. The shell element thickness varies from the actual element thickness at the node point of connectivity to the solid elements to zero at the extended shell node within the solids. The solids are modified similarly. This method was used on the containment, drywell, shield, and weir walls as well as the RPV pedestal.

Input data for nonaxisymmetrical pool swell bubble pressures is described in subsection 3.8.1.3.5.

The containment cylinder wall is analyzed for local pipe rupture forces on penetration wall sleeves, local compartmental runway girders, and enclosure building support brackets by the use of References 1 and 2, as listed at the end of this section. The containment dome is manually analyzed for loadings from embedded enclosure building support brackets/concrete piers by the use of Reference 3.

The containment wall and dome are shielded from the design wind by the auxiliary building, which completely encircles the structure, and by the enclosure building. Since all enclosure building siding will remain intact during severe environmental conditions, and since the auxiliary building is a separate and independent structure, forces from the design wind will be transmitted directly to the containment wall and dome through the support brackets/piers embedded in the shell. Effects of these loads are considered by use of the analytical references listed above.

The containment wall above the auxiliary building roof and the dome will be substantially exposed to a tornado wind, since the enclosure building siding will be assumed to be breached on the windward side. All siding will be stripped on the leeward and the other two sides, and all roof decking will be stripped.

The finite element model used in ASHSD for determining the structural response due to the nonaxisymmetrical safety relief valve loads showing the assumed boundary conditions, demarcation of soil stratum, distribution of shell and solid elements, and the modeling of associated internals, is shown in Figures 3.8-54 and 3.8-55.

The actuation of safety relief valves results in dynamic loads on the suppression pool boundaries. These dynamic loads are formulated by applying a non-dimensional time function to the attenuated pressures in the suppression pool which result from single and multiple valve discharge. These attenuated pressures are calculated based upon the methodology presented in Appendices 6A and 6D. The non-dimensional time function is the oscillation of the bubble pressure within the suppression pool, which is shown in Appendix 6D, Figure 3B.A.5.11. The magnitude of the pressure at any point within the pool decreases with time, with the duration of the load being less than 1 second.

Once this pressure time-history has been formulated, it is represented in terms of a Fourier series and then used as input for ASHSD.

The response of the containment and internals, considering proportional damping in the system, is determined throughout the time domain by the direct integration scheme in ASHSD. This analysis results in displacement, stress, and acceleration time-histories throughout the structure.

Harmonic acceleration time-histories calculated in ASHSD are summed for a full Fourier series and input into Computer Code CE 802 (see Appendix 3D), which calculates in-structure response spectra at critical areas of interest. Soil-structure interaction effects are accounted for in the ASHSD direct integration analysis. Structures adjacent to the containment building will not experience significant effects from suppression pool loads in the containment.

The containment shell, drywell shell, and containment base mat are extremely stiff steel-lined reinforced concrete structures which form the suppression pool boundary. Thus, effects of fluid-structure interaction upon the total containment building response due to dynamic suppression pool boundary loads will be small. Suppression pool boundary loads, as defined in Appendices 6A and 6D, are applied to the ASHSD mathematical model shown in Figures 3.8-54 and 3.8-55 as rigid wall loads. The mass of the suppression pool water has been lumped at those node points of Figure 3.8-54 that form the suppression pool boundary. The effect of negative safety relief valve pool boundary loads upon the suppression pool liner is discussed further in subsections 3.8.1.1.2 and 3.8.1.4.2.

The tornado wind forces will be transmitted to the containment by two simultaneous load mechanisms. The first mechanism will transfer drag forces on the enclosure building frame to the shell through the embedded support brackets/piers. The effects of these loads are analyzed in the same manner as stated previously for the design wind. The second mechanism will consist of a general wind loading on the shell. The structural effects of this load are analyzed by the use of classical shell and beam theory. The two analyses are then superimposed to effect the total structural response of the containment from the tornado wind loading.

3.8.1.4.1.1.2 Axisymmetrical Loads

Axisymmetric loads imposed on the containment structure include the following, and are as defined in subsection 3.8.1.3 of this safety analysis report:

a. Structure dead load

b. Surcharge loads from adjacent structures

c. Hydrostatic load from probable maximum flood

d. Hydrostatic load from normal site water table

e. Hydrostatic load from post-LOCA flooding of the containment

f. Local dead and live loads from embedded brackets, treated as axisymmetrical loads for overall structural response

g. Dead and live loads from internal structures imposed on the foundation mat

h. Normal operating thermal gradients

i. Abnormal plant thermal gradients (including those from DBA, IBA, and SBA)

j. Preoperational test pressure

k. Abnormal plant pressure loads (including those from DBA, IBA, and SBA)

l. Normal external pressure load

The containment structure is analyzed for all of the above loadings by the use of FINEL, a finite element computer program, which performs a linear elastic analysis for stresses, strains, and deflections of an axisymmetrical structure for axisymmetrical static loadings.

For these loads, an isotropic, elastic, linear analysis was performed for static axisymmetrical loadings. The containment structure is idealized in FINEL as an assemblage of constant strain, two dimensional finite elements of triangular or quadrilateral shapes. The liner plate, which is the primary leakage barrier on the interior surface of the containment shell and foundation mat, is modeled integrally with the concrete containment in the finite element model, so as to consider the interaction between the steel plate and the concrete structure. A portion of the drywell wall, beyond its characteristic length, is integrally modeled with the containment foundation mat so as to accomplish the structural interaction between these two elements and to ascertain that the foundation will deflect realistically to the imposed boundary condition of fixity between the drywell wall and the foundation mat. The idealized finite element model

used for these loadings showing the assumed boundary conditions, demarcation of soil strata, and distribution of elements shown in Figures 3.8-56 and 3.8-57.

The design basis accident (DBA), a loss-of-coolant accident (LOCA), results in dynamic loads on the suppression pool boundaries. These dynamic loads are formulated by applying a time function to the attenuated pressures in the suppression pool. The attenuated pressures are calculated based upon the methodology presented in Appendices 6A and 6D. Once the pressure time-histories are formulated, they are represented in terms of a Fourier series and then used as input for ASHSD.

The response of the containment and internals is determined throughout the time domain by the direct integration scheme in ASHSD using the same finite element model as was used for the dynamic nonaxisymmetric safety relief valve loadings described in subsection 3.8.1.4.1.1.1.

3.8.1.4.1.1.3 Major Penetrations

The major penetrations which intersect the containment wall include the equipment hatch and two personnel locks as described in subsection 3.8.1.1. The state of stress around these openings is determined by the use of analytical-numerical techniques as delineated in References 4 and 5. This analysis is based on the split-rigidity concept of stiffened shells. Since the reinforced concrete section will crack from thermal and mechanical loadings, the membrane and bending stiffnesses will change from that of an isotropic, elastic, homogeneous section. In the vicinity of the opening, the containment wall is idealized as having an equivalent reduced membrane thickness/stiffness and an equivalent reduced bending thickness/stiffness. These effective thicknesses are incorporated into the governing differential equations of Reference 4, the equations are then resolved, and certain material-geometric parameters are subsequently modified to be used in conjunction with References 4 and 5. The references above are used to find the stress amplification around the openings from meridional membrane forces, pressure loadings (internal and external), and thermal moments. Reference 6 is used to evaluate the state of stress around the openings from tangential shears which are induced by a seismic event or by wind loadings.

3.8.1.4.1.1.4 Variation of Physical Material Properties

The effects of possible variations in the physical properties of materials on the analytical results are as follows. Since the containment is a non-prestressed, conventionally reinforced concrete structure, significant reduction in the concrete modulus of elasticity from sustained creep effects is not expected. A higher initial value of this modulus is used in the analysis, which will conservatively determine the upper bound effects from thermal stresses in the structure. The variation of the soil modulus of elasticity and Poisson's ratio of concrete is bounded in the analysis in order to constitute the most severe structural analysis.

3.8.1.4.1.2 Design Methods

The final design of the containment shell and foundation mat is preceded by the summation of all axisymmetrical and nonaxisymmetrical loads as required by loading combinations defined in subsection 3.8.1.3.8. The structure is designed for S-loading combinations by the techniques of Section 8.10 (alternate design method) of ACI 318-71. All stress limitations of this section are used, except that the permissible tensile stress for reinforcing is a maximum of 0.5 f_y and not more than 30,000 pounds per square inch.

The structure is designed for U-loading combinations by use of the ultimate design method of ACI 318-71, except for the design of primary tensile reinforcing which is required to resist flexural and/or membrane loads. This reinforcing is designed by use of a modified working stress approach (alternate design method), in which the straight-line theory of stress and strain is applicable. The maximum allowable concrete and reinforcing stresses for the containment shell and foundation mat, when subjected to flexural and/or membrane loads, are as follows:

f_c (allowable concrete stress) = $\emptyset(.85)(f'_c)$

f_s (allowable reinforcing stress) = $\emptyset f_y$

Where,

$\emptyset = 0.9$

$f'_c = 5000$ psi

f_y = 60,000 psi

The structure is designed for load cases which include thermal stresses as follows:

a. Stresses for a cracked section are calculated by the combination of thermal stresses and all membrane forces present by procedures of Appendix A of ACI-349, Criteria for Reinforced Concrete Nuclear Power Containment Structures, Title No. 69-2.

b. Stresses for a cracked section are calculated, for mechanical moments resulting from loads, independently of the effects considered in step a.

c. The results of steps a. and b. are added to obtain the total concrete and reinforcing stresses.

In the above design procedure, stresses from operating thermal gradients are treated as live loads; i.e., the presence or absence of these gradients will always constitute the most severe structural loading.

Primary tensile reinforcement, which terminates in a location where membrane tension or membrane plus flexural tension is predicted in a direction perpendicular to the bars, is anchored by use of special mechanical anchorage details. Primary tensile reinforcement, which is spliced in a region where membrane tension or membrane plus flexural tension is predicted in a direction perpendicular to the bar to be spliced, is spliced with a full positive mechanical splice (Cadweld).

Tangential (membrane) shear acts tangentially to the containment shell. The provisions of Section 11.16 in ACI 318-71 are not applicable to the design for this shear. The containment wall and dome are designed for tangential shear, using the governing results of two design methods.

In Method 1, the tangential shear stress carried by the concrete (v_c) is taken to be equal to 12,000 ρ for $\rho \leq 0.01$ and equal to 93 + 2700ρ for $0.01 \leq \rho \leq 0.025$, up to a maximum of 160 psi, where v_c is given in psi and ρ is the lesser of the reinforcement steel ratios in the meridional and circumferential directions.

When v_u is less than or equal to v_c, the concrete is assumed to completely resist the tangential shear. When v_u is greater than v_c, inclined reinforcement is proportioned to resist 1.5 times the excess shear stress ($v_u - v_c$).

In Method 2, the allowable tangential shear stress carried by the concrete (v_c) is taken as a maximum of 40 psi, provided that the ratio of reinforcement both vertically and horizontally is a minimum of $60/f_y$ (f_y in pounds per square inch), in addition to that required for membrane tension. Also, the ratio of reinforcement both vertically and horizontally shall be a minimum of 0.004, including that required for membrane tension. When vu is less than or equal to v_c (i.e., 40 psi), the concrete is assumed to completely resist the tangential shear. When v_u is greater than v_c, inclined reinforcement is proportioned to resist the excess shear stress ($v_u - v_c$). In this method, v_u is taken to equal v_T/φ where v_T is the tangential shear stress from the analysis and φ is equal to 0.85.

The inclined reinforcing is placed in two mutually perpendicular directions in the form of a helix and is located within the containment wall and a major portion of the dome. In both methods, the nominal shear stress (v_u) does not exceed $8 \sqrt{f_c'}$ (f_c' in pounds per square inch).

The containment structure is designed for radial shear by methods of Chapter 11 of ACI 318-71, except that subsection 11.2.2 is not applicable. As stated in Section 11.2.1 of ACI 318-71, the capacity reduction factor (φ) is included in the nominal shear stress (v_u). Actual stirrup stresses, computed by ACI equation (11-13), are therefore modified by the factor φ and are subsequently compared to ASTM yield strength (f_y).

3.8.1.4.1.3 Comparison of Design Criteria

The following is a comparison of load combinations and design criteria presented in ACI-359-80, Section III, Division 2, Article CC-3000 with those committed to in subsection 3.8.1.3.8. The design criteria comparison is presented in Table 3.8-44.

Equivalent Terms Used in Load Combinations

ACI-359-80 Table CC-3230-1 (Load definitions are found in Section CC-3200)	GG FSAR (Load definitions are found in subsection 3.8.1.3)
D	D
L	L
F	N/A
P_t	$1.15P_{cd}$
P_a	P_{CD}
T_t	T_t
T_o	T_o
T_a	$T_o + T_a$

ACI-359-80 Table CC-3230-1 (Load definitions are found in Section CC-3200)	GG FSAR (Load definitions are found in subsection 3.8.1.3)
E_o	E
E_{ss}	E'
W	W
W_t	W'
R_o	Effect included in T_o term
R_a	Effect included in $(T_o + T_a)$ term
R_r	R
P_v	Z
H_a	B

Comparison of Load Combinations

Service Loads (S-Loads)

Test Case:

CC-3230-1: $S = 1.0D + 1.0L + 1.0F + 1.0P_t + 1.0T_t$

(Continued) Comparison of Load Combinations

FSAR: $S = 1.0D + 1.0L + 1.15P_{CD} + 1.0\ T_t$

Equations are equivalent. This case controls in the containment design.

Construction Case:

CC-3230-1: $S = 1.0D + 1.0L + 1.0F + 1.0T_o + 1.0W$

FSAR: $U = 1.0D + 1.25L + 1.0T_o + 1.25W + 1.0Z$

The overall load combination is represented in the FSAR equation with a higher load factor for live (L) and wind (W) loads (1.25 vs. 1.0) in a factored load combination (U) as opposed to a service load combination (S). This case does not control in the containment design.

Normal Case:

CC-3230-1: $S = 1.0D + 1.0L + 1.0F + 1.0T_o + 1.0R_o + 1.0P_v$

FSAR: $S = 1.0D + 1.0L + 1.0T_o + 1.0Z$

Equations are equivalent. This case controls in the containment design.

Severe Environmental Case:

CC-3230-1: $S = 1.0D + 1.0L + 1.0F + 1.0T_o + 1.0(E_o \text{ or } W) + 1.0R_o + 1.0P_v$

FSAR: $U = 1.0D + 1.25L + 1.0T_o + 1.25(E \text{ or } W) + 1.0Z$

The overall load combination is represented in the FSAR equation with a higher load factor for live (L), wind (W), and OBE (E) (1.25 vs. 1.0), in a factored load combination (U) in place of a service load combination (S). This case does not control in the containment design.

Factored Loads (U-Loads)

Severe Environmental Case:

CC-3230-1: $U = 1.0D + 1.3L + 1.0F + 1.0T_o + 1.5(E_o \text{ or } W) + 1.0R_o + 1.0P_v$

(Continued) Comparison of Load Combinations

FSAR: $U = 1.0D + 1.25L + 1.0T_o + 1.25(E \text{ or } W) + 1.0Z$

The overall load combination is represented in the FSAR equation with a lower load factor in the live load (L) term (1.25 vs. 1.3) and the OBE (E) and wind (W) load terms (1.25 vs. 1.5). Since this load combination is specified under service load combinations, it need not be checked under factored loads per ACI-359. Also, this case does not control in the containment design.

Extreme Environmental Case:

CC-3230-1: $U = 1.0D + 1.0L + 1.0F + 1.0T_o + 1.0E_{ss} + 1.0R_o + 1.0P_v$

FSAR: $U = 1.0D + 1.0L + 1.0T_o + 1.25E + 1.0Z$

Safe Shutdown Earthquake (E') is not considered in combination with a containment differential pressure variation (Z). This case does not control in the containment design.

Extreme Environmental Case:

CC-3230-1: $U = 1.0D + 1.0L + 1.0F + 1.0T_o + 1.0W_t + 1.0R_o + 1.0P_v$

FSAR: $U = 1.0D + 1.0L + 1.0T_o + 1.0W' + 1.0Z$

Equations are equivalent. This case does not control in the containment design.

Abnormal Case:

CC-3230-1: $U = 1.0D + 1.0L + 1.0F + 1.5P_a + 1.0T_a + 1.0R_a$

FSAR: $U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.5\ P_{CD} + 1.0R$

Equations are equivalent. This case controls in the containment design.

Abnormal Case:

CC-3230-1: $U = 1.0D + 1.0L + 1.0F + 1.0P_a + 1.0T_a + 1.25R_a$

FSAR: $U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.5\ P_{CD} + 1.0R$

(Continued) Comparison of Load Combinations

The overall load combination is represented in the FSAR equation with a higher load factor for design pressure (P_{CD}) (1.5 vs. 1.0) and a lower load factor for pipe break loads (R) (1.0 vs. 1.25). This case controls in the containment design.

Abnormal + Severe Environmental Case:

CC-3230-1: $U = 1.0D + 1.0L + 1.0F + 1.25P_a + 1.0T_a + 1.25\ (E_o \text{ or } W) + 1.0R_a$

FSAR: $U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.25\ P_{CD} + 1.0R + 1.25\ (E \text{ or } W)$

Equations are equivalent. This case does not control in the containment design.

Abnormal + Severe Environmental Case:

CC-3230-1: $U = 1.0D + 1.0L + 1.0F + 1.0T_o + 1.0\ (E_o \text{ or } W) + 1.0\ H_a$

FSAR: $U = 1.0D + 1.0L + 1.0(EB \text{ or } W) + 1.0B$

It should be noted that a typographical error occurs in the ACI-359 version of this equation where the load factor "0.5" under E_{ss} is intended to read "or." OBE load (E_B) includes effects of the flooded condition. Containment wall temperatures are considered to be at or near ambient temperatures in the regions of maximum stress (base slab and lower wall). Temperature stresses are considered to be negligible in these areas and are, therefore, not included in the FSAR load combination. As such, these equations are equivalent. This case controls in the containment design.

Abnormal + Extreme Environmental Case:

CC-3230-1: $U = 1.0D + 1.0L + 1.0F + 1.0P_a + 1.0T_a + 1.0E_{ss} + 1.0R_a + 1.0R_r$

FSAR: $U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.0P_{CD} + 1.0R + 1.0E'$

Equations are equivalent. This case controls in the containment design.

3.8.1.4.2 Liner Plate

All analysis and design of the liner plate was done in accordance with the following Bechtel Corporation Topical Reports:

BC-TOP-1

BC-TOP-5A (for liner plate only)

The Grand Gulf liner plate which is backed by concrete does not classify as a pressure boundary and is not designed in accordance with ASME Section III, Division I. However, the suppression pool liner plate is affected by periodic negative pressure loading due to SRV discharge. The suppression pool floor plate, having 10-foot spacings between supports, was designed on the basis of the following assumptions:

a. The initial absolute pressure on the underside of the plate is atmospheric pressure (14.7 psia).

b. The air space beneath the plate acts like a confined adiabatic volume of air with a restrictive opening to the air gap volumes under adjacent plates. This assumption has been substantiated using a one-dimensional transient heat transfer analysis that determines a conservative transient temperature increase in the air gap between the suppression pool liner plate and the concrete floor following a relief valve discharge. The results of the analysis show that, during the transient, the gap vapor temperature increase is insignificant, and the assumption of an adiabatic gap atmosphere is valid.

c. The volume is assumed as an initial prismatic air gap of about 0.3 inch between the plate and concrete. This volume occurs due to the “as built” imperfect concrete surface condition, straightness of the liner plate, and shrinkage of concrete throughout the 40-year plant life. A sensitivity study was performed by varying the gap size. This study showed that the initial gap is not a major parameter affecting the response of the liner. An “as built” survey of the concrete foundation conducted 3 years after placement confirms the conservatisms of this initial condition.

d. The restrictive opening is assumed as a 1/32-inch crack between the T-beam and concrete, and the leakage path is postulated to be between the T-beam and the concrete (see Figure 3.8-116). The actual analysis assumed a crack of constant width throughout the perimeter of a liner plate panel and an unlimited supply of air from the adjacent panel. An "as built" survey of the concrete surface indicates a high-quality finish and very little surface cracking. All cracks found were repaired.

The above assumptions were the basis of the analysis which produced the structural response for the suppression pool floor plates utilizing closed form solutions of plate theory and a computer code that performed a single degree of freedom, non-linear, elastic, dynamic analysis. The results showed that the floor plates did not exceed yield stresses and stayed well below the design criteria maximum of 90 percent of the yield stress. A review of the ASME Code, Section III, Division 2, showed that this particular type of loading was not addressed in the code; therefore, the stress limits of Division 2 were not applicable. For a comparison of stresses found in the suppression pool liner plate for negative pressure loads, see Table 3.8-26.

The analysis for negative pressure loads on the wall was performed in accordance with the procedure outlined in the above-referenced Bechtel Topical Reports. The analysis and design of all plates in the suppression pool considered the effects of buckling, bending, and fatigue. Fatigue analysis for the suppression pool and cylinder wall liner plate was performed using subsections NE and NB of the ASME Code, Section III, Division I, 1971 Edition with Summer of 1973 Addenda. The liner was found to conform with the criteria of the ASME Code.

The liner plate was included in the linear elastic computer model used to analyze the containment concrete cylinder wall for all associated load cases. Based on this linear elastic analysis, the computer program results obtained show stresses in the liner plate in excess of the material yield stress for certain containment load cases. These stresses are not considered significant because the plate stresses will be limited by the material yield stress, and the plate functions as a leaktight barrier instead of a load-carrying element. For these reasons, and because no allowable stresses were stated in any of the documents used for this design, the yield stress of the material has been chosen for comparison in Tables 3.8-4 and 3.8-5. Although

it has not been used for design, the limits for strain from the ASME Boiler and Pressure Vessel Code Section III, Division 2, have also been used for comparison. (Grand Gulf is not designed to the requirements of ASME Section III, Division 2.)

The results of the analysis show that the liner plate does yield for some critical load cases, but the strains are within the limits of those specified in ASME Section III, Division 2. For construction loads the plate remains within the stress limits specified in ASME Section III, Division 2.

3.8.1.4.3 Polar Crane Brackets

Steel design was performed in compliance with the elastic method for steel structures in accordance with the "Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings," (AISC) 1969 Edition. The bracket was designed as a fixed cantilever using the analytical techniques of BC-TOP-1, "Containment Building Liner Plate Design Report."

3.8.1.5 Structural Acceptance Criteria

3.8.1.5.1 Containment Structure

The actual and allowable stresses for the containment structure are found in Tables 3.8-2 and 3.8-3.

3.8.1.5.2 Liner Plate

The structural acceptance criteria for the liner plate is in accordance with the following Bechtel Corporation Topical Reports:

BC-TOP-1

BC-TOP-5A (for liner plate only)

The actual and allowable stresses for the containment liner plate are found in Tables 3.8-4 and 3.8-5.

3.8.1.5.3 Polar Crane Brackets

The polar crane brackets are designed for structural acceptance criteria as limited by subsection 3.8.1.3.9. For the normal operating and severe environmental load combinations, the

allowable stress (F_s) in no case shall exceed 0.90 f_y in bending, 0.85 f_y in axial tension, and 0.50 f_y in shear (where f_y is the material yield stress).

The actual and allowable stresses for the polar crane brackets are found in Table 3.8-6.

3.8.1.6 Materials, Quality Control, and Special Construction Techniques

3.8.1.6.1 General

Specifications covering all materials are in sufficient detail to assure the structural design requirements of the work.

3.8.1.6.2 Concrete

All concrete materials are approved prior to start of construction on the basis of their characteristics in test comparisons using ASTM standard methods. Concrete aggregates and cement, conforming to acceptability criteria of the specifications, are obtained from approved sources. Concrete properties are determined by laboratory tests. Concrete admixtures are used to minimize the mixing water requirements and increase workability. The specified compressive strength at 91 days, or earlier, is:

Structure or Component	Specified Strength (f'_c) (psi)
Precast concrete panels	4000
Containment and internals	5000
All other seismic Category I concrete	4000
Mass concrete fill	2000
RPV shield wall concrete fill	5000
Electrical duct encasement	2000

a. Water used in mixing of concrete, mortar, and grout is tested in accordance with the provisions of American Association of State Highway Officials (AASHO) Standard T26-70, Standard Method of Testing for Quality of Water to be Used in Concrete.

Acidity or alkalinity (pH factor) is within the range of 6.0 to 8.5, and the water does not contain impurities in excess of the following limits:

	Percent	ppm
Total chlorides as Cl	0.025	250
Total sulphates as SO_4	0.10	1,000
Total organic solids	0.05	500
Total inorganic solids	0.20	2,000

b. All concrete work is in accordance with ACI 318-71, Building Code Requirements for Reinforced Concrete, except as modified in subsection 3.8.1.2.2.2, and complies with the following specifications:

Material	ASTM Specification
Cement, Type II low alkali	C 150-72
Normal aggregate	C 33-71a
Heavy aggregate	C 637-73 and C 638-73
Fly ash	C 618-73 (see note below)
Air-entraining admixture	C 260-69
Water-reducing admixture	C 494-71
RPV Shield Wall fill concrete aggregate	C 637-73

Note: Exceptions to ASTM C 618-73: in-process testing frequency shall be for each 200 tons in accordance with ANSI N45.2.5, Table B; in addition, passing the Pozzolanic Activity Index with either cement or lime shall be acceptable.

3.8.1.6.3 Reinforcing Bars

Reinforcing bars for concrete consist of deformed bars meeting the requirements of ASTM A 615-75 (including Supplementary Requirements S-1), Grade 60, with additional requirements as set forth below for tension and bend testing.

All test specimens are full-diameter bars, as rolled.

One tension test is made for each 50 tons, or fraction thereof, of each bar size of reinforcing bars produced from each heat and production rolling for the determination of tensile properties. Acceptance standards for tension testing are in conformance with the requirements of Table 2, ASTM A 615-75, including the retest provisions of Section 11, ASTM A 615-75.

One bend test is made for each test lot and production rolling of each bar size of each heat for the determination of bend properties. Acceptance standards for bend testing are in conformance with the requirements of Table 3 and Supplementary Requirements S-1, ASTM A 615-75.

Normally, testing is accomplished at the mill or shop of the supplier; the purchaser's representative has the option at any time of witnessing the selection, cutting, and testing of test specimens. Any of the required testing which has not been accomplished prior to receipt of the reinforcing bar at the construction site is accomplished as part of the receiving inspection process at the construction site.

3.8.1.6.4 Structural Steel

The structural steel materials conform to all applicable requirements of the AISC Manual of Steel Construction, unless otherwise noted in the job specifications or on the drawings, and also complies with the following specifications:

a. Materials for Interior Structures

Material	ASME or ASTM Specification
Structural steel and connections	A 36
High strength structural steel	A 441, A 572, A 516, A 540
Tubular steel sections	A 53, Types F and S; A 500, Grade A or B; A 501
Anchor bolts	A 36, A 449
Shop fabrication	AWS D1.1, welding or bolting A 325 and A 490 bolts
Field erection	High strength bolts, A 325 and A 490; or welding AWS D1.1

Material	ASME or ASTM Specification
RPV anchor bolts	ASME SA 540

b. Materials for the Containment

Material	ASME or ASTM Specification
Liner plate:	
Above El. 117-10	A 285
Below El. 117-10	A 240, Type 304L
Thickened plate penetrations and inserts above El. 117-10	A 516, Grade 70
Penetrations:	
Welded Pipe	SA 155 KCF, Grade 70
Nozzles	SA 333, Grade 1 or 6; SA 312 A 106
Nozzle Reinforcement	SA 516, Grade 70
Anchoring rings and plates	SA 516, Grade 70
Rolled Shapes:	A 36
Fasteners:	
Structural steel	A 325 or A 490
Anchor bolts	A 307 or A 36
Concrete studs	A 108

[HISTORICAL INFORMATION] [Steel plate is tested at the mill in full conformance to the applicable ASTM specifications, and certified mill test reports are supplied for review and approval. The plate is visually examined for laminations and pitting. Identity of the plate is maintained throughout fabrication.]

3.8.1.6.5 Control Tests for Concrete

a. [HISTORICAL INFORMATION] [Sample Tests

Samples for the following routine in-process tests of concrete are taken at the sampling point, at the frequencies shown, in accordance with ASTM C 172-71. This point shall be at the central mix plant discharge. Correlation test samples shall be taken at the end of the

last piece of conveying equipment at the point of discharge of concrete into the forms. Water shall not be added to any concrete batch after in-process tests are made unless the in-process tests (except unit weight/ yield) are repeated after the water is added. Samples shall not be taken from concrete after it has been deposited in the forms. The sampling frequencies are as follows:

1. Temperature: First batch produced each day and every 50 cubic yards placed for each class of concrete.

2. Slump (ASTM C 143-71): First batch produced each day and every 50 cubic yards placed for each class of concrete.

3. Air Content (ASTM C 231-72T): First batch produced each day and with each set of compressive strength cylinders for each class of concrete.

4. Plastic Unit Weight/Yield (ASTM C 138-71T): Daily during production for each class of concrete.

5. Compressive Strength (ASTM C 31-69, tested in accordance with ASTM C 39-72): Standard 6-inch by 12-inch concrete test cylinders will be molded and tested at 7, and 28 or 91 days from each 100 cubic yards, or a minimum of one set per day for each class of concrete.

Correlation tests for air content, slump, and temperature of concrete shall be made at the end of the last piece of conveying equipment for the first batch produced each day and for each 200 cubic yards or fraction over 100 cubic yards placed of each class of concrete.

b. Concrete Cylinders

A strength test is the average of the strengths of the two specimens from each sample tested at 28 or 91 days. Only one cylinder shall be made and tested at 7 days. The provisions of ACI 318-71, Paragraph 4.3.3, shall apply in determining the satisfactory strength level of the concrete tested at 28 or 91 days.]

3.8.1.6.6 Splices of Reinforcement

a. Provisions of ACI 318-71, Sections 7.5, 7.6, and 7.7, will be met for all reinforcing bar splices. Splices are designed to develop the specified minimum yield strength.

b. All mechanical splices for reinforcing bars are made by Cadweld process, using clamping devices, sleeves, charges, etc., as specified by the Cadweld Splice Instruction Sheets for B and T series connections. C series materials are not permitted.

   Qualification of operators, visual inspection, tensile testing, tensile test frequency, and procedure for substandard tensile test results are in accordance with the Cadweld Instruction Sheets, and will comply with NRC Regulatory Guide 1.10, Revision 1, dated 1/2/73.

3.8.1.7 Testing and Inservice Surveillance Requirements

3.8.1.7.1 Structural Integrity Pressure Test (Unit 1)

[HISTORICAL INFORMATION] [Following construction, the containment was proof-tested at 115 percent of the design pressure. During this test, deflection and concrete crack measurements were made to determine that the actual structural response was within the limits predicted by the design analyses.

A test procedure was issued prior to the Structural Integrity Test and was based on the containment being a non-prototype structure. This procedure was in compliance with the requirements of NRC Regulatory Guide 1.18, to the extent noted in Appendix 3A.

The containment was pressurized pneumatically in four pressure increments from atmospheric pressure to the maximum test pressure of 17.25 psig, as shown in Figure 3.8-122. Normal operational water volumes were present, and the drywell structure was vented through the drywell personnel lock. Pressurization and depressurization were halted at 5.0, 10.0, 15.0, 17.25, 15.0, 10.0, and 5.0 psig to allow acquisition of the required structural integrity test data, i.e., deflection and crack patterns. At these plateaus, the pressure was maintained for a minimum of 1 hour to determine if indicated displacements lagged the actual pressurization/depressurization cycle. In addition, deflection data were recorded at each 2.5 psig change in pressure.

Gross structural deformations were measured using taut invar wire extensometers that spanned between points on the containment wall, dome, and springline and fixed points within the structure. The extensometers were located to measure radial displacements along typical wall sections and around the lower equipment hatch, vertical displacement of the dome relative to the operating floor, and vertical displacement of the spring-line relative to the foundation slab. The layout of the extensometer system is shown in Figures 3.8-123 through 3.8-125.

Concrete crack patterns were mapped in the areas shown in Figure 3.8-126.

The lengths and widths (measured by optical comparator) of all visible cracks within these areas were recorded at specified pressure levels.

A detailed description of the SIT is given in the "Final Report on Primary Reactor Containment Structural Integrity Test Performed at The Grand Gulf Nuclear Station Unit 1 for Mississippi Power and Light Company," dated January 1982.

A summary discussion of the results of the SIT test is given in the following paragraphs.

a. Radial and Vertical Deflections

   Tables 3.8-41 through 3.8-43 provide the radial and vertical deflections observed during the test at each displacement transducer location, and the corresponding predicted values. The actual vertical and radial deflections were below the maximum predicted values for all locations.

   The response of the containment to the maximum test pressure of 17.25 psig is illustrated in Figure 3.8-127, which shows the measured radial and vertical deflections of the cylinder wall and the measured vertical deflection of the dome. Also shown are the predicted deflections based on both elastic and cracked section analyses. The measured vertical deflections of the dome and wall are less than the elastic analysis prediction. This demonstrates that the membrane stresses in the dome and the vertical membrane stress in the cylinder wall were not sufficient to cause significant tensile cracking. Also, the modulus of elasticity used in elastic deflection

prediction calculations was based on the specified concrete compressive strength rather than the actual strength, which is higher, as demonstrated by concrete cylinder break tests. Higher concrete strength, and, consequently, higher modulus of elasticity, will result in deflections lower than the predicted, as observed from the actual vertical deflection measurements.

The radial deflection of the wall is between the elastic and cracked wall section case predictions. This demonstrates that hoop stress caused tensile cracking but did not develop the complete cracked section used as a model in the cracked section analysis.

The wall radial deflections listed and plotted in Figure 3.8-127 show averages of the measurements made on several azimuths at each elevation. The individual measurements made at each elevation show a variation from azimuth to azimuth, which can be attributed to the containment shell not being constructed perfectly axisymmetric. On the dome, the measured deflections show a smooth trend. The measured wall vertical deflection listed in Figure 3.8-127 is the average of the measurements at extensometers V_1, V_3, and V_4 (V_2 malfunctioned). The individual measurements are tightly grouped, as indicated in Table 3.8-41, demonstrating a uniform vertical elongation of the cylindrical portion of the containment.

Figure 3.8-128 also shows the measured radial growth of the wall in the vicinity of the equipment hatch. The deflections measured along the horizontal center plane of the hatch are approximately symmetrical about the vertical center line. Measurements along the vertical center plane of the hatch show that the outward movement in the vicinity of the hatch increased with elevation. This behavior is typical for a large opening located close to the containment base mat.

Measurements of 24-hour residual deflections showed an 85 percent recovery of the maximum recorded vertical deflection at the dome apex (extensometer D_1), an 88 percent recovery of the average maximum vertical deflections recorded at the springline (extensometers V_1, V_3, V_4), and a 72 percent recovery of the average maximum

radial deflections recorded at the elevation that exhibited the maximum average peak radial deflection (extensometers H_{19}, H_{20}, H_{21}).

Recovery values are based on an average of the 24-hour residual deflections recorded at the appropriate extensometers, as tabulated in the "Primary Containment Structural Integrity Test Report," dated January 1982. In all cases, the deflection recovery within 24 hours after complete depressurization, at the points of maximum recorded deflection for the dome vertical direction and for the cylinder wall radial and vertical directions, was greater than 70 percent.

b. Containment Concrete Cracking

The surface concrete crack mapping areas on the containment wall are shown in Figure 3.8-126. A variety of surface cracks with widths less than 0.010 inch were observed immediately prior to pressurization.

Concrete crack widths in the various surveillance areas did not change in measured width by more than 0.010 inch. This is consistent with the low deflections recorded and does not adversely affect the integrity of the containment structure.

During the subsequent ILRT pressurization cycle to 12.5 psig, cracks that appeared in Crack Mapping Area 2 were monitored. Those cracks enlarged slightly from the zero pressure recovery value and returned to hairline width not accurately measurable with the optical comparators. Growth during the ILRT pressurization cycle was estimated at 0.002 inch at the locations observed.]

c. Summary and Conclusions

The deflections met the acceptance criteria set in the Grand Gulf test procedure.

[HISTORICAL INFORMATION] [The containment structural integrity test provided proof of the structure's ability to contain the internal design pressure and provided measurement of structural response to changes in internal pressure. Test measurements for the containment included gross structural deformations and concrete crack growth.

The structure withstood the internal pressures with no observable indications of structural distress. All measured structural deformations were less than the maximum predicted values, and all recovery deflections at the points of maximum average deflection recovered more than 70 percent of their maximum deflection. No cracks with widths equal to or exceeding 0.010 inch were observed in the crack mapping areas prior to pressurization. Changes in concrete cracks did not change in measured width by more than 0.010 inch during pressurization, and those cracks that developed during pressurization closed to below measurable values at zero pressure.]

The results of the structural integrity test provide direct experimental evidence that the containment structure can contain the internal design pressure with sufficient margin of safety and that the gross response to pressure is predictable.

3.8.1.7.2 Preoperational Integrated Leak Rate Test

[HISTORICAL INFORMATION] [Following construction of the containment, an integrated leak rate test of the containment and its penetrations is conducted at 100 percent of the calculated peak pressure. This test is discussed in subsection 6.2.6.]

3.8.1.8 Ultimate Capacity of Containment

The ultimate capacity of the containment is based on the static capacity of the reinforced concrete containment shell, the containment equipment hatches, and the personnel airlocks. The ultimate static capacity is defined as the containment pressure at which a general yield state is reached at a critical structural section. As stated in subsection 6.2.5.1.1n, the design basis of the hydrogen ignition system precludes the necessity of determining the ultimate dynamic pressure capacity of the containment for hydrogen combustion.

The ultimate capacity of the concrete containment shell is calculated from the yield strengths of the hoop reinforcement in the critical structure section of the containment. Analytical results indicate that the ultimate capacity is 56 psig based on a specified material yield strength of 60 ksi for reinforcing steel. The actual material strengths of the reinforcement are determined from the certified mill test report. Using a normal distribution curve, the mean reinforcing steel material yield

strength is taken as the average strength of the population. The lower and upper bound strengths are the values corresponding to a standard deviation of 2.0 and 1.4 respectively, from the mean. The containment ultimate capacities, based on the actual reinforcing steel yield strengths (i.e., lower, mean, and upper bounds), are 62, 67, and 70 psig, respectively. The liner plate strains corresponding to the containment ultimate pressures of 56, 62, 67, and 70 psig are 0.22, 0.24, 0.27, and 0.28 percent, respectively.

The ultimate pressure retaining capacities of the containment equipment hatch and the lower containment personnel airlock are 206.5 and 77.6 psig, respectively. The ultimate capacity of the upper containment airlock is determined to be 65.5 psig after implementation of structural modifications to the airlock.

Each containment airlock door has two inflatable seals that are maintained at a nominal pressure of 70 psig. Opening an airlock door, however, requires for its seals to be deflated. Before the other door on the same airlock can be opened this door must be closed and its seals must be reinflated up to the 60 psig nominal interlock setpoint. This interlock ensures the pressure intergrity of containment is maintained up to 56 psig when the airlocks are in use.

The liner plate in the vicinity of containment penetrations is reinforced to reduce the stress concentration effect of the opening and is typically attached to the penetration sleeve by a full penetration groove weld. The use of E-70XX low hydrogen electrodes for these full penetration welds results in a weldment which has strength and ductility characteristics comparable to the base material. Analysis of a major penetration (personnel airlock) in the vicinity of maximum liner plate strains indicates that the subject weld will be well within the elastic limit and would not constitute a leakage path.

3.8.2 Steel Containment

As described in subsection 3.8.1, the containment is a reinforced concrete structure; therefore this subsection does not apply.

3.8.3 Concrete and Steel Internal Structures of the Concrete Containment

3.8.3.1 Description of the Internal Structures

3.8.3.1.1 Drywell Structure

The majority of the drywell is designed as a reinforced concrete structure. The primary structural aspects of the drywell consist of four major components.

a. A flat, circular reinforced concrete foundation which supports the drywell walls and is common to the containment foundation.

b. A right, vertical, cylindrical wall section which is further subdivided into two structural components which differ in both design and analytical approach. The lower wall has two stiffened steel, concentric, cylindrical, surface plates which are penetrated by a series of horizontal, cylindrical, steel vents. The annulus between the surface plates is stiffened by vertical, radial plates and is filled with concrete. Shear connectors are attached to the embedded surface of the cover plates to develop composite action. The plates are designed to carry all membrane tensile force in this region. This lower section is supported by and anchored to the containment base slab and extends up to El. 117'-10". At El. 109'-9", the lower steel section is connected integrally with the upper wall vertical and diagonal reinforcement. The upper wall is designed as a reinforced concrete cylinder which is supported by the steel, lower wall section and internal concrete as mentioned above and extends up to El. 180'-2" where it supports the drywell roof.

c. A flat, horizontal, circular, reinforced concrete roof contains a circular opening for the drywell head which is located at the axis of symmetry of the roof. The roof is supported by and integrally connected to the cylindrical wall at the outer edge and in turn supports the embedded lower flange of the drywell head at the edge adjacent to the circular opening. The flat roof section is stiffened by two longitudinal reinforced concrete walls which are part of the upper pool wall system.

d. The drywell head is located directly over the reactor pressure vessel. The head is a short stainless steel cylinder with an elliptical head which is part of the drywell pressure retention boundary during plant operation (see Figure 3.8-59). During refueling operations, the drywell head is removed to permit unobstructed access to the top of the reactor pressure vessel. The head is equipped with a pair of horizontal mating flanges. The lower flange is equipped with two compression seals. The annulus between the compression seals (see Figure 3.8-59) can be pressurized and monitored to test the leaktight integrity of the head assembly without necessitating pressurization of the entire drywell. The elements of the drywell completely surround the reactor pressure vessel and the recirculation system as shown in Figure 3.8-60.

The drywell is designed as the temporary pressure retention boundary which separates the reactor pressure vessel and the recirculation system from the suppression pool and the containment annulus. If a LOCA should occur, the drywell temporarily retains the released steam and channels it around the weir wall and through the horizontal vents and into the suppression pool, thus minimizing the energy released to the containment annulus. The drywell wall is designed to be penetrated by piping as shown on Figure 3.8-60 and wall guard pipes as shown on Figure 3.6A-33.

Personnel air lock and equipment hatch openings penetrate the drywell cylindrical wall as shown on Figures 3.8-58 and 3.8-61. Each of the two doors on the personnel air lock is fitted with two inflatable rubber seals to ensure the leaktightness of the lock. The pressure within the seals can be monitored during normal operation to further ensure the integrity of the lock. The personnel lock is equipped with air accumulators which have been sized to keep the seals inflated for 30 days in the unlikely event of loss of the plant air supply. All reinforcement is continued around the lock sleeve as shown in Figure 3.8-58 and, because of this detail, the lock sleeve is not considered to significantly affect the overall structural performance of the cylindrical wall section.

The equipment hatch (see Figure 3.8-61) is designed to be removed during plant maintenance. During plant operation the drywell equipment hatch is part of the drywell pressure retention boundary and as such uses two compression seals around its

periphery to maintain its leaktightness along the mating surfaces. Testing of the annulus between the compression seals can be performed to assure the leaktightness of the hatch seal. The embedded hatch frame is designed for and acts as end anchorage for all of the wall reinforcement in this vicinity and, as such, is considered to be a structural element. During normal operation, the hatchway through the wall is filled with removable nuclear shielding.

Drywell electrical penetrations are designed to provide passage of electrical cables through the drywell wall. In addition, while not part of the fission product barrier, the penetrations must be of sufficient leaktightness so that the overall drywell design basis leakage is not exceeded (see subsection 6.2.1.1.5). To meet these objectives, each electrical penetration consists of steel conduit, not larger than 6 inches in diameter, with a single fitting (see Figure 3.8-62). After cable installation, this fitting is filled with a compound which will solidify in the seal chamber and prevent an exchange of atmosphere between the containment and the drywell. The filler material is designed to resist the highest pressure and temperature which could be attained in the drywell. All materials used in the penetrations are capable of withstanding LOCA conditions. The drywell end of the electrical penetrations is open, and any gases resulting from the unlikely event of an electrical failure within the penetration will be vented inward.

In the unlikely event of separation of a seal fitting from a conduit (or other conceivable mechanism), which would result in destruction or bypass of the seal for any one penetration, the resulting bypass leakage through the total internal cross section of the penetration would not be sufficient to impair the basic drywell function. As shown on Figure 3.8-62, each conduit penetration will be partially filled with cable that will occupy a portion of the cross section and will further reduce the potential bypass capacity.

Typical mechanical and control systems penetrations are shown on Figure 3.8-60. These penetrations are detailed and designed to be leaktight. During normal operation, the leakage past these penetrations will be negligible.

3.8.3.1.2 Weir Wall

The weir wall is a vertical, reinforced concrete, right circular cylinder that is fixed at the bottom and free at the top. It measures 16'-7" in height and has an inside radius of 32'-6" and an outside radius of 34'-0". The outside face of the weir wall is completely covered with a stainless steel clad, carbon steel liner plate, 3/8-inch thick, which forms the inner boundary of the suppression pool (see Figure 3.8-60).

The weir wall supports the steel floor framing at El. 114'-6", pipe supports, and pipe restraints, and it is also designed to withstand the jet impingement force associated with a recirculation line pipe break. For weir wall behavior under loads, see subsection 3.8.3.4.2. The general arrangement of the weir wall is shown on Figure 3.8-60.

3.8.3.1.3 Refueling Pool and Operating Floor

a. Refueling Pool

The reinforced concrete refueling pool walls, located within the containment, are located on and integrally attached to the drywell roof (see Figures 3.8-60 and 3.8-63).

The outer walls form the boundary of the refueling pool. This large pool is subdivided by three interior cross walls into four smaller compartments. A horizontal transfer tube penetration is provided at one end of the pool to transfer fuel elements between the containment and the auxiliary building. The location of the transfer tube penetration is shown on Figure 3.8-1.

The base slab of the refueling pool is common to the drywell roof slab and is stiffened by the pool walls.

The interior face of the pool is lined with a stainless steel liner plate. The stainless steel liner plate is designed to minimize leakage of water from the pools, and to capture, via the leak chase channel system, any leakage that might occur through the plate joints (see Figure 3.8-64).

b. Operating Floor

The operating floor at El. 208'-10" is shown on Figure 3.8-64. It is designed to provide laydown space for refueling perations. Construction materials are either reinforced concrete or steel grating supported by structural steel framing. The containment and upper pool walls are designed as vertical supports for the floor, and the upper pool walls are designed to provide horizontal support for the floor. Connections which allow free horizontal motion are provided for the vertical support points at the containment wall.

3.8.3.1.4 Reactor Supports

The reactor support consists of a welded steel plate girder. It is constructed from 3-inch thick, ASTM A 36, ultrasonically tested, steel plates. The ring girder is fabricated with 120 4-inch diameter holes through which the RPV anchor bolts pass. The RPV anchor bolts have been designed, based on a shear-bearing type connection, to carry all of the reactor support loads induced by either seismic or thermal effects. The reactor supports are shown on Figures 3.8-65 through 3.8-65d.

3.8.3.1.5 Reactor Pressure Vessel (RPV) Support Pedestal and Mat

The RPV support pedestal is a vertical, reinforced concrete, circular cylinder 20'-7" high with a constant outside radius of 16'-4". The inside radius varies from 10'-7" (between El. 93'-0" and El. 114'-2") to 9'-5" (between El. 115'-4" and El. 121'-4"). The pedestal is located directly beneath the RPV and the shield wall. The pedestal also supports the upper and lower control rod drive (CRD) support platforms, the CRD removal platform, pipe whip restraints, pipe hangers, and snubbers. For a general arrangement of the RPV pedestal see Figure 3.8-60.

The vertical cylinder of the pedestal is rigidly connected to the 7'-9" thick, circular, reinforced concrete base mat which is 68'-0" in diameter. The outside face of the mat is covered with a stainless steel clad, carbon steel liner plate which serves as the lower extension of the weir wall forming the interior boundary of the suppression pool directly opposite the suppression pool vents. The outside edge of the pedestal mat acts as the base of the weir wall. The mat is attached to the containment base slab via the reinforcing steel tension ties at radius 32'-4" and shear key at radius 13'-7".

A large structural steel embedment is used in the pedestal to form the penetrations through which the CRD hydraulic piping passes. This structural embed spans over the CRD penetration at the 90 degree and 270 degree azimuths and is designed to carry all the loads associated with the reactor vessel and pedestal at this elevation. Anchorage of this embed is achieved through the use of high strength anchor bolts and steel reinforcing bars which transfer the loads to the body of the concrete pedestal.

The reactor vessel is attached to the concrete pedestal via 120 3-1/2-inch diameter high strength bolts.

3.8.3.1.6 Reactor Shield Wall

The wall is designed as a cylindrical wall which surrounds the RPV (see Figure 3.8-60) and rests on the reactor pedestal. Both surfaces of the wall are lined with a carbon steel plate for strength requirements. High density concrete is placed between the plate surfaces for shielding. This wall also supports steel floors at elevation levels 147'-7" and 161'-10" and resists high energy line break (HELB) forces. Some of the snubbers and hangers for high energy lines and recirculation pump motor are directly attached to the shield wall. The middle and lower pipe penetrations through the shield wall are equipped with shield plugs (except where flow diverters are provided), as shown on Figure 3.8-60, for inservice inspection. Flow diverters (See Figure 3.8-60a) are incorporated in the shield wall penetration sleeves for six feedwater lines and for two recirculation outlet lines. The effect of these flow diverters is to minimize the asymmetric loads associated with annulus pressurization in the unlikely event of a pipe break in any one of the feedwater or recirculation system outlet lines at the pipe-to-nozzle safe end weld. The flow diverters are cylindrical sleeves that are retractable to permit inservice inspection in accordance with Section XI of the ASME Code. On the recirculation lines, radiation shielding (both fixed and retractable) is provided along the outside of the flow diverters. An inspection opening is provided at a lower elevation through the shield wall to allow easy access to the lower portion of the reactor. The analysis procedure for the shield wall and shield plugs is found in subsection 3.8.3.4.5.

3.8.3.1.7 Other Interior Structures

a. Main Steam Pipe Tunnel

The main steam pipe tunnel provides shielding for process piping. It is designed as a reinforced concrete structure cantilevered from the drywell wall (see Figure 3.8-60). A gap is designed to separate the tunnel from the containment wall. The pipe tunnel is vented to allow for pressure decay if the RWCU lines rupture within the tunnel. Doorways and hatchways connect the air volume in the tunnel with the containment air volume. The main steam and feedwater lines are enclosed within the guard pipes in this room. Grating platforms are provided within the tunnel for inservice inspection.

b. Floor at Elevation 120'-10"

Located immediately above the suppression pool at elevation 120'-10" is an access platform fabricated from structural steel and covered with grating (see Figure 3.8-66). The structural steel has been designed from vertical plates and has no flanges to act as targets for the pool swell forces resulting from a LOCA. This design allows the water to wash through the floor unimpeded.

The traversing incore probe (TIP) station located at this level is a concrete slab cantilevered over the suppression pool. This reinforced concrete slab has a steel box structure that projects beneath the slab into the suppression pool (see Figures 3.8-66 and 3.8-66a). This box structure is designed to direct the impact loads, associated with pool swell away from the TIP station slab. Similar box-like structures are installed beneath the projections of the containment personnel lock, the drywell equipment hatch and the drywell personnel lock to serve the same purpose. Details of these structures are shown on Figures 3.8-66 and 3.8-66a.

The platforms at this level allow access from the containment personnel lock to the TIP station and drywell personnel lock or the drywell equipment hatch. The steel in the area of the equipment hatch is designed to support the recirculation pump motor and is designed for removal of the control rods.

The floor steel plates that make up the floor girders have been fabricated from ASTM A 36 stock and are corrosion protected with a galvanized coating in accordance with

ASTM A 123. In addition the ASTM A 325 bolts have been galvanized and the load indicating washers have been shearedized or mechanical galvanized.

All welding of the steel plates was performed prior to galvanizing in accordance with AWS D1.1. The bolted connections have been designed and erected in accordance with the AISC specification for structural steel.

c. Floor at Elevation 135'-4"

In the annulus between the containment and the drywell there is a structural steel and reinforced concrete floor at elevation 135'-4". Two composite concrete slabs are located 180 degrees apart from azimuths 60 degrees to 118 degrees and 240 degrees to 295 degrees and are used to support the hydraulic control units for the CRD system. The balance of the floor is grating on structural steel, and its function is to support various pieces of equipment.

The floor beams are supported by brackets embedded in the containment wall and the drywell wall. The containment wall brackets form part of the containment liner plate pressure boundary and have been designed in accordance with the requirements of the liner plate to ensure leaktight integrity. All beam connections at the containment wall brackets are sliding bolted joints to minimize wall movement differences.

The floor has been designed to withstand all normal operating loads including dead loads, live loads, and any dynamic equipment loads as well as seismic loads. In addition, this floor has been designed to withstand the uplift loads due to the dynamic effects of the pool swell phenomenon identified with the Mark III containment (see Figure 3.8-67). In order to address Humphrey Issue 1.7 (MP&L (SERI) Action Plan 47), it was confirmed that the total open area of this floor exceeds 1500 square feet. This open area is sufficient to prevent excessive differential pressure across this floor during a pool swell event.

The floor steel has been fabricated from rolled, ASTM A 36 steel sections, and the reinforced concrete slabs have been constructed using ASTM A 615, Grade 60, reinforcing steel and 5000 psi concrete.

All welding on the floor was done in accordance with AWS D1.1, and all bolted connections have been designed and erected in accordance with AISC Specification for Structural Joints using ASTM A 325 or ASTM A 490 bolts. All steel surfaces are corrosion protected with a carbozinc II coating applied to a blast-cleaned surface conforming to the requirements of SSPC-SP-10 (near white).

d. Floors at El. 147'-7" and 161'-10", Between the Drywell and the Shield Wall

Both floors consist of steel beams and grating with the beams framed between the drywell and the shield wall. Vertical support points are provided at the drywell and shield wall. Supports for horizontal forces are provided at the drywell wall and sliding connections are provided at the shield wall. The main structural function of these floors is to support pipe whip restraints for high energy lines. The frame and frame supports are constructed of standard steel shapes. The floor at El. 147'-7" supports a frame to which the recirculation pump motor hangers are attached. In addition, these floors support other electro-mechanical equipment, as well as snubbers, anchors, hangers, and the recirculation pump motor removal monorail. Both floors are designed to facilitate removal of the safety relief valves and, if necessary, the inboard containment isolation valves on the main steam line. The arrangement of the floor at El. 147'-7" is shown on Figure 3.8-67. The floor at 161'-10" is similar to this figure in design and construction.

e. Miscellaneous Floors Outside the Drywell (Concrete and Grating Platforms)

The miscellaneous floors consist of both reinforced concrete structures and steel framing with grating. Vertical supports for the floors are provided at the drywell and containment walls. Supports for horizontal forces are provided at the drywell wall, and sliding

connections are provided at the containment wall. The miscellaneous floors are designed to allow access to and provide support for various equipment and piping. These concrete floors and walls also provide shielding. The arrangement of the miscellaneous floors discussed above is shown on Figure 3.8-60. In order to address Humphrey Issue 1.7 (Grand Gulf Action Plan 47), it was confirmed that these floors have open areas in excess of 1500 square feet. These open areas prevent excessive pressure differentials across these floors during a pool swell transient.

3.8.3.2 Applicable Codes, Standards, and Specifications

The design of the drywell and other containment internal structures conforms to the applicable codes, standards, and specifications listed below, except where specifically stated otherwise. Where contradictions occur between codes, the more stringent shall apply.

3.8.3.2.1 Regulations

a. Code of Federal Regulations, Title 10: Energy Part 50, Licensing of Production and Utilization Facilities

b. Code of Federal Regulations (CFR), Title 10: Energy, Part 100, Reactor Site Criteria, (10 CFR Part 100, including Appendix A thereto, "Seismic and Geologic Siting Criteria for Nuclear Power Plants")

c. Occupational Safety and Health Administration (OSHA), Department of Labor, Occupational Safety and Health Standards, Title 29: Labor, Parts 15 and 19

3.8.3.2.2 Codes and Standards

3.8.3.2.2.1 Design Codes

a. American Institute of Steel Construction (AISC), Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings, 1969 edition

An exception to this code is as follows:

Table 1.17.5 - Minimum Size of Fillet Welds

The requirements of this table shall not apply where shown by stress analysis that a smaller weld size may be used.

b. American Concrete Institute (ACI), Building Code Requirement for Reinforced Concrete, ACI 318-71

Tables 3.8-45 and -46 compare load combinations and acceptance criteria required in ACI 349-76 and Regulatory Guide 1.142 with the commitments of subsections 3.8.3.3 and 3.8.3.4, which are based in part on ACI 318-71. The extensive comparisons of load factor combinations and design criteria demonstrate that although the Grand Gulf design of the containment internal structures does not fully comply with the requirements of ACI 349-76 and Regulatory Guide 1.142, the design is conservative with respect to these criteria.

c. American Welding Society (AWS), Code for Welding in Building Construction, AWS D1.1-72*

Exceptions taken to this code are as follows:

Paragraph 1.2 - Base Metal and Paragraph 8.2 - Base Metal

Paragraph 8.2.1 of AWS D1.1 contains a list of ASTM materials which may be used with prequalified welding procedure specifications. This code requires that welding procedure specifications be qualified for each material used for structural applications which are not included in the list. This AWS code requirement is not implemented for the following carbon steels used for structural applications: ASTM A 105; ASTM A 106, Gr. A, B, and C; ASTM A 366; ASTM A 446, Gr. A; ASTM A 516, Gr. 70; ASTM A 526; ASTM A 569; ASTM A 570, Gr. B, C, and D. The above listed materials have chemical compositions and physical properties similar to those carbon steels listed in paragraph 8.2.1 and have welding parameters that would be no different from those welding parameters contained in prequalified welding procedures specifications for the carbon steels listed in paragraph 8.2.1. Stainless steels (including ASTM A 167, Gr. 304; ASTM A 240, TP 304; and

* AWS D1.1-72 was used in the original construction of GGNS. Later editions that are adopted for use are specified in the component's/structure's specification or standard.

ASTM A 276, TP 304) are welded using welding procedure specifications meeting the requirements of the ASME Code, Section IX.

Paragraph 1.3 - Welding Processes

The gas-tungsten arc welding process, as used in the installation of certain miscellaneous structural steel, particularly stainless steel, use welding procedure specifications meeting the requirements of ASME Code, Section IX for both carbon and stainless steels.

Section 2 - Design of Welded Connections

Full penetration groove welds which cannot be back gouged or which cannot use backing strips due to joint configuration or accessibility are welded using Type E6010 electrodes for the root pass, accompanied by welding out the remainder of the weld with Type E7018 electrodes.

Table 2.7 - Minimum Fillet Weld Size for Prequalified Joints

The requirements of this table shall not apply where shown by stress analysis that a smaller weld size may be used.

Paragraph 3.6.4 - Weld Profiles

Prior to June 1, 1987, the requirement is changed to: "For buildings and tubular structures, undercut shall be no more than 1/32-inch (0.8 mm) deep in any direction for material 3/16 inch and greater in thickness."

As of June 1, 1987 the following requirements shall apply:

Paragraph 3.6 - Weld Profiles

Weld profiles acceptance criteria shall be as required by NCIG-01, Revision 2. NCIG-01 is invoked by a modification to AWS D1.1 paragraph 8.15 as discussed below.

Paragraph 4.9.2 - Electrode Storage And Issue

The requirement for E7018 (low hydrogen) electrodes is changed to: "E7018 electrodes that are not used within 14 hours...shall be redried before use."

Section 5 - Qualification and Paragraph 2.6 - Joint Qualification

Welding procedure specifications and welders qualified to ASME Code, Section IX requirements are permitted for structural steel applications in lieu of welding procedure specifications and welders meeting the requirements of AWS D1.1. Joint designs not prequalified to the requirements of AWS D1.1 are permitted using welding procedure specifications and welders qualified to ASME Code, Section IX requirements.

Paragraph 8.15 - Quality of Welds

All welds for Category I structural steel are visually inspected, and the inspection results are documented. Welds for Category I miscellaneous and embedded steel are visually inspected on a surveillance basis, and the surveillance inspection results are documented.

As of June 1, 1987 the following requirements shall apply:

Acceptance criteria for visual inspection of welds shall be per NCIG-01, Revision 2, paragraph 2.0. Inspectors shall be trained in accordance with the elements of NCIG-03, Revision H.

d. American Society of Mechanical Engineers, (ASME) Boiler and Pressure Vessel Code, Division I, Section III, subsection NE, 1974 edition, governs the design and fabrication of the drywell head, personnel airlocks, penetrations, and other steel elements serving pressure vessel functions.

e. American Society of Civil Engineers (ASCE), Paper 3269, Wind Forces on Structures.

f. American National Standards Institute (ANSI) B.31.1, 1973 Edition with addenda through Summer 1973.

g. American Concrete Institute, Criteria for Reinforced Concrete Nuclear Power Containment Structures, ACI Committee 349 Title No. 69-2, Journal of the American Concrete Institute, Jan. 1972.

Exceptions taken to this code are as follows:

1. Chapter 2, subsection 2.3.1: Definition of T_a

   The definition of T_a is changed to: Added thermal effects (over and above operating thermal effects) which may occur during a design accident and which correspond to the unfactored design pressure P. This temperature transient is also applied in conjunction with factored pressure loads, i.e., 1.25P and 1.5P.

2. Chapter 2, subsection 2.3.3: Capacity Reduction Factor

   This section is replaced by: The capacity reduction for reinforcing in the containment shell elements and foundation mat shall be $\varphi = 0.9$ for axial.

3. Chapter 2, subsection 2.2.1: Stress and Strain Limitations

   The first two sentences are replaced by: Under all U-loading combinations as defined in subsection 3.8.1.3 of this SAR, the reinforcement, the primary tensile reinforcement of a section, shall have a maximum predicted stress of not more than f_y.

4. Chapter 2, subsection 2.5.2: Reinforcement Anchorage

   In this section, replace the word "tension" with the phrase, "membrane tension or membrane plus flexural tension."

5. Chapter 2, subsection 2.5.3: Splices in Reinforcement

   In this section, replace the word "tension" with the phrase "membrane tension or membrane plus flexural tension."

6. Chapter 2, subsection 2.4.1: Shells

   The third paragraph is deleted.

Tables 3.8-45 and -46 compare load combinations and acceptance criteria required in ACI 349-76 and Regulatory Guide 1.142 with the commitments of subsections 3.8.3.3 and 3.8.3.4, which are based in part on ACI 349, Title 69-2. The extensive comparisons of load factor combinations

and design criteria demonstrate that although the Grand Gulf design of the containment internal structures does not fully comply with the requirements of ACI 349-76 and Regulatory Guide 1.142, the design is conservative with respect to these criteria.

h. American Society of Civil Engineers (ASCE), Manual of Engineering Practice No. 42 - Design of Structures to Resist Nuclear Weapons Effects.

i. For ductile concrete expansion anchors use Appendix B, Steel Embedments, to the American Concrete Institute, Code Requirements for Nuclear Safety-Related Concrete Structures, ACI 349-80 with the 1984 supplement. The following sections of Appendix B relating specifically to expansion anchors will be applicable; Sections B.7.1 through B.7.5, but the requirements of B.7.2 will be modified by the applicable requirements of NRC IE Bulletin 79-02.

j. Nuclear Construction Issues Group (NCIG), Visual Weld Acceptance Criteria for Structural Welding at Nuclear Plants; NCIG-01, Revision 2.

k. Nuclear Construction Issues Group (NCIG), Training Manual for Inspectors of Structural Welds at Nuclear Power Plants Using the Acceptance Criteria of NCIG-01; NCIG-03, Revision H.

3.8.3.2.2.2 Construction Codes of Practice

The following codes of practice, with indicated exceptions, will establish the standards of construction procedure:

a. ACI 301-72: Specifications for Structural Concrete for Buildings

Exceptions:

Chapter 1, General

Section 1.4, ASTM Standards: later issues of the standards cited may be used, when appropriate.

Section 1.5, Other Reference Standards and Recommended Practices: later issues than those cited may be used, when appropriate.

Chapter 4, Formwork

Section 4.3, Tolerances: The exceptions to the tolerances of Table 4.3.1 are itemized in Table 3.8-25.

Section 4.5, Removal of Forms: The following requirements shall apply instead of the requirements of subsection 4.5.4:

4.5.4 Forms for columns, walls, sides of beams, slabs and girders, and other parts not supporting the weight of the concrete shall be removed as soon as practicable in order to avoid delay in curing and repairing surface imperfections. Wood forms or insulated steel forms for members three feet or greater in thickness shall be stripped or loosened within 24 hours. Forms shall be left in place for seven days when removal of forms after 24 hours exposes surfaces of mass concrete to an ambient temperature below 50 F.

Chapter 5, Reinforcement

Section 5.4, Fabricating and Placing Tolerances: The following requirements shall apply instead of the requirements specified in subsection 5.4.2 and its subsections:

5.4.2 Center to center bar spacing: plus or minus two inches, except that the minimum clear distance specified in ACI 318-71, Chapter 7, Section 7.4 and the requirements of Section 7.3, ACI 318-71, shall be maintained (see Table 3.8-25 for exceptions to Section 7.3) and the total number of bars shall be maintained within the tolerances given below:

5.4.2 Center to center bar spacing: plus or minus two inches, except that the minimum clear distance specified in ACI 318-71, Chapter 7, Section 7.4 and the requirements of Section 7.3, ACI 318-71, shall be maintained (see Table 3.8-25 for exceptions to Section 7.3) and the total number of bars shall be maintained within the tolerances given below:

Number of Bar	Tolerance (Percent)
Greater than required	10
Less than required	5

Chapter 8, Placing

Section 8.2, Conveying: The following requirements shall apply in place of the requirements specified in subsection 8.2.2.4:

8.2.2.4 Pumping or pneumatic conveying equipment shall be of a suitable kind with adequate pumping capacity. Pneumatic placement shall be controlled so that segregation is not apparent in discharged concrete. Concrete shall not be conveyed through pipe made of aluminum or aluminum alloy.

Chapter 14, Massive Concrete

Temperature shall be controlled as specified below for all concrete to be placed when the least dimension is greater than 3 feet.

Section 14.4, Placing: The following requirements shall apply instead of subsections 14.4.1, 14.4.2, and 14.4.3:

14.4.1 Slump: The slump working limit of the structural concrete shall be as found in the project specifications, with an inadvertency margin (tolerance) of plus two inches for individual batches. The inadvertency margin as defined in the project specifications pertains to occasional batches only.

14.4.2 Placing Temperature: The maximum temperature of concrete when deposited shall be 75 F in place of 70 F. Concrete greater than 6 feet in the least dimension shall have a placing temperature as near 65 F as can be obtained.

14.4.3 Depth of Layers: Concrete shall be placed in layers approximately 18 inches thick and not more than 24 inches thick. Vibrator heads shall extend into the previously placed layer.

Section 14.5, Curing and Protection: The following requirements shall apply in place of subsections 14.5.1, 14.5.2 and 14.5.3:

14.5.1 Minimum Curing Period: In place of a minimum curing period of 2 weeks, the minimum curing period shall be 7 days for heavily reinforced massive sections, as specified in ACI 308-71, Section 3.3.3, provided the temperature of the surrounding air is above 40 F and is expected to remain above this temperature for the entire curing period.

14.5.2 When the temperature of the surrounding air is 40 F or below or is expected to drop below 40 F in the next 12 hours, adequate means shall be provided to maintain the temperature of the surrounding air between 50 and 70 F for the remainder of the seven-day curing period.

14.5.3 Moist Curing: As an alternate to keeping the forms and exposed concrete continuously wet for at least the first 48 hours after placing, and also whenever the surrounding air temperature is above 90 F, the provisions of Section 12.2, Preservation of Moisture, ACI 301-72, shall apply.

b. ACI 306-66, Recommended Practice for Cold Weather Concreting

Addition:

Requirements for winter concreting shall conform to ACI Standard 306-66 with the added requirement that curing periods shall be as herein stated under “Exceptions” for Section 14.5 (above) of ACI Standard 301-72.

c. ACI 318-71, Building Code Requirements for Reinforced Concrete

Exceptions:

Chapter 4, Concrete Quality

Subsection 4.2.5 and Table 4.2.5: Concrete air content shall be within 2 to 6 percent by volume for all sizes of coarse aggregates. However, when air content falls below 2 percent, the frequency of correlation testing shall be increased to each 100 cubic yards until test results fall within 2 to 6 percent by volume. If air content is not brought within limits, as determined by the Contractor, the concrete placement shall cease until the problem is corrected.

Subsection 4.3.1: Where applicable, the following requirement shall be used instead of the last sentence: Each strength test result shall be the average of two cylinders from the same sample tested at 28 or 91 days.

Chapter 5, Mixing and Placing Concrete

Section 5.4, Depositing: The following requirements shall apply in place of subsection 5.4.4:

5.4.4 On the bottom of formed beams and slabs, where conditions make consolidation difficult or where reinforcement is congested, batches of grout containing the same proportions of cement and sand as those used in the concrete shall first be deposited in the forms to a depth of approximately 1/4 inch.

Chapter 6, Formwork, Embedded Pipes and Construction Joints

Section 6.3, Conduits and Pipes Embedded in Concrete: The following requirements shall apply in place of subsections 6.3.2.2, 6.3.2.3, and 6.3.2.4:

6.3.2.2 and 6.3.2.3 All pipes carrying liquid, gas, or vapor at temperatures exceeding 150 F or pressures exceeding 200 psia, or both, shall not be embedded in structural concrete. If an embedment is necessary, the pipes shall be provided with suitably designed sleeves.

6.3.2.4 All embedded piping and fittings shall be tested to the specific provisions of ASME Code, Section III, USAS B31.7, ANSI B31.1, and/or local codes, as may be applicable, that govern the system of which the embedded piping is a part. Embedded gravity drainage piping, which is not covered by any of the above codes, shall be hydrostatically tested to 15 psig in accordance with ANSI B31.1. Floor and wall penetrations are not considered embedded piping and therefore are not hydrostatically tested. Where no other code governs a particular pipe embedded in structural concrete, the requirements of ACI 318-71, subsection 6.3.2.4, shall govern.

Subsection 6.4.1: The following requirements shall apply instead of 6.4.1: Joints not indicated on the plans shall be so made and located as not to impair significantly the strength of the structure. Both horizontal and vertical construction joints shall be thoroughly cleaned of all coatings, debris, and other foreign materials. All concrete surfaces to receive new concrete shall be free of surface water, but shall be thoroughly dampened to prevent absorption of water from the new concrete. All horizontal surfaces shall be covered, immediately before the concrete is placed, with a nominal 1/4 inch of grout thoroughly broomed into the surface. For congested areas where brooming is impossible, the grout shall be forced ahead of the concrete. Grout used in construction joints shall have the same sand-cement ratio as the concrete with which it is to be used.

Chapter 7, Details of Reinforcement, Section 7.3, Placing Reinforcement: The requirements that shall apply instead of the tolerances of paragraph 7.3.2 are tabulated in Table 3.8-25.

Chapter 11, Shear and Torsion

Section 11.1.6 - This section is not applicable for tangential shear design.

Chapter 19, Shells and Folded Plate Members

Subsections 19.2.2, 19.3.2, 19.5.1, 19.5.2, 19.5.4, and the last sentence of 19.5.3 are not applicable for containment internals design.

d. ACI 347-68 Recommended Practice for Concrete Formwork

e. ASTM C 94-72 Ready-Mixed Concrete

f. American Concrete Institute, Criteria for Reinforced Concrete Nuclear Power Containment Structures, ACI Committee 349, Title No. 69-2, Journal of the American Concrete Institute, January 1972, for shear and torsion only

g. ACI 305-72 Recommended Practice for Hot Weather Concreting

Exceptions:

Chapter 2, Section 2.2.1: For massive concrete, the placing temperature shall be as near 65 F as jobsite conditions permit, but the maximum placing temperature shall be 75 F.

Chapter 4: Curing shall be in accordance with the provisions of ACI 301-72, Chapters 12 and 14, Section 14.5, with the exceptions to Section 14.5 listed above.

h. ACI 211.1-70, Recommended Practice for Selecting Proportions for Concrete

i. ACI 304-73, Recommended Practice for Measuring, Mixing, Transporting, and Placing Concrete

j. ACI 315-65, Manual of Standard Practice for Detailing Reinforced Concrete Structures

k. ASME, Boiler and Pressure Vessel Code, Section VIII

Subsection B, Requirements Pertaining to Methods of Fabrication of Pressure Vessels

Part UW, Requirements for Pressure Vessels Fabricated by Welding

l. AWS D1.1, Code for Welding in Building Construction

m. ASME Boiler and Pressure Vessel Code, Section III, Division 2, 1975 Edition (ACI Standard 359-74)

This code is referenced only for the purpose of comparing the load and loading combination criteria of Article CC-3000 to the loads and loading combinations listed in subsections 3.8.3.3.1 and 3.8.3.3.2.1. The drywell structure is designed in accordance with the design codes listed in items a through l above.

In every instance, the construction procedure will equal or exceed the recommendations set forth in the foregoing publications except as noted above. The extent to which each detailed process will exceed standard requirements cannot be abstracted without incorporating all applicable job specifications and anticipating hypothetical construction problems and conditions. In general, wherever the applicable code specifies a minimum and an ideal criteria, the ideal will be incorporated into specifications, except as noted above.

3.8.3.2.3 General Design Criteria, Regulatory Guides, Industry Standards, and Topical Reports

a. 10 CFR 50, Appendix A, General Design Criteria for Nuclear Power Plants (conformance is discussed in Section 3.1)

   Criteria Nos. 2, 16, 51, and 53.

b. NRC Regulatory Guides (exceptions are discussed in Appendix 3A)

   Regulatory Guide 1.10, Mechanical (Cadweld) Splices in Reinforcing Bars of Seismic Category I Concrete Structures

   Regulatory Guide 1.15, Testing Reinforcing Bars for Seismic Category I Concrete Structures

   Regulatory Guide 1.18, Structural Acceptance Test for Concrete Primary Reactor Containments

   Regulatory Guide 1.19, Nondestructive Examination of Primary Containment Liner Welds

   Regulatory Guide 1.60, Design Response Spectra for Seismic Design of Nuclear Power Plants

   Regulatory Guide 1.61, Damping Values for Seismic Design of Nuclear Power Plants

Regulatory Guide 1.76, Design Basis Tornado for Nuclear Power Plants

Regulatory Guide 1.102, Flood Protection for Nuclear Power Plants

Regulatory Guide 1.115, Protection Against Low-Trajectory Turbine Missiles

Regulatory Guide 1.122, Development of Floor Design Response Spectra for Seismic Design of Floor Supported Equipment or Components

c. Industry Standards

Nationally recognized industry standards, such as those published by the American Society for Testing and Materials (ASTM) and the American National Standards Institute (ANSI), are used to describe the material properties, testing procedures, fabrication methods, and construction methods.

d. Bechtel Power Corporation Topical Reports

BN-TOP-1, Rev. 1, Testing Criteria for Integrated Leak Rate Testing of Primary Containment Structures for Nuclear Power Plants

BN-TOP-2, Rev. 2, Design for Pipe Rupture Effects

BC-TOP-1, Rev. 1, Containment Building Liner Plate Design

BC-TOP-3A, Rev. 3, Tornado and Extreme Wind Design Criteria for Nuclear Power Plants

BC-TOP-4, Rev. 1, Seismic Analysis of Structures and Equipment for Nuclear Power Plants, including Addendum 1

BC-TOP-9A, Rev. 2, Design of Structures for Missile Impact

3.8.3.2.4 Specifications

Specifications are prepared to cover the areas related to the design and construction of the containment internal structures. These specifications are prepared by Bechtel Power Corporation specifically for these structures. The specifications emphasize important points of the industry standards for these structures

and reduce options such as would otherwise be permitted by industry standards. Unless specifically noted otherwise, these specifications do not deviate from the applicable industry standards and, therefore, are not included in the SAR. These specifications cover the following areas:

a. Furnishing and delivery of concrete.

b. Purchasing, forming, placing, and curing of concrete.

c. Furnishing, detailing, fabricating, delivery, and placing of reinforcing steel.

d. Furnishing, delivery, and installation of exothermic splicing.

e. Furnishing, delivery, and erection of structural steel.

f. Furnishing, delivery, and erection of steel liner plate for water filled cavities (refueling pool, etc.).

g. Furnishing, delivery, and installation of refueling bellows, drywell head, drywell penetrations, and drywell air locks.

3.8.3.3 Loads and Loads Combinations

3.8.3.3.1 Load Definitions

Loads, definition of terms, and nomenclature of all the major loads to be encountered and/or to be postulated in a nuclear power plant are listed below. All the loads listed, however, are not necessarily applicable to all the internal structures. Loads and the applicable load combinations for which each structure is designed depend on the conditions to which the particular structure is subjected.

The containment internal structures are presently designed in accordance with the loads described in Appendices 6A and 6D. These are defined herein, and the effects of these loads have been considered in the design verification of all internal structures as applicable. The loads within the loading combinations are combined using the absolute sum technique. (Those loads which are defined as reversible in algebraic sign are

combined in such a way as to produce the maximum resultant stresses in the structure. All other loads are combined in accordance with their direction of application to the structure.)

3.8.3.3.1.1 Normal Loads

a. D - Dead load of the structure and equipment plus any other permanent loads, including vertical and lateral pressures of liquids.

b. L - Live loads, including any movable equipment loads and other loads which vary in intensity and occurrence, such as forces exerted by the lateral pressure of soil.

c. T_o - Thermal effects during normal operating, startup, or shutdown conditions, including liner plate expansion, equipment and pipe reactions, and thermal gradients, based on the most critical transient or steady state thermal gradient.

d. Z - Pressure difference between the interior and exterior of the containment, considering both interior pressure changes because of heating or cooling, and exterior atmospheric pressure variations. This particular term is considered to be negligible for the containment internal structures and is therefore considered to be zero for all load combinations containing this term.

e. Safety Relief Valve Loads

The containment internal structures experience oscillatory pressure loadings resulting from discharge of safety relief valves into the suppression pool. The development of these loads is in accordance with the methods described in Appendices 6A and 6D. The number and combinations of valves that will open during a reactor vessel pressure transient are as follows:

1. C_1 - Design pressure load on the suppression pool boundary resulting from discharge of one safety relief valve into the suppression pool (positive or negative). Positive pressures for the first actuation and subsequent actuation of this loading to be sustained by the internal structures are shown in Figures 3.8-16a through 3.8-25b.
2. C_2 - Design pressure load on the suppression pool boundary resulting from discharge of two adjacent safety relief valves into the suppression pool (positive or negative). Positive pressures for this loading to be sustained by the internal structures are shown in Figures 3.8-26a through 3.8-30a.
3. C_{11} - Containment design pressure load on the suppression pool boundary resulting from discharge of 11 safety relief valves into the suppression pool (positive or negative). Positive pressures for this loading to be sustained by the internal structures are shown in Figures 3.8-31a through 3.8-35b.
4. C_A - Containment design pressure load resulting from automatic depressurization system (ADS) discharge of eight safety relief valves into the suppression pool (positive or negative). Positive pressures for this loading to be sustained by the internal structures are shown in Figures 3.8-36a through 3.8-40b.
5. C' - Containment design pressure load resulting from discharge of all 20 safety relief valves into the suppression pool (positive or negative). Positive pressures for this loading to be sustained by the internal structures are shown in Figures 3.8-41a through 3.8-45b.

3.8.3.3.1.2 Preoperational Testing Loads

a. P_T - Tests loads are loads which are applied during the structural integrity test. The drywell test pressure is 30 psig.

b. T_t - Thermal effects during the structural integrity test.

3.8.3.3.1.3 Severe Environmental Loads

a. E - Operating Basis Earthquake (OBE) as defined in Section 3.7.

b. W - Loads generated by the design wind specified for the plant site as defined in Section 3.3. This term is negligible for the internal structures and is taken as zero for all load combinations containing this term.

3.8.3.3.1.4 Extreme Environmenta1 Loads

a. E' - Safe shutdown earthquake as defined in Section 3.7.

b. W' - Loads generated by the design tornado specified for the plant site as defined in Section 3.3. This term is negligible for the internal structures and is considered to be zero for all load combinations containing this term.

3.8.3.3.1.5 Abnormal Plant Loads

a. R - Local force, pressure, or reaction on an internal structure caused by rupture of any one pipe as defined in Section 3.6.

b. P_{AP} - Asymmetric pressure due to recirculation suction line, recirculation return line, or feed water line outlet break.

c. P - Pressure equivalent static load within or across a compartment generated by the postulated break and including an appropriate dynamic load factor to account for the dynamic nature of the load.

d. Design Basis Accident (DBA)

The containment internals are designed to sustain the following loadings as a result of a loss of coolant accident (LOCA).

1. P_{CD} - Containment design loss of coolant accident pressure load during a design basis accident as presented in subsection 6.2.1 of this FSAR. The containment design pressure is 15 psig.

2. T_a - Added thermal effects (over and above operating thermal effects) which may occur during a design accident and which correspond to the unfactored design pressure PCD. Accident thermal gradients for the internal structures are shown in Figures 3.8-68 through 3.8-70.

3. P_{DD} - Peak drywell design pressure load during a design basis accident. The drywell design pressure is 30 psig.

4. P_{ND} - Peak drywell design negative pressure load during a design basis accident. Pressure diagrams for this load are shown in Figure 3.8-46.

5. P_{CDC} - Containment and drywell design pressure, during a design basis accident (DBA), concurrent with chugging (this loading is postulated in the interval from t = 30 sec. to t = 100 sec. during the DBA). (See Figure 3.8-48a.)

6. Suppression pool design basis accident (DBA) loads. Immediately following a design basis accident, the drywell wall, weir wall, and structures located above the suppression pool will experience various dynamic loadings as a result of the DBA vent clearing process into the suppression pool. The following are the loadings resulting from the DBA phenomena:

   R_B - Pool swell bubble pressure within the suppression pool (axisymmetric). Pressure diagrams for this load are shown on Figure 3.8-47 and Appendix 6D.

R_{BA} - Asymmetric pool swell bubble pressure on the pool walls. This load has the same pressure ordinate as R_B above and is applied to the suppression pool walls from 0 to 180 degrees in plan.

R_D - Axisymmetric containment wetwell differential pressure due to flow restriction of the hydraulic control unit (HCU) floor to pool swell concurrent with a differential drywell pressure. Pressure diagrams for this load are shown on Figure 3.8-48 and Appendix 6D.

R_P - Impact and drag loads on structures located above the suppression pool due to upward pool swell. Loads on these internal structures are shown in Appendix 6D.

R_F - Fallback loads of cascading water onto structures above the suppression pool immediately following pool swell. Loads on these internal structures are shown in Appendix 6D.

R_C - An oscillatory dynamic loading (chugging) in the top vent and weir annulus or on the suppression pool boundary due to steam condensation inside the top vent or at the top vent exit during periods of low steam mass flow in the top vent only following a LOCA. Pressure distributions for this loading are shown in Appendix 6D.

R_{CO} - oscillatory dynamic loading (condensation on the suppression pool boundary which occurs due to steam condensation at the vent exits during the period of high steam mass flow through the vents following a LOCA. Pressure distributions for this loading are shown in Appendix 6D.

R_W - Inward load through the vents applied to the weir wall due to negative drywell pressure differential. Pressure distributions for this loading are shown in Appendix 6D.

e. Small Break Accident (SBA). The small break accident is deemed to be an accident during which a pipe rupture is not large enough to automatically depressurize the reactor, as presented in subsection 6.2.1 of this FSAR. The internal structures are designed to sustain the following loadings during an SBA:

1. P_{DS} - Containment and drywell design pressure load during SBA. Pressure diagrams for this load are shown in Figure 3.8-49.

2. P_{DN} - Drywell design negative pressure load associated with chugging (water oscillation back and forth in the top row of drywell vents) during SBA. Pressure diagrams for this load are shown in Figure 3.8-50.

3. T_a - Added thermal effects (over and above operating thermal effects) which may occur during a small break accident and which correspond to the unfactored design pressure, P_{DS} Thermal gradients for the internal structures are shown in Figures 3.8-71 and 3.8-72.

4. R_C - See definition in subsection 3.8.3.3.1.5.d above.

f. Intermediate break accident (IBA). The intermediate break accident is deemed to be an accident during which a pipe rupture, albeit smaller than a DBA, is of sufficient size to automatically depressurize the primary system due to loss of fluid and/or automatic initiation of the ECCS systems, as presented in subsection 6.2.1 of this SAR. The internal structures are designed to sustain the following loadings during an IBA.

1. P_{CI} - Containment and drywell design pressure load during IBA. Pressure diagrams for this load are shown in Figure 3.8-52.

2. T_a - Added thermal effects (over and above operating thermal effects) which correspond to the unfactored design pressure, P_{CI}.

3. R_C - See definition in subsection 3.8.3.3.1.5.d above.

4. R_{CO} - See definition in subsection 3.8.3.3.1.5.d above.

3.8.3.3.1.6 Post-Abnormal Plant Loads

For fuel recovery subsequent to a design basis accident, the containment is designed for flooding to a datum level of 55'-6" above the top of the active fuel in the reactor core.

B - Hydrostatic load due to post-LOCA flooding of the containment to a level 115'-10" above the top of the foundation mat.

3.8.3.3.1.7 Site Related Loads

A - Hydrostatic load due to probable maximum flood as defined in Section 3.4 of this SAR. This term is negligible for the internal structures and is considered to be zero for all load combinations containing this term.

Other loads associated with activities near the site are discussed in Section 2.2, and loads from postulated missiles are discussed in Section 3.5, both of this SAR.

3.8.3.3.2 Load Combinations

Notations used in this section are as follows:

S - Required capacity of the section such that the stress limitations of Section 8.0, ACI 318-71, are satisfied, with the exception of the permissible tensile stress for reinforcement, which shall be a maximum of 0.5 f_y and not more than 30,000 psi.

U - Required ultimate load capacity as defined in ACI 318-71.

A comparison of the GGNS load combinations and acceptance criteria with the requirements of ACI 349-76 and Regulatory Guide 1.142 is provided in Tables 3.8-45 and -46.

3.8.3.3.2.1 Drywell

During the evaluation of the loading combinations, a time-dependency of loads was considered. This time-dependency is necessary due to the duration of the accidents and the chronology of the loading events. An example of this time-dependency occurs during a DBA which is defined in Appendix 6D. Although the seismic event is considered for design to occur coincidently with the DBA, a combination of the seismic loads with the maximum negative pressure load (P_{ND}) cannot occur since the negative pressure load occurs several minutes after the initiation of the DBA. The seismic event was considered to occur at the beginning of the accident.

a. Normal Operating Conditions

$U = 1.4D + 1.7L + 1.0T_o + 1.0Z$

$S = 1.0D + 1.0L + 1.0T_o + 1.0Z$

$S = 1.0D + 1.0L = 1.0T_o + 1.0C_{11}\ (\pm)$

$S = 1.0D + 1.0L + 1.0T_o + 1.0C'\ (\pm)$

$S = 1.0D + 1.0L + 1.0T_o + 1.0C_A(\pm)$

$S = 1.0D + 1.0L + 1.0T_o + 1.0C_2(\pm)$

$S = 1.0D + 1.0L + 1.0T_o + 1.0C_1\ (\pm)$

$U = 1.4D + 1.7L + 1.0T_o + 1.7C_{11}\ (\pm)$

$U = 1.4D + 1.7L + 1.0T_o + 1.7C_1\ (\pm)$

$U = 1.4D + 1.7L + 1.0T_o + 1.7C'\ (\pm)$

$U = 1.4D + 1.7L + 1.0T_o + 1.7C_2\ (\pm)$

$U = 1.4D + 1.7L + 1.0T_o + 1.7C_A\ (\pm)$

b. Normal Operating and Severe Environmental Conditions

$U = 1.0D + 1.0L + 1.0T_o + 1.0C'(\pm) + 1.25E$

$U = 1.0D + 1.0L + 1.0T_o + 1.0C_{11}\ (\pm) + 1.25E$

$U = 1.0D + 1.0L + 1.0T_o + 1.0C_2\ (\pm) + 1.25E$

$U = 1.0D + 1.0L + 1.0T_o + 1.0C_1\ (\pm) + 1.25E$

$U = 1.0D + 1.0L + 1.0T_o + 1.0C_A\ (\pm) + 1.25E$

$U = 1.0D + 1.25L + 1.25(E \text{ or } W) + 1.0T_o + 1.0Z$

c. Normal Operating and Extreme Environmental Conditions

$U = 1.0D + 1.0L + 1.0T_o + 1.0A + 1.0W'$

$U = 1.0D + 1.0L + 1.0T_o + 1.0A + 1.0Z$

$U = 1.0D + 1.0L + 1.0T_o + 1.0C'\ (\pm) + 1.0E'$

$U = 1.0D + 1.0L + 1.0T_o + 1.0C_{11}\ (\pm) + 1.0E'$

$U = 1.0D + 1.0L + 1.0T_o + 1.0C_2\ (\pm) + 1.0E'$

$U = 1.0D + 1.0L + 1.0T_o + 1.0C_1\ (\pm) + 1.0E'$

$U = 1.0D + 1.0L + 1.0T_o + 1.0C_A\ (\pm) + 1.0E'$

d. Normal Operating and Abnormal Operating Conditions

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.50P_{CD} + 1.0R$

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.50(R_B \text{ or } R_{BA}) + 1.50P_{DD} + 1.25C_1\ (\pm) + 1.0R$

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.25C_A\ (\pm) + 1.5(P_{DS} \text{ or } P_{DN}) + 1.5R_C$

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.50P_{ND} + 1.0R$

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.50(R_D + R_P + R_F + R_{CO}) + 1.25C_1\ (\pm) + 1.0R$

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.50R_C + 1.50P_{CDC} + 1.0R$

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.25C_2\ (\pm) + 1.5P_{CI} + 1.5R_{CO}$

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.25C_A\ (\pm) + 1.50P_{CI} + 1.50R_C$

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.25C_2\ (\pm) + 1.5(P_{DS}\ \text{or}\ P_{DN}) + 1.5R_C$

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.50(P_{DS}\ \text{or}\ P_{DN}) + 1.25C_1\ (\pm) + 1.5R_C$

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.25C_1\ (\pm) + 1.5P_{CI} + 1.5RC_O$

e. Normal Operating, Abnormal Operating, and Severe Environmental Conditions

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.0R + 1.25P_{CD} + 1.25(E\ \text{or}\ W)$

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.25(R_B\ \text{or}\ R_{BA}) + 1.25P_{DD} + 1.0C_1\ (\pm) + 1.25E + 1.0R$

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.0C_A\ (\pm) + 1.25\ R_C + 1.25(P_{DS}\ \text{or}\ P_{DN}) + 1.25E$

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.25P_{ND} + 1.25E + 1.0R$

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.25(R_D + R_P + R_F + R_{CO}) + 1.0C_1\ (\pm) + 1.25E$

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.0C_2\ (\pm) + 1.25(P_{DS}\ \text{or}\ P_{DN}) + 1.25E + 1.25R_C$

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.0C_A\ (\pm) + 1.25P_{CI} + 1.25\ R_C + 1.25E$

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.0C_2\ (\pm) + 1.25P_{CI} + 1.25\ R_{CO} + 1.25E$

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.25\ R_C + 1.25\ P_{CDC} + 1.25(E\ \text{or}\ W) + 1.0R$

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.25(P_{DS}\ \text{or}\ P_{DN}) + 1.0C_1\ (\pm) + 1.25E + 1.25R_C$

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.0C_1\ (\pm) + 1.25\ P_{CI} + 1.25\ R_{CO} + 1.25E$

f. Normal Operating, Abnormal Operating, and Extreme Environmental Loads

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.0P_{CD} + 1.0R + 1.0E'$

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.0(R_B \text{ or } R_{BA}) + 1.0P_{DD} + 1.0C_1\ (\pm) + 1.0R + 1.0E'$

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.0R_C + 1.0P_{CDC} + 1.0E' + 1.0R$

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.0P_{ND} + 1.0R + 1.0E'$

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.0(R_D + R_P + R_F + R_{CO}) + 1.0C_1\ (\pm) + 1.0E'$

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.0C_A\ (\pm) + 1.0R_C + 1.0(P_{DS} + P_{DN}) + 1.0E'$

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.0P_{CI} + 1.0R_C + 1.0C_A\ (\pm) + 1.0E'$

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.0(P_{DS} \text{ or } P_{DN}) + 1.0R_C + 1.0C_1\ (\pm) + 1.0E'$

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.0C_2\ (\pm) + 1.0R_C + 1.0(P_{DS} + P_{DN}) + 1.0E'$

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.0P_{CI} + 1.0R_{CO} + 1.0C_1\ (\pm) + 1.0E'$

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.0P_{CI} + 1.0R_{CO} + 1.0C_2\ (\pm) + 1.0E'$

g. Structural Pressure Test and Post-LOCA Flooding Conditions

Structural Pressure Test Conditions:

$S = 1.0D + 1.0L + 1.0\ P_T + 1.0T_t$

Post-LOCA Flooding:

$$U = 1.0D + 1.0L + 1.0B + 1.0E$$

h. Values of water weight, D, P, P_{AP}, P_{CD}, P_{DD}, P_{ND}, P_{DS}, P_{DN}, P_{CI}, and T_A shown in a, b, c, d, e, f, and g above are time dependent and hence the values used in the above equations correspond to those occurring at a time which gives the most adverse loading combination. See the discussions and definitions in subsection 3.8.3.3.1.

i. Extent to which the load and load combination criteria comply with Article CC-3000 of the ASME Code, Section III, Division 2.

This code has been referenced only for the purpose of comparing the load and loading combination criteria of Article CC-3000 to the loads and loading combinations listed in subsections 3.8.3.3.1 and 3.8.3.3.2.1. The discussion of this comparison is required by Regulatory Guide 1.70. The drywell structure is designed in accordance with the design codes listed in items a through l of subsection 3.8.3.2.2.

All loads specified in Section CC-3220 have been appropriately incorporated into the drywell design. However, the nomenclature which was used for this design differs from that specified in Section CC-3221. SAR subsection 3.8.3.3.1 adequately defines all loading conditions which apply to the drywell.

There are several differences between the loading combinations listed in subsection 3.8.3.3.2 and those listed on Table CC-3230-1 of the ASME Code. The differences are listed below:

1. The ASME Code Table CC-3230-1 lists unfactored severe environmental loads as service loads. This project did not consider this type of loading combination.

2. 2. Code Table CC-3230-1 lists under factored loads a load case which reads:

$$U = 1.0D + 1.3L + 1.0T_o + 1.5E + 1.0R_o + 1.0P_v$$

This equation was used except 1.25L was used in place of 1.3L and 1.25E was used in place of 1.5E.

3. Code Table CC-3230-1 lists under factored loads a load case for abnormal/severe environmental loading which reads:

   $U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.0E + 1.0W$

   This combination was used, except E or W was considered and not the summation of 1.0E + 1.0W.

   For all other load combinations listed on Table CC-3230-1, the project either meets or surpasses the intent of the given loading combinations.

j. Extent to which the loading and loading combinations for the portions of the drywell comply with Article NE-3000 of Subsection NE of the ASME Code.

   The loads and loading combinations used to design the metal portion of the lower drywell wall are identical to those loading combinations listed at the beginning of subsection 3.8.3.3.2. The loading combination procedure delineated by Article NE-3000 of the ASME Code and by NRC Regulatory Guide 1.57 were not employed.

3.8.3.3.2.2 Reactor Pressure Vessel (RPV) Pedestal andWeir Wall

a. Normal Operating Conditions

   $U = 1.4D + 1.7L$

   $U = 1.4D + 1.7L + 1.4T_o$

   $U = 0.75(1.4D + 1.7L + 1.4T_o)$

   $U = 1.4D + 1.7L + 1.4T_o + 1.7C_1$ (±)

   $U = 1.4D + 1.7L + 1.4T_o + 1.7C_2$ (±)

   $U = 1.4D + 1.7L + 1.4T_o + 1.7C_A$ (±)

   $U = 1.4D + 1.7L + 1.4T_o + 1.7C_{11}$ (±)

   $U = 1.4D + 1.7L + 1.4T_o + 1.7C'$ (±)

b. Normal Operating and Severe Environmental Conditions

$U = 0.75(1.4D + 1.7L + 1.4T_o + 1.9E)$

$U = 1.4D + 1.7L + 1.4T_o + 1.25E + 1.7C_1$ (±)

$U = 1.4D + 1.7L + 1.4T_o + 1.25E + 1.7C_2$ (±)

$U = 1.4D + 1.7L + 1.4T_o + 1.25E + 1.7C_A$ (±)

$U = 1.4D + 1.7L + 1.4T_o + 1.25E + 1.7C_{11}$ (±)

$U = 1.4D + 1.7L + 1.4T_o + 1.25E + 1.7C'$ (±)

c. Normal Operating and Extreme Environmental Conditions

$U = 1.4D + 1.7L + 1.4T_o + 1.0E' + 1.0C_1$ (±)

$U = 1.4D + 1.7L + 1.4T_o + 1.0E' + 1.0C_2$ (±)

$U = 1.4D + 1.7L + 1.4T_o + 1.0E' + 1.0C_A$ (±)

$U = 1.4D + 1.7L + 1.4T_o + 1.0E' + 1.0C_{11}$ (±)

$U = 1.4D + 1.7L + 1.4T_o + 1.0E' + 1.0C'$ (±)

d. Normal Operating and Abnormal Operating Conditions

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.5P_{DD} + 1.5R_B + 1.0R + 1.25C_1$ (±)

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.50(R_D + R_P + R_F + R_{CO}) + 1.0R + 1.25C_1$ (±)

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.5P_{ND} + 1.0R + 1.0R_W$

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.5P_{CI} + 1.25C_1$ (±)+ $1.5(R_{CO}$ or $R_C)$

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.5P_{CI} + 1.25C_2$ (±) + $1.5(R_{CO}$ or $R_C)$

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.5P_{CI} + 1.25C_A$ (±) + $1.5R_C$

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.5(P_{DS}$ or $P_{DN})+ 1.25C_1$ (±) + $1.5R_C$

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.5(P_{DS} \text{ or } P_{DN}) + 1.25C_2\ (\pm) + 1.5R_C$

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.5(P_{DS} \text{ or } P_{DN}) + 1.25C_A\ (\pm) + 1.5R_C$

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.5P_{CDC} + 1.5R_C + 1.0R$

e. Normal Operating, Abnormal Operating, and Severe Environmental Conditions

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.25P_{DD} + 1.25E + 1.0C_1\ (\pm) + 1.0R$

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.25P_{ND} + 1.25E + 1.0R + 1.0R_W$

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.25(R_D + R_{CO}) + 1.25E + 1.0C_1\ (\pm)$

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.25P_{CI} + 1.25E + 1.0C_1\ (\pm) + 1.25\ (R_{CO} \text{ or } R_C)$

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.25P_{CI} + 1.25E + 1.0C_2\ (\pm) 1.25\ (R_{CO} \text{ or } R_C)$

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.25P_{CI} + 1.25E + 1.0C_A\ (\pm) + 1.25R_C$

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.25(P_{DS} \text{ or } P_{DN}) + 1.25E + 1.25R_C + 1.0C_1\ (\pm)$

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.25(P_{DS} \text{ or } P_{DN}) + 1.25E + 1.25R_C + 1.0C_2(\pm)$

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.25(P_{DS} \text{ or } P_{DN}) + 1.25E + 1.25R_C + 1.0C_A\ (\pm)$

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.25P_{CDC} + 1.25E + 1.25R_C + 1.0R$

f. Normal Operating, Abnormal Operating, and Extreme Environmental Loads

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.0P_{DD} + 1.0E' + 1.0R + 1.0C_1$ (±)

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.0P_{ND} + 1.0E' + 1.0R + 1.0R_W$

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.0(R_D + R_{CO}) + 1.0E' + 1.0C_1$ (±)

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.0P_{CI} + 1.0E' + 1.0C_1$ (±) $+ 1.0(R_{CO}$ or $R_C)$

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.0P_{CI} + 1.0E' + 1.0C_2$ (±) $+ 1.0(R_{CO}$ or $R_C)$

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.0P_{CI} + 1.0E' + 1.0C_A$ (±) $+ 1.0R_C$

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.0(P_{DS}$ or $P_{DN}) + 1.0C_1$ (±) $+ 1.0E' + 1.0R_C$

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.0(P_{DS}$ or $P_{DN}) + 1.0C_2$ (±) $+ 1.0E' + 1.0R_C$

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.0(P_{DS}$ or $P_{DN}) + 1.0C_A$ (±) $+ 1.0E' + 1.0R_C$

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.0\ P_{CDC} + 1.0R_C + 1.0R$

3.8.3.3.2.3 Other Internal Structures

For other internal structures, the effect of the safety relief valve loads and loss-of-coolant accident loads is considered in accordance with the loading combinations in subsections 3.8.3.3.2.3.1 and 3.8.3.3.2.3.2.

3.8.3.3.2.3.1 Concrete Structures

a. Normal Operating Conditions

$U = 1.4D + 1.7L$

$U = 1.4D + 1.7L + 1.0T_o$

$U = .75\ (1.4D + 1.7L + 1.4T_o)$

$U = 1.4D + 1.7L + 1.4T_o + 1.7C_1\ (\pm)$

$U = 1.4D + 1.7L + 1.4T_o + 1.7C_2\ (\pm)$

$U = 1.4D + 1.7L + 1.4T_o + 1.7C_A\ (\pm)$

$U = 1.4D + 1.7L + 1.4T_o + 1.7C_{11}\ (\pm)$

$U = 1.4D + 1.7L + 1.4T_o + 1.7C'\ (\pm)$

$S = D + L + T_o$

b. Normal Operating and Severe Environmental Conditions

$U = 1.0D + 1.25L + 1.25E + 1.0T_o$

$U = 1.0D + 1.0L + 1.0E$

$U = .75\ (1.4D + 1.7L + 1.4T_o + 1.9E)$

$U = 1.0D + 1.0L + 1.0T_o + 1.25E + 1.0C_1\ (\pm)$

$U = 1.0D + 1.0L + 1.0T_o + 1.25E + 1.0C_2\ (\pm)$

$U = 1.0D + 1.0L + 1.0T_o + 1.25E + 1.0C_A\ (\pm)$

$U = 1.0D + 1.0L + 1.0T_o + 1.25E + 1.0C_{11}\ (\pm)$

$U = 1.0D + 1.0L + 1.0T_o + 1.25E + 1.0C'\ (\pm)$

c. Normal Operating and Abnormal Operating Conditions

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.5P_{CI} + 1.5(R_{CO} \text{ or } R_C) + 1.25C_1\ (\pm)$

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.5P_{CI} + 1.5(R_{CO} \text{ or } R_C) + 1.25C_2\ (\pm)$

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.5P_{CI} + 1.5R_C + 1.25C_A\ (\pm)$

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.5(P_{DS} \text{ or } P_{DN}) + 1.5R_C + 1.25C_1\ (\pm)$

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.5(P_{DS} \text{ or } P_{DN}) + 1.5R_C + 1.25C_2$ (±)

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.5(P_{DS} \text{ or } P_{DN}) + 1.5R_C + 1.25C_A$(±)

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.5P_{CDC} + 1.5R_C + 1.0R$

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.5(P \text{ or } P_{AP}) + 1.0R + 1.25C_1$ (±)

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.5(P \text{ or } P_{AP}) + 1.5(R_B \text{ or } R_{BA}) + 1.25C_1$ (±)

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.5(P \text{ or } P_{AP}) + 1.5(R_P + R_D) + 1.25C_1$ (±)

d. Normal Operating, and Severe Environmental Conditions

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.25(P \text{ or } P_{AP}) + 1.0R + 1.25E + 1.0C_1$ (±)

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.25(P \text{ or } P_{AP}) + 1.25(R_B \text{ or } R_{BA}) + 1.05E + 1.0C_1$ (±)

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.25(P \text{ or } P_{AP}) + 1.25(R_D + R_P) + 1.25E + 1.0C_1$ (±)

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.25P_{CI} + 1.25(R_{CO} \text{ or } R_C) + 1.25E + 1.0C_1$ (±)

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.25P_{CI} + 1.25(R_{CO} \text{ or } R_C) + 1.25E + 1.0C_2$ (±)

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.25P_{CI} + 1.25R_C + 1.25E + 1.0C_A$ (±)

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.25(P_{DS} \text{ or } P_{DN}) + 1.25R_C + 1.25E + 1.0C_1$ (±)

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.25(P_{DS} \text{ or } P_{DN}) + 1.25R_C + 1.25E + l.0C_2$ (±)

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.25(P_{DS} \text{ or } P_{DN}) + 1.25R_C + 1.25E + 1.0C_A \ (\pm)$

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.25P_{CDC} + 1.25R_C + 1.25E + 1.0R$

e. Normal Operating, Abnormal Operating, and Extreme Environmental Conditions

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.0(P \text{ or } P_{AP}) + 1.0C_1 \ (\pm) + 1.0E' + 1.0R$

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.0(P \text{ or } P_{AP}) + 1.0C_1(\pm) + 1.0E' + 1.0(R_B \text{ or } R_{BA})$

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.0(P \text{ or } P_{AP}) + 1.0C_1 \ (\pm) + 1.0E' + 1.0(R_D + R_P)$

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.0P_{CI} + 1.0C_1(\pm) + 1.0E' + 1.0(R_{CO} \text{ or } R_C)$

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.0P_{CI} + 1.0C_2(\pm) + 1.0E' + 1.0(R_{CO} \text{ or } R_C)$

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.0P_{CI} + 1.0C_A \ (\pm) + 1.0E' + 1.0R_C$

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.0(P_{DS} \text{ or } P_{DN}) + 1.0C_1 \ (\pm) + 1.0E' + 1.0R_C$

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.0(P_{DS} \text{ or } P_{DN}) + 1.0C_2 \ (\pm) + 1.0E' + 1.0R_C$

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.0(P_{DS} \text{ or } P_{DN}) + 1.0C_A \ (\pm) + 1.0E' + 1.0R_C$

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.0P_{CDC} + 1.0R_C + 1.0R + 1.0E'$

3.8.3.3.2.3.6 Steel Structures

The loading combinations and associated allowable stresses for containment internal steel structures are discussed in subsection 3.8.3.5.5.

3.8.3.4 Design and Analysis Procedures

3.8.3.4.1 Drywell

The overall analysis procedure incorporates two phases. The first phase involves the initial proportioning of the various structural elements of the internal structure and has been accomplished by one or a combination of the methods listed below:

a. Classical theory

b. Empirical equations

c. Numerical methods

d. Experience based on completed designs

e. Parametric studies of other structures of a similar nature

After the initial proportioning had been accomplished, several detailed finite element models were created using the information which was compiled during the initial phase. The overall drywell analysis employed three separate finite element models. The first model (see Figure 3.8-56) encompasses the entire containment, subfoundation, and the lower portion of the drywell. The second model encompasses the lower drywell wall including the vent portion (see Figure 3.8-73). The third model encompasses the upper drywell wall, drywell roof, refueling pools and upper pool walls (see Figure 3.8-74). A detailed description of the containment model, associated assumptions, and boundary conditions can be found in subsection 3.8.1.4.

The finite element model for the lower vent section (Figure 3.8-73) consists of an 8-degree segment of the cylindrical shell and was used for the analysis of all axisymmetric loadings. The model was assumed to consist of a linearly elastic, homogeneous, isotropic concrete cylindrical core section which is lined on both sides with a linearly elastic, homogeneous, isotropic steel plate. Vent holes and the steel vent hole liner plates were also modeled.

The following boundary conditions are imposed on the subject model to arrive at the final analysis results.

a. At the lower wall/base slab interface, vertical support was provided for all exterior nodes.

b. The lower wall/base slab interface translations and rotations were taken from the axisymmetric analysis of the containment model and imposed on the nodes at the bottom of the model.

c. The dead weight and appropriate live loads from the upper drywell wall, drywell roof, drywell head, upper pool walls, and upper pool water have been imposed on the upper free end of the model as nodal loads.

All other loading phenomena, including nonlinear thermal gradients, were applied to the model in a standard format. The nonlinear thermal gradients are shown on Figures 3.8-68 through 3.8-72. The computer program used for this portion of the analysis was SAP which is a Bechtel proprietary program discussed in Appendix 3D.

Major drywell penetrations, such as the drywell equipment hatch and the drywell personnel air lock, have had separate analyses performed on them. Descriptions of these analyses can be found in the latter part of this section.

The static and dynamic nonaxisymmetric and axisymmetric loading cases related to the drywell were analyzed using the ASHSD program. Loading conditions falling into this category are discussed in subsections 3.8.1.4.1.1.1, 3.8.1.4.1.12, and 3.8.3.3.1. ASHSD is a Bechtel proprietary program discussed in Appendix 3D of this FSAR. The ASHSD model employed for this analysis is shown on Figure 3.8-54.

The upper drywell and upper pool walls were analyzed with a three dimensional finite element model. The model for this analysis is shown on Figure 3.8-74. Due to bilateral symmetry of loadings and of the structure, only one quarter of the actual structure was modeled. The concrete was assumed to be a linearly elastic, homogeneous, isotropic material which was modeled with plate elements. The boundary conditions for the edges of the model along the lines of symmetry were satisfied with typical conditions of symmetry (i.e., lateral translation and edge rotation are restrained). All loadings listed in subsection 3.8.3.3 were

appropriately incorporated into the above model except the hydrodynamic loads induced on the upper pool walls. Hydrodynamic loads were incorporated by a longhand method delineated in Earthquake Engineering for Nuclear Reactor Facilities by Blume, Sharpe and Kost. SAP was the computer program employed for this analysis. Nonlinear thermal gradients were applied to the plate elements by using a numerical integration which replaced the nonlinear gradient with a gradient and a uniformly varying gradient.

The support conditions for the common boundary of the two drywell wall finite element models are compatible and represent the behavior of a typical section of a long cylinder under axisymmetric loading. The vertical sides of both models are restrained in the tangential direction but are free to translate in the radial and vertical directions. Thus, the models behave as cylinders. Elements at the interface in both models are free to translate in the radial direction and fixed against rotation in all directions. The interface elements are free to translate in the tangential direction except at the vertical edges. Also, the top of the lower drywell is free to deflect vertically, and the bottom of the upper wall model is supported in the vertical direction. The top of the lower drywell wall is restricted in the design so that all points on that surface deflect in a common plane. Thus, the interface between the two models closely represents a typical section through a wall. This boundary condition does not affect the interface stresses in the wall. These stresses are primarily membrane stresses since the bending stresses at the interface are extremely small. The fact that the bottom of the upper wall is supported vertically does not appreciably affect the drywell design since the actual maximum deflection of the top of the lower wall is extremely small. Vertical deflections of the upper wall are relative to those of the lower wall model. The total vertical deflection of a point in the upper wall model is a combination of the deflections of both models.

The results of the two models subjected to equivalent loading combinations were compared for compatibility and equilibrium. At the interface, the displacements, normal forces, and moments in the upper wall model are essentially compatible with those in the lower wall model. In addition, the interface between the two models is located at a point in the structure where the drywell behaves as a long cylinder, i.e., the interface is located at least one characteristic length away from the major

discontinuities in the structure, such as the roof-wall connection, the wall-basemat connection, and the vent holes. No significant discontinuities in the structure are located near the chosen interface. Any small disagreement in stress at this interface is insignificant since the stress resultants away from the interface (at the structural discontinuities) control the design of the drywell. The entire structure was designed for these maximum stress resultants, thereby yielding a conservative and adequate design for the drywell.

The pressure-retaining parts of the drywell head assembly were analyzed and designed to the requirements of Subsection NE (Class MC) of the ASME Boiler and Pressure Vessel Code, 1974 Edition. In addition, the non-pressure-retaining lifting lugs on the head have been analyzed for loads incurred during the lifting and positioning of the head. The interaction between the cylindrical portion of the drywell head and the reinforced concrete drywell roof has been considered. The analyses performed include stress analyses, thermal analyses, shell stability analyses, and dynamic response analyses. The loads included in the above analyses are the design external and internal pressures and temperatures associated with the loss-of-coolant accident, normal operating thermal loads, the hydrostatic head of water in the pool over the drywell, specified steam jet impingement loads, and seismic accelerations due to the SSE and OBE. The computer program ASHSD was used for the majority of the drywell head static and dynamic analyses. Classical methods delineated in ASME Subsection NE were used to check the drywell head manway penetration.

The refueling bulkhead is not part of the drywell pressure boundary during normal, power operation but is subject to pressure and jet impingement loads due to a postulated release of steam from the RPV in the cavity between the reactor head and the drywell head. Pressure is released from the cavity to the drywell through the open bulkhead vents. The refueling bulkhead is subject to hydrostatic and seismic loads during refueling operations when the bulkhead vent covers are installed to form part of the upper pool boundary. These loadings along with temperature and hydrodynamic loads as applicable have been considered in the bulkhead design.

The drywell equipment hatch was analyzed as an integral part of the drywell. A finite element model was used to analyze the steel hatch frame and the surrounding concrete. Both materials were considered to be linearly elastic, homogeneous and isotropic. The SAP program was used to analyze the hatch.

The drywell personnel lock and related openings were analyzed using the closed form solution. A sketch of the lock and lock barrel is shown on Figure 3.8-58.

The drywell penetrations were analyzed by the procedures which are delineated in BC-TOP-1 (see subsection 3.8.3.2.3).

The lower vent portion of the drywell was designed as a reinforced concrete section. Reinforcement was provided by the exterior steel liner plates. Due to the nature of the analysis model, the cracking of the concrete under thermal loading was taken into account by the procedure delineated in Appendix A of Criteria for Reinforced Concrete Nuclear Power Containment Structures, Title No. 69-2, reported by ACI Committee 349. The limiting tensile and compressive stress in the steel lower vent plate assembly were 0.9 F_y for all factored loading combinations. Provisions of the ASME Code, Section III, Division I, were not employed for the design phase of the lower vent plate assembly in lieu of the above criteria.

Continuity between the base mat and the lower vent assembly was achieved by anchoring the liner plates to the base mat as shown on Figure 3.8-75. At the top of the liner plate assembly, the vertical wall reinforcing bars are anchored to the liner plate as shown on Figure 3.8-75.

The reinforced concrete portions of the drywell wall, roof, and upper pool walls are designed to the requirements of ACI 318-71 and American Concrete Institute, Title No. 69-2, Criteria for Reinforced Concrete Nuclear Power Containment Structures, reported by ACI Committee 349.

In view of the recent ongoing work by ACI 359 on diagonal rebar design criteria, this reinforcement was based on the above as interpreted by the paper, Design of Containments for Seismic Loads, by T. E. Johnson, as delivered in the Specialty Conference on Code Requirements for the Nuclear Containment, ASCE, at the December 1974 meeting at Chicago, Illinois.

Bending and normal forces are resisted by reinforcing which was designed and placed vertically and circumferentially on both faces of the drywell wall as shown on Figure 3.8-75. The drywell roof and upper pool walls are reinforced as shown on Figures 3.8-63 and 3.8-75. Radial shear is resisted by horizontal radial shear ties where required (see Figure 3.8-75). Tangential shear is restrained by two perpendicular layers of diagonal spiral reinforcing as shown on Figure 3.8-75.

Due to the nature of the analysis model, the cracking of the concrete under thermal loading was taken into account by the procedure delineated in Appendix A of the Criteria for Reinforced Concrete Nuclear Power Containment Structures ACI Title No. 69-2. For concrete pressure resisting portions of the drywell, the analysis and design procedures employed comply with Article CC-3000 of the ASME Code only to the extent that ACI 318-71 and ACI-349 comply with Article CC-3000. However, the diagonal reinforcement of the drywell wall was designed to the requirements of Article CC-3421.5 of the ASME Code, Section III, Division 2.

3.8.3.4.2 Weir Wall

The weir wall has been designed as a freestanding cylinder rigidly attached to the RPV pedestal mat (see Figure 3.8-60). The analysis techniques include thin shell theory, classical beam theory, plate theory, and finite element computer programs. The design was performed in accordance with ACI 318-71.

One of the major loads to which the weir wall is subjected is a jet force due to a recirculation line break. The effects of the jet force on the weir wall were evaluated using a closed-form solution based on the assumption that, in response to a local load, a shallow shell acts as a flat plate on an elastic foundation. The analysis showed that the weir wall deformation under the jet force consisted primarily of bending, which is the expected response for a thin cylindrical shell under a local load. The jet force also produced severe shear stresses in the weir wall. Classical beam theory was used to analyze and design the weir wall to carry this shear force.

Computer program output shows that the thermal gradient produced by an SBA induces significant moments and bending deformations in the weir wall. The procedures outlined in ACI 349 for cracked section analysis were used to verify that the wall is adequate to carry these loads without loss of function.

Computer program output showed that the loads associated with SRV discharge were not critical to the weir wall. It can be seen from Figure 3.8-60 that only the top drywell vent is directly opposite the weir wall and that the greater part of the SRV loads are carried by the RPV pedestal mat and not the weir wall.

The stainless steel clad, carbon steel liner plate on the suppression pool side of the weir wall is employed only as a watertight barrier. The plate has been fabricated with enough concrete anchors to develop the full strength of the plate as an extra measure of conservatism.

The computer programs used were FINEL for static axisymmetric loads and ASHSD for static and dynamic axisymmetric and non-axisymmetric loads.

3.8.3.4.3 RPV Pedestal

The RPV pedestal has been designed as a freestanding cylinder rigidly attached to the RPV mat. The analysis techniques include shell theory, classical beam theory, and finite element computer programs.

The design of the pedestal was accomplished in several steps. First, the mat was designed using classical beam theory based on the assumption that it was pin supported at radius 13'-7" and roller supported at radius 32'-4". This allowed all radial and tangential shear loads, induced in the pedestal by the reactor or seismic event, to be transferred to the containment base slab at the pin connections. The roller support allows free thermal expansion of the pedestal mat during accident conditions; it also acts to resist the overturning loads that act upon the RPV and pedestal during seismic or accident conditions via the concrete reinforcing steel tension ties (see Figure 3.8-60).

Design loads for the cylindrical shell pedestal wall were obtained from computer program output and were used to size the reinforcing steel which was laid out in a rectangular pattern using vertical and hoop bars. For finite element meshes see Figure 3.8-76. The upper haunched section of the pedestal has been

designed to carry all the loads to the body of the pedestal. The haunch was assumed to act as a corbel and was checked using the methods of ACI 318-71.

The structural steel embed that spans the CRD openings has been designed using classical beam theory in accordance with the AISC specification to carry all of the loads induced by the reactor at this point. The embed is anchored to the pedestal by the use of high strength anchor bolts and concrete reinforcing steel that transfers the loads to the body of the pedestal.

The pedestal shell has been additionally reinforced at major penetration locations to ensure continuity of stress flow and strength. The design of the additional reinforcement in major penetration areas was performed using the area replacement method for interrupted bars.

The computer programs used were FINEL for static axisymmetric loads and ASHSD for static and dynamic axisymmetric and non-symmetric loads.

3.8.3.4.4 Reactor Pressure Vessel (RPV) Supports

The ring girder that supports the RPV was analyzed using classical beam theory and designed in accordance with the AISC specification.

3.8.3.4.5 Other Internal Structures (Outside the Drywell Concrete and Steel Floors)

3.8.3.4.5.1 Miscellaneous Floors (Outside the Drywell)

Miscellaneous floors outside the drywell consist of both reinforced concrete structures and steel framing with grating.

The reinforced concrete floors and walls are supported at the drywell. A gap is provided to separate the concrete structures from the containment wall to allow differential movements between the drywell and containment wall. In this manner, both vertical and horizontal forces are transferred to the drywell. The analysis and design techniques include classical beam and plate theory. Concrete thickness for most of the slabs and walls is governed by shielding requirements rather than the structural requirements.

For the steel floors, vertical supports are provided at the drywell and the containment walls. Supports for horizontal forces are provided at the drywell wall and sliding connections are provided at the containment wall. Classical beam theory is used for the analysis of the steel structures.

The moments and shears determined above were combined according to loading combinations as shown in subsection 3.8.3.3 to obtain the design moments and shears.

Using design moments and shears, concrete reinforcing or steel beam and connections have been provided to ensure that all levels of stress are maintained within the acceptable limits as specified in ACI 318-71 or the AISC Specification, 1969.

3.8.3.4.5.2 Operating Floors

Operating floors consist of both reinforced concrete structures and steel framing with grating.

The reinforced concrete floors are supported by the upper containment pool walls, which are supported by the drywell. A gap is provided to separate the concrete structures from the containment wall to allow differential movements between the upper pool and containment walls. Both vertical and horizontal forces are transferred to the upper pool walls. The analysis and design techniques include classical beam and plate theory.

For the steel floors with grating, vertical supports are provided at the upper pool walls and the containment wall. Supports for horizontal forces are provided at the upper pool walls only, and sliding connections are provided at the containment wall to ensure that no horizontal reactions are transferred to the containment wall. Classical beam theory was used for the analysis of the steel structures.

The moments and shears determined above were combined according to loading combinations as shown in subsection 3.8.3.3 to obtain the design moments and shears.

Using these design moments and shears, the concrete reinforcing or steel beams and connections have been provided to ensure that all levels of stress are maintained within the acceptable limits specified in ACI 318-71 or the AISC Specification (1969).

3.8.3.4.5.3 Floors at El 147'-7" and 161'-10" (Between the Dry Well Wall and the Shield Wall)

Floors at El. 147'-7" and 161'-10" consist of steel box girders and beams with grating. A grid of box girders and wide flange beams span between the drywell and the shield walls and are supported at either end by box-type sockets. The box-type sockets at the drywell and the shield walls are provided with a removable top and/or bottoms to facilitate erection. Classical beam theory was employed to design this framing.

Most of the framing is designed to carry the vertical, lateral, and longitudinal forces which are caused by the postulated breaks in high energy lines. These lines are restrained from pipe whip by U-shaped bars anchored to the heavy box girders and beams. At these elevations, should a pressurized pipe rupture, the escaping jet will accelerate the pipe, and the pipe will impact the U-shape bars. The bars will arrest the motion of the broken pipe. In addition, the floor supports other electromechanical equipment as well as snubbers, anchors, hangers, and the recirculation pump motor removal monorail. The framing is also designed to carry the load from the frame which supports the major hangers for the recirculation pump motor.

Vertical and lateral forces (forces perpendicular to beams) are transferred to the drywell and to the shield wall by bearings between the beams and sockets.

A pin connection between the socket and the beam is provided at the drywell end to transfer reactions due to longitudinal forces (forces parallel to the longitudinal axis of the beams). Beams and girders are free to slide inside the sockets at the shield wall. This is achieved by a gap between the end of the girders or beams and the inside face of the socket. All the longitudinal forces are transferred to the drywell only, thus allowing for differential movements between the drywell and the shield wall during seismic conditions.

The design parameters, determined by using classical beam theory, were combined according to loading combinations shown in subsection 3.8.3.3 for the maximum design parameters. The framing and connections were designed for maximum design parameters and all levels of stress were maintained within the acceptable limits specified in the AISC Specification (1969).

3.8.3.4.5.4 RPV Shield Wall

The reactor shield wall has been designed as a freestanding cylindrical shell rigidly attached to the RPV pedestal. The RPV pedestal is rigidly attached to the RPV mat. The inside and outside diameters of the shield wall surfaces are 28'-8" and 32'-8", and they are lined with 3/4-inch and 1-1/4-inch thick plates, respectively. The cylinder is 51 feet 10-1/2 inches in height. Between the two lined surfaces, 1 inch vertical webs are provided at 15 degree spacing. Additional horizontal and vertical diaphragms are provided as required to accommodate local loads. The entire cylinder is filled with high density concrete.

The shielding plugs are provided around the pipe penetrations through the shield wall for inservice inspection except where flow diverters are provided. The shield wall inspection opening is provided through the shield wall at a lower elevation to allow easy access to the lower portion of the reactor without passing through the core area. It also allows access to the control rod drive support structures and RPV shield wall annulus for ISI. The shield plugs and the fastening system for the shield plugs were designed to sustain full jet force in case of a guillotine break of a pipe. A thrust coefficient equal to 1.25 and a dynamic load factor equal to 2.0 were assumed to calculate the design jet force.

Flow diverters (See Figure 3.8-60a) are provided in the shield wall penetration sleeves for six feedwater lines and for two recirculation outlet (suction) lines. These flow diverters are cylindrical sleeves that are retractable to allow for inservice inspection. These sleeves can carry a maximum pressure equal to the system pressure at high temperature (approximately 535 F). Jet impingement loads are much smaller than the design limit and therefore do not present a design constraint. Locks are provided to prevent the sleeves from sliding out of the shield wall due to the fluid pressure acting on the edge near the RPV. Shielding is provided around the recirculation outlet flow diverter for gamma and neutron attenuation. Due to the flow diverters, a moderate local load acts outward on the shield wall at the feedwater line penetrations and at the recirculation outlet line penetrations. These flow diverters minimize the asymmetric loads associated with annulus pressurization. During operating and hypothetical accident conditions, there is a negligible temperature gradient across the thickness of the flow diverter sleeve walls and hence, no significant thermal stresses result in these sleeves.

Some of the high energy line pipe whip restraints, snubbers, and hangers for main steam line recirculation and recirculation pump motors are directly attached to the shield wall. Loads from these components were applied to the shield wall as concentrated loads. Stresses were calculated assuming localized effect of these loads and using conventional beam theory. Additional horizontal and vertical diaphragms were provided in the region of the concentrated loads to stiffen the cylinder.

Analysis and design techniques include shell theory, classical beam theory, and computer programs: Bechtel 900 Series for seismic analysis and ASHSD for the RPV shield wall asymmetrical pressure loads. The computer programs used to design the internal structures are discussed in Appendix 3D of this FSAR.

Maximum membrane stresses in the shield wall were determined by superposition of stresses due to effective loads, and maximum bending moments and radial shears were determined at the base of the cylinder by assuming continuity at the base. The cylinder is unrestrained in the vertical direction; therefore, it was assumed that no thermal stresses exist in that direction. In the hoop direction, thermal expansion is restrained by various horizontal diaphragms inside the shield wall. Because of the geometric continuity of the structure, thermal stresses were considered in the longitudinal and circumferential directions.

A finite element model was used to perform the structural analysis for the RPV pedestal and the shield wall to account for asymmetrical pressure transient loadings from annulus pressurization loadings. To determine the effects of annulus pressurization loads on the shield wall, the computer program ASHSD (see Appendix 3D, page 3D-10) was utilized. A detailed dynamic analysis was performed on an axisymmetric finite element model of the shield wall RPV pedestal and the RPV. Loadings considered in this analysis are transient pressures created in the annulus region due to: a) recirculation suction line break; b) recirculation return line break; and c) feedwater line break and the transient effects outside of the annulus region due to a main steam line break. Also included in the analysis is the transient loading applied to the shield wall from the pipe whip restraint at the postulated break location. The recirculation suction line break and feedwater line break analyses include the effect of flow diverters, which reduce the pressure in the annulus. These

pressure loads are combined with the associated linear thermal gradient for the accident considered and analyzed as discussed above.

It was assumed that point loads (jet loads, pipe restraints, snubber and hanger loads, penetrations, and floor loads) produce localized effects on the shield wall cylinder, and these localized effects were evaluated using conventional beam theory and local analysis.

The design loadings determined from the above static and dynamic analyses were combined according to the loading combinations as shown in subsections 3.8.3.3 and 3.8.3.5 to obtain the maximum design loadings. Maximum stress conditions from the annulus pressurization analysis are combined by the absolute sum method, using loading combinations in subsections 3.8.3.3 and 3.8.3.5. Feedback loads of LOCA and/or SRV actuations induce very small stresses on the shield wall. These loads, combined with the other normal loads applied to the shield wall, do not govern the design of the shield wall. The shield wall was designed for maximum design loadings, and all resulting levels of stress were maintained within the acceptance limits as specified in the AISC Specification (1969), ACI 318-71, Section 11, and FSAR subsection 3.8.3.5.5.3. The maximum stresses are shown in Table 3.8-16.

3.8.3.4.5.5 Main Steam Pipe Tunnel

The main steam tunnel has been designed as a reinforced concrete structure supported by the drywell wall. The analysis and design techniques include classical beam and plate theory.

The design of the main steam tunnel was accomplished in several steps. The top and bottom slabs of the tunnel were designed using plate theory based on the assumption that slabs were fixed at the drywell wall and at the two side walls and free at the containment wall. The top and bottom slabs were subjected to the same loads, except the bottom slab was subjected to additional loads due to pool swell. For thermal loads, the top and bottom slabs were assumed fixed at the drywell wall. Moments and shears were determined using classical beam theory.

The side walls of the tunnel were designed assuming a closed frame formed by the two side walls and top and bottom slab. This analysis also provided the moments and shears for the top and bottom slab in the longitudinal direction. A small reinforced concrete L-shaped frame (see Figure 3.8-60), which extends from

the containment wall and follows the boundary line of the main steam tunnel, has been provided for shielding. A 2-inch gap between the main steam tunnel and this frame has been provided to account for seismic movements. The L-shaped frame was designed for the same loads as the main steam tunnel.

The moments and shears resulting from the above analysis were combined according to loading combinations, as shown in subsection 3.8.3.3, in order to obtain the design moments and shears.

Using these design moments and shears, concrete reinforcing has been provided to ensure that all levels of stress are maintained within the acceptable limits specified in ACI 318-71.

3.8.3.4.5.6 Floor at Elevation 120'-10"

The unique design of the welded plate girders at El. 120'-10" was dictated by the severe loads resulting from pool swell.

The original analysis, performed by GE and modified by the NRC, produced a static equivalent load of 230 psi, to be applied to structures just above the pool. The loading was used to size the web and bracing plates of the structural steel floor framing in accordance with the AISC Specification for structural steel.

Subsequent testing by GEH showed that the pool swell loading reaches a maximum of 115 psi as shown in Appendix 6D, Figure 3B-74. Since fabrication of this steel was nearing the completion stage, the original design was not altered to reflect this reduced loading.

3.8.3.4.5.7 Floor at Elevation 135'-4”

The characteristics of the pool swell loads dictated that the analysis of the floor at El. 135'-4” be performed using structural dynamic techniques. The following steps were used to analyze and design the structural members of the floor. The natural frequency of a unit width of the concrete slab and each steel member was calculated, and then by using the procedures and graphs published in Biggs “Structural Dynamics,” a dynamic load factor was determined. The dynamic load factor was then applied to the pool swell loads taken from Appendices 6A and 6D, making it a static equivalent load, and it was algebraically added to the critical load case.

The design of the slab was performed using the criteria of ACI-318-71. All structural framing was designed using the criteria of AISC-1969 except bending stresses were taken to 90 percent of yield.

3.8.3.5 Structural Analysis Criteria

3.8.3.5.1 Drywell

The allowable stresses used for this section comply with Article CC-3000 of the ASME Code, Section III, Division 2 only to the extent that ACI 318-71 and ACI Title No. 69-2, Criteria for Reinforced Concrete Nuclear Power Containment Structures comply with the ASME Code.

Allowable stresses for steel pressure retaining portions of the drywell were taken as equal to 0.9 F_y for all factored load cases. The actual and allowable stresses for the drywell are found in Tables 3.8-7 through 3.8-12.

3.8.3.5.2 Weir Wall

The weir wall was designed in accordance with the strength requirements of ACI 318-71. The actual and allowable stresses for critical sections are shown in Table 3.8-13.

The amount of reinforcing steel provided is based on the ultimate strength design methods of ACI 318-71. The basic assumption of the ultimate strength design method is that the reinforcing steel stress equals the yield stress multiplied by an appropriate capacity reduction factor. This method affords no means of determining actual reinforcing steel stress when the area of reinforcing steel provided exceeds the area required by calculation.

Therefore, in order to determine the actual stress in the reinforcing steel and concrete when the area of steel provided exceeds the area required or for load cases other than the critical load case, the techniques outlined in the working stress design methods and in Title 69-2 for crack section analysis are employed.

3.8.3.5.3 Reactor Pressure Vessel (RPV) Support Pedestal and Mat

The RPV support pedestal and mat are designed in accordance with the strength requirements of ACI 318-71. The actual and allowable stresses for critical sections are shown in Tables 3.8-14 and 3.8-15.

The amount of reinforcing steel provided is based on the ultimate strength design methods of ACI 318-71. The basic assumption of the ultimate strength design method is that the reinforcing steel stress equals the yield stress multiplied by an appropriate capacity reduction factor. This method affords no means of determining actual reinforcing steel stress when the area of reinforcing steel provided exceeds the area required by calculation.

Therefore, in order to determine the actual stress in the reinforcing steel and concrete when the area of steel provided exceeds the area required or for load cases other than the critical load case, the techniques outlined in the working stress design methods and in Title 69-2 for crack section analysis are employed.

3.8.3.5.4 Reactor Pressure Vessel (RPV) Supports

The RPV support ring girder is designed to the stress requirements of the AISC Manual of Steel Construction. The stress limits are shown in subsection 3.8.3.5.5.3.

3.8.3.5.5 Other Internal Structures

The actual and allowable stresses for other internal structures are found in Tables 3.8-16 through 3.8-24.

The amount of reinforcing steel provided is based on the ultimate strength design methods of ACI 318-71. The basic assumption of the ultimate strength design method is that the reinforcing steel stress equals the yield stress multiplied by an appropriate capacity reduction factor. This method affords no means of determining actual reinforcing steel stress when the area of reinforcing steel provided exceeds the area required by calculation.

Therefore, in order to determine the actual stress in the reinforcing steel and concrete when the area of steel provided exceeds the area required or for load cases other than the critical load case, the techniques outlined in the working stress design methods and in Title 69-2 for crack section analysis are employed.

3.8.3.5.5.1 Notation

U - Required ultimate load capacity as defined in ACI 318-71

S - Required capacity of the section such that the stress limitations of Section 8.10, ACI 318-71, are satisfied

0 - Capacity reduction factor as defined in ACI 318-71, Section 9.2

F_b - Allowable bending stress for structural steel

F_a - Allowable compression stress for structural steel

F_t - Allowable tensile stress for structural steel

F_v - Allowable shear stress in structural steel

F_y - Yield strength of steel

F_s - Allowable stress for structural steel

F_p - Allowable bearing stress for structural steel

3.8.3.5.5.2 Concrete Structures

All internal concrete structures are designed for the strength requirements of ACI 318-71. The load combinations and applicable section capacities are shown in subsection 3.8.3.3.2.3.

3.8.3.5.5.3 Steel Structures

The following are the load combinations and allowable design limits used for the internal steel structures. To assure that structural integrity will be maintained, limits on the resulting stresses are considered for service and factored loads and various combinations thereof.

a. Normal Conditions

Loading Combinations	Stress Limit
Normal Operating Conditions	
$D + L + T_o$	F_s
$D + L + T_o + C_1$	F_s
$D + L + T_o + C_2$	F_s
$D + L + T_o + C_A$	F_s
$D + L + T_o + C_{11}$	F_s
$D + L + T_o + C'$	F_s
Normal Operating and Severe Environmental Conditions	
$D + L + T_o + E$	$1.25\ F_s^*$
$D + L + T_o + E + C_1$	$1.25\ F_s^*$
$D + L + T_o + E + C_2$	$1.25\ F_s^*$
$D + L + T_o + E + C_A$	$1.25\ F_s^*$
$D + L + T_o + E + C_{11}$	$1.25\ F_s^*$
$D + L + T_o + E + C'$	$1.25\ F_s^*$
For structural elements carrying mainly earthquake forces:	
$D + L + T_o + E$	F_s
$D + L + T_o + E + C_1$	F_s
$D + L + T_o + E + C_2$	F_s
$D + L + T_o + E + C_A$	F_s
$D + L + T_o + E + C_{11}$	F_s
$D + L + T_o + E + C'$	F_s

b. Factored Load Conditions

Normal Operating and Extreme Environmental Conditions	
$D + L + T_o + E'$	(see note)
$D + L + T_o + E' + C_1$	(see note)
$D + L + T_o + E' + C_2$	(see note)
$D + L + T_o + E' + C_A$	(see note)
$D + L + T_o + E' + C_{11}$	(see note)
$D + L + T_o + E' + C'$	(see note)

* The allowable stress for structural steel, Fs, used for these load combinations which include the OBE, is

increased by 25 percent. This increase in the allowable stress was adopted based on the recommendations of the following codes and standards.

a. In AISC "Specification for the Design Fabrication, and Erection of Structural Steel for Buildings" (1969), Article 1.5.6 allows an increase of 1/3 in the allowable stresses when wind or seismic loadings act alone or in combination with the design dead and live loads.

b. In ANSI Standard A58.1-1972, Article 4.2 allows indirectly a similar increase in the allowable stresses when earthquake loads (E) or wind loads (W) are combined with dead load (D) and live load (L).

The 25-percent increase in the allowable stresses for OBE is, therefore, conservative when compared to the allowables specified for design in the above codes and standards.

Moreover, both the PSAR and the FSAR have clearly distinguished the loading case for structural members carrying mainly earthquake forces and does not allow for any increase in the allowable stresses.

Normal Operating and Abnormal Operating Conditions	
$D + L + (T_o + T_a) + R$	(see note)
$D + L + (T_o + T_a) + R + (P \text{ or } P_{AP})$	(see note)
$D + L + (T_o + T_a) + R + (P \text{ or } P_{AP}) + C_1$	(see note)
$D + L + (T_o + T_a) + R + (P \text{ or } P_{AP}) + C_2$	(see note)
$D + L + (T_o + T_a) + R + (R_B \text{ or } R_{BA}) + R_P + C_1$	(see note)
$D + L + (T_o + T_a) + R + R_D + C_1$	(see note)
$D + L + (T_o + T_a) + R + R_F + C_1$	(see note)
$D + L + (T_o + T_a) + R + R_C + C_1$	(see note)
$D + L + (T_o + T_a) + R + R_{CO} + C_1$	(see note)
$D + L + (T_o + T_a) + R_{CO} + C_1$	(see note)
$D + L + (T_o + T_a) + R_{CO} + C_2$	(see note)
$D + L + (T_o + T_a) + R_C + C_A$	(see note)
$D + L + (T_o + T_a) + R_C + C_1$	(see note)
$D + L + (T_o + T_a) + R_C + C_2$	(see note)
Normal Operating, Abnormal Operating, and Severe Environmental Conditions	
$D + L + (T_o + T_a) + E + R$	(see note)
$D + L + (T_o + T_a) + E + R + (P \text{ or } P_{AP})$	(see note)
$D + L + (T_o + T_a) + E + R + (P \text{ or } P_{AP}) + C_1$	(see note)
$D + L + (T_o + T_a) + E + R + (P \text{ or } P_{AP}) + C_2$	(see note)
$D + L + (T_o + T_a) + E + R + (R_B \text{ or } R_{BA}) + R_P + C_1$	(see note)
$D + L + (T_o + T_a) + E + R + R_D + C_1$	(see note)
$D + L + (T_o + T_a) + E + R + R_F + C_1$	(see note)
$D + L + (T_o + T_a) + E + R + R_C + C_1$	(see note)
$D + L + (T_o + T_a) + E + R + R_{CO} + C_1$	(see note)
$D + L + (T_o + T_a) + E + R_{CO} + C_1$	(see note)
$D + L + (T_o + T_a) + E + R_{CO} + C_2$	(see note)
$D + L + (T_o + T_a) + E + R_C + C_A$	(see note)
$D + L + (T_o + T_a) + E + R_C + C_1$	(see note)
$D + L + (T_o + T_a) + E + R_C + C_2$	(see note)
Normal Operating, Abnormal Operating, and Extreme Environmental Conditions	
$D + L + (T_o + T_a) + E' + R$	(see note)

$D + L + (T_o + T_a) + E' + R + (P \text{ or } P_{AP})$	(see note)
$D + L + (T_o + T_a) + E' + R + (P \text{ or } P_{AP}) + C_1$	(see note)
$D + L + (T_o + T_a) + E' + R + (P \text{ or } P_{AP}) + C_2$	(see note)
$D + L + (T_o + T_a) + E' + R + (R_B \text{ or } R_{BA}) + R_P + C_1$	(see note)
$D + L + (T_o + T_a) + E' + R + R_D + C_1$	(see note)
$D + L + (T_o + T_a) + E' + R + R_F + C_1$	(see note)
$D + L + (T_o + T_a) + E' + R + R_C + C_1$	(see note)
$D + L + (T_o + T_a) + E' + R + R_{CO} + C_1$	(see note)
$D + L + (T_o + T_a) + E' + R_{CO} + C_1$	(see note)
$D + L + (T_o + T_a) + E' + R_{CO} + C_2$	(see note)
$D + L + (T_o + T_a) + E' + R_C + C_A$	(see note)
$D + L + (T_o + T_a) + E' + R_C + C_1$	(see note)
$D + L + (T_o + T_a) + E' + R_C + C_2$	(see note)

Note: The following allowable stresses are used for the factored load conditions:

$F_b = 0.9F_Y$ (bending)

F_a = 1.33 times the allowable compressive stress for the working stress design

$F_v = 0.5F_y$ (shear)

F_p = 1.33 times the allowable bearing stress for the working stress design

F_t = 1.33 times the allowable tensile stress for the working stress design

For the RPV shield wall, the following allowable stresses are used for factored load combinations:

$F_b = 0.9F_y$ (bending)

$F_a = 0.85F_y$ (axial tension)

$F_v = 0.5F_y$ (shear)

For factored loading combinations containing "P" or "P_{AP}" loads, stress limits shall be limited to 1.5 F_s except for web shear stress where stress limits shall be limited to 1.4 F_s. The allowable stresses under factored load conditions are within the requirements of Section 3.8.3.II.3 and Section 3.8.3.II.5 of the SRP. For further discussion see Tables 3.8-35 and 3.8-36.

3.8.3.6 Materials, Quality Control, and Special Construction Techniques

3.8.3.6.1 General

Specifications covering all materials are of sufficient detail to assure the structural design requirements of the work.

3.8.3.6.2 Concrete

a. [HISTORICAL INFORMATION] [All concrete materials are approved prior to start of construction on the basis of their characteristics in test comparisons using ASTM standard methods. Concrete aggregates and cement, conforming to acceptability criteria of the specifications, are obtained from approved sources. Concrete properties are determined by laboratory tests. Concrete admixtures are used to minimize the mixing water requirements and increase workability. The specified compressive strength at 91 days, or earlier, is:

Structure of Component	Specified Strength (f'_c) (psi)
Precast concrete panels	4,000
Containment and internals	5,000
All other seismic Category I concrete	4,000
Mass concrete fill	2,000
RPV shield wall concrete fill	5,000
Electrical duct encasement	**2,000**

b. Water used in mixing of concrete, mortar, and grout is tested in accordance with the provisions of the American Association of State Highway Officials (AASHO), Standard T-26-70, Standard Method of Testing for Quality of Water to be Used in Concrete.

Acidity or alkalinity (pH factor) is within the range of 6.0 to 8.5, and the water does not contain impurities in excess of the following limits:

	Percent	ppm
Total chlorides as C1	0.025	250
Total sulphates as SO_4	0.10	1,000
Total organic solids	0.05	500
Total inorganic solids	0.20	2,000

c. All concrete work is in accordance with ACI 318-71, Building Code Requirements for Reinforced Concrete, except as modified in subsection 3.8.3.2.2.2, and complies with the following specifications:

Material	ASTM Specification
Cement, Type II, low alkali	C 150-72
Normal aggregate	C 33-71a
Heavy aggregate	637-73 and C 638-73
Fly ash	C 618-73 (see note below)
Air-entraining admixture	C 260-69
Water-reducing admixture	C 494-71
RPV shield wall fill concrete aggregate	C 637-73

Note: Exceptions to ASTM C 618-73: in-process testing frequency shall be for each 200 tons in accordance with ANSI N45.2.5, Table B; in addition, passing the Pozzolanic Activity Index with either cement or lime shall be acceptable.]

3.8.3.6.3 Reinforcing Bars

[HISTORICAL INFORMATION] [Reinforcing bars for concrete consist of deformed bars meeting the requirements of ASTM 615-75 (including Supplementary Requirements S-1, Grade 60), with additional requirements as set forth below for tension and bend testing.

All test specimens are full-diameter bars, as rolled.

One tension test is made for each 50 tons, or fraction thereof, of each bar size of reinforcing bars produced from each heat and production rolling for the determination of tensile properties. Acceptance standards for tension testing are in conformance with the requirements of Table 2, ASTM A 615-75, including the retest provisions of Section 11, ASTM A 615-75.

One bend test is made for each test lot and production rolling of each bar size of each heat for the determination of bend properties. Acceptance standards for bend testing are in conformance with the requirements of Table 3 and Supplementary Requirements S-1, ASTM A 615-75.

Normally, testing is accomplished at the mill or shop of the supplier; the purchaser's representative has the option at any time, of witnessing the selection, cutting, and testing of test specimens. Any of the required testing, which has not been accomplished prior to receipt of the reinforcing bar at the construction site, is accomplished as part of the receiving inspection process at the construction site.]

3.8.3.6.4 Structural Steel

The structural steel materials conform to all applicable requirements of the AISC Manual of Steel Construction, unless otherwise noted in the job specifications or on the drawings, and also comply with the following specifications:

a. Materials for Interior Structures

Material	ASME or ASTM Specification
Structural steel and connections	A 36
High strength structural steel	A 441, A 572, A 516, A 540
Tubular steel sections	A 53, Types F and S; A 500, Grade A or B; A 501
Anchor bolts	A 36, A 449
Shop fabrication	AWS Dl.1 welding or bolting, A 325 and A 490 bolts
Field erection	High strength bolts, A 325 and A 490, or welding AWS D1.1

RPV anchor bolts	ASME SA 540

b. Materials for the Drywell

Material	ASME or ASTM Specification
Vent assembly below El. 117'-10"	A 240, Type 304L, clads A 537, Grade A
Drywell formwork above El. 117'-10"	A 36
Thickened plate penetrations and inserts above El. 117'-10"	A 516, Grade 70; A 36
Penetrations:	
Seamless pipe	SA 333 Grade 1 or 6
Welded pipe	SA 155 KCF Grade 70
Reinforcement plate	SA 516 Grade 70
Anchoring rings and plates	SA 516 Grade 70
Stainless steel pipe	A 312
Rolled shapes	A 36
Plate, stainless steel	SA 240, Type 304
Plate, carbon steel	SA 516, Grade 70
Forging, carbon steel	SA 350, Grade LF 1
Pipe, carbon steel	A 36
Fasteners:	
Structural steel	A 325 or A 490
Anchor bolts	A 307 or A 36
Concrete studs	A 108

Steel plate is tested at the mill in full conformance with the applicable ASTM specifications, and certified mill test reports are supplied for review and approval. The plate is visually examined for laminations and pitting. Identity of the plate is maintained throughout fabrication.

3.8.3.6.5 Control Tests for Concrete

a. [HISTORICAL INFORMATION] [Sample Tests

Samples for the following routine in-process tests of concrete are taken at the sampling point, at the frequencies shown, in accordance with ASTM C 172-71. This point shall be at the central mix plant discharge. Correlation test samples shall be taken at the end of the last piece of conveying equipment at the point of discharge of concrete into the forms. Water shall not be added to any concrete batch after in-process tests are made, unless the in-process tests (except unit weight/ yield) are repeated after the water is added. Samples shall not be taken from concrete after it has been deposited in the forms. The sampling frequencies are as follows:

1. Temperature: First batch produced each day and every 50 cubic yards placed for each class of concrete.

2. Slump (ASTM C 143-71): First batch produced each day and every 50 cubic yards placed for each class of concrete.

3. Air Content (ASTM C 231-72T): First batch produced each day and with each set of compressive strength cylinders for each class of concrete.

4. Plastic Unit Weight/Yield (ASTM C 138-71T): Daily during production for each class of concrete.

5. Compressive Strength (ASTM C 31-69), tested in accordance with ASTM C 39-72: Standard 6 inch by 12 inch concrete test cylinders will be molded and tested at 7, and 28 or 91 days from each 100 cubic yards, or a minimum of one set per day for each class of concrete.

Correlation tests for air content, slump, and temperature of concrete shall be made at the end of the last piece of conveying equipment for the first batch produced each day and for each 200 cubic yards or fraction over 100 cubic yards placed of each class of concrete.

b. Concrete Cylinders

A strength test is the average of the strengths of the two specimens from each sample tested at 28 or 91 days. Only one cylinder shall be made and tested at 7 days. The provisions of ACI 318-71, paragraph 4.3.3, shall apply in determining satisfactory strength level of the concrete tested at 28 or 91 days.]

3.8.3.6.6 Splices of Reinforcement

a. Provisions of ACI 318-71, Sections 7.5, 7.6, and 7.7, will be met for all reinforcing bar splices. Splices are designed to develop the specified minimum yield strength.

b. All mechanical splices for reinforcing bars are made by the Cadweld process, using clamping devices, sleeves, charges, etc., as specified by the Cadweld Splice Instruction Sheets for B and T series connections. C series materials are not permitted.

Qualification of operators, visual inspection, tensile testing, tensile test frequency, and procedure for substandard tensile test results are in accordance with the Cadweld instruction sheets, and comply with NRC Regulatory Guide 1.10, Revision 1, dated 1/2/73.

3.8.3.7 Testing and Inservice Surveillance Requirements

3.8.3.7.1 Test Requirements

A formal program of testing and inservice surveillance is not planned for the internal structures except the drywell structure for Unit 1 which will undergo a structural integrity test (SIT). The other internal structures are not directly related to the functioning of the containment concept; therefore, no testing or surveillance is required.

The drywell will be subjected to a structural integrity test (SIT) during which the drywell internal pressure will be increased by four equal pressure increments of 7.5 psig from atmospheric pressure to the design pressure of 30 psig. Normal operational water volumes will be present. The containment will be open to atmosphere. After each pressure increment has been attained, the pressure will be held constant for a minimum of 1 hour after which the deflections and strains are recorded.

After the design pressure has been attained, the drywell will be depressurized by the same equal pressure increments, and measurements will be recorded as during the pressurization cycle. Measurements at atmospheric pressure will be recorded before and after the test.

Radial deflections will be measured at three points along each of three meridians which are equally spaced around the drywell. The location of these points of radial measurement will be as shown on Figures 3.8-1 and 3.8-118.

Because the drywell is considered a prototype structure, strain measurements will be made by means of embedded instruments at two opposing meridians as shown on Figures 3.8-118 and 3.8-75.

Where the outer surface of the lower drywell wall has a stress carrying steel plate, orthogonal strain gauges are attached to the outer surfaces. See Figure 3.8-118 for the location of these strain gauges.

Deflection and strain measurements as well as concrete crack pattern measurements will be recorded at the time of the test. Predicted values were found in accordance with the procedure below.

Since the cylindrical drywell is a non-prestressed reinforced concrete structure, the concrete can be expected to exhibit some stress-induced cracking during the SIT. The gross section hoop tension stresses approach the lower bound predictions of the modulus of rupture for the concrete. The predicted deflections, strains, and crack widths and spacings are presented in Tables 3.8-27 through 3.8-30.

The predictions of deflections and strains are based on the finite element analysis described in subsection 3.8.3.4.1. Concrete crack widths were predicted using classical theory of flexure in concrete (Refs. 12 and 13). These values were independently checked and crack spacing predicted using methods developed for pressure-retaining concrete shells (Ref. 11). The results of these two approaches gave reasonable agreement, and the predicted values are given in Table 3.8-30. It should be noted that the predictions of deflections, strains, and crack widths are based on the assumptions of material homogeneity, structural uniformity, and well-behaved structural response. The actual values will be greatly influenced by local structural behavior attributed to (1) variations in concrete material properties,

such as modulus of rupture, modulus of elasticity, compressive strength, and pre-existing shrinkage cracks; (2) structural discontinuities, such as major penetrations and embedded items; and (3) structural components that add stiffness to the shell, such as walls, floors, and slabs.

Due to the complexity of accounting for all of the above contributing factors, a tolerance band of -60 to +40 percent was selected to establish a range into which the predicted values of deflections could fall. This tolerance range serves as an envelope for the predicted responses and accounts for variations of the structural parameters described above.

This tolerance band was established using several published references (Refs. 14 and 15) such as technical literature, laboratory test results of well-behaved components, and review of actual test data from previous structural integrity tests performed on concrete containments. The tolerance values reflect those found in the references, but were increased due to the variations previously discussed and were further adjusted for the anticipated field test conditions as opposed to the controlled laboratory test conditions.

As concrete cracking occurs, the subsequent redistribution of load from the concrete to the reinforcement would cause significant variations in the magnitude of predicted local response. If the cracks in the concrete traverse the instrumented reinforcing steel, strain measurements could increase by a factor of 3 due to local change in stiffness. Local deflections may similarly be affected. Although the local increases in strain to the magnitude stated above would result in reinforcing stresses equal to the allowable stress, this would have no detrimental effect on the structural integrity of the drywell.

Further, if concrete cracking is assumed to consist of all through-thickness cracks having the maximum predicted crack area, the resulting leakage rate is well below the maximum allowable leakage rate of 84,000 scfm discussed in subsection 6.2.6.5.1. Crack mapping areas shown on Figure 3.8-118 have been located in accessible areas which are relatively free of structural discontinuities. Concrete cracks with widths greater than 0.010 inch shall be recorded.

The purpose of the structural integrity test is to demonstrate that the design and construction of the drywell is sufficient to maintain the design pressure load. Failure of the structural integrity test would be defined as gross structural deformation, collapse, or a gross rupture causing a leakage rate in excess of 84,000 scfm.

The measured deflections and strains will be compared with those predicted in the above procedure and as shown in Tables 3.8-27 through 3.8-29.

3.8.3.7.2 Drywell Structural Integrity Test and Integrated Leak Rate Test Results

3.8.3.7.2.1 Structural Integrity Test

The structural integrity test was performed in accordance with the procedures and requirements of subsection 3.8.3.7.1. The drywell internal pressure was increased to the design pressure of 30 psig in accordance with the pressure cycle shown on Figure 3.8-119.

Instrumentation used in the test included invar wire extensometers for measurement of vertical and radial displacements. These extensometers were of the same type as used on 19 other containment structures under conditions similar to Grand Gulf Unit 1 drywell. The ends of the invar wires were anchored to the inside of the drywell wall and outside of the shield wall for radial measurements and to the bottom of the drywell roof and top of the pedestal mat for vertical measurements. Included within the span was a spring-loaded frame incorporating a linear potentiometer.

The strain gauges were the half-bridge type for the liner plate and the full-bridge type for the instrumented reinforcement. The liner gauges were welded directly to the plate, and the bar gauges were attached to milled surfaces on the reinforcement and covered with shrinkable boots for protection against damage.

The extensometers and strain gauges were connected to an external power supply and readout equipment for data acquisition. The extensometers were calibrated with traceability to the National Institute of Standards and Technology. The strain gauges were procured to satisfy the anticipated strains.

Revised Tables 3.8-27 through 3.8-29 summarize the actual instrumented bar strains, liner plate strains, and deflections that resulted from the test. A discussion of the results is given in the following paragraphs.

a. Liner Plate Strains

The measured strains of the drywell structural liner plate are shown in Table 3.8-28. The vertical strains at azimuths 90° and 270° were not measured. Sufficient instrumented reinforcement was used to establish the characteristics of the lower drywell response.

The gauge attached to the outside liner plate at azimuth 90° was found to be not operable just before the test began. The remaining gauges indicated that the drywell structural liner plate was stressed well below the expected response. These results also reveal that the structural plate stress is substantially below the allowable stress (which is one half the yield stress).

b. Instrumented Bar Strains

Table 3.8-27 compares the measured instrumented bar strains with the predicted values at the 16 locations shown. The measured bar strains were well below the predicted values and also well below the allowable service load stress of 30 ksi. This direct measurement of reinforcement stress reveals that the structure is conservatively modeled and designed. With a maximum bar strain of 193 in./in. x 10^{-6}, the structure shows no significant distress.

c. Crack Data

The crack mapping areas shown in Figure 3.8-118 were visually surveyed at the 0, 15, 30, and 0 psi stages during the test. The test procedures required that any cracks greater than 0.01" in width would be recorded; however, no cracking was observed at any of the surveillance areas. Although some cracking was expected to occur, the absence of cracking is consistent with the results obtained from the instrumented reinforcement.

d. Radial and Vertical Deflections

Tables 3.8-28 and 3.8-29 give the comparison of the radial and vertical deflections observed during the test to the predicted values. The actual vertical deflections are in good agreement within the predicted values.

Table 3.8-29 gives the actual radial deflections at each displacement transducer location. The deflection following the 24-hour recovery period is also given in addition to the percent recovery for those deflections outside the tolerance band. Some of the deflections exceeded the predicted values; however, such uncertainties as material nonlinearity, structural discontinuities, and shell out-of-roundness affect not only structural response but also the amount of recovery following the test. The structure recovered more than 80 percent, 24 hours after the test, for each of the deflections outside the tolerance band with the exception of the extensometer located at azimuth 155°, El. 140'-0". The factors discussed above contribute to the lack of total recovery at this and other points in the structure. In addition, the deflection which did not show at least 80 percent recovery is not considered significant since the instrumented reinforcement stresses were very low and no cracking was observed in the mapping areas.

The deflections met both the Grand Gulf and the ASME Boiler and Pressure Vessel Code, Section III Division 2, Article CC-6000, tolerance band criteria with the exceptions noted in the measured data. Figure 3.8-120 shows the maximum deflections plotted against the predicted response.

e. Evaluation of Test Results

The limits of structural response are best revealed by the instrumented reinforcement strains and the absence of cracking in the mapping areas. These results indicate that the structure was not stressed as much as predicted and responded in the elastic stress range. In addition, no visible signs of permanent damage to either the concrete or liner were detected. Therefore, the design and construction of the drywell is sufficient to maintain the design pressure load.

3.8.3.7.2.2 Structural Leakage Rate Test

The structural leakage rate test was performed immediately after the hold period at 30 psig. The structure was allowed to leak with no addition of air volume until the pressure inside the drywell reached 22.5 psig. Readings were taken at equal time intervals and the average leakage was recorded.

a. Test Instrumentation

   Two dial-type precision pressure gauges were used to record the internal pressure of the structure at the time intervals chosen for the test. A Texas Instruments precision pressure quartz gauge was used during the test to measure air flow rate. These gauges were properly calibrated prior to the test. The combined use of these two sets of instruments enabled the observers to measure the leakage rate of the drywell.

b. Leakage Rate

   The measured structural leakage rate was a maximum of 3200 scfm which is substantially less than the allowable rate of 84,000 scfm. In accordance with subsection 6.2.6.5.1, the structure meets the requirements for high pressure leaktightness.

3.8.4 Other Seismic Category I Structures

3.8.4.1 Description of the Structures

The general arrangements and elevations of all other seismic Category I structures are shown in Figure 3.4-1.

These seismic Category I structures include:

a. Auxiliary building

b. Diesel generator buildings

c. Enclosure building

d. Control building

e. Standby service water cooling towers and basins

f. Structures housing or supporting seismic Category I equipment

g. Diesel generator fuel oil storage tanks

Other nonsafety-related structures which have undergone a seismic analysis for safety-related reasons include:

a. Turbine building

b. Radwaste building

3.8.4.1.1 Seismic Category I Structures

3.8.4.1.1.1 Auxiliary Building

The auxiliary building is classified as a seismic Category I structure which completely encircles the containment, enclosing the containment from base mat to mid-height as indicated in Figures 3.8-83 and 3.8-84.

The building is physically separated from the adjacent structures except as described in subsection 3.7.2.1.1.3.1 and noted on Figure 3.8-77. A horizontal waterstop is provided between the control, turbine, and containment foundations and the auxiliary building foundation (Figure 3.8-86). Vertical waterstops are placed in the walls at various locations to provide watertightness. The auxiliary building houses the auxiliary systems, both normal and emergency, of the nuclear steam supply system and the fuel-handling facilities. The normal systems include the residual heat removal (RHR) system, reactor core isolation cooling system (RCIC), part of the control rod drive system (CRD), and the fuel pool cooling and cleanup system. The emergency auxiliary systems include the residual heat removal (RHR), high-pressure core spray (HPCS), low-pressure core spray (LPCS), and the standby gas treatment system (SGTS). The building also houses:

a. Electrical and instrumentation piping penetration rooms

b. Ventilation systems for the auxiliary building, containment, fuel handling area, and standby gas treatment

c. Electrical equipment such as load centers (LC), motor control centers (MCC), and both the normal and emergency cable trays

d. Normal and emergency process piping

The auxiliary building, which is founded on the Catahoula Formation, consists of a reinforced concrete base mat and reinforced concrete/structural steel sub- or superstructure. The building is a multilevel structure with floor slabs and transmits the loadings, through reinforced concrete walls, structural steel columns, or composite columns to the foundation. There are substitutions of reinforced masonry block walls. These occur only where walls are not utilized as shear walls, are used only for shielding/protection purposes, or are used for removable partitions for the installation of equipment. The design and construction of these walls were performed in accordance with "National Concrete Masonry Association" (NCMA) Specification (1970) (Ref. 16), with allowable stresses and analysis techniques meeting the intent of the NRC's "SEB Interim Criteria for Safety-Related Masonry Wall Evaluation," Revision 1 (Ref. 18), whenever the safety of seismic Category I systems and components is involved. The floor and roof systems are of composite construction, utilizing structural steel framework and reinforced concrete slabs.

Typical plan, sections, and details of the auxiliary building reinforced concrete structure are shown in Figures 3.8-77 through 3.8-85.

Removable sections in the auxiliary building consist of three types:

a. Walls (watertight) that resist heavy lateral pressures

   Steel framing with bolted connections is used with horizontally and vertically reinforced solid concrete block walls. A 3/8 inch thick cement mortar joint is used at each course of concrete blocks. This system is sandwiched between steel cover plates to form the removable section.

b. Walls that resist normal loadings

   Steel framing with bolted connections is used with solid concrete block walls without horizontal and vertical reinforcement and mortar. This system is sandwiched between steel cover plates to form the removable section.

c. Hatch covers

Hatch covers within the building may be constructed of steel or concrete. Hatch covers located on the roof are constructed of reinforced concrete and are protected from tornado missiles.

The building contains massive reinforced concrete spent fuel and cask storage pools. The pools have thick concrete walls and floors that are lined on the inside surfaces with stainless steel liner plates. The leaktightness is assured by means of a leak chase system at each field-welded plate joint, and the system is connected to the leak detection system. This system enables continual surveillance to be maintained for leaktightness.

The cask washdown area and the new fuel storage vault are also part of this massive concrete system which is a monolithic part of the auxiliary building. The arrangement of the spent fuel and cask laydown pool is shown in Figure 3.8-85.

The 150-ton fuel cask crane is supported by reinforced concrete walls which straddle the spent fuel pool bay and transmit the load to the foundation through the exterior wall and structural steel columns embedded in concrete.

The floors of the auxiliary building are utilized as follows:

Floor Elevation	Function
93'-0”	Heat exchangers and pumps, small equipment, tanks, and sumps
119'-0”	RWCU pump room, switchgear rooms
139'-0”	Steam tunnel, steam lines, monitoring equipment, motor control centers, and railroad
166'-0”	Steam tunnel roof, bottom of spent fuel pool, and cask storage area
185'-0”	Cask washdown area, new fuel storage area, demineralizers, and fuel transfer canal roof on east side
208'-10”	Top of refueling pool, top of RHR enclosures, steam tunnel shaft above east roof, and small Category I equipment
228'-0”	Intermediate roof

245'-0"	Roof above demineralizers and HVAC units
266'-0"	Roof above refueling pool

The design loads, load combinations, and allowable stresses are those specified in subsections 3.8.6.2 and 3.8.6.3. The ultimate strength design method is used for reinforced concrete design, and the working stress design method is used for steel design in accordance with the applicable codes and regulations described in subsection 3.8.3.2.

3.8.4.1.1.2 Seismic Category I Electrical Manholes

The seismic Category I electrical manholes are located in the yard between the standby service water cooling tower and the control building. These manholes are reinforced concrete structures buried in structural backfill and are provided with a ductile cast iron manhole cover for access. The seismic Category I electrical manholes and duct banks are shown in Figure 3.8-88.

3.8.4.1.1.3 Seismic Category I Electrical Duct Banks

The duct banks are located in the yard between the standby service water cooling towers and the control building, and the diesel generator building and the diesel fuel oil storage tanks. These ductbanks consist of reinforced concrete which encloses the electrical conduits and provides protection from tornado missiles. The duct banks are located a minimum of 4 feet below finished grade in structural backfill for missile protection. Where this requirement is not met, extra concrete thickness is provided at the top of the ductbanks.

3.8.4.1.1.4 Standby Service Water (SSW) Cooling Tower Basins

The SSW cooling tower basins are physically separated by a distance of 40 feet to maintain redundancy as shown in Figure 1.2-1.

Each structure is composed of the following major portions (see Figures 3.8-89 through 3.8-100):

a. A makeup water storage basin consisting of a base slab, exterior walls, interior walls and columns to support the cooling tower, a cover slab, a pumphouse, and a valve room.

b. A pumphouse supported on the basin roof slab houses the SSW and HPCS pumps and related equipment and piping. It consists of a concrete operating floor slab and exterior walls and roof to protect the seismic Category I equipment from tornado winds and missiles.

c. The pipe and valve room consisting of a reinforced concrete enclosure structure above the basin cover slab to protect the seismic Category I piping and valves from tornado winds and missiles.

d. The mechanical draft SSW cooling towers supported by the basin columns and interior walls and consisting of concrete exterior walls, interior walls, columns and beams, a roof slab, and four concrete fan stacks with steel grating over each fan stack top to protect the fans from tornado entrained debris (see Figure 3.8-116). The stacks are designed to provide full horizontal missile protection for the fans.

The entire structure is built of reinforced concrete, designed to withstand the loads and load combinations as specified in subsections 3.8.6.2 and 3.8.6.3, using the ultimate strength design method for reinforced concrete and working stress design for miscellaneous structural steel as outlined in subsection 3.8.6.2.1 and in accordance with the codes and regulations described in subsection 3.8.3.2.

3.8.4.1.1.5 Control Building

The control building is a reinforced concrete structure designed to seismic Category I requirements. It is located adjacent to the turbine and auxiliary buildings. The building location, floor plans, and sectional views are shown in Figures 3.8-101 through 3.8-110.

The dimensions of the building are approximately 142 feet by 96 feet with six stories, two of which are below grade elevation.

The floors of the control building are utilized as follows:

Floor Elevation	Function
93'-0” (base mat)	Access control and hot machine shop

111'-0”	Safeguard, switchgear, and battery rooms
133'-0”	HVAC equipment room
148'-0”	Lower cable spreading, computer, and turbine generator control rooms
166'-0”	Main control room
177'-0” (mezzanine)	Viewing gallery
189'-0”	Upper cable spreading room

All floors are composed of composite sections of concrete slabs resting on steel beams. Most of the interior walls are 8-inch thick reinforced concrete block masonry walls.

The external walls below grade are three feet thick and are physically separated from the turbine and auxiliary building walls by 1-1/2 inches of compressible material. The exterior walls above grade are a minimum of two feet thick and are separated from the adjacent turbine and auxiliary buildings by a minimum of 1-1/2 inches of compressible material.

Except as noted in Section 3.8.4.4.5, the exterior openings (doors and louvers) of the control building are equipped with seals which make the openings capable of withstanding differential atmospheric pressures of 3 psi. In the event of imminent extreme environmental conditions, the openings will be sealed to prevent depressurization of the building.

The louvers that are exposed to externally generated missiles are protected by 24-inch thick concrete barriers.

3.8.4.1.1.6 Diesel Generator Building

The diesel generator building houses three emergency diesel generators and is designed as a seismic Category I building. The general configuration and orientation of the building is shown in Figure 3.8-111.

The building is a three-celled (one for each generator), one story, concrete structure founded on structural backfill. The approximate dimensions of the building are 94'-4 1/2” x 110' x 38'

high. The east wall of the building is located 5'-6” to the west of the auxiliary building with the space between used as a piping and electrical chase.

The exterior and interior walls of the building are reinforced concrete, 2 feet thick.

The platforms at El. 158'-0” are designed as composite sections of concrete slabs resting on steel beams. he roof is a 2-foot thick reinforced concrete slab supported on structural steel beams spanning between concrete walls and designed as composite sections. The building is physically separated from the auxiliary building by a 2-inch expansion joint.

Concrete missile barriers, 24 inches thick, are provided to protect louvers, doors, and other vulnerable areas against tornado generated missiles.

3.8.4.1.1.7 Enclosure Building

The enclosure building is a limited leakage, steel-framed, seismic Category I structure with uninsulated metal siding and insulated roof deck. It completely encloses the portions of the containment above the auxiliary building roof levels and is designed and constructed to ensure that an acceptable upper limit of leakage of radioactive materials into the environment would not be exceeded even in the unlikely event of a loss-of-coolant accident.

The structural steel frame is supported solely by struts founded on the containment shell, thus maintaining structural autonomy of the structure during an earthquake. An internal inspection platform allows access around the periphery of the building.

To maintain the required leakage limits, a flexible seal is provided around the entire periphery of the enclosure/auxiliary building interface. This seal is designed to absorb all anticipated differential seismic movements between the containment and the auxiliary building without loss of the seal's leaktight integrity. During the aforementioned movement, the seal will only transmit a negligible force between the two structures, thus maintaining their seismic autonomy.

The enclosure building is designed to retain its limited leakage function during a LOCA; however, it is not designed to maintain this function during a design basis tornado which is not

postulated to occur simultaneously with a LOCA. The enclosure building superstructure will not collapse onto surrounding structures during a tornado. The following is assumed to occur during a tornado:

a. The exterior siding of the enclosure building is a two-span design connected with 1/8-inch diameter rivets spaced at 14-inch and 15-inch centers. At wind pressures associated with a tornado, 1/8-inch diameter rivets fail in shear and the siding panels are stripped away.

   Vendor tests were performed to demonstrate that the siding and its connections could withstand normal wind pressures, but would strip away at the wind pressures identified for a tornado.

   Therefore, for the design of the enclosure building, it is assumed that the wind acts on the naked steel frame. The members on the leeward side of the containment are shielded by the containment.

b. The enclosure building structural framing and siding are not designed to resist tornado generated missiles. The worst condition is a tornado missile penetrating the enclosure building and impacting the containment dome which is designed for the full spectrum of tornado missiles as discussed in subsections 3.5.1.4 and 3.8.1.3.4. impact of a tornado missile with members of the enclosure building have been discounted for the following reasons:

   1. Due to the redundancy of the steel frame, reaction loads due to tornado missile impact would be distributed throughout the frame. Therefore, reaction loads at the enclosure building support locations on the containment dome would be less than those of a direct impact of a tornado missile on the dome.

   2. Direct impact of a tornado missile on one of the structural framing members could cause severe deformations or failure. However, due to the redundancy of the steel members, even the local collapse of a column would result in a redistribution of load to the other members and no catastrophic collapse of the building could occur.

3. The generation of new missiles, due to the impact of tornado missiles on the structural steel frame of the enclosure building, is possible. However, the energy absorbed during the impact and subsequent elasto-plastic deformation prior to the release of the member, as a missile, would diminish its effects below those of a direct impact of a tornado missile on the dome.

Typical details of the enclosure building are shown in Figures 3.8-112 through 3.8-114.

The annulus between the containment and enclosure building will be maintained at a slightly negative pressure, approximately 1/4 inch of water during the operation of the standby gas treatment system. The air within the annulus is collected and passed through charcoal filters; part of it is returned to the enclosure building and the balance is released through the vent stack.

The loads, loading combinations, and allowable stresses are those specified in subsections 3.8.6.2 and 3.8.6.3.

The working stress design method is used for steel design in accordance with the codes and regulations described in subsection 3.8.3.2.

3.8.4.1.2 Non-seismic Category I Safety-Related Structures

3.8.4.1.2.1 Turbine Building

The turbine building is a non-Category I structure which is situated adjacent to seismic Category I structures. The building is constructed of structural steel with metal roof decking and exterior siding above the operating floor at El. 166'-0” and of reinforced concrete below El. 166'-0”.

The turbine building, including the foundation, is designed to comply with seismic Category I requirements and tornado effects (except the siding and its supporting members) and thus will have no adverse effects on the adjacent Category I structures. Any missiles engendered during a tornado will be no greater than those postulated in subsection 3.5.1.4.

3.8.4.1.2.2 Radwaste Building

The radwaste building is classified as a non-Category I structure which is partially embedded in soil. The entire building has been designed for seismic conditions per NRC Branch Technical Position ETSB No. 11-1 (Revision 1).

This design prevents uncontrolled release of radioactivity from hot waste material to the environment.

The building is physically separated from adjacent structures. A horizontal waterstop is provided between the turbine and radwaste building foundation, and vertical waterstops are placed on the walls of the two buildings to provide watertightness.

The radwaste building houses liquid radwaste, solid radwaste, and offgas systems and the various equipment and tanks associated with the offgas systems. This equipment includes various drain collection and sampling tanks, charcoal adsorbers, dryers, pumps, etc.

The foundation is a reinforced concrete mat that rests on the Catahoula Formation. The superstructure is supported by reinforced concrete walls and structural steel columns.

The building is a multilevel structure with floor slabs which transmit loadings to the foundation through reinforced concrete walls or structural steel columns. There are substitutions for reinforced masonry block walls which occur only where the walls are not utilized as shear walls or where the walls are used only for shielding/protection purposes, or for removable partitions.

The design and construction of concrete masonry walls are performed in accordance with the National Concrete Masonry Association (NCMA), Specification for the Design and Construction of Load Bearing Concrete Masonry, 1970 and the Masonry Design Manual, published by the Masonry Industry Advancement Committee, 1972.

The floor and roof systems are of composite construction, utilizing structural steel framework and reinforced concrete slabs. Removable sections in the radwaste building consist of three types:

a. Walls That Resist Normal Loadings

Steel framing with bolted connections is used with solid concrete block walls that have neither horizontal or vertical reinforcement nor mortar. This system is sandwiched between steel cover plates to form the removable section.

b. Walls Providing Access to Large Tank Rooms

Solid concrete block walls (with mortar) are built into the openings without horizontal or vertical reinforcement.

c. Hatch Covers

Hatch covers may be constructed of steel or concrete.

The design loads, load combinations, and allowable stresses are those specified in subsections 3.8.6.2 and 3.8.6.3 with the exception of loads pertaining to tornado missile design, building depressurization due to the tornado, and seismic design of concrete masonry walls.

The ultimate strength design method is used for reinforced concrete design and the working stress design method is used for steel design in accordance with the applicable codes and regulations described in subsection 3.8.3.2.

3.8.4.1.2.3 Radwaste Building Pipe Tunnel

The pipe tunnels are underground structures which run from the auxiliary to the radwaste buildings. The tunnels are located under the auxiliary, turbine, and radwaste building base mats. Also, a portion of the tunnel runs parallel to and outside of the radwaste building. The tunnels are separated from the building foundations with a horizontal, compressible material (cork) and a continuous waterstop. The tunnels consist of reinforced concrete walls and a base slab. At the outside of the radwaste building, it takes a box shape with a monolithic top cover.

The tunnel base slab rests on the Catahoula Formation and the sides are covered with either compacted backfill or mass concrete.

This structure houses nonsafety-related piping, electrical conduits, ducts, and sumps. The sumps are lined with stainless steel plates and the entire system is waterproofed.

Vertical loads, lateral forces, and hydrostatic loads are considered for the design of the pipe tunnel. The structure is considered as seismic Category I, even though it houses nonsafety-related components.

3.8.4.2 Applicable Codes, Standards, and Specifications

All other seismic Category I structures are designed in accordance with the regulations, codes, standards, and specifications listed in subsection 3.8.3.2 plus the following:

Regulatory Guide 1.76, Design Basis Tornado for Nuclear Power Plants

Regulatory Guide 1.102, Flood Protection for Nuclear Power Plants

International Conference of Building Officials," Uniform Building Code," 1970

American Welding Society, AWS D1.1, 1972, later editions that are adopted for use are specified in the component's/ structure's specification or standard with the following additional exception to those noted in subsections 3.8.1.2.2.1 and 3.8.3.2.2.1: for materials less than 3/16-inch thick, weld undercut shall be allowed only within the limits of good industry standard welding practices such that section thickness is not significantly reduced.

The enclosure building is designed in accordance with the regulations, codes, standards, and specifications listed in subsection 3.8.3.2, except that effects of tornado generated missiles are not considered.

3.8.4.3 Loads and Loading Combinations

All other seismic Category I structures, except the enclosure building, are designed for the loads and loading combinations given in subsection 3.8.6, Structural Design Criteria for Category I Structures.

The enclosure building is designed for the loads and load combinations as described in subsection 3.8.6, except that tornado generated missile loads are not considered.

3.8.4.4 Design and Analysis Procedures

The computer programs used for the design and analysis of other seismic Category I structures are discussed in Appendix 3D of this FSAR.

3.8.4.4.1 Auxiliary Building

The design and analysis of the auxiliary building structure is done in accordance with the regulations, codes, standards, and specifications listed in subsection 3.8.4.2. These codes, etc., were reviewed for the extent of their compliance with ACI 318-71 for concrete structures and with the AISC Steel Manual (1969) for steel structures.

The auxiliary building is designed and analyzed as follows:

a. Reinforced Concrete Foundation

   The design and analysis procedures of the auxiliary building foundation are discussed in subsection 3.8.5.4.2.

b. Structural Steel Columns

   In most cases, structural steel columns are encased in concrete or surrounded by reinforced concrete walls. In general, the vertical design load is transferred totally to the structural steel column; however, in a few cases, the vertical loads are considered supported by a composite system, and a portion of the load is transferred to the concrete encasement or wall while the remainder of the load is transferred to the steel column.

   The AISC steel column design program, CSTOR, is used to analyze the structural steel columns and base plates.

c. Reinforced Concrete Bearing Walls

   1. Exterior Walls

      At the subgrade level, the exterior walls of the auxiliary building are designed for lateral pressures from structural backfill, seismic forces, depressurization loads due to the tornado, and design flood water effects as well as vertical loads.

Above finished grade, the exterior walls are designed for seismic forces, tornado depressurization loads, and tornado missile loads as well as the vertical loads. The exterior walls above finished grade have a minimum thickness of 2 feet which is in accordance with subsection 3.5.3 of this FSAR The Structural Engineering Systems Solver (STRESS, CE-309) and the Structural Design Language (STRUDL, CE-901) computer programs were used along with manual calculations to design the auxiliary building exterior walls.

2. Interior Walls

The auxiliary building interior walls act in combination with the exterior walls to resist shear loads. The interior walls are designed for seismic, tornado depressurization, jet impingement, temperature, and other lateral loads (pressures or suctions) as well as vertical loads. Most of the walls are a minimum of 2 feet thick for shielding purposes.

Only continuous reinforced concrete walls are considered to provide shear resistance. Steel columns and frames are assumed to provide negligible shear resistance. The interior walls were designed for shear by manual, longhand methods.

d. Reinforced Concrete Floors and Structural Steel Framing

Each member of the structural steel beam/girder system is designed as a simple span member to carry the concrete slab dead load and other construction loads. The composite section of the concrete slab acting integrally with the steel beam is designed to resist live loads.

The structural concrete floors are designed as one-way slabs spanning between the steel beams in accordance with ACI 318-71.

e. Structural Steel and Reinforced Concrete Roof and Missile Barrier Walls

The steel beams and structural concrete for the auxiliary building roof are designed using the same analyses and procedures of part d. above. The concrete roof thickness

is a minimum of 2 feet to resist tornado missiles in accordance with subsection 3.5.3. Missile barrier walls are similarly a minimum of 2 feet thick to resist tornado missiles.

3.8.4.4.2 Seismic Category I Electrical Manhole Design

The seismic Category I electrical manholes are buried in structural backfill and designed for the lateral dynamic earth pressures due to the governing seismic event as well as normal lateral earth pressure loads and a surcharge of 250 pounds per square foot.

The walls are 12 inches thick and the removable cover slab is 2 feet thick to protect the electrical cables against tornado missiles.

3.8.4.4.3 Seismic Category I Electrical Ductbanks

A concrete duct bank encloses the 6-inch diameter conduits located in the yard.The stresses in the conduits are within the allowable limits for bending stresses due to seismic shear waves and also within the allowable limits for axial stresses due to seismic compression and shear waves. These forces are developed to the design earthquake using techniques in accordance with Chapter 6 of Bechtel Topical Report BC-TOP-4, Rev. 1, including Addendum 1.

To protect against tornado-borne missile damage to the structure, the top of the duct bank is located a minimum of 4 feet below finished grade. When this 4-foot minimum earth cover requirement is not satisfied, the concrete on top of the duct bank is thickened to give an equivalent amount of cover.

Ductbanks at seismic Category I building interfaces and in the free field are designed as beams on elastic foundations.

Minimum percentages of reinforcement satisfying the provisions of ACI 318-71 are provided in the ductbanks. This is because the ductbanks act only as enclosures for the conduits and because the calculated axial and bending stresses from seismic compression and shear waves are low. The structural acceptance criteria is presented in subsection 3.8.4.5.

3.8.4.4.4 Standby Service Water Cooling Tower Basin

The standby service water cooling tower basin was analyzed elastically by computer using a three-dimensional finite element model compatible with the ICES STRUDL computer program. The structure was partitioned into two major substructures for analysis: The basin structure below plant grade (from El. 78'-6" to 133'-0") and the cooling tower structure above plant grade (from El. 133'-0" to 181'-0"). Appropriate boundary conditions were imposed at the level of partitioning between the two substructures to model the monolithic behavior of the real structure.

The basin and cooling tower structures are doubly symmetrical.

Finite element models were developed for one-half of the total structure to allow analyses of structural behavior for both vertical and lateral loadings simultaneously, using one model. Areas of local discontinuity such as the pumphouse, valve room, and pump pit are analyzed using classical techniques and superimposed elastically with results from the computer analysis to obtain final results for structural behavior in these areas.

The SSW cooling tower basin is analyzed elastically as a buried, monolithically constructed, reinforced concrete structure supported on an elastic foundation (Catahoula Formation). Vertical loads from the cooling tower and the basin are carried to the base mat by columns, beams, interior walls, and exterior walls. Lateral loads are transmitted to the foundation by columns, interior walls, beams, and exterior walls.

The structural design of the standby service water cooling tower basin is based upon strength design concepts in accordance with ACI 318-71, Building Code Requirements for Reinforced Concrete.

Miscellaneous and embedded structural steel is designed using allowable stress concepts in accordance with the AISC Specification for Steel Structures.

3.8.4.4.5 Control Building

The control building has been designed as a monolithically constructed, reinforced concrete structure founded on the Catahoula Formation. The analysis techniques include classical beam and plate theory. The design was performed in accordance with ACI 318-71, Building Code Requirements for Reinforced Concrete,

for concrete structures and with the Specification for Design, Fabrication and Erection of Structural Steel for Buildings, AISC, 1969, for steel structures.

All elevated floors were designed as composite beams with concrete slabs generally 1'-0" thick resting on steel beams. All floor slabs were also designed as rigid diaphragms to transmit lateral loads to the external shear walls.

The control room ceiling system at El. 177'-0" was designed to withstand seismic loadings utilizing a structural steel space truss system. The roof is designed as a composite section with a 2-foot thick slab which is sufficient to resist the most critical load case including tornado generated missiles.

The external walls below grade (3 feet thick) were designed as retaining walls to resist earth and hydrostatic pressures as well as shear walls to resist seismic loads. Exterior walls above grade are 2 and 3 feet thick and are designed to resist the most critical loading case as well as tornado generated missiles. All exterior walls are designed as shear walls to transmit lateral loads to the foundation. All vertical loads from roof and floor slabs are transmitted to the foundation through the exterior walls and interior steel columns.

The control building is designed as a partially sealed building for tornado pressure drop to comply with the extreme environmental condition. A design pressure of 3 psi was used for all exterior walls. Floor drains in the upper and lower cable spreading rooms and a select number of penetrations between the Unit 1 side and the Unit 2 side of the control building may be left open for drainage and cable pulling operations. The structural effects of the tornado pressure drop resulting from these open penetrations were investigated. The resulting differential wind pressures were evaluated during the design of the interior walls. Concrete missile barriers 2 feet thick are provided to protect all louvers and other vulnerable openings against tornado generated missiles except as noted in Table 3.5-8.

The steel and concrete elements of the control building were designed using classical methods of analysis. The composite steel beams and columns were designed elastically for the loads and load combinations of subsection 3.8.6.3 in accordance with the Specification for Design, Fabrication, and Erection of Structural Steel for Buildings, AISC, 1969. The concrete walls, slabs, and

foundation were designed using ultimate strength techniques for loads and load combinations of subsection 3.8.6.2 in accordance with the Building Code Requirements for Reinforced Concrete, ACI-318-71. Concrete block masonry unit (CMU) construction was used for interior walls. These are non-load bearing walls (with a few exceptions where walls support roofs over isolated rooms). The design and construction of these walls were performed in accordance with "National Concrete Masonry Association" (NCMA) Specification (1970) (Ref. 16), with allowable stresses and analysis techniques meeting the intent of the NRC's "SEB Interim Criteria for Safety-Related Masonry Wall Evaluation," Revision 1 (Ref. 18), whenever the safety of seismic Category I systems and components is involved. The Uniform Building Code (Ref. 17) earthquake criteria for Zone 1 was used in the design of non-Category I CMU walls.

3.8.4.4.6 Diesel Generator Building

The diesel generator building has been designed as a monolithically constructed, reinforced concrete structure supported on a structural backfill foundation. The analysis techniques include classical beam and plate theory. The design was performed in accordance with ACI 318-71, Building Code Requirements for Reinforced Concrete, for concrete structures and with the Specification for Design, Fabrication and Erection of Structural Steel for Buildings, AISC, 1969, for steel structures.

The diesel generator building roof was designed as composite beams with 2-foot thick concrete slabs resting on steel beams. The concrete roof serves as a horizontal diaphragm which transfers lateral loads to the exterior and interior walls which are designed as shear walls to transmit lateral loads to the foundation. Vertical loads from the roof and mezzanine slabs are transmitted through the concrete walls to the foundation.

The 2-foot thick roof slab and exterior walls are sufficient to resist the critical load case which includes tornado generated missiles.

The 2-foot thick interior concrete walls which separate the cells housing the diesel generators are designed to resist building depressurization loads as well as the critical load combination.

The diesel generator building is designed as a partially vented building for the pressure drop due to the tornado to comply with extreme environmental conditions. A design depressurization load of 3 psi was used for the walls for a conservative design.

Two-foot thick concrete missile barriers are provided to protect louvers, doors, and other vulnerable interior areas against tornado generated missiles.

The steel and concrete elements of the diesel generator building were designed using classical methods of analysis. The composite steel beams were designed elastically for the loads and load combinations of subsection 3.8.6.3 in accordance with the Specification for Design, Fabrication and Erection of Structural Steel for Buildings, AISC, 1969.

The concrete walls, slabs, and foundation were designed using the ultimate strength techniques for the loads and load combinations of subsection 3.8.6.2 in accordance with the Building Code Requirements for Reinforced Concrete, ACI 318-71.

3.8.4.4.7 Enclosure Building

The enclosure building was designed as a space truss supported by the containment building. A continuous seal between the auxiliary and enclosure buildings is provided by a flexible seal. The structure is assumed to be pin-connected to the containment structure, and the attachment to the auxiliary building roof is used for sealing purposes and carries a negligible load.

The analysis was performed using finite elements in the Static and Dynamic Structural Analysis Systems (STARDYNE) computer program.

The design of the enclosure building was performed in accordance with the elastic methods of the Specification for the Design, Fabrication and Erection of Structural Steel for Buildings, AISC 1969.

The structural framing of the enclosure building is designed to maintain its structural autonomy during all load conditions except for tornado missiles; however, the structure may be stripped of its siding during a tornado. For design, it is assumed that the naked steel frame plus 1/4 of the windward siding will experience the full tornado wind force. The structural framing and siding is not designed to resist tornado generated missiles.

A missile penetrating the enclosure building will strike the concrete containment or the auxiliary building which are designed to resist tornado generated missiles. The allowable stress for the structural steel used in the enclosure building is increased by 33 percent for tornado and hurricane wind loads, but in no case does the allowable stress exceed 0.90 F_y for bending, 0.85 F_y in axial tension, and 0.5 F_y in shear.

Explosion at the hydrogen storage facility will result in stripping of the enclosure building siding and pressure loads applied to the naked steel framing. These loads are bounded by the existing considerations for a tornado. Therefore, based on siting considerations for the bulk hydrogen storage vessels, no special evaluation of the enclosure building steel or its interface with the containment dome are required.

3.8.4.4.8 Diesel Generator Fuel Oil Storage Tanks

The diesel generator fuel oil storage tanks are buried in structural backfill. The bottom elevation of the tank is 110'-0”. The top of the tank at El. 122'-0” has a minimum of 10 feet of cover to protect the tanks from tornado missiles.

A shaft constructed from a 6-foot diameter corrugated metal pipe provides access to the pump. This access shaft is protected from tornado-borne missiles by a 2'-4 1/2” thick reinforced concrete slab provided with a manhole cover. Details of the buried tanks, shafts, and protective slabs are shown in Figure 3.8-87.

3.8.4.5 Structural Acceptance Criteria

All other seismic Category I structures are designed for structural acceptance criteria as defined in subsection 3.8.6.1.5 and shown in subsections 3.8.6.2 and 3.8.6.3.

3.8.4.6 Materials, Quality Control, and Special Construction Techniques

3.8.4.6.1 General

Specifications covering all materials are in sufficient detail to assure the structural design requirements of the work.

3.8.4.6.2 Concrete

a. All concrete materials are approved prior to start of construction on the basis of their characteristics in test comparisons using ASTM standard methods.

Concrete aggregates and cement, conforming to acceptability criteria of the specifications, are obtained from approved sources.

Concrete properties are determined by laboratory tests. Concrete admixtures are used to minimize the mixing water requirements and increase workability. The specified compressive strength at 91 days, or earlier, is:

Structure or Component	Specified Strength (f'_c) (psi)
Precast concrete panels	4,000
All other seismic Category I concrete	4,000
Turbine pedestal	3,000
Turbine building floor slabs Unit 1	3,000
Turbine building foundation and walls	4,000
Radwaste building	4,000
Mass concrete fill	2,000
Electrical duct encasement	2,000

b. Water used in mixing of concrete, mortar, and grout is tested in accordance with the provisions of American Association of State Highway Officials (AASHO), Standard T26-70, Standard Method of Testing for Quality of Water to be Used in Concrete.

Acidity or alkalinity (pH factor) is within the range of 6.0 to 8.5, and the water does not contain impurities in excess of the following limits:

	Percent	ppm
Total chlorides as Cl	0.025	250
Total sulphates as SO_4	0.10	1,000

Total organic solids	0.05	500
Total inorganic solids	0.20	2,000

c. All concrete work is in accordance with ACI 318-71, Building Code Requirements for Reinforced Concrete, except as modified in subsection 3.8.1.2.2.2 and complies with the following specifications:

Material	ASTM Specification
Cement, Type II low alkali	C 150-72
Normal aggregate	C 33-71a
Heavy aggregate	C 637-73 and C 638-73
Fly ash	C 618-73 (see note below)
Air-entraining admixture	C 260-69
Water reducing admixture	C 494-71
RPV shield wall fill concrete aggregate	C 637-73

Note: Exceptions to ASTM C 618-73: in-process testing frequency shall be for each 200 tons in accordance with ANSI N45.2.5, Table B; in addition, passing the Pozzolanic Activity Index with either cement or lime shall be acceptable.

3.8.4.6.3 Reinforcing Bars

Reinforcing bars for concrete consist of deformed bars meeting the requirements of ASTM 615-75, (including Supplementary Requirements S-1), Grade 60, with additional requirements as set forth below for tension and bend testing.

All test specimens are full-diameter bars, as rolled.

One tension test is made for each 50 tons, or fraction thereof, of each bar size of reinforcing bars produced from each heat and production rolling for the determination of tensile properties. Acceptance standards for tension testing are in conformance with the requirements of Table 2, ASTM A 615-75, including the retest provisions of Section 11, ASTM A 615-75.

One bend test is made for each test lot and production rolling of each bar size of each heat for the determination of bend properties. Acceptance standards for bend testing are in conformance with the requirements of Table 3 and Supplementary Requirements S-1, ASTM A 615-75.

Normally, testing is accomplished at the mill or shop of the supplier; the owner's representative has the option, at any time, of witnessing the selection, cutting, and testing of test specimens. Any of the required testing which has not been accomplished prior to receipt of the reinforcing bar at the construction site is accomplished as part of the receiving inspection process at the construction site.

3.8.4.6.4 Structural Steel

The structural steel materials conform to all applicable requirements of the AISC Manual of Steel Construction, unless otherwise noted in the job specifications or on the drawings, and also complies with the following specifications:

Material	ASTM Specification
Structural steel and connections	A 36
High strength structural steel	A 441, A 572, A 516, A 540
Tubular steel sections	A 53, Type F or S
Anchor bolts	A 36, A 449
Shop fabrication	AWS D1.1 welding or bolting, A 325 and A 490 bolts
Field erection	High strength bolts, A 325 and A 490, or welding AWS D1.1

3.8.4.6.5 Control Tests for Concrete

a. [HISTORICAL INFORMATION] [Sample Tests

Samples for the following routine in-process tests of concrete are taken at the sampling point, at the frequencies shown, in accordance with ASTM C 172-71. This point shall be at the central mix plant discharge.

Correlation test samples shall be taken at the end of the last piece of conveying equipment at the point of discharge of concrete into the forms. Water shall not be added to any concrete batch after in-process tests unless the in-process tests (except unit weight/yield) are repeated after the water is added. Samples shall not be taken from concrete after it has been deposited in the forms.

1. Temperature: first batch produced each day and every 50 cubic yards placed for each class of concrete.

2. Slump (ASTM C 143-71): first batch produced each day and every 50 cubic yards placed for each class of concrete.

3. Air Content (ASTM C 231-72T): first batch produced each day and with each set of compressive strength cylinders for each class of concrete.

4. Plastic Unit Weight/Yield (ASTM C 138-71T): daily during production for each class of concrete.

5. Compressive Strength (ASTM C 31-69, tested in accordance with ASTM C 39-72): tandard 6 inch by 12 inch concrete test cylinders will be molded and tested at 7, and 28 or 91 days from each 100 cubic yards, or a minimum of one set per day for each class of concrete.

Correlation tests for air content, slump, and temperature of concrete shall be made at the end of the last piece of conveying equipment for the first batch produced each day and for each 200 cubic yards or fraction over 100 cubic yards placed of each class of concrete.

b. Concrete Cylinders

A strength test is the average of the strengths of the two specimens from each sample tested at 28 or 91 days. Only one cylinder shall be made and tested at 7 days. The provisions of ACI 318-71, paragraph 4.3.3, shall apply in determining the satisfactory strength level of the concrete tested at 28 or 91 days.]

3.8.4.6.6 Splices of Reinforcement

a. Provisions of ACI 318-71, Sections 7.5, 7.6, and 7.7, will be met for all reinforcing bar splices. Splices are designed to develop the specified minimum yield strength.

b. All mechanical splices for reinforcing bars are made by the Cadweld process, using clamping devices, sleeves, charges, etc., as specified by the Cadweld Splice Instruction Sheets for B and T series connections. C series materials are not permitted.

   Qualification of operators, visual inspection, tensile testing, tensile test frequency, and the procedure for substandard tensile test results are in accordance with the Cadweld instruction sheets, and will comply with NRC Regulatory Guide 1.10, Revision 1, dated 1/2/73.

3.8.4.7 Testing and Inservice Surveillance Requirements

Testing and inservice surveillance are not required for seismic Category I structures other than the containment, and a formal program of testing and inservice surveillance is not planned.

3.8.5 Foundations

3.8.5.1 Description of the Foundations and Supports

The foundations of all seismic Category I structures are reinforced concrete base slabs resting on the Catahoula Formation or on caissons which are embedded into the Catahoula. The electrical manholes, electrical duct banks, and the diesel generator building are founded on Category I structural backfill. The foundation of each structure is physically separated from the others.

3.8.5.1.1 Containment Structure Foundation

The containment structure foundation is a reinforced concrete slab 9 feet-6 inches thick and 134 feet in diameter. The bottom of the foundation slab is 49 feet below grade. The containment structure foundation is Shown on Figures 3.8-2 through 3.8-4.

3.8.5.1.2 Auxiliary Building Foundation

The auxiliary building, which surrounds the containment structure, is supported on a reinforced concrete base slab which measures 249 feet-0 inches by 179 feet-0 inches. The base slab is generally 6 feet thick but is 9 feet thick in certain areas, such as immediately adjacent to the containment and in the vicinity of the spent fuel pool. The bottom of the foundation extends 45 feet-6 inches to 48 feet-6 inches below finished grade level.

The auxiliary building foundation is shown in Figure 3.8-77 and is physically separated from adjacent foundations except as noted on Figure 3.8-77 and as described in subsection 3.7.2.1.1.3.1. The separation gap is filled with a compressible material which is 2 inches thick between the auxiliary and containment buildings and 1-1/2 inches thick between the auxiliary building and other surrounding structures. These separations are provided to minimize the structural interaction between the buildings.

Vertical column and wall loads, lateral soil pressure, and seismic loads are considered for the structural characteristics of the base mat. The foundation slab is founded on the Catahoula Formation, and a waterproof membrane separates the structural slab from the non-structural mud slab. The effects of this membrane are considered in the design for sliding of the auxiliary building. Construction joints in the foundation mat are vertically keyed for shearing strength.

3.8.5.1.3 Control Building Foundation

The control building foundation is a reinforced concrete slab, 7 feet thick. The bottom of the slab is 46 feet-6 inches below grade, and the building is approximately 142 feet by 96 feet. The mat is physically separated from adjoining buildings by 1-1/2 inches of compressible material to prevent seismic interaction. The control building foundation is shown on Figure 3.8-101.

3.8.5.1.4 Diesel Generator Building Foundation

The diesel generator building is supported by a reinforced concrete base slab. The slab is 5 feet thick and the bottom of the slab is 4 feet-6 inches below grade. The building is approximately 94 feet 4-1/2 inches by 120 feet 9 inches. The mat is physically separated from the auxiliary building by 1-1/2 inches of compressible material to prevent seismic interaction. The diesel generator building foundation is shown on Figure 3.8-111.

3.8.5.1.5 Standby Service Water Cooling Tower Basins

The standby service water cooling tower basins which support the standby service water cooling towers are supported on reinforced concrete base mats which are 4 feet thick and founded on the Catahoula Formation at El. 78'-6". Each base mat is 150 feet by 150 feet and is physically separated from the other by a distance of 40 feet (see Figure 3.8-89).

Each base mat contains a pump sump also founded on the Catahoula Formation at El. 69'-6". The pump sumps are monolithically connected to the basin base mat by walls 6 feet-2 inches to 10 feet-6 inches thick. The 5-foot thick pump sump base mat is monolithically connected to the pump sump walls.

3.8.5.1.6 Seismic Category I Electrical Manholes

A discussion of the foundations for the seismic Category I electrical manholes is included in subsection 3.8.4.4.2.

3.8.5.1.7 Seismic Category I Electrical Duct Banks

A discussion of the foundations for the seismic Category I electrical duct banks is included in subsection 3.8.4.4.3.

3.8.5.2 Applicable Codes, Standards, and Specifications

The applicable regulations, codes, standards, and specifications are discussed in subsection 3.8.1.2 for the containment and in subsections 3.8.3.2 and 3.8.4.2 for other seismic Category I structures.

3.8.5.3 Loads and Load Combinations

The containment structure foundation loads and load combinations are discussed in subsection 3.8.1.3. The foundation loads and load combinations for other seismic Category I structures are discussed in subsection 3.8.6, Structural Design Criteria for seismic Category I Structures.

3.8.5.4 Design and Analysis Procedures

The foundations of seismic Category I structures are analyzed using well established methods where the transfer of loads from the base slab to the supporting soil is determined by elastic methods.

3.8.5.4.1 Containment Structure

Design and analysis procedures for the containment, including the base slab, are discussed in subsection 3.8.1.4.

3.8.5.4.2 Auxiliary Building

Since the foundation mat is integrally connected to the 3-foot thick exterior walls, the mat is considered fixed and rigid around its edges. The reinforced concrete walls located on and integrally connected to the mat are analyzed as stiffeners for the foundation. The maximum allowable bearing pressure is increased by 50 percent for dynamic loadings, as discussed in subsection 2.5.4.11.1.

The design and analysis of the auxiliary building foundation was performed using the SAP (CE-779) (Structural Analysis Program) computer program and the Bechtel program CE-668, plate bending. These programs are described in Appendix 3D of this FSAR.

The auxiliary building foundation mat was designed based on the strength design concept in accordance with ACI 318-71, Building Code Requirements for Reinforced Concrete. Vertical and lateral loads are transmitted to the base mat by columns, beams, interior walls, and exterior walls.

3.8.5.4.3 Control Building

The foundation for the control building is a reinforced concrete base mat founded on the Catahoula Formation. Vertical and lateral loads are transmitted to the base mat by interior columns and the exterior walls. The lateral forces are transmitted from the structure to the foundation media by friction and adhesive forces, and the overturning moments are transmitted by the bearing forces between the base mat and the soil.

The design and analysis of the control building foundation was performed using well established, classical, longhand methods based on general principles of engineering mechanics. These calculations were checked using the STRUDL computer program which is described in Appendix 3D of this FSAR.

The foundation design was based upon ultimate strength design concepts in accordance with ACI 318-71, Building Code Requirements for Reinforced Concrete.

3.8.5.4.4 Diesel Generator Building

The foundation for the diesel generator building is a reinforced concrete base mat founded on structural backfill. Vertical and lateral loads are transmitted to the base mat by the interior and exterior walls. The lateral forces are transmitted from the structure to the foundation media by friction forces, and the overturning moments are transmitted by bearing forces between the base mat and the soil.

The design and analysis of the diesel generator building foundation was performed using well established, classical, longhand methods based on general principles of engineering mechanics.

The foundation mat was designed based on the ultimate strength design concepts in accordance with ACI 318-71, Building Code Requirements for Reinforced Concrete.

3.8.5.4.5 Standby Service Water Cooling Tower Basin Foundation

The foundation for the SSW Cooling Tower Basin is analyzed as a continuous mat foundation supported on an elastic support (Catahoula Formation).

The foundation analysis considers the mat a monolithic portion of the SSW cooling tower basin using the elastic finite element computer analysis of the basin described in subsection 3.8.4.4.4. The area of local discontinuity at the pump sump is analyzed using classical techniques and superimposed elastically with results from the computer analysis of subsection 3.8.4.4.4 to obtain final results for structural behavior in this area.

Vertical loads on the structure are transmitted to the base mat as described in subsection 3.8.4.4.4 and from the mat to the supporting Catahoula Formation by direct bearing. Lateral loads are transmitted to the base mat as described in subsection 3.8.4.4.4 and from the mat to the supporting Catahoula Formation by direct shear. Overturning moments due to lateral loads are transmitted to the base mat as couples and from the base mat to the supporting Catahoula Formation by direct bearing. The structural design of the foundation mat is done based upon strength design concept in accordance with the ACI 318-71 Building Code Requirements for Reinforced Concrete.

3.8.5.4.6 Seismic Category I Electrical Manholes

The foundations for the electrical manholes consist of 12-inch thick base slabs which were designed as two way slabs supported by the four side walls.

Overturning is not considered since the manholes are buried in structural backfill.

3.8.5.4.7 Seismic Category I Electrical Ductbanks

The seismic Category I electrical ductbanks enclose the conduits and are buried a minimum of 4 feet below finished grade. These ductbanks, through their box-shaped design, act as single foundation units as well as enclosure structures. For a discussion of the design and analysis procedures for the seismic Category I electrical ductbanks, see subsection 3.8.4.4.3.

3.8.5.4.8 Diesel Generator Fuel Oil Storage Tank Foundation

The diesel generator fuel oil storage tanks are buried in structural backfill with a minimum cover of 10 feet. The design and analysis procedures for the concrete slab protecting the access shaft are described in subsection 3.8.4.4.8.

3.8.5.5 Structural Acceptance Criteria

The foundations of all seismic Category I structures are designed to meet the same structural acceptance criteria as the structures themselves. These criteria are discussed in subsections 3.8.1.5, 3.8.6.1, 3.8.6.2, and 3.8.6.3. Factors of safety due to overturning, sliding, and flotation are discussed in subsection 3.8.6 of this UFSAR.

3.8.5.6 Materials, Quality Control, and Special Construction Techniques

The foundations and concrete supports are constructed of reinforced concrete using proven methods common to heavy industrial construction. For further discussion refer to subsections 3.8.1.6 and 3.8.4.6.

3.8.5.7 Testing and Inservice Surveillance Requirements

Testing and inservice surveillance are not required and are not planned for foundations of structures or for concrete supports.

3.8.6 Structural Design Criteria for Seismic Category I Structures Other Than Containment

The criteria given herein are applicable for design for all seismic Category I structures other than the containment.

3.8.6.1 Loads, Definition of Terms, and Nomenclatures

The following nomenclature and definition of terms apply to all the criteria given in subsection 3.8.6.

All the major loads to be encountered and/or to be postulated in a nuclear power plant are listed. All the loads listed, however, are not necessarily applicable to all the structures and their elements in a plant. Loads and the applicable load combinations for which each structure is designed depend on the conditions to which that particular structure could be subjected.

3.8.6.1.1 Normal Loads

Normal loads are those loads to be encountered during normal plant operation and shutdown. hey include the following:

D - Dead load of the structure and equipment plus any other permanent loads, including vertical and lateral pressures of liquids.

L - Live loads, including any movable equipment loads and other loads which vary in intensity and occurrence, such as forces exerted by the lateral pressure of soil.

T_o - Thermal effects during normal operating, startup, or shutdown conditions, including liner plate expansion, equipment and pipe reactions, and thermal gradients, based on the most critical transient or steady-state thermal gradient.

3.8.6.1.2 Severe Environmental Loads

Severe environmental loads are those loads that could be encountered infrequently during plant life. Included in this category are:

E - Loads generated by the operating basis earthquake (OBE). The earthquake is composed of two orthogonal horizontal components and one vertical component. The effects of the three components are combined, based on the square-root-of-the-sum-of-the-squares. Only the actual load and existing live loads are considered in evaluating the seismic response forces.

W - Loads generated by the design wind specified for the plant.

3.8.6.1.3 Extreme Environmental Loads

Extreme environmental loads are those loads which are credible but are highly improbable. They include:

E' - Loads generated by the safe shutdown earthquake (SSE). The earthquake is composed of two orthogonal horizontal components and one vertical component. The effects of the three components are combined, based on the square-root-of-the-sum-of-the-squares. Only the actual load and the existing live loads are considered in evaluating the seismic response forces.

W' - Wind load generated by the design tornado specified for the plant. This includes tornado wind pressure, differential pressure, and energy resulting from tornado missile impact.

A - Hydrostatic loads due to probable maximum flood.

3.8.6.1.4 Abnormal Loads

Abnormal loads are 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 - Design Pressure load within or across a compartment and/or building, generated by the postulated pipe rupture, including the dynamic effects due to pressure time history

T_a - Thermal effects due to thermal conditions generated by the postulated break, excluding T_o

R - Pipe reactions under thermal conditions generated by the postulated break and including piping dead load reactions with fluid

a. Load on a structure generated by the reaction on a ruptured high-energy pipe during the postulated event

b. Load on a structure generated by the jet impingement from a ruptured high-energy pipe during the postulated event

c. The energy resulting from the impact of a ruptured high-energy pipe on a structure or a pipe restraint during the postulated event

3.8.6.1.5 Other Definitions

U - Required ultimate load capacity as defined in ACI 318-71 for concrete structures

Ø - Capacity-reduction factor as defined in ACI 318-71, Section 9.2

F_s - Allowable stress for structural steel as defined in the AISC Manual for Steel Construction, 1969 (Ref. 7)

f_s Calculated stress in structural steel

F_y Yield strength for steel

3.8.6.1.6

3.8.6.2 Load Combinations and Structural Acceptance Criteria for Seismic Category I Concrete Structures

The following presents the load combinations and allowable design limits used for seismic Category I concrete structures. To assure that structural integrity will be maintained, limits on the resulting stresses and the required strength capacities are considered for normal operating conditions, including earthquake (OBE) and wind loads, and for extreme environmental conditions, including earthquake (SSE), tornado, and pipe break effects and various combinations thereof.

3.8.6.2.1 Load Combinations for Normal Operating Conditions

$$U = 1.4\ D + 1.7\ L$$

$$U = .75\ (1.4\ D + 1.7\ L + 1.4\ T_o)$$

$$U = .75\ (1.4\ D + 1.7\ L + 1.4\ T_o + 1.9\ E)$$

$$U = .75\ (1.4\ D + 1.7\ L + 1.4\ T_o + 1.7\ W)$$

For structural elements carrying mainly earthquake forces:

$$U = 1.0\ D + 1.0\ L + 1.9\ E + 1.0\ T_o$$

Subsections 9.3.4 and 9.3.5 of ACI 318 are used for the design of exterior foundation walls below grade and retaining walls.

3.8.6.2.2 Load Combinations for Severe, Extreme, and Abnormal Environmental Conditions

$$U = 1.0\ D + 1.0\ L + 1.0\ (T_a + T_o) + 1.0\ R + 1.25(E \text{ or } W) + 1.25\ P$$

$$U = 1.0\ D + 1.0\ L + 1.0\ A + 1.0\ T_o$$

$$U = 1.0\ D + 1.0\ L + 1.0\ W' + 1.0\ T_o$$

$$U = 1.0\ D + 1.0\ L + 1.0\ (T_o + T_a) + 1.0\ R + 1.0\ E' + 1.0\ P$$

The effects of tornado generated differential pressures and missiles shall be combined in accordance with BC-TOP-3A (Ref. 8).

3.8.6.3 Load Combinations and Structural Acceptance Criteria for Seismic Category I Steel Structures

The following are the load combinations and allowable design limits used for seismic Category I steel structures. To assure that structural integrity will be maintained, limits on the resulting stresses are considered for service and factored loads.

3.8.6.3.1 Normal Operating Conditions Stress Limit

	Stress Limit
$D + L + T_o$	F_s
$D + L + T_o + E$	$1.25\ F_s$
$D + L + T_o + W$	$1.33\ F_s$

For structural elements carrying mainly earthquake forces:

$D + L + T_o + E$	F_s

3.8.6.3.2 Load Combinations for Abnormal Conditions and Severe or Extreme Environmental Conditions

	Stress Limit
$D + L + R + (T_o + T_a) + E' + P$	(Note 2)
$D + L + A + T_o$	(Note 1)
$D + L + T_o + W'$	(Note 1)

Notes:

1. In no case shall the allowable stress exceed 0.9 F_y in bending, 0.85 in axial tension, and 0.5 F_y in shear. Where F_s is governed by requirements of stability (local and lateral buckling), f_s shall not exceed 1.4 F_s.

2. Stress limits shall be limited to 1.5 F_s except for web shear where stress limits shall be limited to 1.4 F_s.

3.8.6.4 Procedures for Determination of the Effect of Missile Impact on Concrete and Steel Structures

Missile barriers, whether of concrete or steel, are designed with sufficient strength to stop the postulated missiles in accordance with subsection 3.5.3.

3.8.6.5 Procedures for Design of Structural Pipe Restraint

Protection of seismic Category I structures, systems, and components from the dynamic effects of postulated high-energy pipe ruptures is discussed in Section 3.6, BC-TOP-9A (Ref. 9), and BN-TOP-2. (Ref. 10)

3.8.6.6 Factors of Safety

Minimum factors of safety for sliding, overturning, and buoyancy for seismic Category I structures are shown in Table 3.8-1.

3.8.7 References

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12. ACI Committee 318, "Building Code Requirements for

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16. Specification for the Design and Construction of Load-Bearing Concrete Masonry, National Concrete Masonry Association, August 1970.

17. Uniform Building Code, International Conference of Building Officials, Vol. I, Sec. 2314, 1970.

18. NRC "SEB Interim Criteria for Safety-Related Masonry Wall Evaluation," Revision 1, July 1981.

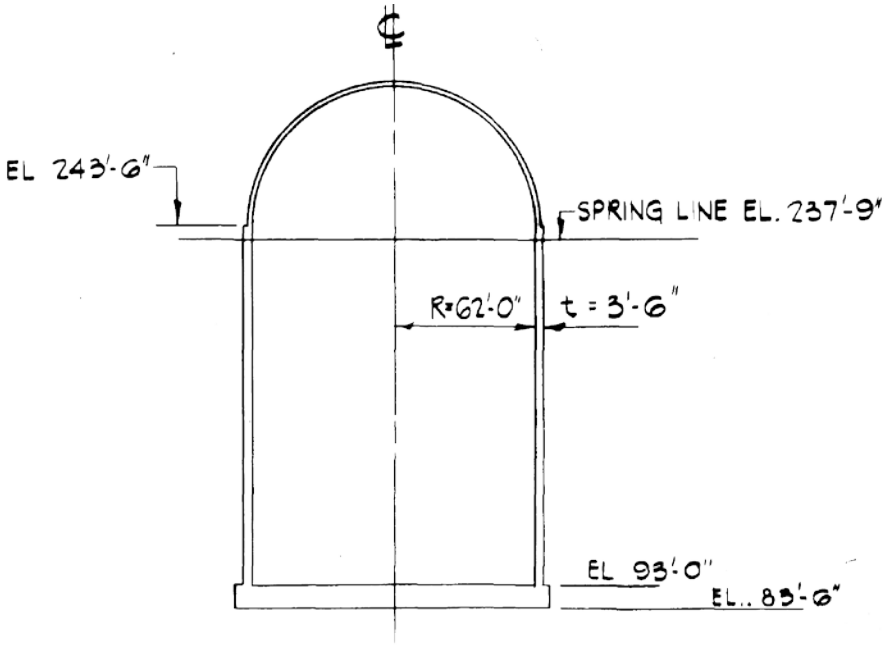
19. "Seismic Analysis of Structures and Equipment for Nuclear Power Plants, "Bechtel Topical Report, BC-TOP-4, September 1972, including Addendum 1 dated April 1973, Bechtel Corporation.

20. Nuclear Construction Issues Group (NCIG), Visual Weld Acceptance Criteria for Structural Welding at Nuclear Plants; NCIG-01, Revision 2.

21. Nuclear Construction Issues Group (NCIG), Training Manual for Inspectors of Structural Welds at Nuclear Power Plants Using the Acceptance Criteria of NCIG-01; NCIG-03, Revision H.

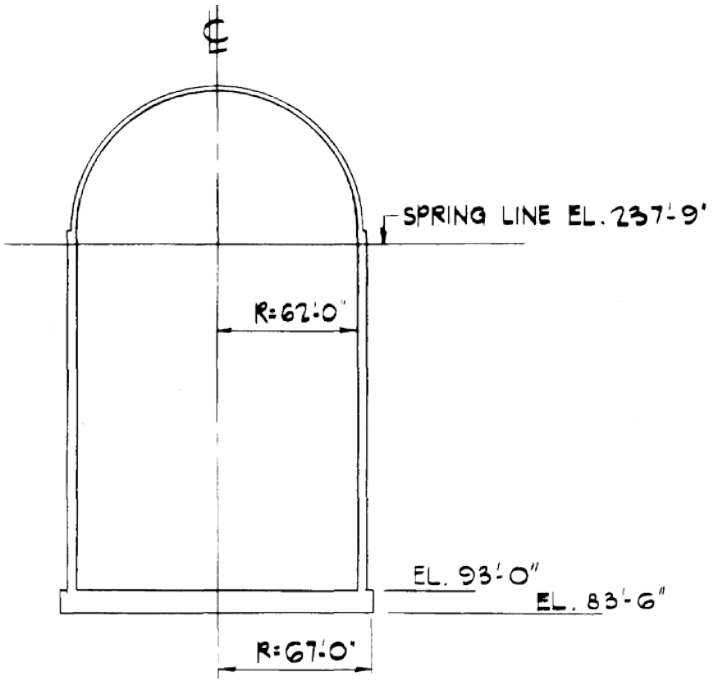
TABLE 3.8-1: FACTORS OF SAFETY FOR SLIDING, OVERTURNING, AND BUOYANCY FOR SEISMIC CATEGORY I STRUCTURES

	Factor of Safety		
Structure	Sliding	Overturning	Buoyancy
Containment building	1.21	1.912	3.847
Auxiliary building	1.109	2.06	2.64
Diesel generator building	1.53	7.7	NA
Control building	1.12	1.92	2.27
SSW cooling towers and basin	NA	NA	1.11



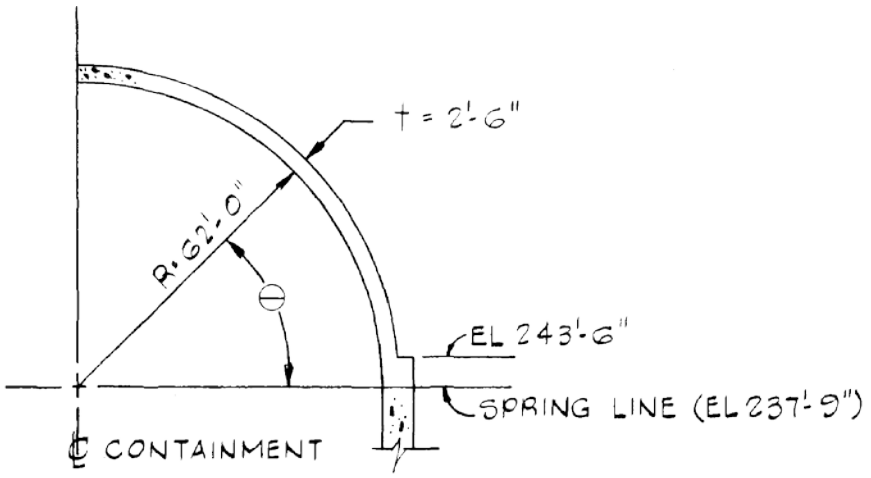
STRESS TABLE 3.8-2

CONTAINMENT STRUCTURE						
CYLINDRICAL WALL						
STRESS COMPONENT			LOCATION	GOVERNING EQUATION	STRESS ACTUAL / ALLOWABLE	REMARKS
CONCRETE STRESS	MEMBRANE + BENDING		EL. 93'-9"	$U = 1.0D + 1.0L + 1.0B + 1.0E$	-3115 psi / -3825 psi	COMPRESSIVE MERIDIONAL STRESS
	RADIAL SHEAR		EL. 93'-9"	$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.0P_{DS} + 1.0C_r + 1.0E' + 1.0R_c$	V_u = 329 psi / V_c = 112 psi	V_u = NOMINAL TOTAL DESIGN SHEAR STRESS V_c = NOMINAL PERMISSIBLE SHEAR STRESS
REINFORCEMENT STRESS	MERIDIONAL	INSIDE	EL. 93'-9"	$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.25P_{CO} + 1.25E$	56,482 psi * / 57,420 psi	* F_y = 63,800 psi MINIMUM PER MATERIAL TEST REPORTS
		OUTSIDE	EL. 93'-9"	$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.0P_{DS} + 1.0C_r + 1.0E' + 1.0R_c$	44,665 psi / 54,000 psi	
	HOOP	INSIDE	EL. 135'-4"	$U = 1.0D + 1.0L + 1.0B + 1.0E$	50,000 psi / 54,000 psi	
		OUTSIDE	EL. 120'-10"	$U = 1.0D + 1.0L + 1.0B + 1.0E$	43,330 psi / 54,000 psi	
	RADIAL SHEAR		EL. 93'-9"	$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.0P_{DS} + 1.0C_r + 1.0E' + 1.0R_c$	56,018 psi ‡ / 60,000 psi	‡ SEE SECT 3.8.1.4.1.2 FOR CAPACITY REDUCTION FACTOR (ϕ)
	DIAGONAL		EL. 101'-2½"	$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.0P_{BO} - 1.0R_{BA} + 1.0C_r(+) + E'$	62,337 psi * ‡ / 68,800 psi	*ALLOWABLE STRESS PER ITEM 130.12 OF LETTER AECM-76/18, 5/30/76



STRESS TABLE 3.8-3

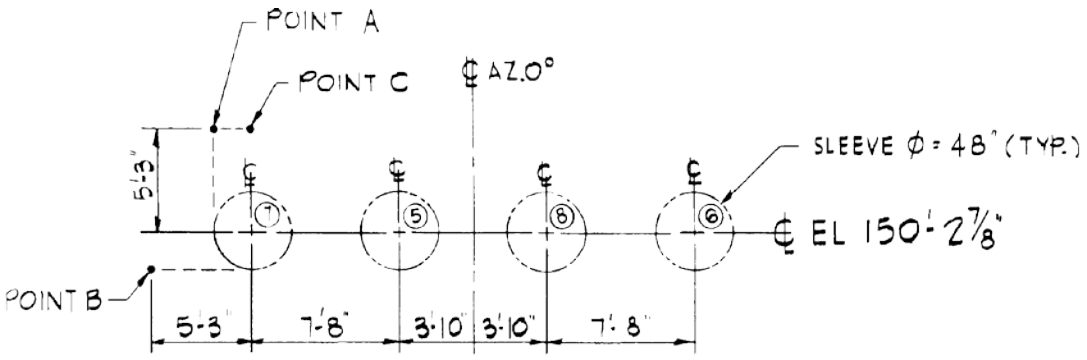
CONTAINMENT STRUCTURE						
FOUNDATION MAT						
STRESS COMPONENT			LOCATION	GOVERNING EQUATION	STRESS ACTUAL / ALLOWABLE	REMARKS
CONCRETE STRESS	MEMBRANE + BENDING		R=5'-5"	$U=1.0D+1.0L+1.0(T_o+T_a)+1.5P_{DS}+1.25C_s+1.5R_c$	-3738 psi / -3825 psi	COMPRESSIVE HOOP STRESS
	RADIAL SHEAR		R=40'-0"	$U=1.0D+1.0L+1.0(T_o+T_a)+1.0P_{CD}+1.0E'$	v_u=274 psi / v_c=97 psi	v_u=NOMINAL TOTAL DESIGN SHEAR STRESS v_c=NOMINAL PERMISSIBLE SHEAR STRESS
REINFORCEMENT STRESS	MERIDIONAL	TOP	R=59'-5"	$U=1.0D+1.0L+1.0B+1.0E$	46,483 psi / 54,000 psi	
		BOTTOM	R=35'-3"	$U=1.0D+1.0L+1.0(T_o+T_a)+1.0P_{CD}+1.0E'$	29,212 psi / 54,000 psi	
	HOOP	TOP	R=59'-5"	$U=1.0D+1.0L+1.0B+1.0E$	24,498 psi / 54,000 psi	
		BOTTOM	R=44'-1"	$U=1.0D+1.0L+1.0(T_o+T_a)+1.0P_{CD}+1.0E'$	21,093 psi / 54,000 psi	
	RADIAL SHEAR		R=40'-0"	$U=1.0D+1.0L+1.0(T_o+T_a)+1.0P_{CD}+1.0E'$	39,930 psi † / 60,000 psi	†SEE SECT. 3.8.1.4.1.2 FOR CAPACITY REDUCTION FACTOR (ø)



DOME SECTION

STRESS TABLE 3.8-3a

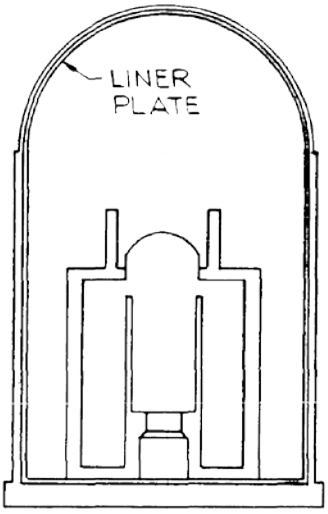
CONTAINMENT STRUCTURE						
DOME						
STRESS COMPONENT			LOCATION	GOVERNING LOADING COMBINATION	STRESS ACTUAL / ALLOWABLE	REMARKS
CONCRETE STRESS	MEMBRANE + BENDING		θ = 4°-40'	S = 1.0D + 1.0L + 1.0T$_o$ + 1.0E	-865 PSI / -2250 PSI	COMPRESSIVE MERIDIONAL STRESS
	RADIAL SHEAR		θ = 0°-30'	U = 1.0D + 1.0L + 1.0 (T$_o$ + T$_a$) + 1.5 P$_{CD}$	V_u = 76 PSI / V_c = 101 PSI	V_u = NOMINAL TOTAL DESIGN SHEAR STRESS V_c = NOMINAL PERMISSIBLE SHEAR STRESS
REINFORCEMENT STRESS	MERIDIONAL	INSIDE	θ = 71°-43'	S = 1.0D + 1.0L + 1.15 P$_{CD}$	22,080 PSI / 30,000 PSI	
		OUTSIDE	θ = 89°-10'	S = 1.0D + 1.0L + 1.15 P$_{CD}$	23,600 PSI / 30,000 PSI	
	HOOP	INSIDE	θ = 29°-55'	S = 1.0D + 1.0L + 1.15 P$_{CD}$	28,800 PSI / 30,000 PSI	
		OUTSIDE	θ = 47°-20'	S = 1.0D + 1.0L + 1.15 P$_{CD}$	29,970 PSI / 30,000 PSI	
	DIAGONAL		θ = 0°-26'	U = 1.0D + 1.0L + 1.0 (T$_o$ + T$_a$) + 1.0 P$_{CD}$ + 1.0E'	35,624 PSI ‡ / 60,000 PSI	‡ SEE SECT 3.8-1.4.1.2 FOR CAPACITY REDUCTION FACTOR (ϕ)
	RADIAL SHEAR			N/A		



ELEVATION
MAIN STEAM AREA

STRESS TABLE 3.8-3b

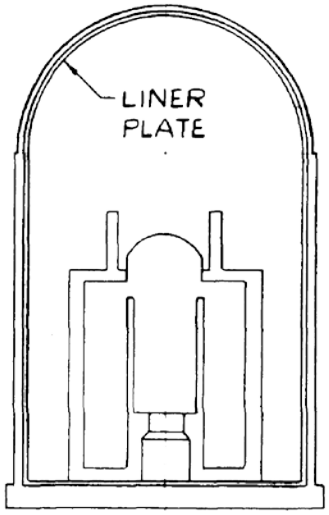
CONTAINMENT STRUCTURE CYLINDRICAL WALL MAIN STEAM PENETRATION AREA						
STRESS COMPONENT			LOCATION	GOVERNING EQUATION	STRESS ACTUAL / ALLOWABLE	REMARKS
CONCRETE STRESS	MEMBRANE + BENDING		POINT A	U=1.0D+1.0L+1.0(T_o+T_a)+1.0P_{CD}+1.0E'-1.0R	-3800 PSI / -3825 PSI	COMPRESSIVE MERIDIONAL STRESS P_{CD} & T_a = 0
	RADIAL SHEAR		POINT C	U=1.0D+1.0L+1.0(T_o+T_a)+1.0P_{CD}+1.0E'+1.0R	V_u=482 PSI / V_c=178 PSI	V_u=NOMINAL TOTAL DESIGN SHEAR STRESS V_c=NOMINAL PERMISSIBLE SHEAR STRESS P_{CD}& T_a=0
REINFORCEMENT STRESS	MERIDIONAL	INSIDE	POINT A	U=1.0D+1.0L+1.0(T_o+T_a)+1.0P_{CD}+1.0E'+1.0R	47,030 PSI / 54,000 PSI	P_{CD} & T_a = 0
		OUTSIDE	POINT A	U=1.0D+1.0L+1.0(T_o+T_a)+1.0P_{CD}+1.0E'+1.0R	44,630 PSI / 54,000 PSI	P_{CD} & T_a = 0
	HOOP	INSIDE	POINT B	U=1.0D+1.0L+1.0(T_o+T_a)+1.0P_{CD}+1.0 E'+1.0R	39,140 PSI / 54,000 PSI	P_{CD} & T_a = 0
		OUTSIDE	POINT B	U=1.0D+1.0L+1.0(T_o+T_a)+1.0P_{CD}+1.0E'+1.0R	44,590 PSI / 54,000 PSI	P_{CD} & T_a = 0
						‡SEE SECT 3.8.1.4.1.2 FOR CAPACITY REDUCTION FACTOR (ϕ)
	RADIAL SHEAR		POINT C	U=1.0D+1.0L+1.0(T_o+T_a)+1.0P_{CD}+1.0E'+1.0R	44,250 PSI / 60,000 PSI ‡	P_{CD} & T_a = 0



STRESS TABLE 3.8-4

CONTAINMENT BUILDING				
LINER PLATE				
LOCATION	STRESS COMPONENT	GOVERNING EQUATION	STRESS ACTUAL / ALLOW	REMARKS
93'-0" HOOP	TENSION	$U=1.0D+1.0L+1.0B+1.0E_B$	4.7 KSI / 30 KSI	
	COMPRESSION	$U=1.0D+1.0L+1.0R+1.0(T_o+T_a)+1.5P_{CO}$	30 KSI / 30 KSI	PLATE IN YIELD CONDITION SEE BC-TOP 1
HOOP	TENSION	CONSTRUCTION LOADS $D+L'$	11.3 KSI / 20 KSI *	L' = PRESSURE DUE TO WET CONCRETE
	COMPRESSION	CONSTRUCTION LOADS $D+L'$	17.76 KSI / 20 KSI *	L' = PRESSURE DUE TO WET CONCRETE
135'-4" HOOP	TENSION	$S=D+L+1.15P_{CO}+1.0T_o$	2.2 KSI / 24 KSI	
	COMPRESSION	$S=D+L+1.15P_{CO}+1.0T_o$	N/A / 24 KSI	

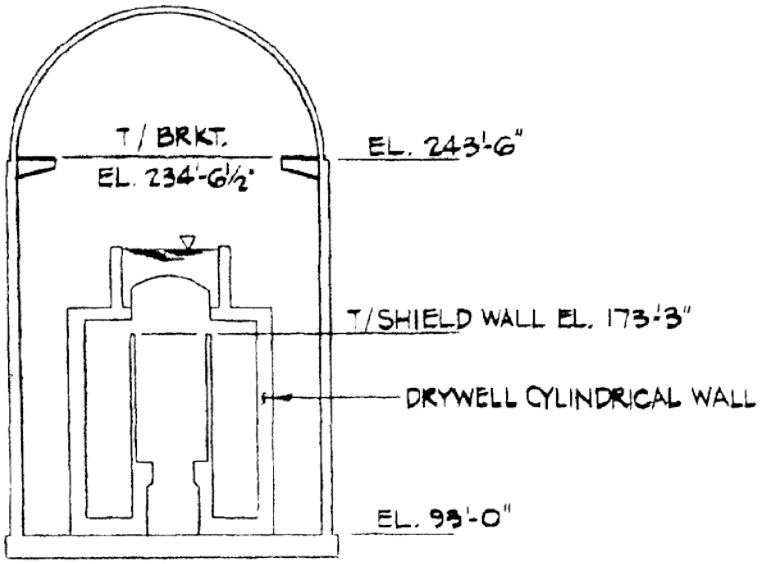
LOCATION	STRAIN COMPONENT	GOVERNING EQUATION	STRAIN ACTUAL / ALLOW	REMARKS
123'-0" HOOP	TENSION	$U=1.0D+1.0L+1.0B+1.0E_B$	177×10^{-6} / $10000^{*}\times10^{-6}$	* LIMITS PER ASME SECT III, DIV 2, SUBSECTION CC-3700
236'-9" HOOP	COMPRESSION	$U=1.0D+1.0L+1.0T_o+1.0R_o+1.0C_1(+)+1.0E'$	172×10^{-6} / $14000^{*}\times10^{-6}$	



STRESS TABLE 3.8-5

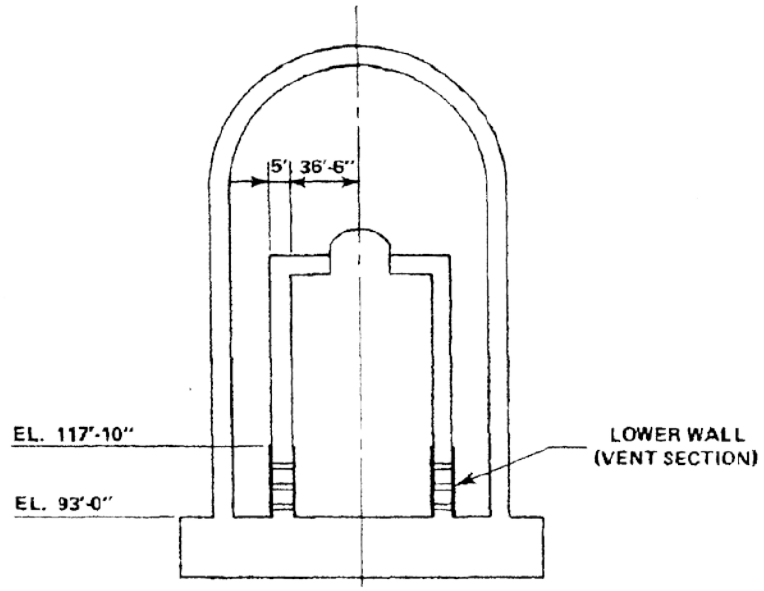
CONTAINMENT BUILDING				
LINER PLATE				
STRESS COMPONENT	LOCATION	GOVERNING EQUATION	STRESS ACTUAL / ALLOWABLE	REMARKS
TENSION	93'-0" LONGITUDINAL	U = 1.0D + 1.0L + 1.0B + 1.0E_B	9394 PSI / 30 KSI	
COMPRESSION	93'-0" LONGITUDINAL	U = 1.0D + 1.0L + 1.0(T_o+T_d) + 1.25P_{CD} + 1.25E + 1.0R	30 KSI / 30KSI	PLATE IN YIELD CONDITION SEE BC- TOP 1
TENSION	LONGITUDINAL	CONSTRUCTION LOADS D + L'	N/A / 20 KSI *	L' = PRESSURE DUE TO WET CONCRETE
COMPRESSION	LONGITUDINAL	CONSTRUCTION LOADS D + L'	3870 PSI / 20 KSI *	"
TENSION	93'-0" LONGITUDINAL	S = D + L + 1.15 P_{CD} + 1.0 T_o	4148 PSI / 30 KSI	
COMPRESSION	93'-0" LONGITUDINAL	S = D + L + 1.15 P_{CD} + 1.0 T_o	N/A / 30 KSI	

STRAIN COMPONENT	LOCATION	GOVERNING EQUATION	STRAIN ACTUAL $\times 10^{-6}$ / ALLOWABLE $\times 10^{-6}$	REMARKS
TENSION	93'-0" LONGITUDINAL	U = 1.0D + 1.0L + 1.0B + 1.0E_B	273 / 10000 *	* LIMITS PER ASME SECT III, DIV 2, SUB SECT CC-3700
COMPRESSION	93'-0" LONGITUDINAL	U = 1.0D + 1.0L + 1.0(T_o) + 1.0Z + 1.25E	557 / 14000 *	



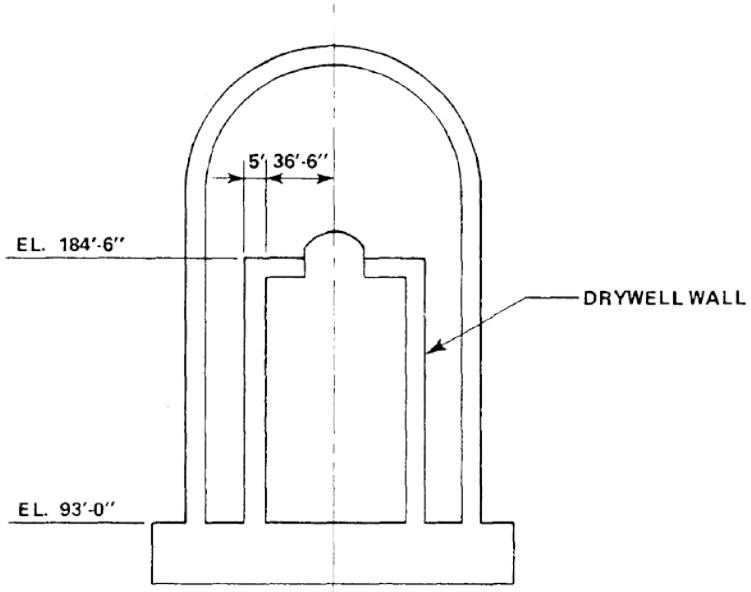
STRESS TABLE 3.8-6

CONTAINMENT POLAR CRANE BRACKET				
T/BRKT EL. 234'-6 1/2"				
STRESS COMPONENT	LOCATION	GOVERNING CONDITION	STRESS ACTUAL / ALLOWABLE	REMARKS
MAXIMUM FLANGE TENSILE OR COMPRESSIVE (F_b)	TOP FLANGE	DEAD LOAD + LIVE LOAD + IMPACT + SAFE SHUTDOWN EARTHQUAKE	33.5 KSI / 34.3 KSI	
MAXIMUM TENSION (F_t)	VERT. EMB. BEARING PL.	d.o.	30.1 KSI / 32.3 KSI	
MAXIMUM SHEAR (F_v)	THICKENED LINER PL. & RIBS	d.o.	16.2 KSI / 18.0 KSI	
MAXIMUM CONCRETE BEARING (F_c)	RIBS	d.o.	2.6 KSI / 2.98 KSI	



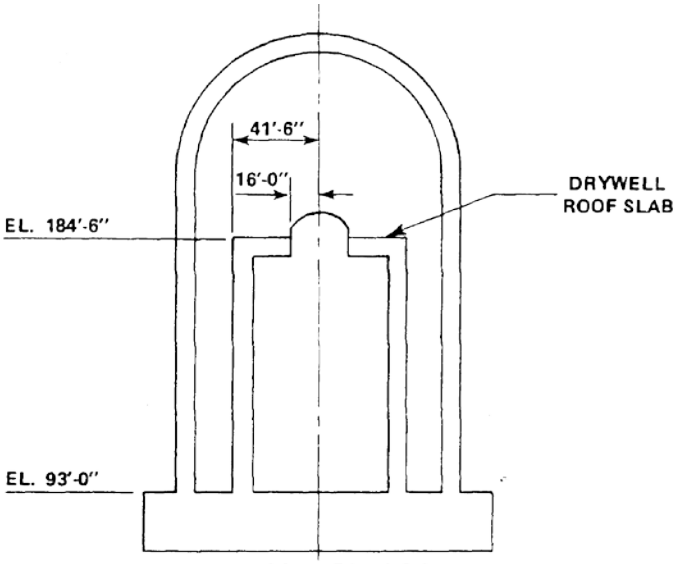
STRESS TABLE 3.8-7

INTERNAL STRUCTURES						
DRYWELL – LOWER WALL (VENT SECTION)						
COMPONENT STRESS			LOCATION	GOVERNING EQUATION	STRESS ACTUAL / ALLOWABLE	REMARKS
CONCRETE STRESS	COMPRESSIVE		ELEMENT 945 (EL 102'-6")	$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.0P_{DS} + 1.0R_c + 1.0C_I + 1.0E'$	3.8 KSI / 3.83 KSI	
	SHEAR		ELEMENT 1 (EL 93'-3")	$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.0P_{DS} + 1.0R_c + 1.0C_H + 1.0E'$	124 PSI / 124 PSI	DIAPHRAM PLATES RESIST EXCESS SHEAR
LINER PLATE STRESS	MERIDIONAL	INSIDE	ELEMENT 1521 (EL 112'-3")	$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.0P_{DS} + 1.0R_c + 1.0C_I + 1.0E'$	26.0 KSI / 45 KSI	COMPRESSIVE
		OUTSIDE	ELEMENT 1297 (EL 106'-4")	$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.0P_{DS} + 1.0R_c + 1.0C_I + 1.0E'$	29.3 KSI / 45 KSI	TENSILE
	HOOP	INSIDE	ELEMENT 1297 (EL 106'-4")	$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.0P_{DS} + 1.0R_c + 1.0C_I + 1.0E'$	43.9ksi / 62 KSI	CHECK OF MATERIAL TEST REPORTS SHOWS YIELD TO BE 68.9KSI THEREFORE
		OUTSIDE	ELEMENT 1521 (EL 112'-3")	$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.0P_{DS} + 1.0R_c + 1.0C_I + 1.0E'$	14.6 KSI / 45 KSI	0.9(68.9)=62 KSI TENSILE
	TANGENTIAL	INSIDE AND OUTSIDE	EL 104'-4" (℄ TOP VENT)	$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.0P_{DD} + 1.0E' + 1.0C_I$	12.0KSI / 20.0KSI	



STRESS TABLE 3.8-8

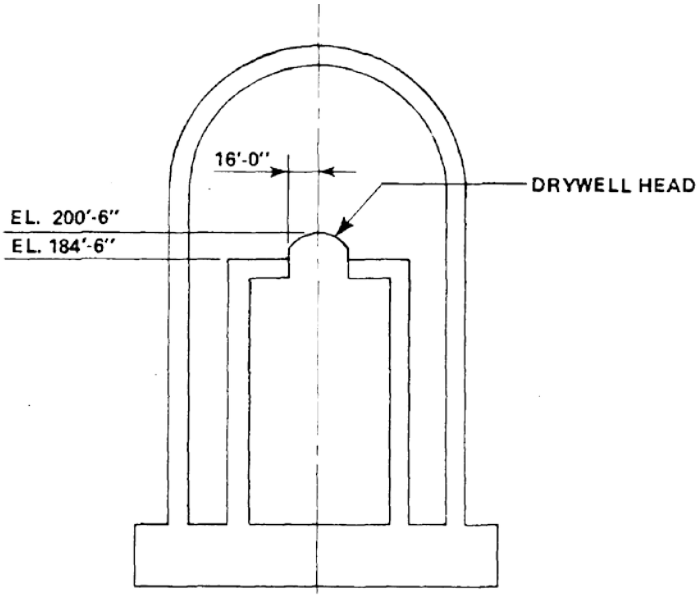
INTERNAL STRUCTURES						
DRYWELL – WALL						
COMPONENT STRESS			LOCATION	GOVERNING EQUATION	STRESS ACTUAL / ALLOWABLE	REMARKS
CONCRETE STRESS	COMPRESSIVE		EL. 102'-6"	$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.0C_1 + 1.0P_{DS} + 1.0R_C + 1.0E'$	3.79 KSI / 3.83 KSI	
	SHEAR		EL. 180'-2"	$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.0P_{DD} + 1.0R_B + 1.0C_1 + 1.0E' + 1.0R$	104 PSI / 104 PSI	SEE RADIAL SHEAR
REINFORCEMENT STRESS	MERIDIONAL	INSIDE	EL. 180'-2"	$S = 1.0D + 1.0L + 1.0P_T$	25.2 KSI / 30.0 KSI	
		OUTSIDE	EL. 106'-4"	$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.0P_{DD} + 1.0R_C + 1.0C_1 + 1.0E'$	29.3 KSI / 54.0 KSI	
	HOOP	INSIDE	EL. 180'-2"	$S = 1.0D + 1.0L + 1.0P_T$	28.6 KSI / 30.0 KSI	
		OUTSIDE	EL. 135'-0"	$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.5P_{DD} + 1.5R_B + 1.25C_1 + 1.0R$	43.4 KSI / 54.0 KSI	
	SHEAR	TANGENTIAL (DIAGONAL)	EL. 109'-3"	$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.0P_{DD} + 1.0E' + 1.0C_1$	19 KSI / 54 KSI	
		RADIAL	EL. 180'-2"	$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.0P_{DD} + 1.0R_B + 1.0C_1 + 1.0R + 1.0E'$	27.3 KSI / 54.0 KSI	



STRESS TABLE 3.8-9

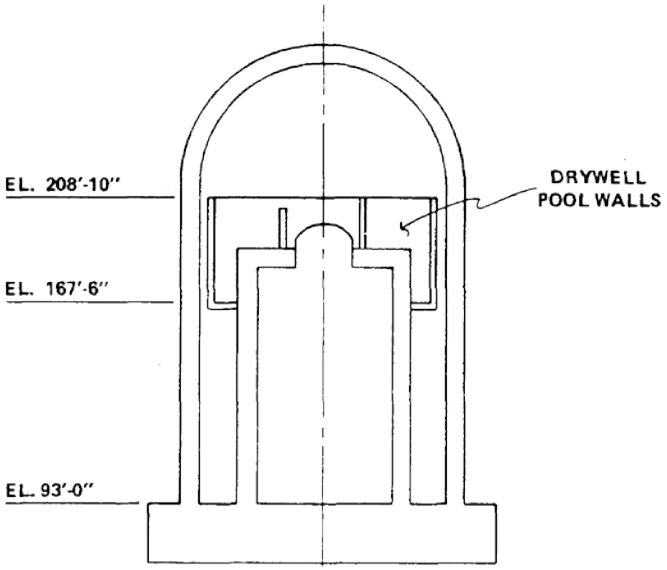
INTERNAL STRUCTURES						
DRYWELL – ROOF SLAB						
COMPONENT STRESS			LOCATION (✱✱)	GOVERNING EQUATION	STRESS ACTUAL / ALLOWABLE	REMARKS
CONCRETE STRESS	COMPRESSIVE		ELEMENT 171	$S=1.0D+1.0L+1.0P_T$	1.6 KSI / 2.25 KSI	
	SHEAR		ELEMENT 197	$U=1.0D+1.0L+1.0(T_o+T_a)+1.0P_{BN}+1.0R_B+1.0C_1+1.0E'+1.0R$	64.2 PSI / 64.2 PSI	SHEAR REINFORCEMENT REQUIRED
REINFORCEMENT STRESS	MERIDIONAL	INSIDE	ELEMENT 167	$S=1.0D+1.0L+1.0P_T$	27.3 KSI / 30.0 KSI	
		OUTSIDE	ELEMENT 171	$U=1.0D+1.0L+1.0(T_o+T_a)+1.5P_{ND}+1.0R+1.0E'$	22.9 KSI / 54.0 KSI	
	HOOP	INSIDE	ELEMENT 228	$U=1.0D+1.0L+1.0(T_o+T_a)+1.5P_{OS}+1.25C_1+1.5R_C$	20.9 KSI / 54 KSI	
		OUTSIDE	ELEMENT 201	$U=1.0D+1.0L+1.0(T_o+T_a)+1.5P_{OD}+1.0R_B+1.0C_1+1.0E'+1.0R$	55.4 KSI / 62.0 KSI	CHECK OF MATERIAL TEST REPORTS SHOWS YIELD TO BE 68.9 KSI THEREFORE, 0.9(68.9)=62 KSI
	SHEAR	TANGENTIAL (DIAGONAL)	NOT APPLICABLE			
		RADIAL	ELEMENT 197	$U=1.0D+1.0L+1.0(T_o+T_a)+1.0P_{OD}+1.0R_B+1.0C_1+1.0E'+1.0R$	49.6 KSI / 54.0 KSI	

✱✱ SEE FIGURE 3.8-74



STRESS TABLE 3.8-10

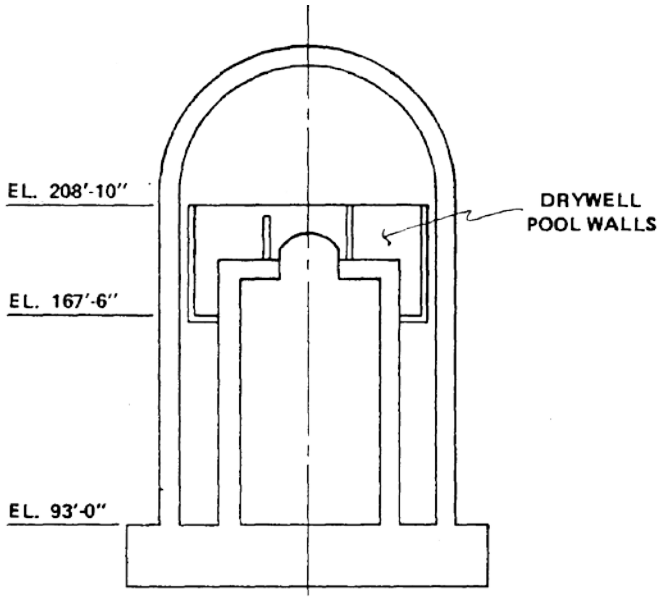
INTERNAL STRUCTURES						
DRYWELL – HEAD						
COMPONENT STRESS			LOCATION	GOVERNING EQUATION	STRESS ACTUAL / ALLOWABLE	REMARKS
PLATE STRESS		MEMBRANE	CROWN OF DRYWELL HEAD	$U=1.0D+1.0L+1.0(T_o+T_a)+1.0P_{DD}+1.0R+1.0E'$	21.4 KSI / 37.0 KSI	
		MEMBRANE PLUS BENDING	KNUCKLE REGION	$U=1.0D+1.0L+1.0(T_o+T_a)+1.0P_{DD}+1.0E'$	9.2 KSI / 27.2 KSI	
		MEMBRANE	CYLINDRICAL SHELL SECTION	$U=1.0D+1.0L+1.0(T_o+T_a)+1.0P_{DD}+1.0E'$	6.6 KSI / 18.2 KSI	
		MEMBRANE PLUS BENDING	CYLINDRICAL SHELL SECTION	$U=1.0D+1.0L+1.0(T_o+T_a)+1.0P_{DD}+1.0E'$	6.6 KSI / 27.2 KSI	



STRESS TABLE 3.8-11

INTERNAL STRUCTURES						
DRYWELL – POOL WALLS						
COMPONENT STRESS			LOCATION	GOVERNING EQUATION	STRESS: ACTUAL / ALLOWABLE	REMARKS
CONCRETE STRESS	COMPRESSIVE		✳	✳	✳	✳
	SHEAR		✳	✳	✳	✳
REINFORCEMENT STRESS	EAST-WEST	TOP	ELEMENT 78 EL. 167'-6"	$S=1.0D+1.0L+1.0P_T$	20.8 KSI / 30.0 KSI	
		BOTTOM	ELEMENT 70 EL. 167'-6"	$U=1.0D+1.0L+1.0(T_o+T_a)+1.0P_{DD}+1.0E'$	53.0 KSI / 54.0 KSI	
	NORTH-SOUTH	TOP	ELEMENT 54 EL. 167'-6"	$U=1.0D+1.0L+1.0(T_o+T_a)+1.5P_{DS}$	52.1 KSI / 54.0 KSI	
		BOTTOM	ELEMENT 54 EL. 167'-6"	$U=1.0D+1.0L+1.0(T_o+T_a)1.5P_{DS}$	45.2 KSI / 54.0 KSI	
	SHEAR	N-S	✳	✳	✳	✳
		E-W	✳	✳	✳	✳

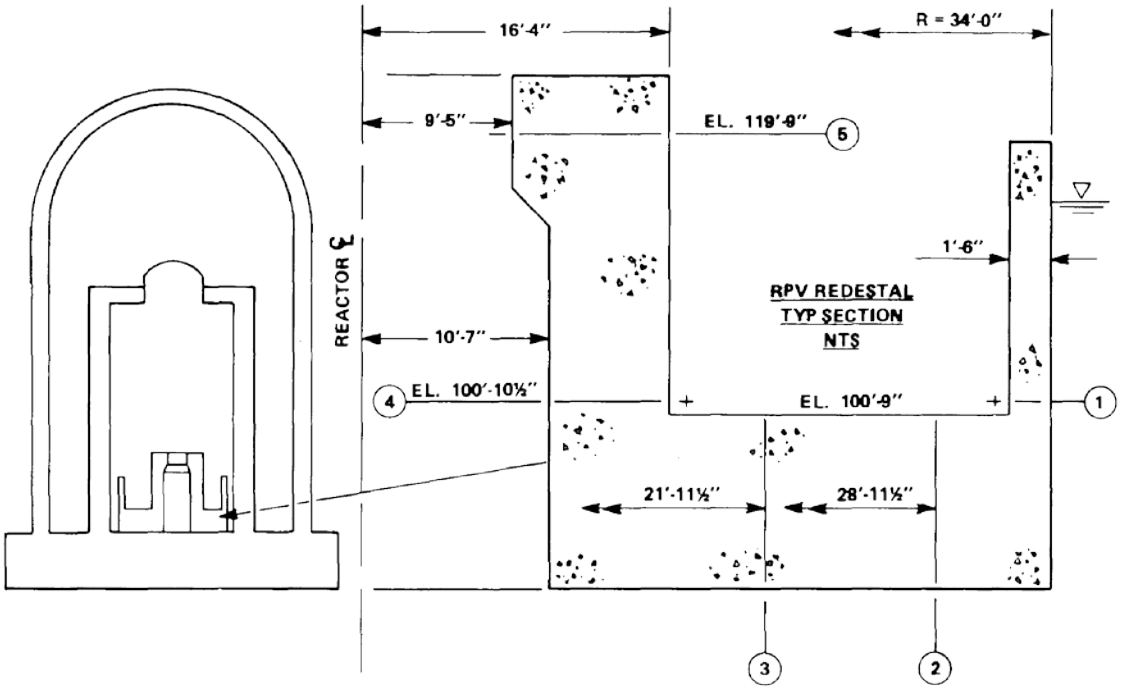
✳ STRESSES ARE NEGLIGIBLE



STRESS TABLE 3.8-12

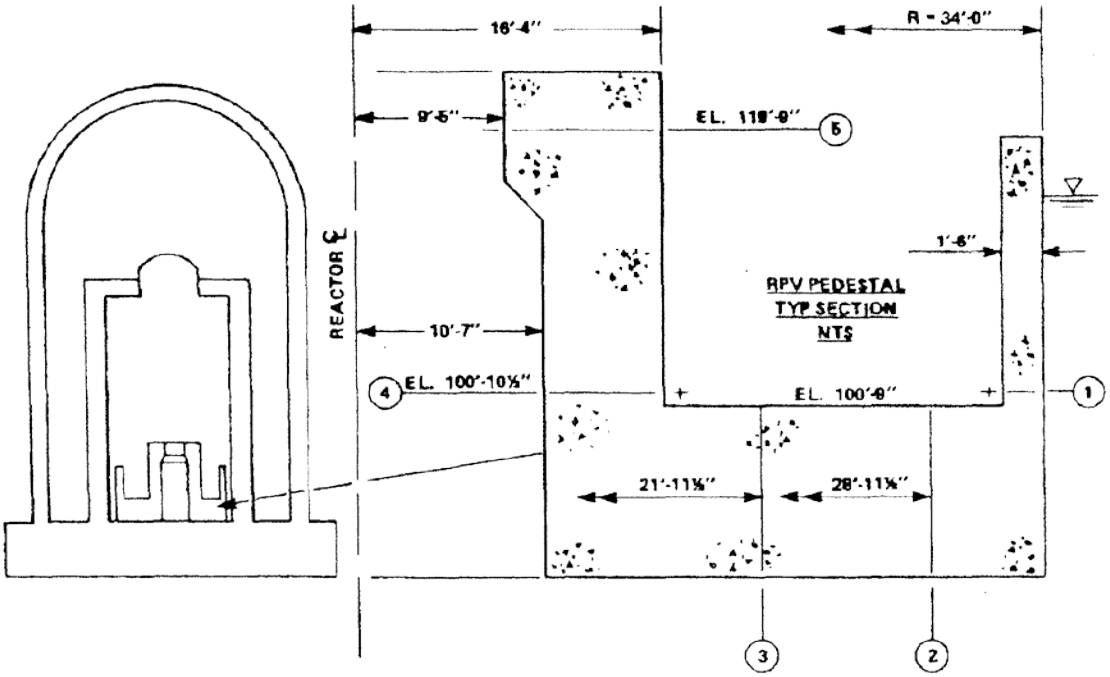
INTERNAL STRUCTURES						
DRYWELL – POOL WALLS						
COMPONENT STRESS			LOCATION	GOVERNING EQUATION	STRESS: ACTUAL / ALLOWABLE	REMARKS
CONCRETE STRESS	COMPRESSIVE		SOUTH POOL WALL ELEMENT 189 EL 182'-4	$S=1.0D+1.0L+1.0T_o$	-2.2 KSI / 2.3 KSI	
	SHEAR		*	*	*	*
REINFORCEMENT STRESS	VERTICAL	INSIDE	SOUTH POOL WALL ELEMENT 172 EL 182'-4"	$S=1.0D+1.0L+1.0P_T$	22.5 KSI / 30.0 KSI	
		OUTSIDE	WEST POOL WALL ELEMENT 118 EL 176'-0"	$U=1.0D+1.0L+1.0(T_o+T_a)+1.0P_{os}+1.0E'$	46.3 KSI / 54.0 KSI	
	HORIZONTAL	INSIDE	SOUTH POOL WALL ELEMENT 206 EL 192'-10"	$S=1.0D+1.0L+1.0T_o$	29.1 KSI / 30.0 KSI	
		OUTSIDE	SOUTH POOL WALL ELEMENT 152 EL 182'-4"	$U=1.0D+1.0L+1.0(T_o+T_a)+1.0P_{os}+1.0E'$	53.0 KSI / 54.0 KSI	
	SHEAR		*	*	*	*
			*	*	*	*

* STRESSES ARE NEGLIGIBLE



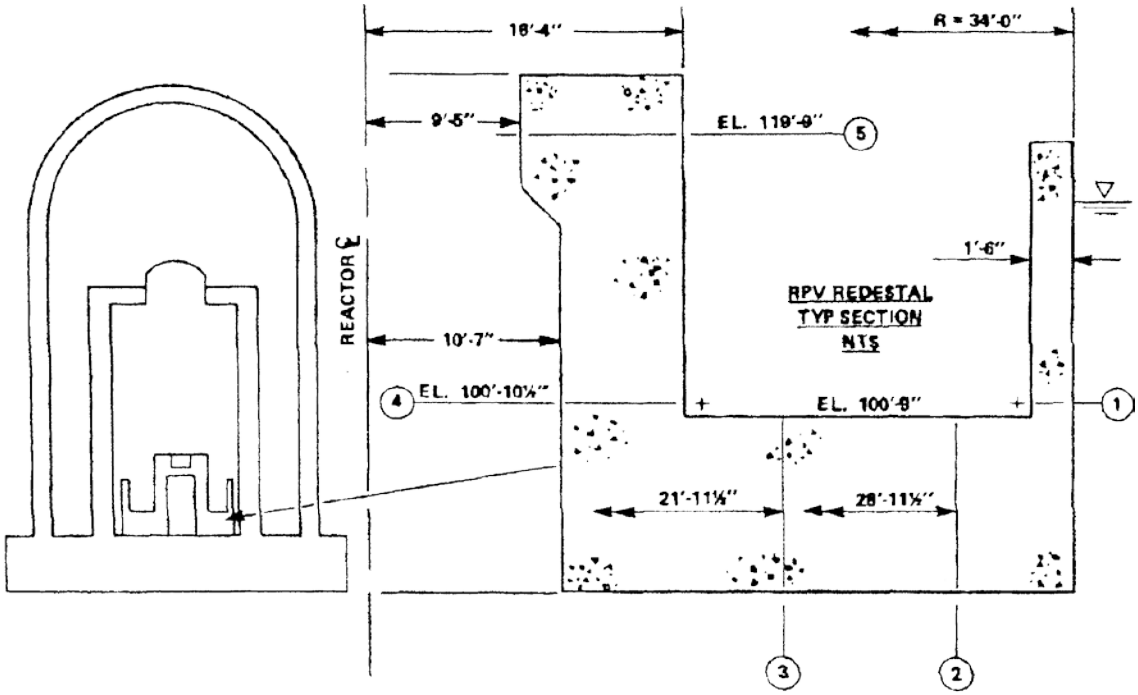
STRESS TABLE 3.8-13

CONTAINMENT INTERNALS						
RPV SUPPORT PEDESTAL WEIR WALL						
COMPONENT STRESS			LOCATION	GOVERNING EQUATION	STRESS: ALLOWABLE / ACTUAL	REMARKS
CONCRETE STRESS	COMPRESSIVE		(1)	$U = D+L+(To+Ta)+R+E'+P_{DD}+C_1$	5000 psi / 3905 psi	
	SHEAR		(1)	$U = D+L+(To+Ta)+R+E'+P_{DD}+C_1$	565 psi / 556.5 psi	SHEAR REINF. PROVIDED
REINFORCEMENT STRESS	MERIDIONAL		(1)	$U = D+L+(To+Ta)+R+E'+P_{DD}+C_1$	63.6 ksi / 59.7 ksi *	*CHECK OF MATERIAL TEST REPORTS SHOWS YIELD TO BE 70.7 KSI. THEREFORE, 0.9 (70.7) = 63.6
	HOOP		(1)	$U = D+L+(To+Ta)+R+E'+P_{DD}+C_1$	63.3 ksi / 65.4 ksi †	† 3% LOCAL OVERSTRESS DUE TO LOCALIZED EFFECT OF JET FORCE WOULD CAUSE NO LOSS OF FUNCTION TO WEIR WALL
	SHEAR		(1)	$U = D+L+(To+Ta)+R+E'+P_{DD}+C_1$	51 ksi / 42.4 ksi	



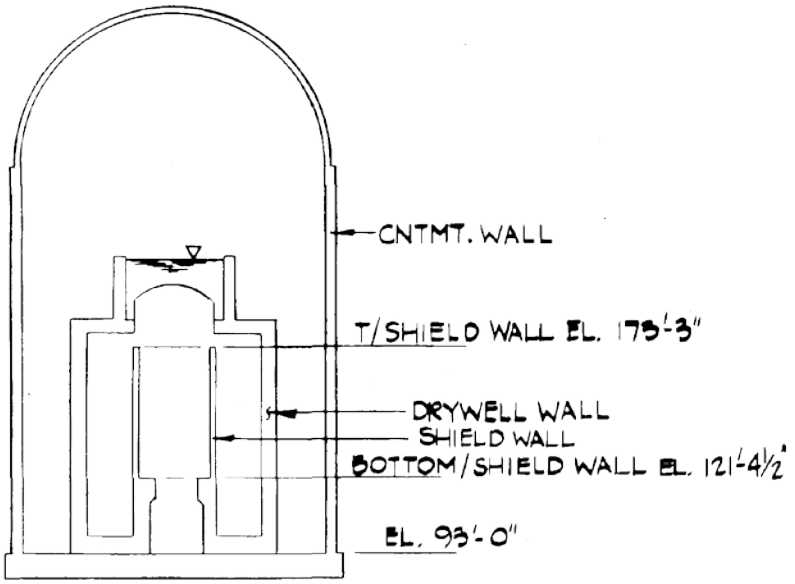
STRESS TABLE 3.8-14

CONTAINMENT INTERNALS						
RPV SUPPORT PEDESTAL						
COMPONENT STRESS			LOCATION	GOVERNING EQUATION	STRESS ALLOWABLE / ACTUAL	REMARKS
CONCRETE STRESS	COMPRESSIVE		4	$U = D+L+(To+Ta)+R+E'+P_{DS}+C_1+Rc$	5000 psi / 1716 psi	
	SHEAR		4	$U = D+R+(To+Ta)+P_{DS}+C_1+E'+Rc$	565 psi / 147 psi	CONCRETE COMPRESSIVE STRENGTH IS CONSISTENTLY HIGHER THAN DESIGN STRENGTH; THEREFORE, OVERSTRESS OF LESS THAN 5.0% IS ACCEPTABLE
REINFORCEMENT STRESS	MERIDIONAL		4	$U = D+L+(To+Ta)+P_{DS}+C_1+E'+Rc+R$	54 ksi / 24.3 ksi	
	HOOP		5	$U = D+L+(To+Ta)+P_{DS}+C_1+E'+Rc$	60.5 ksi / 57.5 ksi	CHECK OF MATERIAL TEST REPORTS SHOWS YIELD FOR THESE BARS TO BE 67.3 ksi. THEREFORE .9 (67.3) ksi = 60.5> 57.5 ksi
	SHEAR		4	$U = D+L+(To+Ta)+P_{DS}+C_1+E'+R_c+R$	N/A	SEE CONCRETE SHEAR



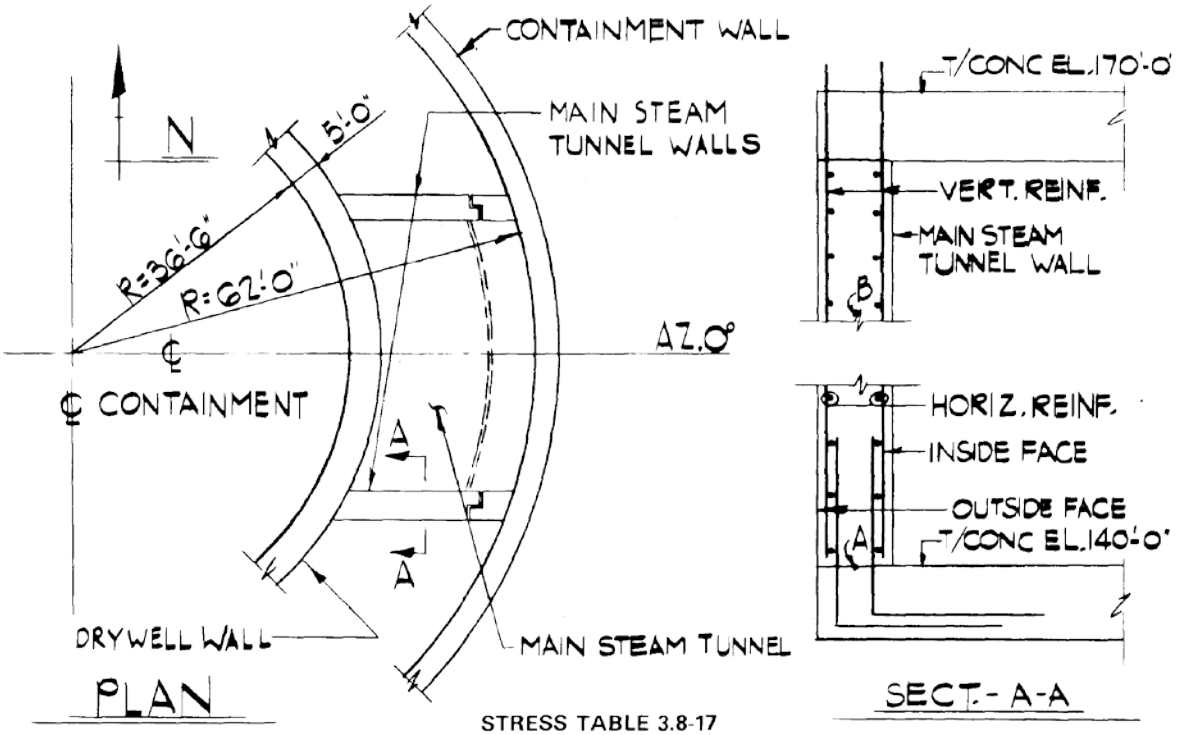
STRESS TABLE 3.8-15

CONTAINMENT INTERNALS						
RPV SUPPORT PEDESTAL MAT						
COMPONENT STRESS			LOCATION	GOVERNING EQUATION	STRESS ALLOWABLE / ACTUAL	REMARKS
CONCRETE STRESS	COMPRESSIVE		(3)	$U = D+L+(To+Ta)+P_{DS}+C_1+E'+Rc$	5000 psi / 2453 psi	
	SHEAR		(2)	$U = D+L+(To+Ta)+P_{DS}+C_1+E'+Rc$	565 psi / 201 psi	SHEAR REINF. PROVIDED
REINFORCEMENT STRESS	MERIDIONAL		(3)	$U = D+L+(To+Ta)+P_{DS}+C_1+E'+Rc$	54 ksi / 43.4 ksi	
			(3)			
	HOOP		(3)	$U = D+L+(To+Ta)+P_{DS}+C_1+E'+Rc$	54 ksi / 32.6 ksi	
	SHEAR		(2)	$U = D+L+(To+Ta)+P_{DS}+C_1+E'+Rc$	54 ksi / 47.8 ksi	



STRESS TABLE 3.8-16

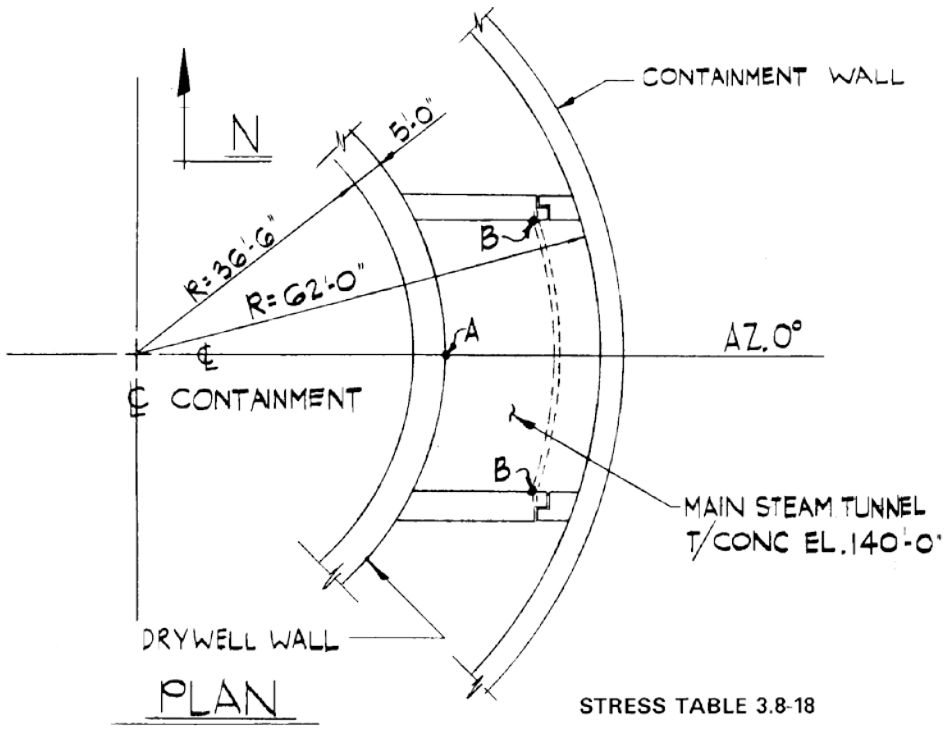
CONTAINMENT INTERNALS						
SHIELD WALL						
COMPONENT STRESS			LOCATION	GOVERNING EQUATION	STRESS ACTUAL / ALLOWABLE	REMARKS
STEEL STRESS	SHEAR	RADIAL	EL. 121'-4½	D+L+(To+Ta)+E'+PAP+R+C1	19.32 KSI / 21.3 KSI	ALLOWABLE BASED ON 1.4 (Fs) PER SECT 3.8.3.5 OF FSAR FOR RADIAL PLATE
					13.5 KSI / 54 KSI	ALLOWABLE BASED ON ACI-318-71 FOR REINFORCING STEEL
	MERIDIONAL	INSIDE	do	do	8.3 KSI / 10.7 KSI	ALLOWABLE IS 1.5 Fs WHERE Fs REDUCED PER APPENDIX C, SECTION C2, AISC SPEC. SINCE $b/t > 176/\sqrt{Fy}$, Fs= 0.6 Fy Qs, WHERE $Qs = \frac{20000}{Fy\,(b/t)^2}$
		OUTSIDE	do	do	8.3 KSI / 10.7 KSI	
	HOOP	INSIDE	do	do	8.8. KSI / 21.7 KSI	
		OUTSIDE	EL. 121'-4½	D+L+(To+Ta)+E'+PAP+R+C1	8.7 KSI / 21.7 KSI	
+ TENSION - COMPRESSION			NOTE: ALL STEEL ASTM A516 GRADE 70			



STRESS TABLE 3.8-17

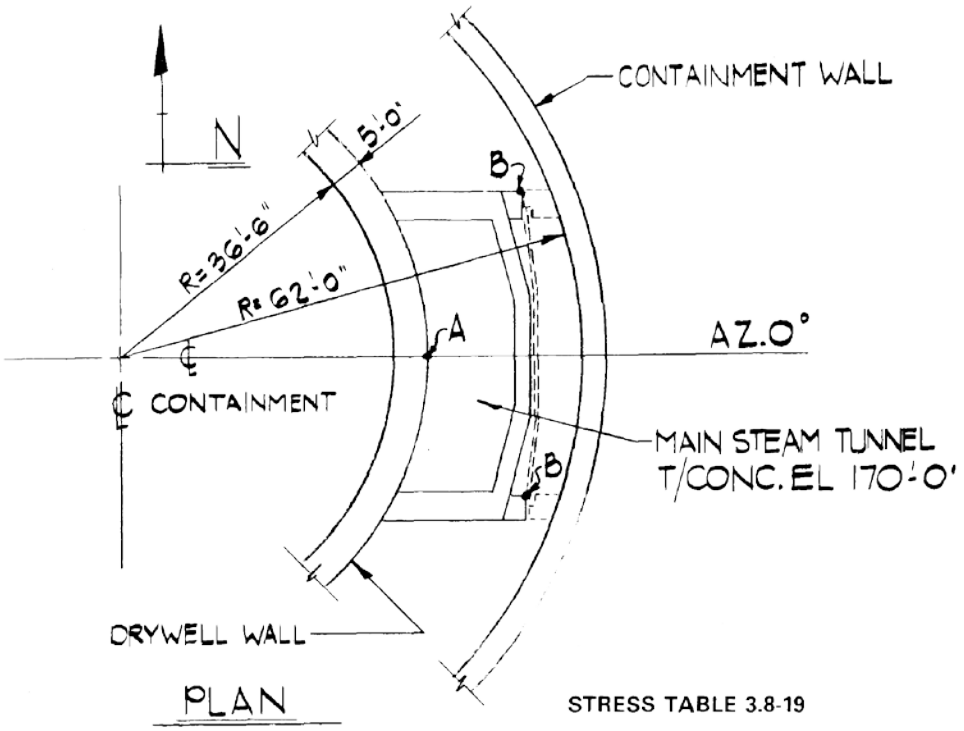
CONTAINMENT INTERNAL CONCRETE STRUCTURES									
MAIN STEAM TUNNEL WALLS									
COMPONENT FORCE OR STRESS			LOCATION	GOVERNING EQUATION	ULTIMATE CAPACITY		REQUIRED CAPACITY		REMARKS
SHEAR STRESS	REINFORCING		——	——	——		——		
	CONCRETE		A	——	141.0 psi		VERY SMALL		
COMBINED AXIAL AND BENDING MOMENT					AXIAL	BENDING	AXIAL	BENDING	
	VERTICAL	INSIDE FACE	B	SEE NOTE BELOW					
		OUTSIDE FACE	A	1.0D+1.0L+1.0(T_o+T_a) +1.5P+1.5Rp+1.25C_1	140 K/FT	857 FT-K/FT	140 K/FT	848 FT-K/FT	
	HORIZONTAL	INSIDE FACE	SEE	NOTE BELOW					
		OUTSIDE FACE	SEE	NOTE BELOW					

NOTE: MINIMUM TEMP & SHRINKAGE REINF. REQ'D.



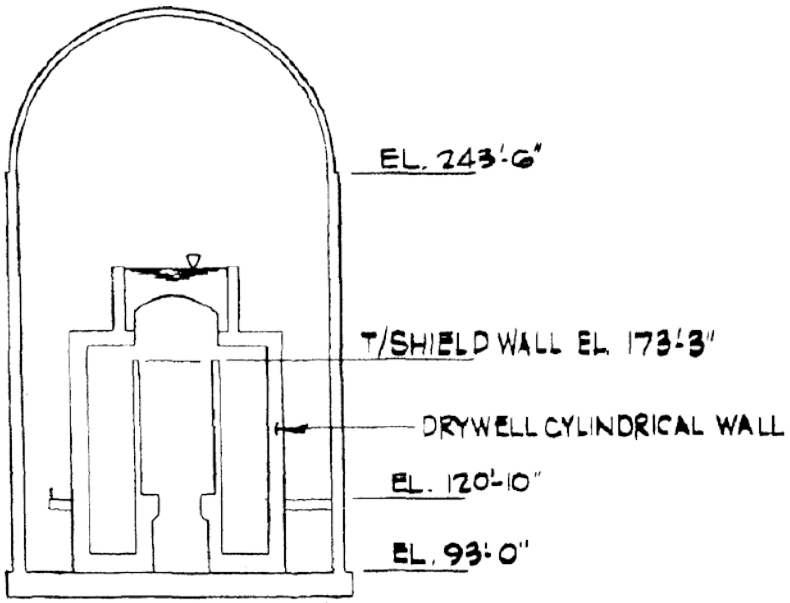
STRESS TABLE 3.8-18

CONTAINMENT INTERNAL CONCRETE STRUCTURES							
MAIN STEAM TUNNEL FLOOR SLAB @ EL 140'-0"							
STRESS COMPONENT			LOCATION	GOVERNING EQUATION	ULTIMATE CAPACITY	REQUIRED CAPACITY	REMARKS
SHEAR STRESS	REINFORCING		A	$1.0D+1.0L+1.0(T_o+T_a)+1.5P+1.5Rp+1.25C_1$	83.3 psi	78.0 psi	
SHEAR STRESS	CONCRETE		A	$1.0D+1.0L+1.0(T_o+T_a)+1.5P+1.5Rp+1.25C_1$	141.0 psi	141.0 psi	
BENDING MOMENT	E-W	TOP	A	$1.0D+1.0L+1.0T_o$	$123^{FTK/FT*}$	$66^{FTK/FT*}$	
BENDING MOMENT	E-W	BOTTOM	A	$1.0D+1.0L+1.0(T_o+T_a)+1.5P+1.5Rp+1.25C_1$	$644^{FTK/FT}$	$477^{FTK/FT}$	
BENDING MOMENT	N-S	TOP	SEE	NOTE BELOW			
BENDING MOMENT	N-S	BOTTOM	B	$1.0D+1.0L+1.0(T_o+T_a)+1.5P+1.5Rp+1.25C_1$	$1150^{FTK/FT}$	$932^{FTK/FT}$	
			* REINFORCING IS PROVIDED USING WORKING STRESS METHOD NOTE: MINIMUM TEMP & SHRINKAGE REINF. REQ'D.				



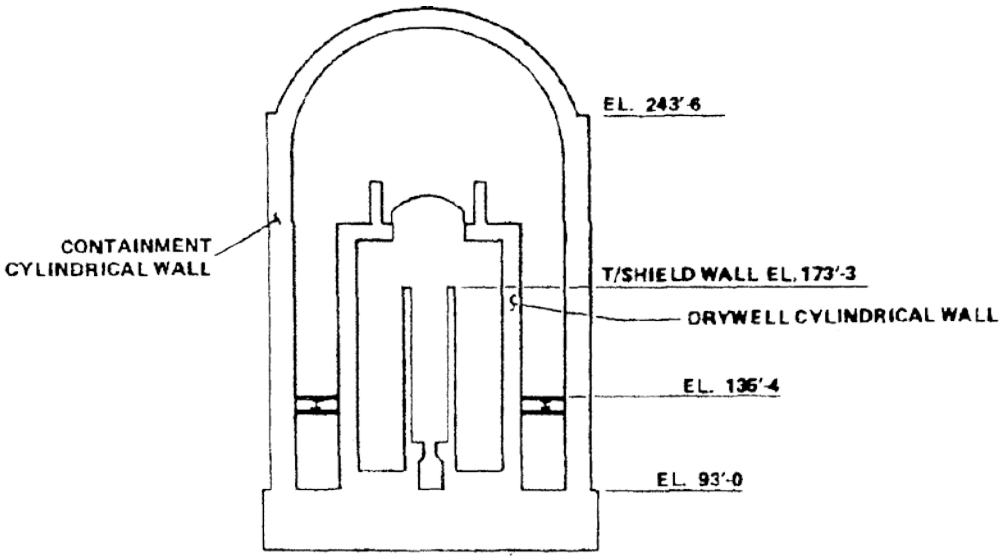
STRESS TABLE 3.8-19

CONTAINMENT INTERNAL CONCRETE STRUCTURES							
MAIN STEAM TUNNEL ROOF SLAB EL. 170'-0"							
STRESS COMPONENT			LOCATION	GOVERNING EQUATION	ULTIMATE CAPACITY	REQUIRED CAPACITY	REMARKS
SHEAR STRESS	REINFORCING		——	——	——	——	
	CONCRETE		A	$1.0D+1.0L+1.0(T_o+T_a)+1.5P+1.0R+1.25C_1$	141.0 psi	135 psi	
BENDING MOMENT	E-W	TOP	A	$1.0D+1.0L+1.0(T_o+T_a)+1.5P+1.0R+1.25C_1$	450 FTK/FT	248 FTK/FT	
		BOTTOM	SEE NOTE BELOW				
	N-S	TOP	B	$1.0D+1.0L+1.0T_o$	*705 FTK/FT	*665 FTK/FT	
		BOTTOM	SEE NOTE BELOW				
			* REINFORCING IS PROVIDED USING WORKING STRESS METHOD. NOTE: MIN TEMP & SHRINKAGE REINF REQ'D BY CODE				



STRESS TABLE 3.8-20

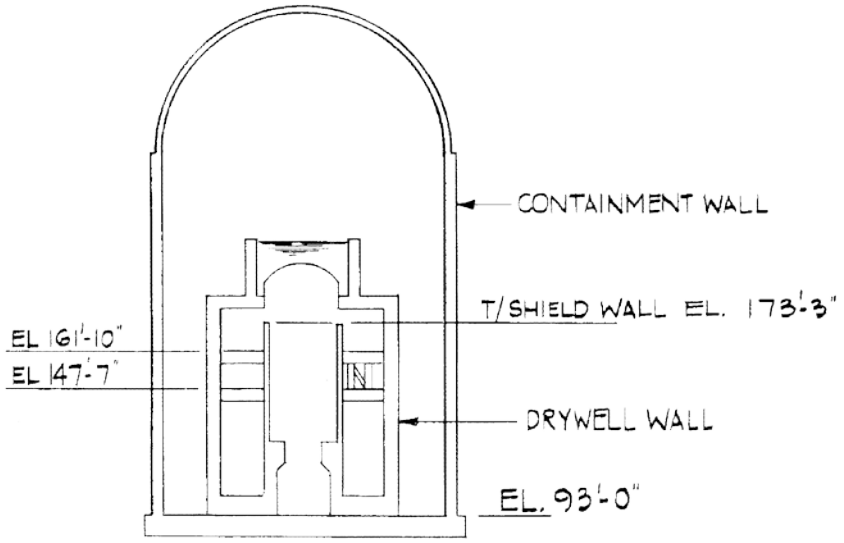
CONTAINMENT INTERNAL FLOOR				
EL. 120'-10"				
STRESS COMPONENT	TYPICAL MEMBER DESIGNATION	GOVERNING CONDITION	STRESS ACTUAL / ALLOWABLE	REMARKS
MAX. BENDING (Fb)	CANTILEVER TWO WEB	D + L + (T_o + T_a) + E' + POOL SWELL	32.8 KSI / 32.4 KSI	
MAX. SHEAR (Fv)	d. o.	d. o.	* / 18.0 KSI	
MAX. BENDING (Fb)	TWO PLATE SPAN (CONTMT. TO DRYWELL)	d. o.	31.4 KSI / 32.4 KSI	
MAX. SHEAR (Fv)	d. o.	d. o.	* / 18.0 KSI	
MAX. BENDING (Fb)	THREE PLATE SPAN (CONTMT. TO DRYWELL)	d. o.	30.6 KSI / 32.4 KSI	
MAX. SHEAR (Fv)	d. o.	d. o.	* / 18.0 KSI	
MAX. BENDING (Fb)	FIVE PLATE SPAN (CONTMT. TO DRYWELL)	d. o.	31.2 KSI / 32.4 KSI	
MAX. SHEAR (Fv)	d. o.	d. o.	* / 18.0 KSI	
	* THE MAGNITUDE OF THIS STRESS IS INSIGNIFICANT IN ALL CASES WHEN COMPARED WITH ALLOWABLE STRESS.			



STRESS TABLE 3.8-21

CONTAINMENT INTERNAL FLOOR				
EL. 135'-4"				
STRESS CATEGORY	TYPICAL MEMBER DESIGNATION	GOVERNING CONDITION	STRESS ALLOWABLE (KSI)	STRESS ACTUAL (KSI)
MAX. BENDING (Fb)	W30 x 116	D+L+(To+Ta)+E'+ POOL SWELL	32.4	24.4
MAX. SHEAR (Fv)	W30 x 116	D+L+(To+Ta)+E'+ POOL SWELL	18.0	11.7
MAX. BENDING (Fb)	W27 x 94	D+L+(To+Ta)+E'+ POOL SWELL	32.4	21.7
MAX. SHEAR (Fv)	W27 x 94	D+L+(To+Ta)+E'+ POOL SWELL	18.0	8.9
MAX. BENDING (Fb)	W24 x 160 WITH B/FLG. PLATE 12"x1"	D+L+(To+Ta)+E'+ POOL SWELL	32.4	25.8
MAX. SHEAR (Fv)	W24 x 160 WITH B/FLG. PLATE 12"x1"	D+L+(To+Ta)+E'+ POOL SWELL	18.0	10.0
MAX. BENDING (Fb)	W14 x 48	D+L+(To+Ta)+E'+ POOL SWELL	32.4	28.8
MAX. SHEAR (Fv)	W14 x 48	D+L+(To+Ta)+E'+ POOL SWELL	18.0	14.3
MAX. BENDING (Fb)	W12 x 40	D+L+(To+Ta)+E'+ POOL SWELL	32.4	27.4
MAX. SHEAR (Fv)	W12 x 40	D+L+(To+Ta)+E'+ POOL SWELL	18.0	14.4

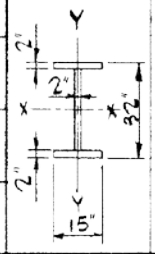
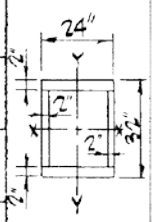
PAGE 1 OF 2.

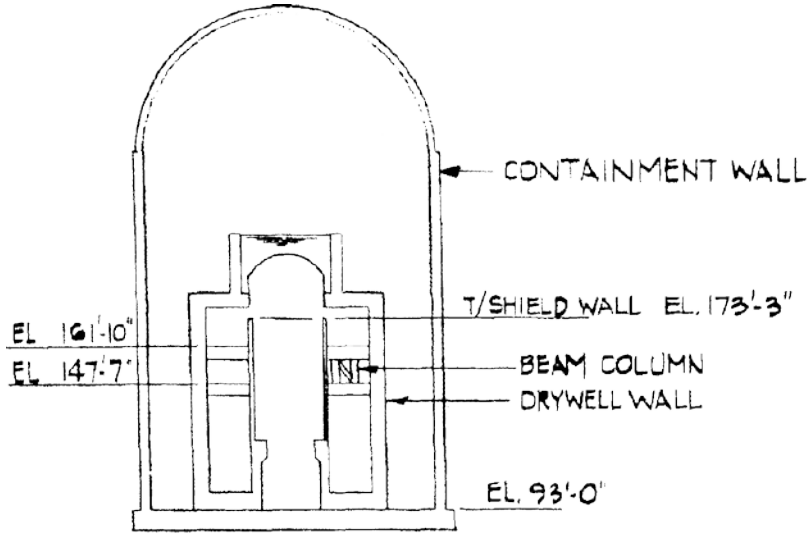


STRESS TABLE 3.8-22

NOTE: ALL STEEL ASTM A-36

* ALLOWABLE STRESSES ARE INCREASED FOR DYNAMIC LOADING [REF (1) BC-TOP-9A, REV. 2, SEPT. 1974, "DESIGN OF STRUCTURES FOR MISSLE IMPACT", TABLE 4-1, (ii) DESIGN OF STRUCTURES TO RESIST NUCLEAR WEAPONS EFFECT - BY ASCE MANUALS OF ENGINEERING PRACTICE NO. 42]

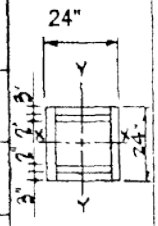
CONTAINMENT DRYWELL FLOORS						
EL 147'-7"						
STRESS CATEGORY	TYPICAL MEMBER DESIGNATION	GOVERNING EQUATION	STRESS ACTUAL (KSI)	STRESS ALLOW (KSI)	ACTUAL $\frac{f_a}{F_a}+\frac{f_{bx}}{F_{bx}}+\frac{f_{by}}{F_{by}}$	ALLOW $\frac{f_a}{F_a}+\frac{f_{bx}}{F_{by}}+\frac{f_{by}}{F_{by}}$
MAX. AXIAL		$D+L+(T_o+T_a)+R+E'$	$f_a = 6.68$	$F_a = 30.57^*$	0.856	1.0
MAX. BENDING IN X-DIR			$f_{bx} = 23.72$	$F_{bx} = 37.26^*$		
MAX. BENDING IN Y-DIR			—	—		
MAX. SHEAR			$f_v = 11.11$	$F_v = 20.16$	N.A.	N.A.
MAX. AXIAL		$D+L+(T_o+T_a)+R+E'$	$f_a = 4.17$	$F_a = 32.47$	0.966	1.0
MAX. BENDING IN X-DIR			$f_{bx} = 28.14$	$F_{bx} = 37.26^*$		
MAX. BENDING IN Y-DIR			$f_{by} = 3.09$	$F_{by} = 37.26^*$		
MAX. SHEAR			$f_v = 11.03$	$F_v = 18.0$	N.A.	N.A.

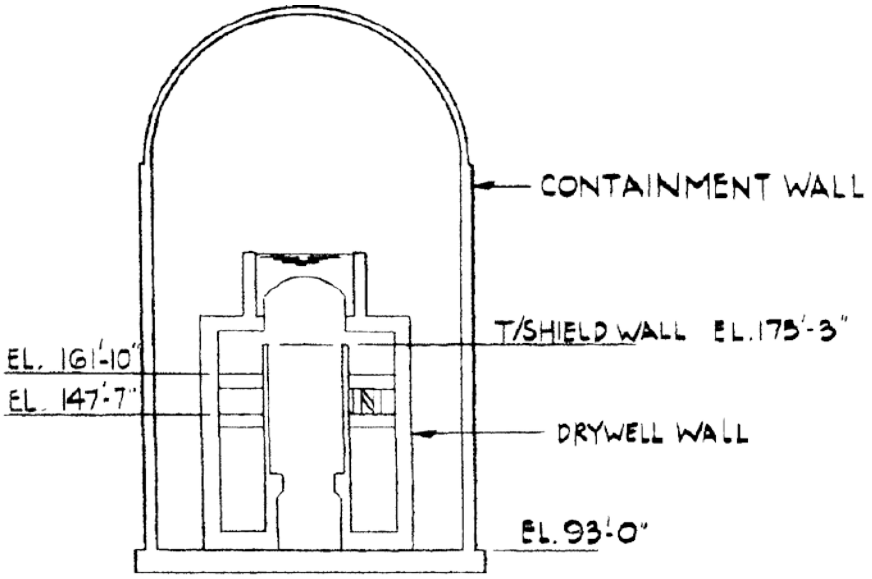


STRESS TABLE 3.8-22 (Cont.)

NOTE : ALL STEEL ASTM A-36

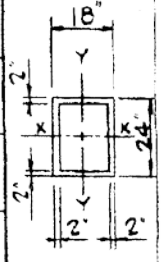
* ALLOWABLE STRESSES ARE INCREASED FOR DYNAMIC LOADING [REF. (1) BC-TOP-9A, REV. 2 SEPT 1974, "DESIGN OF STRUCTURES FOR MISSILE IMPACT" TABLE 4-1, (11) DESIGN OF STRUCTURES TO RESIST NUCLEAR WEAPONS EFFECT - BY ASCE MANUALS OF ENGINEERING PRACTICE NO. 42]

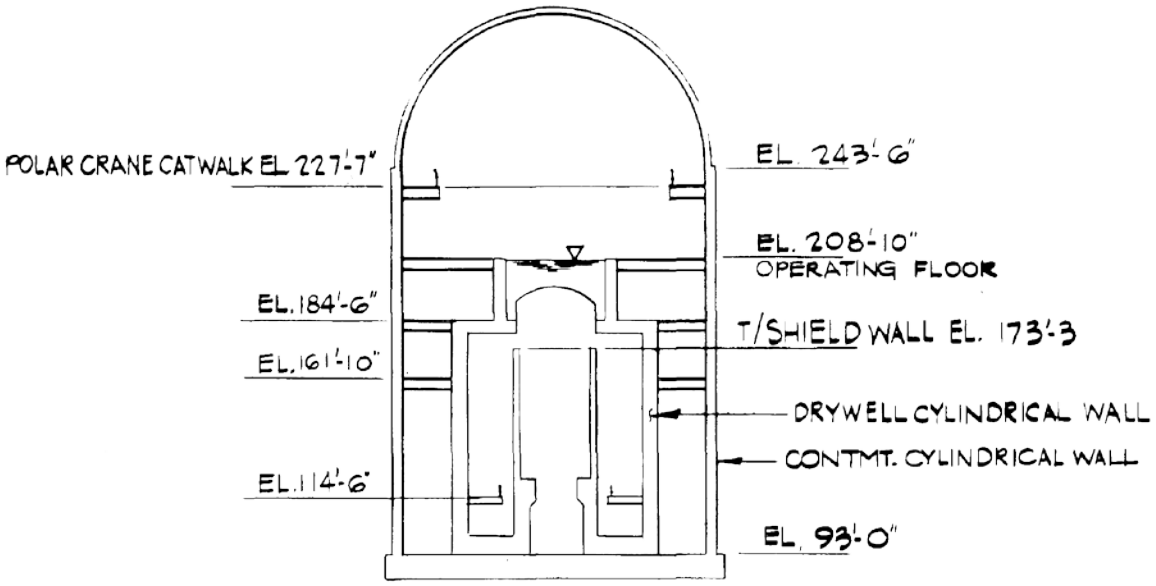
CONTAINMENT DRYWELL FLOORS						
EL. 147'-7"						
STRESS CATEGORY	TYPICAL MEMBER DESIGNATION	GOVERNING EQUATION	STRESS ACTUAL (KSI)	STRESS ALLOW (KSI)	ACTUAL $\frac{f_a}{F_a}+\frac{f_{bx}}{F_{bx}}+\frac{f_{by}}{F_{by}}$	ALLOW $\frac{f_a}{F_a}+\frac{f_{bx}}{F_{by}}+\frac{f_{by}}{F_{by}}$
MAX. AXIAL		$D+L+(T_o+T_a)+R+E'$	—	—	0.956	1.0
MAX. BENDING IN X-DIR			$f_{bx}=28.76$	$F_{bx}=37.26$ *		
MAX. BENDING IN Y-DIR			$f_{by}=6.84$	$F_{by}=37.26$ *		
MAX. SHEAR	BEAM COLUMN		$f_v=8.74$	$F_v=20.16$	N.A.	N.A.
MAX. AXIAL	W30×132	$D+L+T_o$	$f_a=0.41$	$F_a=12.0$	1.0	1.0
MAX. BENDING IN X-DIR			$f_{bx}=17.46$	$F_{bx}=22.0$		
MAX. BENDING IN Y-DIR			$f_{by}=4.03$	$F_{by}=22.0$		
MAX. SHEAR			$f_v=4.75$	$F_v=14.40$	N.A.	N.A.



STRESS TABLE 3.8-23

NOTE: ALL STEEL ASTM A-36

CONTAINMENT DRYWELL FLOORS						
EL. 161'-10"						
STRESS CATEGORY	TYPICAL MEMBER DESIGNATION	GOVERNING EQUATION	STRESS ACTUAL (KSI)	STRESS ALLOW (KSI)	ACTUAL $\frac{f_a}{F_a}+\frac{f_{bx}}{F_{bx}}+\frac{f_{by}}{F_{by}}$	ALLOW $\frac{f_a}{F_a}+\frac{f_{bx}}{F_{by}}+\frac{f_{by}}{F_{by}}$
MAX. AXIAL		$D+L+(T_o+T_a)+R+E'$	f_a= 5.9	F_a=30.6	0.85	1.0
MAX. BENDING IN X-DIR			f_{bx}=16.0	F_{bx}=32.4		
MAX. BENDING IN Y-DIR			f_{by}= 5.1	F_{by}=32.4		
MAX. SHEAR			f_v= 8.3	F_v=18.0	N.A.	N.A.
MAX. BENDING	W14×38	$D+L+T_o$	f_{bx}=22.2	F_{bx}=24	N.A.	N.A.
MAX. SHEAR			f_v=4.7	F_v=14.4	N.A.	N.A.
MAX. BENDING	W18×50	$D+L+T_o$	f_{by}=23.7	F_{by}=24	N.A.	N.A.
MAX. SHEAR			f_v=5.8	F_v=14.4	N.A.	N.A.



STRESS TABLE 3.8-24

CONTAINMENT OPERATING AND MISCELLANEOUS FLOORS				
EL's. 114'-6"; 161'-10"; 184'-6"; 208'-10"; AND 227'-7"				
STRESS COMPONENT	TYPICAL MEMBER DESIGNATION	GOVERNING CONDITION	STRESS ACTUAL / ALLOWABLE	REMARKS
MAX. BENDING (Fb)	W12×27	$D+L+(T_o+T_a)+E'$	29.9 KSI / 32.4 KSI	
MAX. SHEAR (Fv)	W12×27	d. o.	4.6 KSI / 18.0 KSI	
MAX. BENDING (Fb)	W24×76	d. o.	29.6 KSI / 32.4 KSI	
MAX. SHEAR (Fv)	W24×76	d. o.	3.9 KSI / 18.0 KSI	
MAX. BENDING (Fb)	W30×108	d. o.	29.4 KSI / 32.4 KSI	
MAX. SHEAR (Fv)	W30×108	d. o.	4.1 KSI / 18.0 KSI	
MAX. BENDING (Fb)	W36×300	d. o.	27.0 KSI / 32.4 KSI	
MAX. SHEAR (Fv)	W36×300	d. o.	6.0 KSI / 18.0 KSI	

TABLE 3.8-25: TOLERANCES FOR FORMWORK AND REINFORCEMENT

Formed Surface	Tolerance (in.)
	ACI 301-72, Table 4.3.1
A. Variation from plumb:	
1. In the lines and surfaces of columns, piers, walls, and in arrises:	
In any 10 ft. length	1/2
2. For exposed corner columns, control-joint grooves, and other conspicuous lines:	
In any 20 ft. length	1/2
Maximum for the entire length	1
B. Variation from the level or from the grades specified in the contract documents:	
1. In slab soffits, ceilings, beam soffits, and in arrises, measurement before removal of supporting shores:	
In any 10 ft. length	1/2
In any bay or in any 20 ft. length	5/8
Maximum for the entire length	1
2. In exposed lintels, sills, parapets, horizontal grooves, and other conspicuous lines:	
In any bay or in 20 ft. length	3/8
Maximum for the entire length	1

*Not changed from ACI 301; shown for completeness only.

TABLE 3.8-25: TOLERANCES FOR FORMWORK AND REINFORCEMENT (CONTINUED)

C.	Variation in the sizes of sleeves, floor openings, and wall openings	+ 1/2, -1/4 in.*
	Variation in the location of sleeves, floor openings, and wall openings	± 1/2 in.
D.	Variation in cross-sectional dimensions of columns and beams and in the thickness of slabs and walls	+1, -1/4 in.*

ACI 318-71, Chapter 7

Chapter 7, Details of Reinforcement, Section 7.3, Placing Reinforcement, Subsection 7.3.2.1, lists tolerances for clear concrete protection and for depth, d, in flexural members, walls, and compression members. The following will be used in place of that given in the code:

Where Depth Is	Tolerance (in.)
8 in. or less:	+1/2, - 1/4
More than 8 in. but less than 24 in.	+3/4, - 3/8
24 in. or more:	+1, - 1/2

The following sentence shall apply in place of subsection 7.3.2.2: For longitudinal location of bends and ends of bars: ±3 in. except at discontinuous ends of members, where tolerance shall be ±1/2 in.

TABLE 3.8-26: LINER PLATE STRESSES DUE TO NEGATIVE PRESSURE LOADS

Condition	Stress at Max Deflection		
Static Solution		21.1 ksi	
Dynamic Solution Frequencies	5 cps	8 cps	11 cps
1/16" Concrete Crack	18.8 ksi	15.7 ksi	14.7 ksi
1/32" Concrete Crack	4.1 ksi	2.9 ksi	2.2 ksi

TABLE 3.8-27: INSTRUMENTED BAR STRAINS DRYWELL SIT

Elevation	Azimuth	Direction of Stress	Bar Location	Bar (1) Strain in./in. x 10^{-6}	
				Predicted	Actual
94'-0"	90°	Vertical	-	172	13
		Hoop	-	69	-8
	270°	Vertical	-	172	13
		Hoop	-	69	-1
140'-0"	90°	Vertical	Inside face	517	-1
			Center	345	0
			Outside face	138	-5
			Inside face	552	189
		Hoop	Center	517	193
			Outside face	483	170
			Inside face	586	7
	270°	Vertical	Center	379	7
			Outside face	103	4
			Inside face	483	89
		Hoop	Center	483	28
			Outside face	448	35

(1) Local peak strains could be 3 times values shown.

TABLE 3.8-28: LINER PLATE STRAINS DRYWELL SIT

Elevation	Azimuth	Direction of Stress	Strain Gauge Location	Strain (1) in./in. x 10^{-6} Predicted	Actual
94'-0"	90°	Vertical not measured	Inside liner	345	**
			Outside liner	-17	**
		Hoop	Inside liner	172	50
			Outside liner	-34	*
	270°	Vertical not measured	Inside liner	345	**
			Outside liner	-17	**
		Hoop	Inside liner	172	-20
			Outside liner	-34	-6

(1) Local peak strains could be 3 times values shown.

VERTICAL DEFLECTIONS
AT AZIMUTH 270°
DRYWELL SIT

Radial Location from Containment C_L	Deflection (2) (in.)	
	Predicted	Actual
~22'-3"	.074	.049
~28'-9"	.037	.034

(2) Tolerance on deflections is -60% to +40%.

* Strange gauge did not operate

** Not measured

TABLE 3.8-29: RADIAL DEFLECTIONS DRYWELL SIT

		Radial Deflection (1) (in.)			
Elevation	Azimuth	Predicted	Actual	24-Hour Residual	% Recovery
120'-0"	85°	0.020	0.019	.003	*
	150°	0.020	0.031	.005	84
	175°	0.020	0.031	.006	81
	265°	0.020	0.020	.004	*
	330°	0.020	0.028	.005	82
	355°	0.020	0.024	.004	*
140'-0"	30°	0.020	.024	.008	*
	90°	0.030	.017	.008	*
	155°	0.020	.030	.008	74
	210°	0.020	.033	.006	82
	270°	0.030	.022	.008	*
	335°	0.020	.026	.0011	*
167'-0"	30°	0.010	.012	.007	*
	90°	0.030	.012	.006	*
	150°	0.010	.013	.005	*
	210°	0.010	.009	.002	*
	270°	0.030	.018	.004	*
	330°	0.010	.000	.007	*

(1) Tolerance on deflections is -60%, +40%.

* Within above tolerance band.

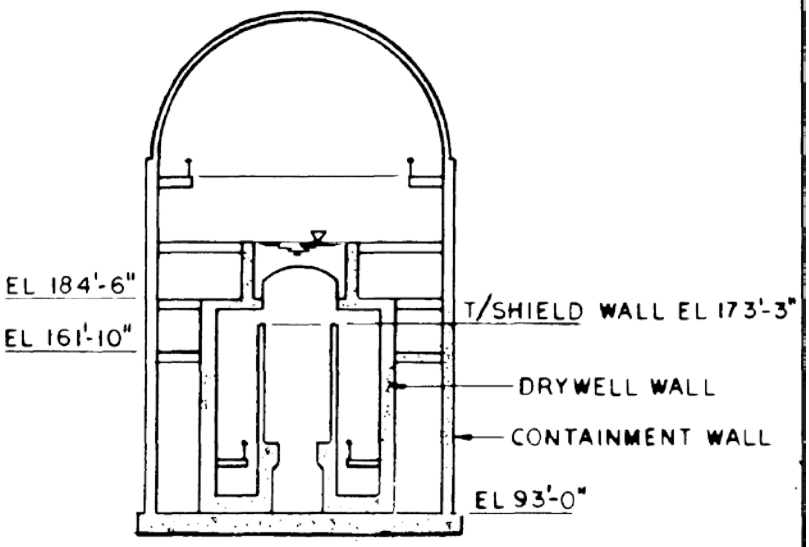
TABLE 3.8-30: PREDICTED CRACK DATA (1) DRYWELL SIT

	Crack Widths (in.)			
Azimuth	Horizonal Cracks (2)	Vertial Cracks	Vertical Crack Length	Vertical Crack Spacing
106° 30'	-	0.035	Approx. 30 ft (Middle 1/3 of drywell height)	24-60 in.
172° 15'	-	0.040		
274° 15'	-	0.030		

(1) Values determined from Refs. 11, 12, and 13.

(2) Expect preexisting shrinkage cracks. Stress cracks in horizontal direction should be minimal and occur predominately at embeds, penetrations, and regions of discontinuity.

Note: No cracks were observed in the mapping areas during the test.



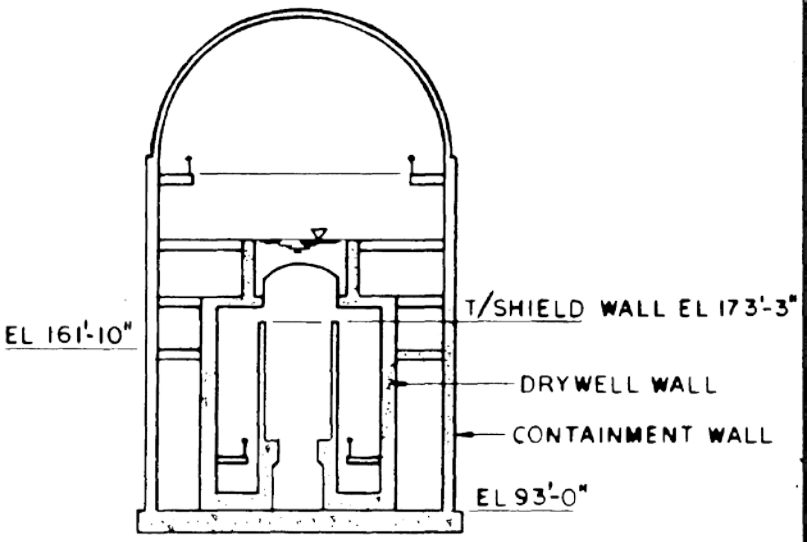
STRESS TABLE 3.8-31

NOTE: ALL STEEL ASTM A-36

CONTAINMENT OPERATING AND MISCELLANEOUS FLOORS

EL 184'-6"

STRESS CATEGORY	TYPICAL MEMBER DESIGNATION	LOAD COMBINATION	STRESS ACTUAL (KSI)	STRESS ALLOW (KSI)	ACTUAL $\frac{f_a}{F_a}+\frac{f_{bx}}{F_{bx}}+\frac{f_{by}}{F_{by}}$	ALLOW $\frac{f_a}{F_a}+\frac{f_{bx}}{F_{bx}}+\frac{f_{by}}{F_{by}}$
MAX. AXIAL	W24x84	D+L+T_o+E (AT 1.0 F_s)	f_a=0.35	F_a=19.42	0.7303	1.0
MAX. BENDING IN X-DIR			f_{bx}=8.42	F_{bx}=24		
MAX. BENDING IN Y-DIR			f_{by}=9.76	F_{by}=27		
MAX. SHEAR			f_{vx}=3.09	F_{vx}=14.4	———	———
MAX. AXIAL		D+L+T_o+E (AT 1.25 F_s) PER FSAR SECT 3.8.3.5.3	f_a=0.35	F_a=24.27	0.5842	1.0
MAX. BENDING IN X-DIR			f_{bx}=8.42	F_{bx}=30.0		
MAX. BENDING IN Y-DIR			f_{by}=9.76	F_{by}=33.75		
MAX. SHEAR			f_{vx}=3.09	F_{vx}=18.0	———	———



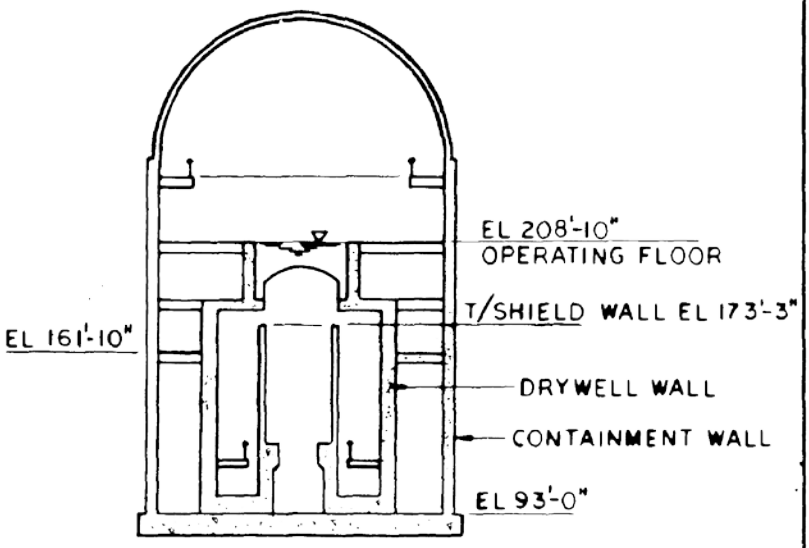
STRESS TABLE 3.8-32

NOTE: ALL STEEL ASTM A-36

CONTAINMENT OPERATING AND MISCELLANEOUS FLOORS

EL 161'-10"

STRESS CATEGORY	TYPICAL MEMBER DESIGNATION	LOAD COMBINATION	STRESS ACTUAL (KSI)	STRESS ALLOW (KSI)	ACTUAL $\frac{f_a}{F_a}+\frac{f_{bx}}{F_{bx}}+\frac{f_{by}}{F_{by}}$	ALLOW $\frac{f_a}{F_a}+\frac{f_{bx}}{F_{bx}}+\frac{f_{by}}{F_{by}}$
MAX. AXIAL	W24x68	D+L+T_o+E (AT 1.0 F_s)	f_a=0.2	F_a=18.95	0.834	1.0
MAX. BENDING IN X-DIR			f_{bx}=10.7	F_{bx}=24.0		
MAX. BENDING IN Y-DIR			f_{by}=10.2	F_{by}=27.0		
MAX. SHEAR			f_{v_x}=3.0	F_{v_x}=14.4	———	———
MAX. AXIAL		D+L+T_o+E (AT 1.25 F_s) PER FSAR SECT 3.8.3.5.5.3	f_a=0.20	F_a=23.69	0.667	1.0
MAX BENDING IN X-DIR			f_{bx}=10.7	F_{bx}=30.0		
MAX. BENDING IN Y-DIR			f_{by}=10.2	F_{by}=33.75		
MAX. SHEAR			f_{v_x}=3.0	F_{v_x}=18.0	———	———



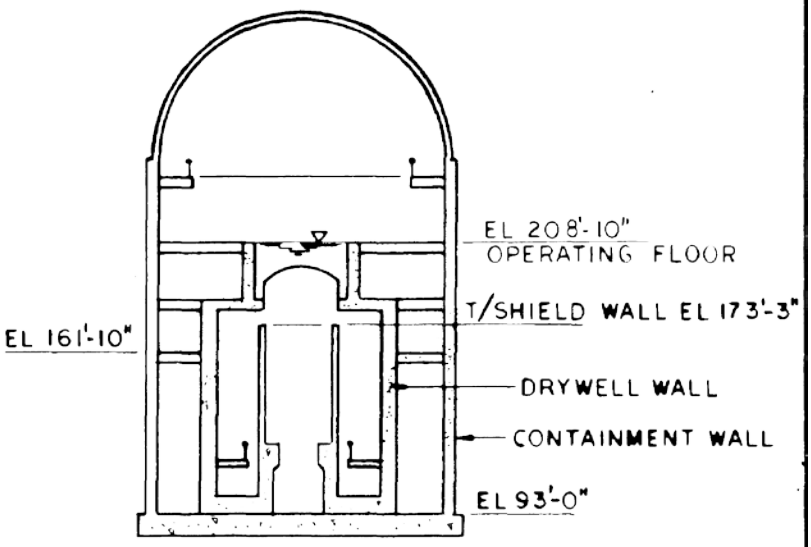
STRESS TABLE 3.8-33

NOTE: ALL STEEL ASTM A-36

CONTAINMENT OPERATING AND MISCELLANEOUS FLOORS

EL 208'-10"

STRESS CATEGORY	TYPICAL MEMBER DESIGNATION	LOAD COMBINATION	STRESS ACTUAL (KSI)	STRESS ALLOW (KSI)	ACTUAL $\frac{f_a}{F_a}+\frac{f_{bx}}{F_{bx}}+\frac{f_{by}}{F_{by}}$	ALLOW $\frac{f_a}{F_a}+\frac{f_{bx}}{F_{bx}}+\frac{f_{by}}{F_{by}}$
MAX. AXIAL	W36x300 (BM-202)	D+L+T_o+E (AT 1.00 F_s)	f_a=0.7	F_a=19.35	0.851	1.0
MAX. BENDING IN X-DIR			f_{bx}=18.50	F_{bx}=24.0		
MAX. BENDING IN Y-DIR			f_{by}=1.18	F_{by}=27.0		
MAX. SHEAR			f_{vx}=5.17	F_{vx}=14.4	———	———
MAX. AXIAL		D+L+T_o+E (AT 1.25 F_s) PER FSAR SECT 3.8.3.5.5.3	f_a=0.7	F_a=24.187	0.681	1.0
MAX BENDING IN X-DIR			f_{bx}=18.50	F_{bx}=30.0		
MAX. BENDING IN Y-DIR			f_{by}=1.18	F_{by}=33.75		
MAX. SHEAR			f_{vx}=5.17	F_{vx}=18.0	———	———



STRESS TABLE 3.8-34

CONTAINMENT OPERATING AND MISCELLANEOUS FLOORS

EL 208'-10"

STRESS CATEGORY	TYPICAL MEMBER DESIGNATION	LOAD COMBINATION	STRESS ACTUAL (KSI)	STRESS ALLOW (KSI)	ACTUAL $\frac{f_a}{F_a}+\frac{f_{bx}}{F_{bx}}+\frac{f_{by}}{F_{by}}$	ALLOW $\frac{f_a}{F_a}+\frac{f_{bx}}{F_{bx}}+\frac{f_{by}}{F_{by}}$
MAX. AXIAL	W36x300 (BM-205)	D+L+T_o+E (AT 1.00 F_S)	f_a=0.8	F_a=18.81	1.033	1.0
MAX. BENDING IN X-DIR			f_{bx}=22.0	F_{bx}=24		
MAX. BENDING IN Y-DIR			f_{by}=2.0	F_{by}=2.7		
MAX. SHEAR			f_{v_x}=5.9	F_{v_x}=14.4	———	———
MAX. AXIAL		D+L+T_o+E (AT 1.25 F_S) PER FSAR SECT 3.8.3.5.5.3	f_a=0.8	F_a=23.26	0.827	1.0
MAX BENDING IN X-DIR			f_{bx}=22.0	F_{bx}=30.0		
MAX BENDING IN Y-DIR			f_{by}=2.0	F_{by}=3.375		
MAX. SHEAR			f_{v_x}=5.9	F_{v_x}=18.0	———	———

TABLE 3.8-35: COMPARISON OF LOAD COMBINATIONS - CONCRETE STRUCTURES

	Grand Gulf (GG)	Stress Limit		Standard Review Plan (SRP)	Stress Limit	Remarks	Comparison
1	1.4D + 1.7L	U	1	1.4D + 1.7L	U		GG Comb.(1)and SRP Comb.(1) are identical.
1a	0.75 (1.4D + 1.7L + $1.4T_o$)	U	1b	0.75 (1.4D + 1.7L + $1.7T_o$ + $1.7R_o$)	U	(i): GG(T_o) is equivalent to SRP (T_o + R_o). SRP (2b) is more severe than SRP (1b)	SRP Comb.(1b) is more conservative than GG Comb. (1a). See Stress Table 3.8-37
2	0.75 (1.4D + 1.7L + $1.4T_o$ + 1.9E)	U	2	1.4D + 1.7L + 1.9E	U	Same as (i); GG takes 0.75 reduction of SRP Comb.(2); GG Load Factor = 1.4 for T_o, SRP Load Factor = 1.7 for (T_o + R_o).	SRP Comb's(2) & (2b) are more conservative than GG Comb. (2). See Stress Table 3.8-37
			2b	0.75 (1.4D + 1.7L + 1.9E + 1.7 T_o + $1.7R_o$)			
3	0.75 (1.4D + 1.7L + $1.4T_o$ + 1.7W)	U	3	1.4 D + 1.7L + 1.7W	U	(ii): (W) is not considered for interior walls and slabs; for exterior walls, extreme environmental and abnormal conditions control the design.	SRP Combinations (3) & (3b) do not control the design. SRP Comb's (2) & (2b) are more severe.
			3b	0.75 (1.4D + 1.7L + 1.7W + $1.7T_o$+ $1.7R_o$)			
4	D + L + 1.0 (T_a + T_o) + R + E' + 1.0P	U	4	D + L + T_o + R_o + E'	U	(iii) by DEF'N; GG(R) is equiv. to SRP (R_a+Y_j+Y_r+Y_m), SRP (T_a) is equiv	By observation, GG Comb.(4) is equivalent to SRP Comb.(8).

TABLE 3.8-35: COMPARISON OF LOAD COMBINATIONS - CONCRETE STRUCTURES (CONTINUED)

	Grand Gulf (GG)	Stress Limit		Standard Review Plan (SRP)	Stress Limit	Remarks	Comparison
			8	$D + L + T_a + R_a + 1.0P_a + 1.0 (Y_j + Y_r + Y_m) + E'$	U	to GG ($T_a + T_o$), GG (T_o) is equiv. to SRP ($T_o + R_o$), GG(P) is equiv. to SRP (Pa). SRP Comb. (8) is more severe than SR Comb than SRP Comb.(4).	
5	$D + L + W' + T_o$	U	5	$D + L + T_o + R_o + W_t$	U	Same as (i).	By observation, GG Comb.(5) is equivalent to SRP Comb.(5).
6	$D + L + 1.0 (T_a +$ To) + R + 1.25 (E or W) + 1.25P	U	6	$D + L + T_a +$ Ra + $1.5P_a$	U	Same as (ii) and (iii). GG Comb. (6) with (E) is more severe than GG Comb.(6) with	By observation, GG Comb.(6)with earthquake loads considered is equivalent to
			7	$D + L + T_a + R_a + 1.25P_a + 1.0 (Y_j + Y_r + Y_m) + 1.25E$	U	(W). SRP Comb.(7) is more severe than SRP Comb.(6).	SRP (Comb.(7)
7	$D + L + A + T_o$	U		Not applicable		There is no SRP load combination to compare directly with GG Comb.(7)	Not applicable

Comb. - combination
DEF'N - definition
equiv. - equivalent

TABLE 3.8-36: COMPARISON OF LOAD COMBINATIONS - STEEL STRUCTURES

Grand Gulf (GG)	Stress Limit	Standard Review Plan (SRP)	Stress Limit	Remarks	Comparison
1 $D + L + T_o$	S	1a $D + L + T_o + R_o$	1.5 S	(i) Grand Gulf (T_o) is equivalent to SRP (T_o + R_o).	By observation, GG Comb. (1) is more conservative than SRP Comb's (1a) and (1).
		1 $D + L$			
2 $D + L + T_o + E$	1.25 S	2a $D + L + T_o + R_o + E$	1.5 S	Same as (i)	By observation, GG Comb. (2) is more conservative than SRP Comb. (2a).
		2 $D + L + E$	S	Same as (i)	SRP Comb. (2) is more conservative than GG Comb.(2). See Stress Table 3.8-38.
3 $D + L + T_o + W$	1.33 S	3a $D + L + T_o + R_o + W$	1.5 S	Same as (i); for interior steel, wind load, (W), is not considered.	Because SRP Load Comb's (2) & (2a) are more severe, SRP Comb's (3) & (3a) do not control the design.
		3 $D + L + W$	S		
4 $D + L + (T_o + T_a) + E'$	Note 1	4 $D + L + T_o + R_o + E'$	1.6 S	Same as (i); GG Comb.(4) includes T_a while SRP Comb. (4) does not.	By observation, GG Comb. (4) is more conservative than SRP Comb. (4).
5 $D + L + T_o + W'$	Note 1	5 $D + L + T_o + R_o + W_t$	1.6 S	Same as (i); stress limits from Note 1 are lower than 1.6 S. Tornado missile loads are designed using the methods described in subsection 3.5.3.1.	By observation, GG Comb. (5) is more conservative than SRP Comb. (5).

TABLE 3.8-36: COMPARISON OF LOAD COMBINATIONS - STEEL STRUCTURES (CONTINUED)

	Grand Gulf (GG)	Stress Limit		Standard Review Plan (SRP)	Stress Limit	Remarks	Comparison
6	$D + L + A + T_o$	Note 1		Not applicable		There is no SRP load combination to compare directly with GG Comb. (6).	Not applicable
7	$D + L + R + (T_o + T_a) + E' + P$	Note 2	6	$D + L + T_a + R_a + P_a$	1.6 S	By definition: SRP (T_a) is equiv. to GG ($T_o + T_a$); GG (P) is equiv.to SRP (P_a); GG (R) is equiv. to SRP ($R_a + Y_j + Y_r + Y_m$). Note 2 Is more severe than SRP stress limits [1.6 S and 1.7 S]	By observation, GG Comb. (7) is more conservative than SRP combinations to(6), (7), & (8).
			7	$D + L + T_a + R_a + P + 1.0 (Y_j + Y_r + Y_m) + E$	1.6 S		
			8	$D + L + T_a + R_a + P_a + 1.0 (y_j + Y_r + Y_m) + E$	1.7 S		

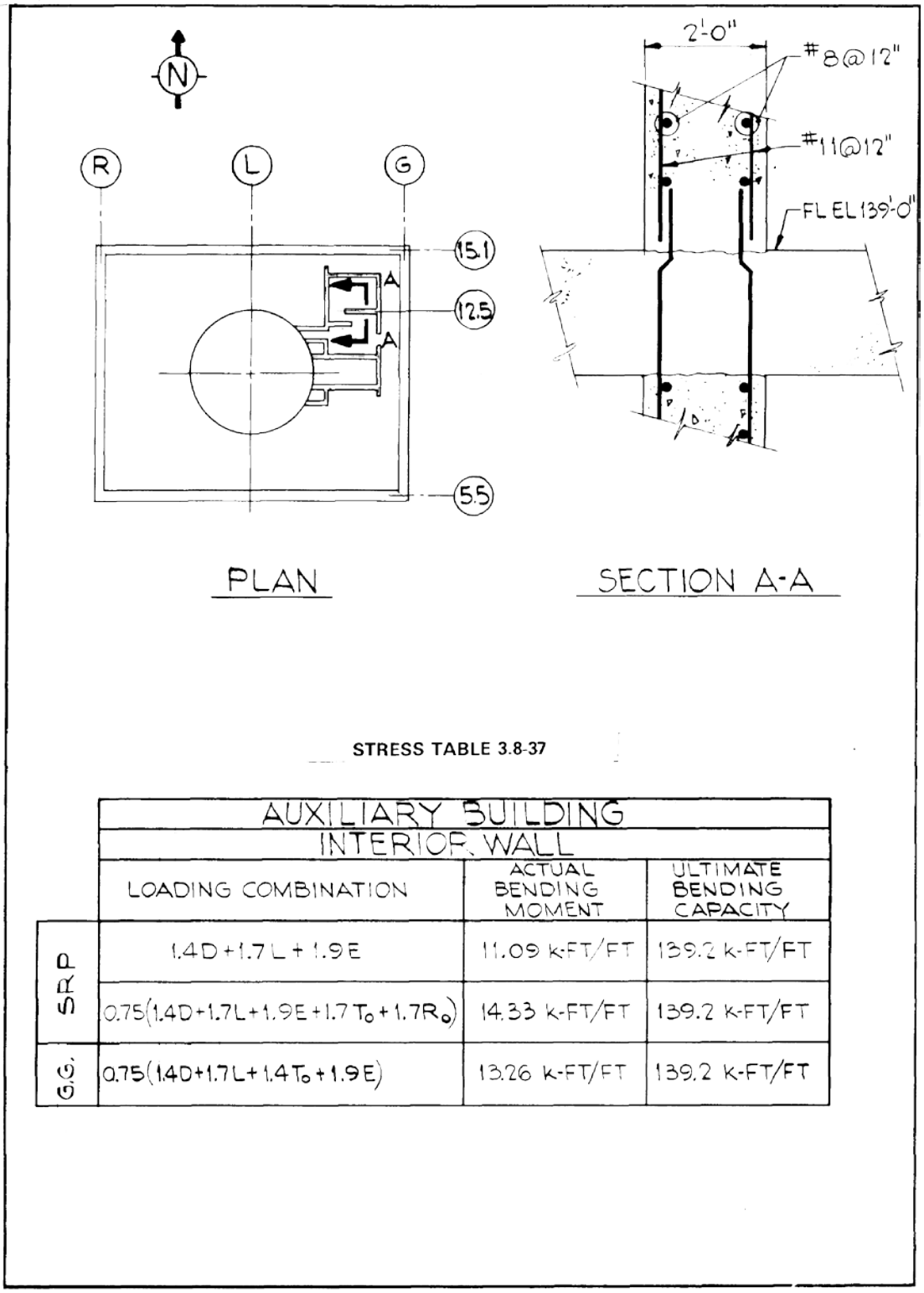
Notes: (1) In no case shall the allowable stress exceed 0.9 F in bending, 0.85 F_y in axial tension, and 0.5 F_y in shear. Where F_s is governed by requirements of stability (local and lateral buckling), f_s shall not exceed 1.4 F_s.

(2) Stress values shall be limited to 1.5 F_s except for web shear, where stress limits shall be limited to 1.4 F_s.

equiv. - equivalent

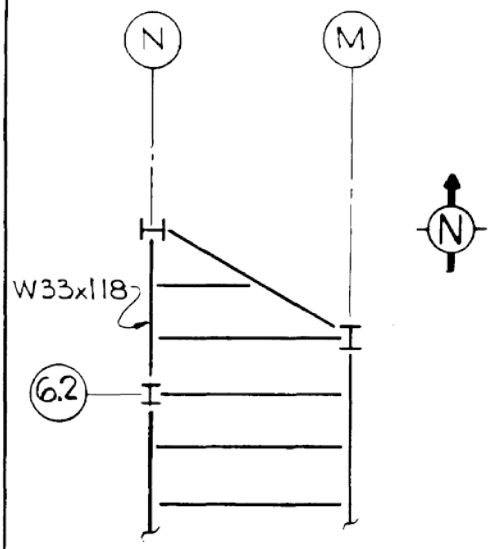
Comb's. - combinations

Comb. - combination

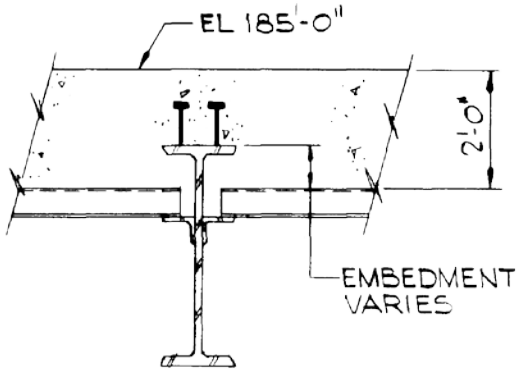


STRESS TABLE 3.8-37

	AUXILIARY BUILDING INTERIOR WALL		
	LOADING COMBINATION	ACTUAL BENDING MOMENT	ULTIMATE BENDING CAPACITY
S.R.P.	1.4D + 1.7L + 1.9E	11.09 k-FT/FT	139.2 k-FT/FT
S.R.P.	0.75(1.4D+1.7L+1.9E+1.7T_o+1.7R_o)	14.33 k-FT/FT	139.2 k-FT/FT
G.G.	0.75(1.4D+1.7L+1.4T_o+1.9E)	13.26 k-FT/FT	139.2 k-FT/FT



PARTIAL PLAN
EL 185'-0"



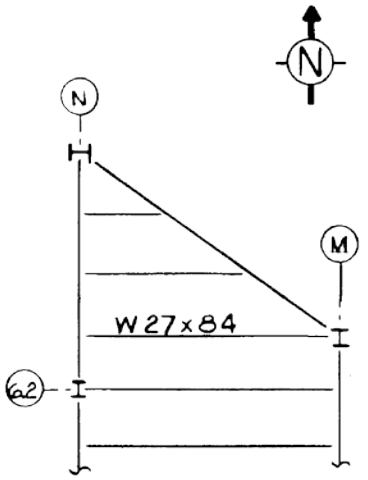
TYP COMPOSITE SECTION

STRESS TABLE 3.8-38

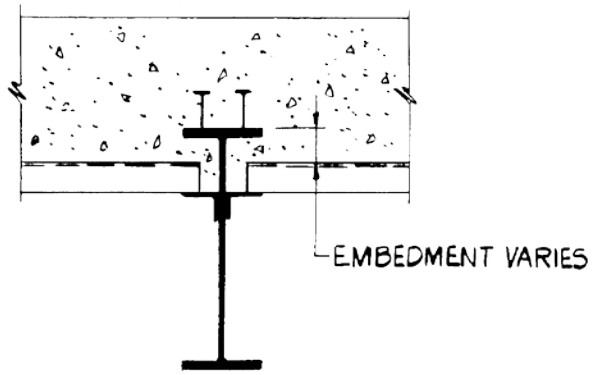
NOTE: ALL STEEL ASTM A-36

AUXILIARY BUILDING
FLOOR FRAMING - STEEL STRESSES

STRESS CATEGORY	MEMBER DESIGNATION	LOAD COMBINATION	ACTUAL STRESS f	ALLOW. STRESS F	f/F
MAX BENDING IN X-DIR	W33x118	1.0D+1.0L < S	f_{bx}=16.24KSI	F_{bx}=24.0KSI	0.68
		1.0D+1.0L+1.0E<S	f_{bx}=17.55KSI		0.73
MAX SHEAR		1.0D+1.0L < S	f_v=0.95KSI	F_v=14.4KSI	0.48
		1.0D+1.0L+1.0E<S	f_v=7.70KSI		0.53
MAX BENDING IN X-DIR		1.0D+1.0L+1.0E<1.25S PER GG FSAR SECT 3.8.6.3	f_{bx}=17.55KSI	F_{bx}=30.0KSI	0.59
MAX SHEAR			f_v=7.70KSI	F_v=18.0KSI	0.43



PARTIAL PLAN - EL 185'-0"

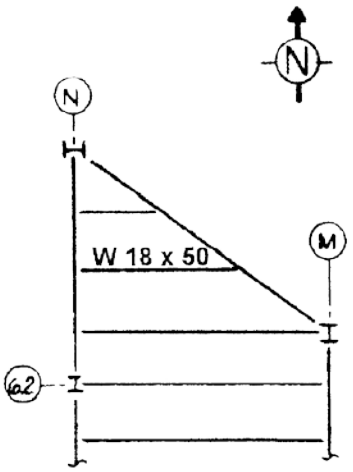


TYPICAL COMPOSITE SECTION

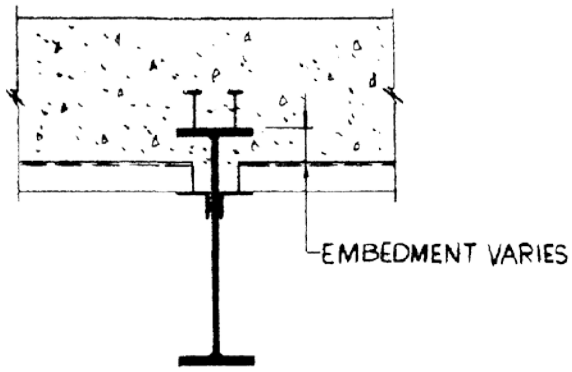
STRESS TABLE 3.8-39

NOTE: ALL STEEL ASTM A-36

AUXILIARY BUILDING FLOOR FRAMING STEEL STRESSES					
STRESS CATEGORY	MEMBER DESIGNATION	LOAD COMBINATION	ACTUAL STRESS f (KSI)	ALLOW STRESS F (KSI)	$\frac{f}{F}$
MAX. BENDING IN X-DIR	W27x84	1.0D+1.0L (AT 1.0S)	f_{bx} = 12.38	F_{bx} = 24.0	0.52
		1.0D+1.0L+1.0E (AT 1.0S) PER SRP SECT 3.8.4 II	f_{bx} = 13.23		0.55
MAX. SHEAR		1.0D+1.0L (AT 1.0S)	f_v = 4.42	F_v = 14.4	0.31
		1.0D+1.0L+1.0E (AT 1.0S)	f_v = 4.89		0.34
MAX. BENDING IN X-DIR		1.0D+1.0L+1.0E (AT 1.25S) PER GG FSAR SECT 3.8.6.3	f_{bx} = 13.23	F_{bx} = 30.0	0.44
MAX. SHEAR			f_v = 4.89	F_v = 18.0	0.27



PARTIAL PLAN - EL 185'-0"



TYPICAL COMPOSITE SECTION

STRESS TABLE 3.8-40

NOTE: ALL STEEL ASTM A-36

AUXILIARY BUILDING FLOOR FRAMING STEEL STRESSES					
STRESS CATEGORY	MEMBER DESIGNATION	LOAD COMBINATION	ACTUAL STRESS f (KSI)	ALLOW STRESS F (KSI)	$\frac{f}{F}$
MAX BENDING IN X-DIR	W 18 x 50	1.0D+1.0L (AT 1.0S)	f_{bx} = 13.88	F_{bx} = 24.0	0.58
		1.0D+1.0L+1.0E (AT 1.0S) PER SRP SECT 3.8.4 II	f_{bx} = 14.84		0.62
MAX. SHEAR		1.0D+1.0L(AT 1.0S)	f_v = 5.77	F_v = 14.4	0.40
		1.0D+1.0L+1.0E(AT 1.0S)	f_v = 6.39		0.44
MAX. BENDING IN X-DIR		1.0D+1.0L+1.0E (AT 1.25S) PER GG FSAR SECT 3.8.6.3	f_{bx} = 14.84	F_{bx} = 30.0	0.49
MAX. SHEAR			f_v = 6.39	F_v = 18.0	0.36

TABLE 3.8-41: VERTICAL DEFLECTIONS OF CONTAINMENT CONTAINMENT SIT [HISTORICAL INFORMATION]

EXTENSOMETER	AZIMUTH (DEGREES)	RADIAL LOCATION FROM CONTAINMENT CENTERLINE	DEFLECTION (INCHES) Predicted[1]	Predicted[2]	Actual
D_1	165°-30'	3'-1"	0.1482	0.541	0.114
D_2	75	31'-0"	0.145	0.507	0.112
D_3	75	45'-0"	0.139	0.463	0.081
V_1	46	61'-8"	0.079	0.197	0.027
V_3	225	61'-8"	0.079	0.197	0.031
V_4	315	61'-8"	0.079	0.197	0.034

[1] Based on uncracked analysis.
[2] Based on cracked analysis.

TABLE 3.8-42: RADIAL DEFLECTIONS OF CONTAINMENT CONTAINMENT SIT [HISTORICAL INFORMATION]

EXTENSOMETER	ELEVATION	AZIMUTH (DEGREES)	DEFLECTION (INCHES) Predicted[1]	Predicted[2]	Actual
H_1	103'0"	30	0.0104	0.080	0.029
H_2	103'0"	90	0.0104	0.080	0.031
H_3	103'0"	150	0.0104	0.080	0.024
H_4	103'0"	210	0.0104	0.080	0.039
H_5	103'0"	270	0.0104	0.080	0.036
H_6	103'0"	330	0.0104	0.080	0.044
H_{16}	131'0"	30	0.0502	0.380	0.056
H_{17}	131'0"	150	0.0502	0.380	0.082
H_{18}	131'0"	270	0.0502	0.380	0.074
H_7	167'0"	30	0.0477	0.420	0.101
H_8	167'0"	90	0.0477	0.420	0.283
H_9	167'0"	150	0.0477	0.420	0.263
H_{10}	167'0"	211°-08'	0.0477	0.420	0.166
H_{11}	167'0"	270	0.0477	0.420	0.168
H_{12}	167'0"	330	0.0477	0.420	0.084
H_{19}	202'0"	30	0.0484	0.420	0.201
H_{20}	202'0"	150	0.0484	0.420	0.271
H_{21}	202'0"	270	0.0484	0.420	0.206
H_{13}*	230'0"	210(30)	0.0402	0.390	0.105
H_{14}*	230'0"	270(90)	0.0402	0.390	0.1245
H_{15}*	230'0"	150(330)	0.0402	0.390	0.092

[1] Based on uncracked analysis.
[2] Based on cracked analysis
*Extensometers H-13, H-14, and H-15 measure differential displacements between six points on the containment wall (i.e., between azimuths 210° and 30°, 270° and 90°, and 150° and 330°, respectively). The deflections indicated for these extensometers are one-half of the measured diametric displacements.

TABLE 3.8-43: RADIAL DEFLECTIONS OF CONTAINMENT AT EQUIPMENT HATCH CONTAINMENT SIT [HISTORICAL INFORMATION]

EXTENSOMETER	ELEVATION	AZIMUTH (DEGREES)	DEFLECTION (INCHES) Predicted[1]	Predicted[2]	Actual
E_1	198'-6"	240	0.0477	0.420	0.233
E_2	191'-2½"	240	0.0477	0.420	0.258
E_3	182'-9"	240	0.0477	0.420	0.232
E_4	161'-9"	240	0.0477	0.420	0.213
E_5	153'-10½"	240	0.0477	0.420	0.187
E_6	146'-6"	240	0.0477	0.420	0.167
E_7	172'-6"	217°-47'	0.0477	0.420	0.200
E_8	172'-3"	223°-14'	0.0477	0.420	0.210
E_9	172'-6"	230°-14'	0.0477	0.420	0.235
E_{10}	168'-2"	247°-07'	0.0477	0.420	0.205
E_{11}	172'-7¼"	259	0.0477	0.420	0.173
E_{12}	172'0"	264	0.0477	0.420	0.170

[1] Based on uncracked analysis.
[2] Based on cracked analysis.

TABLE 3.8-44: COMPARISON OF DESIGN CRITERIA

Note: The provision of ASME/ACI 359-80 and those used in the design of the containment are similar, except as tabulated below.

ASME/ACI 359-80		FSAR: ACI 349 (Title 69-2)
Article CC-3421.1 Factored Allowable Stress for Concrete Compression		
Primary Membrane	$F_c = 0.60f'_c$	
Primary Membrane plus Bending	$F_c = 0.75f'_c$	$F_c = 0.85f'_c$
Primary + Secondary Membrane	$F_c = 0.75f'_c$	
Primary + Secondary Membrane + Bending	$F_c - 0.85f'_c$	
Article CC-3421.5 Factored Allowable Stress for Concrete Tangential Shear		
Method 1:		
$V_c = 12000\rho$ for $\rho \leq 0.01$		
$V_c = 93 + 2700\rho$ for $0.01 \leq \rho \leq 0.025$		Same
$V_c = 160$ psi (max.)		(Used $V_c \leq 160$ psi (max.))
or		
Method 2:		
$V_c = 40$ psi (max.)		$V_c = 40$ psi (max.)
if $\rho \geq \frac{60}{f_y}$ in addition to that required for membrane tension		if $\rho \geq \frac{60}{f_y}$ in addition to that required for membrane tension
$v_u \leq 8\sqrt{f'_C}$ (max.) unless v_c assumed to be zero		$\rho \geq 0.004$ including that required for membrane tension $v_u \leq 8\sqrt{f'_C}$ (max.)
Article CC-3431.1 Service Allowable Stress for Concrete Compression		
Primary Membrane	$F_C = 0.3f'_C$	
Primary Membrane with W or E loads	$F_C = 0.4f'_C$	$F_C = 0.45f'_C$
Primary Membrane + Bending	$F_C = 0.45f'_C$	

TABLE 3.8-44: COMPARISON OF DESIGN CRITERIA (CONTINUED)

ASME/ACI 359-80	FSAR: ACI 349 (Title 69-2)
Primary + Secondary Membrane $F_C = 0.45f'_C$ Primary + Secondary Membrane + Bending $F_C = 0.60f'_C$	
Article CC-3431.3 Service Allowable Stress for Concrete Radial Shear	
Use 50 percent factored allowables except	Same
Use 67 percent factored allowables for - test condition - thermal + mechanical loads provided that section is not less than required for mechanical loads only	No increase in allowable stresses used.
Article CC-3432.2 Service Allowable Stresses for Bar Compression	
$F_S = 0.5f_y$	Same
A 33-l/3 percent increase in F_S is permitted for the test case and when temperature effects are combined with other loads.	No increase in allowable stresses used.
Article CC-3521.1 Factored Tangential Shear Bar Design	
Where $v_u > v_c$: $A_{sh} = \frac{N_h + N_{he} + V_c}{0.9f_y}$ $A_{sv} = \frac{N_v + N_{ve} + V_e}{0.9f_y}$ $A_{si} = \frac{V_u - V_c}{0.9f_y}$	Concrete resists v_c $A_{si} = \frac{FACTOR(V_u - V_c)bs}{f_y(\sin\alpha + \cos\alpha)}$ FACTOR = 1.5 where $v_c \leq 160$ psi (Method 1) FACTOR = 1.0 where v_c = 40 psi max. (Method 2)

TABLE 3.8-44: COMPARISON OF DESIGN CRITERIA (CONTINUED)

ASME/ACI 359-80	FSAR: ACI 349 (Title 69-2)
Where $v_u \leq v_c$: $A_{sh} = \frac{N_h + N_{he} + V_u}{0.9 f_y}$ $A_{sv} = \frac{N_v + N_{ve} + V_u}{0.9 f_y}$	Concrete resists v_u

TABLE 3.8-45: COMPARISON OF LOAD COMBINATIONS FOR CONTAINMENT INTERNAL STRUCTURES

Equivalent Terms Used in Load Combinations

ACI 349-76	GG FSAR
D	D
L	L
F	Effects included in D term
H	Not applicable to internal structures
T_o	T_o
R_o	Effects included in D and T_o terms
E_o	E
W	W
E_{ss}	E'
W_t	W'
P_a	$P_{CD'}$ $P_{DD'}$ $P_{ND'}$ $P_{DS'}$ $P_{DN'}$ $P_{CI'}$ P
T_a	$T_o + T_a$
R_a	R
Y_r	Effect included in R term
Y_j	Effect included in R term
Y_m	Effect included in R term

Comparison of Load Combinations - Drywell

1. Normal Operating Conditions:

ACI 349-76 and Regulatory Guide 1.142:

$$U = 1.4D + 1.7L + 1.7R_o$$

GG FSAR:

$$U = 1.4D + 1.7L + 1.0T_o + 1.0Z$$

This combination is represented in the FSAR with a load factor of 1.0 applied to the thermal effects term (T_o) instead of 1.7. The FSAR combination is identical to combination 2.3.2.1(a) of ACI 349, Title 69-2. The FSAR combination is not conservative with respect to the ACI 349-76 combination; however, this combination does not control in the drywell design.

TABLE 3.8-45: COMPARISON OF LOAD COMBINATIONS FOR CONTAINMENT INTERNAL STRUCTURES (CONTINUED)

2. Severe Environmental Conditions:

ACI 349-76 and Regulatory Guide 1.142:

$$U = 1.4D + 1.4F + 1.7L + 1.7H + 1.9E_o + 1.7R_o$$

GG FSAR:

$$U = 1.0D + 1.0L + 1.0T_o + 1.0C + 1.25E$$

$$U = 1.0D + 1.25L + 1.0T_o + 1.25(E \text{ or } W) + 1.0Z$$

This combination is represented in the FSAR with a load factor of 1.0 applied to the dead load (D) and thermal effects (T_o) terms and load factor of 1.25 for the live load (L) and earthquake or design wind terms. The FSAR combination is identical to combination 2.3.2.1(b) of ACI 349, Title 69-2. The FSAR combination is not conservative with respect to the ACI 349-76 combination; however, this combination does not control in the drywell design.

3. Severe Environmental Conditions:

ACI 349-76 and Regulatory Guide 1.142:

$$U = 1.4D + 1.4F + 1.7L + 1.7H + 1.7W + 1.7R_o$$

This combination is identical to Combination 2 (with the substitution of design wind loads for earthquake loads); however, design wind loads are not applicable to the drywell.

4. Extreme Environmental Conditions:

ACI 349-76 and Regulatory Guide 1.142:

$$U = D + F + L + H + T_o + R_o + E_{ss}$$

GG FSAR:

$$U = 1.0D + 1.0L + 1.0T_o + 1.0C_1 + 1.0E'$$

The combinations are equivalent. This combination controls in the drywell design.

5. Extreme Environmental Conditions:

ACI 349-76 and Regulatory Guide 1.142:

$$U = D + F + L + H + T_o + R_o + W_t$$

GG FSAR:

$$U = 1.0D + 1.0L + 1.0T_o + 1.0A + 1.0W'$$

TABLE 3.8-45: COMPARISON OF LOAD COMBINATIONS FOR CONTAINMENT INTERNAL STRUCTURES (CONTINUED)

The combinations are equivalent; however, tornado loads are not applicable to the drywell design.

6. Abnormal Operating Conditions:

ACI 349-76 and Regulatory Guide 1.142:

$U = D + F + L + H + T_a + R_a + 1.5P_a$

GG FSAR:

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.5(P_{DS} \text{ or } P_{DN}) + 1.25C_1 + 1.5R_C$

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.25C_1 + 1.0R + 1.5P_{DD} + 1.5(R_B \text{ or } R_{BA})$

The combinations are equivalent. The pipe reaction loads for the small break accident case are incorporated into the accident thermal effects terms. This combination controls in the drywell design.

7. Abnormal Operating and Severe Environmental Conditions:

ACI 349-76 and Regulatory Guide 1.142:

$U = D + F + L + H + T_a + R_a + 1.0(Y_r + Y_j + Y_m) + 1.25P_a + 1.25E_o$

GG FSAR:

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.25(R_B \text{ or } R_{BA}) + 1.25P_{DD} + 1.0C_1 + 1.25E + 1.0R$

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.25(P_{DS} \text{ or } P_{DN}) + 1.0C_1 + 1.25E + 1.25R_C$

The combinations are equivalent. The pipe reaction loads for the small break accident case are incorporated into the accident thermal effects term. This combination controls in the drywell design.

8. Abnormal Operating and Extreme Environmental Conditions:

ACI 349-76 and Regulatory Guide 1.142:

$U = D + F + L + H + T_a + R_a + 1.0P_a + 1.0(Y_r + Y_j + Y_m) + 1.0E_{ss}$

TABLE 3.8-45: COMPARISON OF LOAD COMBINATIONS FOR CONTAINMENT INTERNAL STRUCTURES (CONTINUED)

GG FSAR:

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.0(P_{DS} \text{ or } P_{DN}) + 1.0 R_C + 1.0C_1 + 1.0E'$

$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.0(R_B \text{ or } R_{BA}) + 1.0P_{DD} + 1.0C_1 + 1.0R + 1.0E'$

The combinations are equivalent. The pipe reaction loads for the small break accident case are incorporated into the accident thermal effects term. This combination controls in the drywell design.

Factored Load Conditions; ACI 349-76 and Regulatory Guide 1.142.

9. $U = .75(1.4D + 1.7L + 1.7T_o + 1.7R_o)$

10. $U = .75(1.4D + 1.7L + 1.4F + 1.7H + 1.9E_o + 1.7T_o + 1.7R_o)$

11. $U = .75(1.4D + 1.7L + 1.4F + 1.7H + 1.7W + 1.7T_o + 1.7R_o)$

The drywell load combinations are based on the requirements of ACI 349, Title 69-2, which has no load combination similar to the combinations above. Therefore, combinations similar to those above were not considered in the drywell design. Since the above combinations do not incorporate design accident conditions and the load factors are reduced by 0.75, these combinations do not control in the drywell design.

Comparison of Load Combinations - RPV Pedestal and Weir Wall

1. Normal Operating Conditions:

ACI 349-76 and Regulatory Guide 1.142:

$U = 1.4D + 1.7L + 1.7R_o$

GG FSAR:

$U = 1.4D + 1.7L + 1.4T_o$

TABLE 3.8-45: COMPARISON OF LOAD COMBINATIONS FOR CONTAINMENT INTERNAL STRUCTURES (CONTINUED)

This combination is represented in the FSAR with a load factor of 1.4 applied to the thermal effects term (T_o) instead of 1.7. The FSAR combination is identical to the requirements of subsection 9.3.7 of ACI 318-71. The FSAR combination is less conservative than the ACI 349-76 combination; however, this is not a governing load case, and the design is still conservative with respect to this combination.

2. Severe Environmental Conditions:

ACI 349-76 and Regulatory Guide 1.142:

$$U = 1.4D + 1.4F + 1.7L + 1.7H + 1.9E_o + 1.7R_o$$

GG FSAR:

$$U = 1.4D + 1.7L + 1.4T_o + 1.25E + 1.7C_1$$

This combination is represented in the FSAR with a load factor of 1.4 applied to the thermal effects term (T_o) instead of 1.7 (applied to R_o) and a factor of 1.25 applied to the earthquake term instead of 1.9. The FSAR combination is based on the requirements of ACI 318-71. The load factor on the earthquake term typifies the requirements of ACI 349, Title 69-2. The FSAR combination is less conservative than the ACI 349-76 combination; however, this is not a governing load case, and the design is conservative with respect to this combination.

3. Severe Environmental Conditions:

ACI 349-76 and Regulatory Guide 1.142:

$$U = 1.4D + 1.4F + 1.7L + 1.7H + 1.7W + 1.7R_o$$

This combination is identical to Combination 2 above (with the substitution of design wind loads for earthquake loads); however, design wind loads are not applicable to the RPV pedestal and weir wall.

4. Extreme Environmental Conditions:

ACI 349-76 and Regulatory Guide 1.142:

$$U = D + F + L + H + T_o + R_o + E_{ss}$$

GG FSAR:

$$U = 1.4D + 1.7L + 1.4T_o + 1.0E' + 1.0C_1$$

TABLE 3.8-45: COMPARISON OF LOAD COMBINATIONS FOR CONTAINMENT INTERNAL STRUCTURES (CONTINUED)

This combination is represented in the FSAR with a load factor of 1.4 applied to the dead load (D) and temperature effects (T_o) terms and is more conservative than the ACI 349-76 combination.

5. Extreme Environmental Conditions:

ACI 349-76 and Regulatory Guide 1.142:

$$U = D + F + L + H + T_o + R_o + W_t$$

This combination is identical to Combination 4 above (with the substitution of tornado loadings for earthquake loadings); however, tornado loadings are not applicable to the RPV pedestal and weir wall.

6. Abnormal Operating Conditions:

ACI 349-76 and Regulatory Guide 1.142:

$$U = D + F + L + H + T_a + R_a + 1.5P_a$$

GG FSAR:

$$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.5(P_{DS} \text{ or } P_{DN}) + 1.25C_1 + 1.5R_c$$

$$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.25C_1 + 1.0R + 1.5P_{DD} + 1.5R_B$$

The combinations are equivalent. The pipe reaction loads for the small break accident case are incorporated into the accident thermal effects terms.

7. Abnormal Operating and Severe Environmental Conditions:

ACI 349-76 and Regulatory Guide 1.142:

$$U = D + F + L + H + T_a + R_a + 1.0(Y_r+Y_j+Y_m) + 1.25P_a + 1.25E_o$$

GG FSAR:

$$U = 1.0D + 1.0L + 1.0(T_o+T_a) + 1.25P_{DD} + 1.0C_1 + 1.25E + 1.0R$$

$$U = 1.0D + 1.0L + 1.0(T_o+T_a) + 1.25(P_{DS} \text{ or } P_{DN}) + 1.0C_1 + 1.25E + 1.25R_c$$

TABLE 3.8-45: COMPARISON OF LOAD COMBINATIONS FOR CONTAINMENT INTERNAL STRUCTURES (CONTINUED)

The combinations are equivalent. The pipe reaction loads for the small break accident case are incorporated into the accident thermal effects term.

8. Abnormal Operating and Extreme Environmental Conditions:

ACI 349-76 and Regulatory Guide 1.142:

$U = D + F + L + H + T_a + R_a + 1.0P_a + 1.0(Y_r+Y_j+Y_m) + 1.0E_{ss}$

GG FSAR:

$U = 1.0D + 1.0L + 1.0(T_o+T_a) + 1.0(P_{DS} \text{ or } P_{DN}) + 1.0R_c + 1.0C_1 + 1.0E'$

$U = 1.0D + 1.0L + 1.0(T_o+T_a) + 1.0P_{DD} + 1.0C_1 + 1.0R + 1.0E'$

The combinations are equivalent. The pipe reaction loads for the small break accident case are incorporated into the accident thermal effects term. These FSAR combinations are shown by analysis to be the controlling design combinations for the RPV pedestal and weir wall.

9. Normal Operating Factored Load Condition:

ACI 349-76 and Regulatory Guide 1.142:

$U = .75(1.4D + 1.7L + 1.7T_o + 1.7R_o)$

GG FSAR:

$U = .75(1.4D + 1.7L + 1.4T_o)$

This load combination is represented in the FSAR with a load factor of 1.4 applied to the thermal effects term instead of 1.7. The effects of pipe reactions and equipment loads are included in the GG thermal effects term and dead load term, respectively. The FSAR combination is identical to the requirements of subsection 9.3.7 of ACI 318-71.

The FSAR combination is less conservative than the ACI 349-76 combination; however, this is not a governing load case, and the design is still conservative with respect to this combination.

TABLE 3.8-45: COMPARISON OF LOAD COMBINATIONS FOR CONTAINMENT INTERNAL STRUCTURES (CONTINUED)

10. Normal Operating and Severe Environmental Factored Load Condition:

ACI 349-76 and Regulatory Guide 1.142:

$U = .75(1.4D + 1.7L + 1.4F + 1.7H + 1.9E_o + 1.7T_o + 1.7R_o)$

GG FSAR:

$U = .75(1.4D + 1.7L + 1.4T_o + 1.9E)$

This combination is represented in the FSAR with a load factor of 1.4 applied to the thermal effects term instead of 1.7. The effects of pipe reactions and equipment loads are included in the GG thermal effects term and dead load term, respectively. The FSAR combination is identical to the requirements of subsections 9.3.3 and 9.3.7 of ACI 318-71.

The FSAR combination is less conservative than the ACI 349-76 combination; however, this is not a governing load case, and the design is still conservative with respect to this combination.

11. Normal Operating and Severe Environmental Factored Load Condition:

ACI 349-76 and Regulatory Guide 1.142:

$U = .75(1.4D + 1.7L + 1.4F + 1.7H + 1.7W + 1.7T_o + 1.7R_o)$

This combination is identical to Combination 10 above (with the substitution of design wind loads for earthquake loads); however, design wind loads are not applicable to the RPV pedestal and weir wall.

TABLE 3.8-45: COMPARISON OF LOAD COMBINATIONS FOR CONTAINMENT INTERNAL STRUCTURES (CONTINUED)

Comparison of Load Combinations - Other Internal Structures

1. Normal Operating Conditions:

ACI 349-76 and Regulatory Guide 1.142:

$$U = 1.4D + 1.7L + 1.7R_o$$

GG FSAR:

$$U = 1.4D + 1.7L + 1.0T_o$$

This combination is represented in the FSAR with a load factor of 1.0 applied to the thermal effects term (T_o), instead of 1.7. The FSAR combination is consistent with the drywell combination of subsection 3.8.3.3.2.1.a.

The FSAR combination is less conservative than the ACI 349-76 combination; however, this is not a governing load case, and the design is conservative with respect to this combination.

2. Severe Environmental Conditions:

ACI 349-76 and Regulatory Guide 1.142:

$$U = 1.4D + 1.4F + 1.7L + 1.7H + 1.9E_o + 1.7R_o$$

GG FSAR:

$$U = 1.0D + 1.25L + 1.0T_o + 1.25E$$

This combination is represented in the FSAR with a load factor of 1.0 applied to the dead load (D) and thermal effects (T_o) terms and a load factor of 1.25 for the live load (L) and earthquake or design wind terms. The FSAR combination is consistent with the drywell combination specified in subsection 3.8.3.3.2.1.b.

The FSAR combination is less conservative than the ACI 349-76 combination; however, this is not a governing load case, and the design is conservative with respect to this combination.

TABLE 3.8-45: COMPARISON OF LOAD COMBINATIONS FOR CONTAINMENT INTERNAL STRUCTURES (CONTINUED)

3. Severe Environmental Conditions

ACI 349-76 and Regulatory Guide 1.142:

$$U = 1.4D + 1.4F + 1.7L + 1.7H + 1.7W + 1.7R_o$$

This combination is identical to Combination 2 (with the substitution of design wind loads for earthquake loads); however, design wind loads are not applicable to the other interior structures.

4. Extreme Environmental Conditions:

ACI 349-76 and Regulatory Guide 1.142:

$$U = D + F + L + H + T_o + R_o + E_{ss}$$

GG FSAR:

$$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.0P + 1.0E' + 1.0R$$

The combinations are equivalent with the addition of the abnormal load of compartment pressurization (P).

5. Extreme Environmental Conditions:

ACI 349-76 and Regulatory Guide 1.142:

$$U = D + F + L + H + T_o + R_o + W_t$$

This combination is identical to Combination 4 above (with the substitution of tornado loadings for earthquake loads); however, tornado loads are not applicable to other internal structures.

6. Abnormal Operating Conditions:

ACI 349-76 and Regulatory Guide 1.142:

$$U = D + F + L + H + T_a + R_a + 1.5P_a$$

GG FSAR:

$$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.5P + 1.0P$$

The combinations are equivalent.

TABLE 3.8-45: COMPARISON OF LOAD COMBINATIONS FOR CONTAINMENT INTERNAL STRUCTURES (CONTINUED)

7. Abnormal Operating and Severe Environmental Conditions:

ACI 349-76 and Regulatory Guide 1.142:

$$U = D + F + L + H + T_a + R_a + 1.0(Y_r + Y_j + Y_m) + 1.25P_a + 1.25E_o$$

GG FSAR:

$$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.25P + 1.25E + 1.0R$$

The combinations are equivalent. This combination controls in the design of the internal structures.

8. Abnormal Operating and Extreme Environmental Conditions:

ACI 349-76 and Regulatory Guide 1.142:

$$U = D + F + L + H + T_a + R_a + 1.0P_a + 1.0(Y_r + Y_j + Y_m) + 1.0E_{ss}$$

GG FSAR:

$$U = 1.0D + 1.0L + 1.0(T_o + T_a) + 1.0P + 1.0R + 1.0E'$$

The combinations are equivalent. This combination controls in the design of the internal structures.

9. Normal Operating Factored Load Conditions:

ACI 349-76 and Regulatory Guide 1.142:

$$U = .75(1.4D + 1.7L + 1.7T_o + 1.7R_o)$$

GG FSAR:

$$U = .75(1.4D + 1.7L + 1.4T_o)$$

This load combination is represented in the FSAR with a load factor of 1.4 applied to the thermal effects term instead of 1.7. The effects of pipe reactions and equipment loads are included in the GG thermal effects term and dead load term, respectively. The FSAR combination is identical to the requirements of subsection 9.3.7 of ACI 318-71.

The FSAR combination is less conservative than the ACI 349-76 combination; however, this is not a governing load case, and the design is conservative with respect to this load.

TABLE 3.8-45: COMPARISON OF LOAD COMBINATIONS FOR CONTAINMENT INTERNAL STRUCTURES (CONTINUED)

10. Normal Operating and Severe Environmental Factored Load Condition:

ACI 349-76 and Regulatory Guide 1.142:

$$U = .75(1.4D + 1.7L + 1.4F + 1.7H + 1.9E_o + 1.7T_o + 1.7R_o)$$

GG FSAR:

$$U = .75(1.4D + 1.7L + 1.4T_o + 1.9E)$$

This combination is represented in the FSAR with a load factor of 1.4 applied to the thermal effects term instead of 1.7. The effects of pipe reactions and equipment loads are included in the GG thermal effects term and dead load term, respectively. The FSAR combination is identical to the requirements of subsections 9.3.3 and 9.3.7 of ACI 318-71.

The FSAR combination is less conservative than the ACI 349-76 combination; however, this is not a governing load case, and the design is conservative with respect to this load.

11. Normal Operating and Severe Environmental Factored Load Condition:

ACI 349-76 and Regulatory Guide 1.142:

$$U = .75(1.4D + 1.7L + 1.4F + 1.7H + 1.7W + 1.7T_o + 1.7R_o)$$

This combination is identical to Combination 10 above (with the substitution of design wind loads for earthquake loads); however, design wind loads are not applicable to the other internal structures.

TABLE 3.8-45: COMPARISON OF LOAD COMBINATIONS FOR CONTAINMENT INTERNAL STRUCTURES (CONTINUED)

Table 3.8-46 is a comparison of design criteria requirements of concrete internal structures of concrete containments. The Standard Review Plan criteria (ACI 349-76) are listed with the corresponding criteria of the Grand Gulf FSAR (ACI 349, Title 69-2/ACI 318-71). Such conditions as minimum reinforcement requirements for massive concrete (Section 10.17), punching shear stress for walls (Section 11.6), and dynamic loadings imposed on members during transportation (Section 16.2), while not addressed in ACI 318-71, are considered in the design of concrete internal structures.

TABLE 3.8-46: DESIGN CRITERIA COMPARISON

SRP: ACI 349-76	FSAR: ACI 349 (Title 69-2)/ ACI 318-71
Not included in the Code	8.10 Alternate Design Method This method assures that the concrete and steel stresses and strains remain elastic.
9.1, 9.2, 9.3 Required Strength These subsections are discussed in Table 3.8-45.	9.1, 9.2, 9.3 Required Strength These subsections are discussed in Table 3.8-45.
9.4 Design Strengths for Reinforcement Yield strength (f_y) limited to 80,000 psi	9.4 Design Strengths for Reinforcement Yield strength (f_y) limited to 60,000 psi
10.17 Massive Concrete Members Minimum reinforcement require-ments for members more than 48" thick stated.	Massive concrete is not addressed in this code. This concern is considered in the Grand Gulf design.
11.4.3 Permissible Shear Stress Alternative method given: $V_c = 2\left(1 + 0.0005\frac{Nu}{Ag}\right) \sqrt{f'_c}$ This method is used if M_m is not computed.	11.4.3 Permissible Shear Stress Separate procedures for computing v_c for members subject to axial compression.

TABLE 3.8-46: DESIGN CRITERIA COMPARISON (CONTINUED)

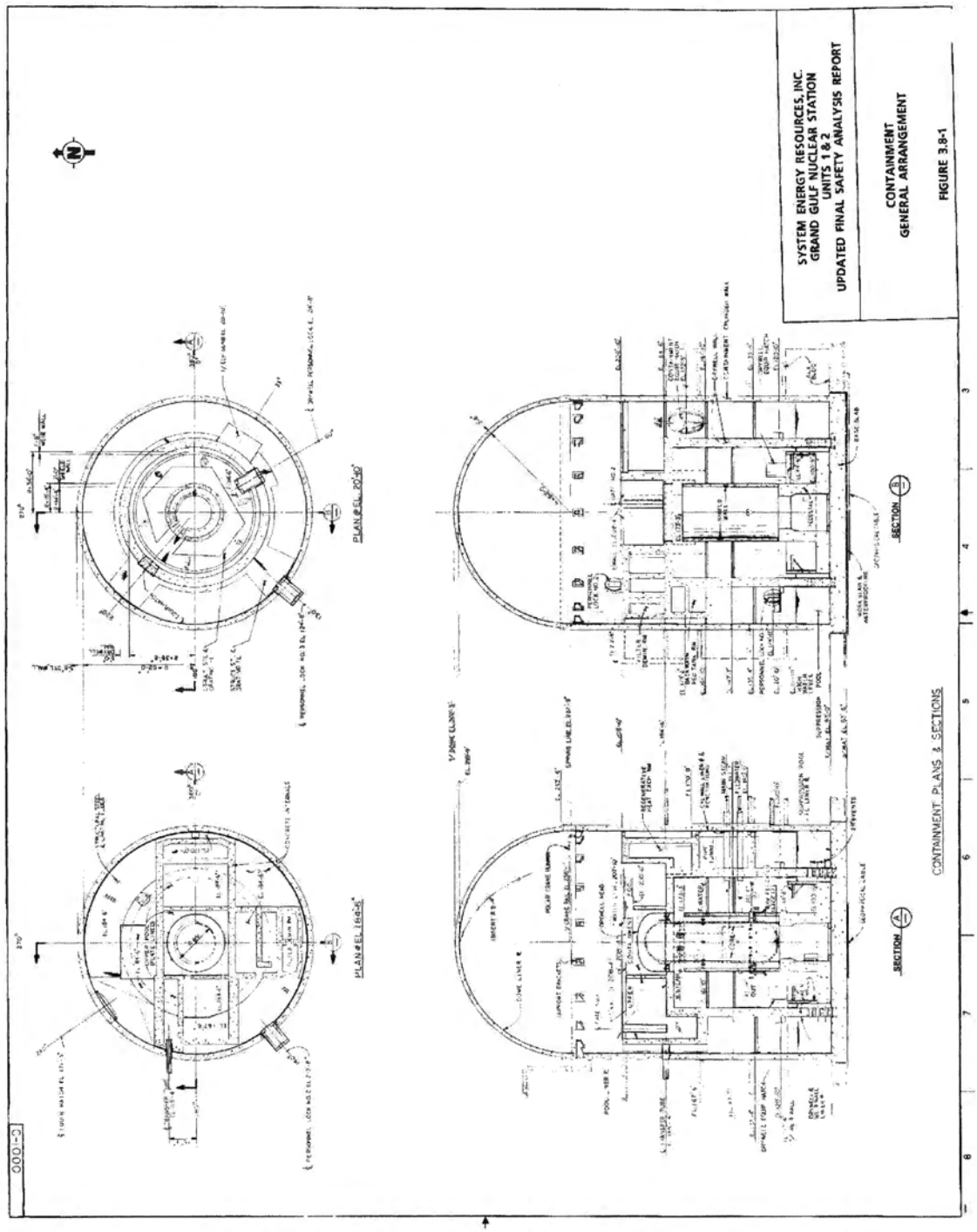
SRP: ACI 349-76	FSAR: ACI 349 (Title 69-2)/ ACI 318-71
11.15.7 Punching Shear Stress The shear stress taken by the concrete on the assumed failure surface shall not exceed: $v_{ch} = 4\sqrt{f'_c}\sqrt{1+f_{mv}}$ $v_{cv} = 4\sqrt{f'_c}\sqrt{\frac{1+f_{mh}}{4f'_c}}$	11.16.6 Design of shear forces perpendicular to the face of the wall is referenced in Section 11.10, where the maximum shear stress permitted in the concrete shall not exceed: $\sqrt{c} = 4\sqrt{f'_c}$
16.2.2 Design “The design shall consider impact and other dynamic loading which may be imposed during transportation or erection.”	16.2 Design Dynamic loadings are not addressed in this section. These loadings are considered to be part of subsection 16.2.1 of this code.
16.4.2 Details Embedded dowels or inserts may be placed while the concrete is in a plastic state provided that they are not required to be hooked or tied to reinforcement within concrete.	16.4.2 Details Lifting devices must have the capacity to support 4 times the members’ weight.
Chapter 18 - Prestressed Concrete This chapter is not applicable to the internal structures.	Chapter 18 - Prestressed Concrete This chapter is not applicable to the internal structures.

TABLE 3.8-46: DESIGN CRITERIA COMPARISON (CONTINUED)

SRP: ACI 349-76	FSAR: ACI 349 (Title 69-2)/ ACI 318-71
Chapter 19 - Shells 19.1.1 This chapter addresses concrete shells with thicknesses equal to or greater than 12 inches.	Chapter 19 - Shells and Folded Plate Members 19.1.3 This chapter addresses thin shells (thicknesses are small compared to their other dimensions).
Elastic analysis not addressed	19.1.5 Elastic analysis is any analysis involving assumptions that are suitable approximations of three-dimensional elastic behavior.
Assumptions are not addressed.	19.2 Assumptions 19.2.1 Assume elastic, homogeneous and isotropic properties for thin shells. 19.2.2 Assume Poisson’s ratio equal to zero.
19.2.4 Compatibility must be satisfied at junction of shell and supporting members and at discontinuities in geometric and material properties. 19.2.6 Effects of openings and penetrations on structural strength must be considered. 19.2.7 Nonlinear stress behavior must be considered.	19.3.2 Approximate methods of analysis are allowed.

TABLE 3.8-46: DESIGN CRITERIA COMPARISON (CONTINUED)

SRP: ACI 349-76	FSAR: ACI 349 (Title 69-2)/ ACI 318-71
19.3.2 Concrete tensile strength shall not be considered.	19.4 $f_y \leq 60{,}000$ psi
19.3.5 Minimum reinforcement requirements are defined.	19.5.1 Minimum reinforcement requirements are defined.
19.3.7 Forces imposed by curved reinforcements shall be considered in the design of local areas.	19.5.3, 19.5.6, 19.5.8 Tensile reinforcement requirements are defined. 19.5.4 Reinforcement require-ments in the direction of principal stress.
Tangential Shear Tangential shear is not addressed.	Tangential Shear The requirements of Section CC-3421 of ASME Section III, Division 2, are satisfied, using the design procedures of the paper “Design of Containments for Seismic loads,” by T. E. Johnson.
Thermal Stress Considerations This code only states that stresses due to thermal loads must be considered (Chapter 9).	Thermal Stress Considerations The analysis of concrete sections for thermal loads using a cracked section approach is identical to the techniques of Appendix A of ACI 349 (Title 69-2).


SYSTEM ENERGY RESOURCES, INC.
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
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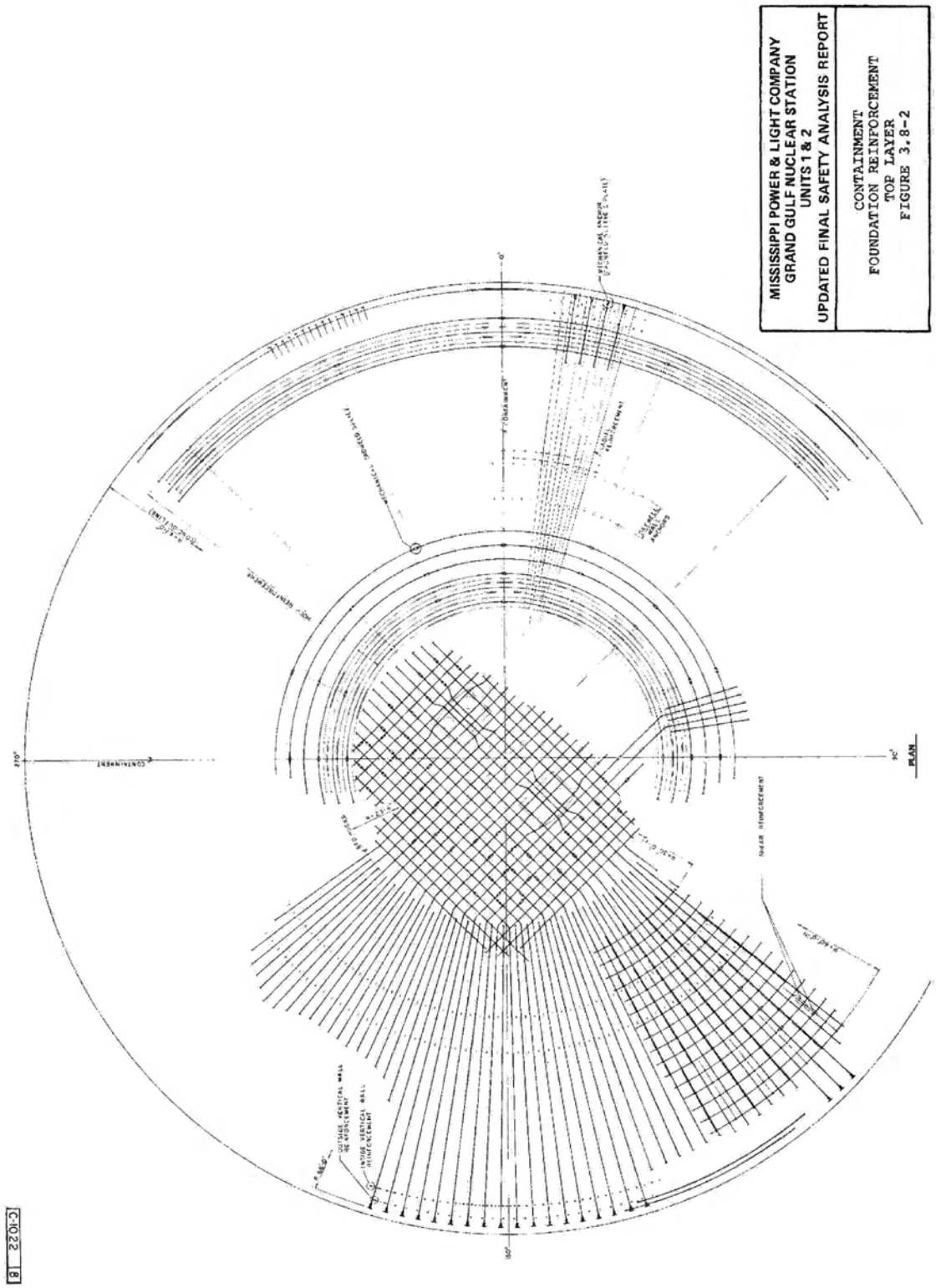
CONTAINMENT
GENERAL ARRANGEMENT
FIGURE 3.8-1
PLAN @ EL. 208'-0"
PLAN @ EL. 166'-0"
SECTION
SECTION
CONTAINMENT PLANS & SECTIONS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UNITS 1 & 2
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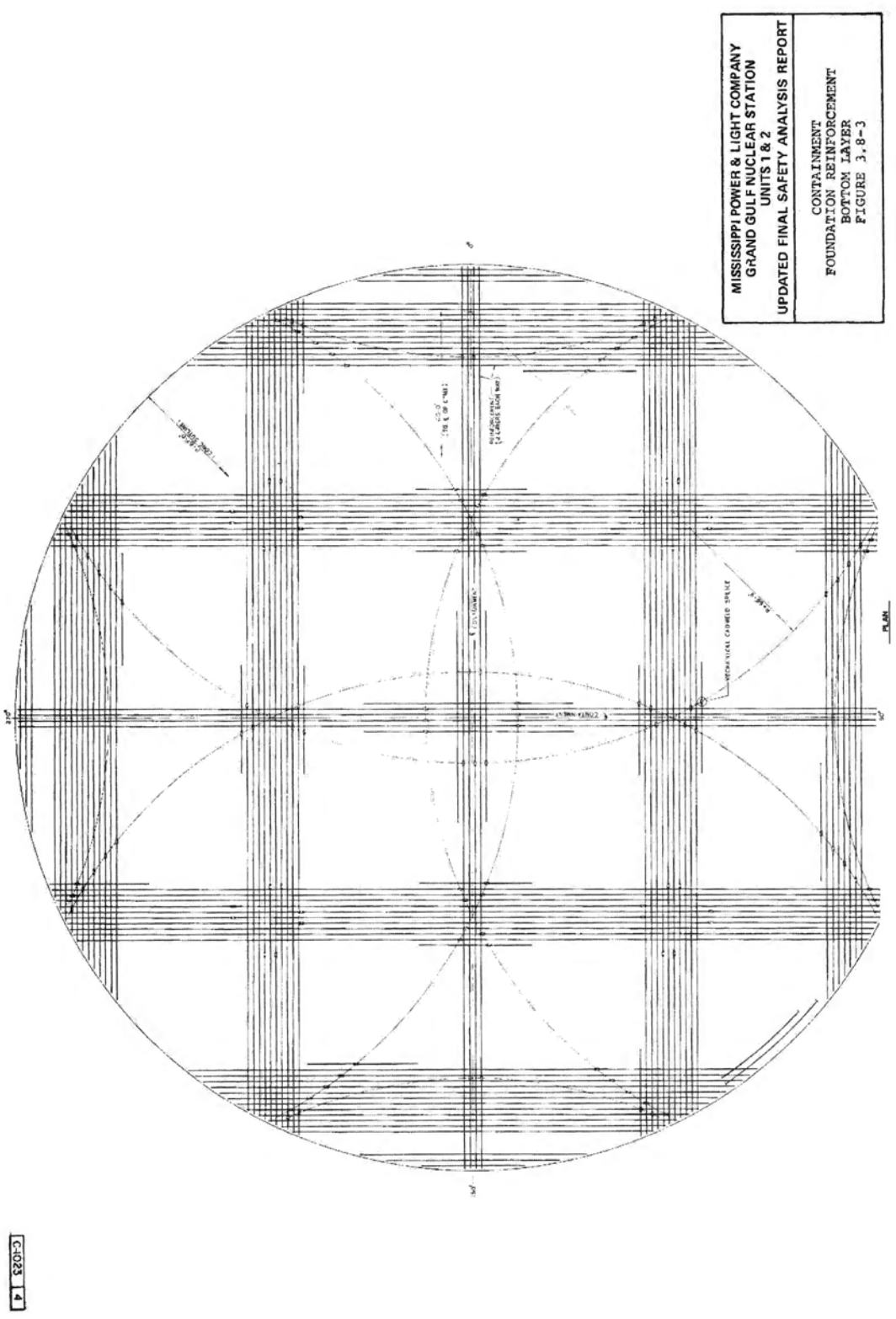
CONTAINMENT
FOUNDATION REINFORCEMENT
TOP LAYER
FIGURE 3.8-2
PLAN
C-1022 8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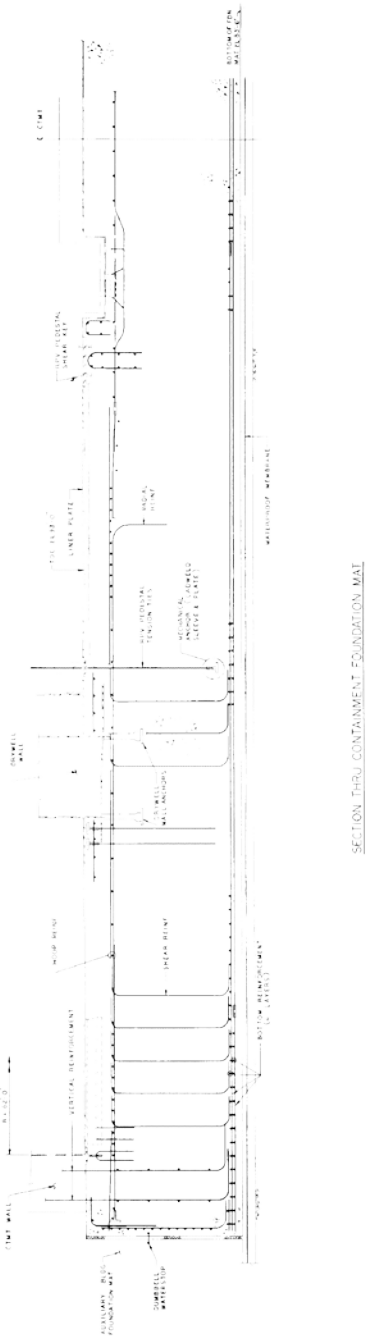
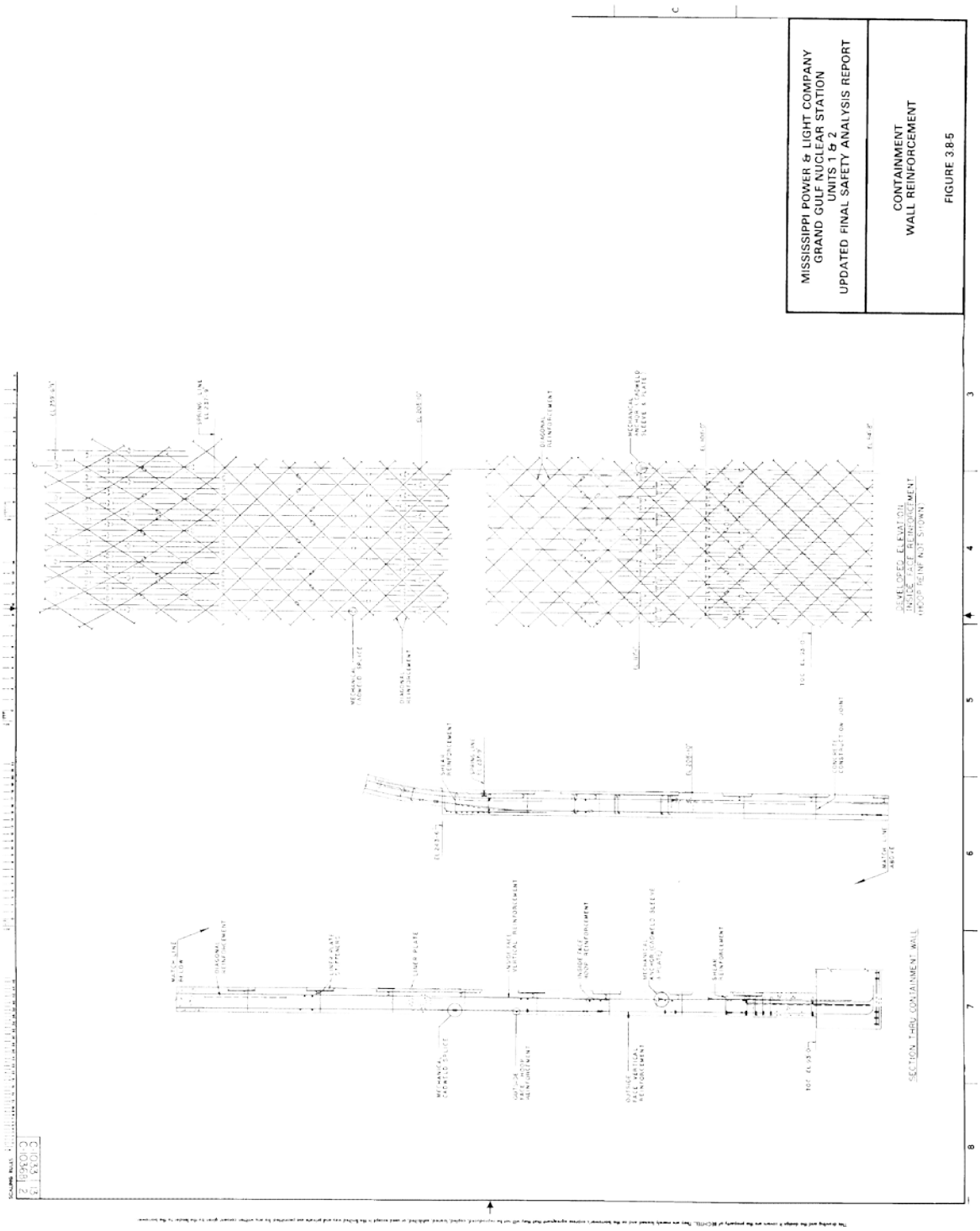
CONTAINMENT
FOUNDATION REINFORCEMENT
BOTTOM LAYER
FIGURE 3.8-3
PLAN
G-4023 4

MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT	CONTAINMENT FOUNDATION REINFORCEMENT SECTION FIGURE 3.8-4



SECTION THRU CONTAINMENT FOUNDATION MAT

C1024 8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UNITS 1 & 2
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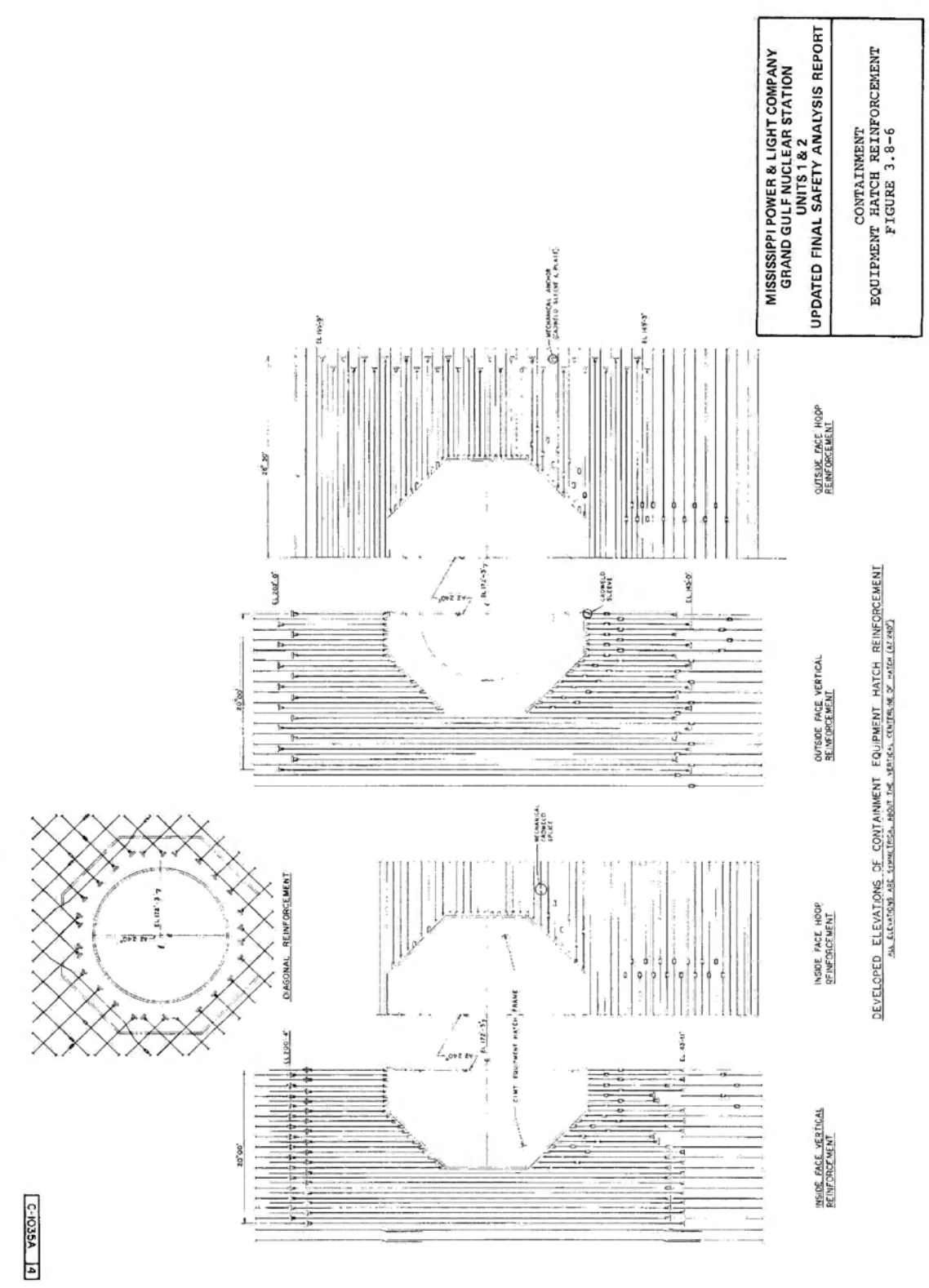
CONTAINMENT
WALL REINFORCEMENT
FIGURE 3.8-5
SECTION THRU CONTAINMENT WALL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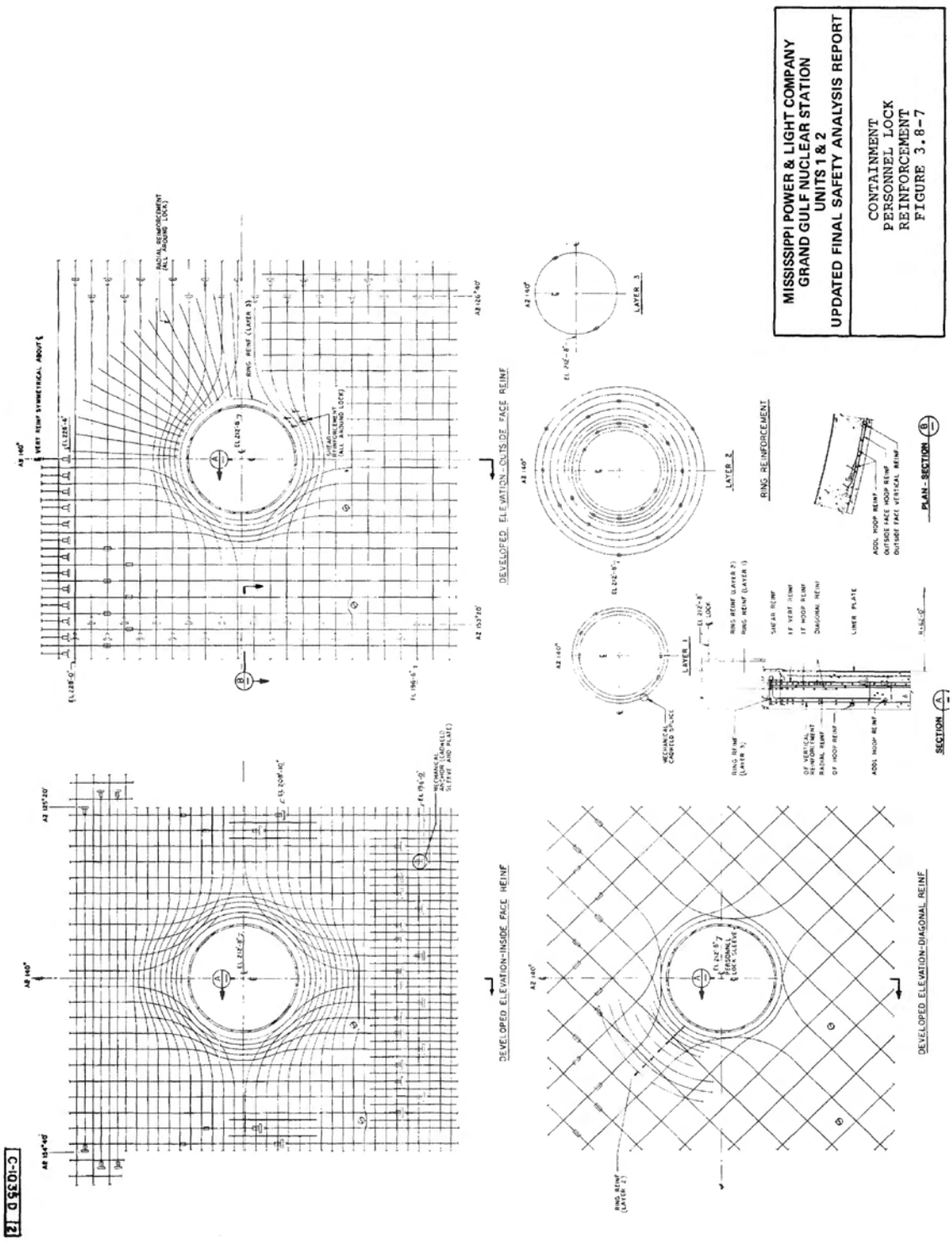
CONTAINMENT
EQUIPMENT HATCH REINFORCEMENT
FIGURE 3.8-6
DEVELOPED ELEVATIONS OF CONTAINMENT EQUIPMENT HATCH REINFORCEMENT
OUTSIDE FACE HOOP REINFORCEMENT
OUTSIDE FACE VERTICAL REINFORCEMENT
INSIDE FACE HOOP REINFORCEMENT
INSIDE FACE VERTICAL REINFORCEMENT
DIAGONAL REINFORCEMENT
C-1035A [4]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CONTAINMENT
PERSONNEL LOCK
REINFORCEMENT
FIGURE 3.8-7
DEVELOPED ELEVATION-OUTSIDE FACE REINF
DEVELOPED ELEVATION-INSIDE FACE REINF
DEVELOPED ELEVATION-DIAGONAL REINF
RING REINFORCEMENT
PLAN-SECTION
SECTION
C-1035 D 12

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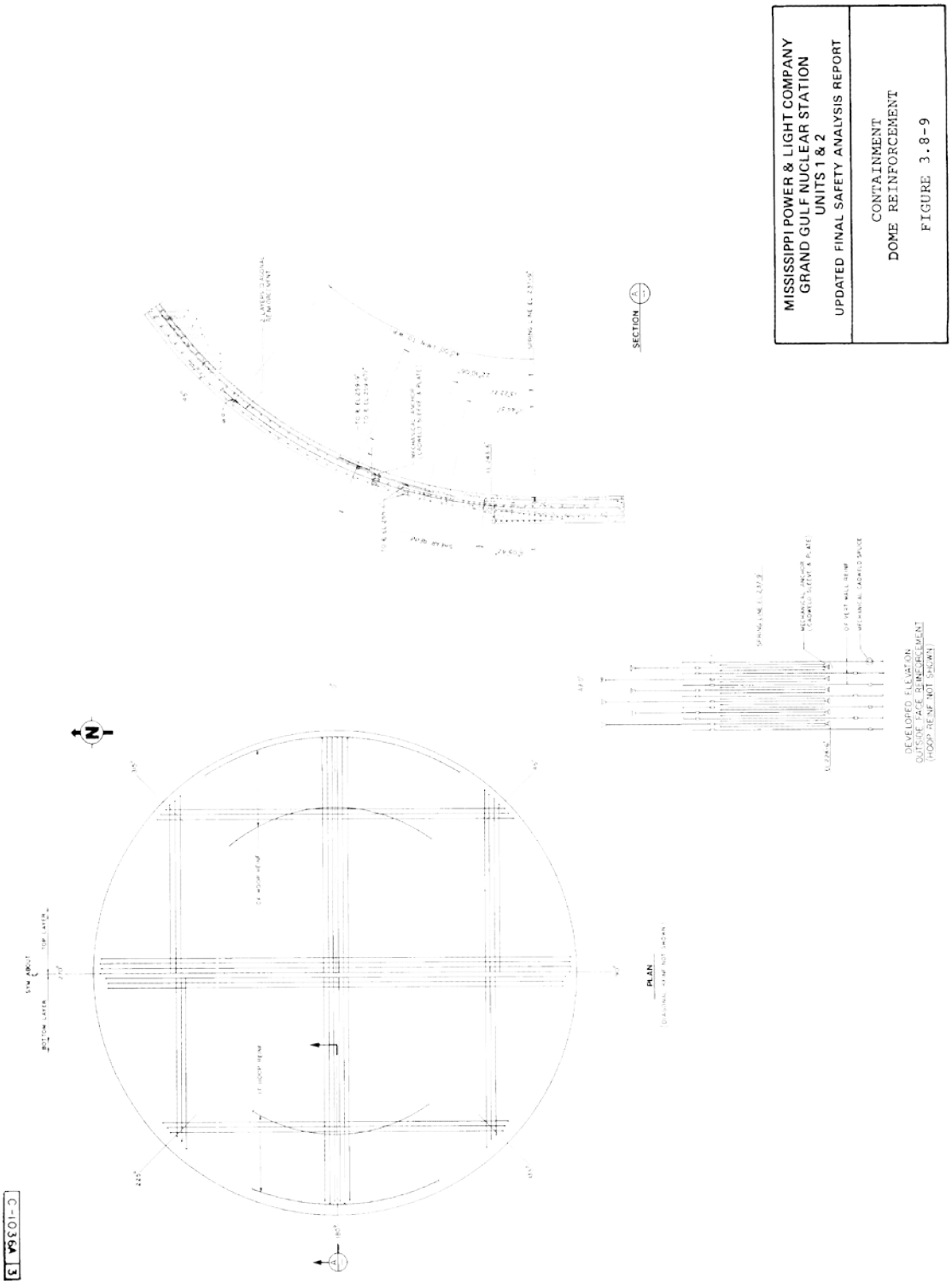
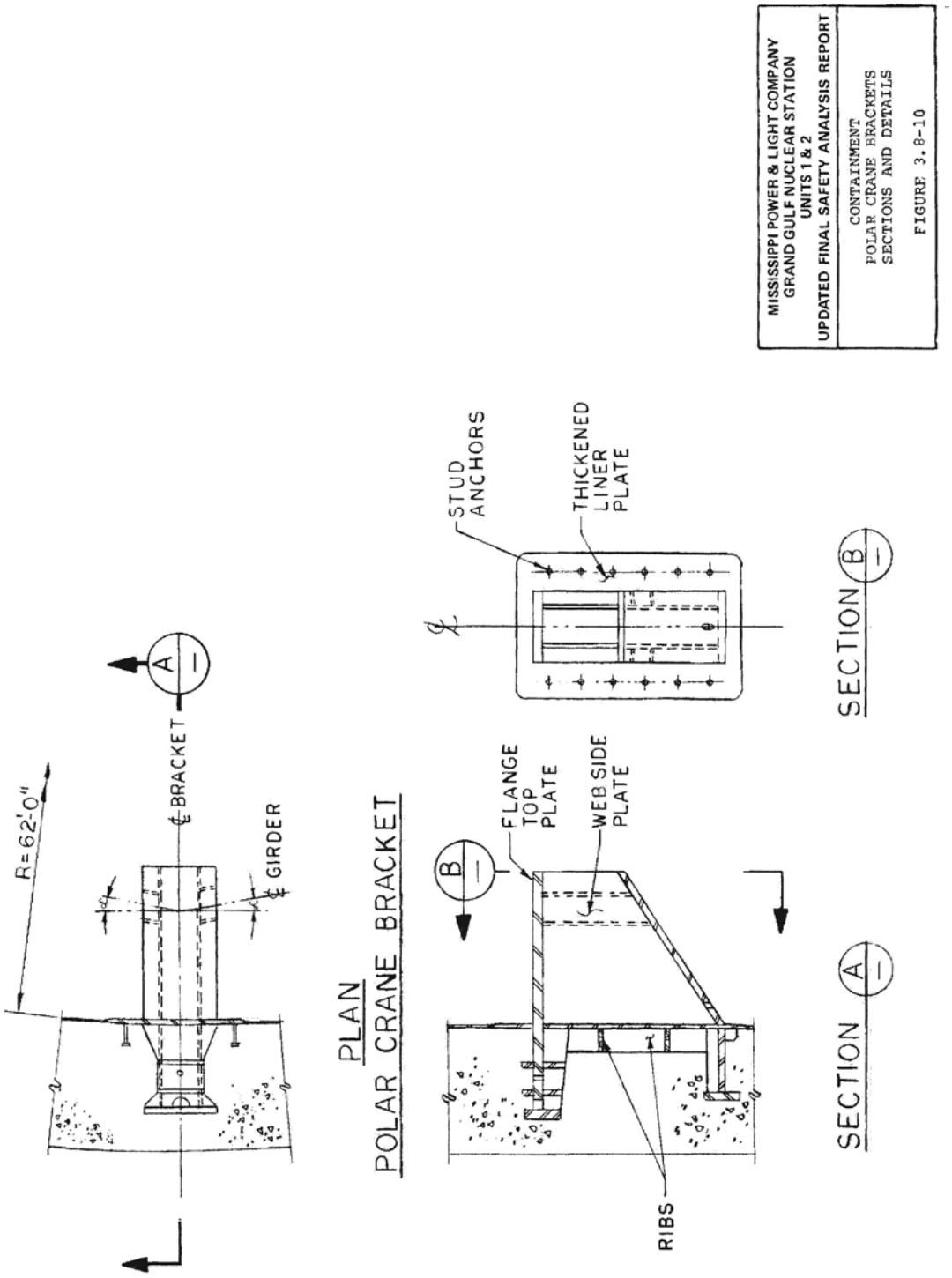
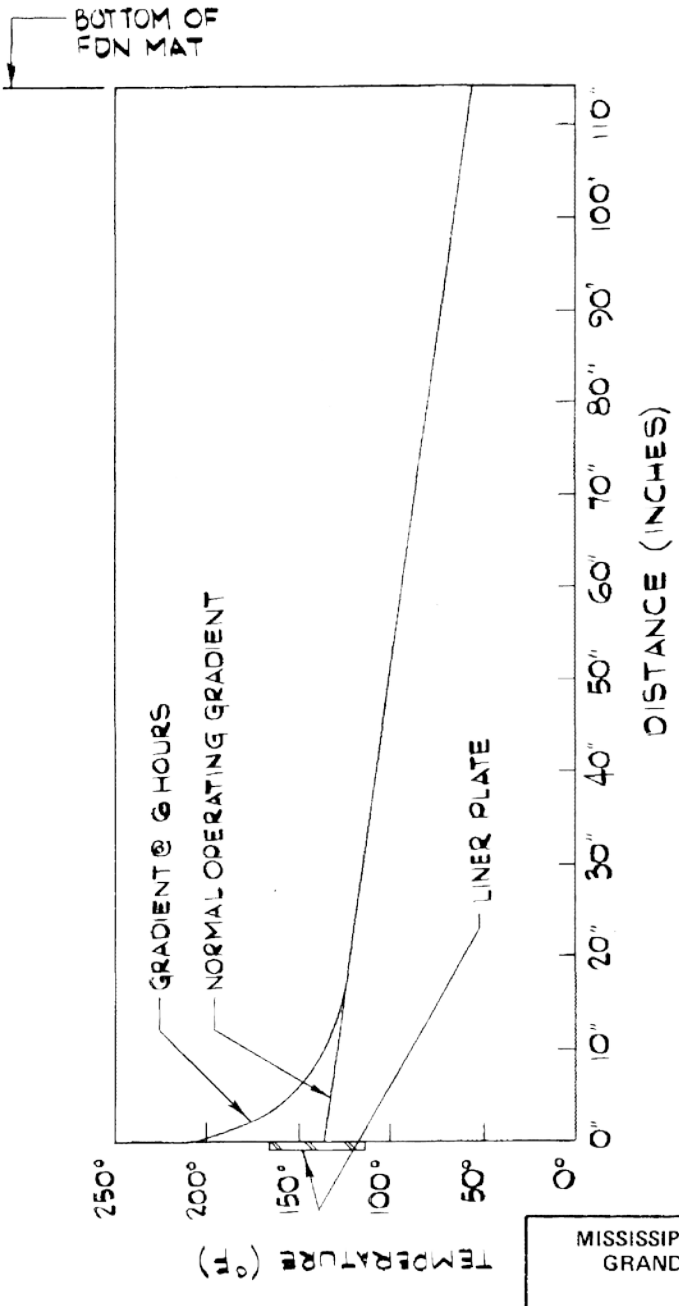
CONTAINMENT
PENETRATION DETAILS
FIGURE 3.8-8

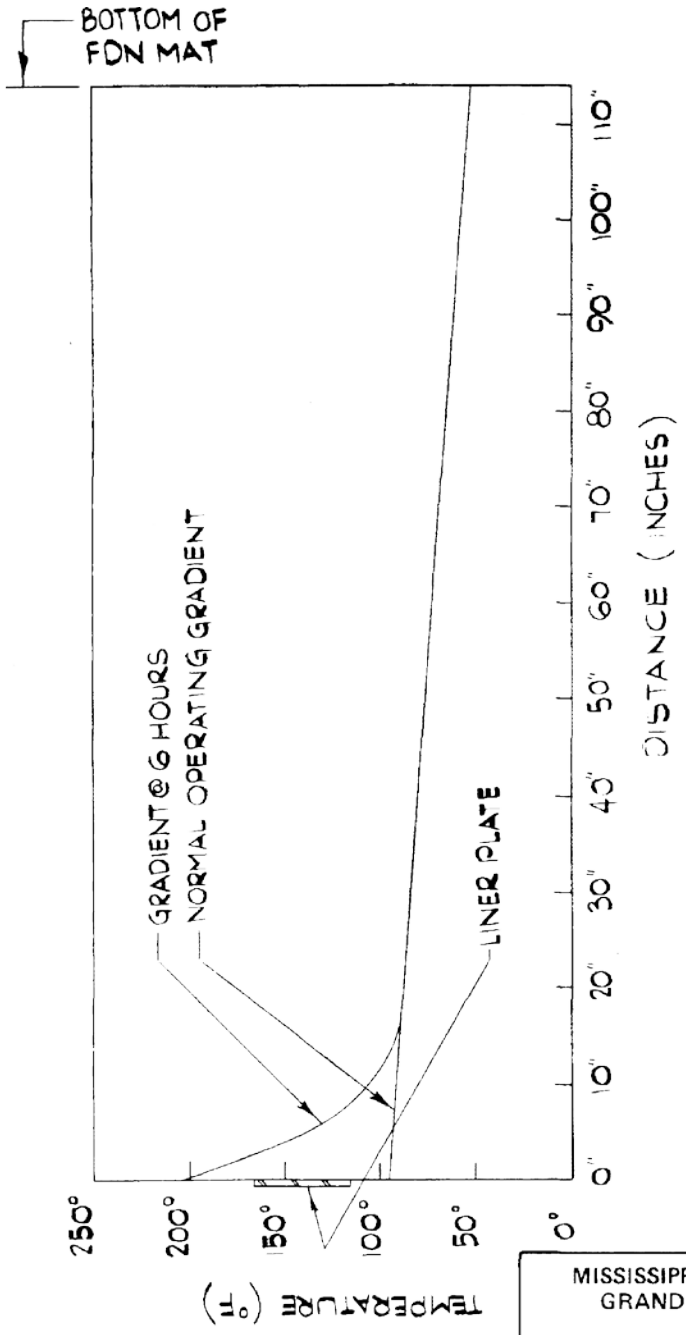

MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT
CONTAINMENT
DOME REINFORCEMENT
FIGURE 3.8-9
SECTION
PLAN
DEVELOPED ELEVATION
OUTSIDE FACE REINFORCEMENT
(HOOP REINF NOT SHOWN)
C-1036A 3


MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT
CONTAINMENT
POLAR CRANE BRACKETS
SECTIONS AND DETAILS
FIGURE 3.8-10
STUD ANCHORS
THICKENED LINER PLATE
SECTION B
PLAN
POLAR CRANE BRACKET
FLANGE TOP PLATE
WEB SIDE PLATE
RIBS
SECTION A
R=62'0"
℄ BRACKET
℄ GIRDER



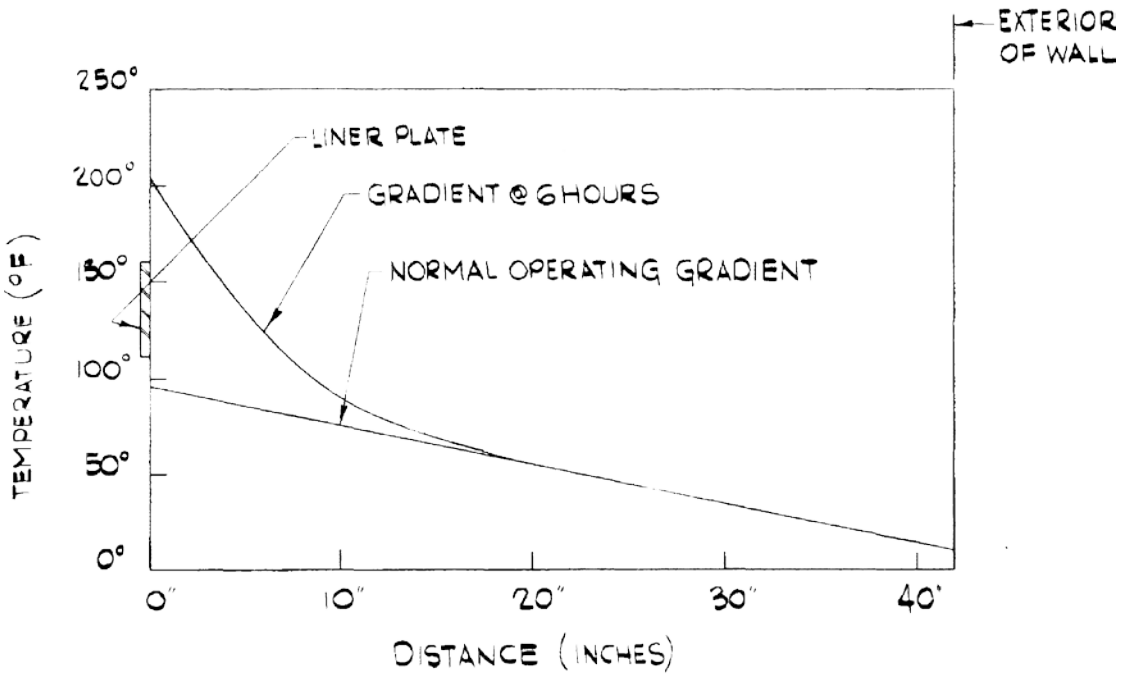
MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT

CONTAINMENT
THERMAL GRADIENT ACROSS FOUNDATION MAT BELOW REACTOR PRESSURE VESSEL
DESIGN BASIS ACCIDENT
FIGURE 3.8-11

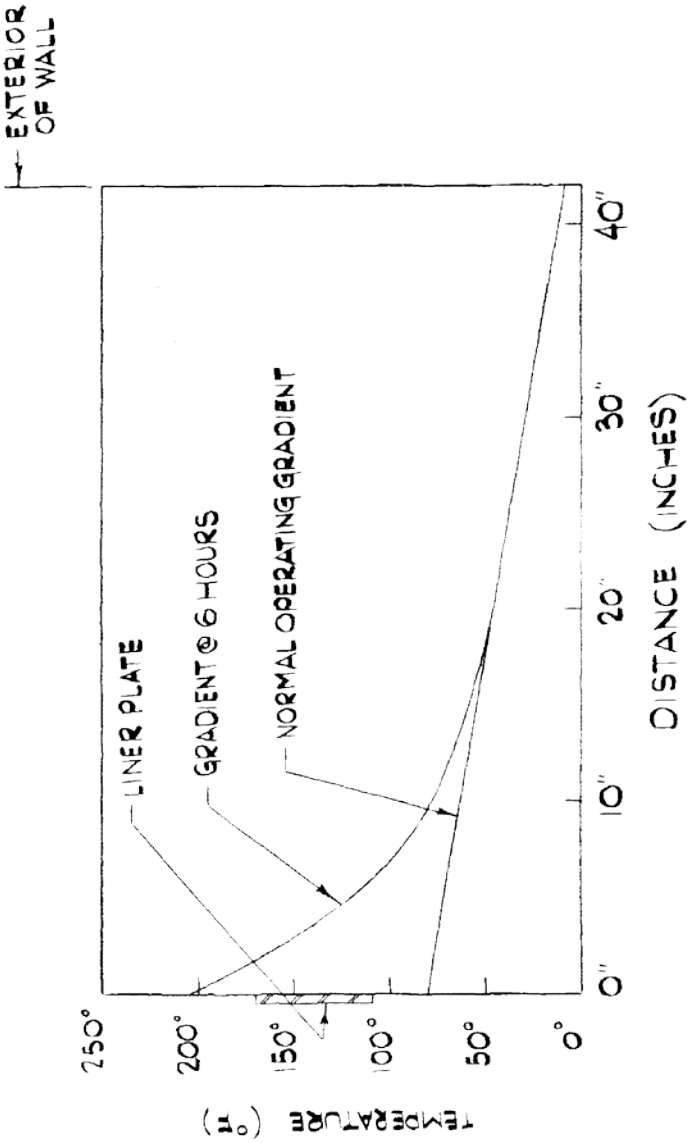


MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT

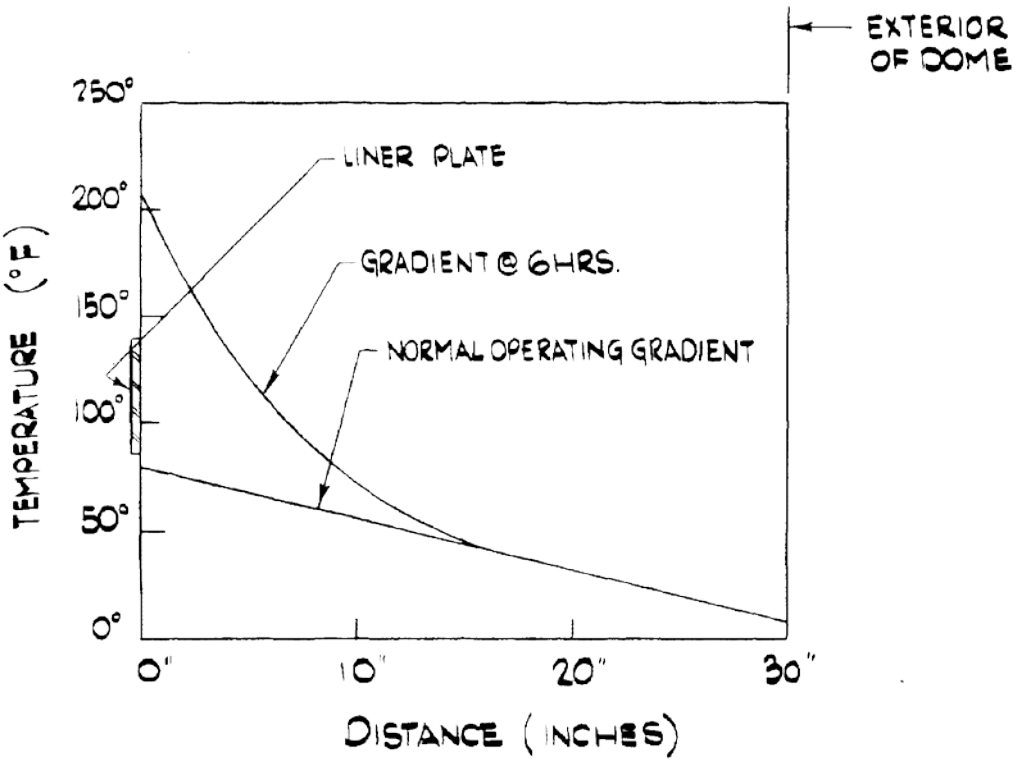
CONTAINMENT
THERMAL GRADIENT ACROSS FOUNDATION
MAT BELOW SUPPRESSION POOL
DESIGN BASIS ACCIDENT
FIGURE 3.8-12



MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT
CONTAINMENT THERMAL GRADIENT ACROSS WALL ADJACENT TO SUPPRESSION POOL DESIGN BASIS ACCIDENT FIGURE 3.8-13



MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT
CONTAINMENT THERMAL GRADIENT ACROSS WALL ABOVE SUPPRESSION POOL DESIGN BASIS ACCIDENT FIGURE 3.8-14



MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT
CONTAINMENT THERMAL GRADIENT ACROSS DOME DESIGN BASIS ACCIDENT FIGURE 3.8-15

Figure 3.8-16

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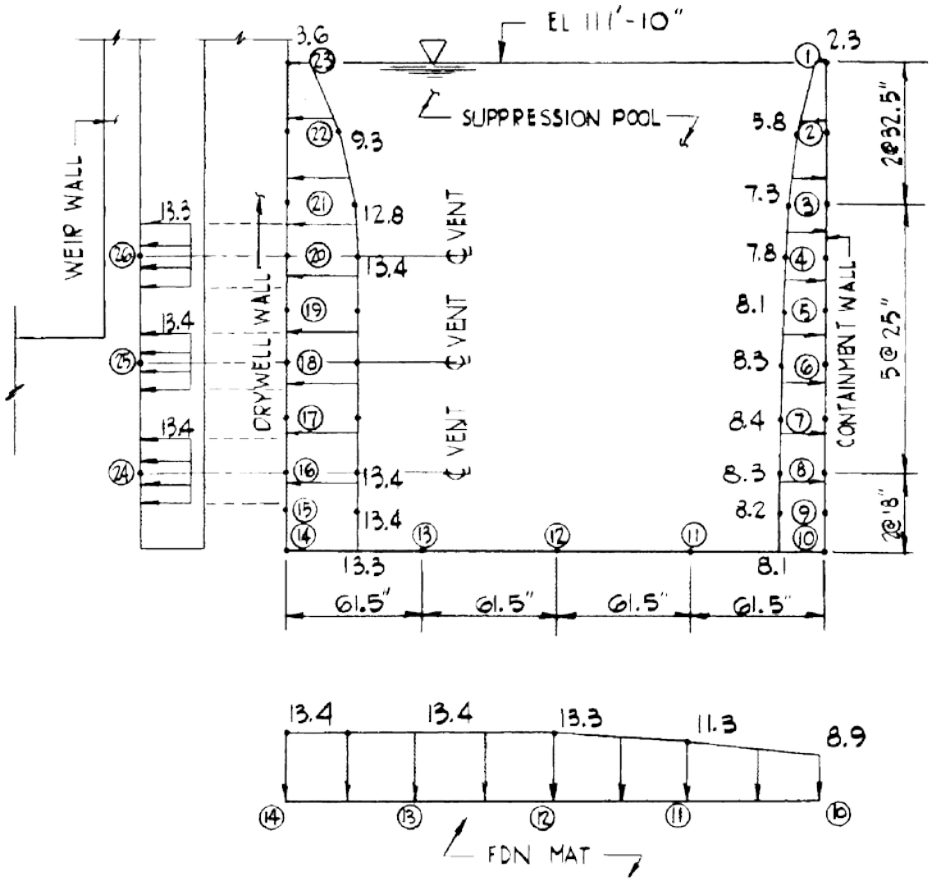
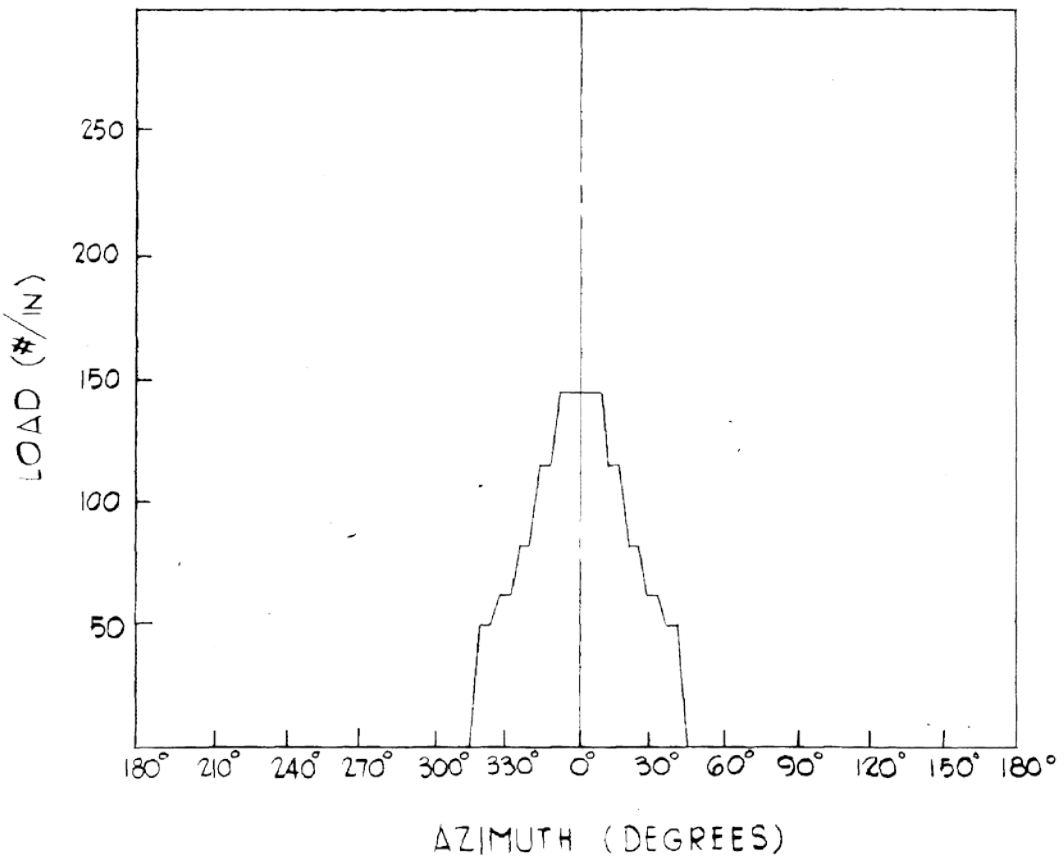
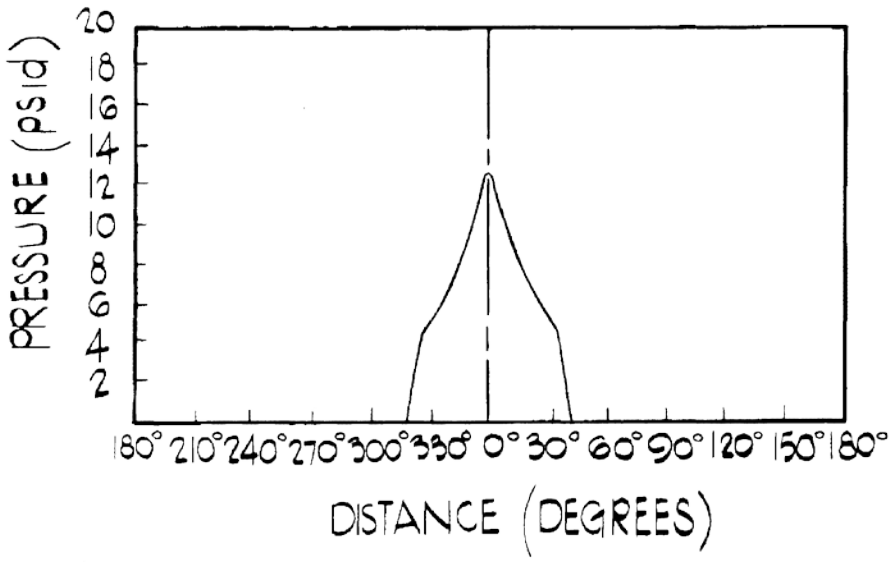


MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT
ONE VALVE FIRST ACTUATION S/R VALVE DISCHARGE LOADS LOCAL PEAK PRESSURE SECT AT 0° (REF APPENDICES 6A AND 6D) FIGURE 3.8 – 16a

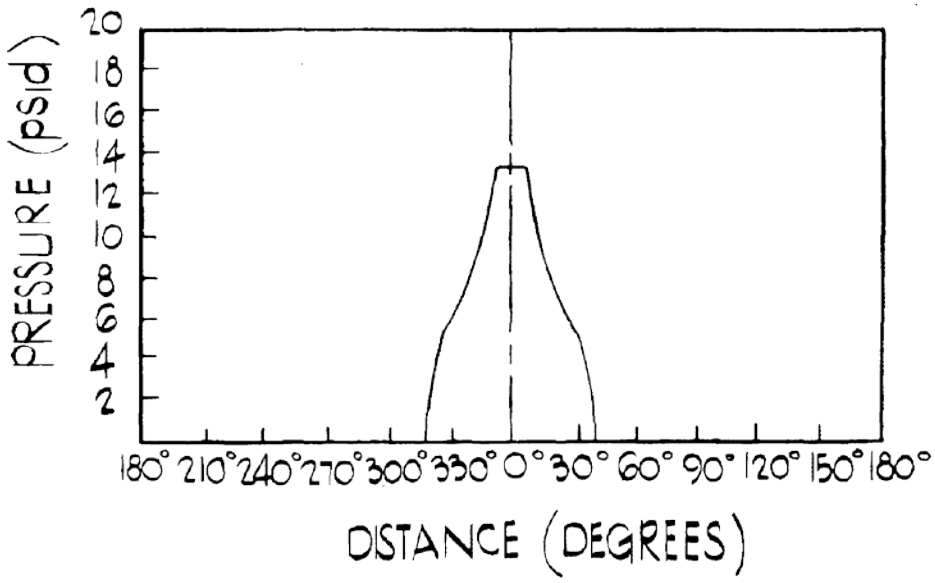


MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT
CONTAINMENT ONE VALVE FIRST ACTUATION CIRCUMFERENTIAL PRESSURE DISTRIBUTION AT REF PT 26 (REF APPENDICES 6A AND 6D) FIGURE 3.8 – 16b



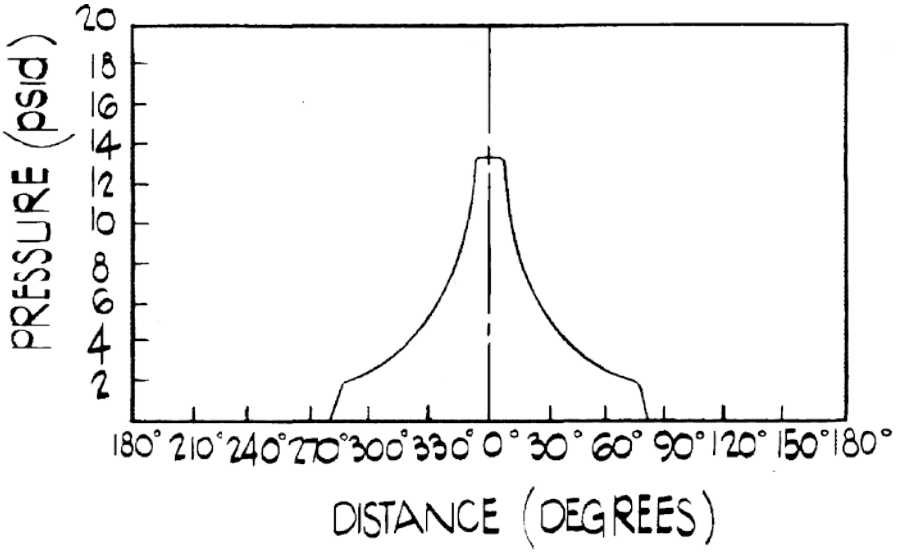
MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT

CONTAINMENT
ONE VALVE FIRST ACTUATION
CIRCUMFERENTIAL PRESSURE
DISTRIBUTION AT REF PT 21
(REF APPENDICES 6A AND 6D)
FIGURE 3.8 – 17



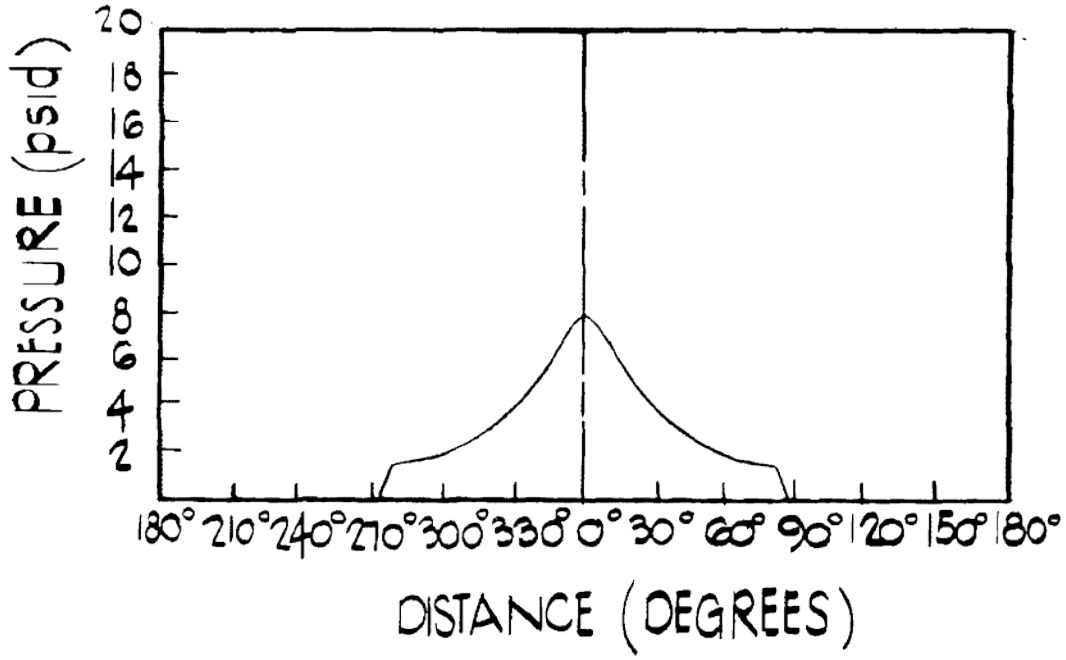
MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT

CONTAINMENT
ONE VALVE FIRST ACTUATION
CIRCUMFERENTIAL PRESSURE
DISTRIBUTION AT REF PT 17
(REF APPENDICES 6A AND 6D)
FIGURE 3.8 – 18



MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT

CONTAINMENT
ONE VALVE FIRST ACTUATION
CIRCUMFERENTIAL PRESSURE
DISTRIBUTION AT REF PT 12
(REF APPENDICES 6A AND 6D)
FIGURE 3.8 – 19



MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT
CONTAINMENT ONE VALVE FIRST ACTUATION CIRCUMFERENTIAL PRESSURE DISTRIBUTION AT REF PT 5 (REF APPENDICES 6A AND 6D) FIGURE 3.8 – 20

Figure 3.8-21

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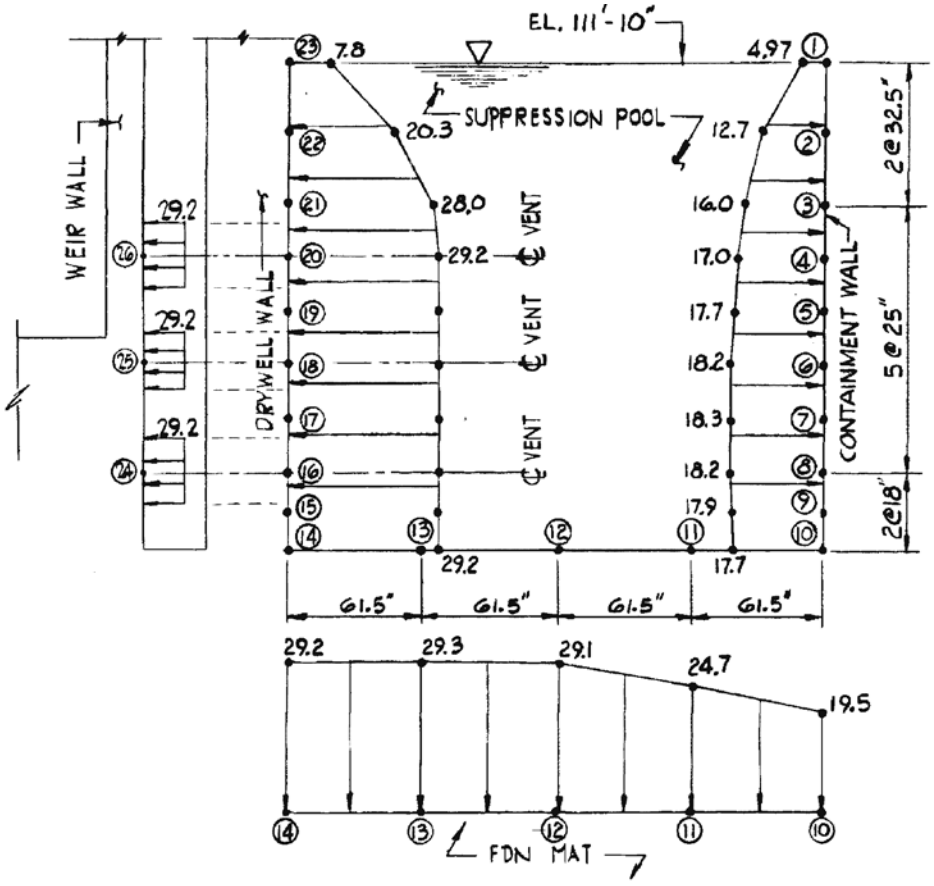

MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT

CONTAINMENT ONE VALVE SUBSEQUENT ACTUATION S/R VALVE DISCHARGE LOAD LOCAL PEAK PRESSURE, SECT AT 0°
(REF APPENDICES 6A AND 6D)
FIGURE 3.8 – 21a

Figure 3.8-22

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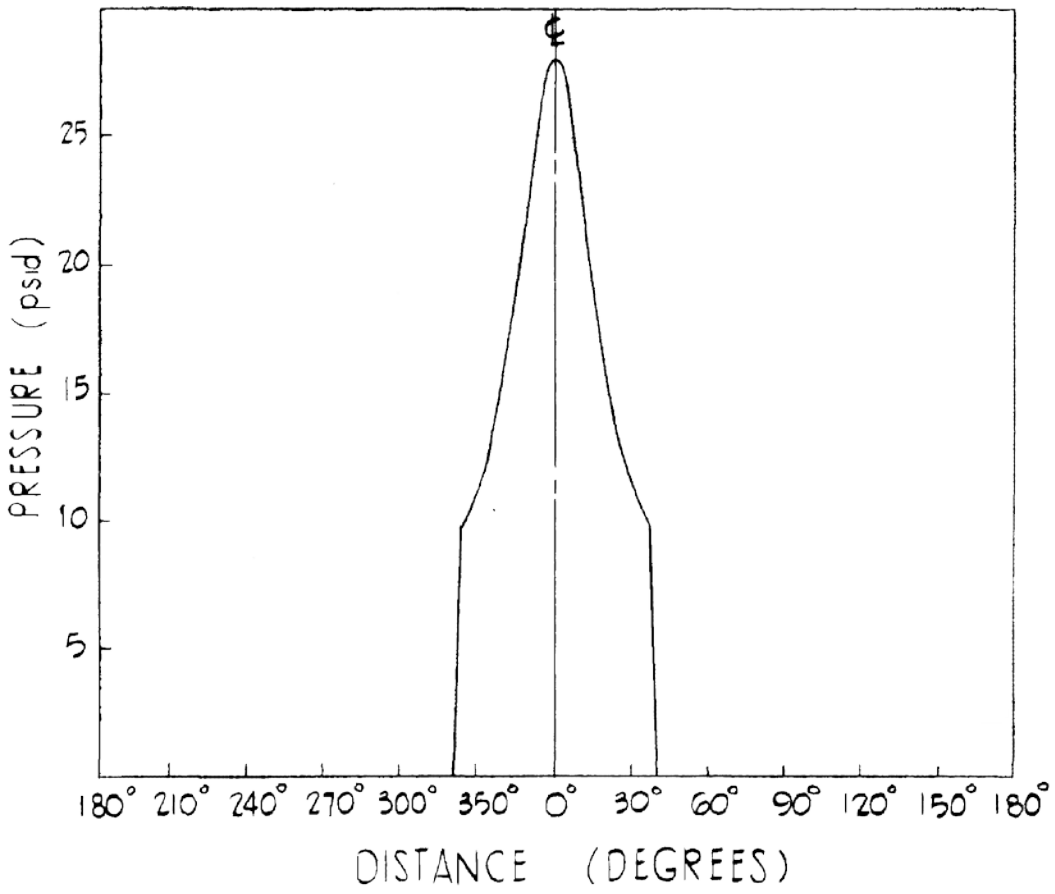

MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT

CONTAINMENT ONE VALVE SUBSEQUENT ACTUATION CIRCUMFERENTIAL PRESSURE DISTRIBUTION AT REF PT 21 (REF APPENDICES 6A AND 6D)

FIGURE 3.8 – 22a

Figure 3.8-23

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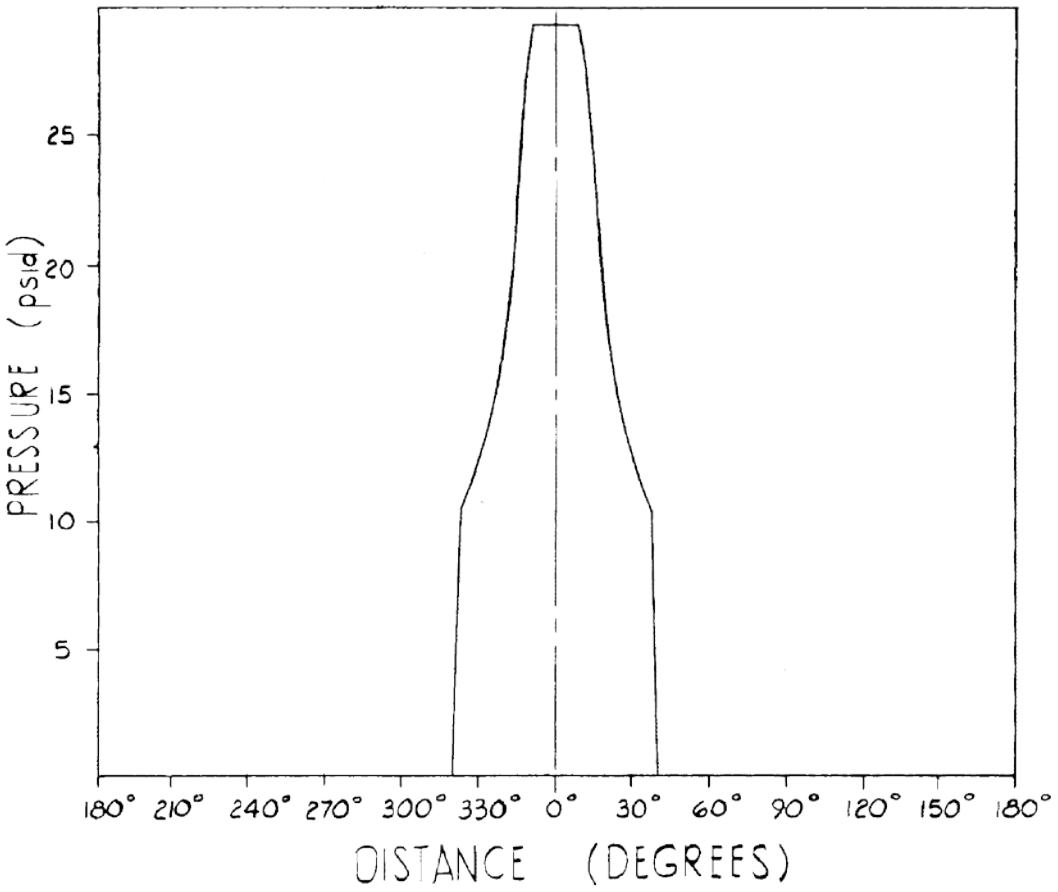

MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT

CONTAINMENT ONE VALVE SUBSEQUENT ACTUATION CIRCUMFERENTIAL PRESSURE DISTRIBUTION AT REF PT 17 (REF APPENDICES 6A AND 6D)

FIGURE 3.8 – 23a

Figure 3.8-24

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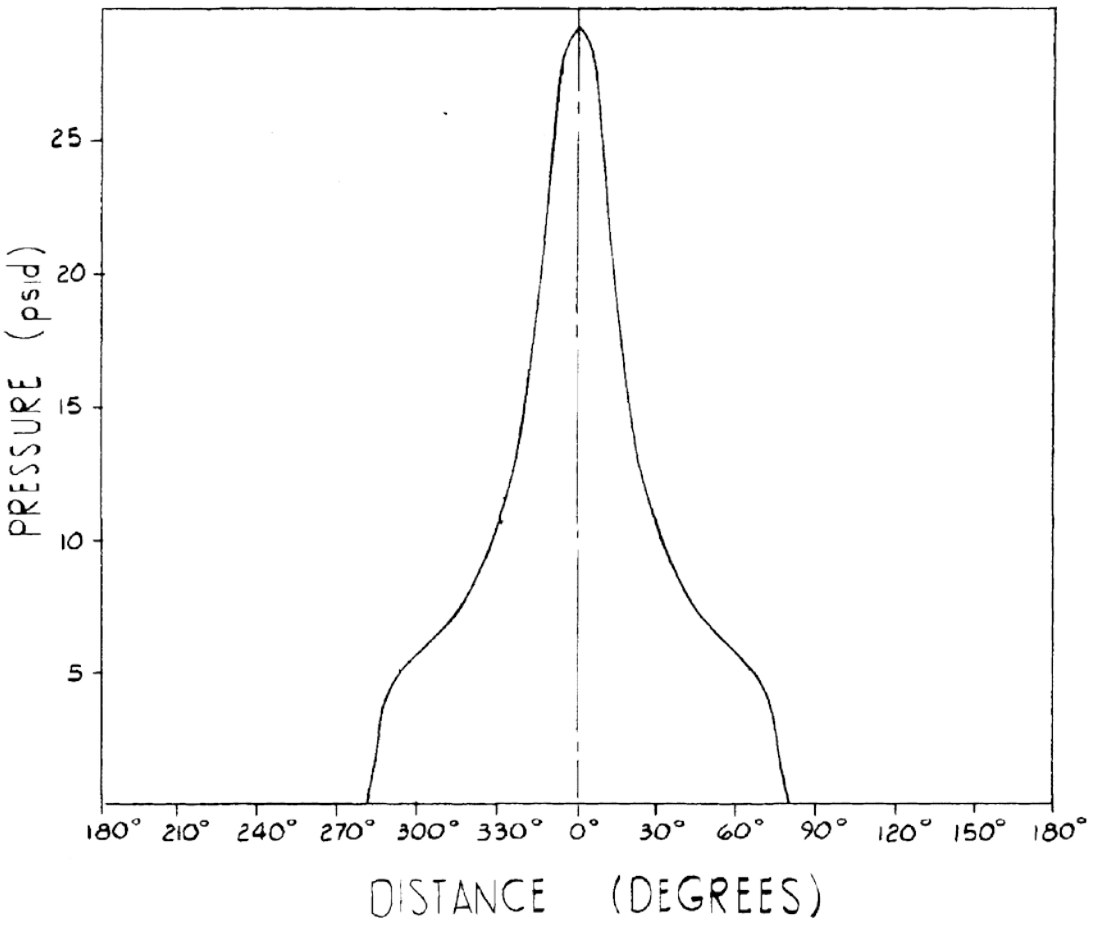

MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT

CONTAINMENT ONE VALVE SUBSEQUENT ACTUATION CIRCUMFERENTIAL PRESSURE DISTRIBUTION AT REF PT 12
(REF APPENDICES 6A AND 6D)

FIGURE 3.8 – 24a

Figure 3.8-25

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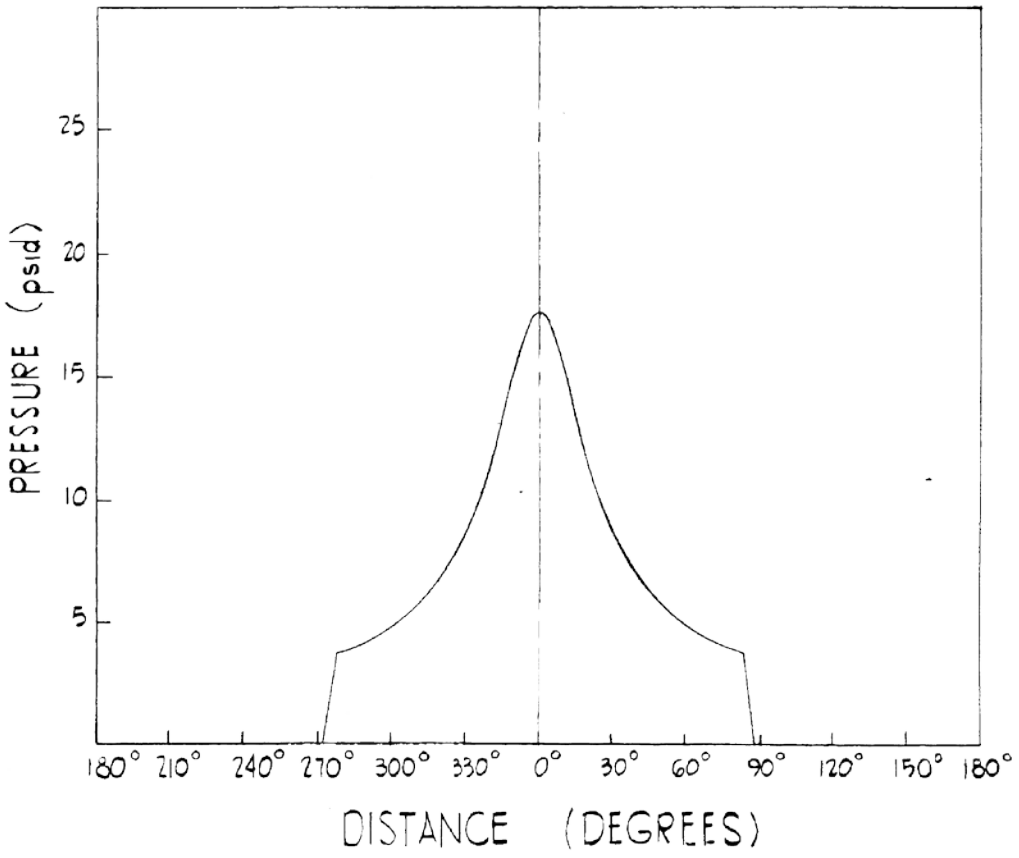
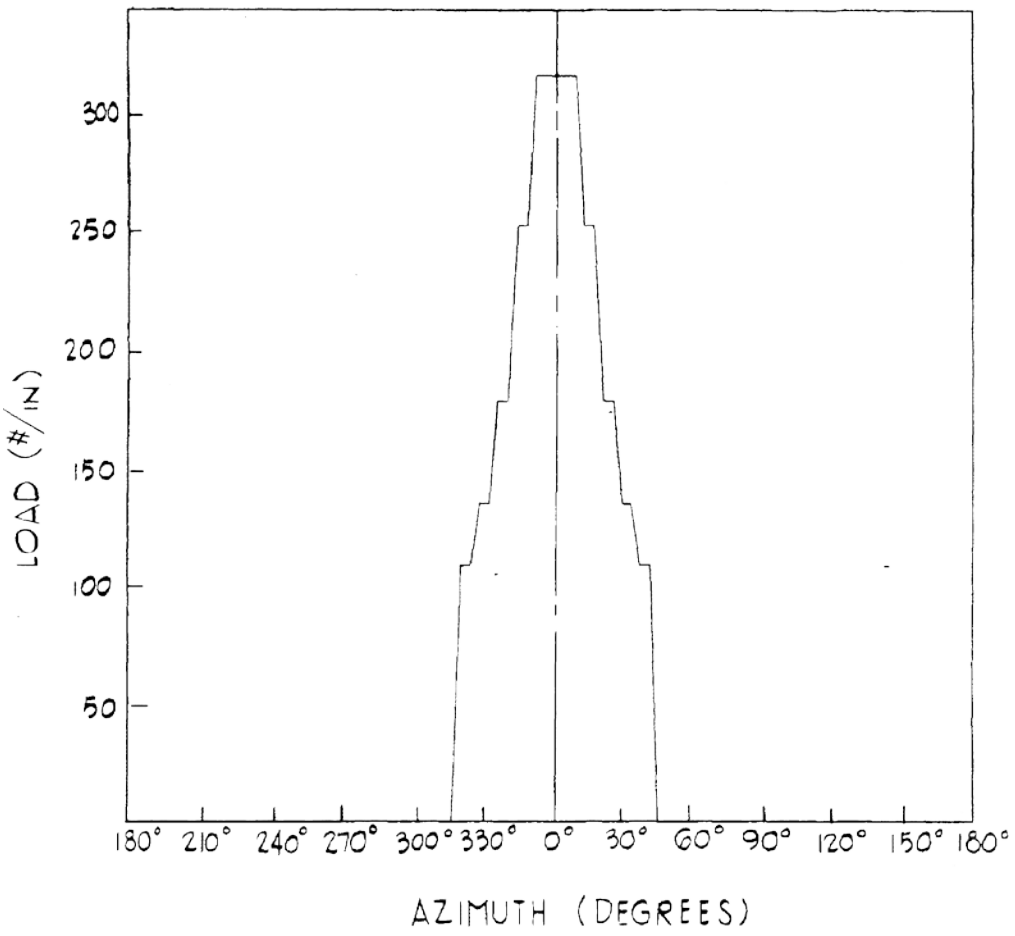


MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT

CONTAINMENT ONE VALVE SUBSEQUENT ACTUATION CIRCUMFERENTIAL PRESSURE DISTRIBUTION AT REF PT 5
(REF APPENDICES 6A AND 6D)

FIGURE 3.8 – 25a



MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT

CONTAINMENT ONE VALVE SUBSEQUENT ACTUATION CIRCUMFERENTIAL PRESSURE DISTRIBUTION AT REF PT 26 (REF APPENDICES 6A AND 6D)

FIGURE 3.8 – 25b

Figure 3.8-26

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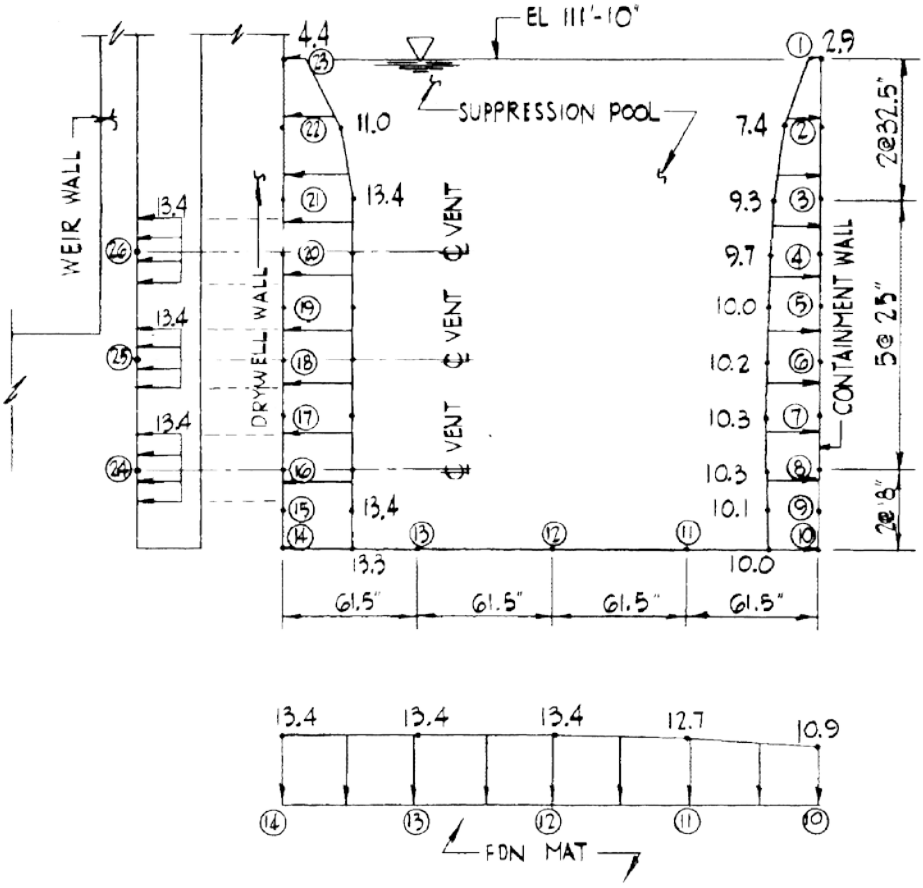

MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT
CONTAINMENT 2 ADJACENT VALVES S/R VALVE DISCHARGE LOADS LOCAL PEAK PRESSURE, SECT AT 0° (REF APPENDICES 6A AND 6D) FIGURE 3.8 – 26a

Figure 3.8-27

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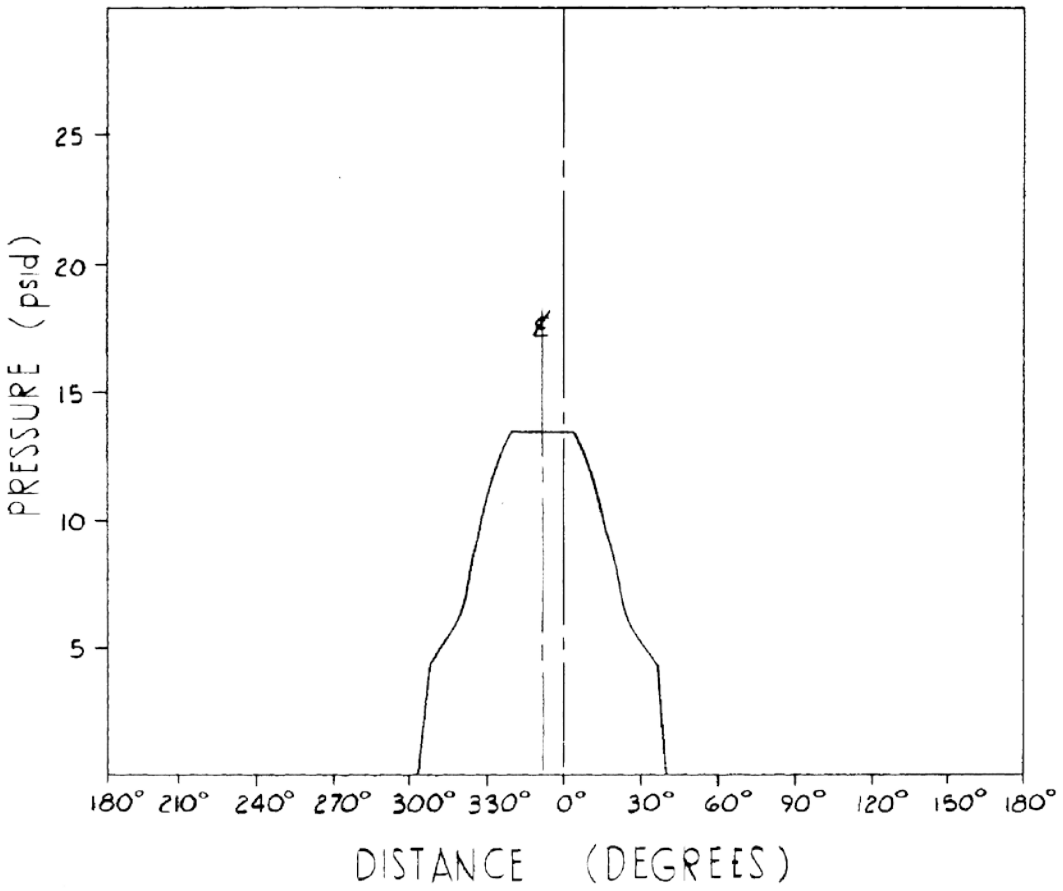

MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT

CONTAINMENT 2 ADJACENT VALVES
CIRCUMFERENTIAL PRESSURE
DISTRIBUTRION AT REF PT 21
(REF APPENDICES 6A AND 6D)

FIGURE 3.8 – 27a

Figure 3.8-28

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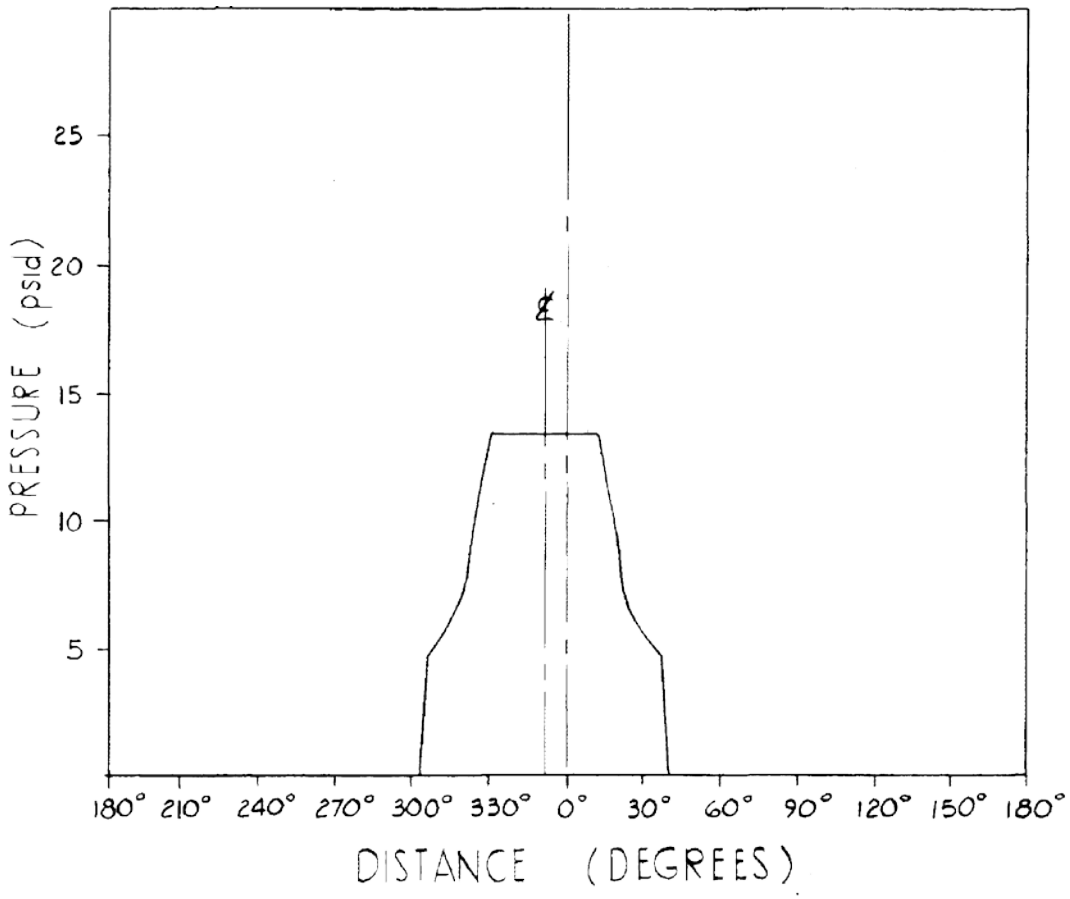
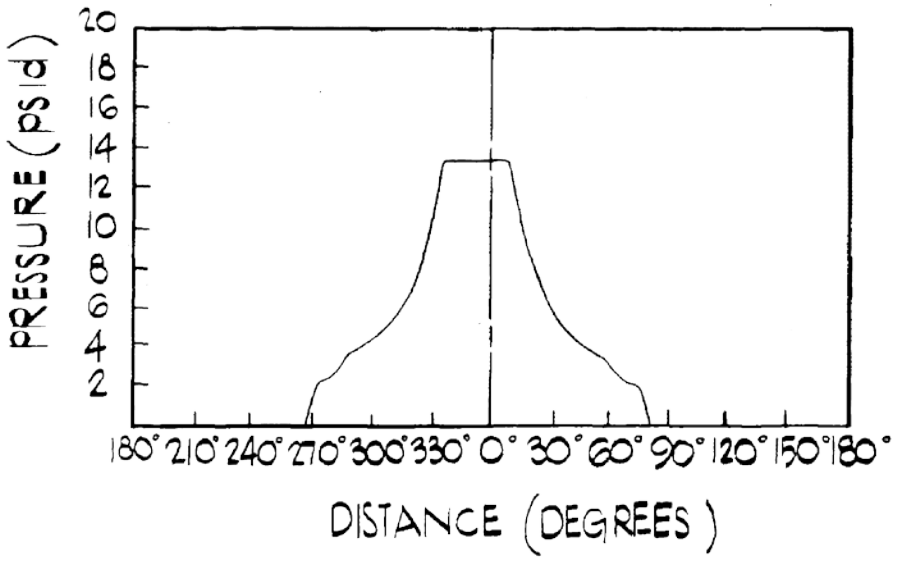


MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT

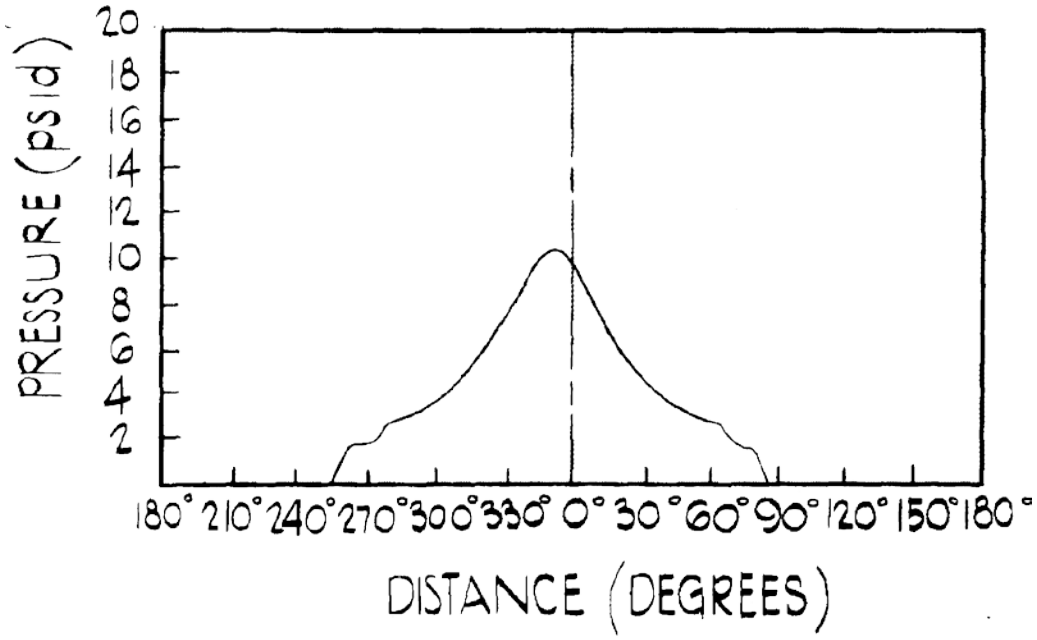
CONTAINMENT 2 ADJACENT VALVES
CIRCUMFERENTIAL PRESSURE
DISTRIBUTION AT REF PT 17
(REF APPENDICES 6A AND 6D)

FIGURE 3.8 – 28a



MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT

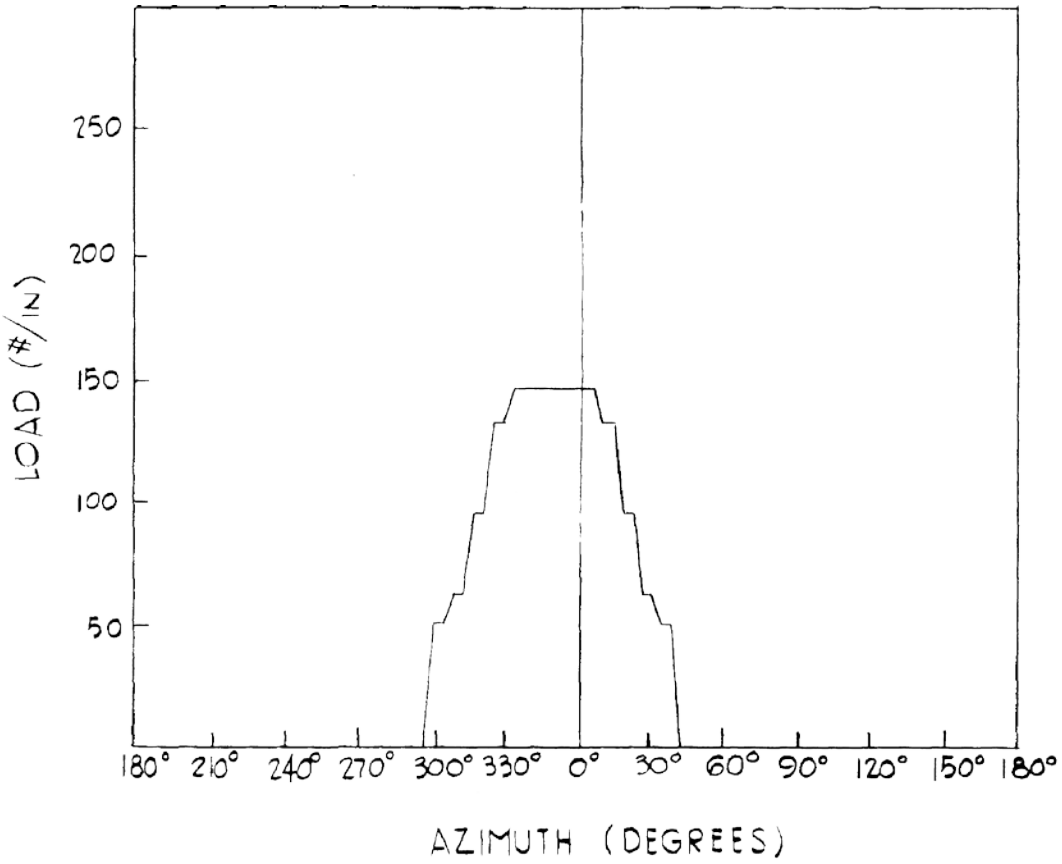
CONTAINMENT 2 ADJACENT VALVES
CIRCUMFERENTIAL PRESSURE
DISTRIBUTION AT REF PT 12
(REF APPENDICES 6A AND 6D)
FIGURE 3.8 – 29



MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT

CONTAINMENT 2 ADJACENT VALVES
CIRCUMFERENTIAL PRESSURE
DISTRIBUTION AT REF PT 5
(REF APPENDICES 6A AND 6D)

FIGURE 3.8 – 30



MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT

CONTAINMENT 2 ADJACENT VALVES
CIRCUMFERENTIAL PRESSURE
DISTRIBUTION AT REF PT 26
(REF APPENDICES 6A AND 6D)

FIGURE 3.8 – 30a

Figures 3.8-31

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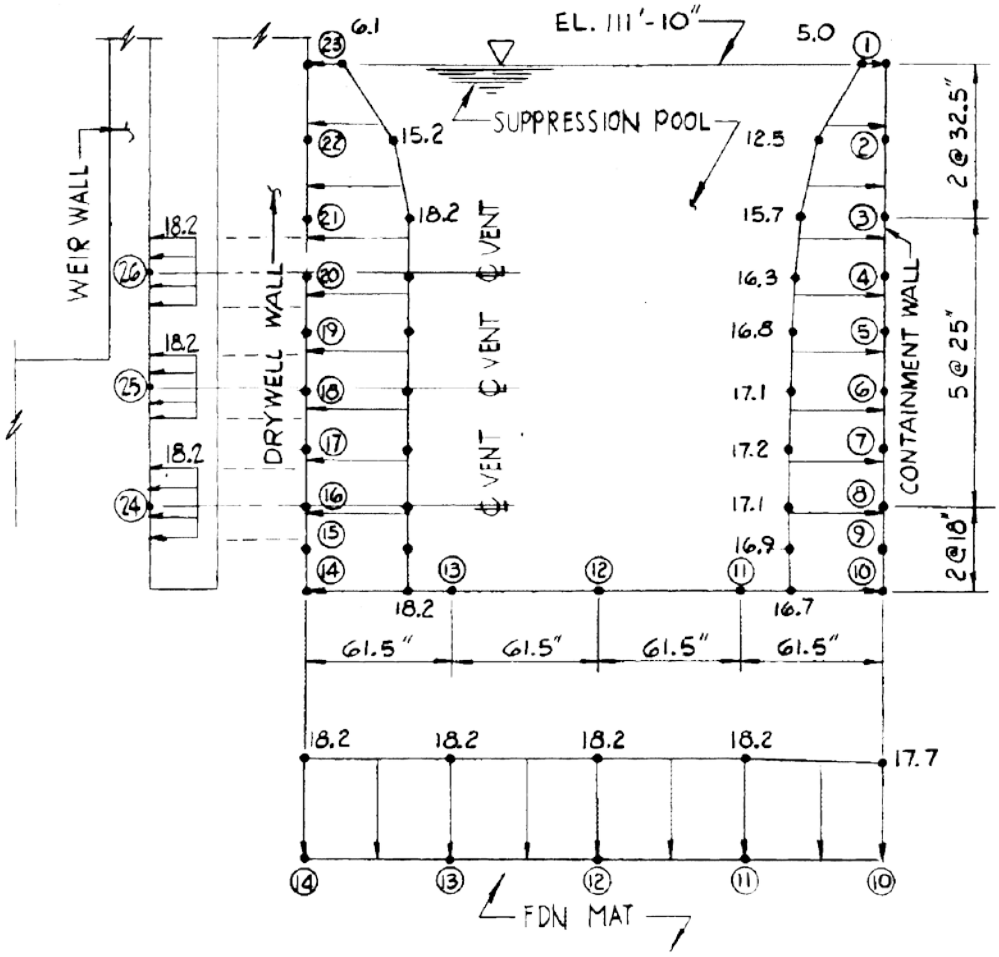

MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT
CONTAINMENT 11 VALVE CASE S/R VALVE DISCHARGE LOADS LOCAL PEAK PRESSURE, SECT AT 0° (REF APPENDICES 6A AND 6D) FIGURE 3.8 – 31a

Figure 3.8-32

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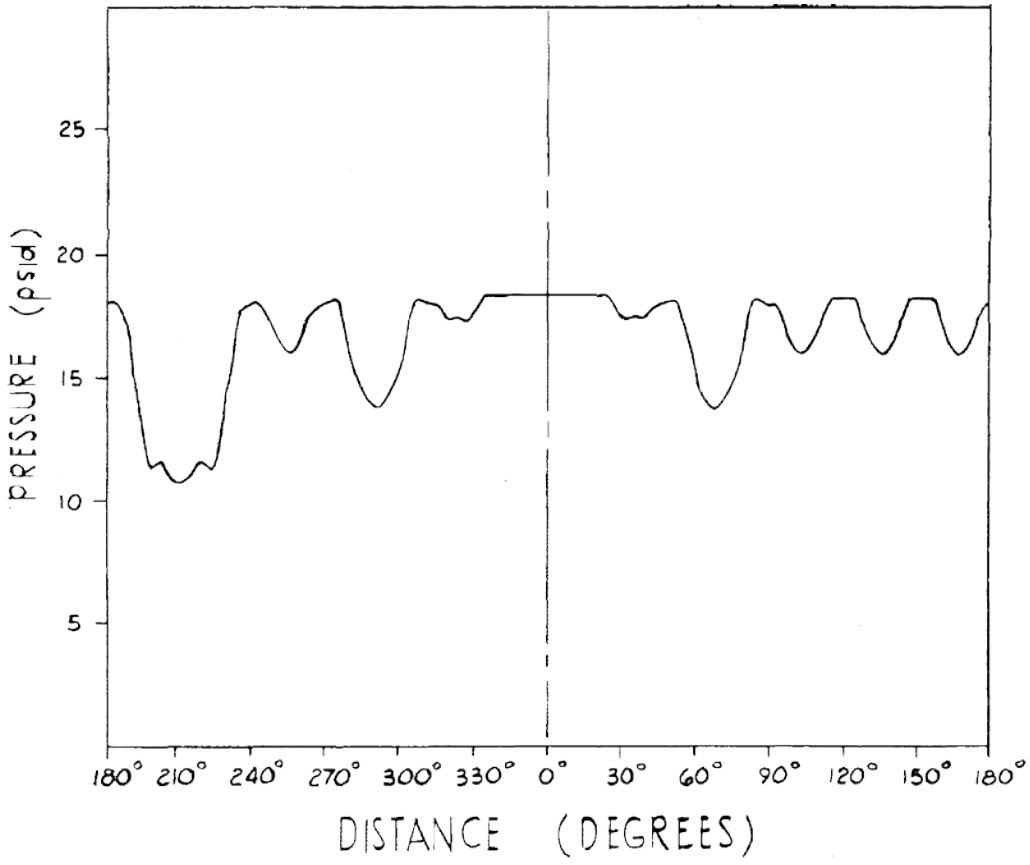

MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT

CONTAINMENT 11 VALVE CASE
CIRCUMFERENTIAL PRESSURE
DISTRIBUTION AT REF PT 21
(REF APPENDICES 6A AND 6D)

FIGURE 3.8 – 32a

Figure 3.8-33

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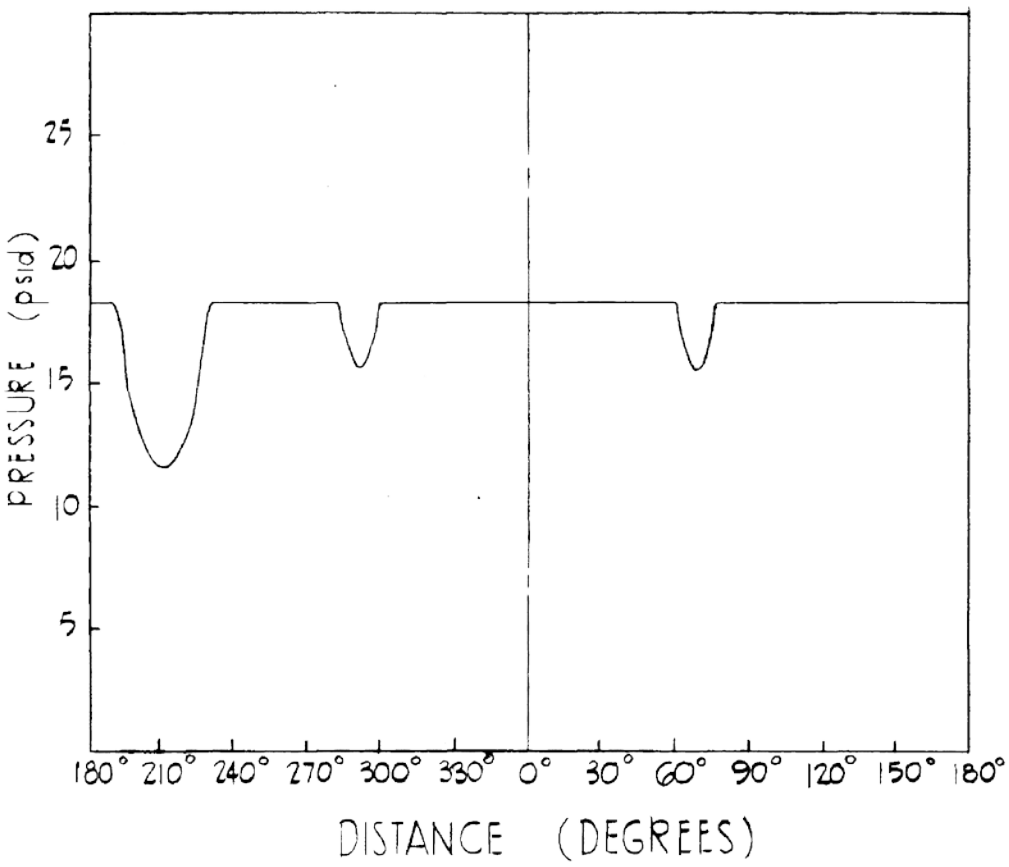

MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT

CONTAINMENT 11 VALVE CASE
CIRCUMFERENTIAL PRESSURE
DISTRIBUTION AT REF PT 17
(REF APPENDICES 6A AND 6D)

FIGURE 3.8 – 33a

Figure 3.8-34

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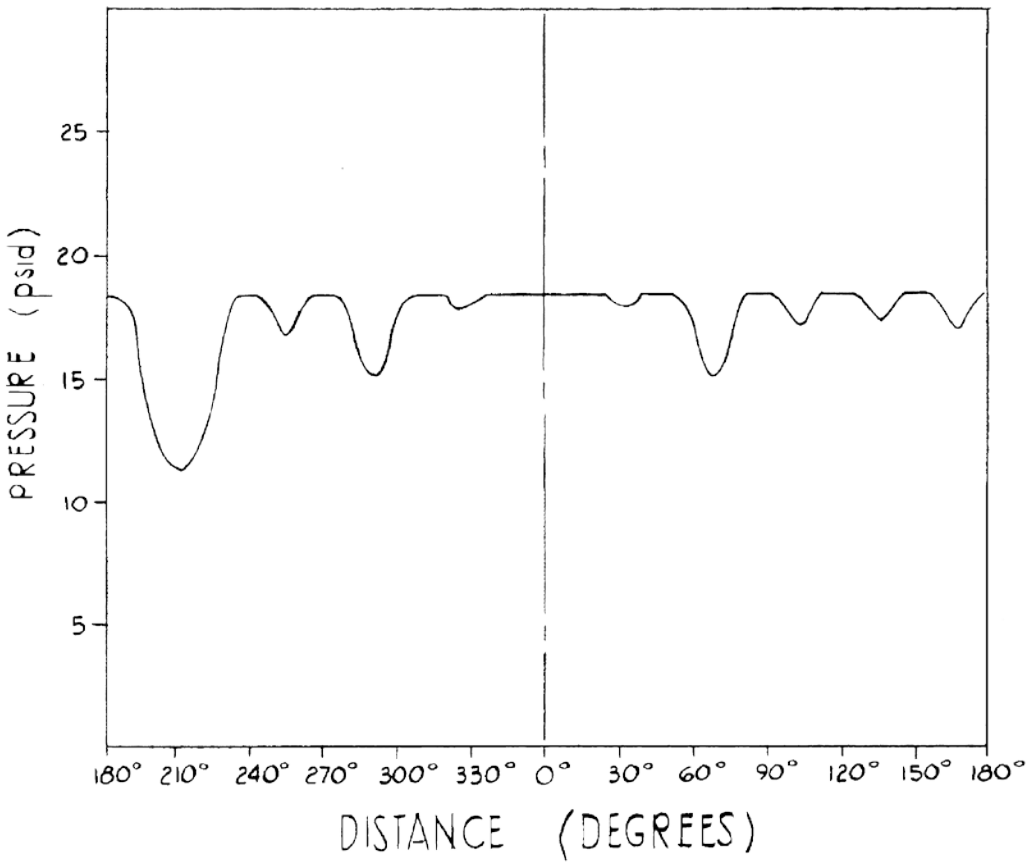

MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT

CONTAINMENT 11 VALVE CASE
CIRCUMFERENTIAL PRESSURE
DISTRIBUTION AT REF PT 12
(REF APPENDICES 6A AND 6D)

FIGURE 3.8 – 34a

Figure 3.8-35

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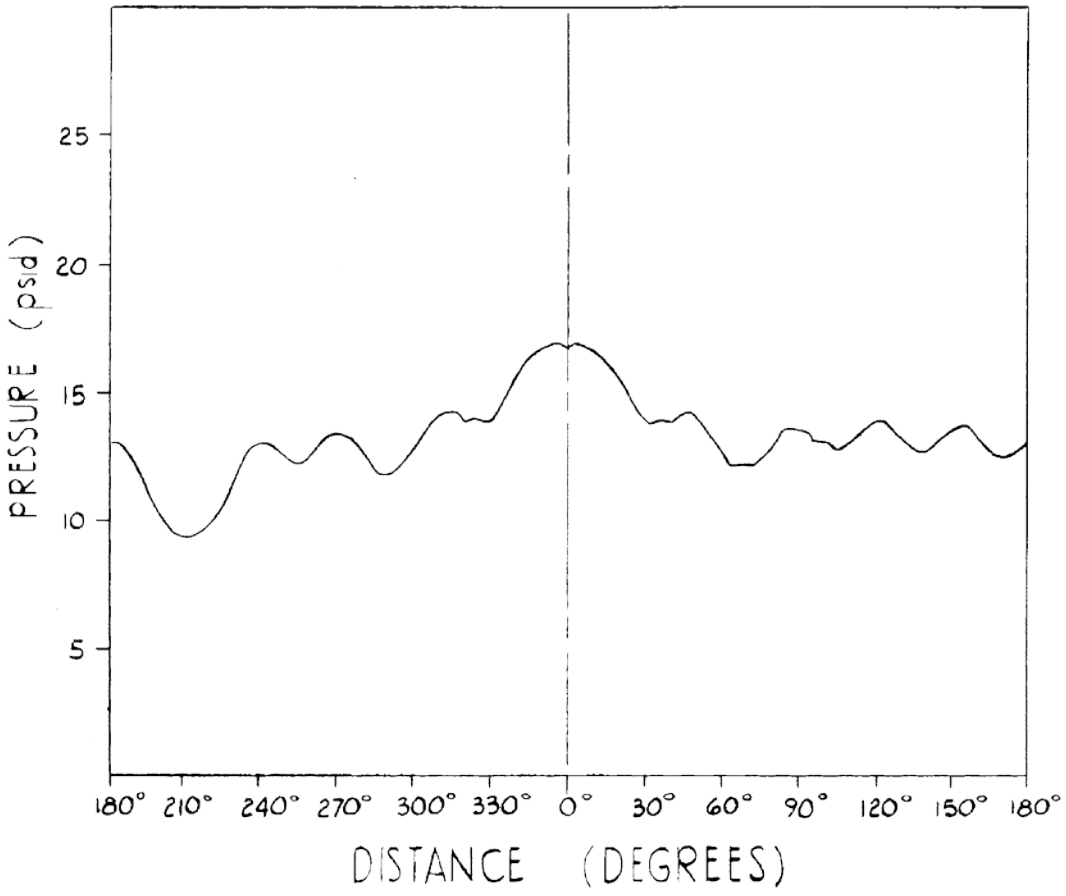
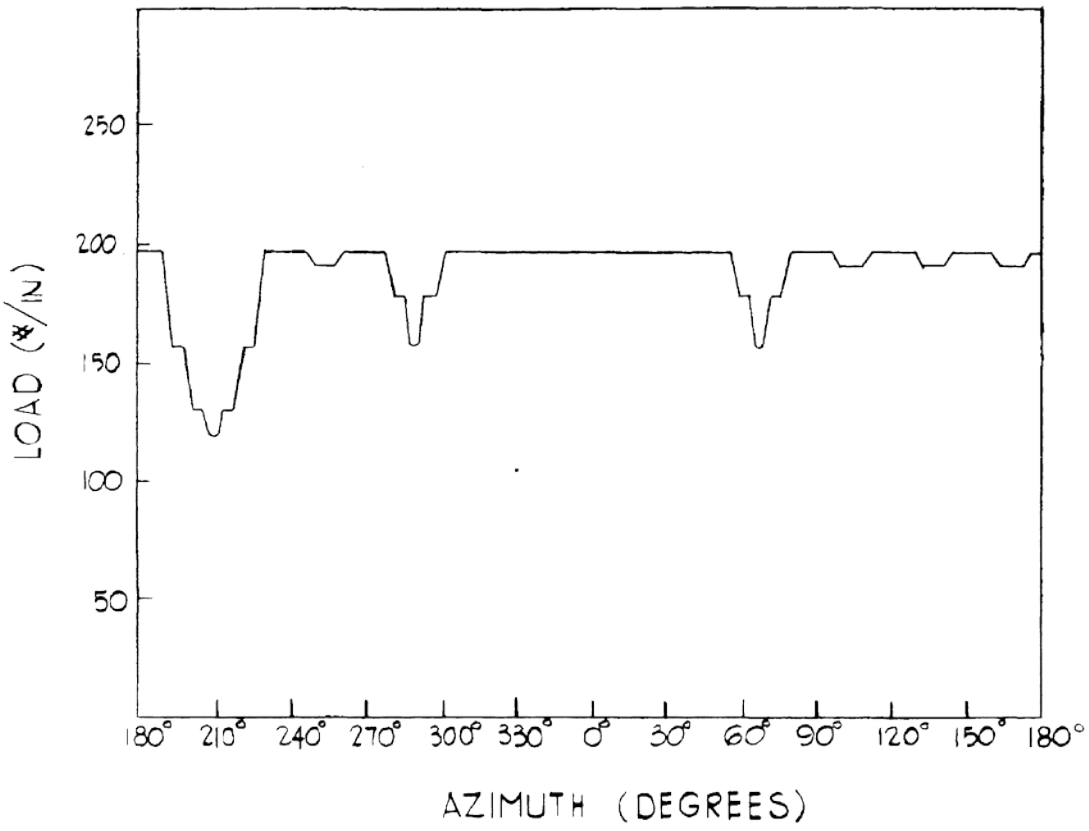


MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT

CONTAINMENT 11 VALVE CASE
CIRCUMFERENTIAL PRESSURE
DISTRIBUTION AT REF PT 5
(REF APPENDICES 6A AND 6D)

FIGURE 3.8 – 35a



MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT

CONTAINMENT 11 VALVE CASE
CIRCUMFERENTIAL PRESSURE
DISTRIBUTION AT REF PT 26
(REF APPENDICES 6A AND 6D)
FIGURE 3.8 – 35b

Figure 3.8-36

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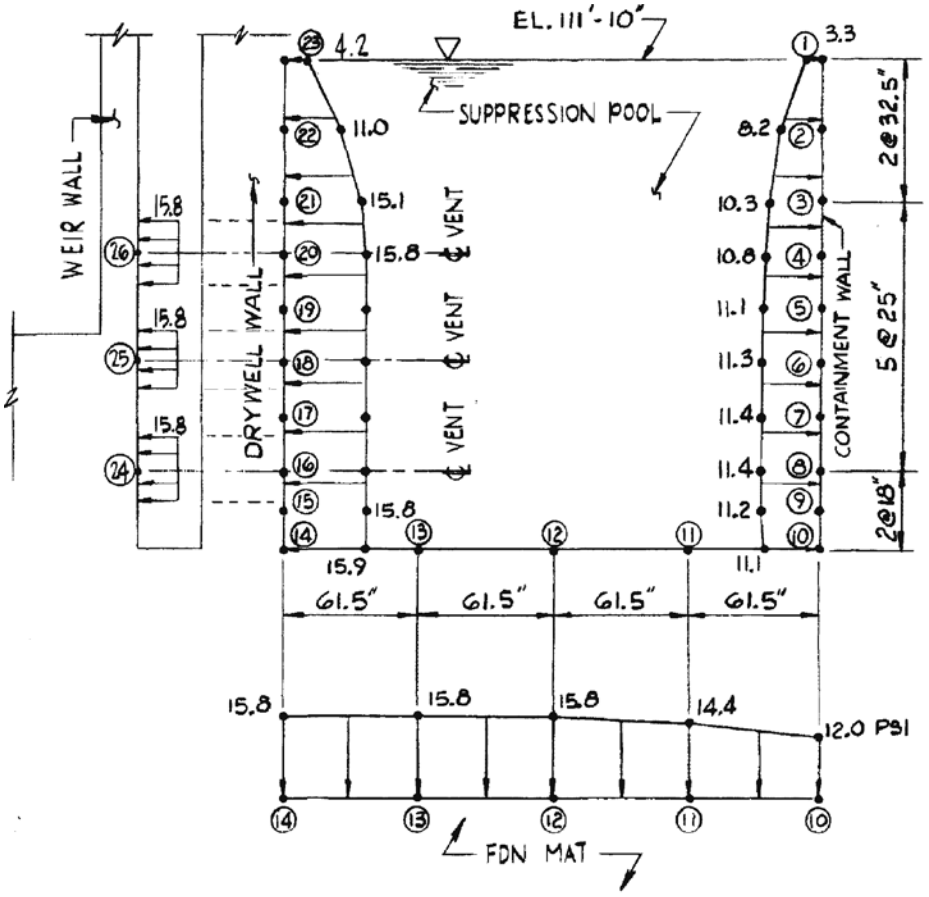

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MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT
CONTAINMENT 8 ADS VALVE CASE S/R VALVE DISCHARGE LOADS LOCAL PEAK PRESSURE, SECT AT 0° (REF APPENDICES 6A AND 6D) FIGURE 3.8 – 36a

Figure 3.8-37

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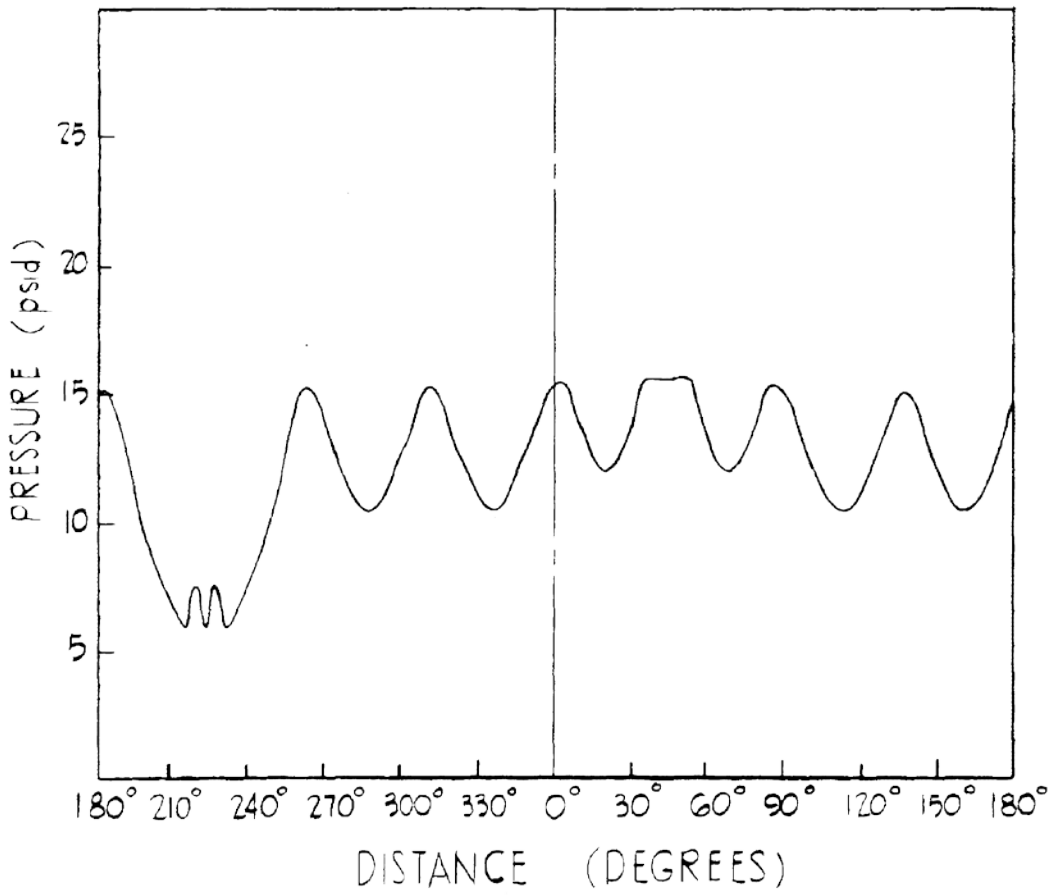

MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT

CONTAINMENT 8 ADS VALVE CASE
CIRCUMFERENTIAL PRESSURE
DISTRIBUTION AT REF PT 21
(REF APPENDICES 6A AND 6D)
FIGURE 3.8 – 37a

Figure 3.8-38

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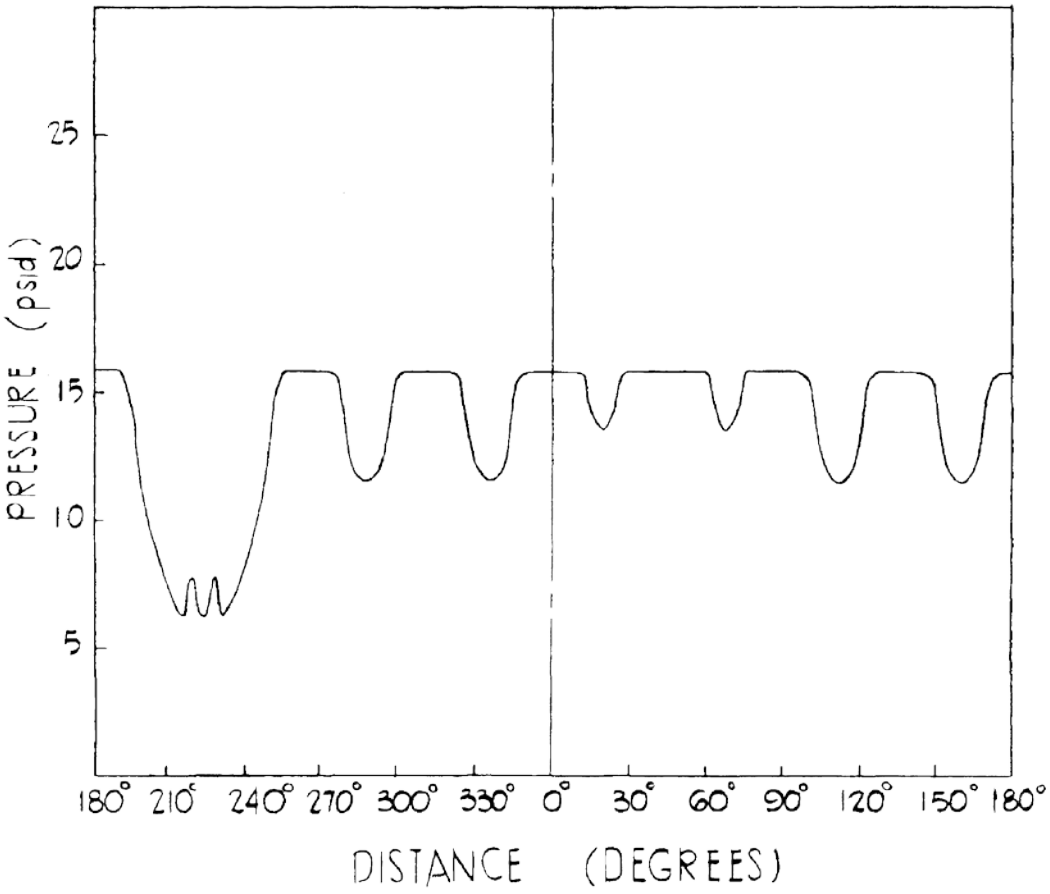

MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT

CONTAINMENT 8 ADS VALVE CASE
CIRCUMFERENTIAL PRESSURE
DISTRIBUTION AT REF PT 17
(REF APPENDICES 6A AND 6D)

FIGURE 3.8 – 38a

Figure 3.8-39

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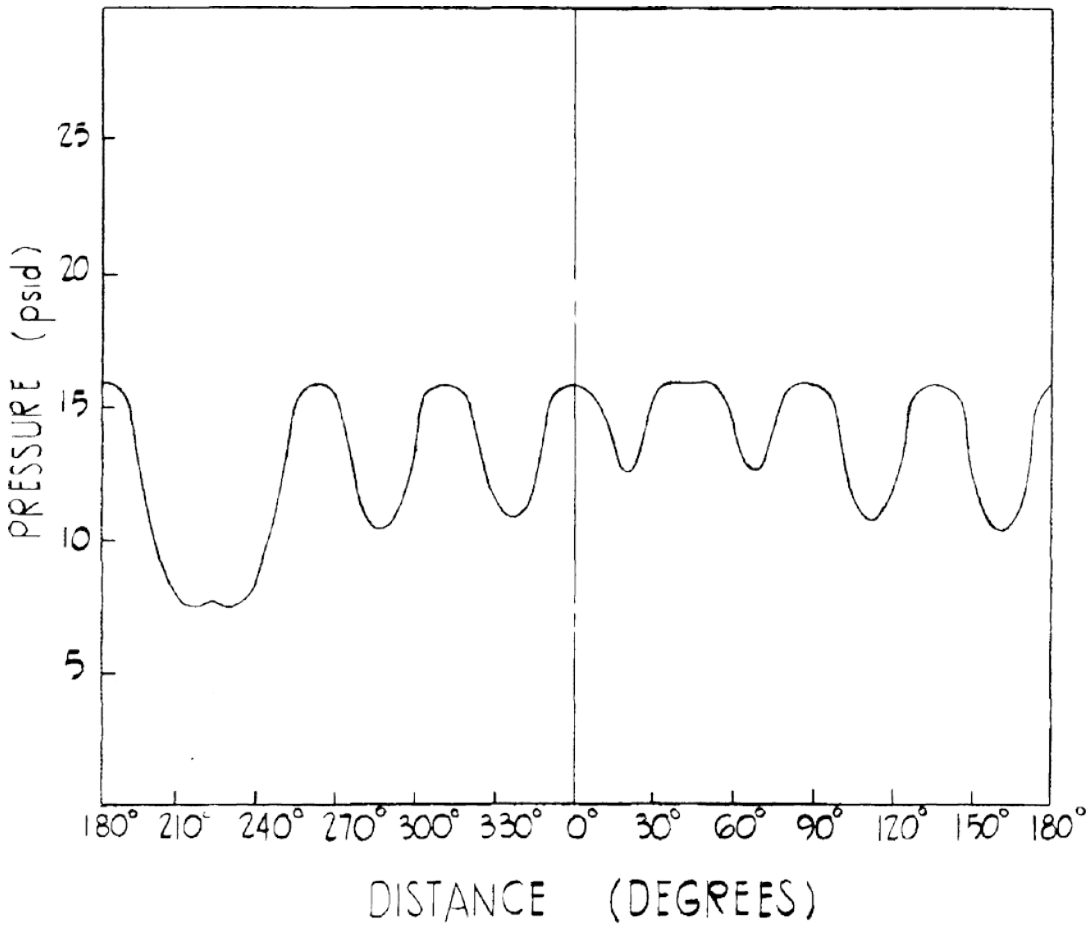

MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT

CONTAINMENT 8 ADS VALVE CASE
CIRCUMFERENTIAL PRESSURE
DISTRIBUTION AT REF PT 12
(REF APPENDICES 6A AND 6D)

FIGURE 3.8 – 39a

Figure 3.8-40

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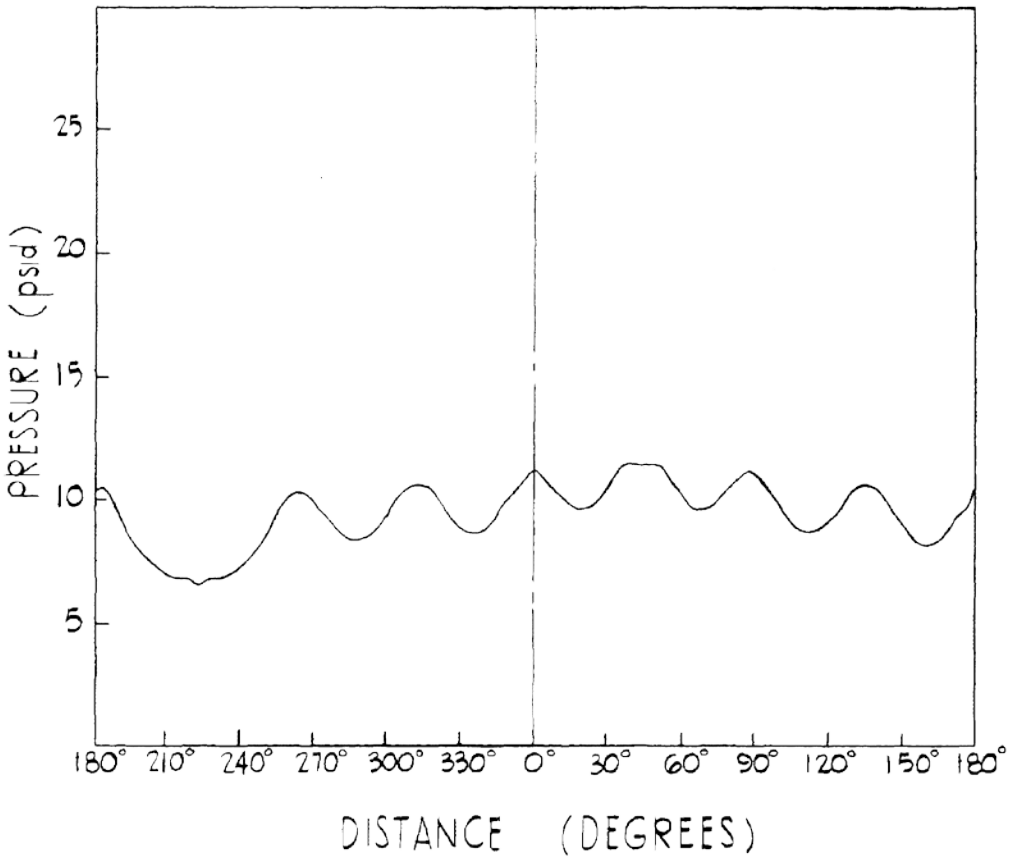
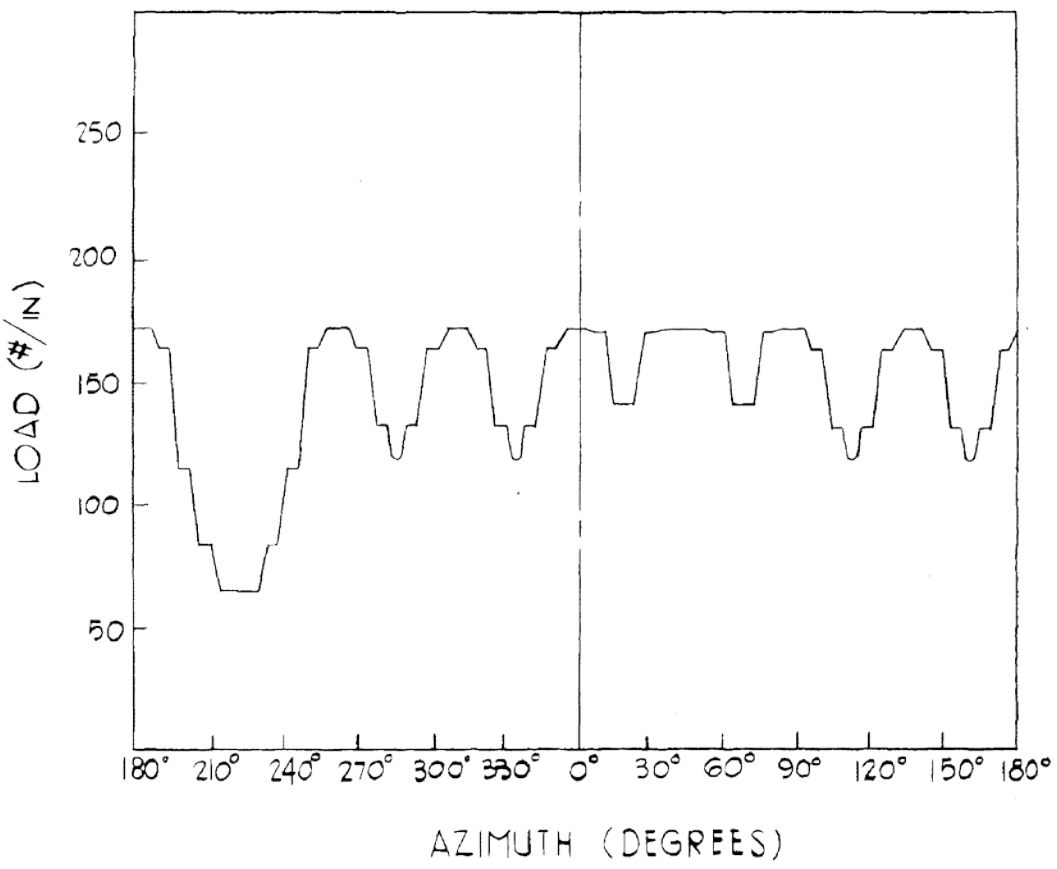


MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT

CONTAINMENT 8 ADS VALVE CASE
CIRCUMFERENTIAL PRESSURE
DISTRIBUTION AT REF PT 5
(REF APPENDICES 6A AND 6D)

FIGURE 3.8 – 40a



MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT
CONTAINMENT 8 ADS VALVE CASE CIRCUMFERENTIAL PRESSURE DISTRIBUTION AT REF PT 26 (REF APPENDICES 6A AND 6D) FIGURE 3.8 – 40b

Figure 3.8-41

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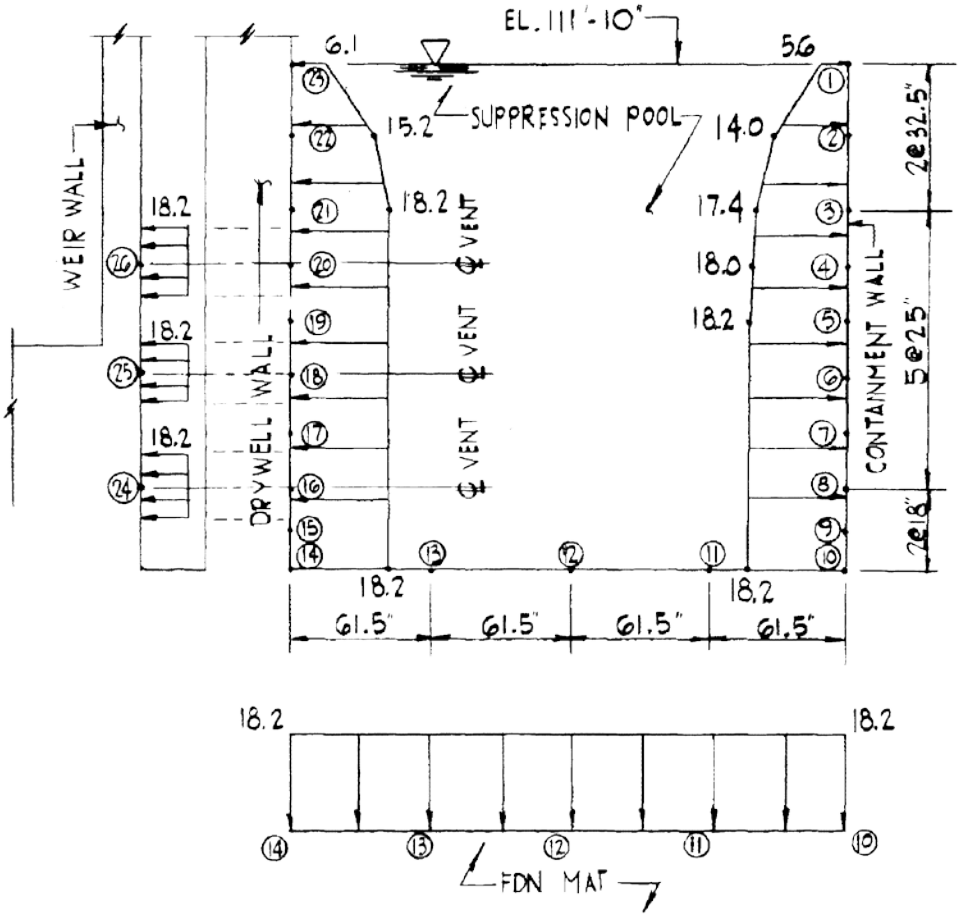

MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT
CONTAINMENT 20 VALVE CASE S/R VALVE DISCHARGE LOADS LOCAL PEAK PRESSURE, SECT AT 0° (REF APPENDICES 6A AND 6D) FIGURE 3.8 – 41a

Figure 3.8-42

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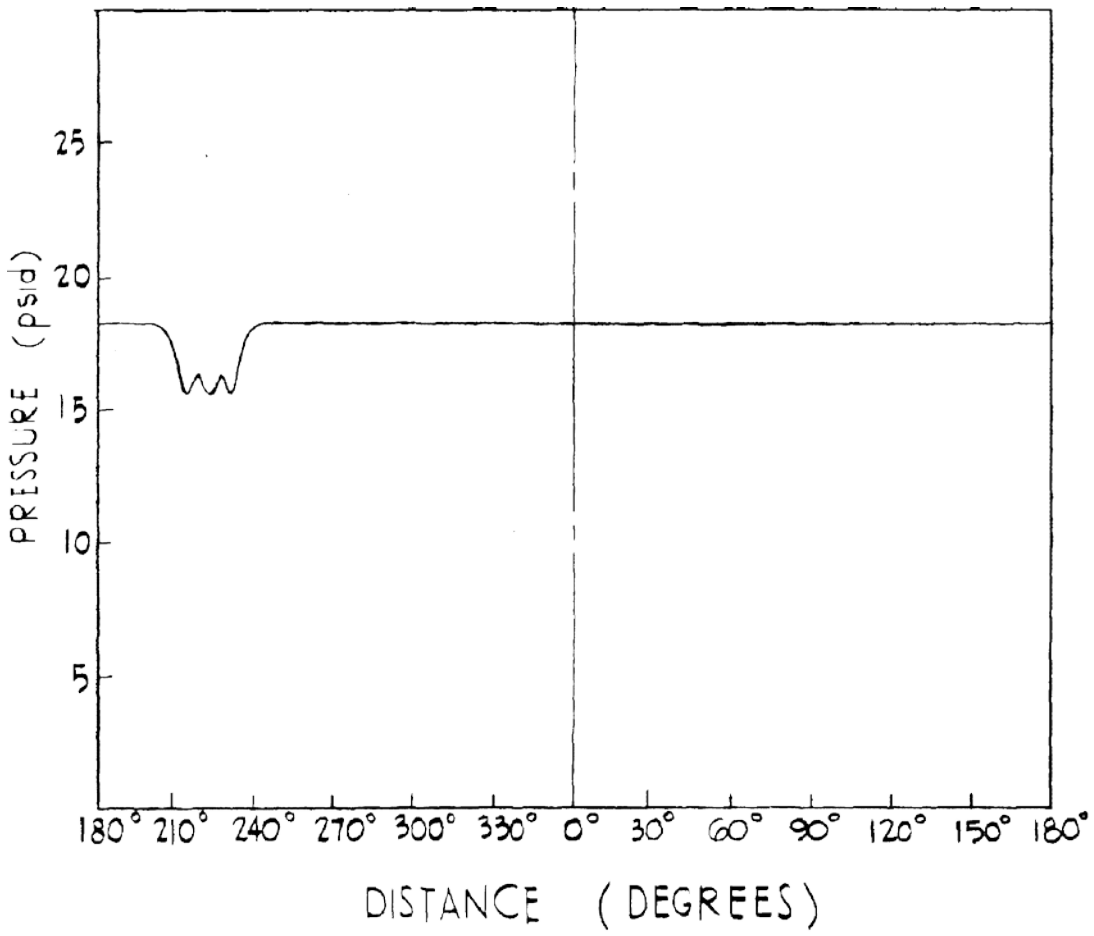

MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT

CONTAINMENT 20 S/R VALVES
CIRCUMFERENTIAL PRESSURE
DISTRIBUTION AT REF PT 21
(REF APPENDICES 6A AND 6D)
FIGURE 3.8 – 42a

Figure 3.8-43

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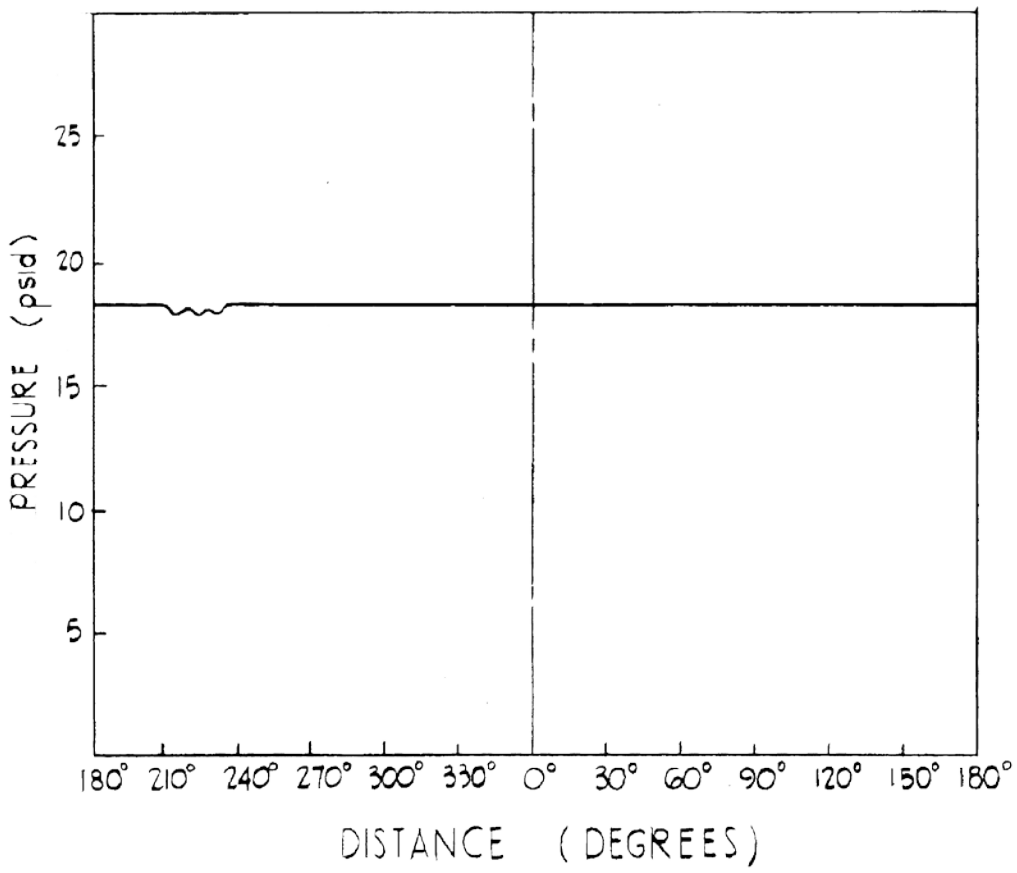

MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT

CONTAINMENT 20 S/R VALVES
CIRCUMFERENTIAL PRESSURE
DISTRIBUTION AT REF PT 17
(REF APPENDICES 6A AND 6D)
FIGURE 3.8 – 43a

Figure 3.8-44

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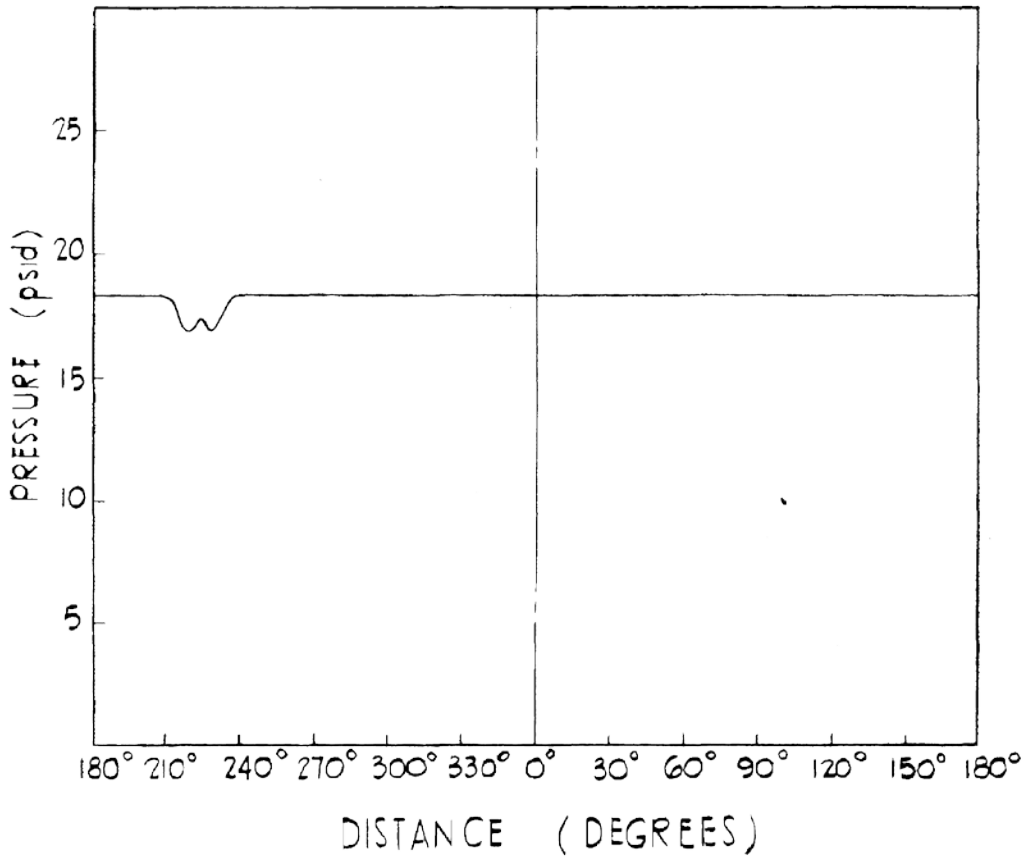

MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT

CONTAINMENT 20 S/R VALVES
CIRCUMFERENTIAL PRESSURE
DISTRIBUTION AT REF PT 12
(REF APPENDICES 6A AND 6D)
FIGURE 3.8 – 44a

Figure 3.8-45

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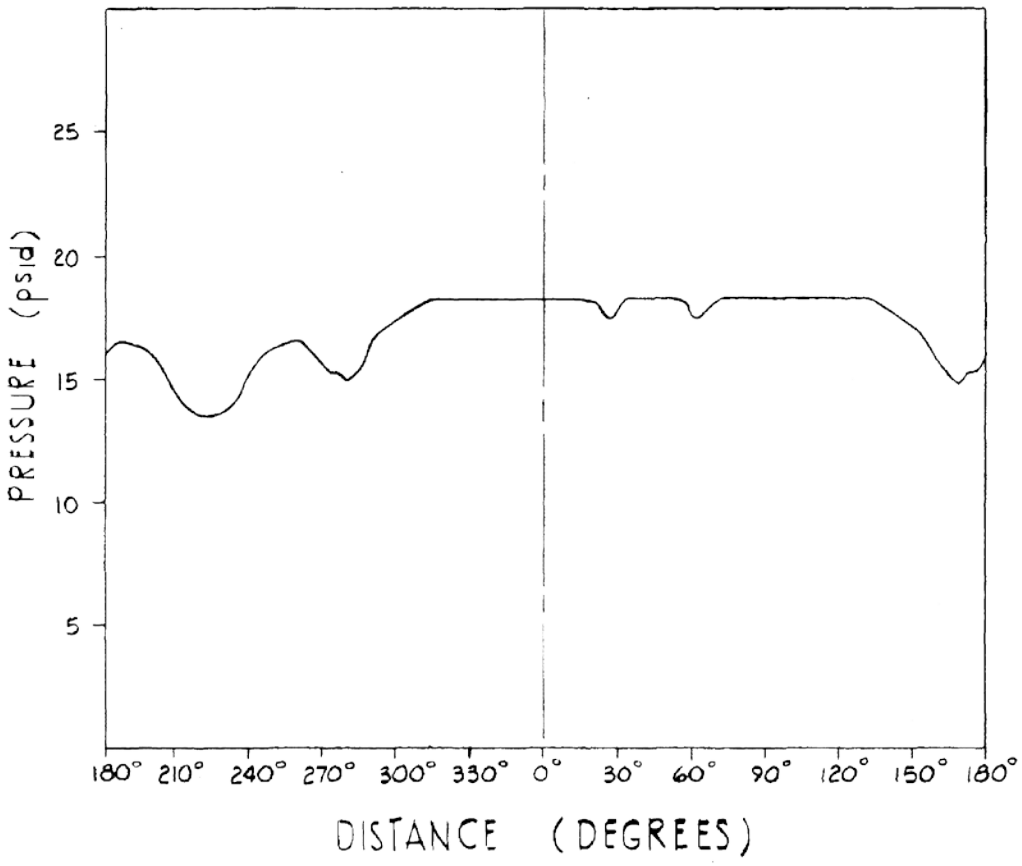
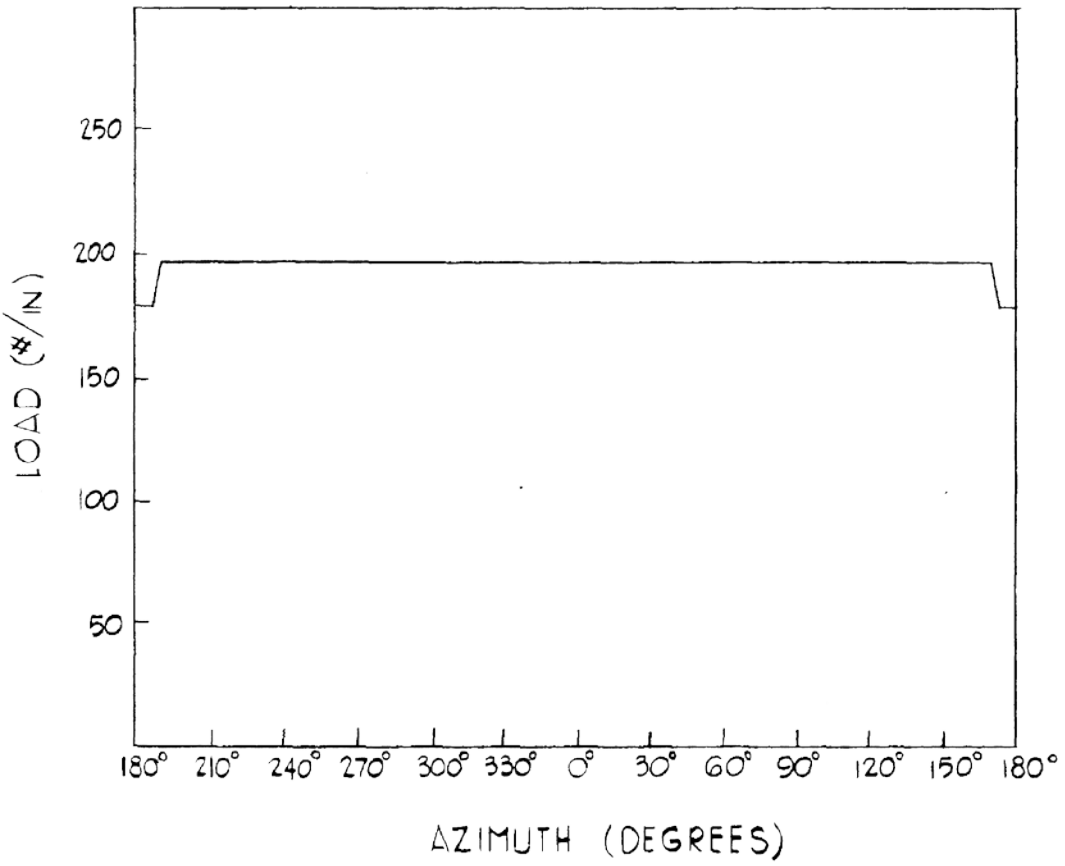


MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT

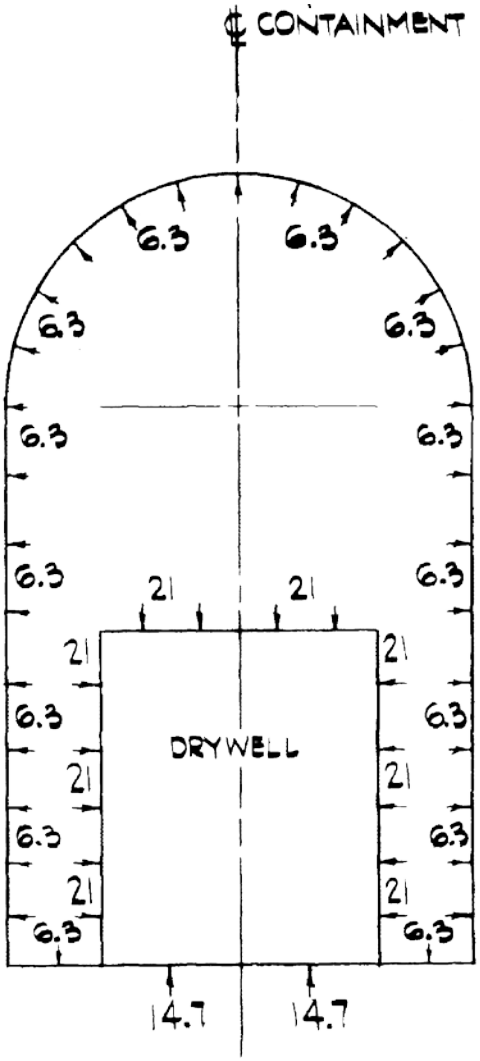
CONTAINMENT 20 S/R VALVES
CIRCUMFERENTIAL PRESSURE
DISTRIBUTION AT REF PT 5
(REF APPENDICES 6A AND 6D)
FIGURE 3.8 – 45a



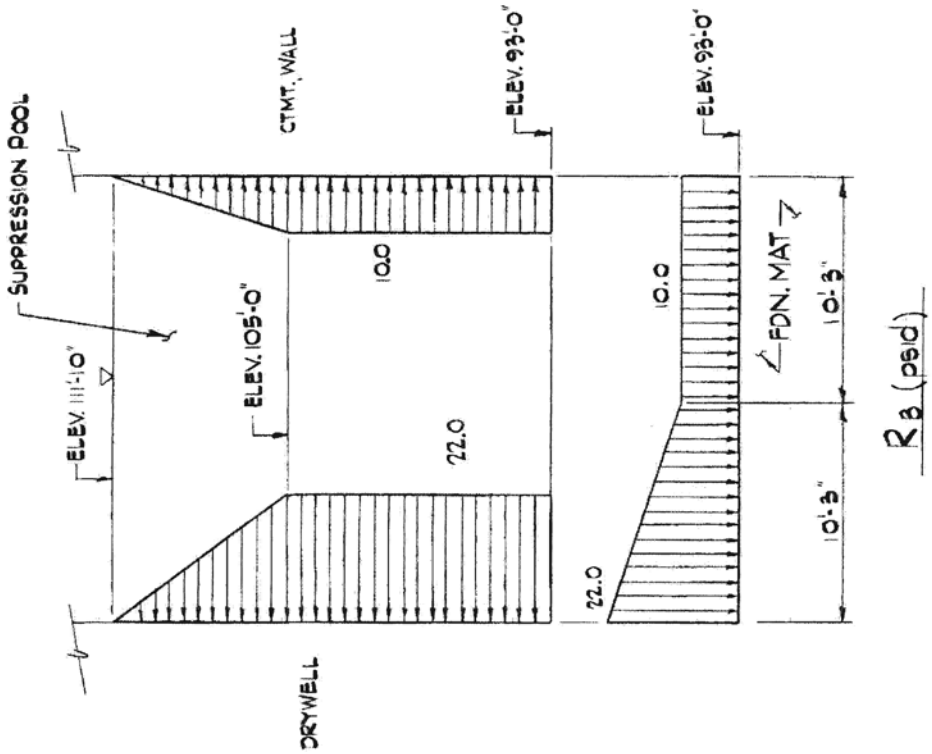
MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT

CONTAINMENT 20 S/R VALVES
CIRCUMFERENTIAL PRESSURE
DISTRIBUTION AT REF PT 26
(REF APPENDICES 6A AND 6D)

FIGURE 3.8 – 45b

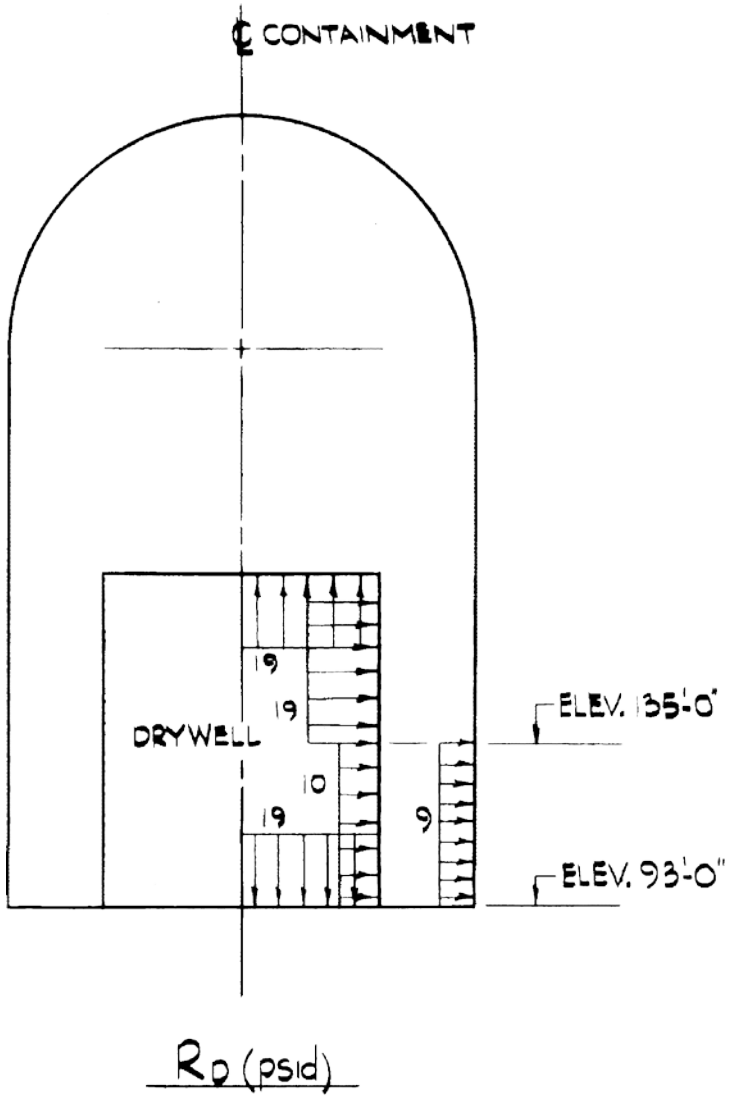


MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT
CONTAINMENT PEAK DRYWELL DESIGN NEGATIVE PRESSURE (P_{ND}) DESIGN BASIS ACCIDENT FIGURE 3.8-46

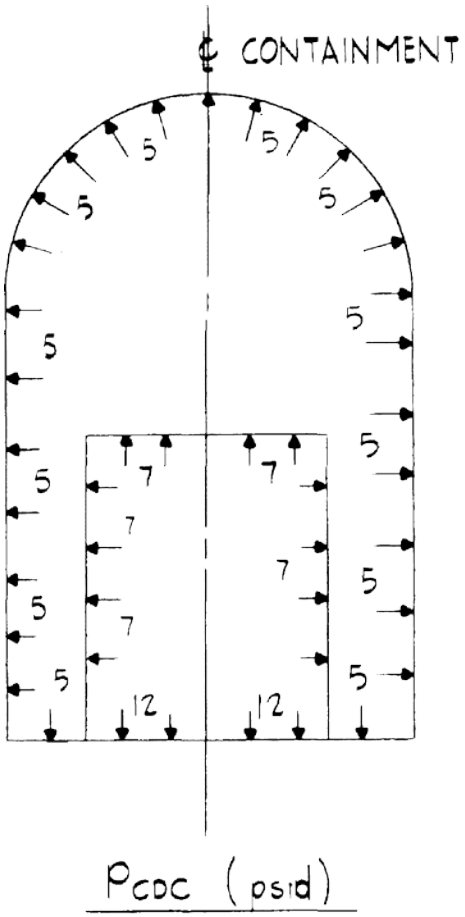


MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT

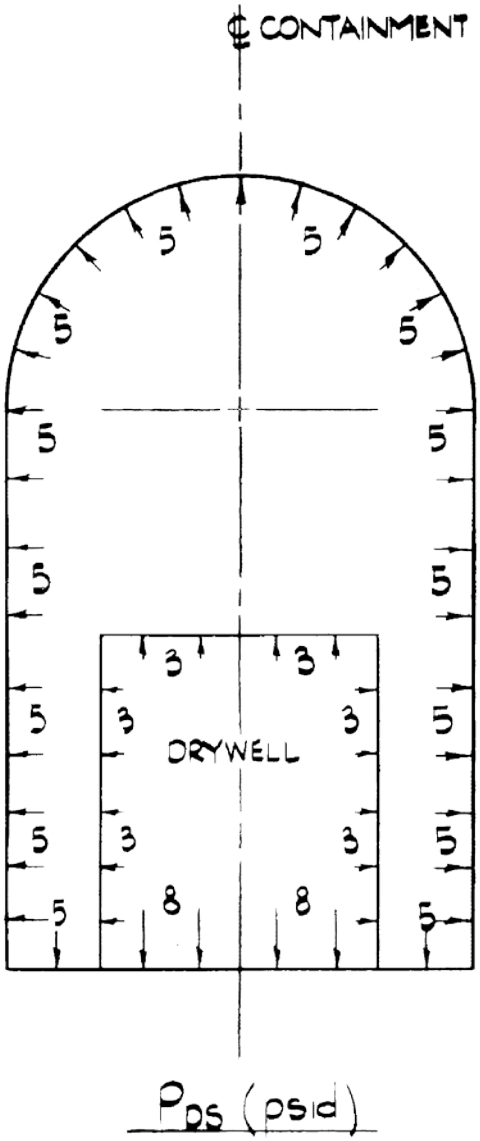
CONTAINMENT
POOL SWELL BUBBLE PRESSURE (R_B)
DESIGN BASIS ACCIDENT
FIGURE 3.8-47



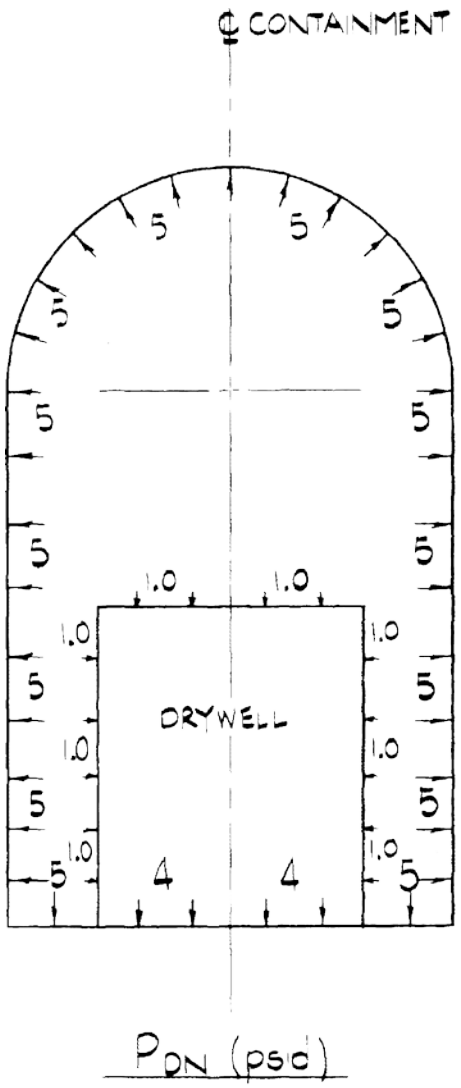
MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT
CONTAINMENT WETWELL DIFFERENTIAL PRESSURE (R_D) DESIGN BASIS ACCIDENT FIGURE 3.8-48



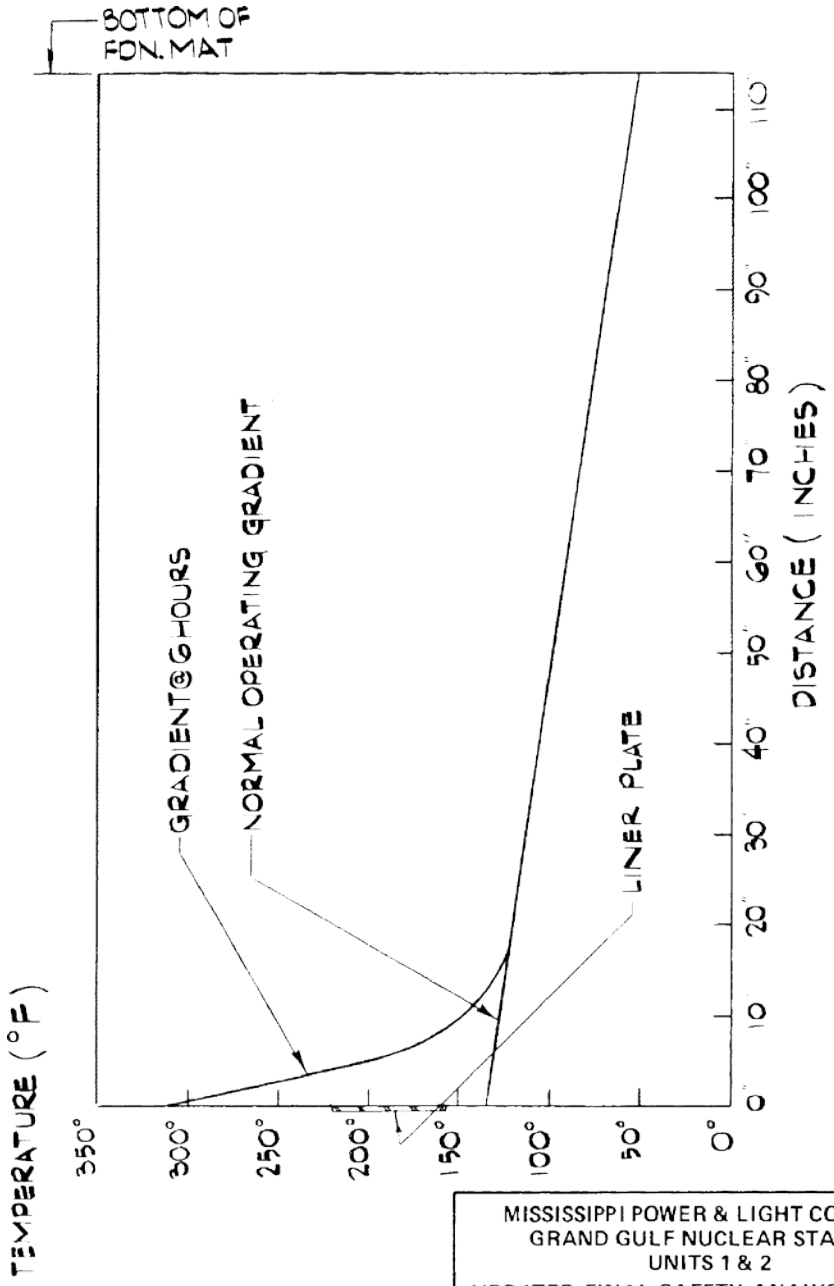
MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT
CONTAINMENT AND DRYWELL DESIGN PRESSURE (PCDC), CONCURRENT WITH CHUGGING, DESIGN BASIS ACCIDENT (REF APPENDICES 6A AND 6D) FIGURE 3.8 – 48a



MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT
CONTAINMENT AND DRYWELL DESIGN PRESSURE (P_{DS}) SMALL BREAK ACCIDENT FIGURE 3.8-49

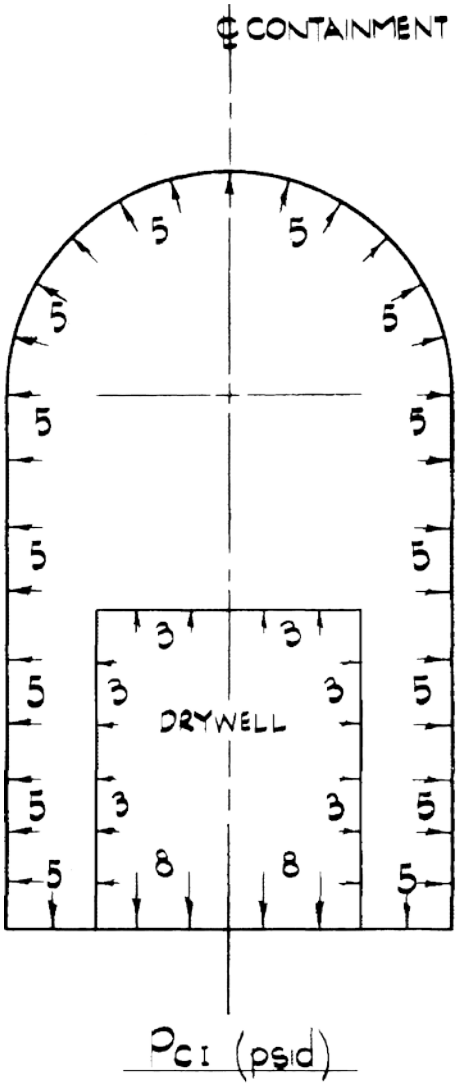


MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT
CONTAINMENT DRYWELL DESIGN NEGATIVE PRESSURE ASSOCIATED WITH CHUGGING (P_{DN}) SMALL BREAK ACCIDENT FIGURE 3.8-50



MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT

CONTAINMENT
THERMAL GRADIENT ACROSS FOUNDATION MAT BELOW REACTOR PRESSURE VESSEL
SMALL BREAK ACCIDENT
FIGURE 3.8-51



MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT
CONTAINMENT AND DRYWELL DESIGN PRESSURE (P_{CI}) INTERMEDIATE BREAK ACCIDENT FIGURE 3.8-52

Figure 3.8-53

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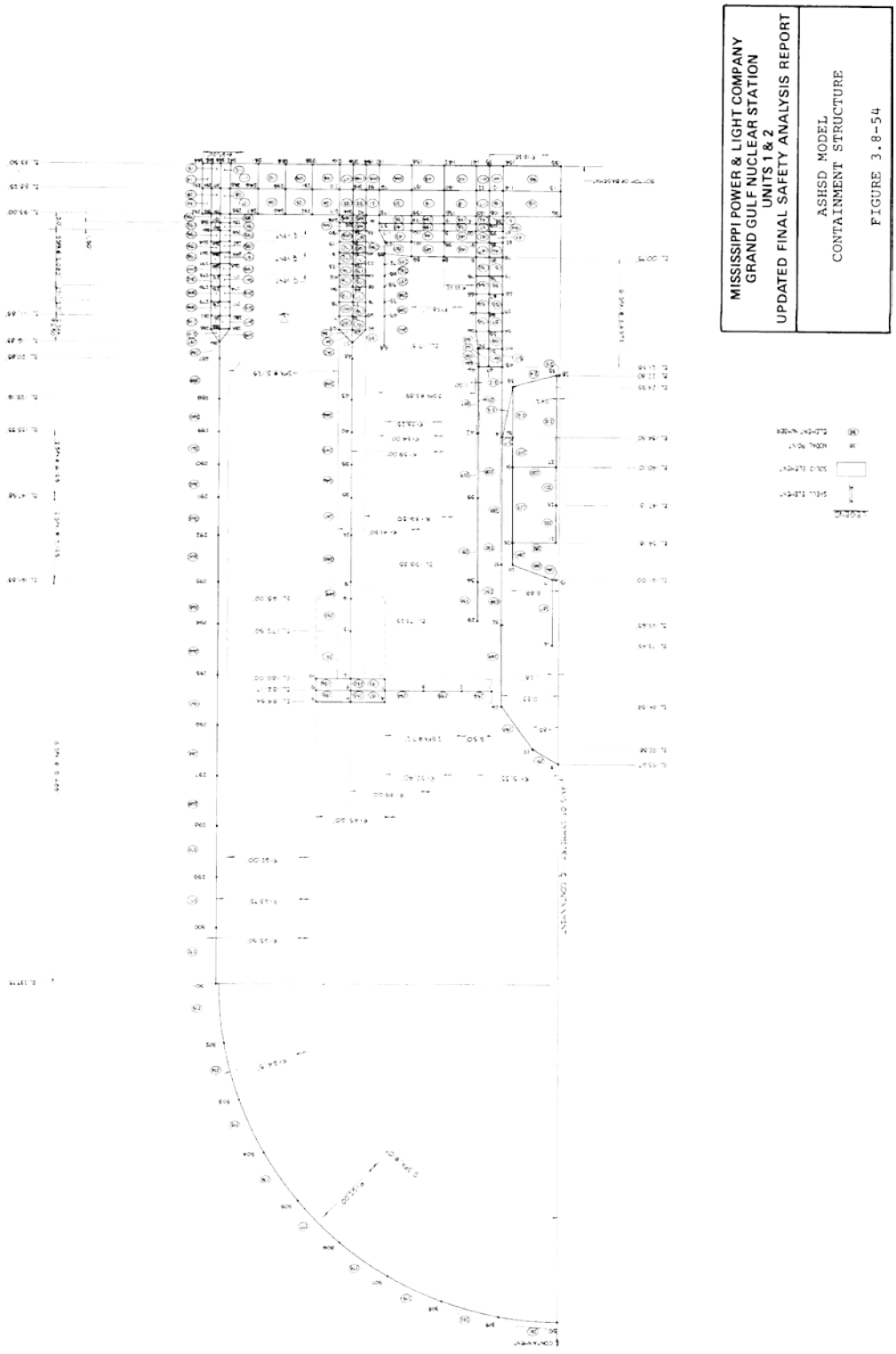
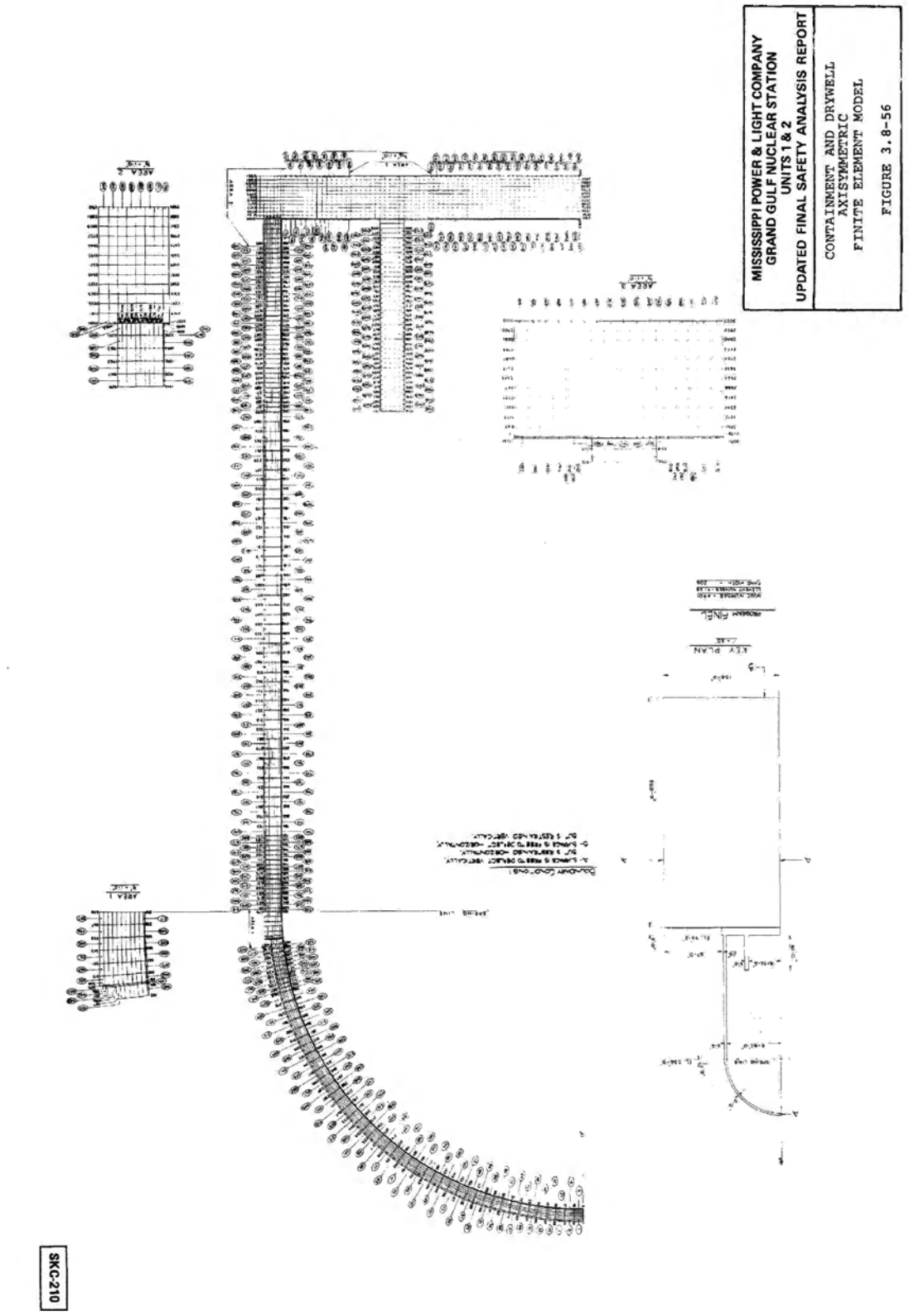
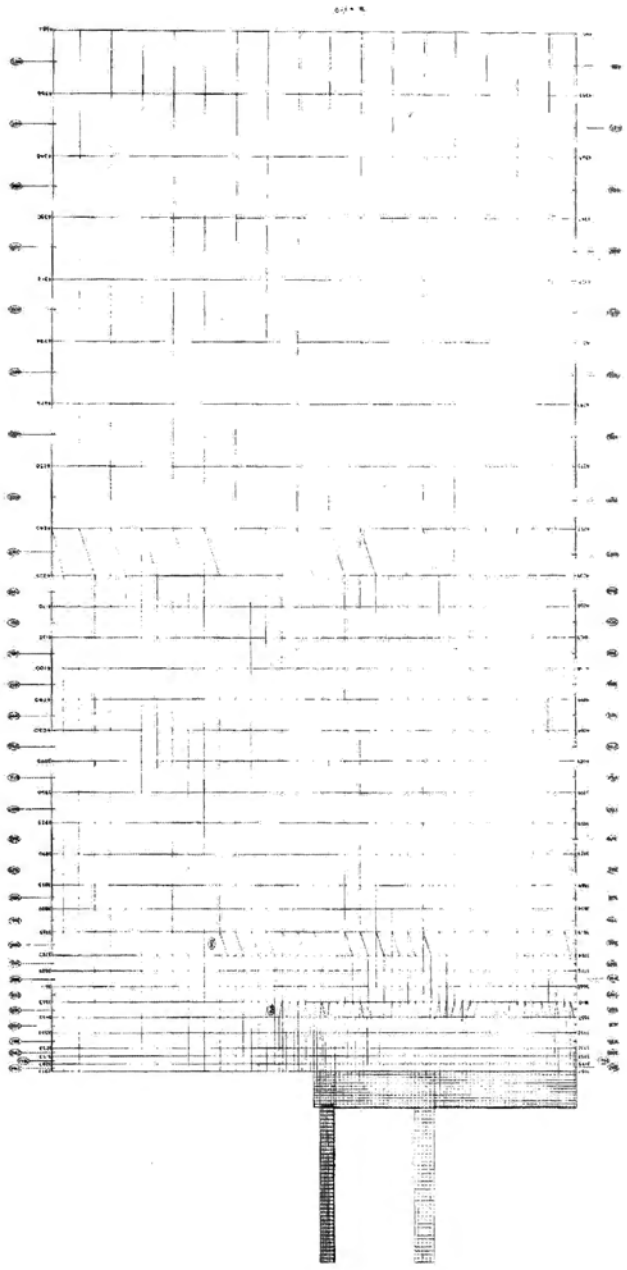

MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT
ASHSD MODEL
CONTAINMENT STRUCTURE
FIGURE 3.8-54

MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT
ASHSD MODEL
CONTAINMENT FOUNDATION
STRUCTURE
FIGURE 3.8-55
LEGEND

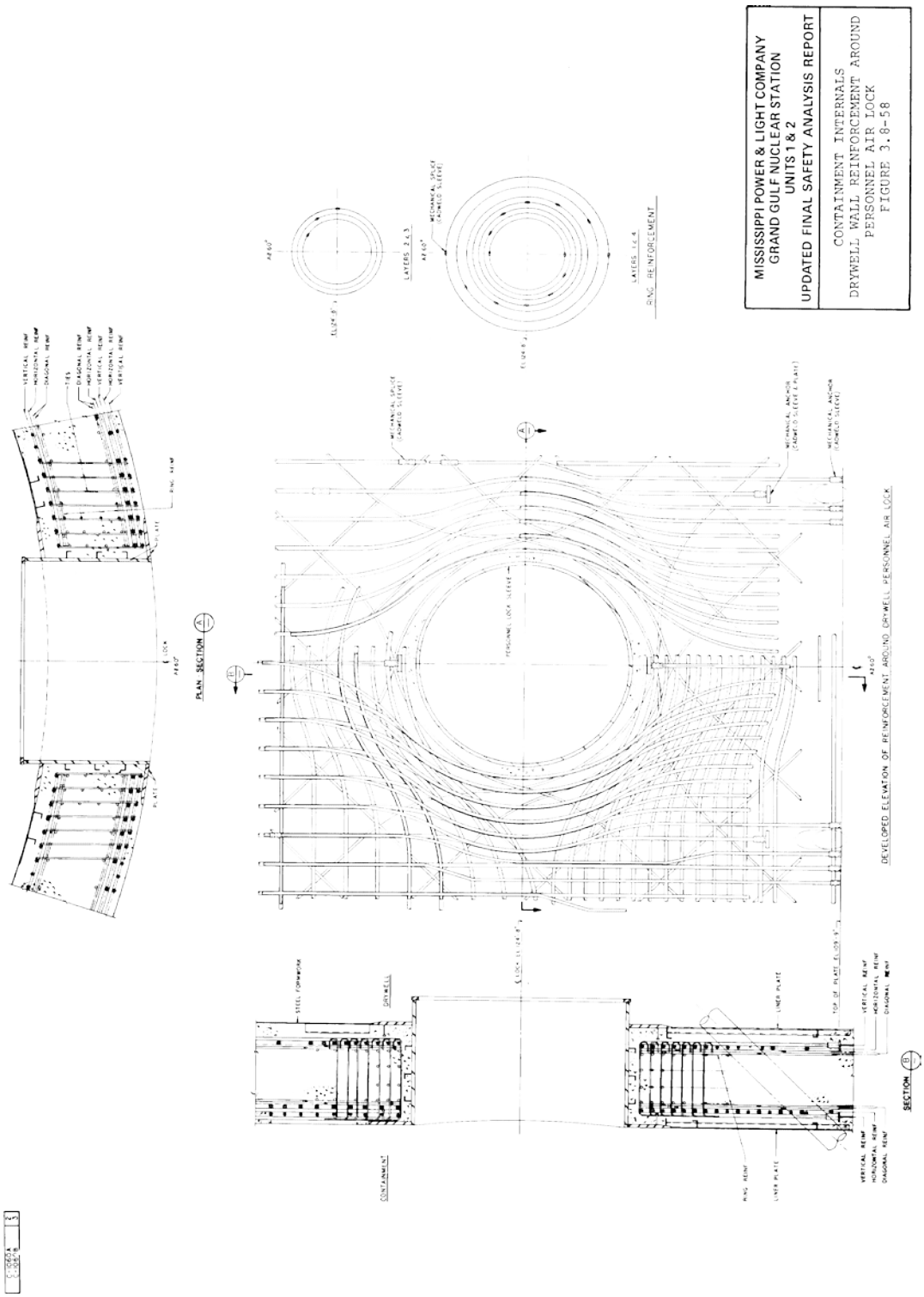

MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT
CONTAINMENT AND DRYWELL
AXISYMMETRIC
FINITE ELEMENT MODEL
FIGURE 3.8-56
SKC-210

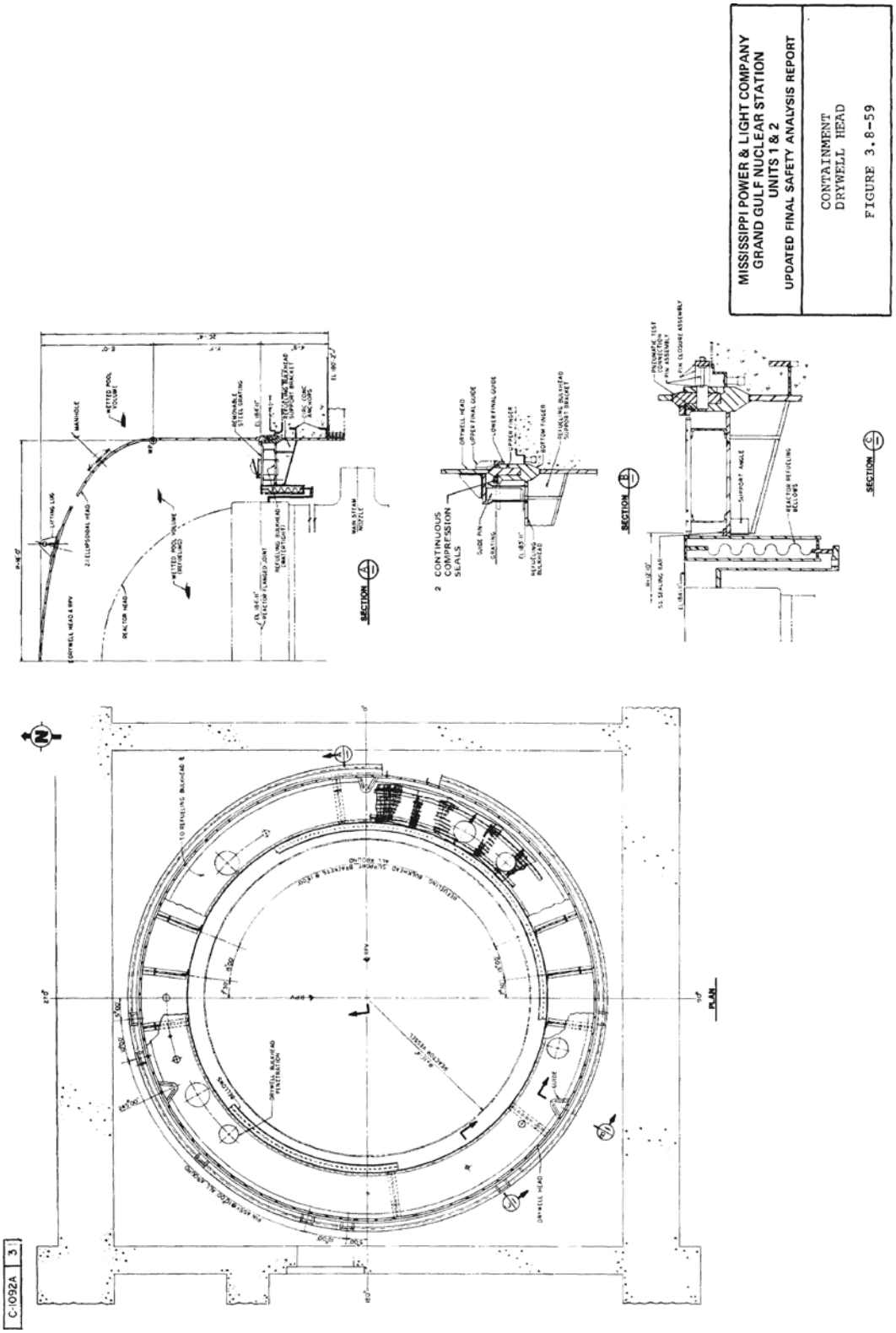


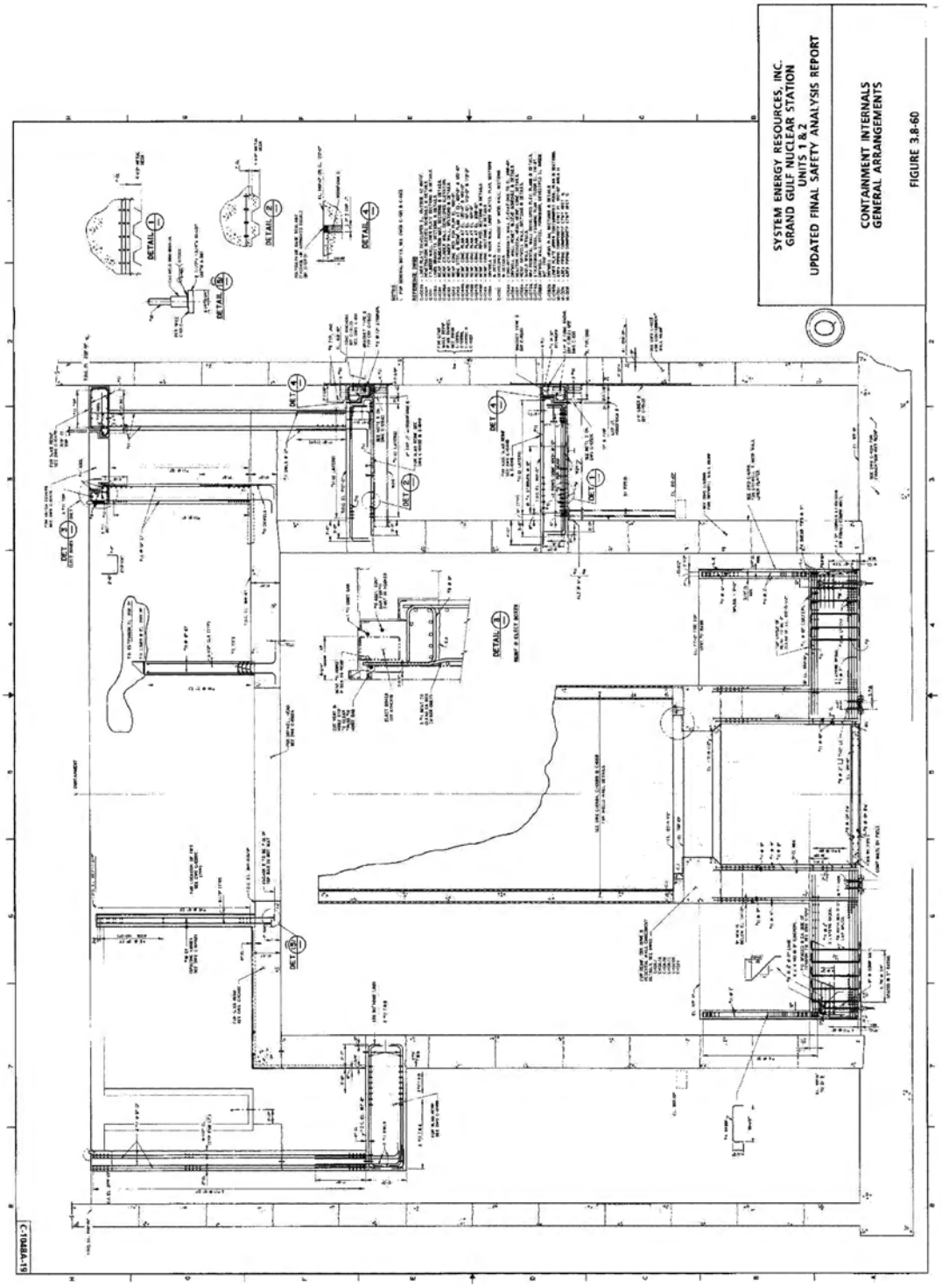
MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT

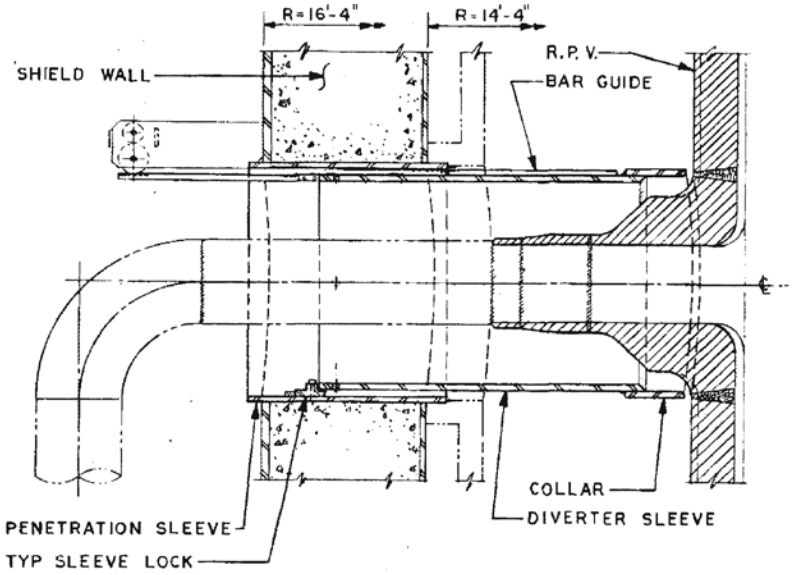
CONTAINMENT AND DRYWELL
AXISYMMETRIC
FINITE ELEMENT MODEL
FIGURE 3.8-57

SKC-211

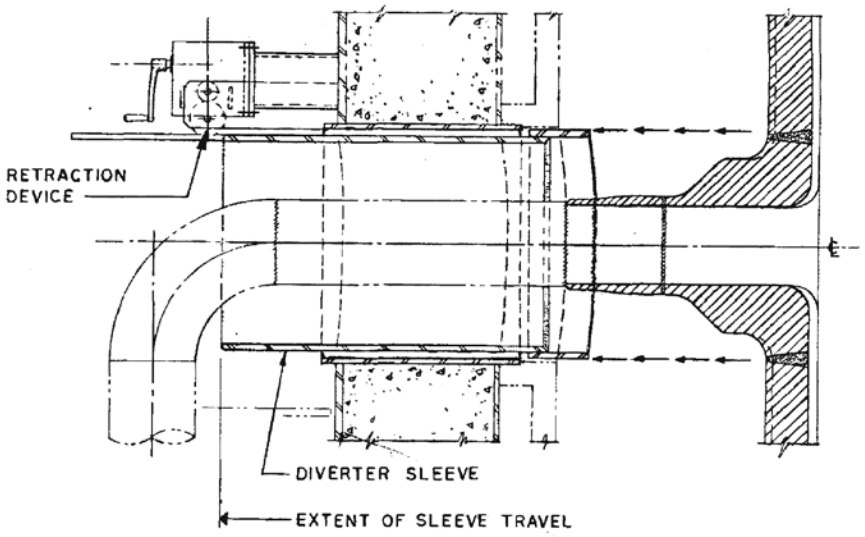

MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT
CONTAINMENT INTERNALS
DRYWELL WALL REINFORCEMENT AROUND
PERSONNEL AIR LOCK
FIGURE 3.8-58
PLAN SECTION
DEVELOPED ELEVATION OF REINFORCEMENT AROUND DRYWELL PERSONNEL AIR LOCK
SECTION


MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT
CONTAINMENT
DRYWELL HEAD
FIGURE 3.8-59
SECTION
SECTION
SECTION
PLAN
2 CONTINUOUS COMPRESSION SEALS
C-1092A 3


SYSTEM ENERGY RESOURCES, INC.
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT
CONTAINMENT INTERNALS
GENERAL ARRANGEMENTS
FIGURE 3.8-60
0

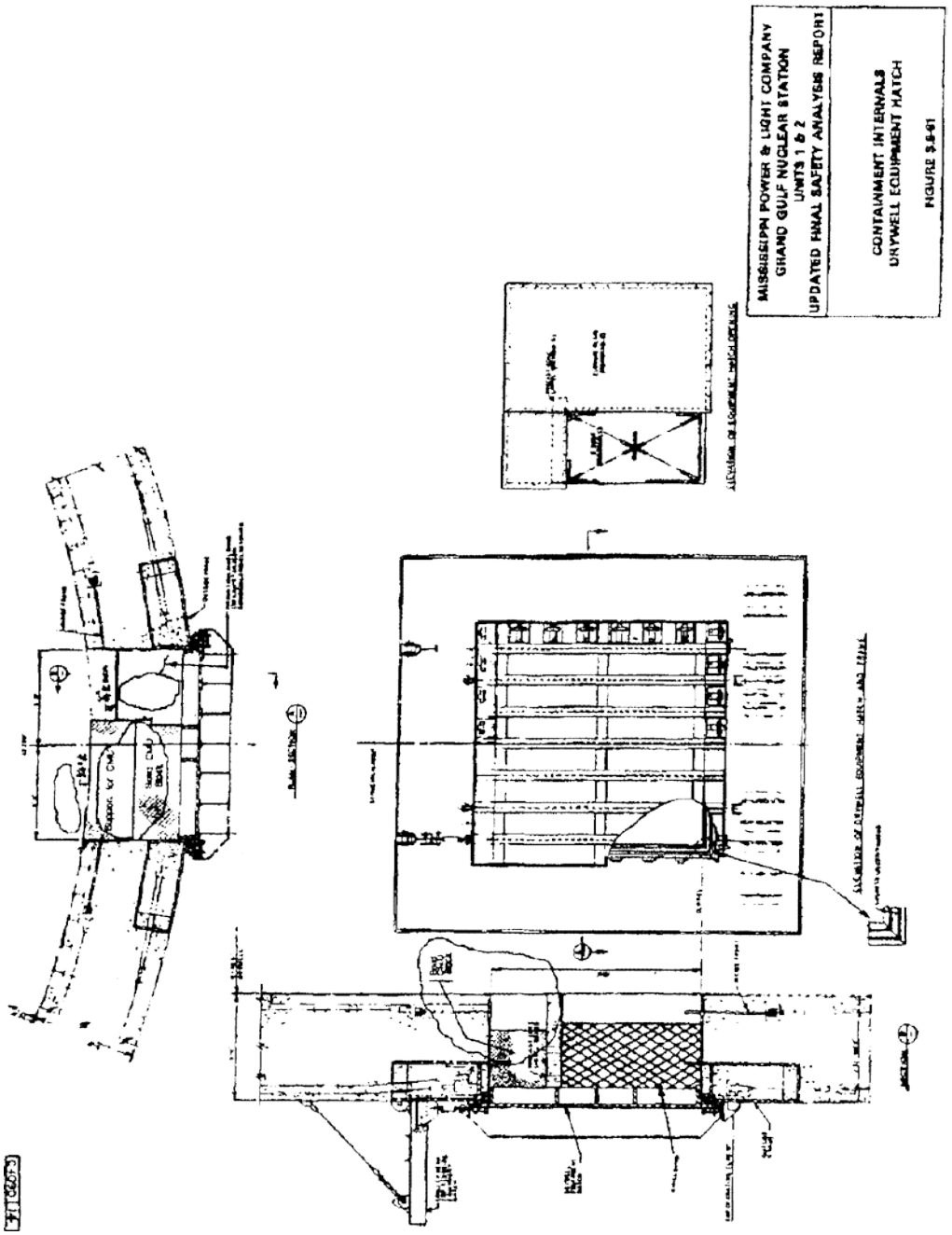


SLEEVE SHOWN IN OPERATING POSITION



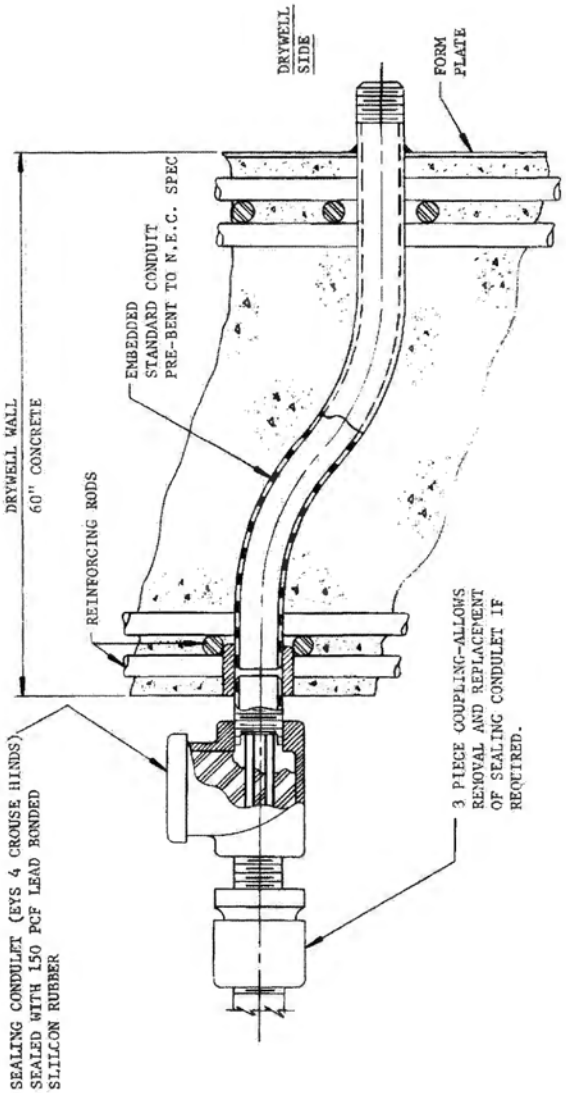
SLEEVE SHOWN IN I.S.I. POSITION

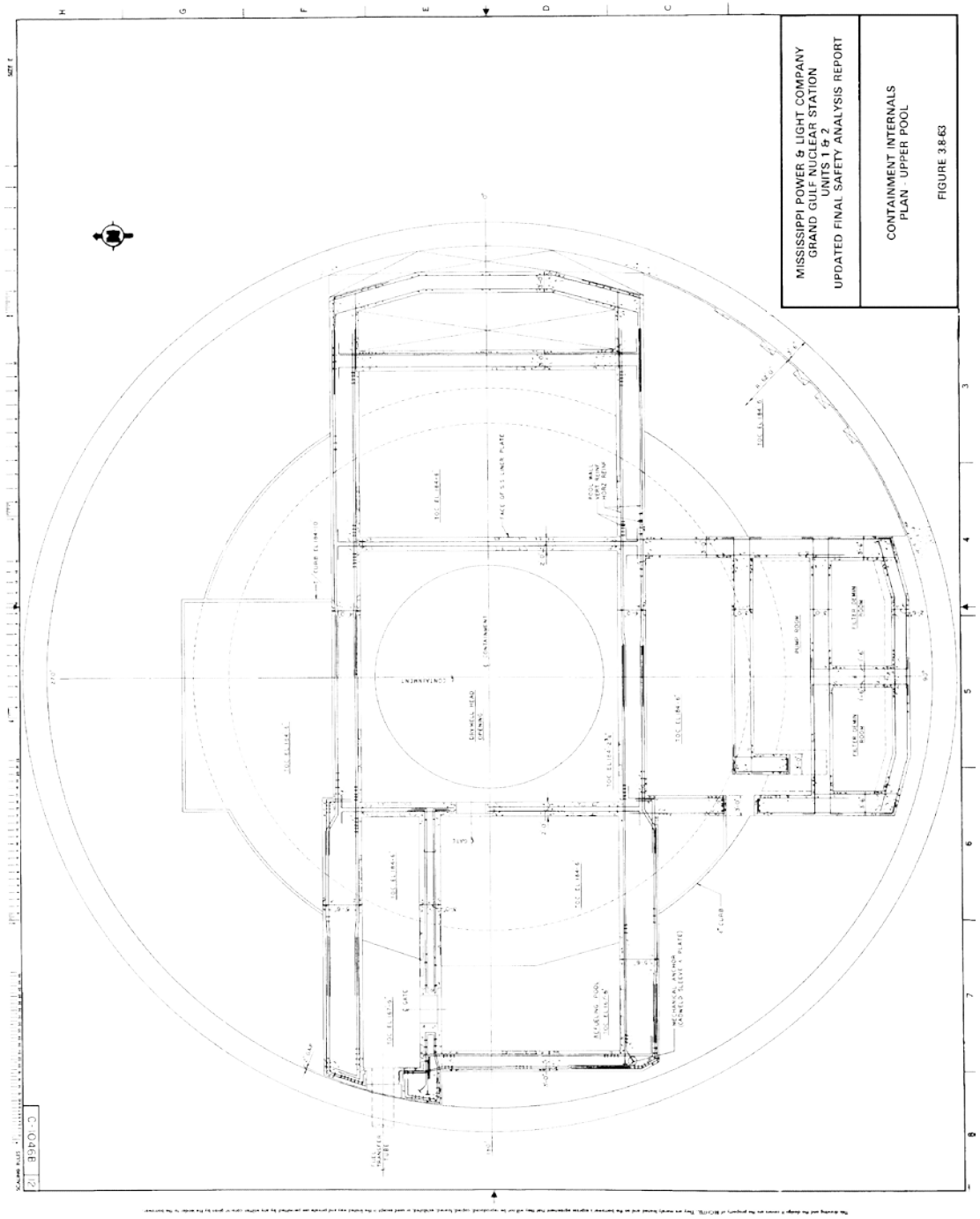
MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT	CONTAINMENT INTERNALS FEEDWATER LINE FLOW DIVERTERS AT SHIELD WALL FIGURE 3.8-60a

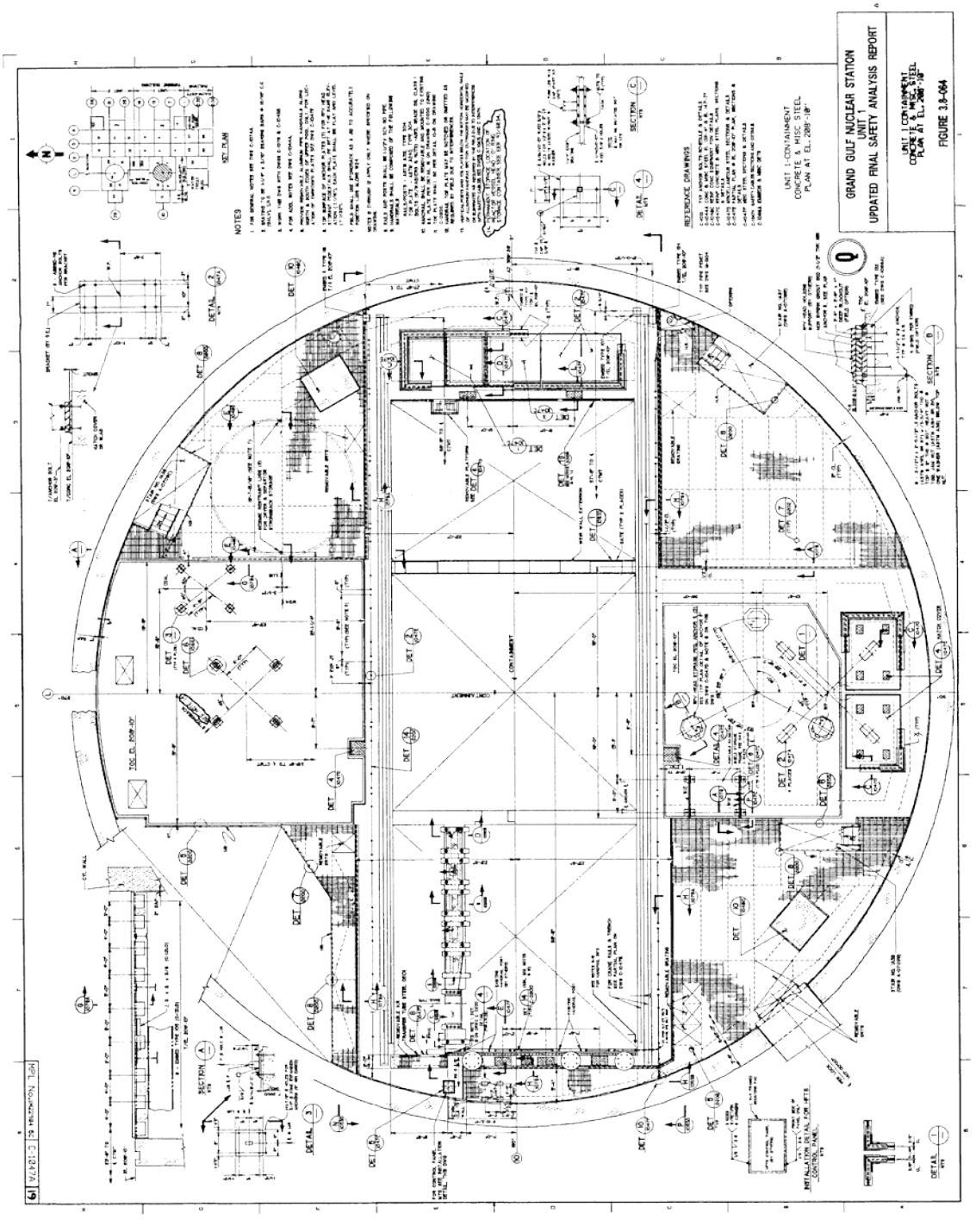

MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT
CONTAINMENT INTERNALS
UNDWELL EQUIPMENT HATCH
FIGURE 3.8-61

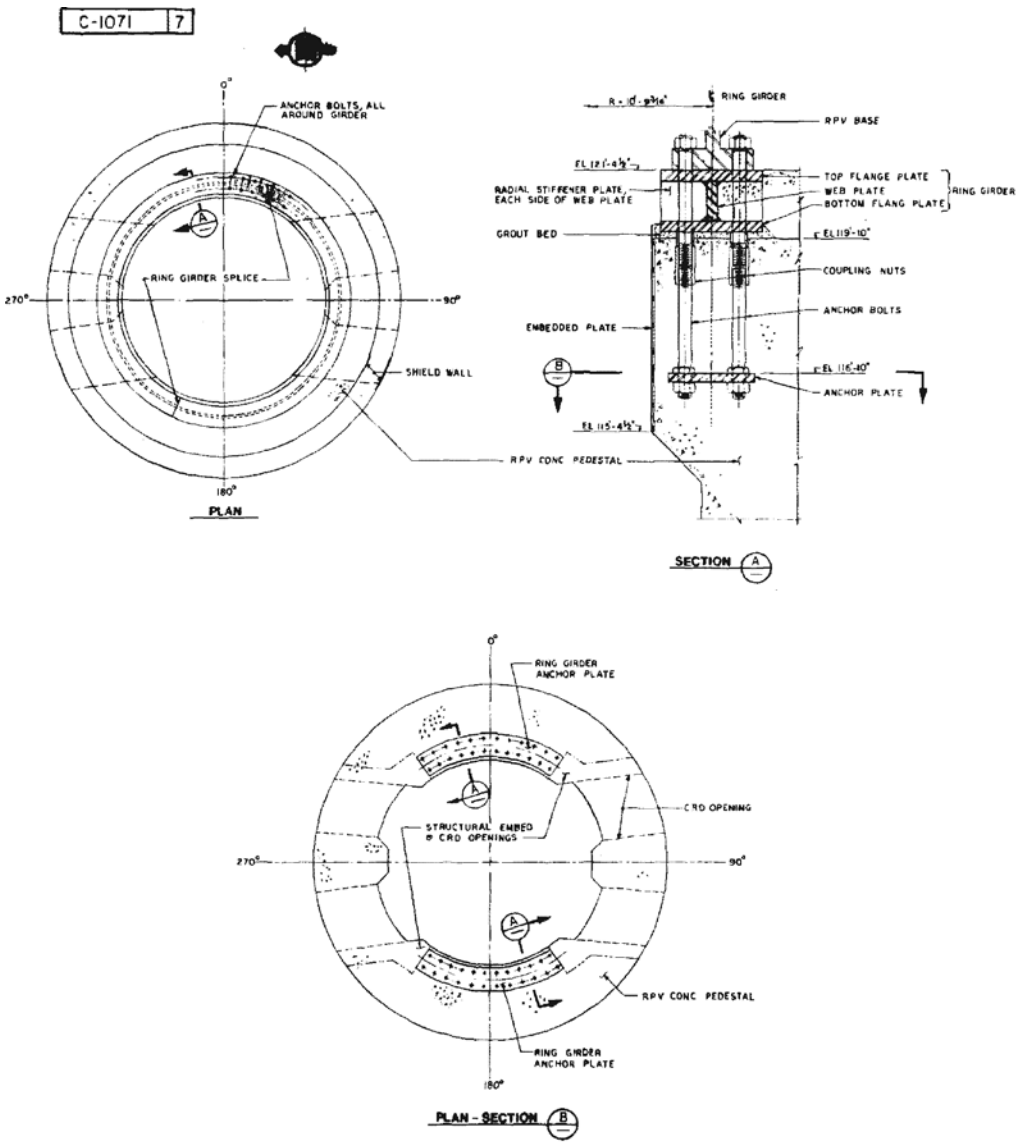
MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT

CONTAINMENT INTERNALS
TYPICAL DRYWELL
ELECTRICAL PENETRATION
FIGURE 3.8-62

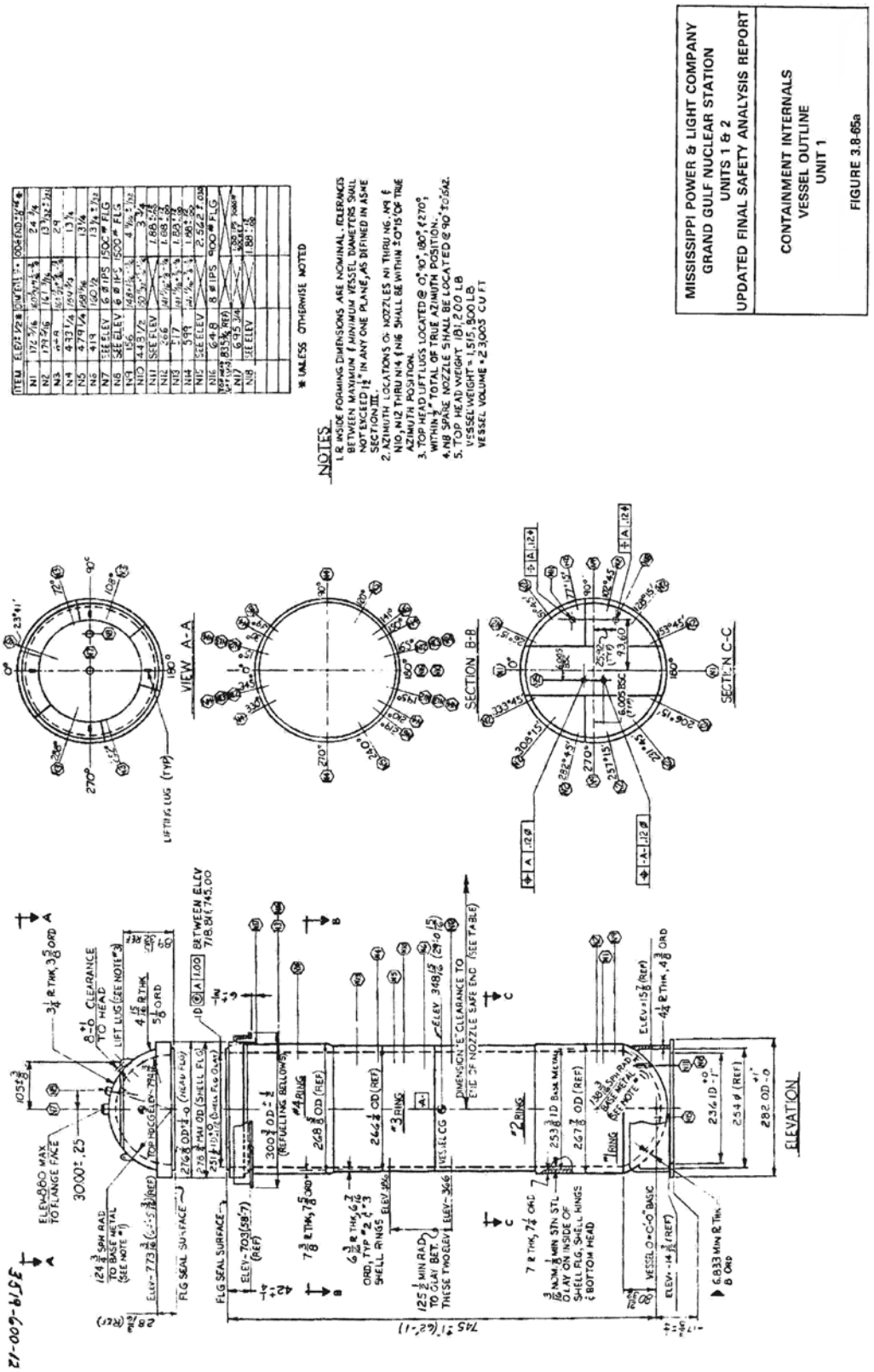


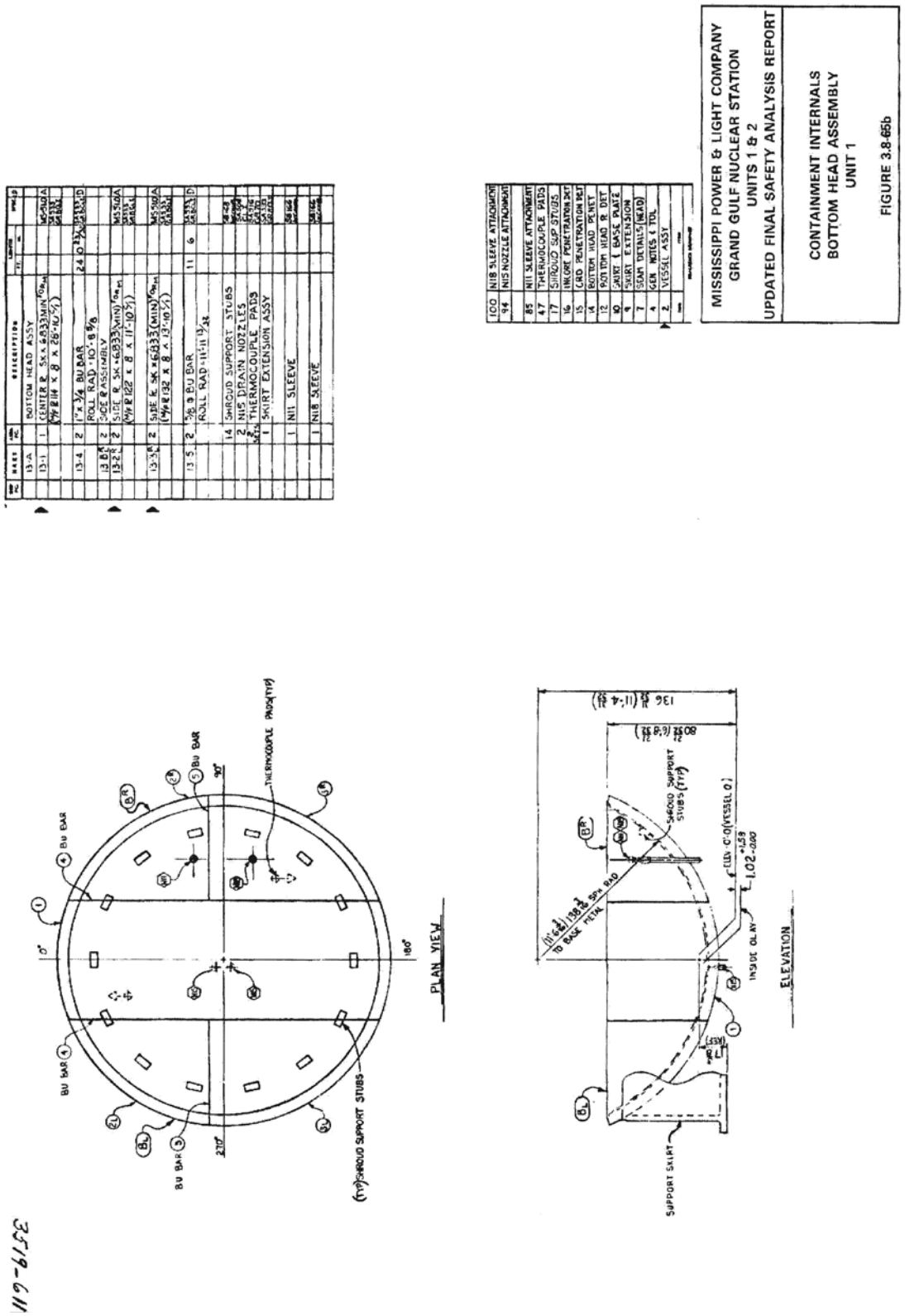

MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT
CONTAINMENT INTERNALS
PLAN UPPER POOL
FIGURE 3.8-63

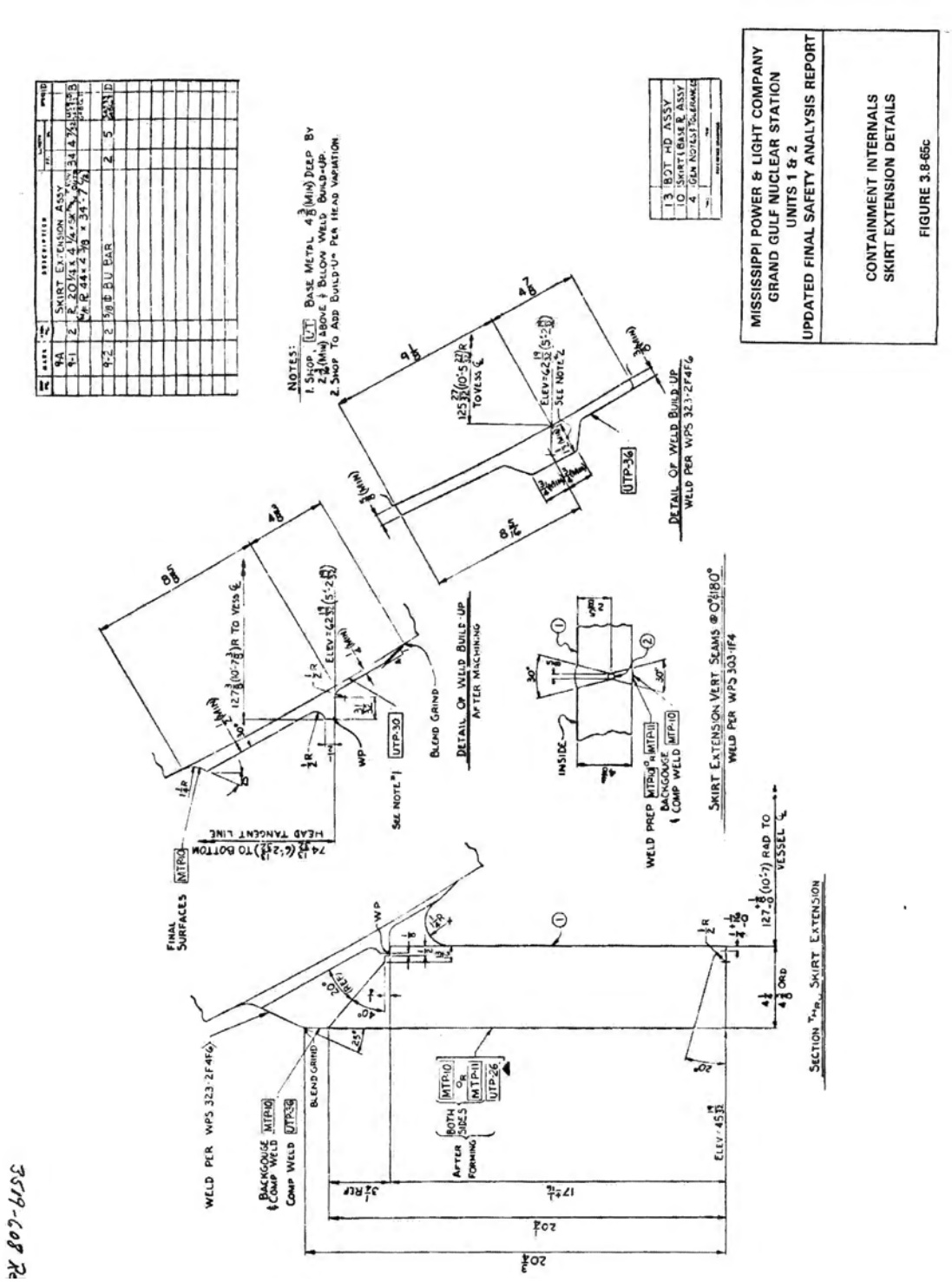

GRAND GULF NUCLEAR STATION
UNIT 1
UPDATED FINAL SAFETY ANALYSIS REPORT
FIGURE 3.8-64
UNIT 1 CONTAINMENT
CONCRETE & MISC STEEL
PLAN AT EL. 208'-10"
NOTES
REFERENCE DRAWINGS
SECTION
DETAIL

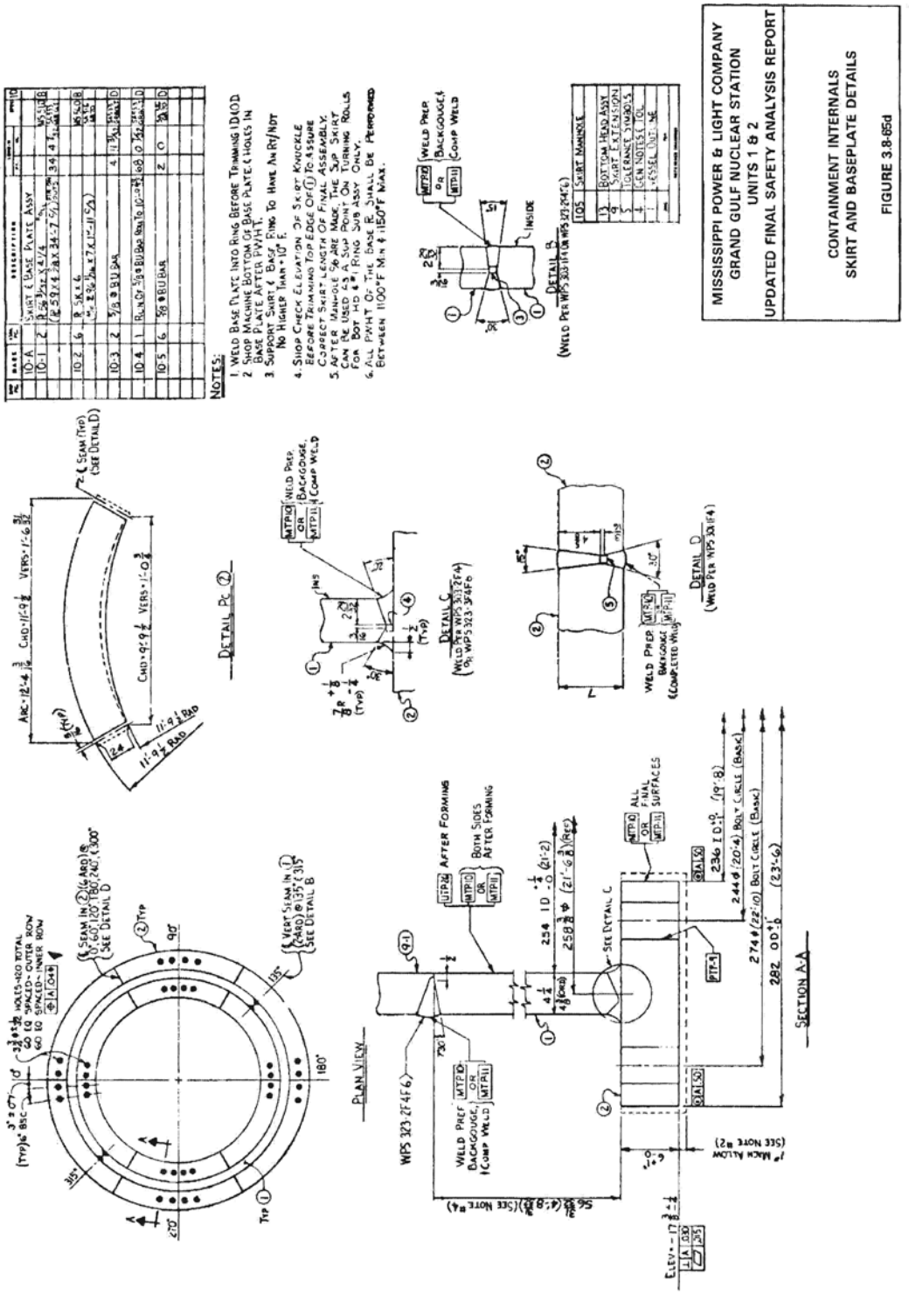


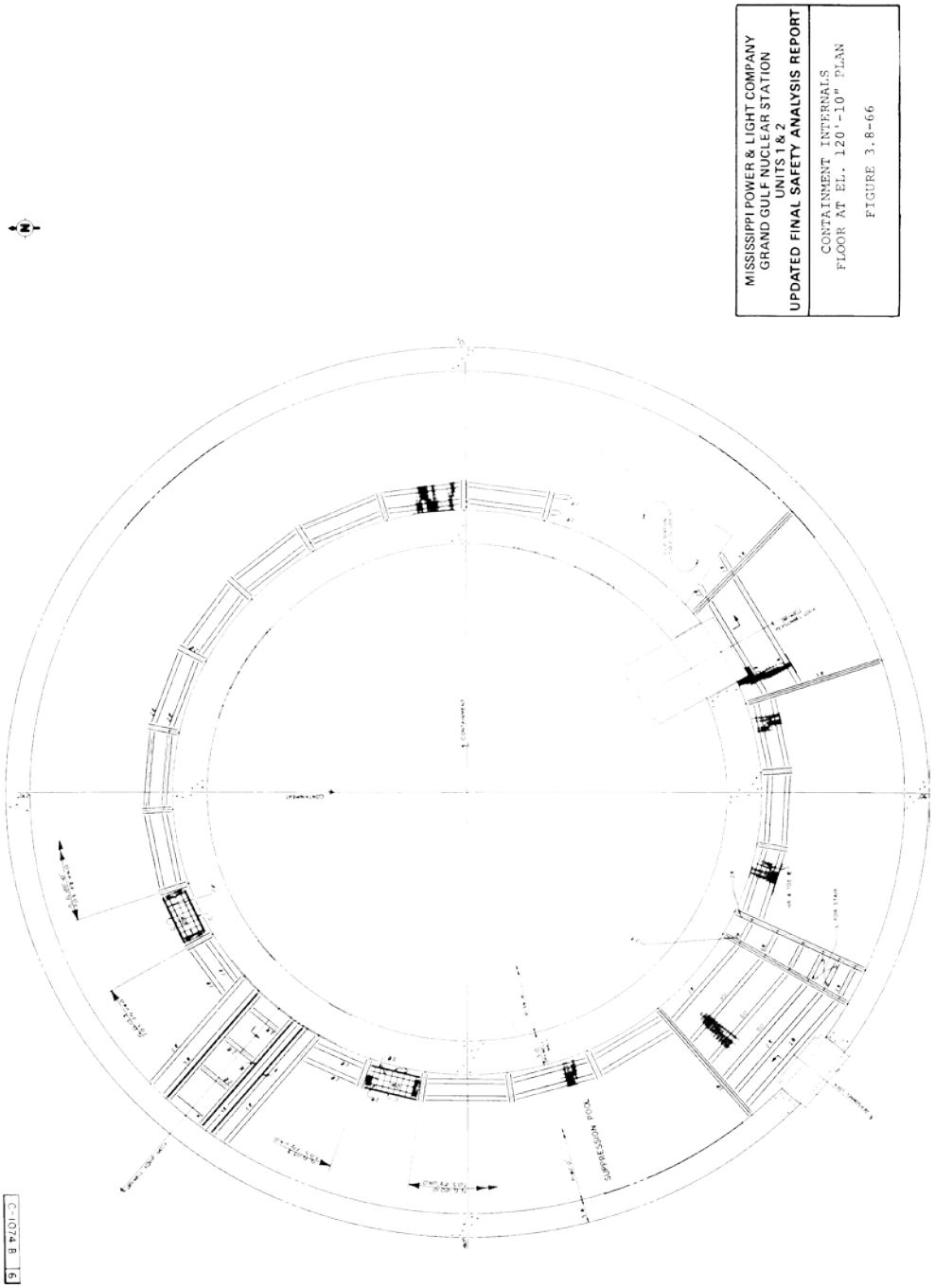
MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT
CONTAINMENT INTERNALS REACTOR SUPPORTS FIGURE 3.8-65

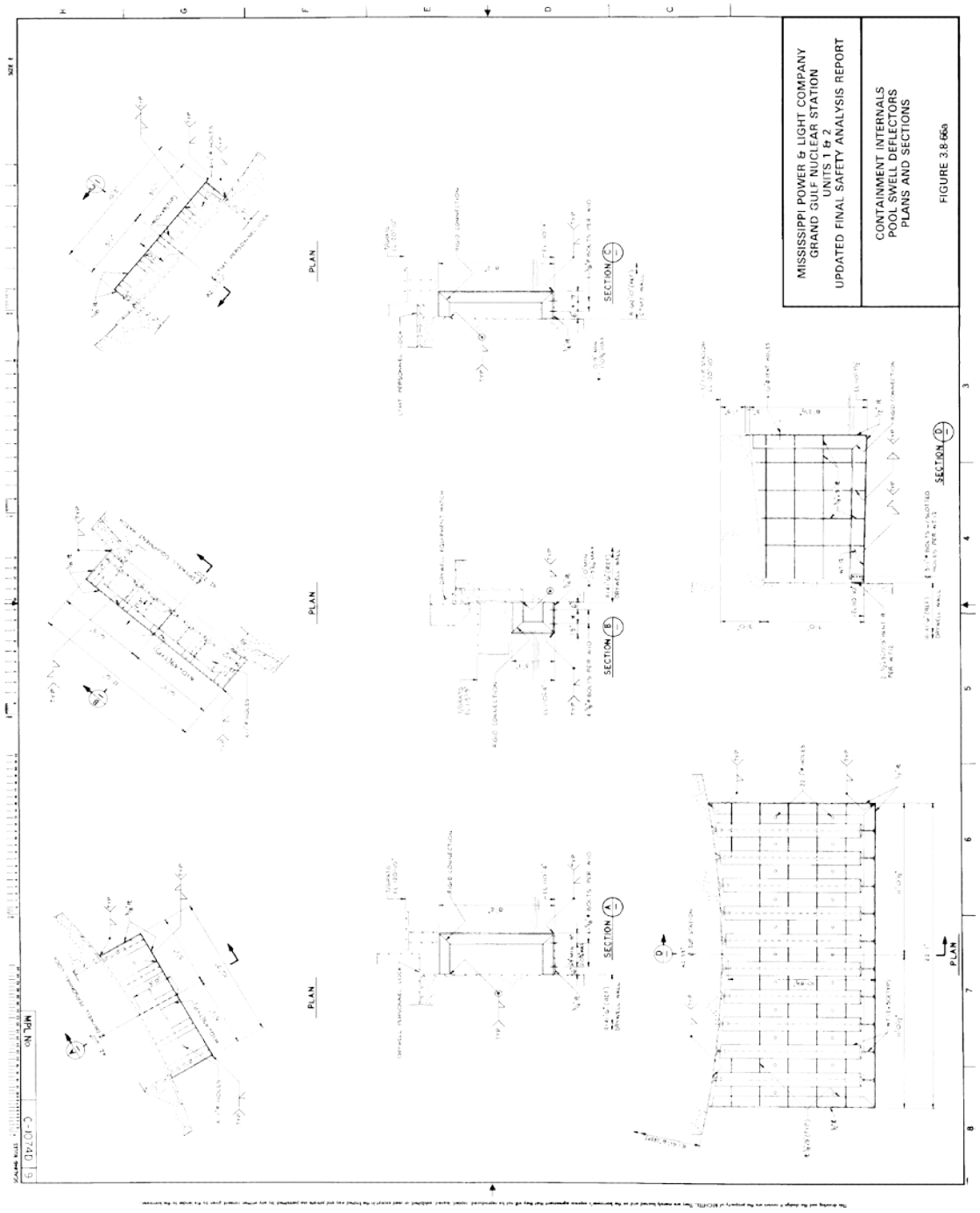

MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT
CONTAINMENT INTERNALS
VESSEL OUTLINE
UNIT 1
FIGURE 3.8-65a
NOTES
1. R INSIDE FORMING DIMENSIONS ARE NOMINAL. TOLERANCES BETWEEN MAXIMUM & MINIMUM VESSEL DIAMETERS SHALL NOT EXCEED 1½" IN ANY ONE PLANE, AS DEFINED IN ASME SECTION III.
2. AZIMUTH LOCATIONS OF NOZZLES N1 THRU N6, N9 & N10, N12 THRU N14 & N16 SHALL BE WITHIN ±0°15' OF TRUE AZIMUTH POSITION.
3. TOP HEAD LIFT LUGS LOCATED @ 0°, 90°, 180°, & 270°, WITHIN ½" TOTAL OF TRUE AZIMUTH POSITION.
4. N8 SPARE NOZZLE SHALL BE LOCATED @ 90° TO N12.
5. TOP HEAD WEIGHT 181,200 LB
VESSEL WEIGHT = 1,515,900 LB
VESSEL VOLUME = 23005 CU FT
* UNLESS OTHERWISE NOTED
ITEM ELEV'S & CENTERLINES* ID & END SIZE*
N1 172 5/16 24 3/4
N2 179 5/16 13 3/32 I.D.
N3 ---- 24
N4 433 1/4 13 3/4
N5 479 1/4 13 3/4
N6 419 13 3/4 I.D.
N7 SEE ELEV 6 @ 11½" 150# RF FLG
N8 SEE ELEV 6 @ 11½" 1500# RF FLG
N9 446 4 3/4 I.D.
N10 443 1/2 3 3/4
N11 SEE ELEV 1.88 I.D.
N12 466 1.88 I.D.
N13 517 1.88 I.D.
N14 599 1.00 I.D.
N16 SEE ELEV 8 @ 11½" 900# RF FLG
N17 SEE ELEV 6 @ 11½" 600# RF FLG
N18 SEE ELEV 1.88 I.D.
VIEW A-A
SECTION B-B
SECTION C-C
ELEVATION
LIFTING LUG (TYP)
FLG SEAL SURFACE
ELEV 703 (58'7") (REF)
ELEV 745'11" (62'-1")
BETWEEN ELEV 718.84 & 745.00
ELEV 348 5/16 (29'-0 5/16")
ELEV 366
ELEV 15 5/8 (REF)
ELEV -14 3/8 (REF)
VESSEL O-C-O BASIC
DIMENSION 'E' CLEARANCE TO END OF NOZZLE SAFE END (SEE TABLE)
8'-0" CLEARANCE TO HEAD
ELEV 860 MAX TO FLANGE FACE
30'00".25
LIFT LUG (SEE NOTE #3)
TOP HEAD
SHELL FLG
BOTTOM HEAD
SHELL RINGS
REFUELING BELOWS
#1 RING
#2 RING
#3 RING
#4 RING
#5 RING
#6 RING
#7 RING
3 NOM. ¾ MIN STN STL OLAY ON INSIDE OF SHELL FLG, SHELL RINGS & BOTTOM HEAD
6.833 MIN R THK @ ORD
3 R THK, 3⅜ ORD
4 15/16 R THK
5⅜ ORD
7 7/8 R THK, 7 3/8 ORD
6 3/8 R THK, 6 7/8 ORD, TYP #2 & #3 SHELL RINGS
7 R THK, 7½ ORD
4¼ R THK, 4⅜ ORD
124 3/4 SPH RAD TO BASE METAL (SEE NOTE #)
125 1/2 MIN RAD TO OLAY BET. THESE TWO ELEV
ELEV 860
ELEV 745'11" (62'-1")
ELEV 703 (58'7") (REF)
ELEV 348 5/16 (29'-0 5/16")
ELEV 366
ELEV 15 5/8 (REF)
ELEV -14 3/8 (REF)
ELEV 860 MAX TO FLANGE FACE
30'00".25
8'-0" CLEARANCE TO HEAD
LIFT LUG (SEE NOTE #3)
TOP HEAD
SHELL FLG
BOTTOM HEAD
SHELL RINGS
REFUELING BELOWS
#1 RING
#2 RING
#3 RING
#4 RING
#5 RING
#6 RING
#7 RING
3 NOM. ¾ MIN STN STL OLAY ON INSIDE OF SHELL FLG, SHELL RINGS & BOTTOM HEAD
6.833 MIN R THK @ ORD
3 R THK, 3⅜ ORD
4 15/16 R THK
5⅜ ORD
7 7/8 R THK, 7 3/8 ORD
6 3/8 R THK, 6 7/8 ORD, TYP #2 & #3 SHELL RINGS
7 R THK, 7½ ORD
4¼ R THK, 4⅜ ORD
124 3/4 SPH RAD TO BASE METAL (SEE NOTE #)
125 1/2 MIN RAD TO OLAY BET. THESE TWO ELEV
FLG SEAL SURFACE
BETWEEN ELEV 718.84 & 745.00
DIMENSION 'E' CLEARANCE TO END OF NOZZLE SAFE END (SEE TABLE)
VESSEL O-C-O BASIC
ELEV 15 5/8 (REF)
ELEV -14 3/8 (REF)
ELEV 703 (58'7") (REF)
ELEV 348 5/16 (29'-0 5/16")
ELEV 366
ELEV 860 MAX TO FLANGE FACE
30'00".25
745'11" (62'-1")

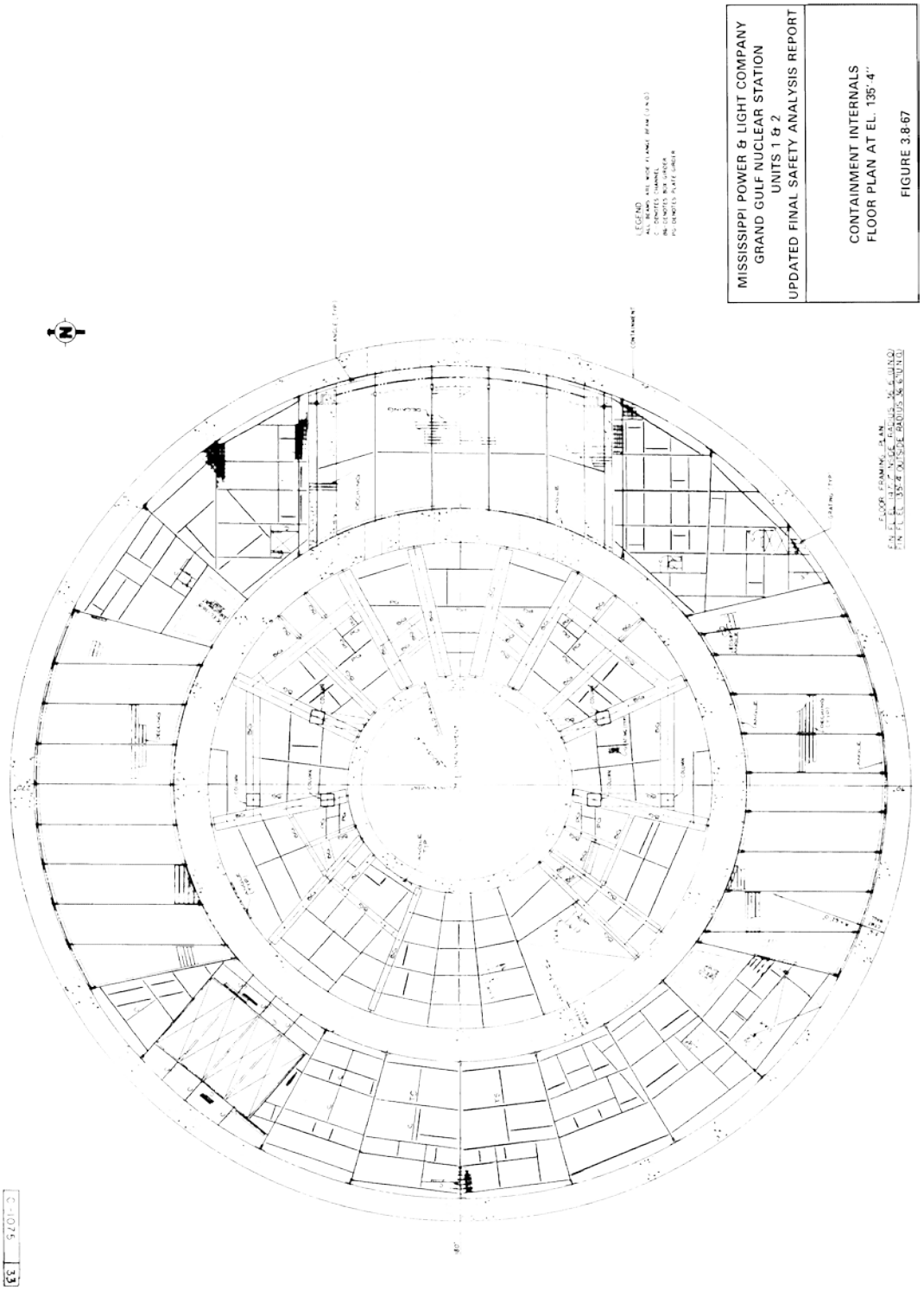

MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT
CONTAINMENT INTERNALS
BOTTOM HEAD ASSEMBLY
UNIT 1
FIGURE 3.8-65b
PLAN VIEW
ELEVATION
BU BAR
THERMOCOUPLE PADS (TYP)
SHROUD SUPPORT STUBS (TYP)
SUPPORT SKIRT

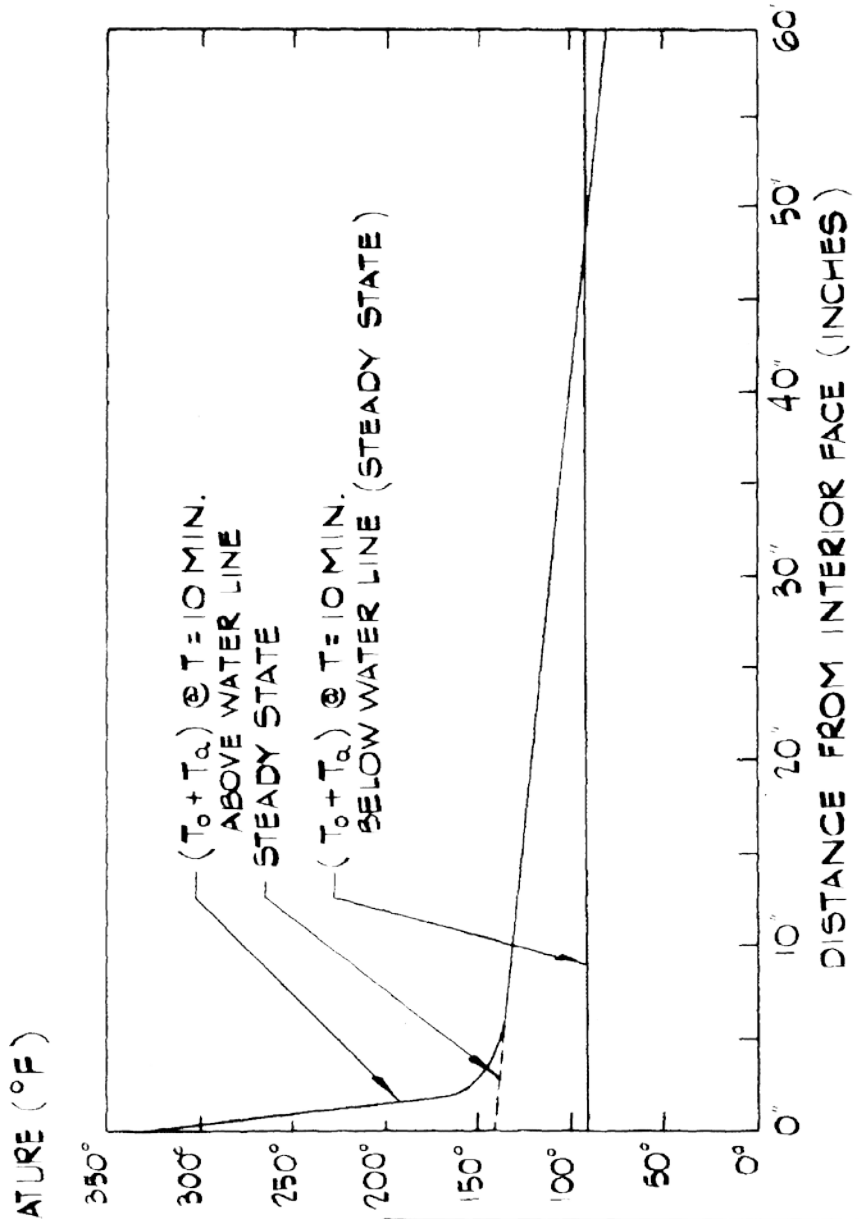

MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT
CONTAINMENT INTERNALS
SKIRT EXTENSION DETAILS
FIGURE 3.8-65c
NOTES:
1. SHOP, UT BASE METPL 4 3/8 (Min) DEEP BY 2 1/2 (Min) ABOVE & BELOW WELD BUILD-UP.
2. SHOP TO ADD BUILD-UP PER HEAD VARIATION
DETAIL OF WELD BUILD UP
WELD PER WPS 323-2F4FG
DETAIL OF WELD BUILD-UP
AFTER MACHINING
SKIRT EXTENSION VERT SEAMS @ 0°/180°
WELD PER WPS 303-1F4
WELD PREP MTPW & MTP-11
BACKGOUGE MTP-10
& COMP WELD
SECTION THRU SKIRT EXTENSION
WELD PER WPS 323-2F4FG
BACKGOUGE MTPW
& COMP WELD UTP-38
BLEND GRIND
AFTER FORMING
BOTH SIDES
MTP-10 OR MTP-11
UTP-26
FINAL SURFACES
MTPW
HEAD TANGENT LINE
BLEND GRIND
SEE NOTE 1
UTP-30
UTP-36
VESSEL ℄
ELEV. 45'-3"
13 BOT HD ASSY
10 SKIRT (BASE) ASSY
4 GEN NOTES (TOLERANCES)
9A SKIRT EXTENSION ASSY
9-1 2 R 20 7/16 x 4 1/4 SA-533 GR B CL 1 W/ R 44 x 2 3/8 x 34'-7 7/8
9-2 2 3/8 ϕ BU BAR 2 5 SA-516 GR 70


NOTES:
1. WELD BASE PLATE INTO RING BEFORE TRIMMING ID/OD
2. SHOP MACHINE BOTTOM OF BASE PLATE. (HOLES IN BASE PLATE AFTER PWHT.
3. SUPPORT SKIRT & BASE RING TO HAVE AN RT/NDT NO HIGHER THAN +10° F.
4. SHOP CHECK ELEVATION OF SKIRT KNUCKLE BEFORE TRIMMING TOP EDGE OF (1) TO ASSURE CORRECT SKIRT LENGTH OF FINAL ASSEMBLY.
5. AFTER MANHOLE 50 ARE MADE, THE SUP SKIRT CAN BE USED AS A SUP POINT ON TURNING ROLLS FOR BOT HD & RING SUB ASSY ONLY.
6. ALL PWHT OF THE BASE R. SHALL BE PERFORMED BETWEEN 1100°F MIN & 1150°F MAX.
MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT
CONTAINMENT INTERNALS
SKIRT AND BASEPLATE DETAILS
FIGURE 3.8-65d
DETAIL B
DETAIL C
DETAIL D
PLAN VIEW
SECTION A-A


MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT
CONTAINMENT INTERNALS
FLOOR AT EL. 120'-10" PLAN
FIGURE 3.8-66
N
SUPPRESSION POOL
6
8 7410-C

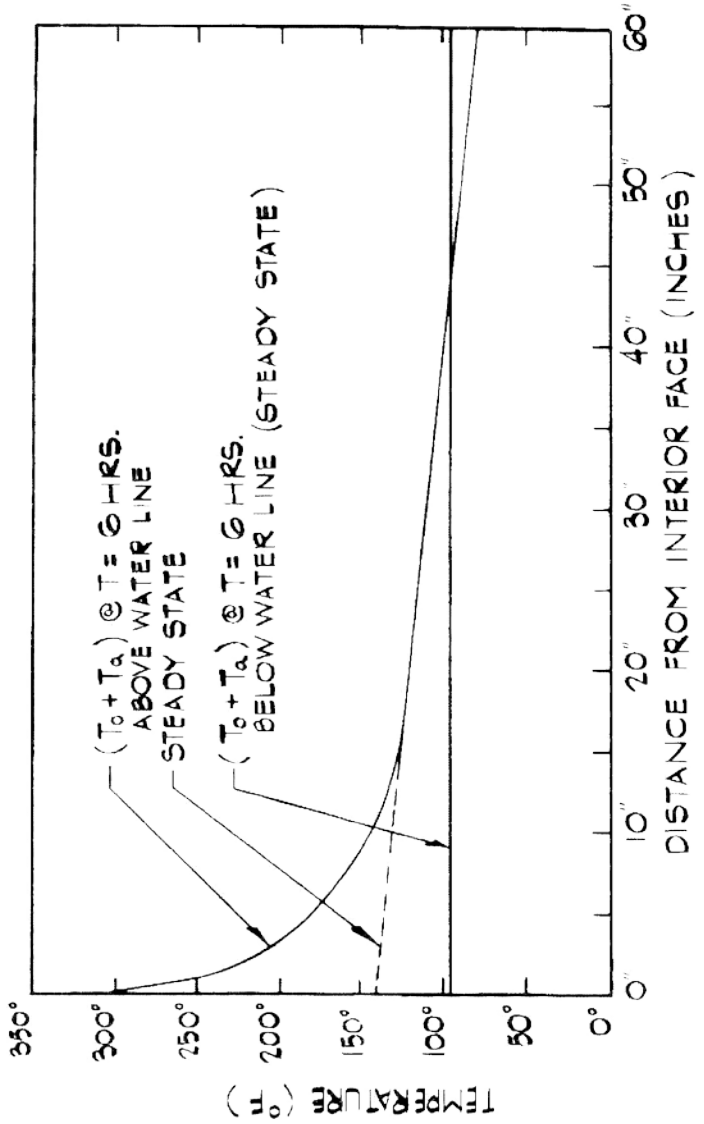

MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT
CONTAINMENT INTERNALS
POOL SWELL DEFLECTORS
PLANS AND SECTIONS
FIGURE 3.8.66a
PLAN
SECTION
C-1074D


MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT
CONTAINMENT INTERNALS
FLOOR PLAN AT EL. 135'-4"
FIGURE 3.8-67
LEGEND
N
E-1075
33

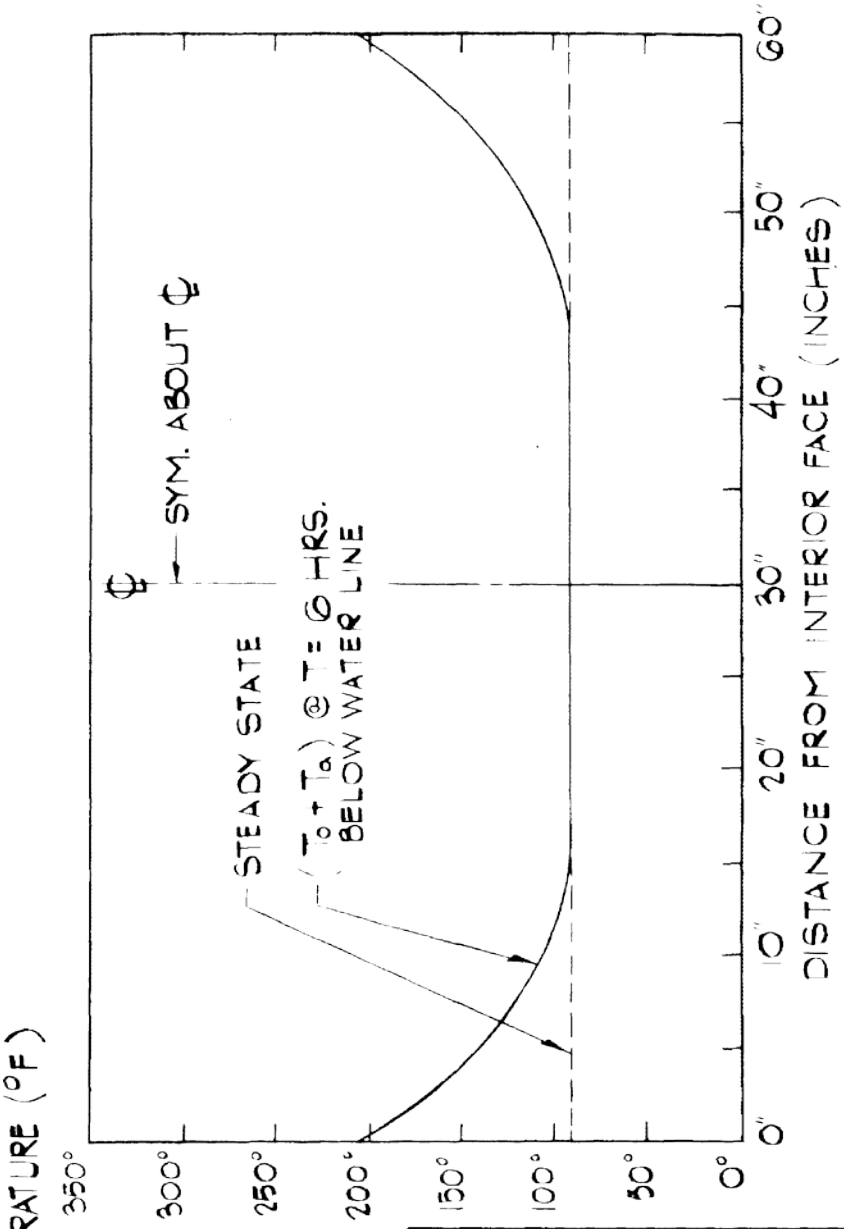


MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT

CONTAINMENT INTERNALS
LOWER DRYWELL THERMAL GRADIENTS
DESIGN BASIS ACCIDENT
t = 10 MINUTES
FIGURE 3.8-68

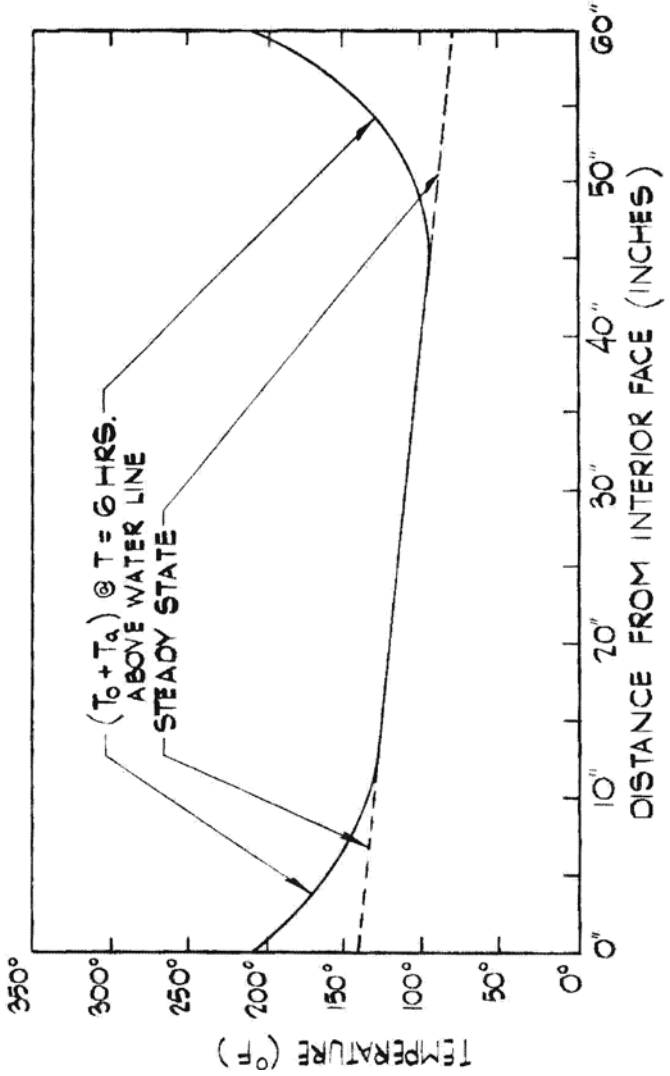


MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT	CONTAINMENT INTERNALS DRYWELL WALL THERMAL GRADIENTS SMALL BREAK ACCIDENT t = 6 HOURS FIGURE 3.8-69

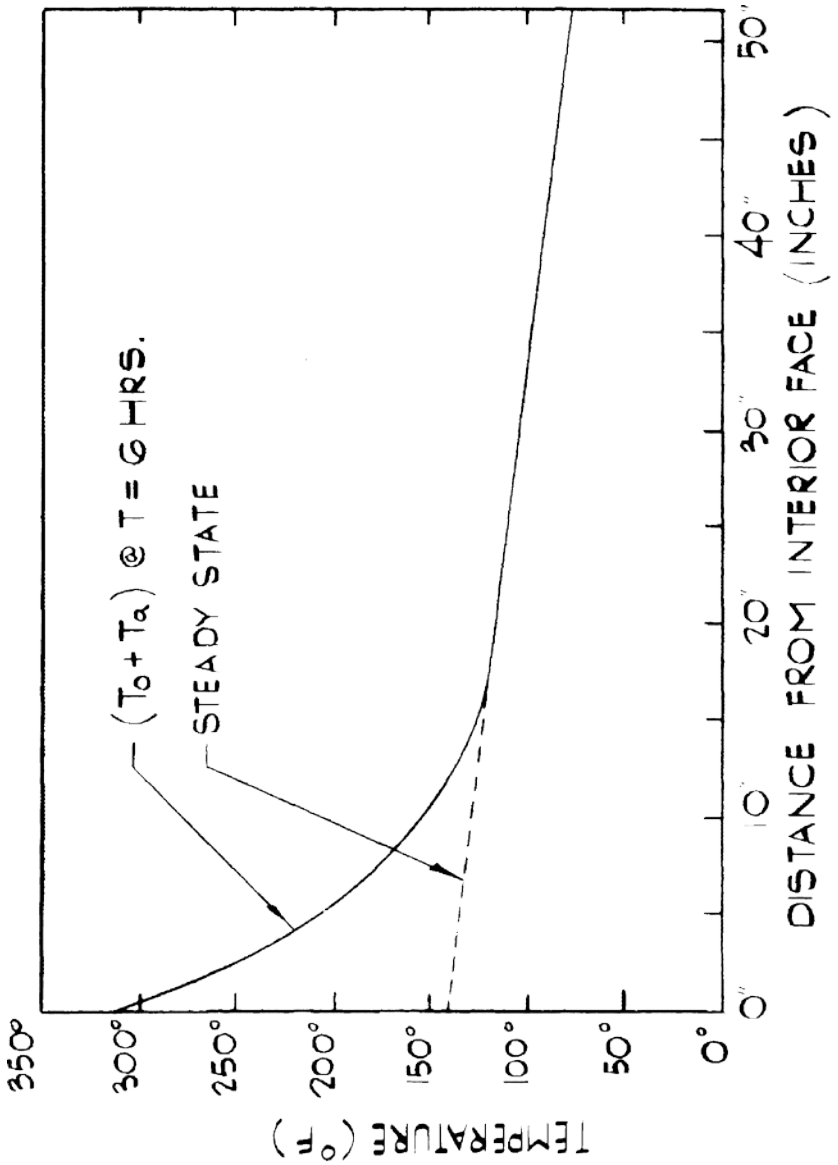


MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT

CONTAINMENT INTERNALS
DRYWELL THERMAL GRADIENTS
BELOW WATERLINE
DESIGN BASIS ACCIDENT
t = 6 HOURS
FIGURE 3.8-70

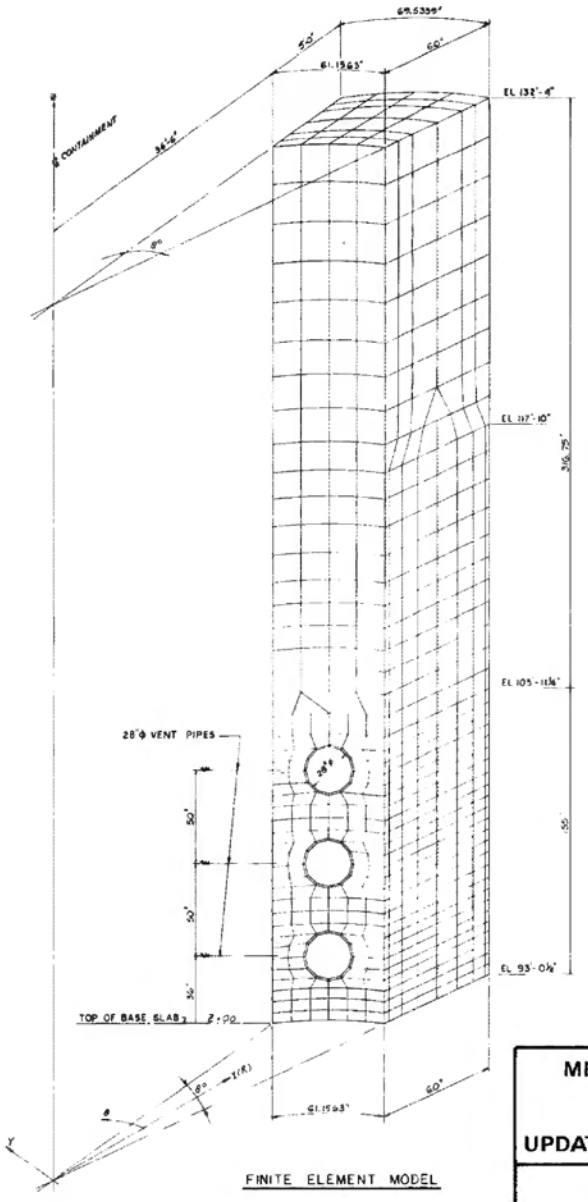


MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT	CONTAINMENT INTERNALS LOWER DRYWELL THERMAL GRADIENTS DESIGN BASIS ACCIDENT t = 6 HOURS FIGURE 3.8-71



MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT
CONTAINMENT INTERNALS DRYWELL ROOF THERMAL GRADIENTS SMALL BREAK ACCIDENT t = 6 HOURS FIGURE 3.8-72

SKC-213 3



FINITE ELEMENT MODEL

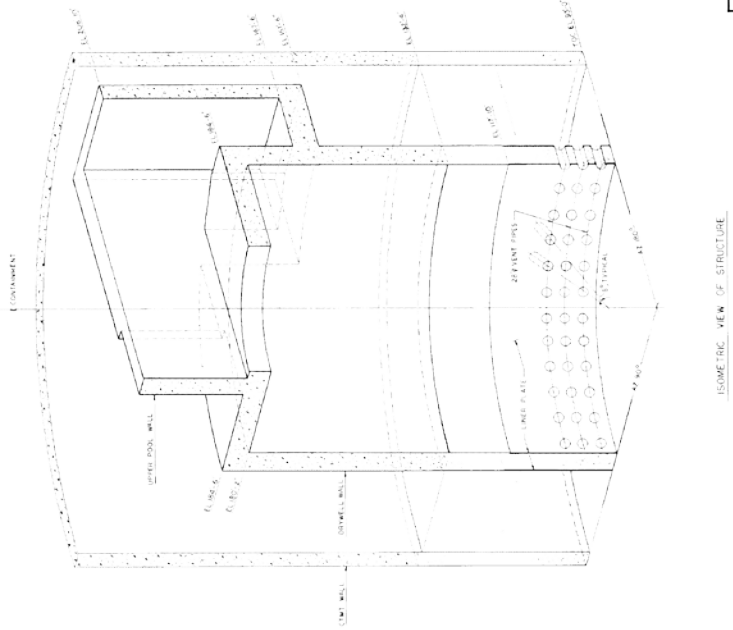
NUMBER OF NODES - 2719
NUMBER OF ELEMENTS - 1912

MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT
CONTAINMENT INTERNALS LOWER DRYWELL WALL FINITE ELEMENT MODEL INCLUDING VENTS FIGURE 3.8-73

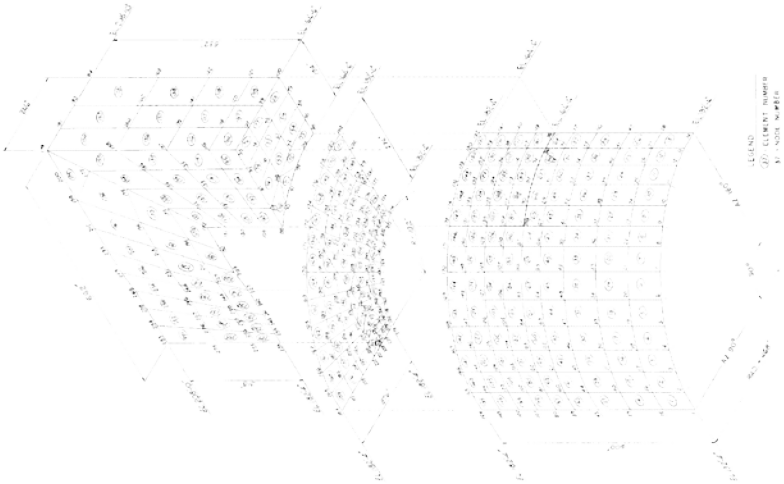
MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT

CONTAINMENT INTERNALS, UPPER DRYWELL WALL, DRYWELL ROOF, REFUELING POOLS AND UPPER POOL WALLS FINITE ELEMENT MODEL

FIGURE 3.8-74

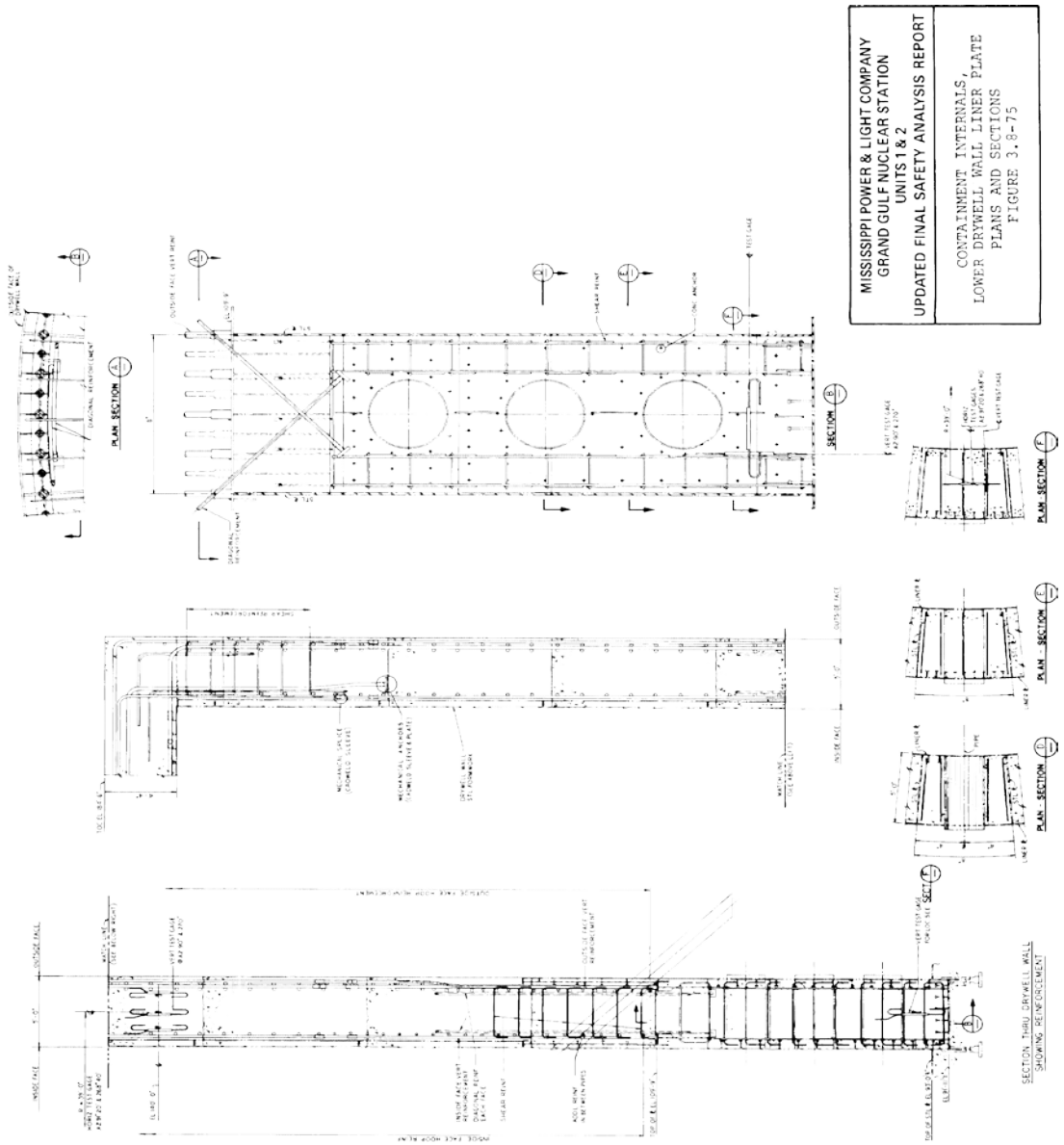


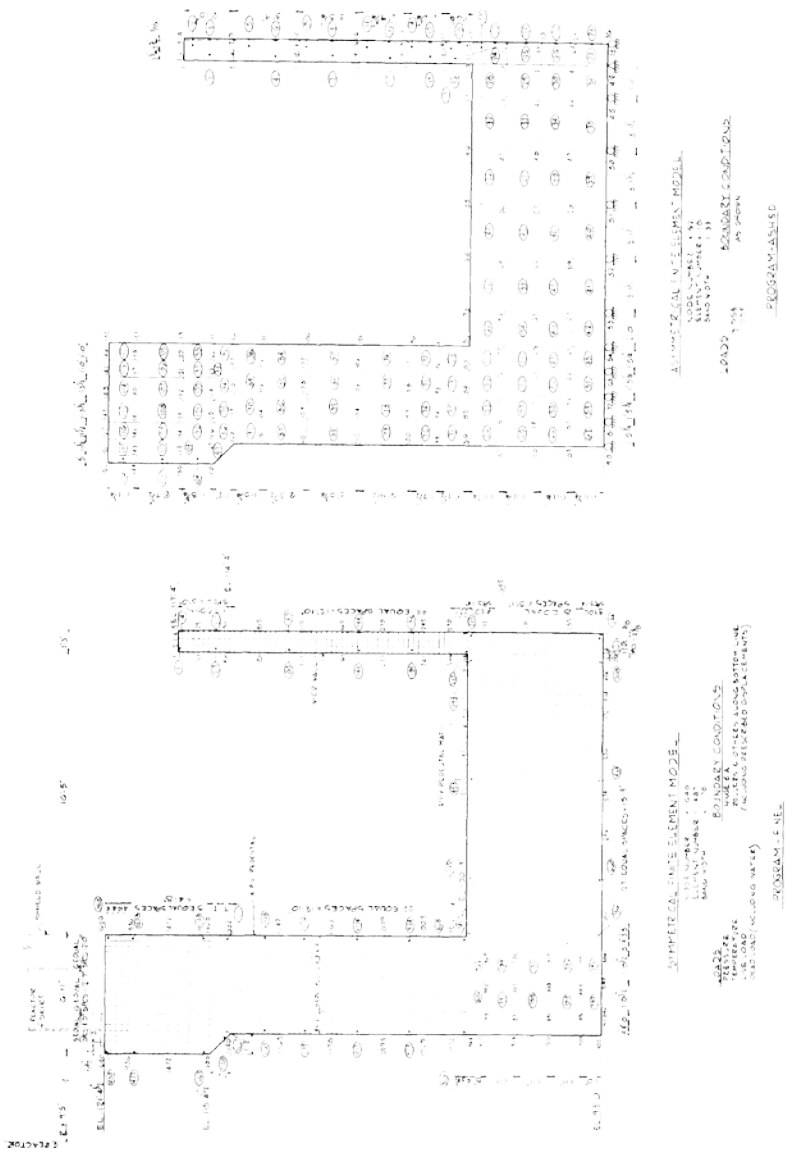
ISOMETRIC VIEW OF STRUCTURE



FINITE ELEMENT MODEL
(EXPLODED VIEW)

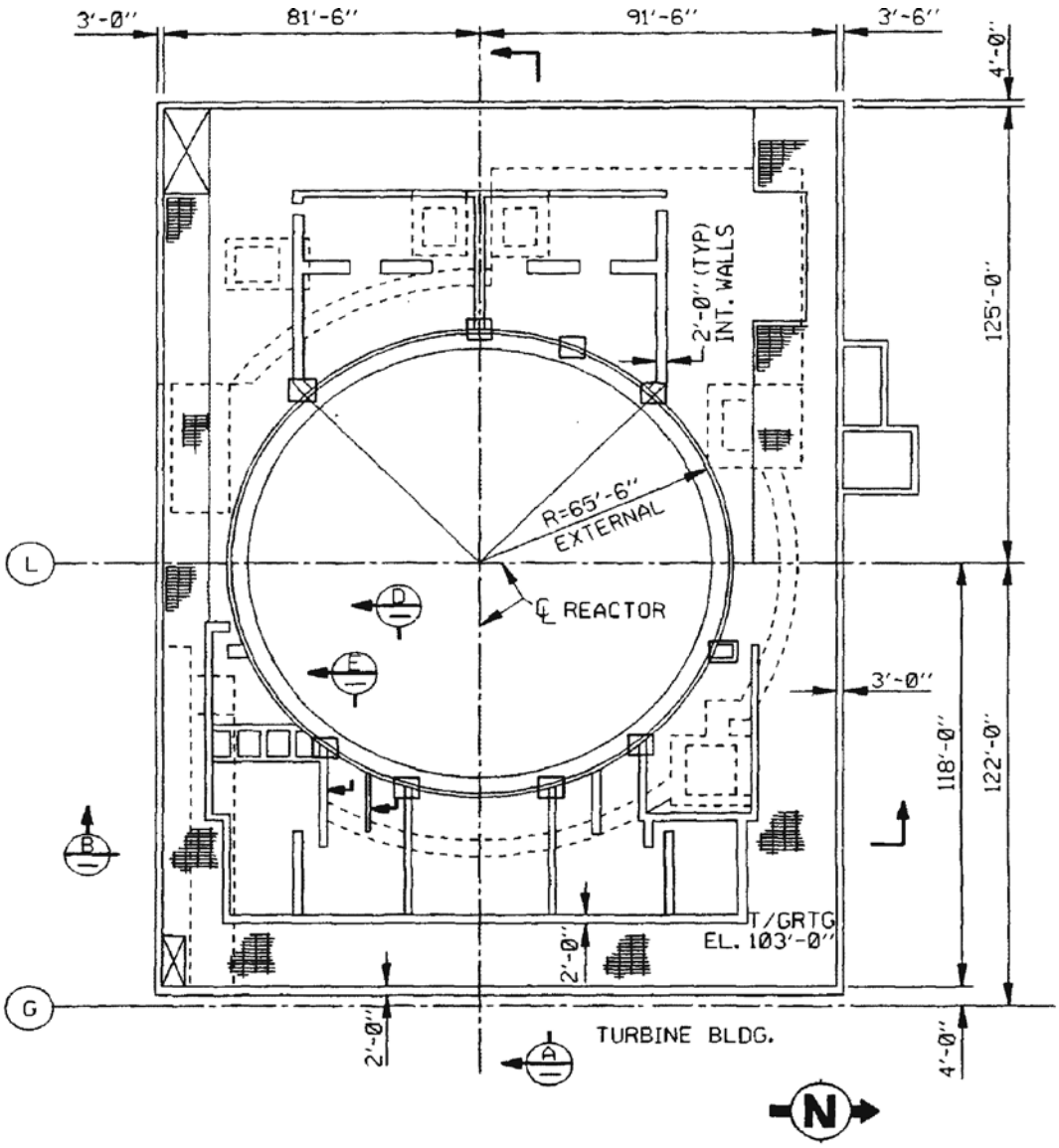
SKC-215


MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT
CONTAINMENT INTERNALS,
LOWER DRYWELL WALL LINER PLATE
PLANS AND SECTIONS
FIGURE 3.8-75
PLAN - SECTION
SECTION
SECTION THRU DRYWELL WALL SHOWING REINFORCEMENT
C-1059A 7



MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT

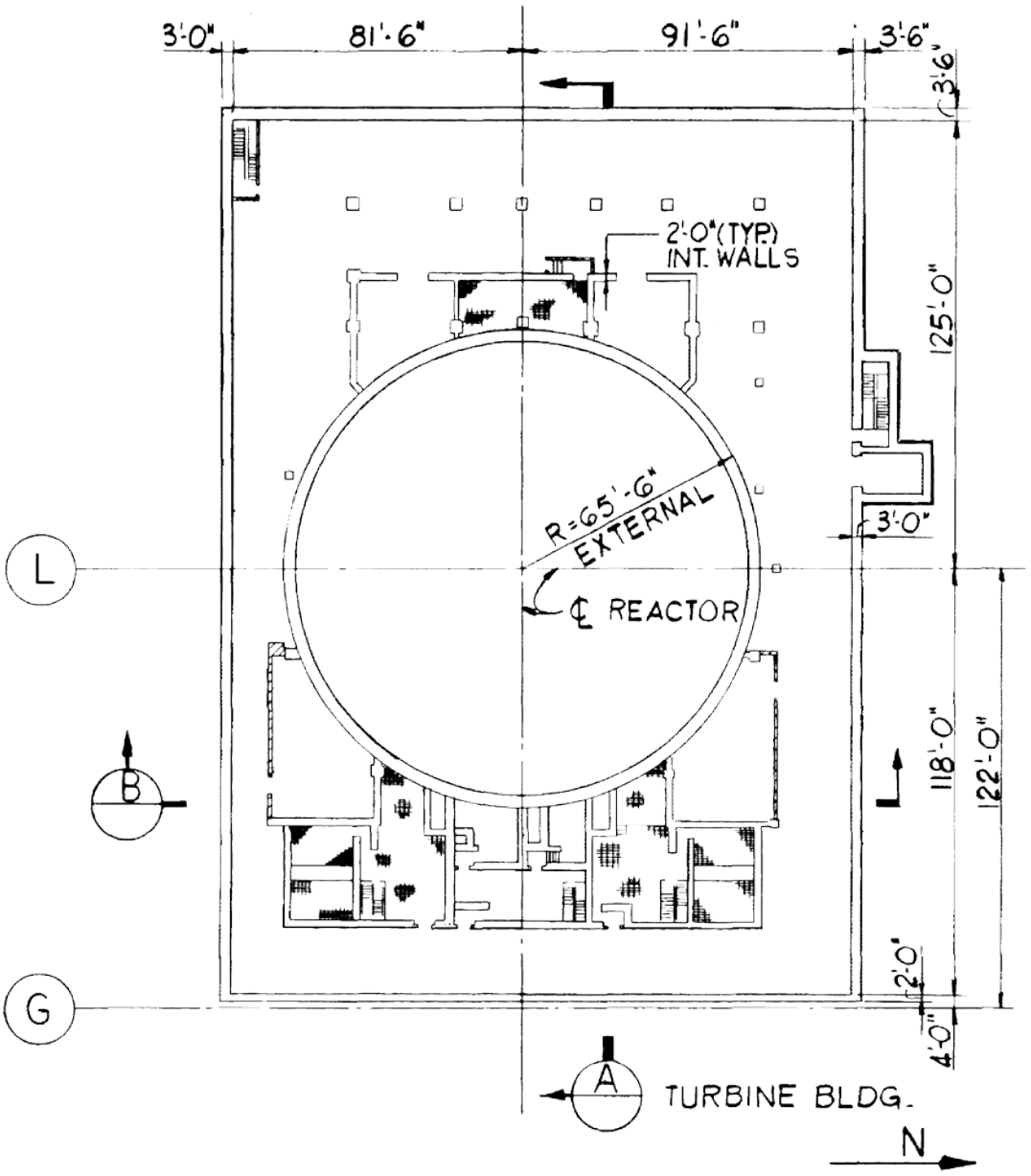
CONTAINMENT INTERNALS
RPV PEDESTAL, PEDESTAL MAT,
AND WEIR WALL
FINITE ELEMENT MODEL
FIGURE 3.8-76



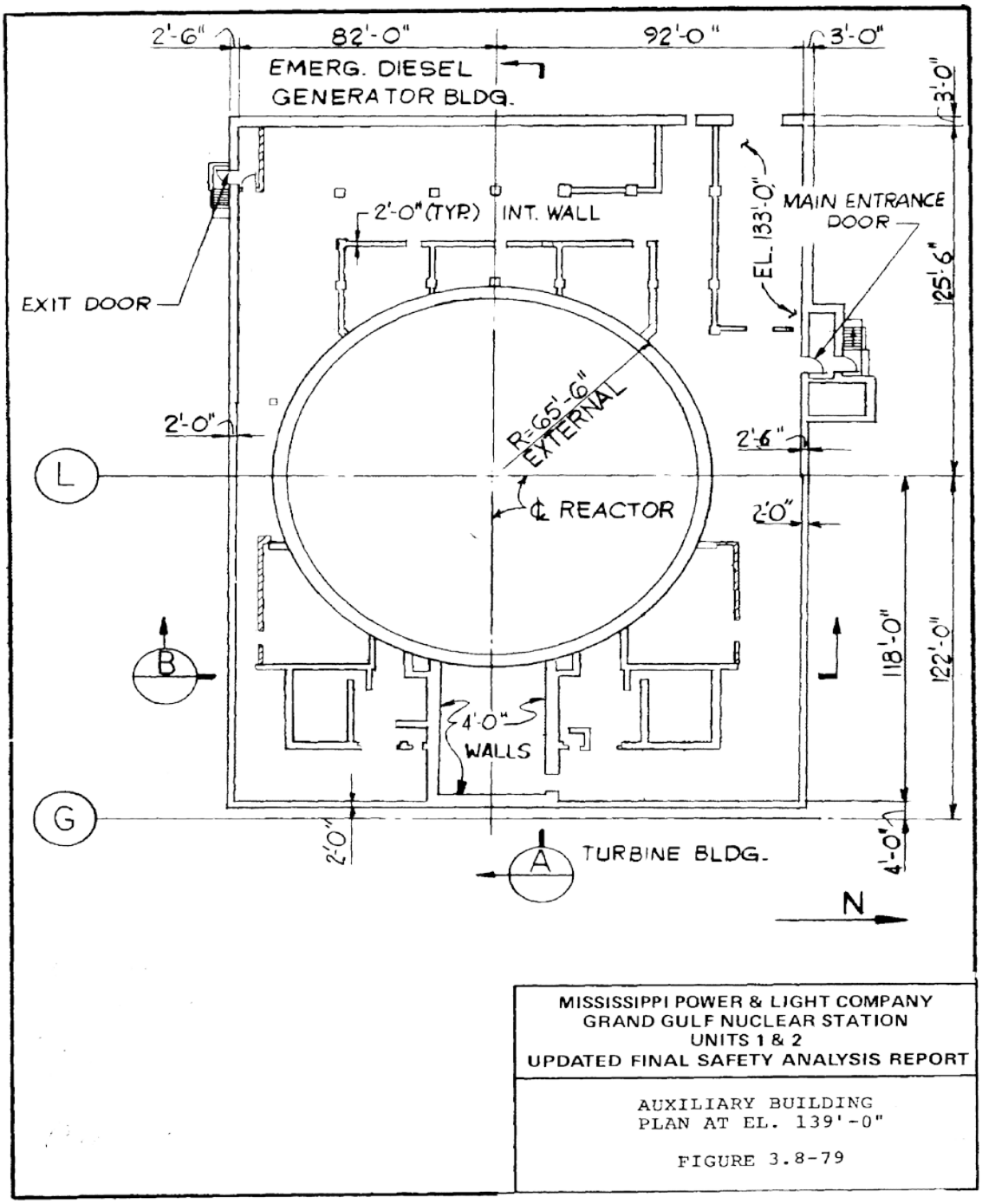
□: LOCATION WHERE NO GAP EXISTS BETWEEN WALL AND SLAB

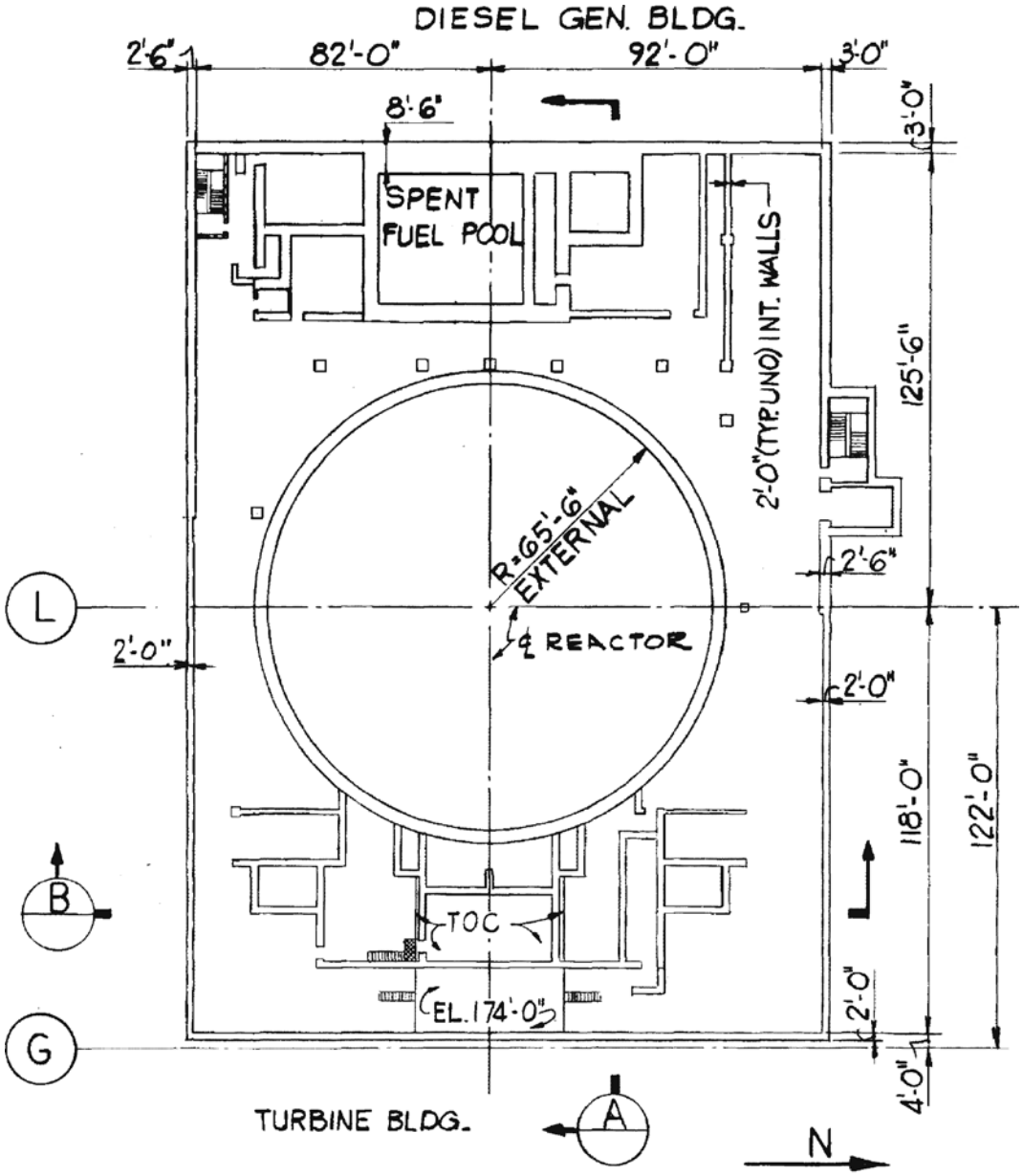
GRAND GULF NUCLEAR STATION
UNIT 1
UPDATED FINAL SAFETY ANALYSIS REPORT

AUXILIARY BUILDING
PLAN EL. 93'-0"
FIGURE 3.8-77



MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT
AUXILIARY BUILDING PLAN AT EL. 119'-0" FIGURE 3.8-78

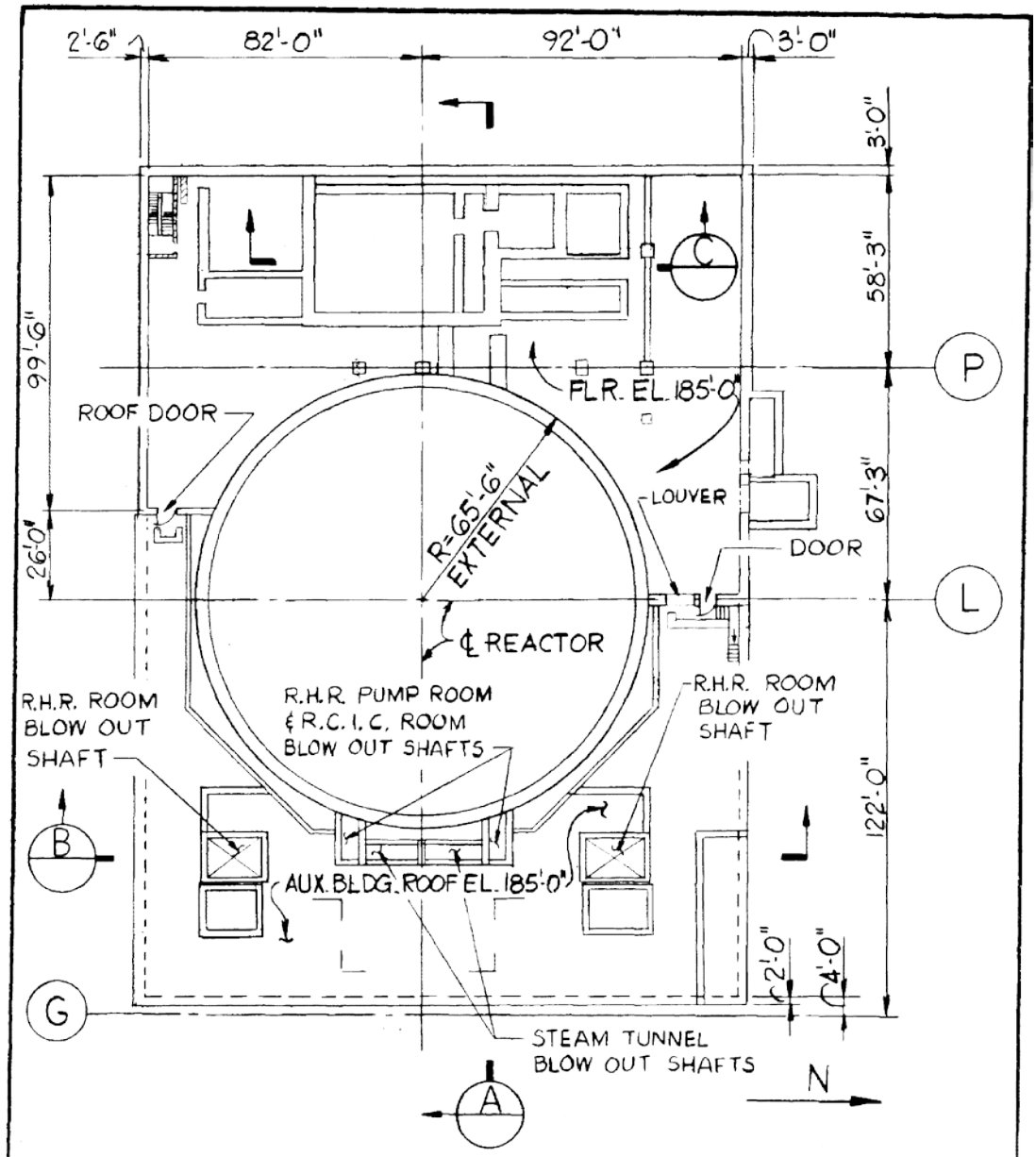

2'-6"
82'-0"
92'-0"
3'-0"
EMERG. DIESEL GENERATOR BLDG.
3'-0"
2'-0" (TYP.) INT. WALL
EL. 133'-0"
MAIN ENTRANCE DOOR
EXIT DOOR
125'-6"
R=65'-6" EXTERNAL
2'-0"
2'-6"
L
℄ REACTOR
2'-0"
B
118'-0"
122'-0"
4'-0" WALLS
G
2'-0"
A
TURBINE BLDG.
4'-0"
N
MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT
AUXILIARY BUILDING
PLAN AT EL. 139'-0"
FIGURE 3.8-79



MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT

AUXILIARY BUILDING
PLAN AT EL. 166'-0"

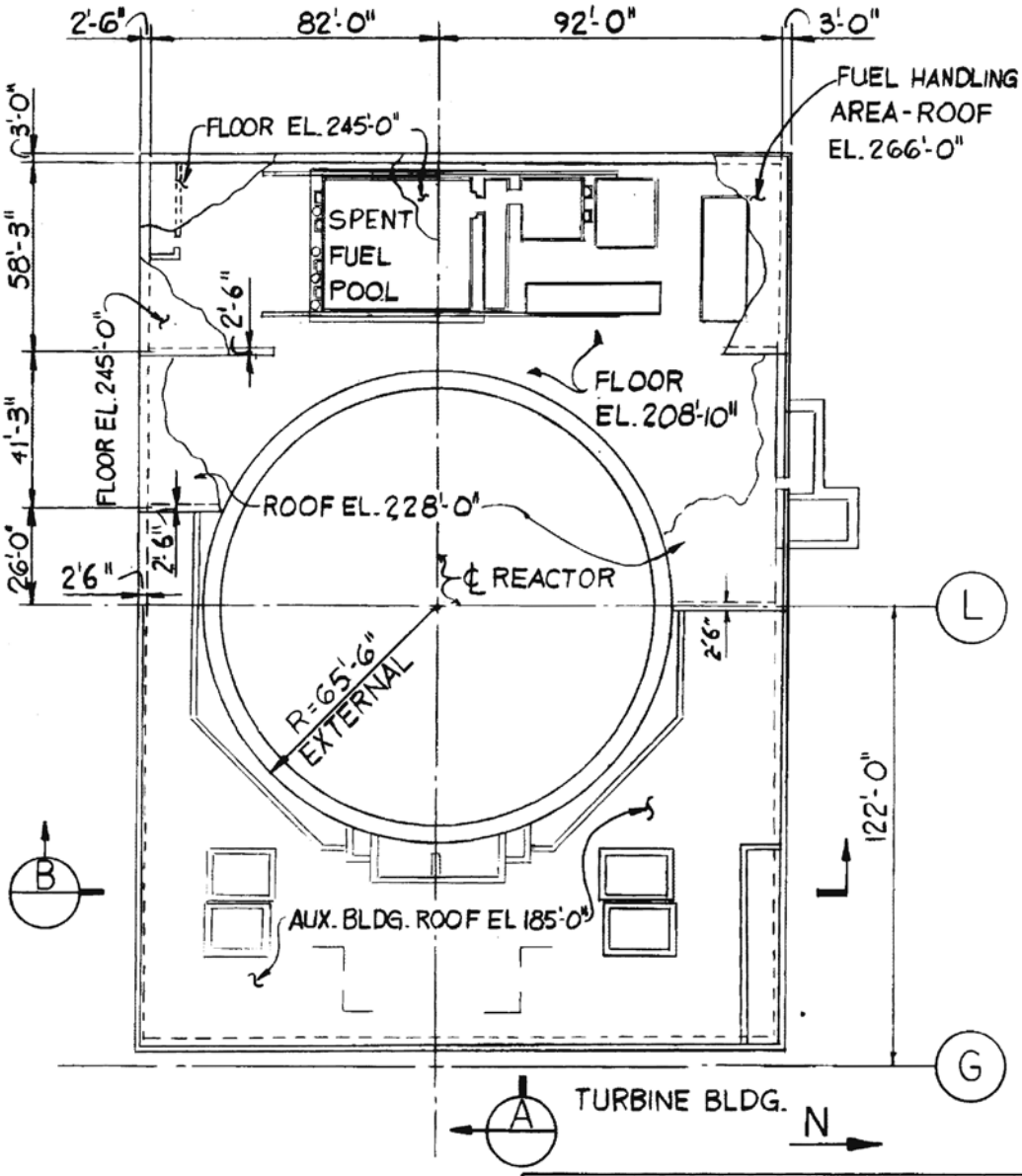
FIGURE 3.8-80



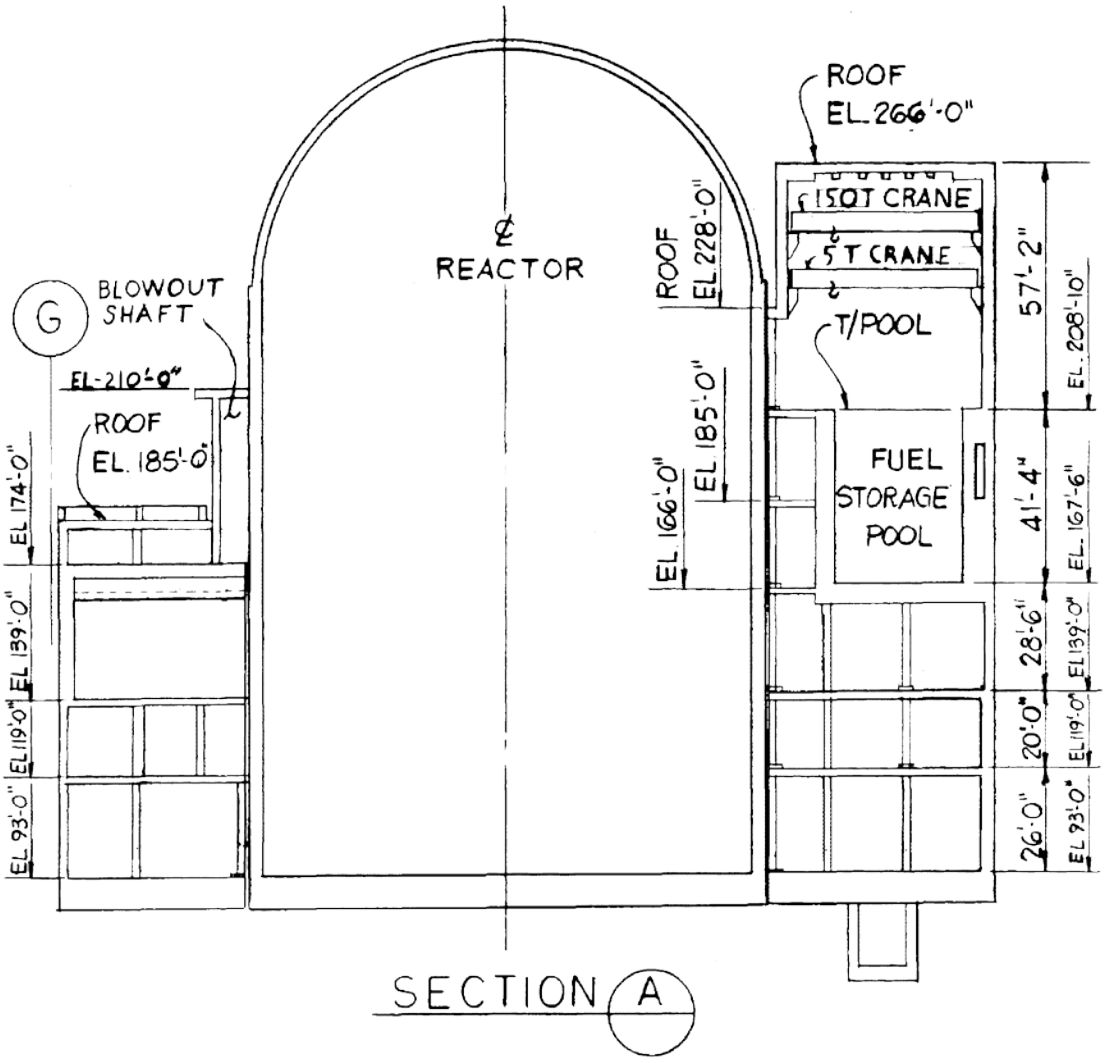
MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT

AUXILIARY BUILDING
PLAN AT EL. 185'-0"

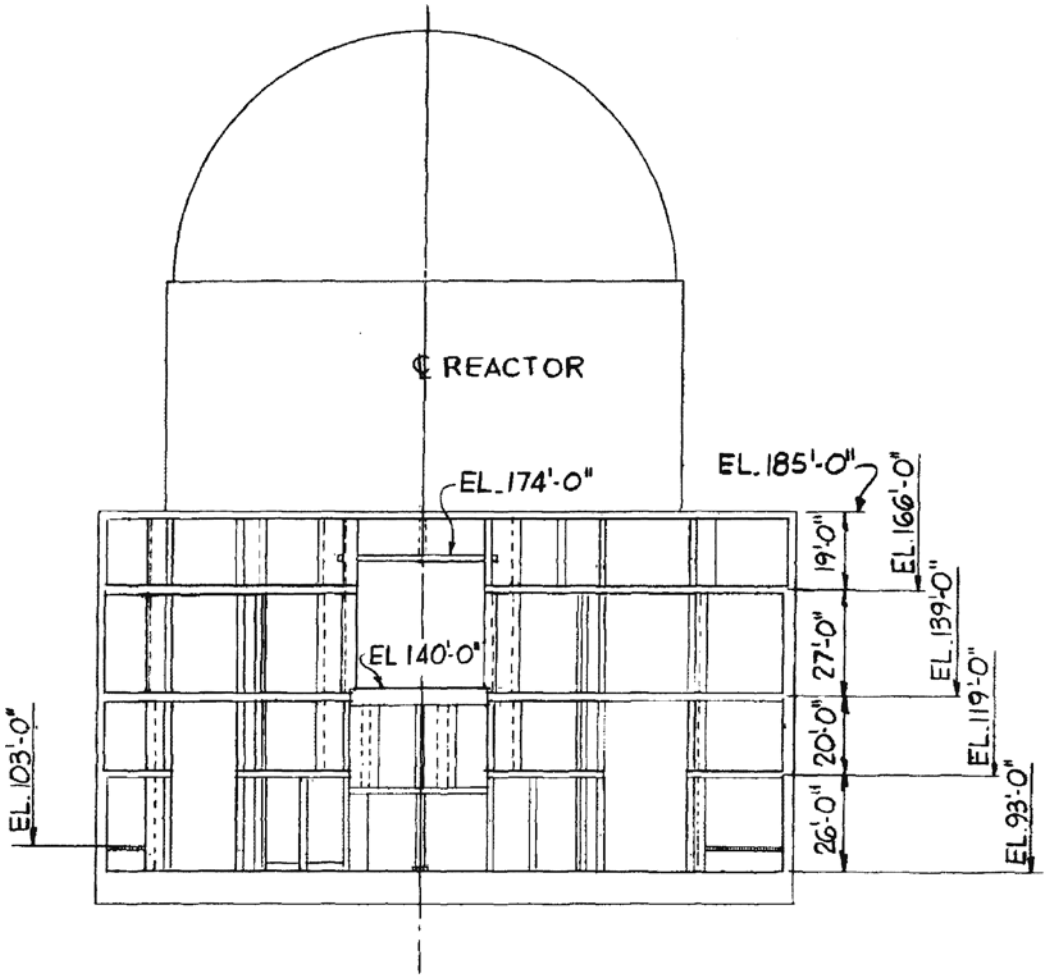
FIGURE 3.8-81



MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT
AUXILIARY BUILDING PLAN AT EL. 208'-10" FIGURE 3.8-82

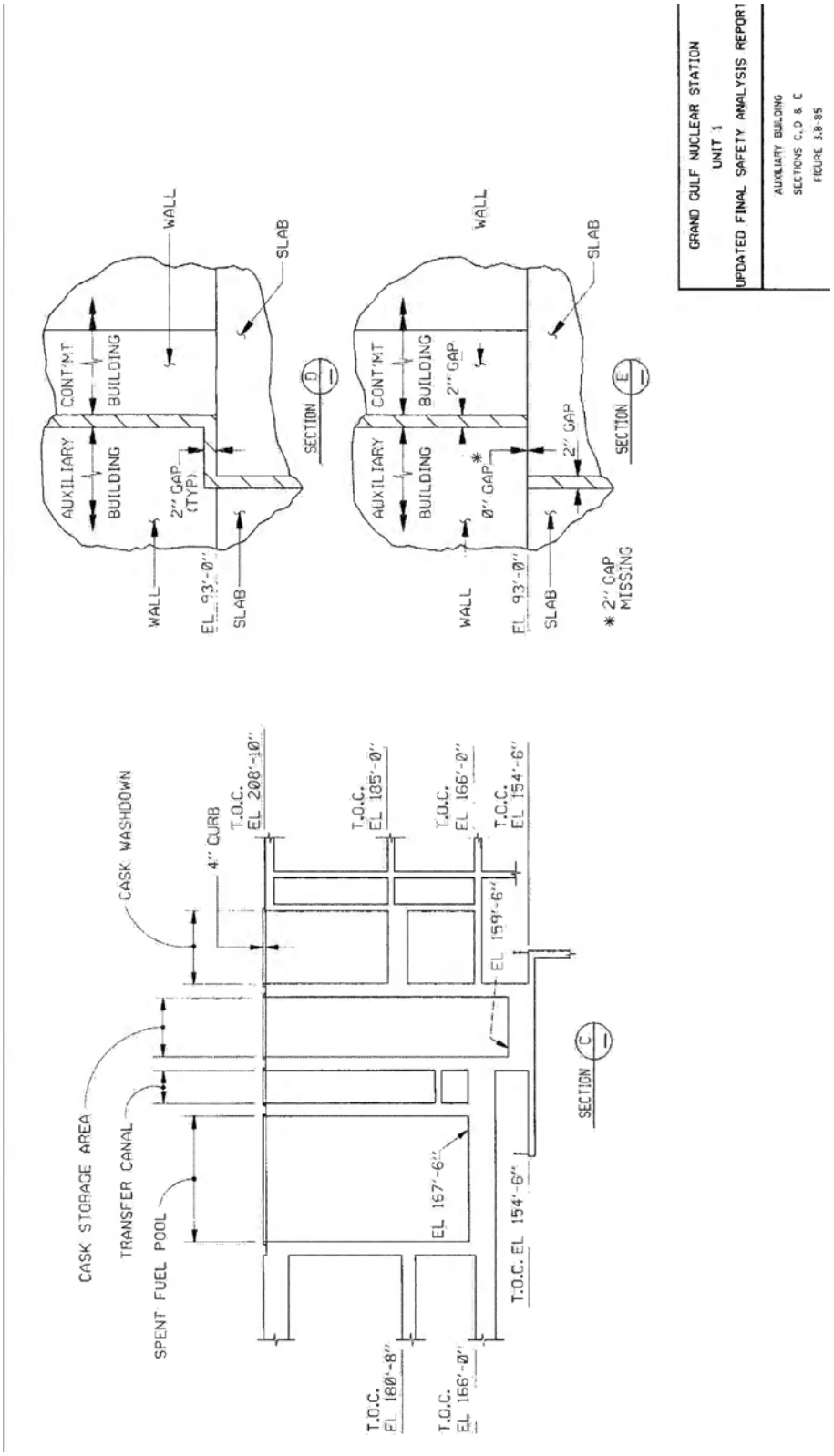


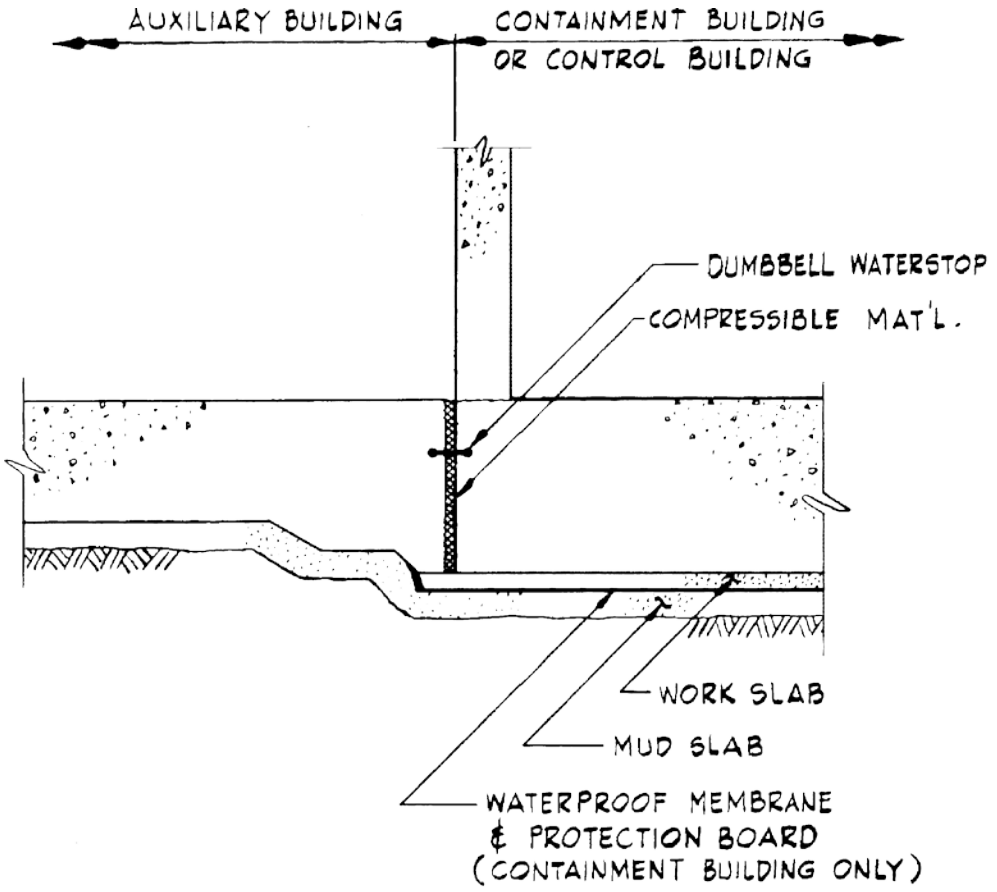
MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT
AUXILIARY BUILDING SECTION A FIGURE 3.8-83



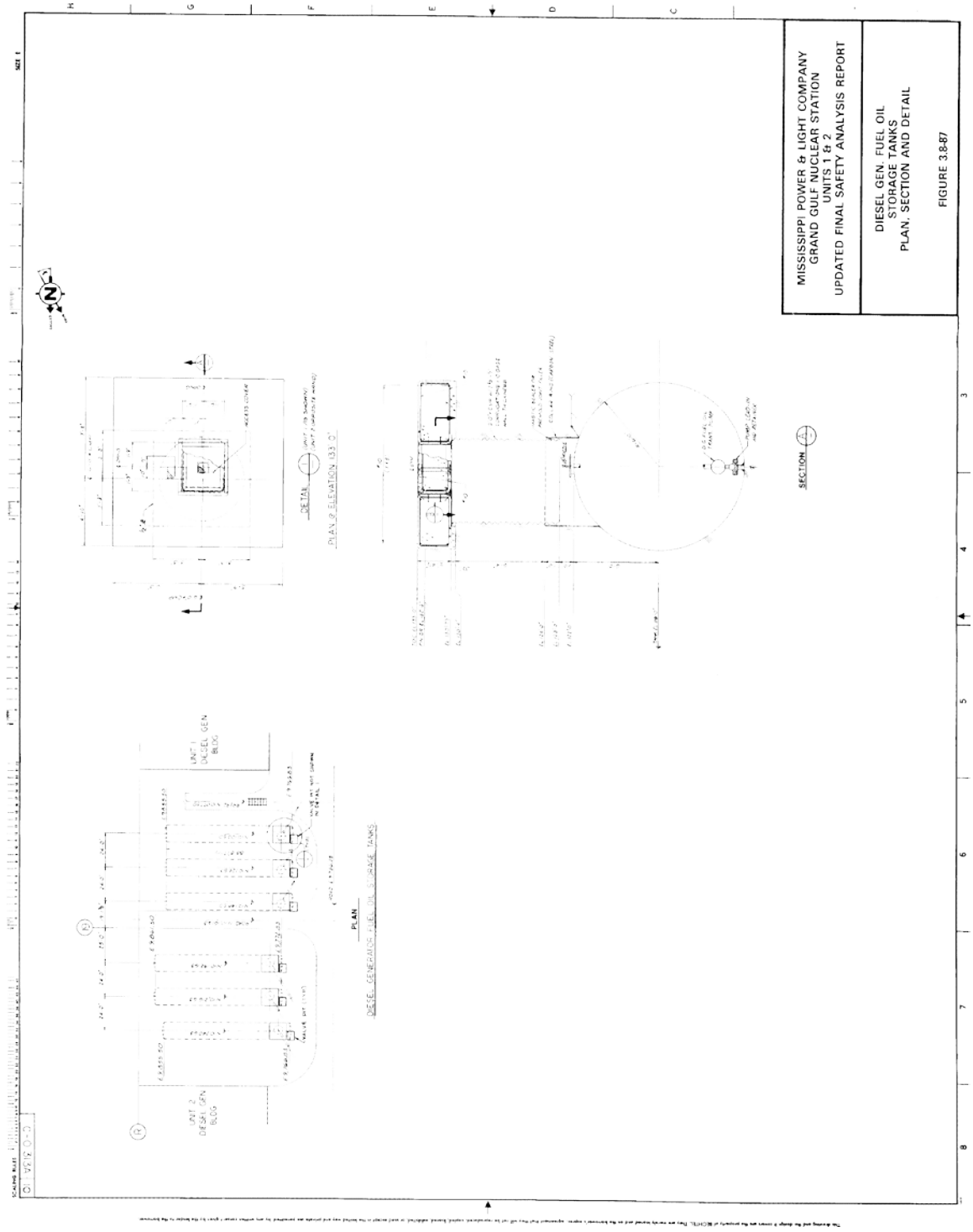
SECTION B

MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT
AUXILIARY BUILDING SECTION B FIGURE 3.8-84

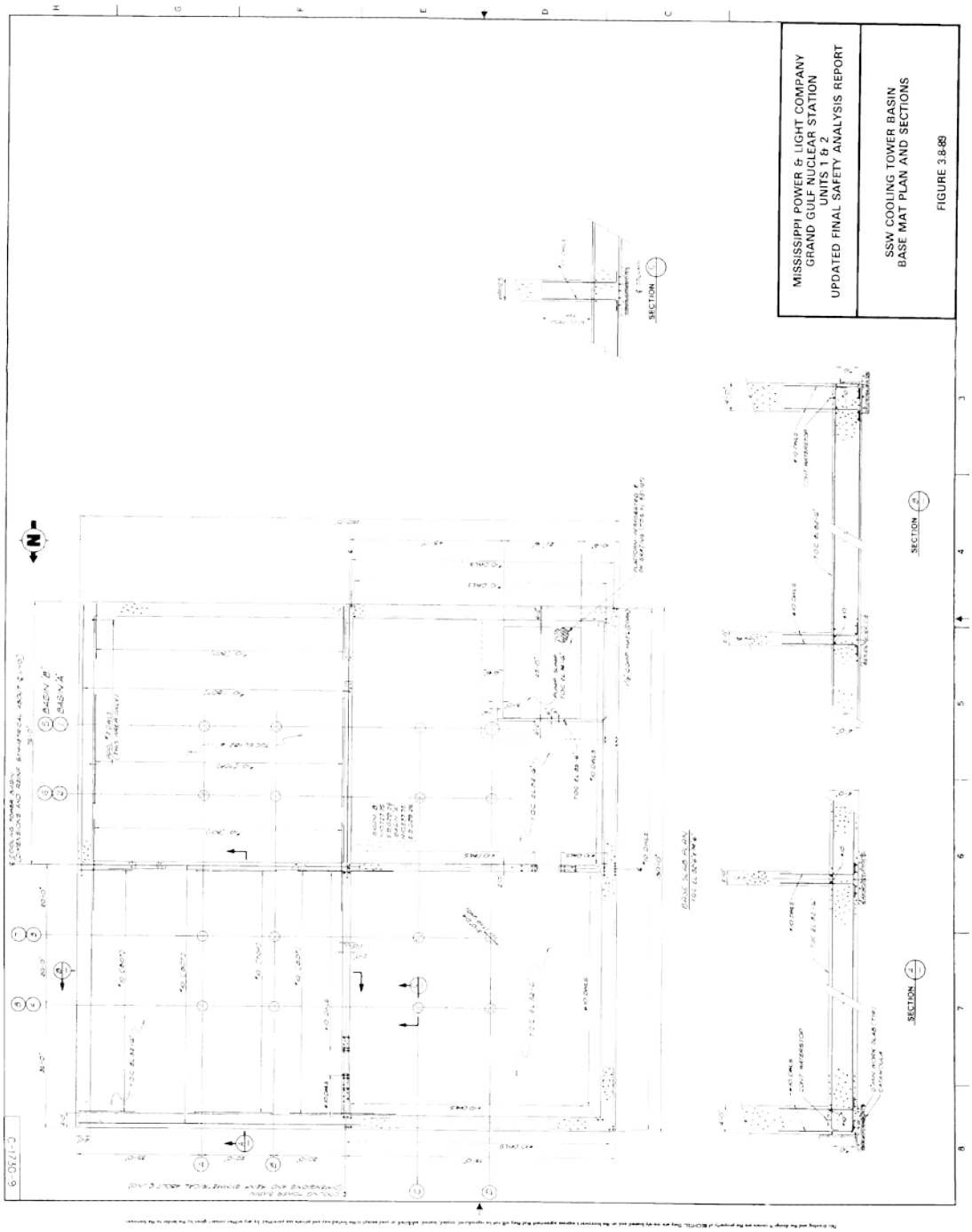

GRAND GULF NUCLEAR STATION
UNIT 1
UPDATED FINAL SAFETY ANALYSIS REPORT
AUXILIARY BUILDING
SECTIONS C, D & E
FIGURE 3.8-65
CONTMT BUILDING
AUXILIARY BUILDING
WALL
SLAB
2" GAP (TYP)
EL 93'-0"
SECTION D
2" GAP
0" GAP
* 2" GAP MISSING
SECTION E
CASK WASHDOWN
4" CURB
T.O.C. EL 208'-10"
T.O.C. EL 185'-0"
T.O.C. EL 166'-0"
T.C.C. EL 154'-6"
EL 159'-6"
CASK STORAGE AREA
TRANSFER CANAL
SPENT FUEL POOL
EL 167'-6"
T.O.C. EL 154'-6"
SECTION C
T.O.C. EL 180'-8"
T.O.C. EL 166'-0"

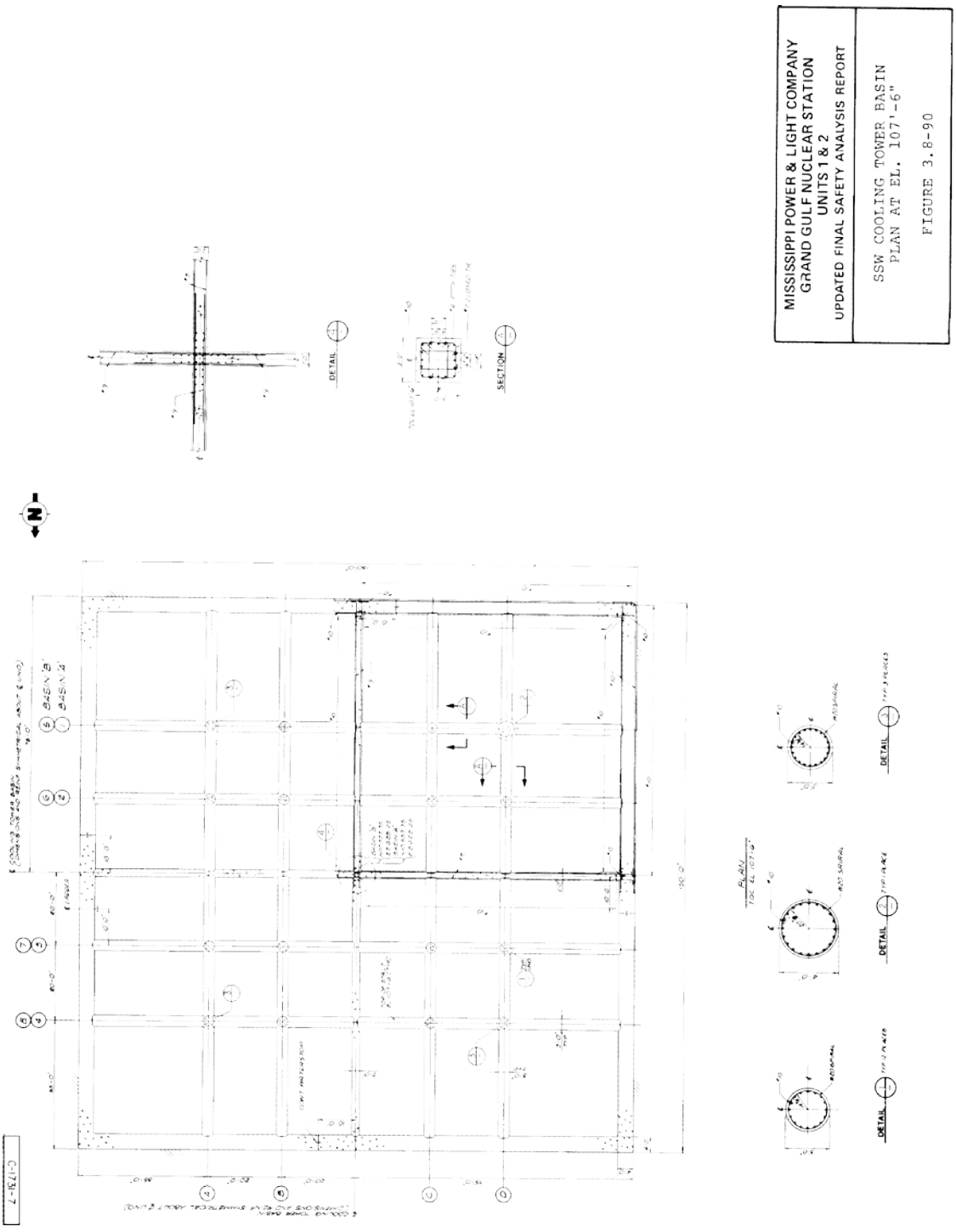


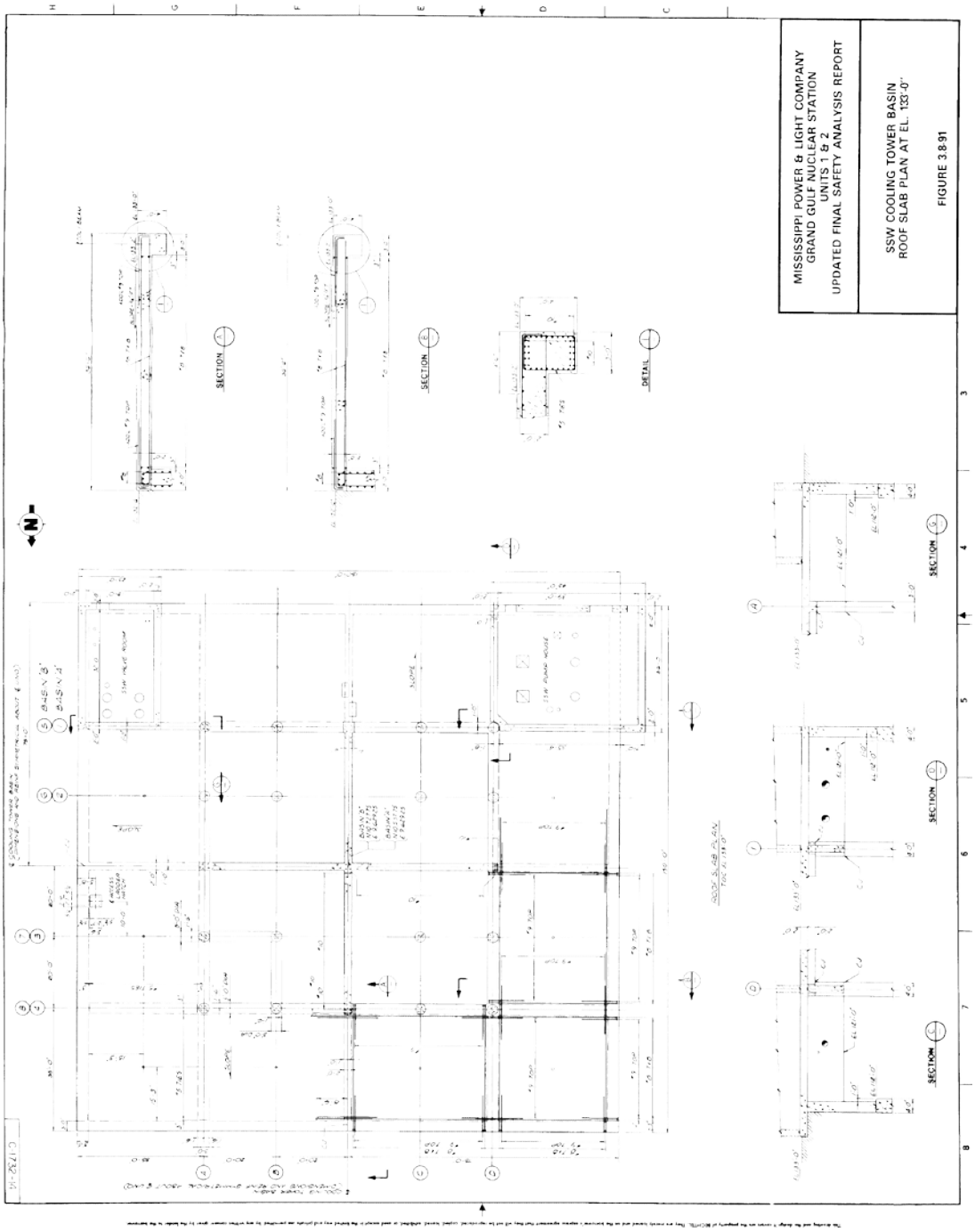
MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT
FOUNDATIONS-TYPICAL DETAIL FIGURE 3.8-86

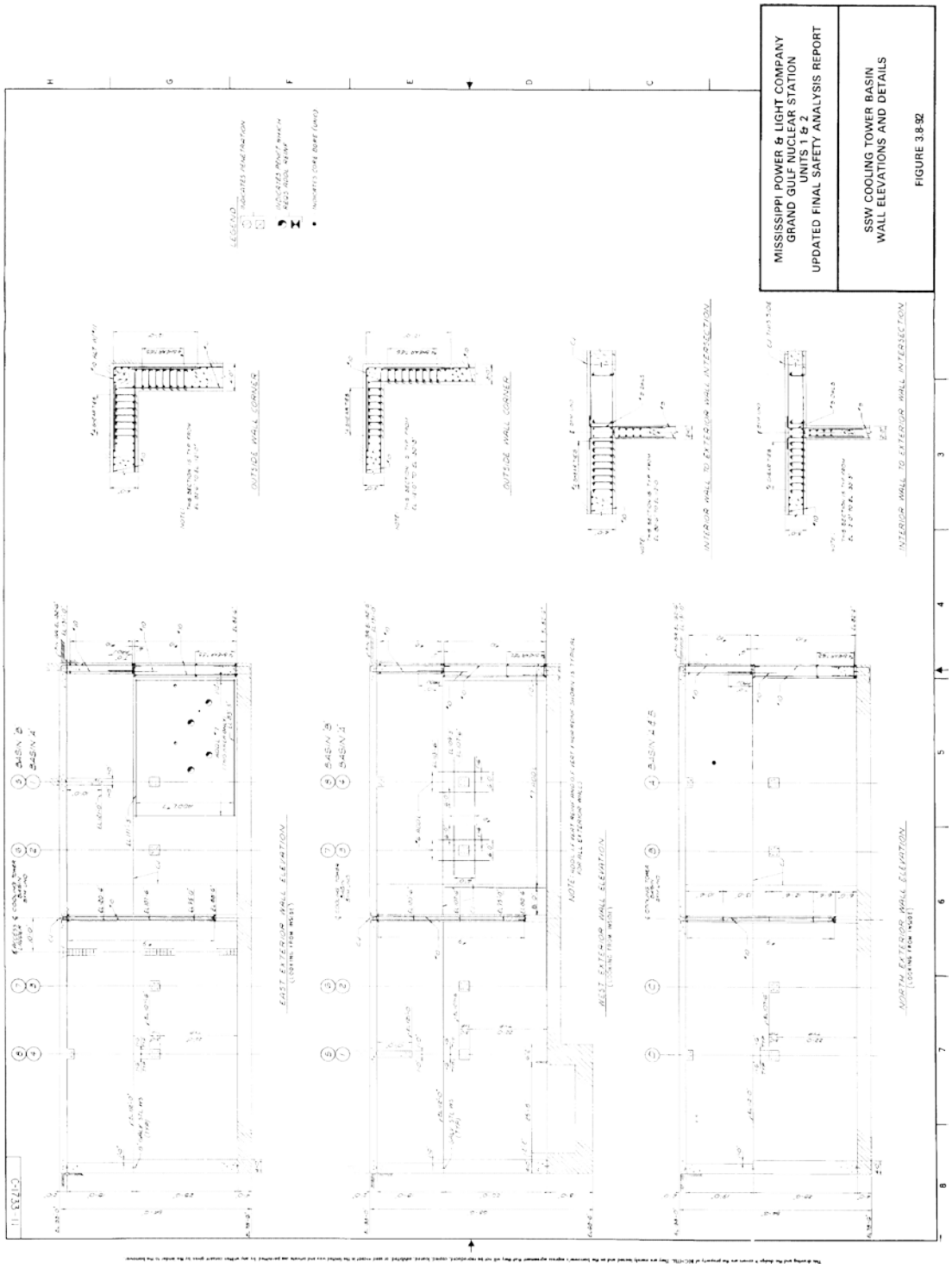

MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT
DIESEL GEN. FUEL OIL
STORAGE TANKS
PLAN, SECTION AND DETAIL
FIGURE 3.8-87
PLAN & ELEVATION (93.0')
DETAIL
SECTION
PLAN
UNIT 1
DIESEL GEN
BLDG
UNIT 2
DIESEL GEN
BLDG

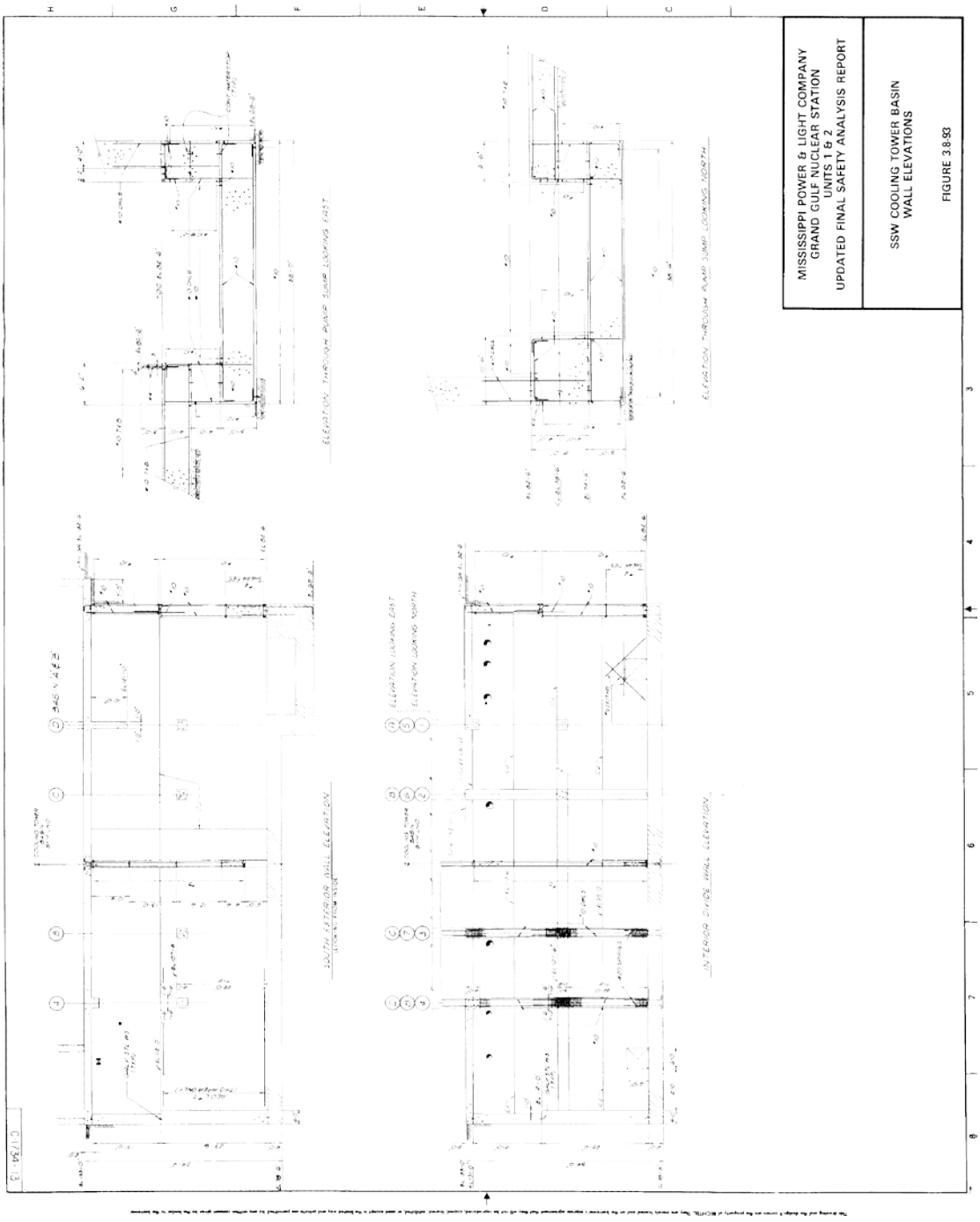
MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT
TYPICAL MANHOLE
AND
DUCT BANK
PLAN AND SECTIONS
FIGURE 3.8-88
FIN GRADE
EL VARIES
STIRRUPS
#4
#4 TOP & BOT
VARIES
T/DUCT
EL VARIES
VARIES
TYPICAL DUCT BANK
REINFORCING
MH COVER
LIFTING LUG
TYP 4 PLACES
VARIES
EQ
EQ
EQ
1'-0"
#6 TOP & BOT
1'-6"
1'-6"
SUMP
VARIES
TYPICAL MANHOLE
PLAN
MANHOLE FRAME
& COVER
ADDL REINF
#6
TYP DUCT BANK &
TRENCH BLOCKOUT
VARIES
FL EL VARIES
#6 TOP & BOT
℄ MH
#6 EL VARIES
SLOPE
ADDL BARS
EACH SIDE
#6
MAT'L.
COMP.
VARIES
1'-10"
SECTION
A
—

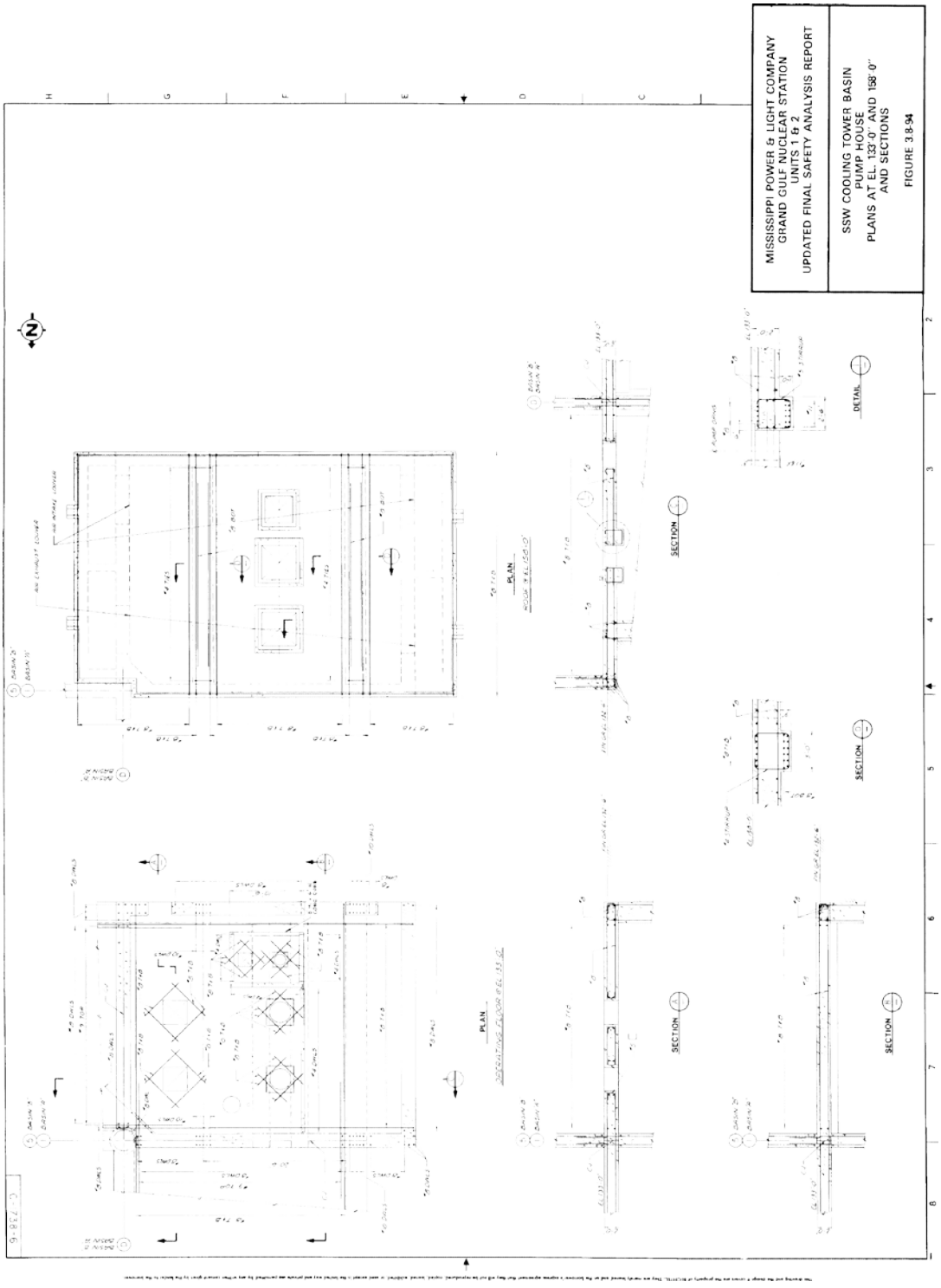

MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT
SSW COOLING TOWER BASIN
BASE MAT PLAN AND SECTIONS
FIGURE 3.8-89
SECTION
SECTION
SECTION

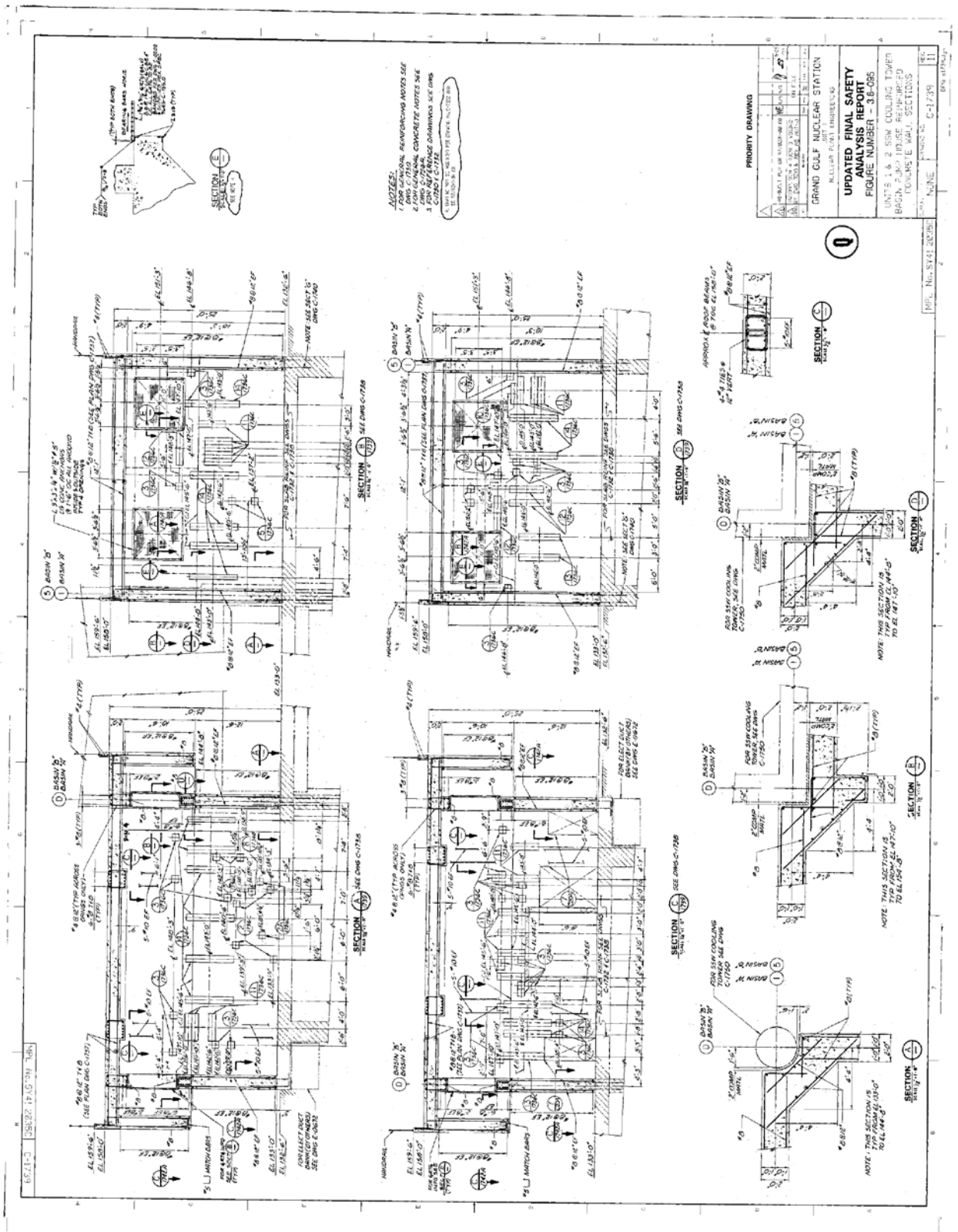

MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT
SSW COOLING TOWER BASIN
PLAN AT EL. 107'-6"
FIGURE 3.8-90
DETAIL
SECTION
N
DETAIL
DETAIL
DETAIL

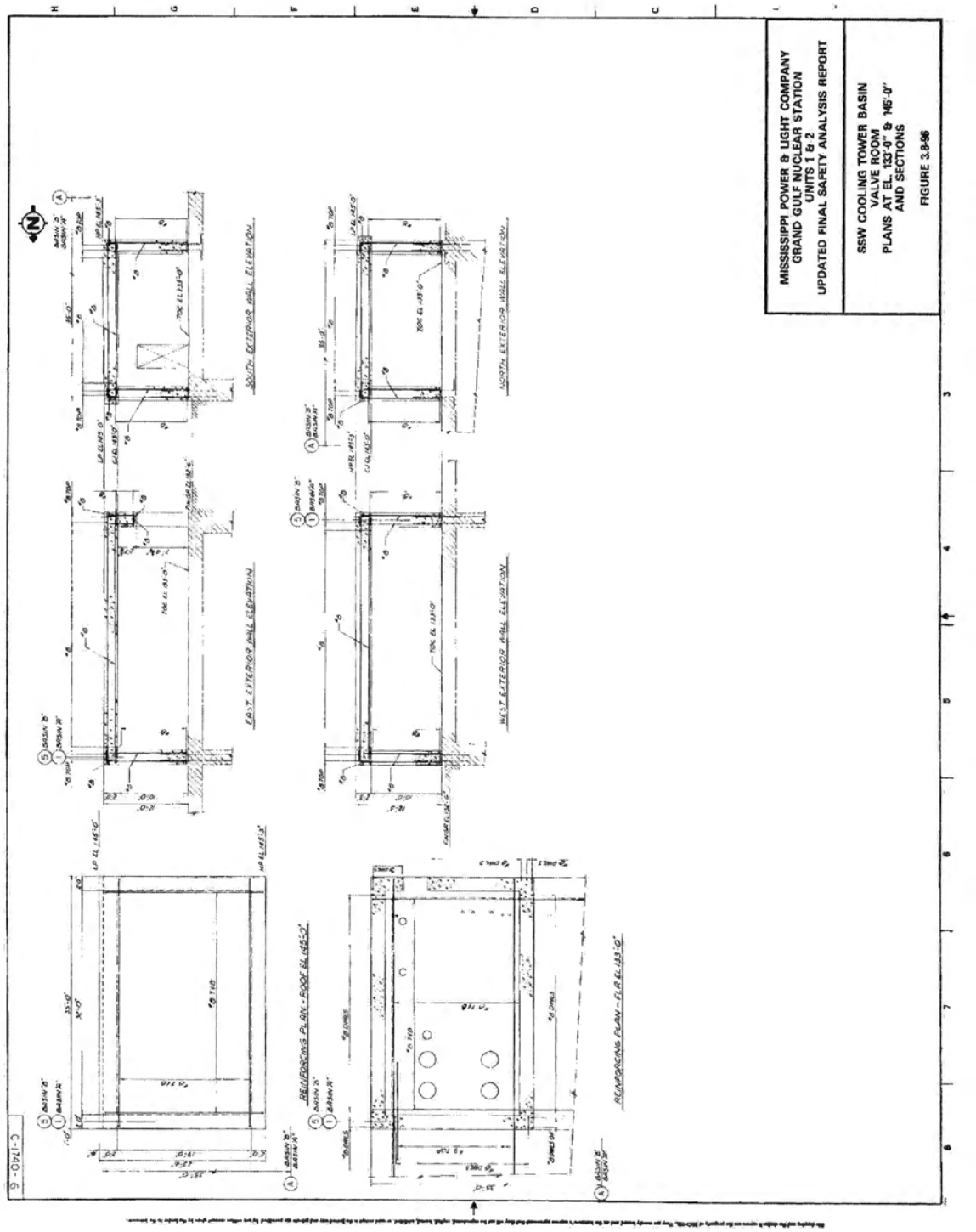

MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT
SSW COOLING TOWER BASIN
ROOF SLAB PLAN AT EL. 133'-0"
FIGURE 3.8.91
SECTION
SECTION
DETAIL
SECTION
SECTION
SECTION
N

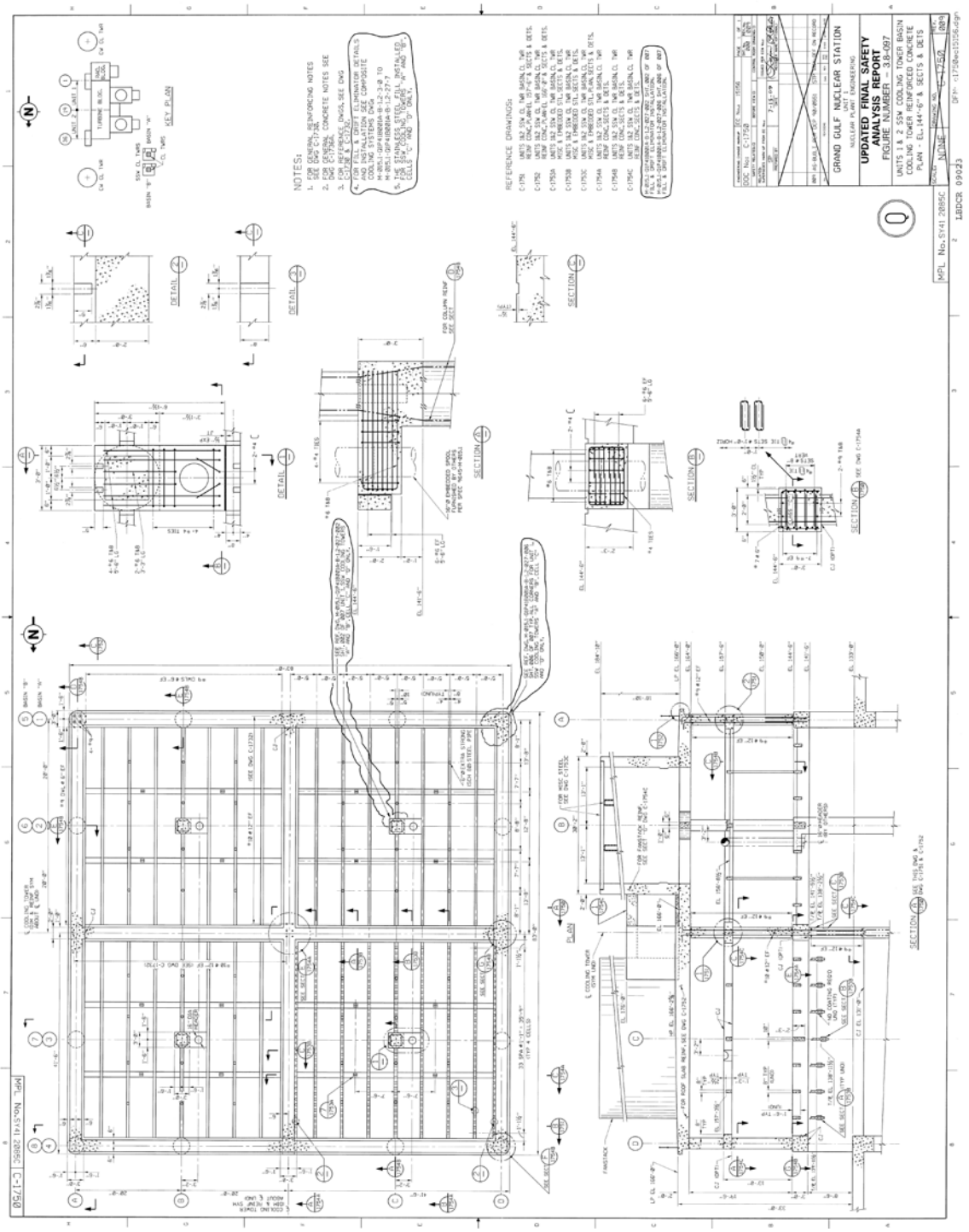

MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT
SSW COOLING TOWER BASIN
WALL ELEVATIONS AND DETAILS
FIGURE 3.8-92
LEGEND
OUTSIDE WALL CORNER
OUTSIDE WALL CORNER
INTERIOR WALL TO EXTERIOR WALL INTERSECTION
INTERIOR WALL TO EXTERIOR WALL INTERSECTION
EAST EXTERIOR WALL ELEVATION
WEST EXTERIOR WALL ELEVATION
NORTH EXTERIOR WALL ELEVATION

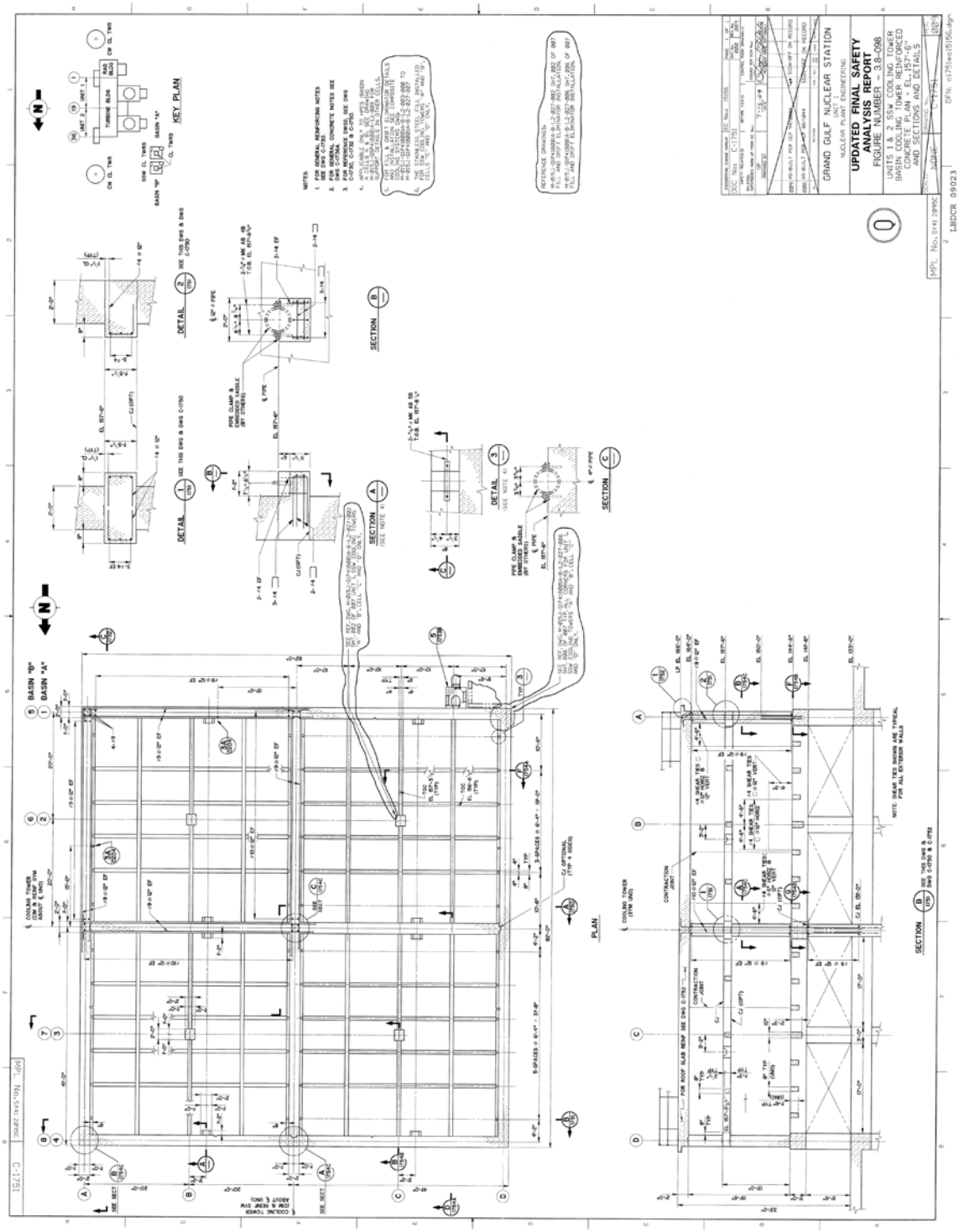

MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT
SSW COOLING TOWER BASIN
WALL ELEVATIONS
FIGURE 3.8-93
ELEVATION THROUGH PUMP SUMP LOOKING EAST
ELEVATION THROUGH PUMP SUMP LOOKING NORTH
SOUTH EXTERIOR WALL ELEVATION
ELEVATION LOOKING EAST
ELEVATION LOOKING NORTH
INTERIOR DIVIDE WALL ELEVATION

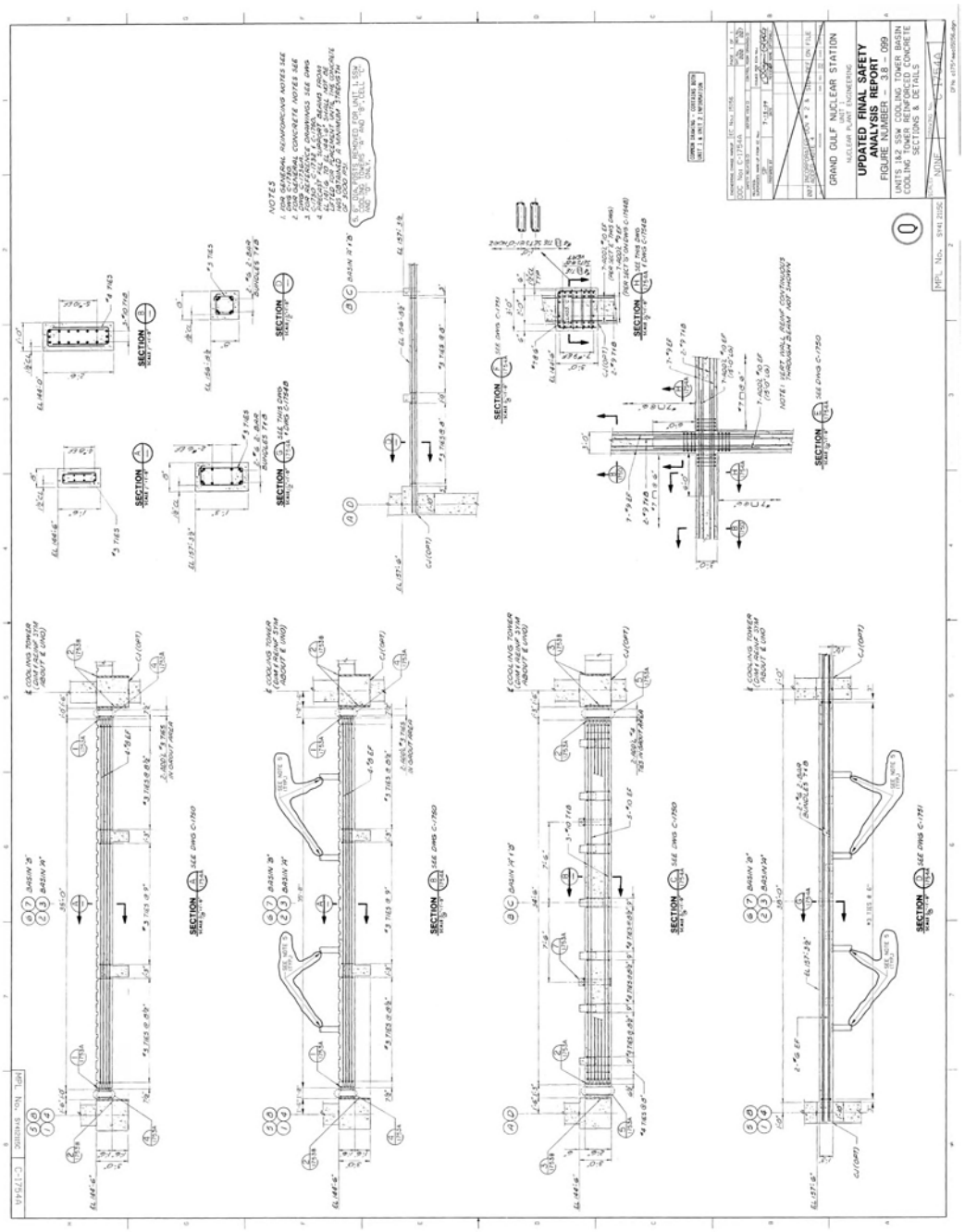

MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT
SSW COOLING TOWER BASIN
PUMP HOUSE
PLANS AT EL 133'-0" AND 158'-0"
AND SECTIONS
FIGURE 3.8-94
PLAN
SECTION
DETAIL

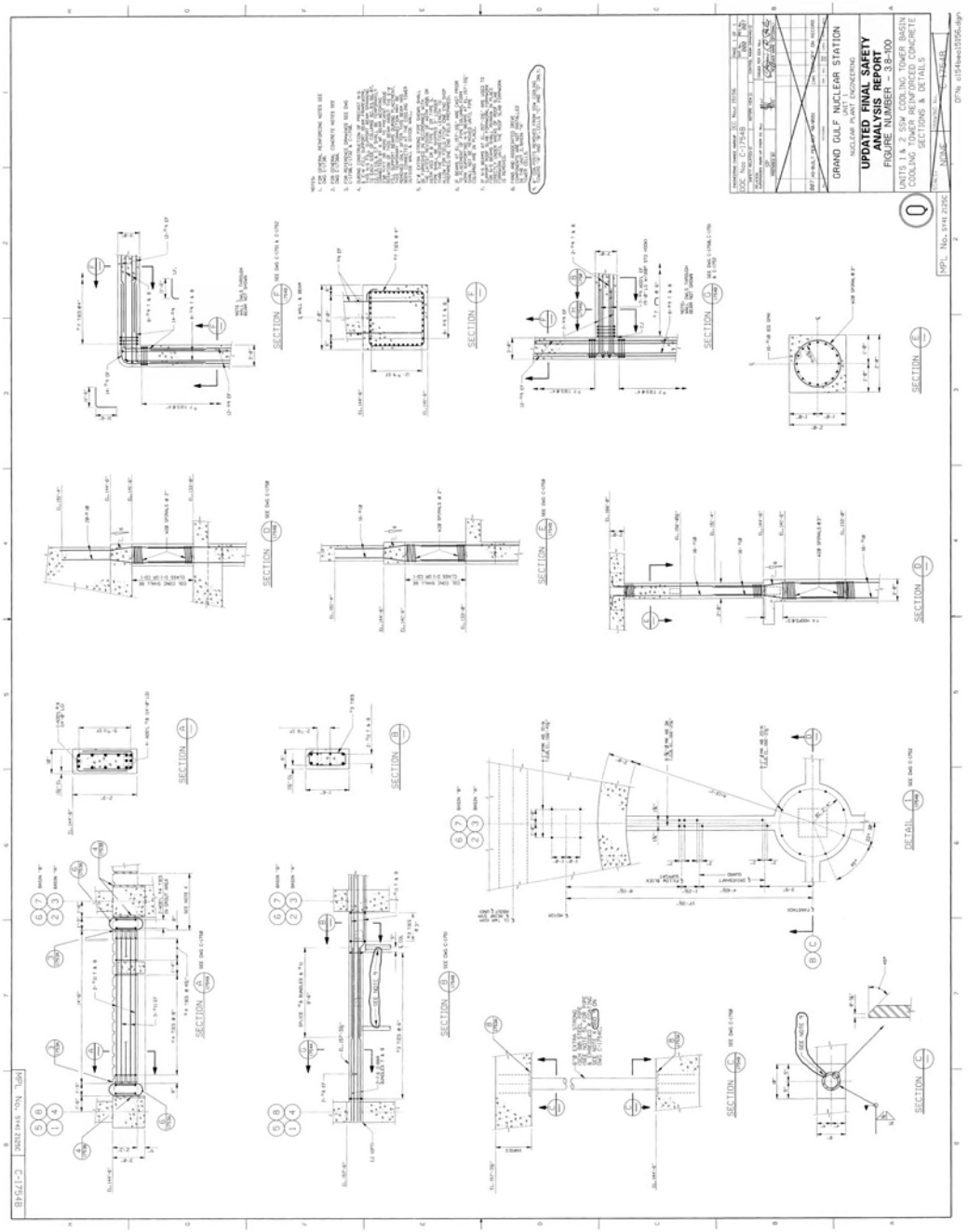

PRIORITY DRAWING
GRAND GULF NUCLEAR STATION
UNIT 1 & 2 SSW COOLING TOWER BASIN PUMP HOUSE REINFORCED CONCRETE WALL SECTIONS
UPDATED FINAL SAFETY ANALYSIS REPORT
FIGURE NUMBER - 3.8-085
NOTES:
SECTION
BASIN 'D'
BASIN 'M'
BASIN 'R'

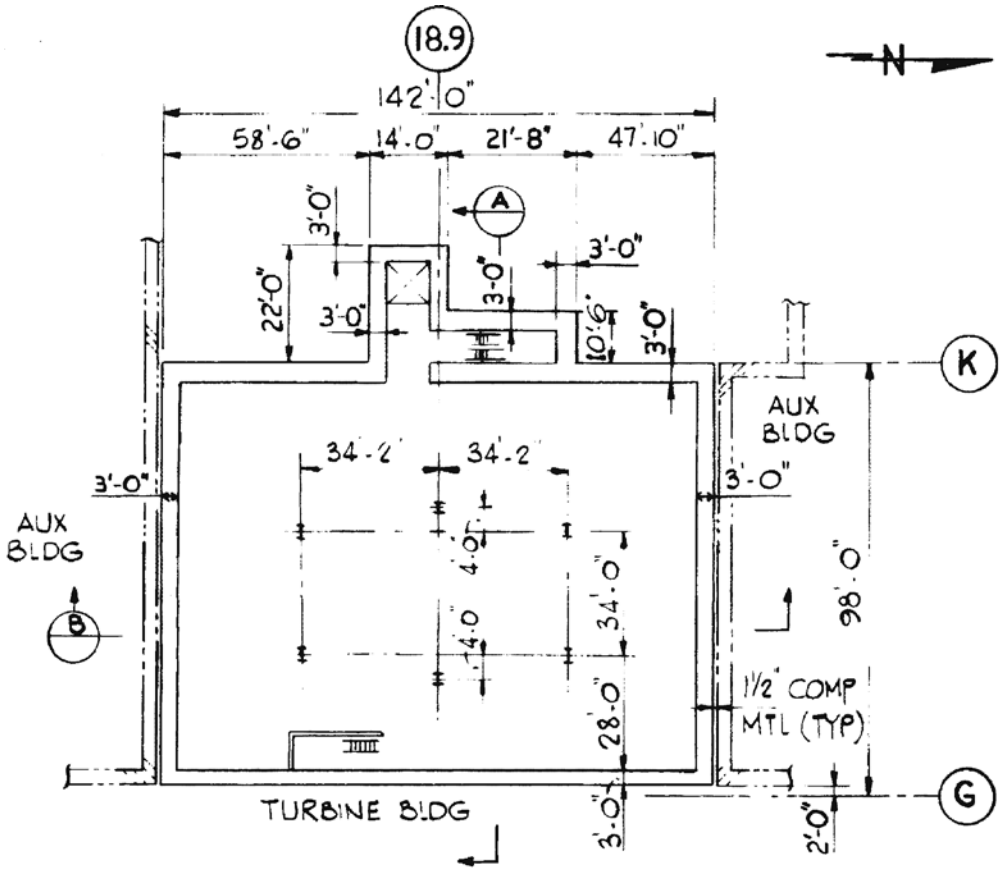

MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT
SSW COOLING TOWER BASIN
VALVE ROOM
PLANS AT EL 133'-0" & 145'-0"
AND SECTIONS
FIGURE 3.8-96
SOUTH EXTERIOR WALL ELEVATION
NORTH EXTERIOR WALL ELEVATION
EAST EXTERIOR WALL ELEVATION
WEST EXTERIOR WALL ELEVATION
REINFORCING PLAN - ROOF EL 145'-2"
REINFORCING PLAN - FLR EL 133'-0"


UPDATED FINAL SAFETY ANALYSIS REPORT
FIGURE NUMBER 3.8-97
UNITS 1 & 2 SSW COOLING TOWER BASIN
COOLING TOWER REINFORCED CONCRETE
PLAN - EL. 144'-6" & SECTS & DETS
GRAND GULF NUCLEAR STATION
NOTES:
REFERENCE DRAWINGS
KEY PLAN
SECTION
DETAIL


KEY PLAN
PLAN
SECTION
DETAIL
GRAND GULF NUCLEAR STATION
UPDATED FINAL SAFETY ANALYSIS REPORT
FIGURE NUMBER 3.8-108
UNITS 1 & 2 SSW COOLING TOWER BASIN COOLING TOWER REINFORCED CONCRETE PLAN - EL 157'-6" AND SECTIONS AND DETAILS

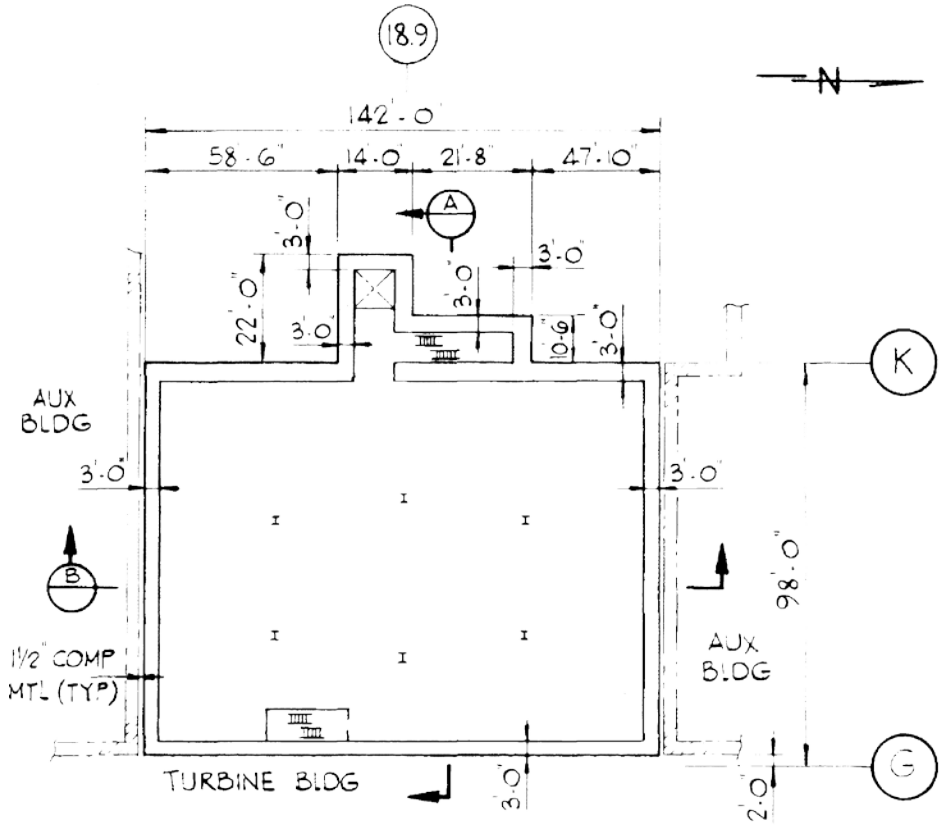

UPDATED FINAL SAFETY ANALYSIS REPORT
FIGURE NUMBER - 3.8 - 099
UNITS 1&2 SSW COOLING TOWER BASIN COOLING TOWER REINFORCED CONCRETE SECTIONS & DETAILS
GRAND GULF NUCLEAR STATION
NOTES


GRAND GULF NUCLEAR STATION
NUCLEAR PLANT ENGINEERING
UPDATED FINAL SAFETY ANALYSIS REPORT
FIGURE NUMBER - 3.8-100
UNITS 1 & 2 SGW COOLING TOWER BASIN
COOLING TOWER REINFORCED CONCRETE
SECTIONS & DETAILS
SECTION
DETAIL



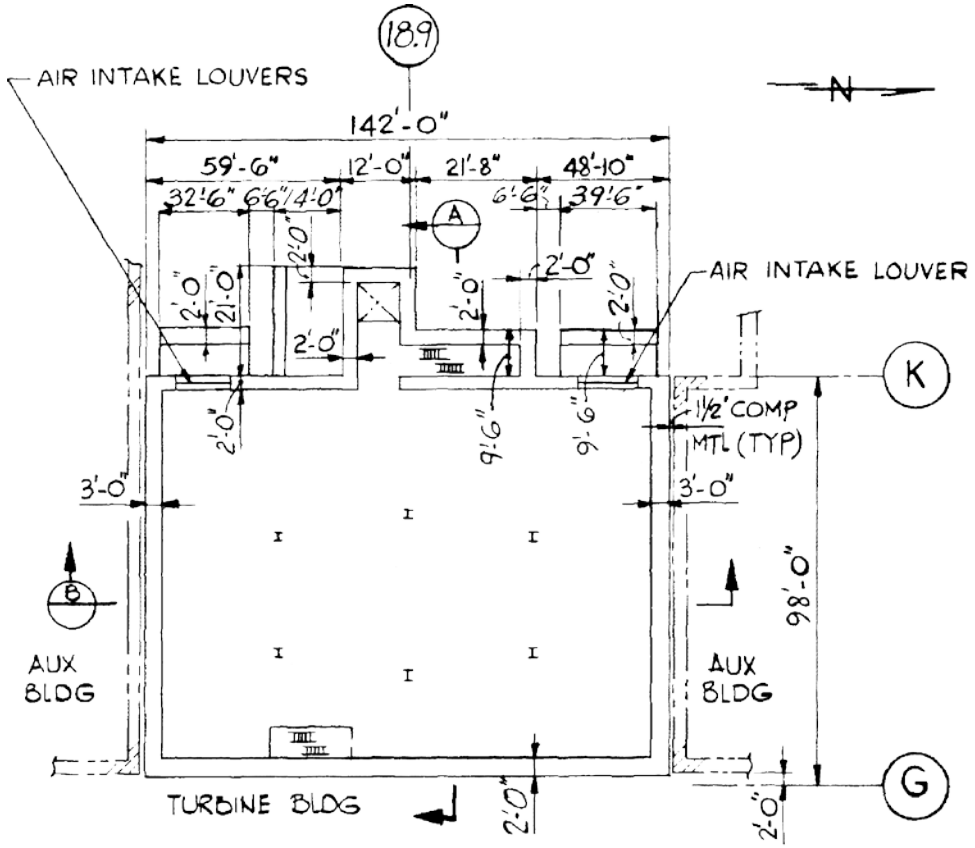
FOUNDATION PLAN
EL 93'0"

MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT
CONTROL BLDG. BASE MAT AT EL. 93'-0" FIGURE 3.8-101



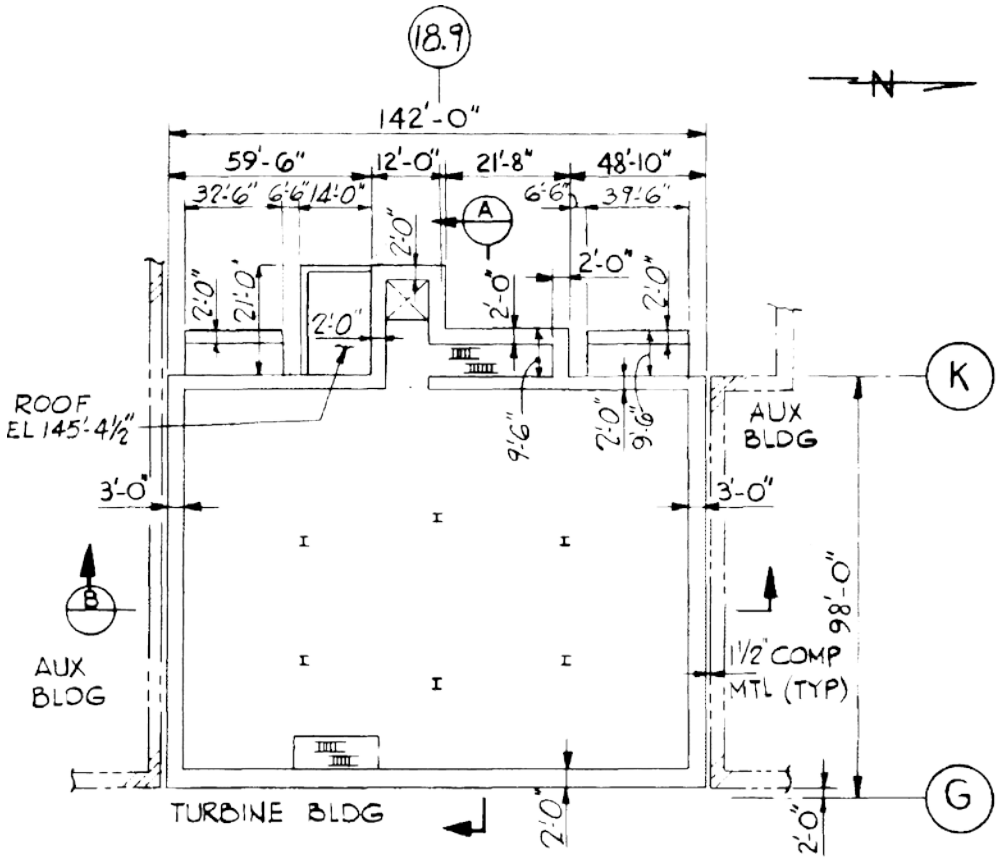
FLOOR PLAN
EL 111'-0"

MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT
CONTROL BLDG. FLOOR PLAN AT EL. 111'-0" FIGURE 3.8-102



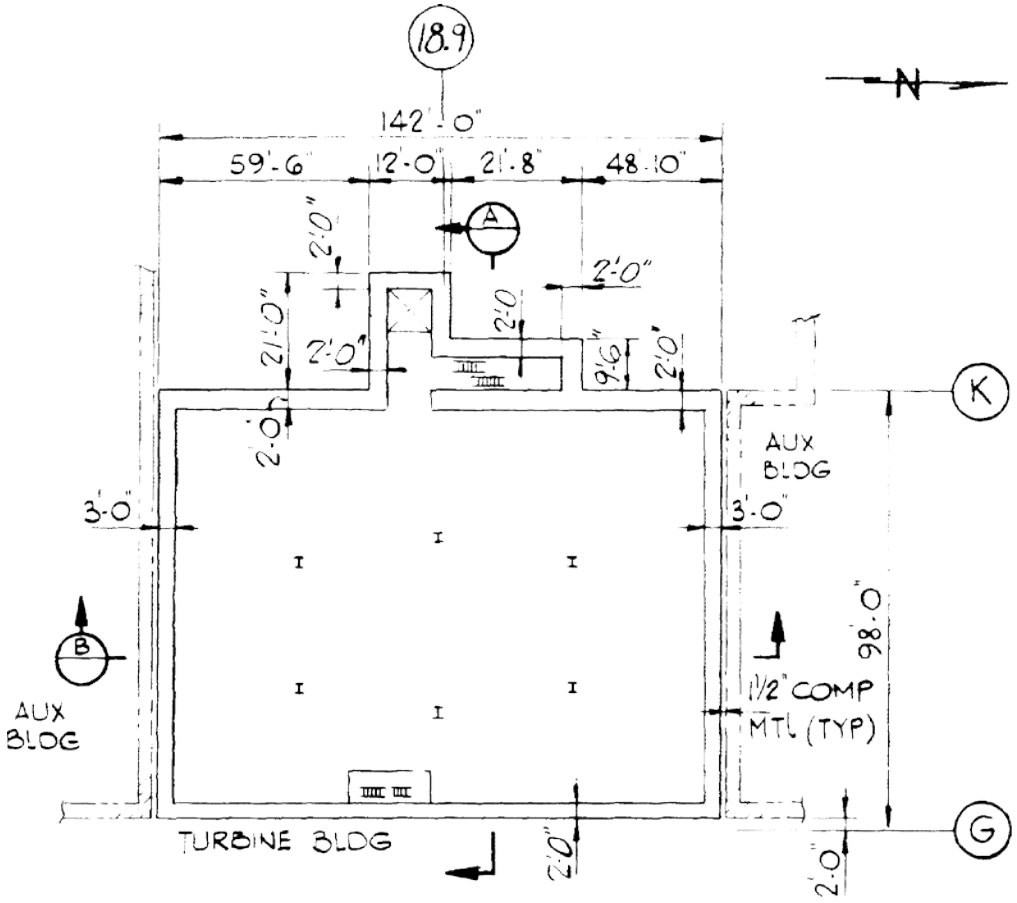
FLOOR PLAN
EL 133'-0"

MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT
CONTROL BLDG. FLOOR PLAN AT EL. 133'-0" FIGURE 3.8-103



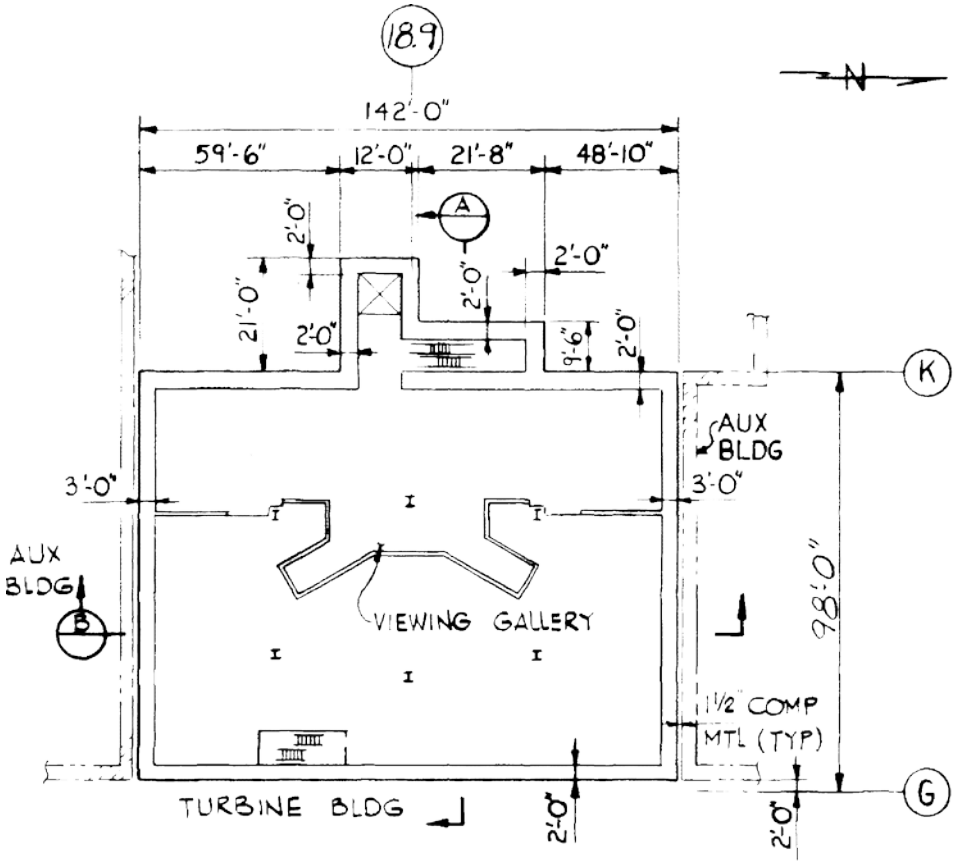
FLOOR PLAN
EL 148'-0"

MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT
CONTROL BLDG. FLOOR PLAN AT EL. 148'-0" FIGURE 3.8-104



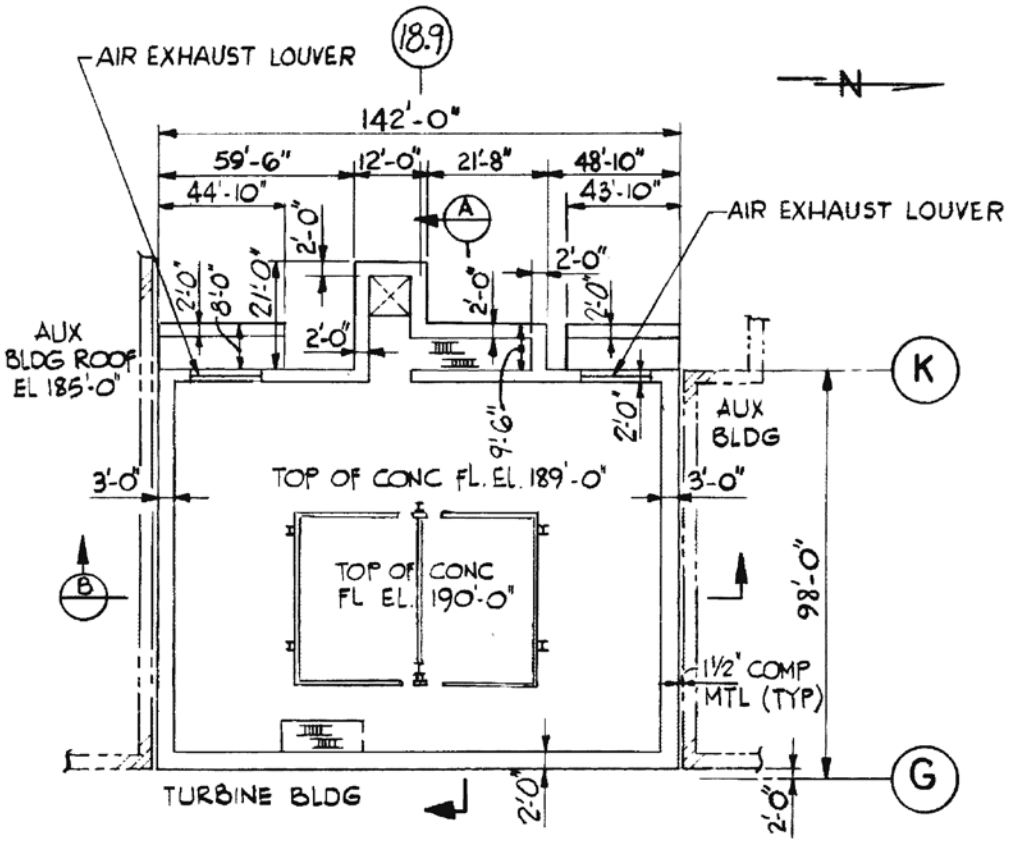
FLOOR PLAN
EL. 166'-0"

MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT
CONTROL BLDG. FLOOR PLAN AT EL. 166'-0" FIGURE 3.8-105



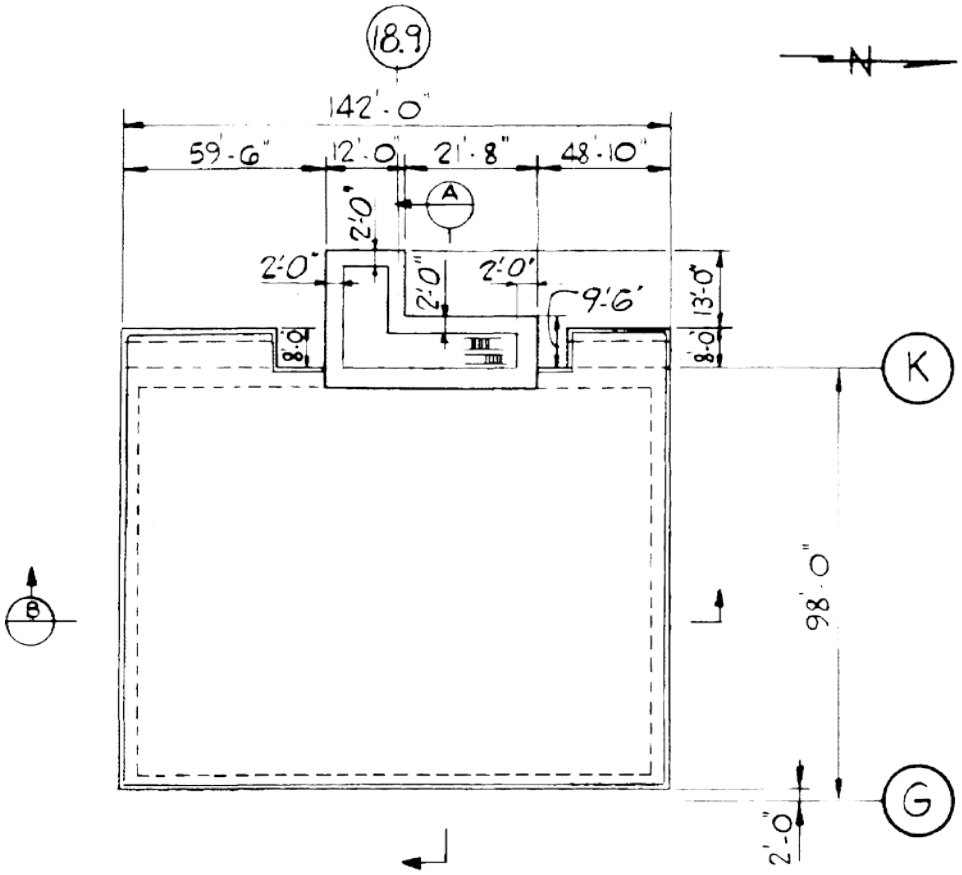
FLOOR PLAN
EL 177'-0"

MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT
CONTROL BLDG. FLOOR PLAN AT EL. 177'-0" FIGURE 3.8-106



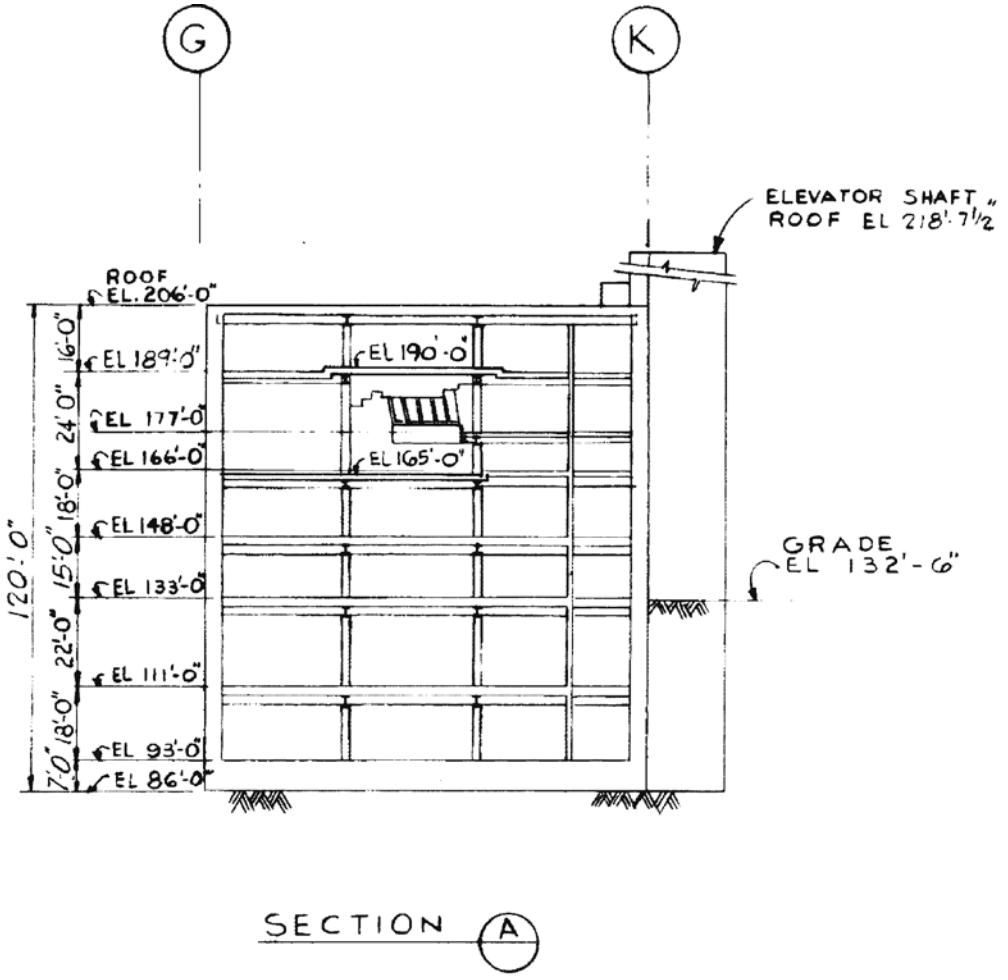
FLOOR PLAN
EL 189'-0"

MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT
CONTROL BLDG. FLOOR PLAN AT EL. 189'-0" FIGURE 3.8-107



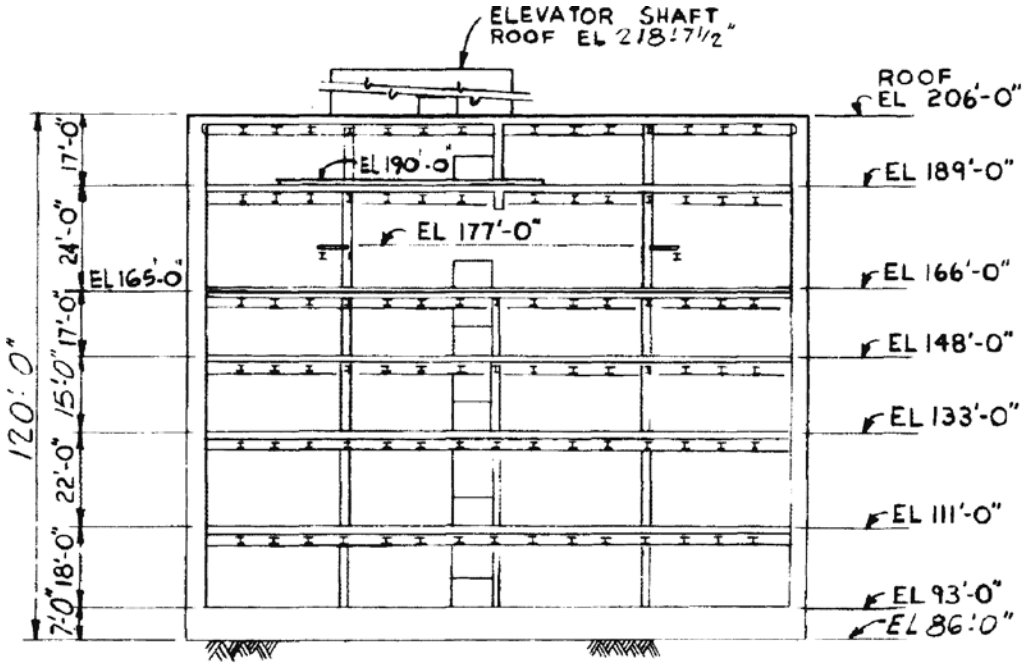
ROOF PLAN
EL 206'-0"

MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT
CONTROL BLDG. ROOF PLAN AT EL. 206'-0" FIGURE 3.8-108



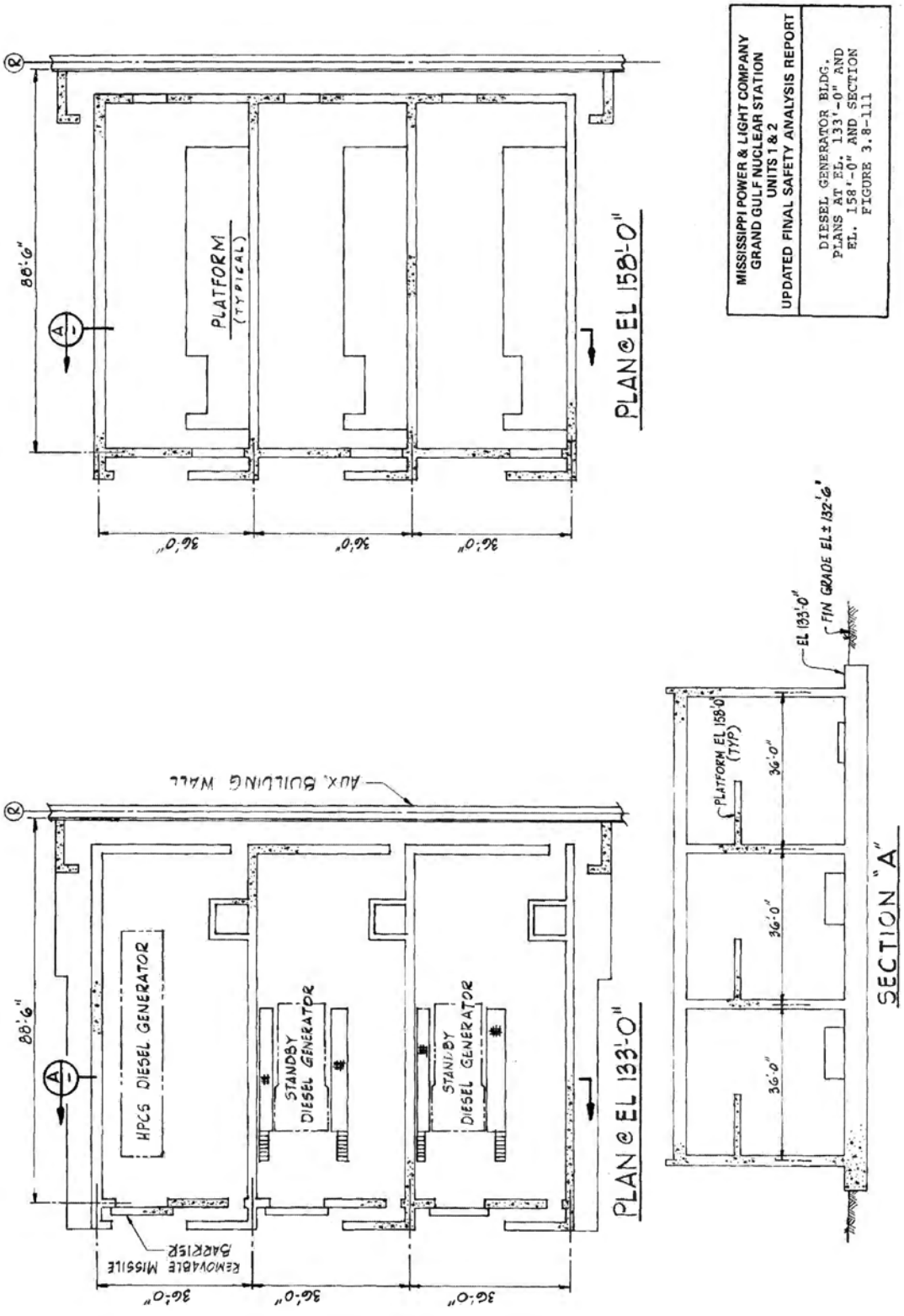
SECTION A

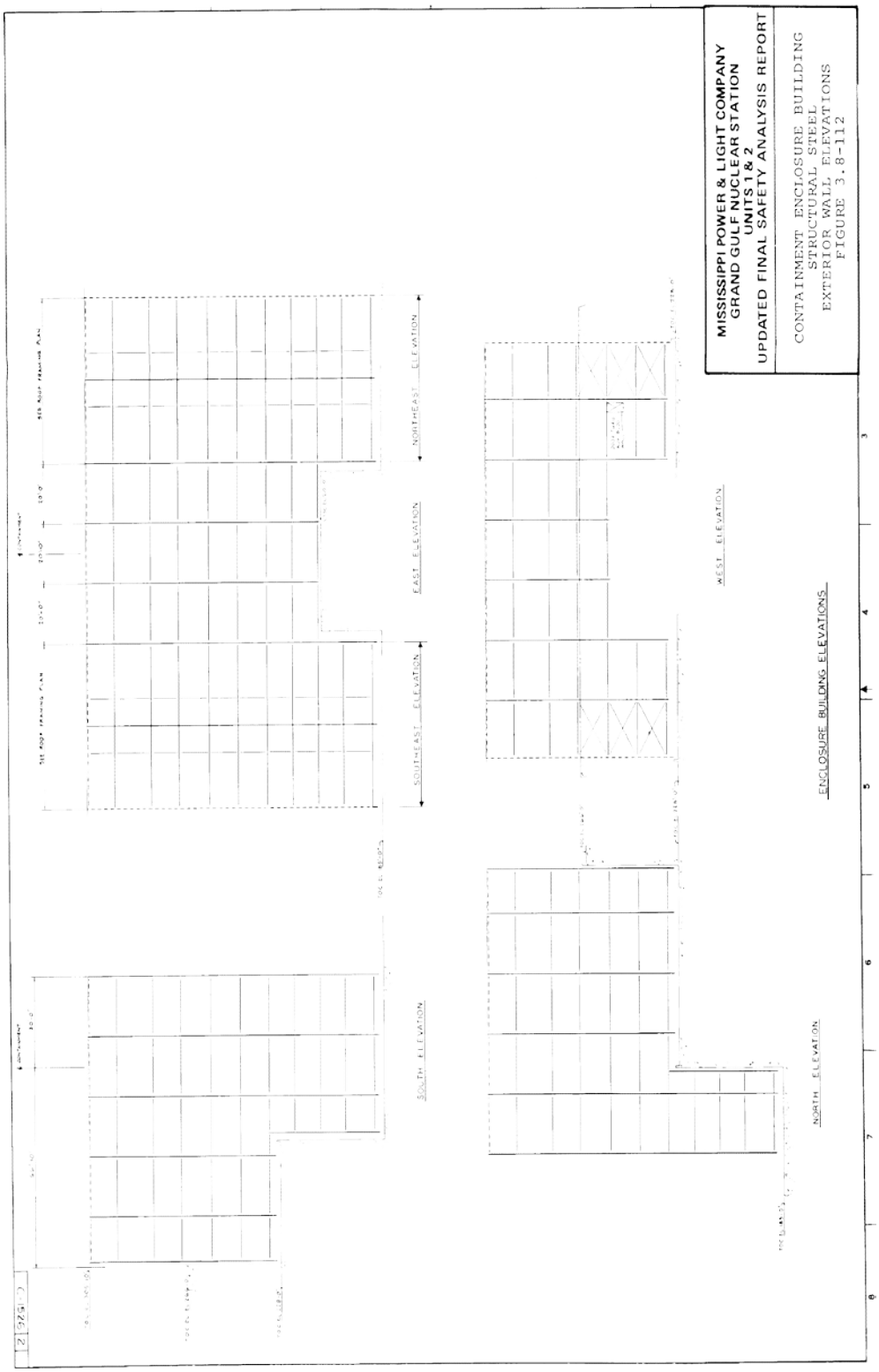
MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT
CONTROL BLDG. SECTION A FIGURE 3.8-109

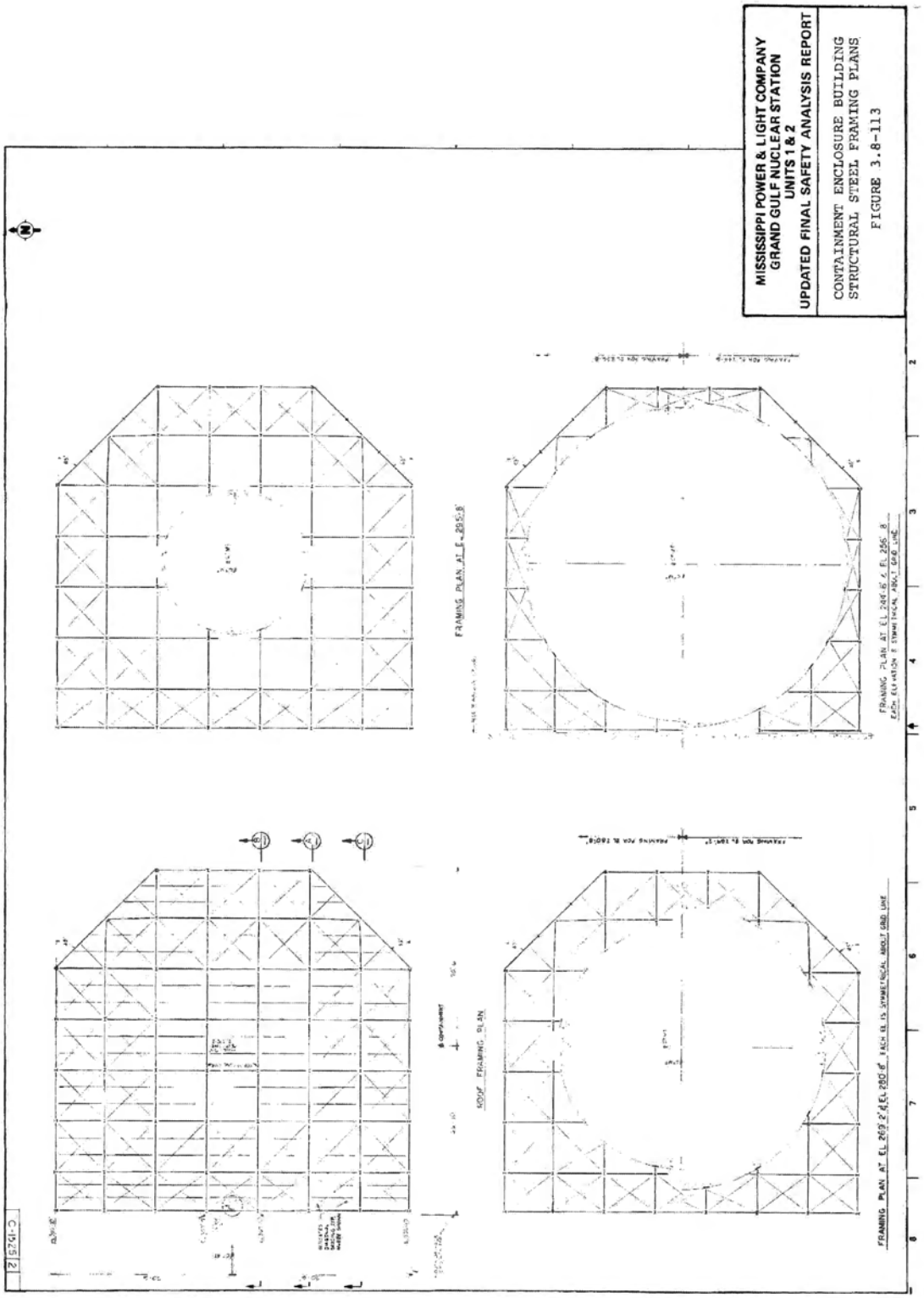


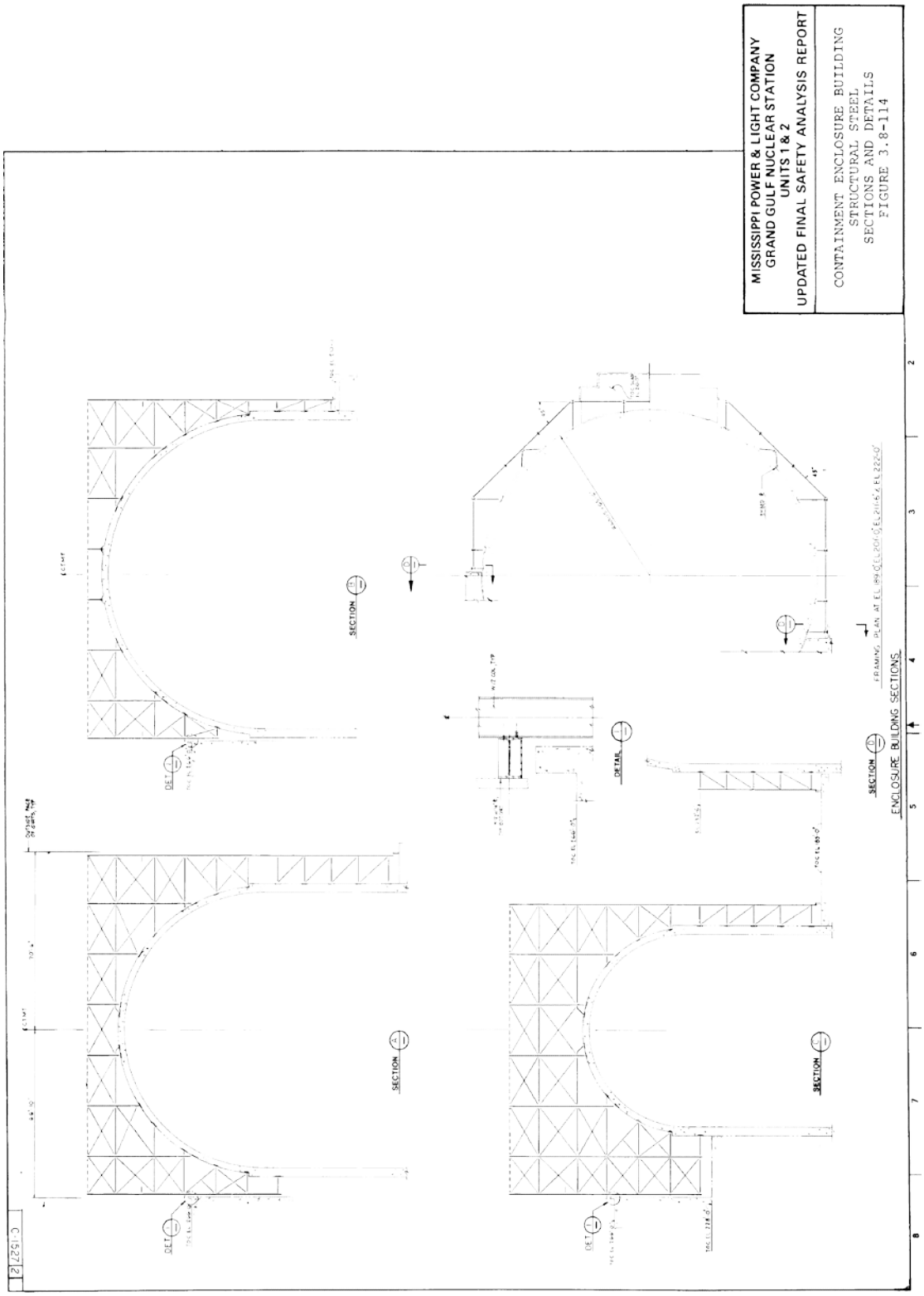
SECTION B

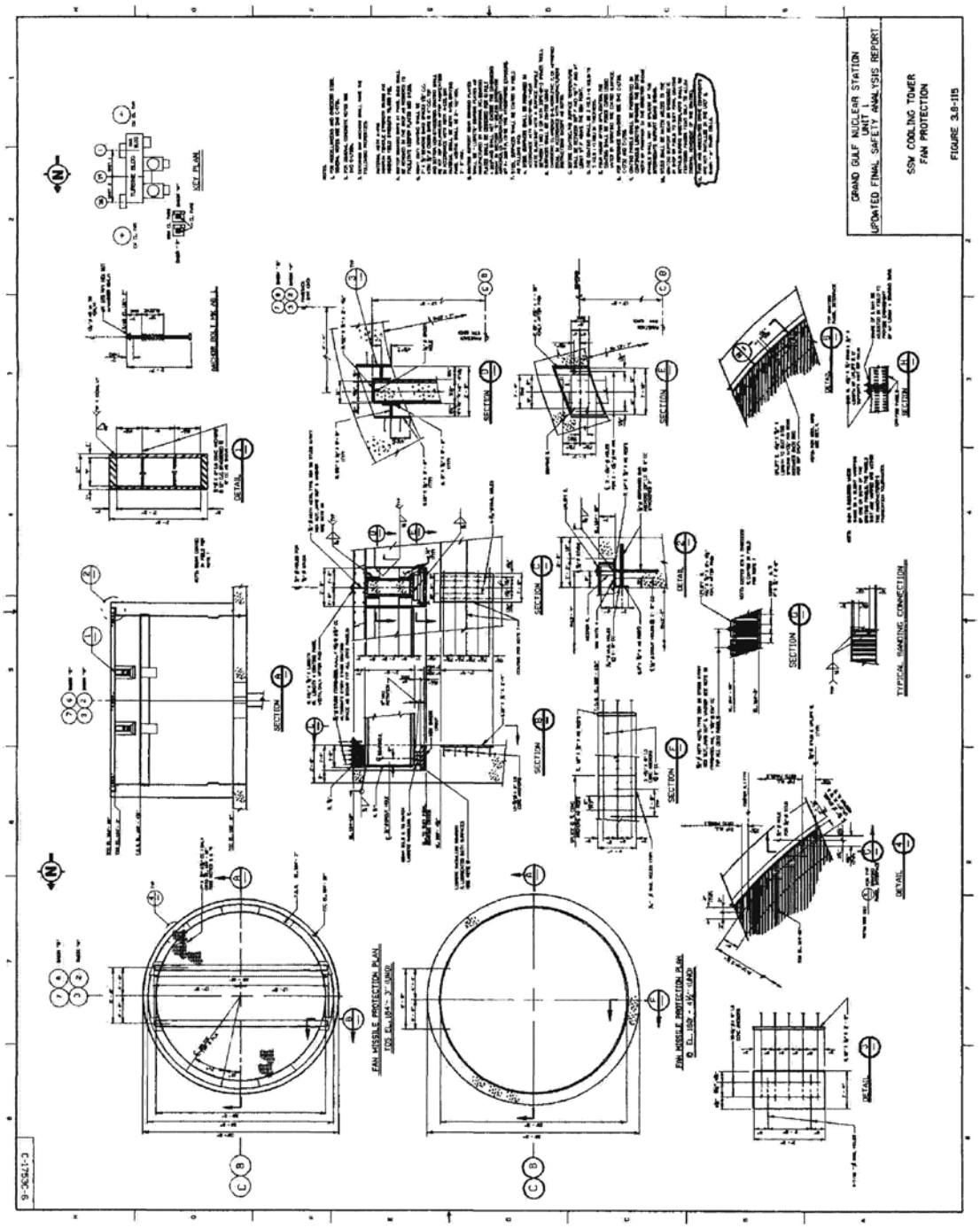
MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT
CONTROL BLDG. SECTION B FIGURE 3.8-110

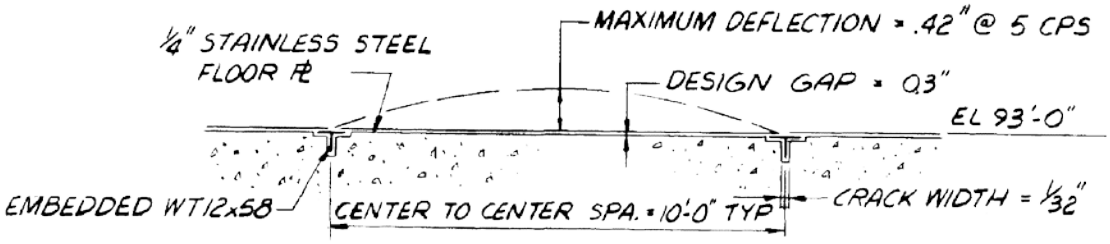

MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT
DIESEL GENERATOR BLDG.
PLANS AT EL. 133'-0" AND
EL. 158'-0" AND SECTION
FIGURE 3.8-111
PLATFORM (TYPICAL)
PLAN @ EL 158'-0"
PLAN @ EL 133'-0"
AUX. BUILDING WALL
HPCS DIESEL GENERATOR
STANDBY DIESEL GENERATOR
STANDBY DIESEL GENERATOR
REMOVABLE MISSILE BARRIER
88'-6"
36'-0"
SECTION "A"
EL 133'-0"
FIN GRADE EL ± 132'-6"
PLATFORM EL 158'-0" (TYP)


MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT
CONTAINMENT ENCLOSURE BUILDING
STRUCTURAL STEEL
EXTERIOR WALL ELEVATIONS
FIGURE 3.8-112
NORTHEAST ELEVATION
EAST ELEVATION
SOUTHEAST ELEVATION
SOUTH ELEVATION
WEST ELEVATION
NORTH ELEVATION
ENCLOSURE BUILDING ELEVATIONS


MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT
CONTAINMENT ENCLOSURE BUILDING
STRUCTURAL STEEL FRAMING PLANS
FIGURE 3.8-113
FRAMING PLAN AT EL 285'-8"
FRAMING PLAN AT EL 244'-6" & EL 256'-8"
ROOF FRAMING PLAN
FRAMING PLAN AT EL 269'-2" & EL 280'-0"

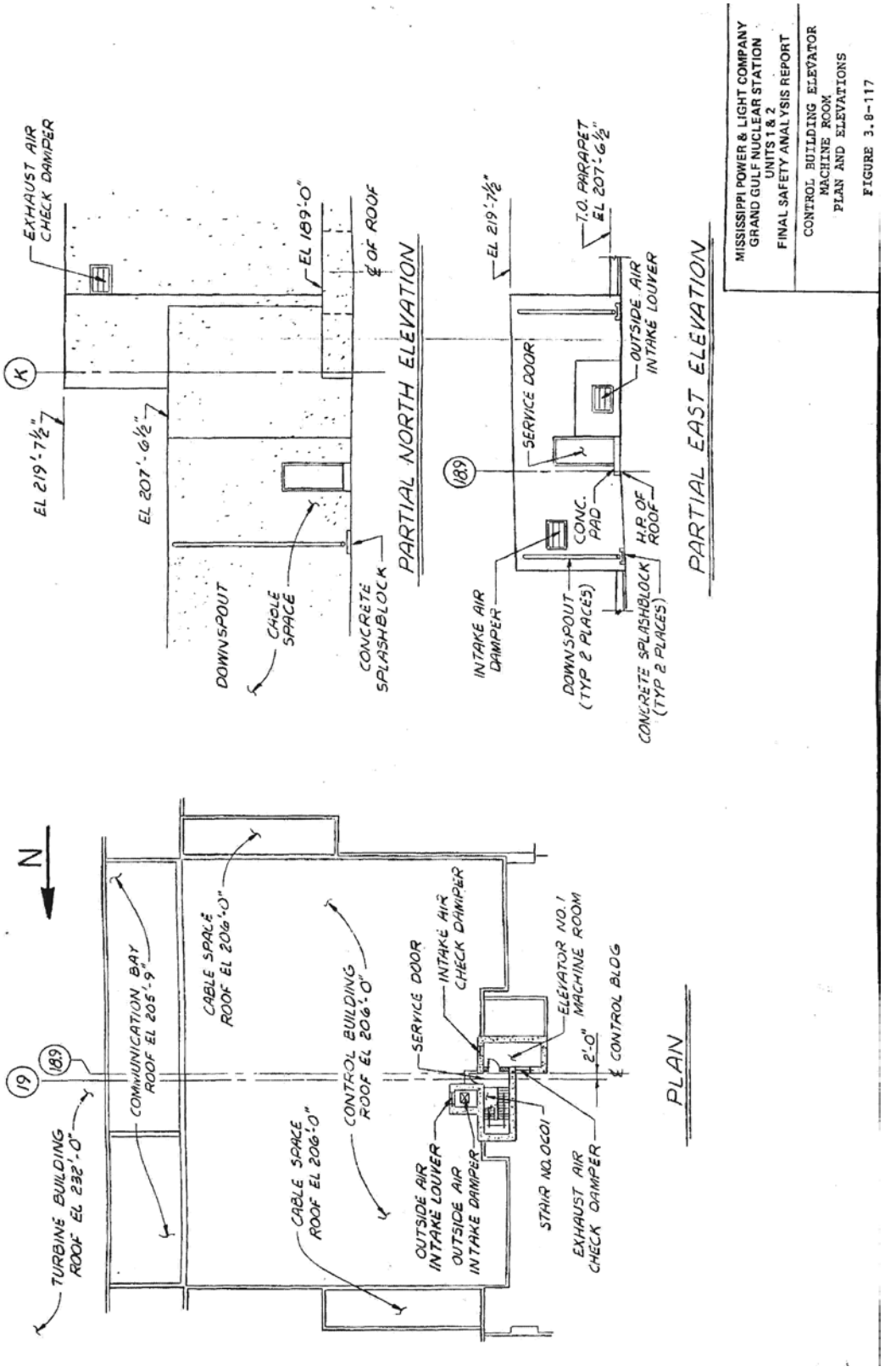

MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT
CONTAINMENT ENCLOSURE BUILDING
STRUCTURAL STEEL
SECTIONS AND DETAILS
FIGURE 3.8-114
SECTION
DETAIL
ENCLOSURE BUILDING SECTIONS

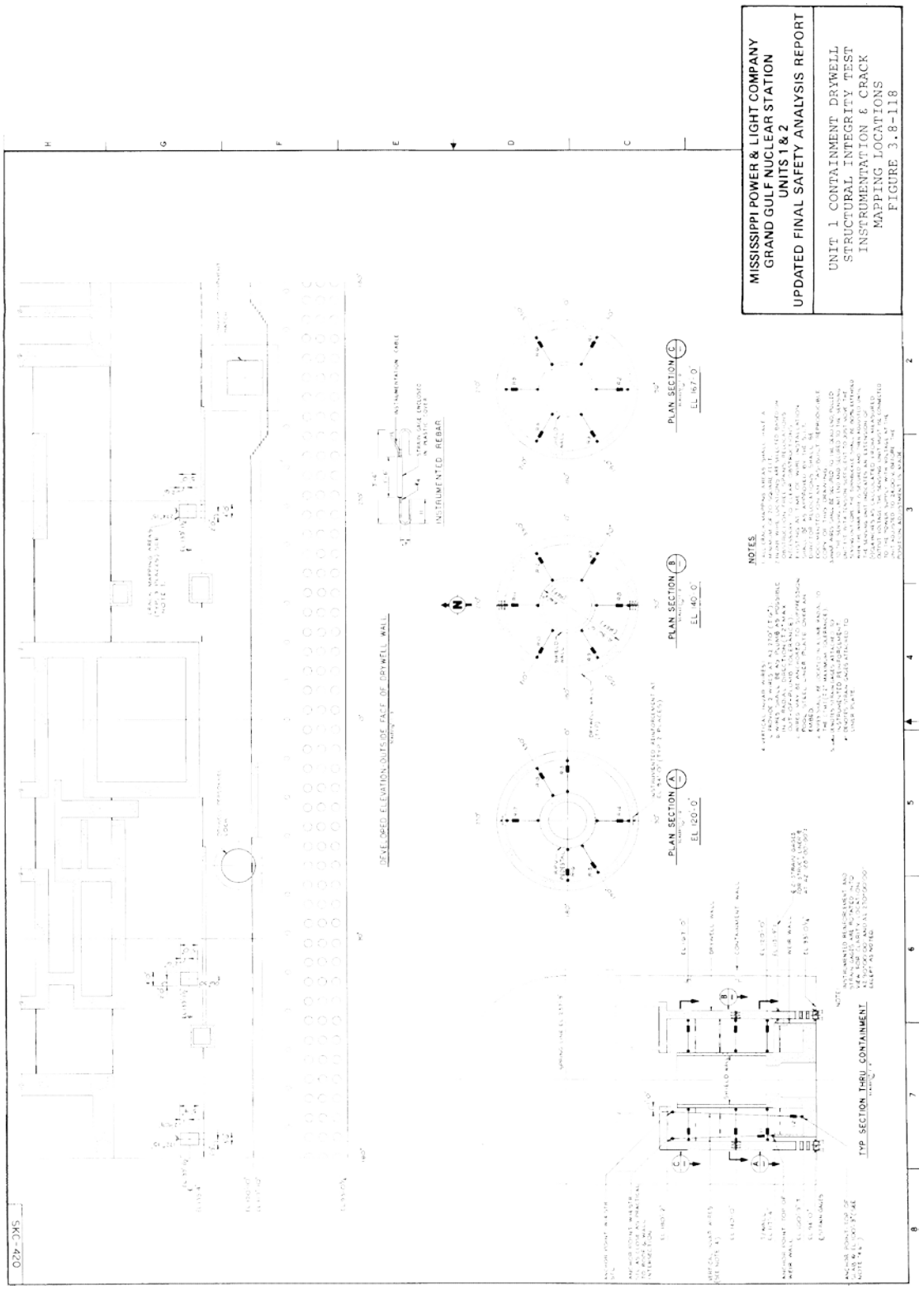

GRAND GULF NUCLEAR STATION
UNIT 1
UFSAR FINAL SAFETY ANALYSIS REPORT
SSW COOLING TOWER
FAN PROTECTION
FIGURE 3.8-115

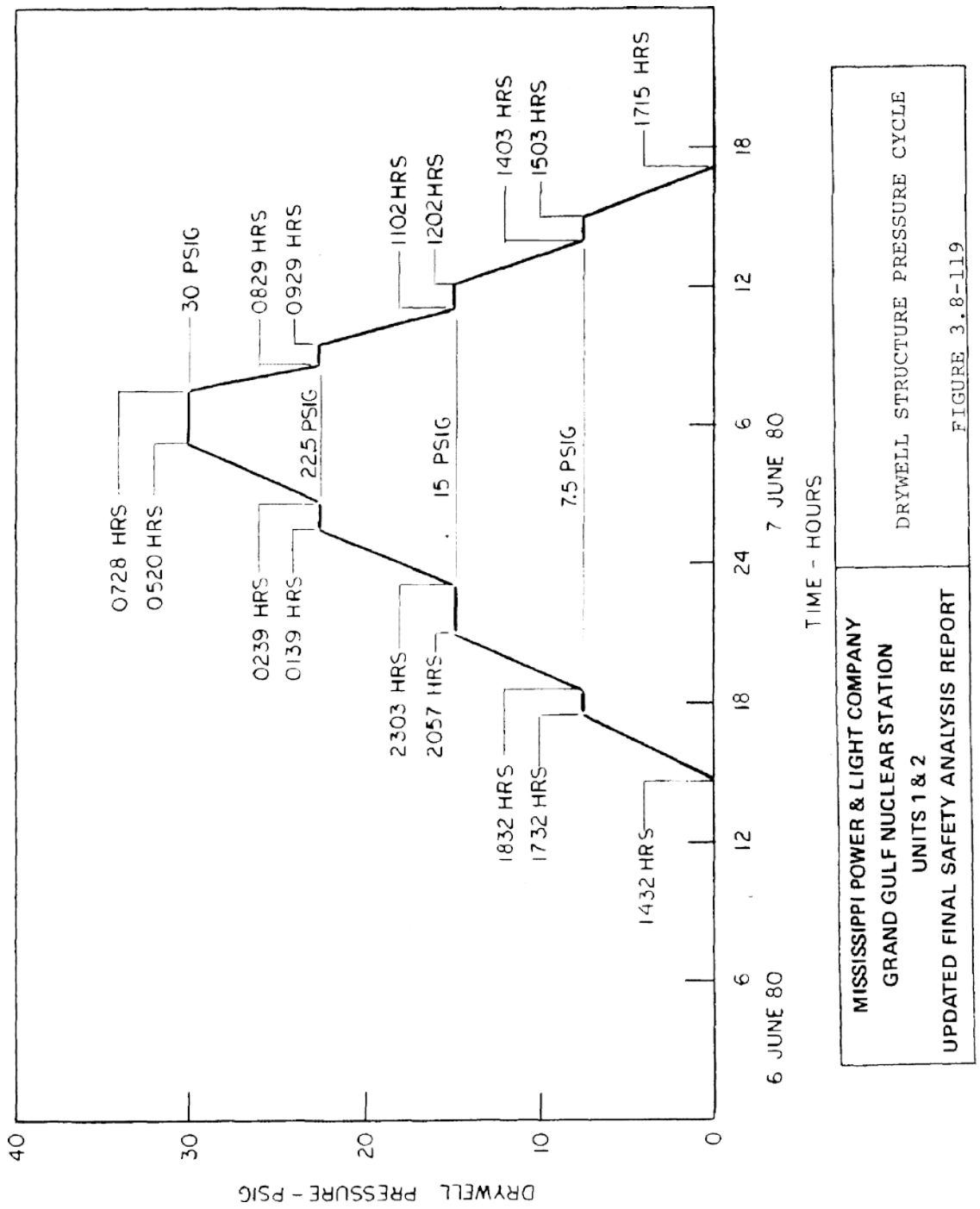


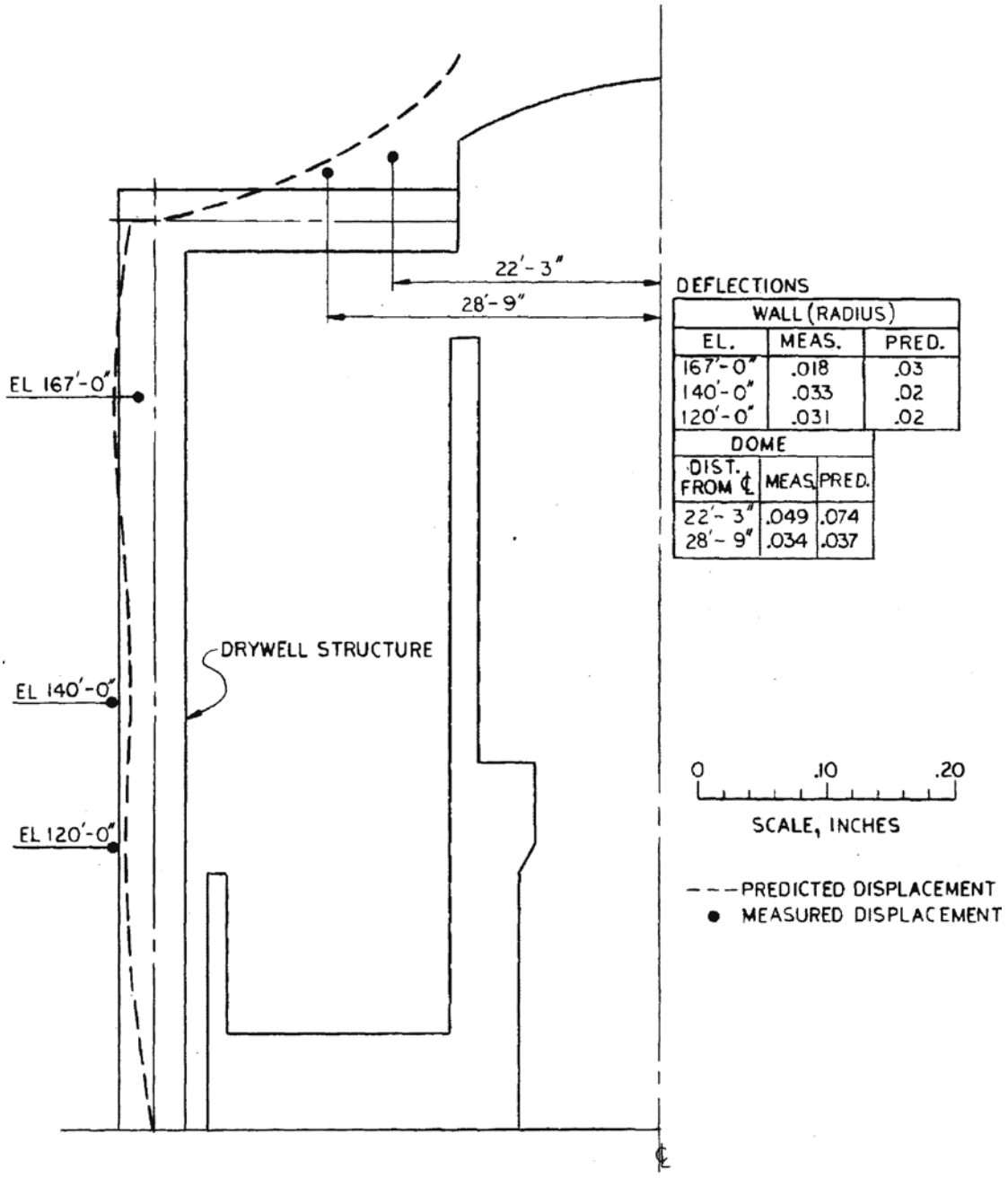
TYPICAL SECTION - SUPPRESSION
POOL FLOOR PLATES
NTS

MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT
SUPPRESSION POOL FLOOR PLATE NEGATIVE PRESSURE ANALYSIS FIGURE 3.8-116


EXHAUST AIR CHECK DAMPER
EL 219'-7½"
EL 207'-6½"
EL 189'-0"
℄ OF ROOF
PARTIAL NORTH ELEVATION
DOWNSPOUT
CABLE SPACE
CONCRETE SPLASHBLOCK
EL 219'-7½"
T.O. PARAPET EL 207'-6½"
SERVICE DOOR
OUTSIDE AIR INTAKE LOUVER
INTAKE AIR DAMPER
CONC. PAD
H.P. OF ROOF
DOWNSPOUT (TYP 2 PLACES)
CONCRETE SPLASHBLOCK (TYP 2 PLACES)
PARTIAL EAST ELEVATION
N
TURBINE BUILDING ROOF EL 232'-0"
COMMUNICATION BAY ROOF EL 205'-9"
CABLE SPACE ROOF EL 206'-0"
CONTROL BUILDING ROOF EL 206'-0"
CABLE SPACE ROOF EL 206'-0"
SERVICE DOOR
INTAKE AIR CHECK DAMPER
ELEVATOR NO.1 MACHINE ROOM
2'-0"
℄ CONTROL BLDG
OUTSIDE AIR INTAKE LOUVER
OUTSIDE AIR INTAKE DAMPER
STAIR NO. 0601
EXHAUST AIR CHECK DAMPER
PLAN
MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT
CONTROL BUILDING ELEVATOR MACHINE ROOM PLAN AND ELEVATIONS
FIGURE 3.8-117


MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT
UNIT 1 CONTAINMENT DRYWELL
STRUCTURAL INTEGRITY TEST
INSTRUMENTATION & CRACK
MAPPING LOCATIONS
FIGURE 3.8-118
NOTES
PLAN SECTION A
EL 120'-0"
PLAN SECTION B
EL 40'-0"
PLAN SECTION C
EL 147'-0"
TYP SECTION THRU CONTAINMENT
SKC-420

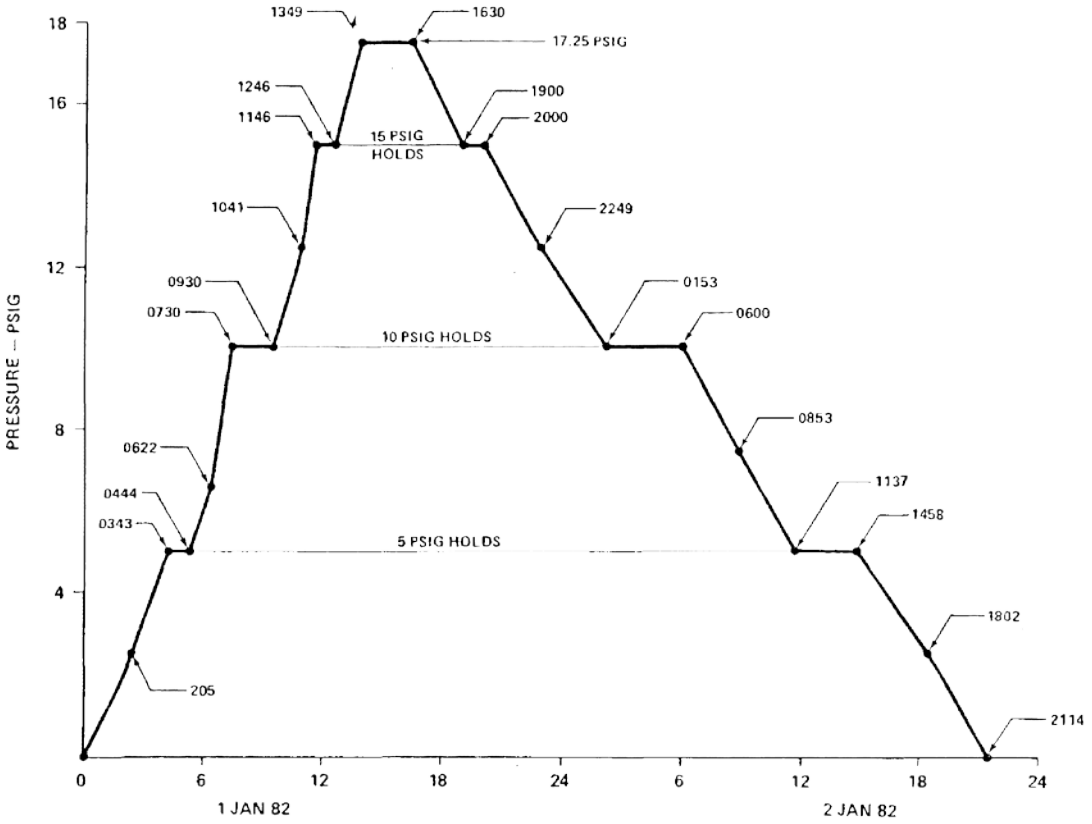

DRYWELL PRESSURE - PSIG
40
30
20
10
0
6 JUNE 80
12
18
24
7 JUNE 80
6
12
18
TIME - HOURS
0728 HRS
0520 HRS
30 PSIG
0829 HRS
0929 HRS
1102 HRS
1202 HRS
1403 HRS
1503 HRS
1715 HRS
0239 HRS
0139 HRS
225 PSIG
2303 HRS
2057 HRS
15 PSIG
1832 HRS
1732 HRS
7.5 PSIG
1432 HRS
MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT
DRYWELL STRUCTURE PRESSURE CYCLE
FIGURE 3.8-119



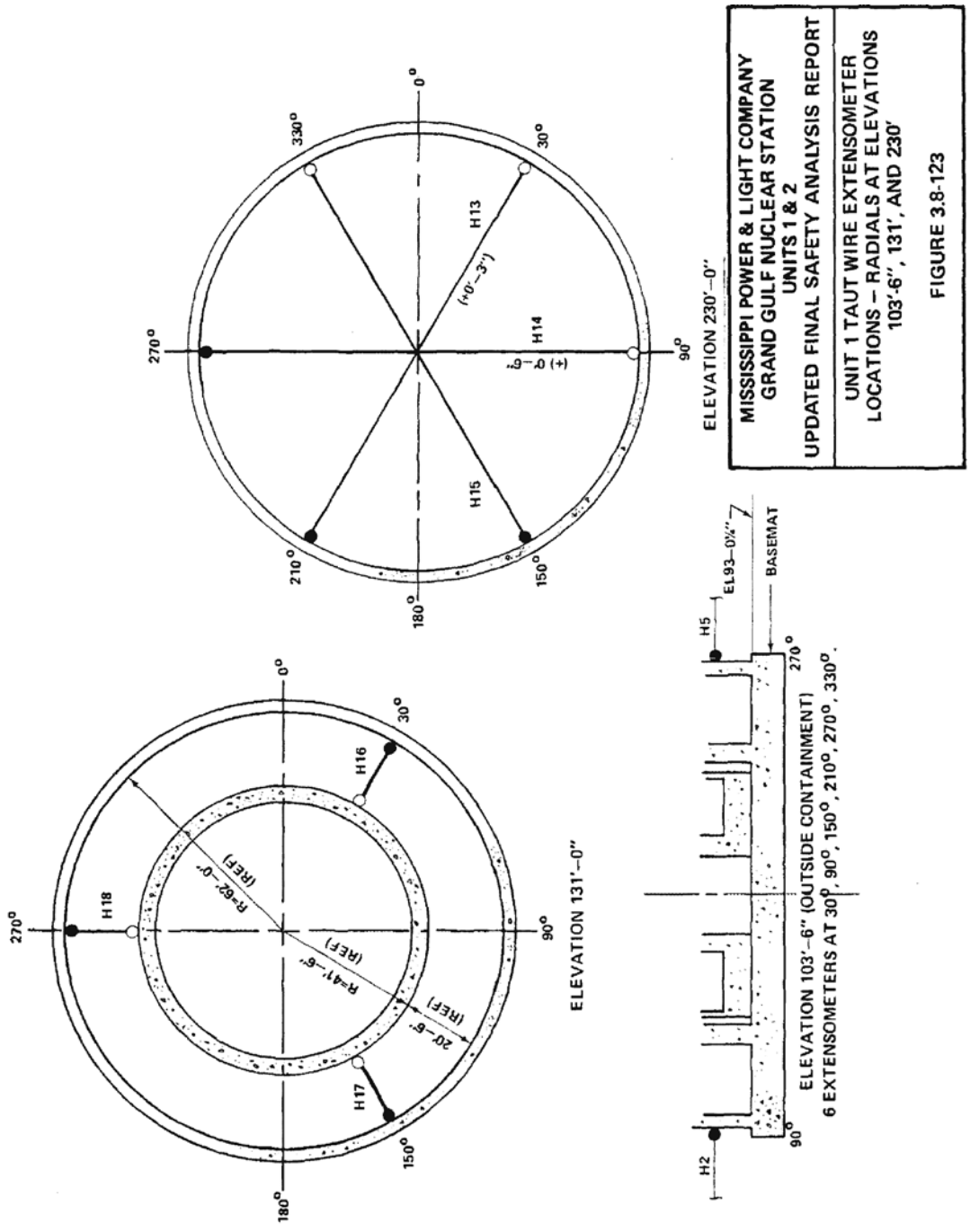
MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT	DRYWELL STRUCTURE DEFLECTION AT 30 PSIG WALL AND ROOF FIGURE 3.8-120

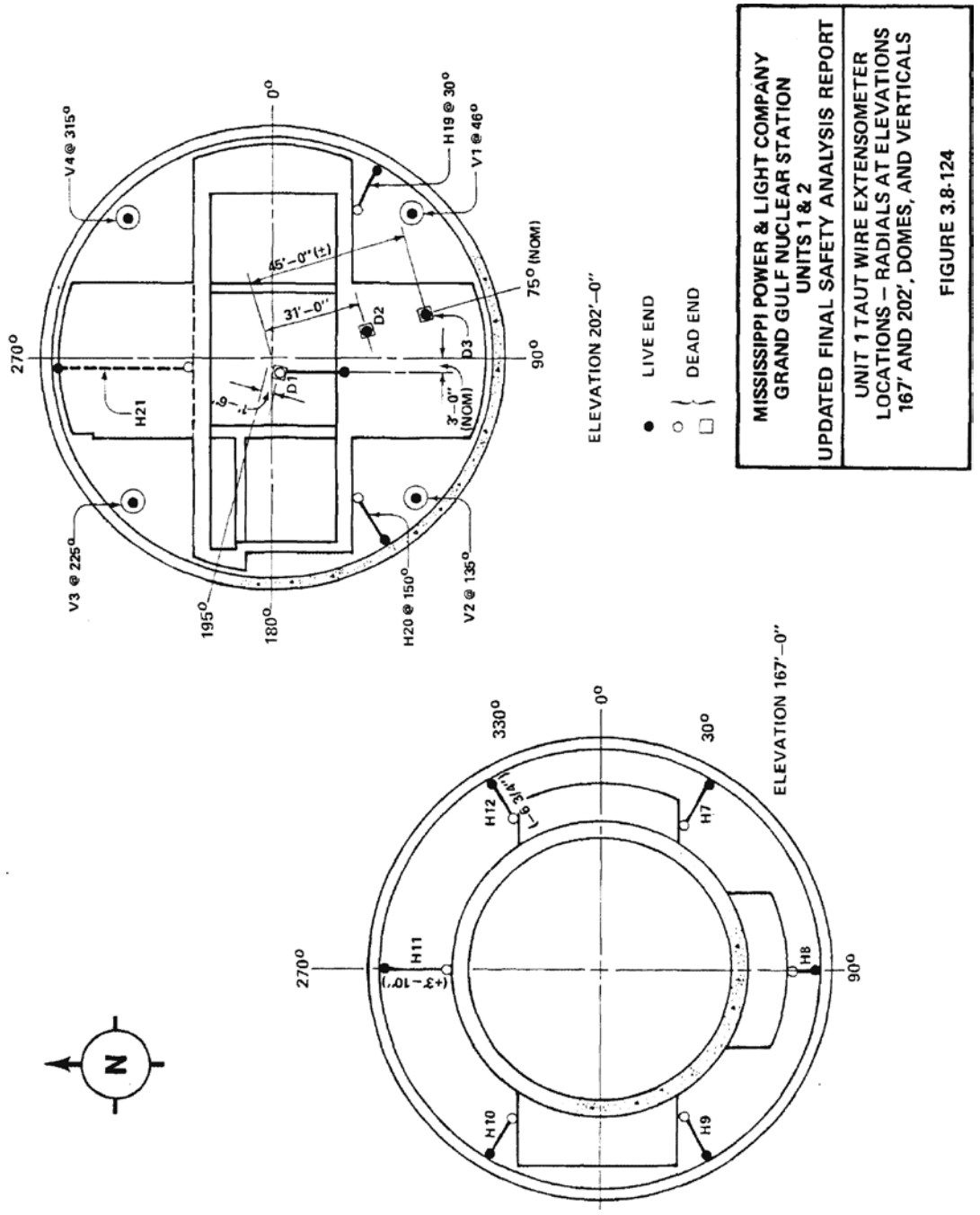
Figure 3.8-121

Deleted

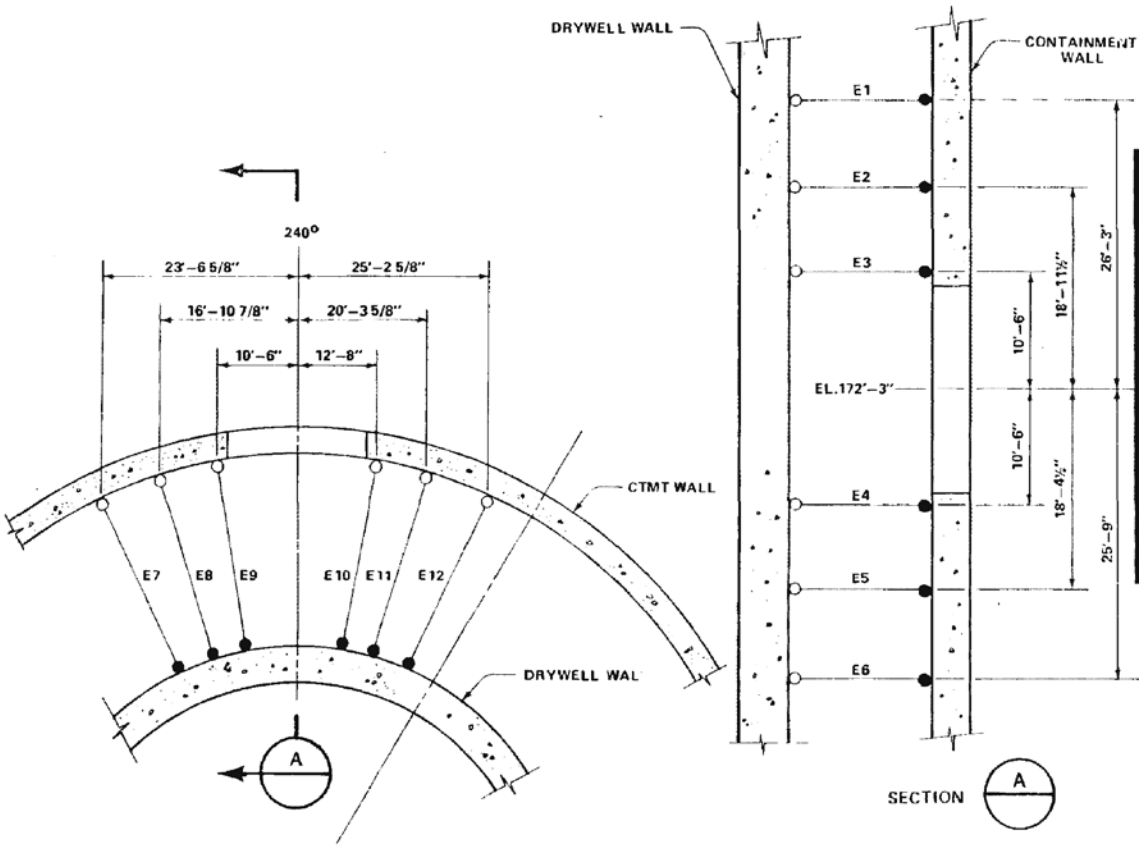


MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT
UNIT 1 SIT PRESSURE CYCLE FIGURE 3.8-122

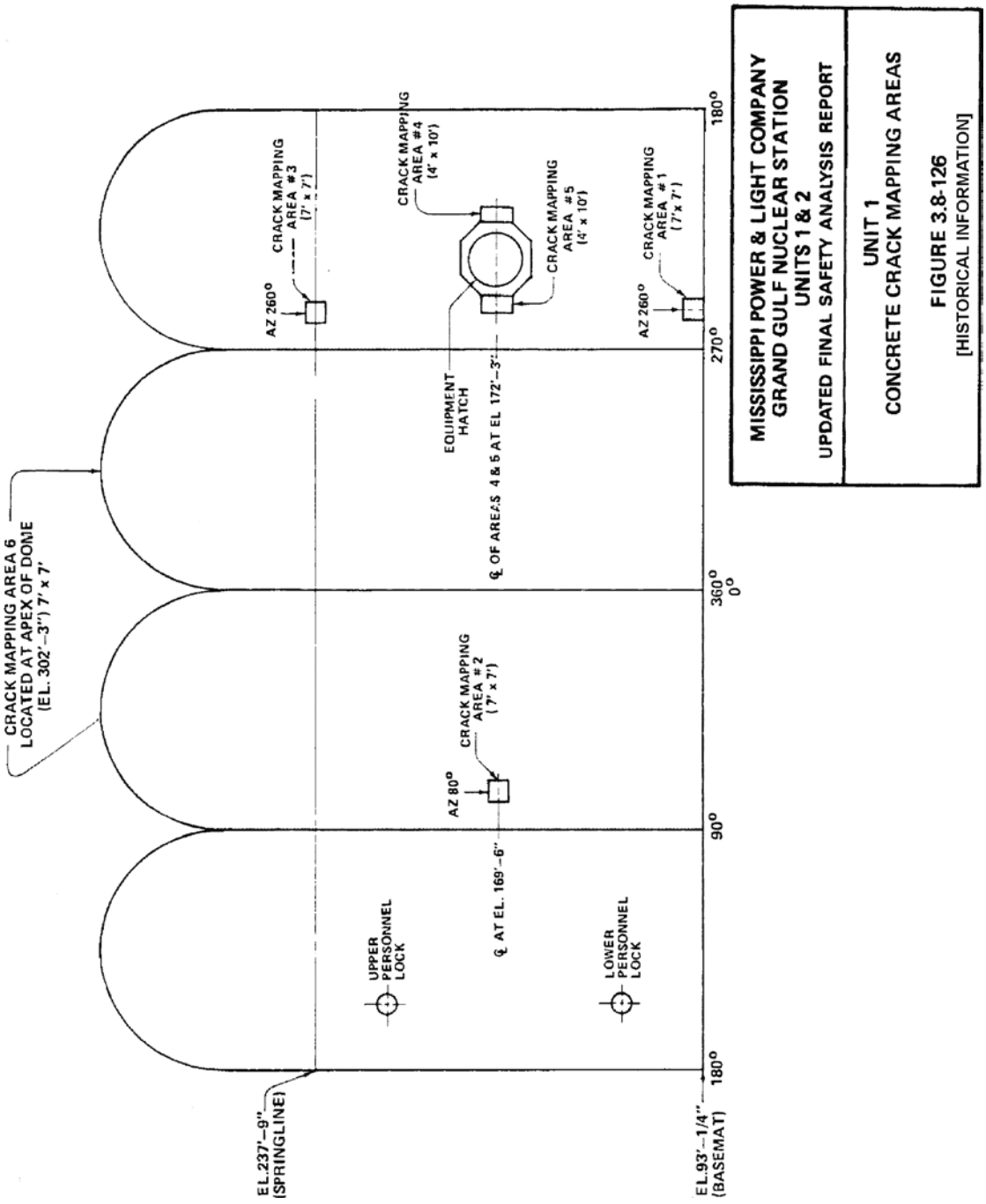

0°
30°
90°
150°
180°
210°
270°
330°
H13
H14
H15
H16
H17
H18
H2
H5
EL93–0½"
BASEMENT
ELEVATION 230'–0"
ELEVATION 131'–0"
ELEVATION 103'–6" (OUTSIDE CONTAINMENT)
6 EXTENSOMETERS AT 30°, 90°, 150°, 210°, 270°, 330°.
MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT
UNIT 1 TAUT WIRE EXTENSOMETER
LOCATIONS – RADIALS AT ELEVATIONS
103'-6", 131', AND 230'
FIGURE 3.8-123



MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT	UNIT 1 TAUT WIRE EXTENSOMETER LOCATIONS – RADIALS AT ELEVATIONS 167' AND 202', DOMES, AND VERTICALS FIGURE 3.8-124

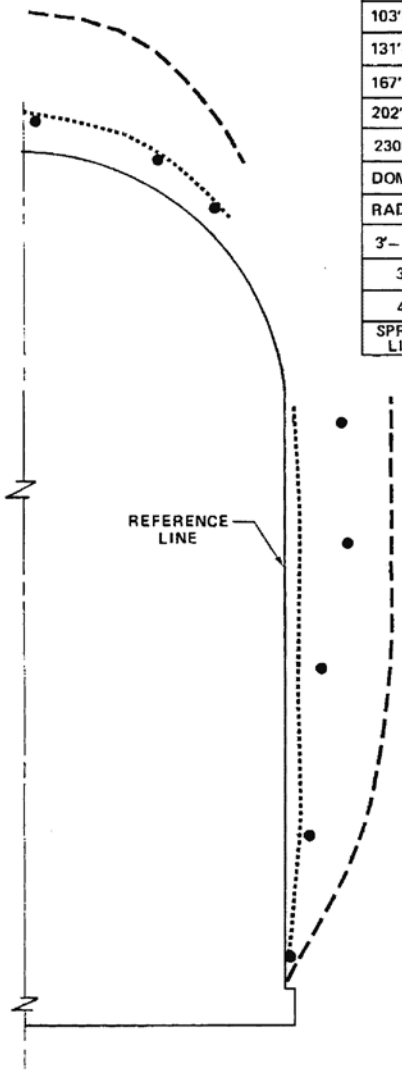


MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT
UNIT 1 TAUT WIRE EXTENSOMETER LOCATIONS – EQUIPMENT HATCH FIGURE 3.8-125


CRACK MAPPING AREA 6
LOCATED AT APEX OF DOME
(EL. 302'-3") 7' x 7'
CRACK MAPPING
AREA #3
(7' x 7')
AZ 260°
CRACK MAPPING
AREA #4
(4' x 10')
CRACK MAPPING
AREA #5
(4' x 10')
EQUIPMENT
HATCH
℄ OF AREAS 4 & 5 AT EL 172'-3"
CRACK MAPPING
AREA #1
(7' x 7')
AZ 260°
180°
270°
360°
0°
AZ 80°
CRACK MAPPING
AREA #2
(7' x 7')
℄ AT EL. 169'-6"
UPPER
PERSONNEL
LOCK
LOWER
PERSONNEL
LOCK
90°
180°
EL.237'-9"
(SPRINGLINE)
EL.93'-1/4"
(BASEMAT)
MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT
UNIT 1
CONCRETE CRACK MAPPING AREAS
FIGURE 3.8-126
[HISTORICAL INFORMATION]

DEFLECTIONS (INCHES)

WALL					
EL	MIN	MAX	AVG	PRED [1]	PRED [2]
103'–6"	.024	944	.034	.08	.01
131'–0"	.056	.082	.071	.38	.05
167'–0"	.084	.283	.178	.42	.05
202'–0"	.201	.271	.226	.42	.05
230'–0" [3]	.092	.1245	.107	.39	.04
DOME					
RADIUS	–	–	DEFL	PRED [1]	PRED [2]
3'–1"	–	–	.114	.54	.15
31'	–	–	.112	.51	.15
45'	–	–	.107	.46	.14
SPRING LINE	.027	.034	.031	.20	.08



0 .20 .40 .60
SCALE – INCHES

CRACKED ANALYSIS DISPLACEMENT

ELASTIC ANALYSIS DISPLACEMENT

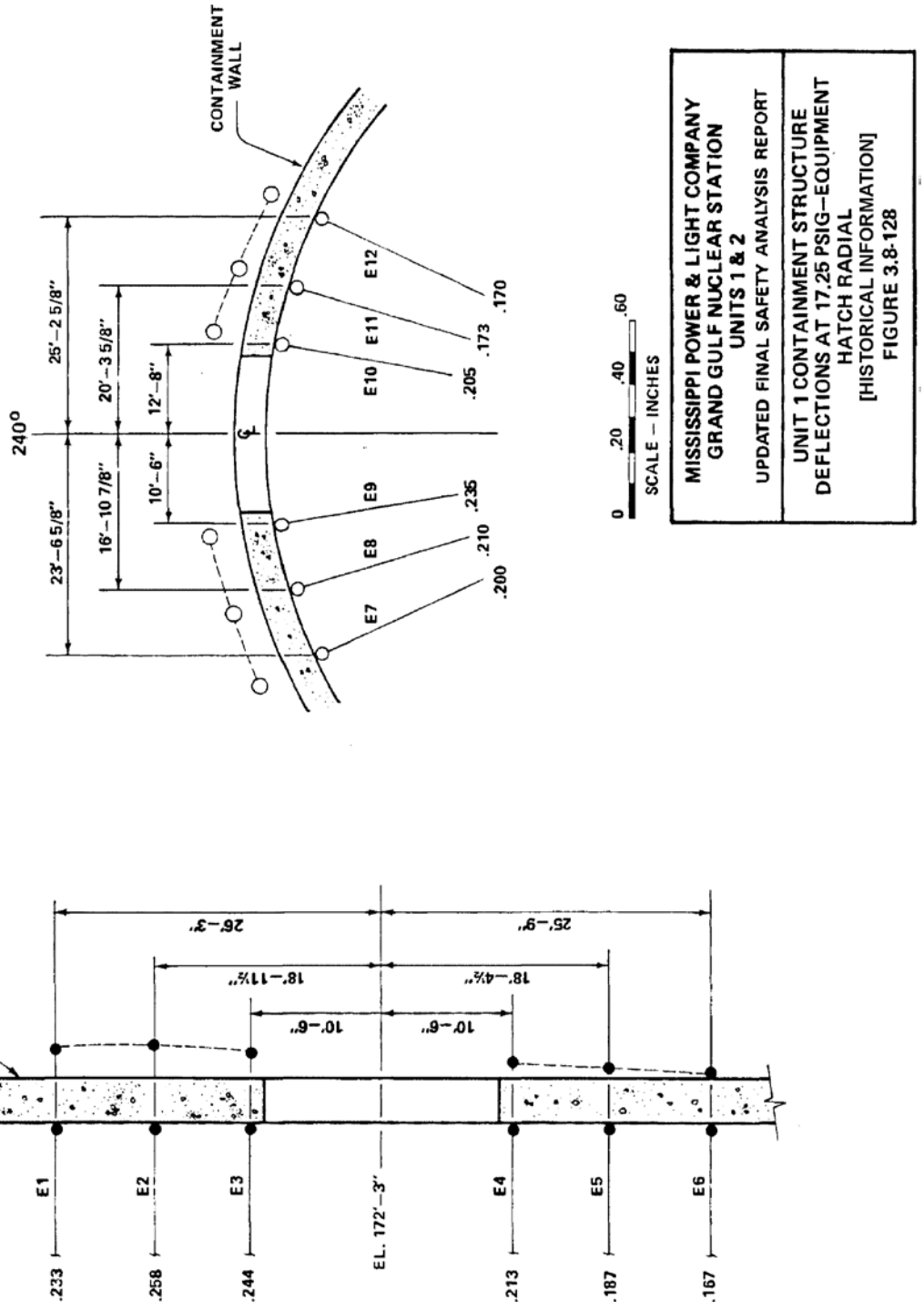
MEASURED VALUE

NOTES:

1. PREDICTED DISPLACEMENT FROM CRACKED ANALYSIS.
2. PREDICTED DISPLACEMENT FROM ELASTIC ANALYSIS.
3. MIN, MAX, AND AVG DEFLECTIONS AT EL. 230' ARE EQUAL TO HALF OF MEASURED DIAMETRIC DISPLACEMENTS.

MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT

UNIT 1 CONTAINMENT STRUCTURE
AVERAGE DEFLECTIONS AT 17.25 PSIG–
WALL AND DOME
[HISTORICAL INFORMATION]
FIGURE 3.8-127


CONTAINMENT WALL
240°
℄
25'–2 5/8"
20'–3 5/8"
12'–8"
16'–10 7/8"
10'–6"
23'–6 5/8"
E12
E11
E10
E9
E8
E7
.170
.173
.205
.235
.210
.200
0
.20
.40
.60
SCALE – INCHES
MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT
UNIT 1 CONTAINMENT STRUCTURE
DEFLECTIONS AT 17.25 PSIG–EQUIPMENT
HATCH RADIAL
[HISTORICAL INFORMATION]
FIGURE 3.8-128
CONTAINMENT WALL
26'–3"
25'–9"
18'–11½"
18'–4½"
10'–6"
10'–6"
E1
E2
E3
EL. 172'–3"
E4
E5
E6
.233
.258
.244
.213
.187
.167

3.9 MECHANICAL SYSTEMS AND COMPONENTS

Analyses supporting GGNS Single Loop Operation (SLO) and operation in the Maximum Extended Operating Domain (MEOD) are detailed in Appendix 15C and 15D, respectively.

3.9.1 Special Topics for Mechanical Components

3.9.1.1 Design Transients

3.9.1.1.1 Design Transients for NSSS Equipment

This section shows the transients which are used in the design of the ASME Code, Section III, Class 1, core support, reactor internals, and control rod drive (CRD) components. The number of cycles or events for each transient are included. The design transients shown in this section are included in the design specifications for the components. Transients or combinations of transients are classified with respect to the component operating condition categories identified as "normal," "upset," "emergency," "faulted," or "testing" in the ASME Code, Section III, Subsection NCA, if applicable. (The first four conditions correspond to Service Levels A, B, C, and D, respectively.)

3.9.1.1.1.1 CRD Transients

The normal and test service load cycles used for the design and fatigue analysis for the 40-year life of the Control Rod Drive are as follows:

	Transient	Category	Cycles
a.	Reactor startup/shutdown	normal/upset	120
b.	Vessel pressure tests	normal/upset	130
c.	Vessel overpressure	normal/upset	10
d.	Scram tests	normal/upset	140
e.	Startup scrams	normal/upset	160
f.	Operational scrams	normal/upset	300
g.	Jog cycles	normal/upset	30,000
h.	Shim/drive cycles	normal/upset	1,000

In addition to the above transients, the following transients have been considered in the design and fatigue analysis of the CRD.

	Transient	Category	Cycles
i.	Scam with inoperative buffer	normal/upset	24
j.	Operation Basis Earthquake (OBE)*	normal/upset	10
k.	Safe Shutdown Earthquake	faulted	1
l.	Scam with struck control blade	faulted	1
m.	Control rod injection faulted	faulted	1

All ASME Class 1 components of the CRD have been analyzed according to ASME Section III Boiler and Pressure Vessel Code.

3.9.1.1.1.2 CRD Housing and Incore Housing Transients

The number of transients, their cycles, and classification as considered in the design and fatigue analysis of the CRD housing and incore housing are as follows:

	Transient	Category	Cycles
a.	Startup and shutdown	normal/upset	120
b.	Design pressure tests	normal/upset	403
c.	Loss of feedwater pumps	normal/upset	10
d.	Relief or safety valve blowdown	normal/upset	8
e.	Scrams	normal/upset	180
f.	Operation Basis Earthquake (OBE)*	normal/upset	10
g.	Safe Shutdown Earthquake (SSE)**	emergency	1
h.	Stuck rod scram	normal/upset	1
i.	Scram no buffer	normal/upset	1

* The frequency of this event would indicate emergency category. However, for conservatism this OBE condition was analyzed as normal and upset event. This event consists of 10 cycles.

** SSE is a faulted condition; however, in the CRD housing stress analysis report, it was treated as an emergency condition with lower stress limits thus making the comparison of results to the allowable more conservative.

3.9.1.1.1.3 Hydraulic Control Unit Transients

The following transients have been considered in the design analysis of the HCU and its components.

	Transient	Category	Cycles
a.	Reactor startup/ shutdown	normal/upset	120
b.	Scram tests	normal/upset	140
c.	Startup scrams	normal/upset	160
d.	Operational scrams	normal/upset	300
e.	Jog cycles	normal/upset	30,000
f.	Shim/drive cycles	normal/upset	1,000
g.	Scram with stuck scram discharge valve	emergency	1
h.	OBE*	normal/upset	10
i.	SSE	faulted	1

3.9.1.1.1.4 Core Support and Reactor Internals Transients

The cycles listed in Table 3.9-1 were considered in the design and fatigue analysis for the reactor internals.

3.9.1.1.1.5 Main Steam System Transients

The following transients are considered in the stress analysis of the main steam piping:

Main Steam Transients**

	Transient	Category	Cycles
a.	Hydrotest	test	40
b.	Leaktest	test	360
c.	Startup	normal	120
d.	Turbine trip	upset	10
e.	Scram and trip isolation valves open	upset	40
f.	Scram	upset	140

*The frequency of this event would indicate emergency category. However, for conservatism this OBE condition was analyzed as normal and upset event. This event consists of 10 cycles.

**The transient cycles shown in this section is the design transient cycles. For operating transient cycles see Mechanical Standard MS-40.

	Transient	Category	Cycles
g.	Shutdown	normal	111
h.	Loss of feedwater pumps isolation valves closed	upset	10
i.	Turbine bypass single relief of safety valve	upset	8
j.	Reactor overpressure delayed scram	emergency	1
k.	Automatic blowdown	emergency	1
l.	Operating Basis Earthquake (OBE)*	normal/upset	10
m.	Turbine stop valve closure (TSV)	upset	600
n.	Relief valve lift (RVL)	upset	5,433

3.9.1.1.1.6 Recirculation System Transients

The following transients are considered in the stress analysis of the recirculation piping:

Recirculation Transients**

Transient		Category	Cycles
a.	Hydrotest	test	40
b.	Startup	normal	120
c.	Turbine trip	upset	10
d.	Partial feedwater heater bypass	upset	70
e.	Turbine generator trip F.W. on isolation valves open	upset	40
f.	Scram	upset	140
g.	Shutdown	normal	111
h.	Loss of feedwater pumps isolation valves closed	upset	10
i.	Turbine bypass single S/RV blowdown	upset	8

*The frequency of this event would indicate emergency category. However, for conservatism this OBE condition was analyzed as normal and upset event. This event consists of 10 cycles.

**The transient cycles shown in this section is the design transient cycles. For operating transient cycles see Mechanical Standard MS-40.

	Transient	Category	Cycles
j.	Reactor overpressure with delayed scram	emergency	1
k.	Automatic blowdown	emergency	1
l.	Operating basis earthquake (OBE)*	normal/upset	10

3.9.1.1.1.7 Reactor Assembly Transients

The reactor assembly includes the reactor pressure vessel, support skirt, shroud support, and shroud plate. The cycles listed in Table 3.9-1 were specified in the reactor assembly design and fatigue analysis.

3.9.1.1.1.8 Main Steam Line Isolation Valve Transients

The main steam line isolation valves are designed for the following service conditions and thermal cycles:

	Transient	Category	Cycles
a.	Startup and shutdown		
	1. Heating cycle @ 100 F/hr	normal/upset	300
	2. Cooling cycle @ 100 F/hr	normal/upset	300
	3. ±29 F between 70 F and 552 F	normal/upset	600
	4. ±50 F step change between 70 F and 552 F	normal/upset	200
b.	Loss of feedwater pump/MSLIV closure		
	1. 552 F to 573 F in 3 seconds (ΔT = 21 F heating)	normal/upset	10
	2. 573 F to 525 F in 9 minutes (ΔT = 48 F cooling)	normal/upset	10
	3. 525 F to 573 F in 6 minutes (ΔT = 48 F heating)	normal/upset	10

*The frequency of this event would indicate emergency category. However, for conservatism this OBE condition was analyzed as normal and upset event. This event consists of 10 cycles.

Transient	Category	Cycles
4. 573 F to 485 F in 7 minutes (ΔT = 88 F cooling)	normal/upset	10
5. 485 F to 573 F in 8 minutes (ΔT = 88 F heating)	normal/upset	10
6. 573 F to 485 F in 7 minutes (ΔT = 88 F cooling)	normal/upset	10
c. Single relief valve blowdown		
1. 552 F to 375 F in 10 minutes (ΔT = 177 F heating)	normal/upset	8
d. Reactor overpressure with delayed scram		
1. 552 F to 586 F in 2 seconds (ΔT = 34 F heating)	emergency	1
2. 586 F to 561 F in 30 seconds (ΔT = 25 F heating)	emergency	1
e. Automatic blowdown (ADS)		
1. 552 F to 375 F in 3.3 minutes (ΔT = 177 F cooling)	emergency	1
2. 375 F to 259 F in 19 minutes (ΔT = 116 F cooling)	emergency	1
f. Pipe rupture and blowdown		
1. 552 F to 259 F in 15 seconds (ΔT = 293 F cooling)	faulted	1

3.9.1.1.1.9 Safety/Relief Valve Transients

The transients used in the analysis of the safety/relief valves are as follows:

Transient	Category	Cycles
a. Heating and cooldown within the temperature limits of 70 F and 552 F at a rate of 100 F/hr	normal/upset	300

	Transient	Category	Cycles
b.	Small temperature changes - of 29 F (either increase or decrease) at any temperature between the limits of 70 F and 552 F	normal/upset	600
c.	50 F temperature changes - (either increase or decrease) at any temperature between the limits of 70 F and 552 F	normal/upset	200
d.	Loss of feedwater pumps, isolation valve closure	normal/upset	10
e.	Turbine bypass, single relief or safety valve blowdown (temperature drops from 552 F to 375 F in 10 minutes)	normal/upset	8
f.	Reactor overpressure with delay scram (temperature rise from 552 F to 586 F in 2 seconds, and the pressure rises from 1050 psig to 1375 psig, immediately followed by a cooling transient in which the temperature drops from 586 F to 561 F in 30 seconds and the pressure drops to 1125 psig)	emergency	1
g.	Automatic blowdown (temperature changes from 552 F to 375 F in 3.3 minutes, immediately followed by a change from 375 F to 259 F in 19 minutes)	emergency	1
h.	Pipe rupture and blowdown - (Temperature drops from 552 F to 259 F in 15 seconds)	faulted	1
i.	Installed hydrotests valve inlet nozzle and disc shall be designed to withstand the following:		

	Transient	Category	Cycles
1.	hydrotests to 1045 psig at 100 F	testing	120
2.	steam line flooding during plant shutdown	other	120

Paragraph NB-3552 of ASME Code Section III excludes various transients and provides means for combining those which are not excluded. Review and approval of the equipment suppliers certified calculations provide assurance of proper accounting of the specified transients.

3.9.1.1.1.10 Recirculation Flow Control Valve Transients

The following pressure and temperature transients were considered in the design of the recirculation system flow control valve:

	Transient	Category	Cycles
a.	Startup (100 F/hr heatup rate 70 F to design temperature)	normal/upset	300
b.	Small temperature changes (29 F step)	normal/upset	600
c.	50 F step changes	normal/upset	200
d.	Safety/relief valve blowdowns (single valve) (552 F to 375 F in 10 minutes)	normal/upset	8
e.	Safety valve transient (110 percent of design pressure)	normal/upset	1
f.	Installed hydrotests		
	1. 1300 psig	testing	130
	2. 1670 psig	testing	3
g.	Automatic blowdown 552 F to 375 F in 3.3 minutes and 375 F to 281 F in 19 minutes	emergency	1
h.	Improper start of pump in cold loop (100 F step to 552 F for 15 seconds)	emergency	1

3.9.1.1.1.11 Recirculation Pump Transients

The list of transients used in the design of the recirculation pump is shown in Table 3.9-1.

3.9.1.1.1.12 Recirculation Gate Valve Transients

The following transients are considered in the design of the recirculation gate valves:

	Transient	Cycles
a.	70 F-575 F-70 F of 100 F/hr	300
b.	±29 F between limits of 70 F and 575 F, instantaneous	600
c.	±50 F between limits of 70 F and 575 F, instantaneous	200
d.	552 F to 375 F, in 10 minutes	8
e.	552 F to 375 F in 3.3 minutes and 375 F to 281 in 19 minutes	1
f.	100 F to 552 F, for 15 seconds	1
g.	110 percent of design pressure at 575F	1
h.	1300 psi at 100 F installed hydrostatic test	130
i.	1670 psi at 100 F installed hydrostatic test	3

3.9.1.1.2 Design Transients for Other Than NSS Systems

Table 3.9-1, Design Plant Transients, gives the plant events that were used for the design and analysis of ASME Code Class 1 components and supports. The table also gives the number of cycles per event and the event classification. The operating basis earthquake (OBE) has been combined with the normal or upset plant condition that causes the most adverse loading and has been used for design. For Grand Gulf, the OBE has been combined with event 10, “Turbine Generator Trip, Feedwater On.” The safe shutdown earthquake is combined with event 10 and is analyzed as a faulted event.

3.9.1.2 Computer Programs Used in Analysis

3.9.1.2.1 Computer Programs Used for NSSS Equipment

The following subsections discuss computer programs used in the analysis of specific components. (Computer programs were not used in the analysis of all components; thus, not all components are listed.)

The NSSS programs can be divided into two categories:

GENERAL ELECTRIC-HITACHI (GEH) PROGRAMS

The verification of the following GE programs has been performed in accordance with the requirements of 10 CFR 50, Appendix B. Evidence of verification is documented in GE Design Record files.

a. SEISM

b. STRESINT

c. MASS

d. DYSEA

e. HEATER

f. FAP-71

g. CREEP-PLAST

h. ANSYS

i. ASSIST

j. CRDSS

k. ANSI

l. RVFOR

m. TSFOR

n. LUGST

o. PDA

p. PISYS

q. SAP4G

r. WBHFN

s. ED-6

t. ED-8

u. GEAPL

v. SPECA

w. ASHSD

x. FTFLG

VENDOR PROGRAMS

The verification of the following vendor programs is assured by contractual requirements between GEH and the vendors. Per the contractual requirements, the verification of these proprietary programs used in the design of N-stamped equipment is in full compliance with 10 CFR 50, Appendix B.

Byron-Jackson Program Recirculation Pump

CB&I Programs

a. 7-11 “GENOZZ”

b. 9-48 “NAPALM”

c. 1027

d. 8-46

e. 7-81 “KALNINS”

f. 979 “ASFAST”

g. 7-66 “TEMAPR”

h. 7-67 “PRINCESS”

i. 9-28 “TGRV”

j. 9-62 “E0962A”

k. 984

l. 992 “GASP”

m. 1037 “DUNHAM'S”

n. 1335

o. 1606 & 1657 “HAP”

p. 1635

q. 953

r. 1666

s. 1684

t. “E1702A”

u. 955 “MESHPLOT”

v. 1028

w. 1038

3.9.1.2.1.1 Reactor Vessel*

The computer programs used in the preparation of the reactor vessel stress report are identified and their use summarized in the following paragraphs.

3.9.1.2.1.1.1 CB&I Program 7-11 - “GENOZZ”

The GENOZZ computer program is used to proportion barrel and double taper type nozzles to comply with the specifications of the ASME Code, Section III and contract documents. The program will either design such a configuration or analyze the configuration

* The RPV text is applicable to plants with RPVs made by CB&I and CBIN only.

input into it. If the input configuration will not comply with the specifications, the program will modify the design and redesign it to yield an acceptable result.

3.9.1.2.1.1.2 CB&I Program 9-48 - "NAPALM"

The basis for the program NAPALM, Nozzle Analysis Program-All Loads Mechanical, is to analyze nozzles for mechanical loads and find the maximum stress intensity and location. The program analyzes at specified locations from the point of application of the mechanical loads. At each location the maximum stress intensity is calculated for both the inside and outside surfaces of the nozzle. The program gives the maximum stress intensity for both the inside and outside surfaces of the nozzle as well as its angular location around the circumference of the nozzle from the 0° reference location. The principal stresses are also printed. The stresses resulting from each component of loading (bending, axial, shear, and torsion) are printed, as well as the loadings which caused these stresses.

3.9.1.2.1.1.3 CB&I Program 1027

This program is a computerized version of the analysis method contained in the "Welding Research Council Bulletin F107, Dec 65."

Part of this program provides for the determination of the shell stress intensities (S) at each of four cardinal points at both the upper and lower shell plate surfaces (ordinarily considered outside and inside surfaces) around the perimeter of a loaded attachment on a cylindrical or spherical vessel. With the determination of each S, there is also determined the components of that S (2 normal stresses, σ_x and σ_y, and one shear stress). This program provides the safe information as the manual computation and the input data is essentially the input of the geometry of the vessel and attachment.

3.9.1.2.1.1.4 CB&I Program 8-46

This program computes the required thickness of a hemispherical head with a large number of circular parallel penetrations by means of the area replacement method in accordance with the ASME Code, Section III.

In cases where the penetration has a counterbore, the thickness is determined so that the counterbore does not penetrate the outside surface of the head.

3.9.1.2.1.1.5 CB&I Program 7-81 - "KALNINS"

This program is a thin elastic shell program for shells of revolution. [HISTORICAL INFORMATION] [This program was developed by Dr. A. Kalnins of Lehigh University. Extensive revisions and improvements have been made by Dr. J. Endicott to yield the CB&I version of this program.

The basic method of analysis was published by Professor Kalnins in the Journal of Applied Mechanics, Volume 31, September 1964,pages 467 through 476.]

The KALNINS shell program (Program 7-81) is used to establish the shell influence coefficient and to perform detail stress analysis of the vessel. The stresses and the deformations of the vessel can be computed for any combination of the following axisymmetric loadings:

a. Preload condition

b. Internal pressure

c. Thermal load

3.9.1.2.1.1.6 CB&I Program 979 - "ASFAST"

ASFAST Program (Program 979) performs the stress analysis of axisymmetric, bolted closure flanges between head and cylindrical shell.

3.9.1.2.1.1.7 CB&I Program 7-66 - "TEMAPR"

This program will reduce any arbitrary temperature gradient through the wall thickness to an equivalent linear gradient. The resulting equivalent gradient will have the same average temperature and the same temperature-moment as the given temperature distribution. Input consists of plate thickness and actual temperature distribution. The output contains the average temperature and total gradient through the wall thickness. The program is written in FORTRAN IV language.

3.9.1.2.1.1.8 CB&I Program 7-67 - "PRINCESS"

The PRINCESS computer program calculates the maximum alternating stress amplitudes from a series of stress values by the method in Section III of the ASME Pressure Vessel Code.

3.9.1.2.1.1.9 CB&I Program 9-28 - "TGRV"

The TGRV program is used to calculate temperature distributions in structures or vessels. Although it is primarily a program for solving the heat conduction equations, some provisions have been made for including radiation and convection effects at the surfaces of the vessel.

[HISTORICAL INFORMATION] [The TGRV program is a greatly modified version of the TIGER heat transfer program written about 1958 at Knolls Atomic Power Laboratory, by A. P. Bray. There have been many versions of TIGER in existence including TIGER II, TIGER II B, TIGER IV, and TIGER V, in addition to TGRV. This manual has been written for use with CB&I's version of TGRV.]

This program utilizes an electrical network analogy to obtain the temperature distribution of any given system as a function of time. The finite difference representation of the three-dimensional equations of heat transfer are repeatedly solved for small time increments and continually summed. Linear mathematics are used to solve the mesh network for every time interval. Included in the analysis are the three basic forms of heat transfer, conduction, radiation and convection, as well as internal heat generation.

Given any odd-shaped structure, which can be represented by a three-dimensional field, its geometry and physical properties, boundary conditions, and internal heat generation rates, TGRV will calculate and give as output the steady state or transient temperature distributions in the structure as a function of time.

3.9.1.2.1.1.10 CB&I Program 962 - "E0962A"

Program E0962A is one of a group of programs (E0953A, E1606A, E0962A, E0992N, E1037N, and E0984N) which are used together to determine the temperature distribution and stresses in pressure vessel components by the finite element method.

Program E0962A is primarily a plotting program. Using the nodal temperatures calculated by program E1606A or Program E0928A, and the node and element cards for the finite element model, it calculates and plots lines of constant temperature (isotherms).

These isotherm plots are used as part of the stress report to present the results of the thermal analysis. They are also very useful in determining at which points in time the thermal stresses should be determined.

In addition to its plotting capability the program can also determine the temperatures of some of the nodal points by interpolation. This feature of the program is intended primarily for use with the compatible TGRV and finite element models that are generated by program E0953A.

3.9.1.2.1.1.11 CB&I Program 984

Program 984 is used to calculate the stress intensity of the stress differences, on a component level, between two different stress conditions. The calculation of the stress intensity of stress component differences (the range of stress intensity) is required by Section III of the ASME Code.

3.9.1.2.1.1.12 CB&I Program 992 - GASP

[HISTORICAL INFORMATION] [The GASP computer program, originated by Professor E. L. Wilson of the University of California at Berkeley, uses the finite element method to determine the stresses and displacements of plane or axisymmetric structures of arbitrary geometry and is written in FORTRAN IV. For a detailed account, see the following reference document.

Wilson, E. L.; "A Digital Computer Program for the Finite Element Analysis of Solids with Non-Linear Material Properties," Aerojet General Corporation, Sacramento, California. Technical Memorandum No. 23, July 1965.]

The program determines the stresses and displacements of plane or axisymmetric structures using the finite element method. The structures may have arbitrary geometry and have linear or non-linear material properties. The loadings may be thermal, mechanical, accelerational, or a combination of these.

The structure to be analyzed is broken up into a finite number of discrete elements or "finite-elements" which are interconnected at finite number of "nodal-points" or "nodes." The actual loads on the structure are simulated by statically equivalent loads acting at the appropriate nodes. The basic input to the program consists of the geometry of the stress-model and the boundary conditions. The program then gives the stress components at the center of each element and the displacements at the nodes, consistent with the prescribed boundary conditions.

3.9.1.2.1.1.13 CB&I Program 1037 - "DUNHAM'S"

DUNHAM'S program is a finite ring element stress analysis program. It will determine the stresses and displacements of axisymmetric structures of arbitrary geometry subjected to either axisymmetric loads or non-axisymmetric loads represented by a Fourier series.

This program is similar to the GASP program (CB&I 992). The major differences are that DUNHAM'S can handle non-axisymmetric loads (which requires that each node have three degrees of freedom) and the material properties for DUNHAM'S must be constant. As in GASP, the loadings may be thermal, mechanical, and accelerational.

3.9.1.2.1.1.14 CB&I Program 1335

To obtain stresses in the shroud support, the baffle plate must be made a continuous circular plate. The program makes this modification and allows the baffle plate to be included in CB&I program 781 as two isotropic parts and an orthotropic portion at the middle (where the diffuser holes are located).

3.9.1.2.1.1.15 CB&I Programs 1606 and 1657 - "HAP"

The HAP program is an axisymmetric nonlinear heat analysis program. It is a finite element program and is used to determine nodal temperatures in a two-dimensional or axisymmetric body subjected to transient disturbances. Programs 1606 and 1657 are identical except that 1606 has a larger storage area allocated and can thus be used to solve larger problems. The model for program 1606 is compatible with CB&I stress programs 992 and 1037.

3.9.1.2.1.1.16 CB&I Program 1635

Program 1635 offers three features to aid the stress analyst in preparing a stress report.

a. Generates punched card input for program 7-67 (PRINCESS) from the stress output of program 7-81 (KALNINS)

b. Writes a stress table in a format such that it can be incorporated into a final stress report

c. Has the option to remove through-wall thermal bending stress and report these results in a stress table similar to the one mentioned above

3.9.1.2.1.1.17 CB&I Program 953

The program is a general purpose program, which does the following:

a. It prepares input cards for the thermal model.

b. It prepares the node and element cards for the finite element model.

c. It sets up the model in such a way that the nodal points in the TGRV model correspond to points in the finite element model. They have the same number so that there is no possibility of confusion in transferring temperature data from one program to the other.

3.9.1.2.1.1.18 CB&I Program 1666

This program is primarily written to calculate the temperature differences at selected critical sections of the nuclear reactor vessel components at different time points of thermal transients during its life of operation and list them all in a tabular form.

This program can be used with any other kind of model that sees thermal transients over a period of time. This program helps ascertain the time points in thermal transients when the thermal stresses may be critical.

3.9.1.2.1.1.19 CB&I Program 1684

This program is written to better expedite the fatigue analysis of nuclear reactor components as required by the ASME Boiler and Pressure Vessel Code, Section III. Specifically, this program is an expansion of an earlier program, 984. The features of this program allow the user to easily perform the complete secondary stress and fatigue evaluation including partial fatigue usage

calculation of a component in one run. An additional option allows the user to completely document the input stress values in a format suitable for a stress analysis report. The program is written to allow for a minimum amount of data handling by the user once the initial deck is established.

3.9.1.2.1.1.20 CB&I Program "E1702A"

This program evaluates the stress-intensity factor K_I due to pressure, temperature, and mechanical load stresses for a number of different stress conditions (times) and at a number of different locations (elements). It then calculates the maximum RT_{NDT} which the actual material can have based on a 1/4T flaw size and compares it with the ordered RT_{NDT}. If the ordered RT_{NDT} is larger than the maximum RT_{NDT}, the maximum allowable flaw size is calculated. The rules of Appendix G are used except that WRC 175 can be used to calculate K_I due to pressure in a nozzle to shell junction.

For a more thorough description of the fracture problem, read WRC Bulletin No. 175, "PVRC Recommendations on Toughness Requirements for Ferritic Materials."

3.9.1.2.1.1.21 Deleted

3.9.1.2.1.1.22 Deleted

3.9.1.2.1.1.23 CB&I Program 955 - "MESHPLOT"

This program plots input data used for finite element analysis. The program plots the finite element mesh in one of three ways: without labels, with node labels, or with element labels. The output consists of a listing and a plot. The listing gives all node points with their coordinates, and all elements with their node points. The plot is a finite element model with the requested labels.

3.9.1.2.1.1.24 CB&I Program 1028

This program calculates the necessary form factors for the nodes of the model which simulates heat transfer by radiation. Inputs are shape-and-dimensions of the head-to-skirt knuckle junction. The program is limited to junctions with a toroidal knuckle part.

3.9.1.2.1.1.25 CB&I Program 1038

This program calculates the loads required to satisfy the compatibility between the shroud baffle plate and the jet pump adaptors in the RPV.

3.9.1.2.1.2 Reactor Internals

3.9.1.2.1.2.1 Fuel Support Loads Program/SEISM

SEISM computes the vertical fuel support loads using the component element methods in dynamics. [HISTORICAL INFORMATION] [The methodology is based on the publication "The Component Element Method in Dynamics," by S. Levy and J. P. D. Wilkinson, McGraw Hill Co., New York, 1976.]

3.9.1.2.1.2.2 Stress Intensities Program/STRESINT

The time-share program STRESINT is used to calculate the principal stresses and intensities of a stress tensor. STRESINT computes the principal stresses by obtaining roots of an eigenvalue cubic equation. Stress intensities are found by taking the absolute values of the differences between the principal stresses.

3.9.1.2.1.2.3 Stress and Vibration Analysis (ASSIST)

ASSIST is a time-sharing computer program for stress and vibration analysis of statically indeterminate structures. The program can be used to obtain load distributions, deflections, critical frequencies and mode shapes in the "in-plane" or "normal-to-plane" modes for planar structures of any orientation. Deformations and resulting load distributions are computed considering all strain energies, i.e., bending, torsion, shear, and direct.

3.9.1.2.1.2.4 Other Programs

The following computer programs are also used in the analysis of the core support structures and other safety-related reactor internals: MASS, DYSEA, HEATER, FAP-71, and CREEP-PLAST. Detailed descriptions of these programs are provided in subsection 4.1.

3.9.1.2.1.3 Piping

3.9.1.2.1.3.1 Component Analysis/ANSI

The ANSI computer program determines stress and accumulative usage factors in accordance with NB-3600 of the ASME Boiler and Pressure Vessel Code, Section III. The program was written to perform stress analyses in accordance with the ASME sample problem, and has been verified by reproducing the results of the sample problem analysis.

3.9.1.2.1.3.2 Relief Valve Discharge Pipe Forces/RVFOR

The relief valve discharge pipe connects the relief valve to the suppression pool. When the valve is opened, the transient fluid flow causes time-dependent forces to develop in the pipe wall. This program computes the transient fluid mechanic parameters and the resultant pipe forces using the method of characteristics.

3.9.1.2.1.3.3 Turbine Stop Valve Closure/TSFOR

The TSFOR program computes the time history forcing function in the main steam piping due to turbine stop valve closure. The program utilizes the method of characteristics to compute fluid momentum and pressure loads at each change in pipe section or direction.

3.9.1.2.1.3.4 Integral Attachment/LUGST

The computer program LUGST evaluates the stresses in the pipe wall that are produced by loads applied to the integral attachments.

3.9.1.2.1.3.5 Piping Dynamic Analysis/PDA

The pipe whip analysis was performed using the PDA computer program. PDA is a computer program used to determine the response of a pipe subjected to the thrust force occurring after a pipe break. The program treats the situation in terms of generic pipe break configuration, which involves a straight, uniform pipe fixed at one end and subjected to a time-dependent thrust force at the other end. A typical restraint used to reduce the resulting deformation is also included at a location between the two ends. Nonlinear and time-dependent stress relations are used to model the pipe and the restraint. Similar to the popular elastic-hinge concept, bending of the pipe is assumed to occur only at the fixed end and at the location supported by the restraint.

Shear deformation is neglected. The pipe bending moment deflection (or rotation) relation used for these locations is obtained from a static nonlinear cantilever beam analysis. Using moment-rotation relations, nonlinear equations of motion are formulated using energy considerations, and the equations are numerically integrated in small time steps to yield the time-history of the pipe motion.

3.9.1.2.1.3.6 Piping Analysis/PISYS

PISYS is a specialized computer code for piping load calculations. It utilizes selected stiffness matrices representing standard piping components, which are assembled to form a finite element model of a piping system. The technique relies on dividing the pipe model into several discrete substructures, called pipe elements, which are connected to each other via nodes called pipe joints. It is through these joints that the model interacts with the environment and loading of the structure becomes possible. PISYS is based on the linear classical elasticity in which the resultant deformation and stresses are proportional to the loading and the superposition of loading is valid.

PISYS has a full range of static and dynamic analysis options which include: distributed weight, thermal expansion, differential support motion modal extraction, response spectra, and time-history analysis by modal or direct integration. The PISYS program has been benchmarked against five Nuclear Regulatory Commission piping models for the option of response spectrum analysis.

3.9.1.2.1.4 Recirculation Pump

The Byron-Jackson proprietary version of SAP using finite element methods is used in the analysis of the recirculation pump casing for the various loading conditions imposed by thermal and mechanical loads during plant operating and postulated conditions.

In general, the finite element techniques are used to solve temperature distribution heat transfer transient analysis problems, and to perform stress analysis for various thermal and mechanical loadings by using the same finite element model representing the pump body. The output of this program is in the

form of temperature profiles, deflections, and stresses at the modal points of the finite element idealization of the pump structure.

Accuracy of the computer program methods is demonstrated by comparison with selected sample problems which lend themselves to alternate computation methods to provide closed-form analytical solutions.

3.9.1.2.1.5 ECCS Pumps and Motors

The following are the computer programs used in the dynamic analysis to ensure the structural and functional integrity of the ECCS pumps and motor assemblies.

3.9.1.2.1.5.1 Structural Analysis/SAP4G

SAP4G is used to analyze the structural and functional integrity of the ECCS pump/motor systems. This is a general structural analysis program for static and dynamic analysis of linear elastic complex structures. The finite element displacement method is used to solve the displacements and stresses of each element of the structure. The structure can be composed of an unlimited number of three-dimensional truss, beam, plate, shell, solid, plate strain-plane stress and spring elements that are axisymmetric. The program can treat thermal and various forms of mechanical loading. The dynamic analysis includes mode superposition, time-history, and response spectrum analysis. Seismic loading and time-dependent pressure can be treated. The program is versatile and efficient in analyzing large and complex structural systems. The output contains displacements of each nodal point as well as stresses at the surface of each element.

3.9.1.2.1.5.2 Effects of Flange Joints Connections/FTFLG

The flange joints connecting the pump bowl casings are analyzed using FTFLG. This program uses the local forces and moments determined by SAP4G to perform flat flange calculations in accordance with the rules set forth in Appendix XI and Section III of the ASME Boiler and Pressure Vessel Code.

3.9.1.2.1.6 RHR Heat Exchangers

Following are the computer programs used in dynamic and static analysis to determine structural and functional integrity of the RHR heat exchanger.

a. RHR Natural Frequency Program (WBHFN)

This program generated by GE calculates the natural frequency of the RHR heat exchanger considering the stiffness of the supporting steel structure. Previously, it was found that the steel frame structure which is normally mounted to a concrete wall or floor was quite flexible and reduced the combined natural frequency (first mode) to the range of 10 to 15 cps. In this program the heat exchanger is modeled as a flexible beam supported on two springs, the two springs being the stiffness characteristics of the upper tie down support and the lower support bracket.

Three different methods of calculating the natural frequency of the RHR heat exchanger could be used. All three methods were used to calculate the natural frequency of a typical system with all results being within 7 percent of each other.

b. Support Load Seismic Analysis (ED-6)

This computer program computes the total loads at the upper and lower supports of the RHR heat exchanger flooded weight, seismic (either OBE or SSE) and the allowable nozzle loads and sets up the worst combination of these loads. By maximizing seismic loads together with nozzle loads, maximum conservative moments and forces at the upper and lower supports are calculated.

c. Stress Analysis of Supports (ED-8)

This program performs a full stress analysis of the upper and lower supports of the RHR heat exchanger. The stresses in the supports (both upper and lower) caused by loads resulting from seismic and nozzle loads are computed in the Support Load Program (ED-6) and are used as input values for this program. This program computes the membrane stresses on the shell of the heat exchanger by the use of the Bijlaard's analysis, as well as the net section stresses (shear, tensile, bearing) on the lower support plate and upper lugs. It also computes the stresses on the welds which hold the supports to the shell of the heat exchanger.

3.9.1.2.1.7 Dynamic Loads Analysis

3.9.1.2.1.7.1 Dynamic Analysis Program/DYSEA

DYSEA simulates a beam model in the annulus pressurization dynamic analysis. A detailed description of DYSEA is provided in Section 4.1. DYSEA employs a preprocessor program named GEAPL. GEAPL converts pressure time-histories into time varying loads and forcing functions for DYSEA. The overall resultant forces and moments time-histories at specified points of resolution can also be obtained from GEAPL.

3.9.1.2.1.7.2 Acceleration Response Spectrum Program/SPECA

SPECA generates acceleration response spectrum for an arbitrary input time-history of piecewise linear accelerations, i.e., it computes maximum acceleration responses for a series of single-degree-of-freedom systems subjected to the same input. It can accept acceleration time-histories from a random file. It also has the capability of generating the broadened/enveloped spectra when the spectral points are generated equally spaced on a logarithmic scale axis of period/frequency. This program is also used in seismic and SRV transient analyses.

3.9.1.2.1.7.3 Axisymmetrical Structure Analysis Program/ASHSD

ASHSD performs dynamic analysis of axisymmetric structures subjected to any arbitrary static or dynamic loading or base acceleration. The finite element methodology is used to generate vibration mode shapes and frequencies, time-history of nodal displacement and acceleration, and element stress for each Fourier term.

3.9.1.2.2 Computer Programs for Non-NSSS Equipment

Table 3.9-11 give computer programs used in the design of non-NSSS components. The program verification is as follows.

3.9.1.2.2.1 ME-632 Program

Purpose

The stresses and loads in piping systems due to restrained expansion, deadweight, seismic anchor movement, and seismic inertia are calculated using the ME-632 computer program (Reference 7).

Method of Analysis

The stiffness method of finite element analysis has been used in this program. In this method, the displacement of the joints of a given structure are considered to be the basic unknowns. The dynamic analysis is performed by the modal synthesis method. The modal synthesis, in principle, exploits known maximum accelerations produced in a single-degree-of-freedom model of certain frequency. The method is described in detail in the references. The program's principal assumptions are:

a. Linearly elastic structure

b. Simultaneous displacement of all supports described by a single time-dependent function

c. Lumped-mass model satisfactorily replaces the structure

d. Modal synthesis is applicable

e. Rotational inertias of the masses have negligible effect

Program Verification

The following is a comparison of the ME-632 program results with the results of the Engineering Data System (EDS) computer program.

The two piping systems selected for stress checks were the following:

a. The Core Spray Piping System - Monticello Nuclear Generating Plant Unit 1

b. Lines 48223-18-HE, 50056-10-HE, 50057-10-HE - SMUD Rancho Seco Unit 1

The analyses by Engineering Data Systems were selected because the EDS computer code is recognized by the industry for yielding a reliable stress analysis and because the above-mentioned systems were available to Bechtel at the time of comparison.

The EDS analysis of the core spray piping system consisted of both dead weight and thermal loading, whereas the SMUD Rancho Seco piping system was a seismic response spectrum analysis.

[HISTORICAL INFORMATION] [The ME-632 piping stress analyses were performed in the period 9/18/72-9/20/72 on Bechtel International Corporation Data Processing Services Honeywell 635 computer. A relocatable binary deck of the program is stored on tape No. 8312 and will be retained indefinitely for documentation purposes.]

Comparison of Results for Thermal and Weight Analysis

A comparison of the ME-632 and EDS analyses is given in Table 3.9-12. Due to differing sign conventions, the reactions have opposite signs. The EDS program prints the effects of the support on the piping system whereas ME-632 prints the effect of the piping system on the support. In some cases, the maximum values for the ME-632 analysis occurred at the middle of the bend. However, since the EDS program does not compute output quantities at the middle of a bend, these maximums are not given in Table 3.9-12. The maximum shown in the table occurred at this same physical point on the piping system in both analyses.

In all cases, the maximum difference in output quantities was less than 5 percent based on the corresponding peak value for the particular load case. This comparison is made after the three-directional components are summed using the square-root-of-sum-of-squares method.

It is therefore concluded that ME-632 correctly performs static and thermal analyses of piping systems that are consistent with the assumptions of elastic beam theory and applicable flexibility and stress intensification factors specified in the ASME Code, Section III.

Comparison of Results for Seismic Analysis

A comparison of maximum stresses, deflections, and reaction forces is given in Table 3.9-13. Unless otherwise noted, the corresponding maximums occurred at identical locations. In all cases, the maximum difference between the two programs was less than 5 percent, based upon the peak deflection, stress, moment, or force for the particular load case.

At the given time, ME-632 did not compute tee stresses per ASME Code, Section III, whenever two consecutive tees existed in the piping system. Otherwise, the stresses computed in the two computer runs compare very favorably.

Whenever two consecutive tees exist in a piping system, the stress levels at the tees are hand calculated using ASME Code, Section III, and the moments computed by ME-632 and documented with the problem summary. Moments computed by ME-632 have been verified to be correct. Therefore, stresses at consecutive tees are computed consistent with ASME Code, Section III.

The natural periods obtained from the two programs are given in Table 3.9-14. Again there is excellent agreement.

It is therefore concluded that the ME-632 computer program correctly performs dynamic analysis of piping systems consistent with the assumptions of the lumped-mass, response spectrum approach for elastic systems, and applicable stress intensification and flexibility factors in accordance with the ASME Code, Section III.

Summary and Conclusions

The following stress analyses were performed using the ME-632 piping stress analysis computer program:

a.	[HISTORICAL INFORMATION] [Monticello Nuclear Power Plant Spray Piping System Unit 1	Dead weight; thermal Core with anchor movements
b.	SMUD Rancho Seco, Unit 1 Lines 48223-18-FE, 50056-10-HE, 50057-10-HE]	Seismic

The resulting forces, moments, deflections, and stresses were compared with independent analyses performed on the same piping systems using the same loadings. A comparison of results showed that differences in the output quantities were less than 5 percent based on the corresponding maximum value.

Based on these results, the ME-632 program may be used with confidence to analyze piping systems per the ASME Code, Section III.

References

1. Deleted

2. Deleted

3. Deleted

4. Deleted

5. Deleted

6. Deleted

7. "Program ME-632, Seismic Analysis of Piping Systems, Users Manual," Pacific International Computing Corp., March 1971.

3.9.1.2.2.2 ME-913 Program

Purpose

The ME-913 program consists of numerical calculations of stress intensity levels for Class 1 nuclear power piping components to validate their design adequacy.

Method of Analysis

The program determines the stress intensity levels of Class 1 nuclear power piping components for equations 9 through 14 of subarticle NB-3650, Analysis of Piping Components of Section III, ASME Boiler and Pressure Vessel Code (Reference 1). The method is described in detail in that subarticle.

Prior to running this program, the user analyzes the piping system using flexibility analysis program (ME-632 or ME-101) and TRHEAT (see subsection 3.9.1.2.2.3). The inputs to ME-913 are the following:

a. Piping configuration

b. Piping and piping component properties

c. Moment reactions due to:

   1. Thermal expansion loads

   2. Weight loads

   3. Earthquake loads

d. The thermal response of the piping system due to the specified transients: T_1, T_2, and (T_a - T_b) values for the key points during system life.

Program Verification

The verification of ME-913 is performed in two phases:

a. Phase I: Comparison of Results Between ASME Sample Problem (Rev. 2) and ME-913

   In order to compare the results of ME-913 and the ASME sample problem, two component locations were selected. They were:

   1. Butt welding tee

   2. Girth butt weld with a change in material and wall thickness.

   A comparison of the results indicated that the calculation of stresses for equations 9, 10, and 11 using ME-913 and the ASME sample problem yields identical values. Calculation of the cumulative usage factor yields results differing by less than 4 percent. The slight difference in allowable cycle determination from the interpolation method introduced different individual fatigue usage factors which in turn result in a slightly different cumulative usage factor.

b. Phase II: Hand-Calculated Verification of the Computer Output

   Several piping systems were selected and analyzed using the ME-913 program. Hand-calculations were performed for the same piping system. Results have been compared to the computer output in order to illustrate the correct

application of Code requirements. Results provided by ME-913 and hand-calculations have only slight numerical differences which would result from round-off errors.

Summary and Conclusion

Stress analyses were performed using ME-913 code compliance computer program for the following examples:

a. Sample Analysis of Class 1 Piping System prepared by Working Group on Piping of the ASME Boiler and Pressure Vessel Code (Ref. 2)

b. Analysis of several piping systems generated by Bechtel Power Corporation.

The resulting stresses were compared with the results from working group calculations and hand computations performed on the same piping systems using the same loadings. A comparison of results shows only slight differences in the output quantities as explained above. Judging from these results, the ME-913 program may be used with confidence to analyze nuclear Class 1 piping systems per ASME Code, Section III.

References

1. ASME Boiler and Pressure Vessel Code, Section III, Nuclear Power Plant Components, 1974 edition.

2. Sample Analysis of a Class 1 Piping System, prepared by the Working Group on Piping (SG-D, Section III) of the ASME Boiler and Pressure Vessel Code, December 1971.

3.9.1.2.2.3 TRHEAT Program

Purpose

Computer program TRHEAT determines the temperature response of a pipe due to a temperature transient in the contained fluid.

Method of Analysis

TRHEAT (Reference 1) computes the equivalent linear and nonlinear pipe wall thermal gradients, ΔT and ΔT_2, along with the temperature differences at locations of piping discontinuities, T_A-T_B, required for calculation of piping stresses in accordance

with the requirements for Class 1 piping specified in the ASME Code, Section III. The method of analysis is a closed form solution to the basic heat transfer partial differential equation.

Program Verification

In order to verify the capabilities and accuracy of solution of computer program TRHEAT, a sample problem was utilized and the following results were verified:

a. Pipe and fluid properties calculated by the program have been compared to standard reference values.

b. Calculated pipe-to-fluid film heat transfer coefficient has been checked against that obtained by using standard reference formulas.

c. TRHEAT calculations for pipe wall thermal gradients and temperature differences at piping discontinuities have been checked with similar closed form solutions provided by reference charts of Brock and McNeill. In addition, TRHEAT results have been checked against those obtained by using the finite elements computer program MARCHEAT (Reference 7).

Generated fluid and pipe thermal and material properties are all within 1.4 percent of standard reference values. The calculated film heat transfer coefficient is identical to that determined using a reference formula. The ΔT_1 and ΔT_2 wall thermal gradients were found to be within 2.6% of those of a closed form solution and within 7.3 percent of a finite element heat transfer code. The T_A-T_B gross discontinuity temperature difference was within 4.4 percent of that of the finite element solution. The deviations that do exist result mostly from differences in numerical integration formulas and round-off errors in numerical operations.

References

1. Nuclear Services Corporation, “TRHEAT: Computer Code for Transient Heat Analysis of Nuclear Piping”; Revision 2, dated November 1972.

2. Deleted

3. Deleted

4. Deleted

5. Deleted

6. Deleted

7. MARC Analysis Corporation, "MARCHEAT: A Finite Element Transient Heat Conduction Program", 1971.

8. Deleted

3.9.1.2.2.4 ANSYS Program

Purpose

The ANSYS program is a large-scale general-purpose computer program for the solution of several classes of engineering analysis problems. Analysis capabilities include static and dynamic; plastic, creep, and swelling; small and large deflections; steady-state and transient heat transfer; and steady-state fluid flow.

Method of Analysis

The matrix displacement method of analysis based upon finite element idealization is employed throughout the program. ANSYS uses the wave front (or frontal) direct solution method for the system of simultaneous linear equations developed by the matrix displacement method.

Program Verification

[HISTORICAL INFORMATION] [The ANSYS program has been developed and verified by Swanson Analysis Systems, Inc. Grand Gulf is using the version dated March 1, 1975. The program is being run on a CDC 6600 computer system.]

3.9.1.2.2.5 ME-101 Program

(See subsection 3.7.3.8.1.3.2.)

3.9.1.2.2.6 SUPERPIPE Program

Purpose

SUPERPIPE is a comprehensive computer program for the structural analysis and design checking of piping systems, with particular emphasis on nuclear power piping. The program is capable of static load analysis, determination of dynamic properties (mode shapes and frequencies), response spectrum analysis (single and multiple excitation analysis), and design checking against Code requirements.

Method of Analysis

a. Static Analysis

The piping system is idealized as a three-dimensional frame and analyzed by the Direct Stiffness method. Only finite length components are considered in the stiffness calculations. All zero-length components are ignored.

Straight pipe elements between user-specified nodes are idealized as single finite elements regardless of length. Stiffness calculations are based upon straight beam-constant section theory and include the effects of shear deformation. Distributed loads are included exactly rather than lumping the loads at discrete nodes.

Bends and elbows are idealized as curved beam elements and their stiffnesses are calculated by an exact procedure and include flexibility factors and shearing deformation effects. Typically, each curved pipe element is analyzed as a single finite element, but, the program may automatically subdivide any bend element into two or more subelements for increased accuracy when performing any distributed load calculation.

b. Dynamic Analysis

Lumped mass modeling techniques are used for mode shapes and frequency calculations. Solution of the resulting eigenvalue problem is accomplished by either an in-core equation solver routine or subspace iteration. Mass point placement is at each user specified node, but additional mass points may be automatically specified based upon

simple beam vibration criteria. Flexible connections can also be used to represent expansion joints, ball joints, etc.

c. Response Spectra Analysis

Simple excitation or multiple excitation response spectra analysis can be performed. For multiple excitation analysis, different response spectra may be specified at different support levels. The total inertia effect is then obtained by combining the effects from each level by the absolute sum method.

Program Verification

This program has been verified using PISOL, NIPIPE, PIPESD, ADLPIPE, and ME101.

3.9.1.3 Experimental Stress Analysis

3.9.1.3.1 Experimental Stress Analysis for NSSS Equipment

The following subsections in this section list the only NSSS components upon which experimental stress analysis was used and provide a discussion of the analysis.

3.9.1.3.1.1 Experimental Stress Analysis of Piping Components

The following components have been tested to verify their design adequacy:

a. Piping seismic shock suppressors

b. Pipe whip restraints

Descriptions of the support and whip restraint tests are contained in subsection 3.9.3.4 and Section 3.6B, respectively.

3.9.1.3.1.2 Orificed Fuel Support, Vertical and Horizontal Load Tests

The BWR 6 Orificed Fuel Support (OFS) under the provisions of the ASME Code, Section III, subsection NG is classified as a core support structure, and therefore, must comply with NA-3352.1. In order to meet this requirement, an analysis was attempted using the finite element method. However, the complexity of the OFS design as well as the nonlinear behavior of the OFS during

analysis preempted the use of finite element analysis. Accordingly, a series of horizontal and vertical load tests were performed in order to conform to the requirements of the code and the design specification. A comparison of the stress results for static and dynamic loadings is provided in Table 3.9-2(b), where a 0.65 quality factor is included in the allowables.

Two separate tests were conducted, and each test was designed to be in conformance with Appendix II of the ASME code Section III. The first test series verified the structural capability of the fuel support casting to sustain vertical design loads. A production fuel support was stress-coated and subjected to an extremely high vertical load to identify the location and principal stress directions of the highest stressed regions. A second fuel support was instrumented with strain gauges: 12 uniaxial gauges were used where the principal stress directions were known from the previous stress-coat test. Six rosettes were used where the principal stress axes could not readily be determined. (All the gauges used in the experimental stress analysis were put in the regions of highest stress as determined by the previous stress-coat test.) The fuel support was mounted in a fixture simulating the geometric characteristics of both the load and support in the reactor. Vertical loads only were supplied, simulating the weight load of the fuel assemblies.

It was found that the fuel support could sustain a vertical load of 104,000 pounds before the onset of yielding in the highest stressed region. This 104,000-pound load represents a safety factor in excess of 35 based on yielding over the normal applied vertical load.

A second series of tests was conducted to investigate the resulting stresses induced in the fuel support by a horizontal (or lateral) load applied by the fuel assemblies during a seismic event. A fuel support was instrumented with 15 three-element rosette strain gauges. The location of these gauges was determined from an initial computer analysis and represented the areas of highest stress plus a few key locations of minimal material thickness.

The test fixtures used were designed to apply equal loads on all four pods. This was achieved by using two hydraulic cylinders to load two spreader bars. The load was transmitted into each spreader bar through balls which prevented moment buildup. Each spreader bar then loaded two arms, which in turn loaded dummy fuel

lower tie plates. At the interface of the tie plates in the fuel support, the dimensions of these dummy tie plates were identical to those used in the production components. During loading, weight was placed at the top of the load arms approximately in the center of the fuel support. This loading simulated a vertical load which would be present due to the fuel assembly weight.

During the initial phases of the testing, it was discovered that the stresses induced by a horizontal load were a maximum when the applied vertical load was a minimum. Because the fuel support is not attached to the guide tube and sits on a chamfered seat on the guide tube provided for that purpose, it was found that an increased downward vertical load actually enhanced the fuel support's ability to sustain a horizontal load. (With increased vertical load, additional rigidity was provided to the fuel support casting by the guide tube.)

A load cell was calibrated and installed on the lower hydraulic cylinder. Load data were recorded on a continuous recorder, and strain gage data were recorded on a multi-channel recorder. The total applied load was twice the load cell readings.

The first horizontal loading applied simulated the ASME code upset condition. For this condition the total vertical load was calculated to be just under 1,000 pounds with a horizontal load of 2,600 pounds being applied. The calculated vertical load applied to the fuel casting included its weight, the upward component of 1/2g seismic load, and the differential pressure across the fuel and the fuel support. The 2,600-pound load was taken from the fuel support design specification for the upset event. A horizontal test load of 3,000 pounds was applied to compensate for possible increased hydraulic piston friction, changes in friction due to small amount of misalignment and/or cocking of the load arm in relation to the piston travel direction.

The test results simulating the upset horizontal loading conditions produced a maximum stress of 10,833 psi. The differential pressure stresses across the castings were computed. The 1,580 psi value obtained from the computation was then added to the test results. (Differential pressures across the fuel support were not simulated in the test program.) The total resultant stress was 12,413 psi for the upset condition. The total stress resultant was less than the ASME code allowable of 15,580 psi for the upset condition.

A second series of test loadings were applied to the support casting and were designed to simulate the faulted conditions. No vertical load was applied during this phase of the testing because of the net result of lg downward force due to gravity and the lg upward component of force due to the safe shutdown seismic faulted event. The horizontal test load was applied to simulate 5,200 pounds of force for the faulted event.

Testing simulating the faulted horizontal loading produced a maximum stress intensity of 21,225 psi. A computed stress value of 1,580 psi for the internal pressure was added to the test result similar to that of the upset event described above. The addition of these two stresses resulted in a maximum stress intensity of 23,505 psi, which is significantly less than the 35,400 psi allowed by ASME code for the faulted conditions.

Paragraph 3228.4 of the NG section of the ASME code and Paragraph F-1320 of Appendix F of the ASME code provide a means by which a maximum loading may be determined such that if the fuel support sustains this loading satisfactorily it will automatically confirm that A, B, and D service limits will be automatically met.

Both the upset and faulted conditions were evaluated, and it was found that the faulted code requirements were more limiting. A maximum load value of 18,875 pounds was computed and applied as a horizontal test load to the fuel support. The formula used to calculate the load value is shown in Table 3.9-34.

The results of the 18,750-pound test indicated that the load was less than the plastic instability load called for in ASME, Section III, Subsection NA, Article F-1321.3(a) and defined in

Article F-1321.1(e). The loading information in Table 3.9-34 confirmed that plastic instability was not reached. First, the applied load did not require continued pumping pressure to be maintained. Second, the horizontal tangent of the load definition curve had not been reached since a further increase to applied load was possible.

3.9.1.3.1.3 Control Rod Drive/CRDSS

The program CRDSS is used to obtain scram performance data for various inputs to the Control Rod Drive Stress Analysis for both code and non-code parts. Experimental data on pressure drops, friction factors, effects of misalignment, etc., are used in

setting up and calibrating the program. Internal drive pressures and temperatures used in the stress analysis are also determined by actual testing of the prototype Control Rod Drives.

3.9.1.3.2 Non-NSSS Equipment

Experimental stress analysis for the design of the non-NSSS equipment was not used.

3.9.1.4 Considerations for the Evaluation of Faulted Conditions

3.9.1.4.1 Evaluation of Faulted Conditions for NSSS Equipment

All seismic Category I equipment is evaluated for the faulted loading conditions. The following paragraphs in subsection 3.9.1.4 show examples of the treatment of faulted conditions for the major components on a component by component basis. Additional discussion of faulted analysis can be found in subsections 3.9.3 and 3.9.5 and Table 3.9-2.

Subsection 3.9.2.2 and Section 3.7 discuss the treatment of dynamic loads resulting from the postulated SSE. Subsection 3.9.2.5 discusses the dynamic analysis of loads affected on NSSS equipment resulting from blowdown. Deformations under faulted conditions have been evaluated in critical areas, and no cases are identified where design limits, such as clearance limits, are violated.

3.9.1.4.1.1 Control Rod Drive System Components

3.9.1.4.1.1.1 Control Rod Drives

The CRD major components that have been analyzed for the faulted conditions are: ring flange, main flange, and indicator tube.

The maximum stresses for these components for various plant operating conditions are given in Table 3.9-2u.

3.9.1.4.1.1.2 Hydraulic Control Unit

The HCU is analyzed for the seismic and hydrodynamic load conditions. Subsection 3.9.2.2.1.6.4 describes the methods of this analysis.

3.9.1.4.1.1.3 Deleted

3.9.1.4.1.2 Standard Reactor Internal Components

3.9.1.4.1.2.1 Control Rod Guide Tube

The maximum calculated stress on the control rod guide tube occurs in the base during the faulted condition. The faulted limit is 2.4 S_m, where S_m is 16,000 psi at 575 F. The analysis and the results for various plant operating conditions are summarized in Table 3.9-2aa.

3.9.1.4.1.2.2 Incore Housing

The maximum calculated stress on the incore housing occurs at the outer surface of the vessel penetration during the faulted condition. The faulted limit is the lesser value of 2.4 S_m or 0.7 S_u. Table 3.9-2ab shows that the calculated stresses are within the allowables.

3.9.1.4.1.2.3 Jet Pump

The elastic analysis of the jet pump shows that the maximum stress occurs in the jet pump riser brace during the faulted condition. Per the ASME Code, Section III, Appendix F, the faulted limit is 3.6 S_m. The analysis and stress results for various plant operating conditions are summarized in Table 3.9-2w.

3.9.1.4.1.2.4 LPCI Coupling

The location of the highest primary stress on the LPCI coupling is at the strut to shroud attachment weld. The smallest margin at the weld is from the faulted loads, resulting from a large line break plus a safe shutdown earthquake. The calculated stress is bounded by the allowable stress which is 2.4 x 1.5 x 0.7 S_m. A weld quality factor of 0.7 is included as required by the ASME Code, Section III, Table NG-3352-1. The analysis and results are summarized in Table 3.9-2y for various plant operating conditions.

3.9.1.4.1.2.5 Orificed Fuel Support

The orificed fuel support is analyzed for the faulted condition. The analysis and testing are described in subsection 3.9.1.3.1.2. Table 3.9-2b summarizing the analytical results shows that the calculated stress is within the allowable.

3.9.1.4.1.2.6 Control Rod Drive Housing

The control rod drive housing is ana1yzed for the faulted condition. The SSE and hydrodynamic loads are considered.

Table 3.9-2v shows that the maximum calculated stresses are within the allowables.

3.9.1.4.1.3 Reactor Pressure Vessel Assembly

The RPV assembly includes the RPV, support skirt, and shroud support. For the faulted condition, the RPV was evaluated using an elastic analysis. The support skirt and shroud support were also evaluated with an elastic analysis; the compressive buckling load was accounted for. The analysis and results are summarized in Table 3.9-2a.

3.9.1.4.1.4 Core Support Structure

The core support structures are evaluated for the faulted load condition on the basis of the seismic and other dynamic events described in Section 3.7 and subsection 3.9.5, respectively. The calculated stresses and the allowables are summarized in Tables 3.9-2a and b for various plant operating conditions.

3.9.1.4.1.5 Main Steam Isolation, Recirculation Gate, and Safety/Relief Valves

Tables 3.9-2g, h, and j indicate the treatment of the safety/ relief, main steam isolation, and recirculation gate valves, respectively.

Standard design rules, as defined in ASME Code, Section III, are utilized in the analysis of pressure boundary components of Seismic Category I valves. Conventional, elastic stress analysis is used to evaluate components not defined in the ASME Code. The Code allowable stresses are applied to determine acceptability of structure under applicable loading conditions including faulted condition.

3.9.1.4.1.6 Recirculation System Flow Control Valve

The recirculation system flow control valve is analyzed for faulted conditions using the elastic analysis criteria from the ASME Code, Section III. The results are summarized in Table 3.9-2f.

3.9.1.4.1.7 Main Steam and Recirculation Piping

For Main Steam and Recirculation System piping, elastic analysis methods are used for evaluating faulted loading conditions. The equivalent allowable stresses using elastic techniques are obtained from ASME Code Section III, Appendix F, "Rules for Evaluation of Faulted Conditions," and these are above elastic limits. Additional information on the main steam and recirculation piping is in Tables 3.9-2d and e.

3.9.1.4.1.8 Nuclear Steam Supply System Pumps, Heat Exchanger, and Turbines

The recirculation, ECCS, RCIC, and SLC pumps, RHR heat exchangers and RCIC turbine have been analyzed for the faulted loading conditions identified in subsection 3.9.1.1.1. In all cases, stresses were within the elastic limits. The analytical methods, stress limits, and allowable stresses are discussed in subsections 3.9.2.2 and 3.9.3.1.

3.9.1.4.1.9 Control Rod Drive Housing Supports

Examples of the analytical methods, calculated stresses and stress limits for faulted conditions for the control rod drive housing supports are shown in Table 3.9-2t.

3.9.1.4.1.10 Deleted

3.9.1.4.1.11 Fuel Assembly (Including Channels)

GE BWR fuel assembly design bases, analytical methods and evaluation results, including those applicable to the faulted conditions, are contained in References 4 and 5 for the initial core. The resulting combined acceleration under various loading conditions is less than the corresponding design basis, as shown in Table 3.9-2.

The analyses of the initial GEH core are applicable to the subsequent Siemens reloads since the fuel bundle structural response is essentially the same (References 10 and 11). The introduction of CarTech channels is addressed in Reference 11. The CarTech channels are similar to the GEH channels and have a strength in excess of that required.

The analyses for the current GEH reloads are addressed in References 13 and 16. The analyses for the AREVA NP reloads are addressed in Reference 14.

3.9.1.4.1.12 Reactor Refueling and Servicing Equipment

Refueling and servicing equipment which is important to safety is classified as essential components per the requirements of 10 CFR 50, Appendix A. This equipment and other equipment whose failure would degrade an essential component is defined in Section 9.1 and is classified as Seismic Category I. These components are subjected to an elastic dynamic finite element analysis to generate loadings. This analysis utilizes appropriate seismic floor response spectra and combines loads at frequencies up to 33 Hz for seismic and 60 Hz for hydrodynamic loads in three directions. Imposed stresses are generated and combined for normal, upset, and faulted conditions. Stresses are compared, depending on the specific safety class of the equipment, to Industrial Codes, ASME, ANSI or Industrial Standards, AISC, allowables.

Examples of the analytical methods, calculated stresses, and stress limits for the faulted conditions for the fuel storage rack fuel preparation machine, refueling platform and fuel transfer tube are shown in Table 3.9-2s.

3.9.1.4.2 Evaluation of Faulted Conditions for Non-NSSS Equipment

For statically applied loads, the stress allowables of Appendix F of ASME Section III are used for Code components. For non-Code components, allowables are based on tests or accepted standards consistent with those in Appendix F of the Code.

Dynamic loads for components loaded in the elastic range are calculated using dynamic load factors, time-history analysis, or any other method that assumes elastic behavior of the component. A component is assumed to be in the elastic range if yielding across a section does not occur. The limits of the elastic range are

defined in paragraph F-1223 of Appendix F for Code components. Local yielding due to stress concentration is assumed not to affect the validity of the assumptions of elastic behavior. The stress allowables of Appendix F for elastically analyzed components are used for Code components. For non-Code components, allowables are based on tests or accepted material standards consistent with those in Appendix F for elastically analyzed components.

In those cases where component stresses exceed yield, simplified methods of analysis based on the conservation of momentum and energy are used. When using simplified methods, the component is designed to absorb the shock or impact energy without exceeding maximum allowable strain limits, based on material properties and stress states. This type of analysis is used in pipe break analysis and is described in Bechtel Topical Report BN-TOP-2.

Where simplified methods of analysis do not adequately characterize the component response, an elastic-inelastic time-history analysis is performed using a verified computer code. Maximum allowable strain limits are based on material type and stress states.

3.9.1.5 Crane Design

Certain cranes as discussed in Section 9.1.4.2 are capable of handling heavy loads over irradiated fuel or safety related equipment. Design, construction, and use of these cranes is in accordance with the provisions of NUREG-0612 to the extent outlined in Appendix 9D.

3.9.2 Dynamic Testing and Analysis

3.9.2.1 [HISTORICAL INFORMATION] [Preoperational Vibration and Dynamic Effects Testing on Piping

3.9.2.1.1 Preoperational Testing of Safety-RelatedMechanical Equipment - (NSSS)

The test program is divided into three phases: preoperational vibration, startup vibration, and operational transients.

3.9.2.1.1.1 Preoperational Vibration Testing

The purpose of the preoperational vibration test phase is to verify that operating vibrations in the recirculation and selected portions of the RHR suction piping are within acceptable limits. This phase of the test uses visual observation to supplement remote measurements. If, during steady-state operation, visual observation indicates that vibration is significant, measurements will be made with a hand-held vibrograph. Visual observations, manual and/or remote measurements will be made during the following steady-state conditions:

a. Recirculation pumps minimum flow

b. Recirculation pumps at approximately 50 percent of rated flow

c. Recirculation pumps at maximum stable preoperational test flow

The acceptance criteria for preoperational transient vibrations (such as those due to pump start) shall be 0.125 inch (peak-to-peak) for nuclear piping.

The Grand Gulf preoperational vibration test program was developed for all ASME Class 1, 2, and 3 systems except for the following:

a. Suppression pool makeup system

b. Diesel fuel oil transfer system

c. Instrument air system associated with the pneumatic air supply to main steam safety/relief valves

d. Condensate refueling storage and transfer system associated with alternate suction to HPCS and RCIC

e. Scram portion of CRD hydraulic system

Systems with low velocity flow (that is, less than 10 feet per second for water or less than 100 feet per second for air) and with low temperature (less than 200 F) have not experienced flow induced vibration problems. The criteria are applicable to Items a through d above. In the case of the scram portion of the CRD

hydraulic system, the piping designer, Reactor Controls, Inc., recommended that monitoring was not required because their experience, which is extensive, has shown no vibration problems. The conditions of the fluid are normally less than 10 feet per second and less than 200 F. However, during a scram, the fluid velocity is in excess of 10 feet per second for less than 5 seconds.

These systems have also been reviewed with respect to mechanically induced external vibration from equipment such as pumps. Items a, c, d, and e above, have been isolated from all sources of mechanical vibration. The diesel fuel oil transfer piping has been designed so that the pump is isolated from the portion of the system that could possibly experience fatigue caused by mechanical vibration. The diesel fuel oil transfer pumps are submersible pumps located inside the fuel oil storage tank. The discharge piping is located underground from the tank to the diesel generator building. Vents, drains, components, instrumentation, etc. are located inside the diesel generator building.

Non-ASME systems that are either high energy or seismic Category I moderate energy were also reviewed and were not included in the preoperational vibration test program. These systems were not included because their failure will not impair safe shutdown of the plant.]

In Appendix 3C of the FSAR, high energy systems have been evaluated for the effects of pipe whip and jet impingement, moderate energy systems were evaluated for spraying caused by cracks, and high and moderate energy systems were evaluated for the effects of flooding. This evaluation included the most limiting cases for ASME and non-ASME systems. Failure of lines excluded from the monitoring program would not affect the ability to achieve a safe shutdown, nor would it increase offsite boundary doses in excess of 10 CFR 100 limits, 10 CFR 50.67 and 10 CFR 50 GDC 19.

3.9.2.1.1.2 Small Attached Piping

[HISTORICAL INFORMATION] [During visual observation of each of the test conditions, special attention will be given to small attached piping and instrument connections to ensure that they are not in resonance with the recirculation pump motors or flow

induced vibrations. If the operating vibration acceptance criteria are not met, appropriate corrective action will be taken. No analysis of small piping will be done.

3.9.2.1.1.3 Startup Vibration

The purpose of this phase of the program is to verify that the main steam, recirculation, and selected portions of the RHR suction piping and RCIC steam piping vibration are within acceptable limits. Because of limited access due to high radiation levels, no visual observation is required during this phase of the test. Remote measurements shall be made during the following steady state conditions:

a. Recirculation flow at minimum flow

b. Recirculation flow at 50 percent of rated

c. Recirculation flow at 75 percent of rated

d. Recirculation and main steam line flow at 100 percent of rated

e. Recirculation system piping and selected portions of the RHR suction piping at 100 percent of rated RHR flow in the shutdown cooling mode

To verify that snubbers are adequately performing their intended function during plant operation, a program for dynamic testing, as a part of normal startup operation testing, is planned.

This dynamic testing is to account for the acoustic wave due to the safety/relief valve lifts (RV1), safety relief valve load resulting from air clearing (RV2), and turbine stop valve closure load (TSVC). The maximum stresses developed in the piping by the RV1, RV2, and TSVC transients analysis is used as a basis for establishing criteria which will ensure proper functioning of the snubbers. If field measurements exceed criteria limits, this may indicate that snubbers are not operating properly. If field measurements are within criteria limits, it will be assumed that snubbers are functioning properly. Sample production snubbers of each size (i.e., 10 kips, 20 kips, 50 kips, etc.) shall also be qualified and tested for design and faulted condition loadings prior to shipment to field. Snubbers shall be tested to allow free piping movements at low velocity. During plant startup, the snubbers shall be checked for improper settings.]

3.9.2.1.1.4 Operating Transient Loads

The purpose of the operating transient test phase is to verify that pipe stresses are within Code limits. The amplitude of displacements and number of cycles per transient of the main steam and recirculation piping will be measured and displacements compared with acceptance criteria. The deflections are correlated with stresses to verify the pipe stresses remain within Code limits. Remote vibration and deflection measurements shall be taken during the following transients:

a. Recirculation pump starts

b. Recirculation pump trip at 100 percent of rated flow

c. Turbine stop valve closure at 100 percent power

d. Manual discharge of each S/R valve at 1,000 psig and at planned transient tests

3.9.2.1.1.5 Test Evaluation and Acceptance Criteria for Main Steam and Recirculation Piping

The piping response to test conditions shall be considered acceptable if the test results verify that the piping responded in a manner consistent with the predictions of the stress report and/or that the tests verify that piping stresses are within code limits, ASME Code, Section III, NB-3600. Acceptable deflection and acceleration criteria are determined after the completion of piping systems stress analysis and are provided in the startup test specifications. To ensure test data integrity and test safety, criteria have been established to facilitate assessment of the test while it is in progress. These criteria, designated Levels 1 and 2, are described in Subsections 3.9.2.1.1.5.1 and 3.9.2.1.1.5.2.

The piping systems are tested for both steady-state vibration and vibration due to operating transients. Different acceptance criteria are established for each type of vibration.

For steady-state vibration, the piping peak stress (zero to peak) due to vibration only (neglecting pressure) will not exceed 10,000 psi for Level 1 criteria and 5,000 psi for Level 2 criteria. These limits are below the piping material fatigue endurance limits as defined in Design Fatigue Curves in Appendix I of the ASME Code for 10^6 cycles.

For operating transient vibration, the piping bending stress (zero to peak) due to the operating transient only will not exceed $1.2S_m$ or pipe support loads will not exceed the Service Level D ratings for Level 1 criteria. The $1.2S_m$ limit ensures that the total primary stress, including pressure and dead weight, will not exceed $1.8S_m$, the new Code Service Level B limit. Level 2 criteria are based on pipe stresses and support loads not to exceed design basis predictions. Design basis criteria require that operating transient stresses and loads not exceed any of the Service Level B limits including primary stress limits, fatigue usage factor limits, and allowable loads on snubbers.

If all Level 2 criteria are satisfied for both steady-state vibration and operation transient vibration, there will be no fatigue damage to the piping system due to steady-state vibration and all operating transient vibrations will be less than the calculated values in the stress report. If any Level 2 limits are not satisfied, detailed engineering evaluation is needed to develop corrective action or to show that the measured results are acceptable. Any resolution must be properly documented and approved.

3.9.2.1.1.5.1 Level 1 Criteria

If in the course of the tests, measurements indicate that the piping is responding in a manner that would make test termination prudent, the test shall be terminated. The Level 1 criteria establish bounds on movement that, if exceeded, make a test hold or termination mandatory. The limits on movement are based on maximum allowable Code stress limits.

3.9.2.1.1.5.2 Level 2 Criteria

Conformance with the Level 2 criteria demonstrates that the piping is responding in a manner consistent with the stress report predictions. Failure to meet Level 2 criteria does not mean that the piping response is unsatisfactory; it means that the system is not responding in accordance with theoretical predictions and further analyses based on test results is necessary. Level 2 criteria are intended to screen out test results that are consistent with predictions and need no analytical review from those that must be evaluated.

3.9.2.1.1.6 Corrective Actions

During the course of the tests, the remote measurements shall be regularly checked to determine compliance with the Level 1 criteria. If trends indicate that the Level 1 criteria may be violated, the measurements shall be monitored at more frequent intervals. The test will be held or terminated as soon as the Level 1 criteria are violated. As soon as possible after the test hold or termination, the following corrective actions will be taken:

a. Installation Inspection. A walkdown of the piping and suspension will be made to identify any obstruction or improperly operating suspension components. Snubbers shall be about the midpoint of the total travel range at the operating temperature. Hangers shall be in their operating range between the hot and cold settings. If vibration exceeds criteria, the source of the excitation must be identified to determine if it is related to equipment failure. Action will be taken to correct any discrepancies before repeating the test.

b. Instrumentation Inspection. The instrumentation installation and calibration will be checked and any discrepancies corrected. Additional instrumentation will be added, if necessary.

c. Repeat Test. If actions a and b identify discrepancies that could account for failure to meet the Level 1 criteria, the test will be repeated.

d. Resolution of Findings. If the Level 1 criteria are violated on the repeat test or no relevant discrepancies are identified in a and b, the test results and criteria shall be reviewed to determine if the test can be safely continued.

If the test measurements indicate failure to meet the Level 2 criteria, the following corrective actions will be taken after completion of the test.

a. Installation Inspection. A walkdown of the piping and suspension shall be made to identify any obstruction or improperly operating suspension components. If vibration

exceeds criteria, the source of the vibration must be identified. Action, such as suspension adjustment, will be taken to correct any discrepancies.

b. Instrumentation Inspection. The instrumentation installation and calibration will be checked and any discrepancies corrected.

c. Repeat Test. If a and b above identify a malfunction or discrepancy that could account for failure to comply with the Level 2 criteria and appropriate corrective action has been taken, the test may be repeated.

d. Documentation of Discrepancies. If the test is not repeated, the discrepancies found under actions a and b above shall be documented in the test evaluation report and correlated with the test condition. The test will not be considered complete until the test results are reconciled with the acceptance criteria.

3.9.2.1.1.7 Measurement Locations

Remote shock and vibration measurements will be made in the three orthogonal directions near the first downstream S/R valve on each steam line; and in the three orthogonal directions on the piping between the recirculation pump discharge and the first downstream valve. During preoperational testing prior to fuel load, visual inspection of the piping will be made, and any visible vibration measured with a handheld instrument.

For each of the selected remote measurement locations, Level 1 and 2 deflection and acceleration criteria will be prescribed in the startup test specification. Level 2 criteria will be based on the results of the stress report adjusted for operating mode and instrument accuracy; Level 1 criteria will be based on maximum allowable Code stress limits.

3.9.2.1.1.8 Startup Thermal Expansion Testing

[HISTORICAL INFORMATION] [The thermal expansion preoperational and start-up testing program performed through the use of potentiometer sensors has been established to verify the normal thermal movement occurs in the systems for the NSSS piping scope of supply. The main purpose of this program is as follows:

a. The piping system during system heatup and cooldown is free to expand and move without unplanned obstruction or restraint in the x, y, and z directions.

b. The piping system does shakedown after a few thermal expansion cycles.

c. The piping system is working in a manner consistent with the assumption of the NSSS stress analysis.

d. There is adequate agreement between calculated values of displacements and measured values of displacement.

e. There is consistency and repeatability in thermal displacements during heatup and cooldown of the NSSS systems.

Limits of thermal expansion displacements have been established prior to start of piping testing to which the actual measured displacements can be compared to determine acceptability of the actual motion. If the measured displacement does not vary from the acceptance limits values by more than the specified tolerance, the piping system is responding in a manner consistent with predictions and is therefore acceptable. Two levels of limits of displacements have been established to check the systems. These are:

a. Level 1 limits which are the maximum limits that specify level of pipe motion which, if exceeded, make a test hold or termination mandatory.

   If a Level 1 limit is exceeded, the plant will be placed in a satisfactory hold condition, and the responsible piping design engineer will be advised. Following resolution, applicable tests must be repeated to verify that the requirements of the Level 1 limits are satisfied.

b. Level 2 is that specified level of pipe motion which, if exceeded, requires that the responsible piping design engineer be advised. If a Level 2 limit is not satisfied, plant operating and start-up testing plans would not necessarily be altered. Investigations of the measurements and of the criteria and calculations used to generate the pipe motion limits would be initiated. An acceptable resolution must be reached by all appropriate and involved parties, including the responsible piping design engineer

of the affected piping system. Depending upon the nature of such resolution, the applicable tests may or may not have to be repeated.

A walkdown of the piping and suspension shall be made to identify any obstruction or improperly operating suspension components. The instrumentation installation and calibration shall be checked and any discrepancies corrected. Snubbers shall be at about the midpoint of the total travel range at operating temperature. Hangers shall be in their operating range between the hot and cold settings.]

3.9.2.1.2 [HISTORICAL INFORMATION] [Preoperational and Startup Testing of Safety-Related Mechanical Equipment - (Non-NSSS)

3.9.2.1.2.1 Preoperational and Startup Vibration Testing

Safety-related piping systems designated as class 1, 2, or 3 were designed in accordance with the ASME Code, Section III. Each system was designed to withstand dynamic loadings from operational transient conditions that will be encountered during expected service as required by NB-3622, NC-3622, and ND-3622 of the Code. A preoperational vibrational testing and dynamic effects on safety-related piping will be conducted during the preoperational testing phase to determine that the piping vibrations are within acceptable limits. Vibration measurements will be conducted for loading conditions such as pump start and stop events. During the startup test phase, vibrational and dynamic effects testing will be conducted during steady-state conditions and during various operational transients, such as pump trips and valve closures. The vibration testing program is further described in Subsection 14.2.12.4. A separate procedure will exist for each of the programs at each test condition at which testing is applicable (see tests 17 and 33 in Table 14.2-3). As each phase testing plateau is completed, the monitoring data will be available for review.

Safety-related small bore and instrumentation piping listed in Table 3.9-33 have been added to the existing vibration monitoring program. This piping will be visually monitored to the extent practical (if plant operating conditions permit) and will be instrumented only where plant operating conditions require. In some cases, there are identical runs of piping within the system (e.g., control rod drive [CRD] piping). For these cases, all of

the lines will not be monitored; a sufficient number of runs will be monitored to demonstrate the adequacy of the design. For the CRD piping, for example, four runs will be monitored because all of the piping runs are designed and mounted in the same manner.

The primary objective of the preoperational testing is to verify that the piping systems behave as predicted by analytical calculations. All the measured displacements at selected locations will be compared with the calculated values. Agreement between the measured and the calculated values within acceptable limits will prove adequate performance of snubbers. The variations between the measured and the calculated values are due to minor differences between actual physical conditions and the mathematical model of the piping systems. If the observed or measured displacements do not agree with the calculated values within the established tolerance, required corrective actions will be taken. Additionally, a few selected snubbers will be instrumented to measure the snubber displacement due to piping thermal expansion and snubber forces during dynamic events.

Calculated thermal displacements and transient vibrations either have been or are being completed for the instrumented systems identified in Table 3.9-33.

The acceptance criteria for preoperational transient vibrations (such as those due to pump start) shall be 0.25 inch (peak-to-peak) for non-nuclear piping.

These limits are established based on previous experience.

Steady state vibration in the piping systems is primarily induced by the flow in the pipe and the equipment motion. In general, the nature of the steady state vibration is not known prior to its observation. Therefore, qualified design engineers with stress analysis experience and familiarity with the subject piping system will evaluate the monitored vibration to determine the system response.

For instrumented systems identified in Table 3.9-33, the systems acceptance criterion for steady state vibration is that the maximum measured amplitude shall not induce a stress in the pipe more than one half the endurance limit of the pipe material.

Material endurance limits will be obtained from the ASME Code, Section III, Appendix I. A stress level corresponding to 10^6 cycles will be used as the endurance limit. Since only one half the endurance limit is used as the acceptance criterion, the possibility of fatigue failure is considered remote.

The acceptance stress level will be converted into an equivalent displacement value using the flexure formula for a simply supported beam. The measured amplitude will be compared against the equivalent displacement values derived from the flexure formula.

3.9.2.1.2.1.1 Flow Modes

Tabulated flow modes for various systems will be provided as a part of the test program. Vibratory dynamic loadings can be placed in two categories: transient induced vibrations and steady-state vibrations. The first is a dynamic system response to a transient, time-dependent forcing function, such as fast valve closure, while the second is a constant vibration, usually flow induced.

3.9.2.1.2.1.2 Measurement Locations

The exact location of measuring devices, visual and inspection points will be supplied in the test program. Measurements taken at these points will show if the stress and fatigue limits are within the acceptable levels.

3.9.2.1.2.1.3 Acceptance Criteria

The acceptance criteria selected for non-NSSS safety-related Class 1, 2, or 3 piping will be rendered in the test program to be provided 90 days before the initiation of the test program.

3.9.2.1.2.1.4 Corrective Actions

If during the vibration test it is noted that the vibrations are beyond the acceptable design level, then additional supports and restraints would be provided. A possibility of piping re-routing would also be considered. A retest would be performed to meet the adequacy of the acceptance criteria.

3.9.2.1.2.2 Startup Thermal Expansion Testing

The object of the thermal expansion test is to verify that the piping system expands in a manner consistent with the design intent.

During the thermal expansion test, pipe deflections will be measured or observed at selected locations. One complete thermal cycle, i.e., cold position to hot position and then to cold position will be monitored. The Thermal Expansion Startup Testing Program is further described in subsection 14.2.12.4.]

3.9.2.2 Seismic and Hydrodynamic Qualification of Safety Related Mechanical Equipment

3.9.2.2.1 Seismic Qualification of Safety-Related-NSSS Equipment

This subsection describes the criteria for dynamic load qualification of safety-related mechanical equipment and also describes the qualification testing and/or analysis applicable to this plant for all the major components on a component by component basis. The methods for combining three component earthquake motions are discussed in subsections 3.7.3.6 and 3.7.3.7. In some cases, a module or assembly consisting of mechanical and electrical equipment is qualified as a unit, for example, motor powered pumps. These modules are generally discussed in this paragraph rather than providing discussion of the separate electrical parts in Sections 3.10 and 3.11. Seismic qualification testing is also discussed in subsection 3.9.3.2. Electrical supporting equipment such as control consoles, cabinets, and panels which are part of the NSSS are discussed in Section 3.10.

3.9.2.2.1.1 Tests and Analysis Criteria and Methods

The ability of equipment to perform its safety-related function during and after the application of dynamic loads is demonstrated by tests and/or analysis. Selection of testing, analysis or a combination of the two is determined by the type, size, shape, and complexity of the equipment being considered. When practical, equipment operability is demonstrated by test. Otherwise, operability is demonstrated by mathematical analysis.

Equipment which is large, simple, and/or consumes large amounts of power is usually qualified by analysis or test to show that the loads, stresses and deflections are less than the allowable maximum. Analysis and/or testing is used to show there are no natural frequencies below 33 Hz for seismic loads. If a natural frequency lower than 33 Hz is discovered, dynamic tests may be conducted and in conjunction with mathematical analysis used to verify operability and structural integrity at the required dynamic input conditions. Similar dynamic test and/or analysis is performed for hydrodynamic loads over a frequency range to include contributions from all significant modes in the total response.

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When the equipment is qualified by dynamic test the response spectrum, or time-history of the attachment point, is used in determining input motion.

Natural frequency may be determined by running a continuous sweep frequency search using a sinusoidal steady-state input of low magnitude. Seismic conditions are simulated by testing using random vibration input or single frequency input (within equipment capability) over the frequency range of interest. Whichever method is used, the input amplitude during testing envelops the actual input amplitude expected during the dynamic load conditions.

The equipment being dynamically tested is mounted on a fixture which simulates the intended service mounting and causes no dynamic coupling to the equipment.

Equipment having an extended structure, such as a valve operator, is analyzed by applying static equivalent dynamic loads at the center of gravity of the extended structure. In cases where the equipment structural complexity makes mathematical analysis impractical, a test is used to determine operational capability at maximum equivalent dynamic load conditions. Pipe-mounted equipment is analyzed in the piping system dynamic analysis.

3.9.2.2.1.2 Random Vibration Input

When random vibration input is used, the actual input motion envelops the appropriate floor input motion at the individual modes. However, single frequency input, such as sine beats, can be used provided one of the following conditions are met:

a. The characteristics of the required input motion is dominated by one frequency.

b. The anticipated response of the equipment is adequately represented by one mode.

c. The input has sufficient intensity and duration to excite all modes to the required magnitude, such that the testing response spectra will envelope the corresponding response spectra of the individual modes.

3.9.2.2.1.3 Application of Input Motion

When dynamic tests are performed, the input motion is applied to one vertical and one horizontal axis simultaneously. However, if the equipment response along the vertical direction is not sensitive to the vibratory motion along the horizontal direction, and vice versa, then the input motion is applied to one direction at a time. In the case of single frequency input, the time phasing of the inputs in the vertical and horizontal directions are such that a purely rectilinear resultant input is avoided.

3.9.2.2.1.4 Fixture Design

The fixture design will simulate the actual service mounting and cause no dynamic coupling to the equipment.

3.9.2.2.1.5 Prototype Testing

Equipment testing is conducted on prototypes of the equipment installed in this plant.

3.9.2.2.1.6 Seismic and Hydrodynamic Qualification of Specific NSSS Mechanical Components

The following sections discuss the testing or analytical qualification of NSSS equipment. Seismic qualification is also described in subsections 3.9.1.4, 3.9.3.1, and 3.9.3.2. Specific input parameters used for this analysis are shown in Table 3.9-31.

3.9.2.2.1.6.1 Jet Pumps

A dynamic analysis of the jet pumps is performed. The stresses resulting from the analysis are below the design allowables.

3.9.2.2.1.6.2 CRD and CRD Housing

The dynamic qualification of the CRD housing (with enclosed CRD) is done analytically, and the stress results of their analysis establish the structural integrity of these components. Preliminary dynamic tests are conducted to verify the operability of the control rod drive during a dynamic event. A simulated test, imposing a static bow in the fuel channels, is performed with the CRD functioning satisfactorily.

3.9.2.2.1.6.3 Core Support (Orificed Fuel Support and CR Guide Tube)

A detailed analysis imposing dynamic effects due to seismic and hydrodynamic events has shown that the maximum stresses developed during these events are much lower than the maximum allowed for the component material.

3.9.2.2.1.6.4 Hydraulic Control Unit (HCU)

The seismic and hydrodynamic loads adequacy of the hydraulic control unit is determined by test and analysis for the BWR 6 Standard Plant design and for Grand Gulf. With the back side of the HCUs mounted on beams, the seismic and hydrodynamic loads are 2 to 5 g's horizontal and 1 to 2 g's vertical in the 2.5 to 100 Hz frequency range. At these frequencies, the maximum HCU capability by test for seismic and hydrodynamic loads is 8 to 12 g's horizontal and 10 to 23 g's vertical.

The HCU analysis, as described above, is not an “elastic-plastic” analysis. GE elastic analysis considered a total of 18 individual HCUs (two back-to-back rows of 9 each) in a three-dimensional model compatible with the computer program “SAMIS.” A full detailed model of two back-to-back HCUs was inserted at the midspan for detailed study of the components and frame of an HCU. The detailed model includes the pipe frames, piston accumulators, nitrogen bottles, scram valves, brackets and main piping. Loads were calculated using response spectrum curves for SSE conditions.

3.9.2.2.1.6.5 Fuel Assembly (Including Channels)

For the initial core, the fuel assembly design bases, analytical methods and evaluation results, including seismic considerations, were addressed in References 4 and 5. The seismic analyses of the initial core are applicable to the subsequent reloads since the fuel bundle structural response is essentially the same (References 10 and 11). The introduction of CarTech channels is addressed in Reference 11. The CarTech channels are similar to the GE channels and have a strength in excess of the required. The analyses for the current GE reloads are addressed in References 13 and 16.

3.9.2.2.1.6.6 Recirculation Pump and Motor Assembly

The recirculation pump, including its appurtenances and supports, individually and as an assembly, is designed to withstand the following dynamic loads:

a. The flooded pump motor assembly is analyzed as a free body supported by constant support hangers from the brackets on the motor mounting member, with hydraulic snubbers attached to brackets on pump case and the top of the motor frame.

b. Primary stresses due to horizontal and vertical dynamic forces are considered. Horizontal and vertical dynamic forces are applied at mass centers and equilibrium reactions were determined for motor and pump brackets.

c. Load, shear, and moment diagrams were constructed to scale, using live loads, dead loads, and calculated snubber reactions. Combined bending, tension, and shear stresses were determined for each major motor flange bolting and pump case.

d. The maximum combined tensile stress in the cover bolting was calculated including tensile stress from design pressure.

e. The brackets, on the pump case, were designed to withstand loads resulting from the building seismic response.

f. Analyses have been completed which demonstrate that the natural frequency of the assembled pump and motor structure under seismic loading is greater than 33 cycles per second.

3.9.2.2.1.6.7 ECCS Pumps and Motors Assembly*

A prototype ECCS pump motor has been seismically qualified via a combination of static analysis and dynamic testing. The complete motor assembly has been seismically qualified via dynamic testing, in accordance with IEEE 344. The qualification test program included demonstration of startup and shutdown capabilities, as well as no load operability during seismic loading conditions.

For static analysis, the seismic forces of each component or assembly are obtained by concentrating its mass at the center of mass of the component or assembly, and multiplying by the seismic acceleration (earthquake coefficient). The magnitude of the earthquake coefficients is 1.25g's vertical and 3g's horizontal (except below mounting flange, 1g horizontal).

The qualification of the pump motor assemblies as units while operating under SSE conditions was provided in the form of a static earthquake acceleration analysis since the natural frequency is above 33 Hz. Under this criteria, the units were considered to be supported as designed and maximum specified vertical and horizontal accelerations being constantly applied simultaneously in the worst case combination and the results of the analysis indicate the pump is capable of sustaining the above loadings without overstressing the pump components.

3.9.2.2.1.6.8 RCIC Pump Assembly*

The RCIC pump construction is a barrel type on a large cross section pedestal. Qualification by analysis was performed. The seismic design analysis is based on 1.5g horizontal and 1.5g vertical accelerations. Results are obtained by using acceleration forces acting simultaneously in two directions, one vertical and one horizontal and calculated using the square root of the sum of the squares method. The pump mass, support system and accessory piping have been shown by analysis to have a natural frequency greater than 33 Hz.

*Hydrodynamic loads have no impact on this equipment because the auxiliary building and the containment are not on the same base-mat.

The RCIC pump assembly has been analytically qualified by static analysis for seismic loading as well as the design operating loads of pressure, temperature and external piping loads. The results of this analysis confirm that the stresses are substantially less than 90 percent of the allowable.

3.9.2.2.1.6.9 RCIC Turbine Assembly*

The RCIC turbine has been seismically qualified via a combination of static analysis and dynamic testing. The turbine assembly consists of rigid masses, wherein static analysis has been utilized, interconnected with control levers and electronic control systems, necessitating final qualification via dynamic testing. Static loading analysis has been employed to verify the structural integrity of the turbine assembly, and the adequacy of bolting under operating and seismic loading conditions. The complete turbine assembly has been seismically qualified via dynamic testing, in accordance with IEEE Std 344. The qualification test program included demonstration of startup and shutdown capabilities, as well as no load operability during seismic loading conditions.

a. Requirements

The specification for seismic qualification of the RCIC turbine and its accessories states that they shall be capable of withstanding the specified seismic accelerations at all frequencies within the range of 0.25 Hz to 33 Hz. Proper performance may be demonstrated by the seller by tests, analysis, or a combination of both. If all natural frequencies of the turbine, the component parts, and the accessories are greater than 33 Hz (as defined by test and/or analysis), a static load analysis may be performed. The seismic forces of each component or assembly are obtained by concentrating its mass at the center of mass of the component or assembly, and multiplying by the seismic acceleration (earthquake coefficient). The magnitude of each of the earthquake coefficients is 1.5g - both horizontal and vertical. If component parts and/or accessories have natural

*Hydrodynamic loads have no impact on this equipment because the auxiliary building and the containment are not on the same basemat.

frequencies below 33 Hz, these parts must be dynamically analyzed or tested, demonstrating satisfaction of the associated floor response spectra.

If the equipment capability is demonstrated by test, the equipment must be subjected to simultaneous horizontal and vertical acceleration inputs of random wave-form motion for a minimum duration of 30 seconds. The random input must envelop the defined floor response spectra.

b. Test Qualification Results

The RCIC turbine assembly was subjected to a total of 33 seismic tests (16 operating basis earthquakes and 17 safe shutdown earthquakes) with an accumulated test time of 905 seconds (-15 minutes). Input to the equipment was random wave-form motion in two directions - one horizontal and the other vertical, with sine beats superimposed at 1/3 octave intervals as necessary to envelop the required response spectra. A 200 psi, 1200 scfm air source was used as the operating medium for the turbine. The electronic governor system, the turbine hydraulic system with interconnecting piping and levers, and all turbine instrumentation were under actual operating conditions during the test program. Nozzle loadings were simulated for the turbine inlet and exhaust piping.

The test results proved the operability - startup, no load steady state, and shutdown capabilities - of the turbine assembly and its components during the required seismic loading conditions. Stress levels were not measured during the test program, since the turbine was not subjected to full pressure and temperature conditions. This qualification has been accomplished analytically, as indicated below.

c. Analytical Qualification Results

The rigid components of the RCIC turbine assembly have been analytically qualified via static analysis for seismic loading, as well as the design operating loads of pressure, temperature, and external piping loads. The results of this analysis confirm that the stresses in all components are substantially below allowable levels, i.e., less than 85 percent of allowable, with the single exception of the turbine governor valve. The “bonnet” of

this valve is stressed to 13,200 psi. Since this component receives 100 percent radiographic examination, the full allowable stress of 17,500 psi (SA216-WCB) could have been used, reducing stress level to 76 percent of "allowable."

3.9.2.2.1.6.10 Standby Liquid Control Pump and Motor Assembly

The SLC positive displacement pump and motor mounted on a common baseplate has been qualified by static analysis.

The seismic design analysis is based on 1.75g horizontal and 1.75g vertical acceleration. Results are obtained by using acceleration forces acting simultaneously in three directions, one vertical and the other two horizontal, and calculated using the square root of the sum of the squares method. The pump/motor/base assembly has been shown by static analysis to have a natural frequency greater than 33 Hz. The SLC pump and motor assembly has been analytically qualified by static analysis for seismic loading as well as the design operating loads of pressure, temperature, and external piping loads. The results of this analysis confirm that the stresses are substantially less than 90 percent of allowable.

3.9.2.2.1.6.11 RHR Heat Exchangers*

A dynamic analysis is performed to verify that the RHR heat exchanger will withstand seismic loadings. Seismic testing is an impractical method of verifying the seismic adequacy of passive equipment when predictable seismic loads can be determined by analysis.

3.9.2.2.1.6.12 Standby Liquid Control Tank

The standby liquid control storage tank is a cylindrical tank 9 feet in diameter and 12 feet high bolted to the concrete floor. The magnitude of the earthquake coefficients for safe shutdown earthquake (SSE) are 1.75g horizontal and 1.75g vertical. The standby liquid control tank has been qualified dynamically by analysis for:

a. Stresses in the tank bearing plate

*Hydrodynamic loads have no impact on this equipment because the auxiliary building and the containment are not on the same basemat.

b. Bolt stresses

c. Sloshing loads imposed at natural frequency of sloshing = 0.58 Hz

d. Minimum wall thickness

e. Buckling

The results of the analysis confirm that stresses at all investigated locations are less than their corresponding allowable values.

3.9.2.2.1.6.13 Main Steam Isolation Valves

The main steam isolation valve structures have been analyzed and representative models statically tested to demonstrate operability at the specified SSE accelerations. Static testing consisted of mechanically loading the extended mass of the valve actuator to equivalent seismic loading while valve closure was performed. Operation of the valve was demonstrated during this test. This testing and analysis satisfies the criteria in the FSAR.

3.9.2.2.1.6.14 Main Steam Safety/Relief Valves

Due to the complexity of the structure and the performance requirements of the valve, the total assembly of the safety/ relief valve (including electrical, pressure devices) were dynamically tested at seismic acceleration equal to or greater than the SSE levels determined for this plant. Tests and analysis satisfy criteria of operability as defined in the FSAR.

3.9.2.2.1.6.15 Fuel Pool Cooling and Cleanup System Pump and Motor Assembly

The analysis performed on the fuel pool cooling and cleanup system pump/motor assembly shows that the fundamental frequency on the combination of pump and motor is greater than 33 Hz, hence, static analysis was used.

This analysis shows that the pump will continue to operate if subjected to a combination of seismic and hydrodynamic loading.

Analysis also ensures that pump running clearances which include deflection of the pump shaft and pump pedestal are met during seismic and hydrodynamic loadings.

3.9.2.2.1.6.16 Standby Liquid Control Valve (Explosive Valve)

The standby liquid control valve has been generically qualified to IEEE 344 for seismic and hydrodynamic loads. The qualification tests demonstrated the absence of natural frequencies within the range of input response spectra. The tests also verified the operability of the valve subjected to dynamic loads in excess of the Required Response Spectra (RRS).

The Grand Gulf initial design conforms to the requirements of IEEE 344-1971 as modified by EICSB Branch Technical Position 10. SQRT review was subsequently performed against IEEE 344-1975. Qualification during the plant operating license stage is in accordance with IEEE 344-1975.

3.9.2.2.2 Seismic Qualification of Safety-Related Non-NSSS Equipment

3.9.2.2.2.1 Seismic Qualification Criteria

The purpose of qualifying seismic Category I mechanical equipment is to demonstrate their ability to perform their safety-related function during and after a postulated seismic occurrence of magnitude up to and including the SSE. The method for combining three component earthquake motions is discussed in subsection 3.7.3.6.

Seismic qualification of mechanical equipment was accomplished by one of the three methods or by a combination method. Refer to Table 3.9-3 for specific pumps and valves qualified and to the SQCF for seismic qualification review summaries which include non-NSSS equipment.

a. Analysis Without Testing

   1. Structural analysis without testing is used if structural integrity alone can assure the design-intended function. Equipment that falls into this category includes:

      Piping
      Ductwork

Tanks and vessels

Heat exchangers
Filters
Valves

The seismic subsystem analysis of piping is described in subsection 3.7.3.

2. Rotational analysis without testing is used to qualify heavy rotating machinery items where it must be verified that deformations resulting from seismic loadings will not cause binding of the rotating element to the extent that the component cannot perform its design-intended function. Components that fall into this category include:

   Pumps and motors
   Fans and motors

   The seismic qualification of pumps is discussed more fully in subsection 3.9.3.2. The procedure discussed therein applies with some variations to other items in this category.

   The qualification of the motors is discussed in subsection 3.9.3.

b. Dynamic Testing

Dynamic testing is used for components containing mechanisms that must change position or maintain position in order to perform their design-intended function and that, because of their complexity, do not lend themselves to analysis. Such components include:

Drywell purge compressor
Electric motor valve actuators, including internal limit switches
Pneumatic and hydraulic valve actuators, including limit switches and solenoid valves
Similar appurtenances for other mechanical equipment

c. Combinations of Analysis with Testing

A combination of analysis, static testing, and dynamic testing is used for seismic qualification of active valves whereby:

1. The manufacturer determines the first natural frequency of the valve assembly by analysis.

2. It is verified by static test that deformation due to seismic loadings will not cause binding of internal valve parts, which prevents valve operations within specified time limits.

3. The motor operator and other electrical appurtenances are qualified by dynamic testing.

The items that are qualified by testing have been mounted and operated in a manner similar to the actual system. These items are subjected to a combined vertical and horizontal input during testing.

3.9.2.2.2.2 Acceptance Criteria

The acceptance criteria which are used are as follows:

a. Tests, when used, demonstrate that the component is not prevented from performing its required safety function during and after the test.

b. Analysis, when used, verifies that stresses do not exceed the allowables specified in Tables 3.9-17 through 3.9-24 for the seismic conditions shown in Table 3.9-1 and that deformations do not exceed those that will permit the component to perform its design-intended function.

The results of typical tests and analysis of safety-related non-NSSS system mechanical equipment are listed in Appendix 3.9.3A.

3.9.2.2.2.3 Seismic Qualification of Specific Non-NSSS Mechanical Equipment

3.9.2.2.2.3.1 Standby Diesel Generator

A combination of analysis and testing has been used for the qualification of the standby diesel generator on a generic basis by DeLaval Turbine.

The following items have been qualified by analysis:

a. Generator rotor

b. Generator stator

c. Intake filter

d. Exhaust silencer

e. Intake silencer

f. Fuel oil day tank

g. Lube oil sump tank

h. Starting air tank

i. Auxiliary skid

j. Standpipe

k. Lube oil cooler

l. Jacket water cooler

m. Check valve related to air start redundancy

The following items have been qualified by shaker table testing:

a. Governor

b. Overspeed governor with shutdown valving

c. Engine-driven lube oil pump

d. Engine-driven jacket water pump

e. Engine-driven fuel oil booster pump

f. Motor-driven B & A lube oil pump

g. Motor-driven lube oil pump (auxiliary)

h. Motor-driven-jacket water pump

i. Motor-driven fuel oil booster pumps

j. Deleted

k. Deleted

l. Lube oil inlet pressure strainer

m. Fuel oil filter

n. Fuel oil strainer

o. Lube oil filter

p. Pre-lube oil filter

q. Fuel oil day tank level sensor

r. Lube oil thermostatic valve

s. Jacket water thermostatic valve

t. Deleted

u. Deleted

v. Deleted

w. Deleted

x. Intercooler

y. Solenoid-operated air start valve

z. Deleted

aa. Turbocharger

bb. Engine control panel

cc. Generator control panel

dd. Local control panel (pump and compressor)

ee. Time overcurrent relay

The items that are qualified by testing have been mounted and operated in a manner similar to the actual system. These items are subjected to a combined vertical and horizontal input during the testing.

3.9.2.2.2.3.2 Piping

The seismic analysis of piping is described in subsection 3.7.3.3.

3.9.2.2.2.3.3 Pumps

The seismic qualification of pumps and the qualification results are discussed in subsection 3.9.3.2. The drywell purge compressor has been qualified by testing and the results are included in Appendix 3.9.3A. The remaining non-NSSS supplied ASME Code, Section III pumps have been qualified by analysis; the results of the analysis are shown in Table 3.9-25.

3.9.2.2.2.3.4 Valves

The seismic qualification of active valves is discussed in subsection 3.9.3.2. The results of the qualification are described in Appendix 3.9.3A.

3.9.2.2.2.3.5 Electrical Equipment

The seismic qualification criteria and methods of qualification of Category I equipment other than those discussed in this seismic section are described in Section 3.10.

3.9.2.2.2.3.6 Tanks

The safety-related tanks have been seismically qualified as follows:

a. The seismic analysis on the Buried Diesel Generator Fuel Oil Storage Tank consisted of the following:

    1. Two-dimensional finite-element soil-structure interaction models were developed

    2. The seismic loads applied to finite element models were obtained as follows:

        (a) Free-field time histories were developed which enveloped the specified design response spectra for the tank.

        (b) Time-history motions at the base of the soil-structure interaction models were obtained by deconvoluting the free-field time histories.

Nonlinear soil properties were considered in deconvolution analyses by using an iterative procedure.

3. The response of the tank to seismic loads was evaluated by applying the deconvoluted time histories to the base of the soil-structure interaction models. These analyses included all significant soil-structure interaction effects. Analyses were performed for two earthquake levels (the Safe Shutdown Earthquake SSE - and the Operating Basis Earthquake - OBE) and for two fluid levels (full tank and half-filled tank).

4. The seismic stresses were added to other operating stresses in accordance with the loading combinations, and evaluated for the adequacy of the tank design.

For allowable stresses see Table 3.9-29b.

b. The seismic analysis on the Fuel Pool Drain Tank, Standby Diesel Generator Fuel Oil Day Tanks, HPCS Diesel Generator Fuel Oil Day Tanks, HPCS Diesel Generator Fuel Oil Day Tanks, and ADS Valve Air Receivers consist of the following:

1. A mathematical analysis of the vessel model has been performed to prove that it has rigid characteristics; i.e; the natural period of vibration of the predominant mode of the supported vessel was in the flat portion of the applicable response spectra curves.

2. The seismic acceleration coefficients were then applied statically and a static analysis was then performed on the equipment and supports. The vertical and horizontal seismic effects were applied simultaneously to the subject vessel at its gravitational center for the seismic load calculation and design.

For allowable stresses on these tanks see Tables 3.9-29a and c.

3.9.2.2.2.3.7 Standby Service Water Cooling Tower Fan

The standby service water cooling tower fan and motor were statically analyzed. Analysis on the fan and motor was performed separately.

a. Motor

This analysis utilizes a static, single-degree-of-freedom approach by applying specified floor acceleration to every major component. This method of analysis is based on "Static Coefficient Analysis."

b. Fan

The fan is statically analyzed with the possible loading modes and attendant stresses realized in the gear reducer assembly for operating and seismic loads occurring simultaneously as per "Static Coefficient Analysis."

The motor and fan results show that they will be able to perform its design intended safety-related function during and after a seismic or tornadic event.

3.9.2.2.2.3.8 Control Room Air Conditioning Units

The control room air-conditioning units have been seismically analyzed by calculating the natural frequency and combined seismic and operating stresses. Where possible some of the ancillary items were tested for structural integrity. All the components listed in the following section c were analyzed/ tested except for the fan-motor and fan where the following procedure was utilized.

a. Fan-Motor

Seismic analysis of the motor has been performed by the following procedures:

1. The stresses from the normal operating conditions when combined with the one-half safe shutdown earthquake (1/2 SSE) stresses was limited to 75 percent of the minimum yield strength of the material.

2. In addition, stresses from the normal operating conditions when combined with the safe shutdown earthquake (SSE) stresses, did not exceed the smaller of (a) the minimum yield strength or (b) 70 percent of the ultimate tensile strength of the material.

3. Two horizontal and one vertical seismic accelerations have been evaluated separately. The calculated results of the three analyses were then superimposed on a square root of the sum of the squares basis.

   The deflections were also evaluated to insure functionality and structural integrity during the safe shutdown earthquake.

   A static coefficient analysis has been performed with static coefficients being defined as 1.5 times the peak value of acceleration from the respective response spectra.

   The results indicate that the motor will be able to perform its safety intended functions during and after a seismic event.

b. Safety-related centrifugal fans have been seismically analyzed as described below:

   1. The static analysis for this centrifugal fan showed that the structure is adequate to withstand a seismic disturbance of up to 4.14 g in the horizontal-axial direction while the fan is in operation. If this disturbance occurs while the fan is not in operation, then an acceleration of 4.38 g in the same direction can be tolerated. All other g values for both operating and non-operating conditions, calculated for this fan are higher than the above figures for accelerations applied individually on each of the three axis. The directions of the acceleration loads are in relation to the shaft axis of the wheel.

   2. A calculation of the natural frequency of the base of this fan shows that it is well above the exciting frequencies produced by a seismic disturbance. The calculated natural frequency of this fan is approximately 40 Hz.

3. Loads and stresses were calculated for both operating and non-operating conditions. The deflection of the moving parts as a unit was also calculated to insure that an interference with non-moving parts would not occur under expected accelerations. Calculations indicate that in excess of 30 g acceleration is required to deflect the moving parts through 75 percent of the standard 3/16 inch clearance.

   This seismic analysis shows that the equipment analyzed will operate satisfactorily during and after the specified seismic event as required in the specification.

c. List of items analyzed/tested.

Item	Mode of Qualification
Piping	Analysis
Compressor	Analysis
Condenser	Analysis
Item	Mode of Qualification
Housing Assembly	Analysis
Control Panel Rack	Analysis
Fan Assembly Mounting	Analysis
Foundation Bolts	Analysis
Ductwork & Supports	Analysis
Instruments	Test
Electrical Control Panel	Test

All stress levels in these units will be below the yield point stress or the maximum operating stress for the materials under the combined effects of operating and dead weight loading and seismic loadings due to a safe shutdown earthquake.

These units will continue to perform its design intended function during and after a seismic event.

3.9.2.2.2.3.9 Control Room Standby Fresh Air System

The seismic qualification of these units are based on a) dynamic analysis and b) static analysis.

When functional components can be considered rigid, qualification is by static analysis. When components cannot be considered rigid, the component is qualified by dynamic analysis and the amplified response is transferred to the rigid structural support. Operating loads are always considered with biaxial seismic loads in the most conservative orientation.

The results indicated that the performance of these units will not be disrupted and they will be structurally safe for the most critical seismic conditions.

All the associated electrical components were seismically tested and were found to be capable of operating in the range of 1 Hz to 45 Hz without experiencing structural aberrations at acceleration levels of 1.20g vertical and 1.20g horizontal applied simultaneously.

3.9.2.2.2.3.10 Safety-Related Ventilation Systems Other Than Control Room

The following specific items have been seismically analyzed:

a. Coolers

   The qualification method chosen for the above coolers was dynamic analysis. The following procedure was used. 1) Mathematical models for the units were developed and 2) A modal/response analysis was performed.

b. Fans

   The following procedure was accepted in the seismic analysis of the fans. 1) Calculation of natural frequency and 2) Calculation of stress, with combined loads.

c. Pneumatic Spring Return Operators

   These units were seismically analyzed with the following procedures: 1) mathematical analysis of the most probable modes of vibration.

d. HVAC Check Dampers

The HVAC Check Dampers were seismically analyzed based on the following procedures: 1) calculation of natural frequency and 2) calculation of combined stresses.

e. Fan-Motor

The fan-motor was analyzed by the following methods: 1) Dynamic analysis or 2) The static coefficient analysis.

The fan-motor was analyzed utilizing a static, single degree of freedom approach by applying specified floor acceleration to every major component.

The results of all of the above components indicated that the units will be able to perform their design-intended functions during and after a seismic event.

3.9.2.2.2.3.11 Standby Gas Treatment System

The standby gas treatment system equipment was analyzed for the earthquake loads and operating loads and was shown not to suffer any deformation or stresses that would interfere with the functioning of the units. A mathematical model was loaded by seismic three directional shock in addition to the other mechanical loads. Natural frequency of the structure was also calculated. The following items have been analyzed:

Item	Mode of Qualification
Charcoal Filter Train	Analysis
Fan-Motors	Analysis
Fans	Analysis
Housing Frame	Analysis
Electric Heater Supporting Frame	Analysis
Plate - Housing Shell	Analysis

The following items have been seismically tested.

Item	Mode of Qualification
Instruments	Test
Heaters	Test
Control Panel, Fire Detectors and Deluge Panels	Test

The results indicate that the above items that have been analyzed/ tested will be able to perform their design-intended functions during and after the seismic event.

3.9.2.3 Dynamic Response of Reactor Internals Under Operational Flow Transients and Steady State Conditions

The major reactor components within the vessel were subjected to extensive testing coupled with dynamic system analyses to properly describe any resulting flow-induced vibration phenomena incurred from normal reactor operation and from anticipated operational transients.

The forcing function for operational flow transients and steady state conditions are not predetermined by detailed analysis. Special analysis of the response signals measured from reactor internals of similar designs are performed to obtain the parameters which determine the amplitude and modal contributions in the vibration responses. These studies are useful for extrapolating the results from tests of components with similar designs.

This vibration prediction method is particularly appropriate where standard hydrodynamic theory cannot be applied due to complexity of the structure and flow conditions. Elements of the vibration prediction method are outlined as follows:

a. Dynamic analysis of major components and subassemblies is performed to identify natural vibration modes and frequencies. The analysis models used for seismic Category I structures are similar to those outlined in subsection 3.7.2, Seismic System Analysis.

b. Data from previous plant vibration measurements is assembled and examined to identify predominant vibration response modes of major components. In general, response modes are similar but response amplitudes vary among BWRs of differing size and design.

c. Parameters are identified which are expected to influence vibration response amplitudes among the several reference plants. These include hydraulic parameters such as velocity and steam flow rates, and structural parameters such as natural frequency and significant dimensions.

d. Correlation functions of the variable parameters are developed which, multiplied by response amplitudes, tend to minimize the statistical variability between plants. A correlation function is obtained for each major component and response mode.

e. Predicted vibration amplitudes for components of the prototype plant are obtained from these correlation functions, based on applicable values of the parameters for the prototype plant. The predicted amplitude for each dominant response mode is stated in terms of a range, taking into account the degree of statistical variability in each of the correlations. The predicted mode and frequency are obtained from the dynamic analyses of paragraph a above.

The dynamic modal analysis also forms the basis for interpretation of the preoperational and initial startup test results (subsection 3.9.2.4). Modal stresses are calculated and relationships are obtained between sensor response amplitudes and peak component stresses for each of the lower normal modes. The allowable amplitude is 10,000 psi.

3.9.2.3.1 Jet Pumps, Core Support, Steam Separators, LPCI Coupling

The magnitude of the jet reaction loads applied to the reactor internal structures caused by acceleration and deceleration of the flow under normal and upset conditions are negligible compared to the differential pressure loads, and generally need not be considered. Jet reaction loads that require consideration are those associated with the jet pump assembly and riser, and within the steam separator itself. The upward jet reaction loads on the separator shroud head assembly are canceled by the downward jet impingement loads at the upper surface of the shroud head. Vibratory loads are continuously applied during normal operation and the stresses are limited to 10,000 psi to prevent fatigue failure. Prediction of vibration amplitudes, mode shapes, and frequencies for normal reactor operations are based on statistical extrapolation of actual measured results on the same or similar components in reactors now in operation.

In order to evaluate the dynamic response of the jet pumps, two locations were chosen for monitoring on an actual model of a prototype plant. These locations were the riser brace and the

nozzle of the jet pump. The reasons for selecting these positions were sensitivity and accessibility. Knowing the stress response at one of these gage locations, the stresses at other locations can be predicted as well as their mode of vibration, natural frequency, and displacement. These values are compared to those of existing plants and thus their acceptability is evaluated.

The load due to cross flow from the jet pumps to the peripheral control rod guide tubes is 620 lbs on the bottom 1/8 of the guide tube length, 345 lbs on the next higher 1/8 of the guide tube length and 130 lbs on the next 1/4 length of the guide tube.

The stresses produced due to vibratory loads are 375.5 psi and are considered negligible.

The dynamic loads due to flow-induced vibration from the feedwater jet impingement would have no significant effect on the steam separator assembly.

Analysis has shown that the impingement feedwater jet velocity is 12 ft./sec., far below the critical velocity of 118 ft./sec. Also, analysis has shown that the excitation frequency of the steam separator skirt is 5.1 Hz, and the natural frequency of the skirt is 50 Hz.

The load due to flow induced vibration will have no effect on the LPCI coupling since the calculated natural frequency of the coupling is over 50 Hz.

The calculated stresses due to the hydrodynamic forces during normal operating conditions are small and considered negligible when compared to the design allowable stresses. Locations for which calculations were made include the weld joints, elbows, and rings.

3.9.2.3.2 Impact of Single Loop Operations on Vibration

The impact of single loop operations (SLO) on the reactor vessel internals is addressed in Appendix 15C. With a recirculation pump flow limit imposed, the vibration levels during SLO should be within the acceptance limits.

3.9.2.3.3 Impact of Maximum Extended Operating Domain on Vibration

The impact of the Maximum Extended Operating Domain (MEOD) on the reactor internals is addressed in Appendix 15D. The effect of increased flow rate on the flow-induced vibration responses of the reactor internals was monitored during startup testing and evaluated to ensure the responses are within the acceptance limits.

3.9.2.4 Preoperational Flow-Induced Vibration Testing of Reactor Internals

[HISTORICAL INFORMATION] [Vibration measurement and inspection programs will be conducted during preoperational and initial startup testing of first of a kind reactor internals configuration*, in accordance with guidelines of Regulatory Guide 1.20 for prototype reactor internals. These programs will be conducted in three phases, described as follows:

a. Preoperational tests prior to fuel loading. Steady state test conditions will include balanced (two-pump) recirculation system operation and unbalanced (single pump) operation, over the full range of flow rates up to rated flow. Transient flow conditions will include single and two-pump trips from rated flow. The specified test duration will be 35 hours of balanced operation, plus 14 hours of single-pump operation of each recirculation loop. This will subject major components to a minimum of 10^6 cycles of vibration at the anticipated dominant response frequency and at the maximum response amplitudes. Vibration measurements will be obtained during this test and the inspection of internals will be conducted before and after the test.

b. Zero-power testing with fuel. This vibration measurement series is conducted with the reactor assembly complete but prior to reactor criticality. Flow conditions will include balanced, unbalanced, and transient conditions as for the first test series. The purpose of this series is to verify the anticipated effect of the fuel on the vibration response of internals. Previous vibration measurements in

*Grand Gulf is expected to be the first BWR6-251 tested.

BWRs have shown that the fuel adds damping and reduces vibration amplitudes of major internal structures. Thus, the first test series (without fuel) is a conservative evaluation of the vibration levels of these structures.

c. Initial startup testing. Vibration measurements will be made during reactor startup at conditions up to 100 percent rated flow and power. Balanced, unbalanced, and transient conditions of recirculation system operation will be evaluated. The primary purpose of this test series is to verify the anticipated effect of two-phase flow on the vibration response of internals. Previous vibration measurements in BWRs have shown that the effect of the two-phase flow is to broaden the frequency response spectrum and diminish the maximum response amplitude of the shroud and core support structures.

Vibration sensor types will include strain gages, displacement sensors (linear variable transformers), and accelerometers. Accelerometers will be provided with double integration signal conditioning to give a displacement output. Sensor locations will include the following:

Top of shroud head, lateral acceleration (displacement)
Top of shroud, lateral displacement
Jet pump riser braces, bending and extension strains
Jet pump diffuser, lateral motion or bending strain
Control rod drive housings, bending strain
Incore housings, bending strain
Core spray internal piping, bending strain

In addition to the above components, vibration of the core spray sparger and the LPCI coupling will be measured during preoperational testing of those respective systems.

In all vibration measurements, only the dynamic component of strain or displacement will be recorded. Data will be recorded on magnetic tape, and provision will be made for selective on-line analysis to verify the overall quality and level of the data. Interpretation of the data will require identification of the dominant vibration modes of each component by the test engineer, using frequency, phase, and amplitude information from the component dynamic analyses. Comparison of measured vibration

amplitudes to predicted and allowable amplitudes is made on the basis of the analytically obtained normal mode which best approximates the observed mode.

The inspections conducted prior to and following preoperational testing are for the purpose of detecting evidence of vibration, wear, or loose parts. At the completion of preoperational testing, the reactor vessel head and the shroud head will be removed, the vessel will be drained, and major components will be inspected on a selected basis. The inspections will cover the shroud, shroud head, and core support structures, the jet pumps, and the peripheral control rod drive and incore guide tubes. Access will be provided to the reactor lower plenum for these inspections.]

3.9.2.5 Dynamic System Analysis of the Reactor Internals Under Faulted Conditions

In order to assure that no significant dynamic amplification of load occurs as a result of the oscillatory nature of the blowdown forces, a comparison will be made of the periods of the applied forces and the natural periods of the core support structures being acted upon by the applied forces. These periods will be determined from a 12 vertical nodes dynamic model of the RPV and internals. Besides the real masses of the RPV and core support structures, account will be made for the water inside the RPV.

The accident analysis method is described in subsection 3.9.5.2.

The time varying pressures are applied to the dynamic model of the reactor internals described above. Except for the nature and locations of the forcing functions and the dynamic model, the dynamic analysis method is identical to that described for seismic analysis and is detailed in subsection 3.7.2.1. This dynamic load is combined with other dynamic loads, including seismic and hydrodynamic, by the absolute sum (ABS) method, although the square root of the sum of squares (SRSS) method is the design basis. The resultant is then combined with other steady-state and static loads on an absolute sum basis to determine the final design load. The result of this dynamic analysis is summarized in Table 3.9-2b.

3.9.2.6 Correlations of Reactor Internals Vibration Tests With the Analytical Results

[HISTORICAL INFORMATION] [Prior to initiation of the instrumented vibration measurement program for the prototype plant*, extensive dynamic analyses of the reactor and internals will be performed. The results of these analyses will be used to generate the allowable vibration levels during the vibration test. The vibration data obtained during the test will be analyzed in detail. The results of the data analysis, vibration amplitudes, natural frequencies and mode shapes, will then be compared to those obtained from the theoretical analysis.

Such comparisons provide the analysts with added insight into the dynamic behavior of the reactor internals. The additional knowledge gained from previous vibration tests has been utilized in the generation of the dynamic models for seismic and LOCA analyses for this plant. The models used for this plant are similar to those used for the vibration analysis of earlier prototype BWR plants.]

3.9.3 ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures

3.9.3.1 Loading Combinations, Design Transients, and Stress Limits

This section delineates the criteria for selection and definition of design limits and loading combinations associated with normal operation, postulated accidents, and specified seismic and hydrodynamic events for the design of safety-related ASME code components (except Class MC components) which are discussed in Section 3.8.

This section also lists the major ASME Class 1, 2, and 3 pressure parts and associated equipment on a component-by-component basis and identifies the applicable loadings, calculation methods, calculated stresses, and allowable stresses. Design transients for ASME Class 1 equipment are addressed in subsection 3.9.1.1. Seismic-related loads are discussed in subsection 3.9.2.2,

*[HISTORICAL INFORMATION] [Grand Gulf is expected to be the first BWR6-251 prototype plant.]

Seismic Qualification Testing of Safety Related Mechanical Equipment, and Section 3.7, Seismic Design. The suppression pool related hydrodynamic loads are described in Section 3.8.

Table 3.9-2 is the major part of this section; it presents in great detail the loading combination, analytical methods (by reference of example) and also the simulated stress or other design values for the most critical areas in the design of each component. These values are also compared with applicable code allowables. Table 3.9-2 also includes the generic loading combinations for the design of NSSS ASME Code Class 1, 2, and 3 components, supports, and safety-related reactor internals and core support structures. Table 3.9-17 presents the design loading combinations for non-NSSS ASME Code Class 1, 2, and 3 components.

A review of the structural integrity of certain pressure retaining components, reactor vessel internals components and their supports, and safety related pumps was performed for implementation of EPU as described in PUSAR Sections 2.2.2, 2.2.3, and 2.2.4. The review focused on the effects of EPU on the design basis input parameters and design basis loads and loads combinations for normal operating, upset, emergency, and faulted conditions. The review covered analyses of flow induced vibration and the analytical methodologies, assumptions, ASME Code editions, and computer programs used for these analyses. The review also included a comparison of the resulting stresses and cumulative fatigue usage factors against the code-allowable limits. Some of the information resulting from the review validated certain of the original design information presented in Table 3.9-2. Furthermore, ongoing tracking of the cyclic and transient occurrences identified in Table 3.9-35 is being done to ensure the reactor vessel and other pressure retaining components are maintained within design limits as required by the plant Technical Specifications. This tracking is accomplished by Mechanical Standard MS-40.

ASME Code Seismic Category I Fluid System Components and supports are required to be designed in accordance with rules and methods specified in the ASME Code. The design stress limits of the ASME Code are selected to ensure the pressure-retaining integrity of safety class equipment. The ASME Code requirements are supplemented by additional requirements in Regulatory Guide 1.48, “Design Limits and Loading Combinations for Seismic Category I Fluid System Components, for Class 1 Active Pumps and Valves, and for Class 2 and 3 Equipment.”

The ASME Code is recognized by industry and in the Code of Federal Regulations, as a standard whose rules and procedures provide a reliable, conservative basis for the design of nuclear safety-related equipment of very high integrity. The design limits of the ASME Code and Regulatory Guide 1.48 to the extent described in Appendix 3A and Table 3.9-4 have been met.

3.9.3.1.1 Loading Combinations, Design Transients, and Stress Limits for NSSS Components and Supports

3.9.3.1.1.1 Plant Conditions

All events that the plant might credibly experience during a reactor year are evaluated to establish a design basis for plant equipment. These events are divided into four plant conditions. The plant conditions described in the following paragraphs are based on event probability (i.e., frequency of occurrence) and correlated design conditions defined in the ASME Code, Section III.

3.9.3.1.1.1.1 Normal Condition

Normal conditions are any conditions in the course of System startup, operation in the design power range, normal hot standby (with condenser available), and System shutdown other than Upset, Emergency, Faulted, or Testing.

3.9.3.1.1.1.2 Upset Condition

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, or an operating basis earthquake. Hot standby with the main condenser isolated is an Upset Condition.

3.9.3.1.1.1.3 Emergency Condition

Those deviations from Normal Conditions which require shutdown for correction of the conditions or repair of damage in the RCPB. The conditions have a low probability of occurrence but are included to provide assurance that no gross loss of structural integrity will result as a concomitant effect of any damage

developed in the system. Emergency condition events include, but are not limited to, transients caused by one of the following: a multiple valve blowdown of the reactor vessel; loss of reactor coolant from a small break or crack which does not depressurize the reactor system nor result in leakage beyond normal makeup system capacity, but requires the safety functions of isolation of containment, and reactor shutdown; and improper assembly of the core during refueling.

3.9.3.1.1.1.4 Faulted Condition

Those combinations of conditions associated with extremely low probability, postulated events whose consequences are such that the integrity and operability of the system may be impaired to the extent that considerations of public health and safety are involved. Faulted conditions encompass events that are postulated because their consequences would include the potential for the release of significant amounts of radioactive material. These postulated events are the most drastic that must be designed against and thus represent limiting design bases. Faulted condition events include, but are not limited to, one of the following: a control rod drop accident, a fuel-handling accident, a main steam line break, a recirculation loop break, the combination of any pipe break plus the seismic motion associated with a safe shutdown earthquake plus a loss of offsite power, or the safe shutdown earthquake.

3.9.3.1.1.1.5 Correlation of Plant Conditions with Event Probability

The probability of an event occurring per reactor year associated with the plant conditions is listed below. This correlation can be used to identify the appropriate plant condition for any hypothesized event or sequence of events.

Plant Conditions	Event Encountered Probability per Reactor Year
Normal (planned)	1.0
Upset (moderate probability)	$1.0 > P > 10^{-2}$
Emergency (low probability)	$10^{-2} > P > 10^{-4}$
Faulted (extremely low probability)	$10^{-4} > P > 10^{-6}$

3.9.3.1.1.1.6 Safety Class Functional Criteria

For any normal or upset design condition event, Safety Class 1, 2, and 3 equipment is capable of accomplishing its safety functions as required by the event and shall incur no permanent changes that could deteriorate its ability to accomplish its safety functions as required by any subsequent design condition event.

For any emergency or faulted design condition event, Safety Class 1, 2, and 3 equipment shall be capable of accomplishing its safety functions as required by the event but repairs could be required to ensure its ability to accomplish its safety functions as required by any subsequent design condition event.

3.9.3.1.1.1.7 Compliance with Regulatory Guide 1.48

Compliance with Regulatory Guide 1.48 is shown in Table 3.9-4.

3.9.3.1.1.1.8 Reactor Pressure Vessel Assembly

The reactor vessel assembly consists of the reactor pressure vessel, vessel support skirt and shroud support.

The reactor pressure vessel and support are constructed in accordance with Section III of the ASME Boiler and Pressure Vessel Code. The shroud support consists of the shroud support plate and the shroud support cylinder and its legs. The reactor pressure vessel assembly components are classified as ASME Class I. Complete stress reports on these components have been prepared in accordance with ASME requirements. Table 3.9-2a summarizes the loading conditions, calculated stresses, and allowable stresses. The stress analysis performed on the reactor vessel, including the faulted conditions, were completed using elastic methods. The shroud support was also evaluated using elastic conditions except as noted in subsection 3.9.1.4.1.3. Load combinations and stress analyses for core support structures and other reactor internals are discussed in subsection 3.9.5.

3.9.3.1.1.1.9 Main Steam Piping

The main steam piping discussed in this paragraph includes that piping extending from the reactor pressure vessel to the outboard main steam isolation valve. This piping is designed in accordance with the ASME Boiler and Pressure Vessel Code, Section III, subsection NB-3600. The load combinations, stress criteria, and calculated and allowable stresses are shown in Table 3.9-2d.

The rules contained in Appendix F of ASME Code, Section III have been used in evaluating faulted loading conditions, independently of all other design and operating conditions. Stresses calculated on an elastic basis have been evaluated in accordance with F-1360.

3.9.3.1.1.1.10 Recirculation Loop Piping

This section discusses the recirculation system piping which is bounded by the reactor pressure vessel nozzles. This piping is designed in accordance with the ASME Boiler and Pressure Vessel Code, Section III, subsection NB-3600. The load combinations and the calculated and allowable stresses are shown in Table 3.9-2e.

The rules contained in Appendix F of ASME Section III have been used in evaluating faulted loading conditions, independently of all other design and operating conditions. Stresses calculated on an elastic basis have been evaluated in accordance with F-1360.

3.9.3.1.1.1.11 Recirculation System Valves

The recirculation system flow control and suction and discharge gate valves are designed in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Class I, subsection NB, Paragraph 3500. These valves are not required to operate under the safe shutdown earthquake. Loading combinations and other stress analysis information are presented in Tables 3.9-2f and j for flow control valves and gate valves, respectively.

3.9.3.1.1.1.12 Recirculation Pump

The recirculation pumps are designed in accordance with the ASME Boiler and Pressure Vessel Code, Section III. These pumps are not required to operate during the safe shutdown earthquake. The loading combinations and stress analysis are summarized in Table 3.9-2i.

3.9.3.1.1.1.13 Standby Liquid Control (SLC) Tank

The standby liquid control tank is designed in accordance with the ASME Code, Section III. The loading condition and calculated and allowable stresses are summarized in Table 3.9-2m.

3.9.3.1.1.1.14 Residual Heat Removal Heat Exchangers

The heat exchangers are designed in accordance with the ASME Boiler and Pressure Vessel Code Section III. The stress analysis and limits for the RHR heat exchangers are shown in Table 3.9-2o.

3.9.3.1.1.1.15 RCIC Turbine

Although not under the jurisdiction of the ASME Code, the RCIC turbine has been designed and fabricated following the basic guidelines for an ASME Code, Class 2 component. As such, the following information is offered:

Operating conditions for the RCIC turbine include:

a. Surveillance Testing - Quarterly operation with reactor pressure at 1000 psig, nominal, and saturated temperature turbine exhaust pressure at 25 psig, peak, and saturated temperature

b. Auto-Startup - 30 cycles per year with reactor pressure at 1150 psig, peak, and saturated temperature, turbine exhaust pressure at 25 psig, peak, and saturated temperature.

Design conditions for the RCIC turbine include:

a. Turbine Inlet - 1250 psig at saturated temperature

b. Turbine Exhaust - 165 psig at saturated temperature

Table 3.9-2q contains a summary of the calculated and allowable loads.

3.9.3.1.1.1.16 RCIC Pump

The RCIC pump has been designed and fabricated following the requirements for an ASME Code, Section III, Class 2 component. Operating conditions for the RCIC pump are simulated during surveillance testing in conjunction with the RCIC turbine. A quarterly operation test is performed where the RCIC pump takes condensate from the aboveground storage tank and at design flow discharges condensate back to the aboveground storage tank via a closed test loop.

Design conditions for the RCIC pump include:

a. Available NPSH - 24 feet
b. Total head - High speed 2980 feet
   Low speed 610 feet
c. Constant flow rate 825 gpm
d. Normal ambient operating temperature - 60 F to 122 F
e. Normal plus Upset conditions which control the pump design include:

   Design pressure - 1525 psig
   Design temperature - 40 F - 140 F
   Operating basis earthquake - 2/3 of SSE

f. Faulted or Emergency conditions include:

   Design pressure - 1525 psig
   Design temperature - 40 F - 140 F
   Safe shutdown earthquake - Horizontal - 1.5g
   Vertical - 1.5g

Table 3.9-2r contains a summary of the design calculations for the RCIC pump components.

3.9.3.1.1.1.17 ECCS Pumps

The RHR, LPCS, and HPCS pumps are designed and fabricated in accordance with the requirements of the ASME Code, Section III. The design conditions are as follows:

	RHR	LPCS	HPCS
Design pressure			
Suction	200 psig	100 psig	100 psig
Discharge	500 psig	600 psig	1575 psig
Design Temperature	40-360 F	40-212 F	40-212 F

a. Normal plus Upset condition:

   Design pressures are tabulated above. The operating basis earthquake, seismic accelerations, 2.0g horizontal and 0.5g vertical. Stress limits for pressure boundary and code allowable stress (1.0S) for general membrane and 1.5S bending (local membrane).

b. Faulted or Emergency condition:

Design pressures are tabulated above. The safe shutdown earthquake, seismic accelerations, 3.0g horizontal and 0.7g vertical. Stress limits for the pressure boundary are 120 percent of ASME Code allowable stress (1.2S) for general membrane and 1.8S for bending (local membrane).

Table 3.9-2n summarizes the design calculations for the ECCS pumps.

3.9.3.1.1.1.18 Standby Liquid Control (SLC) Pump

The standby liquid control pump has been designed and fabricated following the requirements for an ASME Code, Section III, Class 2 component.

Operating conditions for the SLC pump and motor are functional tests by pumping demineralized water through a closed test loop. The SLC pump is capable of injecting the net contents of the storage tank into the reactor in not less than 50 minutes and not more than 125 minutes. The pump is capable of injecting flow into the reactor against zero psig up to the initial set point of the reactor relief valves.

Design conditions for the SLC pump include:

a. Flow rate — 43 gpm

b. Available NPSH, minimum — 8.3 psia

c. Maximum operating discharge pressure — 1300 psig

d. Ambient conditions:

Temperature	70 F - 122 F
Relative humidity	20% - 95%

e. Normal plus upset conditions which control the pump design include:

Design pressure	1700 psig
Design temperature	150 F
Operating basis earthquake	2/3 of SSE

f. Faulted or emergency conditions include:

Design pressure	1700 psig
Design temperature	150 F
Safe shutdown earthquake	Horizontal 1.75g

g. Deleted

A summary of the design calculations for the standby liquid control pump components is provided in Table 3.9-21.

3.9.3.1.1.1.19 Main Steam Isolation and Safety/Relief Valves

These valves are designed in accordance with the requirements of the ASME Code, Section III, Subsection NB-3500, Safety Class I.

Load combination, analytical methods, calculated stresses, and allowable limits are shown in Tables 3.9-2g and 3.9-2h respectively.

3.9.3.1.1.1.20 Safety Relief Valve Piping

The relief valve discharge piping discussed in this paragraph includes only that piping extending from the relief valve discharge flange to the first downstream anchor. This piping is designed in accordance with the ASME Boiler and Pressure Vessel Code, Section III, subsection ND-3600, Class 3. The load combination, stress criteria, and allowable stresses are shown in Table 3.9-2(k). The rules contained in ASME Code Case 1606-1 were used in evaluating faulted loading conditions, independently of all other design and operating conditions.

3.9.3.1.1.1.21 Reactor Water Cleanup (RWCU) System

The RWCU pump and heat exchangers are not part of a safety system and are not designed to Seismic Category I requirements.

However, an analysis was done which shows that the fundamental frequency (first mode) of pump or motor on the combination of pump and motor is greater than 33 cps; hence a static analysis was used.

The static analysis considers static equilibrium forces on the equipment including the effect of seismic forces of 3.0g horizontal and 1.5g vertical. This analysis considers piping loads as well as torsional moment produced by the rotating assembly.

The design loading combinations and limits for the pump include the following:

a. Normal plus upset loads: This includes the simultaneous effect of normal operating loads, design pressure, temperature, nozzle loads, dead weight loads including seismic due to operational basis earthquake (OBE) loads, plus torsional loads due to rotation of the component assembly.

b. Seismic loading: This equipment and supports are designed to withstand the static seismic forces applied at the mass center, assuming that the pump is flooded.

c. Stresses in the supports and the anchor bolts due to seismic loads are combined with the stresses due to other live and dead loads and operating loads. The allowable stress for this combination of loads is based on the allowable stress as set forth in the applicable codes.

d. The ASME Boiler and Pressure Vessel Code, Section III, is used as a guide in calculating the thickness of the pressure retaining parts and for sizing the cover bolting.

e. Identify thermal transients: Equipment operates between 70 - 545 F. Transient analysis is not required for Class III components in this temperature range.

The ASME Code Section III, Class 3 component requirements, are used as guides in designing the RWCU pump and heat exchanger components.

Table 3.9-2c shows calculated values and allowable limits for the heat exchangers.

3.9.3.1.1.1.22 Fuel Pool Cooling and Cleanup System Pump and Heat Exchanger

The pump and heat exchangers are designed in accordance with the ASME Boiler and Pressure Vessel Code Section III. The stress analysis and limits are summarized in Table 3.9-2z and 3.9-2x, respectively. The pump design is also discussed in subsection 3.9.2.2.

3.9.3.1.2 Loading Combinations, Design Transients and Stress Limits for Non-NSSS Components and Supports

The combinations of design loadings categorized with respect to plant operating conditions identified as normal, upset, emergency and faulted which have been specified for the design of ASME Code constructed items are presented in Table 3.9-17. The corresponding stress limits, for each category of plant operating condition which will be specified for each type of ASME Code constructed item, are presented in Tables 3.9-18 through 3.9-24. The specified component operating condition will be the same as the plant operating condition for each transient event.

All non-NSSS piping systems affected by asymmetric pressure loadings have been analyzed using the response spectra technique. Computer codes used for this analysis were ME-101 (refer to subsection 3.9.1.2.2.5) and SUPERPIPE (refer to subsection 3.9.1.2.2.6).

Loading combination and stress level requirements as described in Table 3.9-17 have been met. In addition, all piping components in essential ASME Class 1, 2, and 3 piping systems have met the requirements of the NRC's “Interim Technical Position - Functional Capability of Passive Components” by MEB.

Piping models used in the analysis of the feedwater; RHR Loops A, B and C; HPCS and LPCS are shown in Figures 3.9-13a through 3.9-13e. In addition, Tables 3.9-32a through 3.9-32f show that the stress allowables of Table 3.9-17 have been met.

The allowables have been satisfied for this faulted event, and a final assessment of break locations has been provided in Tables 3.6A-2, 4, 9, 10 and 11; this includes fatigue data.

The non-NSSS seismic Category I mechanical equipment, that is not within the jurisdiction of the ASME Code and is located within the containment and drywell, will be qualified for the most severe combined seismic and hydrodynamic vibratory loadings for the following load combinations:

1. SRV_{ALL} or SRV_x (whichever is controlling) + OBE
2. SRV_{ALL} or SRV_x (whichever is controlling) + SSE
3. SRV_{ADS} + SSE + IBA

4. SSE + DBA

For all NSSS seismic Category I mechanical and electrical equipment located within the containment and drywell, this equipment will be qualified for the load combinations specified above.

The loading combination OBE and SRV has been considered as an upset condition and as such evaluated to Level B service limit criteria. In the fatigue analysis of the reactor vessel, vessel supports, and internals, it is assumed that the connected piping loads are at the maximum allowable dynamic load. The combination of the OBE and SRV loads on the RPV nozzle are required to be limited to the maximum allowable dynamic load. The OBE and SRV events are independently evaluated for fatigue considerations assuming that each event reaches the connected piping maximum allowable load. Therefore, the requirement for treating the OBE and SRV loads in combination has been met.

Safety-related valves, active and passive, are designed in accordance with the requirements of the ASME Code, Section III and Addenda as defined in 10 CFR 50.55a(F). Accordingly, the Code stress limits have been satisfied. Refer to subsection 3.9.3.2 for the additional requirements to assure functional adequacy of "active" valves.

All valves in seismic Category I systems having extended proportions are designed to withstand a 3.0g seismic load in any direction. The piping designer supports the piping such that these accelerations are not exceeded at the valve inlet and outlet support points. (Refer to subsection 3.7.3 for further discussion of piping seismic analysis and the modeling of inline components.) The extended parts have a natural frequency of vibration greater than 33 Hz. This was verified by either tests and/or calculations. Valve motor and air actuators were also required to be designed to withstand a 3.0g seismic load in any direction and have a natural frequency of vibration greater than 33 Hz. Refer to subsection 3.9.3.2 for additional requirements of motor and air-operated valves.

There are no safety-related ASME Sect. III, Class 1, non-NSSS supplied pumps. ASME Sect. III, Class 2 and 3 non-NSSS supplied pumps, both active and passive, are designed in accordance with ASME Section III. The pumps were designed to withstand SSE accelerations. Allowable rotor deflection is limited to the

maximum deflection that will not induce rotor binding. Pumps are designed to possess a natural frequency of 33 Hz or greater. For additional information on pump requirements see subsection 3.9.3.2.

Class 1 piping meets the criteria based on ASME Code, Section III and 10 CFR 50.55a section (d). Classes 2 and 3 meet the criteria of ASME Code, Section III.

ASME Class 1, 2, and 3 piping systems which could be exposed to jet impingement or pipe whip impact loads resulting from postulated pipe breaks in adjacent high energy piping was modeled according to its physical properties and dimensions in Bechtel's stress analysis computer program, ME102/ME632. The external impingement forces were applied to the piping system either as a uniform or concentrated load.

The maximum stress combination resulting from the jet impingement plus normal operating and seismic (SSE) load cases are then examined to ensure that the limits of stress level D have been met. If the system being impinged is required to function under the postulated event, functional capability is assured as described in Table 3.9-17.

Subsection 3.9.1 includes the design transients for ASME Code Class 1, 2 and 3 components, CS structures, and component supports.

A summary description of the computer programs used in the analysis of ASME Code, Class 1 components is given in subsection 3.9.1.2.

Inelastic methods of analysis are not the preferred methods, however, these methods may be used in those cases where it is deemed desirable and appropriate to permit significant inelastic response. In these cases, if any, the system or subsystem analysis performed is modified to include the inelastic strain compatibility in the regions of the components and component supports at which significant inelastic response is permitted.

Subsection 3.9.1.3 discusses experimental stress analysis.

3.9.3.2 Pump and Valve Operability Assurance

3.9.3.2.1 NSSS Active ASME Code Pumps and Valves

The active pumps and valves are listed in Table 3.9-3a.

Active mechanical equipment classified as Seismic Category I are designed to perform their function during the life of the plant under postulated plant conditions. Equipment with faulted condition functional requirements include "active"* pumps and valves in fluid systems such as the emergency core cooling system and the main steam system.

Safety-related valves and pumps are qualified by testing and analysis satisfying the stress and deformation criteria at critical locations. Operability is ensured by satisfying the requirements of the following programs.

3.9.3.2.1.1 ECCS Pumps

All active pumps are qualified for operability by first being subjected to rigid tests before and after installation in the plant. The in-shop tests include (1) hydrostatic tests of pressure-retaining parts to 125 percent of the design pressure times the ratio of material allowable stress at room temperature to the allowable stress value at the design temperature, (2) seal leakage 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 operating tests are bearing temperatures (except water cooled bearings) and vibration levels. Both will be shown to be below specified limits. After the pump is installed in the plant, it undergoes the cold hydro tests, functional tests, and the required periodic inservice inspection and operation. These tests demonstrate reliability of the pump for the design life of the plant.

In addition to these tests, the safety-related active pumps are analyzed for operability during a faulted condition by ensuring that (1) the pump will not be damaged during the faulted event, and (2) the pump will continue operating despite the faulted loads.

*Active equipment must perform a mechanical motion during the course of accomplishing a safety function.

3.9.3.2.1.1.1 Consideration of Loading, Stress, and Acceleration Conditions in the Analysis

In order to avoid damage during the faulted plant condition, the stresses caused by the combination of normal operating loads, SSE, and dynamic system loads will be limited to the material elastic hydrodynamic limit, as indicated in subsection 3.9.3.1 and Table 3.9-2(n). The average membrane stress (σm) for the faulted conditions loads are maintained at 1.2S, or approximately 0.75 σy (σy - yield stress) and the maximum stress in local fibers (σm + bending stress) will be limited to 1.8S, or approximately 1.1σy. The maximum nozzle loads during a faulted event will also be considered in any analysis of the pump supports to ensure that a system misalignment cannot occur.

Performing these analyses with the conservative loads stated and with the restrictive stress limits of Table 3.9-2(n) as allowables will ensure that critical parts of the pump will not be damaged during the faulted condition and that, therefore, the reliability of the pump for post-faulted condition operation will not be impaired.

A dynamic analysis is made to determine the seismic load from the applicable floor response spectra. Analysis is made to check that faulted condition nozzle loads and seismic accelerations will not impair the operability of the pumps during or following the faulted condition. Components of the pump, when having a natural frequency above 33 Hz, will be considered essentially rigid. This frequency is considered sufficiently high to avoid problems with amplification between the component and structure for all seismic areas.

In case the natural frequency is below 33 Hz, an analysis is performed to determine the amplified input accelerations necessary to perform the static analysis. The adjusted accelerations will be determined using the same conservatisms contained in the horizontal and vertical accelerations used for "rigid" structures. The static analysis is performed using the adjusted accelerations; the stress limits stated in Table 3.9-2(n) must be satisfied.

3.9.3.2.1.1.2 Pump/Motor Operation During and Following SSE Loading

Active pump/motor rotor combinations are designed to rotate at a constant speed under all conditions. Motors are designed to withstand short periods of severe overload. The high rotary inertia in the operating pump rotor, and the nature of the random, short duration loading characteristics of the seismic event, will prevent the rotor from becoming seized. In actuality, the seismic loadings will 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 will not shut down during the faulted event load and will operate at the design speed.

The functional ability of the active pumps after a faulted condition is ensured, since only normal operating loads and steady state nozzle loads exist. For the active pumps, the faulted condition is greater than the normal condition only due to seismic and hydrodynamic loads on the equipment. These events are infrequent and of relatively short duration compared to the design life of the equipment. Since it is demonstrated that the pumps would not be damaged during the faulted condition, the post-faulted condition operating loads will be no worse than the normal plant operating limits. This is ensured by requiring that the imposed nozzle loads (steady-state loads) for normal conditions and post-faulted conditions are limited by the magnitudes of the normal condition nozzle loads. The post-faulted condition ability of the pumps to function under these applied loads is proven during the normal operating plant conditions for active pumps.

3.9.3.2.1.2 SLC Pump and Motor Assembly and RCIC Pump Assembly

These equipment assemblies are small, compact, rigid assemblies, with natural frequencies well above 33 Hz. With this fact verified, each equipment assembly has been seismically qualified via static analysis only. This static qualification verifies operability under seismic conditions, and ensures structural loading stresses within Code limitations.

3.9.3.2.1.3 RCIC Turbine

The RCIC turbine has been qualified via a combination of static analysis and dynamic testing. The turbine assembly consists of rigid masses, wherein static analysis has been utilized, interconnected with control levers and electronic control systems, necessitating final qualification via dynamic testing.

Static loading analysis has been employed to verify the structural integrity of the turbine assembly, and the adequacy of bolting under operating and seismic loading conditions. The complete turbine assembly has been seismically qualified via dynamic testing, in accordance with IEEE 344. The qualification test program included demonstration of startup and shutdown capabilities, as well as no load operability during seismic loading conditions.

Subsections 3.9.1.4, 3.9.2.2, 3.9.3.1, and Table 3.9-2(q) also discuss the analysis and testing done on the RCIC turbine.

3.9.3.2.1.4 ECCS Motors

Qualification of the Class 1E motors used for the ECCS motors is in compliance with IEEE Standard 323-1971. The qualification of all motor sizes is based on completion of a type test, followed up with review and comparison of design and material details and seismic and hydrodynamic analysis of production units, ranging from 500 to 3500 bhp, with the motor used in the type test. All manufacturing, inspection, and routine tests by motor manufacturer on production units are performed on the test motor.

The type test has been performed on a 1250 hp vertical motor in accordance with IEEE Standard 323, first simulating normal operation during the design life, then the motor being subjected to a number of dynamic events, and then to the abnormal environmental condition possible during and after a loss of coolant accident (LOCA). The test plan for the type test was as follows:

a. Thermal aging of the motor electrical insulation system (which is a part of the stator only) was based on extrapolation in accordance with the temperature life characteristic curve from IEEE Standard 275 for the insulation type used on the ECCS motors. The amount of aging equaled the total estimated operation days of maximum insulation surface temperature.

b. Radiation aging of the motor electrical insulation equals the maximum estimated integrated dose of gamma during normal and abnormal conditions.

c. The normal operation-induced vibration effect on the insulation system has been simulated by 1.5g horizontal vibration acceleration of current frequency for one hour duration.

d. Normal mechanical wear on the major motor components has been simulated to the extent accomplished, through usage of the type test motor during previous pump performance testing.

e. The dynamic load deflection analysis of the rotor shaft, performed to ensure adequate rotation clearance, has been verified by static loading and deflection of the rotor for the type test motor.

f. Dynamic load aging and testing has been performed on a biaxial test table in accordance with IEEE Standard 344. During this type test the shake table was activated simulating the maximum design limit of the safe shutdown earthquake and hydrodynamic loads with motor starts and operation combination as may possibly occur during a plant life.

g. An environmental test simulating a LOCA condition with 100 days duration time has been performed with the test motor fully loaded, simulating pump operation. The test consisted of startup and six hours operation at 212 F ambient temperature and 100 percent steam. Another startup and operation of the test motor after one hour stand-still in the same environment was followed by sufficient operation at high humidity and temperature, based on extrapolation in accordance with the temperature life characteristic curve from IEEE Standard 275 for the insulation type used on the ECCS motors.

3.9.3.2.1.5 NSSS Active Valves

3.9.3.2.1.5.1 ASME Code Class 1 Active Valves

The Class 1 Active Valves are the main steam isolation valves, and the safety/relief valves. Each of these valves is designed to perform its mechanical motion in conjunction with a design basis accident including hydrodynamic loads. Qualification for operability is unique for each valve type; therefore, each method of qualification is detailed individually below.

3.9.3.2.1.5.1.1 Main Steam Isolation Valve

The MSIVs are evaluated for operability during seismic and hydrodynamic load events by both analysis and test.

a. First, the valve body is designed in accordance with the ASME B&PV Code, Section III, Class 1 (Table 3.9-2h) which limits deformation to be within the elastic limit of the material by limiting pressure and pipe reaction input loads (including seismic and hydrodynamic loads). This limits deformation in the operating area of the valve body precluding any interference with valve operability.

b. The valve actuator is operated by the plant instrument air. However, the valve is designed to be fail-safe and the safety operation of the valve to close does not depend upon the plant instrument air or an electrical operation. When the valve is in its open position, the yoke springs are held in compression. In the event that the solenoids are de-energized or the plant instrument air is interrupted for any reason, the springs are capable of closing the valve. The main steam isolation valve with the actuator is modeled in the piping analysis. The axial force and moment at the body-bonnet centerline are predicted from the analysis for the worst combination of piping loads. These values are ensured not to exceed the maximum allowable values determined from a simplified valve/actuator analysis, which uses the design g-coefficient of the static equivalent seismic load.

   Safety-related limit switches are Class 1E active equipment, which will be qualified in accordance with the requirements of IEEE 382-1980, IEEE 323-1974, and IEEE 344-1975 and NUREG-0588 (Category I).

c. The main steam isolation valves' operability following a down stream line break was demonstrated by test. The test specimen was a 20-inch valve of a design representative of the MSIVs. Operability during seismic and hydrodynamic accelerations is addressed in subsection 3.9.2.2.1.

   Environmental qualification of sensitive electric/ pneumatic equipment to meet performance requirements defined in Tables 3.11-1, 3.11-2, and 3.11-3 has been successfully completed by product design and test evaluation methods and is summarized as follows:

1. Selection of materials and requirements of periodic replacements of environmentally affected materials ensures performance during normal operation.

2. Air valve test in the 340 F steam environment of the design basis accident demonstrated performance of the required function during abnormal plant operation. The detailed qualification results are contained in the General Electric design record files.

3.9.3.2.1.5.1.2 Main Steam Safety/Relief Valves

The S/R valves are qualified by test for operability during seismic and hydrodynamic events. Structural integrity of the configuration during a dynamic event is demonstrated by both code analysis and test.

a. The valve is designed for the maximum moments which may be imposed when installed in service. These moments are resultants due to dead weight plus seismic and hydrodynamic loadings on both valve and connecting pipe, thermal expansion of the connecting pipe, and reaction forces from valve discharge.

b. A mathematical model of this valve is included in the main steam line system analysis in order to ensure that the equipment design limits are not exceeded.

c. The safety/relief valve (SRV) is qualified by type test to IEEE 344-1975. The SRV seismic testing demonstrated operability during the seismic and hydrodynamic events. The SRV is seismically qualified to response spectra induced at the valve inlet flange of 9 g's for the horizontal principal axis and for 6 g's for the vertical principal axis, with a concurrent static moment load of 800,000 in-lbs on the inlet flange and 600,000 in-lbs on the outlet flange. The SRV electrical components and interfaces are qualified by type test to IEEE 323-1974 and IEEE 344-1975. The environmental testing program for the sensitive electrical/pneumatic interface components demonstrated operability after radiation aging (28.88 X 10^6 RADS), thermal aging (840 hours at 304°F), mechanical aging (1100 cycles with a specified load), seismic/

vibration/fatigue aging, and post LOCA steam, pressure, and temperature conditions equivalent to the 100 day accident.

3.9.3.2.1.5.1.3 Deleted

3.9.3.2.1.5.1.4 HPCS Valve

G.E. has one Class 1 active valve. This valve is a motor-operated gate valve. The valve body design analysis, and testing is in accordance with the ASME B&PV Code, Section III, Class 1 requirements. The environmental testing (radiation, thermal, and mechanical aging) per IEEE 382-1972 of a specimen motor actuator will be performed for the equivalent of 60 years of normal environment and 100 days of post-LOCA environment. The generic tests per IEEE 344-1975 were conducted on the valve actuators for seismic qualification without environmental aging. The HPCS piping system analysis results for seismic and hydrodynamic loads do not exceed the qualified acceleration levels of 6.0 g (including gravity) in any of the three axes.

3.9.3.2.1.5.2 ASME Code Class 2 and 3 Active Valves

There are ten Class 2 valves--six HPCS valves, two CRD system valves and two SLC system valves (explosive valves); there are no NSSS ASME Code Class 3 active valves.

The six (6) Class 2 active valves in the HPCS system are motor-operated. The valve body design, analysis, and testing is in accordance with the ASME B&PV Code, Section III, Class 2 requirements. The valve actuators are qualified to IEEE 382-1972 generically. The tested actuator was exposed to thermal aging (at 165 F for 200 hours, 100 percent relative humidity), radiation aging (2×10^7 rads), and mechanical aging (176 cycles during thermal aging and 1800 additional cycles at ambient temperature). The seismic testing was performed after the aging, qualifying the actuator for 6.0 g acceleration in any of the three axes. It was then exposed to a post-LOCA environment of 250 F for one day, plus 200 F for 15 days at 100 percent relative humidity. The HPCS piping system analysis results for seismic and hydrodynamic loads do not exceed the qualified acceleration levels of 6.0 g (including gravity) in any of the three axes.

The two (2) Class 2 active valves in the CRD scram discharge volume vent and drain lines are air-operated. However, the valves are designed to be fail-safe; the safety operation of the valve closure does not depend upon the plant service air or on electrical operation of the controlling solenoid valves. In the event that the solenoid valves that control the globe valves are de-energized, or the plant service air is interrupted for any reason, the yoke springs held in tension are capable of closing the valve. The valves are analyzed, and type tested per IEEE 344-1975 to ensure operability during and after the dynamic loadings due to an earthquake and hydrodynamic loads.

The Class 2 active valves in the Standby Liquid Control System are explosive valves F004 A and B. The SLCS valves are qualified by type test to both IEEE 323-1971 and IEEE 344-1975. The Grand Gulf initial design conforms to the requirements of IEEE 344-1971 as modified by EICSB Branch Technical Position 10. SQRT review was subsequently performed against IEEE 344-1975. Qualification during the plant operating license stage is in accordance with IEEE 344-1975.

The environmental testing demonstrated that the explosive valve did not fire spuriously after the exposure to 100 percent relative humidity and 190 F temperature for 100 days with a radiation dosage of 22,000 rads, representing the required two years of installed life. It was also demonstrated that the explosive valve fired as intended in the normal environment. The valve is required to function only if there is no breach of the reactor coolant boundary.

The seismic and hydrodynamic load testing demonstrated operability after the completion of the dynamic event which imposes a horizontal loading equivalent to 6.5g and a vertical loading equivalent to 4.5g.

3.9.3.2.2 Active ASME Code Class 2 and 3 Pumps and Class 1, 2, and 3 Valves for Non-NSSS Systems

3.9.3.2.2.1 Pumps

The safety-related pumps are tabulated in Table 3.9-3(b). Safety-related active pumps are subjected to in-shop tests which include hydrostatic or pneumatic tests of casings per applicable code requirements, and performance tests to determine total developed head, minimum and maximum head, net positive suction head (NPSH) requirements, and other pump/motor characteristics. Where

applicable, bearing temperature and vibration are monitored during the performance tests. Environmental tests of the motor are performed in accordance with IEEE Standards 323-1971 and 334-1971 (see Section 3.11).

In addition to the required testing, the pumps are designed and supplied in accordance with the following specified criteria:

a. In order to ensure that the active pump will not be damaged during the seismic event, the pump manufacturer is required to demonstrate by test or analysis that the lowest natural frequency of the pump is greater than 33 Hz. The pump, when having a natural frequency above 33 Hz, will be considered essentially rigid. This frequency is considered sufficiently high to avoid problems with amplification between the component and structure for all seismic areas. A static shaft deflection analysis of the rotor is performed. The natural frequency of the support is determined and used in conjunction with the applicable seismic floor response spectra. The deflection determined from the static shaft analysis is compared to the allowable rotor clearances.

   In case the natural frequency is found to be below 33 Hz, an analysis is performed to determine the amplified input accelerations necessary to perform the static analysis. The static deflection analyses are performed using adjusted accelerations.

b. The maximum seismic nozzle loads are also considered in an analysis of the pump supports to ensure that unacceptable system misalignment cannot occur.

c. To complete the seismic qualification procedures, the pump motor and all appurtenances vital to the operation of the pump are independently qualified for operation during the maximum seismic event in accordance with IEEE Standard 344 (see Section 3.10).*

   If the testing option has been chosen, sine-beat or sweep testing for the electrical equipment is justified by satisfying one or more of the following requirements to

*In some cases the pump/motor were qualified as a single unit.

demonstrate that multifrequency response is negligible or the sine-beat or sine-sweep input is of sufficient magnitude to conservatively account for this effect:

1. The equipment response is basically due to one mode.

2. The 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 narrow peak at this frequency.

The degree of coupling in the equipment, in general, determines if a single or multiaxis test is required. Multiaxis testing is required if there is considerable cross-coupling. If coupling is very light, then single-axis testing is justified. Or, 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 this, it is concluded that the safety-related pump/motor assemblies will not be damaged and will continue operating under SSE loadings and will perform their intended functions. These requirements take into account for the complex characteristics of the pump and are sufficient to demonstrate and assure the seismic operability of the active pumps.

3.9.3.2.2.1.1 HPCS and Standby Service Water Pumps Test Summary

A seismic, stress and deflection analysis of the HPCS and standby service water pumps was performed. A dynamic model was developed and a computer analysis was performed. The dynamic model that was developed for the frequency analysis and analytical results are available for review upon request.

The resulting stresses, loads, and clearances are indicated in Tables 3.9-25a and d under various loading conditions.

3.9.3.2.2.1.2 Diesel Fuel Oil Transfer Pump Test Summary

A seismic, stress and deflection analysis of the submersible pump/motor was performed. For allowable stresses see Table 3.9-25c.

3.9.3.2.2.1.3 RHR Jockey Pumps Test Summary (LPCS, HPCS, and RHR Pumps)

A complete seismic stress and deflection analysis of RHR Jockey Pumps was performed in the similar manner and was found to be within the design conditions. For allowable stresses see Table 3.9-25b.

3.9.3.2.2.1.4 Drywell Purge Compressors Test Summary

This equipment was considered to be similar to a pump. A prototype test model was used in the analysis of seismic stress and deflection determination. An abstract of test results is provided in Appendix 3.9.3A. A pump/motor test was conducted as a unit with the envelope conditions described in Appendix 3.9.3A. The drywell purge compressor tests confirmed the ability to perform during adverse environmental as well as under accident conditions. Functional testing will be performed to confirm its ability to operate during emergency conditions.

3.9.3.2.2.2 Valves

The safety related valves are listed in Table 3.9-3c. In addition to the requirements specified in subsection 3.9.3.1, "active" valves are subjected to further analyses/testing to assure functional operability as outlined in this section.

Following installation, active valves will be subjected to functional qualification tests and periodic inservice operations. These tests enhance the reliability of the valves for the design life of the plant. Class 2 and 3 valves shall meet the stress limits of Table 3.9-27.

For summarized test procedures and test results see Appendix 3.9.3A.

3.9.3.2.2.2.1 Qualification of Valve Actuators

Each operator has been qualified by analysis and successful use under similar conditions, or by actual test, to demonstrate its ability to perform its function under all service conditions and environmental conditions. As a minimum, this included:

a. Shock and vibration testing

b. Heat testing

c. Live steam testing

d. Life cycle testing

e. Simulated accident environment testing

Motor and air actuators, including electrical appurtenances, were seismically qualified per IEEE Std. 344-1971 and Paragraph 6.6.6 of IEEE 344-1975. SQRT review was subsequently performed against IEEE 344-1975. Qualification during the plant operating license stage is in accordance with IEEE 344-1975. See Section 3.10. IEEE Std. 382-1972 was used as a guide for the environmental testing. (Boric acid spray was not applicable.)

3.9.3.2.2.2.2 Valve/Actuator Assembly Qualifications

The following testing requirements have been applied to a select number of sizes of production or prototype valves. The data acquired for a valve qualification were extrapolated to qualify valves which fell within a range of sizes as permitted by Table 3.9-26. The qualifications were extrapolated to valves of the next higher or lower pressure rating provided the stresses in the valve rating to which the qualification was extended was lower in all regions of the valve. The extrapolations noted above to any other size or rating was only permitted for valves of the same type (gate for gates, globe for globes) and similar configuration (gasketed bonnet for gasketed bonnet valves, etc.).

The valve/actuator assembly was installed in a suitable test facility. All valves were oriented such that the valve stem was in the horizontal position. A static load was then applied through the center of gravity of the valve operator. The static load was equal to twice the operator weight applied in the vertical direction (downward) and three times the operator weight in a direction parallel to the valve nozzle centerline applied simultaneously. Refer to the Figure 3.9-7 for a typical test arrangement. A resultant force having equivalent loads was permitted in lieu of two specified loads.

The operator that induced the greatest moment on the valve body (that is, the length from the valve nozzle centerline to the operator center of gravity multiplied by the operator weight) was used. This valve was then qualified for all other air and motor operators which may be used with the valve.

In case the frequency of the valve with top works, by test or analysis, is less than 33 Hz, a dynamic analysis of the valve is performed to determine an equivalent acceleration which is to be applied during the static test. The analysis provides the amplification of the input acceleration considering the natural frequency of the valve and frequency content of the plant floor response spectra. The adjusted accelerations are determined using a margin similar to that contained in the 3.0g horizontal and 3.0g vertical accelerations used for "rigid" valves. The adjusted accelerations are used in the static analysis. Should it ever become necessary to restrain the actuator to prevent an overstress in the pipe or a response above 3 g, the actuator will be qualified as a component support.

While statically loaded, the valve was pressurized to the design differential pressure and then cycled once (open-to-closed, closed-to-open). The time required to open or close was verified to meet the design requirements. Proper operation of valve/ actuator accessories was also verified.

Due to the simple characteristics of check valves, testable check valves and other compact valves were not subjected to functional static load testing.

3.9.3.3 Design and Installation Details for Mounting of Pressure Relief Devices

3.9.3.3.1 Devices other than Main Steam Safety/Relief Valves

The design and installation criteria applicable to the mounting of the pressure relieving devices for over-pressure protection of ASME Class 1 and 2 system components follows.

Pressure vessels are protected by pressure relieving devices to meet applicable code requirements. Detailed design of relief valve stations includes the consideration of both the local stresses at the header-to-relief valve inlet piping junction and the stresses in the relief valve inlet piping and header.

Forces and moments on the piping resulting from thrust developed by full opening of the relief valve are considered in the stress analysis. Dynamic load factors are considered in the design. The dynamic load factor is defined as the ratio of the actual (or calculated) deflection under dynamic application of a load to the deflection associated with static application of the same load.

The total forces and moments on the header resulting from simultaneous discharge of all relief valves are considered in the design of the header and its supporting restraint system.

The design of pressure relieving devices can be generally grouped in two categories: open discharge and closed discharge.

a. Open Discharge

An open discharge is characterized by a relief or safety valve discharging to the atmosphere or to a vent stack open to the atmosphere.

Regulatory Position C.1 of Regulatory Guide 1.67 recommends that the magnitude of the reaction force, the anticipated transient behavior, and the basis for their determination should be included in the design specification for the valve. Since all of the above are characteristic to each valve manufacturer, it is not a current practice to stipulate these data. Rather, the manufacturer supplies that information after selection of the valve, orifice size, inlet and outlet diameter, etc., to accommodate the flow-rates required by the design specification. This manufacturer-supplied information is then used in the design of the relief/safety valve station.

The design of open discharge valve stations includes the following considerations:

1. Stresses in the valve header, the valve inlet piping, and local stresses in the header-to-valve inlet piping junction due to thermal effects, internal pressure, seismic loads, and thrust loads are considered. These stresses are calculated in accordance with the applicable subsections of Section III of the ASME Code. These stresses are combined as shown in Tables 3.9-17 through 3.9-24 and composed to appropriate allowable stresses.

2. Thrust forces include both pressure and momentum effects.

3. Where more than one safety or relief valve is installed on the same pipe run, valve spacing is as specified in ASME Code Case 1569.

4. Where more than one safety or relief valve is installed on the same pipe run, the sequence of valve openings that induces the maximum stresses are considered as recommended by Regulatory Guide 1.67.

5. Deleted.

6. The effects of the valve discharge on piping connected to the valve header are considered.

7. The reaction forces and moments used in stress calculations include the effects of a dynamic load factor (DLF) or are the maximum instantaneous values obtained from a dynamic time-history analysis. A dynamic load factor of 2.0 is used, if a dynamic analysis is not performed to determine the dynamic load factor, as recommended by Regulatory Guide 1.67.

b. Closed Discharge

A closed discharge system is characterized by piping between the valve and a tank or some other terminal end. Under steady-state conditions, there are no net unbalanced forces. The initial transient response and resulting stresses are determined using either a time-history computer solution or a conservative equivalent static solution. In calculating initial transient forces, pressure and momentum terms are included.

3.9.3.3.2 Main Steam Safety/Relief Valves

Safety/relief valve lift results in a transient that produces momentary unbalanced forces acting on the discharge piping system for the period form opening of the safety/relief valve until a steady discharge flow from the reactor pressure vessel to the suppression pool is established. This period includes clearing of the water slug from the end of the discharge piping submerged in the suppression pool. Pressure waves traveling through the discharge piping following the relatively rapid opening of the safety/relief valve cause the safety/relief valve discharge piping to vibrate. This in turn produces forces that act on the main steam piping.

The analysis of the relief valve discharge transient consists of a stepwise time-history solution of the fluid flow equation, to generate a time-history of the fluid properties at numerous

locations along the pipe. Simultaneously, reaction loads on the pipe are determined at each location corresponding to the position of an elbow. These loads are composed of pressure-times-area, momentum change, and fluid property and pipe section load transients typical of those produced by relief valve discharge. Figure 3.9-6 shows a typical transient load on a pipe section caused by a relief valve discharge.

The method of analysis applied to determine piping system response to relief valve operation is time-history integration. The forces are applied at locations on the piping system where fluid flow changes direction, thus causing momentary reactions. The resulting loads on the safety/relief valve, the main steam line, and the discharge piping are combined with loads due to other effects as specified in subsection 3.9.3.1. The Code stress limits corresponding to load combinations classification as normal, upset, emergency and faulted, are applied to the steam and discharge pipe.

3.9.3.4 Component Supports

3.9.3.4.1 Piping

Nuclear Piping

Piping supports are designed in accordance with subsection NF of ASME Section III, with the exception that welded supports are fabricated from ASTM A 500-76 Gr B, C, or A 501-76 (structural shapes), and minimum fillet weld sizes shown in Table XVII-2452.1-1 do not apply. In subsection NF, Appendix XVII, Table 2452.1-1, minimum fillet weld size for prequalified joints is not considered mandatory where it is shown, by analysis, that a smaller weld size is adequate. Supports are either designed by load rating per paragraph NF-3260 or to the stress limits for linear supports per paragraph NF-3231. In general, the load combinations for the various operating conditions correspond to those used to design the supported pipe. Design transient cyclic data are not applicable to piping supports as no fatigue evaluation is necessary to meet the Code requirements.

The design criteria and dynamic testing requirements for component supports are as follows:

a. Component Supports

All components supports are designed, fabricated, and assembled so that they cannot become disengaged by the movement of the supported pipe or equipment after they have been installed. Class 1 NSSS component supports are required to be designed in accordance with the rules of ASME Section III, Subsection NF. To avoid buckling in the component supports, Appendices F and XVII of the Code require that the allowable loads be limited to two-thirds of the critical buckling loads. The critical buckling loads for Class 1 component supports subjected to faulted loads which are more severe than normal, upset, and emergency loads are determined by the vendor using the methods described in Appendix F of the Code.

For non-NSSS provided equipment, all component supports are evaluated for buckling in accordance with ASME Section III, Appendix XVII, Paragraph XVII-2213. The allowables for emergency and faulted conditions are increased as specified in ASME Section III, Subsection NF, Subparagraph NF-3231.1(b)(c). NSSS linear type component supports are evaluated in accordance with paragraph F-1370(c) of Appendix F of the ASME Code, Section III.

b. Hangers

The design load on hangers is the load caused by dead weight. The hangers are calibrated to ensure that they support the design load at both their hot and cold load settings. Hangers provide a specified down travel and up travel in excess of the specified thermal movement.

c. Snubbers - (E-Systems Hydraulic)

Requirements for hydraulic snubbers other than E-Systems are specified in their respective design specifications.

The design load on snubbers includes those loads caused by dynamic forces (operating basis earthquake, hydrodynamic loads and safe shutdown earthquake), system anchor movements and reaction forces caused by relief valve discharge, turbine stop valve closure, etc.

The snubbers are designed in accordance with NF-3000 to be capable of carrying the design load for all operating conditions. They are designed to be able to carry the load under normal, upset, emergency, and faulted loading conditions.

The entire piping system, including valves and suspension system between anchor points, is mathematically modeled for complete structural analysis. In the mathematical model, the snubbers are modeled as a spring with a given spring stiffness depending on the snubber size. The analysis determines the forces and moments acting on each component and the forces acting on the snubbers due to all dynamic loading conditions defined in the piping design specification. The design load on snubbers includes those loads caused by seismic forces (operating basis earthquake and safe shutdown earthquake), seismic anchor movements, and reaction forces caused by relief valve discharge, turbine stop valve closure, etc.

The snubber location and loading direction are first decided by estimation so that the stresses in the piping system will have acceptable values. The snubber locations and direction are refined by performing the computer analysis on the piping system as described above.

The spring constant required by the suspension design specification for a given load capacity snubber is compared against the spring constant used in the piping system model. If the spring constants are the same, then the snubber location and load direction have been confirmed. If the spring constants are not in agreement, they are brought in agreement, and the system analysis is redone to confirm the snubber loads. This iteration is continued until all snubber load capacities and spring constants are compatible.

The snubbers are required by the suspension design specification to be designed in accordance with all of the rules and regulations of the ASME Boiler and Pressure Vessel Code, Section III, Subsection NF. This design requirement includes analysis wherein the stresses in the snubber component parts are calculated under normal, upset, emergency, and faulted loads. These calculated

stresses are then compared against the allowable stresses of the material as given in ASME Section III Code, to make sure that they are below the allowable limit.

The snubbers are tested to ensure that they can perform as required during the operating basis earthquake (OBE), the safe shutdown earthquake (SSE), and under anticipated operational transient loads or other mechanical loads associated with the design requirements for the plant.

Two snubbers of each size and each model were tested under upset and faulted loads in the manner described below:

Snubbers were tested dynamically to ensure that they could perform as required under upset loading conditions in the following manner:

1. The snubbers were subjected to a force that varied approximately as the sine wave.

2. The frequency (Hz) of the input force was in increments of 5 Hz within the range of 3 to 33 Hz.

3. The test was conducted with the snubber at room temperature and at 200 F.

4. The peak load in both tension and compression was equal to or higher than the rated load of the snubbers.

5. The duration of the test at each frequency was 10 seconds or more.

Snubbers were tested dynamically to ensure that they could perform as required under emergency and faulted loading conditions in the following manner:

1. The snubbers were subjected to force that varied approximately as a sine wave.

2. The test was conducted with the snubbers at room temperature.

3. The peak load in both tension and compression was equal to 1.5 times the rated load of the snubbers.

4. The duration of the test was 10 seconds.

Upon completion of the above abnormal environmental transient test, the snubber shall be tested dynamically at a frequency within a specified frequency range. The snubber must operate normally during the dynamic test.

In addition to the dynamic testing of the snubbers, GEH has subjected safety relief valve piping systems of various BWR plants to safety relief valve discharge loads while monitoring the piping system for stresses. The safety relief valve discharge creates acoustic waves that propagate through the safety relief valve piping and impose momentary forces on the pipe at each change in direction. These loads have a high frequency content above 33 Hz equivalent to suppression pool dynamic loads. The results of the above testing of the piping systems demonstrate a satisfactory correlation between the actual stresses monitored in the pipe and the analytically predicted stresses in the pipe. The above analyses and testing conclude that the snubbers have been adequately qualified to ensure that they will function in a predicted manner when subjected to various dynamic loads including those at frequencies higher than 33 Hz.

An installation instruction manual is required by the suspension design specification. This manual is required to contain instructions for storage, handling, erection, and adjustments (if necessary) of snubbers. Each snubber has an installation location drawing, which contains the installation location of the snubber on the pipe and structure, the hot and cold settings, and additional information needed to install the particular snubber.

The suspension design specification requires that snubbers be provided with position indicators to identify the rod position. This indicator facilitates the checking of hot and cold settings of the snubber, as specified in the installation manual, during plant preoperational and startup testing.

The suspension design specification requires that the snubber supplier prepare an installation instruction manual. This manual is required to contain complete instructions for the testing, maintenance, and repair of the snubber. It also contains inspection points and the period for inspection. The suspension design specification

requires that hydraulic snubbers be equipped with a fluid level indicator so that the level of fluid in the snubber can be ascertained easily.

To the maximum extent practical, inspection platforms are provided for the inspection, testing, and repair or replacement of the safety-related hydraulic snubbers.

d. Struts

The design load on struts includes those loads caused by deadweight, thermal expansion, primary seismic forces, i.e., operating basis earthquake (OBE) and safe shutdown earthquake (SSE), hydrodynamic loads, system anchor displacements, and reaction forces caused by relief valve discharge, turbine stop valve closure, etc.

Struts are designed in accordance with NF-3000 to be capable of carrying the design load for all operating conditions.

e. Snubbers (Mechanical)

The design load on snubbers includes those loads caused by seismic forces (operating basis earthquake and safe shutdown earthquake), hydrodynamic loads, system anchor movements and reaction forces caused by relief valve discharge, turbine stop valve closure, etc.

Snubbers are generally used in situations where dynamic support is required because thermal growth of the piping prohibits the use of rigid supports. Snubber locations and support directions are first established using engineering judgment. The snubber locations and loading directions are then refined by performing a computer analysis on the piping system.

Using an iterative process, the pipe support is provided in such a manner that the stresses in the piping system meet code requirements.

The entire piping system is mathematically modeled for computer analysis. The model considers all spring supports, rigid supports, and snubbers. The analysis determines the forces and moments acting on each component and the forces acting on the snubbers due to all dynamic

loading conditions defined in the piping design specification. The design load is based on the loading combinations specified in Table 3.9-17.

The snubbers are designed and load rated in accordance with NF-3000 to be capable of carrying the design load for all operating conditions. Faulted condition design used the criteria outlines in Appendix F of the ASME Code. They are designed to be able to carry the load under normal, upset, emergency, and faulted loading conditions.

[HISTORICAL INFORMATION] [The prototype snubbers have been tested dynamically to ensure that they can perform as required in the following manner:

1. The snubber was subjected to force that varied approximately as the sine wave.

2. The frequency (Hz) of the input force was verified at small increments within the specified range.

3. The resulting relative displacements and corresponding loads across the working components including end attachments, were recorded.

4. The test was conducted with the snubber at various temperatures.

5. The peak load in both tension and compression tests was higher than the rated load followed by an operational test.

6. The duration of the tests at each frequency was specified.

7. Snubbers were tested for various abnormal environment conditions, including salt-fog, sand and dust, and humidity followed by operational tests.

   Each snubber was alternately loaded in tension and compression to rated (design, normal, and upset) load with the criteria that the total travel of the unit during cyclic loading, including lost motion and deflection, shall not exceed ±0.060 inch (0.120 inch total). The unit was subjected to load cycling at 100, 75, and 50 percent of rated load at frequencies

of 3, 6, 9, and 12 Hz. In addition, the unit was subjected to 100 percent rated load at frequencies between 15 Hz and 33 Hz by applying a sinusoidal load pulse alternately in tension and compression. The faulted load was applied for 1 minute in tension and compression. The snubbers were not tested for emergency loads.

Selected BOP snubbers have been subjected to force versus time loading at frequencies through the 100 Hz range. A PSA-1 snubber (1,000 pounds capacity) was tested at 60 Hz and 100 Hz frequencies using the 100 percent rated load. Another unit (PSA-10, capacity 15,000 pounds) was also tested at 60 Hz with 75 percent of rated load and at 100 Hz with 40 percent of rated load. The 100 percent rated load was attained at 3 Hz. The capability of the test apparatus (MTS machine) prevented attaining full rated load at higher frequencies.]

Additionally, all snubbers have been qualified by test for lateral loads up to 6g.

Snubbers were disassembled and inspected after the test. Inspection revealed that all units were in satisfactory condition.

The snubbers are required by the suspension design specification to be designed in accordance with all of the rules and regulations of the ASME Boiler and Pressure Vessel Code, Section III, Subsection NF. This design requirement includes analysis wherein the stresses in the snubber component parts are calculated under normal, upset, emergency, and faulted loads. These calculated stresses are then compared against the allowable stresses of the material as given in ASME Code Section III, to make sure that they are below the allowable limit.

An installation instruction manual is required by the suspension design specification. This manual is required to contain instructions for storage, handling, erection, and adjustments (if necessary) of snubbers. Each snubber has an installation location drawing, which contains the installation location of

the snubber on the pipe and structure, the hot and cold settings, and additional information needed to install the particular snubber.

To the extent practical, easy access to individual snubbers has been provided for the inspection, testing, and replacement of safety-related snubbers.

f. Anchors are designed to secure at the desired points of piping in relatively fixed positions. They permit piping to expand and contract freely as directed from the anchor and are structurally capable of withstanding thrusts, moments, and other imposed loads.

Non-Nuclear Piping

The design of all supports for non-nuclear piping shall satisfy the requirements of ANSI B31.1, paragraphs 120 and 121.

3.9.3.4.2 Reactor Pressure Vessel (RPV) Support Skirt

The RPV support skirt is analyzed to meet the requirements of the ASME Code, Section III, Subsection NF. The loading conditions, stress criteria, calculated stresses, and allowable stresses in the critical support areas are summarized in Table 3.9-2a.

The allowable buckling loads for the reactor pressure vessel support skirt were determined from the buckling criteria given below.

Support Skirt Cylinder Buckling Criteria

For the support skirt cylinder, the permissible load "P" and moment "M" at any elevation under consideration shall interact and be limited by the following:

$$\frac{P}{A} + \frac{MC}{I} \leq \frac{2}{3} KSm$$

Where

Sm = Design stress intensity at temperature* for the elevation under consideration
P/A = Primary membrane compressive stress over the entire cross section
MC/I = Maximum primary bending compressive stress due to the moment at elevation under consideration
K = Appropriate event factor as defined below:
= 1.0 for design, testing, normal, and upset conditions
= 1.2 for emergency conditions
= 1.5 for faulted conditions

*This is the material temperature at the section under consideration. The design temperature for the support skirt shall be assumed to vary linearly with elevation from a temperature of 575 F at the support skirt to the bottom head juncture, to 120 F at the support skirt flange base, or the skirt temperature actually determined by the detailed analysis flange may be used.

GE also has performed a generic analysis for the BWR/6 reactor pressure vessel support skirt buckling which shows that the vessel support skirt can comply with paragraph 1370(c) of Appendix F of the ASME Code, Section III, and the loads will not exceed 0.67 times the critical buckling strength of the support skirt.

3.9.3.4.3 NSSS Floor Mounted Equipment (Pumps, Heat Exchangers, SLC Tank, and RCIC Turbine)

The high pressure core spray, low pressure core spray, residual heat removal, reactor core isolation cooling, standby liquid control, fuel pool cooling and cleanup and reactor water cleanup system pumps; the residual heat removal, reactor water cleanup, and fuel pool heat exchangers; the SLC tank; and the RCIC turbine are all analyzed to verify the adequacy of their support structure under various plant operating conditions. In all cases, the stress loads in the critical support areas are within ASME Code allowables. The loading conditions, stress criteria, and allowable stresses in the critical support areas are summarized in Table 3.9-2 under the respective equipment table.

3.9.3.4.4 Supports for ASME Code Class 1, 2, and 3 Active Components

ASME Code Class 1, 2, and 3 active components are either pumps or valves. Since valves are supported by piping and not tied to building structures, pipe design criteria govern.

Seismic Category I active pumps supports are qualified for dynamic (seismic and hydrodynamic) loads by testing when the pump supports along with the pumps are fulfilling the following conditions:

a. Simulate actual mounting conditions

b. Simulate all static and dynamic loadings on the pump

c. Monitor pump operability during testing

d. The normal operation of the pump during and after the test indicates that the supports are adequate. Any deflection or deformation of the pump supports which precludes the operability of the pump is not accepted.

e. Supports are inspected for structural integrity after the test. Any cracking or permanent deformation is not accepted.

Dynamic qualification of component supports is accomplished by checking that the stresses at all support elements and parts such as pump holddown, baseplate holddown bolts, pump support pads, pump pedestal, and foundation are within allowable limits as specified in the ASME Code, Subsection NF.

For normal and upset plant conditions, the deflections and deformations of the supports are assured to be within the elastic limits and not exceed the values permitted by the designer based on his design verification tests to ensure the operability of the pumps.

For emergency and faulted plant conditions, the deformations must not exceed the values permitted by the designer to ensure the operability of the pumps.

3.9.3.4.5 Component Supports (non-NSSS)

The loadings as specified in the design specifications are taken into account in designing component supports for ASME Code constructed items. These loadings include but are not limited to the following.

a. Weight of the component and normal contents under operating and test conditions

b. Weight of the component support

c. Superimposed loads and reactions induced by the adjacent system components

d. Dynamic loads, including loads caused by earthquake vibration

e. Restrained thermal expansion

f. Anchor and support movement effects.

All non-NSSS Class 1, 2, and 3 component supports are evaluated for buckling in accordance with ASME Section III, Appendix XVII, paragraph 2213 and Appendix F, paragraph 1370(c).

Loadings and stress limits for ASME Section III, Class 2 and 3 component supports are discussed in subsection 3.9.3.1. Active component supports are considered in subsection 3.9.3.2. The stress limits are per NF-3000 for design, normal, and emergency conditions and per ASME Section III, Appendix F, for faulted conditions.

Other non-seismic equipment located inside the containment, such as HVAC ducting, conduits, and cable trays are also supported to seismic requirements.

The above design criteria are used to ensure that non-seismic equipment located inside containment will not compromise the integrity of safety-related equipment.

APPENDIX 3.9.3A

I. Preface

The objective of this Appendix is to provide documentation of the operability assurance of equipment that is required to perform its safety-related function during and after a seismic or a LOCA condition. The following section includes typical tests, analysis or a combination of test and analysis to provide the operability of the subject equipment.

Attached are typical results on valves, valve actuators, and other associated equipment.

The drywell purge compressors were qualified by testing; the results are included in this Appendix. Typical qualification by analysis for pumps is shown in Tables 3.9-25a, b, c, and d.

Due to the voluminous accumulation of test programs and test results, a summarized form is addressed in the FSAR. However, if the NRC requires to peruse the documents that are available in the files, they will be forwarded upon request.

APPENDIX 3.9.3A

II. Exhibits

Exhibit: 1

Title: Active ASME Section III, Class 1, 2, or 3 Motor-Actuated Valves 2 Inches and Smaller

Items: Valves, valve actuators, and limit switches

Purpose:

The purpose of this report is to describe the acceptable combinations of design analysis and/or testing procedures implemented to ensure that motor actuated valves will function during and after the most adverse combinations of loadings to which valves may be subjected.

Description of Test Program:

The following tests have been performed to check the adequacy of the valve operability to perform its intended function: a) seismic qualification of valve with its associated components and b) environmental qualification of actuator and its associated components.

a. Seismic Qualification of Valves

    1. By static testing

    2. By natural frequency (seismic stress analysis)

    3. By analysis

b. Environmental Qualification of Valve Actuators

    1. The testing plan as per IEEE-382-72 consists of three basic parts:

        (a) Aging simulation

            (1) Thermal aging

            (2) Mechanical aging

            (3) Radiation aging

Seismic qualification

(c) Accident simulation

Results:

(b)

1. Valve assembly: was subject to static loading and was determined to be within the acceptable requirements. The valve assembly will also be able to perform with the design-intended worst environmental conditions.

2. Valve actuator and its associated components: were subjected to seismic loading and combinations of more severe external loadings (i.e., hydrodynamic loads, safety/relief valve blowdown, etc.) and were found to function satisfactorily during and after the dynamic loading with the natural frequencies within the range of 1 to 100 Hz. Valve actuator and its associated components were capable of performing their required function during and after post-LOCA or HELB conditions.

Consequently, the results indicate that the valve and its actuator will perform its function during and after the most adverse plant condition.

Exhibit: 2

Title: Active Power-Actuated Butterfly Valves

Items: Valves and actuators for motor-actuated and pneumatically actuated 6-inch through 44-inch butterfly valves

Purpose:

The functional qualification consists of tests and/or analyses to demonstrate that a valve assembly will perform its required safety-related function during and after post-LOCA and HELB conditions.

Description of Test Program:

a. The valves were subjected to a series of tests to demonstrate the valves' capability to perform their intended function. These tests included:

   1. Valve leakage test

2. Cold cycle tests

3. Hot cycle tests

4. Temperature-pressure testing

5. Seismic loading test

   This test is part of the operability assurance program. Static loads of at least 6g are applied simultaneously in two orthogonal directions while the valve is cycled to demonstrate operability under the most adverse combinations of external loadings (i.e., hydrodynamic loads, safety/relief valve blowdown, etc.). Static loads of at least 3g are applied simultaneously in two orthogonal directions to valves subjected to seismic loading only.

6. Vibration testing

   This test verifies the valve operability when subjected to seismic vibration. An exploration vibration test is performed by a sweep of the full frequency range (usually the range is from 3.5 Hz to 95 Hz). A variable frequency test is run where loads of at least 0.5g at 12 different frequencies are applied while the valve is operated. A vibration endurance test is also performed where the valve assembly is vibrated for at least 3 minutes at 33 Hz (and at any resonant frequencies).

b. The valve actuator is subjected to environmental qualification tests in accordance with IEEE Std. 382-1972 which consists of:

   1. Aging simulation

      (a) Thermal aging

      (b) Mechanical aging

      (c) Radiation aging

   2. Seismic qualification

   3. Accident simulation

c. Conclusion: There was no evidence of physical damage and/or electrical malfunction as a result of these tests. Therefore, the results indicate that the valves and their actuators will perform their function during and after the most adverse plant conditions.

Exhibit: 3

Title: Active Power-Actuated Gate and Globe Valves -2-1/2 Inches and Larger

Items: Valves and actuators for motor-actuated and pneumatically actuated gate and globe valves

Purpose:

The functional qualification consists of tests and/or analyses to demonstrate that a valve assembly will perform its required safety-related function.

Description of Test Program:

a. The valves were subjected to a static loading test which is part of the operability assurance program. Static loads of at least 6g are applied simultaneously in two orthogonal directions while the valve is cycled to demonstrate operability under the most adverse conditions of external loadings (i.e., hydrodynamic loads, safety/relief valve blowdown, etc.). Static loads of at least 3g are applied simultaneously in two orthogonal directions to valves subjected to seismic loading only.

b. The valve actuator was subjected to environmental qualification tests in accordance with IEEE Std. 382-1972 which consists of:

    1. Aging simulation

        (a) Thermal aging

        (b) Mechanical aging

        (c) Radiation aging

    2. Seismic qualification

    3. Accident simulation

c. Conclusion: There was no evidence of physical damage and/or electrical malfunction as a result of these tests. Therefore, the results indicate that the valves and actuators will perform their function during and after the most adverse plant conditions.

Exhibit: 4

Title: Drywell Purge Compressors

Items: Compressor and Motor

Reference:

Test Report For
"Drywell Purge Compressor (SC-6) Prototype-TBI-77-TR-5"
By
Turbonetics, Inc.
968 Albany Shaker Road
Latham, N.Y. 12110

Purpose:

The objective of this test is to confirm the ability of the prototype compressor/motor to fulfill the requirements as per IEEE Std. 323-1974 and IEEE Std. 334-1974 and to perform a safety-related function during and after an accident condition.

Description of Test Program:

The test program consists of a) thermal test, b) aging test, c) radiation test, and d) seismic qualification test.

	Test Conditions	Plant Conditions
Pressure (psig)	15.3	15.3
Temperature (F)	192*	200*
Radiation (rads)	1.1×10^8	(normal) 1×10^7 (accident) 6.5×10^6

*The fact that the test condition is less than the plant condition (192 degrees being less than 200 degrees) was verified as being satisfactory in accordance with the "Response to NUREG-0588," Specification M-050.1.

Results:

The test results indicate that the said equipment will perform its safety-intended function during and after a LOCA or a seismic condition. Hence, the operability of the equipment is assured.

3.9.4 Control Rod Drive System

This plant is equipped with a hydraulic control rod drive system which includes the control rod drive mechanism (CRDM), the hydraulic control unit (HCU), the condensate supply system and the scram discharge volume and extends to the coupling interface with the control rods.

3.9.4.1 Descriptive Information on CRDS

Descriptive information on the control rod drives as well as the entire control and drive system is contained in Section 4.6.

3.9.4.2 Applicable CRDS Design Specifications

The control rod drive system (CRDS) is designed to meet the functional design criteria as outlined in Section 4.6 and consists of the following:

a. Locking piston control rod drive

b. Hydraulic control unit

c. Hydraulic power supply (pumps)

d. Interconnecting piping

e. Flow, pressure, and isolation valves

f. Instrumentation and electrical controls

Quality group classification is not applicable for the CRD.

The pressurized components of the CRD are designed according to ASME Code Section III.

The quality group classification of the CRD hydraulic system is outlined in Table 3.2-1 of this FSAR, and the components are designed according to the codes and standards governing the individual quality groups.

Pertinent aspects of the design and qualification of the CRD components are discussed in the following locations: transients in subsection 3.9.1.1, faulted conditions in subsection 3.9.1.4, seismic testing in subsection 3.9.2.2.

3.9.4.3 Design Loads, Stress Limits, and Allowable Deformation

The ASME Code components of the CRDs and CRD housings have been evaluated analytically and the design loading conditions, stress criteria, and calculated and allowable stresses are listed in Table 3.9-2u and 3.9-2v. For the non-code components, experimental testing was used to determine the CRD performance under all possible conditions as described in subsection 3.9.4.4.

Deformation has been compared with the allowable and is not a limiting factor in the analysis based upon the numerous tests performed on the drive.

3.9.4.3.1 Control Rod Drive Housing Supports

The control rod drive (CRD) housing support system functions are described in Section 4.6.

The American Institute of Steel Construction (AISC) Manual of Steel Construction, "Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings," was used in designing the CRD housing support system. However, to provide a structure that absorbs as much energy as practical without yielding, the allowable tension and bending stresses used were 90 percent of yield and the shear stress used was 60 percent of yield. These design stresses are 1.5 times the AISC allowable stresses (60 percent and 40 percent of yield, respectively).

For purposes of mechanical design, the postulated failure resulting in the highest force in an instantaneous circumferential separation of the CRD housing from the reactor vessel, with the reactor's operating pressure of 1086 psig (at the bottom of the vessel) acting on the area of the separated housing. The weight of the separated housing, control rod drive, and blade, plus the pressure of 1086 psig acting on the area of the separated housing, gives a force of approximately 32,000 lb. This force is used to calculate the impact force, conservatively assuming that the housing travels through a 1-in. gap before it contacts the supports. The impact force (109,000 lb) is then treated as a

static load in design. The CRD housing supports are designed as seismic Category I equipment. Loading conditions and examples of stress analysis results and limits are shown in Table 3.9-2t.

3.9.4.4 CRD Performance Assurance Program

The CRDs test program consists of the following tests:

a. Development Tests

b. Factory Quality Control Tests

c. 5-Year Maintenance Life Tests

d. 1.5X Design Life Tests

e. Operational Tests

f. Acceptance Tests

g. Surveillance Tests

All of the above tests except c. and d. are discussed in subsections 4.6.3 through 4.6.3.1.1.5. Tests c. and d. are discussed below:

c. 5-Year Maintenance Life Tests

Four Control Rod Drives are normally picked at random from the production stock each year and subjected to various tests under simulated reactor conditions and more than 1/8 of the cycles specified in subsection 3.9.1.1.

Upon completion of the test program, the control rod drive parts are checked to the drawings and all parts must meet or surpass the minimum specified performance requirements.

d. 1.5X Design Life Tests

When a significant design change is made to the components of the drive, the drive is subjected to a series of tests equivalent to 1.5 times the life test cycles specified in subsection 3.9.1.1.

Two CRDs have undergone such testing in 1976. Upon completion of the test program, the CRDs were disassembled and the parts checked to the drawing for wear and/or damage. All parts met or surpassed the minimum specified performance requirements.

3.9.5 Reactor Pressure Vessel Internals

This subsection identifies and discusses the structural and functional integrity of the major reactor pressure vessel internals.

3.9.5.1 Design Arrangements

The core support structures and reactor vessel internals (exclusive of fuel, control rods, and incore nuclear instrumentation) are identified below:

Core Support Structures
- Shroud
- Shroud support cylinder, plate and legs (part of the RPV)
- Core plate, and core plate hardware
- Grid (only that portion below the bottom weld in the cylindrical portion is core support structure. The grid is a part of the top guide assembly.)
- Top guide (including studs, nuts, and pins)
- Orificed fuel supports (except for orifices which do not support or restrain the core)
- CRD housing (only that portion above the first weld that is above the housing, to the pressure vessel weld) Control rod guide tubes

Reactor Internals
- Jet pump assemblies, braces, and instrumentation
- *Feedwater spargers
- Vessel head spray nozzle
- Differential pressure and liquid control lines
- *In-core instrument guide tubes and stabilizers
- *Initial startup neutron sources
- *Surveillance sample holders
- Core spray lines and spargers
- *In-core instrument housings
- LPCI coupling
- *Steam dryers, shroud head, and steam separator assembly

A general assembly drawing of the important reactor components is shown in Figure 3.9-8.

*Non-safety class components

The floodable inner volume of the reactor pressure vessel can be seen in Figure 3.9-9. It is the volume inside the core shroud up to the level of the jet pump suction inlet.

The design arrangement of the reactor internals, such as the jet pumps steam separators and guide tube, is such that one end is unrestricted and thus free to expand.

The LPCI couplings incorporate vertically oriented slip fit joints to allow free thermal expansion.

3.9.5.1.1 Core Support Structures

The core support structures consist of those items listed in subsection 3.9.5.1. These structures form partitions within the reactor vessel, to sustain pressure differentials across the partitions, direct the flow of the coolant water, and laterally locate and support the fuel assemblies. Figure 3.9-9 shows the reactor vessel internal flow paths.

3.9.5.1.1.1 Shroud and Shroud Support

The shroud support, shroud, and top guide make up a stainless steel cylindrical assembly that provides a partition to separate the upward flow of coolant through the core from the downward recirculation flow. This partition separates the core region from the downcomer annulus, thus providing a floodable region following a recirculation line break. The volume enclosed by this assembly is characterized by three regions. The upper portion surrounds the core discharge plenum, which is bounded by the shroud head on top and the top guide's grid plate below. The central portion and the shroud surrounds the active fuel and forms the longest section of the assembly. This section is bounded at the top by the grid plate and at the bottom by the core plate. The lower portion, surrounding part of the lower plenum, is welded to the reactor pressure vessel shroud support.

The shroud support is designed to support the shroud and jet pumps. It provides an annular baffle between the reactor pressure vessel and the shroud. The jet pump discharge diffusers penetrate the shroud to introduce coolant to the lower plenum below the core.

3.9.5.1.1.2 Core Plate

The core plate consists of a circular stainless steel plate with bored holes stiffened with a rim and beam structure. The plate provides lateral support and guidance for the control rod guide tubes, in-core flux monitor guide tubes, peripheral fuel supports, and startup neutron sources. The last two items are also supported vertically by the core support plate.

The entire assembly is bolted to a support ledge on the lower portions of the shroud.

3.9.5.1.1.3 Top Guide

The top guide consists of a circular grid plate with square openings welded to the bottom of the top guide cylinder. Each opening provides lateral support and guidance for four fuel assemblies or, in the case of peripheral fuel, one or two fuel assemblies. Notches are provided in the bottom of the intersections to anchor the in-core flux monitors and startup neutron sources. The top guide is bolted to the shroud. The core spray spargers are installed in the upper portion of the top guide cylinder.

3.9.5.1.1.4 Fuel Support

The fuel supports, shown in Figure 3.9-10 are of two basic types; namely, peripheral supports and four-lobed orificed fuel supports. The peripheral fuel support is located at the outer edge of the active core and is not adjacent to control rods. Each peripheral fuel support will support one fuel assembly and contains a single orifice assembly designed to assure proper coolant flow to the peripheral fuel assembly. Each four-lobed orificed fuel support will support four fuel assemblies and is provided with four orifice plates to assure proper coolant flow distribution to each rod-controlled fuel assembly. The four-lobed orificed fuel supports rest in the top of the control rod guide tubes which are supported laterally by the core plate. The control rods pass through slots in the center of the four-lobed orificed fuel support. A control rod and the four adjacent fuel assemblies represent a core cell. (See subsection 4.4.2, “Fuel System Design Description and Drawings.”)

3.9.5.1.1.5 Control Rod Guides Tubes

The control rod guide tubes, located inside the vessel, extend from the top of the control rod drive housings up through holes in the core plate. Each tube is designed as the guide for a control rod and as the vertical support for a four-lobed orificed fuel support piece and the four fuel assemblies surrounding the control rod. The bottom of the guide tube is supported by the control rod drive housing, which in turn transmits the weight of the guide tube, fuel support, and fuel assemblies to the reactor vessel bottom head. A thermal sleeve is inserted into the control rod housing from below and is rotated to lock the control rod guide tube in place. A key is inserted into a locking slot in the bottom of the control rod drive housing to hold the thermal sleeve in position.

3.9.5.1.2 Reactor Internals

3.9.5.1.2.1 Shroud Head and Steam Separator Assembly

The shroud head and steam separator assembly is bolted to the top of the top guide to form the top of the core discharge plenum. This plenum provides a mixing chamber for the steamwater mixture before it enters the steam separators. Individual stainless steel axial flow steam separators are attached to the top of standpipes that are welded into the shroud head. The steam separators have no moving parts. In each separator, the steam-water mixture rising through the standpipe passes vanes that impart a spin to establish a vortex separating the water from the steam. The separated water flows from the lower portion of the steam separator into the downcomer annulus.

3.9.5.1.2.2 Jet Pump Assemblies

The jet pump assemblies are located in two semi-circular groups in the downcomer annulus between the core shroud and the reactor vessel wall. The design and performance of the jet pump is covered in detail in References 1 and 2. Each stainless steel jet pump consists of driving nozzles, suction inlet, throat or mixing section, and diffuser (see Figure 3.9-11). The driving nozzle, suction inlet, and throat are joined together as a removable unit, and the diffuser is permanently installed. High pressure water from the recirculation pump is supplied to each pair of jet pumps through a riser pipe welded to the recirculation inlet nozzle thermal sleeve. A riser brace consists of cantilever beams welded to a riser pipe and to pads on the reactor vessel wall.

The nozzle entry section is connected to the riser by a metal-to-metal, spherical-to-conical seal joint. Firm contact is maintained by a holddown clamp. The throat section is supported laterally by a bracket attached to the riser. There is a slip-fit joint between the throat and diffuser. The diffuser is a gradual conical section changing to a straight cylindrical section at the lower end.

3.9.5.1.2.3 Steam Dryers

The steam dryers remove moisture from the wet steam leaving the steam separators. The extracted moisture flows down the dryer vanes to the collecting troughs, then flows through tubes into the downcomer annulus. A skirt extends from the bottom of the dryer vane housing to the steam separator standpipe, below the water level. This skirt forms a seal between the wet steam plenum and the dry steam flowing from the top of the dryers to the steam outlet nozzles.

The steam dryer and shroud head are positioned in the vessel during installation with the aid of vertical guide rods. The dryer assembly rests on steam dryer support brackets attached to the reactor vessel wall. Upward movement of the dryer assembly, which would occur only under accident conditions, is restricted by steam dryer hold-down brackets attached to the reactor vessel top head.

3.9.5.1.2.4 Feedwater Spargers

The feedwater spargers are stainless steel headers located in the mixing plenum above the downcomer annulus. A separate sparger is fitted to each feedwater nozzle and is shaped to conform to the curve of the vessel wall. Sparger end brackets are pinned to vessel brackets to support the spargers. Feedwater flow enters the center of the spargers and is discharged radially inward to mix the cooler feedwater with the downcomer flow from the steam separators and steam dryer before it contacts the vessel wall. The feedwater also serves to condense the steam in the region above the downcomer annulus and to subcool the water flowing to the jet pumps and recirculation pumps.

3.9.5.1.2.5 Core Spray Lines

The core spray lines are the means for directing flow to the core spray nozzles which distribute coolant during accident conditions.

Two core spray lines enter the reactor vessel through the two core spray nozzles. (See Section 5.4, Component and Subsystem Design.) The lines divide immediately inside the reactor vessel. The two halves are routed to opposite sides of the reactor vessel and are supported by clamps attached to the vessel wall. The lines are then routed downward into the downcomer annulus and pass through the grid immediately below the flange. The flow divides again as it enters the center of the semicircular sparger, which is routed halfway around the inside of the grid. The two spargers are supported by brackets designed to accommodate thermal expansion. The line routing and supports are designed to accommodate differential movement between the grid and vessel. The other core spray line is identical except that it enters the opposite side of the vessel and the spargers are at a slightly different elevation inside the grid. The correct spray distribution pattern is provided by a combination of distribution nozzles pointed radially inward and downward from the spargers. (See Section 6.3, Emergency Core Cooling System.)

3.9.5.1.2.6 Deleted

3.9.5.1.2.7 Core Support Plate Differential Pressure Lines

The differential pressure lines enter the vessel through two bottom head penetrations and sense the differential pressure across the core support plate (described in Section 5.4, Component and Subsystem Design). One line terminates near the lower shroud with a perforated length below the core support plate. It is used to sense the pressure below the core support plate during normal operation. This line was also used as the Standby Liquid Control System (SLCS) injection sparger until the SLCS injection was rerouted to the HPCS discharge line. The other line terminates immediately above the core support plate and senses the pressure in the region outside the fuel assemblies.

3.9.5.1.2.8 In-Core Flux Monitor Guide Tubes

These tubes provide a means of positioning fixed detectors in the core as well as provide a path for calibration monitors (TIP System).

The in-core flux monitor guide tubes extend from the top of the in-core flux monitor housing (see Section 5.4, Component and Subsystem Design) in the lower plenum to the top of the core support plate. The power range detectors for the power range monitoring units and the dry tubes for the source range monitoring

and intermediate range monitoring (SRM/IRM) detectors are inserted through the guide tubes. A latticework of clamps, tie bars, and spacers give lateral support and rigidity to the guide tubes. The bolts and clamps are welded, after assembly, to prevent loosening during reactor operation.

3.9.5.1.2.9 Surveillance Sample Holders

The surveillance sample holders are welded baskets containing impact and tensile specimen capsules (see Section 5.4, Component and Subsystem Design). The baskets hang from the brackets that are attached to the inside wall of the reactor vessel and extend to mid-height of the active core. The radial positions are chosen to expose the specimens to the same environment and maximum neutron fluxes experienced by the reactor vessel itself while avoiding jet pump removal interference or damage.

3.9.5.1.2.10 Low Pressure Coolant Injection Lines

Three LPCI lines penetrate the core shroud through separate LPCI nozzles. Coolant is discharged inside the core shroud.

3.9.5.2 Design Loading Conditions

3.9.5.2.1 Events to be Evaluated

Examination of the spectrum of conditions for which the safety design basis must be satisfied by core support structures and engineered safety feature components reveals four significant faulted events:

a. Recirculation Line Break: a break in a recirculation line between the reactor vessel and the suction or discharge side of the recirculation pump.

b. Steam line break accident: a break in one main steam line between the reactor vessel and the flow restrictor. The accident results in significant pressure differentials across some of the structures within the reactor.

c. Earthquake: subjects the core support structures and reactor internals to significant forces as a result of ground motion.

d. Safety-relief valve discharge in combination with an SSE

Analysis of other conditions existing during normal operation, abnormal operational transients, and accidents shows that the loads affecting the core support structures and other engineered safety feature reactor internals are less severe than those due to the above postulated events.

The faulted conditions for the reactor pressure vessel internals and core support structures are discussed in subsection 3.9.1.4. Loading combination and analysis for the reactor pressure vessel internals are discussed in subsection 3.9.3.1, Tables 3.9-1 and 3.9-2.

The core support structures are designed in accordance with subsection NG of Section III of the ASME Code.

3.9.5.2.2 Pressure Differential During Rapid Depressurization

A digital computer code is used to analyze the transient conditions within the reactor vessel following the recirculation line break accident and the steam line break accident. The analytical model of the vessel consists of nine nodes, which are connected to the necessary adjoining nodes by flow paths having the required resistance and inertial characteristics. The program solves the energy and mass conservation equations for each node to give the depressurization rates and pressure in the various regions of the reactor. Figure 3.9-12 shows the nine reactor nodes. The computer code used is the General Electric short-term thermal hydraulic model described in Reference 3. This model has been approved for use in ECCS conformance evaluation under 10 CFR 50, Appendix K. In order to adequately describe the blowdown pressure effect on the individual assembly components, three features are included in the model that are not applicable to the ECCS analysis and are, therefore, not described in Reference 3. These additional features are discussed below:

a. The liquid level in the steam separator region and in the annulus between the dryer skirt and the pressure vessel is tracked to more accurately determine the flow and mixture quality in the steam dryer and in the steamline.

b. The flow path between the bypass region and the shroud head is more accurately modeled since the fuel assembly pressure differential is influenced by flashing in the guide tubes and bypass region for a steamline break. In

the ECCS analysis, the momentum equation is solved in this flow path, but its irreversible loss coefficient is conservatively set at an arbitrary low value.

c. The enthalpies in the guide tubes and the bypass are calculated separately, since the fuel assembly ΔP is influenced by flashing in these regions. In the ECCS analysis, these regions are lumped.

3.9.5.2.3 Recirculation Line and Steam Line Break

3.9.5.2.3.1 Accident Definition

Both a recirculation line break (the largest liquid break) and an inside steam line break (the largest steam break) are considered in determining the design basis accident for the engineered safety feature reactor internals. The recirculation line break is the same as the design basis loss-of-coolant accident described in Section 6.3, Emergency Core Cooling Systems. A sudden, complete circumferential break is assumed to occur in one recirculation loop. The pressure differentials on the reactor internals and core support structures are in all cases lower than for the main steam line break.

The analysis of the steam line break assumes a sudden, complete circumferential break of one main steam line between the reactor vessel and the main steam line restrictor. A steam line break upstream of the flow restrictors produces a larger blowdown area and thus a faster depressurization rate than a break downstream of the restrictors. The larger blowdown area results in greater pressure differentials across the reactor internal structures.

The steam line break accident produces significantly higher pressure differentials across the reactor internal structures than does the recirculation line break. This results from the higher reactor depressurization rate associated with the steam line break. Therefore, the steam line break is the design basis accident for internal pressure differentials.

3.9.5.2.3.2 Effects of Initial Reactor Power and Core Flow

For purposes of illustration, the maximum internal pressure loads can be considered to be composed of two parts: steady-state and transient pressure differentials. For a given plant the core flow and power are the two major factors which influence the reactor internal pressure differentials. The core flow essentially

affects only the steady-state part. For a fixed power, the greater the core flow, the larger will be the steady-state pressure differentials. The core power affects both the steady-state and the transient parts. As the power is decreased, there is less voiding in the core and consequently the steady-state core pressure differential is less. However, less voiding in the core also means that less steam is generated in the reactor pressure vessel and thus the depressurization rate and the transient part of the maximum pressure load is increased. As a result, the total loads on some components are higher at low power.

To ensure that the calculated pressure differences bound those which could be expected if a steam line break should occur, an analysis is conducted at a low power-high recirculation flow condition in addition to the standard safety analysis condition at high power, rated recirculation flow. The power chosen for analysis is the minimum value permitted by the recirculation system controls at rated recirculation drive flow (that is, the drive flow necessary to achieve rated core flow at rated power).

This condition maximizes those loads which are inversely proportional to power. It must be noted that this condition, while possible, is unlikely; first, because the reactor will generally operate at or near full power; second, because high core flow is neither required nor desirable at such a reduced power condition.

3.9.5.2.4 Seismic and Hydrodynamic Events

The seismic and hydrodynamic loads acting on the structures within the reactor vessel are based on a dynamic analysis as described in Sections 3.7 and 3.8 and subsection 3.9.2.5, respectively. Dynamic analysis is performed by coupling the mathematical model of the reactor vessel and internals, with the building model to determine the accelerations, forces, and moments time-history in the reactor vessel and internals. This is done using the modal superposition method. Acceleration response spectra are also generated for subsystem analyses of selected components.

3.9.5.2.5 Impact of the Maximum Extended Operating Domain

The impact of the Maximum Extended Operating Domain (MEOD) on the various reactor internal components was evaluated as discussed in Appendix 15D.8. The reactor internals most affected by MEOD are the core plate, guide tube, shroud head, upper shroud, shroud support ring and lower shroud, shroud top guide, fuel channel

wall, steam dryer and jet pump. These components were evaluated under normal, upset, emergency and faulted conditions. It was concluded that the stresses during MEOD are within the allowable design limits.

3.9.5.3 Design Loading Categories

The basis for determining faulted loads on the reactor internals is shown for seismic and hydrodynamic loads in Sections 3.7 and 3.8 and subsection 3.9.2.5, and for pipe rupture loads in subsections 3.9.5.2.3 and 3.9.5.4.3.

Stress intensity and other design limits are discussed in subsection 3.9.5.4.4. The core support structures which are fabricated as part of the reactor pressure vessel assembly are discussed in subsection 3.9.1.3 in conjunction with the reactor pressure vessel.

The design requirements for equipment classified as "other internals," e.g., steam dryers and shroud heads, were specified by the designer with appropriate consideration of the intended service of the equipment and expected plant and environmental conditions under which it will operate. Where possible, design requirements are based on applicable industry codes and standards. If these are not available, the designer relies on accepted industry or engineering practices.

3.9.5.4 Design Bases

3.9.5.4.1 Safety Design Bases

The reactor core support structures and internals meet the following safety design bases:

a. Shall be arranged to provide a floodable volume in which the core can be adequately cooled in the event of a breach in the nuclear system process barrier external to the reactor vessel.

b. Deformation shall be limited to assure that the control rods and core standby cooling systems can perform their safety functions.

c. Mechanical design of applicable structures shall assure that safety design bases a. and b., above, are satisfied so that the safe shutdown of the plant and removal of decay heat are not impaired.

3.9.5.4.2 Power Generation Design Bases

The reactor core support structures and internals are designed to the following power generation design bases:

a. They shall provide the proper coolant distribution during all anticipated normal operating conditions to all power operation of the core without fuel damage.

b. They shall be arranged to facilitate refueling operations.

c. They shall be designed to facilitate inspection.

3.9.5.4.3 Response of Internals Due to Inside Steam Break Accident

It is concluded that the maximum pressure loads acting on the reactor internal components result from an inside steam line break, and on some components the loads are greatest with operation at the minimum power associated with the maximum core flow. This has been substantiated by the analytical comparison of liquid versus steam breaks and by the investigation of the effects of core power and core flow.

It has also been pointed out that, although possible, it is not probable that the reactor would be operating at the rather abnormal condition of minimum power and maximum core flow. More realistically, the reactor would be at or near a full power condition and thus the maximum pressure loads acting on the internal components would be less.

3.9.5.4.4 Stress, Deformation, and Fatigue Limits for Reactor Internals (Except Core Support Structure)

The stress deformation and fatigue criteria listed in Tables 3.9-7, 3.9-8, 3.9-9, and 3.9-10 are used or the criteria are based on the criteria established in applicable codes and standards for similar equipment, by manufacturers' standards, or by empirical

methods based on field experience and testing. For the quantity SFmin (minimum safety factor) appearing in those tables, the following values listed were used.

Service Level	Design Condition	SF min
A	Normal	2.25
B	Upset	2.25
C	Emergency	1.5
D	Fault	1.125

Elastic displacement is considered in the design of reactor internal components in which deflection can affect control rod insertability. No plastic deformation occurs in any permanent core support structure components or the reactor vessel. Radiation induced deformation can occur in the fuel channel over core life. These effects are considered in control rod insertability tests. The forcing functions applicable to the reactor internals are discussed in subsection 3.9.2.5.

3.9.5.4.5 Stress and Fatigue Limits for Core Support Structures

The stress, fatigue, and other limits for the core support structures are in accordance with ASME Section III, subsection NG, and are summarized in Table 3.9-2.

3.9.6 Inservice Testing of Pumps and Valves

Inservice testing of ASME Code Class 1, Class 2, and Class 3 pumps and valves will be performed in accordance with ASME OM Code for Operation and Maintenance of Nuclear Power Plants and applicable Addenda as required by 10 CFR 50, Section 50.55a(f), except where specific written relief has been granted by the NRC pursuant to 10 CFR 50, Section 50.55a(f)(6)(i).

The inservice testing will be in accordance with an approved ASME OM Code for inservice testing of pumps and valves.

In addition, safety and relief valves shall meet the inservice test requirements of the ASME OM Code.

SEP-GGNS-IST-1, GGNS Inservice Testing Bases Document, is the document that defines the bases for the pump and valve inservice testing program. SEP-GGNS-IST-2, GGNS Inservice Testing Plan, is the document that defines the scope and testing plan of the pump and valve inservice testing program.

3.9.6.1 Inservice Testing of Pumps

SEP-GGNS-IST-2, GGNS Inservice Testing Plan, identifies the pumps that will be tested according to the requirements of ASME OM Code for Operation and Maintenance of Nuclear Power Plants. The hydraulic and mechanical parameters to be measured or observed are defined in this plan. SEP-GGNS-IST-2 also contains relief requests from the requirements of ASME OM Code.

3.9.6.2 Inservice Testing of Valves

SEP-GGNS-IST-2, GGNS Inservice Testing Plan, identifies the valves that will be tested according to the requirements of ASME OM Code for Operation and Maintenance of Nuclear Power Plants. Parameters to be measured or observed are defined in this plan. SEP-GGNS-IST-2 also contains relief requests from the requirements of ASME OM Code.

3.9.7 References

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TABLE 3.9-1: TRANSIENTS AND THE NUMBER OF ASSOCIATED CYCLES CONSIDERED IN THE DESIGN AND FATIGUE ANALYSES OF THE RPV ASSEMBLY AND INTERNALS

Transients	No. of Cycles
Normal, Upset, and Testing Conditions	
1. Bolt Up [1]	123
2. Design Hydrostatic Test	40
a. Leak checks at 400 psig prior to power operation, 3 cycles/startup	
3. Startup (100 F/hr Heatup Rate) [2]	120
4. Daily Reduction of 75% Power [1]	10,000
5. Weekly Reduction of 50% Power [1]	2,000
6. Control Rod Patterns Change [1]	400
7. Loss of Feedwater Heaters (80 Cycles Total):	80
8. 50% Safe Shutdown Earthquake Event at Rated Operating Conditions	10[4]
9. Scram:	
a. Turbine Generator Trip, Feedwater On, Isolation Valves Stay Open	40
b. Other Scrams	140
c. Loss of Feedwater Pumps, Isolation Valves closed	10
d. Turbine Bypass, Single Safety or Relief Valve Blowdown	8
10. Reduction to 0% Power, Hot Standby, with Main Condenser available Shutdown (100 F/hr Cooldown Rate) [2]	111
11. Unbolt	123
Emergency Conditions	
1. Scram	
a. Reactor Overpressure with Delayed Scram, Feedwater Stays on, Isolation Valves Stay Open	1[3]
b. Automatic Blowdown (ADS)	1[3]
2. Improper Start of Cold Recirculation Loop	1[3]

TABLE 3.9-1: TRANSIENTS AND THE NUMBER OF ASSOCIATED CYCLES CONSIDERED IN THE DESIGN AND FATIGUE ANALYSES OF THE RPV ASSEMBLY AND INTERNALS (Continued)

Transients	No. of Cycles
3. Sudden Start of Pump in Cold Recirculation Loop	1[3]
4. Hot Standby with Reactor Drain Shut Off Followed by Pump Restart	1[3]
Faulted Condition	
1. Pipe Rupture and Blowdown	1[3]
2. Safe Shutdown Earthquake at Rated Operating Conditions	1[3]

In addition to the above temperature/pressure/flow transients the following dynamic load transients have been considered to the design and/or fatigue evaluation:

Dynamic/Transient Load	Category	Cycles/Event
1. Operating Basis Earthquake (OBE)[5]	Upset	10 cycles
2. Safe Shutdown Earthquake (SSE)[6]	Faulted	1 cycle
3. Turbine Stop Valve Closure (TSV)[7]	Upset	690 cycles
4. Safety Relief Valve Actuation (Acoustic Wave)[7]	Upset	5460 cycles
5. Safety Relief Valve Actuation[8]	Upset	(a) Piping Analysis (i) All Valve Actuation 660 full-range cycles 880 half-range cycles (ii) Single Valve Actuation 4800 full-range cycles 6400 half-range cycles

TABLE 3.9-1: TRANSIENTS AND THE NUMBER OF ASSOCIATED CYCLES CONSIDERED IN THE DESIGN AND FATIGUE ANALYSES OF THE RPV ASSEMBLY AND INTERNALS (Continued)

Transients		No. of Cycles
		(b) RCPB & Internals Analysis 12,600 cycles
6. Loss of Coolant Accident (LOCA):		
Small break LOCA	Emergency/faulted	1 event
Intermediate break LOCA	Faulted	1 event
Large break LOCA	Faulted	1 event

(1) Applies to reactor pressure vessel only.
(2) Bulk average vessel coolant temperature change in any 1-hour period.
(3) The annual encounter probability of the one cycle event in <10-2 for emergency and <10-4 for faulted events over the 40-year plant life.
(4) Includes 10 maximum load cycles per event.
(5) One 50 percent SSE event includes 10 maximum load cycles.
(6) One stress reversal cycle of maximum seismic amplitude.
(7) Applicable to main steam piping system only. (1820 actuations with three acoustic cycles each.)
(8) In total, 1820 actuations are considered; out of which 220 are all valve actuations and 1600 are single valve actuations. Each actuation is accompanied by 7 cycles due to air bubble oscillation. For piping design, the first 3 of 7 cycles are considered for full pressure range; the remainder 4 cycles are considered half range due to the decaying amplitude.

TABLE 3.9-2: INTRODUCTION

This table lists the major mechanical safety-related mechanical components in the plant. Various parts of the table are referenced in Section 3.9. The format in various parts of the table is not consistent since variations on analytical method and depth of detail, necessary to demonstrate the safety aspects of various components, differ.

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LOADING COMBINATIONS, STRESS LIMITS, AND ALLOWABLE STRESSES

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3.9-2e	ASME Code Class 1 Recirculation Loop Piping and Pipe Mounted Equipment - Highest Stress Summary
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3.9-2f	Main Steam Safety/Relief Valves (i) ASME Code Section III, July 1974 (ii) ASME Code Section III, July 1974, Including Addenda through Summer 1976
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TABLE 3.9-2: INDEX
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3.9-2k	ASME Code Class III Safety/Relief Valve Discharge Piping System - Highest Stress Summary, Main Steam Line - "A"
3.9-2l	Standby Liquid Control Pump
3.9-2m	Standby Liquid Control Tank
3.9-2n	ECCS Pumps (i) Residual Heat Removal Pump (ii) Low Pressure Core Spray Pump (iii) High Pressure Core Spray Pump
3.9-2o	Residual Heat Removal (RHR) Heat Exchanger
3.9-2p	Deleted
3.9-2q	RCIC Turbine
3.9-2r	RCIC Pump
3.9-2s	Reactor Refueling and Servicing Equipment (i) Fuel Storage Racks (ii) Refueling Platform (iii) Fuel Preparation Machine (iv) Horizontal Fuel Transfer Tube
3.9-2t	Reactor Pressure Vessel Support Equipment (i) CRD Housing Restraint Beams (ii) CRD Housing Supports - Grid Structure
3.9-2u	Control Rod Drive (i) Main Flange (ii) Ring Flange (2) (iii) Indicator Tube
3.9-2u	Control Rod Drive Housing
3.9-2w	Jet Pumps
3.9-2x	Fuel Pool Heat Exchanger
3.9-2y	LPCI Coupling (Strut-to-Shroud Weld)
3.9-2z	Fuel Pool Cooling and Cleanup System Pump

TABLE 3.9-2: INDEX
LOADING COMBINATIONS, STRESS LIMITS, AND ALLOWABLE STRESSES (Continued)

TABLE	CONTENTS
3.9-2aa	Control Rod Guide Tube
3.9-2ab	Incore Housing
3.9-2ac	High Pressure Core Spray System ASME Code Class 1 Valve

TABLE 3.9-2: DESIGN LOADING COMBINATIONS FOR NSSS ASME CODE CLASS 1, 2, AND 3 COMPONENTS(5)

Load Case(1)	N	SRV_X(4)	SRV_{ADS}	OBE	SSE	SBA/IBA(3)	DBA	ASME Code Service Limits
1	X	X						B
2	X	X		X				B
3	X	X			X			D(2)
4	X		X			X (SBA only)		C(2)
5	X		X	X		X		D(2)
6	X		X		X	X		D(2)
7	X				X		X	D(2)
8	X							A
9	X			X				B

NOTES: (1) See legend for definition of terms.

(2) All ASME Code Class 1, 2, and 3 piping systems which are required to function for safe shutdown under the postulated events are designed to meet the requirements of NRC's "Evaluation of Topical Report - Piping Functional Capability Criteria," by MEB Dated July 17, 1980.

(3) SBA or IBA, whichever is greater, except Case 4.

(4) SRV_{ALL} or SRV_1, whichever is controlling, will be used.

(5) The design basis methodology is the square-root-of-thesum-of-the-squares (SRSS) for combining the dynamic responses in evaluation of these load combinations.

TABLE 3.9-2: LOAD DEFINITIONS LEGEND

N	Normal load consists of pressure, dead weight, and thermal loads.
OBE	Operational basis earthquake loads.
SSE	Loads due to vibratory motion from safe shutdown earthquake loads.
SRV_1	Safety/relief valve discharge induced loads from one valve's subsequent actuation.
SRV_{ALL}	The loads induced by actuation of all safety/relief valves which activate within milliseconds of each other (e.g., turbine trip operational transient).
SRV_{ADS}	The loads induced by the actuation of safety/ relief valves associated with the Automatic Depressurization System which actuate within milliseconds of each other during the postulated small- or intermediate-size pipe rupture.
DBA	Design basis accident is the sudden break of the main steam or recirculation lines (largest postulated breaks). DBA-related loads include main vent clearing and pool swell, chugging, condensation oscillation, and annulus pressurization.
SBA	Small-break accident.
IBA	Intermediate-break accident.

TABLE 3.9-2A: REACTOR PRESSURE VESSEL AND SHROUD SUPPORT ASSEMBLY
(i) VESSEL SUPPORT SKIRT

ASME B&PV Code Sec. III Subsection NB Primary Stress Limit Criteria	Loading	Primary Stress Type	Allowable Stress (psi)	Maximum Calculated Stress (psi)
MATERIAL: SA 533 GR B CL 1				
A. NORMAL & UPSET CONDITION:				
$P_m \leq S_m$	Normal and Upset Condition Loads	PRIMARY MEMBRANE	26,700	16,610
S_m = 26,700 @ 575 F	1. Dead weight			
$P_L + P_b < 1.5S_m$	2. Pressure			
S_m = 26,700 @ 575 F	3. Operating basis earthquake	PRIMARY MEMBRANE PLUS BENDING	40,050	39,320
	4. SRV actuations-all			
B. EMERGENCY CONDITION:				
$P_m \leq S_y$	Emergency Condition Loads	PRIMARY MEMBRANE		
S_y = 42,800 @ 528 F	1. Dead weight			
$P_L + P_b < 1.5S_y$	2. Pressure			
S_y = 42,800 @ 528 F	3. Operating basis earthquake	PRIMARY MEMBRANE PLUS BENDING		
	4. SRV actuations-all			
C. FAULTED CONDITION:				
$P_m \leq S_y$	Faulted Condition Loads	PRIMARY MEMBRANE	42,800	22,460
S_y = 42,800 @ 528 F	1. Dead weight			
$P_L + P_b < 1.5S_y$	2. Safe shutdown earthquake			
S_y = 42,800 @ 528 F	3. Combined pres sure and jet loads	PRIMARY MEMBRANE PLUS BENDING	64,300	52,660

TABLE 3.9-2A: REACTOR PRESSURE VESSEL AND SHROUD SUPPORT ASSEMBLY (CONTINUED)
(i) VESSEL SUPPORT SKIRT

MATERIAL: SA 533 GR B CL 1 (Cont.)

4. Vent clearing
5. Jet reaction

D. MAXIMUM CUMULATIVE USAGE FACTOR: 0.656 At Skirt-Base Junction

E. The fatigue usage factor shown in this table is the design fatigue usage factor. For operating fatigue usage factor see Mechanical Standard MS-40.

F. The structural integrity of the RPV support was evaluated for EPU conditions at 4408 MWt and it was concluded it will continue to meet the applicable structural design acceptance criteria following EPU implementation (Ref. 15).

TABLE 3.9-2A: REACTOR PRESSURE VESSEL AND SHROUD SUPPORT ASSEMBLY
(II) SHROUD SUPPORT

ASME B&PV Code Sec. III Subsection NB Primary Stress Limit Criteria	Loading	Primary Stress Type	Allowable Stress (psi)	Maximum Calculated Stress (psi)
MATERIAL: SB168				
A. NORMAL & UPSET CONDITION:				
$P_m \leq 0.9\ S_m$	Normal and Upset Condition Loads	PRIMARY MEMBRANE	20,970	18,000
S_m = 23,300 @ 575 F	1. Normal Loads			
$P_L + P_b < 1.5(0.9)S_m$	2. Upset Pressure			
S_m = 23,300 @ 575 F	3. Safety Relief Valve-All	PRIMARY MEMBRANE PLUS BENDING	31,450	30,700
	4. Operating Basis Earthquake			
B. EMERGENCY CONDITION:				
$P_m \leq 0.9S_m$	Emergency Condition Loads	PRIMARY MEMBRANE	20,970	18,000
S_m = 23,300 @ 575 F	1. Normal Loads			
$P_L + P_b < 1.5S_m$	2. Upset Pressure			
S_m = 23,300 @ 575 F	3. Safety Relief Valve-All	PRIMARY MEMBRANE PLUS BENDING	31,450	30,700
	4. Chugging			
C. FAULTED CONDITION:				
$P_m \leq (0.7)0.9S_u$	Faulted Condition Loads	PRIMARY MEMBRANE	46,600	35,400
S_u = 74,000 @ 575 F	1. Normal Loads			
$P_L + P_u \leq 1.5(0.7)0.9S_u$	2. Faulted Pressure			

TABLE 3.9-2A: REACTOR PRESSURE VESSEL AND SHROUD SUPPORT ASSEMBLY (CONTINUED)
(II) SHROUD SUPPORT

ASME B&PV Code Sec. III Subsection NB Primary Stress Limit Criteria	Loading	Primary Stress Type	Allowable Stress (psi)	Maximum Calculated Stress (psi)
MATERIAL: SB168 (Cont.)				
S_u = 74,000 @ 575 F	3. Safety Relief Valve-All	PRIMARY MEMBRANE PLUS BENDING	69,900	61,700
	4. Safety Shutdown Earthquake			
	5. Chugging			

D. MAXIMUM CUMULATIVE USAGE FACTOR: 0.098 At Vessel-Support Plate Junction

E. The fatigue usage factor shown in this table is the design fatigue usage factor. For operating fatigue usage factor see Mechanical Standard MS-40.

F. Maximum Cumulative usage factor was obtained from GGNS NSSS new loads design adequacy evaluation report: GE-NEDC-30273.

G. The structural integrity of the shroud support was evaluated for EPU conditions at 4408 MWt and it was concluded the RPV support skirt will continue to meet the applicable structural design acceptance criteria following EPU implementation (Ref. 15).

TABLE 3.9-2A: REACTOR PRESSURE VESSEL AND SHROUD SUPPORT ASSEMBLY
(III) RPV FEEDWATER NOZZLE

ASME B&PV Code Sec. III Subsection NB Primary Stress Limit Criteria	Loading	Primary Stress Type	Allowable Stress (psi)	Maximum Calculated Stress (psi)
MATERIAL: 508 CL-1-Safe End				
A. NORMAL & UPSET CONDITION:				
$P_m < S_m$	Normal and Upset Condition Loads	PRIMARY MEMBRANE	17,700	9,630
S_m = 17,700 @ 575 F	1. Normal Loads			
$P_L + P_b < 1.5Sm$	2. Upset Pressure			
S_m = 17,700 @ 575 F	3. Operating Basis Earthquake	PRIMARY MEMBRANE PLUS BENDING	26,550	22,930
	4. Safety Relief Valve-All			
$P+Q < 3.0S_m$		P+Q - thermal bending	50,700	19,416
S_m = 16,900 @ 550 F				
B. EMERGENCY CONDITION:				
$P_m < S_Y$	Emergency Condition Loads	PRIMARY MEMBRANE	25,900	21,420
S_Y = 25,900 @ 594 F	1. Normal Loads			
$P_L + P_b < 1.5SY$	2. Upset Pressure			
S_Y = 25,900 @ 594 F	3. Chugging	PRIMARY MEMBRANE	38,850	22,400
	4. SRV-ADS	PLUS BENDING		
C. FAULTED CONDITION:				
$P_m \leq 3S_m$	Faulted Condition Loads	PRIMARY MEMBRANE	54,300	16,600

TABLE 3.9-2A: REACTOR PRESSURE VESSEL AND SHROUD SUPPORT ASSEMBLY (CONTINUED)
(III) RPV FEEDWATER NOZZLE

ASME B&PV Code Sec. III Subsection NB Primary Stress Limit Criteria	Loading	Primary Stress Type	Allowable Stress (psi)	Maximum Calculated Stress (psi)
MATERIAL: 508 CL-1-Safe End (Cont.)				
S_m = 18,100 @ 552 F	1. Normal loads			
$P_L + P_b \leq 3S_m$	2. Faulted pressure			
S_m = 18,100 @ 552 F	3. Chugging	PRIMARY MEMBRANE	54,300	33,740
	4. SRV-ADS	PLUS BENDING		
	5. Safe shutdown earthquake			
D. MAXIMUM CUMULATIVE USAGE FACTOR:		0.9503	At Safe End	

E. The fatigue usage factor shown in this table is the design fatigue usage factor. For operating fatigue usage factor see Mechanical Standard MS-40.

F. Summary of maximum stresses and maximum cumulative usage factor were obtained from GGNS NSSS new loads design adequacy evaluation report: GE-NEDC-30273.

G. The structural integrity of the RPV feedwater were evaluated for EPU conditions at 4408 MWt and it was concluded these components will continue to satisfy the applicable stress criteria and the applicable fatigue usage factor criteria following EPU implementation (Ref. 15).

TABLE 3.9-2A: REACTOR PRESSURE VESSEL AND SHROUD SUPPORT ASSEMBLY
(IV) CRD PENETRATION

ASME B&PV Code Sec. III Subsection NB Primary Stress Limit Criteria	Loading	Primary Stress Type	Allowable Stress (psi)	Maximum Calculated Stress (psi)
MATERIAL: Inconel SB-167				
A. NORMAL & UPSET CONDITION:				
$P_m \leq S_m$	Normal and Upset Condition Loads	PRIMARY MEMBRANE	20,000	8,485
S_m = 20,000 @ 550 F	1. Normal loads			
$P_L + P_b \leq S_y$	2. Upset pressure			
S_y = 24,100 @ 575 F	3. SRV actuation-all	PRIMARY MEMBRANE PLUS BENDING	24,100	14,200
	4. Operating basis earthquake			
B. EMERGENCY CONDITION:				
$P_m \leq S_y$	Normal and Upset Condition Loads	PRIMARY MEMBRANE	24,480	8,485
S_y = 24,480 @ 528 F	1. Normal loads			
$P_L + P_b \leq 1.5S_y$	2. Upset pressure			
S_y = 24,480 @ 528 F	3. SRV actuation-all	PRIMARY MEMBRANE PLUS BENDING	36,720	14,200
	4. Operating basis earthquake			
C. FAULTED CONDITION:				
$P_m \leq S_y$	Faulted Condition Loads	PRIMARY MEMBRANE	24,480	10,750
S_y = 24,480 @ 528 F	1. Normal loads			

TABLE 3.9-2A: REACTOR PRESSURE VESSEL AND SHROUD SUPPORT ASSEMBLY (CONTINUED)
(IV) CRD PENETRATION

ASME B&PV Code Sec. III Subsection NB Primary Stress Limit Criteria	Loading	Primary Stress Type	Allowable Stress (psi)	Maximum Calculated Stress (psi)
MATERIAL: Inconel SB-167 (Cont.)				
$P_L + P_b \leq 1.5S_y$ S_y = 24,480 @ 528 F	2. Faulted pressure 3. Jet reaction 4. SCRAM 5. Safe shutdown earthquake	PRIMARY MEMBRANE PLUS BENDING	36,720	24,910
D. MAXIMUM CUMULATIVE USAGE FACTOR:		0.107	At Stub Tube	

E. The fatigue usage factor shown in this table is the design fatigue usage factor. For operating fatigue usage factor see Mechanical Standard MS-40.

F. Summary of maximum stresses and maximum cumulative usage factor were obtained from GGNS NSSS new loads design adequacy evaluation report: GE-NEDC-30273.

G. The structural work integrity of the CRD penetration was evaluated for EPU conditions at 4408 MWt and it was concluded the RPV support will continue to meet the applicable structural design acceptance criteria following EPU implementation (Ref. 15).

TABLE 3.9-2B: REACTOR INTERNALS & ASSOCIATED EQUIPMENT
(I) GRID - HIGHEST STRESSED BEAM

ASME B&PV Code Sec. III Subsection NG Primary Stress Limit Criteria	Loading	Primary Stress Type	Allowable Stress (psi)	Maximum Calculated Stress (psi)
MATERIAL: 304 L SS				
A. NORMAL & UPSET CONDITION:				
$P_m \leq S_m$	Normal and Upset Condition Loads	PRIMARY MEMBRANE	14,300	9,567
S_m = 14,300 @ 550 F	1. Pressure differential			
$P_L + P_b < 1.5S_m$	2. Weight of structure	PRIMARY MEMBRANE PLUS BENDING	21,450	14,734
S_m = 14,300 @ 550 F	3. SRV-all			
	4. OBE-seismic			
B. EMERGENCY CONDITION:				
$P_m \leq 1.5S_m$	Emergency Condition Loads	PRIMARY MEMBRANE	21,450	9,567
S_m = 14,300 @ 550 F	1. Pressure differential			
$P_L + P_b < 2.25S_m$	2. Weight of structure	PRIMARY MEMBRANE PLUS BENDING	32,175	14,734
S_m = 14,300 @ 550 F	3. SRV-ADS			
	4. Annulus pressurization			
C. FAULTED CONDITION:				
$P_m \leq 2.4S_m$	Faulted Condition Loads	PRIMARY MEMBRANE	34,320	19,134
S_m = 14,300 @ 550 F	1. Pressure differential			
$P_L + P_b < 3.6S_m$	2. Weight of structure	PRIMARY MEMBRANE PLUS BENDING	51,480	29,468

TABLE 3.9-2B: REACTOR INTERNALS & ASSOCIATED EQUIPMENT (CONTINUED)
(I) GRID - HIGHEST STRESSED BEAM

ASME B&PV Code Sec. III Subsection NG Primary Stress Limit Criteria	Loading	Primary Stress Type	Allowable Stress (psi)	Maximum Calculated Stress (psi)
MATERIAL: 304 L SS (Cont.)				
S_m = 14,300 @ 550 F	3. Annulus pressurization 4. SRV-all			

D. MAXIMUM CUMULATIVE USAGE FACTOR: 0.717 At Grid Beam

E. The structural work integrity of the RPV internals were evaluated for EPU conditions at 4408 MWt and it was concluded the grid will continue to satisfy the applicable stress criteria following EPU implementation, including consideration of the effects of flow induced vibration and the effects of EPU implementation on the design loads and loading combinations (Ref. 15).

TABLE 3.9-2B: REACTOR INTERNALS & ASSOCIATED EQUIPMENT
(II) CORE PLATE (LIGAMENT IN TOP PLATE)

ASME B&PV Code Sec. III Subsection NG Primary Stress Limit Criteria	Loading	Primary Stress Type	Allowable Stress (psi)	Maximum Calculated Stress (psi)
MATERIAL: 304 L SS				
A. NORMAL & UPSET CONDITION:				
$P_m \leq S_m$	Normal and Upset Condition Loads	PRIMARY MEMBRANE	14,300	12,838
S_m = 14,300 @ 550 F	1. Normal loads			
$P_L + P_b < 1.5S_m$	2. Upset Pressure			
S_m = 14,300 @ 550 F	3. Operating basis earthquake	PRIMARY MEMBRANE PLUS BENDING	21,450	19,299
	4. SRV-all			
B. EMERGENCY CONDITION:				
$P_m \leq 1.5S_m$	Emergency Condition Loads	PRIMARY MEMBRANE	21,450	17,410
S_m = 14,300 @ 550 F	1. Normal loads			
$P_L + P_b < 2.25S_m$	2. Upset Pressure			
S_m = 14,300 @ 550 F	3. Chugging	PRIMARY MEMBRANE PLUS BENDING	32,175	25,965
	4. SRV-all			
C. FAULTED CONDITION:				
$P_m \leq 1.5S_m$	Faulted Condition Loads	PRIMARY MEMBRANE	34,320	30,280
S_m = 14,300 @ 550 F	1. Normal loads			
$P_L + P_b < 2.25S_m$	2. Faulted Pressure			
S_m = 14,300 @ 550 F	3. Chugging	PRIMARY MEMBRANE PLUS BENDING	51,480	45,157
	4. SRV-ADS			

TABLE 3.9-2B: REACTOR INTERNALS & ASSOCIATED EQUIPMENT (CONTINUED)
(II) CORE PLATE (LIGAMENT IN TOP PLATE)

ASME B&PV Code Sec. III Subsection NG Primary Stress Limit Criteria	Loading	Primary Stress Type	Allowable Stress (psi)	Maximum Calculated Stress (psi)
MATERIAL: 304 L SS (Cont.)				
	5. Safe Shutdown Earthquake			
D. MAXIMUM CUMULATIVE USAGE FACTOR:		0.90		At cover Plate Rim and Rim to Cover Plate Weld

E. Summary of maximum stresses and maximum cumulative usage factor were obtained from GGNS NSSS new loads design adequacy evaluation report: GE-NEDC-30273.

F. The structural integrity of the RPV internals were evaluated for EPU conditions at 4408 MWt and it was concluded the core plate will continue to satisfy the applicable stress criteria following EPU implementation, including consideration of the effects of flow induced vibration and effects of EPU implementation of the design loads and loading combinations (Ref. 15).

TABLE 3.9-2B: REACTOR INTERNALS & ASSOCIATED EQUIPMENT
(III) VENT AND HEAD SPRAY NOZZLE

ASME B&PV Code Sec. III Subsection NG Primary Stress Limit Criteria	Loading	Primary Stress Type	Allowable Stress (psi)	Maximum Calculated Stress (psi)
MATERIAL: SA 333, 350 & 420 CS				
A. NORMAL & UPSET CONDITION:				
$P + Q_m \leq 3.0S_m$	Normal and Upset Condition Loads	PRIMARY MEMBRANE PLUS BENDING	53,100	47,400
S_m = 17,700 @ 575 F	1. Pressure	PLUS SECONDARY		
	2. Weight	MEMBRANE		
	3. OBE			
	4. SRV-all			
MATERIAL: SA 333, 350 & 420 CS (Cont.)				
B. EMERGENCY CONDITION:				
$P_L + P_b \leq 2.25S_m$	Emergency Condition Loads	PRIMARY MEMBRANE PLUS BENDING	39,800	26,550
S_m = 17,700 @ 575 F	1. Pressure			
	2. Weight			
	3. OBE			
	4. SRV-all			
C. FAULTED CONDITION:				
	Faulted Condition Loads			
	1. Pressure			
	2. Weight			
$P_L + P_b \leq 0.7S_m$	3. Safe shutdown earthquake	PRIMARY MEMBRANE PLUS BENDING	49,000	26,550

TABLE 3.9-2B: REACTOR INTERNALS & ASSOCIATED EQUIPMENT (CONTINUED)
(III) VENT AND HEAD SPRAY NOZZLE

ASME B&PV Code Sec. III Subsection NG Primary Stress Limit Criteria	Loading	Primary Stress Type	Allowable Stress (psi)	Maximum Calculated Stress (psi)
MATERIAL: SA 333, 350 & 420 CS (Cont.)				
S_m = 70,000 @ 575 F	4. Annulus pressurization 5. Max. of SRV-all and SRV-ADS 6. Jet reaction 7. Chugging			

D. MAXIMUM CUMULATIVE USAGE FACTOR: 0.79 At Nozzle

E. The fatigue usage factor shown in this table is the design fatigue usage factor. For operating fatigue usage factor see Mechanical Standard MS-40.

F. The structural integrity of the RPV internals were evaluated for EPU conditions at 4408 MWt and it was concluded the head spray nozzle will continue to satisfy the applicable stress criteria following EPU implementation, including consideration of the effects of flow induced vibration and the effects of EPU implementation on the design loads and loading combinations (Ref. 15).

TABLE 3.9-2B: REACTOR INTERNALS & ASSOCIATED EQUIPMENT
(IV) ORIFICED FUEL SUPPORT

Criteria	Loading	Limiting Stress Type	Allowable Stress (psi)	Calculated Stress (psi)
Based on ASME Code, Section III Subsection NG S_m @ 550 F = 16,000 psi				
For Normal, Upset, and Emergency Conditions: S_{limit} = 1.5 X .65S_m See notes (1) and (2)	Normal, Upset, and Emergency Loads: 1. Temperature 2. OBE + SRV 3. SCRAM 4. Pressure 5. Weight of components	Primary Membrane Plus Bending	15,600	14,683
For Faulted Condition: S_{limit} = 1.5 X 2.4 X .65S_m See note (2)	Faulted Loads: 1. Temperature 2. SSE +SRV 3. SCRAM 4. Pressure 5. Weight of components	Primary Membrane Plus Bending	37,400	30,794

NOTES:

(1) Normal, upset and emergency loads are analyzed together. Stresses are compared to upset limits.

(2) Casting quality factor = .65 per ASME Code, Section III, NG-2571.2(a).

(3) The structural integrity of the RPV internals were evaluated for EPU conditions at 4408 MWt and it was concluded the orificed fuel support will continue to satisfy the applicable stress criteria following EPU implementation, including consideration of the effects of flow induced vibration and the effects of EPU implementation on the design loads and loading combinations (Ref. 15).

TABLE 3.9-2B: REACTOR INTERNALS & ASSOCIATED EQUIPMENT
(V) CORE SPRAY LINES AND SPARGER

Criteria	Loading	Limiting Stress Type	Allowable Stress (psi)	Calculated Stress (psi)
For type 304 SS, S_m @ 550 F = 16,950 psi				
For type 316 L, S_m @ 550 F = 13,950 psi				
For Normal and Upset Condition:	Normal and Upset Loads:	PRIMARY MEMBRANE	41,850	30,758
S_{limit} - $3.0S_m$	1. SRV	+ BENDING		
	2. OBE	+ SECONDARY		
	3. Normal	MEMBRANE		
	4. Pressure			
For Emergency Condition:	Emergency Loads:	PRIMARY MEMBRANE	31,387	16,772
S_{limit} - $2.25S_m$	1. SRV	+ BENDING		
	2. SSE			
	3. Normal			
(1)	4. Pressure			
For Faulted Condition:	Faulted Loads:	PRIMARY MEMBRANE	61,020	49,130
S_{limit} - $3.6S_m$ (1)	1. A.P.	+ BENDING		
	2. SSE			
	3. Normal			
	4. Pressure			

NOTES:

(1) The component material at the location where the most limiting load occurs under faulted conditions was type 304 SS; thus S_{limit} equals 3.6 X 16,950.

TABLE 3.9-2B: REACTOR INTERNALS & ASSOCIATED EQUIPMENT (CONTINUED)
(V) CORE SPRAY LINES AND SPARGER

Criteria	Loading	Limiting Stress Type	Allowable Stress (psi)	Calculated Stress (psi)
(2) The structural integrity of the RPV internals were evaluated for EPU conditions at 4408 MWt and it was concluded the core spray line and sparger will continue to satisfy the applicable stress criteria following EPU implementation, including consideration of the effects of flow induced vibration and the effects of EPU implementation on the design loads and loading combinations (Ref. 15).				

TABLE 3.9-2B: REACTOR INTERNALS & ASSOCIATED EQUIPMENT
(VI) FUEL ASSEMBLY (INCLUDING CHANNEL)

Acceptance Criteria	Loading[2]	Primary Load Type	Calculated Peak Acceleration	Evaluation Basis Acceleration
Acceleration Envelope	Horizontal Direction: 1. Peak Pressure 2. Safe Shutdown Earthquake 3. Annulus Pressurization	Horizontal Acceleration Profile	1.5 g	(1)
	Vertical Direction: 1. Peak Pressure 2. Safe Shutdown Earthquake 3. Safety Relief Valve 4. SCRAM	Vertical Accelerations	2.3 g[3]	(1)

NOTES:

(1) Evaluation Basis Accelerations and Evaluations are contained in NEDE-21175-3-P.

(2) The fatigue analysis indicates that the fuel assembly has adequate fatigue capability to withstand the loads resulting from multiple SRV actuations and the OBE+SRV event.

(3) These values are determined using the methodology contained in NEDE-21175-3-P. The calculated maximum fuel assembly gap opening for the most limiting load combination is 0.002 inch.

(4) The structural integrity of the RPV internals were evaluated for EPU conditions at 4408 MWt and it was concluded the fuel channel and peripheral fuel support will continue to satisfy the applicable stress criteria following EPU implementation, including consideration of the effects of flow induced vibration and the effects of EPU implementation on the design loads and loading combinations (Ref. 15).

TABLE 3.9-2C: REACTOR WATER CLEANUP HEAT EXCHANGERS
(I) REGENERATIVE

Criteria	Loading	Component	Minimum Thickness Required (inches)	Actual Thickness (inches)
1. Closure Bolting				
Bolting requirements are calculated per rules of ASME Codes, Section III	Design Basic loads consisting of:	Bolting - channel to shell flange	Dia.	Dia.
Primary stress limit for SA-193-B7	a. Design pressure b. Design temperature c. Design gasket load		N/A	N/A
S = 25,000 psi				
2. Wall Thickness				
Wall thickness requirements are calculated per rules of ASME Section III, Class 3 and TEMA Class C.	a. Design pressure b. Design temperature	Shell Shell head Channel shell Channel cover Tube sheet Tubes (# BWG)	0.94 1.082 0.94 1.082 4.8387 0.071	1.00 1.25 1.00 1.25 6.00 0.095
Primary stress limit for:				
Carbon steel				
S = 17,500 psi				
Austenitic stainless steel				
S = 11,900 psi				

TABLE 3.9-2C: REACTOR WATER CLEANUP HEAT EXCHANGERS
(I) REGENERATIVE

Criteria	Loading[3]	Component	Allowable Nozzle Loads	Actual Nozzle Loads
3. Nozzle Loads				
The maximum forces and moments due to pipe reactions shall not exceed the allowable limits	a. Design pressure b. Design temperature c. Dead Weight d. Thermal Expansion e. Seismic (Class II basis)	Nozzle N1 (tube inlet)	[1]Fo = 1520 lb Mo = 1230 ft-lb	836 lb 2,777 ft-lb[2]
		Nozzle N2 (tube outlet)	Fo = 1520 lb Mo = 1230 ft-lb	436 lb 680 ft-lb
		Nozzle N3 (shell inlet)	Fo = 2910 lb Mo = 3530 ft-lb	1240 lb 2758 ft-lb[2]
		Nozzle N4 (shell outlet)	Fo = 2910 lb Mo = 3530 ft-lb	3,337 lb 7,868 ft-lb[2]

(1) Maximum allowable piping loads shall not exceed the following relationship for each nozzle:
[Fi/Fo] + [Mi/Mo] ≤ 1

Where:

Fi (lb) = Maximum of three orthogonal forces (Fx, Fy, Fz).

Mi (ft/lb) = Maximum of three orthogonal moments (Mx, My,Mz).

Fo (lb) = The allowable value of Fi when all moments are zero.

Mo (ft/lb) = The allowable value of Mi when all forces are zero.

(2) Because the calculated nozzle loads exceed the allowable value, a refined analysis was made and the actual loads were within the capability of the nozzle.

(3) Design assessment evaluation for the reactor water cleanup regenerative heat exchangers is not warranted as LOCA and SRV loads are negligible.

TABLE 3.9-2C: REACTOR WATER CLEANUP HEAT EXCHANGERS
(II) NONREGENERATIVE

	Criteria	Loading	Component	Minimum Thickness Required (inches)	Actual Thickness (inches)
1.	Closure Bolting				
	Bolting requirements are calculated per rules of ASME Codes, Section III Primary stress limit for SA-193-B7 S = 25,000 psi	Design basis loads consisted of: a. Design pressure b. Design temperature c. Design gasket load	Bolting - channel to shell flange	____	1.875 Dia.
2.	Wall Thickness				
	Wall thickness requirements are calculated per rules of ASME Section III, Class 3 components and TEMA Class C.	a. Design pressure b. Design temperature	Shell	0.876	1.0
			Shell head	0.2104	0.375
			Channel shell	1.0573	1.125
			Channel cover	1.43	1.5
			Tube sheet	5.231	6.25
	Primary stress limit for: Carbon steel S = 17,500 psi Austenitic stainless steel S = 11,700 psi		Tubes	0.0383	0.058

TABLE 3.9-2C: REACTOR WATER CLEANUP HEAT EXCHANGERS
(II) NONREGENERATIVE

Criteria	Loading[3]	Component	Allowable Nozzle Loads	Actual Nozzle Loads
3. Nozzle Loads				
The maximum forces and moments due to pipe reactions shall not exceed the allowable limits.	a. Design pressure b. Design temperature c. Dead weight d. Thermal expansion e. Seismic (Class II basis)	Nozzle N1 (tube inlet)	Fo = 1520 lb Mo = 1230 ft-lb	Fi = 438 lb Mi = 708 ft-lb
		Nozzle N2 (tube outlet)	Fo = 1520 lb Mo = 1230 ft-lb	Fi = 903 lb Mi = [3]1740 ft-lb
		Nozzle N3 (shell inlet)	Fo = 2910 lb Mo = 4850 ft-lb	Fi = 1599 lb Mi = 3427 ft-lb[3]
		Nozzle N4 (shell outlet)	Fo = 2910 lb Mo = 4850 ft-lb	Fi = 1705 lb Mi = 2070 ft-lb[3]

(1) Maximum allowable piping loads shall not exceed the following relationship for each nozzle:
[Fi/Fo] + [Mi/Mo] ≤ 1

Where: Fi (lb) = Maximum of three orthogonal forces (Fx, Fy, Fz).
Mi (ft/lb) = Maximum of three orthogonal moments (Mx, My, Mz).
Fo (lb) = The allowable value of Fi when all moments are zero.
Mo (ft/lb) = The allowable value of Mi when all forces are zero.

(2) Design assessment evaluation for the reactor water cleanup nonregenerative heat exchanger is not warranted as LOCA and SRV loads are negligible.

(3) These maximum loads are acceptable per DCA 91-101 (GEXI-91-1593)

(4) See SEGE-87128 and GE Drawing 762E530 for qualified loads for Nozzle N2

TABLE 3.9-2D: ASME CODE CLASS 1 MAIN STEAM PIPING AND PIPE MOUNTED EQUIPMENT - HIGHEST STRESS SUMMARY

Acceptance Criteria	Limiting Stress Type	Calculated Stress[1] or Usage Factor	Allowable Limits	Ratio Actual/ Allowable	Loading[1]	Identification[2] of Locations of Highest Stress Points - NODE Point Numbers
ASME B&PV Code Section III, NB-3600						
Design Condition:					1. Pressure	Lug Guide
					2. Weight	Steam Line A
Eq. 9 $\leq 1.5S_m$[3]	Primary	10,590 psi	28,725 psi	0.37	3. OBE	
Service Levels A & B (Normal & Upset) Condition:						
Eq. 12 $\leq 3.0S_m$[3]	Secondary	42,617 psi	57,450 psi	0.74	1. Pressure 2. Weight 3. Thermal expansion 4. OBE 5. SRV (Acoustic wave and structural feedback)	Joint 2N 1st Elbow Steam Line A
Service Levels A & B (Normal & Upset) Condition: Eq. 13 $\leq 3.0S_m$[4]	Primary Plus Secondary (Except Thermal Expansion)	29,772 psi	53,100 psi	0.56	1. Pressure 2. Weight 3. OBE 4. SRV (Acoustic wave and structural feedback)	Joint 28 Steam Line A

TABLE 3.9-2D: ASME CODE CLASS 1 MAIN STEAM PIPING AND PIPE MOUNTED EQUIPMENT - HIGHEST STRESS SUMMARY

Acceptance Criteria	Limiting Stress Type	Calculated Stress(1) or Usage Factor	Allowable Limits	Ratio Actual/ Allowable	Loading(1)	Identification(2) of Locations of Highest Stress Points - NODE Point Numbers
Service Levels A & B (Normal and Upset) Conditions:					1. Pressure 2. Weight 3. OBE	Joint 17 Flow Element Weld Steam Line A
Cumulative Usage Factor (5)	N/A	0.07	1.0	0.07	4. SRV (Acoustic wave and structural feedback)	
Service Level B (Upset) Condition:					1. Pressure 2. Weight 3. OBE	Lug Guide Steam Line A
Eq. 9 $\leq 1.8S_m$(3) & 1.5Sy	Primary	19,861 psi	34,470 psi	0.58	4. SRV (Acoustic wave and structural feedback)	
Service Level C (Emergency) Condition:					1. Pressure 2. Weight 3. Pressure (ATWS) 4. LOCA (SBA/IBA)	Lug Guide Steam Line A
Eq. 9 < $2.25S_m$(3) & 1.8Sy	Primary	19,824 psi	43,088 psi	0.46	5. SRV (Acoustic wave and structural feedback)	

TABLE 3.9-2D: ASME CODE CLASS 1 MAIN STEAM PIPING AND PIPE MOUNTED EQUIPMENT - HIGHEST STRESS SUMMARY

Acceptance Criteria	Limiting Stress Type	Calculated Stress(1) or Usage Factor	Allowable Limits	Ratio Actual/ Allowable	Loading(1)	Identification(2) of Locations of Highest Stress Points - NODE Point Numbers
Service Level D (Faulted) Condition:					1. Pressure 2. Weight 3. SSE	Lug Guide Steam Line A
Eq. 9 < $3.0S_m$(3)	Primary	19,953 psi	57,450 psi	0.35	4. TSVC 5. LOCA (SBA/IBA/DBA) 6. SRV (Acoustic wave and structural feedback)	

TABLE 3.9-2D: ASME CODE CLASS 1 MAIN STEAM PIPING AND PIPE MOUNTED EQUIPMENT - HIGHEST STRESS SUMMARY

Component/Load Type	Highest Calculated Load	Allowable Load	Ratio Calculated/ Allowable	Loading[1]	Identification of Equipment[2] with Highest Loads
Service Level B Loads	47,026 lb	70,000 lb	0.67	1. Pressure 2. Weight 3. OBE 4. SRV (Acoustic wave and structural feedback)	Main Steam Line C Snubber S106
Service Level C Loads	26,911 lb	93,100 lb	0.29	1. Pressure 2. Weight 3. OBE 4. SRV (Acoustic wave & structural feedback)	Main Steam Line C Snubber S106
Service Level D Loads	59,707 lb	75,000 lb	0.80	1. Pressure 2. Weight 3. Large break LOCA	Main Steam Line C Snubber S101
Service Level D Loads Bonnet/Moment	903,192 in-lb	1,780,000 in-lb	0.507	1. Pressure 2. Weight 3. Large break LOCA	Main Steam Line C MSIV

TABLE 3.9-2D: ASME CODE CLASS 1 MAIN STEAM PIPING AND PIPE MOUNTED EQUIPMENT - HIGHEST STRESS SUMMARY

Component/Load Type	Highest Calculated Load	Allowable Load	Ratio Calculated/ Allowable	Loading[1]	Identification of Equipment[2] with Highest Loads
Service Level D Loads Flange Moment	591,721 in-lb	1,360,000 in-lb	0.4351	1. Pressure 2. Weight 3. SSE 4. LOCA (SBA/IBA/DBA) 5. SRV (Acoustic wave & structural feedback)	Main Steam Line C SRV Inlet
Service Level D Loads	187,287 in-lb	600,000 in-lb	0.3121	1. Pressure 2. Weight 3. SSE 4. LOCA (SBA/IBA/DBA) 5. SRV (Acoustic wave & structural feedback)	Main Steam Line C Weight SRV outlet
Service Level D Loads Horizontal Accelerations	6.00 g	9.0 g	0.667	1. Pressure 2. Weight 3. Large break LOCA	Main Steam Line A SRV

TABLE 3.9-2D: ASME CODE CLASS 1 MAIN STEAM PIPING AND PIPE MOUNTED EQUIPMENT - HIGHEST STRESS SUMMARY

Component/Load Type	Highest Calculated Load	Allowable Load	Ratio Calculated/ Allowable	Loading(1)	Identification of Equipment(2) with Highest Loads
Service Level D Loads Vertical Acceleration	5.34 g	6.0 g	0.89	1. Pressure 2. Weight 3. SSE 4. LOCA (SBA/IBA/DBA) 5. SRV (Acoustic wave and structural feedback)	Main Steam Line A SRV

Notes: (1) Appropriate loading combinations of Table 3.9-2 were considered and the calculated stresses are reported for the governing loading combinations.
(2) Refer to Figure 3.6-6a for the identification of node point numbers.
(3) ASME SA-155 Grade KCF-70 @ 575 F, S_m, = 19,150 lb/ft^2
(4) ASME SA-106 Grade B @ 575 F, S_m, = 17,700 lb/ft^2
(5) The fatigue usage factor shown in this table is the design fatigue usage factor. For operating fatigue usage factor see Mechanical Standard MS-40.
(6) The structural integrity of the main steam piping affected by EPU implementation was evaluated for EPU conditions at 4408 MWt and it was concluded the piping will continue to meet the applicable structural criteria to which this system was originally designed (Ref. 15).
(7) Note: MS Lines were evaluated for TPO at 3898 MWt and concluded that all load limits mentioned above were met

TABLE 3.9-2E: ASME CODE CLASS 1 RECIRCULATION PIPING AND PIPE MOUNTED EQUIPMENT - HIGHEST STRESS SUMMARY

Acceptance Criteria	Limiting Stress Type	Calculated Stress(1) or Usage Factor	Allowable Limits	Ratio Actual/ Allowable	Loading	Identification(2) of Locations of Highest Stress Points - NODE Point Numbers
ASME B&PV Code Section III, NB-3600						
Design Condition: Eq. 9 ≤ $1.5S_m$	Primary	14.436 psi	25,013 psi	0.58	1. Pressure 2. Weight 3. OBE	Hanger logs
Service Levels A & B (Normal & Upset) Condition: Eq. 12 ≤ $3.0S_m$	Secondary	31,262 psi	50,025 psi	0.62	1. Pressure 2. Weight 3. OBE 4. SRV (Structural feedback)	RHR tee
Service Levels A & B (Normal & Upset) Condition: Eq. 13 ≤ $3.0S_m$	Primary Plus Secondary (Except Thermal Expansion)	38,089 psi	50,025 psi	0.76	1. Pressure 2. Weight 3. OBE 4. SRV (Structural feedback)	Header sweepolet at azimuth of 257° 15′
Service Levels A & B (Normal and Upset) Condition: Cumulative Usage Factor(3)	N/A	0.10	1.0	0.10	1. Pressure 2. Weight 3. OBE 4. SRV (Structural feedback)	RHR Tee

TABLE 3.9-2E: ASME CODE CLASS 1 RECIRCULATION PIPING AND PIPE MOUNTED EQUIPMENT - HIGHEST STRESS SUMMARY

Acceptance Criteria	Limiting Stress Type	Calculated Stress(1) or Usage Factor	Allowable Limits	Ratio Actual/ Allowable	Loading	Identification(2) of Locations of Highest Stress Points - NODE Point Numbers
Service Level B (Upset) Condition:					1. Pressure 2. Weight 3. OBE 4. SRV (Structural feedback)	
Eq. 9 ≤ 1.8Sm & 1.5S_y	Primary	19,001 psi	28,596 psi	0.66		RHR Tee
Service Level C (Emergency) Condition:					1. Pressure 2. Weight 3. SRV	
Eq. 9 ≤ 2.25S_m & 1.8S_y	Primary	19,222 psi	34,315 psi	0.56		
Service Level D (Faulted) Condition:					1. Pressure 2. Weight 3. SSE 4. SRV	
Eq. 9 ≤ 3.0S_m	Primary	19,487 psi	50,025 psi	0.39		

TABLE 3.9-2E: ASME CODE CLASS 1 RECIRCULATION PIPING AND PIPE MOUNTED EQUIPMENT - HIGHEST STRESS SUMMARY

Component/Load Type	Highest Calculated Load	Allowable Load	Ratio Calculated/ Allowable	Loading(1)	Identification of Equipment with Highest Loads
Service levels A & B Loads	27,538 lb	100,000 lb	0.275	OBE+SRV_{ALL}	Loop B Snubber 371
Service levels C & D Loads	50,488 lb	150,000 lb	0.337	AP+SSE	Loop B Snubber 369
Discharge Valve					
Vert. Acceleration	1.68 g	3.00 g	0.558	AP+SSE	Loop B
Horiz. Acceleration	6.16 g	11.60 g	0.531	AP+SSE	Loop B
Suction Valve					
Vert. Acceleration	1.24 g	3.00 g	0.411	AP+SSE	Loop B
Horiz. Acceleration	3.63 g	10.66 g	0.341	AP+SSE	Loop B

Notes: (1) Appropriate loading combinations of Table 3.9-2 were considered and the calculated stresses are for the governing loading combinations.
(2) Refer to Figure 3.6-4c for the identification of node point numbers.
(3) The fatigue usage factor shown in this table is the design fatigue usage factor. For operating fatigue usage factor see Mechanical Standard MS-40.
(4) The structural integrity of the recirculation piping affected by EPU implementation was evaluated for EPU conditions at 4408 MWt and it was concluded the piping will not experience increases in stresses and fatigue. Furthermore, the 12 percent increase in core flow resulting from EPU is bounded by testing performed for the recirculation system piping during the original power ascension testing, the results of which were found acceptable. Therefore, the flow induced vibration levels within this piping will remain acceptable following EPU implementation (Ref. 15).

TABLE 3.9-2F: RECIRCULATION FLOW CONTROL VALVE 24" SIZE (FISHER)
CODE ASME SECT. III 1971 EDITION, WITH W73 ADDENDA

Para No.	Component/Stress/Loading	Design Procedure	Allowable Lim.	Calculated or Actual Value	Ratio Calc. Allowed
1.0	Body, Housing, Bonnet & Covers				
1.1	Loads - Design Pressure Design Temperature	System Requirement System Requirement		1675 psi 575 F	
1.2	Body Pressure Rating	ASME Sec. III NB3545.1-2	--	985 psi	
1.3	Body Min. Wall Thickness	ASME Sec. III NB3541	t_m = 2.614 in	t_m = 2.710 in	N/A
1.4	Max. Primary Body Membrane Stress	ASME Sec. III NB3545.1	$P_m \leq S_m$ (575 F) = 19,600 psi	P_m = 10,265 psi	0.523
1.5	Max. Primary + Sec. Body Stress	ASME Sec. III NB3545.2	$S_m \leq 3S_m$ ≤ 58,800 psi	S_m = 22,000 psi	0.374
1.6	Housing Min. Wall Thickness	ASME Sec. III NB3541	t_m = 2.549 in	t_m = 2.710 in	N/A
1.7	Max. Primary Housing Membrane Stress	ASME Sec. III NB3545.1	$P_m \leq S_m$ (575 F) ≤ 19,600 psi	P_m = 8,400 psi	0.428
1.8	Max. Primary + Sec. Housing Stress	ASME Sec. III NB3545.2	$S_m < 3S_m$ (575 F) < 58,800 psi	S_m = 23,100 psi	0.392
1.9	Cyclic Requirements	ASME Sec. III NB3545.3	$N_a \geq$ 2,000 cyc	$N_a = 10^6$ cyc	

TABLE 3.9-2F: RECIRCULATION FLOW CONTROL VALVE 24" SIZE (FISHER)
CODE ASME SECT. III 1971 EDITION, WITH W73 ADDENDA

Para No.	Component/Stress/Loading	Design Procedure	Allowable Lim.	Calculated or Actual Value	Ratio Calc. Allowed
1.1	Fatigue Analysis Usage Factor	ASME Sec. III NB3550	$I_t \leq 1.0$	I_t = .0004	
1.11	Body to Housing Flange Max. Stress	ASME Sec. III NB3647.1	S_m = 29,400 psi (1.5 x 19,600)	S_m = 27,440 psi	0.933
1.12	Body to Housing Studs - Area	ASME Sec. III NB3647.1	$A_6 = \geq 29.04$ in²	A_6 = 31.68 in²	1.090
	Body to Housing Primary Stress		S_m = 27,000 psi	S_6 = 24,750 psi	0.916
	Body to Housing Maximum Stress		$3S_m$ = 81,000 psi	S_6 = 66,500 psi	0.820
1.13	Top Housing Cover - Thickness	ASME Sec. III NB3646 & ASME Sec. VIII UG-34	$t_m \geq 3.23$ in	t_m = 4.63 in	1.433
1.14	Top Housing Cover Studs - Area	ASME Sec. III NB3647.1	$A_6 \geq 31.75$ in²	A_6 = 33.60 in²	1.058
	Top Housing Primary Stud Stress		S_m = 27,000 psi	S_6 = 25,900 psi	0.959
	Top Housing Maximum Stud Stress		$3S_m$ = 81,000 psi	S_6 = 45,100 psi	0.556
1.15	Bottom Cover - Thickness	NB3646 & Sec. VIII UG-34	$t_m \geq 1.85$ in	t_m = 3.56 in	1.924
1.16	Bottom Cover Primary Stud Stress	ASME Sec. III NB3647.1	S_m = 27,000 psi	S_6 = 25,000 psi	0.925
	Bottom Cover Maximum Stud Stress		$3S_m$ = 81,000 psi	S_6 = 55,380 psi	0.683
	Bottom Cover Studs - Area		$A_6 \geq 11.09$ in²	A_6 = 13.44 in²	1.211

TABLE 3.9-2F: RECIRCULATION FLOW CONTROL VALVE 24" SIZE (FISHER) CODE ASME SECT. III 1971 EDITION, WITH W73 ADDENDA

Para No.	Component/Stress/Loading	Design Procedure	Allowable Lim.	Calculated or Actual Value	Ratio Calc. Allowed
1.17	Bonnet Cartridge - Thickness	NB3646 & Sec. VIII UG-34	$t_m \geq 1.82$ in	$t_m = 3.125$ in	1.717
1.18	Bonnet Cart. Studs - Area	ASME Sec. III NB3647.1	$A_6 \geq 12.07$ in^2	$A_6 = 18.48$ in^2	1.531
	Bonnet Cart. Primary Stud Stress		$S_m = 27{,}000$ psi	$S_6 = 17{,}630$ psi	0.652
	Bonnet Cart. Maximum Stud Stress		$3S_m = 81{,}000$ psi	$S_6 = 51{,}680$ psi	0.638

Note: The recirculation flow control valves are passive components and therefore not required to operate in emergency or faulted conditions. The valves have been designed for 6 g's vertical and 9 g's horizontal, which exceeds any load condition from Table 3.9-2a. The valves will maintain pressure integrity during and after events imposing these accelerations. The valve internals (ballshaft, linkage, bearings, etc.) have also been designed for faulted, large pipe break conditions. Qualification method is by analysis only.

TABLE 3.9-2G: MAIN STEAM SAFETY/RELIEF VALVES (I) ASME CODE, SECTION III, JULY 1974

Topic	Method of Analysis	Dikkers Analysis	Allowable Value	Calculated
1. Body inlet and outlet flange stresses Note, Topics 1 and 2: Design Pressures: P_d = 1375 psig (inlet) P_b = 625 psig (outlet) These are the max anticipated pressures under all operating conditions. Analyses include applied moments of M=800,000 in/lb (inlet) and M=300,00 in/lb (outlet) The analyses also include consideration of seismic, operational, and flow reaction forces. Since these safety/relief valves are pipe-mounted equipment, refer to the piping analysis for verification that the moments are not exceeded.	$S_H = \frac{fMo}{Lg_1^2B} + \frac{PB}{4g_o} < 1.5Sm$ $S_R = \frac{(4te\,\S\,3+1)Mo}{Lt^2B} < 1.5Sm$ $S_T = \frac{YMo}{t^2B} - ZS_R < 1.5Sm$ where S_H = Longitudinal "Hub" wall stress, psi S_R = Radial "flange" Body Base, Inlet) Stress, psi S_T = Tangential "Flange" Stress, psi	(Uses same notation as codes) Body Material: ASME SA 352 LCB Inlet: Sm @ 585 F = 17,540 psi Outlet: Sm @ 500 F = 18,900 psi	1.5Sm = 26,310 psi (inlet) and = 28,350 psi (outlet)	Inlet: S_H = 1.15 Sm = 0.77 (allowable) S_R = 0.23 Sm = 0.169 (allowable) S_T = 0.98 Sm = 0.66 (allowable) Outlet: S_H = 1.21 Sm = 0.81 (allowable) S_R = 0.79 Sm = 0.53 (allowable) S_T = 0.49 Sm = 0.53 (allowable)

TABLE 3.9-2G: MAIN STEAM SAFETY/RELIEF VALVES (I) ASME CODE, SECTION III, JULY 1974 (CONTINUED)

Topic	Method of Analysis	Dikkers Analysis	Allowable Value	Calculated
2. Inlet and outlet stud area requirements	Total cross-sectional area shall exceed the greater of: $Am_1 = \frac{Wm_1}{Sb}$, or $Am_2 = \frac{Wm_2}{Sa}$ where Am_1 = total required bolt (stud) area for operating condition. Am_2 = total required bolt (stud) area for gasket seating.	(Uses same notation as codes) Bolting Material: ASME SA 193 GR B7	Inlet: $Am_1 (>Am_2) = 12.45$ in^2 Outlet: $Am_1 (>Am_2) = 4.65$ in^2	Inlet: Ab (actual area) = 1.52 Am (required min) Outlet: Am (actual area) = 1.84 Am (required min)
3. Nozzle wall thickness	1. Minimum Wall Thickness Criterion: $t_{min} < t_A$ where t_{min} = minimum calculated thickness requirement, including corrosion allowance	Section near nozzle base: $t_{m_e} < {}^t m_e$ (actual) Nozzle mid-section: $t_{m_c} < {}^t m_c$ (actual) Thin section near valve seat: $t_{m_b} < {}^t m_b$ (actual)	$t_{m_e} = 0.84$ in $t_{m_c} = 0.84$ in $t_{m_b} = 0.79$ in $t_{m_a} = 0.79$ in	${}^t m_e$ (actual) $= 1.58\ {}^t m_e$ ${}^t m_c$ (actual) $= 1.54\ {}^t m_c$ ${}^t m_b$ (actual) $= 1.012\ {}^t m_b$ ${}^t m_a$ (actual) $= 1.68\ {}^t m_a$

TABLE 3.9-2G: MAIN STEAM SAFETY/RELIEF VALVES (I) ASME CODE, SECTION III, JULY 1974 (CONTINUED)

Topic	Method of Analysis	Dikkers Analysis	Allowable Value	Calculated
	t_A = Actual nozzle wall thickness. (Note: This t_{min} is t_m p.er notation of the codes.	Thinnest section at nozzle tip-just below valve seat:		
		$t_{m_a} < t_{m_a}$ (actual) Nozzle Material: ASME SA 350 LF2	Actual thickness greater than t_m at the section under consideration.	
(Refer to subsection 3.9.1.1.1.9 for thermal transients information)	2. Cyclic Rating: Thermal: $It = \Sigma \frac{Nri}{Ni}$	$It = \Sigma \frac{Nri}{Ni}$ (i=1, 2, 3, 4, & 5)	It (max) ≤ 1.0	It = 0.00138 [-0.00138xIt (max)]
	Fatigue Na ≥ 2,000 cycles, as based on Sa, where Sa is defined as the larger of	Na ≥ 2,000 cycles, as based on Sa, where Sa=Sp_1 (>Sp_2)	Na ≥ 2,000 cycles	Na (based on Sa=Sp_2) = 400,000 cycles: criterion satisfied
	$Sp_1 = (2 \S 3)Q_p + \frac{P_{eb}}{2} + {}^QT_3$ + $1.3{}^QT_1$ or $Sp_2 = 0.4Q_p + \frac{K}{2}({}^Peb + 2{}^QT_3)$ where	(Uses same notation as codes)		

TABLE 3.9-2G: MAIN STEAM SAFETY/RELIEF VALVES (I) ASME CODE, SECTION III, JULY 1974 (CONTINUED)

Topic	Method of Analysis	Dikkers Analysis	Allowable Value	Calculated
	Sp_1 = Fatigue stress intensity at inside surface of crotch, psi			
	Sp_2 = Fatigue stress intensity at inside surface of crotch, psi			
4. Bonnet Flange Strength	Flange treated as a loose type flange without hub:		1.5Sm (for max S_H, S_R, and S_T)	S_R = 1.35 Sm = 0.9 (allowable) S_T = 0.53 Sm = 0.35 (allowable)
	$S_R = \pm\frac{6M_p}{t^2(3.14C - nD)}$	(Uses same notation as codes)	= 28,350 psi	(max S_T @ back face of flange)
	$S_T = \pm\frac{5.46M_p}{Bt^2}\left[0.318\left[\frac{C-B}{C+B}+\frac{2h_c}{C+A}\right]+r_B\right]-\frac{E\theta_A t}{B}$			
		Bonnet Material: ASME SA 352 LCB		
	S_R = Radial "Flange" Stress, psi	Sm at 500 F = 18,900 psi		
	S_T = Tangential "Flange" stress, psi			
5. Bonnet bolting area requirements	Total cross-sectional area shall exceed the greater of:		$Am_1 (>Am_2)$=7.399 in^2	A_b (actual area)=1.34 Am (required min)

TABLE 3.9-2G: MAIN STEAM SAFETY/RELIEF VALVES (I) ASME CODE, SECTION III, JULY 1974 (CONTINUED)

Topic	Method of Analysis	Dikkers Analysis	Allowable Value	Calculated
	$Am_1 = \frac{Wm_1}{Sb}$, or $Am_2 = \frac{Wm_2}{Sa}$	$Am_1 = \frac{Wm_1}{Sb}$ $Am_2 = \frac{Wm_2}{Sa}$	where Am (required minimum) is the greater of Am_1 and Am_2; and Ab (actual bolt area) must exceed Am	
	Where Am_1 = Total required bolt (stud) area for operating condition. Am_2 = Total required bolt (stud) area for gasket seating.	Body to Bonnet Bolting Material: ASME SA 193 Gr B7		
6. Disc	The disc stress is calculated based on treating the disc as a flat circular plate, edges supported, uniform load over area with radius r_o; see Bach's Formula on page 414 of reference h. From the reference, $t = 1.2 \frac{w\left[1-\frac{2r_o}{3R}\right]}{S}$ W is based on p-1375 psi under the disc.	W = 27,430 lb r_o = 0.785 inch R = 2.48 inches Disc Material: ASME SA 351 CF3A	t (minimum allowable) = 1.067 inches	Actual t_{min} = 1.068 inches = 1.0009 (required minimum)

TABLE 3.9-2G: MAIN STEAM SAFETY/RELIEF VALVES (I) ASME CODE, SECTION III, JULY 1974 (CONTINUED)

Topic	Method of Analysis	Dikkers Analysis	Allowable Value	Calculated
		Temperature: 585 F Sm (585 F) = 18,235 psi Allowable stress is 1.5 Sm. This is the value "S" in the above formula. [1.5Sm = 27,353 psi]		
7. Seismic Capability	Stress analysis uses F sub (vertical) = (mass of valve) x (4.5 g), and F sub (horizontal) = (mass of valve) x (6.5 g), with 800,000 in-lb and 300,000 in-lb applied at the inlet and outlet, respectively. The actual capability was verified by testing (with the moments concurrently applied), and exceeds these values.			

TABLE 3.9-2H: MAIN STEAM ISOLATION VALVE DESIGN OF PRESSURE RETAINING PARTS - ASME B&PV CODE SECTION III 1974

Item Number	Component/Load Type/Stress Type	Design Procedure	Allowable Value	Design/Calculated Value	Ratio (Calculated/ Allowable)
1.0	Body and Bonnet				
1.1	Loads:				
	Design Pressure	GE System Specification	1375 psi	1375 psi	N/A
	Design Temperature	GE System Specification	586 F	586 F	N/A
	Pipe Reaction Loads	ASME Section III NB - 3545-2 (b) (1)	N/A	N/A	N/A
1.2	Pressure Rating	Table NB 3542.1-2	P_R = 575 lb	P_R = 575 lb	N/A
1.3	Minimum Wall Thickness	Paragraph NB 3542	$t_m \geq$ 1.889 in	t_m = 1.94 in	N/A
1.4	Primary Membrane Stress	Paragraph NB-3545.1 (500 F)	P_m = 19,400 psi	P_m = 11,123 psi	0.57
1.5	Secondary Stress Due to Pipe Reaction	Paragraph NB 3545.2 (b)(1)	P_{ed}, P_{eb}, and P_{et} $\leq 1.5S_m$ (500 F) $1.5S_m$ = 29,100 psi	P_{ed} = 6,010 psi P_{ed} = 13,084 psi P_{et} = 12,727 psi	0.21 0.45 0.44
1.6	Primary Plus Secondary Stress Due to Internal Pressure	Paragraph NB-3545.2(1a)	N/A	Q_b = 26,024 psi	N/A

TABLE 3.9-2H: MAIN STEAM ISOLATION VALVE DESIGN OF PRESSURE RETAINING PARTS - ASME B&PV CODE SECTION III 1974

Item Number	Component/Load Type/Stress Type	Design Procedure	Allowable Value	Design/Calculated Value	Ratio (Calculated/ Allowable)
1.7	Thermal Secondary	Paragraph NB-3545.2(c)	N/A	$^{Q}T_2$= 601 psi	N/A
1.8	Range of Primary Plus Secondary Stress at Crotch Region	Paragraph NB-3545.2	S_n = 58,200 psi	S_n = 33,236 psi	0.57
1.9	Body Shape Rule	Paragraph NB-3544			
	- Radius at Crotch	Paragraph NB-3544.1(a)	$r_2 \geq 0.581$ in	r_2 = 1.125 in	N/A
	- Corner Radius	Paragraph NB-3544.1(b)	$r_4 \leq 0.125$ in	r_4 = 1.0 in	
	- Longitudinal Curvature	Paragraph NB-3544.6	> 0.052 1/in	= 0.123 1/in	
	- No Flat Walls	Paragraph NB-3544.7	N/A	N/A	N/A
	- Minimum Wall at Weld Ends	Paragraph NB-3544.8	> 1.492 in	= 1.636 in	
1.10	Cyclic Requirement for Fatigue Analysis	Paragraph NB-3545.3	$N_a \geq 2000$ Cycles	N_a = 36,000 Cycles	N/A
1.11	Cumulative Usage Factor Requirements for Fatigue Analysis	Paragraph NB-3550	$I_t \leq 1.0$	I_t = 0.0143	0.01
2.0	Body Flange/Bonnet				

TABLE 3.9-2H: MAIN STEAM ISOLATION VALVE DESIGN OF PRESSURE RETAINING PARTS - ASME B&PV CODE SECTION III 1974

Item Number	Component/Load Type/Stress Type	Design Procedure	Allowable Value	Design/Calculated Value	Ratio (Calculated/ Allowable)
2.1	Loads: (1) Design Pressure (2) Design Temperature (3) External Moments Due To Dynamic Loads That Include SSE Accelerations (4) Spring Force				
2.2	Body Flange Stresses	Paragraph NB-3647.1	N/A	P_{FD} = 1.756 psig	N/A
	Longitudinal Hub Stress		$S_H \leq 1.5S_m$ = 26,700 psi	S_H = 21,499 psi	0.80
	Radial Flange Stress		$S_R \leq 1.5S_m$ = 26,700 psi	S_R = 14,439 psi	0.54
	Tangential Flange		$S_T \leq 1.5S_m$ = 26,700 psi	S_T = 7227 psi	0.27
2.3	Bonnet Thickness	Paragraph NB-3646 Corrosion Allowance	$t_m \geq$ 5.7397 in = 0.12 in	t_m = 5.938 in	N/A
2.4	Bonnet Reinforcement	Paragraph NB-3646 (e)	Area 7.902 in^2	Area = 8.205 in^2	
3.0	Bonnet to Body	Appendix XI	$A_b \geq$ 41.62 in^3	A_b = 61.614 in^2	1.48

TABLE 3.9-2H: MAIN STEAM ISOLATION VALVE DESIGN OF PRESSURE RETAINING PARTS - ASME B&PV CODE SECTION III 1974

Item Number	Component/Load Type/Stress Type	Design Procedure	Allowable Value	Design/Calculated Value	Ratio (Calculated/ Allowable)
	Bolting Loads:				
	(1) Design Pressure				
	(2) Design Temperature				
	(3) External Moments Due To Dynamic Loads Which Include SSE Accelerations				
	(4) Actuator Operational Loads				
	(5) Spring Force				
4.0	Valve Poppet				
4.1	Loads:				
	(1) In-Line Pressure Loads				
4.2	Maximum Stress	Roark's Formulae for Stress and Strain (reference i), Cases 13, 14, 21 and 22.	$S_t \leq 17,800$ psi $S_s \leq 10,680$ psi	$S_t = 17,277$ psi $S_s = 3,731$ psi	0.97 0.35
5.0	Valve Stem*				
5.1	Loads:				

TABLE 3.9-2H: MAIN STEAM ISOLATION VALVE DESIGN OF PRESSURE RETAINING PARTS - ASME B&PV CODE SECTION III 1974

Item Number	Component/Load Type/Stress Type	Design Procedure	Allowable Value	Design/Calculated Value	Ratio (Calculated/ Allowable)
	(1) Axial Loads				
5.2	Undercut Thread Stress	Industry Standards	$S_t \leq 92,800$ psi	$S_t = 18,525$ psi	0.20
5.3	Thread Shear Stress	Values Indicated Are for Valves in Closed Position.	$S_s \leq 26,280$ psi	$S_s = 7,177$ psi	0.27
5.4	Buckling Force		F ≤ 63,245 lb	F = 46,951 lb	0.74

*The valve stem is not actually an ASME Code part. Per Subarticle NB-3546.3(a) the valve stem is considered to be a "significantly stressed part whose failure could lead to gross violation of the pressure retaining boundary". Therefore it was evaluated in the valve design stress report and the results from the design report are included in this table.

Note: (1) The structural integrity of the main stream isolation valve was evaluated for EPU conditions at 4408 MWt and it was concluded the valves will continue to meet the applicable structural acceptance criteria to which they were originally designed (Ref. 15).

TABLE 3.9-2I: RECIRCULATION PUMP CASE SUMMARY OF LOAD CLASSIFICATION WITH STRESS LOCATIONS AND LIMIT CRITERIA

Loading Condition ASME Sec. III	Load Combination: Pressure	Load Combination: Mechanical Loads	Criteria (ASME Sec. III) NB-3220	Location	Highest Calc. Stress/ Usage Fact.	Allowable Stress	Ratio (Act./) (All.)
Design (NB-3112)	Design Pressure =1650 psig	1. OBE (conservatively included) 2. Pump Thrust 3. Dead weight 4. Nozzle Loads 5. Gasket Seating Load 6. Lug Loads	Fig. NB-3221-1 P_m 1.0 S_m $P_L + P_b$ 1.5 S_m	Upper Flange	22,451 psi	28,800	0.78
Design (NB-3441.3)	Design Pressure =1650 psig		NB 3444.3	Casing	Minimum Wall Thickness =2.91 in.	N/A	N/A
Normal (NB-3113.1) and Upset (NB-3113.2)	Operating Pressure =1313 psig	1. Dead weight 2. Nozzle Loads 3. Thermal Transient 4. OBE for upset only 5. Gasket seating 6. Lug loads	Fig. NB-3222-1 $P_L + P_b + P_e + Q$ 3.0 S_m P_e 3.0 S_m Elastic-Plastic Analysis NB-3228.3(1)	Upper Case Flange	57,947psi(1)	57,675psi(1)	1.00 All six criteria of NB-3228 are satisfied

TABLE 3.9-2I: RECIRCULATION PUMP CASE SUMMARY OF LOAD CLASSIFICATION WITH STRESS LOCATIONS AND LIMIT CRITERIA (CONTINUED)

Loading Condition ASME Sec. III	Load Combination: Pressure	Load Combination: Mechanical Loads	Criteria (ASME Sec. III) NB-3220	Location	Highest Calc. Stress/ Usage Fact.	Allowable Stress	Ratio (Act./) (All.)
Emergency (NB-3113.3)	Most Severe Emergency Pressure (1630 psig)	1. Dead weight 2. Nozzle Loads 3. Pump Thrust 4. Gasket Seating Load 5. OBE (conservatively included) 6. Lug loads	Fig. NB-3224-1 P_m(1.2 S_m or S_y) P_L (1.8 S_m or 1.5 S_y) P + P_L (1.8 S_m or 1.5 S_y)	Upper Flange	Loading is less severe than design loading and allowable stresses are higher for emergency condition. The stresses are acceptable.		
Faulted (NB-3113.4)	Most Severe Faulted Pressure (1313 psig)	1. Dead weight 2. Nozzle Loads 3. SSE 4. Pump Thrust 5. Gasket Seating Load 6. Lug Loads	Table F-1322.2-1 P_m 1.5 S_m or 0.7 S_y P_m = Lesser of 2.4 S_m or 0.7 S_u	Case Bolting	41,398 psi	52,000 psi	0.795
				Motor stand plate/cover	14,786 psi	44,600 psi	0.351

TABLE 3.9-2J: REACTOR RECIRCULATION SYSTEM GATE VALVES
(i) SUCTION

Paragraph No.	Component/ Load/Stress Type	Design Procedure	Allowable Limit	Design/ Calculated/Value	Ratio Calculated Allowed
1.0	Body and Bonnet				
1.1	Loads:				
	Design Pressure	System Requirement	1250 psi	1250 psi	N/A
	Design Temperature	System Requirement	575 F	575 F	N/A
1.2	Pressure Rating, psi	ASME Section III(1), Figure NB-3545.1-2	P_r = 735 psi	P_r = 735 psi	N/A
1.3	Minimum Wall Thickness, inches	ASME Section III(1), Paragraph NB-3542	t_{min}=1.931 inch	t_{min}=1.931 in	N/A
1.4	Primary Membrane Stress, psi	ASME Section III(1), Paragraph NB-3545.1	$P_m \leq S_m$(500 F) = 19,600 psi	P_m = 6327 psi	0.32
1.5	Secondary Stress Due to Pipe Reaction	ASME Section III(1), Paragraph NB-3545.2 (b)(i)	P_e = Greatest Value of P_{ed} P_{eb} and P_{et} $\leq 1.5S_m$ (500 F) = (1.5)(19,600) = 29,400 psi	P_{ed} = 6300 psi P_{eb} = 13,894 psi P_{et} = 13,894 psi P_e =P_{eb} = 13,894 psi	0.47

NOTE: (1) ASME Section III, 1974 Edition
(2) Valve differential pressure is 50 psid
(3) Maximum expected differential pressure is 34 psid

TABLE 3.9-2J: REACTOR RECIRCULATION SYSTEM GATE VALVES (CONTINUED)
(i) SUCTION

Paragraph No.	Component/ Load/Stress Type	Design Procedure	Allowable Limit	Design/ Calculated/Value	Ratio Calculated Allowed
1.6	Primary Plus Secondary Stress Due To Internal Pressure	ASME Section III(1), Paragraph NB-3545.2 (a)(1)	See Paragraph 1.8	Q_p = 20,632 psi	N/A
1.7	Thermal Secondary Stress	ASME Section III(1), Paragraph NB-3545.2(c)	See Paragraph 1.8	Q_T = 5004 psi	N/A
1.8	Range of Primary Plus Secondary Stress at Crotch Region	ASME Section III(1), Paragraph NB-3545.2	$S_n \leq 3\ S_m$ (500 F) =3(19,600) = 58,800	$S_n = Q_p + P_e + 2Q_T$ = 29,940	
1.9	Cycle Requirements for Fatigue Analysis	ASME Section III(1), Paragraph NB-3545.3	$N_a \geq$ 2000 Cycles	$N_b = 1x10^6$ Cycles	N/A
1.10	Usage Factor Requirements for Fatigue Analysis	ASME Section III(1), Paragraph NB-3550	$I_t \leq 1.0$	I_t = 0.0016	0.0016

TABLE 3.9-2J: REACTOR RECIRCULATION SYSTEM GATE VALVES (CONTINUED)
(i) SUCTION

Paragraph No.	Component/ Load/Stress Type	Design Procedure	Allowable Limit	Design/ Calculated/Value	Ratio Calculated Allowed
2.0	Body to Bonnet Bolting				
2.1	Loads: 1) Design Pressure and 2) Temp., 3) Gasket Loads, 4) Stem Operational Load, 5) Seismic Load (Safe Shutdown Earthquake)	---	---	---	
2.2	Bolt Area	ASME Section III(1), Paragraph NB-3647.1	A_b = 29.73 in²	A_b = 29.73 in²	
			S_b = 28,675 psi	S_b = 28,675 psi	1.00
2.3	Body Flange Stresses	ASME Section III(1), Paragraph NB-3647.1	---	---	---
2.3.1	(1) Operating Condition		$S_h \leq 1.5\ S_m$(575 F) = 28,837 psi	S_h = 13,362 psi	0.46
			$S_r \leq 1.5\ S_m$(575 F) = 28,837 psi	S_r = 12,928 psi	0.45
			$S_t \leq 1.5\ S_m$(575 F) = 28,837 psi	S_t = 3,925 psi	0.14

TABLE 3.9-2J: REACTOR RECIRCULATION SYSTEM GATE VALVES (CONTINUED)
(i) SUCTION

Paragraph No.	Component/ Load/Stress Type	Design Procedure	Allowable Limit	Design/ Calculated/Value	Ratio Calculated Allowed
2.3.2	Gasket Seating Condition	ASME Section III(1), Paragraph NB-3647.1	$S_h \leq 1.5\ S_m(100\ F)$ = 30,000 psi	S_h = 16,319 psi	0.55
			$S_r \leq 1.5\ S_m(100\ F)$ = 30,000 psi	S_r = 16,830 psi	0.56
			$S_t \leq 1.5\ S_m(100\ F)$ = 30,000 psi	S_t = 5,108 psi	0.17
2.4	Bonnet Flange Stresses	ASME Section III(1), Paragraph NB-3647.1	---	---	---
2.4.1	Operating Condition	ASME Section III(1), Paragraph NB-3647.1	$S_h \leq 1.5\ S_m(575\ F)$ = 28,838 psi	S_h = 16,720 psi	0.58
			$S_r \leq 1.5\ S_m(575\ F)$ = 28,838 psi	S_r = 10,621 psi	0.37
			$S_t \leq 1.5\ S_m(575\ F)$ = 28,838 psi	S_t = 4,909 psi	0.17

TABLE 3.9-2J: REACTOR RECIRCULATION SYSTEM GATE VALVES (CONTINUED)
(i) SUCTION

Paragraph No.	Component/ Load/Stress Type	Design Procedure	Allowable Limit	Design/ Calculated/Value	Ratio Calculated Allowed
2.4.2	Gasket Seating Condition	ASME Section III(1), Paragraph NB-3647.1	$S_h \leq 1.5\ S_m$ (100 F) = 30,000 psi	S_h = 20,986 psi	0.7
			$S_r \leq 1.5\ S_m$(100 F) = 30,000 psi	S_r = 14,064 psi	0.47
			$S_t \leq 1.5\ S_m$(100 F) = 30,000 psi	S_t = 6,503 psi	0.22
3.0	Stresses In Stem				
3.1	Loads: 1) Operator Thrust and 2) Torque	---	---	---	---
3.2	Stem Thrust Stress	Calculate Stress Due To Operator Thrust in Critical Cross Section	$S_t \leq S_m$ = 42,275 psi	S_t = 10,500 psi	0.25
3.3	Stem Torque Stress	Calculate Shear Stress Due To Operator Torque In Critical Cross Section	$S_m \leq 0.6\ S_m$ = 25,365 psi	S_m = 4,800 psi	0.19

TABLE 3.9-2J: REACTOR RECIRCULATION SYSTEM GATE VALVES (CONTINUED)
(i) SUCTION

Paragraph No.	Component/ Load/Stress Type	Design Procedure	Allowable Limit	Design/ Calculated/Value	Ratio Calculated Allowed
3.4	Buckling on Stem	Calculate Slenderness Ratio. If Greater Than 30, Calculate Allowable Load From Rankine's Formula Using Safety Factor of 4 (see reference i).	Max. Allowable Load = 33,638 lb	Slenderness Ratio = 115 Actual Load on Stem = 24,549 lb Therefore, No Buckling	0.73
4.0	Disc Analysis				
4.1	Loads: Maximum Differential Pressure(2)	---	---	---	---
4.2	Maximum Stress in the Disc	ASME Section III(1), Paragraph NB-3215 and ASME Section III(1), Paragraph NB-3221.3	$S_{max} \leq 1.5S_m$(575 F) = 27,487 psi	Max Stress = 20,225 psi	0.74

TABLE 3.9-2J: REACTOR RECIRCULATION SYSTEM GATE VALVES (CONTINUED)
(i) SUCTION

Paragraph No.	Component/ Load/Stress Type	Design Procedure	Allowable Limit	Design/ Calculated/Value	Ratio Calculated Allowed
5.0	Yoke and Yoke Connections				
5.1	Loads: Stem Operational Loads	Calculate Stresses in the Yoke and Yoke Connections to Acceptable Structural Analysis Methods	---	---	---
5.2	Tensile Stress In Yoke Leg Bolts	---	$S_{max} \leq 2S_m(100\ F)$ = 70,000 psi	S_{max} = 32,224 psi	0.46
5.3	Bending Stress of Yoke Legs	---	$S_b \leq 1.5\ S_m(185\ F)$ = 33,165 psi	S_b = 18,743 psi	0.57

TABLE 3.9-2J: REACTOR RECIRCULATION SYSTEM GATE VALVES
(II) DISCHARGE (24″)

Paragraph No.	Component/ Load/Stress Type	Design Procedure	Allowable Limit	Design/ Calculated/Value	Ratio Calculated Allowed
1.0	Body and Bonnet				
1.1	Loads:				
	Design Pressure	System Requirement	1650 psi	N/A	N/A
	Design Temperature	System Requirement	575 F		
	Pipe Reaction	Not Specified			
	Thermal Effects	Not Specified			
1.2	Pressure Rating, psi	ASME Section III(1), Figure NB-3545.1-2	P_r = 735 psi	P_r = 735 psi	N/A
1.3	Minimum Wall Thickness, inches	ASME Section III(1), Paragraph NB-3542	t_{min}=1.931 inch	t_{min}=1.931 in	N/A
1.4	Primary Membrane Stress, psi	ASME Section III(1), Paragraph NB-3545.1	$P_m \leq S_m$(500 F) = 19,600 psi	P_m = 6327 psi	0.32
1.5	Secondary Stress Due to Pipe Reaction	ASME Section III(1), Paragraph NB-3545.2	P_e = Greatest Value of P_{ed} P_{eb} and P_{et} $\leq 1.5S_m$ (500 F) = (1.5)(19,600) = 29,400 psi	P_{ed} = 6300 psi P_{eb} = 13,894 psi P_{et} = 13,894 psi $P_e = P_{eb}$ = 13,894	0.47

NOTES: (1) ASME Section III, 1974 Edition
(2) Valve differential pressure is 50 psid
(3) Maximum expected differential pressure is 34 psid

TABLE 3.9-2J: REACTOR RECIRCULATION SYSTEM GATE VALVES (CONTINUED)
(II) DISCHARGE (24″)

Paragraph No.	Component/ Load/Stress Type	Design Procedure	Allowable Limit	Design/ Calculated/Value	Ratio Calculated Allowed
1.6	Primary Plus Secondary to Stress Due to Internal Pressure	ASME Section III(1), Paragraph NB-3545.2 (a)(1)	See Paragraph 1.8	Q_p = 20,632 psi	N/A
1.7	Thermal Secondary Stress	ASME Section III(1), Paragraph NB-3545.2 (c)	See Paragraph 1.8	Q_T = 5004 psi	N/A
1.8	Range of Primary Plus Secondary Stress at Crotch Region	ASME Section III(1), Paragraph NB-3545.2	$S_n \geq 3\ S_m$ (500 F) = 3(19,600) = 58,800	$S_n = Q_p + P_e + 2Q_T$ = 29,940	
1.9	Cycle Requirements for Fatigue Analysis	ASME Section III(1), Paragraph NB-3545.3	$N_a \geq$ 2000 Cycles	$N_b = 1 \times 10^6$ Cycles	N/A
1.10	Usage Factor Requirements for Fatigue Analysis	ASME Section III(1), Paragraph NB-3550	$I_t \leq 1.0$	$I_t \leq 0.0016$	0.0016
2.0	Body to Bonnet Bolting				
2.1	Loads: 1) Design Pressure and 2) Temp., 3) Gasket Loads, 4) Stem Operational Load (Safe Shutdown Earthquake)		---	---	---

TABLE 3.9-2J: REACTOR RECIRCULATION SYSTEM GATE VALVES (CONTINUED)
(II) DISCHARGE (24″)

Paragraph No.	Component/ Load/Stress Type	Design Procedure	Allowable Limit	Design/ Calculated/Value	Ratio Calculated Allowed
2.2	Bolt Area	ASME Section III(1), Paragraph NB-3647.1	A_b = 29.73 in^2	A_b = 29.73 in^2	
			S_b = 28,675 psi	S_b = 28,675 psi	1.00
2.3	Body Flange Stresses	ASME Section III(1), Paragraph NB-3647.1	---	---	---
2.3.1	(1) Operating Condition		$S_h \leq 1.5\ S_m$(575 F)	S_h = 13,362 psi	0.46
			= 28,837 psi		
			$S_r \leq 1.5$ Sm(575 F)	S_r = 12,928 psi	0.45
			= 28,837 psi		
			$S_t \leq 1.5$ Sm(575 F)	S_t = 3,925 psi	0.14
			= 28,837 psi		
2.3.2	Gasket Seating Condition	ASME Section III(1), Paragraph NB-3647.1	$S_h \leq 1.5$ Sm(100 F)	S_h = 16,319 psi	0.55
			= 30,000 psi		
			$S_r \leq 1.5$ Sm(100 F)	S_r = 16,830 psi	0.56
			= 30,000 psi		
			$S_t \leq 1.5$ Sm(100 F)	S_t = 5,108 psi	0.17
			= 30,000 psi		
2.4	Bonnet Flange Stresses	ASME Section III(1), Paragraph NB-3647.1	---	---	---

TABLE 3.9-2J: REACTOR RECIRCULATION SYSTEM GATE VALVES (CONTINUED)
(II) DISCHARGE (24″)

Paragraph No.	Component/ Load/Stress Type	Design Procedure	Allowable Limit	Design/ Calculated/Value	Ratio Calculated Allowed
2.4.1	Operating Condition	ASME Section III(1), Paragraph NB-3647.1	$S_h \leq 1.5$ Sm(575 F) = 28,838 psi	Sh = 16,720 psi	0.58
			Sr ≤ 1.5 Sm(575 F) = 28,838 psi	S_r = 10,621 psi	0.37
			$S_t \leq 1.5\ S_m$(575 F) = 28,838 psi	S_t = 4,909 psi	0.17
2.4.2	Gasket Seating Condition	ASME Section III(1), Paragraph NB-3647.1	$S_h \leq 1.5\ S_m$ (100 F) = 30,000 psi	S_h = 20,986 psi	0.70
			$S_r \leq 1.5\ S_m$(100 F)	S_r = 14,064 psi	0.47
			$S_t \leq 1.5$ Sm(100 F) = 30,000 psi	S_t = 6,503 psi	0.22
3.0	Stresses in Stem				
3.1	Loads: Operator Thrust and Torque	---	---	---	---
3.2	Stem Thrust Stress	Calculate Stress Due To Operator Thrust in Critical Cross Section	$S_t \leq S_m$ = 42,275 psi	S_t = 10,500 psi	0.25

TABLE 3.9-2J: REACTOR RECIRCULATION SYSTEM GATE VALVES (CONTINUED)
(II) DISCHARGE (24″)

Paragraph No.	Component/ Load/Stress Type	Design Procedure	Allowable Limit	Design/ Calculated/Value	Ratio Calculated Allowed
3.3	Stem Torque Stress	Calculate Shear Stress Due To Operator Torque In Critical Cross Section	$S_m \leq 0.6\ S_m$ = 25,365 psi	S_m = 4,800 psi	0.19
3.4		Calculate Slenderness Ratio. If Greater Than 30, Calculate Allowable Load From Rankine's Formula Using Safety Factor of 4 (see reference i).	Max. Allowable Load = 33,638 lb	Slenderness Ratio = 115 Actual Load on Stem = 24,549 lb Therefore, No Buckling	0.73
4.0	Disc Analysis				
4.1	Loads: Maximum Differential Pressure(2)	---	---	---	---
4.2	Maximum Stress in the Disc	ASME Section III(1), Paragraph NB-3215 and ASME Section III(1), Paragraph NB-3221.3	$S_{max} \leq$ 1.5Sm(575 F) = 27,487 psi	Max Stress = 20,225 psi	0.74
5.0	Yoke and Yoke Connections				

TABLE 3.9-2J: REACTOR RECIRCULATION SYSTEM GATE VALVES (CONTINUED)
(II) DISCHARGE (24″)

Paragraph No.	Component/ Load/Stress Type	Design Procedure	Allowable Limit	Design/ Calculated/Value	Ratio Calculated Allowed
5.1	Loads: Stem Operational Loads	Calculate Stresses in the Yoke and Yoke Connections to Acceptable Structural Analysis Methods	---	---	---
5.2	Tensile Stress In Yoke Leg Bolts		$S_{max} \leq 2S_m$(100 F) = 70,000 psi	S_{max} = 32,224 psi	0.46
5.3	Bending Stress of Yoke Legs	---	$S_b \leq 1.5\ S_m$(185 F) = 33,165 psi	S_b = 18,743 psi	0.57

TABLE 3.9-2K: ASME CODE CLASS III SAFETY/RELIEF VALVE DISCHARGE PIPING SYSTEM - HIGHEST STRESS SUMMARY MAIN STEAM LINE - "A"

Acceptance Criteria	Limiting Stress Type	Calculated Stress	Allowable Limits	Radio Actual/ Allowable	Loading	Identification(2) of Locations of Highest Stress Points - NODE Point Numbers
Based on ASME Code, Section III, ND-3600						
Design Condition:						
Eq. 8 ≤ 1.0 S_h	Sustained Loads	5018	15,000	0.335	1. Pressure 2. Weight 3. OBE	Joint 98
Service Levels A & B (Normal & Upset) Conditions						
Eq. 9 ≤ 1.2 S_h	Occasional Loads	6420	18,000	0.357	1. Pressure 2. Weight 3. OBE 4. TSVC 5. SRV (Acoustic wave & structural feedback)	Joint 104
Eq. 10 ≤ S_a	Thermal Expansion	4734	22,500	0.210	1. Pressure 2. Weight 3. OBE 4. TSVC 5. SRV (Acoustic wave & structural feedback)	Joint 104
Eq. 11 ≤ S_a + S_h		8864	37,500	0.236		

TABLE 3.9-2K: ASME CODE CLASS III SAFETY/RELIEF VALVE DISCHARGE PIPING SYSTEM - HIGHEST STRESS SUMMARY MAIN STEAM LINE - "A" (CONTINUED)

Acceptance Criteria	Limiting Stress Type	Calculated Stress	Allowable Limits	Radio Actual/ Allowable	Loading	Identification[2] of Locations of Highest Stress Points - NODE Point Numbers
Service Level C (Emergency) Condition:						
Eq. 9 ≤ 1.8 S_h	Primary Loads	6056	27,000	0.224	1. Pressure 2. Weight 3. Pressure (ATWS) 4. LOCA (SBA/IBA) 5. SRV (Acoustic wave & structural feedback	Joint 104
Service Level D (Faulted) Condition: ASME Code Case 160 B-1						
Eq. 9 ≤ 2.4 S_h	Primary loads	6160	36,000	0.171	1. Pressure 2. Weight 3. LOCA	Joint 104

Notes: (1) Appropriate loading combination of Table 3.9-2a were considered and the calculated stresses are reported for the governing loading combination.
(2) Refer to Figure 3.6-6a for the identification of Node point numbers.
(3) Safety Relief Valve Discharge piping were evaluated for TPO at 3898 MWt and concluded that all load limits mentioned above were met.
(4) The structural integrity of the safety/relief valve discharge piping was evaluated for EPU conditions at 4408 MWt and it was concluded the valves will continue to meet the applicable structural acceptance criteria to which they were originally designed (Ref. 15).

TABLE 3.9-2L: STANDBY LIQUID CONTROL PUMP

Criteria/ Loading	Component	Limiting Stress Type	Allowable Stress (psi)	Calculated Stress (psi)
Based on ASME B&PV Code Section III.				
Pressure boundary parts:				
1) Fluid cylinder - SA182-F304,		S_y = 30,000 psi		
2) Discharge valve stop and cylinder head extension SA 479-304,		S_y = 30,000 psi		
3) Discharge valve cover, cylinder head & stuffing box flange plate, SA 240-304,		S_y = 30,000 psi		
4) Stuffing box gland, 8A 564-630,		S_y = 115,000 psi		
5) Studs, SA 193B7,		S_y = 115,000 psi		
6) Dowel pins (2) alignment, SAE 4140,		S_A = 23,400 psi		
7) Studs, cylinder tie, SA 193-B7,		S_A = 25,000 psi		
8) Pump holddown bolts, SAE GR. 8,		T_A = 30,000 psi Q_A = 37,500 psi		

TABLE 3.9-2L: STANDBY LIQUID CONTROL PUMP

Criteria/ Loading		Component	Limiting Stress Type	Allowable Stress (psi)	Calculated Stress (psi)
9) Power frame, foot area, cast iron,	S_A = 15,000 psi				
10) Motor holddown bolts, SAE GR. 1,	T_A = 12,000 psi Q_A = 15,000 psi				
11) Motor frame foot area, cast iron,	S_A = 7,500 psi				
Normal & Upset Condition Loads:					
1. Design Pressure		1. Fluid cylinder	General Membrane	See Note (5)	
2. Design Pressure		2. Discharge valve stop	General Membrane	See Note (5)	See Note (3)
3. Operating Basis Earthquake		3. Cylinder head extension	General Membrane	See Note (5)	
4. Nozzle Loads (1)		4. Discharge valve cover	General Membrane	See Note (5)	
5. SRV Discharge		5. Cylinder head	General Membrane	See Note (5)	
		6. Stuffing box flange plate	General Membrane	See Note (5)	
		7. Stuffing box gland	General Membrane	See Note (5)	
		8. Cylinder head studs	Tensile	See Note (5)	
		9. Stuffing box studs	Tensile	See Note (5)	
Emergency Condition:					
1. Design Pressure		1. Fluid cylinder	General Membrane	See Note (5)	See Note (5)
2. Design Temperature		2. Discharge valve stop	General Membrane	See Note (5)	
3. Dead Weight		3. Cylinder head extension	General Membrane	See Note (5)	

TABLE 3.9-2L: STANDBY LIQUID CONTROL PUMP

Criteria/ Loading	Component	Limiting Stress Type	Allowable Stress (psi)	Calculated Stress (psi)
4. Thermal Expansion	4. Discharge valve cover	General Membrane	See Note (5)	
5. Nozzle Loads	5. Cylinder head	General Membrane	See Note (5)	
6. Safety/Relief Valve Discharge	6. Stuffing box flange plate	General Membrane	See Note (5)	
7. LOCA	7. Stuffing box gland	General Membrane	See Note (5)	
Faulted Condition:				
1. Design Pressure	1. Cylinder head studs	Tensile	25,000	See Note (5)
2. Design Temperature	2. Stuffing box studs	Tensile	25,000	See Note (5)
3. Nozzle Loads	3. Dowel pins(2)	Shear only (2)	23,400	See Note (5)
4. Safety/Relief Valve Discharge	4. Studs, cylinder tie	Tensile(2)	25,000	See Note (5)
5. LOCA	5. Pump holddown bolts	Shear	30,000	See Note (5)
6. Safe shutdown earthquake	6. Pump holddown bolts	Tensile	37,500	See Note (5)
	7. Power frame-foot area	Shear	15,000	See Note (5)
	8. Power frame-foot area	Tensile	15,000	See Note (5)
	9. Motor holddown bolts	Shear	30,000	See Note (5)
	10. Motor holddown bolts	Tensile	37,500	See Note (5)
	11. Motor frame-foot area	Shear	7,500	See Note (5)
	12. Motor frame-foot area	Tensile	7,500	See Note (5)

TABLE 3.9-2L: STANDBY LIQUID CONTROL PUMP

Criteria/ Loading	Component	Limiting Stress Type	Allowable Stress (psi)	Calculated Stress (psi)
Faulted Condition Dynamic Loads				
1. SRV	SLC Pump Assembly	Acceleration	1.75 g 1.75 g	horizontal 0.83g vertical 0.49g
2. SSE				
3. LOCA				

TABLE 3.9-2L: STANDBY LIQUID CONTROL PUMP

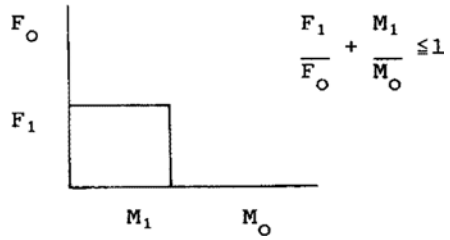
Criteria/ Loading	Component	Limiting Stress Type	Allowable Loads (lb, ft,-lb)	Calculated Loads (lb, ft,-lb)
Nozzle Load Definition: Unit Forces- lb s: Moments- Ft-lb Allowable combination of forces and moments are as follows  Where: F_1 = The largest absolute valve of the three actual external orthogonal forces (F_x, F_y, F_z) that may be imposed by the interface pipe, and, M = The largest absolute value of the three actual internal orthogonal moments (M_x, M_y, M_z) permitted from the pipe when they are combined simultaneously for a specific condition.				
Normal and Upset Condition Loads: 1. Design Pressure 2. Design Temperature Operating Basis 3. Earthquake 4. Nozzle Loads 5. SRV Discharge	F_o = Allowable value of F_1 when all moments are zero		SUCTION: F_o = 770 M_o = 490	See Note (6)

TABLE 3.9-2L: STANDBY LIQUID CONTROL PUMP

Criteria/ Loading	Component	Limiting Stress Type	Allowable Loads (lb, ft,-lb)	Calculated Loads (lb, ft,-lb)
			DISCHARGE:	
	M_o = Allowable value of M_1 when all forces are zero.		F_o = 370 M_o = 110	See Note (6)
Emergency or Faulted Condition Loads:			SUCTION:	
1. Design Pressure			F_o = 920 M_o = 590	See Note (6)
2. Design Temperature				
3. Nozzle Loads				
4. SRV Discharge			DISCHARGE:	
5. LOCA			F_o = 440 M_o = 130	See Note (6)
6. Safe Shutdown Earthquake				

Notes: (1) Nozzle loads produce shear loads only.

(2) Dowel pins take all shear.

(3) Calculated stresses for emergency or faulted condition are less than the allowable stresses for the normal and upset condition loads; therefore, the normal and upset condition is not evaluated.

(4) Operability: The sum of the plunges and rod assembly, pounds mass times 1.75, acceleration is much less than the thrust loads encountered during normal operating conditions. Therefore, the loads during the faulted condition have no significant effect on pump operability.

(5) Reference the latest revision of GGNS Document VPF 3579-GGNS-10-1-0 for the calculated stress values and allowables.

(6) Reference the latest revisions of Stress Problems 2048A and 2048B for the calculated loads.

TABLE 3.9-2M: STANDBY LIQUID CONTROL TANK

Component/Loading	Method of Analysis	Allowable Stress or Min. Thickness Reqd or Load			Actual Stress or Thickness or Load	
1. Shell Thickness						
Loads: Normal & Upset Design Pressure and Temperature	Brownell & Young "Process Equipment Design" $t = \frac{PR}{SE-0.6P}$	0.016 in.			0.25 in.	
Stress Limit	ASME Section III	18,300 psi			1203 psi	
2. Nozzle Loads						
Loads: Normal & Upset Design Pressure and Temperature	The maximum moments due to pipe reaction and maximum forces shall not exceed the allowable limits.	Fo (lb)	Mo (ft-lb)			
Overflow Nozzle		770	490	142	157	
Discharge Nozzle		770	490	20	46	
Loads: Faulted Dead Weight, Thermal Expansion and SSE Earthquake	The maximum moments due to pipe reaction and maximum forces shall not exceed the allowable limits.	Fo (lb)	Mo (ft-lb)			
Overflow Nozzle		925	590	163	187	
Discharge Nozzle		925	590	25	57	
3. Anchor Bolts	ASME Section III	25,000 psi			4221 psi	

TABLE 3.9-2M: STANDBY LIQUID CONTROL TANK

Component/Loading	Method of Analysis	Allowable Stress or Min. Thickness Reqd or Load	Actual Stress or Thickness or Load
4. Dynamic Loads SRV SSE LOCA	Equivalent Static	1.75g horizontal 1.75g vertical	0.645g horizontal 0.378g vertical

Note: Design assessment shows stresses resulting from additional loads are lower than originally calculated loads.

TABLE 3.9-2N: ECCS PUMPS
(I) RESIDUAL HEAT REMOVAL PUMP

Location	Loading Condition	Criteria	Calculated Stress (psi) or Actual Thickness (in.)	Allowable Stress (psi) or Min. Thickness (in.)	
Discharge Head Shell	FAULTED CONDITION Design Pressure Nozzle Loads Seismic Loads	ASME Boiler &Pressure Vessel Code, Section VIII, Division 1, Para. UG-27	1.00 in.	0.281 in.	|
Discharge Head Cover	Design Pressure	ASME Boiler &Pressure Vessel Code, Section VIII, Division 1, Para. UG-34, UG-39 & UG-40	2.875 in.	2.071 in.	|
Nozzle Shell Inter Section	FAULTED CONDITION Design Pressure Nozzle Loads Seismic Load	ASME Boiler & Pressure Vessel Code, Section VIII, Division 1, Para. UG-37	(Suction) 28,356 psi (Discharge) 20,728 psi	31,500 psi	|
Discharge Pipe	FAULTED CONDITION Design Pressure Nozzle Loads	ASME Boiler & Pressure Vessel Code, Section VIII, Division 1, Para. UG-27	0.562 in.	0.442 in.	|
Discharge Head Bolting	FAULTED CONDITION	Bolting Loads & Stresses per "Rules for Bolted Flange Connections" ASME Section VIII, App. II	37,342 psi	37,500 psi	|

TABLE 3.9-2N: ECCS PUMPS (CONTINUED)
(I) RESIDUAL HEAT REMOVAL PUMP

Location	Loading Condition	Criteria	Calculated Stress (psi) or Actual Thickness (in.)	Allowable Stress (psi) or Min. Thickness (in.)
Motor Bolting	FAULTED CONDITION Seismic Load	Bolting Loads & Stresses per "Rules for Bolted Flange Connections" ASME Section VIII, App. II	2,193 psi	25,000 psi

NOTES:
1. Operability demonstrated by analysis.
2. ECCS pumps are not affected by hydrodynamic loads because the Auxiliary Building and the Containment are on different basemats.
3. The structural integrity of the residual heat removal pump was evaluated for EPU conditions at 4408 MWt and it was concluded the pump design and IST Program requirements are not affected by EPU (Ref. 15).

TABLE 3.9-2N: ECCS PUMPS
(II) LOW PRESSURE CORE SPRAY PUMP

Location	Loading Condition	Criteria	Calculated Stress (psi) or Actual Thickness (in.)	Allowable Stress (psi) or Min. Thickness (in.)
Discharge Head Shell	FAULTED CONDITION Design Pressure Nozzle Loads Seismic Load	ASME Boiler & Pressure Vessel Code, Section VIII Division 1, Para. UG-27	1.25 in.	.21 in.
Discharge Head Cover	Design Pressure	ASME Boiler &Pressure Vessel Code, Section VIII Division 1, Para. UG-34, UG-39, &UG-40	3.0 in.	1.79 in.
Nozzle Shell Inter Section	FAULTED CONDITION Design Pressure Nozzle Loads Seismic Load	ASME Boiler & Pressure Vessel Code, Section VIII Division 1, Para. UG-37	(Suction) 24,999 psi (Discharge) 22,050 psi	31,500 psi
Discharge Pipe	FAULTED CONDITION Design Pressure Nozzle Loads	ASME Boiler &Pressure Vessel Code, Section VIII Division 1, Para. UG-27	.500 in.	.394 in.
Discharge Head Bolting	FAULTED CONDITION Design Pressure Nozzle Loads Seismic Load	Bolting Loads & Stresses per "Rules for Bolted Flange Connections" ASME Section VIII, App. II	34,961 psi	37,500 psi

TABLE 3.9-2N: ECCS PUMPS (CONTINUED)
(II) LOW PRESSURE CORE SPRAY PUMP

Location	Loading Condition	Criteria	Calculated Stress (psi) or Actual Thickness (in.)	Allowable Stress (psi) or Min. Thickness (in.)
Motor Bolting	FAULTED CONDITION Seismic Load	Bolting Loads & Stresses per "Rules for Bolted Flange Connections" ASME Section VIII, App. II	1610 psi	50,000 psi

NOTES: 1. Operability demonstrated by analysis.
2. ECCS pumps are not affected by hydrodynamic loads because the Auxiliary Building and the Containment are on different basemats.
3. The structural integrity of the low pressure core spray pump was evaluated for EPU conditions at 4408 MWt and it was concluded the pump design and IST Program requirements are not affected by EPU (Ref. 15).

TABLE 3.9-2N: ECCS PUMPS
(III) HIGH PRESSURE CORE SPRAY PUMP

Location	Loading Condition	Criteria	Calculated Stress (psi) or Actual Thickness (in.)	Allowable Stress (psi) or Min. Thickness (in.)
Discharge Head Shell	FAULTED CONDITION Nozzle Loads Seismic Load	ASME Boiler & Pressure Vessel Code, Section VIII Division 1, Para. UG-27	1.25 in.	.231 in.
Discharge Head Cover	Design Pressure	ASME Boiler & Pressure Vessel Code, Section VIII Division 1, Para. UG-34, UG-39, & UG-40	3.25 in.	2.119 in.
Nozzle Shell Inter Section	FAULTED CONDITION Design Pressure Nozzle Loads Seismic Load	ASME Boiler & Pressure Vessel Code, Section VIII Division 1, Para. UG-37	(Suction) 16,310 psi (Discharge) 18,160 psi	31,500 psi
Discharge Pipe	FAULTED CONDITION Design Pressure Nozzle Loads	ASME Boiler &Pressure Vessel Code, Section VIII Division 1, Para. UG-27	1.5 in.	.72 in.
Discharge Head Bolting	FAULTED CONDITION Design Pressure Nozzle Loads Seismic Load	Bolting Loads & Stresses per "Rules for Bolted Flange Connections" ASME Section VIII, App. II	34,417 psi	50,500 psi
Motor Bolting	FAULTED CONDITION Seismic Load	Bolting Loads & Stresses per "Rules for Bolted Flange Connections" ASME Section VIII, App. II	2,980 psi	37,500 psi

TABLE 3.9-2N: ECCS PUMPS (CONTINUED)
(III) HIGH PRESSURE CORE SPRAY PUMP

Location	Loading Condition	Criteria	Calculated Stress (psi) or Actual Thickness (in.)	Allowable Stress (psi) or Min. Thickness (in.)

NOTES: 1. Operability demonstrated by analysis.
2. ECCS pumps are not affected by hydrodynamic loads because the Auxiliary Building and the Containment are on different basemats.
3. The structural integrity of the low pressure core spray pump was evaluated of EPU conditions at 4408 MWt and it was concluded the pump design and IST Program requirements are not affected by EPU (Ref. 15).

TABLE 3.9-2O: RESIDUAL HEAT REMOVAL (RHR) HEAT EXCHANGER

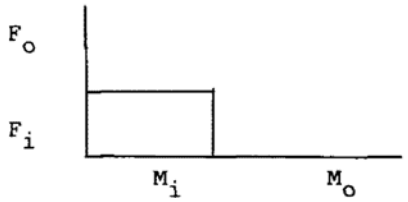
Loading/Component	Criteria/Location	Allowable Stress or Min. Thickness Reqd.	Calculated Stress or Thickness
1. Closure Bolting	Bolting Loads and Stresses calculated per "Rules for Bolted Flange Connections" ASME Section III, App. XI		
Loads: Normal			
Design Pressure and Temperature Design Gasket Load	a. Shell to Tube Sheet Bolts	25,000 psi	23,700 psi
	b. Channel Cover Bolts	25,000 psi	24,560 psi
2. Wall Thickness	Shell Side ASME Section III, Class 2 and TEMA, Class C		
Loads: Normal			
Design Pressure and Temperature	Tube Side ASME Section III, Class 3 and TEMA, Class C		
Stress limit per ASME Section III	a. Shell	0.954 in.	1.0 in.
	b. Shell Cover	0.942 in.	0.942 in. min.
	c. Channel Ring	0.982 in.	1.0 in.
	d. Tubes	0.0515 in.	0.054 in. min
	e. Channel Cover	8.82 in.	8.88 in.
	f. Tube Sheet	7.59 in.	7.62 in.

TABLE 3.9-20: RESIDUAL HEAT REMOVAL (RHR) HEAT EXCHANGER

Loading/Component	Criteria/Location	Allowable Stress or Min. Thickness Reqd.	Calculated Stress or Thickness
3. Nozzle Loads: Faulted	The maximum moments due to pipe reaction and the maximum forces shall not exceed the allowable limits.	See below (a) (b)	See below (c)
Design Pressure and Temperature Dead Weight, Thermal Expansion, Safe Shutdown Earthquake, SRV, LOCA	Primary Stress Smaller of 0.70 S or 2.4 S per ASME S&%tion III allowable.		

(a) The following expression relates the allowable combination of forces and moments:

$$\left|\frac{F_i}{F_o}\right| + \left|\frac{M_i}{M_o}\right| \leq 1$$



Where:

F_i = The largest of any of the three actual external orthogonal forces (Fx, Fy, and Fz)

M_i = The largest of any of the three actual external orthogonal moments (Mx, My, and Mz)

F_o = The allowable value of Fi when all moments are zero

M_o = The allowable value of Mi when all forces are zero

NOTE: One coordinate axis must be the nozzle centerline. Another coordinate axis must be parallel with the heat exchanger centerline, except where the heat exchanger centerline is parallel with the nozzle centerline. In this case, the coordinate axis must be orthogonal to the nozzle centerline and at O°-180° or 90°-270° azimuth.

TABLE 3.9-2O: RESIDUAL HEAT REMOVAL (RHR) HEAT EXCHANGER

(b) Allowable Limits (Faulted)

	N1	N2	N3	N4
F_x =	13,000 lb	13,000 lb	15,500 lb	15,500 lb
F_y =	13,000 lb	13,000 lb	15,500 lb	15,500 lb
F_z =	13,000 lb	13,000 lb	15,500 lb	15,500 lb
M_x =	46,000 ft-lb	46,000 ft-lb	60,000 ft-lb	60,000 ft-lb
M_y =	46,000 ft-lb	46,000 ft-lb	60,000 ft-lb	60,000 ft-lb
M_z =	46,000 ft-lb	46,000 ft-lb	60,000 ft-lb	60,000 ft-lb

(c) Actual Nozzle Loads (Faulted)

	N1 (HX-2A)	N2 (HX-1A)	N3 (HX-1A)	N4 (HX-2A)
F_i =	3,190 lb	3,814 lb	4,441 lb	7,426 lb
M_i =	9,531 ft-lb	25,616 ft-lb	31,936 ft-lb	26,240 ft-lb

TABLE 3.9-2P:

" Table 3.9-2P (Sheets 1 through 2)

TABLE 3.9-2Q: RCIC TURBINE

Criteria/Loading	Component	Limiting Stress Type	Allowable Stress (psi)	Calculated Stress (psi)
The highest stressed sections of the various components of the RCIC Turbine assembly are identified. Allowable stresses are based on ASME Code, Section III, for:				
Pressure Boundary Castings SA216-NCB: S = 14,000 psi				
Pressure Boundary Boltings, SA193-B7 S = 25,000 psi				
Alignment Dowel Pins: AISI4037, Rc28-35 τ_a = 61,000 psi				
Normal and Upset Condition Loads:	Castings: 1) Stop valve	General Membrane	14,000	See Note 1
1. Design Pressure	2) Governor valve	General Membrane	14,000	
2. Design Temperature	3) Turbine inlet	Local Bending	21,000	
3. Operating Basis Earthquake (Horizontal & Vertical)	4) Turbine case	Local Bending	21,000	
	Pressure containing bolts:	Tensile	25,000	
	Structure alignment pins:	Shear	61,100	
4. Inlet Nozzle Loads				
5. Exhaust Nozzle Loads				
Emergency or Faulted Condition:	Castings: 1. Stop valve	General Membrane	16,800	9,800
1. Design Pressure	2. Governor valve	General Membrane	16,800	13,200
2. Design Temperature	3. Turbine inlet	Local Bending	25,200	15,300
3. Safe Shutdown Earthquake (Horizontal & Vertical)	4. Turbine case	Local Bending	25,200	18,000
	Pressure containing bolts	Tensile	25,000	20,100
	Structure alignment pins	Shear	61,100	46,800
4. Inlet Nozzle Loads				
5. Exhaust Nozzle Loads				

TABLE 3.9-2Q: RCIC TURBINE (Continued)

Criteria/Loading	Component	Limiting Stress Type	Allowable Stress (psi)	Calculated Stress (psi)
Nozzle Load Definition: Turbine vendor has defined allowable nozzle loads for the turbine assembly. The above calculated stresses assume these allowable nozzle loads have been satisfied. Normal & Upset Condition Loads: 1. Design Pressure 2. Design Temperature 3. Weight of Structure 4. Thermal Expansion 5. Operating Basis Earthquake			ALLOWABLE LOAD CRITERIA (4) Inlet: $F = \frac{(3500 - M)}{3}$ Exhaust: $F = \frac{(7000 - M)}{3}$	CALCULATED LOADS (4) Inlet: F = 472 lb M = 526 ft-lb Exhaust: F = 590 lb M = 4031 ft-lb

TABLE 3.9-2Q: RCIC TURBINE (CONTINUED)

Criteria/Loading	Component	Limiting Stress Type	Allowable Stress (psi)	Calculated Stress (psi)
Emergency & Faulted Condition Loads:			Inlet:	F = 710 lb(3)
1. Design Pressure				M = 634 ft-lb
2. Design temperature			$F = \frac{(4200 - M)}{3}$	
3. Weight of structure				
4. Thermal expansion			Exhaust:	F = 696 lb
5. Safe shutdown earthquake				M = 4257 ft-lb
			$F = \frac{(8400 - M)}{3}$	

NOTES:

(1) Calculated stresses for the emergency or faulted condition are lower than the allowable stresses for the normal plus upset condition, therefore, the normal and upset condition is not evaluated.

(2) Operability: Analysis indicates that shaft deflection with faulted loads is 0.006 inch, which is fully acceptable; and maximum bearing Load with faulted condition is 80 percent of allowable. Furthermore, as indicated in subsection 3.9.2.2.1.6.9, the turbine assembly has been seismically qualified via dynamic testing, enveloping the response spectra defined. This qualification included demonstration of start-up and shutdown capabilities, as well as no load operability during seismic loading conditions.

(3) Because the calculated nozzle loads exceed the allowable loads, a refined analysis was made and the actual toads were within the capability of the nozzle.

(4) F = resultant force (lbs); M = resultant moment (ft-lbs).

TABLE 3.9-2R: RCIC PUMP

Criteria/Loading	Component	Limiting Stress type	Allowable Stress (psi)	Calculated Stress (psi)
Pressure boundary stress limits of the various components for the RCIC pump assembly are based on the ASME B&PV Code Section III, for pressure boundary parts @ 140 F. 1. Forged barrel, SA105 GR. II S_y = 36,000 psi 2. End cover plates, SA105 GR. II S_y = 36,000 psi 3. Nozzle connections, SA105 GR. II S_y = 36,000 psi 4. Aligning pin, SA105 GR. II S_y = 36,000 psi 5. Closure bolting, SA193-87 S_y = 105,000 psi 6. Pump holddown bolting, SA325 S_y = 77,000 psi 7. Taper pins, SA108 GR. B1112 S_y = 75,000 psi				
NORMAL & UPSET CONDITION LOADS:				
1. Design pressure 2. Design temperature 3. Operating basis earthquake 4. Suction nozzle loads 5. Discharge nozzle loads	1. Forged barrel	General membrane	17,500	
	2. End cover (Suction)	General membrane	17,500	
	3. End cover (Discharge)	General membrane	17,500	See Note 1
	4. Nozzle reinforcement	General membrane	17,500	
	5. Alignment pin	Shear	18,000	

TABLE 3.9-2R: RCIC PUMP (Continued)

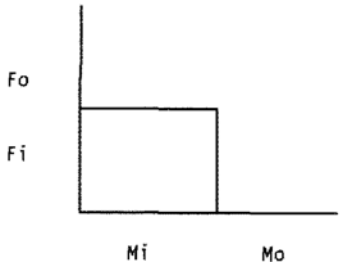
Criteria/Loading	Component	Limiting Stress type	Allowable Stress (psi)	Calculated Stress (psi)
	6. Closure bolting	Tensile	25,000	
	7. Taper pins	Shear	15,000	
	8. Pump holddown bolts	Tensile	40,000	
EMERGENCY OR FAULTED CONDITION LOADS:				
1. Design pressure	1. Forged barrel	General membrane	21,000	13,300
2. Design temperature	2. End cover (suction)	General membrane	21,000	10,760
3. Safe shutdown earthquake	3. End cover (discharge)	General membrane	21,000	15,900
4. Suction nozzle loads	4. Nozzle reinforcement at barrel	General membrane	21,000	13,180
5. Discharge nozzle loads				
	5. Alignment pin	Shear	18,000	9,378
	6. Closure bolting	Tensile	25,000	21,000
	7. Taper pin (baring housing)		18,000	2,280
	8. Pump holddown bolts	Tension	48,000	9,107

NOZZLE LOAD DEFINITION:

Units: Forces - lbs
Moments - ft-lbs

The allowable combinations of forces and moments are as follows:

TABLE 3.9-2R: RCIC PUMP (Continued)

Criteria/Loading	Component	Limiting Stress type	Allowable Stress (psi)	Calculated Stress (psi)
 $\frac{Fi}{Fo} + \frac{Mi}{Mo} \leq 1$ $\frac{Fi}{Fo} + \frac{Mi}{Mo} \leq 1$				

Where:

Fi = Largest absolute value of the three actual external orthogonal forces (Fx, Fy, Fz) that may be imposed by the interface pipe and,

Mi = Largest absolute value of the three actual external orthogonal moments (Mx, My, Mz) permitted from the interface pipe when they are combined simultaneously for a specific condition.

TABLE 3.9-2R: RCIC PUMP (CONTINUED)

Criteria/Loading	Component	Limiting Stress type	Allowable Loads (lb, ft-lb))	Calculated Loads (lb, ft-lb)
NORMAL & UPSET CONDITION LOADS:				
1. Design pressure		Fo = Allowable value of Fi when all moments are zero	SUCTION:	
2. Design temperature			Fo = 1940	Fi = 782
3. Weight of structure			Mo = 2460	Mi = 1421
4. Thermal expansion				
5. Operating basis earthquake		Mo = Allowable value of Mi when all forces are zero	DISCHARGE:	
			Fo = 3715	Fi = 1085
			Mo = 4330	Mi = 1470
EMERGENCY OR FAULTED CONDITION LOADS:				
1. Design pressure			SUCTION:	
2. Design temperature			Fo = 2325	Fi = 855
3. Weight of structure			Mo = 2950	Mi = 1432
4. Thermal expansion				
5. Safe shutdown earthquake			DISCHARGE:	
			Fo = 4450	Fi = 1132
			Mo = 5200	Mi = 1690

NOTES: (1) Calculated stresses for emergency or faulted condition are less than the allowable for normal plus upset condition. Therefore, the normal and upset condition is not evaluated.
(2) Operability: State analysis for emergency or faulted condition show that the maximum shaft deflection is 0.002 in. with 0.006 in. allowable, shaft stresses are 3080 psi with 25000 psi allowable, and bearing loads of, drive end 98 lbs, with 7670 lbs allowable and thrust end 765 lbs with 17600 lbs allowable.

TABLE 3.9-2S: REACTOR REFUELING AND SERVICING EQUIPMENT[6]
(I) FUEL STORAGE RACKS[5]

Criteria/Loading	ALLOWABLE STRESSES (7) Tension	Shear	Compression	Bending	Combined Flexure & Compression or Tension	Thermal
The following data on the physical properties[1] of the rack material are obtained from the ASME Code[2]:						
Young's Modulus - E = 28.3 x 10^6 psi						
Yield Strength - S_y = 25,000 psi						
Ultimate Strength - S_u = 71,000 psi						
Allowable Stress - S = 17,800 psi						
For Normal and Upset Conditions	XVII-2211(a)[3]	XVII-2212.1[3]	XVII-2213.1[3]	XVII-2214.4[3]	XVII-2215.1[3]	3Sm[4]
For Faulted Condition	F-1370[3]	F-1370[3]	F-1370[3]	F-1370[3]	F-1370[3]	

Notes: (1) Evaluated at 200 F, which is higher than pool temperature.
(2) ASME Code, Section III, Division I, 1980 edition up to and including winter 1981 addenda. This code was chosen because it provides the most consistent set of limits for the various stress types and various loading conditions.
(3) Reference Article of Appendices of Code listed in Note 2.
(4) Allowable stress intensity of the rack material at the maximum operating temperature.
(5) This sheet shows the allowable fuel storage rack stresses for Unit 1.
(6) Hydrodynamic loads have no impact on this component.
(7) For actual values of allowable stresses and calculated stresses, refer to Attachment 2 of AECM-85/0143.

TABLE 3.9-2S: REACTOR REFUELING AND SERVICING EQUIPMENT (1)
(ii) REFUELING PLATFORM

Acceptance Criteria	Loading	Primary Stress Type	Allowable Stress (psi)	Calculated Stress (psi)
The allowable axial plus bending loads stress are based on AISC Part 5 Section 1.5 for ASTM A36 Structural Steel				
For type S_u = 58,000 psi S_y = 36,000 psi				
For normal condition: S_{limit} = 0.66 S_y	For normal condition: 1. Normal operating load	Axial load plus bending	23,760	18,162
For upset condition: S_{limit} = 0.9 S_y	For upset condition: 1. Normal operating loads 2. Operating basis earthquake	Axial load plus bending	32,400	30,194
For faulted condition: S_{limit} = 0.7 S_u	For faulted condition: 1. Normal operating loads 2. Safe shutdown earthquake	Axial load	40,600	35,860

Note: Operability assurance is demonstrated by analysis.
(1) Hydrodynamic loads have no impact on the component.

TABLE 3.9-2S: REACTOR REFUELING AND SERVICING EQUIPMENT (1)
(III) FUEL PREPARATION MACHINE

Acceptance Criteria	Loading	Primary Stress Type	Allowable Stress (psi)	Calculated Stress (psi)
The allowable axial loads stress is based on ASME Code Section III, ASTM A157 for Type 302 Annealed S.S. S_y = 30,000 psi S_u = 75,000 psi S_m = 17,800 psi				
For normal condition: S_{limit} = 1.0 S_m	For normal condition: 1. Normal operating loads	Axial load	17,800	16,030 psi
For upset condition: S_{limit} = 0.8 S_y	For upset condition: 1. Normal operating loads 2. Operating basis earthquake	Axial load	24,000	16,148 psi
For faulted condition: S_{limit} = 1.2 S_y	For faulted condition: 1. Normal operating loads 2. Safe shutdown earthquake	Axial load	36,000	16,132 psi

Note: Operability assurance is demonstrated by analysis.
(1) Hydrodynamic loads have no impact on the component.

TABLE 3.9-2S: REACTOR REFUELING AND SERVICING EQUIPMENT
(IV) HORIZONTAL FUEL TRANSFER TUBE

Acceptance Criteria	Loading	Primary Stress Type	Allowable Stress (psi)	Calculated Stress (psi)
The allowable axial plus bending loads are based on ASME Code Section III, 1974 Edition, Subsection NC, for type SA 358 Gr. 304 SS S_u = 75,000 psi S_y = 30,000 psi				
For normal condition: S_{limit} = 0.6 S_y	For normal condition: 1. Normal operating loads	Axial load plus bending	17,950	2,626
For emergency condition: S_{limit} = 0.6 x 1.2 S_y	For emergency condition: 1. Normal operating loads 2. Operating basis earthquake 3. SRV and LOCA	Axial load plus bending	21,540	13,881
For faulted condition: S_{limit} = 0.6 x 2.4 S_y	For faulted condition: 1. Normal operating loads 2. Safe shutdown earthquake 3. SRV and LOCA	Axial load plus bending	43,080	31,402

Note: Operability assurance is demonstrated by analysis.

TABLE 3.9-2T: REACTOR PRESSURE VESSEL SUPPORT EQUIPMENT
(i) CRD HOUSING RESTRAINT BEAMS

Criteria	Loading	Limiting Stress Type	Allowable Stress (psi)	Calculated Stress (psi)
Based on AISC Specification for the design, fabrication, and erection of structural steel for buildings.				
F_y @ 50 F = 36,000 psi				
For normal and upset condition:	Normal and Upset Loads:			
$F_a = 0.60\ F_y$ (Tension)				
$F_b = 0.66\ F_y$ (Bending)	Negligible (2)			
$F_v = 0.40\ F_y$ (Shear)				
For emergency condition:	Emergency Loads:			
	Negligible (2)			
For faulted condition:	Faulted Loads			
$F_{a(limit)} = 1.5 \times 0.60 \times F_y$ (Tension)	1. Weight of structure	Top Chord	33,000	F_a = 14,000
$F_{b(limit)} = 1.5 \times 0.66 \times F_y$ (Bending)	2. Impact force from failing of CRD housing	Top Chord	33,000	F_b = 20,000
$F_{v(limit)} = 1.50 \times 0.40 \times F_y$ (Shear)		Bottom Chord	33,000	F_a = 1,700
			33,000	F_b = 21,000

NOTES: (1) F_y = Material yield strength.
(2) Deadweights, earthquake, and hydrodynamic loads are very small compared to impact force.

TABLE 3.9-2T: REACTOR PRESSURE VESSEL SUPPORT EQUIPMENT
(II) CRD HOUSING SUPPORTS - GRID STRUCTURE

Criteria	Loading	Limiting Stress Type	Allowable Stress (psi)	Calculated Stress (psi)
Based on AISC Specification for the design, fabrication and erection of structural steel for buildings.				
F_y @ 150 F = 46,000 psi				
For Normal and Upset condition:	Normal and Upset Loads:			
$F_a = 0.60\ F_y$ (Tension)	Negligible (2)			
$F_b = 0.66\ F_y$ (Bending)				
$F_v = 0.40\ F_y$ (Shear)				
For Emergency condition:	Emergency Loads:			
	Negligible (2)			
For Faulted condition:	Faulted Loads:			
$F_{a(limit)} = 1.5 \times 0.60 \times F_y$ (Tension)	1. Weight of structure	Bending	41,500	F_b = 40,500
$F_{b(limit)} = 1.5 \times 0.66 \times F_y$ (Bending)	2. Impact force from failure of CRD housing	Shear	27,500	F_v = 11,600
$F_{v(limit)} = 1.5 \times 0.40 \times F_y$ (Shear)				

NOTES: (1) F_y = Material yield strength.
(2) Deadweights, earthquake, and hydrodynamic loads are very small compared to impact force.

TABLE 3.9-2U: CONTROL ROD DRIVE
(I) MAIN FLANGE (2)

Criteria	Loading	Primary Stress Type	Allowable Stress (psi)	Calculated Stress (psi)
Allowable Primary Membrane Stress plus Bending Stress is based on ASME Boiler & Pressure Vessel Code, Section III for type F304 Stainless Steel @ 575 F S_a = 16,700 psi				
For normal and upset condition: $S_{allow} = 1.5 \times S_m$	For normal & upset condition: 1. Normal loads (1) 2. Scram with OBE 3. Scram with no buffer	General Membrane + Bending	25,000	5,813
For emergency condition: $S_{allow} = 1.8 \times S_m$	For emergency condition: 1. Normal loads (1) 2. Scram at emergency vessel pressure condition 3. Scram with accumulator at overpressure	General Membrane + Bending	30,000	4,300
For faulted condition: $S_{allow} = 3.6 \times S_m$	For faulted condition: 1. Normal loads (1) 2. Scram with SSE 3. Scram with stuck rod	General Membrane + Bending	60,000	7,294

NOTES: (1) Noraml loads include pressure, temperature, weight and mechanical loads.
(2) Hydrodynamic loads do not significantly contribute to the flange stresses.

TABLE 3.9-2U: CONTROL ROD DRIVE
(II) RING FLANGE (2)

Criteria	Loading	Primary Stress Type	Allowable Stress (psi)	Calculated Stress (psi)
Allowable Primary Membrane Stress plus Bending Stress is based on ASME Boiler & Pressure Vessel Code, Section III for type F304 Stainless Steel @ 250 F S_m = 20,000 psi				
For normal and upset condition: $S_{allow} = 1.5 \times S_m$	For normal & upset condition: 1. Normal loads (1) 2. Scram with OBE and no buffer	General Membrane + Bending	30,000	17,922
For emergency condition: $S_{allow} = 1.8 \times S_m$	For emergency condition: 1. Normal loads (1) 2. Scram with accumulator at over-pressure	General Membrane + Bending	36,000	1,838
For faulted condition: $S_{allow} = 3.6 \times S_m$	For faulted condition:	General Membrane + Bending	72,000	4,041

NOTES: (1) Normal loads include pressure, temperature, weight and mechanical loads.
(2) Hydrodynamic loads do not significantly contribute to the flange stresses.

TABLE 3.9-2U: CONTROL ROD DRIVE
(III) INDICATOR TUBE

Criteria	Loading	Primary Stress Type	Allowable Stress (psi)	Calculated Stress (psi)
Stress plus Bending Stress is based on ASME Boiler & Pressure Vessel Code, Section III for type 316 Stainless Steel @ 250 F S_m = 20000 psi				
For normal and upset condition: s_{allow} = 1.5 x S_m	For normal & upset condition: 1. Normal Loads (1) 2. Scram with OBE and no buffer	General Membrane + Bending	30,000	15,939
For emergency condition: S_{allow} = 1.8 x S_m	For emergency condition: 1. Normal Loads (1) 2. Failure of pressure regulation system 3. Scram with accumulator at over-pressure	General Membrane + Bending	36,000	23,826
For faulted condition: S_{allow} = 2.4 x S_m	For faulted condition: 1. Normal Loads (1) 2. Scram with SSE 3. Chugging and LOCA 4. SRV	General Membrane + Bending	48,000	11,778

NOTES: (1) Normal loads include pressure, temperature, weight and mechanical loads.
(2) For normal, upset and emergency conditions the calculated stresses due to hydrodynamic loads are less than the originally calculated stresses.

TABLE 3.9-2V: CONTROL ROD DRIVE HOUSING

Criteria	Loading	Primary Stress Type	Allowable Stress (psi)	Calculated Stress (psi)
Primary Stress Limit - The allowable nrimarv membrane stress is based 'on the ASME Boiler and Pressure Vessel Code, Section III, for Class I vessels, for type 304 stainless steel.				
For normal and upset condition:	Normal and upset condition loads:	Maximum membrane stress intensity occurs at the tube to the tube weld near the center of the housing for normal, upset, and emergency conditions.	16,600	10,400
S_{limit} = 1.0 S_m = 16,600 psi @ 575 F	1. Design pressure			
	2. Stuck rod scram loads			
	3. Operational basis earthquake, with housing lateral support installed			
S_{limit} = 1.5 S_m @ 575 F	4. Safety relief valve			
		Primary Membrane + Bending	24,900	19,600
For faulted conditions (1):	Faulted conditions loads:			
S_{limit} = 2.4 S_m	1. Design pressure			
	2. Stuck rod scram loads	Primary Membrane	39,840	10,700
S_{limit} = 3.6 S_m	3. Safe shutdown earthquake, with housing lateral support installed			
	4. Safety relief valve	Primary Membrane + Bending	59,760	20,600
	5. Chugging			
	6. Jet reaction			

TABLE 3.9-2V: CONTROL ROD DRIVE HOUSING

NOTE: (1) The calculated stresses under the hydrodynamic loads are lower than the original design calculations.
(2) The structural integrity of the control rod drive housing was evaluated for EPU conditions at 4408 MWt and it was concluded that component will continue to satisfy the applicable stress criteria following EPU implementation, including consideration of the effects of flow induced vibration and the effects of EPU implementation on the design loads and loading combinations (Ref. 15).

TABLE 3.9-2W: (SUPPLEMENT 1) JET PUMPS

Criteria	Loading Combinations	Stress Type	Allowable Stress (psi)	Calculated Stress (psi)
PRIMARY MEMBRANE PLUS BENDING STRESS BASED ON ASME CODE SECTION I11 304 S.S.				
FOR SERVICE LEVELS A & B (NORMAL AND UPSET) CONDITION: S_{limit} = 3. 0 S_m psi	Service level A & B (normal & upset) 1. Normal loads 2. Operating basis earthquake 3. SRV (all)	PRIMARY MEMBRANE PLUS SECONDARY MEMBRANE PLUS BENDING EXCLUDING THERMAL BENDING	50,700	22,636
FOR SERVICE LEVEL C (EMERGENCY) CONDITION: S_{limit} = 2.25 S_m psi	Service level C (Emergency) Loads 1. Normal loads 2. Operating basis earthquake 3. SRV (all)	PRIMARY MEMBRANE PLUS BENDING	38,025	13,874
FOR SERVICE LEVEL D (FAULTED) CONDITION: S_{limit} = 3.6 S_m psi	Service level D (faulted) loads 1. Normal loads 2. Safe shutdown 3. Annulus Pressurization 4. SRV (all) 5. Jet Reactions	PRIMARY MEMBRANE PLUS BENDING	61,884 (Note 3)	61,678

Note 1: The fatigue usage is less than 1.0.

Note 2: A mismatch of 15% was evaluated for long-term operation - (Ref. M-001.0-QlBl3DOO6-8.0-1-0)

TABLE 3.9-2W: (SUPPLEMENT 1) JET PUMPS

Criteria	Loading Combinations	Stress Type	Allowable Stress (psi)	Calculated Stress (psi)

Note 3: This allowable based on operating temperature versus design temperature.

Note 4: The structural integrity of the jet pump was evaluated for EPU conditions at 408 MWt and it was concluded the component will continue to satisfy the applicable stress criteria following EPU implementation, including consideration of the effects of flow induced vibration and the effects of EPU implementation on the design loads and loading combinations (Ref. 15).

TABLE 3.9-2X: FUEL POOL HEAT EXCHANGER

Component Loading	Criteria	Allowable Stress or Minimum Thickness Required	Calculated Stress or Thickness
1. Frame Plate & Pressure Plate			
Design Pressure and Temperature			
Pressure Plate (No holes)	ASME Section III Class 3, ND-3333	3.10 in	3.14 in
Pressure plate (with holes)	ASME Section III Class 3, ND-3333	3.51 in	4.12 in
2. Combined Loadings Including Local Nozzle Loading			
Normal Condition	Stress Limit per ASME Section III		
Frame Plate		17.5 ksi	7.6 ksi
Tightening Bolt		25.0 ksi	20.0 ksi
Upset Condition	Stress Limit per ASME Section III		
Frame Plate		19.3 ksi	9.2 ksi
Tightening Bolt		27.5 ksi	21.0 ksi
Faulted Condition	Stress Limit per ASME Section III		
Frame Plate		35.0 ksi	11.4 ksi
Tightening Bolt		50.0 ksi	22.3 ksi
3. Closure Bolting Loads			
Stud Bolt Required Area	ASME Section III, Appendix XI, XI-3220 and ND-3647		
Gasket Seating Condition		5.03 squin (Actual Area of Bolt)	2.04 squin
Normal Condition		5.03 squin (Actual Area of Bolt)	4.06 squin
Upset Condition		5.03 squin (Actual Area of Bolt)	5.15 squin
Fault Condition		5.03 squin (Actual Area of Bolt)	2.65 squin

TABLE 3.9-2X: FUEL POOL HEAT EXCHANGER (Continued)

Component Loading	Criteria	Allowable Stress or Minimum Thickness Required	Calculated Stress or Thickness
4. Structural Support			
Evaluation for Frame Plate and Pressure Plate Bolts	ASME Section III, Appendix XVII Subsection-2460	Interaction Ratio allowable	Computed Interaction Ratio
Normal Condition			
Plate bolts		1.0	0.51
Anchor Bolts		1.0	0.36
Upset Condition			
Plate bolts		1.0	0.97
Anchor Bolts		1.0	0.67
Faulted Condition			
Plate bolts		1.0	0.66
Anchor Bolts		1.0	0.45

TABLE 3.9-2X: FUEL POOL HEAT EXCHANGER (Continued)

Loading	Criteria	Allowable Nozzle Forces and Moments Force in lb., Moment in ft-lb	Actual Nozzle Loads
5. Nozzle Loads Design Pressure and Temperature Dead Weight, Thermal Expansion, Operational basis earthquake, Safe Shutdown Earthquake	The maximum moments due to pipe reaction and the maximum forces shall not exceed the allowable limits.	See Table below	See Table below

Heater Exchanger A, Nozzle S1, S2

HTX, A Nozzle S1 (PDS0166 Data Pt. 1255)

Condition	**Fx lb**	**Fy lb**	**Fz lb**	**Mx ft-lb**	**My ft-lb**	**Mz ft-lb**
$Allowable_{Normal}$	3118	3118	3118	8405	8405	8405
$Calculated_{Normal}$	1082	2805	559	1475	370	6914
$Allowable_{Upset}$	3898	3898	3898	10086	10086	10086
$Calculated_{Upset}$	1313	2937	782	1965	602	7505
$Allowable_{Faulted}$	4678	4678	4678	12608	12608	12608
$Calculated_{Faulted}$	1534	3048	998	2438	819	8061

HTX, A Nozzle S2 (PDS0166 Data Pt. 1490)

Condition	**Fx lb**	**Fy lb**	**Fz lb**	**Mx ft-lb**	**My ft-lb**	**Mz ft-lb**
$Allowable_{Normal}$	3118	3118	3118	8405	8405	8405
$Calculated_{Normal}$	427	2631	268	629	425	5651
$Allowable_{Upset}$	4678	4678	4678	12608	12608	12608

TABLE 3.9-2X: FUEL POOL HEAT EXCHANGER (Continued)

$Calculated_{Upset}$	629	2852	447	847	1217	6238

Heater Exchanger A, Nozzle S3 & S4

HTX, A Nozzle S3 (PDS0085 Data Pt. 1610)

Condition	**Fx lb**	**Fy lb**	**Fz lb**	**Mx ft-lb**	**My ft-lb**	**Mz ft-lb**
$Allowable_{Normal}$	3118	3118	3118	8405	8405	8405
$Calculated_{Normal}$	120	1310	146	402	454	3541
$Allowable_{Upset}$	3898	3898	3898	10086	10086	10086
$Calculated_{Upset}$	589	1793	432	1664	4794	5884
$Allowable_{Faulted}$	4678	4678	4678	12608	12608	12608
$Calculated_{Faulted}$	812	2068	502	2380	7270	7225

HTX, A Nozzle S4 (PDS0086 Data Pt. 1675)

Condition	Fx lb	Fy lb	Fz lb	Mx ft-lb	My ft-lb	Mz ft-lb
$Allowable_{Normal}$	3118	3118	3118	8405	8405	8405
$Calculated_{Normal}$	185	289	8	17	19	217
$Allowable_{Upset}$	3898	3898	3898	10086	10086	10086
$Calculated_{Upset}$	715	495	195	371	890	671
$Allowable_{Faulted}$	4678	4678	4678	12608	12608	12608
$Calculated_{Faulted}$	1039	520	282	527	1321	940

Heater Exchanger B, Nozzle S1, S2

HTX, B Nozzle S1 (PDS0166 Data Pt. 1375)

Condition	**Fx lb**	**Fy lb**	**Fz lb**	**Mx ft-lb**	**My ft-lb**	**Mz ft-lb**
$Allowable_{Normal}$	3118	3118	3118	8405	8405	8405

TABLE 3.9-2X: FUEL POOL HEAT EXCHANGER (Continued)

$Calculated_{Normal}$	386	4009	212	759	499	6926
$Allowable_{Upset}$	3898	3898	3898	10086	10086	10086
$Calculated_{Upset}$	548	4124	331	1261	976	7329
$Allowable_{Faulted}$	4678	4678	4678	12608	12608	12608
$Calculated_{Faulted}$	696	4193	440	1744	1419	7645

HTX, B Nozzle S2 (PDS0166 Data Pt. 1605)

Condition	**Fx lb**	**Fy lb**	**Fz lb**	**Mx ft-lb**	**My ft-lb**	**Mz ft-lb**
$Allowable_{Normal}$	3118	3118	3118	8405	8405	8405
$Calculated_{Normal}$	127	1637	113	376	185	5113
$Allowable_{Upset}$	3898	3898	3898	10086	10086	10086
$Calculated_{Upset}$	268	1861	253	622	942	6093
$Allowable_{Faulted}$	4678	4678	4678	12608	12608	12608
$Calculated_{Faulted}$	397	2028	365	739	1379	6753

Heater Exchanger B, Nozzle S3 & S4

HTX, B Nozzle S3 (PDS0085 Data Pt. 1795)

Condition	Fx lb	Fy lb	Fz lb	Mx ft-lb	My ft-lb	Mz ft-lb
$Allowable_{Normal}$	3118	3118	3118	8405	8405	8405
$Calculated_{Normal}$	124	2976	168	1583	787	4891
$Allowable_{Upset}$	3898	3898	3898	10086	10086	10086
$Calculated_{Upset}$	378	3207	363	1981	1790	5236

TABLE 3.9-2X: FUEL POOL HEAT EXCHANGER (Continued)

$Allowable_{Faulted}$	4678	4678	4678	12608	12608	12608
$Calculated_{Faulted}$	441	3292	414	2190	2339	5355

HTX, B Nozzle S4 (PDS0086 Data Pt. 1845)

Condition	**Fx lb**	**Fy lb**	**Fz lb**	**Mx ft-lb**	**My ft-lb**	**Mz ft-lb**
$Allowable_{Normal}$	3118	3118	3118	8405	8405	8405
$Calculated_{Normal}$	114	185	11	34	72	124
$Allowable_{Upset}$	3898	3898	3898	10086	10086	10086
$Calculated_{Upset}$	670	419	162	366	565	722
$Allowable_{Faulted}$	4678	4678	4678	12608	12608	12608
$Calculated_{Faulted}$	946	478	231	579	880	1086

TABLE 3.9-2Y: LPCI COUPLING (STRUT-TO-SHROUD WELD)

Criteria	Loading Combinations	Stress Type	Allowable Stress (psi)	Calculated Stress (psi)
PRIMARY MEMBRANE PLUS BENDING STRESS BASED ON ASME B&PV CODE SECTION III FOR TYPE 316L, NG - 3000 S_m @ 528 F = 14,150 psi				
FOR SERVICE LEVELS A & B (NORMAL & UPSET) CONDITION: S_{LIMIT} = 1.5 X 0.7 S_m	NORMAL AND UPSET LOADS 1. Normal Loads 2. Operating basis earthquake 3. SRV (all)	PRIMARY MEMBRANE + BENDING	14,858	14,033
FOR SERVICE LEVEL D (FAULTED) CONDITION: S_{LIMIT} = 2.25 X 0.7S_m	EMERGENCY LOADS 1. Normal loads 2. Chugging 3. SRV (ADS)	PRIMARY MEMBRANE + BENDING	22,290	20,249
FOR SERVICE LEVEL D (FAULTED) CONDITION: S_{LIMIT} = 2.4(1.5)(.7)S_m	FAULTED LOADS 1. Normal loads 2. Jet reactions 3. Annulus pressurization 4. Safe shutdown earthquake	PRIMARY MEMBRANE + BENDING	35,658	33,359

TABLE 3.9-2Z: FUEL POOL COOLING AND CLEANUP SYSTEM PUMP

Component	Loading Condition	Stress Criteria	Stress Type	Allowable Stress (psi)	Calculated Stress (psi)
Suction Nozzle	Design Pressure & Design Temperature	$S \leq S_a$	General Membrane	14,000	1,350
Discharge Nozzle	250 psig/150 F	$S \leq S_a$	General Membrane	14,000	1,350
Cover Bolting	250 psig/150 F	$S \leq S_a$	General Membrane	25,000	8,054
Pump Casing	250 psig/150 F	$S \leq S_a$	General Membrane	Minimum Wall .224 in.	Actual Wall .625 in.
Pump Cover Flange	250 psig/150 F	$S \leq S_a$	General Membrane	14,000	6,342
Pump Cover	250 psig/150 F	$S \leq S_a$	General Membrane	14,000	2,309
Pedestal Bolts (Tensile)	Pressure Loads Thermal Loads Nozzle Load Seismic Load Dead Weight Torsional Load	$S \leq S_a$	General Membrane	59,860	16,420
Pedestal Bolts	Pressure Loads Thermal Loads Nozzle Loads Seismic Loads Torsional Loads	$S \leq S_a$	Shear	22,005	8,530
Pedestal Bolts	Preload	$S \leq .9S_y$	General Membrane	132,030	68,707

TABLE 3.9-2Z: FUEL POOL COOLING AND CLEANUP SYSTEM PUMP (Continued)

Component	Loading Condition	Stress Criteria	Stress Type	Allowable Stress (psi)	Calculated Stress (psi)
Bearing Bracket Support	Thermal Load Nozzle Load Torsion Load	$S \leq S_a$ $S \leq S_a$	General Membrane	10,800	3,779

TABLE 3.9-2Z: FUEL POOL COOLING AND CLEANUP SYSTEM PUMP (Continued)

Component	Loading Condition	Stress Criteria	Stress Type	Allowable Stress (psi) or Allowable Deflection	Calculated Stress (psi) or Calculated Displacement
Motor Hold-Down Bolts (Shear)	Pressure Loads Thermal Loads Seismic Loads Dead Weight Torsional Load	$S \leq S_a$	General Membrane	30,000 psi	4,604 psi
Motor Hold-Down Bolts (Tensile)	Pressure Loads Thermal Loads Seismic Loads Dead Weight Torsional Loads	$S \leq S_a$	General Membrane	37,500 psi	7,911 psi
Foundation Bolts (Tensile)	Pressure Loads Thermal Loads Seismic Loads Dead Weight Nozzle Loads Torsional Loads	General Electric Design	General Membrane	10,000 psi	7,561 psi
Foundation Bolts (Shear)	Pressure Loads Thermal Loads Seismic Loads Dead Weight Nozzle Loads Torsional Loads	General Electric Design Spec.	Shear	10,000 psi	5,038 psi
Shaft Deflection at Pump Wear Ring	Pressure Loads Thermal Loads Seismic Loads Dead Weight Nozzle Loads Torsional Loads	Industry Standard	N/A	10 mils	6.4 mils

TABLE 3.9-2Z: FUEL POOL COOLING AND CLEANUP SYSTEM PUMP (Continued)

Component	Loading Condition	Stress Criteria	Stress Type	Allowable Stress (psi) or Allowable Deflection	Calculated Stress (psi) or Calculated Displacement
Shaft Deflection at Pump Coupling	Pressure Loads Thermal Loads Seismic Loads Dead Weight Nozzle Loads Torsional Loads	Industry Standard	N/A	10 mils	2.2 mils

Nonmenclature: S_a = Allowable general membrane stress per ASME Section III, 1974 Edition
S = Calculated Stress

TABLE 3.9-2AA: CONTROL ROD GUIDE TUBE

Criteria	Loading	Primary Stress Type	Allowable Stress (psi)	Calculated Stress (psi)(2)
CONTROL ROD GUIDE TUBE FLANGE				
Primary Stress Limit				
The allowable primary membrane stress plus bending stress is based on the ASME Boiler and Pressure Vessel Code, Section 111, Class CS for type 304 stainless steel material				
For service levels A & B (normal and upset) (Note 1) conditions:	(Note 1)		24,000	15,861
$1.5\ S_m = 1.5 \times 16,000 = 24,000$ psi	1. External Pressure 2. Metal and water weight 3. OBE + SRV 4. Scram	Primary Membrane + Bending		
For service level D (faulted) condition:			38,400	32,575
$S_{limit} = 2.4\ S_m = 2.4 \times 16,000 = 38,400$ psi	1. External Pressure 2. Metal and water weight 3. SSE + SRV 4. Scram	Primary Membrane + Bending		

Note: (1) Normal, upset, and emergency loads were analyzed together and compared to normal/upset allowables.
(2) Calculated Stress was obtained from GGNS NSSS new loads design adequacy evaluation report: GE-NEDC-30273.
(3) The structural integrity of the control rod guide tube was evaluated for EPU conditions at 4408 MWt and it was concluded the component will continue to satisfy the applicable stress criteria following EPU implementation, including consideration of the effects of flow induced vibration and the effects of EPU implementation on the design loads and loading combinations (Ref. 15).

TABLE 3.9-2AB: INCORE HOUSING

Criteria	Loading	Primary Stress Type	Allowable Stress (psi)	Calculated Stress (psi)(2)
Primary Stress Limit - The allowable primary membrane stress is based on ASME Boiler and Pressure Vessel Code, Section III for Class I vessels for type Inconel 600 austenitic high nickel alloy steel				
For service levels A & B (normal and upset) (Note 1) condition: S_{limit} = 1.0 S_m = 23,300 psi at 575 F	1. Design pressure 2. OBE 3. SRV	Maximum membrane stress intensity occurs at the outer surface of the vessel penetration	23,300	18,055
Service level D (Faulted) Condition Stress limit is the lesser of 0.7 Su = 0.7 x 80,000 = 56,000 or 2.4 Sm = 2.4 x 23,000 = 55,920	1. Design Pressure 2. Static Weights 3. Safe Shutdown Earthquake 4. LOCA 5. Annulus Pressurization	General membrane plus bending	55,920	21,725

Note: (1) Normal, upset, and emergency loads are analyzed together and compared to the normal/upset allowables.
(2) Calculated Stress was obtained from GGNS NSSS new loads design adequacy evaluation report: GE-NEDC-30273.

TABLE 3.9-2AC: HIGH PRESSURE CORE SPRAY SYSTEM ASME CODE CLASS 1 VALVE

Component/Load/Stress/Type		Design Procedure	Allowable Limit	Design/ Calculated Value	Ratio Calculated/ Allowable
1.	Body				
1.1	Loads:				
	Design Pressure	System Requirement	1575 psi	1575 psi	N/A
	Design Temperature	System Requirement	575 F	575 F	N/A
	Pipe Reaction	Not Specified			
	Thermal Effects	Not Specified			
1.2	Pressure Rating, psi	ASME Section III(1) Figure NB 3545.1-2	P_r = 900 psi	P_r = 900 psi	N/A
1.3	Minimum Wall Thickness, inches	ASME Section III(1) Para NB-3542	t (nominal) = 1.250 inches	t_m = 1.250 min inches	N/A
1.4	Primary Membrane Stress, psi	ASME Section III(1) Para NB-3545-1	$P_m \leq S_m$(500 F) = 19,400 psi	P_m = 10,510 psi	0.54
1.5	Secondary Stress Due to Pipe Reaction	ASME Section III(1) Para NB-3545.2.(b) (S = 30,000 psi)(1)	P_e = Greatest value of P_{ed}, P_{eb} and P_{et} $\leq$1.5 S_m(500 F) 1.5 (19,400) = 29,100 psi	P_e = 15,790 psi	0.54
NOTES: (1) ASME Section III 1971 Edition					
1.6	Primary Plus Secondary Stress Due to Internal Pressure	ASME Section III(1) Para NB-3545.2(a)(1)	Q_p<3Sm = 58,200 psi	Q_p = 23,280 psi	0.40
1.7	Thermal Secondary Stress	ASME Section III(1) Para NB-3545.2(c)		Q_T = 1,149.5 psi	

TABLE 3.9-2AC: HIGH PRESSURE CORE SPRAY SYSTEM ASME CODE CLASS 1 VALVE (Continued)

Component/Load/Stress/Type		Design Procedure	Allowable Limit	Design/ Calculated Value	Ratio Calculated/ Allowable
1.8	Sum of Primary Plus Secondary Stress	ASME Section III(1), Para NB-3545.2	$S_n \leq 3S_m(500\ F)$ = 3 (19,400) = 58,200 psi	$S_n = O_p+P_e+2Q_T$ = 39,370 psi	0.676
1.9	Fatigue Requirements	ASME Section III(1) Para NB-3545.3	Na ≥ 2000 cycles	$N_a = 3.636x10^4$ cycles	N/A
1.10	Cyclic Rating	ASME Section III(1), Para NB-3550	$I_t \leq 1.0$	$I_t = 0.0775$	0.0775
NOTES: (1) ASME Section III 1971 Edition					
2.0	Bonnet				
2.1	Loads: Design Pressure and Temperature	ASME Section III(1) Para NC-3324.8			
2.2	Thickness	ASME Section III(1) Para NC-3324.8	1.83 in.	0.5928 in.	
2.3	Body Neck Stresses	ASME Section III(1) Para NB-3647.1	-	-	-
2.3.1	(1) Operating Conditions	(3) (4)	1.1Sm=20,020 psi 1.65Sm=30,030 psi	20,020 psi 22,680 psi	1.0 0.76

NOTES: (1) ASME Section III 1971 Edition
(3) Valve component stresses based on 1575 psid
(4) Based on Stem Thrust is of $1.246x10^6$lb.
Limiting Stem Thrust is 156,700 lb.

TABLE 3.9-2AC: HIGH PRESSURE CORE SPRAY SYSTEM ASME CODE CLASS 1 VALVE (Continued)

Component/Load/Stress/Type	Design Procedure	Allowable Limit	Design/ Calculated Value	Ratio Calculated/ Allowable
2.4 Pressure Seal Stresses				
2.4.1 (I) Body - outer lip is governing	ASME Section III(1), (3) (4)	Membrane Stress = 1.1 Sm = 20,020 psi	16,150 psi	0.81 Close Mode
		Mem+Bending Stress = 1.65 Sm = 30,030 psi	30,030 psi	1.0 Close Mode
		= 1.1 Sm = 20,020 psi	20,020 psi	1.0 Open Mode
		= 1.65 Sm = 30,030 psi	30,030 psi	1.0 Open Mode
		= 1.1 Sm = 20,020 psi	14,500 psi	0.72 Back seating Mode
		= 1.65 Sm = 30,030 psi	23,910 psi	0.80 Back seating Mode
2.4.2 (II) Gasket Retainer Condition	ASME Section III(1), (3)	= 1.1 Sm = 21,070 psi	15,480 psi	0.73 Close Mode
		= 1.65 Sm = 31,600 psi	23,210 psi	0.73 Close Mode
		= 1.1 Sm = 21,070 psi	15,480 psi	0.73 Open Mode
		= 1.65 Sm = 31,600 psi	23,210 psi	0.73 Open Mode
		= 1.1 Sm = 21,070 psi	21,060 psi	1.0 Back seating Mode
		= 1.65 Sm = 31,600 psi	31,600 psi	1.0 Back seating Mode

NOTES: (1) ASME Section III 1971 Edition
(3) Valve Component Streses based on 1575 psid
(4) Stem Thrust is = 264,200 lb. (Close Mode)
= 343,000 lb. (Open Mode)
= 81,670 lb. (Back seating Mode)

3. Stresses in Stem					
3.1 Loads: Operator Thrust and Torque	NB-3546.3(1)	-	-	-	

TABLE 3.9-2AC: HIGH PRESSURE CORE SPRAY SYSTEM ASME CODE CLASS 1 VALVE (Continued)

Component/Load/Stress/Type		Design Procedure	Allowable Limit	Design/ Calculated Value	Ratio Calculated/ Allowable
3.2	Stem Thrust Stress	Calculate Stress Due to max. Operator Thrust in Critical Cross Section (3)	$S_T < 1.1S_m$ = 50,020 psi	S_T=48,740 psi	0.97
3.3	Stem Torque Stress	Calculate Shear Stress due to Operator Torque in Critical Cross Section	$S_S \leq 0.66\ S_m$ = 30,010 psi	S_S = 28,520 psi	0.95
3.4	Buckling on Stem	Calculate Slenderness Ratio. If greater than 30, calculate allowable load from Rankine's Formula using safety factor of 2.5	Max allowable load = 207,900 lb	Slenderness ratio = 25.44 Actual load on Stem = 162,000 lb Therefore, no buckling.	0.78
4.	Disc. Analysis				
4.1	Loads: Maximum Differential Pressure (2)	-	-	-	-
4.2	Maximum allow. T-head Thrust (Open)	ASME Section III(1) ASME Section III(1)		169,700 lb.	
5.	Yoke and Yoke Clamp				
5.1	Loads: Stem Operational Load, Seismic Load	Calculate stresses in the yoke and yoke clamp to acceptable structural analysis methods (3)			

NOTES: (1) ASME Section III 1971 Edition
(2) Valve differential pressure is 1575 psid
(3) Stem Thrust is 162,000 lb.

Note: These changes supersede the changes made to the same numbers by FSAR Change Request No. NPEFSAR 93/0020.

TABLE 3.9-2AC: HIGH PRESSURE CORE SPRAY SYSTEM ASME CODE CLASS 1 VALVE (Continued)

Component/Load/Stress/Type	Design Procedure	Allowable Limit	Design/ Calculated Value	Ratio Calculated/ Allowable
5.2 Tensile stress in Yoke Legs		$S_{max} \leq 1.65$ Sm (100 F) = 33,990 psi	S_{max} = 30,800 psi	0.90
5.3 Maximum Stress of Yoke Flange	(3)	$S_s \leq 1.65$ Sm = 30,030 psi or Sy = 27,230 psi	S_s = 27,220 psi	1.0
5.4 Maximum Bolting Stress Actuator mounting Cap screw has limiting Thrust of 156,700 lb.		135,000 psi	135,000 psi	1.0
5.5 Yoke Clamp		1.65 Sm =31,6000 psi	28,750 psi	0.91

Notes: (1) ASME Section III 1971 Edition
(2) Deleted
(3) Based on Stem Thrust is 247,700 lb

TABLE 3.9-3A: NSSS SEISMIC ACTIVE PUMPS AND VALVES

COMPONENT NAME	MASTER PARTS LIST NUMBER
Main Steam Isolation Valves	B21F022 B21F028
Main Steam Safety/Relief Valves	B21F051 B21F041 B21F047
Control Rod Drive Globe Valves	C11F010 C11F011 C11F180 C11F181
Standby Liquid Control Pumps*	C41C001
Standby Liquid Control Valve*	C41F004
RHR Pump	E12C002
LPCS Pump and Motor	E21C001
HPCS Pump and Motor	E22C001
RCIC Pump	E51C001
RCIC Turbine	E51C002
Fuel Pool Pump	G41C001
MSLIV Leakage Control Blower*	E32C001** E32C002

* Required to operate following, but not during, SSE
** E32C001 has no active function; however, it is installed in the plant for seismic considerations.

TABLE 3.9-3B: SUMMARY OF ACTIVE PUMPS (NON-NSSS)

FSAR Fig.	Equip. No.	System Name	Equipment Name	ASME Section III Code Class	Active Function	Qualification Method	Summary of Results
9.2-1, 2, 3, 4	C001A-A C001B-B	P41 - Standby service water system	Standby service water pumps	See Table 3.2-1	Provides cooling to the safety-related equipment	Analysis	Tables 3.9-25a and 3.9-25d
	C002-C		HPCS service water pump	See Table 3.2.1			
5.4-16, 17	C003A-A C003B-B C003C-B	E12 - RHR system	RHR Jockey Pumps	See Table 3.2.1	Maintains main pump discharge line filled with water at all times This prevents occurrence	Analysis	Table 3.9-25b
6.3-4	C002-A	E21 - LPCS system jockey pump	LPCS	See Table 3.2.1	of a water hammer in a partially filled line upon initiation	Analysis	Table 3.9-25b
6.3-1	C003-C	E22 - HPCS system jockey pump	HPCS	See Table 3.2.1		Analysis	Table 3.9-25b
9.5-11, 12	C002A-A C002B-B	P75 - Standby diesel generator sys.	Dsl. Gen. fuel oil transfer pump	See Table 3.2.1	Supplies F.O. to day tank	Analysis	Table 3.9-25c
6.2-81	C001A	E-61 - Combustible gas control	Drywell purge compressor	See Table 3.2.1	Maintains conc. in the drywell post-LOCA	Prototype Test	Appendix 3.9-3a
9.5-14, 15, 16, 17, 18	C001-C	P81 - HPCS diesel generator sys.	Diesel generation fuel oil transfer pump	See Table 3.2.1	Supplies fuel to HPCS diesel generator day tank	Analysis	Table 3.9-25c

TABLE 3.9-3C: SUMMARY OF ACTIVE VALVES(NON-NSSS)

FSAR Fig	Valve No.	System Name	Valve Size (inches)	Valve Type (See Note 1)	ASME Section III Code Class	Active Function	Test Type	Summary of Test Results
5.2-6, 7,8	F010A/B	Nuclear boiler system (B21)	24.0	Check	1	Contmnt. isolation	See Note 2	See subsection 3.9.3 and Appendix 3.9-3A
	F016		3.0	Gate - MO	1			
	F019		3.0	Gate - MO	1			
	F032A/B		24.0	Check - AO	1			
	F065A/B		24.0	Gate - MO	2			
	F067A/B		1.5	Globe - MO	1			
	F067C/D		1.5	Globe - Mo	1			
	F098A/B		28.0	Gate - MO	2	System isolation		
	F113		4.0	Gate - AO	3	Aux. bldg. isolation		
	F098C/D		28.0	Gate - MO	2	System isolation		
	F114		4.0	Gate - AO	3	Aux. bldg. isolation		
	F037H, D,F,L, R,N,A, U		10.0	Check	3	ADS-SRV Discharge vacuum relief		
	F078H, D,F,L, R,N,A, U		10.0	Check	3	ADS-SRV Discharge vacuum relief		
	F130A/B		2.0	Check	3	System isolation ADS-SRV air supply		

TABLE 3.9-3C: SUMMARY OF ACTIVE VALVES(NON-NSSS) (CONTINUED)

FSAR Fig	Valve No.	System Name	Valve Size (inches)	Valve Type (See Note 1)	ASME Section III Code Class	Active Function	Test Type	Summary of Test Results
	F036A, B,C,D, E,F,G, H,J,K, L,M,N, P,R,S, T,U,V, W		1.0	Check	3	System isolation SRV air accumulators		
	F039D, F,H,J, P,R,S, T		1.0	Check	3	System isolation ADS-SRV air accumulators		
	F147A/B		0.75	Gate - MO	3	System control		
	F100A, B,C		3.0	Check	3	ADS-SRV vent line vacuum relief		
	F100D, E,F,G, H,J,K, L,M,N, P,R,S, T,U,V, W		2.5	Check	3			

TABLE 3.9-3C: SUMMARY OF ACTIVE VALVES(NON-NSSS) (CONTINUED)

FSAR Fig	Valve No.	System Name	Valve Size (inches)	Valve Type (See Note 1)	ASME Section III Code Class	Active Function	Test Type	Summary of Test Results
5.4-2,3	F019	Recirc. system (B33)	3/4	Globe - MO	2	Drywell isolation	See Note 2	See subsection 3.9.3 and Appendix 3.9-3A
	F020		3/4	Globe - MO	2			
	F017A/B		3/4	Check	2			
	F013A/B		3/4	Check	2			
	F125		3/4	Globe - MO	2	Contmnt. isolation		
	F126		3/4	Globe - MO	2			
	F127		3/4	Globe - MO	2			
	F128		3/4	Globe - MO	2			
4.6-7, 8,9	F083	CRD hydraulic system (C11)	2.0	Globe - MO	2	Contmnt. isolation		
	F322		6.0	Globe - MO	3	Aux. bldg. isolation		
	F122		2.0	Check	2	Contmnt. isolation		
9.3-24	F001A/B	Standby-Liquid Control System (C41)	3.0	Globe - MO	2	Pump suction from storage tank		
	F006		1.5	Check	1	Drywell isolation		

TABLE 3.9-3C: SUMMARY OF ACTIVE VALVES(NON-NSSS) (CONTINUED)

FSAR Fig	Valve No.	System Name	Valve Size (inches)	Valve Type (See Note 1)	ASME Section III Code Class	Active Function	Test Type	Summary of Test Results
9.3-24	F007	Standby-Liquid Control System (C41)	1.5	Check	1	Drywell isolation	See Note 2	See subsection 3.9.3 and Appendix 3.9-3A
↓	F033A/B	↓	1.5	Check	2	System control isolation	↓	↓
↓	F222	↓	1.5	Check	1	System isolation		
7.5-5	F591	Fission product monitoring sys. (D23)	3/4	Globe - MO	2	Contmnt. isolation		
↓	F592	↓	3/4	Globe - MO	2	↓		
↓	F593	↓	3/4	Globe - MO	2			
↓	F594	↓	3/4	Globe - MO	2			
5.4-16, 17	F003A/B	RHR System (E12)	18.0	Globe - MO	2	System control isolation		
↓	F004 A/B/C	↓	24.0	Globe - MO	2	Contmnt. isolation		
↓	F006A/B	↓	18.0	Globe - MO	2	Pump suction from reactor sys.		
↓	F008	↓	20.0	Gate - MO	1	Contmnt. isolation		

TABLE 3.9-3C: SUMMARY OF ACTIVE VALVES(NON-NSSS) (CONTINUED)

FSAR Fig	Valve No.	System Name	Valve Size (inches)	Valve Type (See Note 1)	ASME Section III Code Class	Active Function	Test Type	Summary of Test Results
5.4-16, 17	F009	RHR system (E12)	20.0	Gate - MO	1	Contmnt. isolation	See Note 2	See subsection 3.9.3 and Appendix 3.9-3A
	F011A/B		4.0	Globe - MO	2			
	F021		14.0	Globe - MO	2			
	F024A/B		18.0	Gate - MO	2			
	F026A/B		4.0	Gate - MO	2	System control		
	F027A/B		18.0	Gate - MO	2	Contmnt. isolation		
	F028A/B		18.0	Gate - MO	2	Contmnt. spray cooling		
	F031 A/B/C		18.0	Check	2	System check		
	F037A/B		12.0	Globe - MO	2	RHR 'A' control system and contmnt. isolation		
	F040		4.0	Globe - MO	2	System control and isolation		
	F041A/B		14.0	Check - AO	1	Drywell isolation		
	F042A/B		14.0	Gate - MO	1	Contmnt. isolation		

TABLE 3.9-3C: SUMMARY OF ACTIVE VALVES(NON-NSSS) (CONTINUED)

FSAR Fig	Valve No.	System Name	Valve Size (inches)	Valve Type (See Note 1)	ASME Section III Code Class	Active Function	Test Type	Summary of Test Results
5.4-16, 17	F046 A/B/C	RHR system (E12)	4.0	Check	2	System check	See Note 2	See subsection 3.9.3 and Appendix 3.9-3A
	F047A/B		18.0	Gate - MO	2	System control		
	F048A/B		18.0	Globe - MO	2			
	F049		4.0	Gate - MO	2			
	F050A/B		12.0	Check	2			
	F051A/B		8.0	Globe - MO	2			
	F052A/B		8.0	Globe - MO	2			
	F053A/B		12.0	Globe - MO	2			
	F054A/B		4.0	Check	2	System check		
	F064 A/B/C		4.0	Gate - MO	2	Contmnt. isolation		
	F066A/B		18.0	Gate - MO	2	System isolation		
	F065A/B		3.0	Globe - AO	2	System control		
	F073A/B		2.0	Globe - MO	2	Contmnt. isolation		
	F074A/B		2.0	Globe - MO	2	System isolation		
	F082A/B		1.5	Globe - MO	2	RHR jockey pump operation		
	F290A/B		1.5	Globe - MO	2	RHR jockey pump operation and contmnt isolation		

TABLE 3.9-3C: SUMMARY OF ACTIVE VALVES(NON-NSSS) (CONTINUED)

FSAR Fig	Valve No.	System Name	Valve Size (inches)	Valve Type (See Note 1)	ASME Section III Code Class	Active Function	Test Type	Summary of Test Results
5.4-16, 17	F087A/B	RHR system (E12)	8.0	Globe - MO	2	System control	See Note 2	See subsection 3.9.3 and Appendix 3.9-3A
	F203		4.0	Gate - AO	3	Aux. bldg. isolation		
	F308		3/4	Check	1	Contmnt. isolation		
	F346		1.0	Globe - MO	2			
	F041C		12.0	Check - AO	1			
	F042C		12.0	Gate - MO	1			
	F103A/B		1.5	Check	2			
	F104A/B		1.5	Check	2			
5.4-16	F394		6.0	Gate - MO	1			
5.4-17a	F416		8.0	Check	2	System isolation		
9.2-4	F094		18.0	Gate - MO	2			
	F096		18.0	Gate - MO	2			
	F098		18.0	Check	2			
6.3-4	F001	LPCS system (E21)	24.0	Gate - MO	2	Contmnt. isolation		
	F003		16.0	Check	2	To keep line down-stream of valve filled with water		

TABLE 3.9-3C: SUMMARY OF ACTIVE VALVES(NON-NSSS) (CONTINUED)

FSAR Fig	Valve No.	System Name	Valve Size (inches)	Valve Type (See Note 1)	ASME Section III Code Class	Active Function	Test Type	Summary of Test Results
6.3-4	F005	LPCS system (E21)	14.0	Gate - MO	1	Safety injection contmnt. isolation	See Note 2	See subsection 3.9.3 and Appendix 3.9-3A
	F006		14.0	Check - AO	1	Contmnt. isolation		
	F011		4.0	Gate - MO	2			
	F012		14.0	Globe - MO	2			
6.3-1	F002	HPCS system (E22)	18.0	Check	2	System check		
	F005		14.0	Check - AO	1	Contmnt. isolation		
	F016		24.0	Check	2	System check		
	F024		16.0	Check	2	To keep line down-stream of valve filled with water		
6.2-82	F001A/B	Suppression pool makeup system (E30)	30.0	Butterfly - MO	2	Upper contmnt. pool dump to suppression pool		
	F002A/B		30.0	Butterfly - MO	2			
	F003A/B		0.75	Check	2			
	F591A/B		3/4	Globe - MO	2	Contmnt. isolation		
	F592A/B		3/4	Globe - MO	2			
	F593A/B		3/4	Globe - MO	2			
	F594A/B		3/4	Globe - MO	2			

TABLE 3.9-3C: SUMMARY OF SUMMARY OF ACTIVE VALVES(NON-NSSS) (CONTINUED)

FSAR Fig	Valve No.	System Name	Valve Size (inches)	Valve Type (See Note 1)	ASME Section III Code Class	Active Function	Test Type	Summary of Test Results
6.7-1	F001A	MSIV - leakage control system (E32)	1.5	Globe - MO	1	Contmnt. isolation to control long-term leakage through MSIVs	See Note 2	See subsection 3.9.3 and Appendix 3.9-3A
	F001E		1.5	Globe - MO	1			
	F001J		1.5	Globe - MO	1			
	F001N		1.5	Globe - MO	1			
	F002A		1.5	Globe - MO	2	To control long-term leakage through MSIVs		
	F002E		1.5	Globe - MO	2			
	F002J		1.5	Globe - MO	2			
	F002N		1.5	Globe - MO	2			
	F006		2.0	Glove - MO	2			
	F007		2.0	Globe - MO	2			
	F008		2.0	Globe - MO	2			
	F009		2.0	Globe - MO	2			
	F010		3/4	Check	2	System drain		
	F011		3/4	Check	2			
6.7-5	F001B	Feedwater leakage control system (E38)	1.5	Globe - MO	1	Contmnt. isolation to control long-term leakage in feedwater		
	F001A		1.5	Globe - MO	2			
	F002A/B		1.5	Check	1			

TABLE 3.9-3C: SUMMARY OF ACTIVE VALVES(NON-NSSS) (CONTINUED)

FSAR Fig	Valve No.	System Name	Valve Size (inches)	Valve Type (See Note 1)	ASME Section III Code Class	Active Function	Test Type	Summary of Test Results
6.7-5	F003A/B	Feedwater leakage control system (E38)	1.5	Check	2	To control long-term leakage through feedwater lines	See Note 2	See subsection 3.9.3 and Appendix 3.9-3A
5.4-10, 11	F004	RCIC - system (E51)	1.0	Globe - MO	2	System isolation		
	F005		1.0	Globe - MO	2			
	F010		6.0	Globe - MO	2			
	F011		6.0	Check	2			
	F013		6.0	Gate - MO	2			
	F015		2.0	Globe - PVC	2	System control		
	F019		2.0	Globe - MO	2	Minimum flow bypass		
	F022		4.0	Globe - MO	2	Pump test return		
	F025		1.5	Globe - AO	2	System isolation		
	F026		1.5	Globe - AO	2			
	F030		6.0	Check	2			
	F031		6.0	Gate - MO	2	Contmnt. isolation		
	F040		20.0	Check	2			
	F045		6.0	Globe - MO	2	System control		
	F046		2.0	Glove - MO isolation	2	Pump cooling loop		
	F059		4.0	Gate - MO	2	System isolation		
	F063		10.0	Gate - MO	1	Contmnt. isolation		

TABLE 3.9-3C: SUMMARY OF ACTIVE VALVES(NON-NSSS) (CONTINUED)

FSAR Fig	Valve No.	System Name	Valve Size (inches)	Valve Type (See Note 1)	ASME Section III Code Class	Active Function	Test Type	Summary of Test Results
5.4-10, 11	F064	RCIC - system (E51)	10.0	Gate - MO	1	Contmnt. isolation	See Note 2	See subsection 3.9.3 and Appendix 3.9-3A
	F065		6.0	Check	2	System control		
	F068		20.0	Gate - MO	2	Contmnt. isolation		
	F076		1.0	Globe - MO	1	Steam warmup line isolation		
	F078		1.5	Globe - MO	2	Contmnt. isolation		
	F077		2.5	Gate - MO	2			
	F095		2.0	Globe - MO	2	System control		
	F204		6.0	Check	2	System isolation		
6.2-81	F100A/B	Combustible gas control system (E61)	10.0	Check	2	System Check		
	F002A/B		10.0	Check	2	Drywell isolation		
	F003A/B		10.0	Butterfly - MO	2			
	F004A/B		10.0	Check	2			
	F005A/B		10.0	Butterfly - MO	2			
	F007		8.0	Butterfly - AO	2			
	F009		6.0	Butterfly - AO	2	Contmnt. isolation		
	F010		6.0	Butterfly - AO	2			
	F020		8.0	Gate - AO	2	Drywell isolation		

TABLE 3.9-3C: SUMMARY OF ACTIVE VALVES(NON-NSSS) (CONTINUED)

FSAR Fig	Valve No.	System Name	Valve Size (inches)	Valve Type (See Note 1)	ASME Section III Code Class	Active Function	Test Type	Summary of Test Results
6.2-81	F595 A/B/C/D	Combustible gas control system (E61)	3/4	Globe - MO	2	Contmnt. isolation	See Note 2	See subsection 3.9.3 and Appendix 3.9-3A
	F596 A/B/C/D		3/4	Globe - MO	2			
	F597 A/B/C/D		3/4	Globe - MO	2			
	F598 A/B/C/D		3/4	Globe - MO	2			
	F056		6.0	Butterfly - AO	2			
	F057		6.0	Butterfly - AO	2			
9.4-12	F001		6.0	Gate - MO	1			
	F004		6.0	Gate - MO	1			
	F028		4.0	Gate - MO	2			
	F034		4.0	Gate - MO	2			
	F039		6.0	Gate - MO	2			
5.4-21	F040	Reactor water cleanup system (G33)	6.0	Gate - MO	2			
	F051		6.0	Check	2	System isolation		
	F052A/B		6.0	Check	2			
	F053		4.0	Gate - MO	2	Contmnt. isolation		
	F054		4.0	Gate - MO	2			
	F234		4.0	Gate - AO	3	Aux. bldg. isolation		
	F235		4.0	Gate - AO	3			
	F250		6.0	Gate - MO	1	Drywell isolation		

TABLE 3.9-3C: SUMMARY OF ACTIVE VALVES(NON-NSSS) (CONTINUED)

FSAR Fig	Valve No.	System Name	Valve Size (inches)	Valve Type (See Note 1)	ASME Section III Code Class	Active Function	Test Type	Summary of Test Results
5.4-21	F251	Reactor water cleanup system (G33)	6.0	Gate - MO	1	Drywell isolation	See Note 2	See subsection 3.9.3 and Appendix 3.9-3A
	F252		6.0	Gate - MO	1	Contmnt. isolation		
	F253		6.0	Gate - MO	2	Drywell isolation		
5.4-24	F101	Filter/ demineralizer system (G36)	4.0	Gate - AO	2	Contmnt. isolation		
	F106		4.0	Gate - AO	2			
	F108		4.0	Gate - AO	3	Aux. bldg. isolation		
	F109		4.0	Gate - AO	3			
9.1-26	F029	Fuel pool cooling & cleanup system (G41)	8.0	Gate - MO	2	Contmnt. isolation		
	F019		10.0	Butterfly - AO	3	Fuel pool filter/ demineralizer isolation and bypass		
	F021		10.0	Globe - MO	3	System isolation		
	F022		10.0	Check	3			
	F032		12.0	Gate - AO	3			
	F033		12.0	Gate - AO	3			
	F048		10.0	Check	3			

TABLE 3.9-3C: SUMMARY OF ACTIVE VALVES(NON-NSSS) (CONTINUED)

FSAR Fig	Valve No.	System Name	Valve Size (inches)	Valve Type (See Note 1)	ASME Section III Code Class	Active Function	Test Type	Summary of Test Results
9.1-26	F043	Fuel pool cooling & cleanup system (G41)	10.0	Globe - MO	3	Fuel pool filter/ demineralizer isolation and bypass	See Note 2	See subsection 3.9.3 and Appendix 3.9-3A
	F045		10.0	Butterfly - AO	3			
	F044		8.0	Gate - MO	2	Contmnt. isolation		
	F040		8.0	Check	2			
	F028		8.0	Globe - MO	2			
9.1-28	F253	Filter/ demineralizer system (G46)	3.0	Gate - AO	3	Aux. bldg. isolation		
9.4-11, 13	F007	Containment cooling system (M41)	20.0	Butterfly - AO	3			
	F008		20.0	Butterfly - AO	3			
	F011		20.0	Butterfly - AO	2	Contmnt. isolation		
	F012		20.0	Butterfly - AO	2			
	F013		20.0	Butterfly - AO	2	Drywell isolation		
	F015		20.0	Butterfly - AO	2			
	F016		20.0	Butterfly - AO	2			
	F017		20.0	Butterfly - AO	2			
	F034		20.0	Butterfly - AO	2	Contmnt. isolation		

TABLE 3.9-3C: SUMMARY OF ACTIVE VALVES(NON-NSSS) (CONTINUED)

FSAR Fig	Valve No.	System Name	Valve Size (inches)	Valve Type (See Note 1)	ASME Section III Code Class	Active Function	Test Type	Summary of Test Results
9.4-11, 13	F035	Containment cooling system (M41)	20.0	Butterfly - AO	2	Contmnt. isolation	See Note 2	See subsection 3.9.3 and Appendix 3.9-3A
↓	F036	↓	20.0	Butterfly - AO	3	Aux. bldg. isolation	↓	↓
↓	F037	↓	20.0	Butterfly - AO	3	↓	↓	↓
	F591A/B	Containment & drywell instrumentation and control sys (M71)	3/4	Globe - MO			↓	↓
7.5-5					2	Contmnt. isolation	↓	↓
↓	F592A/B	↓	3/4	Globe - MO	2	↓	↓	↓
↓	F593	↓	3/4	Globe - MO	2	↓	↓	↓
↓	F594	↓	3/4	Globe - MO	2	↓	↓	↓
↓	F595	↓	3/4	Globe - MO	2	↓	↓	↓
	F047	Condensate & refueling water storage & transfer (P11)	12.0	Gate - AO			↓	↓
10.3-5, 6,7					3	Aux. bldg. isolation	↓	↓
↓	F061	↓	12.0	Gate - AO	3	↓	↓	↓
↓	F062	↓	8.0	Gate - AO	3	↓	↓	↓
↓	F063	↓	8.0	Gate - AO	3	↓	↓	↓
↓	F064	↓	12.0	Gate - AO	3	↓	↓	↓
↓	F065	↓	12.0	Gate - AO	3	↓	↓	↓
↓	F066	↓	12.0	Gate - AO	3	↓	↓	↓
↓	F067	↓	12.0	Gate - AO	3	↓	↓	↓
↓	F075	↓	6.0	Gate - AO	2	Contmnt. isolation	↓	↓

TABLE 3.9-3C: SUMMARY OF ACTIVE VALVES(NON-NSSS) (CONTINUED)

FSAR Fig	Valve No.	System Name	Valve Size (inches)	Valve Type (See Note 1)	ASME Section III Code Class	Active Function	Test Type	Summary of Test Results
10.3-5, 6,7	F004	Condensate & refueling water storage & transfer (P11)	6.0	Check	2	Contmnt. isolation	See Note 2	See subsection 3.9.3 and Appendix 3.9-3A
	F130		12.0	Butterfly - AO	2			
	F131		12.0	Butterfly - AO	2			
	F024	Makeup water treatment system (P21)	4.0	Gate - AO	3	Aux. bldg. isolation		
9.2-12	F017		2.0	Globe - MO	2	Contmnt. isolation		
	F018		2.0	Globe - MO	2			
9.2-1, 2,3,4	F001A/B	Standby service water system (P41)	24.0	Butterfly - MO	3	Req'd open		
	F005A/B		24.0	Butterfly - MO	3			
	F006A/B		20.0	Butterfly - MO	3	SSW line isolation		
	F007A/B		4.0	Gate - MO	3			
	F011		10.0	Butterfly - MO	3	Req'd open		
	F014A/B		18.0	Butterfly - MO	3			
	F015A/B		4.0	Gate - MO	3	SSW line isolation		
	F016A/B		4.0	Gate - MO	3			
	F018A/B		10.0	Butterfly - MO	3	Req'd open		

TABLE 3.9-3C: SUMMARY OF ACTIVE VALVES(NON-NSSS) (CONTINUED)

FSAR Fig	Valve No.	System Name	Valve Size (inches)	Valve Type (See Note 1)	ASME Section III Code Class	Active Function	Test Type	Summary of Test Results
9.2-1, 2,3,4	F068A/B	Standby service water system (P41)	18.0	Butterfly - MO	3		See Note 2	See subsection 3.9.3 and Appendix 3.9-3A
	F113		2.0	Globe - MO	3	Req'd open SSW line isolation		
	F119A/B		1.0	Globe - MO	3			
	F120A/B		1.0	Check	3			
	F121A/B		1.0	Globe - MO	3			
	F122A/B		1.0	Globe - MO	3	SSW line isolation		
	F159A/B		2.0	Globe - MO	2	Contmnt. isolation		
	F160A/B		2.0	Globe - MO	2			
	F168A/B		2.0	Globe - MO	2			
	F169A/B		2.0	Check	2			
	F239		3.0	Gate - AO	3	System isolation		
	F240		3.0	Gate - AO	3			
	F238		3.0	Gate - MO	3	Req'd open		
	F236		3.0	Check	3	System isolation		
	F241		3.0	Gate - MO	3			
	F064A/B		3.0	Gate - MO	3	Req'd open		
	F081A/B		3.0	Gate - MO	3			
	F074A/B		3.0	Gate - MO	3	System isolation		
	F237		3.0	Gate - MO	3	Req'd open		

TABLE 3.9-3C: SUMMARY OF ACTIVE VALVES(NON-NSSS) (CONTINUED)

FSAR Fig	Valve No.	System Name	Valve Size (inches)	Valve Type (See Note 1)	ASME Section III Code Class	Active Function	Test Type	Summary of Test Results
9.2-1, 2,3,4	F189	Standby service water system (P41)	6.0	Gate - MO	3	System isolation	See Note 2	See subsection 3.9.3 and Appendix 3.9-3A
	F154		3.0	Gate - MO	3			
	F174		3.0	Check	3			
	F155A/B		3.0	Gate - MO	3			
	F125		6.0	Gate - MO	3			
	F066A/B		3.0	Gate - MO	3	SSW line isolation		
9.2-9, 10	F028A/B	Component cooling water system (P42)	8.0	Butterfly - MO	3	System isolation		
	F032A/B		8.0	Butterfly - MO	3			
	F035		10.0	Check	2	Contmnt. isolation		
	F066		10.0	Gate - MO	2			
	F067		10.0	Gate - MO	2			
	F068		10.0	Gate - MO	2			
	F105		10.0	Butterfly - MO	3	System isolation		
	F114		8.0	Gate - MO	2	Drywell isolation		
	F115		8.0	Check	2			
	F116		8.0	Gate - MO	2			
	F117		8.0	Gate - MO	2	Drywell isolation		
	F200A/B		8.0	Butterfly - MO	3	System isolation		

TABLE 3.9-3C: SUMMARY OF ACTIVE VALVES(NON-NSSS) (CONTINUED)

FSAR Fig	Valve No.	System Name	Valve Size (inches)	Valve Type (See Note 1)	ASME Section III Code Class	Active Function	Test Type	Summary of Test Results
9.2-9, 10	F201A/B	Component cooling water system (P42)	8.0	Butterfly - MO	3	System isolation	See Note 2	See subsection 3.9.3 and Appendix 3.9-3A
	F203		8.0	Butterfly - MO	3	Train isolation		
	F204		8.0	Butterfly - MO	3	System isolation		
	F205		10.0	Butterfly - MO	3			
9.2-22, 23	F042	Plant service water system (P44)	8.0	Butterfly - MO	3			
	F052		8.0	Check	3			
	F054		8.0	Butterfly - MO	3			
	F067		8.0	Butterfly - MO	3			
9.3-9, 10,11,12	F003	Floor & equipment drain system (P45)	3.0	Gate - AO	2	Drywell isolation		
	F004		3.0	Gate - AO	2			
	F009		3.0	Gate - AO	2			
	F010		3.0	Gate - AO	2			
	F061		6.0	Gate - AO	2	Contmnt. isolation		
	F062		6.0	Gate - AO	2			
	F067		6.0	Gate - AO	2			
	F068		6.0	Gate - AO	2			
	F096		1.5	Globe - MO	2	Drywell isolation		

TABLE 3.9-3C: SUMMARY OF ACTIVE VALVES(NON-NSSS) (CONTINUED)

FSAR Fig	Valve No.	System Name	Valve Size (inches)	Valve Type (See Note 1)	ASME Section III Code Class	Active Function	Test Type	Summary of Test Results
9.3-9, 10,11,12	F097	Floor & equipment drain system (P45)	1.5	Globe - MO	2	Drywell isolation	See Note 2	See subsection 3.9.3 and Appendix 3.9-3A
↓	F098		3.0	Gate - AO	2	Contmnt. isolation		
9.2-22, 23	F099	Floor & equipment drain system (P45)	3.0	Gate - AO	2	↓		
	F158	↓	8.0	Gate - AO	3	Aux. bldg. isolation		
	F159		8.0	Gate - AO	3	↓		
	F160		8.0	Gate - AO	3			
	F161		8.0	Gate - AO	3	Aux. bldg. isolation		
	F163		3.0	Gate - AO	3	↓		
	F273		4.0	Gate - MO	2	Contmnt. isolation		
	F274		4.0	Gate - MO		↓		
9.3-3,4	F105	Service air system (P52)	3.0	Gate - AO	22			
	F122	↓	3.0	Check	2	↓		
	F160A/B		4.0	Gate - AO	3	Aux. bldg. isolation		
	F195		2.0	Globe - MO	2	Drywell isolation		
	F196		2.0	Check	2	↓		
	F221A/B		4.0	Gate - AO	3	Aux. bldg. isolation		
9.3-1,2	F001	Instrument air supply system (P53)	2.5	Gate - AO	2	Contmnt. isolation		
↓	F002	↓	2.5	Check	2	↓		
	F003		1.0	Globe - MO	2			

TABLE 3.9-3C: SUMMARY OF ACTIVE VALVES(NON-NSSS) (CONTINUED)

FSAR Fig	Valve No.	System Name	Valve Size (inches)	Valve Type (See Note 1)	ASME Section III Code Class	Active Function	Test Type	Summary of Test Results
9.3-1,2	F006	Instrument air supply system (P53)	0.75	Check	2	Contmnt. isolation	See Note 2	See subsection 3.9.3 and Appendix 3.9-3A
	F007		2.0	Globe - MO	2	Drywell isolation		
	F008		2.0	Check	2			
9.3-23	F001	Suppression pool cleanup system (P60)	12.0	Gate - AO	2	System isolation		
	F003		12.0	Gate - AO	3	Aux. bldg. isolation		
	F004		12.0	Gate - AO	3			
	F007		12.0	Gate - AO	3			
	F008		12.0	Gate - AO	3			
	F009		12.0	Gate - AO	2	Contmnt. isolation		
	F010		12.0	Gate - AO	2			
	F021		12.0	Gate - AO	2	System isolation		
9.2-13, 14	F029A/B	Domestic water system (P66)	3.0	Gate - AO	3	Aux. bldg. isolation		
9.2-17, 18,19,20, 21	F148	Plant chilled water system (P71)	4.0	Gate - AO	2	Contmnt. isolation		
	F149		4.0	Gate - AO	2			
	F150		4.0	Gate - AO	2			
	F151		4.0	Check	2			
9.2-23	F165	Drywell chilled water system (P72)	4.0	Check	2			
	F121		4.0	Gate - MO	2			
	F122		4.0	Gate - MO	2			

TABLE 3.9-3C: SUMMARY OF ACTIVE VALVES(NON-NSSS) (CONTINUED)

FSAR Fig	Valve No.	System Name	Valve Size (inches)	Valve Type (See Note 1)	ASME Section III Code Class	Active Function	Test Type	Summary of Test Results
9.2-23	F123	Drywell chilled water system (P72)	4.0	Gate - MO	2	Contmnt. isolation	See Note 2	See subsection 3.9.3 and Appendix 3.9-3A
	F124		4.0	Gate - MO	2	Drywell isolation		
	F147		4.0	Check	2			
	F125		4.0	Gate - MO	2			
	F126		4.0	Gate - MO	2			
	F006	Auxiliary bldg. ventilation system (T41)	24.0	Butterfly - AO	3	Aux. bldg. isolation		
9.4-10	F007		24.0	Butterfly - AO	3			
	F003	Fuel handling area ventilation system (T42)	44.0	Butterfly - AO	3			
9.4-2,3	F004		44.0	Butterfly - AO	3			
	F011		8.0	Butterfly - AO	3			
	F012		8.0	Butterfly - AO	3			
	F019		36.0	Butterfly - AO	3			
	F020		36.0	Butterfly - AO	3			
	F006	Standby gas treatment system (T48)	6.0	Butterfly - MO	2	System isolation		
6.5-2,3	F005		6.0	Butterfly - MO	2			

TABLE 3.9-3C: SUMMARY OF ACTIVE VALVES(NON-NSSS) (CONTINUED)

FSAR Fig	Valve No.	System Name	Valve Size (inches)	Valve Type (See Note 1)	ASME Section III Code Class	Active Function	Test Type	Summary of Test Results
6.5-2,3	F023	Standby gas treatment system (T48)	18.0	Butterfly - MO	2	System isolation	See Note 2	See subsection 3.9.3 and Appendix 3.9-3A
	F024		18.0	Butterfly - MO	2			
	F025		18.0	Butterfly - MO	2			
	F026		18.0	Butterfly - MO	2			
9.4-1	F001	Control room HVAC (Z51)	24.0	Butterfly - AO	3	Control room isolation		
	F002		24.0	Butterfly - AO	3			
	F003		18.0	Butterfly - AO	3			
	F004		18.0	Butterfly - AO	3			
	F007		20.0	Butterfly - MO	3			
	F008		20.0	Butterfly - MO	3	System isolation		
	F010		18.0	Butterfly - AO	3	Control room isolation		
	F011		18.0	Butterfly - AO	3			
	F014		20.0	Butterfly - MO	3	System isolation		
	F016		20.0	Butterfly - MO	3	Control room isolation		

Notes

1. MO - Motor-operated

   AO - Air-operated

   PCV - Self-contained pressure control

2. Valves were analytically qualified. In some cases combination of test and analysis were performed. For typical test results, see Appendix 3.9-3A.

3. The environmental conditions for which the air- and motor-operated valves are qualified are described in GGNS Environmental Qualification Files. The check valves have been designed to operate in the following limiting environments:

	Normal Drywell	Containment	Other
Temperature, F	150	135	40 to 90
Pressure, psig	-2.0 to 2.0	-0.5 to 2.0	0
Relative humidity, %	90	90	90
Radiation dosage (rads) (integrated over 40 yrs) Gamma	1.95×10^7	3.5×10^7	5×10^6
	Accident		
Temperature, F	330	330	212
Pressure, psig	30	15	0.3 to 2.0
Relative humidity, %	All steam	All steam	All steam
Radiation dosage (rads) (integrated over 40 yrs) Gamma	1.93×10^7	1.1×10^7	See Table 3.11-2

TABLE 3.9-4: NSSS COMPLIANCE WITH REGULATORY GUIDE 1.48

Component	Plant Condition	Loading Combination(1)	Design Limit	Regulatory Guide Paragraph
Class 1 Vessels	Upset (U)	[NPC or UPC] + 0.5 SSE	NB-3223	1.a
	Emergency (E)	EPC	NB-3224	1.b
	Faulted (F)	NPC + SSE + DSL	NB-3225	1.c
Class 1 Piping	U	[NPC or UPC] + 0.5 SSE	NB-3654	1.a
	E	EPC	NB-3655	1.b
	F	NPC + SSE + DSL	NB-3656	1.c
Class 1 Pumps (Inactive)	U	[NPC or UPC] + 0.5 SSE	NB-3223	2.a
	E	EPC	NB-3224	2.b
	F	NPC + SSE + DSL	NB-3225	2.c
Class 1 Pumps (Active)	U	[NPC or UPC] + 0.5 SSE	NB-3222	4.a
	E	EPC	NB-3222	4.a
	F	NPC + SSE + DSL	NB-3222	4.a
Class 1 Valves (Inactive) by analysis	U	[NPC or UPC] + 0.5 SSE	NB-3223	2.a
	E	EPC	NB-3224	2.b
	F	NPC + SSE + DSL	NB-3225	2.c
Class 1 Valves (Inactive) Designed by either std. or alternative design rules	U	[NPC or UPC] + 0.5 SSE	1.1 Pr	3.a
	E	EPC	1.2 Pr	3.b
	F	NPC + SSE + DSL	1.5 Pr	3.c
Class 1 Valves (Active) by analysis	U	[NPC or UPC] + 0.5 SSE	NB-3222	4.a
	E	EPC	NB-3222	4.b
	F	NPC + SSE + DSL	NB-3222	4.c

TABLE 3.9-4: NSSS COMPLIANCE WITH REGULATORY GUIDE 1.48 (CONTINUED)

Component	Plant Condition	Loading Combination(1)	Design Limit	Regulatory Guide Paragraph
Class 1 Valves	U	[NPC or UPC] + 0.5 SSE	1.0 Pr	5.a
(Active)	E	EPC	1.0 Pr	5.b
Designed by either std. or alternative design rules	F	NPC + SSE + DSL	1.0 Pr	5.c

(1) An OBE or 0.5 SSE intensity is classified as an emergency event. However, for design purposes it is treated as an upset condition as shown in Table 3.9-2.

TABLE 3.9-4: NSSS COMPLIANCE WITH REGULATORY GUIDE 1.48 (CONTINUED)

Component	Plant Condition	Loading Combination[1]	Code Allowable Stresses	ASME Section III Reference	Comparison with NRC Regulatory 1.48
Class 1 Vessels	U	[NPC or UPC] + 0.5 SSE	3.0 S_m (includes secondary stresses)	NB-3223	Agree
	E	EPC	1.8 S_m or 1.5 S_y	NB-3224	
	F	NPC + SSE + DSL	App. F - Sect. III	NB-3225	
Class 1 Piping	U	[NPC or UPC] + 0.5 SSE	3.0 S_m (includes secondary stresses)	NB-3654	Agree
	E	EPC	2.25 S_m	NB-3655	
	F	NPC + SSE + DSL	3.0 S_m	NB-3656	
Class 1 Pumps	U	[NPC or UPC] + 0.5 SSE	1.65 S_m (includes secondary stresses)	NB-3223	Agree
	E	EPC	1.8 S_m	NB-3224	
	F	NPC + SSE + DSL	App. F - Sect. III		
Class 1 Pumps	U	[NPC or UPC] + 0.5 SSE	Not Applicable	Not	Not
(Active)	E	EPC		Applicable	Applicable
	F	NPC + SSE + DSL			
Class 1 Valves	U	[NPC or UPC] + 0.5 SSE	Not Applicable	Not	Not
(Inactive) by analysis	E F	EPC NPC + SSE + DSL		Applicable	Applicable

TABLE 3.9-4: NSSS COMPLIANCE WITH REGULATORY GUIDE 1.48 (CONTINUED)

Component	Plant Condition	Loading Combination(1)	Code Allowable Stresses	ASME Section III Reference	Comparison with NRC Regulatory 1.48
Class 1 Valves (Inactive) Designed by either std. or alternative design rules	U	[NPC or UPC] + 0.5 SSE	1.1 PR	NB-3525	Agree
	E	EPC	1.2 PR	NB-3526	
	F	NPC + SSE + DSL	1.5 PR	NB-3527	
Class 1 Valves (Active) by analysis	U	[NPC or UPC] + 0.5 SSE	Not Applicable	Not Applicable	Not Applicable
	E	EPC			
	F	NPC + SSE + DSL			
Class 1 Valves (Inactive) Designed by either std. or alternative design rules	U	[NPC or UPC] + 0.5 SSE	1.0 Pr	NB-3525	Agree
	E	EPC	1.0 Pr	NB-3526	
	F	NPC + SSE + DSL	1.0 Pr	NB-3527	

(1) An OBE or 0.5 SSE intensity is classified as an emergency event. However,for design purposes it is treated as an upset condition as shown in Table 3.9-2.

TABLE 3.9-4: NSSS COMPLIANCE WITH REGULATORY GUIDE 1.48 (CONTINUED)

Component	Plant Condition	Loading Combination[1]	Design Limit	Regulatory Guide Paragraph
Class 2 & 3 Vessels (Division 1) of Section III of the ASEM Code	U	[NPC or UPC] + 0.5 SSE	1.1S	6.a
	E	EPC	1.1S	6.b
	F	NPC + SSE + DSL	1.5S	6.c
Class 2 Vessels ASME (Division 2) of Section VIII of the Code	U	[NPC or UPC] + 0.5 SSE	NB-3223	7.a
	E	EPC	NB-3224	7.b
	F	NPC + SSE + DSL	NB-3225	7.c
Class 2 & 3 Piping	U	[NPC or UPC] + 0.5 SSE	NC-3611.1(b) (4) (c) (b) (1)	8.a
	E	EPC	NC-3611.1(b) (4) (c) (b) (1)	8.a
	F	NPC + SSE + DSL	NC-3611.1(b) (4) (c) (b) (2)	8.b
Class 2 & 3 Pumps (Inactive)	U	[NPC or UPC] + 0.5 SSE	$\sigma m \leq 1.1S \geq (\sigma m+\sigma b)/1.5$	9.a
	E	EPC	$\sigma m \leq 1.1S \geq (\sigma m+\sigma b)/1.5$	9.a
	F	NPC + SSE + DSL	$\sigma m \leq 1.2S \geq (\sigma m+\sigma b)/1.5$	9.b
Class 2 & 3 Pumps (Active)	U	[NPC or UPC] + 0.5 SSE	$\sigma m \leq 1.0S \geq (\sigma m+\sigma b)/1.5$	10.a
	E	EPC	$\sigma m \leq 1.0S \geq (\sigma m+\sigma b)/1.5$	10.a
	F	NPC + SSE + DSL	$\sigma m \leq 1.0S \geq (\sigma m+\sigma b)/1.5$	10.a
Class 2 & 3 Valves (Inactive)	U	[NPC or UPC] + 0.5 SSE	1.1 Pr	11.a
	E	EPC	1.1 Pr	11.a
	F	NPC + SSE + DSL	1.2 Pr	11.b

TABLE 3.9-4: NSSS COMPLIANCE WITH REGULATORY GUIDE 1.48 (CONTINUED)

Component	Plant Condition	Loading Combination(1)	Design Limit	Regulatory Guide Paragraph
Class 2 & 3 Valves	U	[NPC or UPC] + 0.5 SSE	1.0 Pr	12.a
(Active)	E	EPC	1.0 Pr	12.a
	F	NPC + SSE + DSL	1.0 Pr	12.a

(1) An OBE or 0.5 SSE intensity is classified as an emergency event. However, for design purposes it is treated as an upset condition as shown in Table 3.9-2.

TABLE 3.9-4: NSSS COMPLIANCE WITH REGULATORY GUIDE 1.48 (CONTINUED)

Component	Plant Condition	Loading Combination[1]	Code Allowable Stresses	ASME Section III Reference	Comparison with NRC Regulatory 1.48
Class 2 & 3 Vessels (Division 1) of Section III of the ASME Code	U	[NPC or UPC] + 0.5 SSE	σm = 1.1S	code case 1607-1 NC/ ND-3300 more conservative	Agree except for Faulted Condition; NRC
	E	EPC			
	F	NPC + SSE + DSL	σm = 2.0S		
Class 2 Vessels (Division 2) of Section VIII of the ASME Code	U	[NPC or UPC] + 0.5 SSE	Not Applicable	Not Applicable	Not Applicable
	E	EPC			
	F	NPC + SSE + DSL			
Class 2 & 3 Piping	U	[NPC or UPC] + 0.5 SSE	1.2 Sh	NC/ND-3611.3 (b)	NRC more conservative. reflects industry position
	E	EPC	1.8 Sh	NC/ND-3611.3(c) code case-1606-1	
	F	NPC + SSE + DSL	2.4 Sh		
Class 2 & 3 Pumps (Inactive)	U	[NPC or UPC] + 0.5 SSE	Not Applicable	Not Applicable	Not Applicable
	E	EPC			
	F	NPC + SSE + DSL			
Class 2 & 3 Pumps (Active)	U	[NPC or UPC] + 0.5 SSE	σm = 1.1S	code case 1626 NC/ND-3423	Agree
	E	EPC			

TABLE 3.9-4: NSSS COMPLIANCE WITH REGULATORY GUIDE 1.48 (CONTINUED)

Component	Plant Condition	Loading Combination[1]	Code Allowable Stresses	ASME Section III Reference	Comparison with NRC Regulatory 1.48
	F	NPC + SSE + DSL	σm = 1.1S		
Class 2 & 3 Valves (Inactive)	U	[NPC or UPC] + 0.5 SSE	σm = 1.1S	code case 1625 NC/ND-3423	Equally Conservative
	E	EPC	σm = 2.0S		
	F	NPC + SSE + DSL	σm = 1.1S		
Class 2 & 3 Valves (Active)	U	[NPC or UPC] + 0.5 SSE	σm = 1.1S	code case 1625 NC/ND-3423	Equally Conservative
	E	EPC			
	F	NPC + SSE + DSL	σm = 1.2S		

(1) An OBE or 0.5 SSE intensity is classified as an emergency event. However, for design purposes it is treated as an upset condition as shown in Table 3.9-2.

TABLE 3.9-5: DELETED

TABLE 3.9-6: DELETED

TABLE 3.9-7: DEFORMATION LIMIT
(for reactor internal structures only)

Either One Of (Not Both)	General Limit
a. $\left[\frac{\text{Permissible deformation, DP}}{\text{Analyzed deformation causing loss of function, DL}}\right]$	$\leq \frac{0.9}{SF_{min}}$
b. $\left[\frac{\text{Permissible deformation, DP}}{\text{Experiment deformation causing loss of function, DE}}\right]^{1}$	$\leq \frac{1.0}{SF_{min}}$

where

DP = permissible deformation under stated conditions of normal, upset, emergency or fault

DL = analyzed deformation which could cause a system loss of function[2]

DE = experimentally determined deformation which could cause a system loss of function

[1] Equation b will not be used unless supporting data are provided to the NRC.

[2] "Loss of Function" can only be defined quite generally until attention is focused on the component of interest. In cases of interest, where deformation limits can affect the function of equipment and components, they will be specifically delineated. From a practical viewpoint, it is convenient to interchange some deformation condition at which function is assured with the loss of function condition if the required safety margins from the functioning conditions can be achieved. Therefore, it is often unnecessary to determine the actual loss of function condition because this interchange procedure produces conservative and safe designs. Examples where deformation limits apply are: control rod drive alignment and clearances for proper insertion, core support deformation causing fuel disarrangement or excess leakage of any component.

TABLE 3.9-8: PRIMARY STRESS LIMIT (FOR REACTOR INTERNAL STRUCTURES ONLY)

Any One Of (No More Than One Required)	General Limit
a. $\left[\frac{\text{Elastic evaluated primary stresses, PE}}{\text{Permissible primary stresse, PN}}\right]$	$\leq \frac{2.25}{SF_{min}}$
b. $\left[\frac{\text{Permissible load, LP}}{\text{Largest lower bound limit load, CL}}\right]$	$\leq \frac{1.5}{SF_{min}}$
c. $\left[\frac{\text{Elastic evaluated primary stresses, PE}}{\text{Conventional ultimate strength at temperature, US}}\right]$	$\leq \frac{0.75}{SF_{min}}$
d. $\left[\frac{\text{Elastic plastic evaluated nominal primary stresses, EP}}{\text{Conventional ultimate strength at temperature, US}}\right]$	$\leq \frac{0.9}{SF_{min}}$
e. $\left[\frac{\text{Permissible load, LP}}{\text{Plastic instability load, PL}}\right]^{1}$	$\leq \frac{0.9}{SF_{min}}$
f. $\left[\frac{\text{Permissible load, LP}}{\text{Ultimate load from fracture analysis, UF}}\right]^{1}$	$\leq \frac{0.9}{SF_{min}}$
g. $\left[\frac{\text{Permissible load, LP}}{\text{Ultimate load or loss of function load from test, LE}}\right]^{1}$	$\leq \frac{1.0}{SF_{min}}$

[1] Equations e., f., and g. will not be used unless supporting data are provided to the NRC by General Electric.

TABLE 3.9-8: PRIMARY STRESS LIMIT (FOR REACTOR INTERNAL STRUCTURES ONLY) (Continued)

where

PE = primary stresses evaluated on an elastic basis. The effective membrane stresses are to be averaged through the load carrying section of interest. The simplest average bending, shear or torsion stress distribution which will support the external loading will be added to the membrane stresses at the section of interest.

PN = permissible primary stress levels under normal or upset conditions under ASME Boiler and Pressure Vessel Code, Section III.

LP = permissible load under stated conditions of normal, upset, emergency or fault.

CL = lower bound limit load with yield point equal to 1.5 Sm where Sm is the tabulated value of allowable stress at temperature of the ASME III code or its equivalent. The “lower bound limit load” is here defined as that produced from the analysis of an ideally plastic nonstrain hardening) material where deformations increase with no further increase in applied load. The lower bound load is one in which the material every where satisfies equilibrium and nowhere exceeds the defined material yield strength using either a shear theory or a strain energy of distortion theory to relate multiaxial yield to the uniaxial case.

US = conventional ultimate strength at temperature or loading which would cause a system malfunction, whichever is more limiting.

EP = elastic plastic evaluated nominal primary stress. Strain hardening of the material may be used for the actual monotonic stress strain curve at the temperature of loading or any approximation to the actual stress strain curve which everywhere has a lower stress for the same strain as the actual monotonic curve may be used. Either the shear or strain energy of distortion flow rule may be used.

PL = plastic instability load. The “plastic instability load” is defined here as the load at which any load bearing section begins to diminish its cross-sectional area at a faster rate than the strain hardening can accommodate the loss in area. This type analysis requires a true stress-true strain curve or a close approximation based on monotonic loading at the temperature of loading.

TABLE 3.9-8: PRIMARY STRESS LIMIT (FOR REACTOR INTERNAL STRUCTURES ONLY) (Continued)

UF = ultimate load from fracture analyses. For components which involve sharp discontinuities (local theoretical stress concentration <3) the use of a "fracture mechanics" analysis where applicable utilizing measurements of plane strain fracture toughness may be applied to compute fracture loads. Correction for finite plastic zones and thickness effects as well as gross yielding may be necessary. The methods of linear elastic stress analysis may be used in the fracture analysis where its use is clearly conservative or supported by experimental evidence. Examples where "fracture mechanics" may be applied are for fillet welds or end of fatigue life crack propagation.

LE = ultimate load or loss of function load as determined from experiment. In using this method, account shall be taken of the dimensional tolerances which may exist between the actual part and the tested part or parts as well as differences which may exist in the ultimate tensile strength of the actual part and the tested parts. The guide to be used in each of these areas is that the experimentally determined load shall use adjusted values to account for material property and dimension variations, each of which has no greater probability than 0.1 of being exceeded in the actual part.

TABLE 3.9-9: BUCKLING STABILITY LIMIT (FOR REACTOR INTERNAL STRUCTURES ONLY)

Any One Of (No More Than One Required)	General Limit
a. $\left[\frac{\text{Permissible load, LP}}{\text{Code normal event permissible load, PN}}\right]$	$\leq \frac{2.25}{SF_{min}}$
b. $\left[\frac{\text{Permissible load, LP}}{\text{Stability analysis load, SL}}\right]$	$\leq \frac{0.9}{SF_{min}}$
c. $\left[\frac{\text{Permissible load, LP}}{\text{Ultimate buckling collapse load from test, SE}}\right]$	$\leq \frac{0.75}{SF_{min}}$

where

LP = permissible load under stated conditions of normal, upset, emergency or fault.

SL = stability analysis load. The ideal buckling analysis isoften sensitive to otherwise minor deviations from ideal geometry and boundary conditions. These effects shall be accounted for in the analysis of the buckling stability loads. Examples of this are ovality in externally pressurized shells or eccentricity on column members.

SE = ultimate buckling collapse load as determined from experiment. In using this method, account shall be taken of the dimensional tolerances which may exist between the actual part and the tested part. The guide to be used in each of these areas is that the experimentally determined load shall be adjusted to account for material property and dimension variations, each of which has no greater probability than 0.1 of being exceeded in the actual part.

[1] Equation c. will not be used unless supporting data are provided to the NRC

TABLE 3.9-10: FATIGUE LIMIT (FOR REACTOR INTERNAL STRUCTURES ONLY)

Summation of fatigue damage usage with design and operation loads following Miner hypotheses . . .[1]

Any One Of (No More Than One Required	Limit for Normal and Upset Design Conditions
a. Mean fatigue [2,3] cycle usage from analyses	≤ 0.05
b. Mean fatigue [2,3] cycle usage from test	≤ 0.33
c. Design fatigue cycle usage from analysis using the method of Table 4.2.10	≤ 1.0

[1] Miner, M.A., "Cumulative Damage in Fatigue," Journal of Applied Mechanics, Vol. 12, ASME, Vol. 67, pp A159-A164, September 1945.

[2] Fatigue failure is defined here as a 25 percent area reduction for a load carrying member which is required to function, or excess leakage, whichever is more limiting.

[3] Equations a. and b. will not be used unless supporting data are provided to the NRC.

TABLE 3.9-11: COMPUTER PROGRAMS UTILIZED IN STRESS ANALYSIS

Code No.	Name	Documentation Traceability	Remarks
ME-632	Piping System Analysis	Bechtel International Corp. Data Processing Serv.	Performs linear elastic stress for piping systems in the thermal, deadweight, seismic, and dynamic transient cases.
ME-913	LOTEMP	Bechtel International Corp. Data Processing Serv.	Calculates piping stresses in accordance with the simplified method of ASME Section III, NB-3600
TRHEAT	Computer Code for Transient Heat Analysis of Nuclear Piping	Nuclear Services Corporation	Calculates the equivalent linear and nonlinear pipe wall temperature gradients, and the discontinuity temperature differences, required for calculation of piping stresses in accordance with the requirements of Class 1 piping specified in the ASME Boiler and Pressure Vessel Code, Section III, Nuclear Power Plant Components
ANSYS	Engineering Analysis System	Swanson Analysis Systems, Inc.	Performs analysis of several types of engineering problems using the finite element method. A variety of elements gives the ANSYS program the capability of analyzing frame structures, piping systems, flat plates, axisymmetric and three-dimensional shells, and nonlinear problems.
ME-101	See FSAR subsection 3.7.3		

TABLE 3.9-12: SUMMARY OF MAXIMUM DEFLECTIONS, STRESSES AND REACTIONS CORE SPRAY PIPING SYSTEM

		Deadweight		Thermal	
Max Deflections		ME-632	EDS	ME-632	EDS
X	inches	-0.0323	-0.0327	-0.236	-0.244
Y		-0.0714	-0.0722	1.622	1.622
Z		-0.0148	-0.0151	-0.625	-0.651
Max Stress σ_{ff} - psi		2133	2100	16099	15990
Max Reactions					
F_x	lb	± 72	± 73	± 441	± 426
F_y		-2949	2956	± 2692	± 2650
F_z		± 34	± 35	± 296	± 383
M_x	lb - ft	4110	4031	-31804	31584
M_y		- 933	945	- 5913	5950
M_z		1110	1122	- 5929	5828

TABLE 3.9-13: SUMMARY OF MAXIMUM DEFLECTIONS, STRESSES, AND REACTIONS

		X+Y Seismic		Z+Y Seismic	
Max Deflections		ME-632	EDS	ME-632	EDS
X	inches	0.0505	0.0496	0.0080	0.0117
Y	inches	0.0086	0.0084	0.0033	0.0036
Z	inches	0.0040	0.0054	0.0460	0.0437
Max Stress					
σ_{ff}	psi	1396*	1377	1644	1564
Max Reactions					
F_x	lb	871	881	892	963
F_y	lb	377	372	118	149
F_z	lb	664	663	3195	3218
M_x	lb - ft	272	268	119	122
M_y	lb - ft	269	349	1964	1919
M_z	lb - ft	1668	1646	269	394

Note: *The peak stress shown here occurred at the beginning of the bend defined by tangent intersection point 20. A higher stress occurred at the middle of this bend, but EDS output does not give stresses at the middle of the bends.

TABLE 3.9-14: COMPARISON OF NATURAL PERIODS

(Periods-Seconds)

	ME-632	EDS
1	0.1077	0.1060
2	0.1035	0.1030
3	0.0658	0.0656
4	0.0561	0.0569
5	0.0532	0.0552
6	0.0509	0.0524
7	0.0502	0.0509

TABLE 3.9-15: COMPARISON OF RESULTS BETWEEN ME-913 AND ASME SAMPLE PROBLEM

		Equations						
Components	Programs	Eq.9 Stress (1)	Eq.10 Stress	Eq.11 Stress	Eq.12 Stress	Eq.13 Stress	Eq.14 Stress	Total Usage Factor
Girth butt weld	ME-913	25,950	63,112	111,833	6,563	49,526	55,917	0.0555
(Location 19)	Sample Problem	23,400	52,549	80,677	(2)	(2)	40,338	0.0126
Butt weld tee	ME-913	24,600	65,567	128,920	39,536	23,152	135,937	0.2511
(Location 10)	Sample Problem	23,400	65,596	128,950	39,564	23,155	135,977	0.3699

Notes:

(1) All stresses are in psi.

(2) Because S_n, calculated by equation 10, is less than $3S_m$, (52800 psi for type 304 at 400F), equations 12 and 13 are satisfied.

TABLE 3.9-16: HAND-CALCULATED RESULTS COMPARED WITH ME-913 OUTPUTS

		Equations						
Components	Programs	Eq.9 Stress (1)	Eq.10 Stress	Eq.11 Stress	Eq.12 Stress	Eq.13 Stress	Eq.14 Stress	Total Usage Factor
Tee	ME-913 Outputs	23,650	87,446	154,318	27,805	20,433	164,256	0.7745
(Location 115)	Hand Calculated Results	23,646	87,358	154,270	27,789	20,419	164,220	(2)

Notes:

(1) All stresses are in psi.

(2) The total usage factor in hand-calculated column is left empty because of large involvements in hand computations.

TABLE 3.9-17: DESIGN LOADING COMBINATIONS FOR NON-NSSS ASME CODE CLASS 1, 2, AND 3 COMPONENTS (6)

Load Case (1)	N	SRV_X (4)	SRV_{ADS}	OBE	SSE	SBA/IBA (3)	DBA	Acceptance Criteria
1	X	X						Upset
2	X	X		X				Upset
3	X	X			X			Faulted (2)
4	X		X			X		Faulted (2)
5	X		X	X		X		Faulted (2)
6	X		X		X	X		Faulted (2)
7	X				X		X	Faulted (2)
8	X							Normal
9	X			X				

Notes:

(1) See legend on Sheet 2 for definition of terms.

(2) All ASME Class 1, 2, and 3 piping systems which are required to function for safe shutdown under the postulated events shall meet the NRC's "Interim Technical Position - Functional Capability of Passive Components" by MEB.

(3) SBA or IBA, whichever is greater.

(4) SRVALL or SRV (whichever is controlling) will be used.

(5) Deleted

(6) Loads, except SSE and DBA, have been added using the absolute summation method of combination. SSE and DBA have been added using the square root of the sum of the squares methods.

TABLE 3.9-17: DESIGN LOADING COMBINATIONS FOR NON-NSSS ASME CODE CLASS 1, 2, AND 3 COMPONETS (6) (CONTINUED)

LOAD DEFINTION LEGEND

N	-	Normal load consists of pressure, dead weight, and thermal loads.
OBE	-	Operational basis earthquake loads
SSE	-	Loads due to vibratory motion from safe shutdown earthquake loads.
SRV_1	-	Safety/relief valve discharge induced loads from one valve's subsequent actuation.
SRV_{ALL}	-	The loads induced by acuation of all safety/ relief valves which activate within milliseconds of each other (e.g., turbine trip operational transient).
SRV_{ADS}	-	The loads induced by the actuation of safety/ relief valves associated with Automatice Depressurizaton System which actuate within milliseconds of each other during the postulated small- or intermediate-size pipe rupture.
DBA	-	Design basis accident is the sudden break of a main steam of recirculation line (largest postulated breaks). DBA-related loads include main vent clearing and pool swell, chugging, condensation oscillation, and annulus pressurization.
SBA	-	Small-break accident
IBA	-	Intermediate-break accident

TABLE 3.9-18: DESIGN CRITERIA FOR ASME CODE CLASS 1 NON-NSSS PIPING

Condition	Stress Limits[1]
Normal and upset	NB-3654
Emergency	NB-3655
Faulted	NB-3656

Note:

1. As specified by ASME Code, Section III, 1974 Edition including the Summer, 1975 Addenda, with the exceptions noted in Appendix 3A, Project Position for Reg. Guide 1.48

TABLE 3.9-19: DESIGN CRITERIA FOR ASME CODE CLASS 1 NON-NSSS VALVES

Condition	Stress Limits[1]
Normal and upset	NB-3200 or NB-3500 (Standard Design Rules)
Emergency[2]	NB-3526
Faulted[2]	NB-3527

Notes:

1. As specified by ASME Code, Section III, 1974 Edition, including the Summer, 1975 Addenda, with exceptions noted in Appendix 3A, Project Position for Reg. Guide 1.48.

2. Where valve function must be assured (active valve) during the emergency or faulted conditions, the specified emergency or faulted condition for the plant shall be considered as the normal condition for the valve.

TABLE 3.9-20: DESIGN CRITERIA FOR ASME CODE CLASS 2 AND 3 VESSELS (DESIGNED TO NC-3300 AND ND-3300)

Condition	Stress Limits
Design and normal	The vessel shall conform to the requirements of NC-3300 and ND-3300
Upset, emergency, and faulted	The vessel shall conform to the requirements of ASME Code Case 1607-1

TABLE 3.9-21: DESIGN CRITERIA FOR ASME CODE CLASS 2 NON-NSSS VESSELS (DESIGNED TO ALTERNATE RULES OF NC-3200)

Condition	Stress Limits[1] [2]
Design and normal	The vessel shall conform to the requirements of NC-3200
Upset[3]	$P_e \leq 3\ S_m$ $P_m \leq 1.1\ S_m$ $(P_m \text{ or } P_L) + P_b \leq 1.65\ S_m$
Emergency	$P_m \leq$ greater of $1.2\ S_m$ or $1.0\ S_y$ $(P_m$ or $P_L) + P_b \leq$ greater of $1.8\ S_m$ or $1.5\ S_y$
Faulted[4]	$P_m\ 2.0\ S_m$ $(P_m \text{ or } P_L) + P_b \leq 2.4\ S_m$

Notes

1. Definition of Symbols:

P_m = General primary membrane stress intensity. This stress intensity is derived from the average value across the solid section under consideration. Excluded discontinuities and concentrations. Produced only by pressure and other mechanical loads.

P_L = Local primary membrane stress intensity. Same as Pm except that discontinuities are considered.

P_b = Primary bending stress intensity. Component of primary stress intensity proportional to distance from centroid of solid section. Excludes discontinuities and concentrations. Produced only by pressure and other mechanical loads.

P_e = Secondary stress intensity range. Developed by constraint of adjacent parts or by self-constraint of a structure. Considers discontinuities but not concentrations. Produced by mechanical loads and by thermal expansion.

S_m = Design stress intensity value, Appendix I, Table I-1.0.

S_y = Yield strength value, Appendix I, Table I-2.0.

TABLE 3.9-21: DESIGN CRITERIA FOR ASME CODE CLASS 2 NON-NSSS VESSELS (DESIGNED TO ALTERNATE RULES OF NC-3200) (Continued)

2. These limits do not take into account either local or general buckling that might occur in thin-wall vessels. Such buckling shall be considered for upset conditions, but need not be considered for emergency or faulted conditions unless required by the design specification.

3. Fatigue analysis requirements of NC-3219 and Appendix XIV shall also be considered.

4. As an alternative to satisfying these limits, the faulted condition stress limits of Appendix F may be applied provided that a complete analysis in accordance with NC-3211.1(c) is performed.

TABLE 3.9-22: DESIGN CRITERIA FOR ASME CODE CLASS 2 AND 3 NON-NSSS PIPING

Condition	Stress Limits
Design and normal, upset, and emergency	The piping shall conform to the requirements of Section III, paragraphs NC-3600 and ND-3600
Faulted	The piping shall conform to the requirements of ASME Code Case 1606-1

TABLE 3.9-23: DESIGN CRITERIA FOR ASME CODE CLASS 2 AND 3 NON-NSSS PUMPS

Condition	Stress Limits
Design and Normal	The pump shall conform to the requirements of Section III, Paragraphs NC-3400 and ND-3400.
Upset, emergency, [1] and faulted[1]	The pump shall conform to the requirements of ASME Code Case 1636-1.

Note

1. Where pump function must be assured (active pumps) during the emergency or faulted condition, the pumps nozzle loads due to the specified emergency or faulted plant conditions shall be considered in satisfying the normal condition stress limits for the pump.

TABLE 3.9-24: DESIGN CRITERIA FOR ASME CODE CLASS 2 AND 3 NON-NSSS VALVES

Condition	Stress Limits[2][3][4][5]
Design and Normal	The valve shall conform to the requirements of Section III, Paragraphs NC-3500 and ND-3500.
Upset, emergency,[1] and faulted[1]	The pump shall conform to the requirements of ASME Code Case 1635-1.

Notes

1. Where valve function must be assured (active valve) during the emergency or faulted condition, the specified emergency or faulted conditions for the plant shall be considered as the normal condition for the valve.

2. Design requirements listed in this table are not applicable to valve discs, stems, seat rings, or other parts of the valve which are contained within the confines of the body and bonnet.

3. These rules do not apply to Class 2 and 3 safety or relief valves.

4. Where valves are provided with operators having extended structures, and where these structures are essential to maintaining pressure integrity, the stress limits listed in this table shall be met by performing an analysis based on static forces resulting from equivalent earthquake accelerations acting at the centers of gravity of the extended masses, for all conditions for which seismic loadings are specified in the design specification.

5. Valve nozzle (piping load) stress analysis is not required when both the following conditions are satisfied by calculation:

   (1) the section modulus and metal area at a plane normal to the flow passage through the region at the valve body crotch (for definition of crotch, see Section A-A of Figure NB-3545.2-1) are at least 10 percent greater than the section modulus and metal area of the piping connected (or joined) to the valve body inlet and outlet nozzles; and (2) the allowable stress S for the valve body material is equal to or greater than the allowable stress S of connected piping material. If the valve body material allowable stress is less than that of the connected piping, the section modulus and metal area at the valve body crotch as calculated in (1) above shall be multiplied by the ratio using stress values at the pipe-to-valve nozzle interface in satisfying condition (1) above. In lieu of these requirements, the design-by-analysis procedures of NB-3545.2 may be followed.

TABLE 3.9-25A: SUMMARY OF RESULTS EQUIPMENT: HPCS SERVICE WATER PUMPS

1. Normal Case (1)

	Actual	Allowable
Maximum column stress, psi	3,641	15,700
Max. column flange stress, psi	3,373	15,700
bolt stress, psi	8,409	28,000
Max. pump flange stress, psi	3,373	17,500
bolt stress, psi	8,409	28,000
Max. pump casing stress, psi	4,490	17,500
Max. shaft stress, psi	27,785	33,750
Motor hold down bolt stress, psi		
tensile	0	28,000
shear	0	14,000
Shaft deflection, inches	0	0.030
Nozzle stress, psi	14,12	15,700
Nozzle flange stress, psi	9,828	28,000
bolt stress, psi	2,10	12,500
Discharge head hold down bolts, psi		
tensile	14,144	28,000
shear	2,375	14,000
Discharge head stress, psi	10,189	15,700
Discharge head flange stress, psi	3,506	15,700

2. Upset Case (1)

Maximum column stress, psi	9,902	25,905
Max. column flange stress, psi	15,353	25,905
bolt stress, psi	19,418	46,200
Max. pump flange stress, psi	15,353	28,875
bolt stress, psi	19,418	46,200
Max. pump casing stress, psi	4,028	28,875
Max. shaft stress, psi	27,785	55,688

NOTES:

1. Reference GGNS-M-931,0-Q1P41C002-7.0-1, Attachment 1
2. Reference C-C811, Rev. 0, Supplement 3

TABLE 3.9-25A: SUMMARY OF RESULTS EQUIPMENT: HPCS SERVICE WATER PUMPS

2. Upset Case (1) (cont.)

	Actual	Allowable
Shaft deflection, inches	0.008	0.030
Nozzle stress, psi	14,905	25,905
Nozzle flange stress, psi	10,009	46,200
bolt stress, psi	2,106	13,750
Anchor bolt stress, psi (2)		
tensile	18,220	44,0000
shear	6,004	13,552
Discharge head hold down bolts, psi		
tensile stress	21,135	46,200
shearing stress	2,284	23,100
Discharge head stress, psi	13,138	25,905
Discharge head flange stress, psi	7,704	25,905

3. Faulted Cases (1)

	Actual	Allowable
Maximum column stress, psi	17,803	37,680
Max. column flange stress, psi	30,338	37,680
bolt stress, psi	30,338	67,200
Max. pump flange stress, psi	30,338	42,000
bolt stress, psi	30,338	67,200
Max. pump casing stress, psi	4,107	42,000
Max. shaft stress, psi	30,081	81,000
Motor hold down bolt stress, psi		
tensile	0	67,200
shear	0	33,600
Shaft deflection, inches	0.016	0.030
Nozzle stress, psi	19,955	37,680
Nozzle flange stress, psi	14,666	67,200
bolt stress, psi	3,082	33,600

NOTES:

1. Reference GGNS-M-931,0-Q1P41C002-7.0-1, Attachment 1
2. Reference C-C811, Rev. 0, Supplement 3

TABLE 3.9-25A: SUMMARY OF RESULTS EQUIPMENT: HPCS SERVICE WATER PUMPS

3. Faulted Cases (1)(cont.)

	Actual	Allowable
Anchor bolt stress, psi (2)		
tensile	34,934	80,000
shear	8,970	24,640
Discharge head hold down bolts, psi		
tensile stress	35,927	67,000
shearing stress	8,970	33,600
Discharge head stress, psi	21,092	37,680
Discharge head flange stress, psi	14,199	37,680

NOTES :
1. Reference GGNS-M-931.0-Q1P41C002-7.0-1-0, Attachment 1
2. Reference C-C811, Rev. 0, Supplement 3

TABLE 3.9-25B: SUMMARY OF RESULTS EQUIPMENT: JOCKEY PUMPS (RHR, HPCS, & LPCS)

1. Normal Steady State RHR B, RHR C, LPCS

Component	Actual	Allowable
Motor hold down bolts stress, psi		
shear	297	10,000
tensile	91	20,000
Pump hold down bolts stress, psi		
shear	3,073	10,000
tensile	7,954	20,000
Anchor bolt stress, psi		
shear	1,540	10,000
tensile	3,491	20,000
Shaft stress, psi	3,485	17,500
Frame stress, psi	3,245	21,750
Thrust retainer bolt stress, psi	2,264	20,000
Pump frame bolt stress, psi		
shear	2,167	10,000
tensile	5,019	20,000
Frame adapter flange stress, psi	17,118	26,250
bolt stress, psi	12,245	15,408
Maximum nozzle stress, psi		
discharge	7,611	16,500
suction	7,892	16,500
Nozzle flange stress, psi		
150 lb. discharge	16,779	24,750
150 lb. suction	12,831	24,750
300 lb. discharge	11,184	24,750
300 lb. suction	11,992	24,750

TABLE 3.9-25B: SUMMARY OF RESULTS EQUIPMENT: JOCKEY PUMPS (RHR, HPCS, & LPCS) (CONTINUED)

1. Normal Steady State (RHR) RHR C, LPCS) (Cont.)

Component	Actual	Allowable
Frame weld stresses, psi	492	18,000
Pump bearing loads, lbs		
inboard	292	4,443
outboard	1,350	8,855
Flexible coupling misalignment, radians	.00350	.017
Impeller connector stress, psi		
shear	1,651	8,750
tensile	1,570	17,500
Impeller relative deflection, inches	.00136	.032

2. Upset Case (RHR B, RHR C, LPCS))

Motor hold down bolts stress, psi		
shear	1,631	11,000
tensile	3,516	22,000
Pump hold down bolts stress, psi		
shear	4,261	11,000
tensile	9,765	22,000
Anchor bolt stress, psi		
shear	2,058	11,000
tensile	4,482	22,000
Shaft stress, psi	4,343	19,250
Frame stress, psi	4,451	23,925
Thrust retainer bolt stress, psi	2,458	22,000
Pump frame bolt stress, psi		

TABLE 3.9-25B: SUMMARY OF RESULTS EQUIPMENT: JOCKEY PUMPS (RHR, HPCS, & LPCS) (CONTINUED)

2. Upset Case (RHR B,RHR C, LPCS) (Cont.)

Component	Actual	Allowable
shear	2,691	11,000
tensile	7,076	22,000
Frame adapter flange stress, psi	9,870	28,875
bolt stress, psi	15,036	16,949
Maximum nozzle stress, psi		
discharge	7,658	27,225
suction	7,892	27,225
Nozzle flange stress, psi		
150 lb. discharge	18,744	27,225
150 lb. suction	15,320	27,225
300 lb. discharge	12,610	27,225
300 lb. suction	14,316	27,225
Frame weld stresses, psi	627	19,800
Pump bearing loads, lbs.		
inboard	328	4,443
outboard	1,601	8,855
Flexible coupling misalignment, radians	.00474	.017
Impeller connector stress, psi		
shear	1,651	9,625
tensile	1,570	19,250
Impeller relative deflection, inches	.00163	.032

3. Faulted Case (RHR B, RHR C, LPCS)

Motor hold down bolts stress, psi		
shear	3,171	20,000
tensile	6,941	40,000

TABLE 3.9-25B: SUMMARY OF RESULTS EQUIPMENT: JOCKEY PUMPS (RHR, HPCS, & LPCS) (CONTINUED)

3. Faulted Case (RHR B, RHR C, LPCS) (Cont.)

Component	Actual	Allowable
Pump hold down bolts stress, psi		
shear	5,765	20,000
tensile	11,576	39,766
Anchor bolt stress, psi		
shear	2,611	20,000
tensile	5,473	40,000
Shaft stress, psi	5,203	35,000
Frame stress, psi	5,657	34,800
Thrust retainer bolt stress, psi	2,652	40,000
Pump frame bolt stress, psi		
shear	3,551	20,000
tensile	9,133	40,000
Frame adapter flange stress, psi	11,251	42,000
bolt stress, psi	15,036	30,816
Maximum nozzle stress, psi		
discharge	7,658	39,600
suction	7,892	39,600
Nozzle flange stress, psi		
150 lb. discharge	16,928	39,600
150 lb. suction	13,019	39,600
300 lb. discharge	11,400	39,600
300 lb. suction	12,168	39,600
Frame weld stresses, psi	813	36,000
Pump bearing loads, lbs.		
inboard	367	4,443

TABLE 3.9-25B: SUMMARY OF RESULTS EQUIPMENT: JOCKEY PUMPS (RHR, HPCS, & LPCS) (CONTINUED)

3. Faulted Case (RHR B, RHR C, LPCS) (Cont.)

Component	Actual	Allowable
outboard	1,894	8,855
Flexible coupling misalignment, radians	.00598	.017
Impeller connector stress, psi		
shear	1,651	17,500
tensile	1,570	35,000
Impeller relative deflection, inches	.00192	.032

TABLE 3.9-25B: SUMMARY OF RESULTS EQUIPMENT: JOCKEY PUMPS (RHR, HPCS, & LPCS)

4. Normal Case (HPCS, RHR A)

COMPONENT	HPCS ACTUAL	RHR A ACTUAL	ALLOWABLE
Motor Hold Down Bolts Stress, PSI (Shear)	297	384	10,000
(Tensile)	91	384	20,000
Pump Hold Down Bolts Stress, PSI (Shear)	3073	4456	10,000
(Tensile)	7954	11,533	18,115
Anchor Bolt Stress, PSI (Shear)	1540	2233	10,000
(Tensile)	3491	5062	20,000
Shaft Stress, PSI	3485	4883	17,500
Frame Stress, PSI	3245	4705	21,750
Thrust Retainer Bolt Stress, PSI	2258	2258	20,000
Pump Frame Bolt Stress, PSI (Shear)	2167	3142	10,000
(Tensile)	5019	7278	20,000
Frame Adapter Flange Stress, PSI	5221	6096	26,250
Bolt Stress, PSI	7024	8245	25,000
Maximum Nozzle Stress, PSI (Discharge)	7642	11,269	16,500
(Suction)	8821	8821	16,500
Nozzle Flange Stress, PSI (Discharge)	6356	8403	24,750
(Suction)	4755	4755	24,750
Frame Weld Stresses, PSI	492	713	18,000
Pump Bearing Loads, Lbs. (Inboard)	292	379	4158
(Outboard)	1243	1399	6544
Flexible Coupling Misalignment, Radians	0.00350	0.00452	0.01700
Impeller Connector Stress, PSI (Shear)	1649	1649	8,750
(Tensile)	1572	1572	17,500
Impeller Relative Deflection, Inches	0.00136	0.00196	0.01100

TABLE 3.9-25B: SUMMARY OF RESULTS EQUIPMENT: JOCKEY PUMPS (RHR, HPCS, & LPCS)

5. Upset Case (HPCS, RHR A)

COMPONENT	HPCS ACTUAL	RHR A ACTUAL	ALLOWABLE
Motor Hold Down Bolts Stress, PSI (Shear)	1631	1718	11,000
(Tensile)	3516	3809	22,000
Pump Hold Down Bolts Stress, PSI (Shear)	4261	6178	11,000
(Tensile)	9765	14,159	22,000
Anchor Bolt Stress, PSI (Shear)	2058	2984	11,000
(Tensile)	4482	6499	22,000
Shaft Stress, PSI	4343	5743	19,250
Frame Stress, PSI	4451	6454	23,925
Thrust Retainer Bolt Stress, PSI	2376	2376	22,000
Pump Frame Bolt Stress, PSI (Shear)	2691	3902	11,000
(Tensile)	7076	10,260	22,000
Frame Adapter Flange Stress, PSI	10,770	12,305	28,875
Bolt Stress, PSI	14,236	16,309	27,500
Maximum Nozzle Stress, PSI (Discharge)	7778	11,424	27,225
(Suction)	8821	8821	27,225
Nozzle Flange Stress, PSI (Discharge)	7945	10202	27,225
(Suction)	6599	6843	27,225
Frame Weld Stresses, PSI	627	909	19,800
Pump Bearing Loads, Lbs. (Inboard)	328	414	4158
(Outboard)	1401	1554	6544
Flexible Coupling Misalignment, Radians	0.00474	0.00575	0.01700
Impeller Connector Stress, PSI (Shear)	1649	1649	9,625
(Tensile)	1572	1572	19,250
Impeller Relative Deflection, Inches	0.00163	0.00222	0.01100

TABLE 3.9-25B: SUMMARY OF RESULTS EQUIPMENT: JOCKEY PUMPS (RHR, HPCS, & LPCS)

6. Faulted Case (HPCS, RHR A)

COMPONENT	HPCS ACTUAL	RHR A ACTUAL	ALLOWABLE
Motor Hold Down Bolts Stress, PSI (Shear)	3171	3237	20,000
(Tensile)	6941	7162	40,000
Pump Hold Down Bolts Stress, PSI (Shear)	5765	7725	20,000
(Tensile)	11,576	15,512	40,000
Anchor Bolt Stress, PSI (Shear)	2611	3499	20,000
(Tensile)	5473	7334	40,000
Shaft Stress, PSI	5203	6261	35,000
Frame Stress, PSI	5657	7580	34,800
Thrust Retainer Bolt Stress, PSI	2494	2494	40,000
Pump Frame Bolt Stress, PSI (Shear)	3551	4758	20,000
(Tensile)	9133	12,238	40,000
Frame Adapter Flange Stress, PSI	6763	7426	42,000
Bolt Stress, PSI	9131	10,056	50,000
Maximum Nozzle Stress, PSI (Discharge)	7653	9892	39,600
(Suction)	8821	8821	39,600
Nozzle Flange Stress, PSI (Discharge)	6492	7536	39,600
(Suction)	4913	4913	39,600
Frame Weld Stresses, PSI	813	1089	36,000
Pump Bearing Loads, Lbs. (Inboard)	367	431	4158
(Outboard)	1586	1697	6544
Flexible Coupling Misalignment, Radians	0.00598	0.00674	0.01700
Impeller Connector Stress, PSI (Shear)	1649	1649	17,500
(Tensile)	1572	1572	35,000
Impeller Relative Deflection, Inches	0.00192	0.00235	0.01100

TABLE 3.9-25B: SUMMARY OF RESULTS EQUIPMENT: JOCKEY PUMPS (RHR, HPCS, & LPCS)

5. Upset Case (HPCS, RHR A)

COMPONENT	HPCS ACTUAL	RHR A ACTUAL	ALLOWABLE
Motor Hold Down Bolts Stress, PSI (Shear)	1631	1718	11,000
(Tensile)	3516	3809	22,000
Pump Hold Down Bolts Stress, PSI (Shear)	4261	6178	11,000
(Tensile)	9765	14,159	22,000
Anchor Bolt Stress, PSI (Shear)	2058	2984	11,000
(Tensile)	4482	6499	22,000
Shaft Stress, PSI	4343	5743	19,250
Frame Stress, PSI	4451	6454	23,925
Thrust Retainer Bolt Stress, PSI	2376	2376	22,000
Pump Frame Bolt Stress, PSI (Shear)	2691	3902	11,000
(Tensile)	7076	10,260	22,000
Frame Adapter Flange Stress, PSI	10,770	12,305	28,875
Bolt Stress, PSI	14,236	16,309	27,500
Maximum Nozzle Stress, PSI (Discharge)	7778	11,424	27,225
(Suction)	8821	8821	27,225
Nozzle Flange Stress, PSI (Discharge)	7945	10202	27,225
(Suction)	6599	6843	27,225
Frame Weld Stresses, PSI	627	909	19,800
Pump Bearing Loads, Lbs. (Inboard)	328	414	4158
(Outboard)	1401	1554	6544
Flexible Coupling Misalignment, Radians	0.00474	0.00575	0.01700
Impeller Connector Stress, PSI (Shear)	1649	1649	9,625
(Tensile)	1572	1572	19,250
Impeller Relative Deflection, Inches	0.00163	0.00222	0.01100

TABLE 3.9-25B: SUMMARY OF RESULTS EQUIPMENT: JOCKEY PUMPS (RHR, HPCS, & LPCS)

6. Faulted Case (HPCS, RHR A)

COMPONENT	HPCS ACTUAL	RHR A ACTUAL	ALLOWABLE
Motor Hold Down Bolts Stress, PSI (Shear)	3171	3237	20,000
(Tensile)	6941	7162	40,000
Pump Hold Down Bolts Stress, PSI (Shear)	5765	7725	20,000
(Tensile)	11,576	15,512	40,000
Anchor Bolt Stress, PSI (Shear)	2611	3499	20,000
(Tensile)	5473	7334	40,000
Shaft Stress, PSI	5203	6261	35,000
Frame Stress, PSI	5657	7580	34,800
Thrust Retainer Bolt Stress, PSI	2494	2494	40,000
Pump Frame Bolt Stress, PSI (Shear)	3551	4758	20,000
(Tensile)	9133	12,238	40,000
Frame Adapter Flange Stress, PSI	6763	7426	42,000
Bolt Stress, PSI	9131	10,056	50,000
Maximum Nozzle Stress, PSI (Discharge)	7653	9892	39,600
(Suction)	8821	8821	39,600
Nozzle Flange Stress, PSI (Discharge)	6492	7536	39,600
(Suction)	4913	4913	39,600
Frame Weld Stresses, PSI	813	1089	36,000
Pump Bearing Loads, Lbs. (Inboard)	367	431	4158
(Outboard)	1586	1697	6544
Flexible Coupling Misalignment, Radians	0.00598	0.00674	0.01700
Impeller Connector Stress, PSI (Shear)	1649	1649	17,500
(Tensile)	1572	1572	35,000
Impeller Relative Deflection, Inches	0.00192	0.00235	0.01100

TABLE 3.9-25C: ALLOWABLE STRESSES(1), (2)

EQUIPMENT: DIESEL GENERATOR FUEL OIL TRANSFER PUMPS (ASME SEC. III)

Component	Plant Condition	
	Design Normal	Upset
Plate: SA-240, TP. 316	18,800	20,680
Shaft: 316 S/S	18,800	20,680
Seamless pipe: SA-312, TP. 316	18,800	20,680
Casting: SA-351, CF8M	17,500	19,250
Tubing: SA-213, TP. 316	18,800	20,680
Bolting: SA-193, B8	15,000	16,500

Notes

1. These values are based on values shown in Section III of the Code for a design temperature of 100 F. The design pressure is 150 psi.

2. Acceleration values for the safe shutdown earthquake have been chosen as .21g in the horizontal direction and .14g in the vertical direction. In determining impeller and rotor deflections, 3.0g was used in the horizontal direction and 2.0g in the vertical direction.

TABLE 3.9-25D: SUMMARY OF RESULTS EQUIPMENT: STANDBY SERVICE WATER PUMPS

	Actual[2]	Allowable[2]
1. Normal Case		
Maximum column stress, psi	9,206	15,700
Maximum column flange stress, psi	4,676	15,700
bolt stress, psi	13,654 [3]	28,000[3]
Maximum pump flange stress, psi	4,676	17,500
bolt stress, psi	13,654	28,000
Maximum pump casing stress, psi	10,920	17,500
Maximum shaft stress, psi	27,364	33,750
Motor hold down bolt stress, psi		
tensile	(4)	28,000
shear	(4)	14,000
Shaft key stress, psi	-	-
Shaft deflection, inches	0	0.030
Impeller deflection (clearance), inches	-	-
Nozzle stress, psi	13,222	15,700
Nozzle flange stress, psi	12,751	15,700
bolt stress, psi	1,142	12,500
Anchor bolt stress, psi		
tensile	2,271 [5]	40,000[5]
shear	4,327 [5]	12,320[5]

Note
1. Deleted
2. From Enertech Seismic Qualification and Stress Analysis MA20506 (M 931.0-Q1P41C001B-7.0-1-0), unless otherwise noted
3. Studs are ASME SA564 TP 630.
4. Value is not significant relative allowable
5. From calculation C-C811, Revision 1, Supp 2.

TABLE 3.9-25D: SUMMARY OF RESULTS EQUIPMENT: STANDBY SERVICE WATER PUMPS (Continued)

	Actual[2]	Allowable[2]
1. Normal Case (Cont.)		
Discharge head stress, psi	12,669	15,700
Discharge head flange stress, psi	4,563	15,700
Discharge head weld stress, psi	7,949	31,400
Discharge head hold down bolts, psi		
tensile	21,257	33,000
shear	11,211	16,500
2. Upset Case		
Maximum column stress, psi	15,176	25,905
Maximum column flange stress, psi	15,637	25,905
bolt stress, psi (SA-540 B21 CI. 1)	27,284[3]	46,200[3]
bolt stress, psi (SA-193 B7)	-	-
Maximum pump flange stress, psi	15,637	28,875
bolt stress, psi	27,284	46,200

Note

1. Deleted
2. From Enertech Seismic Qualification and Stress Analysis MA20506 (M-931.0-Q1P41C001B-7.0-1-0), unless otherwise noted.
3. Studs are ASME SA564 TP 630.
4. Value is not significant relative allowable.
5. From calculation C-C811, Revision 1, Supp 2.

TABLE 3.9-25D: SUMMARY OF RESULTS EQUIPMENT: STANDBY SERVICE WATER PUMPS (Continued)

	Actual[2]	Allowable[2]
2. Upset Case (Cont.)		
Maximum pump casing stress, psi	9,459	28,875
Maximum shaft stress, psi	27,979	55,688
Motor hold down bolt stress, psi		
tensile	(4)	46,200
shear	(4)	23,100
Shaft key stress, psi	-	-
Shaft deflection, inches	0.006	0.03
Impeller deflection (clearance), inches	-	-
Nozzle stress, psi	14,702	25,905
Nozzle flange stress, psi	16,303	25,905
bolt stress, psi	1,142	13,750
Anchor bolt stress, psi		
tensile	19,103[5]	44,000[5]
shear	7,175[5]	13,552[5]
Discharge head stress, psi	21,208	25,905
Discharge head flange stress, psi	13,046	25,905
Discharge head weld stress, psi	-	-

Note
1. Deleted
2. From Enertech Seismic Qualification and Stress Analysis MA20506 (M-931.0-Q1P41C001B-7.0-1-0), unless otherwise noted.
3. Studs are ASME SA564 TP 630.
4. Value is not significant relative allowable.
5. From calculation C-C811, Revision 1, Supp 2.

TABLE 3.9-25D: SUMMARY OF RESULTS EQUIPMENT: STANDBY SERVICE WATER PUMPS (Continued)

	Actual[2]	Allowable[2]
2. Upset Case (cont.)		
Discharge head hold down bolts, psi		
tensile	46,812	54,450
shear	13,563	27,225
3. Faulted Case		
Maximum column stress, psi	22,535	37,680
Maximum column flange stress, psi	27,698	37,680
bolt stress, psi (SA-540 B21 CI. 1)	41,750[3]	67,200[3]
bolt stress, psi (SA-193 B7)	-	-
Maximum pump flange stress, psi	27,698	42,000
bolt stress, psi	41,750	67,200
Maximum pump casing stress, psi	9531	42,000
Maximum shaft stress, psi	28,594	81,000
Motor hold down bolt stress, psi		
tensile	(4)	67,200
shear	(4)	33,600

Note
1. Deleted
2. From Enertech Seismic Qualification and Stress Analysis MA20506 (M-931.0-Q1P41C001B-7.0-1-0), unless otherwise noted.
3. Studs are ASME SA564 TP 630.
4. Value is not significant relative allowable.
5. From calculation C-C811, Revision 1, Supp 2.

TABLE 3.9-25D: SUMMARY OF RESULTS EQUIPMENT: STANDBY SERVICE WATER PUMPS (Continued)

	Actual[2]	Allowable[2]
3. Faulted Case (cont.)		
Shaft key stress, psi	-	-
Shaft deflection, inches .05	0.012	0.030
Impeller deflection (clearance), inches	-	-
Nozzle stress, psi	23,361	37,680
Nozzle flange stress, psi	22,249	37,680
bolt stress, psi	1,142	25,000
Anchor bolt stress, psi		
tensile	42,810[5]	80,000[5]
shear	11,608[5]	24,640[5]
Discharge head stress, psi	32,631	37,680
Discharge head flange stress, psi	22,691	37,680
Discharge head weld stress, psi	-	-
Discharge head hold down bolts, psi		
tensile	78,683	79,200
shear	16,082	39,600

Note

1. Deleted
2. From Enertech Seismic Qualification and Stress Analysis MA20506 (M-931.0-Q1P41C001B-7.0-1-0), unless otherwise noted.
3. Studs are ASME SA564 TP 630.
4. Value is not significant relative allowable.
5. From calculation C-C811, Revision 1, Supp 2.

TABLE 3.9-26: EXTRAPOLATION OF VALVE SIZES FOR QUALIFICATION

Qualification Extends To:

	1/2	1	1-1/2	2	3	4	6	8	10	12	14	16	18	20	22	24	26	28	30	36
1/2	X	X																		
1	X	X	X																	
1-1/2		X	X	X																
2			X	X	X															
3				X	X	X														
4					X	X	X													
6						X	X	X												
8							X	X	X	X										
10								X	X	X	X									
12								X	X	X	X	X								
14									X	X	X	X	X	X						
16										X	X	X	X	X	X					
18											X	X	X	X	X	X				
20											X	X	X	X	X	X	X	X		
22													X	X	X	X	X	X	X	
24													X	X	X	X	X	X	X	
26														X	X	X	X	X	X	X
28														X	X	X	X	X	X	X
30															X	X	X	X	X	X
36																	X	X	X	X

Note

The qualification may be extrapolated to valves of the next higher or lower pressure rating provided the stresses in the valve rating that the qualification is extended to are lower in all regions of the valve.

TABLE 3.9-27: STRESS LIMITS - CLASS 2 & 3 ACTIVE VALVES

Condition	Stress Limits[1-5]	Pmax[6]
Design and normal	The valve shall conform to the requirements of Section III, paragraphs NC-3500, or ND-3500, as applicable	
Upset	$S_m \leq 1.1\ S$ $(S_m \text{ or } S_L) + S_b \leq 1.65\ S$	1.1
Emergency	$S_m \leq 1.5\ S$ $(S_m \text{ or } S_L) + S_b \leq 1.8\ S$	1.2
Faulted	$S_m \leq 2.0\ S$ $(S_m \text{ or } S_L) + S_b \leq 2.4\ S$	1.5

Notes

1. Definitions of symbols:

   S_m = General membrane stress. This stress is equal to the average stress across the solid section under consideration. Excludes discontinuities and concentrations. Produced only by mechanical loads.

   S_L = Local membrane stress. This stress is the same as Sm except that it includes discontinuities.

   S_b = Bending stress. This stress is equal to the linear varying portion of the stress across the solid section under consideration. Excludes discontinuities and concentrations. Produced only by mechanical loads.

   S = Allowable stress value given in ASME Section III, Appendix I, Tables I-7.1, I-7.2, I-8.1 and I-8.2, corresponding to the highest metal temperature at the section during the condition under consideration.

TABLE 3.9-27: STRESS LIMITS - CLASS 2 & 3 ACTIVE VALVES (Continued)

2. "Stress" means the maximum normal (or direct) primary stress. Classical bending and direct stress formulae, where free body diagrams determine a simple stress distribution that is in equilibrium with the applied loads, or any design formulae that had been demonstrated to be satisfactory, may be used.

3. A casting quality factor of 1 shall be used in satisfying these limits.

4. These requirements for the acceptability of valve design are not intended to ensure the functional adequacy of the valve. These requirements are intended to ensure the structural integrity of the pressure boundary only.

5. Design requirements listed in this table are not applicable to valve discs, stems, seat rings, or other parts of the valve which are contained within the confines of the body and bonnet.

6. The maximum pressure during the condition under consideration shall not exceed a value equal to the design pressure multiplied by the corresponding factor listed under Pmax for that condition.

TABLE 3.9-28: ASME BOILER AND PRESSURE VESSEL CODE CASES

1.	1606-1	Stress Criteria Section III, Classes 2 and 3 Piping subject to upset, emergency, and faulted operating conditions
2.	1607-1	Stress Criteria for Section III, Classes 2 and 3 vessels designed to NC/ND-3300 excluding the NC-3200 alternate
3.	1636-1	Stress Criteria for Section III, Class 2 and 3 Pumps subjected to upset, emergency, and faulted operating conditions
4.	1635-1	Stress Criteria for Section III, Class 2 and 3 Valves subjected to upset, emergency, and faulted operating conditions
5.	1569	Design of Piping for Pressure Relief Valve Station, Section III
6.	1541-1, 1541-3	Hydrostatic testing of embedded Class 2 and 3 piping for Section III construction
7.	1588	Electro-etching of Section III code symbols
8.	1644-4, 1644-5, 1644-6, 1644-7	Additional materials for component supports and alternate design requirements for bolted joints, Section III [Note (1)]
9.	1682	Alternate rules for material manufacturers and suppliers, Section III, Subarticle NA 3700
10.	1516-1, 1516-2	Welding of seats or minor internal permanent attachments in valves for Section III application
11.	1567	Testing lots of carbon and low alloy steel covered electrodes, Section III
12.	1580-1	Butt welded alignment tolerances and acceptable slopes for concentric centerlines for Section III, Class 1, 2, and 3 construction
13.	1622	PWHT of repair welds in carbon steel castings, Section III, Class 1, 2, and 3
14.	1718	Design of structural connections for linear type component supports Section III, Divisions 1, 2, 3 and MC
15.	1818	Welded joints in component standard supports, Section III, Division 1
16.	1677	Clarification of flange design loads Section III, Class 1, 2, and 3

TABLE 3.9-28: ASME BOILER AND PRESSURE VESSEL CODE CASES (Continued)

17.	1614	Hydrostatic testing of piping prior to or following the installation of spray nozzles for Section III, Classes 1, 2, and 3 Piping Systems
18.	N-192	Use of flexible hose for Section III, Division 1, Class 1, 2, and 3 Construction
19.	1388-2	Requirements for stainless steel precipitation hardening, Section III
20.	1683	Bolt holes for Section III Class 1, 2, 3 and MC component supports
21.	1651	Interim requirements for certification of component supports, Section III, subsection NF
22.	1690	Stock materials for Section III construction, Section III, Division 1
23.	N-224	Use of ASTM A 500 Grade B and ASTM A 501 structural tubing for welded attachments for Section III Class 2 & 3 construction
24.	N-225	Certification and identification of material for component supports, Section III, Division 1
25.	N-240	Hydrostatic testing of open-ended piping
26.	N-241	Hydrostatic testing of piping
27.	N-242	Materials certification [Note (2)]
28.	1555	Certification of safety relief valves on liquids
29.	1574	Hydrostatic test pressure for safety relief valves
30.	N-252	Low energy capacitive discharge welding method for temporary or permanent attachments to components and supports (Note (3))
31.	1335-10	Requirements for Bolting Materials, Section III
32.	N-188-1	Use of welded alloy G25 tubing Section III, Division 1, Class 2 & 3
33.	N-310	Certification of Bolting Materials
34.	N-71-10	Additional materials for component supports fabricated by welding [Note (1)]
	N-71-11	Additional materials for component supports fabricated by welding, Section III, Division 1, Subsection NF, Class 1, 2, 3, and MC component supports

TABLE 3.9-28: ASME BOILER AND PRESSURE VESSEL CODE CASES (Continued)

N-71-12 Additional materials for component supports fabricated by welding, Section III, Division 1, Class 1, 2, 3, and MC

N-71-13 Additional materials for subsection NF, Class 1, 2, 3, and MC component supports fabricated by welding, Section III, Division 1

N-71-14 Additional materials for subsection NF, Class 1, 2, 3, and MC component supports fabricated by welding, Section III, Division 1

N-71-15 Additional materials for subsection NF, Class 1, 2, 3, and MC component supports fabricated by welding, Section III, Division 1

35. N-249-1 Additional materials for component supports fabricated without welding

36. N-71-8 Additional materials for component supports, Section III, Division 1, Subsection NF, Class 1, 2, 3 and MC component supports

37. N-71-9 Additional materials for component supports fabricated by welding, Section III, Division 1, Subsection NF, Class 1, 2, 3 and MC component supports

38. N-249-2 Additional materials for Subsection NF Class 1, 2, 3 and MC component supports fabricated without welding, Section III, Division 1

N-249-3 Additional materials for Subsection NF, Class 1, 2, 3, and MC component supports fabricated without welding, Section III, Division 1

N-249-4 Additional materials for Subsection NF, Class 1, 2, 3, and MC component supports fabricated without welding, Section III, Division 1

N-249-5 Additional materials for Subsection NF, Class 1, 2, 3, and MC component supports fabricated without welding, Section III, Division 1

N-249-6 Additional materials for Subsection NF, Class 1, 2, 3, and MC component supports fabricated without welding, Section III, Division 1

N-249-7 Additional materials for Subsection NF, Class 1, 2, 3, and MC component supports fabricated without welding, Section III, Division 1

N-249-8 Additional materials for Subsection NF, Class 1, 2, 3, and MC component supports fabricated without welding, Section III, Division 1

TABLE 3.9-28: ASME BOILER AND PRESSURE VESSEL CODE CASES (Continued)

	N-249-9	Additional materials for Subsection NF, Class 1, 2, 3, and MC component supports fabricated without welding, Section III, Division 1
	N-249-10	Additional materials for Subsection NF, Class 1, 2, 3, and MC component supports fabricated without welding, Section III, Division 1
	N-249-11	Additional materials for Subsection NF, Class 1, 2, 3, and MC component supports fabricated without welding, Section III, Division 1
39.	N-247	Certified design report summary for component standard supports, Section III, Division 1, Class 1, 2, 3 and MC
40.	N-196-1	Exemption from the shakedown requirements when plastic analysis is performed for Section III, Division 1, Class 1 construction
41.	Deleted	
42.	N-411	Alternative Damping Values for Seismic Analysis of Class 1, 2, and 3 Piping Sections, Section III, Division I
43.	Deleted	
44.	N-392	Procedure for Evaluation of the Design of Hollow Circular Cross Section Welded Attachments on Class 2 and 3 Piping Section III, Division 1.
45.	N-318-3	Procedure for Evaluation of the Design of Rectangular Cross Section Attachments on Class 2 or 3 Piping Section III, Division 1 (Note (5))
46.	Deleted	
47.	Deleted	
48.	Deleted	
49.	Deleted	
50.	Deleted	
51.	Deleted	
52.	Deleted	
53.	Deleted	
54.	Deleted	

TABLE 3.9-28: ASME BOILER AND PRESSURE VESSEL CODE CASES (Continued)

Restrictions set forth in Regulatory Guides 1.84 and 1.85 are followed as delineated in Appendix 3A.

Note (1) The maximum measured ultimate tensile strength of the component support material has not exceeded 170 ksi per Regulatory Guide 1.85.

(2) Components and supports requiring the use of Code Case N-242, paragraphs 1.0 through 4.0, are as follows:

a. ASME Section III, Class 3, submersible pumps (diesel generator fuel oil transfer pumps)

b. ASME Section III, Class 2 and 3, pressure relief valves (miscellaneous relief valves)

c. ASME Section III, Class 2, piping and supports (control rod drive hydraulic system)

d. Containment and drywell personnel air locks with penetration rings and appurtenances

e. Containment equipment hatches with stiffener rings and appurtenances

f. Drywell heads with flange assemblies, drywell bulkheads, and appurtenances

g. Drywell penetrations

(3) ASME Section III, 1980 edition thru Summer 1982 addenda has been adopted for the capacitive discharge method of welding (NX 4311.2 and NX 4311.3).

(4) Deleted

(5) As required by Regulatory Guide 1.84, the following conditions are applicable to the use of Code Case N-318-3.

TABLE 3.9-28: ASME BOILER AND PRESSURE VESSEL CODE CASES (CONTINUED)

h. Lugs analyzed using Code Case N-318-3 shall be integrally attached by welding.

i. Code Case N-318-3 shall apply to ASME Section III, Class 2 and 3 piping systems. The specific system shall be identified in the stress analysis.

j. The stress analysis shall identify the specific location where attachments have been analyzed to the rules of the code case.

(6) ASME Section III Code Cases listed in Table 3.9-28 are those used in the original construction of GGNS. Other Code Cases that are adopted for use, as approved by Regulatory Guides 1.147, 1.84, 1.85 or specifically approved by the Regulatory Authority for use at GGNS, are specified in the component's design specification as required by ASME Section III NA-3250.

(7) ASME Section XI Code Cases approved for use at GGNS are specified in the Program Section for ASME Section XI, Div. I Inservice Inspection Program.

TABLE 3.9-29A: ALLOWABLE STRESSES STRESS SUMMARY FOR FUEL POOL DRAIN TANK

Plant Condition	Normal/Allowable Stress (psi)	Combined/Allowable Stress (psi)(4)
Upset		
Head	2,900/20,410	19,110/30,610
Shell	600/20,410	
Faulted		
Head	4,500/37,100	21,420/44,520
Shell	780/37,100	

Notes

1. σ_m = membrane stress
2. σ_b = bending stress
3. $\sigma_m+\sigma_b$ = combined stress
4. On tank heads only normal stresses were considered in upset/faulted loading combinations.

TABLE 3.9-29B: ALLOWABLE STRESSES STRESS SUMMARY FOR BURIED DIESEL GENERATOR FUEL OIL STORAGE TANK

Tank Shell

Plant Condition	Limiting Stress (psi)	Allowable Stress (psi)
Faulted	4,158	27,400
Upset	3,219	15,070

Tank Heads

Pressure on the tank heads:	13.3 psi
Allowable pressure for tank heads:	101 psi

Notes

1. Seismic loads were performed based on time-history analyses with two-dimensional finite element models.
2. Stress analysis was performed for applicable loading combinations.

TABLE 3.9-29C: ALLOWABLE STRESSES STRESS SUMMARY FOR STANDBY DIESEL GENERATOR FUEL OIL DAY TANKS, AND HPCS DIESEL GEN. FUEL OIL DAY TANKS

Plant Condition	Normal/Allowable Stress (psi)	Combined Allowable Stress (psi)
Normal		
Head	442/13,700	10,492/13,700
Shell	357/13,700	3,617/13,700
Upset		
Head	489/15,070	10,539/22,605
Shell	459/15,070	3,719/22,605
Emergency		
Head	442/20,550	15,192/24,660
Shell	357/20,550	3,987/24,660
Faulted		
Head	536/27,400	20,206/32,880
Shell	562/27,400	5,392/32,880

Notes

1. σ_m = membrane stress
2. σ_b = bending stress
3. Combined Stress = $\sigma_m + \sigma_b$

TABLE 3.9-29D: ALLOWABLE STRESSESSTRESS SUMMARY FOR ADS VALVE AIR RECEIVER

Plant Condition		Normal/Allowable Stress (psi)	Combined Allowable Stress (psi)
Upset			
	Head	10,530/15,070	11,480/22,610
	Shell	8,580/15,070	10,260/22,610
Faulted			
	Head	12,500/27,400	14,420/32,880
	Shell	10,180/27,400	13,530/32,880

Notes

1. σ_m = membrane stress
2. σ_b = bending stress
3. $\sigma_m + \sigma_b$ = combined stress
4. Based on code case 1607-1 (ASME Sec. III, 1974 Edition)

TABLE 3.9-30: DELETED

TABLE 3.9-31: SEISMIC AND HYDRODYNAMIC QUALIFICATIONS SUMMARY OF MAJOR SAFETY-RELATED MECHANICAL EQUIPMENT

Component	Qualification Method[1]	Input Motion[2]	Qualification Summary[3]
Jet Pump	T/A	(4)	3.9-2w
Control Rod Drive	T/A	(4)	3.9-2u
CRD Housing	A	(4)	3.9-2v
CR Guide Tube	A	(4)	3.9-2aa
Orificed Fuel Support	Static Test	5200 lbs horizontal[5] 1.0 g vertical	3.9-2b
Main Steam Isolation Valve	Analysis	(4)	3.9-2h and 3.9-2d
Main Steam Safety/Relief Valve	Dynamic Test	9.0g Horizontal 6.0g Vertical	3.9-2g and 3.9-2d
Recirculation Pump and Motor	Analysis	4.5g Horizontal 3.0g Vertical	3.9-2i
Hydraulic Control Unit	T	See Subsection 3.9.2.2.1.6.4	See Subsection 3.9.2.2.1.6.4
SLC Pump and Motor	Static Test	1.75g Horizontal 1.75g Vertical	3.9-2l

TABLE 3.9-31: SEISMIC AND HYDRODYNAMIC QUALIFICATIONS SUMMARY OF MAJOR SAFETY-RELATED MECHANICAL EQUIPMENT (Continued)

Component	Qualification Method[1]	Input Motion[2]	Qualification Summary[3]
SLC Tank	Analysis	1.75g Horizontal 1.75g Vertical	3.9-2m
RHR Pump and Motor	T/A	3.0g Horizontal 1.25g Vertical	3.9-2n
RHR Heat Exchanger	A	(4)	3.9-2o
LPCS Pump and Motor	T/A	2.0g Horizontal .83g Vertical	3.9-2n
HPCS Pump and Motor	T/A	3.0g Horizontal 1.25g Vertical	3.9-2n
RCIC Pump Assembly	Static Test	1.5g Horizontal 1.5g Vertical	3.9-2r
RCIC Turbine	T/A	(4)	3.9-2q
FPCCU Pump and Motor	Static Analysis	1.5g Horizontal 1.5g Vertical	3.9-2z
Fuel Assembly (including Analysis Channel)	Analysis	(4)	3.9-2b

TABLE 3.9-31: SEISMIC AND HYDRODYNAMIC QUALIFICATIONS SUMMARY OF MAJOR SAFETY-RELATED MECHANICAL EQUIPMENT (Continued)

Component	Qualification Method[1]	Input Motion[2]	Qualification Summary[3]
FPCCU Heat Exchanger	Analysis	OBE 1.95 g Horizontal 1.43 g Vertical SSE 3.15 g Horizontal 2.33 g Vertical	3.9-2x

NOTES:

1. Qualification Method:

   T = Testing

   A = Analysis

   T/A = Testing/Analysis

2. Input Motion: Response spectrum if column entry is a figure number
   Static g-level if column is numeric.

3. Qualification Summary: Refer to applicable table number.

4. Multi-response spectrum input to equipment support points is based on Reactor Pressure Vessel and reactor building dynamic model analysis.

5. Pound unit was used for horizontal load to obtain highest loading.

TABLE 3.9-32A: LPCS SYSTEM - HIGHEST STRESS SUMMARY

Acceptance Criteria	Limiting Stress Type	Calculated Stress or Usage Factor(1)	Allowable Limits	Ratio Actual/ Allowable	Loading(1)	Identification of location of Highest Stress Points - NODE Point Numbers(2)
ASME B&PV Code Section III, NB-3600						
Design Condition:						
Eq. 9 ≤ 1.5S_m	Primary	28,049 psi	34,950 psi	0.80	1. Pressure 2. Weight 3. OBE 4. SRV	Node 130 14" Elbow/ Butt Weld
Service Level A & B (Normal & Upset) Conditions:						
Eq. 12 ≤ 3.0S_m	Secondary	1,090 psi	60,530 psi	0.02	1. Thermal expansion	Node 27 Butt Weld at Valve
Service Level A & B (Normal & Upset) Conditions:						
Eq. 13 ≤ 3.0S_m	Primary plus Secondary (except thermal expansion)	27,800 psi	60,530 psi	0.46	1. Pressure 2. Weight 3. OBE 4. SRV 5. SAM	Node 27 Butt Weld at Valve
Service Level A & B (Normal & Upset) Conditions:						

TABLE 3.9-32A: LPCS SYSTEM - HIGHEST STRESS SUMMARY (CONTINUED)

Acceptance Criteria	Limiting Stress Type	Calculated Stress or Usage Factor(1)	Allowable Limits	Ratio Actual/ Allowable	Loading(1)	Identification of location of Highest Stress Points - NODE Point Numbers(2)
Cumulative Usage(3) Factor	N/A	0.164	1.0	0.164	1. Pressure 2. Weight 3. OBE 4. SRV 5. Thermal expansion 6. SAM	Node 130 14" Elbow/ Butt Weld
Service Level B (Upset) Condition:						
Eq. 9 ≤ $1.5S_m$	Primary	28,890 psi	30,260 psi	0.95	1. Pressure 2. Weight 3. OBE 4. SRV	Node 130 14"Elbow/ Butt Weld
Service Level D (Faulted) Condition:						
Eq. 9 ≤ $3.0S_m$	Primary	33,531 psi	69,900 psi	0.48	1. Pressure 2. Weight 14" 3. SSE 4. SRV 5. LOCA (SBA/IBA/DBA)	Node 130 Elbow/ Butt Weld

TABLE 3.9-32A: LPCS SYSTEM - HIGHEST STRESS SUMMARY (CONTINUED)

NOTES:

(1) Appropriate loading combinations of Table 3.9-2 were considered and the calculated stresses are reported for the governing loading combinations.

(2) Refer to Figure 3.9-13a for the identification of node point numbers.

(3) The fatigue usage factor shown in this table is the design fatigue usage factor. For operating fatigue usage factor see Mechanical Standard MS-40.

TABLE 3.9-32B: HPCS SYSTEM - HIGHEST STRESS SUMMARY

Acceptance Criteria	Limiting Stress Type	Calculated Stress or Usage Factor[1]	Allowable Limits	Ratio Actual/ Allowable	Loading[1]	Identification of location of Highest Stress Points - NODE Point Numbers[2]
ASME B&PV Code Section III, NB-3600						
Design Condition:						
Eq. 9 £ 1.5Sm	Primary	22,788 psi	31,602 psi	0.72	1. Pressure 2. Weight 3. OBE 4. SRV	Node 221 12" Elbow/ Butt Weld
Service Level A & B (Normal & Upset) Conditions:						
Eq. 12 ≤ 3.0Sm	Secondary	14,740 psi	60, 530 psi	0.24	1. Thermal expansion	Node 10 14" Elbow/ Butt Weld
Service Level A & B (Normal & Upset) Conditions:						

TABLE 3.9-32B: HPCS SYSTEM - HIGHEST STRESS SUMMARY (CONTINUED)

Acceptance Criteria	Limiting Stress Type	Calculated Stress or Usage Factor[1]	Allowable Limits	Ratio Actual/ Allowable	Loading[1]	Identification of location of Highest Stress Points - NODE Point Numbers[2]
Eq. 13 ≤ 3.0Sm	Primary plus Secondary (except thermal expansion)	46,840 psi	60,530 psi	0.77	1. Pressure 2. Weight 3. OBE 4. SRV 5. SAM	Node 25 Butt Weld at Valve
Service Level A & B (Normal & Upset) Conditions: Cumulative Usage(3) Factor	N/A	0.3225	1.0	0.3225	1. Pressure 2. Weight 3. OBE 4. SRV 5. Thermal expansion 6. SAM	Node 27 Butt Weld at Valve
Service Level B (Upset) Condition:						

TABLE 3.9-32B: HPCS SYSTEM - HIGHEST STRESS SUMMARY (CONTINUED)

Acceptance Criteria	Limiting Stress Type	Calculated Stress or Usage Factor[1]	Allowable Limits	Ratio Actual/ Allowable	Loading[1]	Identification of location of Highest Stress Points - NODE Point Numbers[2]
Eq. 9 ≤ 1.5Sm	Primary	26,810 psi	30,260 psi	0.89	1. Pressure 2. Weight 3. OBE 4. SRV	Node 221 12"Elbow/ Butt Weld
Service Level D (Faulted) Condition:						
Eq. 9 ≤ 3.0Sm	Primary	36,990 psi	63,204 psi	0.59	1. Pressure 2. Weight 3. SSE 4. SRV 5. LOCA (SBA/IBA/DBA)	Node 98 14" Elbow/ Butt Weld

Notes:

1. Appropriate loading combinations of Table 3.9-2 were considered and the calculated stresses are reported for the governing loading combinations.
2. Refer to Figure 3.9-13b for the identification of node point numbers.
3. The fatigue usage factor shown in this table is the design fatigue usage factor. For operating fatigue usage factor see Mechanical Standard MS-40.

TABLE 3.9-32C: RHR LPCI SYSTEM (LOOPS A, B & C) - HIGHEST STRESS SUMMARY

Acceptance Criteria	Limiting Stress Type	Calculated Stress or Usage Factor(1)	Allowable Limits	Ratio Actual/ Allowable	Loading(1)	Identification of location of Highest Stress Points - NODE Point Numbers(2)
ASME B&PV Code Section III, NB-3600						
Design Condition:						
Eq. 9 ≤ 1.5Sm	Primary	25,721 psi	34,950 psi	0.74	1. Pressure 2. Weight 3. OBE 4. SRV	Node 130 12" Elbow/ Butt Weld (LOOP C)
Service Level A & B (Normal & Upset)						
Conditions:						
Eq. 12 ≤ 3.0Sm	Secondary	23,840 psi	60,525 psi	0.39	1. Thermal expansion	Node 200 12" Elbow/ Butt Weld (LOOP A)
Service Level A & B (Normal & Upset)						
Conditions:						

TABLE 3.9-32C: RHR LPCI SYSTEM (LOOPS A, B & C) - HIGHEST STRESS SUMMARY (Continued)

Acceptance Criteria	Limiting Stress Type	Calculated Stress or Usage Factor[1]	Allowable Limits	Ratio Actual/ Allowable	Loading[1]	Identification of location of Highest Stress Points - NODE Point Numbers[2]
Eq. 13 ≤ 3.0Sm	Primary plus Secondary (except thermal expansion)	34,200 psi	60,525 psi	0.57	1. Pressure 2. Weight 3. OBE 4. SRV 5. SAM	Node 200 Butt Weld to RPV Nozzle (LOOP B)
Service Level A & B (Normal & Upset) Conditions:						
Cumulative Usage(3) Factor	N/A	0.3421	1.0	0.3421	1. Pressure 2. Weight 3. OBE 4. SRV 5. Thermal expansion 6. SAM	Node 260 14" Elbow/ Butt Weld to Flued Head (LOOP B)

TABLE 3.9-32C: RHR LPCI SYSTEM (LOOPS A, B & C) - HIGHEST STRESS SUMMARY (Continued)

Acceptance Criteria	Limiting Stress Type	Calculated Stress or Usage Factor(1)	Allowable Limits	Ratio Actual/ Allowable	Loading(1)	Identification of location of Highest Stress Points - NODE Point Numbers(2)
Service Level B (Upset) Condition:						
Eq. 9 $\leq 1.5S_m$	Primary	24,740 psi	34,950 psi	0.71	1. Pressure 2. Weight 3. SSE 4. SRV 5. LOCA (SBA/IBA/DBA)	Node 130 12" Elbow/ Butt Weld (LOOP C)

Notes:

1. Appropriate loading combinations of Table 3.9-2 were considered and the calculated stresses are reported for the governing loading combinations.
2. Refer to Figure 3.9-13c and 3.9-13e for the identification of node point numbers.
3. The fatigue usage factor shown in this table is the design fatigue usage factor. For operating fatigue usage factor see Mechanical Standard MS-40.

TABLE 3.9-32D: FEEDWATER SYSTEM - HIGHEST STRESS SUMMARY

Acceptance Criteria	Limiting Stress Type	Calculated Stress or Usage Factor[1]	Allowable Limits	Ratio Actual/ Allowable	Loading[1]	Identification of location of Highest Stress Points - NODE Point Numbers[2]
ASME B&PV Code Section III, NB-3600						
Design Condition:						
Eq. 9 ≤ 1.5S_m	Primary	23,981 psi	33,960 psi	0.71	1. Pressure 2. Weight 3. OBE 4. SRV	Node 60 18" x 12" Tee
Service Level A & B (Normal & Upset) Conditions:						
Eq. 12 ≤ 3.0S_m	Secondary	51,520 psi	67,920 psi	0.76	1. Thermal expansion	Node 60 18" x 12" Tee
Service Level A & B (Normal & Upset) Conditions:						
Eq. 13 ≤ 3.0S_m	Primary plus Secondary (except thermal expansion)	44,090 psi	67,920 psi	0.65	1. Pressure 2. Weight 3. OBE 4. SRV 5. SAM	Node 110 Butt Weld at Valve

TABLE 3.9-32D: FEEDWATER SYSTEM - HIGHEST STRESS SUMMARY (Continued)

Acceptance Criteria	Limiting Stress Type	Calculated Stress or Usage Factor[1]	Allowable Limits	Ratio Actual/ Allowable	Loading[1]	Identification of location of Highest Stress Points - NODE Point Numbers[2]
Service Level A & B (Normal & Upset) Conditions:						
Cumulative Usage(3) Factor	N/A	0.1894	1.0	0.1894	1. Pressure 2. Weight 3. OBE 4. SRV 5. Thermal expansion 6. SAM	
Service Level B (Upset) Condition:						
Eq. 9 $\leq 1.5S_m$	Primary	23,760 psi	33,960 psi	0.70	1. Pressure 2. Weight 3. OBE 4. SRV	Node 60 18" x 12" Tee

TABLE 3.9-32D: FEEDWATER SYSTEM - HIGHEST STRESS SUMMARY (Continued)

Acceptance Criteria	Limiting Stress Type	Calculated Stress or Usage Factor(1)	Allowable Limits	Ratio Actual/ Allowable	Loading(1)	Identification of location of Highest Stress Points - NODE Point Numbers(2)
Service Level D (Faulted) Condition:						
Eq. 9 ≤ $3.0S_m$	Primary	30,451 psi	67,920 psi	0.45	1. Pressure 2. Weight 3. SSE 4. SRV 5. LOCA (SBA/IBA/DBA)	Node 60 18" x 12" Tee

Notes:

1. Appropriate loading combinations of Table 3.9-2 were considered and the calculated stresses are reported for the governing loading combinations.
2. Refer to Figure 3.9-13d for the identification of node point numbers.
3. The fatigue usage factor shown in this table is the design fatigue usage factor. For Operating fatigue usage factor see Mechanical Standard MS-40.

TABLE 3.9-32E: DELETED

TABLE 3.9-32F: DELETED

TABLE 3.9-33: VIBRATION, THERMAL, AND DYNAMIC EFFECTS TESTING FOR BOP PIPING

Item	System	Flow Mode(s)	Thermal Monitoring	Vibration	Dynamic	Accessible - A Inaccessible - IA
1	Nuclear boiler					
	a) Feedwater piping	100% rated flow for vibration, approximately 5%, 25%, and 100% of rated plant power for thermal monitoring; pump trip, MSIV closure for dynamic testing	X	X	X	IA
	b) Main steam piping (Class 2 only)	25%, 50%, 75%, and 100% of rated flow for vibration; approximately 5% and 25% plant power and at rated plant power for thermal monitoring; at 45-65%, 95-100% steam flow for dynamic testing. (two lines only)	X	X	X Turbine stop valve closure and main steam isolation valve closure	IA
	c) Safety relief valve piping (discharge only)	Operational testing at pressure set points (two lines only)	X		X Safety/ relief valve operation	IA

TABLE 3.9-33: VIBRATION, THERMAL, AND DYNAMIC EFFECTS TESTING FOR BOP PIPING (Continued)

Item	System	Flow Mode(s)	Thermal Monitoring	Vibration	Dynamic	Accessible - A Inaccessible - IA
	d) Feedwater leakage control piping	Normal system operation		X		A (During plant shutdown)
	e) Main steam isolation valve leakage control piping	Normal system operation	X	X		A
	f) Main steam turbine bypass piping	Bypass valve operation	X	X	X	IA
	g) Reactor vessel level instrumentation	Reactor normal operating temperature		X		A
	h) Main steam flow instrumentation	25%, 50%, 75% and 100% of rated flow		X		IA
2	High-pressure core spray and jockey pump discharge piping	System test at rated flow (suction from condensate storage tank), pump START and STOP (6)	X (Class 1 only)	X		IA (2)

TABLE 3.9-33: VIBRATION, THERMAL, AND DYNAMIC EFFECTS TESTING FOR BOP PIPING (Continued)

Item	System	Flow Mode(s)	Thermal Monitoring	Vibration	Dynamic	Accessible - A Inaccessible - IA
3	Low-pressure core spray and jockey pump discharge piping	System test at rated flow, pump START and STOP (6)	X (Class 1 only)	X		IA (2)
4	Residual heat removal system and jockey pump discharge piping (Loops A or B and C)	(Except containment spray) system test at rated flow, pump START and STOP (6)	X (High temp only)	X		A (1)(2)
5	Standby liquid control piping to reactor	System test at rated flow, pump START and STOP (6)	X (Class 1 portion only)	X (Will be done during preop testing)		IA (2)
6	Reactor core isolation cooling piping	System at rated flow (suction from condensate storage tank), pump START and STOP (6)	X (High temp only)	X		IA
7	Reactor water cleanup system piping	System at rated flow	X (Excluding filter-demineralizers)	X (ASME Section III only)		IA

TABLE 3.9-33: VIBRATION, THERMAL, AND DYNAMIC EFFECTS TESTING FOR BOP PIPING (Continued)

Item	System	Flow Mode(s)	Thermal Monitoring	Vibration	Dynamic	Accessible - A Inaccessible - IA
8	Fuel pool cooling and cleanup piping	System test at rated flow, pump START and STOP (6)		X (Excluding filter-demineralizers)		A
9	Deleted					
10	Combustible gas control - piping from drywell purge compressors	System test at rated flow, pump START and STOP (6)		X		A
11	Standby service water - piping (including small piping)	System test at rated flow, pump START and STOP (6)		X (5)		A
12	Condensate	System test at rated flow, pump START and STOP (6)	X	X		IA
13	Feedwater (B31.1)	System test at rated flow, pump START and STOP	X	X		IA

TABLE 3.9-33: VIBRATION, THERMAL, AND DYNAMIC EFFECTS TESTING FOR BOP PIPING (Continued)

Item	System	Flow Mode(s)	Thermal Monitoring	Vibration	Dynamic	Accessible - A Inaccessible - IA
14	Reactor core isolation cooling steam supply to turbine flow monitoring small piping	Rated flow to turbine		X		IA
15	Control rod drive (insert and withdrawal piping)	Scram			X	IA
16	Instrument air (small piping for safety/relief valve air accumulator, ADS valve air receivers, and MSIV air accumulators)	Safety/relief valve and MSIV operation			X	IA
17	Reactor recirculation (instrumentation piping, vents, and drains)	Rated flow		X		A/IA (4)

TABLE 3.9-33: VIBRATION, THERMAL, AND DYNAMIC EFFECTS TESTING FOR BOP PIPING (Continued)

Item	System	Flow Mode(s)	Thermal Monitoring	Vibration	Dynamic	Accessible - A Inaccessible - IA
18	ECCS and RCIC (instrumentation piping, vents, drains, and test connections)	Rated flow		X		A

Notes:

(1) Steam condensation mode test will be inaccessible. The steam condensing mode of RHR is not licensed by NRC for GGNS.

(2) Piping in the drywell will be inaccessible. All other locations will be accessible unless otherwise noted.

(3) All data points on systems (or portions of systems) designated as inaccessible will be instrumented.

(4) A = During preoperational testing

IA = During power operational testing

(5) Visual inspection for small piping only.

(6) Pump start and stop will be recorded and compared with the dynamic acceptance criteria.

TABLE 3.9-34: CALCULATION OF MAXIMUM HORIZONTAL LOAD TO SHOW CONFORMANCE WITH FAULTED CODE "D" LIMITS

$$P = \frac{5200\ lb}{0.8 \times 0.65} \times \frac{39,500}{18,800} \times \frac{(0.327)^2}{(0.345)^2}$$

P = 18,875

5,200 lb	Design Faulted Load
0.65	Quality Factor (Reduction in strength to account for the fact that all the castings from one heat are not fully volumetrically examined)
0.8	Reduction Factor on Collapse Load per Paragraph F-1380, Appendix F, Section III, ASME Code
0.327"	Maximum Wall Thickness of Casting Actually Tested
0.345"	Minimum Design Wall Thickness for Fuel Support Casting (The stress produced at the most highly stressed region is in bending and is assumed to be directly proportional to the inverse of the thickness squared.)
39,500 psi	Room Temperature Yield Strength of Material Tested (from Material Cert.)
18,800 psi	Section III Yield Strength at 550° F (Collapse assumed to be a function of yield strengths)

TABLE 3.9-35: COMPONENT CYCLIC OR TRANSIENT LIMITS[1]

Component	Cyclic or Transient Limit	Design Cycle or Transient
Reactor	120 heatup and cooldown cycles	70°F to 560°F to 70°F
	80 step change cycles	Loss of feedwater heaters
	200 reactor trip cycles	100% to 0% of RATED THERMAL POWER
	40 hydrostatic pressure or leak tests	Pressurized to $\geq$ 930 psig and $\leq$ 1250 psig

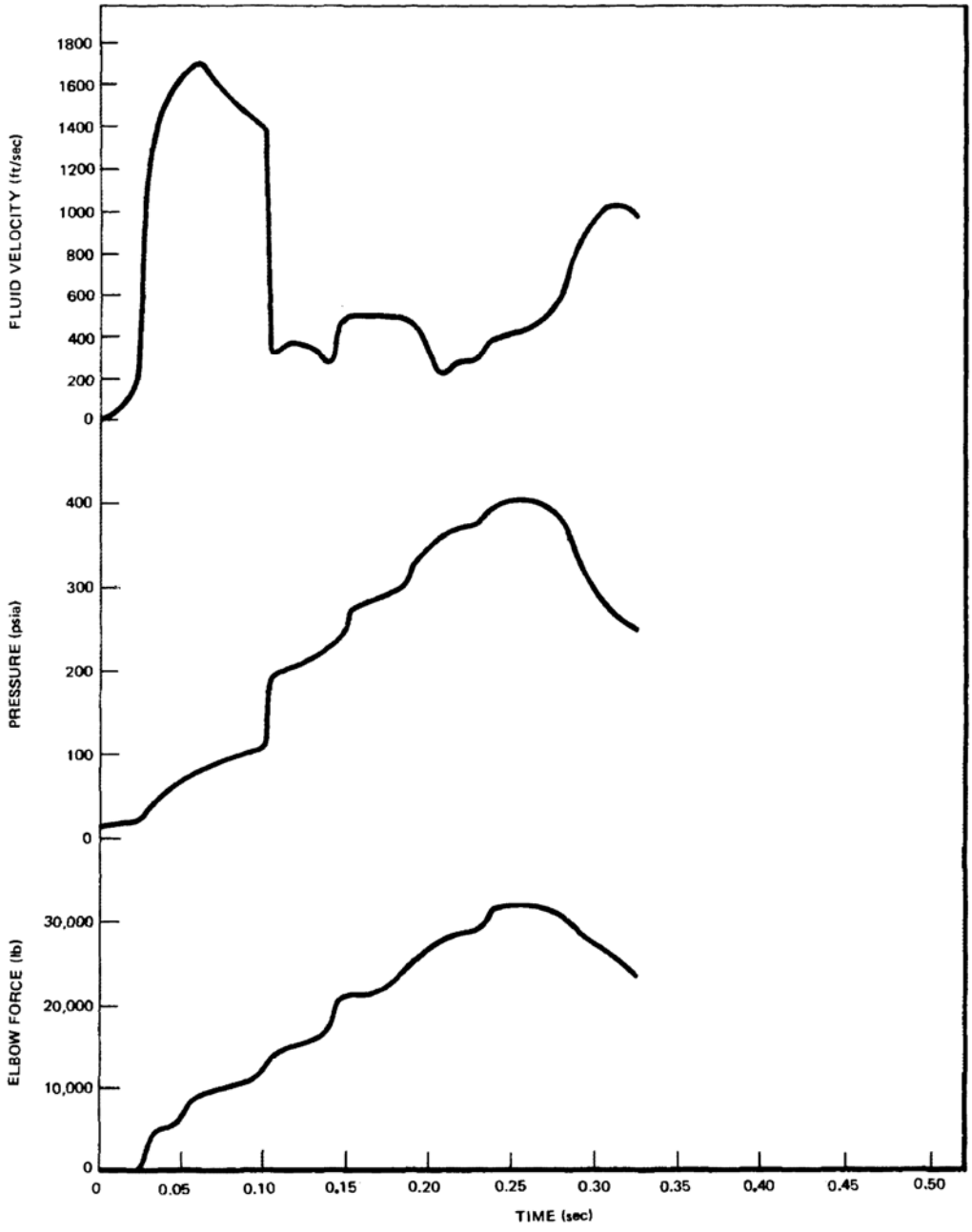
Note:

[1] The transient limit (fatigue cycle) shown in this table is the design transient limit (fatigue cycle). For operating fatigue cycle see Mechanical Standard MS-40.

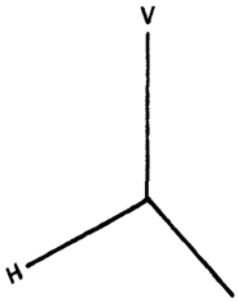
Figures 3.9-1 through 3.9-4 and

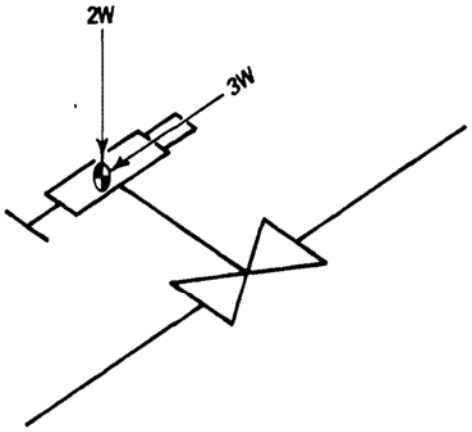
Figures 3.9-5a and 3.9-5b

DELETED



MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT	TYPICAL RELIEF VALVE TRANSIENT FIGURE 3.9-6





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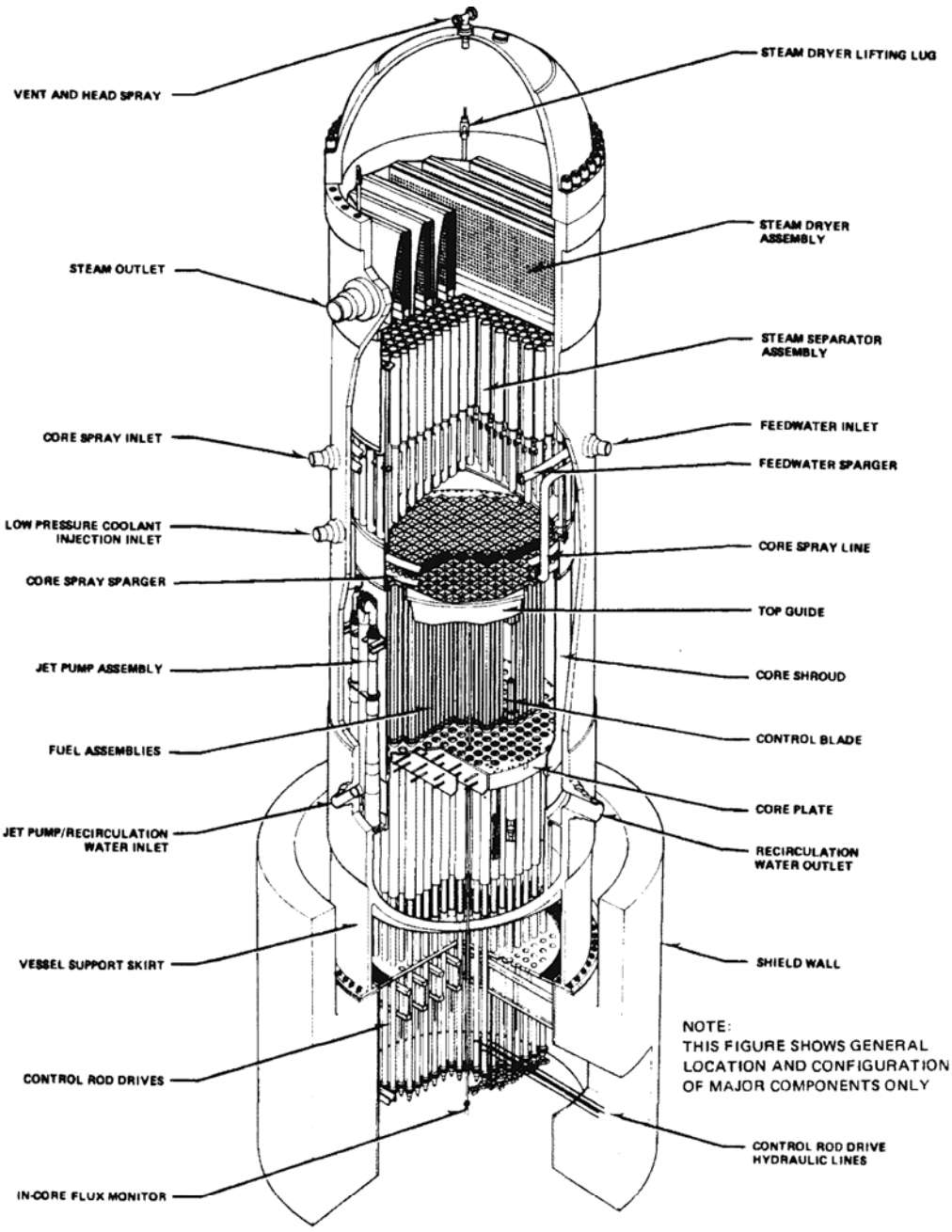
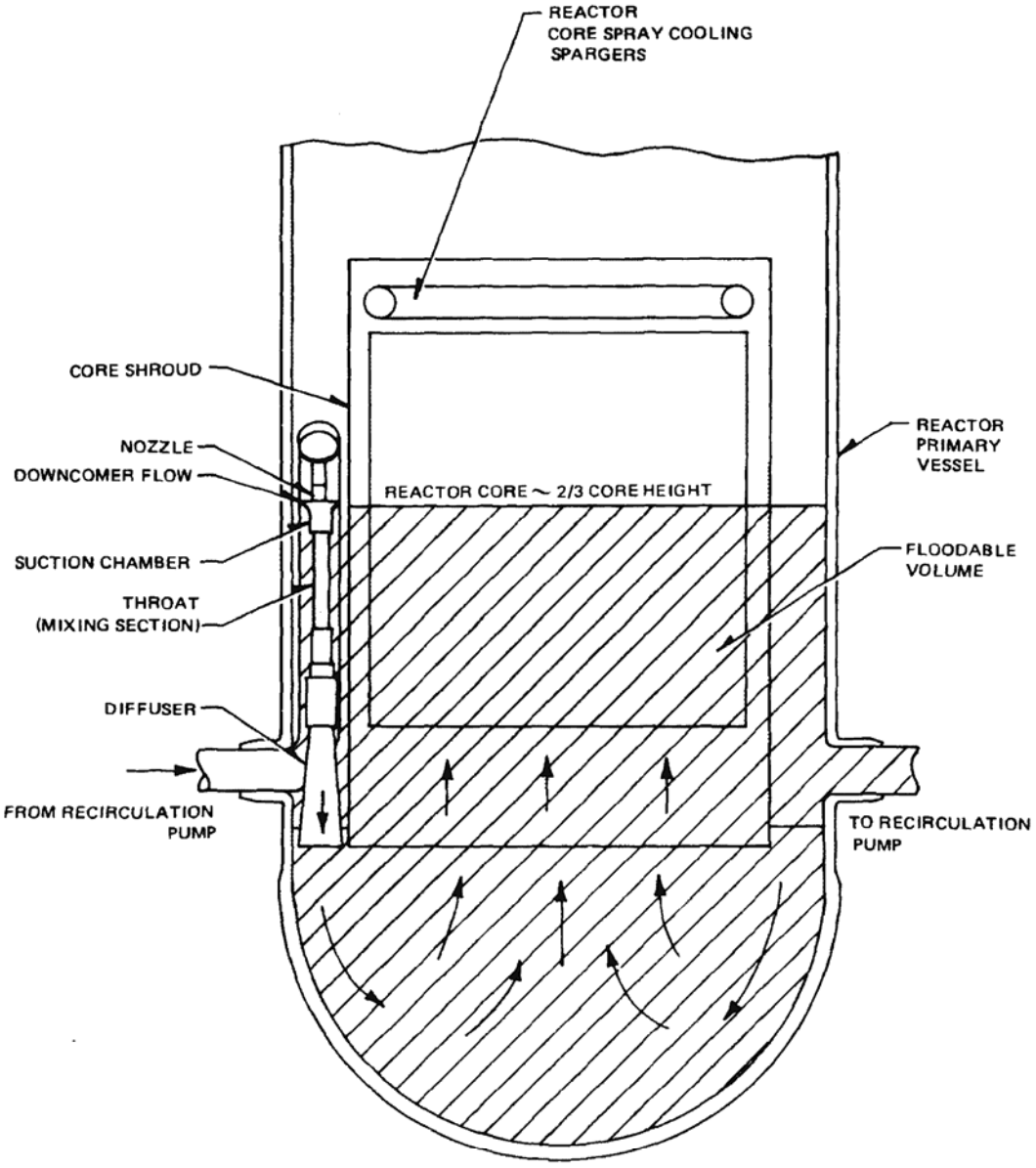
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT

TYPICAL TESTING ARRANGEMENT FOR
QUALIFICATION TESTING OF
ACTIVE VALVE/ACTUATOR ASSEMBLIES

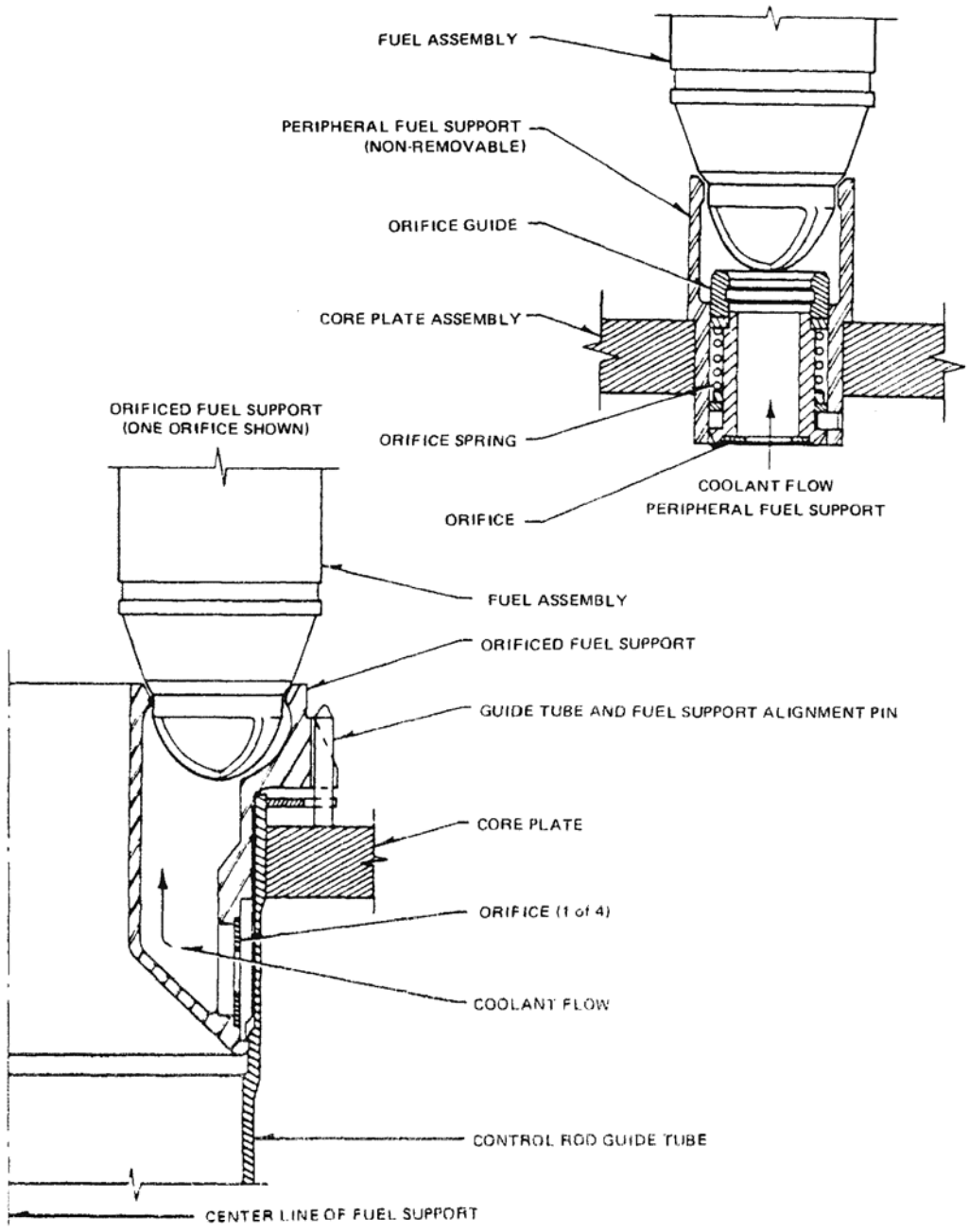
FIGURE 3.9-7



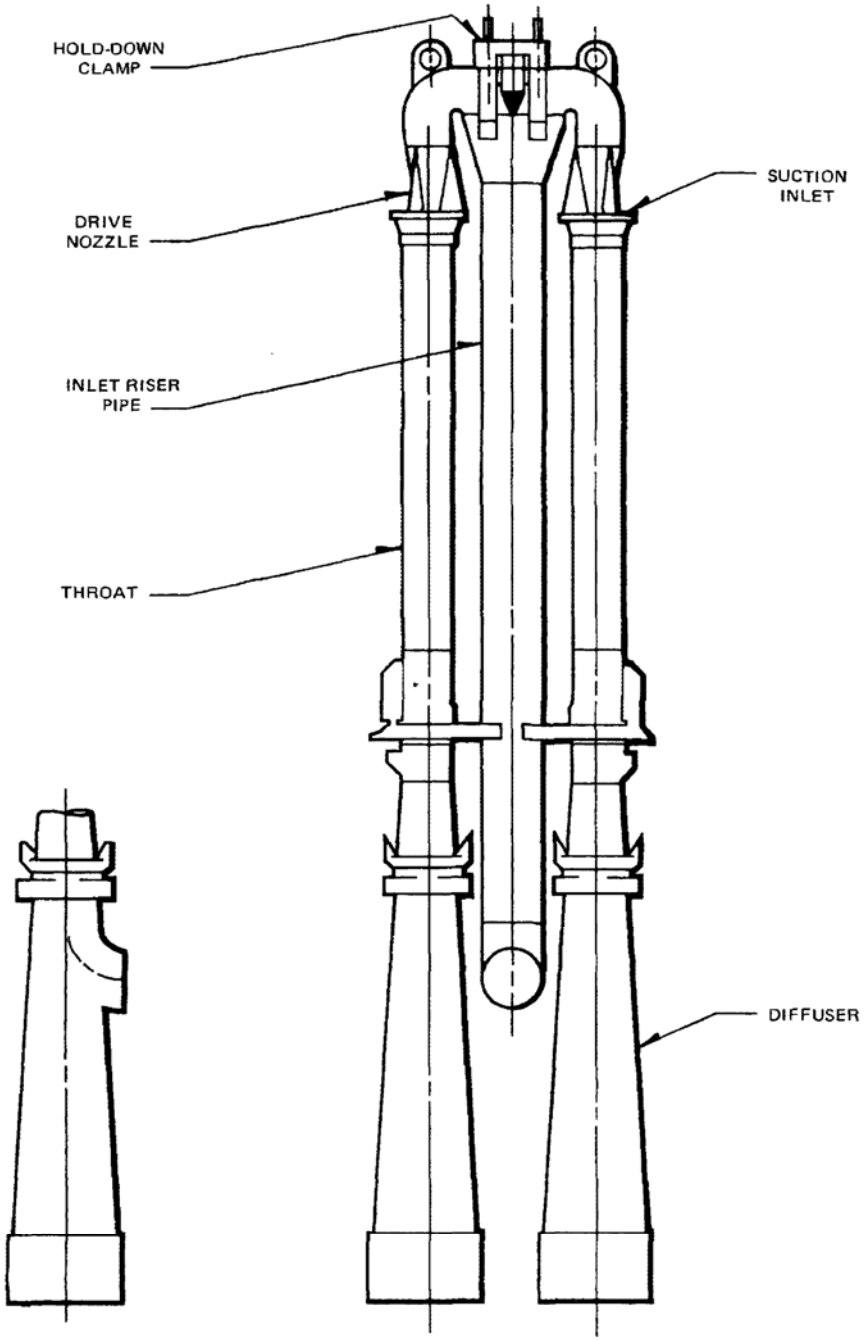
MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT	REACTOR VESSEL CUTAWAY FIGURE 3.9-8



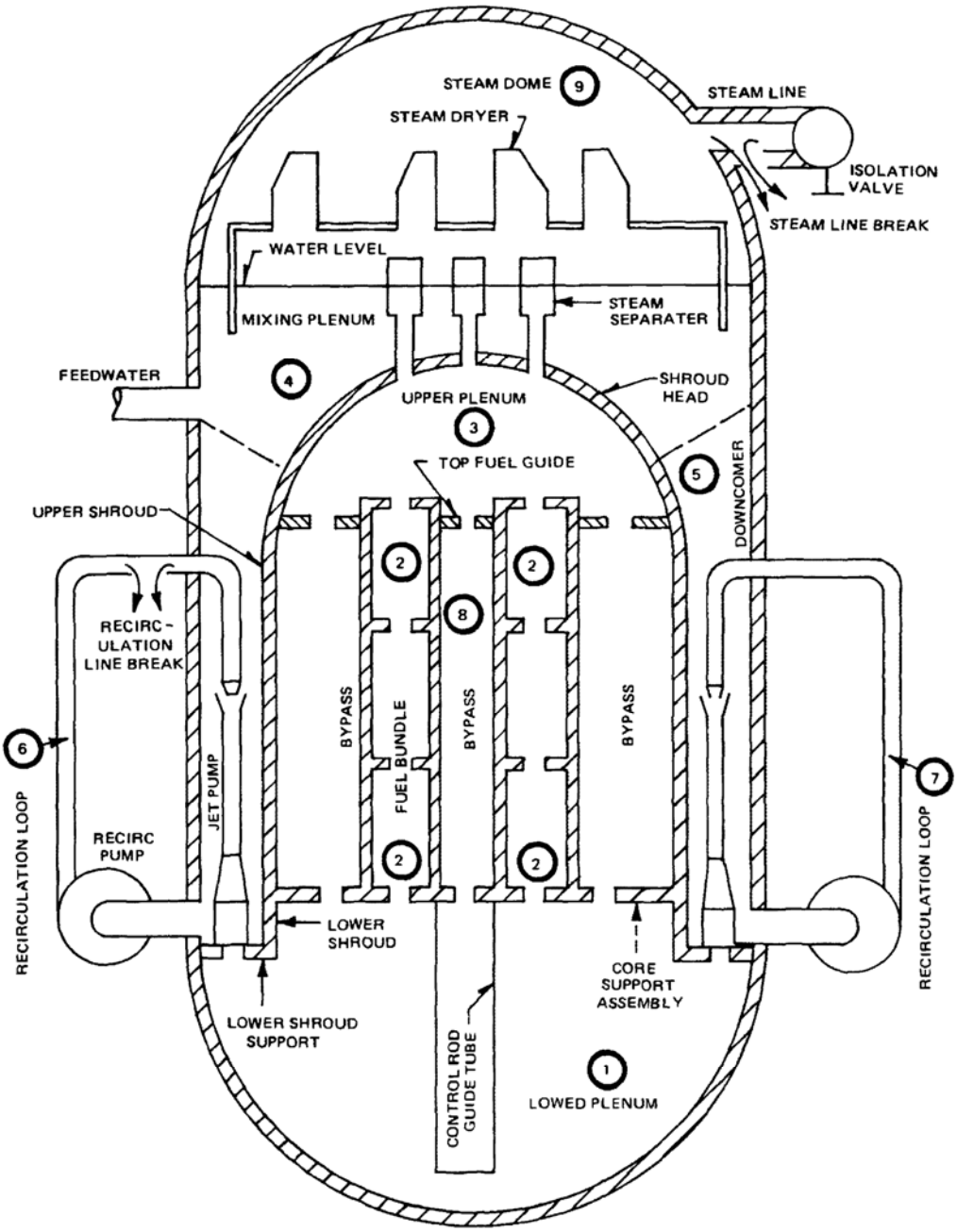
MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT	REACTOR INTERNALS FLOW PATHS FIGURE 3.9-9



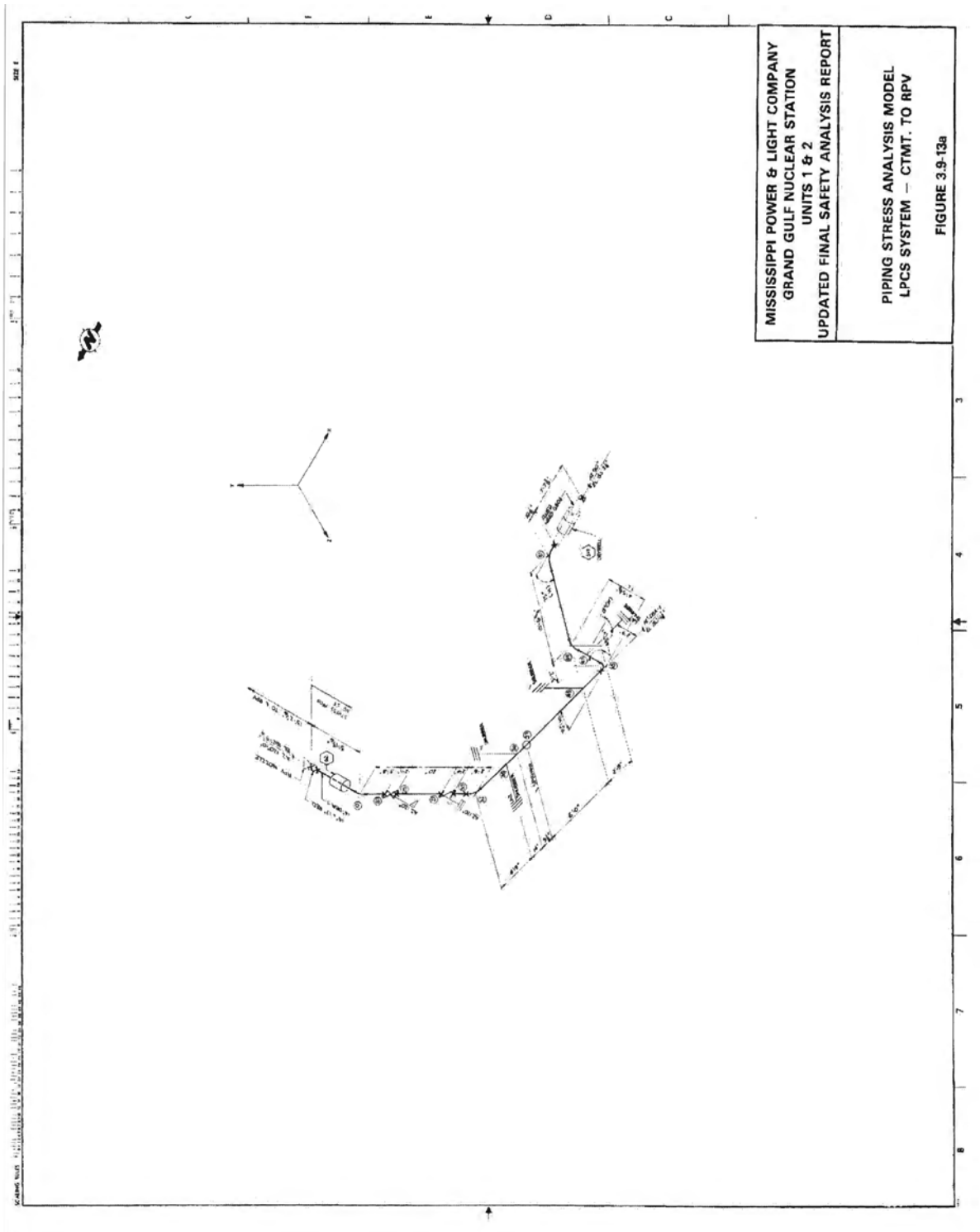
MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT	FUEL SUPPORT PIECES FIGURE 3.9-10

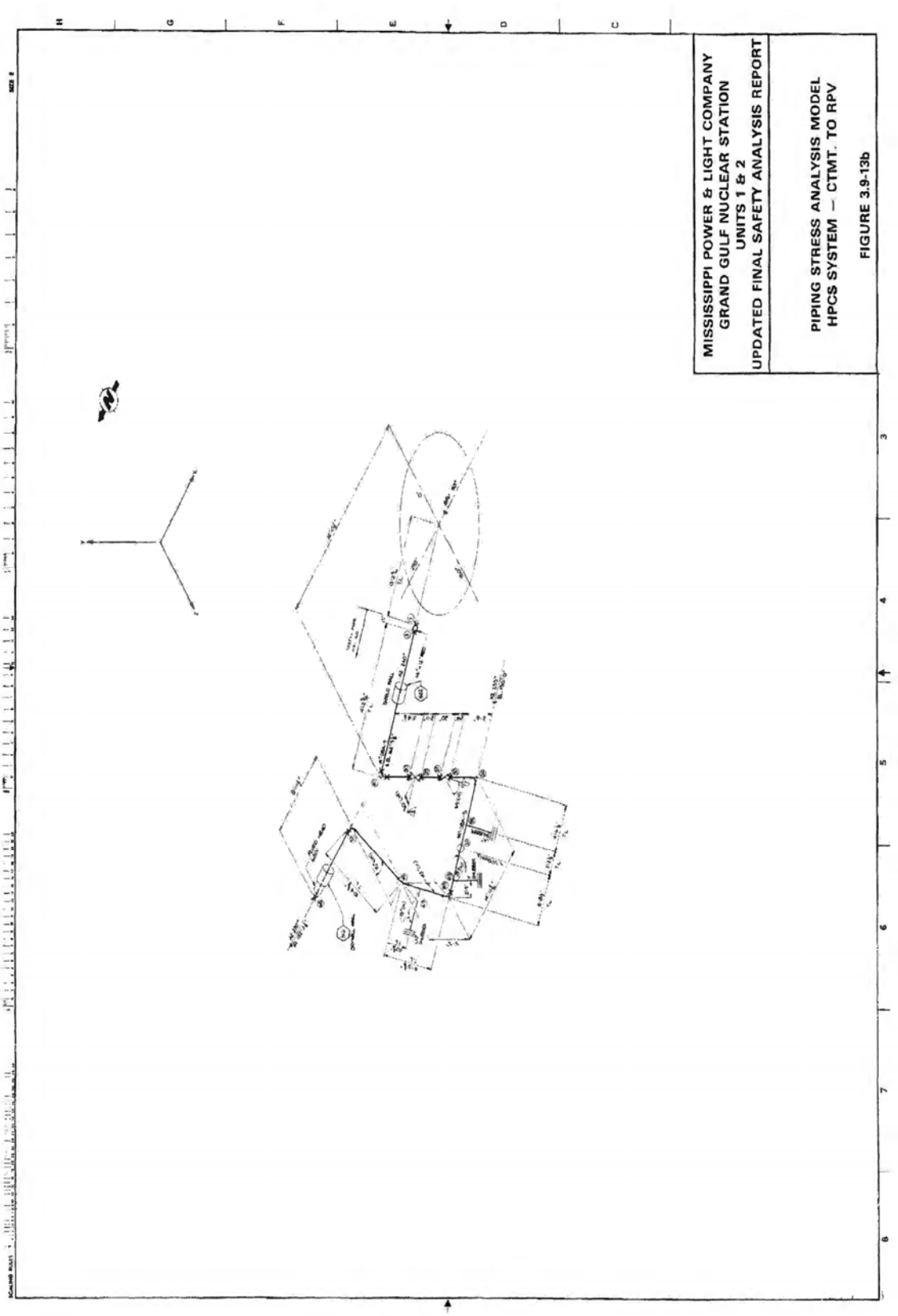


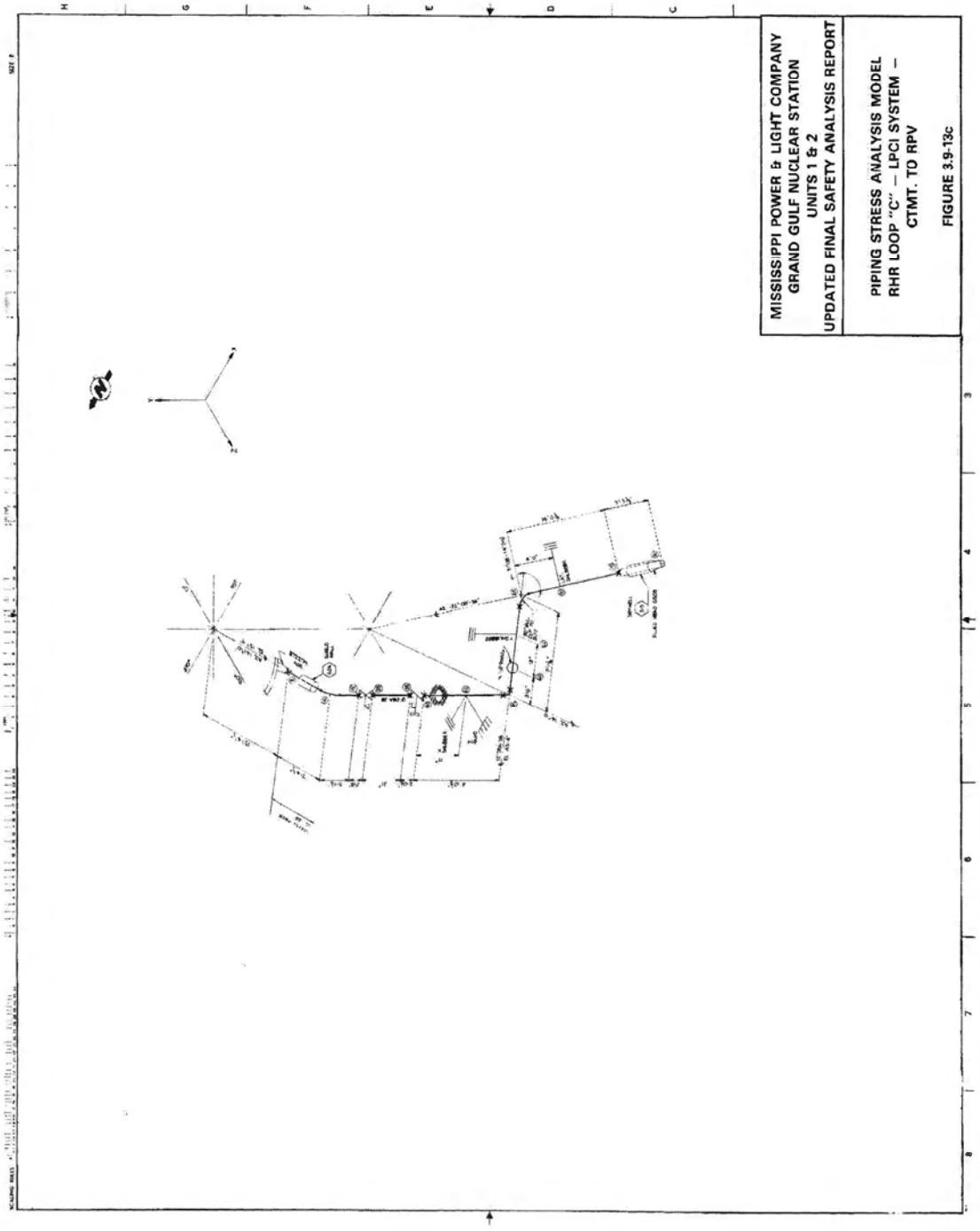
MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT	JET PUMP FIGURE 3.9-11

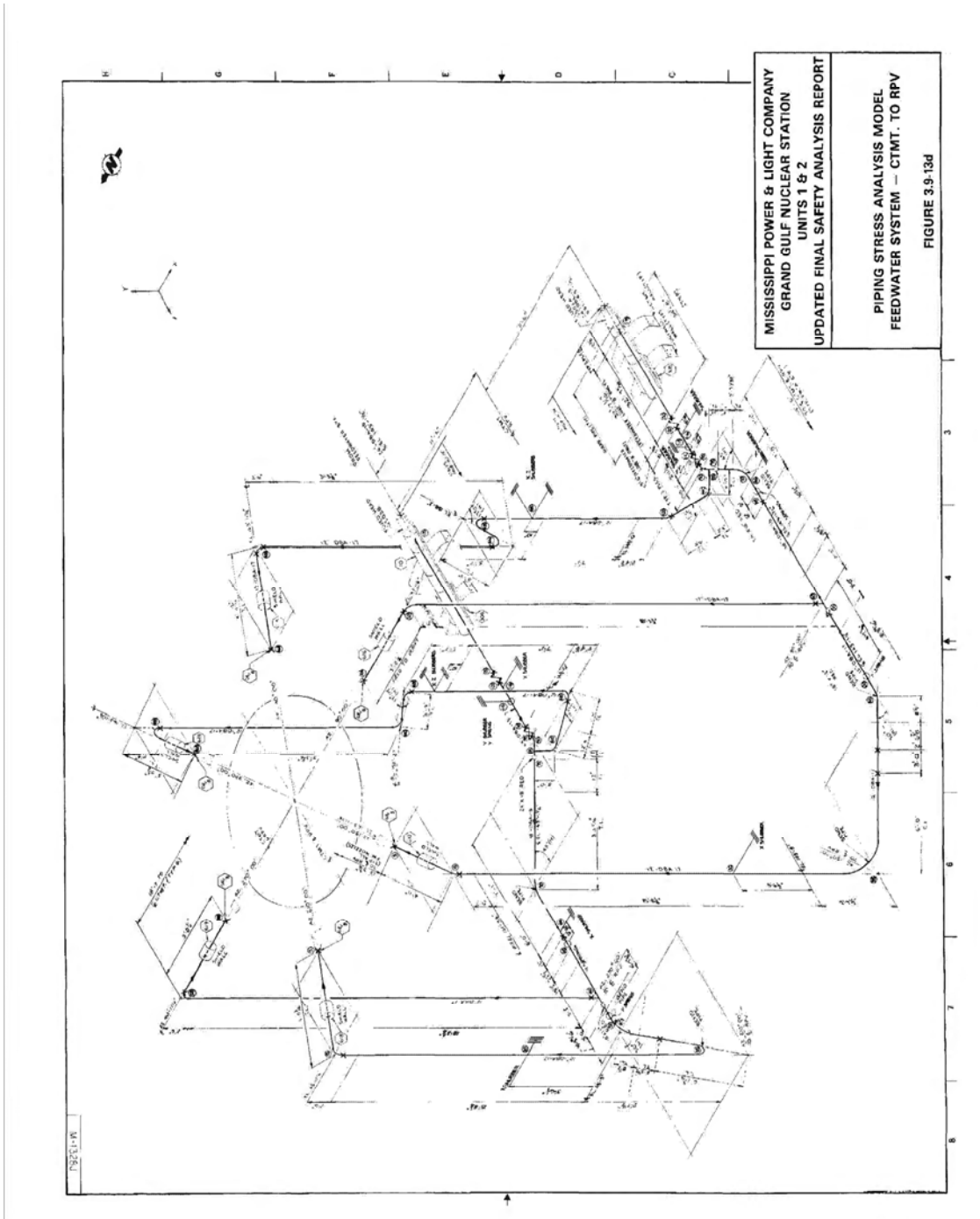


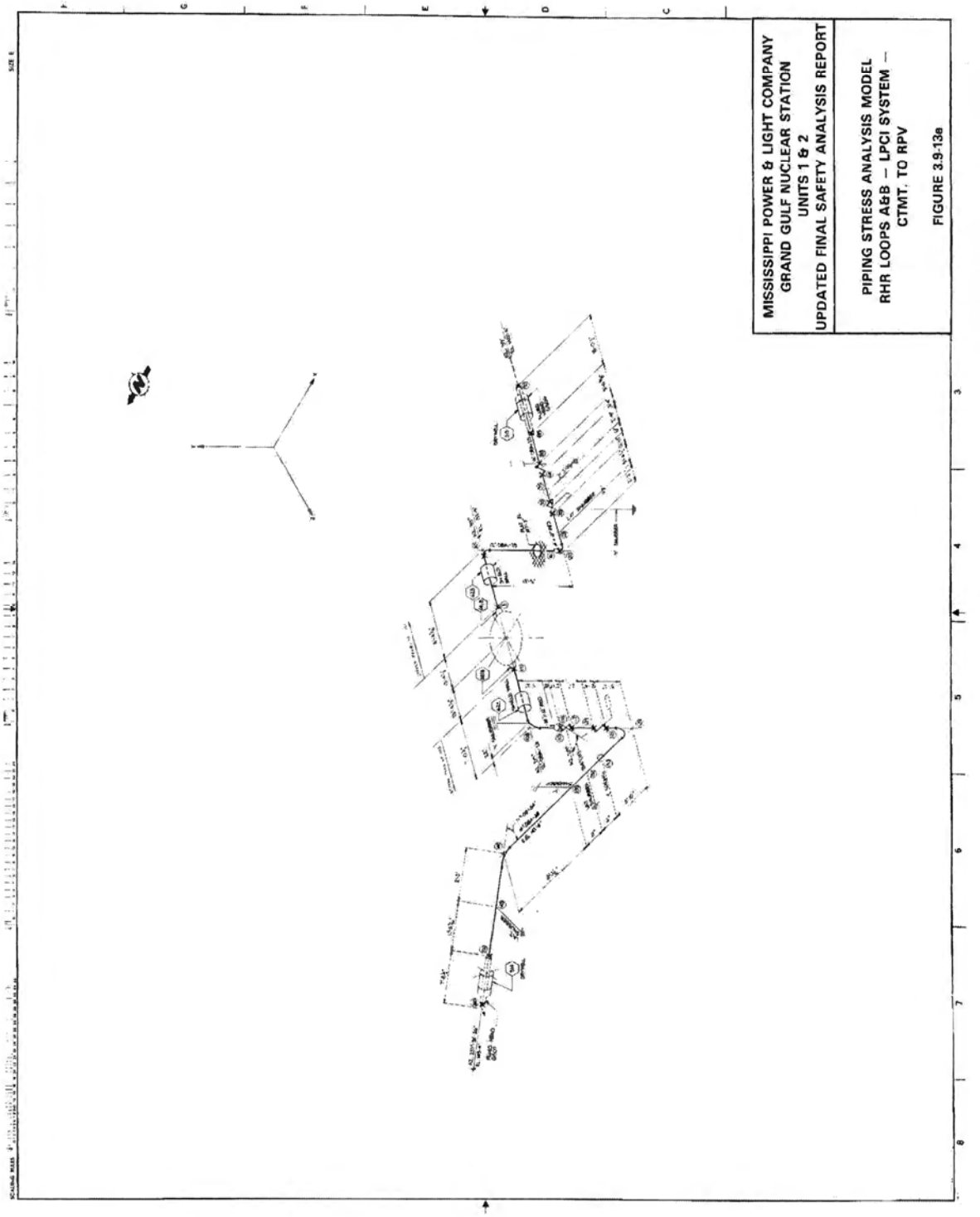
MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT	PRESSURE NODES USED FOR DEPRESSURIZATION ANALYSIS FIGURE 3.9-12


MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT
PIPING STRESS ANALYSIS MODEL
LPCS SYSTEM – CTMT. TO RPV
FIGURE 3.9-13a


MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT
PIPING STRESS ANALYSIS MODEL
HPCS SYSTEM — CTMT. TO RPV
FIGURE 3.9-13b


MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT
PIPING STRESS ANALYSIS MODEL
RHR LOOP "C" – LPCI SYSTEM –
CTMT. TO RPV
FIGURE 3.9-13c


MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT
PIPING STRESS ANALYSIS MODEL
FEEDWATER SYSTEM — CTMT. TO RPV
FIGURE 3.9-13d


MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT
PIPING STRESS ANALYSIS MODEL
RHR LOOPS A&B – LPCI SYSTEM –
CTMT. TO RPV
FIGURE 3.9-13a

3.10 SEISMIC AND DYNAMIC QUALIFICATION OF MECHANICAL AND ELECTRICAL EQUIPMENT

3.10.1 Organization, Definitions, Equipment Identification, Exceptions, Design Criteria

3.10.1.1 Organization

Section 3.10.1 contains general information about seismic and dynamic qualification of equipment. Several important terms denoting safety level such as "safety-related", "Class 1E", "Seismic Category I", "Seismic Category II/I", "PAM", "QF1", "QF2", "QF3", "QF5", and "QP" are defined. Safety related and PAM equipment requiring qualification is identified. Some equipment by virtue of its construction is exempted from the need for formal qualification. Design criteria which apply to seismic and dynamic equipment qualification are listed and discussed.

Section 3.10.2 outlines the methods and procedures by which qualification was initially established and by which qualification is maintained throughout the operation license stage of GGNS. The section includes actions taken by vendors providing original NSSS and non-NSSS safety related equipment and those continuing actions by Entergy Operations to maintain this qualification and to establish qualification for new or replacement equipment and any PAM equipment requiring seismic qualification.

Section 3.10.3 addresses the seismic analysis and/or testing of support structures to which safety-related and seismically qualified PAM equipment is mounted.

3.10.1.2 Definitions

3.10.1.2.1 Mechanical and Electrical Equipment

This group of equipment includes all safety related electrical, mechanical, instrumentation, and control devices.

3.10.1.2.2 Safety-Related

Safety-related equipment or components are defined as equipment or components necessary to ensure the integrity of the reactor coolant pressure boundary, the capability to shut down the reactor and maintain it in a safe shutdown condition, or the capability to prevent or mitigate the consequences of accidents

that could result in potential offsite exposures comparable to the guideline exposures of 10 CFR Part 100. This includes equipment associated with systems that are essential to emergency reactor shutdown, containment isolation, reactor core cooling, and containment and reactor heat removal, or otherwise are essential in preventing significant release of radioactive material to the environment. Specifically, the equipment is safety-related if it performs the above functions automatically, if it is used by operators to perform these functions manually, or if its failure can prevent the satisfactory accomplishment of one or more of the above safety functions.

3.10.1.2.3 Class 1E

The safety class assigned to electrical equipment or components which perform a safety-related function as defined in 3.10.1.2.2.

3.10.1.2.4 Seismic Category I

The seismic category assigned to structures, equipment, or components which perform a safety related function as defined in 3.10.1.2.2.

3.10.1.2.5 Seismic Category II/I

The seismic category assigned to structures, equipment, or components that are not seismic Category I but whose collapse, failure, or malfunction could result in loss of required function of seismic Category I structures, equipment, or components. This term generally applies to malfunctions of non-safety related equipment which would cause safety-related equipment to malfunction.

3.10.1.2.6 PAM

Post Accident Monitoring (PAM) equipment as defined in 10CRF50.49 and Regulatory Guide 1.97 covers certain equipment required to monitor post accident conditions as appropriate to ensure adequate safety. Some of this equipment requires seismic qualification to the extent indicated by the applicable regulatory guide category. Table 7.5-2 lists all PAM equipment and indicates whether the device is seismically qualified for performance of its required PAM function.

3.10.1.2.7 QF1

Terminology used to refer to Seismic Category I instrumentation which must retain functional capability during and after an SSE.

3.10.1.2.8 QF2

Terminology used to refer to Seismic Category I instrumentation which must retain functional capability post-SSE only.

3.10.1.2.9 QP

Terminology used to refer to passive instrumentation which are not required to function but must retain pressure and structural integrity.

3.10.1.2.10 QF3

Terminology used to refer to passive instrumentation which are not required to function but must maintain structural integrity in order to maintain the integrity of the Class 1E circuit.

3.10.1.2.11 QF5

Terminology used to refer to passive instrumentation which are not required to function but must maintain structural integrity in order to maintain the integrity of the Class 1E circuit and also retain pressure and structural integrity.

3.10.1.3 Equipment Identification

3.10.1.3.1 Equipment List

All safety-related, Class 1E components are designed to the requirements of Table 3.2-1 and are listed by MPL number in the Component Data Base (CDB). The CDB indicates whether a component requires seismic qualification and includes the reference to the Seismic Qualification Central File (SQCF) Qualification Package, which contains the pertinent information on the type of qualification performed. The CDB includes all component MPL numbers listed in the original NSSS and Non-NSSS tables generated for SQRT except as modified to include information pertaining to plant hardware changes made after the original tables were generated.

All PAM equipment requiring seismic qualification is identified in Table 7.5-2.

3.10.1.3.2 Seismic Qualification Central File

Detailed information pertaining to the qualification of seismically qualified devices and PAM devices listed in Table 7.5-2 is maintained in the Seismic Qualification Central File (SQCF). The CDB fields containing the seismic qualification data and the SQCF are maintained by controlled administrative procedures, auditable under the GGNS QA program. All equipment listed in the SQCF has been qualified to an extent commensurate with the required degree of functionality. The SQCF contains qualification summaries which cover each device listed in the CDB as requiring seismic qualification and for seismically qualified devices listed in Table 7.5-2. Each summary references the appropriate qualification documentation. If the documentation is not maintained in the SQCF, it is available from the equipment supplier for review.

3.10.1.4 Exceptions

3.10.1.4.1 Compact Devices

Certain devices by virtue of their small size, low mass, compact construction and/or lack of moving parts, are considered adequate to perform their required function in a seismic/hydrodynamic event without the need to formally demonstrate this capability through testing or analysis. These devices are listed below with a brief description of rationale for this exception. Some devices are also included because they meet the definition of seismically insensitive in EPRI NP-7484, EPRI TR 104871, and EPRI TR 105849.

3.10.1.4.1.1 Terminal Blocks

Exempt due to lack of moving parts and obvious structural integrity based on low mass and the high stiffness associated with vendor supplied mounting hardware.

3.10.1.4.1.2 Fuse Holders - see 3.10.1.4.1.1

3.10.1.4.1.3 Fuses

Exempt due to lack of moving parts. Low mass creates very low inertia forces not sufficient for the fuse to overcome retention forces exerted by the fuse holder.

3.10.1.4.1.4 Check Valves and Other Compact Valves Without Extended Top Works or Power Actuators

Exempt since operability, when required, is not affected by seismic/hydrodynamic events due to compact characteristics and the fact that valve bodies are at least as strong as the system in which they are installed. Inertia forces acting on check valve disks are not considered significant enough to affect operation of the disk.

3.10.1.4.1.5 Signal Resistor Units

Exempt for reasons cited in 3.10.1.4.1.1. Signal resistor units are essentially a terminal block with resistors installed across certain terminals. The resistors are of such minimal mass that their displacement in a seismic event is not considered a tenable possibility.

3.10.1.4.1.6 Flow Orifices and Flow Elements

Exempt since these are compact devices with no moving parts that are routinely designed to function in the piping system while resisting all applicable piping, seismic, and hydrodynamic loads. This analysis is an integral part of the piping system analysis which considers all loads on the flanges and bodies of these devices. Since this analysis demonstrates piping system structural integrity and since structural integrity ensures operability of the flow orifices and elements, no additional specialized qualification effort is required for these devices.

3.10.1.4.1.7 Thermowells

Exempt since these are compact, low mass devices with no moving parts that are routinely designed to maintain the pressure integrity of the piping system when subject to all anticipated service loads including seismic and hydrodynamic. Since pressure integrity along ensures operability of the thermowells, no additional specialized qualification effort is required for these devices.

3.10.1.4.1.8 Diesel Generator Tachometer Pickup - Exempt for same reason as in 3.10.1.4.1.1

3.10.1.4.1.9 Compact Items Classified as Seismically Insensitive by EPRI TR 105849

- Mechanical O Rings (Static Seal Type) – Static seal O-rings are a type of packing used to prevent or minimize leakage of a fluid through mechanical clearances in either a static or dynamic state for pressures up to the design pressure of the static seal O-rings.
- Motor/Engine Oils – Motor/Engine Oils are petrochemical fluids use for motor/engine lubrication.
- Printed Circuit Board Items – Printed circuit board (PCB) items include most electrical components that are mounted on the board. This includes items such as: resistors, diodes, rectifiers, varistors, and integrated circuits.
- These items shall have low mass and small size relative to their host PC board assembly.
- Solenoid Coils – a solenoid coil consists of magnetic wire spirally wound in a cylindrical form and encapsulated with epoxy or silicone varnish. Two lead wires or spade terminals extend from the coil for electrical terminations.
- Indicator Bulbs - The indicator bulb including the electrical connections required for electrical and control panels.
- Indicating Light Assemblies - An indicating light assembly is a lamp holder including the electrical connections and mounting but excluding bulbs and contact blocks. The assemblies may be either porcelain or phenolic with metallic pin contacts, wire leads and solder, screws or quick connect terminals. Assemblies hold bulbs in place with screw threads, speed nut friction clips, or expandable latching fingers.
- Mechanical Seals - Mechanical seals prevent leakage of fluid or gas between moving surfaces on shafts such as pumps and large valves. Mechanical seals consist of at least a single and sometimes multiple stage sealing faces.
- Filter Elements - filter elements cover types of cartridge, roll, or wafer type filters. They can be either HVAC type, inline on a piping system, or contained in a stand-alone container or vessel. Typical materials for filters are glass fibers, nylon, paper, cotton, and rayon (metal elements are excluded from this evaluation). The filter cartridge may

contain a metal mesh surrounding the element to maintain shape and protect the element from larger particles. This evaluation overs only the filter element itself and not the housing.

- Diaphragm - any thin membrane which provides a seal between two mechanical components. typically used in valves, regulators, pressure switches, and transmitters

3.10.1.4.2 Non-NSSS Class IE Equipment Not Tested for Spurious Operation During Seismic Testing

3.10.1.4.2.1 Electronic Signal Isolating Devices

These devices serve to isolate Class IE circuits from non-Class IE circuits. In an SSE event, their total function is to isolate. [HISTORICAL INFORMATION] [They were successfully tested functionally before and after SSE testing and each was tested with 1,500 volts directly applied.] These devices are considered free from a capability for causing a spurious signal into the Class IE system.

3.10.1.4.2.2 Deleted

3.10.1.4.2.3 Pitot Tube Flow Elements

Pitot tubes are analytically qualified for surviving an SSE event. There are no moving parts. No spurious signals are expected due to a fault with these devices.

3.10.1.4.3 Deluge and Sprinkler Systems

The control systems for the automatic actuation of the charcoal filter train deluge systems have been removed and the systems are manually activated after a high ambient temperature is detected. (Refer to Section 9.5 for a discussion of the deluge systems.) Therefore, spurious actuation of this non-Class 1E system is not a credible event.

The spurious actuation of the preaction sprinkler system in the diesel generator areas will not adversely affect Class 1E systems. Spurious signals may activate the alarm and flood the sprinkler piping, but no water can be released. High temperatures are required to open the fusible sprinkler heads and release water.

3.10.1.5 Seismic Design Criteria

3.10.1.5.1 Design Criteria for Originally Supplied NSSS Equipment

The seismic criterion used in the design and subsequent qualification of all Class IE instrumentation and electrical equipment supplied by GEH was as follows:

> The Class IE equipment shall be capable of performing all safety-related functions during (1) normal plant operation, during (2) anticipated transients, during (3) design basis accidents, and during (4) post-accident operation while being subjected to, and after the cessation of the accelerations resulting from the SSE at the point of attachment of the equipment to the building or supporting structure.

The criteria for each of the devices used in the Class IE systems depends on the use in a given system. Since GEH supplies many devices for many applications, the approach taken was to qualify the device in all modes in which it was to be used. In this way, the capability of protective action initiation and the proper operation of fail-safe circuits are assured.

Seismic design criteria for originally supplied NSSS mechanical equipment is covered in Section 3.9.2.2.1.

3.10.1.5.2 Design Criteria for Originally Supplied Equipment Other Than NSSS

3.10.1.5.2.1 General

Seismic Category I instrumentation and electrical equipment are designed to withstand the effects of the safe shutdown earthquake (SSE) and to remain functional during and after accident conditions. Seismic design criteria for originally supplied non-NSSS mechanical equipment is covered in section 3.9.2.2.2.

3.10.1.5.2.2 Conformance With IEEE 344-1971

The IEEE Std 344-1971 “Guide for Seismic Qualification of Class I Electric Equipment for Nuclear Power Generating Stations” is a basic part of this subsection except as specified and amended in the following. Paragraph numbers in this subsection conform to

the paragraph numbers listed in IEEE Standard 344-1971 or are paragraphs added by Entergy Operations to provide additional information not in the standard.

1. Scope

All seismic Category I instrumentation and electrical systems and equipment (assemblies and devices) supplied for this plant must withstand the postulated seismic occurrence specified herein. This subsection contains criteria for acceptable methods and procedures for the seismic qualification of seismic Category I equipment. The equipment vendor is responsible for ensuring that the equipment and systems operate safely under the postulated seismic conditions, and he shall verify that the equipment will meet the stated functional requirements for continued operation without any malfunction or loss in function during and after the specified events.

2. Definitions

2.1 Seismic Category I Equipment

Class I and seismic Category I equipment are synonymous as defined in IEEE 344, Paragraph 2.1.

2.2 Design Basis Earthquake

Design basis earthquake and safe-shutdown earthquake are synonymous as defined in IEEE 344, Paragraph 2.2. An additional requirement is imposed on the design and qualification of equipment in the form of an operational basis earthquake (OBE). The OBE is defined as that earthquake that produces the ground motion for which those features of the nuclear power plant necessary for continued operation without undue risk to the health and safety of the public are designed to remain functional. Unless otherwise specified, it is taken as 50 percent of the SSE.

2.3 Natural Frequency

Applicable as published

2.4 Response Spectrum

The following additional definition is applied for specification of equipment performance:

A. Required response spectrum (RRS)

The response spectrum issued by the Buyer as part of his specification for proof testing equipment, which specifies frequency content and severity of vibratory motion.

B. Test response spectrum (TRS)

The response spectrum developed based on the actual motion of the shaker table.

2.5 Floor Acceleration

Applicable as published

2.6 Ground Acceleration

Applicable as published

2.7 Sine Beats

Applicable as published

2.8 Malfunction

Equipment malfunction or functional impairment is the failure of equipment to operate in the same manner in which it would have operated in the absence of a seismic disturbance. For protective systems, malfunction is the loss of capability to initiate a protection action.

3. Procedures

Add to end of paragraph: When the malfunctioning of seismic Category I equipment is considered, testing is the method recommended to verify the functional capabilities.

Analysis without testing may be acceptable only if structural integrity alone can assure the intended function. When a complete seismic testing is impracticable, a combination of tests and analyses may be acceptable.

3.1 Analysis

3.1.1 Add to end of paragraph: The number of masses shall be sufficient to define the dynamic behavior of the equipment (the mathematical model shall be shown even for a single degree of freedom system).

3.1.2 Add to end of paragraph: The natural frequencies as determined shall be assumed to have a minimum variation of 10 percent. The actual variation shall depend on the expected accuracy of the calculations.

3.1.3 Applicable as published

3.1.4 Add the following paragraph to end of existing paragraph: An alternate method of analysis (static coefficient analysis) which allows a simpler technique in return for added conservatism may be used. No determination of natural frequencies is made, but rather, the response of the equipment is assumed to be the peaks of RRS at a conservative and justifiable value of damping. The response is then multiplied by a static coefficient of 1.5 which has been established from experience to take into account the effects of both multifrequency excitation and multimode response. A lower static coefficient may be used if it can be shown to yield conservative results. The seismic forces on each component of the equipment are obtained by multiplying the values of the mass and acceleration and the resulting force distributed over the component in a manner proportional to its mass distribution. The static coefficient analysis is performed in three orthogonal directions, and the resulting seismic effects are combined by means of the square-root-of-sum-of-squares method to obtain the total structural response due to the seismic input.

3.1.5 Applicable as published

3.2 Testing

3.2.1 General

Applicable as published

3.2.2 Devices

3.2.2.1 Add to end of paragraph: Single axis tests are allowed as described above if the tests are designed to conservatively reflect the seismic event at the equipment mounting locations, or if the equipment being tested can be shown to respond independently in each of the three (3) orthogonal axes, or otherwise withstand the seismic event at its mounting locations. This is the case if the coupling is zero or very low or if other justification can be provided.

If the preceding considerations do not apply, simultaneous multiaxis testing will be used on a minimum of two axes (biaxial).

Independent random inputs are preferred and when used, the tests are performed in two steps with the equipment rotated 90 degrees in the horizontal plane for the second step. If in-phase inputs are used, four tests are run: first, with the inputs in-phase; next, with one input 180 degrees out-of-phase; next, with the equipment rotated 90 degrees horizontally and the

inputs in-phase; and finally, with the same equipment orientation but with one input 180 degrees out-of-phase.

3.2.2.2 Applicable as published

3.2.2.3 Applicable as published

3.2.2.3.1 Applicable as published

3.2.2.4 Add to end of paragraph: If it can be shown that the equipment has no resonances, or only one resonance, or that multiple resonances do not have closely spaced frequencies, or are widely spaced and do not interact to reduce the highest level of input excitation that an equipment can withstand without failure, malfunction, or out of tolerance performance, then single frequency tests may be used to fully test the equipment.

Single frequency testing may also be applicable if the input has sufficient intensity and duration to excite all modes to the required magnitude. The testing shall be such that the testing response spectra envelops the corresponding required response spectra of the individual modes.

In most cases, the required response spectrum exhibits broadband characteristics, in which case multiple frequency testing is applicable and is the preferred method of qualification. In addition,

multiple frequency testing is applicable and the preferred method for general qualification as long as the spectrum generated from an analysis of the test table motion equals or exceeds the required response spectrum.

3.2.2.4.1 Add to end of paragraph:

The minimum time duration for each condition and direction shall be 20 seconds, and in no case shall the number of cycles be less than 40. If the location of the device necessitates a longer duration for a conservative test, the above duration should be increased accordingly.

3.2.2.4.2 Applicable as published

3.2.2.4.3 Add to end of paragraph:

Before other tests are used, the test procedure and justification shall be submitted to the buyer for review. If the buyer concurs with the proposed

tests and procedures, theymay be used.

3.2.3 Assemblies

Add to end of paragraph: Biaxial or multiaxial testing may be required to meet the requirements of paragraph 3.2.2.1.

3.2.3.2 Applicable as published

3.2.3.3 Applicable as published

3.2.3.4 Applicable as published

4. Documentation

4.1 General

3.2.3.1

Applicable as published

4.2 Analytical Data

Applicable as published

4.3 Test Data

1. Applicable as published
2. Applicable as published
3. Applicable as published
4. Applicable as published
5. Applicable as published
6. (added) Data analysis and evaluation (including the resultant spectra for the surface upon which the equipment was mounted when tested)
7. (added) Summary and conclusions
8. Certifying signature and data of signature

9. (added) Calibration history of test equipment

4.4 References

Applicable as published

3.10.1.5.2.3 Load Combinations

Seismic Category I equipment located within the containment and drywell will be qualified for the most severe combined seismic and hydrodynamic vibratory loading for the following load combinations:

1. SRV_{ALL} or SRV_x (whichever is controlling) + OBE
2. SRV_{ALL} or SRV_x (whichever is controlling) + SSE
3. SRV_{ADS} + SSE + IBA
4. SSE + DBA

3.10.1.5.2.4 Response Spectra

Parameters used to develop seismic loadings and criteria for seismic Category I structures, systems, and components are described in Section 3.7. From the basic ground input data, a series of response spectra curves at various building elevations was developed. The magnitude and frequency of the SSE loadings that each component will experience are determined by the locations within the plant. These response spectra curves were included in the purchase specifications for seismic Category I equipment and systems. For instrumentation located at various areas throughout the plant, the purchase specification included one response spectrum curve which envelops the response spectra at all locations where the instrumentation is used. The performance requirements of the seismic Category I electrical and instrumentation items and their respective supports are functional.

3.10.1.5.2.5 Operability Classification

Seismic Category I instrumentation is divided into three classifications: 1) equipment that is designed to maintain its functional capability during and after an SSE; 2) equipment that is designed to perform its functional capability post-SSE only; and 3) equipment that, although not required to maintain its

functional capability, is designed to maintain the pressure boundary integrity of the system of which it is a part during and after an SSE. The structural requirements for instrumentation equipment and systems, which are required to maintain pressure boundary integrity, are in accordance with the ASME Code, Section III, 1974, and the Winter 1975 addendum or ANSI Std B31.1, as applicable.

3.10.1.5.2.6 Analysis

Seismic analyses alone, without testing, were performed only on equipment whose functional operability was assured by its structural integrity alone. When complete seismic testing is impractical, a combination of tests and analyses is performed.

3.10.1.5.3 Design Criteria for Maintenance of Existing Qualification and Establishment of New Qualification for NSSS and Non-NSSS Equipment

3.10.1.5.3.1 General

All safety related equipment as defined in section 3.10.1.2.2 and any PAM equipment requiring seismic qualification as identified in Table 7.5-2 must be capable of withstanding the effects of all postulated seismic/hydrodynamic events without compromise of the level of performance required for the equipment to accomplish its function. The design intent of the following standards and regulatory guides is recognized as criteria acceptable for establishment of seismic and dynamic qualification of mechanical and electrical equipment.

IEEE Standard 344-1975
Reg. Guide 1.61, 10/73
Reg. Guide 1.92, 2/76
Reg. Guide 1.100, 3/76

3.10.1.5.3.2 Determination of Qualification Methods

The choice of qualification method depends upon parameters such as function, type, size, shape, and complexity of the equipment. Testing is recognized as the preferred and recommended method for demonstrating seismic/hydrodynamic qualification. However, this is not always practicable, possible or necessary due to various combinations of the above parameters. In this case, some combination of analysis, evaluation, and sub-component testing may be performed to demonstrate qualification. Analysis and

evaluation without testing may only be acceptable for the qualification of equipment if structural integrity of the device alone can ensure the design intended function.

3.10.1.5.3.3 Load Combinations and Required Response Spectra

GGNS unique design response spectra have been generated for numerous locations on safety related structures throughout the plant where equipment requiring seismic/hydrodynamic qualification is located. RRS are maintained for OBE, SSE, SRV actuation, and LOCA events. In addition, RRS are generated on an as-need basis to capture the amplified response of equipment racks, panels, and other substructures attached to the building. There RRS are used in the qualification of new or replacement devices mounted to these sub-structures.

The RRS are utilized in the establishment of the combined effect of loadings from the various seismic/hydrodynamic events. Applicable load combinations are listed in Sections 3.8 and 3.9.

3.10.1.5.3.4 Acceptance Criteria

Efforts to qualify equipment are considered acceptable if the following criteria are met:

a. Approach and method meet the intent of IEEE 344-75.

b. The test/analysis is appropriate for and applicable to the equipment type, model, etc., i.e., consistency exists between the tested/analyzed conditions, design drawings, and field installations.

c. Loading is conservative, i.e., correct RRS used, load combinations correct, amplification of floor spectra considered, conservative damping values used as applicable.

d. Conclusions of qualification testing/analysis are supported by the results of these tests or analyses.

e. Disposition of test anomalies, when applicable, is adequate.

3.10.2 Methods and Procedures Utilized to Establish and Maintain Qualification for NSSS and Non-NSSS Instrumentation, Electrical and Mechanical Equipment

3.10.2.1 Methods and Procedures Utilized to Establish Initial Qualification for Non-NSSS Equipment

Qualification and documentation procedures used for seismic Category I equipment and/or systems meets the provisions of IEEE 344-1971, as supplemented by Branch Technical Position EICSB-10. Discussion regarding the qualification of mechanical equipment is included in Section 3.9.

The methods by which the supplier can qualify equipment are as follows:

a. Type-testing (prototype)

b. Testing

c. Analysis

d. Combination of a. or b. and c.

Results of individual tests and analyses are given in the GGNS Seismic Qualification Central File (SQCF).

3.10.2.1.1 Testing

[HISTORICAL INFORMATION] [Seismic tests were performed by subjecting equipment to vibratory motion that conservatively simulates the SSE at the equipment mounting. Vibratory tests were conducted over the range of 1 to 33 Hz. See Section 3.10.2.1.4 for discussion of hydrodynamic loads.

To the greatest extent practicable, seismic qualification tests on equipment were performed while the equipment was subjected to normal operating loads. Where it was demonstrated that operating loads such as pressure, torque, flow, voltage, current, and thermal did not cause significant stress loads within the equipment, or where such operating loads are not significant to a determination of equipment operability, operation under load was not required. The equipment was observed and evaluated during and after the test for malfunction or failure and, upon completion of the test, was thoroughly inspected for damage.

During seismic qualification testing of multi-component equipment, auxiliary components such as relays, switches, and instruments necessary for proper operation were, as far as possible, mounted in a manner similar to which it was to be installed, and qualification testing was conducted on the entire assembly. This was the preferred procedure. In some other cases, seismic qualification was performed by a combination of tests and analyses. When this procedure was employed, the major component was qualified by analyses, and the motors, operators, and appurtenances necessary for operation qualified by testing. When this procedure was employed, the auxiliary equipment was tested and qualified to the acceleration level at its mounted location, and its equivalent seismic loading was applied to the major component being analyzed.

The input motion was applied simultaneously to the vertical axis and one principal horizontal axis (or two orthogonal horizontal axes). The maximum input motion acceleration was equal to or exceeded the maximum seismic acceleration expected at the equipment mounting.

In some cases, where it was found that equipment had similar characteristics, the test program was based upon testing of a prototype equipment. The test reports, furnished by the equipment supplier, were reviewed by the architect/engineer for compliance.]

3.10.2.1.2 Analysis

Analysis without testing was acceptable only if structural integrity alone could assure the design-intended function. The analysis was performed by the equipment supplier or a qualified consultant and was checked by the architect/engineer.

In the case of large electrical motors used as pump drivers, if it was shown that the motors had no resonances in the frequency range below the high-frequency asymptote, Zero Period Acceleration of the Required Response Spectrum, they were considered rigid and were analyzed statically without testing for the purpose of qualification.

3.10.2.1.3 Combined Analysis and Testing

When the equipment could not be practically qualified by analysis or testing alone because of its size and complexity, combined analysis and testing were used as discussed in subsection 3.10.2.1.1.

3.10.2.1.4 Evaluation of Equipment Other Than NSSS to the Seismic Qualification Review Team Requirements

In response to an NRC letter dated January 27, 1981 requesting information concerning equipment qualification for seismic and hydrodynamic loads at the Grand Gulf Nuclear Station, a review was performed in order to assess the adequacy of the existing non-NSSS qualification based on the NRC Seismic Qualification Review Team (SQRT) requirements. These requirements include:

a. IEEE 344-1975

b. Regulatory Guide 1.92

c. Regulatory Guide 1.100

d. Standard Review Plan 3.9.2

e. Standard Review Plan 3.10

f. Grand Gulf unique building vibratory response to seismic and hydrodynamic loads

Prior to the start of the SQRT evaluation, a list of nuclear safety-related non-NSSS equipment was developed. This list included all safety-related mechanical components and electrical, instrumentation, and control equipment, including valve actuators and other appurtenances of active pumps and valves. This list included the following information:

a. Equipment description

b. System in which the equipment is located

c. Hot standby/cold shutdown requirements

d. Location of the equipment

e. Availability for inspection

f. Vendor/model number

g. Specification number

Equipment qualification results and technical specifications were reviewed in order to assess the adequacy of non-NSSS equipment qualification in light of the SQRT requirements. The qualification information was summarized in tabular form and included the following information:

a. Specification number

b. Qualification method

c. Equipment description

d. Compliance to original qualification criteria

e. Compliance to NRC SQRT criteria and justification

f. Plans for re-evaluation/requalification

All portions of the above lists and tables relevant to the continuing maintenance of required equipment qualification is incorporated into the Seismic Qualification Central File (SQCF) as described in section 3.10.1.3.2.

The majority of the non-NSSS equipment is not located in the containment building and, hence, is not affected by hydrodynamic loads. In most cases, non-NSSS equipment located outside the containment building has been qualified by static analyses or random motion testing. Careful consideration has been given to operability requirements and natural frequency information.

In most cases, non-NSSS equipment located inside the containment building has been qualified by dynamic analysis or random motion testing. The appropriate seismic and hydrodynamic loads have been accounted for in the qualification program. The effects of fatigue on the equipment due to seismic loads have been considered in the tests and analyses which have been performed.

a. In the cases where analyses have been performed on Class 1, 2, and 3 components in accordance with the ASME Code, fatigue effects have been considered. For Class 1 components, the fatigue requirements are described in Section NB-3222.4. The non-NSSS Class 1 equipment has been

analyzed in accordance with this section of the code. For Class 2 and 3 components, fatigue is considered in Section NC-3112.4-C. The following footnote appears in this section:

> "It is recognized that high localized and secondary stresses may exist in components designed and fabricated in accordance with the rules of this subsection; however, insofar as practical, design rules for details have been written to hold such stresses at a safe level consistent with experience."

High localized and secondary stresses are the causes of fatigue failures in equipment. The non-NSSS Class 2 and 3 equipment has been analyzed in accordance with the ASME Code. Since the actual stresses have been found to be less than the allowables and the allowables have been established at a safe level, the equipment is qualified to fatigue criteria.

b. In the cases where testing has been performed, the effects of fatigue have been considered for non-NSSS equipment located outside the containment building. In these locations, only the effects of seismic fatigue needs to be considered. The tests include operating basis and safe shutdown earthquake loading conditions. For non-NSSS equipment located inside the containment building, further study of equipment response due to the hydrodynamic loads was conducted in part to address fatigue concerns related to repetitive safety relief valve (SRV) hydrodynamic loadings.

It was concluded that GGNS safety related equipment located in the containment will not be adversely affected when subject to repetitive SRV-induced dynamic loadings. This conclusion is based on results of the evaluation of the effects of preliminary qualification testing, the results of the SRV in-plant test, and calculation of cumulative damage factors based on the controlling load combination for equipment qualified by analysis.

The report which supports these conclusions was submitted to the NRC via AECM-86/0334, dated October 23, 1986.

3.10.2.2 Methods and Procedures Utilized to Establish Initial Qualification for NSSS Equipment

3.10.2.2.1 Methods of Showing NSSS Equipment Compliance with IEEE Std. 344-1971

a. Scope

The CDB lists all qualified NSSS equipment, indicates the Qualification Package which contains the summary, which indicates the qualification methodology used and gives the results of individual tests and analysis performed.

b. Procedures

GEH-supplied Class IE equipment meets the requirement that seismic qualification should demonstrate the capability to perform the required function during and after the SSE. Both analysis and testing were used but most equipment was tested. Analysis was primarily used to determine the adequacy of mechanical strength (mounting bolts, etc.) after operating capability was established by testing.

1. Analysis

GEH-supplied Class IE equipment performing primarily a mechanical safety function (pressure boundary devices, etc.) was analyzed since the passive nature of their critical safety role usually made testing impractical. Analytical methods sanctioned by IEEE 344-1971 were utilized in such cases.

2. Testing

GEH-supplied Class IE equipment having primarily an active electrical safety function was tested in compliance with IEEE 344-1971, Section 3.2.

c. Documentation

Available documentation verifies that the seismic qualification of GEH-supplied Class IE equipment is in accordance with the requirements of IEEE 344-1971, Section 4.

3.10.2.2.2 Testing Procedures for Qualifying Electrical Equipment and Instrumentation (Excluding Motors and Valve-Mounted Equipment)

[HISTORICAL INFORMATION] [The test procedure required that the devices be mounted on the table of the vibration machine in a manner similar to which it was to be installed. The device was tested in the operating states that it is to be used in performing its Class IE functions and these states were monitored before, during, and after the test to assure proper function and absence of spurious function. In the case of a relay, both energized and deenergized states and normally open and normally closed contact configurations were tested if the relay is used in those configurations in its Class IE functions.

The seismic excitation was a single frequency "continuous" test in which the applied vibration was a sinusoidal table motion at a fixed peak acceleration and a discrete frequency at any given time. Each frequency and acceleration combination was maintained for about 30 seconds except when a resonance search was made. (See IEEE 344-1971, Paragraph 3.2.2.4.1). The vibratory excitation was applied in three orthogonal axes individually with the axes chosen as those coincident with the most probable mounting configuration.

The first step was to search for resonances in each device. This was done since resonances cause amplification of the input vibration and are the most likely cause of malfunction. The resonance search was usually run at low acceleration levels (0.2G) to avoid destroying the test sample in case a severe resonance was encountered. The resonance search was run from 5 Hz to 33 Hz in accordance with IEEE 344 in no less than 7 minutes; if the device was large enough, the vibrations were monitored by accelerometers placed at critical locations from which resonances were determined by comparing the acceleration level with that at the table of the vibration machine. Usually, the devices were either too small for an accelerometer, had their critical parts in an inaccessible location, or had critical parts that would be adversely affected by the mounting of an accelerometer. In these cases, the resonances were detected by visual (strobe light), or audible observation, or performance.

Following the frequency scan and resonance determination, the devices were tested to determine their malfunction limit. The malfunction limit test was run at each resonant frequency as

determined by the frequency scan. In this test, the acceleration level was gradually increased until either the device malfunctioned or the limit of the vibration machine was reached. If no resonances were detected (as was usually the case), the device was considered to be rigid (all parts move in unison) and the malfunction limit was therefore independent of frequency. To achieve maximum acceleration from the vibration machine, rigid devices were malfunction tested at the upper test frequency (33 Hz) since that allowed the maximum acceleration to be obtained from deflection-limited machines.

The above procedures were required of purchased devices as well as those made by GEH. Vendor test results were reviewed and if unacceptable, the tests were repeated either by GEH or the vendor. If the vendor tests were adequate, the device was considered qualified to the limits of the test.]

3.10.2.2.3 Qualification of Valve-Mounted Equipment

The piping analyses establish the response spectra, power spectral density function or time-history characteristics, and develop a horizontal and a vertical acceleration for the pipe-mounted equipment. Pipe-mounted valves are normally furnished with a natural frequency greater than or equal to 33 Hz. Class IE motor-operated valves actuators were qualified in accordance with IEEE 382-1972.

The safety/relief valves, including the electrical components mounted on the valves, are subjected to a dynamic seismic test. This testing is described in subsections 3.9.2.2 and 3.9.3.2.

3.10.2.2.4 Qualification of NSSS Motors

Seismic qualification of the ECCS motors is discussed in subsection 3.9.2.2 in conjunction with the ECCS pump and motor assembly.

Seismic qualification of the standby liquid control (SLC) motor is discussed in subsection 3.9.2.2 in conjunction with the SLC pump motor assembly.

3.10.2.2.5 Evaluation of NSSS Equipment to the Seismic Qualification Review Team Requirements

In response to an NRC letter dated January 27, 1981, requesting information concerning equipment qualification for seismic and hydrodynamic loads at the Grand Gulf Nuclear Station, a review was performed in order to assess the adequacy of the existing NSSS qualification based on the NRC Seismic Qualification Review Team (SQRT) requirements. These requirements include:

a. IEEE 344-1975

b. Regulatory Guide 1.92

c. Regulatory Guide 1.100

d. Standard Review Plan 3.9.2

e. Standard Review Plan 3.10

f. Grand Gulf unique building vibratory response to seismic and hydrodynamic loads

Prior to the start of the SQRT evaluation, a list of nuclear safety-related NSSS equipment was developed. This list included all safety-related mechanical components and electrical, instrumentation, and control equipment, including valve actuators and other appurtenances of active pumps and valves. This list included the following information:

a. Equipment description

b. System in which the equipment is located

c. Hot standby/cold shutdown requirements

d. Location of the equipment

e. Availability for inspection

f. Vendor/model number

g. Specification number

Equipment qualification results were reviewed in order to assess the adequacy of the NSSS equipment qualification in light of the SQRT requirements.

The qualification information was summarized in tabular form and included the following information:

a. Specification number

b. Qualification method

c. Equipment description

d. Compliance to original qualification criteria

e. Compliance to NRC SQRT criteria and justification

f. Plans for reevaluation/requalification

All portions of the above lists and tables relevant to the continuing maintenance of required equipment qualification is incorporated into the Seismic Qualification Central File as described in section 3.10.1.3.2.

Fatigue effects have been considered in the cases where analyses have been performed on Class 1, 2, and 3 components in accordance with the ASME Code. For Class 1 components, the fatigue requirements are described in Section NB-3222.4. The NSSS Class 1 equipment has been analyzed in accordance with this section of the Code. For Class 2 and 3 components, fatigue is considered in Section NC-3112.4-C. The following footnote appears in this section:

> "It is recognized that high localized and secondary stresses may exist in components designed and fabricated in accordance with the rules of this subsection; however, insofar as practical, design rules for details have been written to hold such stresses at a safe level consistent with experience."

Localized and secondary stresses are considered in the fatigue evaluation of equipment. The NSSS Class 2 and 3 equipment has been analyzed in accordance with the ASME Code. Since the actual stresses must be less than the allowables and the allowables have been established at a safe level, the equipment addresses fatigue criteria.

In the cases where testing has been performed, the effects of fatigue have been considered for NSSS equipment located outside the containment building. In these locations, only the effects of

seismic fatigue need to be considered. The tests include operating basis and safe shutdown earthquakes loading conditions. For NSSS equipment located inside the containment building, the building response due to the fatigue loading caused by repetitive safety relief valve (SRV) actuations was investigated.

It was concluded that GGNS safety related equipment located in the containment will not be adversely affected when subject to repetitive SRV-induced dynamic loadings. This conclusion is based on results of the evaluation of the effects of preliminary qualification testing, the results of the SRV in-plant test, and calculation of cumulative damage factors based on the controlling load combination for equipment qualified by analysis.

The report which supports these conclusions was submitted to the NRC via AECM-86/0334, dated October 23, 1986.

3.10.2.3 Methods and Procedures for Maintenance of Existing Qualification and Establishment of New Qualification for NSSS and non-NSSS Equipment

3.10.2.3.1 Documentation

Documentation of qualification for safety-related, Seismic Category I equipment and any PAM equipment requiring seismic qualification is maintained in the Seismic Qualification Central File (SQCF) and is controlled by Engineering Procedures. The SQCF is subdivided into qualification packages which contain as a minimum a qualification summary for equipment covered by the package. Source documentation such as test reports and analyses referenced in the summary are either part of the package or available at the equipment supplier's office. All the above equipment is covered by at least one qualification package. The SQCF is accessed through the CDB which lists equipment numbers and the corresponding qualification package number for components which require seismic qualification.

[HISTORICAL INFORMATION] [Qualification packages need not be maintained for Seismic Category II/I or QP devices. However, the affect of these devices on seismically qualified systems or components is considered in the design process as documented in design calculations and formal in-house correspondence generated in support of the design process.]

3.10.2.3.2 Design Control

Incorporation of seismic qualification design considerations into all design work conducted by Entergy Operations is assured through GGNS administrative procedures. A review is conducted for all design documents which result in changes to the plant facility or which document the existence of a change. The purpose of the review is to determine the affect of the change on existing qualification bases or to determine what new qualification bases may be required. Any changes required to the SQCF or the CDB are incorporated subsequent to implementation of the facility design changes.

3.10.2.3.3 Qualification Methods

Four basic qualification methods are utilized, listed as follows:

a. Testing
b. Analysis
c. Combination of Testing or Analysis
d. Similarity Analysis based on Type Testing or Analysis

Acceptable standards for conducting qualification by the above methods are delineated in IEEE 344-1975 and Reg. Guides 1.61, 1.92, and 1.100. The most appropriate method or methods is chosen based on a knowledge of equipment operation and functionality requirements. Analysis without testing may be acceptable only if structural integrity alone can assure the intended function. When complete seismic testing is impracticable, a combination of tests and analyses may be acceptable.

3.10.3 Methods and Procedure of Seismic Analysis or Testing of Supports of Electrical Equipment and Instrumentation

3.10.3.1 NSSS Equipment (Other Than Motors and Valve-Mounted Equipment)

The Class IE equipment supplied by GEH is used in many systems on many different plants under widely varying seismic requirements. The seismic qualification tests were performed at all frequencies from 5 to 33 Hz. (The actual qualification range was 0.25 to 33 Hz but since test facility capability usually limited the lower frequency test to 5 Hz, a combination of test and analysis was used to assure there were no untested resonances.)

Some GEH-supplied Class IE devices were qualified by analysis only. A sample of such analysis is shown in Appendix 3.10A. Analysis was used for passive mechanical devices and was sometimes used in combination with testing for larger assemblies containing Class IE devices. For instance, a test might have been run to determine if there were natural frequencies in the equipment within the critical seismic frequency range. (See IEEE 344-1971, Paragraph 3.2.2.3.1.) If the equipment was determined to be free of natural frequencies, then it was assumed to be rigid and a static analysis was performed as shown in Appendix 3.10B. (See IEEE 344-1971, Paragraph 3.2.3.4.) If it had natural frequencies in the critical frequency range, then calculations of transmissibility and responses to varying input accelerations were determined to see if Class IE devices mounted in the assembly would operate without malfunctioning. A sample analysis is shown in Appendix 3.10C. In general, the testing of Class IE equipment was accomplished using the following procedure. See section 3.10.2.2.5 for discussion of evaluation of these procedures to IEEE 344-1975 among other standards.

[HISTORICAL INFORMATION] [Assemblies (i.e., control panels) containing devices which have had seismic malfunction limits established were tested by mounting the assembly on the table of a vibration machine in the manner it was to be mounted when in use and vibration testing it by running a low-level resonance search. As with the devices, the assemblies were tested in the three major orthogonal axes. The resonance search was run in the same manner as described for devices. If resonances were present, the transmissibility between the input and the location of each Class IE device was determined by measuring the accelerations at each device location and calculating the magnification between it and the input. Once known, the transmissibilities could be used analytically to determine the response at any Class IE device location for any given input. (It was assumed that the transmissibilities were linear as a function of acceleration even though they actually decrease as acceleration is increased-therefore, a conservative assumption.) As long as the device input accelerations were determined to be below their malfunction limits by a specified margin, the device was considered qualified for that location.]

Figures 3.10-1 through 3.10-4 illustrate the four basic panel types referenced above and show typical accelerometer locations. The results of the seismic tests on the Class IE panels supplied by GEH for GGNS are summarized in the SQCF.

The full acceleration level tests described above disclosed that most of the panel types had more than adequate mechanical strength and that a given panel design acceptability was just a function of its amplification factor and the malfunction levels of the devices mounted in it. Subsequent panels were, therefore, tested at lower acceleration levels and the transmissibilities measured to the various devices as described above. By dividing the devices' malfunction levels by the panel transmissibility between the device and the panel input the panel seismic qualification level could be determined. Several high level tests have been run on selected generic panel designs to assure the conservativeness in using the transmissibility analysis described.

3.10.3.2 Seismic Category I Electrical Equipment and Instrumentation Support Other Than NSSS

3.10.3.2.1 Battery Racks, Instrument Racks, Control Consoles, Cabinets, and Panels

Response spectra for floors where Category I equipment is located have been supplied to the vendor. The vendor submitted test data, operating experience, and/or calculations to verify that the equipment will not suffer any loss of function before, during, or after the specified seismic disturbance. Analysis and/or testing procedures are in accordance with subsection 3.10.2.1 which encompasses IEEE Standard 344-1971 as supplemented by BTP EICSB-10. Evaluation to IEEE Standard 344-1975 is discussed in section 3.10.2.1.4.

In essence, these supports are inseparable from their supported items and were qualified with the items. During testing, the supports were fastened to the test table with fastening devices or methods as used in the actual installation, thereby qualifying the total installation.

3.10.3.2.2 Cable Trays and Conduit Supports

With the exceptions of supports for non-ESF raceway installed in the BOP computer rooms, El. 148' (OC403 and OC404), all cable trays and conduit supports are designed by the response spectrum method. The preceding exceptions are not seismically supported unless non-seismic supports would violate non-seismic criteria. Where non-ESF raceway is installed over ESF raceway (over seismic), that portion of the non-ESF raceway supports is installed in accordance with seismic criteria. Analysis and seismic restraint measures are based on combined limiting values

for static load, span length, and response to excitation at the natural frequency. Restraint against lateral and longitudinal movement to limit excursion uses the structural capacity of the tray or conduit to determine the spacing of the fixed support points. Provisions for differential motion between buildings are made by breaks in the trays and flexible connections in the conduit.

Supports for the main steam line isolation system (subsection 7.2.1.1.4.4.3) and the reactor protection system (subsection 7.2.2.1.2.3.1.2) raceway conduits installed in the turbine building are designed as seismic Category I supports. These supports are designed by the response spectrum method utilizing an enveloping SSE response spectrum curve developed by Bechtel in accordance with Reference 1.

The following criteria are used in the design of cable tray and conduit supports:

a. Cable Tray Support Spacings

   Tray support spacings for horizontal or vertical runs do not exceed 10 feet unless noted otherwise on design drawings (design drawings generally locate supports not more than 8 feet 0 inches apart with a longitudinal tolerance of 1 foot 0 inches to avoid interferences).

   Tray supports provide resistance to three earthquake directions by means of vertical, transverse, and longitudinal support and bracing systems.

b. Support Locations

   Tray and conduit supports are anchored in the following manner:

   1. Support members may be attached to existing structural steel.

   2. Support members may be attached to supplementary steel members spanning between existing floor beams (or between existing floor beams and walls).

   3. Support members may be attached to concrete wall/ columns/floor using:

(a) Embedded steel plates with shear connectors

(b) Steel plates or strut sections attached to concrete with concrete expansion anchors

c. Loads

1. Dead loads: Tray load = 10 lbs/lin ft (including cover plate and connection details)

2. Live loads: Cable load = 50 lbs/lin ft (24-inch wide tray)*

(Dead load plus live load for conduit is based on maximum amount of cables placed in the conduit.)

3. Concentrated live load: 200 lbs placed at any point in the span (for cable tray supports only)

4. Earthquake loads: SSE or OBE

5. Containment loads

(a) SRV (safety relief valve) loads: Loads resulting from discharge of one or several valves (refer to subsection 3.8.3.3.1.1 and Appendix 6A)

(b) LOCA (Loss-of-Coolant-Accident) loads:

(1) DBA - Load due to design basis accident (includes pool swell, vent clearing, chugging, and condensation oscillation - refer to subsection 3.8.3.3.1.5.d)

*25 lb/lin ft live load for 12-inch-wide trays
75 lb/lin ft live load for 36-inch-wide trays
Other widths have proportionate loads.

(2) SBA - Load due to small break accident (includes chugging - refer to subsection 3.8.3.3.1.5.e)

(3) IBA - Load due to intermediate break accident (includes chugging and condensation oscillation - refer to subsection 3.8.3.3.1.5.f)

Parameters used to develop seismic loadings and criteria for Category I cable tray supports are described in Section 3.7. Development of SRV and LOCA loads is in accordance with the method described in Appendix 6A and GESSAR II, Appendix 3B. (UFSAR Appendix 6D).

d. Load Combinations

SRV loads and LOCA loads are combined with earthquake loads by the square-root-of-the-sum-of-the-squares (SRSS) method.

e. Seismic Analysis

1. Regardless of cable tray or bus duct function, all supports are designed to meet seismic Category I requirements. Seismic loads are determined by dynamic analysis using appropriate seismic response spectra.

2. Floor Response Spectra

(a) Floor response spectra to be used are those generated for the supporting floor. In case supports are attached to the walls or to two different locations, the upper bound envelope spectra obtained by superimposing are used.

(b) In many cases, to facilitate the design, several floor response spectra are combined by an upper bound envelope obtained by superimposing.

f. Special Considerations

For cable tray support members which are stress-free or in tension during normal operating conditions, the minimum safety factor for compression is 1.1.

3.10.3.3 Local Instrument Supports

For field-mounted instruments, the following is applicable:

a. The mounting structures for seismic Category I instruments have a fundamental frequency of 33 Hz or higher which corresponds to the maximum floor acceleration or zero period acceleration.

b. The stress level in the mounting structure does not exceed the material allowable stress when the mounting structure is subjected to the maximum acceleration level for its location.

3.10.3.4 Instrument Tubing Supports

The following bases are used in the seismic design and analysis of instrument tubing supports:

a. All instrument tubing supports are qualified by the response spectrum method.

b. Analysis and seismic restraint measures for tubing supports are based on combined limiting values for static load, span length, and computed seismic response.

c. The seismic Category I instrument tubing systems are supported so that the allowable stresses permitted by Section III of the ASME Boiler and Pressure Vessel Code are not exceeded when the tubing is subjected to the loads specified in subsection 3.9.2.1 for Class 2 and 3 piping.

3.10.4 Operating License Review

3.10.4.1 NSSS Control and Electrical Equipment (Other Than Motors and Valve-Mounted Equipment)

Summaries of all tests and analyses performed to demonstrate seismic/dynamic qualification of NSSS equipment are maintained in the GGNS Seismic Qualification Central File. In some cases copies of the actual test report or analysis are available in this on-site file. Reports not maintained on-site are maintained in a permanent file by GEH and can be readily audited in all cases. The equipment used in Class IE applications passed the prescribed tests. [UNNECESSARY INFORMATION] [Where equipment failed to pass the tests, it was rejected. In some cases, equipment which failed one test was modified or repaired to meet the performance requirements and retested. If the retested equipment passed the latter test, it could be used in a Class IE application.]

3.10.4.2 NSSS Motors

Seismic qualification test results for the ECCS motors are discussed in subsection 3.9.2.2 in conjunction with the ECCS pump and motor assembly. Seismic qualification test results for the standby liquid control (SLC) motor is discussed in subsection 3.9.2.2 in conjunction with the SLC pump motor assembly.

3.10.4.3 Valve-Mounted Equipment

The safety/relief valves, including the electrical components mounted on the valve, are subjected to dynamic tests. The results of these tests are discussed in subsections 3.9.2.2 and 3.9.3.2.

3.10.4.4 Class IE Equipment Supplied by Vendors Other Than GEH

The results of all seismic tests and analyses performed by outside vendors required review and approval by Bechtel or GGNS personnel prior to release for shipment. These results accompanied the equipment in shipment and have become a permanent onsite record. Summaries of seismic test and/or analysis results are maintained in the SQCF. These summaries show that all seismic Category I equipment has satisfactorily passed seismic qualification. Actual test reports and analyses for non-NSSS equipment are also maintained in the SQCF. Where this is not the case, reports are available from the non-NSSS supplier, Bechtel Power Corporation.

3.10.5 Reference

1. "Standardized Seismic Design Spectra for Nuclear Plant Equipment," by N. C. Tsai and W. S. Tseng. Transaction of the 4th International Conference on Structural Mechanics in Reactor Technology, San Francisco, California, August 1977.

TABLE 3.10-1: Deleted

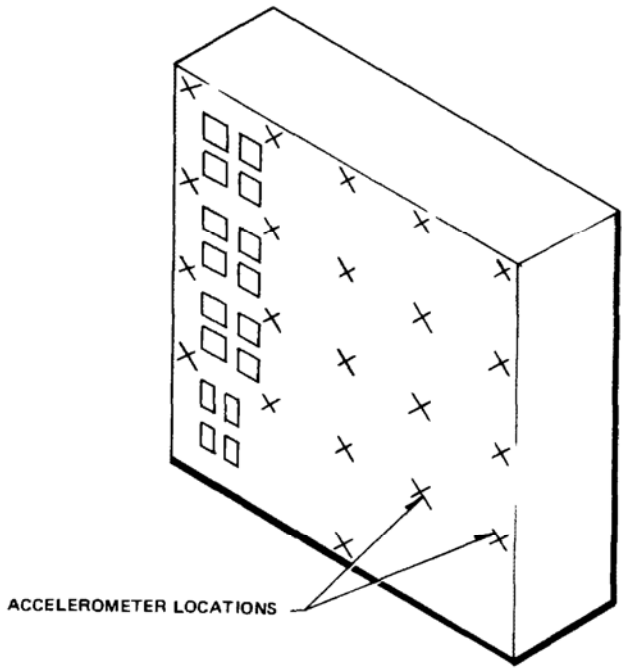
TABLE 3.10-2: Deleted

TABLE 3.10-3: Deleted

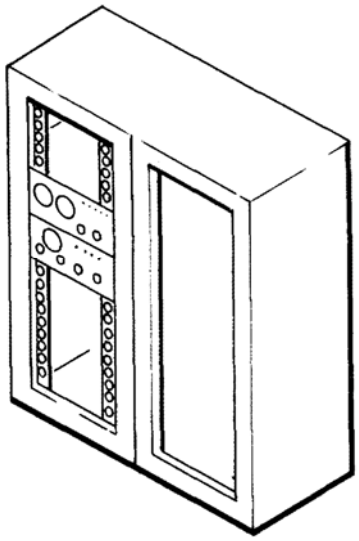
TABLE 3.10-4: Deleted

TABLE 3.10-5: Deleted

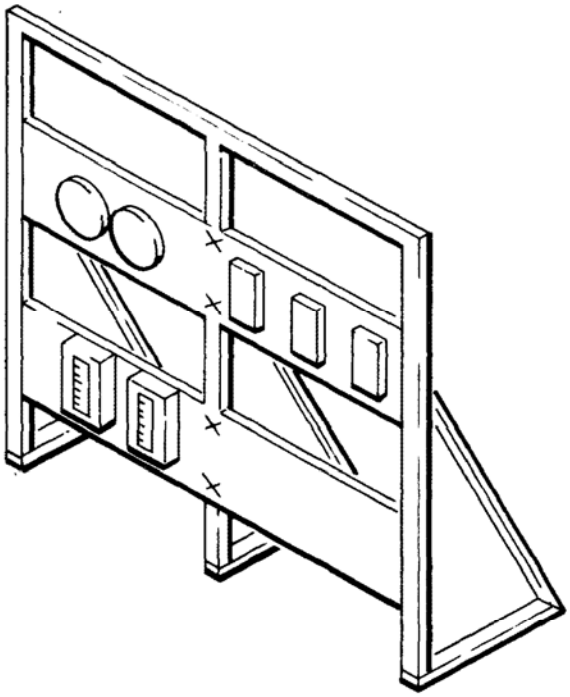
TABLE 3.10-6: Deleted



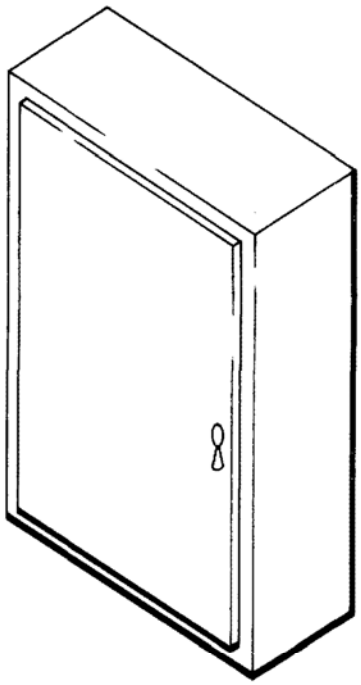
MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT	TYPICAL VERTICAL BOARD (BENCHBOARD WOULD BE THE SAME WITH A BENCH SECTION PROTRUDING OUT ABOUT HALF-WAY DOWN) FIGURE 3.10-1



MISSISSIPPI POWER & LIGHT COMPANY **GRAND GULF NUCLEAR STATION** **UNITS 1 & 2** **UPDATED FINAL SAFETY ANALYSIS REPORT**	INSTRUMENT RACK (CABINET WOULD CONTAIN PAGES OR OTHER SPECIAL INSTRUMENTS INSTEAD OF SIMPLY DRAWER TYPE INSTRUMENTS (BWR 4, 5, 6) FIGURE 3.10-2



MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT	TYPICAL LOCAL RACK (PIPING AND OTHER EXTERNAL CONNECTIONS NOT SHOWN) BWR 4, 5, 6 FIGURE 3.10-3



MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT	NEMA TYPE-12 ENCLOSURE (INSTRUMENTS MOUNTED INSIDE INTERNAL MEMBRANE MOUNTED ON STANDOFFS ATTACHED TO BACK) FIGURE 3.10-4

APPENDIX 3.10A DYNAMIC ANALYSIS BY RESPONSE SPECTRUM METHOD NSSS EQUIPMENT

The system stiffness and mass matrices are generated using standard techniques. A seismic analysis is performed using the following equations of motion and procedure to uncouple these equations:

The equations of motion in matrix form are as follows:

$$M(\ddot{X}+\ddot{Y})+C\dot{X}+KX=O \qquad (1)$$

where

M = mass matrix, n x n (this includes the hydrodynamic mass)

X = column vector of displacement relative to ground (n x 1)

C = damping matrix (n x n)

K = stiffness matrix (n x n)

$\ddot{Y}$ = column vector of ground accelerations (n x 1)

• = first derivative with respect to time

¨ = second derivative with respect to time

It should be noted that for equipment containing fluid, a hydrodynamic mass coupling exists between real structural masses. This hydrodynamic mass appears as diagonal and off-diagonal terms in the mass matrix. The overall system stiffness matrix K is determined by either the matrix force method or the matrix displacement method. The resulting stiffness matrix is similar.

Removing the driving-point acceleration vector to the right side of equation (1), the equation reduces to the classical form:

$$M\ddot{X}+C\dot{X}+KX=-M\ddot{Y} \qquad (2)$$

In order to uncouple equation (2), we set:

$$X=\phi q \qquad (3)$$

$$M\phi\ddot{q} + C\phi\dot{q} + K\phi q = -M\ddot{Y} \qquad (4)$$

Pre-multiplying (4) by ϕ^T, the transpose of ϕ, and performing the coordinate transformation described in (4) such that ϕ is defined by the following orthogonality conditions:

$$\phi^T M\phi = I \qquad (5)$$

$$\phi^T K\phi = \left[\omega^2\right] \qquad (6)$$

where I is an identify matrix (n x n) and $\left[\omega^2\right]$ is a diagonal matrix of the eigenvalues. Then (4) becomes

$$\phi^T M\phi\ \ddot{q} + \phi^T C\phi\dot{q} + \phi^T K\phi q = \phi^T M\ddot{Y} \qquad (7)$$

$$\ddot{q} + \phi^T C\phi\dot{q} = \omega^2\ \ q = -\phi^T M\ddot{Y} \qquad (8)$$

The above procedure for uncoupling the equation of motion by using the modal matrix of the undamped system assumes that damping in the system is small. It will further be assumed that the damping matrix C is such that $\varphi^T C\varphi$ is a diagonal matrix. The elements of this diagonal-matrix are the modal damping values.

With the above assumptions, equation (8) may be written in the following uncoupled form:

$$\ddot{q}_i + 2\ \beta_i\ \omega_i\dot{q}_i + \omega_i^{\ 2}\ q_i = S_i\ \ddot{U}_g$$

$$i = 1,\ 2,\ ---\ n \qquad (9)$$

where

$$X_i = \begin{bmatrix} X_{1i} \\ X_{2i} \\ . \\ . \\ . \\ . \\ X_{ni} \end{bmatrix} \qquad \phi_i = \begin{bmatrix} \phi_{1i} \\ \phi_{2i} \\ . \\ . \\ . \\ . \\ \phi_{ni} \end{bmatrix}$$

The maximum physical displacement for each mass is then taken to the square root of the sums of the squares of each of the maximum displacement responses for each mode, i.e.,

$$(X)\ \text{max.} = \left[\sum_{j=1}^{n} x_{ij}^{2} \right]^{1/2}, \quad i = 1, 2, \ldots m$$

where: (X) maximum is the column vector of maximum displacements. Similarly, the maximum load response for the i^{th} mode is found from

$$L_{ji} = \beta_j \; X_i$$

$$L_{ji} = \begin{bmatrix} L_{1i} \\ L_{2i} \\ . \\ . \\ . \\ L_{mi} \end{bmatrix}$$

where

β_j is the stress matrix for element j, j=1,...m

m = total number of elements.

where

β_i = damping ratio for the i^{th} mode expressed as percent of critical damping

ω_i = i^{th} natural angular frequency of the system

S_i = modal participation factor the i^{th} mode = $-\phi_i^t M D$

$\ddot{U}_g$ = ground or floor acceleration time history

ϕ_i^t = transpose of the i^{th} mode shape

D = earthquake direction vector

The response is calculated using the response spectra specified for the location of the input to the analytical model. The analytical procedure is described briefly in the following paragraphs.

The system of one-degree-of-freedom equations represented by equation (8) or (9) can be solved by the response spectrum method. With this method, the maximum modal response for each natural frequency of interest is found from the applicable response spectra. Response spectrum curves are essentially plots of the maximum responses of single-degrees-of-freedom systems described by equation (9) with $S_i = 1.0$ as a function of their natural frequencies.

Having found the maximum modal displacements, q, i = 1...m, the maximum physical displacement for the i^{th} mode is given by:

$$X_i = \phi_i S_i q_i$$

The maximum load response is taken to be the square root of the sums of the squares of each of the maximum responses for each mode, i.e.,

$$(L_j)\ \max = \left[\sum_{i=1}^{n} L_{ji}^2 \right]^{1/2}, \quad j = 1, 2, \ldots m$$

where (L) max. is the column vector of maximum loads.

The accelerations for each mode are determined by multiplying the displacements vector for that mode (X_i) by the natural frequency (w_i^2) of that mode.

$$A_i = X_i \ \omega_i^2$$

The maximum accelerations are then determined by

$$(A)\ \max = \left[\sum_{i=1}^{n} A_i^2 \right]^{1/2}$$

APPENDIX 3.10B SAMPLE SEISMIC STATIC ANALYSIS NSSS EQUIPMENT

PART I

Part I presents a set of curves from which static seismic analysis of standard enclosures can be quickly performed. A standard enclosure is any enclosure listed in the Enclosure Standards Manual. The enclosures are assumed to be floor mounted, using all mounting holes with 5/8 inch steel bolts or studs each having an effective area of 0.2256 in^2. Using an elastic limit of one-half the ultimate strength, the bolts are assumed to have a maximum safe tension stress and maximum safe shear stress of 28,000 psi and 21,000 psi, respectively. The curves are based on a design basis earthquake having a horizontal acceleration of 1.5G and a vertical acceleration of 0.5G. It is assumed that each enclosure is mounted alone and not coupled directly to any other enclosure.

The static analysis consists of determining the maximum allowable safe weight of the enclosure and its components for which the mounting bolt stresses are not exceeded. The curves of Figure 3.10B-1 have been derived for this purpose. To use the curves given in Figure 3.10B-1, first determine from Table 3.10B-1 the curve designation of the enclosure being considered. Next, using the corresponding curve in Figure 3.10B-1, determine the maximum safe weight per bolt for a given height of the center of gravity. The maximum safe enclosure weight is then determined by multiplying the weight per bolt by the total number of enclosure mounting bolts. Comparison with the actual weight of the enclosure and its components then indicates whether or not the mounting bolt stresses are exceeded. If the comparison shows that the maximum safe weight per bolt is exceeded, steps should be taken to increase the effective bolt area by welding the enclosure to its mounting, increasing the number of mounting bolts, adding top braces to a wall, or using another appropriate method to ensure safe operation during seismic disturbance.

PART II

Part II presents the necessary assumptions and equations for the calculation of the maximum normal and shear stresses in the mounting bolts of any enclosure under seismic disturbance. The following assumptions and conventions are made:

1. The enclosure under consideration is assumed to be a rigid body in equilibrium with respect to its mounting.

2. The forces on the enclosure due to seismic accelerations are assumed to act through the enclosure's center of gravity.

3. The enclosure is assumed to have a known weight W as well as a known center of gravity located at X, Y, Z with respect to a right-handed coordinate system.

4. The right-handed coordinate system is arbitrarily assumed to be located at the front left-hand lower corner of the enclosure with the positive X-axis to the right along the front edge, the positive Y-axis toward the back of the enclosure, and the positive Z-axis toward the top of the enclosure.

5. The stresses on the enclosure mounting bolts are assumed to be greatest when the horizontal component of the floor acceleration is perpendicular to a side of the enclosure and the vertical component of the acceleration is downward.

6. It is assumed that the enclosure tends to rotate about an axis parallel to either the X-axis or the Y-axis, dependent upon the direction of the horizontal acceleration. The location of the axis of rotation is dependent upon the mounting configuration of the enclosure.

7. There is assumed to be no friction between the enclosure and its mounting.

8. The horizontal shear force due to the horizontal component of the acceleration is assumed to be distributed equally among the mounting bolts.

9. All mounting bolts are assumed to be identical.

The following procedure outlines the equations involved in determining the mounting bolt stresses.

From the geometric configuration of the mounting bolts it is found that the tension forces in the bolts are related by

$$F_i = \frac{d_i}{d_j} F_j \ , \qquad (1)$$

where F_i and F_j are the tension forces acting on the i-th and the j-th bolts, respectively, and d_i and d_j are the perpendicular distances of the i-th and the j-th bolts, respectively, from the axis about which the enclosure tends to rotate. When the enclosure is mounted directly to the floor, the axis of rotation will be an edge of the enclosure. For other mounting configurations, care must be exercised in determining this axis.

Summing moments about the enclosure's axis of rotation, the equation relating the unknown bolt tension forces to known quantities is found to be

$$F_1 d_1 + F_2 d_2 + \ . \ . \ . + F_N d_n = W[A1 \ . \ Z + (A2 - 1)L], \qquad (2)$$

where N is the number of mounting bolts, A1 and A2 are the relative magnitudes of the horizontal and vertical components of the floor acceleration, respectively, and L is the perpendicular distance between the line of action of the vertical acceleration through the center of gravity and the axis about which the enclosure tends to rotate.

Substituting (1) into (2), the j-th tension force is

$$F_J = \frac{d_j \cdot W[A1 \cdot Z + (A2 - 1)L]}{d_1^2 + d_2^2 + \cdots + d_N^2} \qquad (3)$$

The other tension forces are determined using Equation (1).

The tension stress T_i is related to the tension force by

$$T_i = \frac{F_i}{A} \qquad (4)$$

Where A is the effective cross-section area of a mounting bolt.

Summing forces in the direction of the horizontal force acting upon the enclosure and making use of assumptions 7 and 8, the shear stress on the i-th bolt is

$$S_i = \frac{W \cdot A1}{N \cdot A} \tag{5}$$

Due to the combined tension and shear stresses, the maximum tension stress, $(T_i)_{max}$, and the maximum shear stress, $(S_i)_{max}$, present in the i-th bolt are

$$(T_i)_{max} = \frac{T_i}{2} + \sqrt{\frac{T^2_i}{2} + S^2_i} \tag{6}$$

and

$$(S_i)_{max} = \sqrt{\frac{T^2_i}{2} + (S_i)^2} \tag{7}$$

For a detailed derivation of Equations (6) and (7), the reader is directed to Strength Of Materials, by Ferdinand L. Singer, Chapter 9, Section 6.

To apply the above equations to determine the maximum tension and shear stresses, the following is required:

Total Weight	W Pounds
Center of Gravity	X, Y, Z Inches
Horizontal Seismic Acceleration	A1 - G
Vertical Seismic Acceleration	A2 - G
Distance to CG (see eq. (2))	L Inches
Number of Bolts	N
Area Each Bolt	A Square Inches
Bolt Distance from Axis of of Rotation	d_1, d_2 ... d_N Inches

PROCEDURE:

1. Determine the axis about which the cabinet tends to rotate for a given floor motion.

2. Determine, using Equation (3), the tension force acting on the j-th mounting bolts (arbitrarily choose one).

3. Determine the tension forces acting on the remaining mounting bolts from application of Equation (1).

4. Calculate the tension stress acting on each bolt using Equation (4) and the results of Step 3.

5. Calculate the horizontal shear stress from Equation (5).

6. Determine the maximum tension stresses using Equation (6) and the results of Steps 4 and 5.

7. Determine the maximum shear stresses using Equation (7) and the results of Steps 4 and 5.

8. Compare these maximum stresses and allowable stresses of one half the ultimate strength (in psi) for the bolt material.

PART III

I.PURPOSE

The purpose of Part III is to document a static seismic analysis which was performed to verify that the mounting bolts of the standard cabinets are capable of withstanding seismic environment.

II.SCOPE

The scope of this report is limited to the static analysis of the mounting bolt stresses of five (5) standard cabinets. The standard cabinets are:

1. Area Radiation Monitor, 236x400 (911)

2. TIP Control, 236x401 (913)

3. Start-up Neutron Monitor, 236x402 (936)

4. Power Range Monitor, 236x403 (937)

5. Rod Position Information System, 236x404 (927)

III.DISCUSSION

The Seismic Design Guide, 225A4582, was used in conducting the static seismic analysis. Each cabinet was assumed to be floor mounted using 5/8" bolts in all mounting holes. The maximum safe tension stress and maximum safe shear stress were assumed to be 28,000 psi and 21,000 psi, respectively.* The design basis earthquake was assumed to have a horizontal acceleration of 1.5G and a vertical acceleration of 0.5G. The weight of each cabinet was estimated using the weight of each major component listed in the parts lists for each cabinet. The height of the center of gravity of each cabinet was calculated using the weight and center of gravity of each of the major components.

The following data sheets include the necessary information for determining the factor of safety for each cabinet.

_____

*Equal to one-half the ultimate strength as given in Machinery's Handbook, Fourteenth Edition.

SEISMIC DESIGN VERIFICATION DATA SHEET

Cabinet Name: Area Radiation Monitor 236x400 (911)

Applied Horizontal Acceleration	1.5 G
Applied Vertical Acceleration	0.5 G
Tension Stress (Maximum Safe)	28,000 psi
Shear Stress (Maximum Safe)	21,000 psi
Weight of Cabinet	675 Lbs
Number of Mounting Bolts	4
Height of Center of Gravity	48 Inches
Maximum Allowable Weight Per Bolt (From Curve No. C1 on Page 8 of Seismic Design Guide, 225A4582)	830 Lbs/Bolt
Maximum Allowable Cabinet Weight 830 Lbs/Bolt * 4 Bolts =	3,320 Lbs
Factor of Safety = $\frac{\text{Maximum Allowable Weight}}{\text{Weight}}$ =	4.9

Cabinet Name: TIP Control, 236x401 (913)

Applied Horizontal Acceleration	1.5 G
Applied Vertical Acceleration	0.5 G
Tension Stress (Maximum Safe)	28,000 psi
Shear Stress (Maximum Safe)	21,000 psi
Weight of Cabinet	755 Lbs
Number of Mounting Bolts	8
Height of Center of Gravity	50 Inches
Maximum Allowable Weight Per Bolt (From Curve No. C3 on Page 8 of Seismic Design Guide, 225A4582)	1,110 Lbs
Maximum Allowable Cabinet Weight 1,110 Lbs/Bolt * 8 Bolts =	8,880 Lbs
Factor of Safety = $\frac{\text{Maximum Allowable Weight}}{\text{Weight}}$ =	11.7

Cabinet Name: Start-Up Neutron Monitor, 236x402 (936)

Applied Horizontal Acceleration	1.5 G
Applied Vertical Acceleration	0.5 G

SEISMIC DESIGN VERIFICATION DATA SHEET (Continued)

Tension Stress (Maximum Safe)	28,000 psi
Shear Stress (Maximum Safe)	21,000 psi
Weight of Cabinet	1,910 Lbs
Number of Mounting Bolts	12
Height of Center of Gravity	50 Inches
Maximum Allowable Weight Per Bolt (From Curve No. C3 on Page 8 of Seismic Design Guide, 225A4582)	1,110 Lbs/Bolt
Maximum Allowable Cabinet Weight 1,110 Lbs/Bolt * 12 Bolts =	13,320 Lbs
Factor of Safety = $\frac{\text{Maximum Allowable Weight}}{\text{Weight}}$ =	11.9

Cabinet Name: Power Range Monitor, 236x403 (937)

Applied Horizontal Acceleration	1.5 G
Applied Vertical Acceleration	0.5 G
Tension Stress (Maximum Safe)	28,000 psi
Shear Stress (Maximum Safe)	21,000 psi
Weight of Cabinet	4,345 Lbs
Number of Mounting Bolts	40
Height of Center of Gravity	46 Inches
Maximum Allowable Weight Per Bolt (From Curve No. C3 on Page 8 of Seismic Design Guide, 225A4582)	1,210 Lbs/Bolt
Maximum Allowable Cabinet Weight 1,210 Lbs/Bolt * 40 Bolts =	48,400 Lbs
Factor of Safety = $\frac{\text{Maximum Allowable Weight}}{\text{Weight}}$ =	11.1

Cabinet Name: Rod Position Information System, 236x404 (927)

Applied Horizontal Acceleration	1.5 G
Applied Vertical Acceleration	0.5 G
Tension Stress (Maximum Safe)	28,000 psi
Shear Stress (Maximum Safe)	21,000 psi

SEISMIC DESIGN VERIFICATION DATA SHEET (Continued)

Weight of Cabinet	2,500 Lbs
Number of Mounting Bolts	20
Height of Center of Gravity	45 Inches
Maximum Allowable Weight Per Bolt (From Curve No. C3 on Page 8 of Seismic Design Guide, 225A4582)	1,225 Lbs/Bolt
Maximum Allowable Cabinet Weight 1,225 Lbs/Bolt * 20 Bolts =	24,500 Lbs
$\text{Factor of Safety} = \frac{\text{Maximum Allowable Weight}}{\text{Weight}} =$	9.8

IV.CONCLUSION

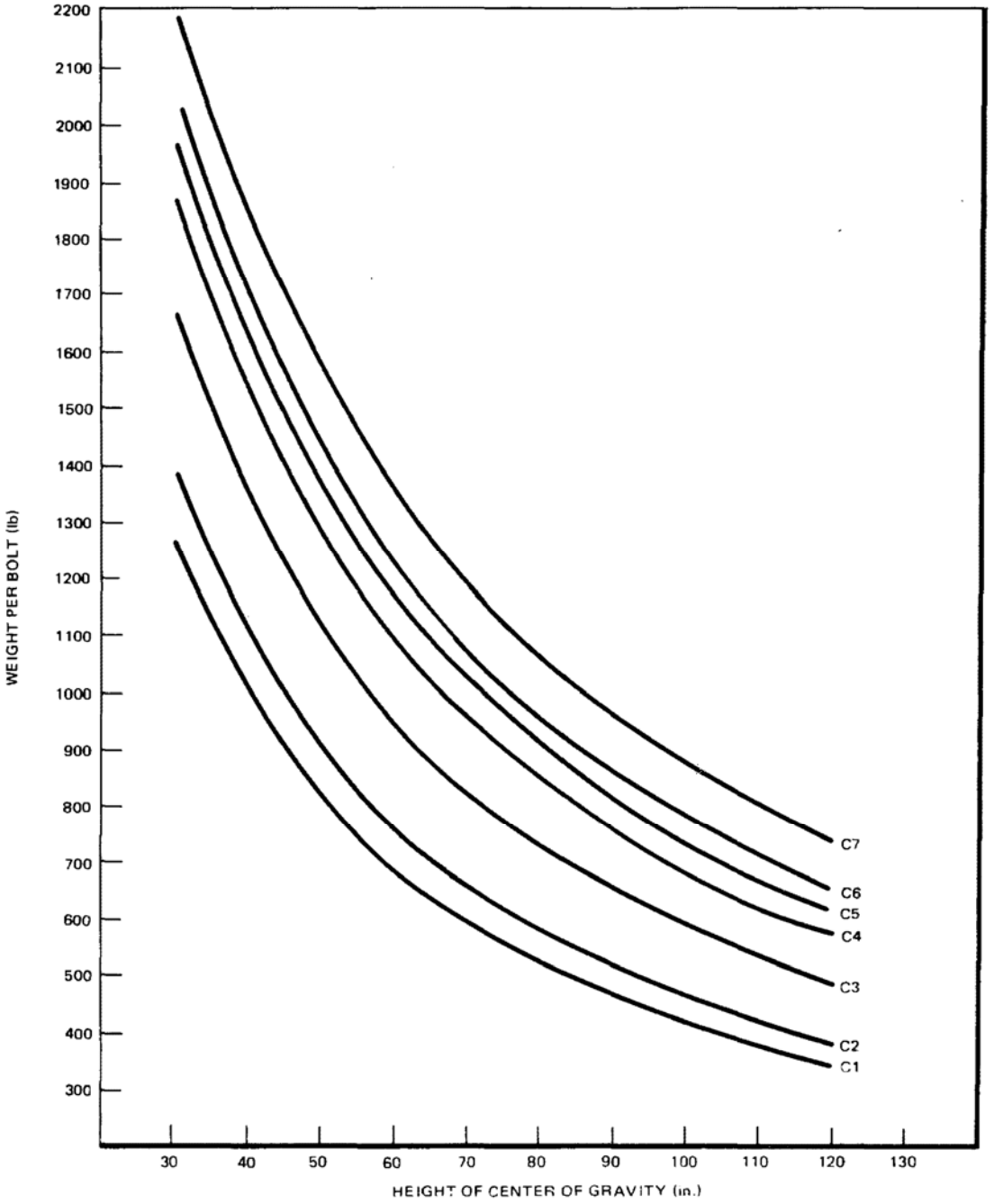
Review of the Factor of Safety of each standard cabinet indicates that the mounting bolts of each cabinet are capable of withstanding seismic disturbance as specified in the Seismic Design Guide.

TABLE 3.10B-1: STANDARD ENCLOSURES

-Curve	Enclosure	Width	Depth	Mode of Failure
C1	Instrument Rack	24"	24"	
	Instrument Rack	24"	30"	
	Vertical Board	24"	24"	Side to Side
	Vertical Board	24"	30"	
	Benchboard	24"	48"	
	Benchboard	24"	54"	
C2	Instrument Rack	30"	30"	
	Instrument Rack	30"	24"	
	Instrument Rack	48"	24"	
	Instrument Rack	60"	24"	Front to Back
	Instrument Rack	72"	24"	OR
	Instrument Rack	96"	24"	Back to Front
	Vertical Board	36"	24"	
	Vertical Board	48"	24"	
	Vertical Board	60"	24"	
	Vertical Board	72"	24"	
	Vertical Board	96"	24"	
C3	Instrument Rack	48"	30"	
	Instrument Rack	60"	30"	
	Instrument Rack	72"	30"	
	Instrument Rack	96"	30"	
	Vertical Board	36"	30"	Front to Back
	Vertical Board	48"	30"	OR
	Vertical Board	60"	30"	Back to Front
	Vertical Board	72"	30"	
	Vertical Board	96"	30"	
C4	Console	96"	42"	Back to Front

TABLE 3.10B-1: STANDARD ENCLOSURES (CONTINUED)

-Curve	Enclosure	Width	Depth	Mode of Failure
C5	Benchboard	48"	54"	Side to Side
	Benchboard	48"	48"	
C6	Benchboard	72"	48"	Front to Back
	Benchboard	96"	48"	
	Console	96"	48"	
C7	Benchboard	72"	54"	Back to Front
	Benchboard	96"	54"	



MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT	MAXIMUM SAFE WEIGHT PER BOLT FOR STANDARD ENCLOSURE AS A FUNCTION OF THE HEIGHT OF THE CENTER OF GRAVITY FIGURE 3.10B-1

APPENDIX 3.10C SAMPLE PANEL FREQUENCY ANALYSIS NSSS EQUIPMENT

The method of analysis used to determine the resonant frequency of the panel is as follows:

1. Calculate the moment of inertia of the corner post structure.

2. First assume a simplified structure and calculate the frequency using the expression:

   $f = 1/2\pi \; Kg/w = \; g/2\pi k/w = 3.13/\sqrt{w/k}$

   $\therefore f = 3.13/\sigma$

   where

   f = frequency

   g = 386 in./sec^2

   k = spring rate #/in.

   w = weight #

   σ = deflection = w/k

   weight distribution is assumed to be uniform.

3. Additional structural components are added and the moment and frequency recalculated.

The calculated resonant frequency of 7.4 Hz for the panel and 5.9 Hz for the benchboard was obtained using only the corner posts and the top. The addition of skin (3/8-in. steel) and 2-in. x 1/4-in. steel stiffeners will raise the frequency further. This proves that resonances cannot exist in the unstable region below 5 hertz.

FIRST APPROXIMATION

For First Approximation lump the four corner posts together and assume the panel is a cantilever beam fixed on one end and uniformly loaded.

The natural frequency is 2.6 Hz so we will have to use more of the structure.

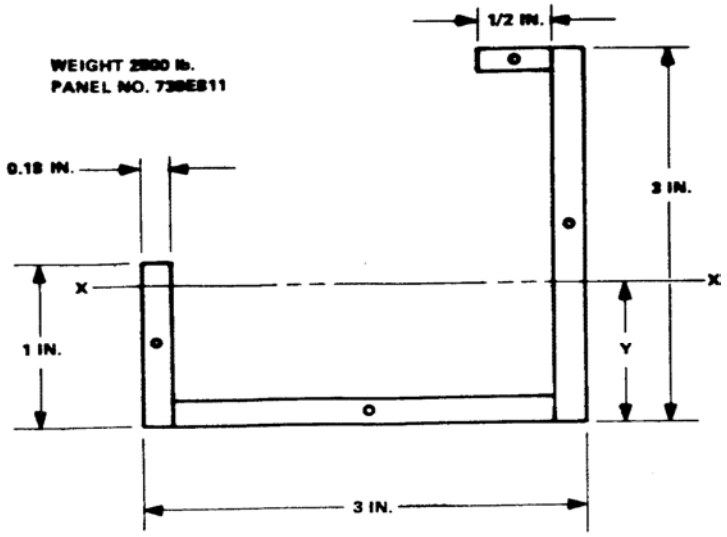


FIGURE 3.10C-1 CORNER POST

SECOND APPROXIMATION

For a second approximation, consider two 0.18" x 30" barriers in addition to the corner posts. The plan view of the panel is shown in Figure 3.10C-2.

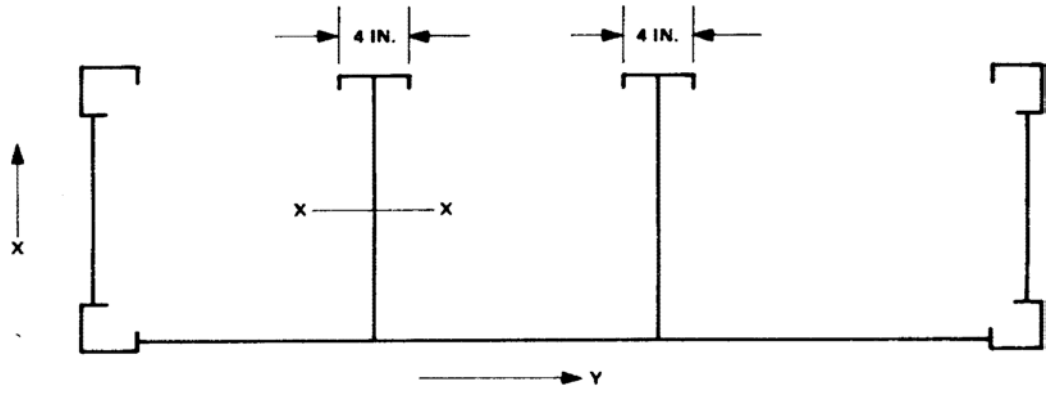


FIGURE 3.10C-2 PLAN VIEW OF PANEL

In the X direction just one barrier will raise the frequency to 30 Hz. Use 4 inches of the back panel for each of the two barriers (see Figure 3.10C-3) and the natural frequency in the Y direction becomes 4 Hz.

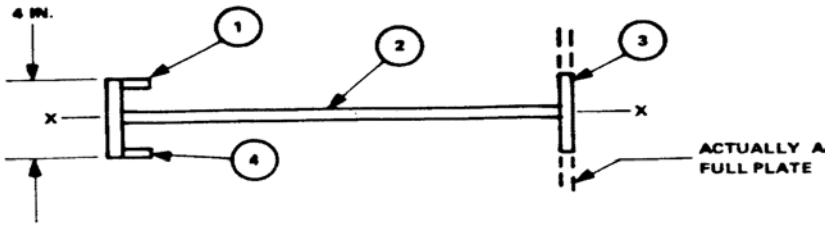


FIGURE 3.10C-3 BARRIER WITH TWO END PLATES

The deflection equation used so far is very conservative: it assumes that the 4 corner posts are lumped together and that the structure can reflect like a simple cantilever beam. Actually the corners are separated by an angle frame which is stiffer than the corner posts. This will force the structure to deflect as shown in Figure 3.10C-4.

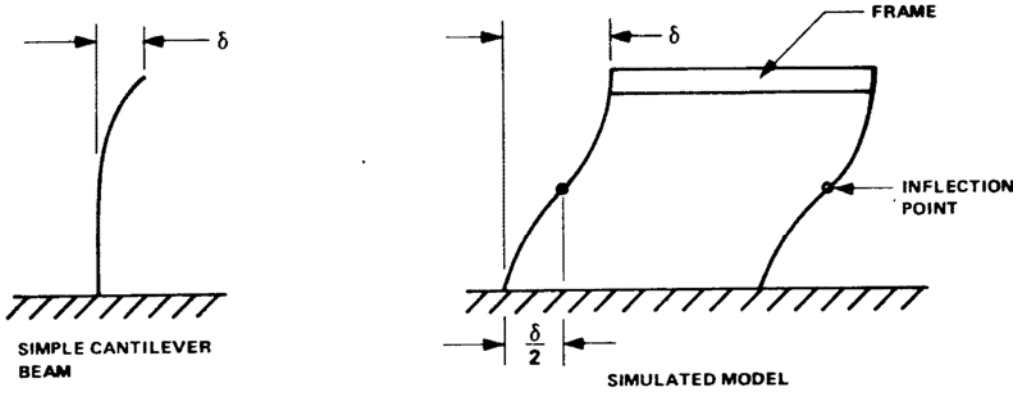
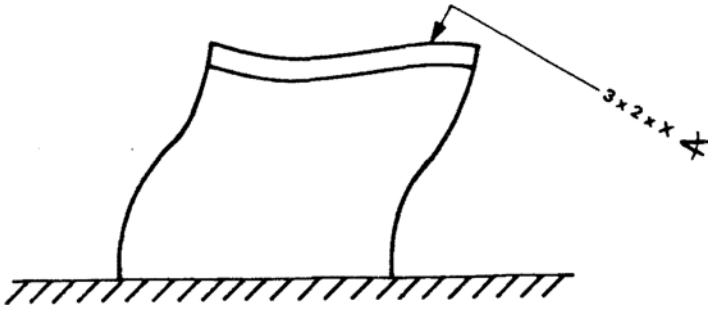


FIGURE 3.10C-4 PANEL DEFLECTIONS

In the simulated model we are not conservative (if we used all of the members) but we are very close. The reason we are not quite correct is because the stiff top frame will deflect slightly as shown below. The calculated frequency is 7.4 Hz which is above the necessary 5 Hz.



The benchboard H11-P601 which weighs 4000 pounds, the calculated natural frequency is 5.9 Hz which is still above the 5 Hz test frequency minimum.

NOTE: This neglects the barriers, the end and front panels, top plate, the stiffening of the lower part of the structure due to the bench board geometry, and all other members of the structure.

APPENDIX 3.10D DELETED

3.11 ENVIRONMENTAL DESIGN OF SAFETY-RELATED MECHANICAL AND ELECTRICAL EQUIPMENT

Environmental design criteria for the facilities conform to 10 CFR 50, Appendix A, General Design Criterion 4, Environmental and Missile Design Bases. For normal operation, systems and components required to mitigate the consequences of a design basis accident, or for hot or cold shutdown from the control room, are designed to remain functional after exposure to the following environmental conditions:

a. Representative operating temperatures maintained at the equipment location during normal operation by the ventilating and cooling systems. Refer to Section 9.4 for a description of the ventilating and cooling systems.

b. Pressure at operating plant conditions.

c. Maximum expected integrated radiation exposures for 60 years at the equipment location during normal operation.

The environmental conditions in different areas of the plant are identified in Tables 3.11-1 and 3.11-2.

3.11.1 Equipment Identification and Environmental Conditions

Equipment is subjected to adverse environmental conditions, including high temperature, pressure, humidity, radiation, containment spray and flooding or froth following a design basis accident (DBA). Tables 3.11-1, 3.11-2, 3.11-3 and Figures 3.11-2 through 3.11-30 give the normal and accident environmental parameters for the three major areas of the plant involved in equipment qualification: drywell, containment (outside drywell), and auxiliary buildings.

The equipment subjected to the requirements of 10CFR50.49 are qualified to perform their safety function during exposure to the environmental conditions of the plant area in which it is installed. Plant design ensures that equipment, not governed by 10CFR50.49 and installed within a harsh environmental area, would not fail in a manner detrimental to plant safety.

Detailed qualification information for 10CFR50.49 equipment is contained in the Environmental Qualification file. This file is both active and maintained in an auditable form.

3.11.1.1 Safety-Related Equipment and Components

The Class IE equipment and associated components must operate or be in a fail-safe condition during the thermal and radiological environment given in Tables 3.11-3 and 3.11-2,respectively.

3.11.2 Qualification Tests and Analyses

The Class IE electrical equipment has been designed, fabricated, and qualified to meet applicable Federal regulations and guide lines. GDC 1 of 10 CFR 50 Appendix A requirements are achieved by incorporation of performance, design, construction, and testing requirements into equipment specifications and by the establishment of a system of reviews to assure conformance with these specified requirements.

The environmental requirements of GDC 4 are addressed in subsections 3.11.1, 3.9.2.2, and 3.9.3.2. The Class IE electrical equipment meets the requirements of GDC 4. GDC 23 is primarily addressed in Chapter 7; however, subsection 3.11.1 describes the normal, abnormal, and accident environments to which the Class IE equipment is subjected and qualified. The requirements of GDC 23 are met.

GDC 50 requirements are achieved by analysis and test of pressure boundary components to assure containment integrity. Inservice inspection is performed to demonstrate leaktight integrity of components such as seal seats and seals.

Non-NSSS safety-related control and instrumentation equipment is qualified considering the requirements of Regulatory Guide 1.89 and IEEE Std. 323-1971 or 1974 (as appropriate).

NSSS components that are Class 1E equipment are qualified in accordance with IEEE Standard 323-1971 or 1974, either by test or analysis.

3.11.2.1 NSSS Instrumentation and Electric Equipment

Instruments and control panels which are used in Class IE applications are qualified under the program described in this section.

Qualification requirements of GEH specifications address tests, analyses, and formal records, thereof, to show compliance with the performance requirements. Life testing and qualified life determinations are pursued with each supplier and device.

3.11.2.1.1 Location of Equipment and Components

The safety-related equipment and components that cannot tolerate the expected accident basis environment of subsection 3.11.1 have been located in a controlled environment, isolated from the accident environment, which assures reliable operation. Reactor protection system and engineered safety features equipment are capable of functioning for the required duration listed in accident environment tables and shall subsequently remain in a fail-safe condition when subjected to the local (close proximity to break) steam environmental conditions if the equipment is:

a. Required to detect a steam or water line accident condition

b. Required to perform a steam line isolation function

c. Required to perform a water line isolation function and could be subjected to the steam environment such as electrical cable or valve operator

d. Required for safety system operation and is located so a steam line break in some other system exposes the safety system equipment to the local accident environment

e. Required to track the post-accident environment condition such as pressure, temperature, and radiation monitors.

Isolation valves and associated equipment required to perform the isolation function per items b and c above have been specified to remain functional for 1 hour in the local steam environment and shall subsequently remain in a safe (closed) condition.

Equipment for leak detection, post-accident surveillance, and other engineered safety features and reactor protection system equipment exposed to the local steam environment (per items a, d, and e above) have been specified to remain functional for the entire duration of the accident environment.

3.11.2.2 Testing of NSSS Instrumentation and Control Equipment (Excluding Motors and Valve-Mounted Equipment)

3.11.2.2.1 Test Sequence

[HISTORICAL INFORMATION] [Equipment is subjected to a one-time design qualification test (environmental and seismic) in accordance with test procedures. (Since control panels only provide mechanical support for the devices, they are limited to just seismic testing.)] Upon successful completion of design qualification testing, equipment is subjected to qualified life testing until either an end-of-life or 60 years occurs, whichever happens first. Following is a summary of the sequence and description of the steps to be taken.

3.11.2.2.1.1 Design Qualification Testing

3.11.2.2.1.1.1 Inspect Equipment

[HISTORICAL INFORMATION] [Look over each device and note physical conditions. Photograph test setup before operating.]

3.11.2.2.1.1.2 Operate Under Normal Conditions

[HISTORICAL INFORMATION] [Operate the device under ambient conditions.]

3.11.2.2.1.1.3 Operate Under Plant Operational Environmental Extremes

[HISTORICAL INFORMATION] [Subject device to those applicable operating extremes, plus the appropriate margin (re: +15 F for temperature) and establish that the instrument performs its operating functions within the spans allowed by its design specifications.]

3.11.2.2.1.1.4 Subject to Seismic Vibration

[HISTORICAL INFORMATION] [Vibrate the device while it is operating and verify the operability of its Class IE function during vibration.]

3.11.2.2.1.1.5 Repeat Operational Test Under Normal Conditions

[HISTORICAL INFORMATION] [Verify that the device still operates under normal conditions (subsection 3.11.2.2.1.1.2) after being subjected to seismic vibration.]

3.11.2.2.1.1.6 Operated Under SSE Environmental Conditions

[HISTORICAL INFORMATION] [Demonstrate that the device will operate and provide its Class IE function during exposure to appropriate SSE environmental extreme.]

3.11.2.2.1.1.7 Repeat Operational Test Under Normal Conditions

[HISTORICAL INFORMATION] [Verify that the device still operates under normal conditions (subsection 3.11.2.2.1.1.2) after being subjected to an SSE.]

3.11.2.2.1.1.8 Establish Replacement Schedule

[HISTORICAL INFORMATION] [Define failure, establish end-of-life, and provide a replacement plan for the equipment.]

3.11.2.2.2 Test Methods

All of the electronic modules and practically all of the process sensors are located either in the control room or other locations outside the drywell. Such being the case, then most all of the Class IE equipment is exposed to environments having negligible levels of radiation, pressure, and chemical atmosphere effluents insofar as electronic equipment operation is concerned. All of the significant environmental effects, then, are primarily due to humidity, temperature, and seismic vibration. It was the plan to provide only the significant environmental effects in the actual test environment and include written justification in the qualification report for each instrument for not including the negligible environmental parameters. Each exception which may be encountered was handled in a way appropriate to the equipment and its unique environment.

3.11.2.2.2.1 Temperature, Humidity Testing

Temperature and humidity environments under standard atmospheric pressure have been provided simultaneously by a standard laboratory-type chamber specially designed for that purpose. Devices under test are operated inside the chamber as various combinations of temperature/humidity are maintained. In the few cases where the pressure might be significantly above atmospheric for a particular device during an SSE, an autoclave is used instead. It was not planned to conduct the SSE temperature/ humidity environmental test simultaneously with a seismic test because of the improbability that they might occur exactly at the same time; because it was not felt that the quantitative effects on the equipment are particularly interdependent on the environments being produced simultaneously; and finally, it is difficult to justify because of the high expense in conducting such a test.

3.11.2.3 NSSS Valve-Mounted Instrumentation and Electrical Equipment

This subsection discusses the testing and analysis of the safety-related instrumentation and electrical equipment supplied with NSSS valve assemblies. Active safety-related valve actuators have been qualified based on performance requirements using IEEE 323, Regulatory Guide 1.89, and IEEE 344 as guidelines.

In the generic qualification, one of a family of actuators is type tested for the following conditions:

a. Aging simulation

    1. Thermal aging

    2. Radiation aging

    3. Operational aging

b. Seismic simulation aging

c. Accident and post-accident simulation

    1. Radiation

    2. Temperature

3. Pressure

4. Humidity/steam condition

5. Chemical exposure

The aging simulation is designed to put the test specimen in the end-of-life or 60-year plant service condition. The seismic simulation is to put the test specimen through the expected OBE events and SSE event estimated for the service life. The accident and post-accident test is to put the test specimen through the design basis event. The duration of the last test will be related to the length of time that the actuator has to perform its safety function.

IEEE Std. 382-1972 provided the outline for valve actuator qualification. Specific quantification of actuator qualification is embodied in the qualification test reports. The procedures for the following aging simulations are as specified in IEEE Std 382-1972.

a. Thermal: Refer to IEEE Std 382-1972, Part II, Section 2, page 10

b. Radiation: Refer to IEEE Std 382-1972, Part II, Section 1, page 10

c. Operation: Refer to IEEE Std 382-1972, Part II, Section 3, page 10

d. Seismic: Refer to IEEE Std 382-1972, Part I, Section 4, par. 4.3, page 8

The NSSS actuators are qualified for thermal aging in accordance with the industry standard IEEE Std 382-1972, with the aging acceleration rate justified by the application of Arrhenius methodology. The radiation aging is consistent with the requirements of IEEE Std 382-1972.

In the specific qualification, an actuator was selected from the qualified family to meet the specific requirements. It has been demonstrated that the selected actuator is indeed similar to the test specimen, that no further type test is required. Also, the selected actuator is manufactured and quality-controlled the same way as the test specimen. It is also demonstrated that the selected actuator satisfied the performance requirement for the

application. The qualification of a Limitorque mid-size unit (SMB-0) as reported in Limitorque document B0058 was used to generically qualify all sizes of Limitorque operators for the environmental test conditions in accordance with IEEE Std 382-72.

The service conditions of the specific application are less than the test conditions of the test specimen by at least the margin as discussed in the applicable IEEE standard. The selected actuator is also within the extrapolation limit of the test actuator as defined in the applicable IEEE standard. The test actuator specimen was subjected to at least 500 cycles as required by IEEE Std 382-1972. Documentation for the qualification has been prepared in accordance with the applicable IEEE standard.

3.11.2.4 Motors (NSSS Supplied)

The environmental qualification of the ECCS motors is discussed in subsections 3.9.2 and 3.9.3 in conjunction with the ECCS pump and motor assembly. Environmental qualification of the standby liquid control (SLC) motor is discussed in subsections 3.9.2 and 3.9.3 in conjunction with the SLC pump motor assembly.

3.11.2.5 Qualification of Safety-Related Non-NSSS Supplied Equipment

For most equipment, special qualification tests to verify operability at normal operating temperature, pressure, and humidity conditions are not required. Certification for this equipment is based on proven operating capability in similar environments in industrial and previous nuclear power plant applications.

3.11.2.5.1 Application of Industry Codes for Equipment Qualification in a Post-Accident Environment

Equipment purchased before November 15, 1974, has been qualified in accordance with IEEE Std. 323-1971. Equipment purchased after November 15, 1974, has been qualified in accordance with IEEE Std. 323-1974. Exceptions to these codes and standards are some electric valve motor operators which have been qualified in accordance with IEEE Std. 382-1972 and the drywell purge compressors which were qualified in accordance with IEEE Std. 334-1971.

3.11.2.5.2 General Description of Qualification Programs for Safety-Related Non-NSSS Supplied Equipment

Safety-related equipment located outside the containment purchased prior to November 1974 has been qualified, in general, by previous experience. The post-accident environment of the subcompartment or building that houses the equipment is not very severe and is similar to environments experienced in industrial and previous power plant applications.

In general, safety-related equipment located inside the containment purchased after November 1974 has been qualified by testing of an actual production component or by testing of a prototype model.

The seismic qualification of mechanical equipment (including pump and valve operator motors) is discussed in subsections 3.9.2 and 3.9.3. Seismic qualification of controls and electrical equipment is discussed in Section 3.10.

3.11.2.5.3 Regulatory Guides

The recommendations contained in the regulatory guides listed in a. through e. below, and other applicable regulatory guides, have been utilized as described in Appendix 3A.

a. Regulatory Guide 1.30, Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment.

b. Regulatory Guide 1.40, Qualification Tests of Continuous Duty Motors Installed Inside the Containment of Water-Cooled Nuclear Power Plants.

   Continuous duty motors used inside the containment are type tested under simulated LOCA conditions. IEEE Std. 334-1971, Trial-Use Guide for Type Tests of Continuous Duty Class I Motors Installed Inside the Containment of Nuclear Power Generating Stations, is used. Insofar as practicable, auxiliary equipment that is part of the installed motor assembly is likewise qualified in accordance with IEEE Std. 334 under simulated design basis event conditions.

c. Regulatory Guide 1.63, Electrical Penetration Assemblies in Containment Structures for Water-Cooled Nuclear Power Plants.

   Electrical containment penetrations are tested in accordance with IEEE Std. 317-1972, Electrical Penetration Assemblies in Containment Structures for Nuclear Power Generating Stations. Fiber Optic containment penetration modules are tested in accordance with IEEE 317-1983.

d. Regulatory Guide 1.73, Qualification Tests of Electric Valve Operators Installed Inside the Containment of Nuclear Power Plants.

   Motor-operated valves used inside the containment are type tested in accordance with IEEE Std. 382-1972 (ANSI N41.6), Trial Use Guide for the Type-Test of Class I Electric Operated Valve Operators for Nuclear Power Generating Station.

e. Regulatory Guide 1.89, Qualification of Class IE Equipment for Nuclear Power Plants.

3.11.3 Qualification Test Results

The qualification status of electrical equipment was first given in MP&L's (SERI's) initial submitted response to NUREG-0588 under AECM-81/231 dated July 1, 1981. Updates to this initial status have been given in subsequent transmittals. Detailed qualification documentation for NUREG-0588/ 10 CFR 50.49 electrical equipment is contained within the environmental qualification files located in engineering offices at the GGNS site.

3.11.3.1 NSSS Instrumentation and Electrical Equipment

This paragraph discusses the test results for safety-related instrumentation and electrical equipment in the NSSS, except that which is supplied with NSSS motors and valves. The test results for GEH safety-related equipment are maintained in a permanent file by GE and can be readily audited. [UNNECESSARY INFORMATION] [In all cases, the equipment used in Class IE applications passed the prescribed tests. Where equipment failed to pass the tests, it was rejected. In some cases, equipment which failed one test was modified or repaired to meet the performance requirements and then retested. If the retested equipment passed the latter test, it could be used in a Class IE application.]

3.11.3.2 NSSS Valve-Mounted Electrical Equipment

The environmental qualification tests and test results for the main steam isolation and safety/relief valves are discussed in subsections 3.9.3 and 3.9.2.

3.11.3.3 NSSS Motors

Environmental qualification of the ECCS motors is discussed in subsections 3.9.2 and 3.9.3 in conjunction with the ECCS pump and motor assembly. Environmental qualification of the standby liquid control (SLC) motor is discussed in subsections 3.9.2 and 3.9.3 in conjunction with the SLC pump motor assembly.

3.11.3.4 Qualification of Non-NSSS Equipment

Qualification of non-NSSS equipment has been accomplished by previous operating experience, analysis, testing, or a combination of these methods. [UNNECESSARY INFORMATION] [For equipment qualification, if any equipment failed to pass the tests, the equipment was either rejected or suitable modifications were made and the equipment was then retested.]

3.11.4 Loss of Ventilation

The maximum temperatures considered in the sizing of air conditioning systems serving safety-related systems are determined by analysis of the following factors:

a. Maximum outdoor design temperatures for the geographical area of the plant (both wet bulb and dry bulb readings)

b. Maximum internal piping thermal loads, if applicable, for the room, using maximum operating temperatures for the pipe contents and maximum footage of active pipe for each mode of operation (this value may be negative)

c. Maximum internal electrical load, assuming full lighting for the room, and using, if applicable, the maximum control and equipment resistance losses for each mode of operation

d. Maximum heat transfer from miscellaneous equipment surfaces, if applicable (e.g., outer surface of the diesel generator)

e. Maximum heat transfer from the surfaces of open pools and tanks, if applicable, using the maximum operating temperature of the contents

f. Maximum heat transfer from the surfaces of the room including walls, floor, and ceiling, or roof (this may be negative)

Seismic Category I air conditioning and ventilation systems, described in Section 9.4, are powered from the Class IE electrical power supplies and are provided for the following locations:

a. Control room

b. Diesel generator buildings

c. ESF switchgear rooms

d. ECCS pump rooms

e. SSW pumphouse

f. Safeguard switchgear and battery rooms

3.11.4.1 Control Room HVAC System

The controls and instrumentation required for a safe plant shutdown are located in the control room. Each one of the two 100-percent capacity independent trains of the control room HVAC system is capable of maintaining the control room space conditions at 72 F. This system has been designed to meet seismic Category I requirements and, in the event of loss of offsite power, each train receives its electrical power automatically from a separate emergency diesel generator. Since no single failure can cause a loss of control room air conditioning, the control room HVAC system design ensures that the operability of the safety-related control and instrumentation equipment will not be impaired due to environmental conditions and will continue to function in a satisfactory environment. Therefore, no special environmental design requirements have been incorporated in the qualification of equipment located in the control room.

3.11.4.2 Diesel Generator Building Ventilation System

Each diesel generator room is provided with its own, totally independent, ventilation system designed to meet seismic Category I requirements. The intake and exhaust for the ventilation systems are protected from tornado missiles. Each ventilation system is connected to the emergency power bus associated with its respective standby or HPCS diesel generator unit upon loss of offsite power. Each ventilation system is capable of maintaining its diesel generator room between 120 and 40 F during diesel operation. The failure of one ventilation system will affect its respective diesel only. The shutdown of one diesel will not impair the capability to shutdown the reactor in a controlled manner.

3.11.4.3 ESF Electrical Switchgear Room Ventilation System

Emergency switchgear for safety-related equipment are located in separate rooms in the auxiliary building. Each switchgear room is provided with one 100-percent capacity independent fan-coil unit. The fan-coil units located in switchgear rooms 1A308 and 1A309, El. 139'-0", will maintain the temperature of the rooms at less than 90 F during all modes of normal plant operation and at less than 105 F during all modes of emergency plant operation. The fan-coil units located in the remaining switchgear rooms will maintain the room temperature at less than 108 F during all modes of normal and emergency plant operation. The ventilation systems for the emergency switchgear rooms are designed to seismic Category I requirements and each system is supplied with power from its associated emergency bus during loss of offsite power and during a LOCA. No single failure can result in loss of cooling to more than one division of ESF electrical switchgear rooms.

3.11.4.4 ECCS Pump Room Ventilation System

The ECCS pumps and associated piping and valves are located in separate rooms in the auxiliary building. Each room is provided with one 100-percent capacity fan-coil unit which maintains a room temperature of less than 155 F during ECCS pump and equipment operation. During normal plant operation, the auxiliary building fan-coil units, in conjunction with the fuel handling area exhaust fans, will maintain a room temperature of less than 105 F. The pump room ventilation units are designed to meet seismic Category I requirements and each fan-coil unit is supplied with

power from its associated emergency bus upon loss of offsite power and during a LOCA. No single failure can result in loss of cooling to more than one division of ECCS pump rooms.

3.11.4.5 Standby Service Water Pumphouses

The A standby service water pumphouse contains the Division I SSW pump and the HPCS service water pump. The B standby service water pumphouse contains the Division II SSW pump.
The standby service water pumphouse ventilation system consists of three, full-capacity fans utilizing separate ducting, dampers, and controls to maintain the space design temperature of the greater of 104 °F maximum or during periods of extreme ambient temperature the room temperature may be equal to outdoor air temperature +10 °F, not to exceed 111° F. Two of the fans are located in standby service water pumphouse A and the other fan is located in pumphouse B. In pumphouse A one fan is supplied with power from the Division I engineered safety features (ESF) bus and starts when its associated SSW pump starts or on high area temperature. The other fan is supplied with power from the Division II ESF bus and starts only in response to a high area temperature. The fan in pumphouse B is supplied with power from the Division II ESF bus and starts when its associated SSW pump starts or on high area temperature. The ventilation system is designed to meet seismic Category I requirements and is supplied with power from the associated emergency bus upon loss of offsite power and during a LOCA. The failure of any one of the fans will cause loss of cooling to its respective SSW pump only. The shutdown of one of the two redundant SSW trains does not impair the capability to shut down the reactor in a controlled manner.

3.11.4.6 Safeguard Switchgear and Battery Rooms Ventilation System

The ventilation system serving the safeguard switchgear and battery rooms consists of four 50-percent capacity air handling units with supply fans and four 50-percent capacity exhaust fans. The system has been designed to maintain the area room temperature between 58 F and 104 F during normal operation. During a LOCA the temperature will increase to 110 F. The ventilation system is designed to meet seismic Category I requirements and is supplied with onsite emergency power upon loss of offsite power. Since a single failure of one of the 50-percent capacity air handling units or one of the 50-percent capacity exhaust fans or associated emergency power source will not cause loss of safeguard switchgear and battery room ventilation, special environmental

design requirements have not been incorporated for the equipment within these rooms.

3.11.4.7 Technical Specifications

Should a redundant ventilation train or one of the individual room coolers become inoperative during normal operation or periodic testing, sufficient equipment is still available to mitigate the consequences of a DBA. In this event, action is taken in accordance with Technical Requirements Manual (TRM).

3.11.5 Estimated Chemical and Radiation Environment

3.11.5.1 Chemical Environment

Engineered safety feature (ESF) systems are designed to perform their safety-related functions in the temperature, pressure, and humidity conditions described in subsection 3.11.1, and Sections 6.2 and 6.3.

Grand Gulf will not utilize any chemical additives in the water that is recirculated by the emergency core cooling systems (ECCS) and the containment spray portion of the residual heat removal (RHR) system.

Following an accident, the containment and drywell atmospheres are maintained below 4 percent (by volume) hydrogen in accordance with Regulatory Guide 1.7 as discussed in subsection 6.2.5.

3.11.5.2 Radiation Environment

ESF systems and components are designed to perform their safety-related functions after the normal operational radiation exposure plus the accident radiation exposure. The normal operational exposure is based on the design source terms presented in December 1979, issue of NUREG-588 and Regulatory Guide 1.89.

Post-accident ESF system and component radiation exposures are dependent on equipment location. The integrated accident doses are based on the assumptions and analyses used in NUREG-588 and Regulatory Guide 1.89. The integrated accident doses are given for gamma doses only.

Although beta radiation is present in the containment, due to the low penetrating ability of beta particles, there is no impact on the integrity and operability of engineered safety features. Cables for engineered safety features inside the containment are in cable trays or conduits where they are completely covered for protection from the operation of containment sprays. In addition,

motor casings are similarly designed for protection from the sprays. The metal conduits, trays, and/or cable insulation, and motor casings are sufficiently thick to eliminate beta radiation damage as a concern.

The integrated accident doses (gamma only) are given in Table 3.11-2.

3.11.5.3 Suppression Pool, Residual Heat Removal System and Emergency Core Cooling System Water Quality

The water in these systems shall not be chemically inhibited. The maximum limits for the suppression pool water have been established to be compatible with those of the primary coolant and are listed below:

Parameter	Pressure Reactor Water Limits Shutdown Condition	Suppression Pool Water Quality Expected	Suppression Pool Water Maximum Limit
Conductivity	≤10 µmho/cm @ 25 C	≤3 µmho/cm @ 25 C	≤10 µmho/cm @ 25 C
Chlorides (as C1)	≤0.5 ppm	≤0.2 ppm	≤0.2 ppm
pH	5.3 to 8.6 @ 25 C	5.3 to 8.6 @ 25 C	5.3 to 8.6 @ 25 C

Monitoring of suppression pool water quality over a period of several years in suppression pools with and without coatings indicates that the feed and bleed to radwaste occurring during normal system testing and level adjustments could maintain the water quality well within the above limits.

In addition, for Grand Gulf, a suppression pool cleanup system as described in subsection 9.3.6 is available for maintaining the suppression pool water quality within the prescribed limits.

During reactor shutdown cooling, the RHR system is lined up with the reactor coolant pressure boundary. Therefore, to insure that the reactor water quality is maintained, the shutdown cooling piping and equipment in the RHR system are flushed with demineralized water from the condensate storage tank prior to commencing shutdown cooling.

During layup, the RHR system will be filled with water of the following limits:

Parameter	RHR System Maximum Limit
Conductivity	≤3 µmho/cm @ 25 C
Chlorides (as Cl)	≤0.05 ppm
pH	5.3 to 7.5 @ 25 C

TABLE 3.11-1: NORMAL PLANT OPERATIONAL ENVIRONMENTAL CONDITIONS

	Area	Pressure (as noted)	Temperature (F) Normal (2)	Relative Humidity % Min/Max
I.	Inside drywell			
A.	Area above reactor vessel shield to top of drywell	-2.0 to +2.0 psig	135 (ave)	20/90
B.	Region adjacent to core	"	"	"
C.	Under RPV, inside shield wall	"	112	"
D.	Vicinity of recirculation pump motors	"	135 (ave)	"
II.	Containment			
A.	Area above suppression pool	-0.10 to 1.0 psid	90 (ave)	60
B.	General floor area	"	"	"
C.	Main steam line tunnel	"	125	10/90
D.	Reactor water cleanup system 1. Heat exchangers 2. Filters and tanks	"	105	20/90

TABLE 3.11-1: NORMAL PLANT OPERATIONAL ENVIRONMENTAL CONDITIONS (Continued)

Area	Pressure (as noted)	Temperature (F) Normal (2)	Relative Humidity % Min/Max
III. Auxiliary Building			
A. RHR eqmt rooms	-1.0 to 0.25 in. w.g.	105 (1)	20/90
B. HPCS eqmt room	"	" (1)	"
C. RCIC eqmt room	"	" (1)	"
D. LPCS eqmt room	"	" (1)	"
E. SGTS room	"	87	"
F. Pipe penetration rooms	"	105 (1)	"
G. Electrical penetration rooms (switchgear rooms)	"	104 (3)	20/50
H. Purge and ventilation rooms	"	80 (8)	"
I. Drain pump rooms	"	"	"

TABLE 3.11-1: NORMAL PLANT OPERATIONAL ENVIRONMENTAL CONDITIONS (Continued)

	Area	Pressure (as noted)	Temperature (F) Normal (2)	Relative Humidity % Min/Max
J.	CCW pump and heat exchanger room	"	80	"
K.	CRD pumps and filter room	"	"	"
L.	CRD repair room	"	"	"
M.	Fuel pool eqmt room	"	"	"
N.	Fuel pool demineralizer rooms	Atmos ±1.0 in. w.g.	90	
O.	Fuel handling area fan room	"	"	"
P.	Fuel handling area	"	"	"
Q.	All other areas	"	" (4)	50
R.	Vicinity of steamlines	"	125	10/90
IV.	Control Building			

TABLE 3.11-1: NORMAL PLANT OPERATIONAL ENVIRONMENTAL CONDITIONS (Continued)

	Area	Pressure (as noted)	Temperature (F) Normal (2)	Relative Humidity % Min/Max
A.	Control room	Atmos +1/4 in. -0 in. w.g.	72	20/50
B.	Other rooms containing batteries, switch-gear, cables, etc. having safety-related functions	Atmos ±1.0 in. w.g.	104	10/60
V.	Other Areas			
A.	Diesel generator areas	"	120 (5)	30/90
B.	Outdoor areas	--	95 Summer Design (6)	--
C.	Turbine building	Atmos ±1.0 in. w.g.	105 max	50
D.	SSW pumphouse	"	104 (7)	--

NOTES

(1) 150 F during hot standby and plant shutdown

TABLE 3.11-1: NORMAL PLANT OPERATIONAL ENVIRONMENTAL CONDITIONS (Continued)

(2) Minimum temperature inside the plant structures containing safety-related equipment will be 65 F or greater with an outside temperature of 20 F, with the exception of the safeguard battery and switchgear room, which could be as low as 58 F at winter design outdoor temperatures.

(3) The temperature shown applies to all electrical penetration rooms except for rooms 1A308 and 1A309, El. 139'-0". The normal temperature will be 90 F and the accident temperature will be 105 F for these rooms.

(4) 90 F for corridor areas at elevation 119'-0".

(5) 80 F (normal) during all modes when diesel generator is not operating.

(6) The time for which temperature exceeds 95 F is expected to be approximately 2.5 percent of the total yearly hours. Equipment in areas cooled by safety related HVAC system has been evaluated for outdoor air temperatures up to 102.5F.

(7) For extreme outdoor weather conditions, this temperature could reach 111 F.

(8) The temperature shown applies to all purge and ventilation rooms except room 1A406. The normal temperature for this room will be 95°F.

TABLE 3.11-2: RADIATION ENVIRONMENTAL CONDITIONS NORMAL AND ACCIDENT

I. Drywell

Equipment or Area	Radiation Type	Operating Dose Rate (1)	Intergrated Dose (3) Normal (40 EFPY)	Normal (60 EFPY)	Accident
Above Core	Gamma	37.0	1.3×10^{7}	1.95×10^{7}	1.72×10^{7}
*(Zone 1)	Neutron	4.6×10^{3}	5.9×10^{12}	8.8×10^{12}	
Core Region	Gamma	77	2.7×10^{7}	4.1×10^{7}	2.08×10^{7}
*(Zone 2)	Neutron	2.0×10^{5}	2.6×10^{14}	3.9×10^{14}	
Under Vessel	Gamma	10.6	3.1×10^{6}	5.0×10^{6}	2.08×10^{7}
*(Zone 3)	Neutron	3.6×10^{-8}	4.7×10^{1}	7.0×10^{1}	
Near Recirc. Pumps	Gamma	37	3.1×10^{7}	1.95×10^{7}	2.08×10^{7}
*(Zone 4)	Neutron	4.6×10^{3}	5.9×10^{12}	8.8×10^{12}	

TABLE 3.11-2: RADIATION ENVIRONMENTAL CONDITIONS NORMAL AND ACCIDENT (Continued)

II. Containment (Outside Drywell)

Equipment or Area	Radiation Type	Operating Dose Rate (1)	Intergrated Dose (3) Normal (40 EFPY)	Normal (40 EFPY)	Accident (4)
Suppression Pool *(Zone 6)	Gamma	0.3	1.1×10^5	1.8×10^5	9.46×10^6
General Floor Area	Gamma	0.3	1.1×10^5	1.8×10^5	5.6×10^6
Cleanup Systems					
a) Heat Exchanger	Gamma	108	3.8×10^7	5.9×10^7	1.79×10^6
b) Filters & Tanks	Gamma	108	3.8×10^7	5.9×10^7	1.36×10^5

TABLE 3.11-2: RADIATION ENVIRONMENTAL CONDITIONS NORMAL AND ACCIDENT (Continued)

III. Auxiliary Building

Equipment or Area	Radiation Type	Operating Dose Rate(1) Plant Oper	Intergrated Dose (3) Normal	Intergrated Dose (3) Accident (4)
HPCS and RCIC Area	Gamma	0.018	5.7×10^{3}	5.84×10^{7}
RHR A	Gamma	1.2	3.4×10^{5}	6.09×10^{7}
LPCS Area	Gamma	0.054	1.7×10^{4}	1.39×10^{7}
RHR B	Gamma	1.2	3.4×10^{5}	7.74×10^{7}
RHR C	Gamma	0.054	1.7×10^{4}	1.52×10^{6}
Steam Tunnel Outside Containment	Gamma	18	5.5×10^{6}	5.5×10^{7}(4)
Standby gas treatment system	Gamma	0.0012	3.8×10^{2}	2.62×10^{8}(4)

TABLE 3.11-2: RADIATION ENVIRONMENTAL CONDITIONS NORMAL AND ACCIDENT (Continued)

NOTES

(1) Gamma Dose Rate — Rads (Carbon)/hour
Neuron flux — Neuts/cm^2-sec

(2) Deleted

(3) Gamma Dose — Rads (Carbon)
Neutron fluence — Neutrons/cm^2 (NVT)
Normal Conditions — Integrated over 40 years - 100 percent load factor at rated power.
Integrated over 60 years – 85% percent load factor at rated power (See Calculation XC-Q1111-20001 for additional information).

Accident Conditions — Integrated over 180 days - For all doses other than Standby Gas Treatment the LOCA analysis was based on the assumptions that 100 percent of the noble gases, 50 percent of the halogens, and 1 percent of the solid fission products were released from the core. Standby Gas Treatment doses were conservatively based on the releases documented in Table 15.6-10 (See Section 15.6.5 for details).

(4) See Equipment Qualifications files for individual component doses and dose rates.

TABLE 3.11-3: ACCIDENT ENVIRONMENT
INSIDE DRYWELL, CONTAINMENT, AND AUXILIARY BUILDING

Inside Drywell-Pressure[1]

Pressure psia	14.7-25.0 (ramp)	44.7	44.7-36.0 (ramp)	36[5]	32.7	29.5-29.4 (ramp)	29.4-14.7 (ramp)
Duration[4]	0-1.5sec	1.5-40 sec	40-43 sec	43 sec-4hr	4-55.6hr	55.6-72hr	3-100 days

Inside Drywell - Temperature[1]

Temperature, F	135-330 (ramp)	330	310	250	250-244 (ramp)	244 (ramp)	244-238 (ramp)	238-156 (ramp)	156-150 (ramp)	150-148 (ramp)	148-144 (ramp)
Rel. Humidity	Steam	Steam	100%	100%	100%	100%	100%	100%	100%	100%	100%
Duration[4]	0-1.5 sec	1.5 sec-3hr	3-6hr	6-24 hr	24-30hr	30-36hr	36-36.11hrs	36.11-475hrs	475-1000hrs	1000hrs -50days	50-100days

Flood to 117'-4" for 100 days; flood from 117'-4" to 127'7" for 10 seconds

Containment (Outside Drywell) - Temperature[2]

Temperature, F	238	238-155 (ramp)	155	155-120 (ramp)	120-105 (ramp)
Rel. Humidity	100%	100%	100%	100%	100%
Duration[4]	0-288sec	288sec-30min	30min-36hr	36-72hr	3-100day

Containment (Outside Drywell) - Pressure [2]

Pressure, psia	30	30-20.0 (ramp)	20.0-14.7 (ramp)
Duration[4]	0sec-30hr	30-300hr	300hr-100day

Flood to 130'-10" for 5 seconds; flood from 130'-10" to 141'-10" for 5 seconds; spray at 185 at 1.3 gpm/ft² for 30 days

Auxiliary Building [3]

Pressure, psia(6)	14.7-25.0 (ramp)	25.0-20.0 (ramp)	20.0-21.0 (ramp)	21.30 (ramp)	30-22.4 (ramp)	22.4-22.8 (ramp)	22.8	22.8-14.7	14.7

TABLE 3.11-3: ACCIDENT ENVIRONMENT
INSIDE DRYWELL, CONTAINMENT, AND AUXILIARY BUILDING (Continued)

Duration[4]	0-0.03sec	0.03-0.4sec	0.4-1.2sec	1.2-3.4sec	3.4-4.3sec	4.3-9.8sec	9.8-120sec	120-121.3sec	121.3sec-100days		
Auxiliary Building - Temperature [3]											
Temperature, F	304	304-312 (ramp)	312-304 (ramp)	304	304-312 (ramp)	312-304 (ramp)	304	304-216 (ramp)	216	216-136	136-125
Rel. Humidity	100%	100%	100%	100%	100%	100%	100%	100%	100%	100%	100%
Duration[4]	0-0.35sec	0.35-0.5sec	0.5-1.0sec	1.0-3.6sec	3.6-6sec	6-16sec	16-180sec	180-475sec	475-1850sec	1850sec-1.4days	1.4-100days

(1) Enveloping profile for drywell
(2) Enveloping profile for containment
(3) Enveloping profile for auxiliary building
(4) Durations are expressed as "From Time Equals - To Time Equals" (e.g., 10 to 30 hours)
(5) Drywell negative transient can occur around 350 sec as a result of the condensation of steam (not listed - see Figure 3.11-5 for typical event)
(6) Actual pressure change should be obtained from Equipment Qualification files

TABLE 3.11-4: Deleted
(Sheets 1 and 2)

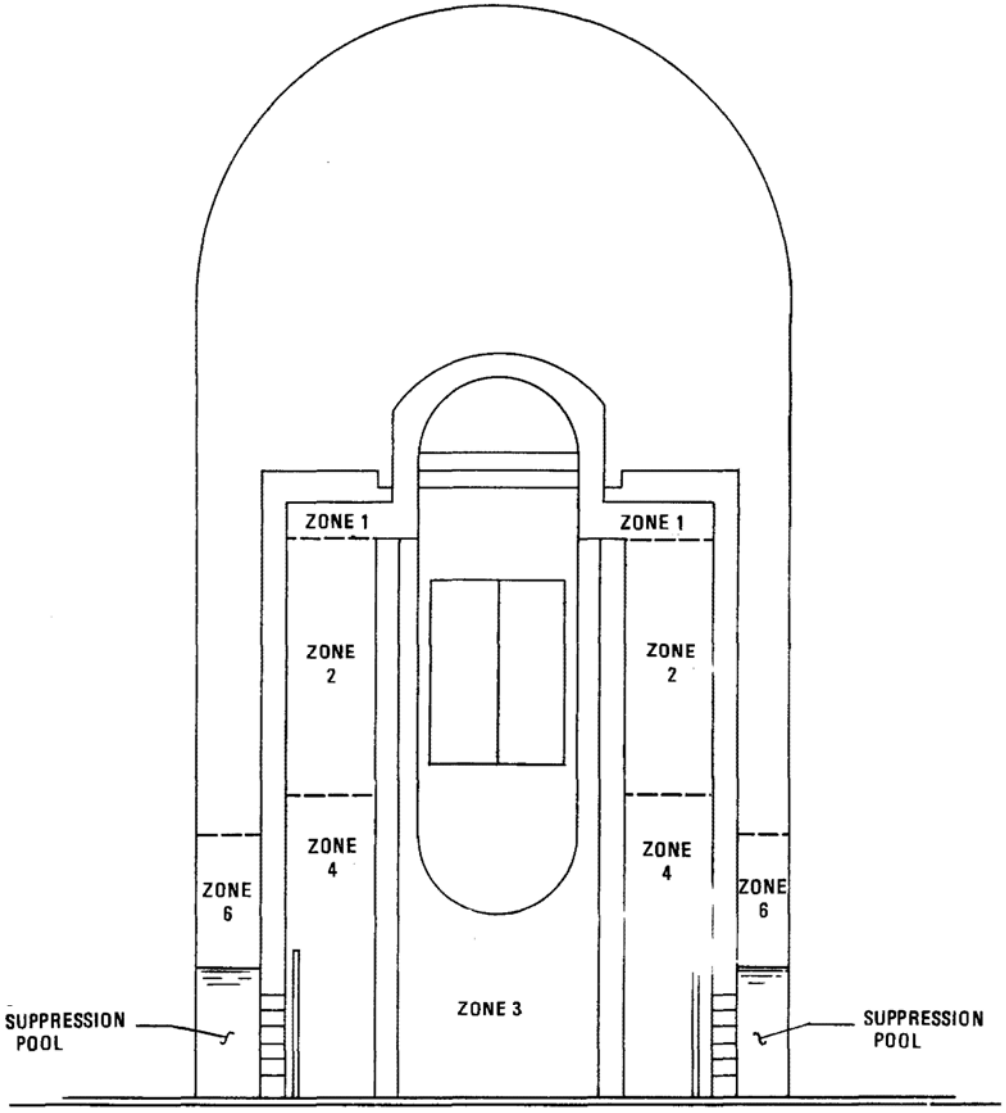
TABLE 3.11-5: Deleted
(Sheets 1 through 3)

TABLE 3.11-6: Deleted

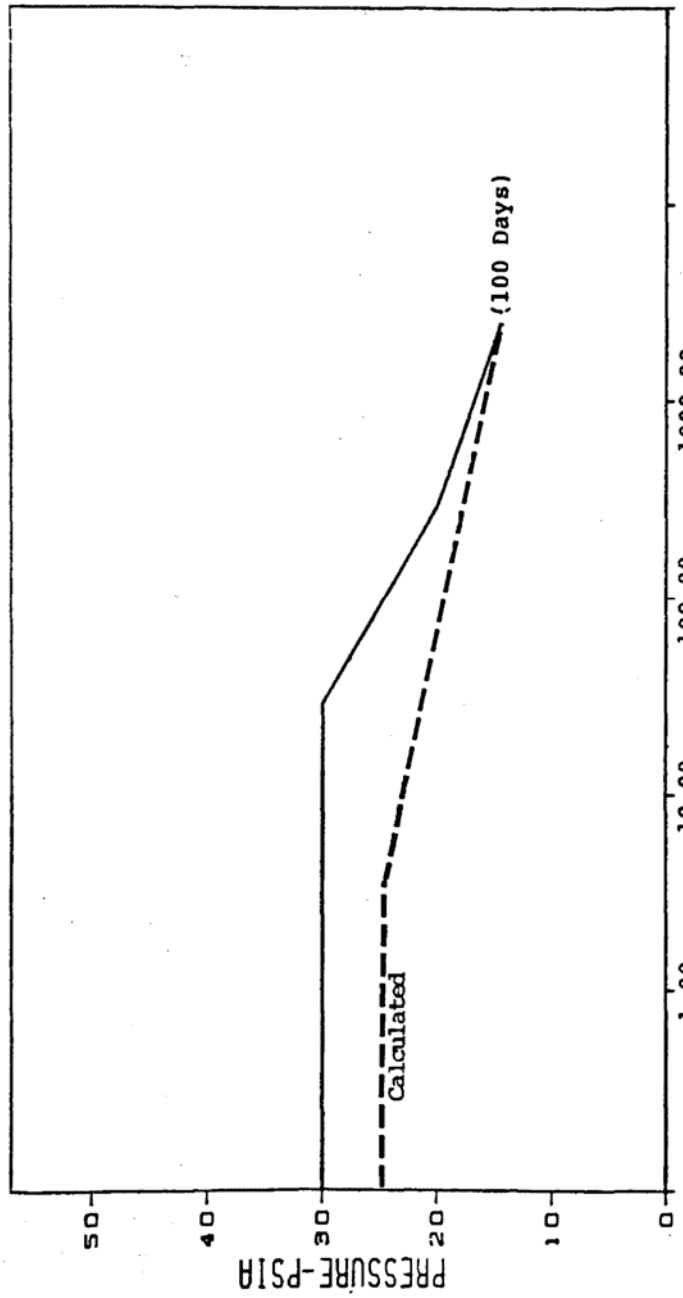
TABLE 3.11-7: DELETED

(Sheets 1 through 69)

The GGNS Environmental Qualification files are maintained as separate controlled documents. The qualification status of electrical equipment was first given in MP&L's (SERI's) initial submitted response to NUREG-0588 under AECM-81/231 dated July 1, 1981. Updates to this initial status have been given in subsequent transmittals.



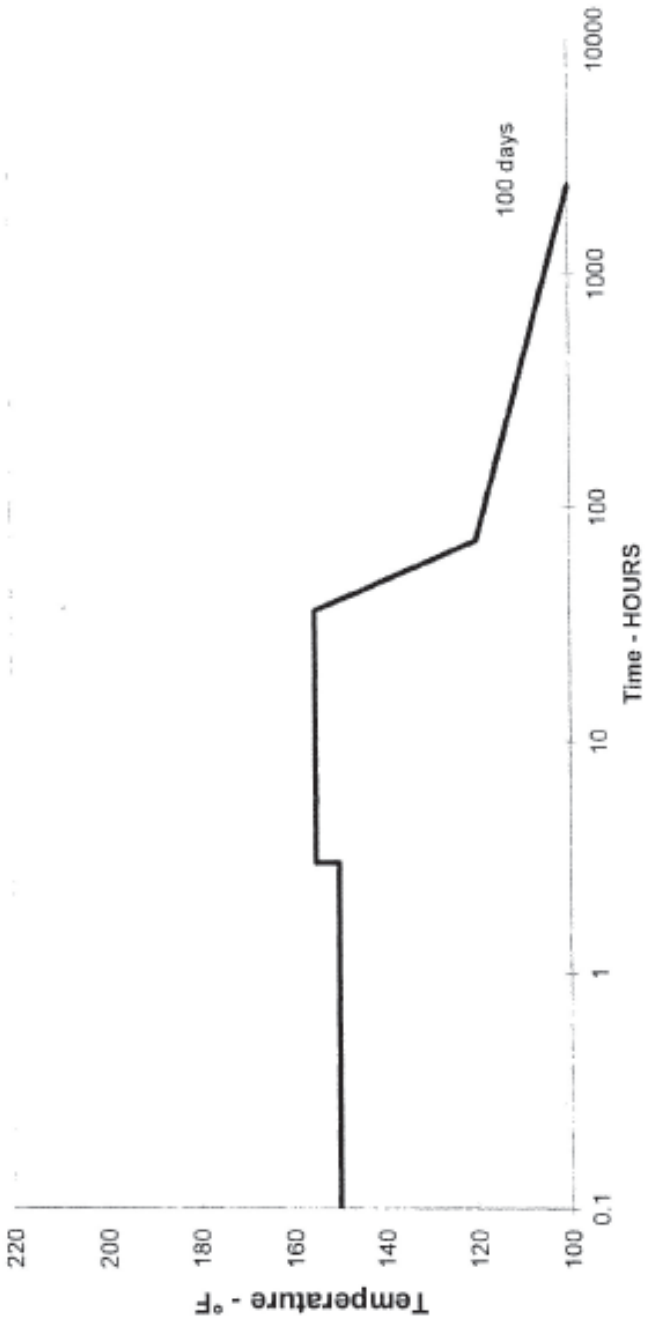
MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT	DRYWELL RADIATION ZONES FIGURE 3.11-1



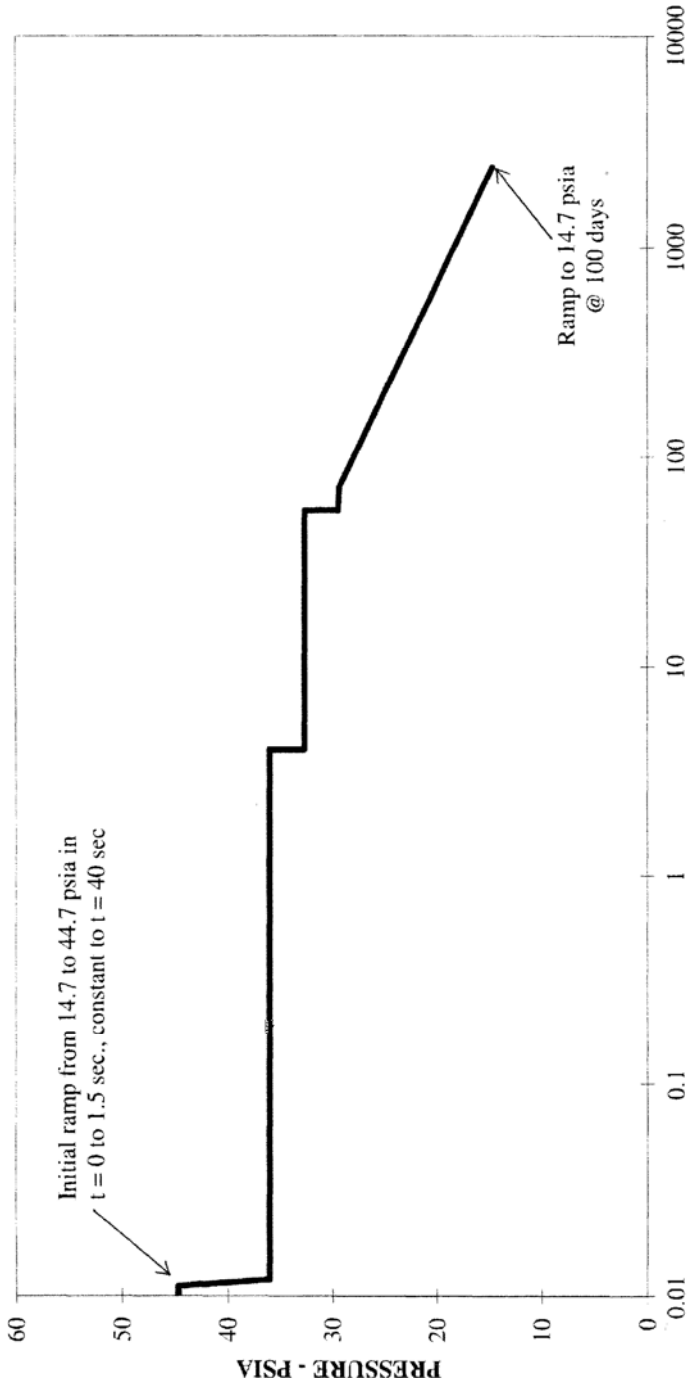
MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT

ENVELOPING PRESSURE PROFILE
FOR ROOMS: 1A110, 1A212, 1A311,
1A313, 1A411, 1A419, 1A421, 1A425, 1A509,
1A510, 1A512, 1A520, 1A601

FIGURE 3.11-2

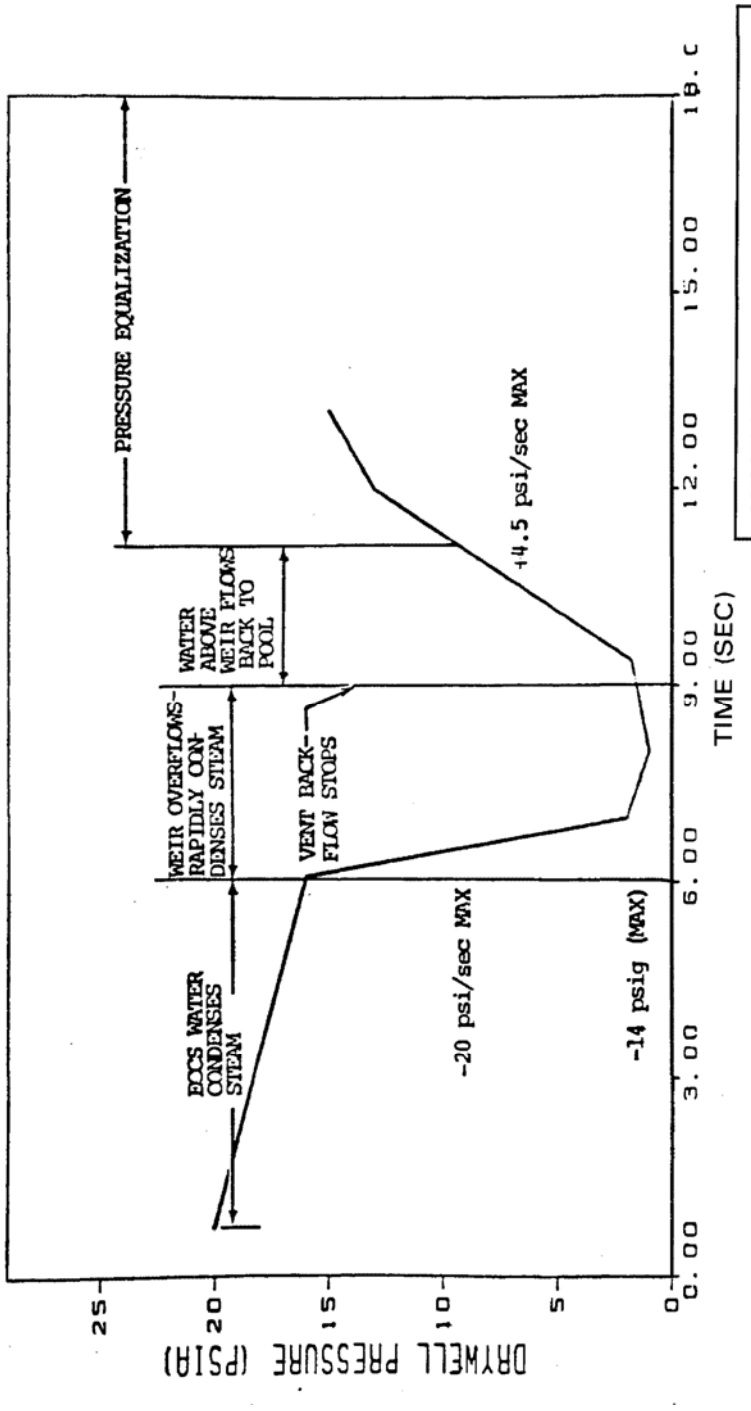


GRAND GULF NUCLEAR STATION
UNIT 1
UPDATED FINAL SAFETY ANALYSIS REPORT
ENVELOPING TEMPERATURE PROFILE
FOR ROOMS: 1A110, 1A212, 1A311, 1A313
1A411, 1A419, 1A421, 1A425, 1A509, 1A510,
1A520, 1A601
FIGURE 3.11-3



GRAND GULF NUCLEAR STATION
UNIT 1
UPDATED FINAL SAFETY ANALYSIS REPORT

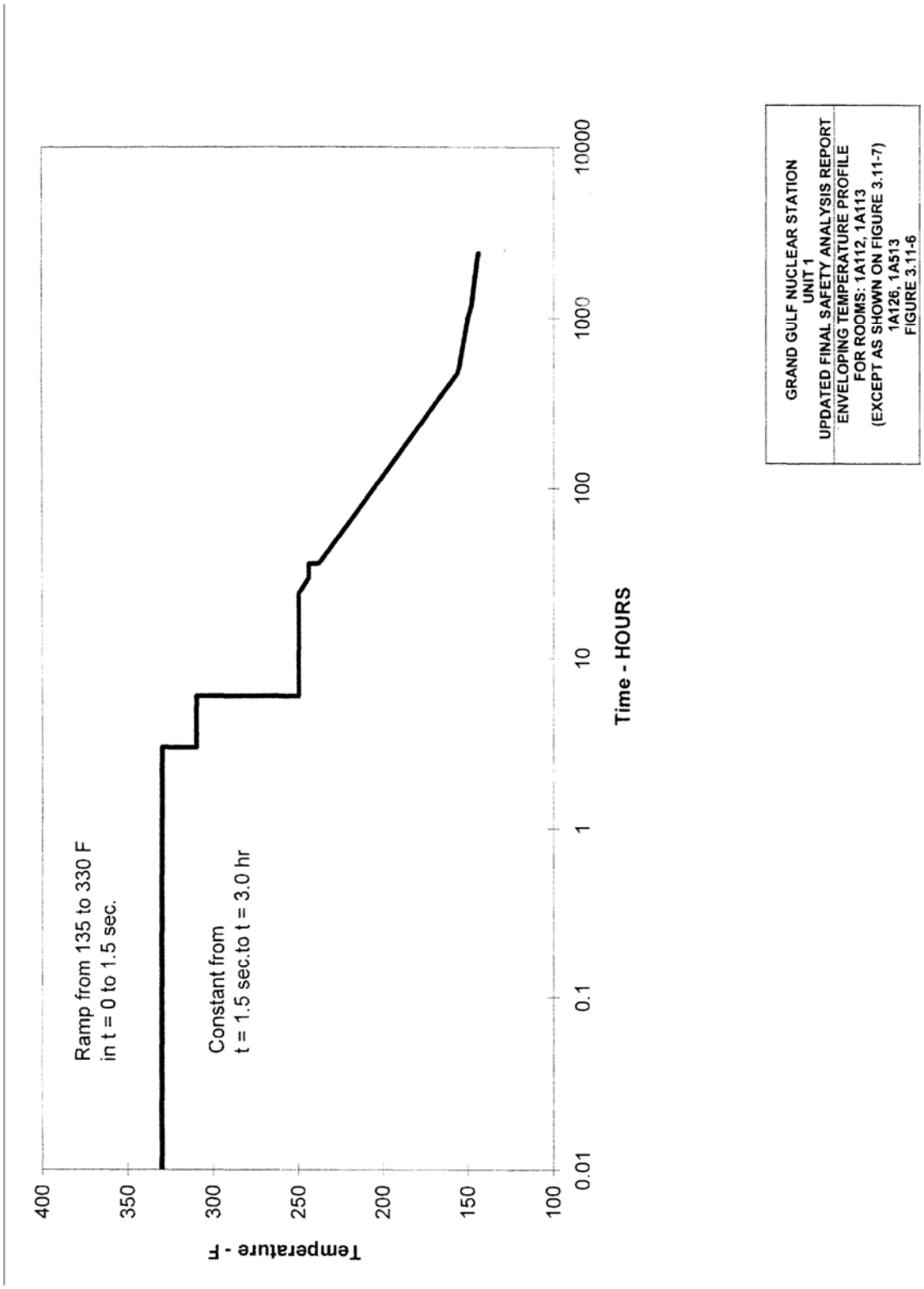
ENVELOPING PRESSURE PROFILE
FOR ROOMS:
1A112, 1A113, 1A126, 1A513
FIGURE 3.11-4

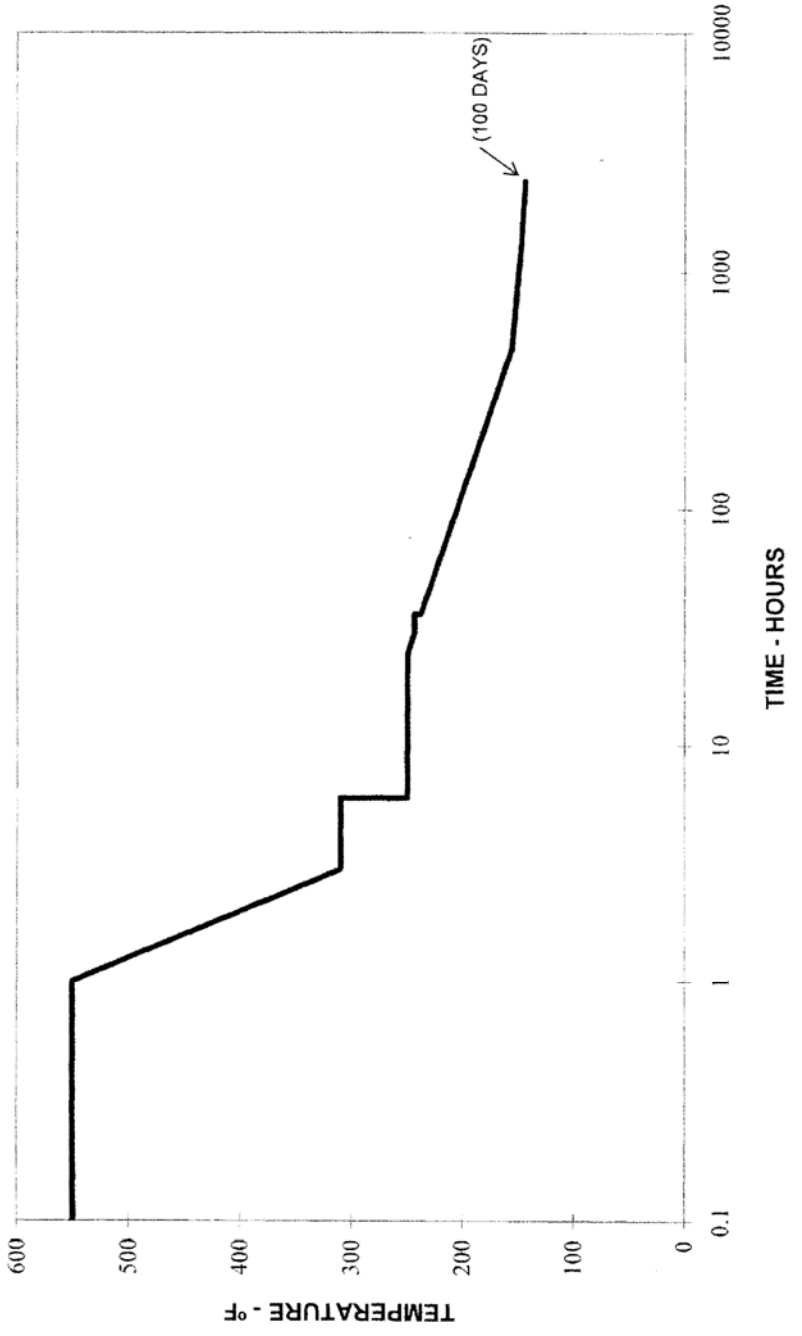


MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT

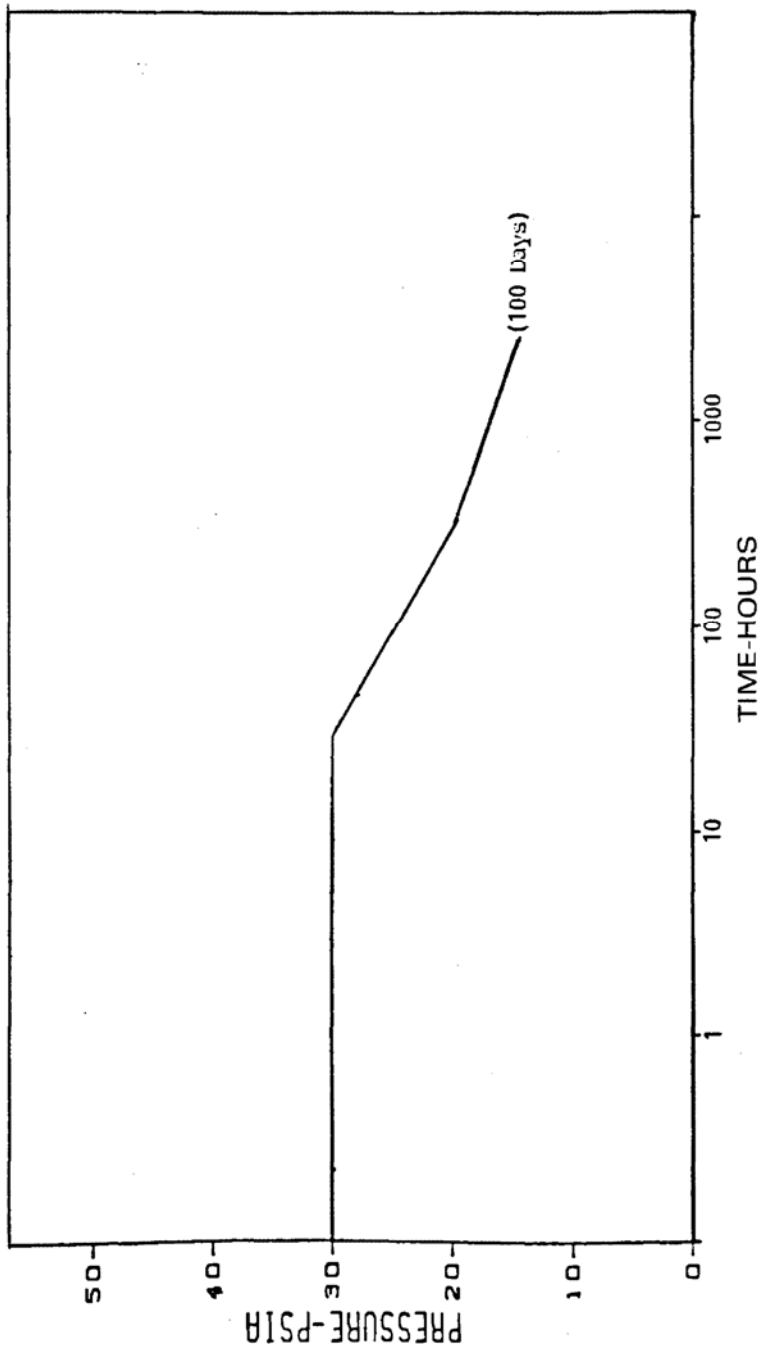
SUPPLEMENTAL PRESSURE PROFILE
FOR ROOMS:
1A112, 1A113, 1A126, 1A513

FIGURE 3.11-5

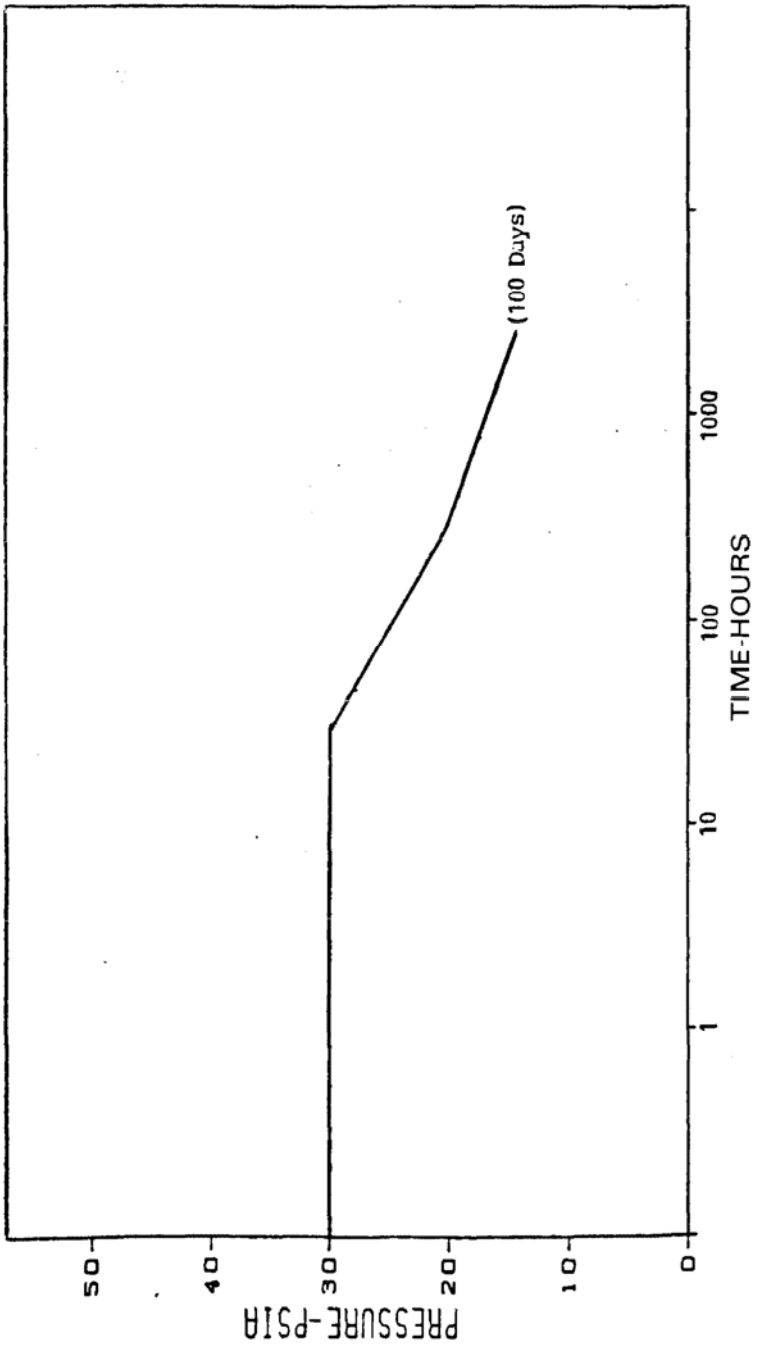

Temperature - F
400
350
300
250
200
150
100
0.01
0.1
1
10
100
1000
10000
Time - HOURS
Ramp from 135 to 330 F
in t = 0 to 1.5 sec.
Constant from
t = 1.5 sec.to t = 3.0 hr
GRAND GULF NUCLEAR STATION
UNIT 1
UPDATED FINAL SAFETY ANALYSIS REPORT
ENVELOPING TEMPERATURE PROFILE
FOR ROOMS: 1A112, 1A113
(EXCEPT AS SHOWN ON FIGURE 3.11-7)
1A126, 1A513
FIGURE 3.11-6



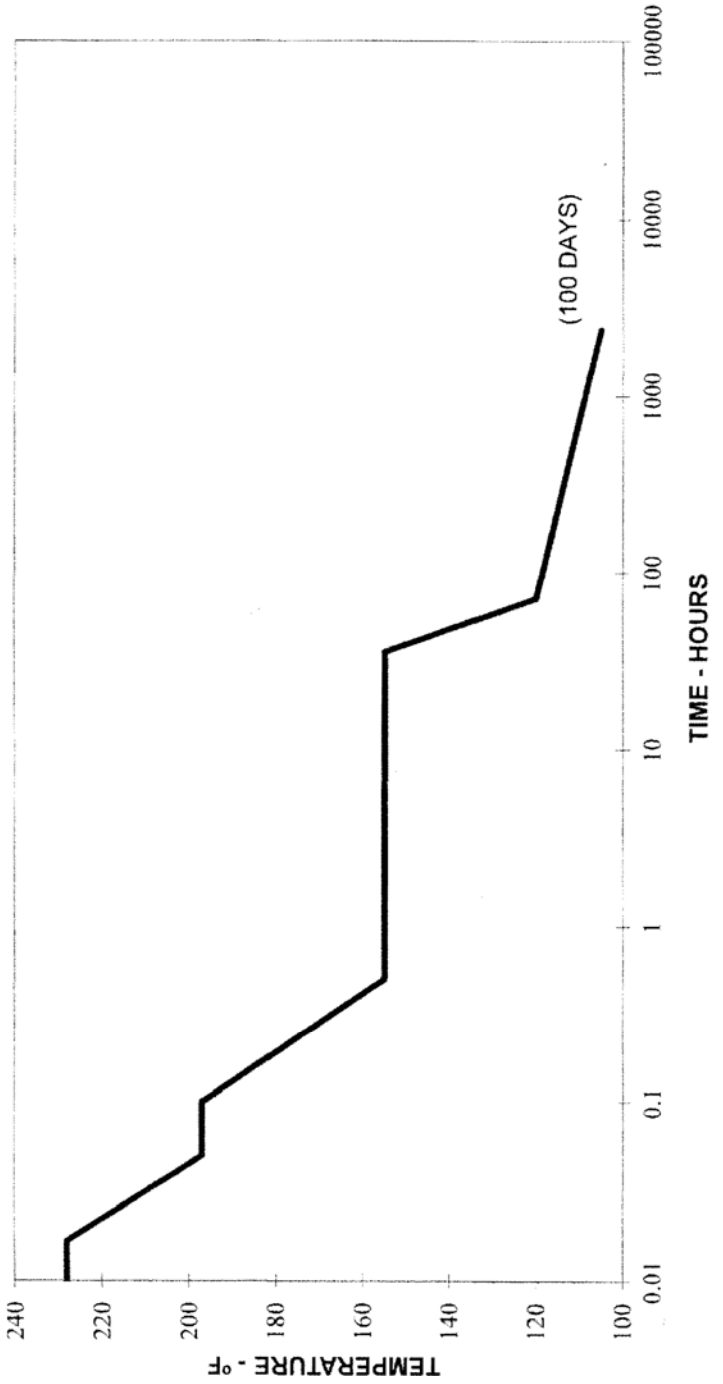
GRAND GULF NUCLEAR STATION
UNIT 1
UPDATED FINAL SAFETY ANALYSIS REPORT
ENVELOPING TEMPERATURE PROFILE
FOR ROOM: 1A113
(BETWEEN PRV AND SACRIFICIAL SHIELD
EL. 172.6' AND EL. 180.2')
FIGURE 3.11-7



MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT	ENVELOPING PRESSURE PROFILE FOR ROOM: 1A310 FIGURE 3.11-8

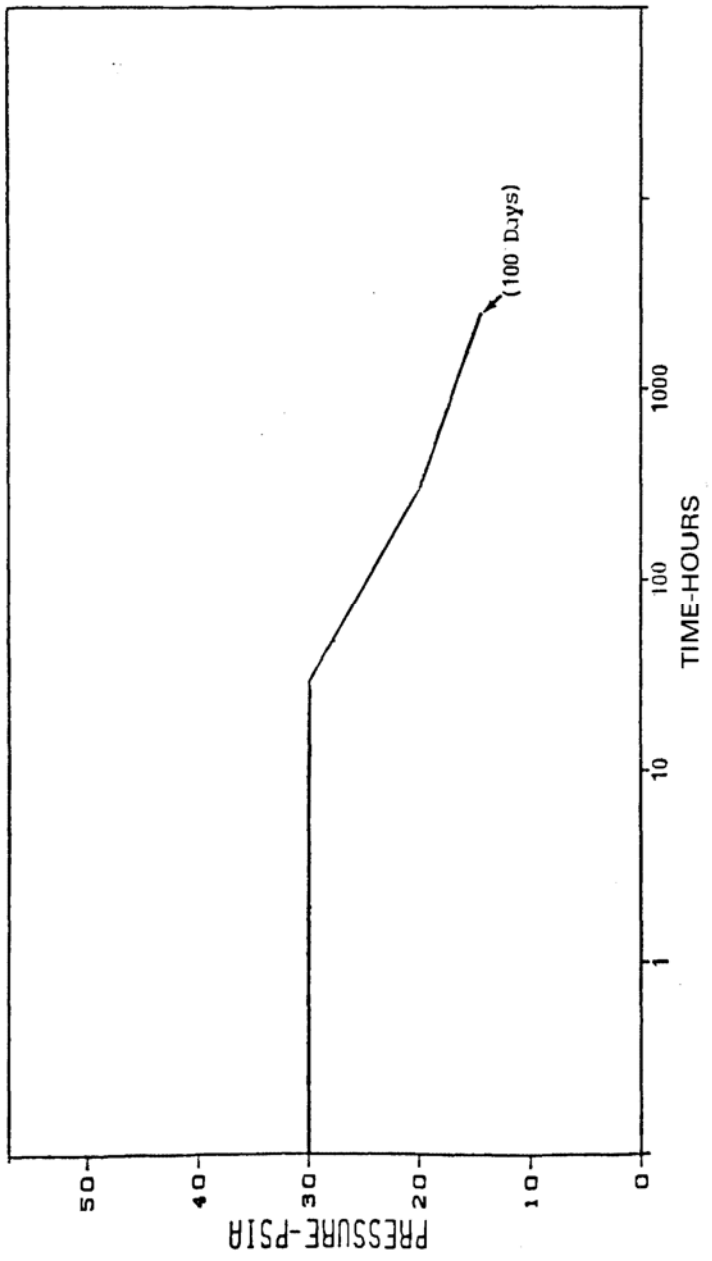


MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT	ENVELOPING PRESSURE PROFILE FOR ROOMS: 1A414, 1A507 FIGURE 3.11-9

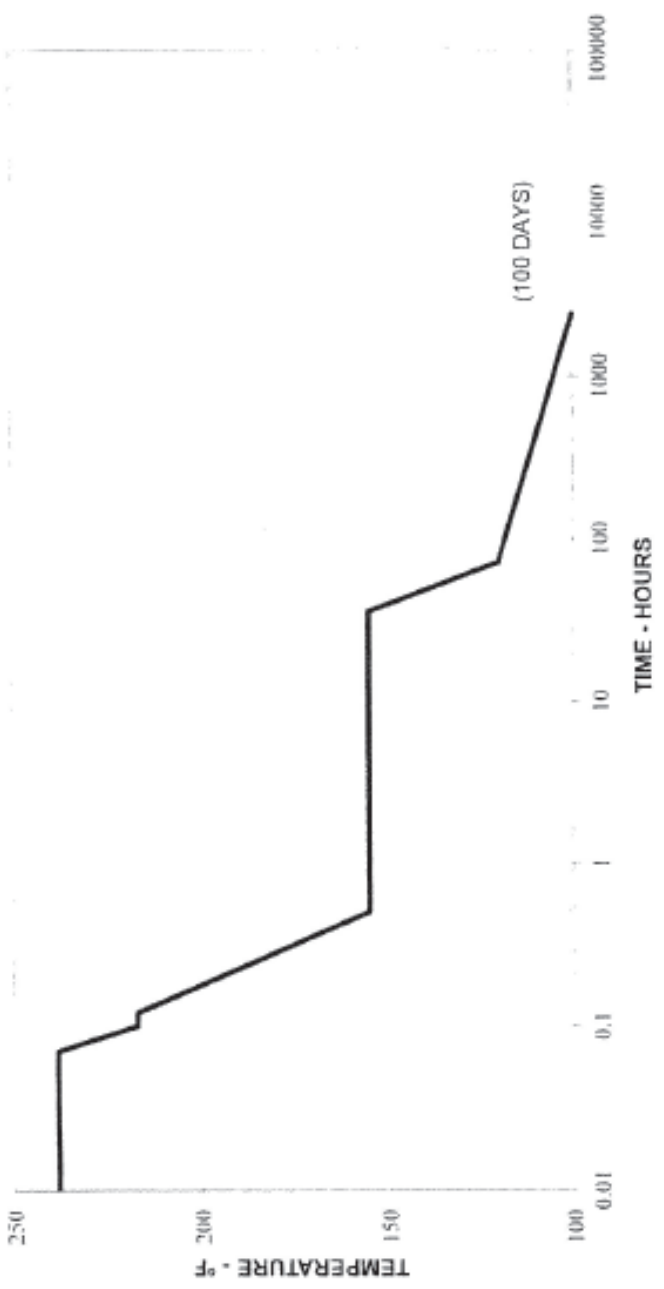


GRAND GULF NUCLEAR STATION
UNIT 1
UPDATED FINAL SAFETY ANALYSIS REPORT

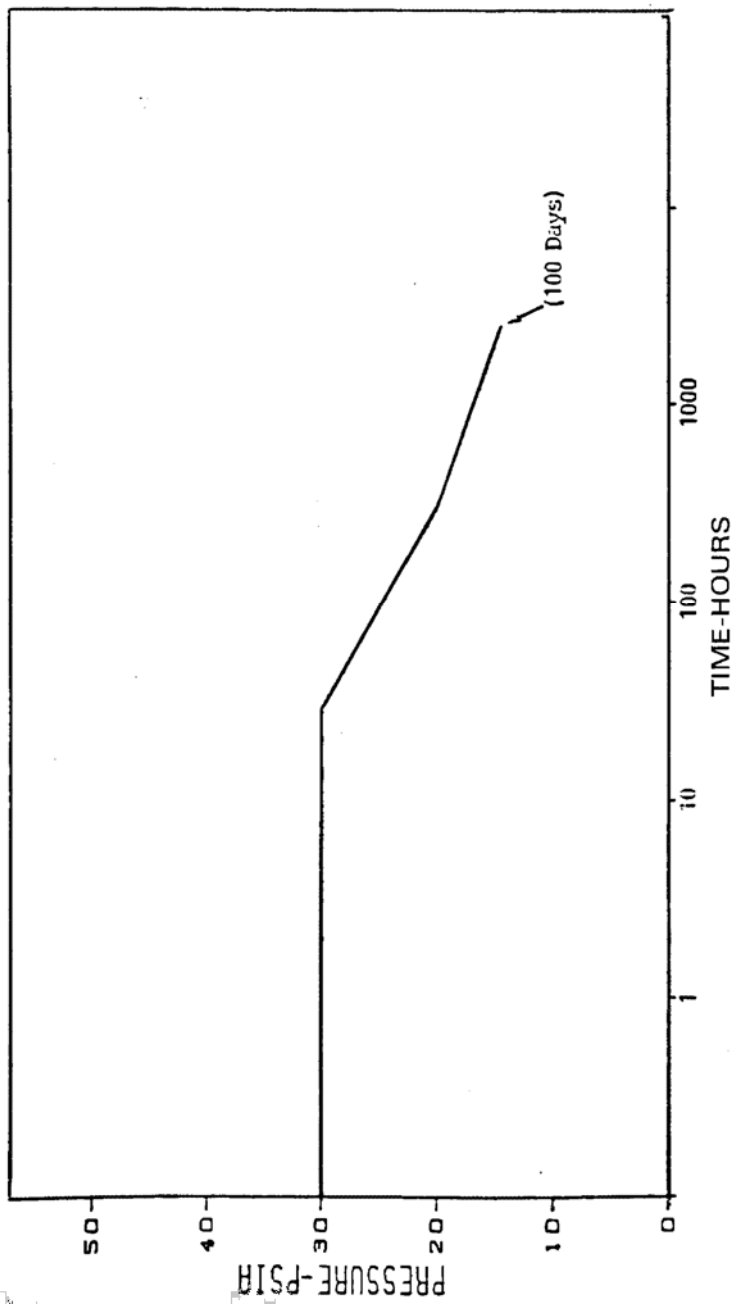
ENVELOPING TEMPERATURE PROFILE
FOR ROOMS:
1A310, 1A414, 1A507
FIGURE 3.11-10



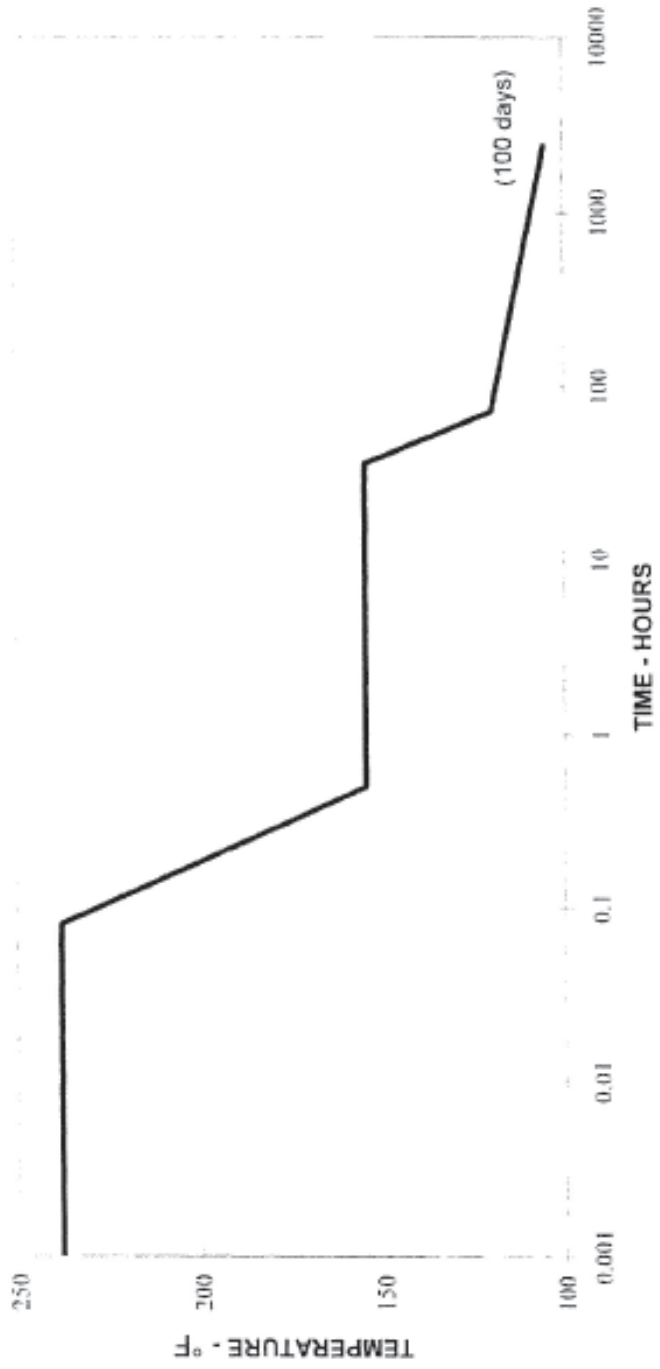
MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT
ENVELOPING PRESSURE PROFILE FOR ROOMS: 1A516, 1A517 FIGURE 3.11-11



GRAND GULF NUCLEAR STATION
UNIT 1
UPDATED FINAL SAFETY ANALYSIS REPORT
ENVELOPING TEMPERATURE PROFILE
FOR ROOMS:
1A318, 1A517
Figure 3.11-12

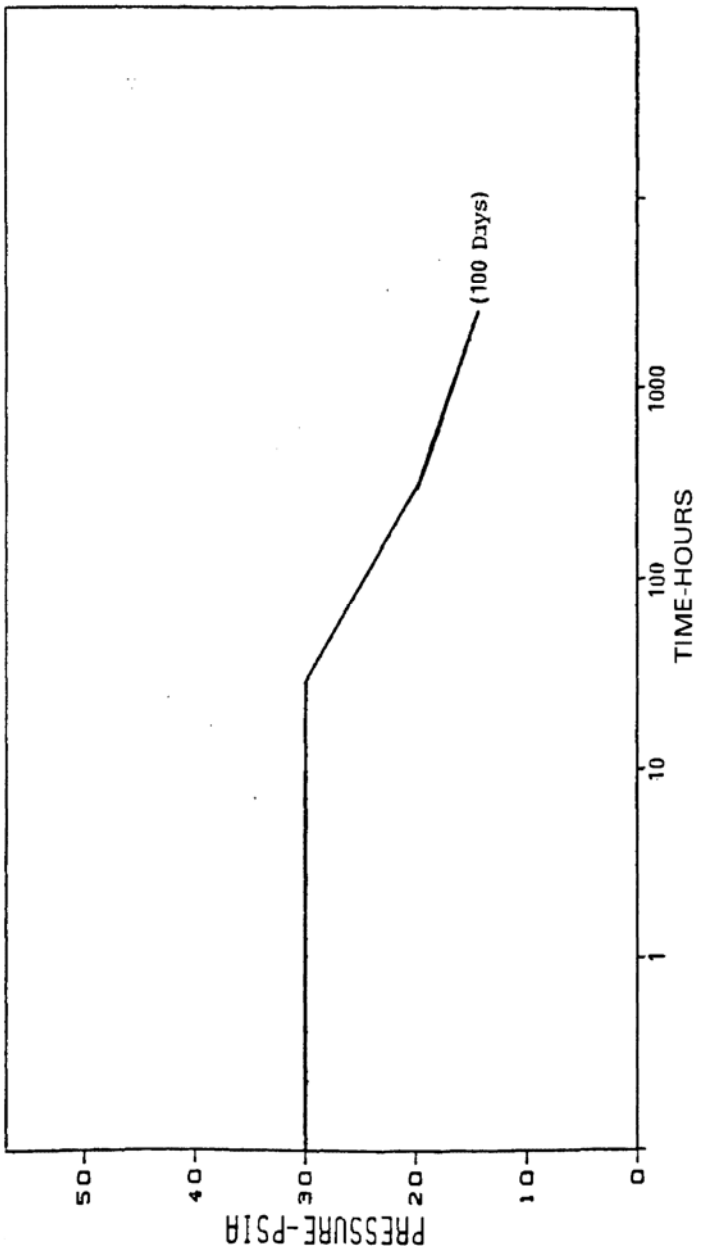


MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT	ENVELOPING PRESSURE PROFILE FOR ROOMS: 1A514, 1A515 FIGURE 3.11-13

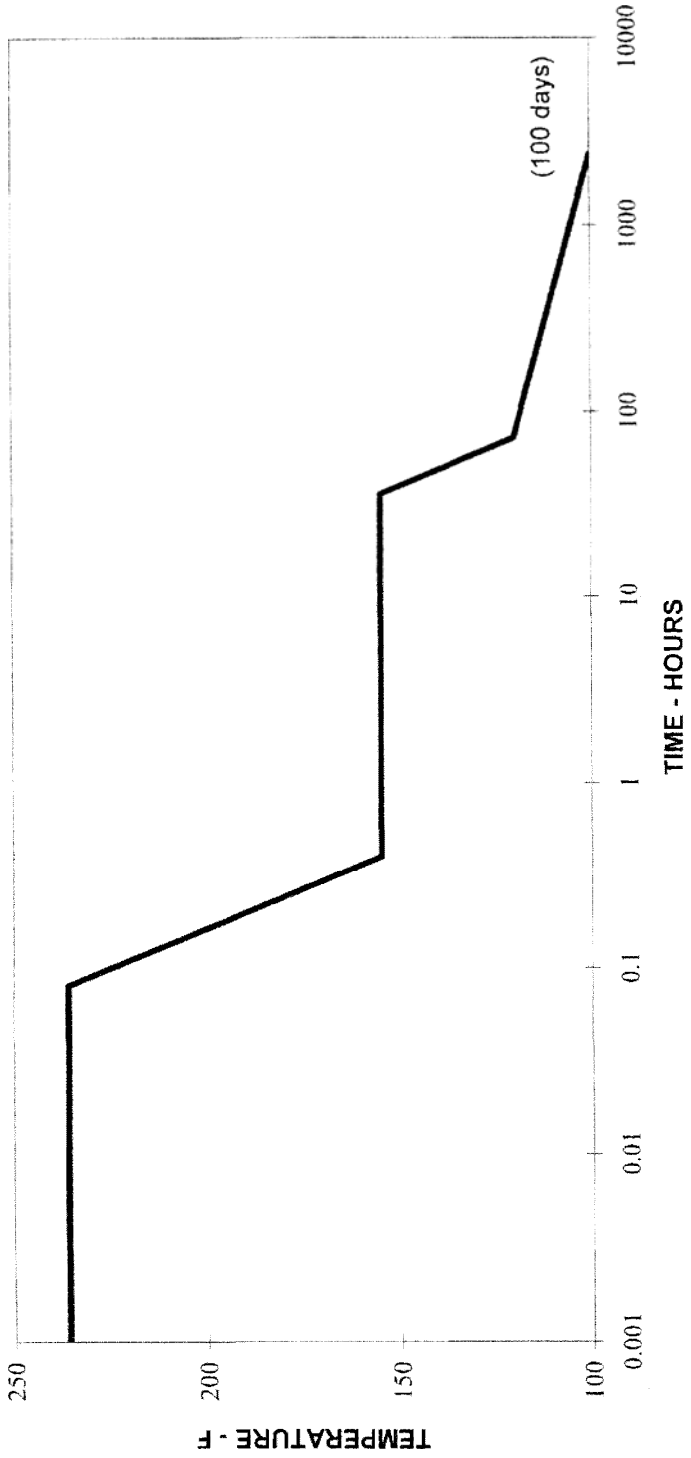


GRAND GULF NUCLEAR STATION
UNIT 1
UPDATED FINAL SAFETY ANALYSIS REPORT

ENVELOPING TEMPERATURE PROFILE
FOR ROOMS:
1A514, 1A515
FIGURE 3.11-14

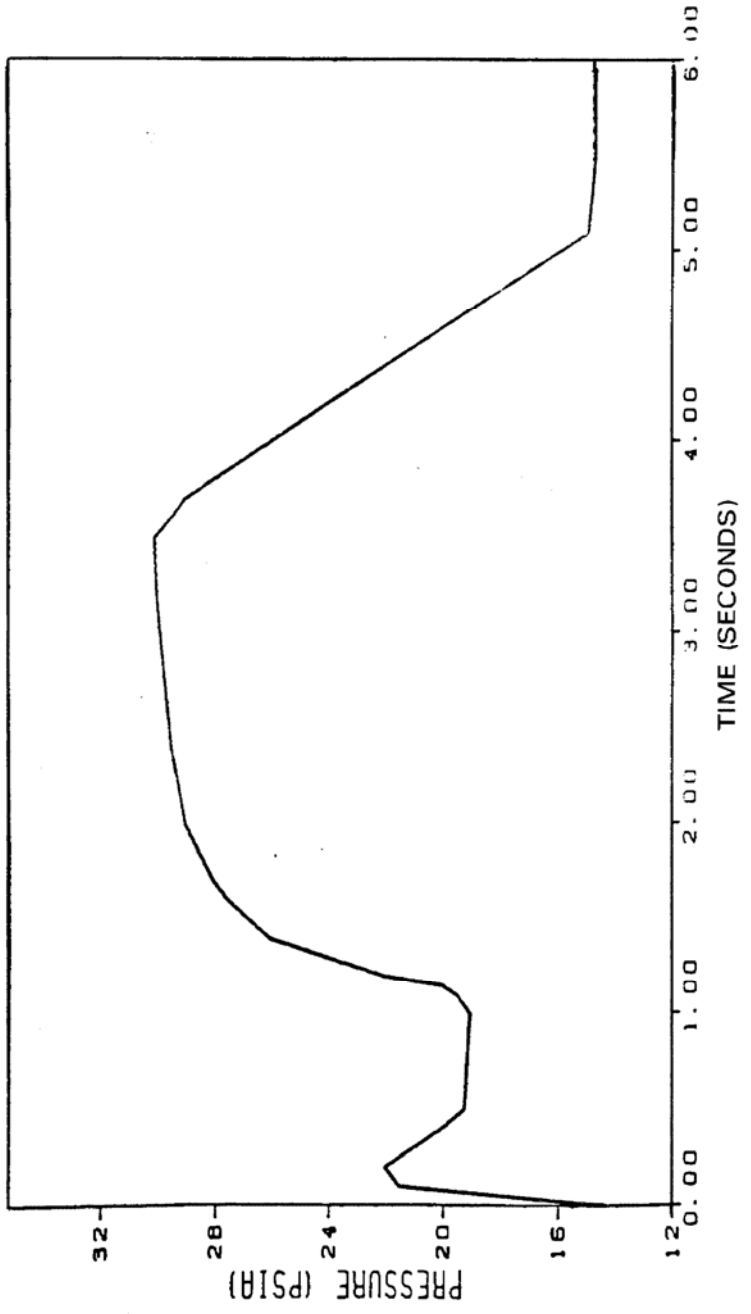


MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT
ENVELOPING PRESSURE PROFILE FOR ROOM: 1A443 FIGURE 3.11-15



GRAND GULF NUCLEAR STATION
UNIT 1
UPDATED FINAL SAFETY ANALYSIS REPORT

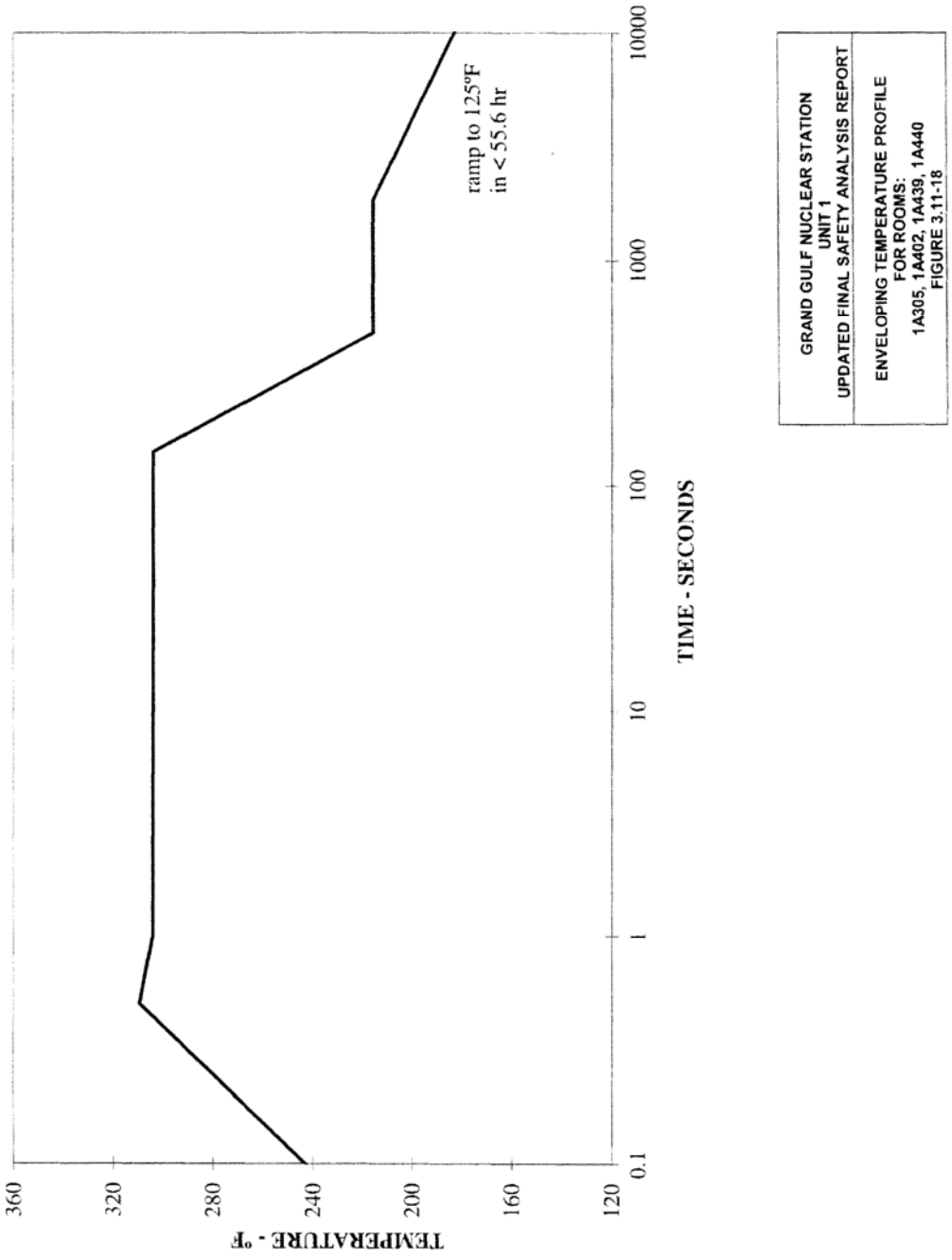
ENVELOPING TEMPERATURE PROFILE
FOR ROOMS:
1A443
FIGURE 3.11-16

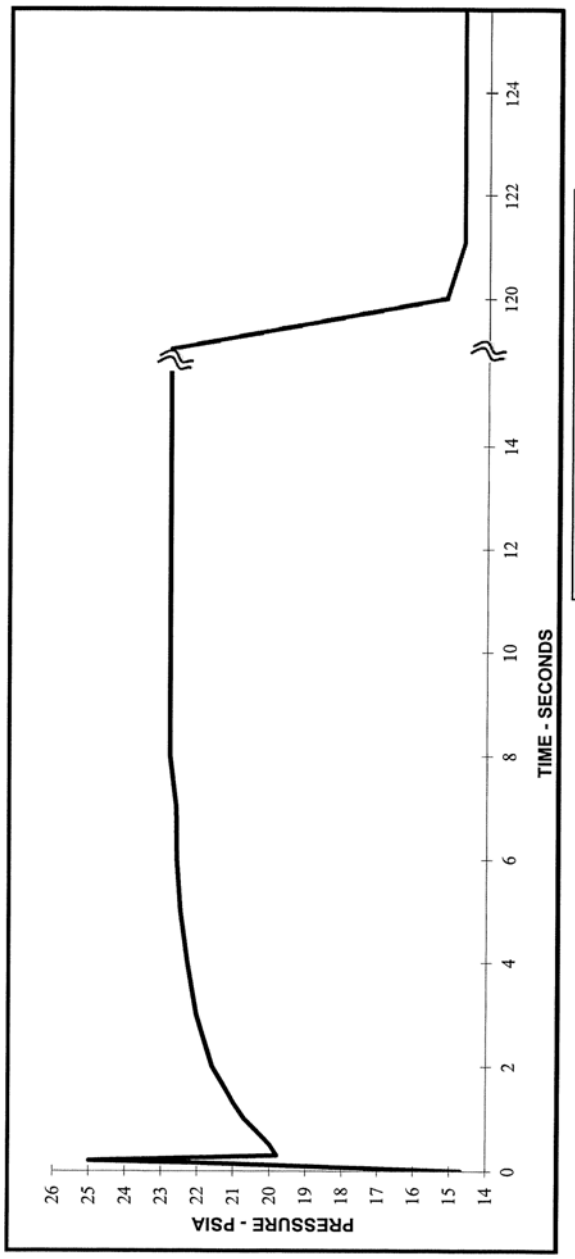


MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT

ENVELOPING PRESSURE PROFILE
FOR ROOMS:
1A305, 1A402, 1A439, 1A440

FIGURE 3.11-17


TEMPERATURE - °F
360
320
280
240
200
160
120
0.1
1
10
100
1000
10000
TIME - SECONDS
ramp to 125°F
in < 55.6 hr
GRAND GULF NUCLEAR STATION
UNIT 1
UPDATED FINAL SAFETY ANALYSIS REPORT
ENVELOPING TEMPERATURE PROFILE
FOR ROOMS:
1A305, 1A402, 1A439, 1A440
FIGURE 3.11-18

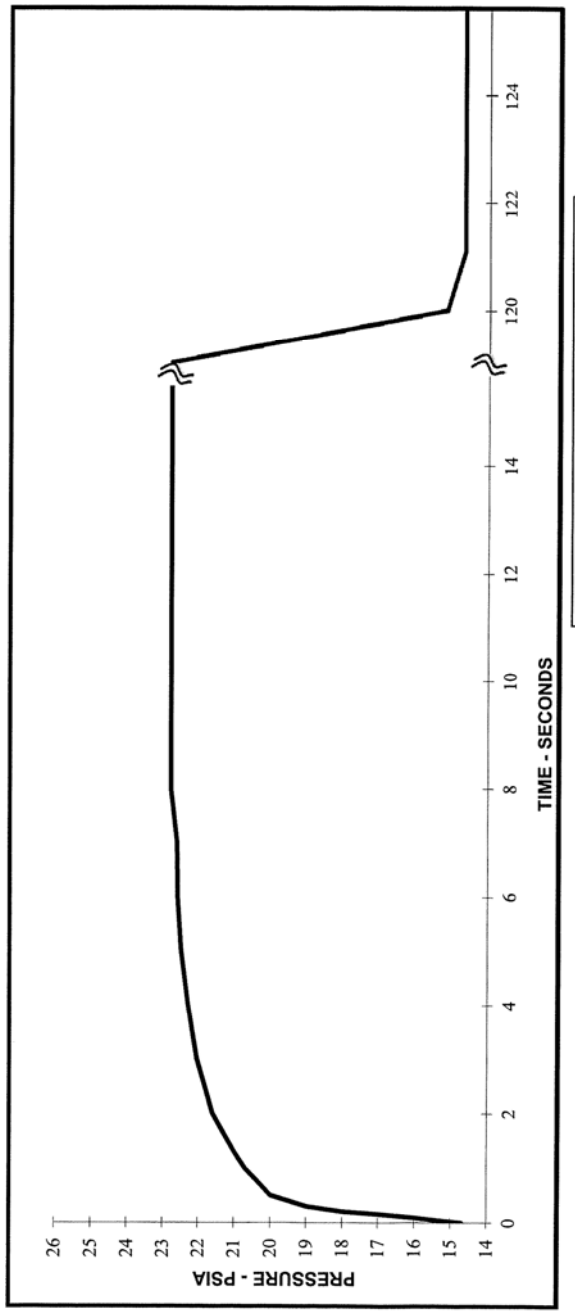


GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT

ENVELOPING PRESSURE PROFILE
FOR ROOMS:
1A104
FIGURE 3.11-19

Rev. 0

3.11-19

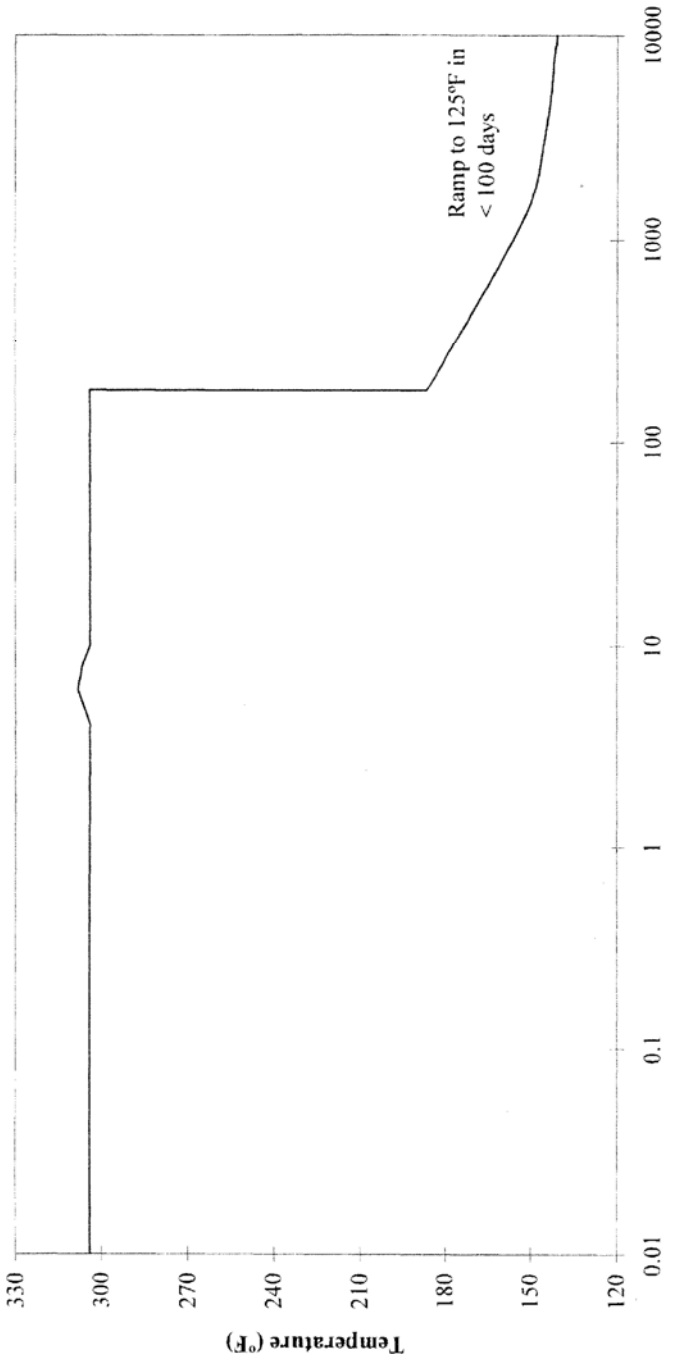


GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT

ENVELOPING PRESSURE PROFILE
FOR ROOMS:
1A204
FIGURE 3.11-20

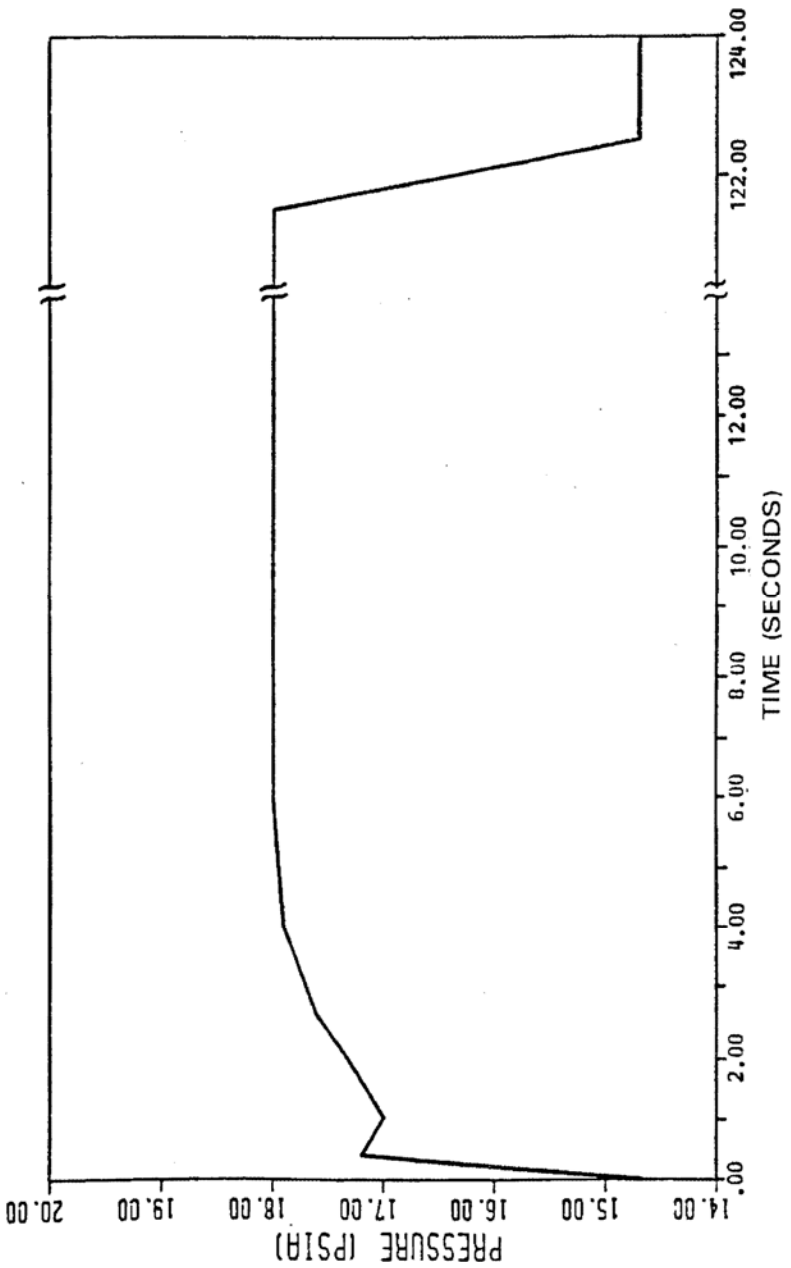
Rev. 0

3.11-20



GRAND GULF NUCLEAR STATION
UNIT 1
UPDATED FINAL SAFETY ANALYSIS REPORT

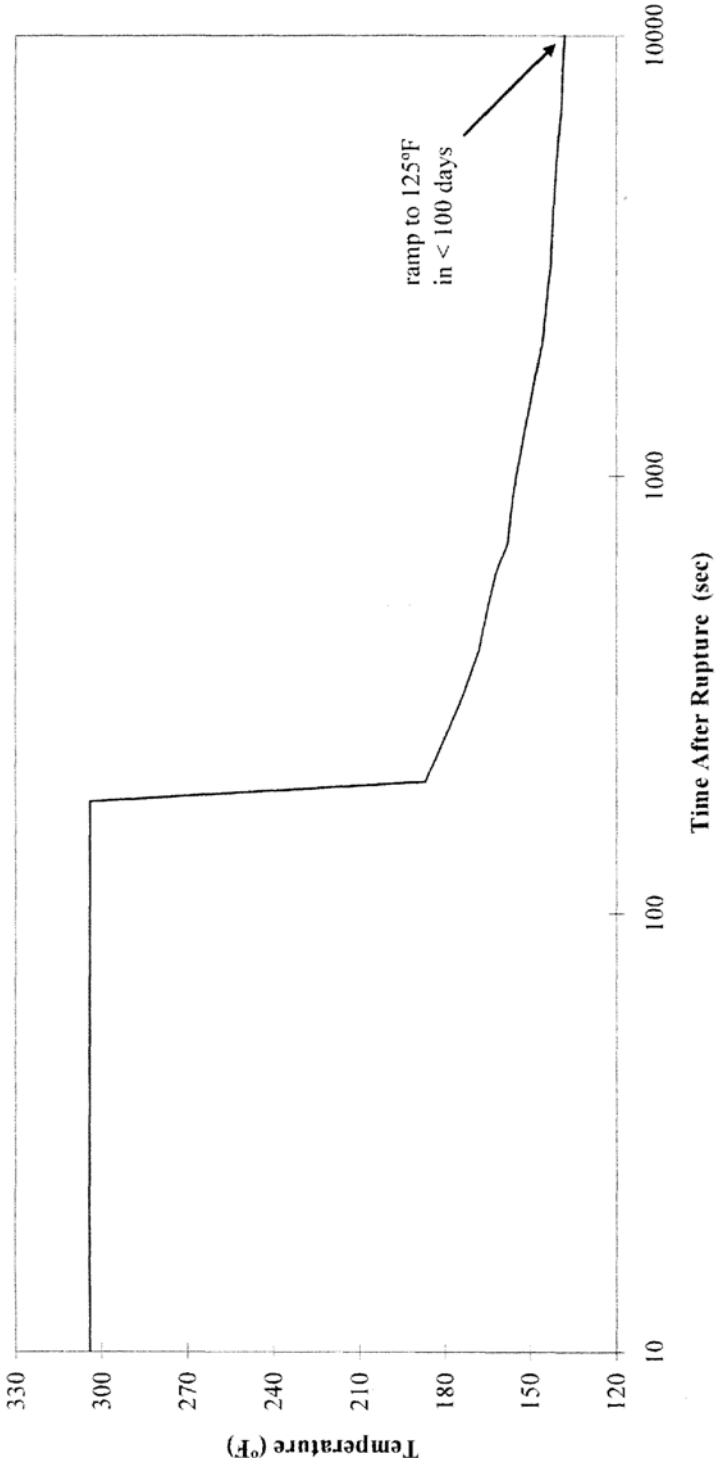
ENVELOPING TEMPERATURE PROFILE
FOR ROOMS:
1A104, 1A204
FIGURE 3.11-21



MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT

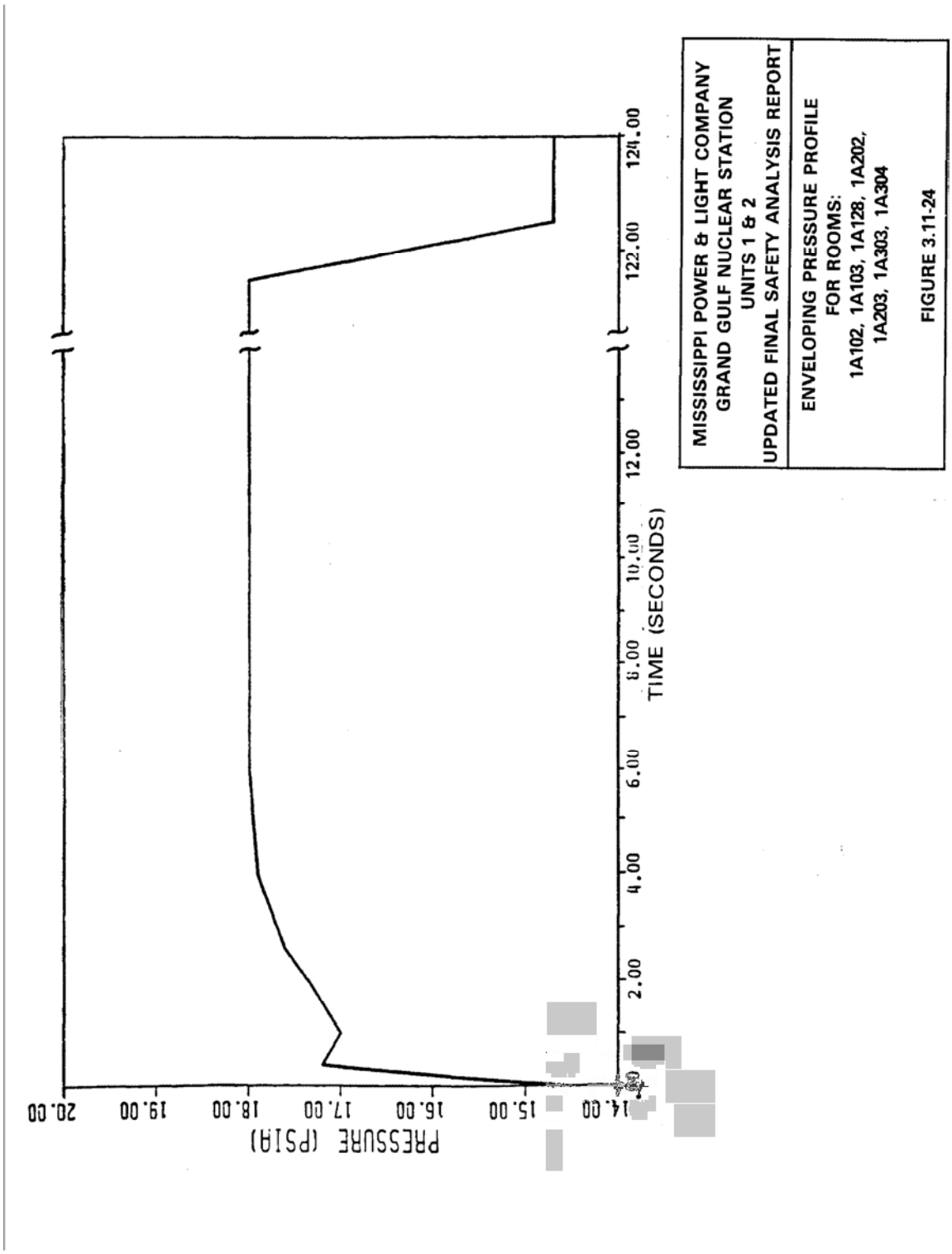
ENVELOPING PRESSURE PROFILE
FOR ROOMS: 1A124, 1A125

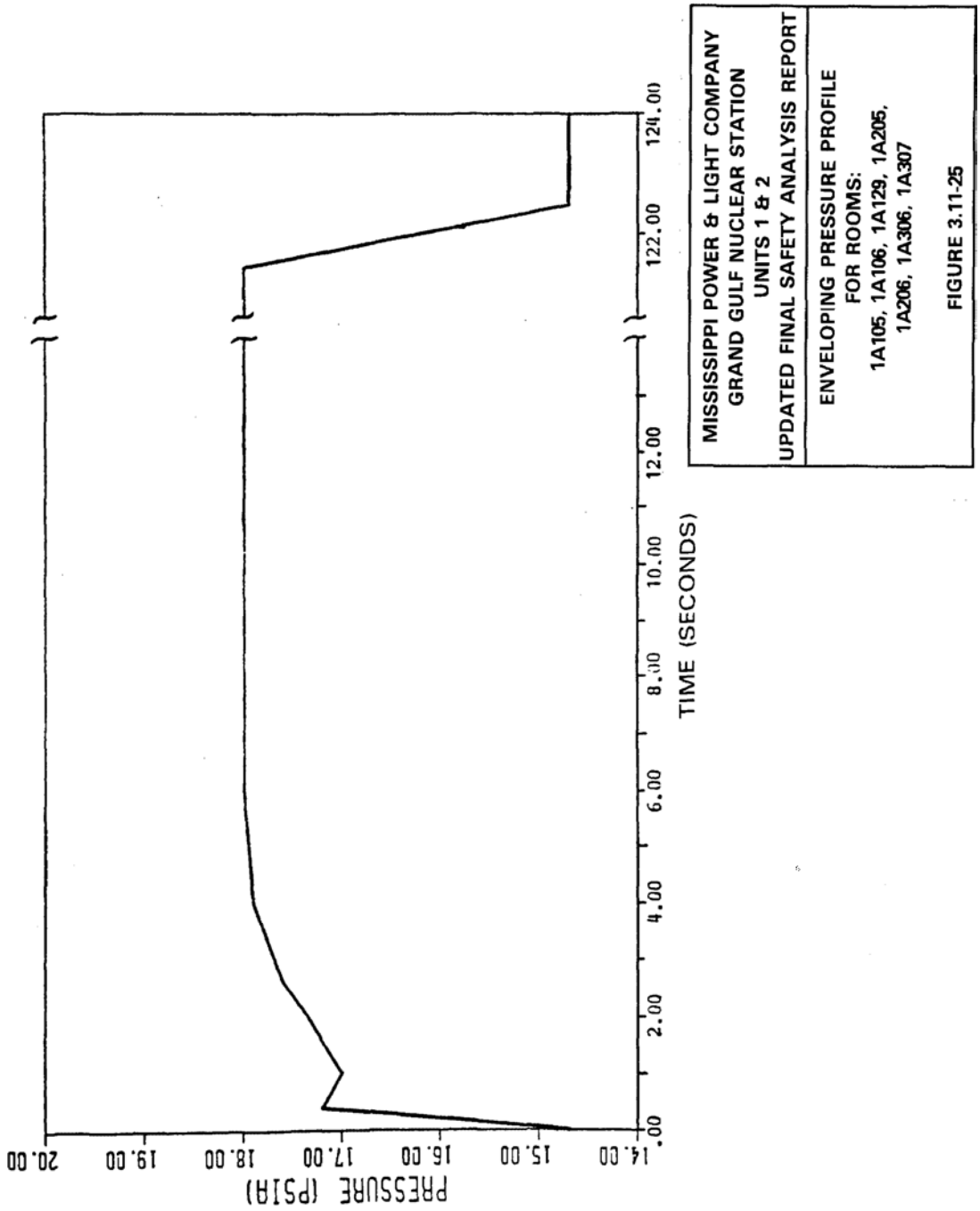
FIGURE 3.11-22

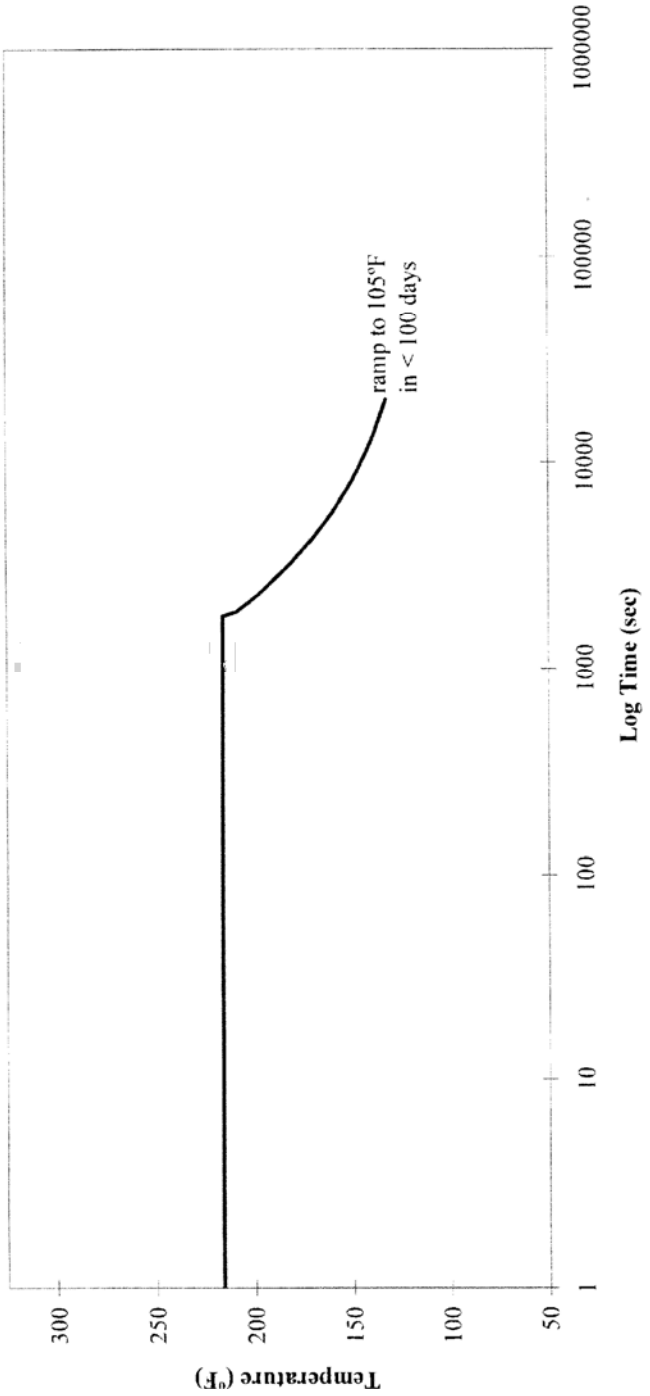


GRAND GULF NUCLEAR STATION
UNIT 1
UPDATED FINAL SAFETY ANALYSIS REPORT

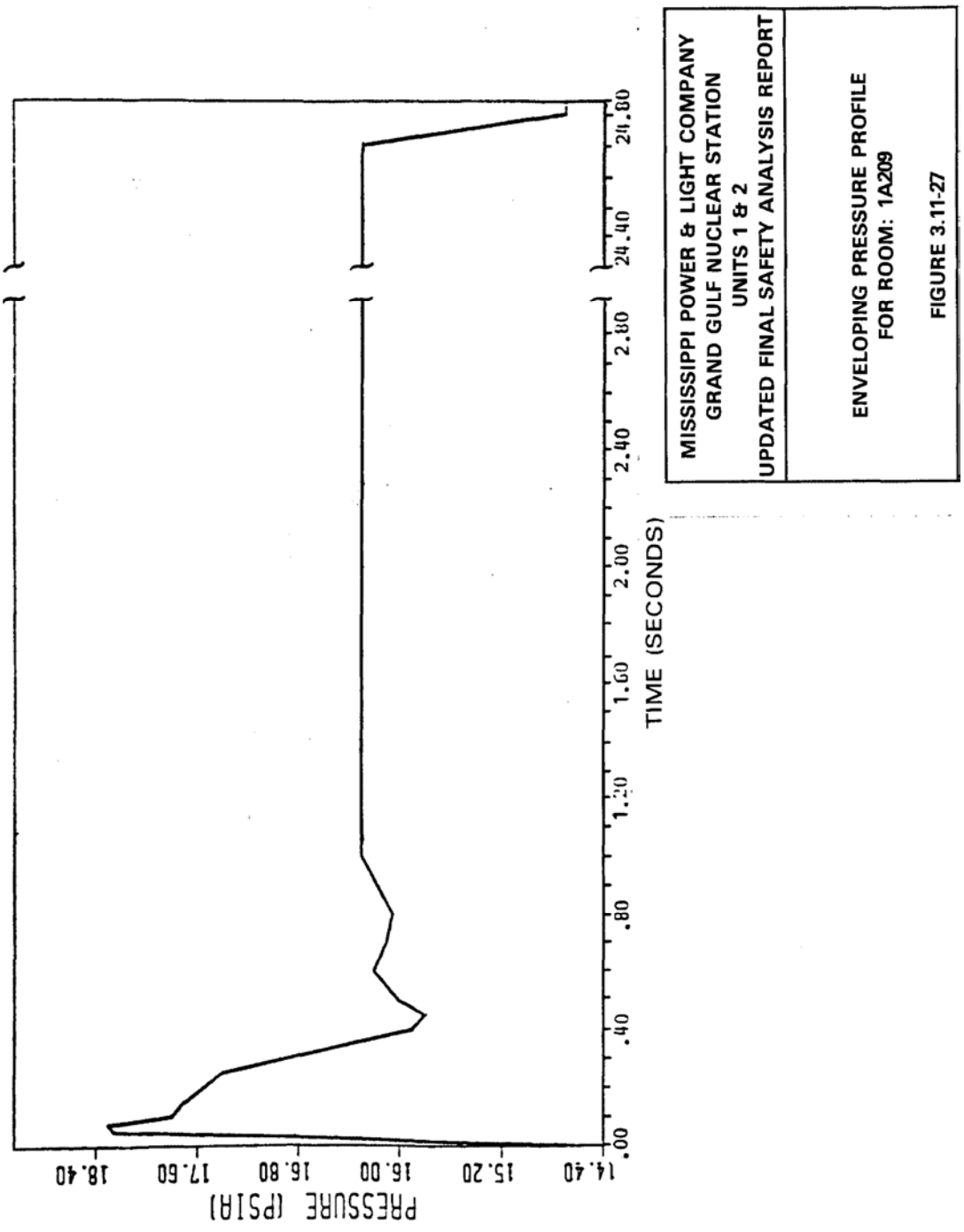
ENVELOPING TEMPERATURE PROFILE
FOR ROOMS:
1A124, 1A125,
FIGURE 3.11-23

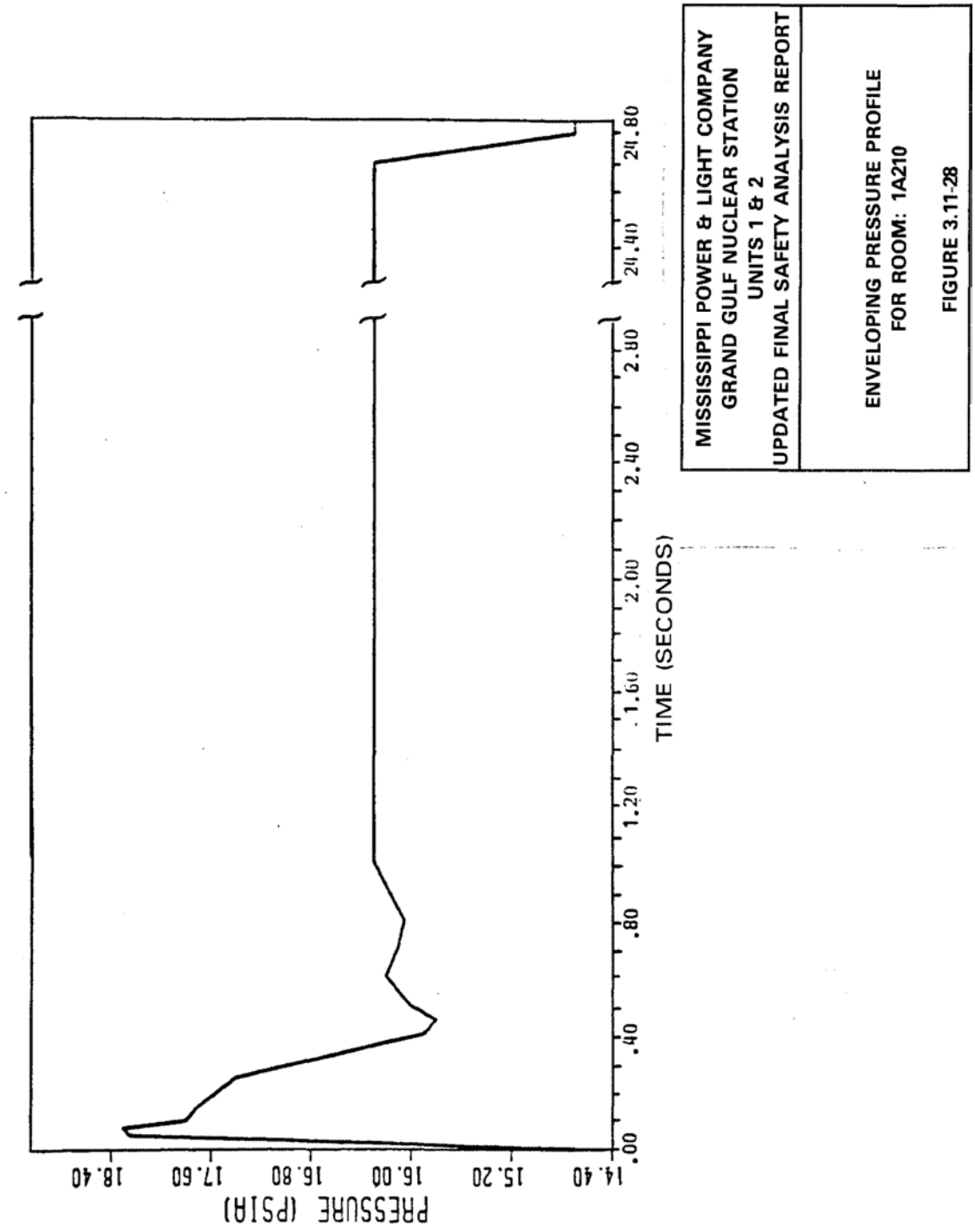

PRESSURE (PSIA)
14.00
15.00
16.00
17.00
18.00
19.00
20.00
TIME (SECONDS)
2.00
4.00
6.00
8.00
10.00
12.00
122.00
124.00
MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT
ENVELOPING PRESSURE PROFILE
FOR ROOMS:
1A102, 1A103, 1A128, 1A202,
1A203, 1A303, 1A304
FIGURE 3.11-24

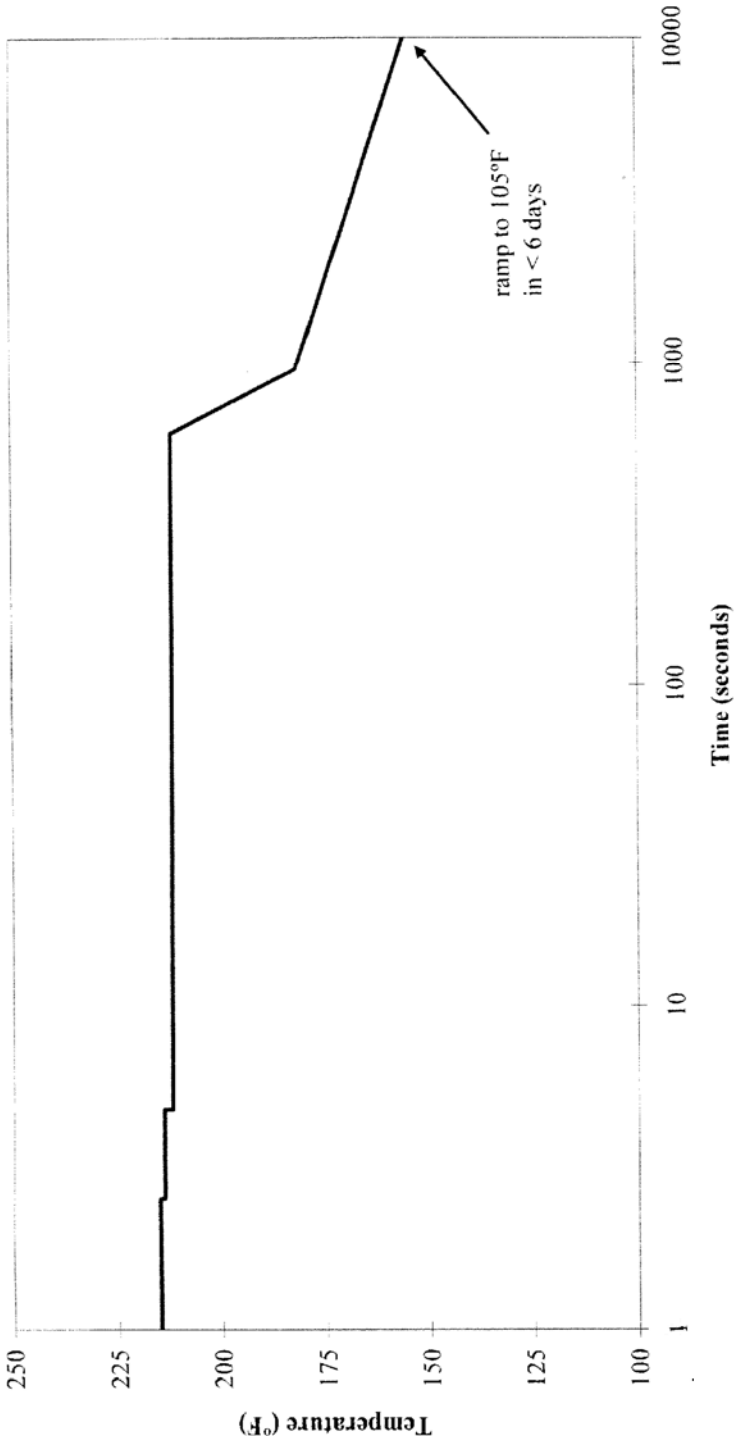

PRESSURE (PSIA)
14.00
15.00
16.00
17.00
18.00
19.00
20.00
.00
2.00
4.00
6.00
8.00
10.00
12.00
122.00
124.00
TIME (SECONDS)
MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT
ENVELOPING PRESSURE PROFILE
FOR ROOMS:
1A105, 1A106, 1A129, 1A205,
1A206, 1A306, 1A307
FIGURE 3.11-25



GRAND GULF NUCLEAR STATION
UNIT 1
UPDATED FINAL SAFETY ANALYSIS REPORT
ENVELOPING TEMPERATURE PROFILE
FOR ROOMS: 1A102, 1A103, 1A105, 1A106, 1A128, 1A129, 1A202, 1A203, 1A205, 1A206, 1A303, 1A304, 1A306, 1A307
FIGURE 3.11-26


PRESSURE (PSIA)
14.40
15.20
16.00
16.80
17.60
18.40
TIME (SECONDS)
.00
.40
.80
1.20
1.60
2.00
2.40
2.80
24.40
24.80
MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT
ENVELOPING PRESSURE PROFILE
FOR ROOM: 1A209
FIGURE 3.11-27


PRESSURE (PSIA)
14.40
15.20
16.00
16.80
17.60
18.40
.00
.40
.80
1.20
1.60
2.00
2.40
2.80
20.40
20.80
TIME (SECONDS)
MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT
ENVELOPING PRESSURE PROFILE
FOR ROOM: 1A210
FIGURE 3.11-28



GRAND GULF NUCLEAR STATION
UNIT 1
UPDATED FINAL SAFETY ANALYSIS REPORT

ENVELOPING TEMPERATURE PROFILE
FOR ROOMS:
1A209, 1A210
FIGURE 3.11-29

Parameter	Range	Duration
PRESSURE:	Atmospheric ± 1 inch wg	0-100 days
TEMPERATURE:	140 F	0-24 hours
	140 F to 80 F (ramp)	1-100 days
RELATIVE HUMIDITY:	100%	0-100 days

MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT

POST-ACCIDENT
PRESSURE AND TEMPERATURE PARAMETERS
IN AUXILIARY BUILDING (BREAK IN ROOM)
ROOMS: 1A117, 1A211, 1A316, 1A319

FIGURE 3.11-30

APPENDIX 3A CONFORMANCE WITH NRC REGULATORY GUIDES

This appendix discusses the conformance of plant design with the guidelines presented in the NRC Regulatory Guides. A reference to the FSAR section in which the applicable design features are described is also provided. Where the design differs from the Regulatory Guide, alternative methods of providing an equivalent level of safety have been utilized; these differences are either discussed in this appendix or reference is made to the appropriate FSAR section(s) in which they are discussed.

The following criteria were utilized to determine which Regulatory Guides were to be addressed in the FSAR:

a. Only new Regulatory Guides and subsequent revisions to existing Regulatory Guides issued through August 1977 are to be addressed.

b. Regulatory Guides issued after September 4, 1974 (the Grand Gulf Construction permit date), which are directly related to construction practices at the Grand Gulf jobsite, are not addressed.

c. Regulatory Guides issued after September 4, 1974 (the Grand Gulf construction permit date), which are related to quality assurance and quality control, are addressed only towards the operating license stage quality assurance program. These guides are addressed in Section 17.2 of the Grand Gulf FSAR.

d. Regulatory Guides which are excluded from mandatory NRC Staff review based on Grand Gulf's construction permit docket date, i.e., November 1972, as discussed in Section D, "Implementation," of the Regulatory Guide, are treated as an option and are only addressed when we wish to use the Regulatory Guide as it applies to our current design.

The following table of contents lists the Regulatory Guides and revisions which have been issued through August 1977. An asterisked note beside a Regulatory Guide indicates that the Regulatory Guide meets one of the above criteria and has, therefore, been included or excluded from this appendix accordingly.

DIVISION 1 REGULATORY GUIDES

Number	Title	Date	Page
1.1	Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps	12/70	3A/1.1-1
1.2	Thermal Shock to Reactor Pressure Vessels	12/70	3A/1.3-1
1.3	Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Boiling Water Reactors	6/74	3A/1.3-1
1.4	Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors	6/74	3A/1.4-1
1.5	Assumptions Used for Evaluating the Potential Radiological Consequences of a Steam Line Break Accident for Boiling Water Reactors	3/10/71	3A/1.5-1
1.6	Independence Between Redundant Standby (Onsite) Power Sources and Between Their Distribution Systems	3/10/71	3A/1.6-1
1.7	Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident	9/76	3A/1.7-1
***1.8	Personnel Selection and Training	9/75	3A/1.8-1
***1.8	Personnel Qualification and Training (Applicable only to Quality Programs Personnel)	12/79	3A/1.8-1a

*** Is applicable to Quality Programs (QP) personnel only as discussed in Section 17.2.

Number	Title	Date	Page
1.9, Rev.3	Selection, Design, Qualification and Testing of Emergency Diesel Generator Units used as Class 1E Onsite Electric Power Systems at Nuclear Power Plants	7/93	3A/1.9-1
1.10	Mechanical (Cadweld) Splices in Reinforcing Bars of Category I Concrete Structures	1/2/73	3A/1.10-1
1.11	Instrument Lines Penetrating Primary Reactor Containment	3/10/71	3A/1.11-1
1.12	Instrumentation for Earthquakes	4/74	3A/1.12-1
1.13	Spent Fuel Storage Facility Design Basis	12/75	3A/1.13-1
1.14	Reactor Coolant Pump Flywheel Integrity	8/75	3A/1.14-1
1.15	Testing of Reinforcing Bars for Category I Concrete Structures	12/28/72	3A/1.15-1
1.16	Reporting of Operating Information - Appendix A Technical Specifications	8/75	3A/1.16-1
1.17	Protection of Nuclear Power Plants Against Industrial Sabotage	6/73	3A/1.17-1
1.18	Structural Acceptance Test for Concrete Primary Reactor Containments	12/28/72	3A/1.18-1
1.19	Nondestructive Examination of Primary Containment Liner Welds	8/11/72	3A/1.19-1
1.20	Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing	5/76	3A/1.20-1
1.21	Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light Water Cooled Nuclear Power Plants	6/74	3A/1.21-1

Number	Title	Date	Page
1.22	Periodic Testing of Protection System Actuation Functions	2/17/72	3A/1.22-1
1.23	Onsite Meteorological Programs	2/17/72	3A/1.23-1
1.24	Assumptions Used for Evaluating the Potential Radiological Consequences of a Pressurized Water Reactor Radioactive Gas Storage Tank Failure	3/23/72	3A/1.24-1
1.25	Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors	3/23/72	3A/1.25-1
***1.26	Quality Group Classifications and Standards for Water, Steam, and Radioactive Waste Containing Components of Nuclear Power Plants	2/76	3A/1.26-1
1.27	Ultimate Heat Sink for Nuclear Power Plants	1/76	3A/1.27-1
****1.28	Quality Assurance Program Requirements (Design and Construction)	6/7/72	3A/1.28-1
***1.29	Seismic Design Classification	9/78	3A/1.29-1
***1.30	Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment	8/11/72	3A/1.30-1
1.31	Control of Stainless Steel Welding	6/73	3A/1.31-1
1.32	Criteria for Safety-Related Electric Power Systems for Nuclear Power Plants	2/77	3A/1.32-1

*** Is addressed as it relates to operational phase QA program only as discussed in Section 17.2.

**** Is replaced by ANSI N18.7 for operational phase QA program only as discussed in Section 17.2.

Number	Title	Date	Page
***1.33	Quality Assurance Program Requirements (Operation)	2/78	3A/1.33-1
1.34	Control of Electroslag Weld Properties	12/28/72	3A/1.34-1
1.35	Inservice Inspection of Ungrouted Tendons in Prestressed Concrete Containment Structures	1/76	3A/1.35-1
1.36	Nonmetallic Thermal Insulation for Austenitic Stainless Steel	2/23/73	3A/1.36-1
***1.37	Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants	3/73	3A/1.37-1
***1.38	Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage, and Handling of Items for Water-Cooled Nuclear Power Plants	5/77	3A/1.38-1
***1.39	Housekeeping Requirements for Water-Cooled Nuclear Power Plants	9/77	3A/1.39-1
1.40	Qualification Tests of Continuous-Duty Motors Installed Inside the Containment of Water-Cooled Nuclear Power Plants	3/73	3A/1.40-1
1.41	Preoperational Testing of Redundant Onsite Electric Power Systems to Verify Proper Load Group Assignments	3/73	3A/1.41-1
1.42	Interim Licensing Policy on As-Low-As Practicable for Gaseous Radioiodine Releases from Light-Water-Cooled Nuclear Power Reactors	(withdrawn 3/22/76)	

*** Is addressed as it relates to operational phase QA program only as discussed in Section 17.2.

Number	Title	Date	Page
1.43	Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components	5/73	3A/1.43-1
1.44	Control of the Use of Sensitized Stainless Steel	5/73	3A/1.44-1
1.45	Reactor Coolant Pressure Boundary Leakage Detection Systems	5/73	3A/1.45-1
1.46	Protection Against Pipe Whip Inside Containment	5/73	3A/1.46-1
1.47	Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems	5/73	3A/1.47-1
1.48	Design Limits and Loading Combination for Seismic Category I Fluid System Components	5/73	3A/1.48-1
1.49	Power Levels of Nuclear Power Plants	12/73	3A/1.49-1
1.50	Control of Preheat Temperature for Welding of Low-Alloy Steel	5/7	33A/1.50-1
1.51	Inservice Inspection of ASME Code Class 2 and 3 Nuclear Power Plant Components	(withdrawn 7/21/75)	
1.52	Design, Testing, and Maintenance Criteria for Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants	7/76	3A/1.52-1
1.53	Application of the Single-Failure Criterion to Nuclear Power Plant Protection Systems	6/73	3A/1.53-1
1.54	Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants	6/73	3A/1.54-1
* 1.55	Concrete Placement in Category I Structures	6/73	N/A

* Is not addressed in FSAR - relates to construction practices after the CP.

Number	Title	Date	Page
1.56	Maintenance of Water Purity in Boiling Water Reactors	7/78	3A/1.56-1
1.57	Design Limits and Loading Combination for Metal Primary Reactor Containment System Components	6/73	3A/1.57-1
***1.58	Qualification of Nuclear Power Plant Inspection, Examination, and Testing Personnel	8/73	3A/1.58-1
1.59	Design Basis Floods for Nuclear Power Plants	8/77	3A/1.59-1
1.60	Design Response Spectra for Seismic Design of Nuclear Power Plants	12/73	3A/1.60-1
1.61	Damping Values for Seismic Design of Nuclear Power Plants	10/73	3A/1.61-1
1.62	Manual Initiation of Protective Actions	10/73	3A/1.62-1
****1.63	Electric Penetration Assemblies in Containment Structures for Water-Cooled Nuclear Power Plants	10/73 and 2/87	3A/1.63-1
***1.64	Quality Assurance Requirements for the Design of Nuclear Power Plants	6/76	3A/1.64-1
1.65	Materials and Inspections for Reactor Vessel Closure Studs	10/73	3A/1.65-1
1.66	Nondestructive Examination of Tubular Products	(withdrawn 9/28/77)	3A/1.66-1
1.67	Installation of Overpressure Protective Devices	10/73	3A/1.67-1
1.68	Initial Test Programs for Water-Cooled Reactor Power Plants	1/77	3A/1.68-1
1.68.1	Preoperational and Initial Start-up Testing of Feedwater and Condensate Systems for Boiling Water Reactor Power Plants	1/77	3A/1.68.1-1

*** Is addressed as it relates to operational phase QA program only as discussed in Section 17.2.

****Regulatory Guide 1.63 February 1987 applies only to fiber optic penetration assemblies.

Number	Title	Date	Page
1.68.2	Initial Startup Test Program to Demonstrate Remote Shutdown Capability for Water-Cooled Nuclear Power Plants	1/77	3A/1.68.2-1
*1.69	Concrete Radiation Shields for Nuclear Power Plants	12/73	N/A
1.70	Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants-LWR Edition	9/75	3A/1.70-1
1.71	Welder Qualification for Areas of Limited Accessibility	12/73	3A/1.71-1
1.72	Spray Pond Plastic Piping	12/73	3A/1.72-1
1.73	Qualification Tests of Electric Valve Operators Installed Inside the Containment of Nuclear Power Plants	1/74	3A/1.73-1
***1.74	Quality Assurance Terms and Definitions	2/74	3A/1.74-1
1.75	Physical Independence of Electric Systems	1/75	3A/1.75-1
1.76	Design Basis Tornado for Nuclear Power Plants	4/74	3A/1.76-1
1.77	Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors	4/74	3A/1.77-1
1.78	Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release	6/74	3A/1.78-1
1.79	Preoperational Testing of Emergency Core Cooling Systems for Pressurized Water Reactors	9/75	3A/1.79-1

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* Is not addressed in FSAR - relates to construction practices after the CP.

*** Is addressed as it relates to operational phase QA program only as discussed in Section 17.2.

Number	Title	Date	Page
1.80	Preoperational Testing of Instrument Air Systems	6/74	3A/1.80-1
1.81	Shared Emergency and Shutdown Electric Systems for Multi-Unit Nuclear Power Plants	1/75	3A/1.81-1
1.82	Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident	5/96	3A/1.82-1
1.83	Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes	7/75	3A/1.83-1
1.84	Code Case Acceptability-ASME Section III Design and Fabrication	5/79	3A/1.84-1
1.85	Code Case Acceptability-ASME Section III Materials	5/79	3A/1.85-1
1.86	Termination of Operating Licenses for Nuclear Reactors	6/74	3A/1.86-1
1.87	Guidance for Construction of Class 1 Components in Elevated-Temperature Reactors (Supplement to ASME Section III Code Cases 1592, 1593, 1594, 1595, and 1596)	6/75	3A/1.87-1
***1.88	Collection, Storage, and Maintenance of Nuclear Power Plant Quality Assurance Records	10/76	3A/1.88-1
1.89	Qualification of Class IE Equipment for Nuclear Power Plants	11/74	3A/1.89-1
**1.90	Inservice Inspection of Pre-stressed Concrete Containment Structures with Grouted Tendons	8/77	3A/1.90-1

** Is not addressed in FSAR - Section D of R.G. indicates not applicable based on CP docket date.

*** Is addressed as it relates to operational phase QA program only as discussed in Section 17.2.

Number	Title	Date	Page
**1.91	Evaluation of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plant Sites	1/75	N/A
1.92	Combining Modal Responses and Spatial Components in Seismic Response Analysis	2/76	3A/1.92-1
1.93	Availability of Electric Power Sources	12/74	3A/1.93-1
***1.94	Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants	4/76	3A/1.94-1
1.95	Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release	1/77	3A/1.95-1
1.96	Design of Main Steam Isolation Valve Leakage Control Systems for Boiling Water Reactor Nuclear Power Plants	6/76	3A/1.96-1
1.97	Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident	12/80	3A/1.97-1

** Is not addressed in FSAR - Section D of R.G. indicates not applicable based on CP docket date.

*** Is addressed as it relates to operational phase QA program only as discussed in Section 17.2.

Number	Title	Date	Page
1.98	Assumptions Used for Evaluating the Potential Radiological Conse-quences of a Radioactive Offgas System Failure in a Boiling Water Reactor	3/76	3A/1.98-1
1.99	Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials	5/88	3A/1.99-1
1.100	Seismic Qualification of Electric Equipment for Nuclear Power Plants	3/76	3A/1.100-1
1.101	Emergency Planning for Nuclear Power Plants	3/77	3A/1.101-1
1.102	Flood Protection for Nuclear Power Plants	9/76	3A/1.102-1
1.103	Post-Tensioned Prestressing Systems for Concrete Reactor Vessels and Containments	10/76	3A/1.103-1
1.104	Overhead Crane Handling Systems for Nuclear Power Plants	2/76	3A/1.104-1
**1.105	Instrument Set Points	11/76	N/A
1.106	Thermal Overload Protection for Electric Motors on Motor-Operated Valves	3/77	3A/1.106-1
1.107	Qualifications for Cement Grouting for Prestressing Tendons in Containment Structures	2/77	3A/1.107-1
1.108	Periodic Testing of Diesel Generators Used as Onsite Electric Power Systems at Nuclear Power Plants	(with-drawn 7/19/93)	3A/1.108-1

** Is not addressed in FSAR - Section D of R.G. indicates not applicable based on CP docket date.

Number	Title	Date	Page
1.109	Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I	10/77	3A/1.109-1
**1.110	Cost-Benefit Analysis for Radwaste Systems for Light-Water-Cooled Nuclear Power Reactors	3/76	N/A
1.111	Methods for Estimating Atmospheric Transports and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors	7/77	3A/1.111-1
1.112	Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light-Water-Cooled Power Reactors	4/76	3A/1.112-1
1.113	Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I	4/77	3A/1.113-1
1.114	Guidance on Being Operator at the Controls of a Nuclear Power Plant	11/76	3A/1.114-1
1.115	Protection Against Low-Trajectory Turbine Missiles	7/77	3A/1.115-1
***1.116	Quality Assurance Requirements for Installation, Inspection, and Testing of Mechanical Equipment and Systems	6/76	3A/1.116-1
1.117	Tornado Design Classification	4/78	3A/1.117-1

** Is not addressed in FSAR - Section D of R.G. indicates not applicable based on CP docket date.

*** Is addressed as it relates to operational phase QA program only as discussed in Section 17.2.

Number	Title	Date	Page
**1.118	Periodic Testing of Electric Power and Protection Systems	6/76	N/A
1.119	Withdrawn		N/A
**1.120	Fire Protection Guidelines for Nuclear Power Plants	6/76	N/A
1.121	Bases for Plugging Degraded PWR Steam Generator Tubes	8/76	3A/1.121-1
1.122	Development of Floor Design Response Spectra for Seismic Design of Floor-Supported Equipment or Components	9/76	3A/1.122-1
***1.123	Quality Assurance Requirements for Control of Procurement of Items and Services for Nuclear Power Plants	7/77	3A/1.123-1
**1.124	Design Limits and Loading Combinations for Class 1 Linear-Type Component Supports	11/76	N/A
**1.125	Physical Models for Design and Operation of Hydraulic Structures and Systems for Nuclear Power Plants	3/77	N/A
1.126	An Acceptable Model and Related Statistical Methods for the Analysis of Fuel Densification	3/77	3A/1.126-1
1.127	Inspection of Water-Control Structures Associated with Nuclear Power Plants	3/78	3A/1.127-1
**1.128	Installation Design and Installation of Large Lead Storage Batteries for Nuclear Power Plants	4/77	N/A

** Is not addressed in FSAR - Section D of R.G. indicates not applicable based on CP docket date.

*** Is addressed as it relates to operational phase QA program only as discussed in Section 17.2.

Number	Title	Date	Page
1.129	Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Nuclear Power Plants	4/77	3A/1.129-1
**1.130	Design Limits and Loading Combinations for Class 1 Plate-and-Shell-Type Component Supports	7/77	N/A
**1.131	Qualification Tests of Electric Cables, Field Splices, and Connections for Light-Water-Cooled Nuclear Power Plants	8/77	N/A
1.137	Fuel-Oil Systems for Standby Diesel Generators	1/78	3A/1.137-1
1.144	Auditing of Quality Assurance Programs for Nuclear Power Plants	9/80	
1.146	Qualification of Quality Assurance Program Audit Personnel for Nuclear Power Plants	8/80	
1.183	Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors	7/00	3A/1.183-1

** Is not addressed in FSAR - Section D of R.G. indicates not applicable based on CP docket date.

DIVISION 4 REGULATORY GUIDES

Number	Title	Date	Page
*4.15	Quality Assurance for Radiological Monitoring Programs (Normal Operations) - Effluent Streams and the Environment	2/79	3A/4.15-1

* Is addressed as it relates to operational phase QA program only as discussed in Section 17.2.

DIVISION 8 REGULATORY GUIDES

Number	Title	Date	Page
8.1	Radiation Symbol	2/2/73	3A/8.1-1
8.2	Guide for Administrative Practices in Radiation Monitoring	2/2/73	3A/8.2-1
8.3	Film Badge Performance Criteria	2/2/73	3A/8.3-1
8.4	Direct-Reading and Indirect-Reading Pocket Dosimeters	6/11	3A/8.4-1
8.5	Immediate Evacuation Signal	2/16/73	3A/8.5-1
8.6	Standard Test Procedure for Gieger-Mueller Counters	5/73	3A/8.6-1
8.7	Instructions for Recording and Reporting Occupational Radiation Exposure Data	6/92	3A/8.7-1
8.8	Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be as Low as is Reasonably Achievable	3/77	3A/8.8-1
8.9	Acceptable Concepts, Models, Equations, and Assumptions for a Bioassay Program	7/93	3A/8.9-1
8.10	Operating Philosophy for Maintaining Occupational Radiation Exposures as Low as is Reasonably Achievable	9/75	3A/8.10-1
8.11	Applications of Bioassay for Uranium	6/74	3A/8.11-1
8.12	Criticality Accident Alarm Systems	12/74	3A/8.12-1
8.13	Instruction Concerning Prenatal Radiation Exposure	6/99	3A/8.13-1
8.14	Personnel Neutron Dosimeters	6/76	3A/8.14-1
8.15	Acceptable Programs for Respiratory Protection	10/76	3A/8.15-1

Number	Title	Date	Page
8.34	Monitoring Criteria and Methods to Calculate Occupational Radiation Doses	6/92	3A/8.34-1
8.35	Planned Special Exposures	6/92	3A/8.35-1
8.36	Radiation Dose to the Embryo/ Fetus	6/92	3A/8.36-1

Regulatory Guide 1.1, (December 1970)

Net Positive Suction Head for Emergency Cooling and Containment Heat Removal System Pumps

Project Position - Comply.

FSAR Subsections - 6.3.2, 5.4.7.2, 6.2.2.3

Regulatory Guide 1.2, (December 1970)

Thermal Shock to Reactor Pressure Vessels

Project Position - Comply.

FSAR Subsection - 5.3.3

Regulatory Guide 1.3, Rev. 2 (June 1974)

Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Boiling Water Reactors

Project Position - Superseded by Appendix A to Regulatory Guide 1.183.

FSAR Section/Subsection - 15.6, 3.11.5

Regulatory Guide 1.4, Rev. 2 (June 1974)

Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors

Project Position - Not applicable to BWRs.

Regulatory Guide 1.5, (March 1971)

Assumptions Used for Evaluating the Potential Radiological Consequences of a Steam Line Break Accident for Boiling Water Reactors

Project Position - Comply.

FSAR Section - 15.6

Regulatory Guide 1.6, (March 10, 1971)

Independence Between Redundant Standby (Onsite)
Power Sources and Between Their Distribution Systems

Project Position - Comply.

FSAR Subsections - 8.3.1 and 7.2.1.6

Regulatory Guide 1.7, Rev. 1 (September 1976)

Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident

Project Position - Comply.

FSAR Subsection - 6.2.5

Regulatory Guide 1.8, Rev. 1 (September 1975)

Personnel Selection and Training

Project Position - Grand Gulf complies with Regulatory Guide 1.8, Revision 1, with the following clarifications:

(1) Reference: Paragraph C of Regulatory Guide 1.8. When the phrase "bachelor's degree or equivalent" is used, the following qualifications are considered equivalent to a bachelor's degree:

a. 4 years of formal schooling in science or engineering,

b. 4 years of applied experience at a nuclear facility in the area for which qualification is sought,

c. 4 years of operational or technical experience/training in nuclear power, or

d. any combination of the above totaling 4 years.

(2) TS 5.3.1 requires staff qualifications to meet ANSI/ANS-3.1-1978, except as clarified in the Quality Assurance Program Manual. TS 5.2.2 also contains an exception to ANSI/ANS-3.1-1978 for the STA position.

FSAR Subsection - 12.5.1.1, 13.1.2

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Regulatory Guide 1.9, Revision 3 (July, 1993)

Selection, Design, Qualification, and Testing of Emergency Diesel Generator Units used as Class 1E Onsite Electric Power Systems at Nuclear Power Plants

Project Position - Grand Gulf complies with Regulatory Guide 1.9, Rev. 3, with the following exceptions:

Reference: Paragraph C.1.4 of the Regulatory Guide states, "During recovery from transients caused by the disconnection of the largest single load, the speed of the diesel generator set should not exceed the nominal speed plus 75 percent of the difference between nominal speed and the overspeed trip setpoint or 115 percent of nominal, whichever is lower." The largest single load for GGNS Division 2 Diesel Generator is the Standby Service Water (SSW) pump which supplies cooling water for the diesel generator as well as other systems. This load is not considered a transient because the loss of the SSW pump, whether or not it was the largest single load, will result in the loss of the diesel. The largest single load which satisfies the intent of this regulatory guide is either RHR B or C pump. Rejection of either load in testing required by the plant technical specifications verifies diesel generator capability to reject the load and maintain engine speed within specified values.

Paragraph C.1.4 of the Regulatory Guide also states, "Frequency should be restored to within 2 percent of nominal in less than 60 percent of each load-sequence interval for stepload increase and in less than 80 percent of each load-sequence interval for disconnection of the single largest load. "The Division 11 and 12 standby diesel generators restore frequency to within 2 percent of the pre-sequence value. The load sequencing is occurring during the diesel generator return to nominal speed after the initial overshoot upon startup. Therefore frequency is slightly higher than nominal plus 2 percent, but remains within five percent of nominal with

satisfactory recovery after each load application. Frequency stabilizes within 2 percent of nominal after approximately one minute of operation. Evaluation of operating for a short duration at a slightly higher frequency (within 5 percent of nominal) proves that there are no adverse effects to the diesel generators or to the safety-related loads energized by them. The quality of power provided by the diesel generators is such that the voltage and frequency is maintained "within limits that will not degrade the performance of any of the loads comprising the design load below their minimum requirements, including the duration of transients caused by load application or load removal" as stated in IEEE 387-1984, which is endorsed by Regulatory Guide 1.9, Rev. 3.

A statement in Regulatory Guide Paragraph 2.2 states that jumpers and other nonstandard configurations or arrangements should not be used subsequent to initial startup testing. This requirement is impractical since, for example, actual ECCS injection to the vessel is not desired during testing. Jumpers are utilized to simulate trip or start signals while manual valve manipulation prevents injection from actually occurring. This method meets the surveillance/testing requirement while providing a tightly controlled performance option to facilitate testing. Use and removal of these options is controlled by specific procedural step signoff for installation and removal.

Paragraph C.2.2.10 of the Regulatory Guide states in part, "Demonstrate hot re-start functional capability at full-load temperature conditions (after it has operated for 2 hours at full load) by verifying that the emergency diesel generator starts on a manual or autostart signal, attains the required voltage and frequency within acceptable limits and time, and operates for longer than 5 minutes."

Hot re-start functional capability is demonstrated after the emergency diesel generator has operated ≥1 hour or until operating temperatures have stabilized at full-load conditions. This pre-conditioning period (based on manufacturer's recommendations) has been

previously reviewed and approved. The current pre-conditioning measures meet the intent of the 2 hour run time required by C.2.2.10 of the Regulatory Guide.

There is also an exception to the specified 5 minute hot restart run time because operating evidence indicates that running the unloaded EDG longer than 5 minutes as stated can be detrimental to the machine. To meet the intent of the statement, surveillance procedures require verification that the Division I and II EDGs run for at least one minute (less than two minutes for Division III). As operating experience has demonstrated, these shorter time periods protect the EDGs from excessive oil residue buildup in the exhaust systems (a potential source of ignition during startup) while still providing assurance that they meet functional requirements during hot restart.

The Principal Design Criteria in Section 5 of IEEE 387-1977 was compared to that in IEEE 387-1984. The only difference is that the 1984 edition gives specifics when it comes to design considerations where the 1977 edition covers the same areas but list the overall area. In the area of fire protection, both editions list the Fire Protection System as a design and application consideration. The 1984 edition lists specifics that were apparently inherently understood in the 1977 edition. Examples include, separation between units, and possible damaging effects of actual or inadvertent fire protection system actuation. These examples were factored into the original design of the GGNS Diesel Generators. The design of the GGNS EDGs and associated fire protection was reviewed and approved by the NRC in NUREG-0831, the GGNS SER. Given these facts, GGNS remains committed to Section 5, “Principal Design Criteria,” of IEEE 387-1977 endorsed by Regulatory Guide 1.9, Revision 2.

GGNS Division 3 diesel generator bypasses the nonessential trips (i.e., trips other than engine overspeed and generator differential current) on a loss of bus voltage to satisfy IEEE 387-1977 Section 5.3, "Interactions", as endorsed by Regulatory Guide 1.9, Revision 2. This is implemented to preclude mechanical and electrical system interactions due to a postulated room depressurization by a tornado that would prevent the unit from meeting requirements of IEEE 387-1977, Section 5.1 for any design basis event.

FSAR Subsection - 8.3.1, 8.3.1.1.4.1.c, 8.3.1.2.1.a.4

Regulatory Guide 1.10, Rev. 1 (January 1973)

Mechanical (Cadweld) Splices in Reinforcing Bars of Category I Concrete Structures

Project Position - Comply.

FSAR Subsection - 3.8.1.6.6

Regulatory Guide 1.11, (March 10, 1971)

Instrument Lines Penetrating Primary Reactor Containment

Project Position - Comply.

FSAR Subsection - 6.2.4.2.4

Regulatory Guide 1.12, Rev. 1 (April 1974)

Instrumentation for Earthquakes

Project Position - The seismic instrumentation program complies with NRC Regulatory Guide 1.12, Rev. 1, except for the items listed below:

(1) Reference: Paragraph C.1.a(2) of the Regulatory Guide. The triaxial peak recording accelerographs on the reactor piping will not serve any meaningful purpose for post-seismic damage evaluation. The PRAs mounted on the reactor piping are subjected to accelerations greater than that predicted for a seismic event. Since the seismic event cannot be separated from the system transient response, no meaningful and conclusive evaluation can be performed. There are sufficient seismic instruments which will provide required information for post-seismic damage evaluation without the triaxial PRAs on the reactor piping. (Ref. AECM-86/0006 and MAEC-86/0169)

(2) Reference: Paragraph C.1.b and C.1.c of the Regulatory Guide. Response spectrum recorders are not supplied as discrete instruments. A response spectrum analyzer permanently installed in the control room presents more complete information than that presented by response spectrum recorders. Data from the strong-motion accelerometers are fed into the response spectrum analyzer to produce earthquake spectra immediately following an earthquake. All locations where response spectrum recorders are required by the Regulatory Guide are monitored by strong-motion accelerometers.

(3) Reference: Paragraph 5.2.2 of ANSI Standard N18.5-1974. The seismic instrumentation system does not consider frequencies below 1 Hz since the lower design limit of seismic Category I structures, systems, and devices is 1 Hz.

Seismic monitoring instrumentation is provided with a response that is essentially flat or equivalently correctable by computational techniques over the frequency range of 1 to 30 Hz.

FSAR Subsection - 3.7.4

Regulatory Guide 1.13, Rev. 1 (December 1975)

Spent Fuel Storage Facility Design Basis

Project Position - Comply.

FSAR Subsections - 9.1.2, 9.1.3, 9.1.4

Regulatory Guide 1.14, Rev. 1 (August 1975)

Reactor Coolant Pump Flywheel Integrity

Project Position - Not applicable to BWRs.

Regulatory Guide 1.15, Rev. 1 (December 28, 1972)

Testing of Reinforcing Bars for Category I Concrete Structures

Project Position - Testing of reinforcing bars for concrete seismic Category I structures complies with Regulatory Guide 1.15, Rev. 1. Retest requirements for tension of reinforcing bars conforms to ASTM A 615-75 as discussed in subsections 3.8.1, 3.8.3, and 3.8.4.

FSAR Subsections - 3.8.1, 3.8.3, 3.8.4

Regulatory Guide 1.16, Rev. 4 (August 1975)

Reporting of Operating Information - Appendix A
Technical Specifications

Project Position – With the approval of Operating License Amendment No. 167, effective 5/3/05, which deleted the Monthly Operating Report and the Occupational Radiation Exposure Report from Technical Specifications, this Regulatory Guide is no longer applicable to Grand Gulf Nuclear Station.

Regulatory Guide 1.17, Rev. 1 (June 1973)

Protection of Nuclear Power Plants
Against Industrial Sabotage

Regulatory Guide 1.17 became obsolete with the promulgation of 10CFR73.56 and Regulatory Guide 5.66. GGNS is committed to Regulatory Guide 5.66, as described in the Plant Security Plan.

Regulatory Guide 1.18, Rev. 1 (December 28, 1972)

Structural Acceptance Tests for Concrete Primary Reactor Containments

Project Position - The testing requirements of the containment, specified by Regulatory Guide 1.18, are discussed here (see subsections 3.8.1.7 and 3.8.3.7 for testing of internal structures). Regulatory Guide 1.18 establishes a systematic approach to testing wherein quantitative information is obtained concerning structural response to pressurization. Compliance with this guide is achieved insofar as is practicable. The following exceptions are considered to be within the intent of this Regulatory Guide.

(1) Reference: Paragraph C.2 of the Regulatory Guide. It is intended to select the number and distribution of measuring points for monitoring radial deflection so that the as-built condition can be considered in the assessment of out of tolerance ovaling and general shell response. Measurements are made at points similar to those shown in Figure 3.8-1. However, to obtain the most significant data, the measuring point locations may be changed to those where the as-built containment is at the limit of tolerances if such points exist. Accordingly, an arbitrary selection of measurement points is not intended.

(2) Reference: Paragraph C.3 of the Regulatory Guide. Measurement of tangential deflections is not planned. The magnitude of the expected local tangential deformation under the test pressure conditions is so negligibly small that, combined with the difficulty in obtaining fixed reference lines for local measurements, it is impractical to attempt measurement of local tangential deflection.

(3) Reference: Paragraph C.5 of the Regulatory Guide. This containment structure is not a prototype. Therefore, this paragraph is not applicable.

(4) Reference: Paragraph C.9 of the Regulatory Guide. It is intended to schedule structural integrity testing for periods when extremely inclement weather is not forecast. Should, despite the forecast, snow, heavy rain, or strong

wind occur during the test, the test results will be considered valid unless there is evidence to indicate otherwise.

(5) Reference: Paragraph C.10 of the Regulatory Guide. Should, due to an unexpected condition, the test pressure drop, it is intended to continue the test, without a restart at atmospheric pressure, unless the structural response deviates significantly from that expected.

(6) Reference: Appendix A.1(e). Main reinforcing is considered to include that which is required to resist internal pressure loads.

FSAR Subsection - 3.8.1.7

Note: The Grand Gulf project position is in compliance with the provisions of proposed Regulatory Guide 1.13b and Article CC-6000 of the 1980 edition of ASME Boiler and Pressure Vessel Code Section III, Division 2, as follows:

Article CC-6110: The project position complies to the extent that the Structural Integrity Test will be performed.

Article CC-6120: The project position complies in that test results and conclusions will be documented in the final test report.

Article CC-6211: The project position complies, except that four (4) pressure increments will be used.

Article CC-6212: The project position complies to the extent that the containment is not a prototype.

Article CC-6213: The project position complies in that specific acceptance criteria will be issued in the test procedure.

Article CC-6214: The project position complies.

Article CC-6221: The project position complies in that a complete test and instrumentation plan will be issued in the test procedure, which is a document separate from a design or construction specification.

Article CC-6222: The project position complies.

Article CC-6223: The project position complies.

Article CC-6231: The project position complies in that only deflection measurements will be taken as required fornon-prototype containments.

Article CC-6232a: The project position commits to several points to be located as noted in FSAR Appendix 3A, Project Position (1) for Regulatory Guide 1.18. The exact number and position of measurement points will be specified in the test procedure.

Article CC-6232b: The project position complies.

Article CC-6232c: The project position commits to a minimum of one vertical deflection measurement at the dome springline.

Article CC-6232d: The project position commits to a vertical deflection measurement at the apex of the dome.

Article CC-6232.1: The project position complies in that any reliable system may be used. Instrument specifications will be issued in the test procedure.

Article CC-6233: The project position complies.

Article CC-6234: Not applicable as the containment is not a prototype.

Article CC-6235: Not applicable as the containment is not prestressed.

Article CC-6236: Not applicable as strain measurements are not required for non-prototype containments.

Article CC-6237: The project position complies.

Article CC-6238: The project position complies in that reliable instrumentation will be used.

Article CC-6240: The project position complies in that a thorough examination of the structure will be made prior to pressure testing.

Article CC-6241: The project position complies.

Article CC-6242: Not applicable as strain measurements are not required for non-prototype containments.

Article CC-6251: The project position complies with the intent of this paragraph.

Article CC-6252: The project position complies, except as noted in FSAR Appendix 3A, Project Position (5) for Regulatory Guide 1.18.

Article CC-6253: The project position complies.

Article CC-6261: The project position complies with the exception that the acceptance criteria used will be specified in the test procedure.

Article CC-6262: The project position complies.

Article CC-6263: The project position complies.

Regulatory Guide 1.19, Rev. 1 (August 11, 1972)

Nondestructive Examination of Primary Containment Liner Welds

Project Position - The Project complies with NRC Regulatory Guide 1.19 with the following exceptions:

(1) Reference: Paragraph C.1.b. On those areas of liner seam welds where radiography is not feasible or where the weld is located in areas which will not be accessible after construction, the entire weld is examined by the magnetic-particle method in accordance with the techniques prescribed in Appendix VI of ASME Section VIII, Division 1, or the liquid-penetrant method in accordance with techniques prescribed in Appendix VIII of ASME Section VIII, Division 1.

(2) Reference: Paragraphs C.2.a and C.2.c. All nondestructive examinations are performed in accordance with the techniques prescribed in Paragraph NE-5120 of ASME Section III.

(3) Reference: Paragraph C.4. All nondestructive examination personnel are qualified in accordance with the requirements of SNT-TC-lA.

(4) Reference: Paragraphs C.2.b and C.7.d. All radiographic examinations are performed in accordance with the techniques prescribed in Article 3 of ASME Section V and the acceptance standards of Paragraph UW-51 of ASME Section VIII, Division 1. Alternatively, magnetic-particle examination in accordance with the procedures and acceptance standards prescribed in Appendix VI of ASME Section VIII, Division 1, or liquid-penetrant examination in accordance with the procedures and acceptance standards prescribed in Appendix VIII of ASME Section VIII, Division l, are acceptable.

FSAR Section - 3.8

Regulatory Guide 1.20, Rev. 2 (May 1976)

Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing

Project Position - Comply.

FSAR Subsections - 1.5.1, 3.9.2, 14.2.12.3.30

Regulatory Guide 1.21, Rev. 1 (June 1974)

Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants

Project Position - Comply with the following exception. RG 1.21 endorses ANSI N13.1-1969. The turbine building occasional release point does not meet all the ANSI N13.1 guidelines for sampling airborne radioactive materials. The guidelines recommend multiple sample points for ducts of this size and longer undisturbed upstream and down stream lengths for obtaining representative samples.

In modes 1, 2, 3, the radioactive particles to be sampled are very small and uniformly mixed with the air. Multiple probes or undisturbed lengths of duct are not required to obtain a representative sample in this specific application. The southeastern most smoke hatch, with proper monitoring equipment, will be used.

Comply with the following exception. In Modes 4 and 5 up to four turbine building roof hatches may be opened without direct monitoring. Alternate means for monitoring Turbine Building Hatch flow rates will be used. The radionuclide concentration from the Turbine Building Ventilation fan discharge will be used for the source term. For this mode of operation the filter bypass will be open. The source term will be periodically monitored and limited to ≤30% of the ODCM 6.11.4 and 6.11.6 dose limit.

Regulatory Guide 1.22, (February 1972)

Periodic Testing of Protection
System Actuation Functions

Project Position - Comply.

FSAR Subsections - 7.1.2.6.3, 7.2.2.1

Regulatory Guide 1.23, (February 1972)

Onsite Meteorological Program

Project Position - Comply.

FSAR Subsection - 2.3.3

Regulatory Guide 1.24, (March 1972)

Assumptions Used for Evaluating the Potential Radiological Consequences of Pressurized Water Reactor Radioactive Gas Storage Tank Failure

Project Position - Not applicable to BWRs.

Regulatory Guide 1.25, (March 1972)

Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors

Project Position - Superseded by Appendix B to Regulatory Guide 1.183.

FSAR Section - 15.7

Regulatory Guide 1.26, Rev. 3 (February 1976)

Quality Group Classification and Standards for Water, Steam, and Radioactive Waste Containing Components of Nuclear Power Plants

Project Position - Complies with this guide with clarifications as noted below and FSAR Section 17.2.

(1) Reference: Paragraph C.2.c of the Regulatory Guide. The guide applies Quality Group C to those systems that are connected to the reactor pressure boundary by two normally closed or automatically closed valves. There are a number of small lines that are open to the atmosphere beyond the normally closed second valve; e.g., vent lines, drain lines, and sample connections. The gain to public safety by making these lines Quality Group C is not evident. The lines are procured and installed to Quality Group D requirements.

(2) Reference: Paragraph C.2.d of the Regulatory Guide. Radwaste systems are classified as Quality Group D augmented as discussed in Table 3.2-1 and Chapter 11.

(3) GGNS may choose not to use the specific A, B, C, and D level classification system set forth in this guide. GGNS generally followed the requirements of this guide in developing the list of structures, systems and components ("Q" List) to which the program will apply. The "Q" List will describe the items to which the Quality Assurance Program will apply in lieu of the guidance contained in this regulatory guide.

FSAR Section - 3.2, Chapter 11

Regulatory Guide 1.27, Rev. 2, (January 1976)

Ultimate Heat Sink for Nuclear Power Plants

Project Position - The Grand Gulf Project complies with the contents of the Regulatory Guide with two items of clarification as discussed below.

(1) Reference: Paragraph C.1 of the Regulatory Guide. The subject position of the Regulatory Guide states that the ultimate heat sink (UHS) should be capable of providing cooling for at least 30 days following initiation of reactor shutdown and that procedures should be available to ensure continuous cooling after 30 days.

The Grand Gulf UHS basins have a combined cooling water makeup capacity in excess of the makeup required for 30 days following shutdown initiation of the nuclear unit. Procedures exist for ensuring that a cooling water makeup supply is available within 30 days following shutdown initiation. This makeup supply will be satisfied by either 1) restoration of the normal makeup supply to the UHS (plant service water [PSW] system) or 2) installation of a temporary makeup supply system utilizing available onsite water sources (main circulating water basin; various onsite basins and ponds; onsite natural water sources, i.e., Hamilton and Gin Lakes or the Mississippi River).

The total makeup water requirement for the UHS for the 60 days beyond the initial 30 days is 13.3×10^6 gallons. This quantity is based on the conservative assumption of 221,000 gpd used on day 30 for the remaining 60 days.

The circulating water basin can provide up to an additional 30-day supply of makeup water to the UHS assuming 100 percent availability of its total 6.98×10^6 gallon capacity. A 24-day makeup is available if the circulating tower basin is only 80 percent full.

The two onsite construction water wells have been sized to provide 500,000 gpd for construction water demand (subsection 2.4.13.2.3). During operation these wells can be made available as a source of makeup water. Electric or diesel pumps can be used to deliver the 221,000 gpd necessary to the UHS. These wells can supply all or part of the additional 60-day water demand.

Suction from an onsite basin constructed on Stream B could also be used. This stream provides an average flow rate of 1 million gallons/day of low quality water and would only be used as a last resort.

In the event that makeup from the PSW system is unavailable, there is sufficient time to install a temporary portable pump and piping system to permit transfer of water from one of the other onsite water sources. In case of a loss of offsite power, Grand Gulf will be given preferential reconnection and should have complete power avail ability after 30 days. These alternate sources of water will ensure a continued cooling water supply to the UHS.

(2) Reference: Paragraph C.2 of the Regulatory Guide. The guide states that ultimate heat sink features which are constructed specifically for the nuclear power plant and which are not required to be designed to withstand the Safe Shutdown Earthquake or the Probable Maximum Flood should at least be designed and constructed to withstand the effects of the Operating Basis Earthquake and waterflow based on severe historical events in the region. Makeup water to the cooling towers is not immediately required since the tower basins have a combined cooling water makeup capacity in excess of that required for the 30-day plant shutdown period. A sufficient alternate makeup source can be incorporated if necessary, during the latter part of the 30-day period (see item (1) above.) Therefore, the failure of the nonseismic makeup source will not affect the capability of the ultimate heat sink to perform its design function.

FSAR Subsections - 9.2.1, 9.2.5, 2.3.1.2.7

Regulatory Guide 1.28, (June 7, 1972)

Quality Assurance Program Requirements
(Design and Construction)

Bechtel Position - Complies with this guide with clarifications as noted in FSAR Section 17.2.

SAR Section - 17.1 of the Grand Gulf PSAR

Regulatory Guide 1.29, Rev. 3 (September 1978

Seismic Design Classification

Project Position - Complies with this guide with clarifications as noted below and in FSAR Section 17.2.

(1) Reference: Paragraph C.1.b. of the Regulatory Guide. Application of this guide is limited to the reactor vessel internals which are engineered safety features.

(2) Reference: Paragraph C.1.e of the Regulatory Guide. Grand Gulf has incorporated into the design of the main steam system a third slow-shutting isolation valve and has classified the main steam lines downstream of the first seismic restraint from the third isolation valve as nonsafety class, nonseismic. This is consistent with Regulatory Guide 1.26, Revision 3, and Standard Review Plan 3.2.1, Figure 3.2.1.

(3) Reference: Paragraph C.1.h of the Regulatory Guide. The component cooling water portions of the reactor recirculation pumps are nonseismic since this does not involve a safety function.

(4) Reference: Paragraph C.1.p of the Regulatory Guide. Radwaste systems are nonseismic. The offgas system supports have been seismically analyzed as discussed in Table 3.2-1 and Section 11.3.

(5) Reference: Paragraph C.2 of the Regulatory Guide. This paragraph requires that items that would otherwise be classified as nonseismic "but whose failure could reduce the functioning" of items important to safety "to an unacceptable safety level" are to be "designed and constructed so that the SSE would not cause such failure." In addition, paragraph C.4 of the guide requires that "the pertinent quality assurance requirement of Appendix B to 10 CFR Part 50 should be applied to the safety requirements" of such items. Both these requirements are considered to be adequately met by applying the following practices to such items:

a. Design and design control for such items are carried out in the same manner as that for items directly important to safety. This includes the performance of appropriate design reviews.

b. Field work is performed under the direction of experienced field construction superintendents and is inspected by the staff of field engineers stationed at the site. The field engineers are responsible for verifying that construction is performed in accordance with the design drawings and specifications and with applicable standard codes and specifications. Audits of work plans and inspection reports are periodically performed by Quality Assurance personnel.

Such items are not included on the Q list.

6. Reference: Paragraph C.3 of the Regulatory Guide. Seismic Category I design requirements are required to be extended "to the first seismic restraint beyond the defined boundaries." Because seismic analysis of a piping system requires division of the systems into discrete segments terminated by fixed points, this means that the seismic design cannot be terminated at a seismic restraint, but is extended to the first point in the system that can be treated as an anchor to the plant structure. In addition, paragraph C.4 of the guide requires that "the pertinent quality assurance requirement of Appendix B to 10 CFR Part 50 should be applied to the safety requirements" of such items. Both these requirements are considered to be adequately met by applying the following practices to such items:

   a. Design and design control for such items are carried out in the same manner as that for items directly important to safety. This includes the performance of appropriate design reviews.

   b. Field inspections are performed by representatives of the originating design group to assure that the final installation of such items is in accordance with documents that formed the basis for the seismic analysis of the items.

   Such items are not included on the Q list.

FSAR Sections - 3.2, 3.7, 7.1.2.6, 8.3.1, 11.3

Regulatory Guide 1.30, (August 11, 1972)

Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment

Operational Stage Position - Complies with this guide with clarifications as noted in FSAR Section 17.2.

FSAR Section - 17.2

Regulatory Guide 1.31, Rev. 1 (June 1973)

Control of Stainless Steel Welding

Project Position - Control of stainless steel welding complies with the Interim Position on Regulatory Guide 1.31 (Branch Technical Position MTEB 5-1) (Reference 1) except as discussed below:

(1) Reference: Paragraph B.1.b of Interim Position. Austenitic stainless steel welding materials used in the fabrication and installation of ASME Section III, Class 1, 2, and 3 parts and components are controlled to deposit weld metal containing 8 to 25 percent delta ferrite, except for Type 309 and 309L welding materials, which are controlled to deposit weld metal containing 5 to 15 percent delta ferrite and are used only for welding carbon and low-alloy steels to austenitic stainless steels. Use of Type 309L welding materials is further limited to the overlay deposit on carbon steel component nozzles or connecting pipe when postweld heat treatment is required.

These limits for delta ferrite in austenitic stainless steel welding materials comply with the Interim Position because the upper limit of 20 percent delta ferrite does not apply for welds that are not heat treated after welding (paragraph 3b) except for solution annealing heat treatment. Solution heat treatment, although not required after welding, is permitted in order to avoid sensitization.

The procedure for determining the amount of delta ferrite in each heat or lot of austenitic stainless steel welding material does not comply with the Interim Position. Determination of delta ferrite is in accordance with ASMF Section III, Division 1, Paragraph NB-2433, 1971, with Summer 1973 Addenda except that an undiluted weld deposit is required for each heat or bare wire used with the gas-metal-arc (GMA) process.

(2) Reference: Paragraph B.2 of Interim Position. This paragraph is complied with for all tests and examinations required by ASME Section III, Division 1, 1971, edition with Summer 1973 Addenda.

(3) Reference: Paragraph B.3.a of Interim Position. This paragraph is not complied with. Magnetic measurements of production welds for delta ferrite content are unnecessary when austenitic stainless steel welding materials are controlled to deposit 8 to 25 percent delta ferrite, based on chemistry, except for Type 309 and 309L welding materials, which are controlled to deposit 5 to 15 percent delta ferrite based on chemistry.

(4) Paragraph B.3.b of Interim Position. This paragraph is complied with for welding material certification.

(5) Reference: Paragraphs B.4.a, B.4.b, and B.4.c of Interim Position. These paragraphs are not complied with because measurement of production welds for delta ferrite is not performed.

(6) Refer to subsections 4.5.2.4 and 5.2.3.4 for further discussion on the application of the guide to the reactor coolant pressure boundary.

Reference 1: Branch Technical Position MTEB 5-1

FSAR Section/Subsections - 6.1, 5.2.3.4, 4.5.2.4

Regulatory Guide 1.32, Rev. 2 (February 1977)

Criteria for Safety-Related Electric Power Systems for Nuclear Power Plants

Project Position - Comply.

FSAR Sections - 7.1.2.6, 8.1, 8.3

Regulatory Guide 1.33, Rev. 2 (February 1978)

Quality Assurance Program Requirements (Operation)

Project Position - Complies with this guide with clarifications as noted in FSAR Section 17.2.

FSAR Section - 17.2

Regulatory Guide 1.34, (December 28, 1972)

Control of Electroslag Weld Properties

Project Position - Electroslag welding is not used for the fabrication of ASME Section III components; therefore, Regulatory Guide 1.34 is not applicable to Grand Gulf.

FSAR Subsections - 5.2.3, 5.3.1.4

Regulatory Guide 1.35, Rev. 2 (January 1976)

Inservice Inspection of Ungrouted Tendons
in Prestressed Concrete Containment Structures

Project Position - Not applicable. Grand Gulf has a reinforced concrete containment.

Regulatory Guide 1.36, (February 1973)

Nonmetallic Thermal Insulation for
Austenitic Stainless Steel

Project Position - The use of nonmetallic insulation materials on safety-related stainless steel piping/tubing and components will be limited to 2-inch and smaller sizes. This size limitation does not apply to the reactor recirculation system, where nonmetallic insulation materials are used for removable inservice inspection covers or substituted for metal reflective insulation due to clearance/thermal efficiency requirements. For these cases, only those nonmetallic insulation materials will be used on Grand Gulf which have been qualified to meet the material property requirements of Regulatory Guide 1.36.

FSAR Section/Subsections - 6.1, 4.5.2.4, 5.2.3, 5.3.3

Regulatory Guide 1.37, (March 16, 1973)

Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants

Project Position - Complies with this guide with clarifications as noted in FSAR Section 17.2.

FSAR Sections/Subsections - 17.2, 6.1, 5.2.3.4, 4.5.1, 4.5.2

Regulatory Guide 1.38, Rev. 2 May 1977

Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage, and Handling of Items for Water-Cooled Nuclear Power Plants

Project Position - Complies with this guide with clarifications as noted below and FSAR Section 17.2.

(1) Reference: Section A3.4.1 of the Appendix to ANSI N45.2.2-1972, Contact Preservatives. During printing of the standard a transposition occurred between the last sentence of A3.4.1(4) and A3.4.1(5). The correct requirements are:

4. "However, preservatives for inaccessible inside surfaces of pumps, valves and pipe systems containing reactor coolant water shall be the water flushable type."

5. "The name of the preservative used shall be indicated to facilitate touch up."

(2) Reference: Section A3.4.2 of the Appendix to ANSI N45.2.2-1972, Inert Gas Blankets. Inert gas blankets are currently used on the reactor pressure vessel (RPV) and on some of the heat exchangers supplied. Provisions are made for measuring and maintaining the RPV blanket pressure within the required range during shipment and storage. Heat exchangers or tanks containing carbon steel which require an inert gas blanket will be inerted prior to shipment. During storage, provisions shall be made for measuring and maintaining the inert gas blanket pressure within the required range within each pressurized purged item or container.

(3) Reference: Section A3.5.2 of the Appendix to ANSI N45.2.2-1972, Tapes and Adhesives. Tapes will meet a sulphur limit of 0.30 percent by weight instead of 0.10 percent as specified in A3.5.2(1)(a). This limit is reasonable based upon the chemical content of commercially available tapes. When used on item surfaces, tapes will be of a contrasting color rather than "Brightly Colored" as required by A3.5.2(3).

(4) Reference: Section A3.6.2 of the Appendix to ANSI N45.2.2-1972, Vapor Barrier Material. The last sentence is considered unnecessary guidance and not an appropriate requirement since vapor barrier materials are usually brown, cream, or black in color.

(5) Reference: Section A3.7.1 of the Appendix ANSI N45.2.2-1972, Fiberboard Boxes. In lieu of A3.7.1(3) and (4), the following will be imposed: Fiberboard boxes shall be securely closed either with a water resistant adhesive applied to the entire area of contact between the flaps, or all seams and joints shall be sealed with not less than 2-inch wide, water resistant tape.

FSAR Section - 17.2

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Regulatory Guide 1.39, Rev. 2 (September 1977

Housekeeping Requirements for Water-Cooled Nuclear Power Plants

Project Position - Complies with this guide with clarifications as noted in FSAR Section 17.2.

Regulatory Guide 1.40, (March 16, 1973

Qualification Tests of Continuous-Duty Motors Installed Inside the Containment of Water-Cooled Nuclear Power Plants

Project Position - Comply.

FSAR Section/Subsection - 3.11, 3.9.3

Regulatory Guide 1.41, (March 1973)

Preoperational Testing of Redundant Onsite Electrical Power Systems to Verify Proper Load Group Assignments

Project Position - Comply.

FSAR Chapter - Chapter 14

Regulatory Guide 1.42, Rev. 1 (March 1974)

Interim Licensing Policy on As-Low-As Practicable for Gaseous Radioiodine Releases from Light-Water-Cooled Nuclear Power Reactors

Project Position - NRC withdrew Regulatory Guide 1.42 on March 22, 1976 (Ref 1).

Reference 1: Federal Register Notice, 41 FR 11891, 3/22/76

Regulatory Guide 1.43, (May 1973)

Control of Stainless Steel Cladding of
Low-Alloy Steel Components

Project Position - Regulatory Guide 1.43 prescribes qualification and production cladding controls for ASME SA 508-2 material made to COARSE GRAIN practice. This material is not used for any of the safety class components within the NSSS. ASME SA 508-2 composition material employed on the reactor pressure vessel for this plant is produced to FINE GRAIN practice. Therefore, this Regulatory Guide is not applicable to the NSSS components in this plant.

FSAR Subsection - 5.3.1.4

Regulatory Guide 1.44, (May 1973)

Control of the Use of Sensitized Stainless Steel

Bechtel Position - The requirements of the regulatory guide are met except as follows:

Reference: Paragraph C.6 of the Regulatory Guide. All welding performed on austenitic stainless steel is with low heat input welding processes. Intergranular corrosion testing is not performed on a routine basis since all the materials used are in the solution annealed condition and the following additional safeguards are taken:

a. The preheat and interpass temperature used during the welding of austenitic stainless steel is kept to 350 F maximum.

b. Postweld heat treatment in the range 800 to 1500 F is strictly forbidden. Solution annealing heat treatment after welding, although not required, is permitted.

Because severe sensitization is avoided by these safeguards, testing to determine susceptibility to intergranular attack is not performed.

General Electric-Hitachi (GEH) Position - Regulatory Guide 1.44's purpose is to address 10 CFR 50, Appendix A, GDCs 1 and 4, and Appendix B requirements to control "the application and processing of stainless steel to avoid severe sensitization that could lead to stress corrosion cracking." The guide purposes that this should be done by limiting sensitization due to welding as measured by ASTM A262 Practice A or E, or another method that can be demonstrated to show nonsensitization in austenitic stainless steels.

Tests by GEH indicate that the test specified by A262 A or E (Detecting Susceptibility to Intergranular Attack in Stainless Steel) detects sensitization in only a cross way, and that the tests do not provide a precise method of predicting susceptibility to stress corrosion cracking in the BWR

environment. Further, it should be recognized that sensitization alone is not enough to cause stress corrosion cracking in the BWR environment. High stress is also required.

As an alternate to performing the above tests, GEH has employed process controls to minimize sensitization and thus stress corrosion. In addition special remedies have been employed on piping lines shown by field experience in earlier General Electric Plants to be "service sensitive." Cracking in these lines, all 10-inch diameter or less, has been shown to be due to high stress. In the lines where cracking was observed, a safety hazard was never present due to austenitic stainless steels inherent toughness, and to early detection. The remedies employed on the Grand Gulf Project for these lines are listed below.

- Recirculation valve bypass - This line has been removed and capped.
- Core spray piping - Piping external to the vessel is carbon steel.
- CRD return - This line has been removed and capped.

To further reduce the probability of cracking in small lines, 3-inch and smaller diameter lines are being fabricated from type 3161, even though they are classified as "service sensitive." This extra precaution has been taken because piping cracks have been confined to smaller piping.

When stress corrosion is not predicated, due to low stress, General Electric has chosen to minimize susceptibility. These controls include (but are not limited to) reduced weld heat input, control of cold work, and control of solution heat treatment.

As summarized above, GEH has complied with the intent of R.G. 1.44 by controlling processing of stainless steel to avoid severe sensitization that could lead to stress corrosion cracking. In addition, areas where a concern for stress corrosion exists due to cracks in earlier designs have been redesigned to eliminate the possibility of stress corrosion cracking.

FSAR Section/Subsections - 6.1, 5.2.3.4, 5.3.1.4, 4.5.1.3, 4.5.2.4, 4.2.3.4.1

Regulatory Guide 1.45, (May 1973)

Reactor Coolant Pressure Boundary
Leakage Detection Systems

Project Position - Comply with the exception of regulatory positions 1a and 6 (See Subsection 7.6.2.4.2.1.)RG 1.45 does not require an alarm from a 1 gpm leakage increase.

FSAR Subsections - 5.2.5, 7.6.1.4, 7.6.2.4.2.1

Regulatory Guide 1.46, (May 1973

Protection Against Pipe Whip Inside Containment

Bechtel Position - Grand Gulf complies with Regulatory Guide 1.46 with the following exception. This exception has been approved by the NRC at the referenced meeting (Ref. 1).

Reference: Paragraph C.1.b of the Regulatory Guide. Stress limits defined in paragraph B.1.b of Reference (2) are used to determine intermediate pipe break locations in ASME Code Class 1 piping instead of those listed in the Regulatory Guide.

References

(1) NRC(MEB) - Bechtel meeting on January 7, 1976 (see Bechtel letter dated 2/18/76 from R.L. Ashley to R.R. Macary of the NRC).

(2) Branch Technical Position MEB 3-1, Postulated Break and Leakage Locations in Fluid System Piping Outside Containment.

(3) ANS N176

GEH Position - This Regulatory Guide is not considered a design requirement for this project. However, pipe whip requirements for fluid system piping within the primary containment that, under normal operation has service temperatures 200°F or pressures ≥ 275 psig, comply with ANS N176 (Ref. 3), Design Basis for Protection Against Pipe Whip, and Regulatory Guide 1.46, except as delineated in the following criteria for no breaks in Class 1 piping:

(1) If Equation 10 of NB-3653-1, ASME Code III results in $S < 2.4 S_m$ for ferritic or austenitic steels, no other requirement need be met. Stress range should be calculated between any two load sets (including zero load set) according to NB-3600 for upset and an OBE event transient.

(2) If Equation 10 results in $2.4 \leq S \leq 3.0 S_m$ for ferritic or austenitic steels, the cumulative usage factor, U, calculated on the basis of Equation 14 of NB-3653.6, must be < 0.1.

(3) If Equation 10 results in S > 3.0 S_m for ferritic or austenitic steels, then the stress value in Equations 12 and 13 of NB-3653.6 must not be ≥ 2.4 S_m.

FSAR Section - 3.6

Regulatory Guide 1.47 (May 1973)

Bypassed and Inoperable Status Indication
for Nuclear Power Plant Safety Systems

Project Position - Comply.

FSAR Subsection - 7.1.2.6, 7.5.1.3, 8.3.1

Regulatory Guide 1.48 (May 1973)

Design Limits and Loading Combinations for Seismic Category I Fluid System Components

GEH Position - NSSS equipment is in compliance with the regulatory guide except as noted in attached Table 1.

Bechtel Position - The extent of compliance with Regulatory Guide 1.48 is as follows:

(1) Reference: Paragraph C.1 of the Regulatory Guide. There are no Class 1 vessels in the non-NSSS equipment. Class 1 piping does comply with this position.

(2) Reference: Paragraph C.2 of the Regulatory Guide. There are no nonactive Class 1 pumps. Nonactive Class 1 valves have not been designed by analysis.

(3) Reference: Paragraph C.3 of the Regulatory Guide. Grand Gulf satisfies the requirements of the ASME Code, Section III, 1971 Edition, Winter 1972 Addendum for nonactive Class 1 valves 2-1/2 inches and larger. Grand Gulf satisfies the requirements of the ASME Code, Section III, 1974 Edition, Winter 1974 Addendum for nonactive Class 1 valves 2 inches and smaller.

In addition to the code requirements, all Class 1 nonactive valves having extended proportions are designed to withstand 3.0g seismic load in any direction and have a frequency of vibration greater than 33 Hz. This is verified by either test and/or calculation.

(4) Reference: Paragraph C.4 of the Regulatory Guide. Not applicable to Grand Gulf.

(5) Reference: Paragraph C.5 of the Regulatory Guide. Grand Gulf satisfies the requirements of the ASME Code, Section III, 1971 Edition, Winter 1972 Addendum for active Class 1 valves 2-1/2 inches and larger. Grand Gulf satisfies the requirements of the ASME Code, Section III, 1974 Edition, Winter 1974 Addendum for active Class 1 valves 2 inches and smaller. Accordingly, the Code stress limits have been satisfied.

In addition to the code requirements, additional testing/ analysis as specified in subsection 3.9.3.2 of the FSAR was performed to assure functional operability. Following installation, active Class 1 valves will also be subjected to functional qualification testing and periodic inservice operations. These tests enhance the reliability of the valves for the design life of the plant.

(6) Reference: Paragraph C.6 of the Regulatory Guide. There are no Class 2 vessels in the non-NSSS equipment. Grand Gulf complies with position 6.a with the exception that the Class 3 vessels are designed to ASME Section III. The Class 3 vessels were analyzed with 50 percent SSE in the normal condition and SSE in the faulted condition in combination with the nozzle loads, deadweight, and operating pressures.

(7) Reference: Paragraph C.7 of the Regulatory Guide is not applicable to Grand Gulf.

(8) Reference: Paragraph C.8 of the Regulatory Guide. Grand Gulf takes exception to this paragraph in that ASME Code Case 1606-1 is used for evaluating non-ECCS Class 2 and 3 piping for the faulted plant condition (refer to Table 3.9-22). OBE and SSE are combined with other loading conditions as defined in FSAR Table 3.9-17.

(9) Reference: Paragraph C.9 of the Regulatory Guide. Not applicable to Grand Gulf.

(10) Reference: Paragraph C.10 of the Regulatory Guide. Active Class 2 and 3 pumps satisfy the criteria specified by ASME Code Case 1636-1.

(11) Reference: Paragraph C.11 of the Regulatory Guide. Same as Position (3)

(12) Reference: Paragraph C.12 of the Regulatory Guide. Grand Gulf satisfies the requirements of the ASME Code, Section III 1971 Edition, Winter 1972 Addenda for gate, globe, and check valves, and Winter 1971 Edition, Summer 1973 Addenda for Butterfly Valves for active Class 2 and 3 valves 2-1/2 inches and larger. Grand Gulf satisfies the requirements of the ASME Code, Section III 1974 Edition, Winter 1974 Addenda for active Class 2 and 3 valves 2 inches and smaller.

In addition, the requirements of Code Case 1635-1 have been satisfied. The additional testing/analysis as specified in subsection 3.9.3.2 of the FSAR was performed to assure functional operability. Following installation, active Class 2 and 3 valves will be subjected to functional qualification testing and periodic inservice operations. These tests enhance the reliability of the valves for the design life of the valves.

FSAR Subsections - 3.9.1, 3.9.2, and 3.9.3

Notes for Regulatory Guide 1.48, Table 1

Numerical indicators in the regulatory guide portion of the table correspond to the footnotes of Regulatory Guide 1.48.

Alphabetical indicators in this table correspond to the following:

a. In addition to compliance with the design limits specified, assurance of operability under all design loading combinations is in accordance with subsection 3.9.4.

b. Referenced paragraphs of code are currently in course of preparation.

c. The design limit for local membrane stress intensity or primary membrane plus primary bending stress intensity is 150 percent of that allowed for general membrane (except as limited to 2.4S for inactive components under faulted condition). Refer to Tables 3.9-2 and 3.9-17 through 3.9-24.

d. When selecting plant events for evaluation, the choice of the events to be included in each plant condition is selected based on the probability of occurrence of the particular load combination. The combination of loads are those identified in Table 3.9-2.

TABLE 1: NSSS COMPLIANCE WITH REGULATORY GUIDE 1.48

	NRC REGULATORY GUIDE 1.48				GRAND GULF			
Component	Plant Condition	Loading Combination[1/]	Design Limit	Regulatory Guide Paragraph	Loading Combination[(d)]	Code Allowable Stresses	ASME Section III Reference	Comparison with NRC Regulatory Guide 1.48
Class 1 Vessels	Upset (U)	[NPC or UPC] + 0.5 SSE[(1)]	NB-3223	1.a	[NPC or UPC] + 0.5 SSE[(1)]	3.0 Sm (includes secondary stresses)	NB-3223	Agree
	Emergency (E)	EPC	NB-3224 2/	1.b	EPC	1.8Sm or 1.5Sy	NB-3224	
	Faulted (F)	NPC + SSE + DSL	NB-3225	1.c	NPC + SSE + DSL	App. F - Sect. III	NB-3225	
Class 1 Piping	U	[NPC or UPC] + 0.5 SSE	NB-3654	1.a	[NPC or UPC] + 0.5 SSE	3.0 Sm (includes secondary stresses)	NB-3223	Agree
	E	EPC	NB-3655 2/	1.b	EPC	2.23 Sm	NB-3224	
	F	NPC + SSE + DSL	NB-3656	1.c	NPC + SSE + DSL	3.0Sm	NB-3225	
Class 1 Pumps (Inactive)	U	[NPC or UPC] + 0.5 SSE	NB-3223 5/	2.a	[NPC or UPC] + 0.5 SEE	1.65 Sm (excludes secondary stresses)	NB-3223	Agree
	E	EPC	NB-3224 1/	2.b	EPC	1.8Sm	NB-3224	
	F	NPC + SSE + DSL	NB-3225	2.c	NPC + SSE + DSL	App. F - Sect. III	NB-3225	
Class 1 Pumps (Active)	U	[NPC or UPC] + 0.5 SSE	NB-3222 5/	4.a	[NPC or UPC] + 0.5 SEE			
	E	EPC	NB-3222 6/	4.b	EPC	Not Applicable	Not Applicable	Not Applicable
	F	NPC + SSE + DSL	NB-3222 7/ 8/	4.c	NPC + SSE + DSL			
Class 1 Pumps (Inactive) by analysis	U	[NPC or UPC] + 0.5 SSE	NB-3223 5/	2.a	[NPC or UPC] + 0.5 SEE			
	E	EPC	NB-3224 4/	2.b	EPC	Not Applicable	Not Applicable	Not Applicable
	F	NPC + SSE + DSL	NB-3225 2/	2.c	NPC + SSE + DSL			
Class 1 Valves (Inactive) Designed by either Std. or alternative design rules	U	[NPC or UPC] + 0.5 SSE	1.1 Pr	3.a	[NPC or UPC] + 0.5 SEE	1.1 Pr	NB-3225	
	E	EPC	1.2 Pr	3.b	EPC	1.2 Pr	NB-3226	Agree
	F	NPC + SSE + DSL	1.5 Pr	3.c	NPC + SSE + DSL	1.5 Pr	NB-3227	
Class 1 Valves (Active) by analysis	U	[NPC or UPC] + 0.5 SSE	NB-3222 5/	4.a	[NPC or UPC] + 0.5 SEE			
	E	EPC	NB-3222 6/	4.b	EPC	Not Applicable	Not Applicable	Not Applicable
	F	NPC + SSE + DSL	NB-3222 7/ 8/	4.c	NPC + SSE + DSL			

TABLE 1: NSSS COMPLIANCE WITH REGULATORY GUIDE 1.48 (CONTINUED)

		NRC REGULATORY GUIDE 1.48			GRAND GULF			
Component	Plant Condition	Loading Combination[1/]	Design Limit	Regulatory Guide Paragraph	Loading Combination[(d)]	Code Allowable Stresses	ASME Section III Reference	Comparison with NRC Regulatory Guide 1.48
Class 1 Valves (Active) Designed by Std. or alternative design rules	U	[NPC or UPC] + 0.5 SSE	1.0 Pr	5.a	[NPC or UPC] + 0.5 SSE	1.0 Pr	NB-3225	Agree
	E	EPC	1.0 Pr [6/]	5.b	EPC	1.0 Pr (a)	NB-3226	
	F	NPC + SSE + DSL	1.0 Pr	5.c	NPC + SSE + DSL	1.0 Pr	NB-3227	
Class 2 & 3 Vessels (Division 1) of Section of the ASME Code	U	[NPC or UPC] + 0.5 SSE	1.1S	6.a	[NPC or UPC] + 0.5 SSE	$\sigma_m = 1.1S$ (c)	code case 1607-1 NC/ND-3300	Agree except for Faulted Condition NRC move conservative
	E	EPC	1.1S [9/]	6.b	EPC	$\sigma_m = 2.0S$ (c)		
	F	NPC + SSE + DSL	1.5S	6.c	NPC + SSE + DSL			
Class 2 Vessels (Division 2) of Section VIII of the ASME Code	U	[NPC or UPC] + 0.5 SSE	NB-3223	7.a	[NPC or UPC] + 0.5 SSE			
	E	EPC	NB-3224 [2/]	7.b	EPC	Not Applicable	Not Applicable	Not Applicable
	F	NPC + SSE + DSL	NB-3225	7.c	NPC + SSE + DSL			
Class 2 & 3 Piping	U	[NPC or UPC] + 0.5 SSE	NC-3611.1(b)(4)(c)(b)(1)	8.a	[NPC or UPC] + 0.5 SSE	1.2 Sh	NC/ND-3611.3(b)	NRC more conservative
	E	EPC	NC-3611.1(b)(4)(c)(b)(1) [10/]	8.b	EPC	1.8 Sh	(NC/ND-3611.3(c)	GE reflects industry position
	F	NPC + SSE + DSL	NC-3611.1(b)(4)(c)(b)(2)	8.c	NPC + SSE + DSL	2.4 Sh		
Class 2 & 3 Pumps (Inactive)	U	[NPC or UPC] + 0.5 SSE	$\sigma m < 1.1S \geq \frac{\sigma m + \sigma b}{1.5}$	9.a	[NPC or UPC] + 0.5 SSE			
	E	EPC	$\sigma m \leq 1.1S \geq \frac{\sigma m + \sigma b}{1.5}$	9.b	EPC	Not Applicable	Not Applicable	Not Applicable
	F	NPC + SSE + DSL	$\sigma m \leq 1.2S \geq \frac{\sigma m + \sigma b}{1.5}$	9.c	NPC + SSE + DSL			
Class 2 & 3 Pumps (Active)	U	[NPC or UPC] + 0.5 SSE	$\sigma m \leq 1.0S \geq \frac{\sigma m + \sigma b}{1.5}$ [11/]	10.a	[NPC or UPC] + 0.5 SSE	$\sigma_m = 1.1S$ (a) (c)	code case 1636, NC/ND-3423 (see Note (b))	Agree (a)
	E	EPC	$\sigma m \leq 1.0S \geq \frac{\sigma m + \sigma b}{1.5}$	10.a	EPC			
	F	NPC + SSE + DSL	$\sigma m \leq 1.0S \geq \frac{\sigma m + \sigma b}{1.5}$	10.a	NPC + SSE + DSL	$\sigma_m = 1.2S$		

TABLE 1: NSSS COMPLIANCE WITH REGULATORY GUIDE 1.48 (CONTINUED)

NRC REGULATORY GUIDE 1.48					GRAND GULF			
Component	Plant Condition	Loading Combination[1/]	Design Limit	Regulatory Guide Paragraph	Loading Combination[(d)]	Code Allowable Stresses	ASME Section III Reference	Comparison with NRC Regulatory Guide 1.48
Class 2 & 3 Valves (Inactive)	U	[NPC or UPC) + 0.5 SSE	1.1 Pr	11.a	[NPC or UPC] + 0.5 SSE	$\sigma_m = 1.1S$ (c)	code case 1635, NC/ND-3423 (see Note (b))	Equally conservative
	E	EPC	1.1 Pr	11.a	EPC			
	F	NPC + SSE + DSL	1.2 Pr	11.a	NPC + SSE + DSL	$\sigma_m = 2.0S$ (c)		
Class 2 & 3 Valves (Active)	U	[NPC or UPC) + 0.5 SSE	1.0 Pr	12.a	[NPC or UPC] + 0.5 SSE	$\sigma_m = 1.1S$ (a) (c)	code case 1635, NC/ND-3421 (see Note (b))	Equally conservative (c)
	E	EPC	1.0 Pr 11/	12.a	EPC			
	F\	NPC + SSE + DSL	1.0 Pr	12.a	NPC + SSE + DSL	$\sigma_m = 1.2S$ (a) (c)		

Regulatory Guide 1.49, Rev. 1 (December 1973)

Power Levels of Nuclear Power Plants

Project Position - Grand Gulf complies with Regulatory Guide 1.49 to the extent that the plant was originally licensed to operate at 3833 MWt and uprated by 15% to 4408 MWt.

Regulatory Guide 1.50 (May 1973)

Control of Preheat Temperature for Welding of Low-Alloy Steel

Project Position - Comply with the regulatory guide except as follows:

(1) Reference: Paragraph C.1.b of the Regulatory Guide. Welding procedures are qualified with the test assembly preheat temperature maintained within the range of minimum preheat temperature to minimum preheat temperature plus 50 F. The preheat temperature requirement above applies only to new welding procedure qualifications.

(2) Reference: Paragraph C.2. of the Regulatory Guide. For each applicable weld joint, post-weld treatment is either:

a. extended hold at preheat temperature, or

b. maintenance of preheat to post-weld heat treatment.

(3) Reference: Paragraph C.4 of the Regulatory Guide. Soundness will be verified for production welds not consistent with Paragraph C.1.b. Verification was performed by nondestructive examination; the procedure's sensitivity exceeds ASME Code requirements.

FSAR Subsection - 5.2.3.3

Regulatory Guide 1.51, (May 1973)

Inservice Inspection of ASME Code Class 2 and 3 Nuclear Power Plant Components

Project Position - This Regulatory Guide was withdrawn by the NRC (7/21/75).

Reference - FR 40-30510, July 21, 1975

Regulatory Guide 1.52, Rev. 2 (March 1978)

Design, Testing, and Maintenance Criteria for Post Accident Engineered-Safety Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants

Project Position - Refer to following pages for project positions.

FSAR Subsections - 6.2.3, 6.5.1, 6.5.3, and 9.4.1.

Reg. Guide 1.52 (Cont.)

Regulatory Position	Discussion
C.1. Environmental Design Criteria	C.1.a. The Grand Gulf ESF filter systems are designed for the maximum environments, resulting from the postulated DRA, to which the systems will be exposed. Environmental conditions are specified in the applicable equipment specification (refer to Table 3.11-7 which tabulates the method of environmental qualification of equipment).
a. The design of an engineered-safety-feature ,atmosphere cleanup system should be based on the maximum pressure differential, radiation dose rate, relative humidity, maximum and minimum temperature, and other conditions resulting from the postulated DRA and on the duration of such conditions.	
b. The design of each system should be based on the radiation dose to essential services in the vicinity of the absorber section integrated over the 30-day period following the postulated DRA. The radiation source term should be consistent with the assumptions found in Regulatory Guides 1.3 (Ref. 4), 1.4 (Ref. 5). and 1.25 (Ref. 6). Other engineered safety features, including pertinent components of essential services such as power, air, and control cables, should be adequately shielded from the atmosphere cleanup systems.	b. Refer to Table 3.11-7 for applicable radiation doses. Source assumptions are consistent with Regulatory Guides 1.3, 1.4, and 1.25. Other ESF equipment and services are adequately shielded from the ESF filter systems.
c. The design of each absorber should be based on the concentration and relative abundance of the iodine species (elemental, particulate, and organic), which should be consistent with the assumptions found in Regulatory Guides 1.3 (Ref. 4), 1.4 (Ref. 5), and 1.25 (Ref. 6).	c. Refer to Table 3.11-7 for applicable radiation doses. Assumptions are consistent with Regulatory Guides 1.3, 1.4, and 1.25.
d. The operation of any atmosphere cleanup system should not deleteriously affect the operation of other engineered safety features such as a containment spray system, nor should the operation of other engineered safety features such as a containment spray system deleteriously affect the operation of any atmosphere cleanup system.	d. The operation of the ESP filter systems is compatible with the operation of other ESF systems.
e. Components of systems connected to compartments that are unheated during a postulated accident should be designed for post accident effects of both the lowest and highest predicted temperatures.	e. Components of ESF filter systems have been designed for the post-accident affects of both the highest and lowest predicted temperatures that the units would be exposed to.

Reg. Guide 1.52 (Cont.)

Regulatory Position	Discussion
2. System Design Criteria	C.2.a. The ESF filter systems are redundant. Filter systems consist of the following sequential components: (1) demisters, (2) electric heaters, (3) prefilter, (4) HEPA filters, (5) impregnated activated carbon iodine absorbers, (6) HEPA filters after the absorbers, and (7) fans.
a. Atmosphere cleanup systems designed and installed for the purpose of mitigating accident doses should be redundant. The systems should consist of the following sequential components: (1) demisters, (2) prefilters (demisters may serve this function), (3) HEPA filters before the absorbers, (4) iodine absorbers (impregnated activated carbon or equivalent absorbent such as metal zeolites), (5) HEPA filters after the absorbers, (6) ducts and valves, (7) fans, and (8) related instrumentation. Heaters or cooling coils used in conjunction with heaters should be used when the humidity is to be controlled before filtration.	
b. The redundant atmosphere cleanup systems should be physically separated so that damage to one system does not also cause damage to the second system. The generation of missiles from high-pressure equipment rupture, rotating machinery failure, or natural phenomena should be considered in the design for separation and protection.	b. Redundant filter trains are physically separated from each other, and are protected from internally-generated missiles.
c. All components of an engineered-safety-feature atmosphere cleanup system should be designated as seismic Category I (see Regulatory Guide 1.29 (Ref. 7)) if failure of a component would lead to the release of significant quantities of fission products to the working or outdoor environments.	c. All components of ESF filter systems are seismic Category I (refer to Table 3 .2-1).
d. If the atmosphere cleanup system is subject to pressure surges resulting from the postulated accident, the system should be protected from such surges. Each component should be protected with such devices as pressure relief valves so that the overall system will perform its intended function during and after the passage of the pressure surge.	d. Pressure surges are not applicable to the Grand Gulf ESF filter systems.

Reg. Guide 1.52 (Cont.)

Regulatory Position	Discussion
e. In the mechanical design of the system, the high radiation levels that may be associated with buildup of radioactive materials on the system components should be given particular consideration. System construction materials should effectively perform their intended function under the postulated radiation levels. The effects of radiation should be considered not only for the demisters, heaters, HEPA filters, absorbers, and fans, but also for any electrical insulation, controls, joining compounds, dampers, gaskets, and other organic containing materials that are necessary for operation during a postulated DBA.	The effects of radiation have been taken into account in the design of the ESF filter systems. All materials are suitable for the radiation levels encountered in this type of service.
f. The volumetric air flow rate of a single cleanup train should be limited to approximately 30,000 cfm. If a total system air flow in excess of this rate is required, multiple trains should be used. For ease of maintenance, a filter layout three HEPA filters high and ten wide is preferred.	f. No ESF filter train capacity exceeds 4000 cfm ±10%. HEPA filter layout is 2 high by 2 wide.
g. The atmosphere cleanup system should be instrumented to signal, alarm, and record pertinent pressure drops and flow rates at the control room.	g. Differential pressure is indicated locally for the demister, electric heater, and prefilter banks for each ESF filter train. Differential pressure is recorded, and alarmed on high level, in the control room for the upstream HEPA bank, the charcoal absorber section, and the downstream HEPA bank for each ESF filter train.
h. The power supply and electrical distribution system for the atmosphere cleanup system described in Section C.2.a above should be designed in accordance with Regulatory Guide 1.32 (Ref. 8) All instrumentation and equipment controls should be designed to IEEE Standard 279 (Ref. 9). The system should be qualified and tested under Regulatory Guide 1.89 (Reg. 10). To the extent applicable, Regulatory Guide 1.30 (Ref. 12), 1.100 (Ref. 13), and 1.118 (Ref. 14) and IEEE Standard 334 (Ref. 15) should be considered in the design.	h. Electrical power systems are per Regulatory Guide 1.32, and controls are per IEEE-279. Electrical components important to safety are qualified as Class IE. The ESF filter trains and electrical items meet QA requirements of 10 CFR 50, Appendix B. Motors are in accordance with IEEE-334 (1974). Seismic qualification of electrical items conforms to the requirements of IEEE 344-1971 as modified by EICSB Branch Technical Position 10. SQRT review was subsequently performed against IEEE 344-1975. Qualification during the plant operating license stage is in accordance with IEEE 344-1975. Periodic testing is per the Grand Gulf Technical Specifications.
e.	

Regulatory Position	Discussion
i. Unless the applicable engineered-safety-feature atmosphere cleanup system operates continuously during all times that a DBA can be postulated to occur, the system should be automatically activated upon the occurrence of a DBA by (1) a redundant engineered-safety-feature signal (i.e., temperature, pressure) or (2) a signal from redundant Seismic Category I radiation monitors.	The Standby Gas Treatment System automatically starts and operates in response to any of the following signals: (1) high pressure in the drywell, (2) low water level in the reactor vessel, and (3) high radiation in the fuel handling area ventilation system exhaust or fuel pool sweep system exhaust. The Control Room Emergency Filtration System is automatically initiated upon any of the following signals: (1) high radiation in the outside air intake duct, (2) loss-of-coolant accident as indicated by 1) high drywell pressure, or 2) low reactor water level.
j. To maintain radiation exposures to operating personnel as low as is reasonably achievable during plant maintenance, ESF atmosphere cleanup systems should be designed to control leakage and facilitate maintenance in accordance with the guidelines of Regulatory Guide 8.8 (Ref. 16). The ESF atmosphere cleanup train should be totally enclosed. Each train should be designed and installed in a manner that permits replacement of the train as an intact unit or as a minimum number of segmented sections without removal of individual components.	j. The guidelines of Regulatory Guide 8.8 have been considered in the physical design of the ESF filters. ESF filter trains are to tally enclosed. Overall design considerations include reductions in radiation exposures during routine maintenance and testing insofar as effectively possible. It is envisioned, however, that workers will not handle filter units immediately after a design basis accident, and will thereby avoid exposures associated with immediate post accident filter handling. As such, Grand Gulf filter trains cannot be removed intact or in segments without physical destruction of major building walls or the filters themselves. Filter units are designed for ease of maintenance in accordance with the guideline or Regulatory Guide 8.8. All internal components are accessible through maintenance doors located in the filter housings. Controls and instrumentation are located outside the housings. The tray-type adsorbers are easily replaced by removing the trays as individual units. The vertical bed adsorbers incorporate a feature which allows the adsorbent to be removed and replaced without requiring access to the housing interior.
i.	

Reg. Guide 1.52 (Cont.)

Regulatory Position	Discussion
k. Outdoor air intake openings should be equipped with louvers, frills, screens, or similar protective devices to minimize the effects of high winds, rain, snow, ice, trash, and other contaminants on the operation of the system. If the atmosphere surrounding the plant could contain significant environmental contaminants, such as dusts and residues from smoke cleanup systems from adjacent coal burning power plants or industry, the design of the system should consider these contaminants and prevent them from affecting the operation of any atmosphere cleanup system.	Under accident conditions, ESF filter trains do not normally take in outside air; however, after a prolonged period of isolation, outdoor air may be brought into the main control room through the control room standby fresh air filter systems. These outside air intake openings are provided with suitable louvers designed to minimize the entry of rain, snow, ice, or trash. The atmosphere surrounding the Grand Gulf plant does not contain significant amounts of other contaminants referred to.
(1) Atmosphere cleanup system housing and ductwork should be designed to exhibit on test a maximum total leakage rate as defined in Section 4.12 of draft standard ANSI N509 (Ref. 1). Duct and housing leak tests should be performed in accordance with the recommendations of Section 6 of ANSI N510-1975 (Ref. 2).	(1) Standby gas treatment system filter housings have been shop tested at an internal negative pressure of 1 psig, to exhibit a leakage rate of no more than 0.5 cfm per 1000 cfm of system flow. Control room standby fresh air systems have been shop tested at an internal negative pressure of 1 psig, to exhibit a leakage rate of no more than 0.5 cfm per 1000 cfm of system flow. Tests were performed in accordance with approved procedures utilizing either a flow measurement or pressure-decay-vs.-time method. System ductwork is of all-welded construction, with bolted-flange transverse joints, and is pressure tested to exhibit a leakage no greater than 0.5% of system flow at system design pressure.
3. Component Design Criteria and Qualification Testing	C.3.a. Demisters meet UL Class I requirements, and the intent of the referenced guide.
a. Demisters should be designed, constructed, and tested in accordance with the requirements of Section 5.4 of ANSI N509-1976 (Ref.1). Demisters should meet Underwriters' Laboratories (UL) Class 1 (Ref. 17) requirements.	
k.	

Reg. Guide 1.52 (Cont.)

Regulatory Position	Discussion
b. Air heaters should be designed, constructed, and tested in accordance with the requirements of Section 5.5 of ANSI N509-1976 (Ref. 1).	For SGTS only, heating elements are capable of reducing the relative humidity of incoming air from 100% to 70% under worst possible DBA conditions. Grand Gulf ESF filter train heating elements are not located in adverse environments of pressure, temperature, humidity, or chemical sprays such as exist in containment environments. The Grand Gulf heaters are Class 1E environmentally qualified. Heater controls are designed to minimize malfunction, and have been qualified to operate during and after the design basis earthquake. Heating elements are shielded from upstream and downstream components so as not to pose a potential source of ignition to the adsorbent. Control Room Standby Fresh air heaters are not required due to removing requirement for charcoal filters per Technical Specification Amendment 145 (GNRI-2001/00032)
c. Materials used in the prefilters should withstand the radiation levels and environmental conditions prevalent during the postulated DBA. Prefilters should be designed, constructed, and tested in accordance with the provisions of Section 5.3 of ANSI N509-1976 (Ref. 1).	c. Materials used in the prefilters are of the type commonly used in this application, and are suitable for the environmental conditions at Grand Gulf. Prefilters are UL Class I, and are listed in the UL Building Materials List. Prefilters have not less than 40% atmospheric dust spot efficiency, per ASHRAE standard 52.
d. The HEPA filters should be designed, constructed, and tested in accordance with Section 5.1 of ANSI N509-1976 (Ref. 1). Each HEPA filter should be tested for penetration of di-octyl phthalate (DOP) in accordance with the provisions of MIL-F-51068 (Ref. 19) and MIL-STD-282 (Ref. 20)[2].	d. HEPA filters are steel cased, designed, and individually tested as described in the regulatory position. Iodine remover sprays are not applicable to Grand Gulf.
e. Filter and absorber mounting frames should be constructed and designed in accordance with the provisions of Section 5.6.3 of ANSI N509-1976 (Ref. 1)	e. Mounting frames are designed and constructed in accordance with applicable portions of Section 4.3 of ORNL-NSIC-65.
f. Filter and absorber banks should be arranged in accordance with the recommendations of Section 4.4 of ERDA 76-21 (Ref. 3).	f. Where applicable, Section 4.4 of ORNL-NSIC-65 guidelines are used in design of the ESF filter and absorber banks. The standby gas treatment system filters use vertical-type absorber beds, normally 8" deep. The SGTS filter trains are manufactured by CVI Corporation.
b.	

Reg. Guide 1.52 (Cont.)

Regulatory Position	Discussion
g. System filter housing, including floors and doors, should be constructed and designed in accordance with the provisions of Section 5.6 of ANSI N509-1976 (Ref. 1).	To the extent applicable, housings are designed and constructed per the referenced documents of Regulatory Position C.3(g). Housings are all steel, and proof-tested at an internal negative pressure of 1 psig. Calculations take into account pressure loads as well as seismic loads. Housing joints are continuously welded.
h. Water drains should be designed in accordance with the recommendations of Section 4.5.6 of ERDA 76-21 (Ref. 3).	h. Separate drains are provided for each section of the filter housings.

Regulatory Position	Discussion
i. The absorber section of the SF atmosphere cleanup system may contain any absorbent material demonstrated to remove gaseous iodine (elemental iodine and organic iodides) from air at the required efficiency. Since impregnated activated carbon is commonly used, only this absorbent is discussed in this guide. Each original or replacement batch of impregnated activated carbon used in the absorbent section should meet the qualification and batch test results summarized in Tabled 5.1 of ANSI N509-1976 (Ref. 1). In this table, a "qualification test" should be interpreted to mean a test that establishes the suitability of a product for a general application, normally a onetime test reflecting historical typical performance of material.In this table, a "batch test" should be interpreted to mean a test made on a production batch of product to establish suitability for a specific application. A "batch of activated carbon" should be interpreted to mean a quantity of material of the same grade, type, and series that has been homogenized to exhibit, within reasonable tolerance, the same performance and physical characteristics and for which the manufacturer can demonstrate by acceptable tests an quality control practices such uniformity. All material in the same batch should be activated, impregnated, and otherwise treated under the same process conditions and procedures in the same process equipment and should be produced under the same manufacturing release and instructions. Material produced in the same charge of batch equipment constitutes a batch; material produced in different charges of the same batch equipment should be included in the same batch only if it can be homogenized as above. The maximum batch size should be 350 ft' of activated carbon. If an absorbent other than impregnated activated carbon is proposed or if the mesh size distribution is different from the specifications in Table 5.1 Of ANSI N509-1976 (Ref. 1), the proposed absorbent should have demonstrated the capability to perform as well as or better than activated carbon in satisfying the specifications in Table 5.1 of ANSI N509-1976 (Ref. 1).	The SGTS filters contain 8" of impregnated, activated charcoal.

Regulatory Position	Discussion
If impregnated activated carbon is used as the absorbent, the absorber system should be designed for an average atmosphere residence time of 0.25 sec per two inches of absorbent bed. The absorption unit should be designed for a maximum loading 2.5 mg of total iodine (radioactive plus stable) per gram of activated carbon. No more 5% of impregnant (50 mg of impregnant per gram of carbon) should be used. The radiation stability of the type of carbon specified should be demonstrated (see Section C.1.b of this guide for the design source term).	Design residence time is 0.25 seconds per two inches of absorbent bed, and the absorber has the capacity to load 2.5 mg of total iodine (radioactive plus stable) per gram of activated carbon. The carbon supplied for Grand Gulf is of the type generally used in nuclear filtration applications, and is suitable for the intended radiation service.
j. Absorber cells should be designed, constructed, and tested in accordance with the requirements Section 5.2 of ANSI N509-1976 (Ref. 1).	j. Hopper-type absorbers for Standby Gas Treatment and Tray type absorbers for the control room standby fresh air system charcoal filter trains are in accordance with the guidelines. The absorber has been seismically qualified, by physical testing. Written procedures for filling and packing the absorber have been provided by the manufacturers.
k. The design of the absorber section should consider possible iodine desorption and autoignition that may result from radioactivity-induced heat in the absorbent and concomitant temperature rise. Acceptable designs include a air bleed system, cooling oils, water sprays for the absorber section, or other cooling mechanisms. Any cooling mechanism should satisfy the single-failure criterion. A low-flow air bleed system should satisfy the single-failure criterion low-humidity (less than 70% relative humidity) cooling air-flow.	k. To preclude iodine desorption, the filter housings are equipped with thermal detectors and manual water spray systems. Based on radio-iodine loadings, absorbent autoignition is not considered feasible
l. The system fan, its mounting, and the connections should be designed, constructed, and tested in accordance with the requirements of 5.7 And 5.8 of ANSI N509-1976 (Ref. 1).	l. System fans and ductwork are designed in accordance with the applicable portions of Section 2.7 of ORNL-NISC-65. The ANSI standard includes the design requirements which are commensurate with the recommendations set forth in the ERDA document.
m. The fan or blower used on the cleanup system should be capable of operating under the conditions postulated, including radiation.	m. The fans and motors have been qualified to the environmental conditions of the postulated environment. Refer to Table 3.11-7.
n. Ductwork should be designed, constructed and tested in accordance with the provisions of Section 5.10 of ANSI N509-1976 Ref. 1).	n. Ductwork design exceeds the requirements of SMACNA for high pressure, high-velocity. Ductwork is unpainted, welded-seam, with bolted, flanged transverse joints. Hangers and supports are seismically designed

Reg. Guide 1.52 (Cont.)

Regulatory Position	Discussion
o. Ducts and housings should be laid out with a minimum of ledges, protrusions, and crevices that could collect dust and moisture and that could impede personnel or create a hazard to them in the performance of their work. Straightening vanes should be installed where required to ensure representative air flow measurement and uniform flow distribution through cleanup components.	This is complied with. Physical layout is such that straightening vanes are not deemed necessary.
p. Dampers should be designed, constructed, and tested in accordance with the provisions of Section 5.9 of ANSI N509-1976 (Ref. 1).	p. Dampers were not supplied with the ESP filter systems.
4. Maintenance	
a. Accessibility of components and maintenance should be considered in the design cleanup systems in accordance with the provisions Section 2.3.8 of ERDA 76-21 (Ref. 3) and Section 4.7 of ANSI N509-1976 (Ref. 1)	C.4.a. Consistent with filter housing dimensions, access doors are 20" x 50".
b. For ease of maintenance, the system design should provide for a minimum of three linear feet from mounting frame to mounting frame between banks of components. If components are to dimension to be provided should be of the maximum length of the component plus a minimum of 3 feet.	b. Spacing between components is shown in Figures 12.3-18 and 12.3-19. All components which require servicing or removal have been provided with adequate access space, consistent with manufacturer's recommendations.
c. The system design should provide for permanent test probes with external connections in accordance with the provisions of 4.11 of ANSI N509-1976 (Ref. 1)	c. Permanent connections for test probes are provided; however, they are not manifolded at a single location
d. Each atmosphere cleanup train should operated at least 10 hours per month, with the theaters on (if so equipped), in order to reduce the buildup of moisture on the adsorbers and HEPA filters.	d. This requirement is incorporated into the Grand Gulf Technical Specifications. Refer to Chapter 16.0.
e. The cleanup components (i.e., HEPA filters, and adsorbers) should not be installed while active construction is still in progress.	e. Comply, except for the control room fresh air charcoal filter trains. Because the Grand Gulf control room is common for Units 1 and 2, Unit 1 will be operational, while moderate amounts of construction are still underway in the Unit 2 portion of the control room.

Regulatory Position	Discussion
5. In-Place Testing Criteria	C.5.a. These testing requirements have been incorporated into the Grand Gulf Technical Specifications. "Operating cycle" interpreted to mean refueling cycle. (Note: Visual inspections are prerequisites for some Specification testing).
a. A visual inspection of the ESP atmosphere cleanup system and all associated components before each in-place airflow distribution test. DOP test, or activated carbon absorber section leak test in accordance with the provisions of Section 5 N510-1975 (Ref. 2).	
b. The airflow distribution to the HEPA filters and iodine absorbers should be tested in place for uniformity initially and after maintenance affecting the flow distribution. The distribution should be within ±20% of the average flow per unit. The testing should be conducted in accordance with the provisions of Section 9 of "Industrial Ventilation" (Ref. 21) and Section 8 of ANSI N510-1975 (Ref. 2).	b. See item (a), above (Note Initial testing was part of startup testing)
c. The in-place DOP test for HEPA filters should conform to Section 10 of ANSI N510-1975 (Ref. 2). HEPA filter sections should be tested in place (1) initially, (2) at least once per 18 months thereafter, and (3) following painting, fire, or chemical release in any ventilation zone communicating with the system to confirm a penetration of less than 0.05% at rated flow. An engineered-safety-feature air filtration system satisfying this condition can be considered to warrant a 99% removal efficiency for particulates in accident dose evaluations. HEPA filters that fail to satisfy this condition should be replaced with filters qualified pursuant to regulatory position C.3.d of this guide. If the HEPA filter bank is entirely or only partially replaced, an in-place DOl? test should be conducted.	c. See item (a) above. Refer to Chapter 15.0 for removal efficiencies utilized in accident dose evaluations.
If any welding repairs are necessary on, within, or adjacent to the ducts, housing, or mounting frames, the filters and absorbers should be removed from the housing during such repairs. The repairs should be completed prior to periodic to sting, filter inspection, and in-place testing. The use of silicone sealants or any other temporary patching material on filters, housing, mounting frames, or ducts should not be allowed.	Welding repairs will not be conducted on housings with filters and absorbers installed. It is intended that silicone sealants will not be used on the ESF filter systems.

Reg. Guide 1.52 (Cont.)

Regulatory Position	Discussion
d. The activated carbon absorber section should be leak tested with a gaseous halogenated hydrocarbon refrigerant in accordance with Section 12 of ANSI N510-1975 (Ref. 2) to ensure that bypass leakage through the absorber: section is less than 0.05%. After the test is completed, air flow through the unit should be maintained until the residual refrigerant gas in the effluent is less than 0.01 ppm. Absorber leak testing should be conducted (1) initially, (2) at least once per 18 months thereafter, (3) following removal of an absorber sample for laboratory testing if the integrity of the absorber section is affected, and (4) following painting, fire, or chemical release in any ventilation zone communicating with the system.	d. Carbon absorber bypass testing is incorporated into the Grand Gulf Technical Specifications.
6. Laboratory Testing Criteria for Activated Carbon	C.6. Note: Laboratory testing of carbon is performed in accordance with ASTM D3803.
a. The activated carbon absorber section of the atmosphere cleanup system should be assigned the decontamination efficiencies given in Table 2 for elemental iodine and organic iodides if the following conditions are met: (1) The absorber section meets the conditions given in regulatory position C.5d of this guide, (2) New activated carbon meets the physical property specifications given in Table 5.1 of ANSI N509-1976 (Ref. 1), and (3) Representative samples of used activated carbon pass the laboratory tests given in Table 2. If the activated carbon fails to meet any of the above conditions, it should not be used in engineered-safety-feature absorbers.	a. Refer to Chapter 15.0 for decontamination efficiencies used in accident dose calculations. (1) Absorber bypass testing in incorporated into Grand Gulf Technical Specifications. (2) Comply. (3) Samples will be tested in accordance with Table 2 requirements. If a sample fails to comply, it will be removed from ESF filter service.

Reg. Guide 1.52 (Cont.)

Regulatory Position	Discussion
b. The efficiency of the activated carbon absorber section should be determined by laboratory testing of representative samples of the activated carbon exposed simultaneously to the same service conditions as the absorber section. Each representative sample should be not less than two inches in both length and diameter, and each sample should have the same qualification and batch test characteristics as the system absorbent. There should be a sufficient number of representative samples located in parallel with the absorber section for estimating the amount of penetration of the system absorbent throughout its service life. The design of the samplers should be in accordance with the recommendations of Appendix A of draft standard ANSI N509 (Ref. 1). Where the system activated carbon is greater than two inches deep, each representative sampling station should consist of enough two-inch samples in series to equal the thickness of the system absorbent. Once representative samples are removed for laboratory test, their positions in the sampling array should be blocked off. Laboratory tests of representative samples should be conducted, as indicated in Table 2 of this guide, with the test gas flow in the same direction as the flow during service conditions. Similar laboratory tests should be performed on an absorbent sample before loading into the absorbers to establish an initial point for comparison of future test results. The activated carbon absorber section should be replaced with new unused activated carbon meeting the physical property specifications of Table 5.1 of ANSI N509-1976 (Ref. 1) if (1) testing in accordance with the frequency specified in Footnote c of Table 2 results in a representative sample failing to pass the applicable test in Table 2 or (2) no representative sample is available for testing.	b. Absorbent samples have been designed to be exposed to the same service conditions as the main absorber. Charcoal used in the sample canisters is from the same lot as used in the main absorber, and is representative of the thickness of the main bed. Once removed, sample positions are sealed to prevent absorber bypass. If samples fail to meet minimum requirements of Table 2, the absorber section will be replaced with charcoal meeting the physical properties of Table 5.1 of ANSI N509-1976.

TABLE 2: REG. GUIDE 1.52 TABLE 2 HAS BEEN DELETED

Regulatory Guide 1.53, (June 1973)

Application of the Single-Failure Criterion to Nuclear Power Plant Protection Systems

Project Position - Comply.

FSAR Subsection - 7.1.2.6.11, 8.3.1

Regulatory Guide 1.54, (June 1973)

Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants

Bechtel Position - ANSI N101.4, where applicable, is used only as a guideline for the basis as complying with quality assurance requirements of Appendix B, 10 CFR Part 50.

These guidelines were followed for organic protective coatings selected and evaluated in accordance with pertinent sections of ANSI N101.2 when applied to interior surfaces of the containment and drywell.

The quality assurance program for protective coatings includes the planned and systematic actions necessary to provide adequate confidence that shop or field coatings for nuclear facilities will perform satisfactorily in service.

GEH Position - The only significant organic materials on GEH-supplied equipment would be the protective coatings used on carbon steel components. These coatings are specified to meet the essential technical requirements of Regulatory Guide 1.54 and are qualified using the standard ANSI tests. However, because of the impracticability of using these special coatings on all equipment, certain exemptions are allowed. These exemptions are restricted to small-size equipment where, in case of a LOCA, the plant debris would in no way be a safety hazard. Exemptions are for such items as electronic/electrical trim, covers, face plates, valve handles, etc. Coatings on equipment shipped after early 1977 meet the requirements of Regulatory Guide 1.54 except for the small parts discussed above.

The quality assurance requirements of Regulatory Guide 1.54 are met to the extent specified below for protective coatings applied to exposed NSSS equipment or structural surfaces within the Mark III primary containment.

		QA Requirement	
	Category	**Equipment Surfaces Not Covered By Insulation**	**Structural Surfaces**
A.	Potential Post-LOCA Debris Pathway to Suppression Pool Exists	Certificate of Compliance	Under Development
B.	Small Equipment[2]	Manufacturer Standard	
C.	No Potential Post-LOCA Debris Pathway to Suppression Pool	Regulatory Guide 1.54 Not Applicable	
D.	Other NSSS equipment	RG 1.54 Partially Used[3]	

[1] Product Quality Certifications to Procurement Specifications (including Coating Process Specifications requiring coating manufacturer's paint certifications).

[2] Face plates, valve handles, solenoid valve covers, small-size pipe whip restraints, etc.

[3] See Section 6.1.2.

Operational Stage Position

The Operational Quality Assurance Program complies with this guide with the following clarification:

(1) Entergy Operations shall comply with the regulatory position established in this regulatory guide in that QA programmatic/administrative requirements included therein shall apply to maintenance and modification activities even though such requirements were not in effect originally. Technical requirements associated with maintenance and modifications shall be the original requirements or better

(e.g., code requirements, material properties, design margins, manufacturing processes, and inspection requirements).

(2) The guidance of Regulatory Guide 1.54 shall be followed for organic protective coatings selected and evaluated in accordance with pertinent sections of ANSI N101.2 when applied to interior surfaces of the containment and drywell. The supplier's quality assurance program shall be approved prior to implementation. Quality assurance documentation may not be similar to records and documents listed in Sections 7.4 through 7.8 of ANSI N101.4 but will be evaluated to assure that they provide at least the same degree of documentation as required by this standard.

FSAR Subsection - 6.1.2

Regulatory Guide 1.56, Rev. 1 (July 1978)

Maintenance of Water Purity in Boiling Water Reactors

Project Position - Comply, with the following exception:

Reference: Regulatory Position C.1, para. 2 of the Regulatory Guide.

The condensate sampling system serves no safety function and is not addressed in the plant Technical Specifications; however, a chemical analysis is performed for chlorides, silica, iron, copper, nickel, chrome, turbidity, and others as deemed necessary for efficiency and performance checks as required by Plant Administrative Procedures and based on System classification. The conductivity of the condensate is continuously monitored to provide indications of dissolved impurities. The feedwater is routinely analyzed for turbidity in order to ensure feedwater quality commensurate to that of the reactor coolant.

FSAR Subsections - 5.2.3.2.2, 7.7.2.8.2, 10.4.6

Regulatory Guide 1.57, (June 1973)

Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components

Project Position - Grand Gulf guard pipe design complies with the requirements of Regulatory Guide 1.57, with the following exception:

Jet/force impingement loads are not applicable to Grand Gulf's design.

The Grand Gulf drywell design does not employ the loading criteria of Article NE-3000 of Subsection NE of the ASME Code, Section III, Division I as augmented by Regulatory Guide 1.57.

FSAR Subsection - 3.8.3, 3.6

Regulatory Guide 1.58, (August 1973)

Qualification of Nuclear Power Plant
Inspection, Examination, and Testing Personnel

Operational Stage Position - Complies with this guide with clarifications as noted below and in FSAR Section 17.2.

(1) The guidance of Regulatory Guide 1.58 shall be followed as it pertains to the qualifications of personnel who verify conformance of work activities to quality requirements. The qualifications of plant operating personnel concerned with day-to-day operation, maintenance, and certain technical services shall conform to Regulatory Guide 1.8.

(2) Not all personnel who approve inspection and test procedures will be certified as meeting the Level III capability requirements of ANSI N45.2.6-1973, but personnel who approve inspection and test procedures will be determined by management, through evaluation of their education, training, and experience, to be fully qualified and competent to approve such procedures. The basis for the determination will be documented.

FSAR Section - 17.2

Regulatory Guide 1.59, Rev. 2 (August 1977)

Design Basis Floods for Nuclear Power Plants

Project Position - The project position on Regulatory Guide 1.59 (Rev. 2) for the Grand Gulf site is as follows:

For the Grand Gulf site, the Mississippi River probable maximum flood (PMF) estimate is based on the U. S. Army Corps of Engineers design project flood (DPF), which is defined as "the most severe flood on the Mississippi River which can reasonably be expected to occur." This design project flood (DPF) is analogous to standard project flood (SPF), which is about 40 to 60 percent of PMF. The PMF level at the site is controlled by the levee on the west bank of the Mississippi River, which is approximately at El. 103 ft. The plant site is considered a dry site because the plant grade elevation (132.5 ft msl) is about 30 feet higher than the PMF level of the Mississippi River (Ref. 1-2).

The PMF for two local streams close to the plant site has been estimated based on the unit hydrograph method in accordance with Regulatory Guide 1.59 (Rev. 2), Appendix A (Ref. 3).

References:

(1) Standard Project Flood Determinations, EM-1110-2-1411, U.S. Army Corps of Engineers, March 1965.

(2) Department of Army, Mississippi River Commission, U. S. Army Corps of Engineers, letter to Woodward-Envicon Inc., on March 16, 1972.

(3) ANSI N-170, "Standards for Determining Design Basis Flooding at Power Reactor Sites," Draft, August 1976.

FSAR Subsections - 2.4.3, 2.4.4

Regulatory Guide 1.60, Rev. 1 (December 1973)

Design Response Spectra for Seismic Design of Nuclear Power Plants

Bechtel Position - Grand Gulf design response spectra in relation to the subject Regulatory Guide are as follows:

(1) Reference: Paragraph C.1 of the Regulatory Guide. Refer to subsection 3.7.1 for a discussion of the horizontal ground design response spectra.

(2) Reference: Paragraph C.2 of the Regulatory Guide. The design response spectra for the vertical component of the SSE and OBE are specified as two-thirds the value of the corresponding horizontal component throughout the frequency range.

GEH Position - The attached Figure 1 is a clarification of Figure 2 of the Regulatory Guide. This figure was used for the vertical design response spectra for NSSS equipment. The response spectra differ for NSSS and non-NSSS equipment because GE utilizes a conservative calculation which bounds numerous sites. The GEH approach is consistent with NRC requirements specifically called out in Regulatory Guide 1.60.

FSAR Section - 3.7.1

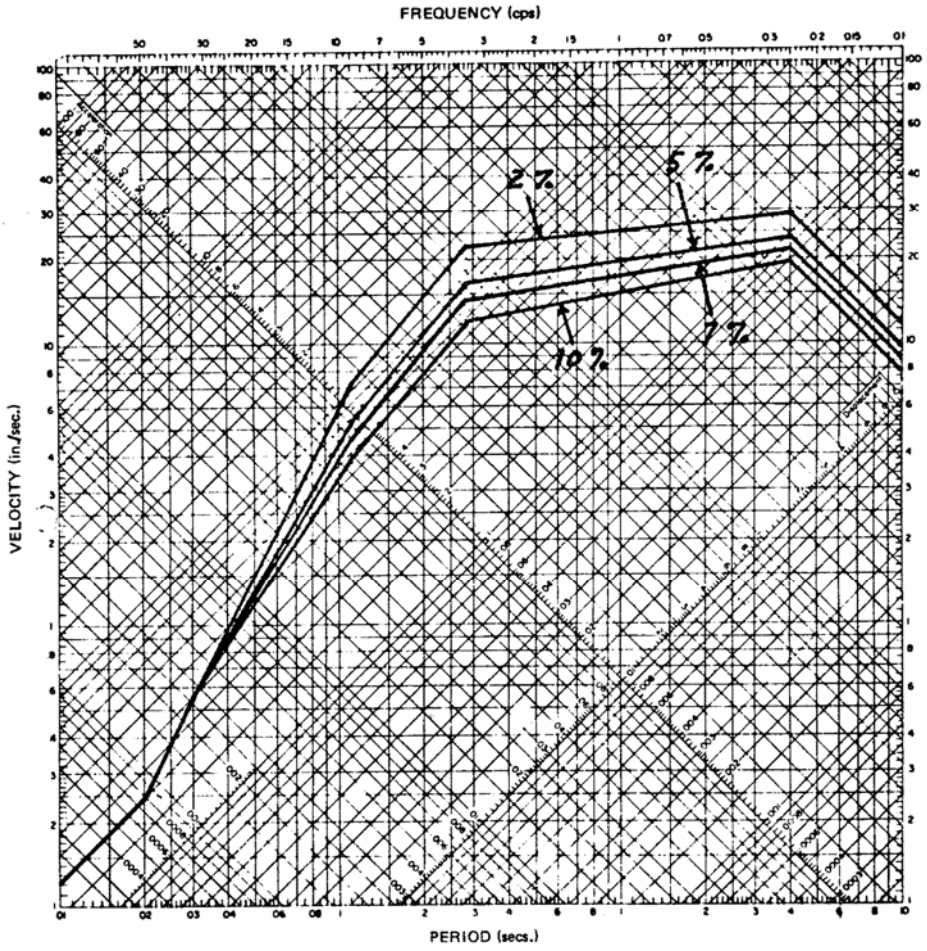


FIGURE 1. VERTICAL DESIGN RESPONSE SPECTRA - SCALED TO 1g HORIZONTAL GROUND ACCELERATION

Regulatory Guide 1.61, (October 1973)

Damping Values for Seismic Design of Nuclear Power Plants

Bechtel Position - Comply

Reference: BC-TOP-4, Rev. 1

GEH Position - General Electric conforms to the subject Regulatory Guide as follows:

Reference: Paragraph C.3 of the Regulatory Guide. Requires that lower damping values be used if the maximum combined stresses are significantly lower than yield or ½ yield stresses. The GEH Position is consistent with current industrial practice and has been accepted by the NRC.

FSAR Section - 3.7 (Table 3.7-3)

Regulatory Guide 1.62, (October 1973)

Manual Initiation of Protective Actions

Project Position - Comply.

FSAR Subsection - 7.1.2.6

Regulatory Guides 1.63, (October 1973 and February 1987)

Electric Penetration Assemblies in Containment Structures for Water-Cooled Nuclear Power Plants

Project Position - Electric penetration assemblies are in conformance with the objectives of IEEE 317-1972 and Regulatory Guide 1.63, October 1973. Only fiber optic penetration assemblies are in conformance with the objectives of IEEE 317-1983 and Regulatory Guide 1.63, February 1987. Refer to subsections 8.1.4 and 8.3.1.2.3 for a discussion of how position C.1 is met.

FSAR Subsections - 8.1.4, 8.3.1.2.3

Regulatory Guide 1.64, Rev. 2 (June 1976)

Quality Assurance Requirements for the Design of Nuclear Power Plants

Operational Stage Position - Complies with this guide with clarifications as noted in FSAR Section 17.2.

FSAR Section - 17.2

Regulatory Guide 1.65, (October 1973)

Materials and Inspections for Reactor Vessel Closure Studs

Project Position - This regulatory guide is not considered a requirement for this project. However, we meet the requirements specified in Regulatory Guide 1.65 with the following clarifications on the referenced paragraphs:

(1) Reference: Paragraph C.2 of the Regulatory Guide. The reactor pressure vessel closure studs are SA450 Grade B23 or B24 (AISI 4340) and are specified to have a maximum ultimate tensile strength of 170 ksi. Additionally, GE specified the bolting material must have Charpy V-notch impact properties of 45 ft.-lbs. minimum with 25 mils lateral expansion. Nondestructive examination before and after threading is specified to be in accordance with subarticle NB-2580 ASME Section III, which complies with Regulatory Position C.2. Subsequent to fabrication, the studs are manganese phosphate coated and are lubricated with a graphite/alcohol or a nickel powder base lubricant.

(2) Reference: Paragraph C.2.b. of the Regulatory Guide. Examination is performed after heat treatment and is in accordance with NB-2584. Straight beam examination is performed on 100 percent of the cylindrical surface and from both ends of each stud using a 3/4-inch (19 mm) maximum diameter transducer. Radial scan calibration is based on a 1/2-inch (12.7 mm) diameter flat bottom hole of a depth equal to 10 percent of the material thickness. End scan calibration is in accordance with NB-2585. Angle beam examination is performed on the outer cylindrical surface of nuts and washers in accordance with ASME SA-388 in both axial and circumferential directions. Any indication greater than the indication from the applicable calibration feature is unacceptable. A distance-amplitude correction curve in accordance with NB-2585 is used for the longitudinal wave examination.

(3) Reference: Paragraph C.3 of the Regulatory Guide. GEH practice allows exposure of stud-bolting surfaces to higher purity fill water; nuts and washers are dry stored during refueling.

FSAR Subsection - 5.3.1.7

Regulatory Guide 1.66, Rev. 0 (October, 1973)

Nondestructive Examination of Tubular Products

Project Position - This regulatory guide was withdrawn on September 28, 1977 by the NRC. Therefore, no assessment is provided (Ref. 1).

Reference 1: Letter from Mr. Robert B. Minogue, Director, Office of Standards Development.

Regulatory Guide 1.67, (October 1973)

Installation of Overpressure Protective Devices

Project Position - The main steam line safety/relief valves relieve to closed discharge systems, as described in subsection 5.2.2. The guidelines delineated in the regulatory guide are not applicable to that situation.

Regulatory Guide 1.68, Rev. 1 (January 1977)

Initial Test Programs for Water-Cooled Reactor Power Plants

Project Position

(1) Reference: Page 1.68-1, Section B, first paragraph, line eight. To the extent practical, the GGNS Startup Program will verify the analytical model assumptions used for predicting plant responses to anticipated transients and postulated accidents. GGNS is not designed or sufficiently instrumented to validate analytical models. GE has a continuing program of model vibration using test reactors and facilities and evaluating operating and special test data from their existing operating BWRs. Preoperational and startup testing will typically verify that the assumptions of the accident analysis are met, i.e., minimum pump flow rates, equipment response times, etc.

(2) Reference: Page 1.68-1, Section B, first paragraph, items (3) and (4). During preoperational testing, tests are conducted, where practical, following plant operating, abnormal, and emergency procedures with the intent of verifying their adequacy. However, familiarizing the plant staff with plant operations and conducting trial use of plant procedures are not primarily functions of the test program.

(3) Reference: Page 1.68-2, first column, fourth paragraph, first sentence. Operating experience reports, including reportable occurrences relevant to startup testing, when considered appropriate, will be used in developing and conducting the test program.

(4) Reference: Page 1.68-2, first column, next to last paragraph. Certain preoperational tests such as control rod drive timing, control rod drive friction and functional tests, and SRM neutron response cannot be completed or sometimes even begun until fuel loading.

(5) Reference: Page 1.68-2, second column, first paragraph, last sentence. The sequence of startup tests will be ordered so that the safety of the plant is never dependent on the

performance of untested structures, systems, and components. However, during component testing and preoperational testing, certain components, systems, and structures may be tested with their supporting component's, system's, or structure's testing incomplete. When this is done, due regard will be given to plant safety and to the effect on the validity of test results. For example, in the component tests of the instrument air compressor, temporary cooling water may be provided if the normal cooling from the Turbine Building Cooling Water System is unavailable. Such an arrangement is in no way unsafe and in no way invalidates the component test.

(6) Reference: Page 1.68-3, Section C.2, first paragraph, last sentence. Construction-related inspections and tests will be completed to the degree that the outstanding inspections and tests could not reasonably affect the validity of test results.

(7) Reference: Page 1.68-3, Section C.2, second paragraph, first sentence. Tests designated as preoperational tests in the FSAR will be completed prior to issuance of an operating license. It should be realized that FSAR Section 14.2 provides very basic and preliminary test descriptions. Testing may be conducted in a manner equivalent to or better than as is described in FSAR Section 14.2. Test results essential to the safe conduct of fuel loading will be evaluated and approved prior to issuance of an operating license.

(8) Reference: Page 1.68-3, Section C.2, second paragraph, second sentence. The proper operation of interlocks, set points, and other protective features used to meet surveillance requirements will be demonstrated. The actual surveillance tests themselves are not included with the scope of the initial test program.

(9) Reference: Page 1.68-3, Section C.3, first sentence. To the extent practical, limiting control system and other limiting equipment failures or malfunctions reasonably expected to occur during the plant lifetime will be simulated.

(10) Reference: Page 1.68-3, Section C.4, sixth sentence. See position No. (3).

(11) Reference: Page 1.68-3, Section C.4, seventh sentence. Test procedures will be available on site approximately 60 days prior to their intended use. Approved test procedures will be available on site approximately 30 days prior to their intended use. In exceptional cases, such as extra tests, special tests, or retests, no specified availability is stated. Copies of all tests will be made available to regulatory personnel for their use.

Approved procedures will not normally be available until 30 days prior to use to allow flexibility in making changes, to maximize review, and to maintain tight control over the issuance of procedures. The circumstances through which extra tests, special tests, or retests may come about preclude their availability at a specified time prior to use.

(12) Reference: Page 1.68-3, Section C.5. Startup schedules for preoperational and initial startup test phases will provide a sufficient amount of time to perform orderly and comprehensive testing. Specification of arbitrary time limitations is not necessary for conduct of an adequate test program. For example, this might preclude application of additional manpower, working extra shifts and overtime.

(13) Reference: Page 1.68-3,4, Section C.7. See position No. (7).

(14) Reference: Page 1.68-4, Section C.8, last sentence. Hold points will be established at appropriate power level test conditions, e.g., 25 percent, 50 percent, and 75 percent.

(15) Reference: Page 1.68-4, Section C.9. The completed test procedures, data forms, graphs, photographs, etc. constitute the official historical record of the preoperational test program. A Startup Summary Report will be submitted in accordance with Regulatory Guide 1.16. In regard to item b, applicable test data will be compared with acceptance criteria. Data are often taken during the performance of a test that does not directly apply to the acceptance criteria. If further detail is desired, it will be available from the plant files.

(16) Reference: Page 1.68-5, App. A, 1.a.(1). Vessel displacements will not be quantitatively measured. Startup is unaware of any criteria to apply to such vessel

displacement measurements. Hot testing is generally not possible on BWR reactor coolant systems and is deferred to heatup during the startup test phase.

(17) Reference: Page 1.68-6, App. A, 1.b.(3). The SLC Sampling System will be tested assuming proper mixing upon solution addition, air agitation, etc., assures sample adequacy.

(18) Reference: Page 1.68-6, App. A, 1.c, first sentence. The definitions of ECCS System responses time, isolation system response time, and reactor trip system response time in the Technical Specifications state: "[the response time] is that time interval from when the monitored parameter exceeds its trip set point at the channel sensor until [the desired action occurs]." Therefore, measurements of response times will not include, "...associated hardware between the measured variable and input to the sensor..." Response times will be obtained for those instruments listed in the Technical Specifications that require response time testing.

(19) Reference: Page 1.68-6, App. A, 1.c, last sentence. Startup takes exception to any statements regarding testing for ATWS type fixes. If required, such testing will be addressed when the ATWS issue is settled for GGNS.

(20) Reference: Page 1.68-6, App. A, 1.d, 1.e, 1.f. Only those structures, systems, components, and design features conforming with the criteria stated in Section C.1 of Regulatory Guide 1.68 will be subject to preoperational testing. Additionally, circulating water and makeup water systems do not require expansion and restraint tests since they are cold water systems.

(21) Reference: Page 1.68-7,8, App. A, 1.g. Only safetyrelated Class IE portions of normal power systems, and emergency power systems will be subject to preoperational tests. The criteria of Section C.1 will be used in making the determination.

The term "load-carrying capability" is ambiguous. The test program demonstrates the ability of system components and cables to adequately supply system electrical loads.

(22) Reference: Page 1.68-7, App. A., 1.g(1), second paragraph. The integrated system test will demonstrate system performance when subjected to a simulated full loss of

offsite power and a coincident simulated accident signal. This reflects the requirements of Regulatory Guide 1.41, Section C.2.

(23) Reference: Page 1.68-7, App. A., 1.g(2), second paragraph. Emergency loads will be tested to demonstrate operation with the nominal design voltage available. No practical method exists to vary voltages to obtain maximum and minimum design voltages.

In addition, "equilibrium conditions" are integrated to mean "electrical equilibrium conditions" (also in 1.g(3)).

(24) Reference: Page 1.68-7,8, App. A, 1.g(4). Maximum dc loads will be applied in sequence during a "maximum credible accident (MCA)" and would be impossible to simulate. The actual load currents will be checked during the loss-of-power test and will be compared with design loads to ensure agreement. The loss-of-power test will not provide total dc system loads.

(25) Reference: Page 1.68-8, App. A, 1.h, second paragraph. Protective devices, such as leaktight covers or housings, will not be preoperationally tested. Rather, credit will be taken for the manufacturer's environmental qualification tests.

(26) Reference: Page 1.68-8, App. A, 1.h(1)(c). "Pump runout conditions" is ambiguous. Testing will demonstrate, to the extent practical, pump operability in all expected operation modes including design pump conditions and the capability to inject at required flow and pressure.

(27) Reference: Page 1.68-8, App. A, 1.h(1)(d). In testing over-pressure features, credit will be taken for bench testing of safety and relief valves. Pressure switches, etc., may be tested in the field.

(28) Reference: Page 1.68-8, App. A, 1.h(1)(e). This does not apply to a BWR test program. Accordingly, no testing is planned.

(29) Reference: Page 2.68-8, App. A, 1.h(5). The prompt relief trip feature is not used on GGNS. Accordingly, no testing is planned.

(30) Reference: Page 1.68-9, App. A, 1.h(6). Cold water interlocks are a PWR design feature and are not present on GGNS. Accordingly, no testing is planned.

(31) Reference: Page 1.68-9, App. A, 1.h(9). We interpret "transfer devices" to be "pumps."

(32) Reference: Page 1.68-9, App. A, 1.h(10). Drywell and containment fans are not required to operate in a post-accident environment to remove heat. Therefore, the drywell and containment fans will not be tested at containment peak design pressure.

(33) Reference: Page 1.68-9, App. A, 1.i, second paragraph, next to last sentence. Testing large isolation devices for closure under "containment overpressure" is prohibitive. This is due to the requirement to establish a test pressure in the upstream volume with the isolation device initially open.

(34) Reference: Page 1.68-9, App. A, 1.i(2). Valve closure times will be measured from fully opened to fully closed exclusive of initiating sensor response time.

(35) Reference: Page 1.68-9, App. A, 1.i(13). The Mark III containment used at GGNS is not inerted. Accordingly, no inerting tests are required.

(36) Reference: Page 1.68-10, App. A, 1.j(15). GNS will not use an automatic dispatcher control system. Hence, no testing is planned.

(37) Reference: Page 1.68-10, App. A, 1.k(2), 1.k(3). Personnel monitors and radiation survey instrument tests are not considered preoperational tests and are done by the Radiation Protection Section. Testing of laboratory equipment used in radiochemical analysis is not a preoperational test. Such tests are the responsibility of the Plant Chemistry Section.

(38) Reference: Page 1.68-10, App. A, 1.1. Tank volumes and capacities can be easily verified by calculations and will not be tested. Flow rates will be measured and holdup times can then be calculated.

We interpret "spiked samples" to mean "radioactive samples." Radioactive samples might be useful only in testing radiation detectors, and their contaminating presence complicates the correction of any identified deficiencies. Accordingly, "spiked" samples will not be used in testing radwaste systems.

(39) Reference: Page 1.68-11, 12, App. A, 1.n, 1.o. Only those structures, systems, components, and design features conforming with the criteria stated in Section C.1 of Regulatory Guide 1.68 will be subject to preoperational testing.

(40) Reference: Page 1.68-12, App. A, Section 2, first paragraph. Requirements for continuous monitoring of neutron flux throughout the core loading will be established so that all changes in the multiplication factor are observable. The objective of neutron monitoring is to detect expected and unexpected changes in core reactivity. However, the multiplication factor or the magnitude of its change cannot be "known" from monitoring data.

(41) Reference: Page 1.68-12, App. A, Section 2, second paragraph. Measured nuclear instrumentation responses will be compared with expected trends. Criteria and action requirements to be taken if the measured values deviate from the expected trends will be established. Shutdown margin verifications will be performed at appropriate intervals.

Predictions of neutron monitoring instrument response cannot be made with any reasonable degree of certainty. However, general response characteristics based on previous similar core loadings can be utilized for expected responses.

(42) Reference: Page 1.68-12, App. A, Section 2, paragraph 4, subparagraph b, last sentence. An acceptable deviation limit on excessive scram times will be established and retest criteria to demonstrate acceptability will be developed.

(43) Reference: Page 1.68-12, App. A, Section 2, paragraph 4, subparagraph g. Intermediate range monitors cannot be calibrated to indicate reactor power until low power testing. Operability checks of alarms and protective trip functions may be performed.

(44) Reference: Page 1.68-15, App. A, 5.h. Selected control rod scram times will be measured during plant transients which result in scrams.

(45) Reference: Page 1.68-15, App. A, 5.k. Simulated automatic actuation of ECCS systems will not be done at power. To do so causes a cold water injection resulting in power transients and increased fuel and reactor pressure vessel thermal cycling.

(46) Reference: Page 1.68-15, App. A, 5.q. No failed fuel is expected during power ascension testing. Accordingly, the main steam line radiation monitoring system will only be subject to routine operational and surveillance checks and calibrations.

(47) Reference: Page 1.68-15, App. A, 5.t. Verification of relief valve set points and relief valve reset pressures, etc., during power testing necessarily must be limited to only the valves which actuate during a transient test or which may be remotely actuated.

(48) Reference: Page 1.68-15, App. A, 5.dd. The remote shutdown panel does not have scram capabilities; it is assumed the reactor would be scrammed before the control room is evacuated.

(49) Reference: Page 1.68-16, App. A, 5.gg. See position No. (19) and position No. (29).

(50) Reference: Page 1.68-16, App. A, 5.ii. The dynamic response of the plant to recirculation pump trips will be done in the midpower range (50 percent - 75 percent) and at full power (100 percent). Flow coastdown rate is not strongly power dependent; hence, more flexibility in initial power level is warranted for the midpower range test.

(51) Reference: Page 1.68-16, App. A, 5.jj. This test may also be done by actually tripping the turbine and interrupting the offsite power feed in lieu of simulation.

(52) Reference: Page 1.68-16, App. A, 5.kk. Loss of feedwater heater tests result in power increases. Accordingly, they will be started at less than 100 percent power to maintain conformance to the Technical Specifications.

Loss of a single feedwater heater at low to mid-power levels results in a mild transient due to lower initial feedwater temperature and reduced feedwater flow. Dynamic response of the plant is verified at the high power test; hence, the 50 percent power test is inappropriate.

(53) Reference: Page 1.68-16, App. A, 5.nn. The generator output breakers will be tripped. However, the requirement for producing a "maximum credible overspeed" has no bearing on the safety of the NSSS.

(54) Reference: Page 1.68-17, App. C, 1.e. It is not practical to simulate the environmental conditions of accident situations during field tests. Credit will be taken for the environmental testing a vendor must successfully complete to qualify his product for nuclear service.

(55) Reference: Page 1.68-19, App. C, 1.i. Test records will identify the data recorder. In most test situations the data recorder and observer are the same person.

(56) Reference: Page 1.68-20, App. C, 2.c(2). If an increment is reduced because of an indication of excessive subcritical multiplication, it shall only be increased again in accordance with an approved fuel loading procedure.

(57) Reference: Page 1.68-21, App. C, Section 3, last paragraph, last sentence. High flux scram trips will be set at their lowest values.

(58) Reference: Page 1.68-21, App. C, Section 4, first paragraph, last sentence. Tests will be conducted at appropriate power level test conditions, e.g., 1-5 percent, 10 percent, 25 percent, 50 percent, 75 percent, and 100 percent power levels.

(59) Reference: Page 1.68-21, App. C, Section 4, last paragraph, subparagraph e. Reactor vessel temperatures will be monitored at various locations. However, temperature surveys of the core internal components will not be performed.

(60) Reference: Page 1.68-21, App. C, Section 4, last paragraph, subparagraph h. Core thermal limits and physics parameters will be evaluated for conformance to the Technical Specifications and core performance test acceptance criteria prior to ascent to the next power plateau.

Regulatory Guide 1.68.1, Rev. 1 (January 1977)

Preoperational and Initial Startup Testing of Feedwater and Condensate Systems for Boiling Water Reactor Power Plants

Project Position - Comply.

FSAR Chapter - Chapter 14

Regulatory Guide 1.68.2, (July 1978)

Initial Startup Test Program to Demonstrate Remote Shutdown Capability for Water-Cooled Nuclear Power Plants

Project Position - Comply

FSAR Subsection - 14.2.12.3.25

Regulatory Guide 1.70, Rev. 2 (September 1975)

Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants - LWR Edition

Project Position - Comply.

FSAR Section - 1.1

Regulatory Guide 1.71, (December 1973)

Welder Qualification for Areas of Limited Accessibility

Project Position - Grand Gulf complies with R.G. 1.71 with the following clarifications:

(1) Reference: Paragraph C.1 of the Regulatory Guide. Performance qualifications for personnel who weld under conditions of limited access, as defined in Regulatory Position C.1, are maintained in accordance with the applicable requirements of ASME Sections III and IX. Additionally, responsible site supervisors are required to assign only the most highly skilled welders to limited-access welding. Of course, welding conducted in areas of limited access is subjected to the required nondestructive examination, and no waiver or relaxation of examination methods or acceptance criteria because of the limited access is permitted.

(2) Reference: Paragraph C.2 of the Regulatory Guide. Requalification is required when any of the essential variables of ASME Section IX are changed or when any authorized inspector questions the ability of the welder to perform satisfactorily to the requirements of ASME Sections III and IX.

(3) Reference: Paragraph C.3 of the Regulatory Guide. Production welding is monitored and welding qualifications are certified in accordance with (1) and (2) above.

FSAR Subsections - 5.2.3, 4.5.2

Regulatory Guide 1.72, (December 1973)

Spray Pond Plastic Piping

Project Position - Not applicable; Grand Gulf does not have spray ponds.

Regulatory Guide 1.73, (January 1974)

Qualification Tests of Electric Valve Operators Installed Inside the Containment of Nuclear Power Plants

Project Position - Comply.

FSAR Subsection - 3.11.2

Regulatory Guide 1.74, (February 1974)

Quality Assurance Terms and Definitions

Project Position - Complies with this guide with the clarifications as noted in FSAR Section 17.2.

FSAR Section - 17.2

Regulatory Guide 1.75, Rev. 1 (January 1975)

Physical Independence of Electric Systems

Project Position - Grand Gulf complies with R.G. 1.75 with the following exceptions:

(1) Reference: Paragraph C.1 of the Regulatory Guide. The following position is taken with respect to the connection of non-Class 1E loads to the Class 1E distribution system:

a. Grand Gulf provides Class 1E power to these non-Class 1E loads and treats the circuits to them as "associated" as defined in position C.4 of the guide, or as "non-Class 1E" as defined in position C.7 of the guide.

b. Isolation of non-Class 1E loads is provided by either:

1. Tripping the Class 1E feeder breaker to the individual load or load group upon receipt of a LOCA signal and allowing the manual reconnection of the load to the Class 1E power supply and operation of the non-Class 1E load at the discretion of the operator, or

2. Providing two Class 1E circuit breakers in series between the Class 1E power supply and the load, thereby providing redundant overcurrent protection which can be coordinated with the source breaker for the Class 1E power supply and allowing the associated non-Class 1E load to remain energized during and after a LOCA.

A list of specific non-Class 1E loads connected to Class 1E power supplies and isolation methods are provided in Tables 8.3-10 and 8.3-11.

(2) Reference: Paragraph C.3 of the Regulatory Guide. The following position is taken with respect to the method of separation of safety function equipment:

GGNS standards for implementing requirements of Regulatory Guide 1.75 and the GGNS design basis that separation would be accomplished via divisional separation are conservative. Nonconformance with GGNS standards for separation does not necessarily cause violation of Regulatory Guide 1.75 requirements. Safety related equipment of one division can be common (i.e., not separated) with safety related equipment of another division provided that the equipment does not perform the same (i.e., redundant) safety function to which it is common. If divisional separation is not satisfied for redundant safety function equipment, then barriers are required in order to satisfy separation criteria. Assessment of divisional and system interdependencies for safety functions must be evaluated when divisional separation is not satisfied.

FSAR Subsections - 7.1.2.2, 7.1.2.6, 8.3.1.4, and 9.3.1.3

Regulatory Guide 1.76, (April 1974)

Design Basis Tornado for Nuclear Power Plants

Project Position - Grand Gulf complies with Regulatory Guide 1.76 except as indicated below:

Reference: Paragraph C.2 of the Regulatory Guide. The Grand Gulf site is located within Region I as indicated in Figure 1 of the guide. The design basis tornado has a maximum wind speed of 360 mph as required in Table 1 of the guide; however, the rotational speed is 290 mph and the translational speed is 70 mph as discussed in the referenced topical report (Ref. 1).

Reference 1: Bechtel Topical Report BC-TOP-3A, Revision 3.

FSAR Subsection/Section - 3.3.2, 3.8

Regulatory Guide 1.77, (April 1974)

Assumptions Used for Evaluating a
Control Rod Ejection Accident for PWRs

Project Position - Not applicable to BWRs.

Regulatory Guide 1.78, (June 1974)

Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During A Postulated Hazardous Chemical Release

Project Position - Grand Gulf complies with the Regulatory Guide 1.78 with following clarification:

There are currently no assumed accidents, involving hazardous chemicals, that could affect the Control Room in which applicable toxicity limits could be exceeded as specified in UFSAR Section 2.2.3.1.2, "Toxic Chemicals." Therefore, the actions specified in Paragraph C.13 of Regulatory Guide 1.78 will not be taken. Should conditions change for existing analyzed chemicals or for any new chemical that would cause hazardous concentrations within the Control Room, NRC Regulatory Guide 1.78, June 1974, paragraph C.13 requirements would have to be considered.

FSAR Sections - 2.2, 6.4

Regulatory Guide 1.79, Rev. 1 (September 1975)

Preoperational Testing of Emergency Core Cooling Systems for Pressurized Water Reactors

Project Position - Not applicable to BWRs.

Regulatory Guide 1.80, (June 1974)

Preoperational Testing of Instrument air Systems

Project Position - The project complies with the Regulatory Guide 1.80 with the following exceptions.

Paragraph C.8 - A loss-of-instrument-air test will be conducted using the branch header concept. This concept is the most practicable and most manageable and is less disruptive to the normal operating condition of the plant. Pneumatically operated valves which have a safety function are listed in FSAR Tables 6.2-44, 9.3-1, and 9.3-2 and will be the ones monitored during loss of air to the branch headers.

The capacity of individual air accumulators provided for safety system valves will be demonstrated in the performance of the respective system preoperational tests by valve operation using the accumulators.

The acceptance test does not address a check to verify that branches of smaller pipe sizes are not starved by flow to branches of larger capacity as recommended in position C.8. There is no practical procedure to demonstrate this in a meaningful fashion. Additionally, this exception is justified because the instrument air system is not safety related.

Paraqraph C.9 - The test described for this position is similar to the test described in C.8 with the exception that the air supply is shut off slowly to simulate a plug of ice forming in an air header. This test is not addressed in the acceptance test because the test of position C.8 provides the same results with the possible exception of line depressurization rates and test duration.

Paragraph C.10 - This position would repeat the test of C.8 with the exception that the only valves to be demonstrated are those valves normally aligned in the non-failed position. The test is not performed since it only reduces the number of valves already tested by C.8. No different valves are cycled.

Regulatory Guide 1.81, Rev. 1 (January 1975)

Shared Emergency and Shutdown Electric Systems
for Multi-Unit Nuclear Power Plants

Project Position - No longer applies to GGNS due to the cancellation of Unit 2.

FSAR Subsection - 8.3.1

Regulatory Guide 1.82, (May 1996)

Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident

Project Position - Comply.

FSAR Subsections - 6.2.2.2, 6.3.2.2

Regulatory Guide 1.83, Rev. 1 (July 1975)

Inservice Inspection of PWR Steam Generator Tubes

Project Position - Not applicable to BWRs.

Regulatory Guide 1.84, Rev. 15 (May 1979)

Code Case Acceptability - ASME Section III Design and Fabrication

Project Position - The revision of Regulatory Guide 1.84 used for the duration of a purchase order is determined by the component date.

Grand Gulf complies with Regulatory Guide 1.84 with the following exceptions:

(1) Reference: Paragraph C.1 of the Regulatory Guide. It is intended to comply with Position C.1 of Regulatory Guide 1.84 by specifying or permitting only those code cases listed in Regulatory Position C.1 of the guide for the design and fabrication of ASME Section III components. This position applies to ASME Section III components procured after the initial issue of Regulatory Guide 1.84 (July 1, 1974) and with the understanding that application to the commission for acceptance of selected code cases issued after the latest revision of the guide may be made.

(2) Reference: Paragraph C.2 of the Regulatory Guide. It is intended to comply with Position C.2 of Regulatory Guide 1.84 with the understanding that acceptable code cases annulled by action of the ASME Council, but which were specified in procurement of components or in performance specifications prior to annulment, shall remain valid for the duration of that purchase order or that job.

(3) Reference: Paragraph C.3 of the Regulatory Guide. It is intended to comply with Position C.3 of Regulatory Guide 1.84 with the understanding that code cases revised by action of the ASME Council, but which were specified in procurement of components prior to revision, remain valid for the duration of that purchase order.

FSAR Section - Tables 3.9-28 and 5.2-1

Regulatory Guide 1.85, Rev. 15 (May 1979)

Code Case Acceptability - ASME Section III Materials

Project Position - The revision of Regulatory Guide 1.85 used for the duration of a purchase order is determined by the component date.

Grand Gulf complies with R. G. 1.85 with the following exceptions:

(1) Reference: Paragraph C.1 of the Regulatory Guide. It is intended to comply with Position C.1 of Regulatory Guide 1.85 by specifying or permitting only those code cases listed in Regulatory Position C.1 of the guide for materials for ASME Section III components. This position applies to ASME Section III components procured after the initial issue of Regulatory Guide 1.85 (July 1, 1974) and with the understanding that application to the commission for acceptance of selected code cases issued after the latest revision of the guide may be made.

(2) Reference: Paragraph C.2 of the Regulatory Guide. It is intended to comply with Position C.2 of Regulatory Guide 1.85 with the understanding that acceptable code cases annulled by action of the ASME Council, but which were specified in procurement of components or in performance specifications prior to annulment, shall remain valid for the duration of that purchase order or that job.

(3) Reference: Paragraph C.3 of the Regulatory Guide. It is intended to comply with Position C.3 of Regulatory Guide 1.85 with the understanding that code cases revised by action of the ASME Council, but which were specified in procurement of components prior to revision, remain valid for the duration of that purchase order.

(4) Code Cases approved by later revisions of the Regulatory Guide 1.85 including restrictions may be used when identified in the applicable purchase or design specifications.

FSAR Section - Tables 3.9-28 and 5.2-1

Regulatory Guide 1.86, (June 1974)

Termination of Operating Licenses for Nuclear Reactors

Project Position - Comply.

Reference - License Application

Regulatory Guide 1.87, Rev. 1 (June 1975)

Guidance for Construction of Class 1 Components in Elevated-Temperature Reactors

Project Position - Not applicable to BWRs.

Regulatory Guide 1.88, Rev. 2 (October 1976)

Collection, Storage, and Maintenance of Nuclear Power Plant Quality Assurance Records

Project Position - Complies with this guide with clarifications as noted in FSAR Section 17.2.

FSAR Section - 17.2

Regulatory Guide 1.89, (November 1974)

Qualification of Class IE Equipment
for Nuclear Power Plants

Bechtel Position - Grand Gulf complies with Regulatory Guide 1.89 with the following exceptions:

(1) Reference: Paragraph C.2 of the Regulatory Guide. The specific environment for which individual equipment is qualified depends on its installed location; namely, inside the containment or outside the containment. It is unnecessary to qualify equipment to the environmental levels to which the equipment will not be exposed if the equipment performs its intended function.

(2) Reference: Paragraph D of the Regulatory Guide. Only purchase orders executed for Class IE equipment on or after November 15, 1974, require compliance with IEEE 323-1974. Purchase orders executed prior to November 15, 1974, require compliance with IEEE 323-1971 as well as the specific guidance in IEEE 334-1971, 382-1972, and 383-1974 for motors, valve operators, and cables, respectively.

GEH Position - This regulatory guide as well as IEEE Std. 323-1974 was issued after most of the design of the NSSS Class IE components for this plant were complete. However, the NSSS complies in part with these documents. Class IE equipment is designed in accordance with IEEE Std. 323-1971 and in accordance with IEEE Std. 344-1971 as described in subsection 7.1.2.4. A few NSSS Class IE components are qualified to the later editions of IEEE 344 and 323 as stated in the applicable component discussions. The seismic and environmental qualification of specific components is discussed in Sections 3.10 and 3.11, respectively. Table 3.11-2 defines the containment radiation environment under accident conditions. The source terms used in developing Table 3.11-2 are in accordance with Regulatory Guide 1.3, i.e., 100 percent N.G., 25 percent halogens, and 1 percent solids.

The impact of the alternative source term described in 10CFR50.67 and Regulatory Guide 1.183 has been evaluated and confirmed to be within the current design basis.

FSAR Subsections - 3.11, Table 7.1-3, 7.1.2, 8.3.1

Regulatory Guide 1.90, (August 1977)

Inservice Inspection of Prestressed Concrete Containment Structures with Grouted Tendons

Project Position - Not applicable; Grand Gulf has a reinforced concrete containment.

Regulatory Guide 1.92, Rev. 1 (February 1976)

Combining Modal Responses and Spatial Components In Seismic Response Analysis

Project Position (Subsystems Other Than NSSS) - For all safety-related piping systems required to safely shut down the plant during an earthquake, the modal responses are combined as required by this guide, using the grouping method.

(NSS Systems) - When the response spectrum method of modal analysis is used, all modes except the closely spaced modes (i.e., the difference between any two natural frequencies is equal to or less than 10 percent) are combined by the square-root-of-the-sum-of-the-squares (SRSS) method as described in subsection 3.7.3.7.2.1. The treatment of closely spaced modes is described in subsection 3.7.3.7.2.2.

FSAR Subsections - 3.7.3.7.1, 3.7.3.7.2

Regulatory Guide 1.93, Rev. 0 (December 1974)

Availability of Electric Power Sources

Project Position - Grand Gulf complies with Regulatory Guide 1.93 with the following exceptions:

(1) Reference: Paragraph C.1 of the Regulatory Guide. The time limits for continued power operation with one onsite power source unavailable (i.e., one inoperable emergency diesel generator) are established by the Technical Specifications. The Technical Specification allowed out-of-service times (AOTs) may exceed the 72-hour limit recommended by the Regulatory Position.

(2) Reference: Paragraph C of the Regulatory Guide. The operating time limits established by the Technical Specifications are not explicitly for corrective maintenance activities only. The operating time limits may be applied to preventive maintenance or testing activities performed during unit operation.

FSAR Sections - 8.1, 8.3.1

Regulatory Guide 1.94, (April 1976)

Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete and Structural Steel During Construction Phase of Nuclear Power Plants

Project Position - Complies with this guide with clarifications as noted in FSAR Section 17.2.

FSAR Section - 17.2

Regulatory Guide 1.95, Rev. 1 (January 1977)

Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release

Project Position - Grand Gulf complies with Positions C.1 through C.6 with the following clarifications:

(1) Reference: Paragraph C.4.b.3 of the Regulatory Guide. The Grand Gulf control room has no exits leading directly to the outside of the building.

(2) Reference: Paragraph C.5 of the Regulatory Guide. Periodic leakage testing is not planned for the control room.

(3) Reference: Paragraph C.3 of the Regulatory Guide. The control room categories described in Table 1 of the guide are low leakage control rooms and do not envelope the GGNS control room characteristics. The GGNS control room has been analyzed for the effect of a chlorine cylinder rupture considering no isolation and the resulting control room concentration is within acceptable (R.G. 1.78) limits. The GGNS control room is provided with manual isolation capabilities. The analysis demonstrates compliance with the intent of Position C.3 for adequate operator protection.

(4) Reference: Paragraph C.4.d of the Regulatory Guide. The Grand Gulf control room is not equipped with chlorine detectors, therefore there is no automatic isolation of the control room upon detection of chlorine.

FSAR Subsection/Section - 2.2.3, 6.4

Regulatory Guide 1.96, Rev. 1 (June 1976)

Design of Main Steam Isolation Valve Leakage Control Systems for Boiling Water Reactor Nuclear Power Plants

Project Position - Comply.

FSAR Section - 6.7.1

Regulatory Guide 1.97, (December 1980)

Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident

Project Position - Entergy Operations is in compliance with the positions and criteria referenced under Regulatory Guide 1.97, item C "Regulatory Position," with the following clarifications (Ref. 6):

(1) Reference: Paragraph C.1.3.2.a of the Regulatory Guide.

Seismic Qualification of Category 2 Instrumentation - Category 2 instruments which perform a safety-related function will meet the requirements of Regulatory Guide 1.100. Category 2 instruments which are part of a safety-related system for maintaining a pressure boundary will be seismically qualified to ensure pressure boundary integrity during accident conditions if the loss of integrity could jeopardize the operation of the fluid system. The electrical components and control room indication for these variables will not be seismically qualified. Category 2 instruments that are not included in the clarifications discussed above will not be designed to meet the seismic requirements of Regulatory Guide 1.100.

Environmental Qualification of Category 2 Instrumentation - Instruments for Category 2 variables will be specified and designed to remain functional in a postulated accident environment. The qualification of these instruments will depend on the instrument usage. Category 2 instruments which are functionally safety-related (i.e., Class IE) and are located in harsh environments will be qualified in accordance with 10 CFR 50.49. Category 2 instruments which actively form part of the pressure boundary of a safety-related system or whose failure could otherwise affect a safety-related system will be qualified to the extent necessary to ensure that such failures do not occur. No additional environmental qualification will be required for the remainder of the Category 2 instruments.

(2) Reference: Paragraph C.1.3.1.e and C.1.3.2.d of the Regulatory Guide.

Quality Assurance (QA) Requirements - Previously installed Category 1 and 2 instruments as identified in Table 7.5-2 were installed under the existing QA programs in effect at that time, which met the requirements of 10 CFR 50, Appendix B. Instrumentation being installed for Regulatory Guide 1.97 implementation which is required to meet 10 CFR 50 Appendix B will be installed under the QA program existing at the time of installation. GGNS's QA program is contained in the Quality Assurance Program Manual.

Entergy Operations will evaluate Category 2 instrumentation for inclusion in the GGNS Q-List for both safety-related and nonsafety-related application. The GGNS operating QA program will then be applied to the appropriate portions of the Category 2 instrumentation as delineated in the Q-List.

(3) Reference: Paragraph 1.4.b of the Regulatory Guide.

Instrument Identification - The identification of instruments for post-accident monitoring must take into consideration other NUREG-0737 Supplement 1 activities such as detailed control room design review and emergency operating procedures upgrade. By incorporating these activities and Regulatory Guide 1.97 into an integrated project (NUREG-0737 Supplement 1), the NRC has ensured that human factors engineering and Regulatory Guide 1.97 integration is achieved. GGNS will develop a philosophy regarding instrument channel identification as part of the overall Emergency Response Capability program, and it will be implemented as part of the GGNS Detailed Control Room Design Review (DCRDR).

(4) Reference: Paragraph 1.5.a of the Regulatory Guide.

Servicing, Testing, and Calibration - Servicing, testing, and calibration programs of instruments will be established and implemented on a frequency necessary to maintain instrumentation operability.

(5) Instrumentation for Neutron Flux Monitoring complies with NEDO-31558-A, Licensing Topical Report; "Position on NRC Regulatory Guide 1.97, Revision 3 Requirements for Post-Accident Neutron Monitoring System" instead of with Regulatory Guide 1.97.

(6) Exceptions taken for certain specific variables shown in Table 1 of Regulatory Guide 1.97 with the supporting justifications were provided in AECM-85/0059, dated February 28, 1985, AECM-86/0030 dated February 14, 1986, and GNRO-93/00043 dated April 21, 1993. The NRC's acceptance of these exceptions was provided in the NRC Safety Evaluation Report dated January 12, 1987 (MAEC-87/0013) and GNRI-94/00058 dated March 7, 1994. Identification of the GGNS Regulatory Guide 1.97 variables are provided in Table 7.5-2.

FSAR Subsection - 7.5, 18.2.4

Regulatory Guide 1.98, (March 1976)

Assumptions Used for Evaluating the Potential Radiological Consequences of a Radioactive Offgas System Failure in a Boiling Water Reactor

Project Position - Comply, except for Regulatory Position C.2.a of RG 1.98. The position states that: "The release from the SJAE is assumed to occur from a break in the delay line just downstream of the SJAE." The SJARE is assumed to operate for a period of 1 hour after the accident. A delay of 5 minutes stipulated by RG 1.98 to account for stream transit to the break in the delay line is not assumed since the source term is based on the activity at the offgas recombiner effluent which is effectively the break location.

Regulatory Guide 1.99, (May 1988)

Effects of Residual Elements on Predicted
Radiation Damage to Reactor Vessel Materials

Project Position - Comply to extent discussed in referenced sections.

FSAR Subsections - 5.3.1.4, 5.3.1.6

Regulatory Guide 1.100, (March 1976)

Seismic Qualification of Electrical Equipment
For Nuclear Power Plants

Project Position -

Initial qualification of equipment for seismic and hydrodynamic loads was accomplished in accordance with either IEEE Standard 344-1971, as supplemented by BTP EICSB-10 or IEEE Standard 344-1975. Review of all equipment requiring seismic/hydrodynamic qualification was conducted for compliance with IEEE 344-1975 in support of SQRT and compliance with this standard was demonstrated. Qualification efforts conducted during the plant operating license stage comply with IEEE 344-1975. Chapter 3.10 discusses in detail the process of performing seismic qualification for safety-related and post accident monitoring equipment. The Component Data Base (CDB) indicates the components required to be qualified for postulated GGNS seismic/hydrodynamic events. Details of qualification bases are maintained in the SQCF and the CDB as controlled by an auditable GGNS administrative procedure.

FSAR Subsection - 3.10

Regulatory Guide 1.101, Rev. 1 (March 1977)

Emergency Planning for Nuclear Power Plants

Project Position - Comply with the intent of the regulatory guide. See Section 13.3 for details of the Emergency Plan.

Regulatory Guide 1.102, Rev. 1 (September 1976)

Flood Protection for Nuclear Power Plants

Project Position - Comply.

FSAR Section - 3.4

Regulatory Guide 1.103, Rev. 1 (October 1976)

Post-Tensioned Prestressing Systems for Concrete Reactor Vessels and Containments

Project Position - Since Grand Gulf has a reinforced concrete containment and steel reactor vessel, this regulatory guide does not apply.

Regulatory Guide 1.104, (February 1976)

Overhead Crane Handling Systems for Nuclear Power Plants

Regulatory Guide 1.104 was superseded by NUREG 0554. EC-12920 documents the spent fuel cask handling crane meets the requirements of NUREG 0554.

FSAR Section - 9.1

Regulatory Guide 1.106, Rev. 1 (March 1977)

Thermal Overload Protection for
Electric Motors on Motor-Operated Valves

Project Position - Comply.

FSAR Subsections - 7.1.2.6.22, 8.3.1.1

Regulatory Guide 1.107, Rev. 1 (February 1977)

Qualifications for Cement Grouting for Prestressing Tendons in Containment Structures

Project Position - Not applicable since Grand Gulf does not have prestressed tendons in its containment structures.

Regulatory Guide 1.108, (August 1977)

Periodic Testing of Diesel Generators Used as Onsite Electric Power Systems at Nuclear Power Plants

Project Position - Withdrawn

FSAR Subsection - Superseded by Regulatory Guide 1.9, Rev. 3

Regulatory Guide 1.109, Rev. 1 (October 1977)

Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I

Project Position - Comply.

FSAR Sections - 11.2, 11.3

Regulatory Guide 1.111, Rev. 1 (July 1977)

Methods for Estimating Atmospheric Transports and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors

Project Position - Methods comply with Regulatory Guide 1.111 with the following clarification:

> Terrain recirculation factors were not considered for the long-term diffusion analysis. Since there are no pronounced hills to the east nor deep, narrow valleys to the west of the plant site to cause any air flow recirculation or hinder the diffusion rate, and three years of onsite meteorological data show no conclusive or systematic up and down or cross valley flow, it would be inappropriate to apply recirculation factors as indicated in Regulatory Guide 1.111 in long-term diffusion estimates for the plant site.

FSAR Section/Subsection - 11.3, 2.3.5.2.

Regulatory Guide 1.112, (April 1976)

Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light-Water-Cooled Power Reactors

Project Position - Comply.

FSAR Sections - Chapters 11 and 15

Regulatory Guide 1.113, Rev. 1 (April 1977)

Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I

Project Position - In the event of an accidental release of radioactive material at the site, the material would encounter the regional water table beneath the site and would move laterally toward the Mississippi River to the west. The concentration of the radioactive material would be reduced during the migration by the processes of ion exchange, dispersion, and radioactive decay. Upon reaching the Mississippi River edge, the concentration of these materials would have been reduced to well below the limits set in 10 CFR 20 as presented in subsection 2.4.13. Therefore, no dispersion analysis of the diluted radioactive material in the Mississippi River was made.

FSAR Subsection/Sections - 2.4.13, 11.2, 15.7

Regulatory Guide 1.114, Rev. 1 (November 1976)

Guidance on Being Operator at the Controls of a Nuclear Power Plant

Project Position - Comply.

FSAR Subsection -

Regulatory Guide 1.115, Rev. 1 (July 1977)

Protection Against Low-Trajectory Turbine Missiles

Project Position:

Grand Gulf complies with NRC Regulatory Guide 1.115 with the following exception:

The protection of essential systems against the impact of low trajectory, turbine-generated missiles as required by Regulatory Guide 1.115 is provided by a concrete barrier located within the turbine generator building. The barrier was designed using a time-history analysis technique developed from experimental data reported in References 1 and 2.

References:

1. Vassalo, F.A., "Missile Impact of Reinforced Concrete Panels," Calspan Corporation, Report No. HC-5609-D-1, Buffalo, N.Y., 1975.

2. Stephenson, A.E., and Sliter, G.E. "Full-Scale Tornado-Missile Impact Tests," Transactions of the 4th International Conference on Structural Mechanics in Reactor Technology, San Francisco, Ca., 1977.

FSAR Section - 3.5

Regulatory Guide 1.116, (June 1976)

Quality Assurance Requirements for Installation, Inspection, and Testing of Mechanical Equipment and Systems

Project Position - Complies with this guide with clarifications as noted in FSAR Section 17.2.

FSAR Section - 17.2

Regulatory Guide 1.117, (April, 1978)

Tornado Design Classification

Project Position - Section D of the Guide indicates that implementation of this guide is not applicable to Grand Gulf based upon the GGNS docket date. Therefore, Grand Gulf complies with the requirements of the guide only to the extent discussed in the referenced section. The Tornado Missile Risk Evaluator (TMRE) methodology may only be applied to discovered conditions where tornado-missile protection was required by the plant's original licensing basis but was not provided. The TMRE methodology cannot be used either to remove existing tornado-missile protection, or to avoid providing tornado-missile protection when required in support of the plant modification process.

FSAR Subsection - 3.5.2.5, 3.5.3.3

Regulatory Guide 1.121, (August 1976)

Bases for Plugging Degraded PWR Steam Generator Tubes

Project Position - Not applicable to BWRs.

Regulatory Guide 1.122, Rev. 0 (September 1976)

Development of Floor Design Response Spectra
For Seismic Design of Floor-Supported Equipment
or Components

Project Position - Comply.

FSAR Section - 3.7

Regulatory Guide 1.123, Rev. 1 (July 1977)

Quality Assurance Requirements for Control of Procurement of Items and Services for Nuclear Power Plants

Project Position - Complies with this guide with clarifications as noted in FSAR Section 17.2.

FSAR Section - 17.2

Regulatory Guide 1.126, (March 1977)

An Acceptable Model and Related Statistical Methods for the Analysis of Fuel Densification

Project Position - GE meets the requirements of this regulatory guide with the exception of the method of derivation of the densification values from re-sintering tests.

GE feels that the NRC method in the regulatory guide represents a significant departure from previously approved GE methods.

Specifically, this departure requires:

a. an increase in the level of statistical confidence level from 50 percent to 95 percent employed to infer population parameters from the sampling results, and

b. identification and individual characterization of lots (powder types) within the quantity of material of initial interest (initial core or reload).

This departure from the GE approach previously approved by the Staff is interpreted as the introduction of additional conservatism in the GE densification analyses, and such a Staff position is considered inappropriate in recognition of the GE/NRC agreed upon conservatisms incorporated into the individual densification models during their formulation.

FSAR Section - 4.2

Regulatory Guide 1.127, (March 1978)

Inspection of Water-Control Structures Associated with Nuclear Power Plants

Project Position - Inspections performed as discussed in Technical Requirements Manual (TRM)

FSAR Subsection - 2.4.3.5.2

Regulatory Guide 1.129, (April 1977)

Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Nuclear Power Plants

Project Position - Grand Gulf complies with NRC Regulatory Guide 1.129 with the following exception:

The Regulatory Guide endorses IEEE Standard 450-1975, "IEEE Recommended Practice for Maintenance, Testing and Storage of Large Lead Storage Batteries for Generating Stations and Substations" as adequate for complying with the Commissions regulations with respect to the maintenance, testing and replacement of large lead storage batteries. Grand Gulf Technical Specification testing requirements may be in accordance with later editions of IEEE Standard 450.

FSAR Section - 8.3

Regulatory Guide 1.137, (January 1978)

Fuel-Oil Systems For Standby Diesel Generators

Project Position - Comply with the following exceptions and clarifications:

(1) Reference: Paragraph C.1.b of the Regulatory Guide. Refer to Appendix 3A, page 3A/1.28-1.

(2) Reference: Paragraph C.1.e of the Regulatory Guide. Comply except for underground portions of the system.

(3) Reference: Paragraph C.2.a of the Regulatory Guide.

The requirements to ANSI N195-1976, Appendix B as referenced by this position are explained as follows:

Third paragraph of Appendix B, N195-1976.

The fuel oil quality shall be maintained to the limits specified in ASTM D975, Table 1, for No. 2-D grade of diesel fuel oil. Fuel oil contained in the storage tanks not meeting the requirements stated above will be replaced in a short time period (about a week).

Fourth paragraph of Appendix B, N195-1976.

Diesel fuel oil will be analyzed before being transferred into the storage tanks as specified in the explanation to position C.2.b below.

(4) Reference: Paragraphs C.1.f and C.2.b of the Regulatory Guide.

Fuel oil will be sampled by the supplier at the supplier's bulk storage facility and analyzed for properties 1-4. The proper analysis documentation shall be provided at the time of fuel oil delivery with the fuel oil shipment. The receipt inspection of the fuel oil shall include a review of the analysis documentation.

Analysis of the required fuel properties listed in ASTM D975-92a that are not performed by the supplier's laboratory will be completed as soon as possible following the transfer (normally within 2 weeks, except total insolubles testing must be completed within 7 days).

A sample of the fuel will be drawn from the transport vehicle. The sample will be analyzed for water, sediment, and viscosity to provide any indication of contamination of the fuel oil between the bulk storage facility and the Grand Gulf storage location. Also, total insolubles testing is required.

(5) Reference: Paragraph c.2.d of the Regulatory Guide. Comply, with the following exception: in lieu of dewatering the F.O. Storage Tank on the day after new fuel addition, the sampling program ensures that the water content of each new fuel addition is less than .05 volume percent.

(6) Reference Paragraph 6.1 of ANSI N195-1976, last sentence.

The suction of the Day tank is located on the tank bottom. Diesel fuel oil is monitored for water and the Fuel Oil Storage tank filters the fuel before it reaches the Day Tank. Therefore exception is taken for the Day Tank having a suction located above the tank bottom.

(7) An extension to the surveillance is granted by ER 2003-0234-01. Div II maybe inspected as late as March 2006. This does not apply to ASME Section XI pressure test. An extension to the surveillance is granted by EC 64460 for the HPCS (Division III) inspection until December 2016.

(8) Reference: Paragraph c.2.h (1) and c.2.h(3) comply, with the following exceptions:

Paragraph (1): Change 12 to 24 months.
Paragraph (3): Change 2 to 8 months.

FSAR Section - 9.5

Regulatory Guide 1.144, Rev. 1 (September 1980)

Auditing of Quality Assurance Programs
for Nuclear Power Plants

Project Position - Complies with this guide with clarifications as noted in FSAR Section 17.2.

FSAR Section - 17.2

Regulatory Guide 1.146, Rev. 0 (August 1980)

Qualification Of Quality Assurance Program Audit Personnel For Nuclear Power Plants

Project Position - Complies with this guide with clarifications as noted in FSAR Section 17.2.

FSAR Section - 17.2

Regulatory Guide 1.160, Rev. 3 (May 2012)

Monitoring the effectiveness of maintenance at Nuclear Power Plants

Project Position: Complies with this guide.

FSAR Section:

Regulatory Guide 1.174, Rev. 2 (May 2011)

An Approach For Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis

Project Position: Complies with this guide.

FSAR Section:

Regulatory Guide 1.177, Rev. 1 (May 2011)

An approach For Plant-Specific, Risk-Informed Decision Making: Technical Specifications

Project Position: Complies with this guide.

FSAR Section: FSAR Section 16.1.2

Regulatory Guide 1.181, Rev. 0 (September 1999)

Content of the Updated Final Safety Analysis Report in Accordance With 10 CFR 50.71(e)

Project Position: Comply.

FSAR Section: N/A

Regulatory Guide 1.183, Rev. 0 (July 2000)

Alternative Radiological Source Terms For Evaluating Design Basis Accidents at Nuclear Power Reactors

Project Position: Complies with this guide.

FSAR Section: 15.4.9, 15.6.5, 15.7.4

Regulatory Guide 4.15, Rev. 1 (February 1979)

Quality Assurance For Radiological Monitoring Programs (Normal Operations) - Effluent Streams And The Environment

Project Position - Comply with Regulatory Guide for Quality Assurance Program for effluent and environmental monitoring procedures required by Technical Specification 5.4.1.

FSAR Section - 17.2

Regulatory Guide 8.1, (February 2, 1973)

Radiation Symbol

Project Position - Comply.

Regulatory Guide 8.2, (February 2, 1973)

Guide for Administrative Practices in Radiation Monitoring

Project Position - Comply. Grand Gulf Nuclear Station complies with Regulatory Guide 8.2 with the following clarification:

Reference: Paragraph C.2 of Regulatory Guide 8.2. The reference to paragraph 20.401 of 10CFR Part 20 changes to 10CFR20 Subpart L.

FSAR Subsections - 12.3.4.1, 12.3.4.2, 12.5.1

Regulatory Guide 8.3, (February 2, 1973)

Film Badge Performance Criteria

Project Position - This guide does not apply to Grand Gulf; Dosimeters processed in accordance with the National Voluntary Laboratory Accreditation Program (NVLAP) operated by the National Institute of Standards and Technology (Dosimeter of Legal Record) will be used instead of film badges for dose assessment.

Regulatory Guide 8.4 (June 2011 Revision 1)

Personnel Monitoring Device-- Direct Reading Pocket Dosimeters

Project Position - Grand Gulf Nuclear Station complies with Regulatory Guide 8.4.

FSAR Subsection - 12.5.2

Regulatory Guide 8.5, (February 16, 1973)

Immediate Evacuation Signal

Project Position - Grand Gulf Nuclear Station complies with Regulatory Guide 8.5 with the following exception:

Reference: Paragraph 2.5.1 of ANSI Std. N2.3 - 1967. In order to prevent inadvertent actuation of an evacuation signal, the system must be manually initated by the control room operator.

Regulatory Guide 8.6, (May 1973)

Standard Test Procedure for Geiger-Mueller Counters

Project Position - The tests described in ANSI N42.3/IEEE 309-1970 do not provide any information affecting the calibration accuracy of the survey instruments containing Geiger-Mueller counters which are used at GGNS. The calibration techniques used at GGNS, performed in accordance with ANSI N323-1978, provide reasonable assurance of the accuracy of our G-M instruments. Therefore, the GGNS position is not to comply with Regulatory Guide 8.6-1973.

FSAR Subsection - 12.5.2.2.2

Regulatory Guide 8.7, Rev. 2 (June 1992)

Instructions for Recording and Reporting
Occupational Radiation Exposure Data

Project Position - Comply.

FSAR Subsection - 12.5.1

Regulatory Guide 8.8, Rev. 2 (March 1977)

Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be as Low as Is Reasonably Achievable

Project Position - See subsection 12.1.1.3 for a discussion on Regulatory Guide 8.8.

FSAR Subsections - 12.1.1.3, 12.3.4.2.4

Regulatory Guide 8.9, Revision 1 (July 1993)

Acceptable Concepts, Models, Equations, and Assumptions for a Bioassay Program

Project Position - Comply.

FSAR Subsection - 12.5.1

Regulatory Guide 8.10, (September 1975)

Operating Philosophy for Maintaining Occupational Radiation Exposures as Low as Is Reasonably Achievable

Project Position - Comply.

FSAR Subsections - 12.1.3, 12.5.1

Regulatory Guide 8.11, (June 1974)

Applications of Bioassay for Uranium

Project Position - This guide does not apply to Grand Gulf.

Regulatory Guide 8.12, (December 1974)

Criticality Accident Alarm Systems

Project Position: Guide withdrawn. Criticality accident requirements governed by 10 CFR 50.68

FSAR Subsection: 12.3.4.1

Regulatory Guide 8.13, (June 1999)

Instruction Concerning Prenatal Radiation Exposure

Project Position - Comply.

FSAR Subsection - 12.5.1

Regulatory Guide 8.14, (June 1976)

Personnel Neutron Dosimeters

Project Position - Grand Gulf does not use a separate dosimeter for monitoring neutron exposure. The dosimetry system used monitors all types of radiation to which personnel may be exposed at Grand Gulf, and meets the requirements for accreditation by NVLAP as specified in 10CFR20.202. Therefore, the project position is not to comply with this Reg. Guide.

Regulatory Guide 8.15, (October 1976)

Acceptable Programs for Respiratory Protection

Project Position - Complies with this guide with the following clarification(s):

The guidance of Regulatory Guide 8.15 shall be followed except:

1. As it pertains to the use of contact lenses. Use of contact lenses is permitted with respirator wear provided the individual has previously demonstrated that he or she has had successful experience wearing contact lenses. The contact lens wearer shall be required to have practice wearing the respirator while wearing the contact lenses.

2. As it pertains to the selection of respiratory equipment stated in Section C.2.S1 & 2. Respiratory equipment that provides a protection factor less than the multiple by which peak concentrations of radioactive materials are expected to exceed the values specified in 10CFR20, Appendix B (to paragraphs 20.1001 - 20.2401) Table 1 may be used if their use maintains the TEDE ALARA.

FSAR Subsections - 12.5.1, 12.5.2.2.4, 12.5.3.5

Regulatorv Guide 8.34, (July 1992)

Monitoring Criteria and Methods to Calculate
Occupational Radiation Doses

Proiect Position - Comply

FSAR Subsection - 12.5

Regulatory Guide 8.35, (June 1992)

Planned Special Exposures

Proiect Position - Comply

FSAIX Subsection - 12.5

Regulatorv Guide 8.36, (July 1992)

Radiation Dose to the Embryo/Fetus

Project Position - Comply

FSAR Subsection - 12.5

APPENDIX 3B PROCEDURE FOR COMPUTING THE PROBABILITY FACTOR (P_2) OF A MISSILE STRIKE DUE TO A TURBINE FAILURE

3B.1 INTRODUCTION

This appendix describes the procedure used in computing the probability factor (P_2) of a missile strike due to a turbine failure.

The system of descriptive coordinates shown in Figure 3B-1 is used to formulate the turbine missile trajectories. From inspection of these coordinates, it is apparent that a successful combination of the angles α, β, θ, and H must occur for a missile to be on target.

The range of the angle β usually is specified by the turbine manufacturer. For computational convenience, the value of β used is measured from the horizontal plane of the turbine shaft. Thus, the -7 to +19 degree range is used as a 71 to 97 degree range.

The range of the angle α is the full 360 degree circle of wheel rotation. For computational convenience, the range of α is limited to a single quadrant (i.e., 0 to 90 degrees) to avoid negative values of the cosine function. Limiting the range of the angle to be considered at one time does not decrease the probability range, but instead means that the trajectories under consideration at any one time fall within a single quadrant.

The angle θ, which describes the orientation of the target in the horizontal plane with respect to the vertical plane through the turbine shaft, also is limited to a single quadrant. Thus, only angles from 0 to 90 degrees are considered at any one time.

The angle of departure H required to strike a specific target is a function of the departure velocity V of the missile and the distance L of the target from the wheel.

3B.2 FORMULATION OF PROCEDURE

3B.2.1 Trajectory of the Missile

It is assumed that the missile traverses a trajectory described by the gravity parabola,

$$Y = (\tan H)L - g(L^2/2V^2\cos^2 H) \qquad \text{(Eq. 3B-1)}$$

where

Y = elevation above (+) or below (-) the departure point

H = vertical plane angle of departure

L = horizontal distance from point of departure

V = translational velocity of departure

g = gravitational acceleration (32.2 ft/sec/sec)

Only gravitational forces are considered to be acting on the missile from the point of departure onward. While air drag doubtlessly will decrease the impactive velocity, especially on distant targets, numerous computations have shown that it has a negligible effect on probabilities. For this reason, air drag on the missile is not considered.

3B.2.2 Relationships of the Four Angles (α, β, θ, and H)

The mathematical relationships among the four descriptive angles α, β, θ, and H are,

$$\tan H = \frac{\tan \beta \cos \alpha}{\sqrt{1 + \tan^2 \beta \sin^2 \alpha}} \text{ and} \qquad \text{(Eq. 3B-2)}$$

$$\tan \theta = \tan \beta \sin \alpha \qquad \text{(Eq. 3B-3)}$$

For convenience in the computation of the limits for the combination of angles, the expression for β, derived from Eq. 3B-2, is

$$\tan^2\beta = \frac{\tan^2 H}{(1-\sin^2 \alpha\ (1+\tan^2 H))} \qquad \text{(Eq. 3B-4)}$$

These mathematical expressions are combined with the gravitational parabola (Eq. 3B-1) to determine the probability of all of them occurring simultaneously to produce a missile on target.

3B.2.3 Computational Sequence

The steps of the computational procedure are performed in the following sequence:

a. Input

   1. Minimum and maximum distances of the target area, L_1 and L_2, respectively

   2. Minimum and maximum values of angle θ for the horizontal circular sector that includes the target area (θ_1 and θ_2, respectively)

   3. Size of the target area

   4. Minimum and maximum values of departure velocity (V_1 and V_2, respectively)

b. Determination of the limits (H_1 and H_2) of the departure angle for the conditions

   $L_1 < L < L_2$ and $V_1 < V < V_2$

c. Determination of the limits of the angles α and β

   1. One limit from Eq. 3B-4

   2. Two limits from Eq. 3B-3

d. Double summation of α and β over the ranges of α and β that jointly satisfy Eqs. 3B-2 and 3B-3

e. Adjustment of the summation result to account for the conditional probability of distance due to the velocity of departure. This is accomplished by introduction of a velocity range factor (VRF)

f. Adjustment of the results, which includes all possible angles, to account for the probability of occurrence of those angles (i.e., 26 degrees for β and 360 degrees for α)

g. Adjustment of the probability that a missile will strike in the sector to the probability that it will strike on target. This adjustment is accomplished through introduction of the area factor, A_f, where A_f = (target area)/(sector area)

The final result of this computational procedure is the probability factor P_2 that a missile will be on target in the target area under consideration.

3B.3 DETAILED COMPUTATIONAL PROCEDURE

3B.3.1 Input Data

The physical arrangement data are determined readily from a scale drawing. For determination of the sector under consideration, it is sufficient to measure the minimum and maximum distances to the target area to the nearest foot, and the minimum and maximum values of θ are measured either to the left or to the right of the turbine shaft. But as indicated previously, it is computationally inconvenient to include angles both to the left and to the right of the turbine shaft in the same fore, each quadrant about the shaft-wheel origin is treated separately. The net area of the target is determined either from a layout drawing or from a construction drawing.

The range of maximum missile velocities is specified by the manufacturer of the turbine, which is approximately from 332 to 488 feet per second. It should be noted that the weights of missiles are not considered, since the missile energy is not used in the computation of the probability factor P_2.

3B.3.2 Determination of Departure Angle Limits

Setting the gravitational parabola equation (Eq. 3B-1) equal to zero produces:

$$\sin 2H = gL/V^2 \qquad \text{(Eq. 3B-5)}$$

And considering the velocity and distance limits, the limits of angle H are:

$$\sin 2H = gL_2/V_1^2 \qquad \text{(Eq. 3B-6a)}$$

and

$$\sin 2H = gL_1/V_2^2 \qquad \text{(Eq. 3B-6b)}$$

Each of these equations produces two positive values of sin 2H. One value is for a high trajectory parabola and the other is for a low trajectory parabola. The high trajectory parabolas have values of H from 45 to 90 degrees, while the low trajectory parabolas have values of H from 0 to 45 degrees.

A tremendous amount of work can be saved by treating the two types of parabolas separately, since the probability is very remote that a missile traversing a low trajectory parabola will hit a target at an elevation below the turbine wheel. Because of this, many targets are excluded from the low trajectory computations on the basis of probability. Stated differently, many targets will be found to have zero probability for the low trajectory simply by inspection of the equation of the parabola. The inspection proceeds as follows:

For targets below wheel elevation, the negative values of Y from Eq. 3B-1 are of interest. The most negative values for a given distance L are attained when H = 0 and Eq. 3B-1 becomes,

$$Y = -gL^2/2V^2 \qquad \text{(Eq. 3B-7)}$$

By choosing the maximum distance L_2 and the minimum velocity V_1, the most negative elevation for a target situation is attained. If the target lies below the value Y computed (i.e., if the difference in elevation between the wheel and the target is greater than the value of Y calculated from Eq. 3B-7), a missile on a low trajectory cannot strike the target, and the low trajectory is excluded from consideration. Further, because of the usual arrangement of installing condensers below the turbine, the shaft elevation is usually much higher than other structures.

There are other cases, called "direct hits" (see Section 3B.4), where the Y values of the target have a range extending both above and below zero elevation. These must be treated as special cases of low trajectory, and in actual layouts these cases are avoided as they engender high probabilities.

3B.3.3 Determination of the Limits of α and β

From inspection of the coordinate system used, the limits of α and β with respect to H are

$$H_{min} < \beta < 90°$$

$$0° < \alpha < 90° - H_{min}$$

Further restriction of these ranges is feasible by refining the definitions to include the relationship of α, β, and θ as given in Eq. 3B-3 below:

$$\tan \theta = \tan \beta \sin \alpha \qquad \text{(Eq. 3B-3)}$$

By rearrangement and selection of the variables to deduce β_{min}, equation 3B-3 becomes:

$$\tan \beta_{min} = \frac{\tan \theta_{min}}{\sin(90 - H_{min})}$$

If the value of β_{min} is less than H_{min}, β_{min} may be substituted as the lower limit of β in the definition above, and the range of is reduced accordingly.

By similar rearrangement and selection, the value of β_{max} may be deduced using:

$$\tan \beta_{max} = \frac{\tan \theta_{max}}{\sin(90 - H_{max})}$$

It should be noted that when $\alpha = 0$, $\beta_{max} = H_{max}$.

The larger of these two values of β_{max} may then be substituted for the 90 degree upper limit for β, thereby reducing the range of β which must be considered.

When the value of β_{max} has been determined, the value of α_{min} may be determined from:

$$\sin \alpha_{min} = \frac{\tan \theta_{min}}{\tan \beta_{max}}$$

The value of α_{max} may be determined by rearrangement and substitution in Eq. 3B-2 so that:

$$\sin \alpha_{max} = \sqrt{\frac{1-(\tan^2 H_{min} \tan^2 \beta_{max})}{1+\tan^2 H_{min}}}$$

3B.3.4 Double Summation Over the Ranges of α and β

To determine the total angular space that must be considered as a source of possible on-target missiles, it is necessary to calculate the double summation of the intervals of and that jointly satisfy the conditions of Eqs. 3B-2 and 3B-3. This summation is represented by:

$$D = \sum_{\beta_2}^{\beta_1} \sum_{\alpha_2}^{\alpha_1} f(\alpha, \beta) d\alpha d\beta$$

This double summation satisfies Eqs. 3B-2 and 3B-3 when $H_1 < H < H_2$ and $\theta_1 < \theta < \theta_2$.

Because the conditions of the double summation do not constrain the velocity, the results represent a situation in which all velocities between V_1 and V_2 may occur simultaneously. Since only one velocity can actually occur for any given case, the result of the double summation must be adjusted accordingly.

3B.3.5 Velocity Range Factor (VRF)

The ranges of missile departure velocity and target distance are specified as:

$$V_1 < V < V_2 \text{ and } L_1 < L < L_2$$

The limits of the angle of departure that satisfy the conditions are calculated by:

$$\sin 2H_1 = gL_2 V^2_1$$

and

$$\sin 2H_2 = gL_1 V^2_2$$

These values represent the limits of the angle of departure, outside of which it is not possible for the missile to hit the target for any velocity between V_1 and V_2. However, this does not mean that between the angles H_1 and H_2 all velocities between V_1 and V_2 will produce missiles on target. In fact, for angle H_1, any velocity greater than V_1 will overshoot the target, and for angle H_2, any velocity less than V_2 will undershoot the target. This situation gives rise to what is known as "the conditional probability of the departure velocity."

To solve this problem, the velocity range may be divided into groups on the basis of the kinetic energy of the missiles. Each representative velocity V_n of each group produces an on-target missile within a specific range of angle H, which must be determined for each V_n. The cumulative range of all velocities is then divided by n, the number of the velocity group used, to obtain the range of the conditional angle of departure. This conditional range is in turn divided by the total angle of departure range H_2 - H_1 to obtain the velocity range factor (VRF).

The computational procedure can be expressed as:

$$M = \frac{\int_{H_1, L_2}^{H_2, L_1} V_n dH}{n}$$

then, $$VRF = \frac{M}{H_2 - H_1}$$

The relationship between L and H is evaluated by applying the gravitational parabola equation,

$$L = \frac{(\tan H)\ 2V^2\cos^2 H}{g}$$

The calculated velocity range factor is applied to the probability of a successful combination of the angles α, β, θ, and H when H is taken between H_1 and H_2.

3B.3.6 Adjustment for the Probability of Occurrence of Possible Angles

The range of occurrence of the angles α and β is not the same for all wheels and manufacturers and so must be determined accordingly. For example, the range of occurrence of β may be from 71 to 97 degrees, a total range of 26 degrees. To adjust the double integration to account for this range of β, the integration result must be divided by 0.454 radians. Similarly, the integration result must be divided by 2π to account for the total range of α. This normalization operation is expressed mathematically as:

$$P_s = \frac{D(VRF)}{2\pi(0.454)}$$

where P_s = the probability of a missile strike in the sector bracketing the target area

3B.3.7 Adjustment for Actual Target Area

The probability P_s that the missile will strike in the bracketing sector is not the same as the probability that it will strike the target, due to the difference in the two areas. Adjustment of the probability P_s to account for this difference is accomplished most readily by dividing the target area by the sector area. The division produces the area adjustment factor,

A_f = area of target/area of sector

The final probability of a missile strike on target, P_2, may be stated by the cumulative expression:

$$P_2 = \frac{D(VRF)A_f}{2\pi(0.454)}$$

3B.4 DIRECT HIT - A SPECIAL CASE OF THE LOW TRAJECTORY

3B.4.1 Computational Procedure

A direct hit is an instance when a missile strikes a vertical surface. This is a special case of the low trajectory parabola wherein values for the angle of departure may not be computed by setting the gravitational parabola equation equal to zero.

For this case, there is only one distance L from the wheel to the vertical surface, and the limits of the angle of departure H must be evaluated from the gravitational parabola,

$$Y = (\tan H)L - g(L^2/2V^2\cos^2 H)$$

by solving for H at the limits of Y_1 and Y_2 of the vertical surface. These two values of Y are respectively, the relative elevations of the bottom and top of the vertical surface with respect to the elevation of the turbine wheel.

Evaluation for the limits of H may be simplified by taking cos H = 1, which is the limit of the low trajectory parabola, to produce:

$$\tan H = \frac{(gL^2/2V^2) + Y}{L}$$

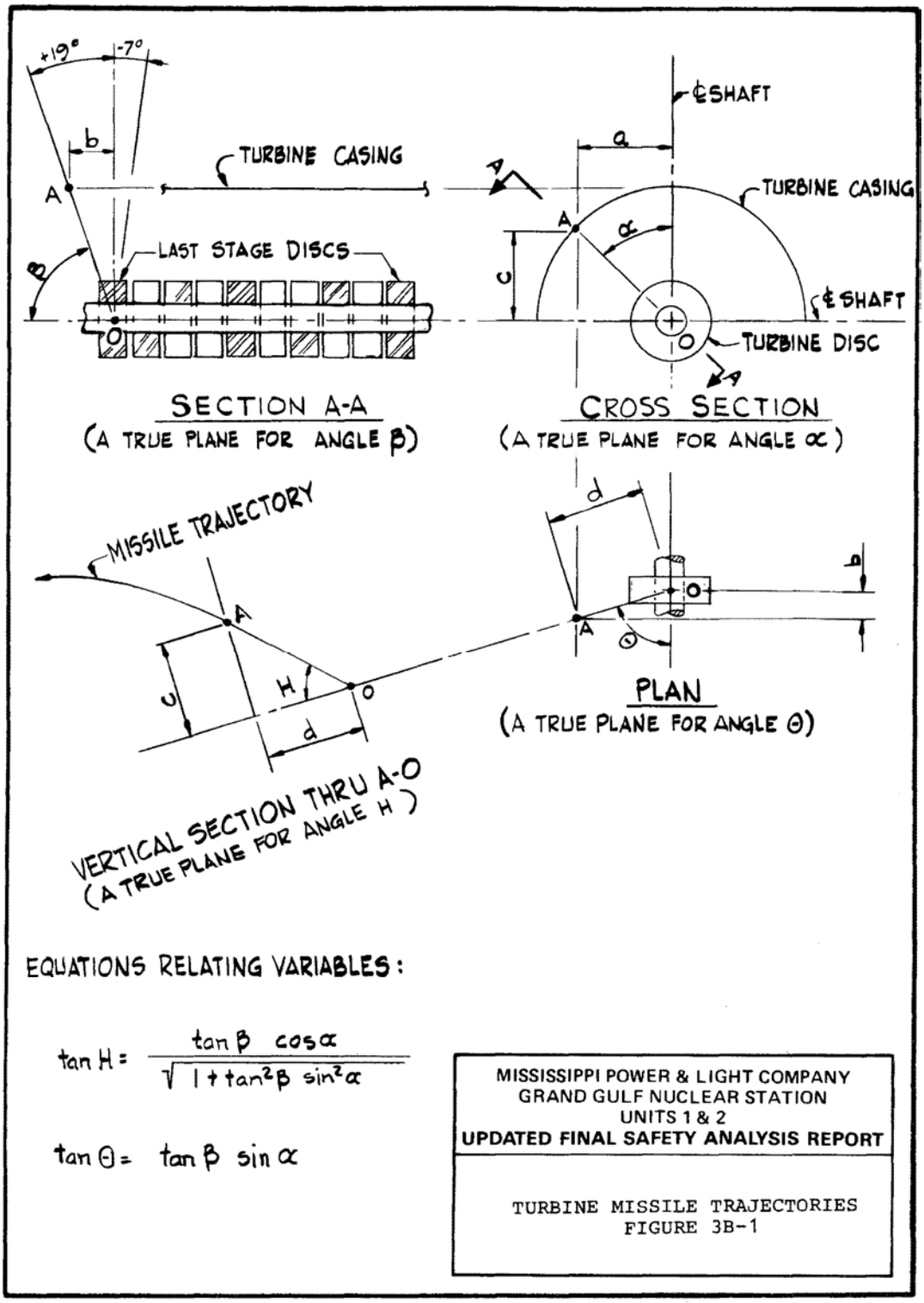
The limits of H are then obtained by solving for H at V_{max} and V_{min}.

The remainder of the procedure is similar to that in Section 3B.3, with the following modifications:

a. Double summation of the ranges of α and β is not applicable.

b. The velocity range factor VRF is computed for the single range value L between the limits Y_1 and Y_2.

c. The area factor is not applicable.


+19°
-7°
b
TURBINE CASING
A
LAST STAGE DISCS
β
O
SECTION A-A
(A TRUE PLANE FOR ANGLE β)
℄SHAFT
a
TURBINE CASING
A
α
c
℄SHAFT
O
TURBINE DISC
CROSS SECTION
(A TRUE PLANE FOR ANGLE α)
MISSILE TRAJECTORY
A
c
H
O
d
VERTICAL SECTION THRU A-O
(A TRUE PLANE FOR ANGLE H)
d
b
O
A
θ
PLAN
(A TRUE PLANE FOR ANGLE θ)
EQUATIONS RELATING VARIABLES:
$\tan H = \frac{\tan\beta \cos\alpha}{\sqrt{1+\tan^2\beta \sin^2\alpha}}$
$\tan\theta = \tan\beta \sin\alpha$
MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT
TURBINE MISSILE TRAJECTORIES
FIGURE 3B-1

APPENDIX 3C FAILURE MODE ANALYSIS FOR PIPE BREAKS AND CRACKS

3C.1 GENERAL

This appendix describes the specific pipe failure protection provided to satisfy the requirements of subsection 3.6A.1 and demonstrates that the essential systems, components, and equipment are not adversely affected by pipe breaks or cracks.

The information is divided into three parts: 1) a discussion of high-energy pipe breaks and the effects of whip and jet impingement; 2) a discussion of moderate-energy pipe cracks and the effects of spraying; and 3) a discussion of flooding as a result of breaks or cracks. Subcompartment pressurization is discussed in detail in subsection 6.2.1.2 (for inside the containment) and in Appendix 3E (for outside the containment).

This appendix does not address the specific protection of field-routed essential instrument tubing or electrical conduit. However, these items are protected in accordance with the requirements of subsection 3.6A.1.

For a detailed discussion of break/crack locations and types, break exclusion areas (no break zones), guard pipes, and whip restraints which are frequently mentioned in this appendix, refer to subsection 3.6A.2.

3C.2 HIGH-ENERGY PIPING

The following systems are described in the noted subsection:

SYSTEM	SUBSECTION
Main Steam	3C.2.1
Feedwater (including RWCU and RHR Injection Piping)	3C.2.2
Reactor Recirculation and RHR Suction	3C.2.3
Steam to RCIC and RHR	3C.2.4
Reactor Water Cleanup(RWCU)	3C.2.6
LPCS/HPCS	3C.2.7
LPCI Mode of RHR	3C.2.8
3 Inch and Smaller High-Energy Piping	3C.2.9

Each section will reference appropriate isometric drawings with break location and restraints. In addition, composite drawings showing pipe/equipment/room configurations have been provided, but are not specifically referenced. Figures 3.6A-14 through 3.6A-22 show the containment and Figures 3.6A-23 through 3.6A-30 show pertinent portions of the auxiliary building.

The only pipe breaks of concern in the nonseismic Category I turbine building were those having a potential to impact safety-related equipment in adjacent buildings. Although the reactor protection SCRAM sensors for the turbine stop and control valves are located in the turbine building, they are not considered essential in the evaluation of low probability events, such as pipe breaks. Other SCRAM signals, not related to the turbine building, are available as backup. Refer to the responses to IEN 79-22 (Refs. 1-3) for an evaluation of the consequences of pipe breaks in the turbine building.

3C.2.1 Main Steam Piping

The locations of postulated pipe breaks and of pipe whip restraints are shown on Figures 3.6A-1A, 3.6A-1B, and 3.6A-2. The results of the associated stress calculations are provided in the documents referenced on the figures. The referenced documents are maintained in accordance with engineering administrative procedures.

General

Each of the four 28-inch main steam lines is welded to the appropriate reactor nozzle (El. 176-10, slightly above the top of the shield wall) after an elbow, and continues downward to approximately El. 151, and then horizontally to the drywell and containment penetrations, through the auxiliary building steam tunnel, and out into the turbine building. The portion of the line from the inner isolation valve (F022) through the drywell wall, the annular area of the containment above the suppression pool (the containment steam tunnel), and through the containment wall to the second isolation valve is fully enclosed in a guard pipe.

The no-break zone, as described in subsection 3.6A.2.1.c.2, starts inboard of the moment-limiting restraint adjacent to the inboard isolation valve and extends just beyond the east wall of the auxiliary building steam tunnel which contains the outboard moment-limiting restraint. The piping from the reactor vessel

through the second isolation valve satisfies all requirements of ASME Code Section III, Class 1 (NRC Quality Group A). After the second isolation valve, the piping is Class 2 (Quality Group B) through the steam tunnel east wall, after which it complies with ANSI Std. B31.1 (NRC Quality Group D).

A total of 20 safety/relief valves are mounted on the horizontal runs between the reactor and the first isolation valves inside the drywell. The discharge piping from these valves is normally unpressurized; therefore, there is no potential for pipe whip or similar hazards.

In addition, a 10-inch line branching from main steam line A supplies steam to the RCIC turbine. This line, which passes through the steam tunnels, is discussed in the analysis of the RCIC system subsection 3C.2.4.

Inside Containment

Conceivably, an unrestrained whipping main steam pipe could impact several items, some of which would be useful in mitigating the effects of such a break. For example, the 10-inch RCIC pipe, 12-inch LPCI pipe, the 14-inch LPCS pipe, the 14-inch HPCS pipe, possibly the CRD hydraulic system piping, and the relief/ADS valve system could be damaged. However, several of these items are not essential. The ruptured steam line would rapidly depressurize the reactor (as discussed in Chapter 15), so that the RCIC, HPCS, and ADS systems would be unnecessary.

To preclude any likelihood of loss of a system required for a safe plant shutdown, a total of 26 restraints and guides have been installed inside the drywell.

The 12-inch LPCI lines which enter the reactor at El. 157-9 are protected by restraints PBR-MS-2B, 2C, 3B, and 3C. The 14-inch LPCS line is protected by restraints PBR-MS-2C, 3C, and 4C. The CRD hydraulic piping necessary for SCRAM is protected by ten restraints (PBR-MS-2A, 3A, 2B, 3B, 4B, 2C, 3C, 4C, 2D, and 3D). The 12-inch feedwater piping is protected by restraints PBR-MS-2A and 2D. The 14-inch HPCS piping is protected by restraints PBR-MS-2B, 3B, and 4B. Additionally, a close gap restraint adjacent to each inboard isolation valve protects the valve from excessive loads caused by a pipe rupture within the drywell.

The 26 restraints and guides, spatial separation, and intervening barriers such as feedwater piping and the grating at El. 147 provide ample protection of the CRD piping from direct jet impingement which could impair SCRAM operation.

The environmental conditions are the same as, or less severe than, the LOCA environment in the drywell for which all essential equipment in the drywell has been qualified (refer to Section 3.11 for environmental qualification).

Inside the Steam Tunnels

The main steam piping, from the moment-limiting restraint inboard of the first isolation valve (inside the drywell) through the moment-limiting restraint at the east wall of the auxiliary building steam tunnel, has been qualified as a no-break zone.

However, to comply with the agreement reached during the "Grand Gulf Auxiliary Steam Tunnel Design" meeting, March 17, 1976 (Ref. 2, subsection 3.6A.2.6), a nonmechanistic slot crack with equivalent cross-sectional area of one main steam line has been evaluated. As a result of that evaluation the steam tunnel has been modified with blowout panels to allow sufficient venting. The results of the pressurization analyses are presented in Appendix 3E. Both the compartment boundaries and the essential components inside are qualified for the calculated accident environment. The blowdown will terminate at 5.5 seconds upon main steam isolation valve closure. The potential for flooding is prevented by isolation valve closure and floor drainage.

In the Turbine Building

After the no-break zone, the piping is designed and constructed in accordance with ANSI B31.1; consequently, breaks have been postulated at each fitting. Because there is no essential equipment in the turbine building, the only area of concern is along the auxiliary building wall. In this area, the four main steam lines run northward, next to the wall, and then turn east. This wall is actually a double wall: the two-foot-thick auxiliary building wall, a small space, and the two-foot-thick turbine building wall. Calculations showed that, with the conservative assumptions used, the whipping pipe could damage the turbine building wall and strike the auxiliary building wall behind it. For conservatism, no contact with the auxiliary building wall is permitted; therefore, a bumper-type restraint has been added for each of the four steam pipes near the northeast corner of the

auxiliary building to prevent pipe contact with the turbine building wall. The pipe configuration near the wall penetration is such that no protection is necessary. Jet impingement against the turbine wall causes negligible damage to the wall.

Conclusions

Using very conservative assumptions and criteria, no postulated failure of the main steam lines will cause additional damage which would impair the ability to safely shut down the reactor, or which would increase the offsite radiation effects beyond the limits of 10 CFR 100.

3C.2.2 Feedwater Piping (Including RHR and RWCU Injection Piping)

The piping, break locations, and restraints are shown in Figures 3.6A-5A, 3.6A-5B and 3.6A-6. The stress analyses used to determine the break locations are provided in the documents referenced on these figures. The referenced documents are maintained in accordance with engineering administrative procedures.

General

From the turbine building, each of the two 24-inch feedwater lines enters the auxiliary building steam tunnel, on through the containment steam tunnel enclosed in guard pipes, into the drywell at El. 143-1-1/2 where they divide into three 12-inch risers and pass through the RPV shield to the RPV nozzles at El. 163-11-1/4. In the auxiliary building tunnel, the 6-inch reactor water cleanup system (RWCU) return line joins the 12-inch RHR shutdown cooling injection piping which in turn joins the feedwater lines.

The no-break zone starts at the RPV side of the moment-limiting restraint, includes the inner isolation valve, extends through the guard pipe and the auxiliary building steam tunnel, and ends just beyond the moment-limiting restraint located in the east wall of the auxiliary building. For the 12-inch RHR branch, the no-break zone ends just below the restraint under the penetration through the tunnel floor. For the RWCU section, the no-break zone starts at the inlet to the inner isolation valve F040, includes the containment penetration flued head and the outer isolation valve F039, and continues to the connection to the RHR line to the feedwater piping.

Inside the Drywell

A circumferential break was postulated at each feedwater connection to the reactor nozzles. Conceivably, the outward movement of the severed pipe could endanger other high-energy piping in the vicinity such as the RCIC piping and possibly some ECCS piping or the main steam relief valves. To ensure that no unacceptable damage could result, a restraint was installed on each riser, near El. 161-5-1/4, attached to the RPV shield wall. Jet impingement and spray localized in the vicinity of the RPV nozzles pose no safety hazard. The pressurization of the annulus between the RPV and its shield wall is discussed in subsection 6.2.1.2.

The postulated longitudinal and circumferental breaks at the tees resulted in the installation of restraints to ensure the safety of the 12-inch recirculation system risers, the 10-inch RCIC steam line, the ECCS piping, the control rod drive piping, and the main steam relief valves.

A close gap whip restraint is installed at the beginning of the no-break zone near the RPV end of valves F011A and F011B. These restraints prevent excessive torsional and bending moments, resulting from a pipe failure between the restraint and the RPV, from causing unacceptable stresses or strains in the isolation valve or flued penetration.

The CRD piping has been analyzed for jet impingement loads caused by feedwater line breaks and has been shown to remain functional.

All equipment inside the drywell whose operation during or after a LOCA is required for safe shutdown is qualified for the post-LOCA drywell environment as discussed in Section 3.11. The environment from the postulated feedwater line breaks will be no worse than that following the design basis LOCA; consequently, no essential equipment will be disabled by such breaks.

Containment Steam Tunnel

All the feedwater piping inside the containment steam tunnel is within the previously mentioned guard pipes. The RHR piping is not in this tunnel. The RWCU piping in this tunnel is mostly nonnuclear and, consequently, a failure is postulated to occur at any fitting as discussed in detail in subsection 3C.2.6. The end of the no-break zone of the RWCU piping is in this tunnel, continuing into the auxiliary building steam tunnel. No moment-

limiting device is required to protect the flued penetration, as the flued head is conservatively designed to withstand the effects of postulated breaks. Should the outer RWCU isolation valve (F039) be assumed to fail (single failure), the outer feedwater isolation valves (F065A and F065B) would ensure isolation.

Outside Containment

For the feedwater piping, the no-break zone continues through the auxiliary building and the restraints located in the east wall of the auxiliary building. However, a nonmechanistic crack in the main steam line has been evaluated for design of the compartment and components within. Because the main steam crack is more severe than a feedwater crack, no evaluation of the feedwater crack is provided. Beyond these restraints (now in the turbine building), the piping changes to nonnuclear; consequently, breaks are postulated at any fitting. There is no essential equipment in the turbine buildings. The only safety concern was for the integrity of the east wall of the auxiliary building. Conservative analysis indicated that the end of the pipe near the north corner of the auxiliary building could penetrate the turbine building wall and contact the auxiliary building wall, with the possibility of damaging that wall. To eliminate the possibility of damaging the auxiliary building wall, bumpers were installed to prevent the feedwater pipe from striking the turbine building wall. Because of the pipe geometry, the south end of the piping needs no restraint.

All the RWCU piping within the auxiliary building steam tunnel is classified as no-break zone piping, from the restraint in the containment building steam tunnel through the attachment to the RHR piping.

The RHR injection piping within the auxiliary building steam tunnel through the tunnel floor is classified as no-break piping. The pipe continues beyond the restraint, through a normally closed check valve which terminates the high-energy portion of this piping. Circumferential breaks were postulated at the end of the no-break zone and at the terminal end (check valve outlet). Also, because an anchor point exists in the horizontal run below the floor, that anchor constitutes another terminal-end-break location. In addition, in the short line between this anchor point and the check valve, intermediate breaks were postulated at both ends of the elbow.

Equipment in the vicinity of this RHR piping includes the 20-inch RHR suction line from the recirculation system used during the shutdown cooling mode. Although the 20-inch piping itself is too large to be seriously damaged by a whipping 12-inch pipe, it includes two motor-operated valves whose operability would be threatened. Consequently, restraints have been installed to prevent possible damage. A postulated pipe failure in either loop A or loop B of the RHR system would not prevent shutdown, because the other loop would be available. According to paragraph B.3.b(3) of BTP APCSB 3-1, additional single failure need not be postulated in the redundant train of the RHR system.

Other essential equipment in the general area of this RHR line, such as electrical circuitry for the RHR loop equipment, is not close enough to the jet from the failed pipe to be of concern.

Conclusions

Because of spatial separation and the installation of pipe-whip restraints, no postulated failure of this piping will prevent the safe shutdown of the plant. Jet impingement has been evaluated and, because of distance and the restrained movement of the piping, no unacceptable damage is foreseen. All cable trays and conduits required for the shutdown will be usable.

3C.2.3 Reactor Recirculation System and RHR Suction

For the recirculation system, the locations of the postulated breaks and restraints are shown on Figure 3.6B-4a. The results of the associated stress calculations are provided in the documents referenced on this figure. The referenced documents are maintained in accordance with engineering administrative procedures. Comparable information for the RHR suction is on Figure 3.6A-9.

General

Each of the two reactor recirculation loops leaves the RPV at El. 137-2-1/4 as a 24-inch line, drops vertically to El. 103-5-¼, turns horizontally through suction isolation valve F023A (or F023B for loop B), and turns up into the pump suction port. The 24-inch pump discharge line runs horizontally at El. 107-4-¼ through flow control valve F060A (or F060B) and isolation valve F067A (or F067B) and turns up to El. 129-9-¼ where it joins the C-shaped 16-inch horizontal header. From this header, six 12-inch risers go up and enter the RPV at El. 137-9-¼. In addition, from

loop B only, a 20-inch RHR suction line branches off from the vertical run between the reactor outlet and valve F023B at El. 112-8-3/4, turns up through the normally open valve F010 and normally closed valve F009, and turns out to leave the drywell and containment at El. 133-0.

This RHR suction line is all in a sing1e vertical plane at azimuth 0 degrees.

The recirculation loops and the RHR suction line up to valve F009 are classified as high energy; from valve F009 out, the RHR suction line is moderate energy.

General Electric-Hitachi (GEH) (the NSSS supplier) is responsible for the location and design of restraints for the recirculation system. Their discussion is presented in Section 3.6B. However, some additional evaluation has been made, particularly as related to neighboring equipment, and is discussed below.

Recirculation Loop A

The GEH analysis resulted in 13 postulated circumferentia1 break locations. Of these, one was at each of the six nozzles at which the 12-inch risers join the RPV; one was at each of the junctions between the 12-inch risers and the 16-inch header; and the other was at the nozzle where the 24-inch pump suction line joins the RPV.

A restraint has been installed on the vertical leg of each riser to limit the travel of the ruptured pipe in the direction of the RPV radius. Six restraints have been installed on the 16-inch header to limit both radial and downward travel. One restraint has been installed on the 16-inch suction line below the first elbow to limit horizontal travel.

Recirculation Loop B

The postulated break locations and resulting restraints are the same for both loops A and B, with the exception that loop B has the RHR connection. This connection results in the postulation of another circumferential failure and longitudinal failure, both at the joint. To protect against these postulated failures, two restraints are installed, one just above and one just below the tee.

RHR Suction Line

The high-energy portion of this line is a short L-shaped run extending about 5 feet horizontally and about 13 feet upward. Circumferential breaks were postulated at both terminal ends (the junction with the recirculation loop B and the inlet to the normally closed valve F009). Intermediate longitudinal and circumferential breaks were postulated at the elbow. A restraint was installed to limit the outward movement of the vertical leg.

Jet Impingement

The effects of jet impingement on essential equipment were considered. Twelve of the 13 breaks postulated for each recirculation line are within the RPV shield, with no essential equipment to be damaged. Outside the RPV shield, the main concern was for the CRD piping. This piping has been analyzed and shown to be functional when exposed to jet impingement loads.

Conclusion

The essential equipment required to mitigate the consequences of this design basis loss-of-coolant accident, as discussed in Chapter 6.0, has been adequately protected from the pipe break effects.

3C.2.4 Steam to RCIC Turbine and RHR

The piping, break locations, and restraints are shown on Figures 3.6A-3 and 3.6A-4. The results of the associated stress analyses are provided in the documents referenced on these figures. The referenced documents are maintained in accordance with engineering administrative procedures.

A 10-inch branch line from main steam line A delivers steam to drive the RCIC turbine and to the RHR heat exchangers.

Inside the drywell, the line starts at El. 167-11 on the vertical leg of main steam line A, azimuth 72 degrees; follows the outer radius of the RPV shield wall horizontally to azimuth 0 degree; drops vertically to El. 143-2-1/4; and goes horizontally through inner isolation valve F063 through the drywell and containment penetrations into the auxiliary building steam tunnel. The steam piping in the containment building steam tunnel is enclosed in a guard pipe.

Inside the auxiliary building steam tunnel, after the second isolation valve (F064), the line branches into three legs: two 8-inch lines and a 6-inch line which supplies the RCIC turbine. One of the 8-inch lines goes north through the steam tunnel wall at El. 147-8, turns west, then north, then west again and terminates at a blind fitting. To this fitting, the line is classified as high energy. The other 8-inch line is essentially a mirror image, with the high-energy portion terminating at the blind fitting. The 6-inch line goes down through the tunnel floor (El. 140-0, inside a vertical pipe chase to El. 119-0, on down to El. 98-8, then essentially horizontally to the normally closed valve F045). To this valve, the line is classified as high energy; beyond this valve to the RCIC turbine, as moderate energy.

From the moment-limiting device adjacent to the inner isolation valve F063 through the guard pipe, outer isolation valve F064, the steam tunnel, and to a point slightly below the floor at El. 140-0 for the RCIC steam line and slightly outside the walls of the tunnel for the two 8-inch lines, the piping meets the requirements of a no-break zone.

Inside the Drywell

For stress analyses, the RCIC steam line within the drywell was modeled as a branch of the main steam piping. Circumferential breaks were postulated at the 28" x 28" x 10" tee of the main steam line, at the first elbow and second elbow, and at the beginning of the no-break zone by the inlet of the inner isolation valve moment-limiting device. The significant source of energy is from the main steam line and reactor; the volume of steam in the line to the RCIC turbine is far less significant.

Three restraints are installed in the horizontal run along the RPV shield wall to prevent possible impact on the main steam safety/relief/ADS valves and a guide has been provided near the inlet of the isolation valve to minimize the probability of valve malfunction resulting from pipe-imposed forces and moments.

Three additional restraints have been provided on the vertical run which prevent pipe whip and minimize jet impingement targets.

Jet impingement effects from the escaping steam (or the later steam-water mixture) have been reviewed; no damage which could prevent a safe shutdown will result. The essential equipment inside the drywell has been qualified for necessary operation in the LOCA and post-LOCA environments.

Inside the Auxiliary Building Steam Tunnel

As all of the RCIC/RHR steam piping inside the auxiliary building steam tunnel is classified as no-break-zone quality, no failure-effects discussion is appropriate (See Ref. 2 of Section 3.6A).

For the 6-inch line, in addition to breaks at each terminal end, circumferential breaks were postulated at the inlet weld of the elbow at the bottom of the vertical leg and at the second tee. Because there is no essential equipment located in this area, no restraints are required below the steam tunnel floor. A bumper-type restraint has been installed within the steam tunnel to preclude pipe whip into the tunnel and to protect the isolation valve. Jet impingement can cause no significant effects. The structure of the concrete pipe chase and of the auxiliary building interior walls will not be damaged to the extent that essential functions are jeopardized or that secondary missiles are produced.

For the two 8-inch lines, in addition to breaks at each of the terminal ends, circumferential breaks were postulated at the elbows outside the no-break zone. An additional circumferential break was also postulated at the end of the no-break zone.

Again, there is little essential equipment located in the vicinity of this piping. The only items of concern outside the steam tunnel are two valves of the RHR system whose continued operability is desirable. Consequently, a restraint was added to each line to protect these valves (Ref. Figures 3.6A-29 and 3.6A-30). These restraints also reduce the area subject to jet impingement. Additional restraints are provided within the steam tunnel to prevent pipe whip into the tunnel and to protect the isolation valve.

Conclusions

A combination of spatial separation and restraints ensures that no postulated failure of either the steam piping to the RCIC turbine or the steam piping to RHR will prevent a safe shutdown of the plant or cause radioactive releases in excess of prescribed limits.

3C.2.5 Deleted

3C.2.6 RWCU Piping

The locations of postulated pipe breaks and of pipe whip restraints are shown on Figures 3.6A-7, 3.6A-7A, and 3.6A-8. The results of the associated stress calculations are provided in the documents referenced on these figures. The referenced documents are maintained in accordance with engineering administrative procedures.

General

For many purposes, the reactor water cleanup system can be considered as two nearly separate subsystems: the loop from the reactor, through the heat exchangers, through the filter/ demineralizers, back through the regenerative heat exchangers, and return to the reactor; and the auxiliary subsystem which removes the used demineralizer resin and replaces it with new resin. The first is almost entirely classified as a high-energy system; the second, even during the small fraction of the time it is in operation, is classified as moderate energy. Neither is required for safe plant shutdown. From the standpoint of piping failure, the only concern is the possible detrimental effects on other equipment.

One 4-inch line from each reactor recirculation loop and the 4-inch line from the reactor lower head drain join into a 6-inch line inside the drywell at El. 107-6. This line progresses upward, leaving the drywell at El. 141-6. Enclosed in a guard pipe, this line passes through the drywell wall and the containment wall into the auxiliary building steam tunnel. Isolation valves are located near each end of this guard pipe. The line, now enlarged to an 8-inch size, then goes down through a pipe chase to El. 116-10, where it divides into the 4-inch lines supplying the two pumps located in separate rooms. From the pumps' discharge, the two 4-inch lines join again into a 4-inch line at El. 120-3 which goes up through the same pipe chase into the steam tunnel and then through a penetration (with two isolation valves) into the containment. The sections of suction piping from the inlet of the inboard isolation valve, through the guard pipe, to the restraints just outside the pipe chase at El. 116-10 and from the restraints at El. 120-3 just outside the pipe chase for the discharge line, to the penetration through which the

pipe reenters the containment (inside the steam tunnel and the vertical pipe chase) to the inboard isolation valve, are qualified as no-break zones.

After reentering the containment, specifically, into the containment steam tunnel, the 4-inch line runs horizontally to near the tunnel north wall, then west, then up through the floor (El. 170-0) of the heat exchanger room, and then into the heat exchangers. After leaving the heat exchangers, the pipe leaves the room at El. 179-3, entering a horizontal pipe chase following the outer curve of the drywell to the filter/demineralizer room in the south side of the containment. After the filter/ demineralizers, the pipe (now a 6-inch size) returns via the same pipe chase to the regenerative heat exchangers. From there, it drops through the floor at El. 170-0 into the containment steam tunnel and through the containment penetration (with two isolation valves) where it joins the RHR piping to the feedwater and returns to the RPV. From the inner isolation valve, through the penetration, and the RHR fitting, this piping is again qualified as no-break zone. (Refer to subsection 3C.2.2 for the discussion on the feedwater piping.)

The high energy RWCU 6"-DBA-89 bypass piping to the heat exchangers penetrates the drywell at penetration 337 and two isolation valves F250 and F251 provide isolation capability in the event of a high energy line break outside the drywell. In order to maintain the capability for the isolation of the drywell and to prevent the possible overpressurization of the RWCU heat exchanger room due to postulated breaks in the vicinity of the outboard isolation valve, the piping between the drywell isolation valves has been qualified as a no-break zone similar to the qualification requirements for piping between containment isolation valves per subsection 3.6A.2.1.C.2. Refer to Figure 3.6A-7.

Inside the Drywell

Circumferential breaks have been postulated at all terminal ends (i.e., the junctions at the RPV drain line at the recirculation piping) and at the boundary of the no-break zone. Intermediate circumferential breaks have been postulated at each weld of the 6-inch x 4-inch tee where usage factors exceed 0.1. A longitudinal break is also postulated in the 6-inch run of the tee. An intermediate circumferential break is also postulated at the dissimilar metal weld in the 4-inch piping.

Because of the arrangement of the piping, very little essential equipment could be damaged by a failed pipe. The only significant item is the control rod drive hydraulic piping inside the reactor pedestal below the RPV. To prevent possible crimping of the insert or withdraw piping, which could impair SCRAM operability, five restraints are installed within the pedestal. Outside of the pedestal, but still within the drywell, a series of restraints has been installed in lines 6"-DBA-9, 6"-DBA-89, and 6"-DBA-90, to and from the RWCU filter demineralizer, to protect isolation valve F001 and to preclude excessive moments at the containment penetration flued head. In addition, although no vulnerable essential equipment is nearby, restraints have been installed to limit the movement of the pipe end and thus minimize the arc of jet impingement.

Because all essential equipment in the drywell has been qualified for the necessary post-LOCA operation, the less severe environment resulting from a postulated rupture of a 4-inch or 6-inch RWCU pipe will cause no safety hazard.

Inside the Auxiliary Building Steam Tunnel and Pipe Chase

All RWCU piping in the auxiliary building steam tunnel and the vertical pipe chase between the tunnel and the RWCU pump room is qualified as a no-break zone. Consequently, no break analysis is required.

Inside the RWCU Pump Rooms

Starting with valves F043A and F043B, slightly beyond the restraint at the end of the no-break zone, the piping has been fabricated and installed to nonnuclear standards. Consequently, circumferential breaks have been postulated at every fitting, weld attachment, and valve; however, there is no essential equipment in these rooms. All equipment is associated with the nonessential RWCU system. All pipe whip effects and jet impingement effects would be confined to these rooms.

Inside Containment Tunnel and Heat Exchanger Room

From the auxiliary building steam tunnel, the piping penetrates the containment, includes isolation valve F053, and goes up through the floor at El. 170-0, into the RWCU heat exchanger room. From valve F053 through the floor, the piping is nonnuclear; consequently, failures were postulated at each fitting starting at the valve-to-pipe weld, except for the region in the vicinity

of isolation valve F251. In this region breaks were postulated based on stress analysis data of the piping system. Refer to Figure 3.6A-7 for additional information. However, there is no essential equipment in the tunnel or heat exchanger room which could be jeopardized by the whipping 4-inch and 6-inch piping.

Inside the Filter/Demineralizer Room

From the heat exchanger room, the piping (still 4-inch nonnuclear) goes through a horizontal pipe chase to the filter/ demineralizer room. As in the heat exchanger and pump rooms, circumferential breaks were postulated at each fitting. Again, there is no essential equipment in the room.

In the Turbine Building

One 4-inch moderate-energy line goes from the auxiliary building steam tunnel to the main condenser in the turbine building. However, there is no high-energy RWCU piping in the turbine building nor any essential equipment.

Conclusions

Partly, by the addition of restraints, but primarily by physical separation, no postulated failure of the reactor water cleanup system will prevent the necessary operation of essential equipment.

3C.2.7 LPCS and HPCS

These systems do not operate during normal plant operation; consequently, only part of the piping which is normally exposed to reactor pressure is classified as high energy.

The piping, break locations, and restraints for the LPCS are shown on Figure 3.6A-11; the results of the associated stress calculations are provided in the documents referenced on this figure. For the HPCS, the comparable reference is Figure 3.6A-10. The referenced documents are maintained in accordance with engineering administrative procedures.

LPCS

The 14-inch LPCS piping is attached to the RPV nozzle at El. 162-9-¼ and azimuth 120 degrees. After passing through the RPV shield wall, it drops vertically downward through locked-open valve F007 and normally-closed check valve F006, at which the high-energy portion stops.

Terminal end circumferential breaks have been postulated at the RPV nozzle and at the high-pressure (i.e., RPV) side of check valve F006. The two intermediate breaks were postulated to occur at each end of valve F036.

Three whip restraints are provided to restrain the LPCS piping. These restraints prevent pipe whip and minimize jet impingement targets. To assure that no whipping of the vertical leg can occur, a restraint has been installed slightly below the elbow at the top of that leg. In addition, to assure no damage occurs to the isolation valve (located in the moderate-energy part of the piping) a restraint has been installed (El. 150-1-1/4) on the low-pressure side of check valve F006. Damage to the RPV shield wall at the penetration is precluded by the energy-absorbent material installed in the wall opening.

HPCS

The 14-inch HPCS piping is very similar to the LPCS piping just discussed, except that it is located at azimuth 240 degrees, and the related valves are locked-open F205 and normally-closed check valve F005. Pipe whip is prevented with similar restraints. An additional difference is the 3 inch branch located just above valve F005 which provides an injection path for the Standby Liquid Control System. The effects of the pipe breaks in this piping are discussed in Section 3C.2.9.4.

Conclusions

No postulated failure of either LPCS or HPCS can prevent safe shutdown by pipe whip, jet impingement, or environmental effects.

3C.2.8 LPCI Mode of RHR

The piping, break locations, and restraints for the LPCI function of the RHR system are shown on Figure 3.6A-12; the results of the associated stress analyses are provided in the documents referenced on this figure. The referenced documents are maintained in accordance with engineering administrative procedures.

The LPCI subsystems are not in use during normal plant operation; consequently, only the piping between the RPV and the first normally-closed valve is pressurized, and thus classified as high energy.

LPCI A

From the RPV nozzle (El. 157-9, azimuth 39 degrees) the line goes through the RPV shield wall, then vertically down to El. 140-4, then horizontally through locked-open valve F039A and normally closed check valve F041A to the drywell penetration. The portion from the outlet (RPV end) of valve F041A is classified as high energy.

In addition to the breaks postulated at the terminal ends, a circumferential break was postulated at the RPV end of the top elbow of the vertical leg and at the valve end of the lower elbow. A restraint has been installed just below the upper elbow to limit travel of the vertical pipe resulting from either postulated failure between that restraint and the RPV. Slightly above the lower elbow, an energy absorbing restraint has been added to assure that the pipe does not cause significant damage to the RPV shield wall. A third restraint, also of the energy-absorbing type, is located between the lower elbow and valve F039A to limit downward travel. By limiting movement of the broken end of pipe, the area which could be impacted by the jet of steam-water mixture is also reduced. No essential components are adversely affected by pipe whip or jet impingement from postulated breaks in the LPCI A piping.

LPCI B

Loop B piping starts at the RPV nozzle (El. 157-9, azimuth 219 degrees), passes through the RPV shield wall, and vertically downward. Unlike loop A, two loop B valves (locked-open valve F039B and normally-closed check valve F041B) are in this vertical leg. The high-energy portion ends at the top of valve F041B at El. 148-10.

Circumferential breaks were postulated at each terminal end and at each end of the elbow. A restraint was installed on the straight pipe near each end of the elbow to limit the travel of the pipe. A restraint was also installed on the moderate-energy line (the low-pressure side of check valve F041B) so that failure of the high-energy portion will not cause unacceptable movement of the moderate-energy piping, and conceivably impair operation

of the isolation valve. High-energy piping in the vicinity has been analyzed for jet impingement loads and has been shown not to be adversely affected.

LPCI C

Loop C piping is very similar in configuration to Loop B; major differences include the azimuth 141 degrees of the nozzle, the pipe size (12-inch), and the valve numbers (locked-open valve F039C and normally-closed check valve F041C). Postulated break locations and restraints are equivalent to Loop B. High-energy piping in the vicinity has been analyzed for jet impingement loads and has been shown not to be adversely affected.

Drywell Environment Effects

Environmental effects within the drywell, from temperature, humidity, etc., will be no more severe than that for which essential equipment in the drywell has been qualified in accordance with Section 3.11.

Conclusion

The LPCI piping is adequately restrained to prevent pipe whip and jet impingement or environmental effects do not adversely affect the function of essential components in the vicinity. Therefore, no postulated failure of the LPCI piping could prevent the safe shutdown of the plant.

3C.2.9 Three-Inch and Smaller High-Energy Lines

3C.2.9.1 General

This portion of Appendix 3C describes the effects of pertinent 3-inch and smaller high-energy piping and the protection of essential components.

Small (3-inch and under) high-energy lines were routed with maximum possible separation and the length of piping runs were minimized to shorten high-energy boundaries whenever possible. As a result, the failure effects evaluation/protection was performed on a fitting by fitting basis in lieu of final stress data for break locations except for the 3-inch main steam drains, the main steam drains upstream of the outboard MSIV's, the Feedwater Leakage Control and the 3-inch SLC injection into the HPCS header. In the case of the 3-inch main steam drains,

portions are considered no-break zone quality. Stress data documents referenced on Figures 3.6A-13A, 13B and 10 verify that the requirements of MEB 3-1 are met. The referenced documents are maintained in accordance with engineering administrative procedures.

For the main steam drains upstream of the outboard MSIV's and the Feedwater Leakage Control, they are no break zone quality to meet the requirements of Reg. Guide 1.96. Stress data documents are referenced on Figures 3.6A-13e and 3.6A-13i and are maintained in accordance with engineering administrative procedures.

The 3-inch CRD drive water return previously identified as high energy in the FSAR has been deleted, and the RPV nozzle capped. No further discussion is required. In addition, no discussion is provided for small high-energy lines located in areas containing no essential shutdown components.

3C.2.9.2 Main Steam and MSIV Leakage Control

3C.2.9.2.1 Main Steam Drains Upstream of Inboard MSIV's

The main steam drains consist of a series of 2-inch branch lines of the main steam system at various locations in the drywell and auxiliary building main steam tunnel.

The 2-inch drains originating in the drywell connect to a 3-inch pipe which is encapsulated by guard pipe between the drywell and containment walls. The 3-inch pipe continues from the guard pipe inside the auxiliary building steam tunnel and exits the steam tunnel where it becomes moderate energy at valve F021. The portion of this 3-inch piping just upstream of F016B to the fourth restraint downstream of F019A is classified as no-break zone quality. The pipe route, with stress data points, is shown on Figures 3.6-13A and 13B. Documents which provide stress data are referenced on these figures. The referenced documents are maintained in accordance with engineering administrative procedures. For the portion of the drains in the drywell which are not part of the no break zone, restraints are provided to prevent any whipping or jet impingement arcs which may affect the inboard containment isolation valves or electrical supplies. The four restraints downstream of F019A have been provided to reduce pipe movements following postulated breaks downstream from the restraints. For the piping identified as no break zone quality, a failure effects discussion is not required (see ref. 2 of section 3.6A).

In conclusion, the restraints have minimized the adverse effects resulting from whip and jet impingement so as not to impair the operation of essential components.

3C.2.9.2.2 Main Steam Drains Upstream of Outboard MSIV's and MSIV Leakage Control

The main steam drains consist of a series of 2-inch branch lines of the main steam system at various locations in the drywell and auxiliary building main steam tunnel.

The 2-inch drains, upstream of the outboard MSIV's, originate in the auxiliary building steam tunnel and reduce to 1½-inch. From this 1½-inch line, it branches for inboard system of the MSIVLC system and continues to a common 1½-inch line for all four of the drains. This common line continues to the main condenser. This piping is classified as no break zone quality from the main steam lines to valves as shown on Figures 3.6A-13e, sheets 1 through 5, in order to comply with Reg. Guide 1.96, Appendix A. Documents which provide stress data are referenced on these figures. The referenced documents are maintained in accordance with engineering administrative procedures. Downstream of valves F001A,E,J & N, the piping is classified as moderate energy. Restraints were provided to preclude the whipping or to direct jet impingement of the 1½-inch line away from outboard isolation valves or their controls. For the piping identified as no break zone quality, a failure effects discussion is not required (see ref. 2 of section 3.6A).

In conclusion, the restraints have minimized the adverse effects resulting from whip and jet impingement so as not to impair the operation of essential components.

3C.2.9.2.3 Feedwater Leakage Control

This piping is classified as no break zone quality from the feedwater lines to valves as shown on Figures 3.6A-13i in order to comply with Reg. Guide 1.96, Appendix A. Documents which provide stress data are referenced on these figures. The referenced documents are maintained in accordance with engineering administrative procedures. Downstream of valves F003A & B, the piping has been classified as moderate energy.

Since this piping is identified as no break zone quality, a failure effects discussion is not required (see Ref. 2 of section 3.6A).

3C.2.9.2.4 Standby Liquid Control Injection into HPCS Discharge

The high-energy portion of the SLCS injection line consists of a short length of 1½" and 3" pipe prior to connecting to the 14" HPCS discharge line.

Breaks were postulated based on stress data documents referenced on Figure 3.6A-10. The only significant essential equipment in the region is the 14" LPCI line. The effects of these breaks on the LPCI line are enveloped by existing analysis of breaks from HPCS. Therefore, no postulated failure of this piping will prevent safe shutdown of the plant.

3C.2.9.2.5 RPV Drains

The two 2-inch RPV drain pipes are located at the bottom of the RPV and are routed horizontally (one at azimuth 0 degrees and the other at 180 degrees) until dropping vertically down to their connection to the 4-inch RWCU piping as described in 3C.2.6 and shown on Figure 3.6A-7.

These pipes were routed 90 degrees away from CRD discharge SCRAM lines for optimum separation. The breaks at the RPV connections which may result in downward pipe and fluid jet movement pose no hazard. However, breaks further away from the RPV have a constant energy source, longer moment arm, and the potential to move in a plane towards the CRD. This portion of the piping is restrained to prohibit any adverse effects to the SCRAM function.

An additional drain exists as a result of the rerouting of the SLC injection line to the HPCS discharge. This line still serves to sense the differential pressure across the core support plate.

This high-energy piping is primarily concentrated below the reactor pressure vessel in the pedestal region. The piping penetrates the pedestal wall at E1. 112, azimuth 187 degrees and extends into the drywell for a distance of approximately 5 feet. At that point the high-energy boundary terminates at the first check valve closed to reactor pressure.

The only significant essential equipment in the pedestal region is the CRD discharge SCRAM lines. These are bundled and routed 90 degrees away from this drain piping thereby being removed from the routing of the high-energy piping.

As a result a break at the drain connection at the bottom of the RPV, which may cause downward pipe movement and fluid jet, poses no hazard. However, breaks further away from the RPV have a constant energy source, longer moment arm, and the potential to move in a plane towards the CRD. This portion of the piping is, therefore, restrained to prohibit any adverse effects to SCRAM capability.

The high-energy portion outside the pedestal presents no safe shutdown hazard because of its short length and routing.

3C.2.9.2.6 Two-Inch Line from RPV Head Nozzle to Main Steam A

The 2-inch high-energy pipe connected to the head nozzle is routed above the RPV in the drywell head region and is supported by the seismic insulation structure. Thereafter it penetrates the drywell bulkhead into the drywell area below.

At El. 170, azimuth 72 degrees, it connects to main steam line A. The 2-inch vent line branching from this spray line has a normally closed valve located about 1 foot from the tee, and causes no additional hazards.

Restraints were provided above the RPV head to prevent any whipping pipes from impacting the drywell head or the seismically supported insulation structure. The piping below the bulkhead at the elevations mentioned is not near any essential components needed to mitigate the consequences of this type of break. As a result, further protection is not warranted.

3C.3 MODERATE-ENERGY PIPING FAILURE

Through-wall leakage cracks were postulated to occur in accordance with subsection 3.6.2A. These cracks were assumed to result in low-velocity wetting of any equipment in the area, whether above or below the crack, and at substantial distances from the crack. In addition, the possibility of compartment flooding has been considered; this aspect is discussed in Section 3C.4.

Each room, compartment, or area has been evaluated. The initial approach was to assume the wetted equipment is rendered inoperable, and to evaluate the safety effects of its failure. If the effects were not shown to be acceptable, a more detailed review was made to determine the actual vulnerability of the item to spray damage, and if necessary, to assure its safety. Each

compartment in the auxiliary building, plus the containment, control building, and standby service water pump houses are discussed. No cracks are postulated in the moderate energy piping inside the diesel generator building based on the exemption criteria discussed in subsection 3.6A.2.1.c.4(d)(1). These areas can be identified by reference to Figures 1.2-2 through 1.2-8.

The synergistic effects of an independent single failure have been evaluated, and found acceptable subject to any limitations specifically discussed.

For some systems (e.g., RHR and standby service water), a single failure in the redundant system has been excluded in accordance with subsection B.3.b(3) of BTP APCSB 3-1; in all such cases, the redundant train meets the qualification of that NRC position.

The availability of offsite power has been assumed only when the direct effects of the postulated piping failure could be expected to cause a trip of the turbine-generator system or reactor protection system. In all cases, at the least, either one of the two emergency diesel-generator systems or offsite power is available.

3C.3.1 Containment

All equipment within the containment (and drywell) which must operate during or after a LOCA is qualified for the appropriate environmental conditions, as described in Section 3.11. The wetting associated with any postulated failure of any moderate-energy piping is within the bounds of that qualification. Consequently, no item-by-item discussion of this less severe event is required to show that the ability to safely shut down the plant will not be impaired.

3C.3.2 Auxiliary Building

Each compartment or significant area will be discussed in sequence, starting at the lowest elevation (93-0). Most of the essential equipment is located on the lowest two levels, in compartments near the east and west ends of the containment.

3C.3.2.1 RHR Pump Room A, Elevation 93-0 and 119-0

It has been assumed that any piping failure in this room would disable all equipment in the room. This would result in loss of RHR pump A, but little else; RHR B, RHR C, RCIC, and all other

ECCS would remain operable, subject to the additionally postulated single active failure, and the consequences would be within the range of events analyzed in Chapters 6 and 15.

In addition to RHR A, some electrical circuits of Division 1 would be wetted. This was assumed to remove from service one (of four) suppression pool level monitors, and several containment isolation valves. However, even should any (or all) of these valves fail in the worst position, no significant release to the public of radioactivity would occur; there would be no substantial amount of free radioactive material to escape; and the standby gas treatment system and auxiliary building boundary would still be functional.

There is one exception to the above. Also in this room is the dc-powered, normally open valve in the RCIC turbine exhaust steam to the suppression pool. Should this valve fail in a manner which would cause it to close, the RCIC system would be inoperable. However, operation of the RCIC is not required to recover from the effects of any piping failure in the RHR pump room.

3C.3.2.2 RHR Pump Room B, Elevation 93-0 and 119-0

Because of the similarity between the arrangements of RHR A and RHR B pump rooms, the basic analysis and conclusions of RHR A apply. Specific differences are the presence mostly of electrical Division 2 circuits, and the absence of the above-mentioned RCIC steam valve.

3C.3.2.3 RCIC Pump Room, Elevation 93-0

It has been assumed that all equipment in this room would be inoperable. The only equipment in this room is that associated with the RCIC system, which is not required for mitigation of the effects of any piping failure in the room.

3C.3.2.4 RHR Pump C, Elevation 93-0

All of the equipment in this room is associated with RHR C except for one (of four) suppression pool level monitor and heat exchangers, valves, and piping associated with the RHR ADHRS subsystem. Electrical circuits are part of Division 2. All of this equipment is assumed to be inoperable. RHR C is not required for plant shutdown.

3C.3.2.5 HPCS Room, Elevation 93-0

All of the equipment in this room is associated with the HPCS, and its Division 3 circuits, and is assumed to be inoperable. Even assuming an additional single failure in the ADS, the plant could be safely shut down without the use of the HPCS.

3C.3.2.6 LPCS Pump Room, Elevation 93-0

All of the equipment in this room is associated with the LPCS except for one (of four) suppression pool level monitor. Electrical circuits are part of Division 1. All of this equipment is assumed to be inoperable. LPCS is not required for plant shutdown.

3C.3.2.7 Piping Penetration Room, Elevation 93-0

There is no safety-related equipment in this room; however, there are several valves, including containment isolation valves, and their operator power and control circuits of both Divisions 1 and 2. None of these valves is directly associated with reactor operation, and they are not required to function for mitigation of the effects of failure of any piping in the room.

3C.3.2.8 Electrical Switchgear Rooms, Elevation 119-0 and 139-0

There are four rooms containing ESF switchgear and containment electrical penetrations. As shown by Figures 1.2-2 through 1.2-8, each room is 90 degrees apart and completely separate from the other. Because the floor at El. 139-0 is not designed to be watertight, it has very conservatively been assumed that moderate-energy piping failures in either the upper or the lower level will spray both levels. The only piping which could be of concern is the water piping servicing the room coolers. Although most electric penetrations of the switchgear cabinets are fairly well sealed (for dust protection, etc.), it has not been proven that the switchgear will remain fully operable under spray conditions. In general, the upper level room contains instrumentation and control circuitry, while the lower level room contains circuitry associated with the same systems. The rooms in the region of the auxiliary building between 0 and 180 degrees are related to Division 1 electrical only and completely separate from the rooms between 180 and 360 degrees which are related to Division 2 electrical. Consequently, it has been assumed that the failure in a Division 1 related room results in a loss of the switchgear and the equipment powered or controlled by it.

However, Division 2 is not affected by this failure or its consequences and vice versa for failures in Division 2 rooms. Adding to the loss an assumed independent single failure, the plant can be safely shut down.

3C.3.2.9 RWCU Pump Rooms, Elevation 115-0 and 119-0

The RWCU pumps are not required for plant shutdown; no essential equipment is in either of these two rooms. The doors to these rooms are designed to withstand the compartment pressures resulting from a ruptured pipe (see Appendix 3E for discussion on compartment pressure); consequently, failure of a pipe in these rooms would have no safety consequences.

The effects of pipe failure in the adjacent rooms, immediately east of the RWCU pump rooms, have also been considered. In this room are two isolation valves associated with the RCIC system.

One of these valves is for a 2-inch RCIC minimum flow line, normally closed, whose failure in either the open or closed position would cause no harm. The second valve is a 20-inch normally-closed valve in the line from the recirculation system to the RHR pump section. Failure of either or both of these valves, in the closed position, would reduce the operational flexibility of the RHR system, but would not preclude safe shutdown by use of the RCIC system and/or other operational modes of the RHR system.

3C.3.2.10 Piping Penetration Room, West End, Elevation 139-0

Other than several containment isolation valves, there is no safety-related equipment in this room. The failure of any or all of these isolation valves, in either the open or shut position, would not prevent a safe plant shutdown.

3C.3.2.11 RHR A Pump and Heat Exchanger Removal Area, Elevation 139-0

The only safety-related items in this area are valves and electrical circuitry associated with the RHR A system. As it has been assumed for the high-energy pipe failure analysis that this system is lost as a direct result of a piping failure, with an additional assumed single failure, no additional effects of spray would worsen the consequences. A safe plant shutdown could be achieved.

3C.3.2.12 RHR B Pump and Heat Exchanger Removal Area, Elevation 139-0

This area is much like RHR A area in subsection 3C.3.2.11, with the exception that this area contains a 6-inch butterfly valve which, if it failed shut, would prevent SGTS train B from drawing from the auxiliary building steam tunnels. However, as (1) train A would be available, and (2) the SGTS is not required for this abnormal condition, no safety hazard exists.

3C.3.2.13 Main Steam Pipe Tunnel, Elevation 140-0

The only items having safety significance in the steam tunnel are the numerous isolation valves. The major pipes in this steam tunnel are no-break zone quality; however, there are also several moderate-energy pipes installed to ANSI B31.1 standards for which failures are postulated. It has been assumed that these failures could cause failure, in either the open or the shut position, of any electrically or pneumatically operated valve in the tunnel. No such valve failure, or combination of such failures, could prevent a safe shutdown or cause unacceptable radiation release.

3C.3.2.14 Elevations 161-10 and 166-0, and Above

At these upper levels, there is relatively little piping or essential equipment. Essential equipment reviewed for potential damage included ESF motor controls MCC 15B21 and MCC 16B41, the room coolers for those MCCs, the drywell and containment hydrogen analyzers, the spent fuel pool cooling pumps and heat exchangers, the enclosure building recirculation fans, containment isolation valves, and electrical Divisions 1 and 2 including the power and controls for the above items.

Using the conservative assumption that any wetted item is rendered inoperable, with isolation valves failing to either the open or closed position, no effect of pipe cracks combined with an assumed independent single failure would prevent the safe shutdown of the plant or endanger the public.

3C.3.3 Control Building

Piping in the control building was located with the intention of minimizing the probability of equipment damage from pipe leaks. This approach, combined with the spatial separation of redundant equipment and systems, results in a very low probability of any hazardous effects of leakage.

Service and chilled water supply and return lines are routed, to the extent feasible, in pipe chases and other areas where their postulated failure could not wet vulnerable equipment. No piping is located in the ceiling over the main control room, nor over the emergency shutdown station at El. 111-0.

3C.3.4 Diesel Generator Building

Each of the two standby diesel generators, and the one HPCS diesel generator, has its own room within the building. There are no doors or similar openings in the concrete walls separating them. However, any fluid accumulation in one room could communicate to other rooms through the common hallway. Since pipe cracks are not postulated inside the diesel generator building (see section 3C.3), equipment required for safe shutdown of the plant will not be affected.

Standby Service Water Pump Houses

Each of the two SSW basins has a pump house. The A standby service water pump house contains the Division I SSW pump and the HPCS service water pump. The B standby service water pump house contains the Division II SSW pump. Consequently, no postulated pipe failure in either pump house would disable the redundant pump for that unit, located in the other pump house. In accordance with Section B.3.b(3) of APCSB 3-1, it is not necessary to assume a single failure in the redundant service water train.

3C.4 COMPARTMENT FLOODING

General

In accordance with Section 3.6A, leakage cracks have been postulated at any point on all moderate-energy piping larger than 1-inch-nominal diameter. The leakage area is assumed to be equal to the product of one-half the pipe wall thickness and one-half the pipe inside diameter, but in a circular shape. Resulting flow rates have been approximated using Crane Technical Paper 410 (1957), equation 2-14, with a flow coefficient of 0.6 and normal operating pressure for the pipe. This leads to flows from near zero to about 826 gpm (from the 36-inch plant service water line).

Flooding can also occur as a result of high-energy piping failures. In general, the flooding effects are less a concern than are the immediate effects of pipe whip and jet impingement; however, each room containing high-energy piping has been

considered, and the flood effects evaluated. Flooding from a high-energy pipe failure would proceed more rapidly (i.e., inflow more gpm) than from a moderate-energy line; however, the failure would be detected and isolated more rapidly, by either automatic or manually initiated action, and the level to which a room might flood would be lower for the high-energy piping failure.

All areas of concern are equipped with drain sumps and at least two sump pumps. Although these pumps would be expected to operate, they are not designed to ESF standards; consequently, to provide additional conservatism, they have not been considered to be in operation until after the leak has been detected, at least partially identified, and operator action taken. Initial leak detection would generally be by the sump level instrumentation, with radiation and temperature instrumentation also available for appropriate systems. In addition, most areas within the plant (those not prohibited by radiation or other conditions) would either be routinely manned or patrolled each shift, so that many small leaks could be detected visually. Larger leaks, as postulated for high-energy piping, would often be detected by the effect on flow and/or pressure inside the failed pipe using the leak detection system described in subsections 5.2.5 and 7.6.1.4.

For all areas, a single independent failure was assumed. The end result was the same as for the analysis of moderate-energy piping failures, in that it was assumed that all equipment in the room containing the failed pipe was inoperable. For all events, a safe shutdown can be achieved, with no hazard to the public.

3C.4.1 Containment

The drywell is designed to be flooded, with water flowing over the weir wall to the suppression pool, during a postulated LOCA; any non-LOCA flooding would be substantially less.

There are few pipes which could leak into the containment steam tunnel; most are enclosed in guard pipes. The 4-inch RWCU backwash transfer line is used only a small fraction of the time, and if it failed then, would only leak at about 41 gpm. No postulated leakage could prevent a safe shutdown.

In the rooms housing the RWCU heat exchangers, filter-demineralizers, backwash tank, pumps, and valves, there are several lines which could be postulated to leak; the largest is the 6-inch RWCU line, which at the normal operating pressure of

1013 psig could leak at as much as 472 gpm. However, there is no essential equipment located in these rooms, and a safe shutdown would not be prevented. Refer to subsection 3C.2.6 for details.

Flooding from a larger leak, or full circumferential break, would be no worse even if not detected earlier. Although events would occur faster, the same end results would develop.

3C.4.2 Auxiliary Building

3C.4.2.1 Main Corridors, Elevation 93-0

In this area, around the perimeter of the building, the largest postulated leak would be from the 36-inch plant service water system, with a maximum leakage rate in the order of 826 gpm. The floor area which would be flooded is about 18,389 square feet; at these conditions, assuming no sump pumps, the water would accumulate at about 4.32 inches per hour. Detection and isolation would prevent safety hazards. There are also four sump pumps available, each rated at 100 gpm; use of any or all of these would decrease the rate of accumulation.

3C.4.2.2 RHR Rooms A and B, Elevation 93-0

In each of these rooms, the greatest threat would be the postulation of a leak in the 20-inch RHR suction line between the suppression pool and the isolation valve. To flood to the top of the blowout shaft barrier at El. 108-0 would require over 30 hours, more than enough time to take corrective action. Thus, although the single RHR train (A or B) in that room is presumed to be lost, no other safety-related equipment would be affected.

3C.4.2.3 RCIC Room, Elevation 93-0

Since the RCIC system is normally not operating, the pump suction line poses the only concern. A postulated crack in the 6-inch RCIC pump suction line would give a leakage rate of 84.5 gpm and a flooding rate of 111.3 minutes per foot. The floor drain system would remove the water and a flood level switch, mounted on the wall, will provide an alarm in the control room. With this rate of flooding and with the water level alarm, ample opportunity has been provided to take any necessary corrective actions.

3C.4.2.4 HPCS Room, Elevation 93-0

In this room, the largest postulated leakage flow rate would be 445 gpm, from the 24-inch pump suction line, supplied from the condensate storage tank.This would result in flooding at the rate of 16.8 minutes per foot depth, or one hour to flood to 46 inches. Again, there is ample time to detect the failure, by the sump level alarms or by visual observations. If the sump pumps are operable (two, at 100 gpm each), the rate of increase in water level would be slower. The watertight door on the room would prevent flooding of neighboring areas of the auxiliary building. Although the HPCS would be lost under these assumptions, the plant can be safely shut down.

3C.4.2.5 LPCS Room, Elevation 93-0

Similar analysis of the LPCS room showed that the 24-inch pump suction line, at 75 psig, was the greatest concern, with a leakage flow of 431.4 gpm (the same as the HPCS).However, the larger room size gives a flooding rate of only 33.9 minutes per foot. The watertight door will prevent flooding outside of the room.

3C.4.2.6 RHR Room C, Elevation 93-0

For this room, again the pump suction line posed the greatest concern. Postulated leakage through this 24-inch line, at 20 psig, is 328.1 gpm for a flooding rate of 44.57 minutes per foot. Again, the watertight doors will prevent outside flooding.

3C.4.2.7 Elevations Above 93-0

Above the lowest floor of the auxiliary building, and with the exception of the railroad bay area at elevation 139'-0", all areas and rooms open directly to the main corridors, and any significant quantities of water in those areas would drain freely to the elevation 93'-0" floor. Floor drains in the railroad bay have been sealed to assist in SGTS draw-down, and as a result, water coming from ruptured fire water piping would drain to the plant yard. Therefore, the effects would be the same as or less severe than those described in detail for elevation 93'-0".

3C.4.2.8 The ESF Switchgear Rooms

Four each at El. 119-0 and 139-0 have no significant sources of water. The combined discharge of the component cooling water pump (at El. 93-0) could produce a postulated leakage rate of 274 gpm

(14-inch pipe at 85 psig); any branch lines throughout the building would be less. This is less severe than the plant service water line previously discussed, as would be the leakage from the spent fuel pool cooling system.

3C.4.3 Control Building

The principal potential sources of leakage water in the control building would be from the chilled water system (used for HVAC) and the fire protection system. Neither of these is in the immediate vicinity of the control room. Leakage from either of these systems would travel toward the lowest elevation of the building (El. 93-0) where there is no essential equipment. The floor drain system would remove the water at numerous points in the building.

3C.4.4 Diesel Generator Building

All four moderate energy piping systems inside the diesel generator building, i.e., plant chilled water, fire protection, standby service water and domestic water systems, have been reviewed and are exempted from postulation of pipe cracks (see section 3C.3).

3C.4.5 Standby Service Water Pump House

There is no normally operating piping in this area which could cause flooding. The standby service water system itself, when running, could conceivably leak; however, even if the leakage did not escape from the pump house, only the one train of the system would be affected. Safe shutdown could be achieved using the other standby service water system, with pumps located in an entirely separate pump house.

3C.5 REFERENCES

(1) Letter, L. F. Dale, MP&L to Harold R. Denton (NRC), dated April 26, 1982 (AECM-82/171).

(2) Letter, L. F. Dale, MP&L to Harold R. Denton (NRC), dated October 25, 1982 (AECM-82/454).

(3) Letter, L. F. Dale, MP&L to Harold R. Denton (NRC), dated October 18, 1984 (AECM-84/404).

APPENDIX 3D COMPUTER PROGRAMS USED FOR STRUCTURAL AND SEISMIC ANALYSES BY BECHTEL POWER CORPORATION

A. INTRODUCTION

A number of computer programs were used in the structural analyses of the seismic Category I structures. These programs are described and documented in this appendix.

B. COMPUTER PROGRAMS USED FOR STRUCTURAL AND SEISMIC ANALYSES BY BECHTEL POWER CORPORATION

3D.1 BECHTEL CE 309, STRUCTURAL ENGINEERING SYSTEMS SOLVER (STRESS)

1. Description

STRESS is a programming system for the solution of structural engineering problems. The system is capable of executing the linear, elastic, static analyses of 2- and 3-dimensional framed structures of the following types:

a. Plane truss

b. Plane frame

c. Plane grid

d. Space truss

e. Space frame

[HISTORICAL INFORMATION] [The programming system was originally developed at Massachusetts Institute of Technology in 1964 and is now in the public domain (Ref. 1)].

2. Validation

The program has been verified by the ICES STRUDL II program. A sample problem of space frame analysis was run using the CE 309 program and the commercially available versions (Version 1 and Version 2) of the ICES STRUDL II program. The results from these runs were found to be

identical.[HISTORICAL INFORMATION] [Document traceability is available at Bechtel International Corporation.]

3. Extent of Application

The program was used to perform structural analysis for both steel and concrete structures.

4. Reference

1. Fenjes, S. J., Logcher, R. D., and Mauch, S. P., Stress Reference Manual, the M.I.T. Press, Cambridge, Mass., 1964.

3D.2 BECHTEL CE 601, COMPOSITE BEAM DESIGN

1. Description

The program selects steel beams from stored lists within the program to satisfy the input of floor/ roof loads and slab thicknesses using the methods and allowable stress provisions of the 1969 AISC Specification for the Design, Fabrication and Erection of Structural Steel for Buildings. The program also computes the number of shear studs and their spacing for the beams selected.

2. Validation

The solutions to the program have been demonstrated to be substantially identical to the results obtained by manual calculations. [HISTORICAL INFORMATION] [Document traceability is available at Bechtel Power Corporation.]

3. Extent of Application

The program was used to design composite beams in the auxiliary building.

3D.3 BECHTEL CE 650, BEAM PROPERTIES

1. Description

The program computes section properties of beam segments. Properties computed are cross-sectional area, shear area, moment of inertia, centroidal distances, and segment weights.

2. Validation

The solutions to the problem have been verified to be substantially identical to the results obtained by manual calculations. [HISTORICAL INFORMATION] [Document traceability is available at the Bechtel Power Corporation.]

3. Extent of Application

The program was used to compute the section properties of beam segments.

3D.4 BECHTEL CE 668, PLATE BENDING ANALYSIS

1. Description

The program performs the linear elastic analysis of a plate with arbitrary shape and supports, stiffener beams, and elastic subgrade, under loads normal to lateral finite elements for analysis (Ref. 1).

Element forces and displacements are solved for applied loads. Concentrated loads or pressures may be applied.

2. Assumptions

The plate theory uses the thin plate assumption that material points which lie on a normal to the middle plane through a point of the middle plane before deformation, will lie on the normal to the deformed middle plane through the same point after deformation has taken place.

3. Validation

The solutions of the program have been verified to be substantially identical to the results of manual calculations. [HISTORICAL INFORMATION] [Document traceability is available at Pacific International Computing Corporation.]

4. Extent of Application

The program was used to compute stresses and deflections in the auxiliary building foundation mat due to gravity, pressure, uplift, and thermal loads.

5. Reference

1. Timoshenko, S., Woinowsky-Krieger, S., Theory of Plates and Shells, 2nd Edition, 1959, McGraw-Hill.

3D.5 BECHTEL CE 779, STRUCTURAL ANALYSIS PROGRAM (SAP)

1. Description

The program performs the static and dynamic analyses of linear, elastic, three-dimensional structures using the finite element method. The finite element library contains truss and beam elements, plane and solid elements, plate and shell elements, axisymmetric (torus) elements, and special boundary (spring) elements.

Element stresses and displacements are solved for either applied loads or temperature distributions. Concentrated loads, pressures, or gravity loads may be applied. Temperature distributions are assigned as an appropriate uniform temperature change in each element. Prestressing may be simulated by using artificial temperature changes on rod elements.

Dynamic response routines are available for solving arbitrary dynamic loads or seismic excitations, using either modal superposition or direct integration. The program can also perform response spectrum and time-history analyses.

2. Validation

The solutions to test problems have been demonstrated to be essentially identical to the results obtained using the ASKA program, [HISTORICAL INFORMATION] [which was developed by Prof. A. J. Argyris (Institut fur Statik and Dynamik, Stuttgart) and to the Chan and Fermin program. The test problem solutions have also been compared to, and found to be in agreement with, the solutions of the programs from the ASME Library of Benchmark Computer Problems and Solutions(Ref. 4). Document traceability is available at Bechtel International Corporation.]

3. Extent of Application

The program was used to perform structural analysis for concrete structures.

3D.6 BECHTEL CE 800, BECHTEL STRUCTURAL ANALYSIS PROGRAM (BSAP)

1. Description

The program performs the static and dynamic analyses of linear, elastic, three-dimensional structures using the finite element method (Ref. 1-3). The finite element library contains truss and beam elements, plane and solid elements, plate and shell elements, axisymmetric (torus) elements, and special boundary (spring) elements.

Element stresses and displacements are solved for either applied loads or temperature distributions. Concentrated loads, pressures or gravity loads maybe applied. Temperature distributions are assigned as an appropriate uniform temperature change in each element. Prestressing may be simulated by using artificial temperature changes on rod elements.

Dynamic response routines are available for solving arbitrary dynamic loads or seismic excitations, using either modal superposition or direct integration. The program can also perform response spectrum and time-history analyses.

2. Validation

The solutions to test problems have been demonstrated to be essentially identical to the results obtained using the following recognized public-domain computer programs:

• EASE	Elastic Analysis Corporation
• STARDYN	• Mechanics Research Incorporated
• MARC/CDC	• MARC Analysis Corporation
• ICES/STRUDL	• McDonnell-Douglas Automation
• ASKA	• Institut fur Statik and Dynamik, Stuttgart, Prof. A. J. Argyris

Agreement has also been established between BSAP program results and the results presented in the ASME Library of Benchmark Computer Problems and Solutions (Ref. 4) and in recognized technical journals. [HISTORICAL INFORMATION] [Document traceability is available at Bechtel Power Corporation.]

3. The program was used to perform structural analysis for concrete structures.

4. References:

1. Wilson, E. L., "SAP, A General Structural Analysis Program," University of California Structural Engineering Laboratory, Report No. UCSESM 70-20, September, 1970.

2. Wilson, E. L., "SOLID SAP - A Static Analysis Program for Three-Dimensional Solid Structures," University of California, Berkeley, Department of Civil Engineering, SESM Report No. 71-19, September, 1971.

3. Wilson, E. L., "SAP-IV-A Structural Analysis Program for Static and Dynamic Response of Linear Systems," University of California, Berkeley, EERC Report No. 73-11, June, 1973.

4. "Pressure Vessel and Piping - 1972 Computer Programs Verification," ASME Committee on Computer Technology, Pressure Vessel and Piping Division.

3D.7 BECHTEL CE 901, THE STRUCTURAL DESIGN LANGUAGE (ICES STRUDL)

1. Description

STRUDL is a structural analysis program with the capability to perform frame analysis and finite element analysis. A wide variety of loads may be accommodated by the program. The program also has the capabilities to perform dynamic analysis as well as static analysis. The STRUDL program performs both steel and concrete design and checks the applicable code in each case.

2. Assumptions

The program assumes a linear, elastic, static, small displacement analysis, member properties are required, and the program treats the joint displacements as unknowns.

3. Validation

The solutions of the program have been verified to be substantially identical to the results of the Bechtel SAP Program Version 1.9C.0.7. for beam analysis, truss analysis, and plane strain finite elements, and the results of the NASTRAN Program for flat plate finite elements.

4. Extent of Application

The program was used in the structural analysis of seismic cable tray and duct supports and pipe anchors, the exterior walls of the auxiliary and radwaste buildings, the roof over the moisture separator, and the turbine pedestal embedded trusses.

3D.8 BECHTEL CE 917, MODAL DYNAMIC ANALYSIS

1. Description

The program computes the reduced stiffness matrix from the basic geometry input for plane frame or truss models, or accepts the reduced stiffness matrix for any structure as input. It calculates mode shapes, frequencies, participation factors, and modal damping values for a lumped mass model.

Special Features:

a. Can accept either diagonal or full mass matrices.

b. Generates output tape for input to Bechtel CE 920 and Bechtel CE 931.

c. Can be used for horizontal or vertical earthquakes with minimal input changes.

2. Validation

The solutions to the program have been demonstrated to be substantially identical to the results obtained by manual calculations. [HISTORICAL INFORMATION] [Document traceability is available at Bechtel Power Corporation.]

3. Extent of Application

The program was used to obtain the mode shapes and natural frequencies of Category I structures.

3D.9 BECHTEL CE 918, RESPONSE SPECTRUM ANALYSIS

1. Description

This program is supplemental to the modal dynamic analysis program (Bechtel CE 917). It computes the modal response of general plane frame or truss models. Response spectrum technique is used, and output is expressed in terms of displacements, accelerations, support reactions, member forces and moments, and spring forces.

2. Validation

The solutions to the program have been demonstrated to be substantially identical to the results obtained by manual calculations. [HISTORICAL INFORMATION] [Document traceability is available at Bechtel Power Corporation.]

3. Extent of Application

The program was used to calculate responses of structures, acceleration, shear, moment, displacement, etc.

3D.10 BECHTEL CE 920, TIME-HISTORY ANALYSIS OF STRUCTURES

1. Description

The program performs the earthquake response time-history analysis of lumped mass models, using modal superposition. Program input consists of frequencies, mode shapes, modal damping, and the base acceleration time history.

2. Validation

The solutions to the program have been demonstrated to be substantially identical to the results obtained by manual calculations. [HISTORICAL INFORMATION] [Document traceability is available at Bechtel Power Corporation.]

3. Extent of Application

The program was used to generate the time histories at seismic Category I equipment locations in the structures.

3D.11 BECHTEL CE 921, RESPONSE SPECTRUM CALCULATIONS

1. Description

The program calculates response acceleration, velocity, and displacement spectra for a specified acceleration time history. It can produce printed plots of the calculated response spectra.

2. Validation

The solutions to the program have been demonstrated to be substantially identical to the results obtained by manual calculations. [HISTORICAL INFORMATION] [Document traceability is available at Bechtel Power Corporation.]

3. Extent of Application

The program was used to generate acceleration, velocity, and displacement spectra at Category I equipment locations and to print plots of these response spectra.

3D.12 BECHTEL CE 931, COMPOSITE DAMPING, SOIL-STRUCTURE INTERACTION

1. Description

The program computes the frequencies, mode shapes, and composite modal damping values of a lumped soil-structure system for seismic analysis. The major input to this program includes fixed base-structural frequencies, fixed base-structural damping per mode, fixed base-structural mode shapes, other fixed base-structural properties, and foundation properties.

Special features:

a. Input may consist of output tape from the Bechtel CE 917 program or direct input of fixed base structural properties.

b. Can be used for analyzing horizontal or vertical motions of earthquakes.

2. Validation

The solutions to the program have been demonstrated to be substantially identical to the results obtained by manual calculations and/or a parallel run of Bechtel CE 917, Modal Dynamic Analysis, which is an independently verified program (see Paragraph 3D.8). [HISTORICAL INFORMATION] [Document traceability is available at Bechtel Power Corporation.]

3. Extent of Application

The program was used to obtain composite modal damping ratios for soil-structure interacted dynamic primary systems.

3D.13 AXISYMMETRIC SHELL AND SOLID COMPUTER PROGRAM (ASHSD)

1. Description

The program performs the static and dynamic analyses of linear, elastic, axisymmetric structures with axisymmetric or non-axisymmetric loadings, utilizing the finite element technique. The program computes the element stresses and nodal displacements due to uniform, concentrated, or pressure loads, or temperature distributions, either over the surface area or through the wall thickness. Prestress forces may be simulated by applying the forces as equivalent concentrated temperature gradients.

2. Validation

The solutions of the program for various loadings have been demonstrated to be essentially identical to the results obtained by manual calculations, and to those obtained from accepted experimental tests or analytical results published in technical literature. (See Refs. 1 and 2.)

3. Extent of Application

The program was used in the analysis of the containment and drywell for non-axisymmetric loadings.

4. References:

1. Ghosh, S., Wilson, E. L., "Dynamic Stress Analysis of Axisymmetric Structures under Arbitrary Loading," Report No. EERC 69-10, University of California, Berkeley, September 1969, pp 69-81.

2. "Topical Report on Dynamic Analysis of Reactor Vessel Internals under Loss-of-Coolant Accident Conditions with Application of Analysis to CE 800 Mwe Class Reactors," Combustion Engineering Report CENPD-42, Combustion Engineering, Inc., Nuclear Power Department, Combustion Division, Windsor, Conn. Appendix A.

3D.14 AISC STEEL COLUMN DESIGN (CSTOR)

1. Description

The program selects steel columns from stored lists within the program to satisfy the input of axial load plus end moments about major and/or minor axes using the allowable stress provisions of the 1969 AISC Specification for the design, fabrication, and erection of structural steel for buildings. The program also selects base plates for the columns selected using the same loads and moments as input.

2. Validation

The solutions to the program have been demonstrated to be substantially identical to the results obtained by manual calculations. [HISTORICAL INFORMATION] [Document traceability is available at Bechtel Power Corporation.]

3. Extent of Application

The program was used to design steel columns in the auxiliary building.

3D.15 FINITE ELEMENT STRESS ANALYSIS (FINEL)

1. Description

The program performs the static analyses of plane or axisymmetric structures using the finite element method, in which a structure is idealized as an assemblage of finite elements. The finite elements are of either triangular or quadrilateral shape and connected at their corners (nodal points). The tributed, or inertial, or may be temperature distributions. At boundaries, displacements may be forced.

The program develops the force-displacement relationship (element stiffness matrix) for each individual element from its geometry and material properties. The element relationships are then assembled into an overall structure force-displacement relationship (structure stiffness matrix). Equilibrium equations are developed for each degree of freedom at each nodal point in terms of the structure force-displacement relationship, the unknown nodal point displacement, and the externally applied nodal point forces. Finally, these equations are solved simultaneously for the unknown nodal point displacements by a modified Gaussian elimination scheme. Once the nodal point displacements are known, element stresses are calculated.

2. Assumptions

The stress and the strain are assumed to be constant within each element.

3. Validation

The program has been verified by manual calculations. [HISTORICAL INFORMATION] [Document traceability is available at Pacific International Computing Corporation.]

4. Extent of Application

The program was used to compute stresses in the containment structure due to gravity, pressure, and thermal loads.

3D.16 BECHTEL CE 988, A COMPUTER PROGRAM FOR SEISMIC SOIL-STRUCTURE INTERACTION ANALYSIS (FLUSH)

1. Description

The program uses finite element techniques to analyze soil-structure interaction effects during earthquakes especially for embedded structures. The program provides consideration of variations of ground motion with depth in the soil-structure

a. Plain strain quadrilateral elements for modeling of soils and structures.

b. Beam elements for modeling of structures.

c. Multiple nonlinear soil properties for equivalent linear analysis.

d. An approximate 3-dimensional ability making it possible to perform meaningful structure soil-structure interaction analyses.

e. Generates output time histories of acceleration and bending moments.

f. Computation of maximum shear forces in beam elements.

g. Generates acceleration and velocity response spectra.

2. Validation

[HISTORICAL INFORMATION] [The program was developed as Report No. EERC 75-30 at the College of Engineering, University of California, Berkeley, California by J. Lysmer, T. Udaka, C.F. Tsai, and H. B. Seed (Ref. 1)].

The program is a recognized program in the public domain and has had sufficient history of use to justify its applicability and validity without further demonstration.

3. Extent of Application

The program was used to develop the floor response spectra for the standby service water cooling tower basin.

4. Reference:

1. Report No. EERC 75-30 at the College of Engineering, University of California, Berkeley, California by J. Lymer, T. Udaka, C. F. Tsia, and H. B. Seed

3D.17 THE STRUCTURAL DESIGN LANGUAGE (ICES-STRUDL BY McDONNELL-DOUGLAS AUTOMATION VERSION)

1. Description

The program performs structural analysis. Frame members can be used in conjunction with finite elements. Some special features include a built-in table for rolled steel wide flange shapes, a member forced concrete member design and checking capability, and a dynamic analysis capability.

2. Validation

The program has been verified [HISTORICAL INFORMATION] [and document traceability is available at McDonnell-Douglas Automation.]

3. Extent of Application

The program was used to perform structural analysis for both steel and concrete structures.

3D.18 IMPEDANCE FUNCTIONS FOR A RIGID CIRCULAR FOUNDATION ON A LAYERED VISCOELASTIC MEDIUM (LUCON)

1. Description

LUCON is a program developed to evaluate the impedance functions for a rigid circular foundation placed on a layered viscoelastic medium. The program computes the vertical, rocking, and horizontal impedance functions and their reciprocals, the compliance functions, for any given set of frequencies with site characteristics, and the foundation geometry. The foundation medium may be layered or may be a uniform elastic half-space. The two types of material damping in the soil are constant hysteretic- type damping and Voigt-type damping. The type of damping must be the same for all layers, but the values of the damping constants may differ from layer to layer.

2. Validation

The solutions of the program for various loadings have been demonstrated to be essentially identical to analytical results published in technical literature. (See Refs. 1 through 5.)

3. Extent of Application

The program was used to compute the impedance functions for the diesel generator building foundation supported by a two-layered soil medium.

4. References:

1. Veletsos, A. S., and Verbic, B., "Vibration of Viscoelastic Foundations," Report No. 18, Dept of Civil Engineering, Rice University, Houston, Texas, April 1973.

2. Shah, P. M., "On the Dynamic Response of Foundation Systems," PH.D. Thesis, Rice University, Houston, Texas, 1968.

3. Veletsos, A. S., and Wei, Y.T., "Lateral and Rocking Vibration of Footings," Journal of the Soil Mechanics and Foundations Division, ASCE, Vol. 97,1971.

4. Luco, J. E. and Westmann, R. A., "Dynamic Response of Circular Footings," Journal of the Engineering Mechanics Division, ASCE, Vol. 97, 1971.

5. Luco, J. E., "Impedance Functions for a Rigid Foundation on a Layered Medium," Nuclear Engineering and Design, 1974.

3D.19 BECHTEL CE 915, A COMPUTER PROGRAM FOR EARTHQUAKE RESPONSE ANALYSIS OF HORIZONTALLY LAYERED SITES (SHAKE)

1. Description

The program computes the responses in a system of homogeneous, viscoelastic layers of infinite horizontal extent subjected to vertically travelling shear waves. The nonlinearity of the shear modulus and damping is accounted for by the use of equivalent, linear, soil properties using an iterative procedure to obtain valves for modulus and damping compatible with the effective strains in each layer. The program handles systems with variation in both moduli and damping and takes into account the effect of the elastic base. The motion used as a basis for analysis, the object motion, can be given in any one layer in the system, and new motions can be computed in any other layer.

2. Validation

[HISTORICAL INFORMATION] [The program was developed as Report No. EERC 72-12 at the College of Engineering, University of California, Berkeley, California by P. B. Schnabel, J. Lysmer, and H. B. Seed (Ref. 1)].

3. Extent of Application

The program was used to develop the base rock time history. The control motion (free field time motion) is input into the program, and the output generated is the base rock time history. This base rock time history is then input into the

FLUSH program to develop the floor response spectra for the standby service water cooling tower basin.

4. Reference:

1. Report No. EERC 72-12 at the College of Engineering, University of California, Berkeley, California by P. B. Schnabel, J . Lymer, and H. B. Seed

3D.20 STATIC AND DYNAMIC STRUCTURAL ANALYSIS SYSTEMS (MRI/ STARDYNE)

1. Description

The STARDYNE (Ref.1) system utilizes the finite element approach to analyze both static and dynamic systems. The output for these analyses include member deformations and member loads/ stresses. The input for the system includes thermal loads, displacements and nodal applied loads for static systems, and transient, steady-state harmonic, random, and shock spectra excitation.

2. Validation

The program [HISTORICAL INFORMATION] [is available from the Control Data Corporation and] is a recognized program in the public domain which has had sufficient history of use to justify its applicability and validity without further demonstration.

3. Extent of Application

The program was used for the creation of the stiffness matrix for the turbine pedestal which was used for input into the Modal Dynamic Analysis Program (Bechtel CE 917). The program was used for the seismic analysis of the turbine buildings. The program was also used for the tornado analysis and eigenvalves solution of the enclosure building.

4. Reference

1. User Information Manual, MRI/STARDYNE, Static and Dynamic Structural Analysis Systems, Cybernet Services, Control Data Corporation, copyright 1976.

3D.21 DOCUMENT M-43, STRESS ANALYSIS OF SEISMIC DUCTS

1. Description

The following programs are included:

a. FREQUENCY*DUCT.KHAN

b. STRESS*DUCT.KHAN

The first program calculates the frequencies of the rectangular (or circular) duct in the transverse and longitudinal directions, section properties of the duct, and effective uniformly distributed load of the duct. The output from this program, seismic accelerations and other required information, are input for the second program. This stress program gives all the critical stresses, including stresses in the stiffeners, and analysis for buckling. The program also assists in the location of duct supports since the duct frequencies are restricted by an upper bound frequency.

[HISTORICAL INFORMATION] [The programs were developed by Bechtel Power Corporation for seismic duct design.]

2. Validation

The solutions to the program have been demonstrated to be substantially identical to the results obtained by manual calculations. [HISTORICAL INFORMATION] [Document traceability is available at Bechtel Power Corporation.]

3. Extent of Application

The programs were used to design seismic Category I ducts and to design and locate duct supports.

3D.22 OTHER COMPUTER PROGRAMS USED IN STRUCTURAL ANALYSIS

In the course of generating structural design calculations, several programs were used to conserve design efforts. These programs were limited in scope and were developed solely to assist the designer in making lengthy, repetitious calculations, thereby saving design efforts. The programs were validated by generating example problems or performing manual design checks. These validations were incorporated into the project design calculation books. These programs are not itemized here due to their simplicity and nature of use.

APPENDIX 3E PRESSURE ANALYSIS FOR SUBCOMPARTMENTS OUTSIDE CONTAINMENT

3E.1 DESIGN BASIS

a. A pressure response analysis was performed for subcompartments outside the containment which contain high-energy piping, the failure of which could endanger essential systems, components, and equipment.

   The definition of high energy and the criteria for postulating breaks is outlined in subsection 3.6A.2.

   The break which produced the greatest blowdown mass and enthalpy during normal operation or hot standby was selected for the design.

b. All break locations were considered to be fully double-ended in the case of circumferential breaks and no credit for limiting blowdown generation due to pipe restraint locations was taken.

   The effective cross-sectional flow area of the pipe was used in the jet discharge evaluation for circumferential and longitudinal breaks.

3E.2 DESIGN FEATURES

Figures 3.6A-23 through 3.6A-30 show the piping and equipment in the compartments. Figures 3.6A-31 and 3.6A-32 specifically show the blowout panels and shafts. The blowout panels were assumed to open instantaneously at 1 psid in the pressure analysis calculations, however, the panels are actually designed to open at 1/2 psid. In addition, the blowout panel openings used in the pressure calculations were slightly smaller than the final design dimensions shown on Figures 3.6A-31 and -32.

The analytical treatment of blowout panels is described in Reference 1.

3E.3 DESIGN EVALUATION

The input data and results for the pressure response analysis are provided in the tables and figures listed on the following pages for the respective subcompartments.

Subcompartment	Break	Tables	Figures
RCIC pump room (El. 93'-0")	6-inch steam to RCIC 12-inch RHR to FW* (in piping room above)	3E-1,2,3,4	3E-1,2,3,4,5
Piping penetration room above RCIC room (El. 119'-0")	12-inch RHR to FW	3E-3,4	3E-4,5,6,7
RHR pump rooms A & B** (El. 93'-0")	12-inch RHR to FW	3E-5,6	3E-8,9,10
RHR heat exchanger** Rooms A & B (El. 93'-0")	12-inch RHR to FW (in RHR pump room)	3E-5,6	3E-8,9,10
RWCU pump rooms A & B (El. 115'-0")	4-inch pump suction 12-inch RHR to FW in pipe penetration room above RCIC room*	3E-9,10	3E-14,15
RWCU pipe chase	8-inch pump suction 28-inch main steam* in connecting steam tunnel	3E-11,12,15,16	3E-16,17,18,22 23,24
RCIC pipe chase	6-inch steam to RCIC 28-inch main* steam in connecting steam	3E-13,14,15,16	3E-19,20,21,22, 23,24

Subcompartment	Break	Tables	Figures
RCIC pump room (El. 93'-0")	6-inch steam to RCIC 12-inch RHR to FW* (in piping room above)	3E-1,2,3,4	3E-1,2,3,4,5
Piping penetration room above RCIC room (El. 119'-0")	12-inch RHR to FW	3E-3,4	3E-4,5,6,7
RHR pump rooms A & B** (El. 93'-0")	12-inch RHR to FW	3E-5,6	3E-8,9,10
RHR heat exchanger** Rooms A & B (El. 93'-0")	12-inch RHR to FW (in RHR pump room)	3E-5,6	3E-8,9,10
RWCU pump rooms A & B (El. 115'-0")	4-inch pump suction 12-inch RHR to FW in pipe penetration room above RCIC room*	3E-9,10	3E-14,15
RWCU pipe chase	8-inch pump suction 28-inch main steam* in connecting steam tunnel	3E-11,12,15,16	3E-16,17,18,22 23,24
RCIC pipe chase	6-inch steam to RCIC 28-inch main* steam in connecting steam tunnel	3E-13,14,15,16	3E-19,20,21,22, 23,24
Main steam tunnel in auxiliary bldg.	28-inch main steam (slot break)	3E-15,16	3E-22,23,24

Subcompartment	Break	Tables	Figures
Main steam blowout shaft	28-inch main steam (slot break)	3E-15,16	3E-22,23,24
Blowout shafts from** RHR, RCIC, RWCU	12-inch RHR to FW in pipe penetration room above RCIC room*	3E-3,4	3E-4,5,6,7
Main steam tunnel in turbine bldg.	28-inch main steam (circumferential break)	3E-17,18	3E-25,26,27

* Worst break.

** Since these compartments are symmetrical, the design pressure of both compartments was based on the analysis performed for one of the compartments.

The information contained in the tables and figures is discussed in the following paragraphs.

a. A complete description of the methods used in the Bechtel computer program COPDA is reported in Bechtel Topical Report, BN-TOP-4, Rev. 1, which was filed with the NRC in October, 1977 (Ref. 1).

   The blowdown mass fluxes and enthalpies are based on the Calculated Moody Blowdown Model. This model is applicable to liquid blowdown at an enthalpy lower than the saturated liquid enthalpy (Ref. 4).

   Descriptions of flow correlations, treatment of air/steam mixtures, and other pertinent calculational information will be found in BN-TOP-4, Rev. 1 (Ref. 1).

   Figures 3E-1, 4, 8, 11, 14, 16, 19, 22, and 26 provide schematic flow diagrams for the systems with postulated breaks.

b. The room initial conditions are given in Tables 3E-1, 3, 5, 7, 9, 11, 13, 15, and 17.

c. A description and justification of the subsonic and sonic flow models and degree of entrainment used in vent flow calculations are given in BN-TOP-4, Rev. 0 (Ref. 1).

d. Break locations for high-energy line ruptures are given in Figures 3.6A-1A through 3.6A-13I. Break areas for postulated line ruptures are given in Tables 3E-1, 3, 5, 7, 9, 11, 13, 15, and 17. Room volumes and vent areas are also presented in Tables 3E-1 through 3E-18.

e. Tables 3E-2, 4, 6, 8, 10, 12, 14, 16, and 18, which tabulate the compartment nodalization schemes, were selected to maximize differential pressures across node boundaries. Restrictions resulting from structural components on equipment placement were selected as node boundaries for the flow model. The differential pressure transients resulting from these restrictions were used to determine the adequacy of structural and component support design.

f. Graphs of the pressure responses of all subnodes as functions of time are given in Figures 3E-2, 3, 5, 6, 7, 9, 10, 12, 13, 15, 17, 18, 20, 21, 23, 24, and 27.

g. The mass and energy release data for the subcompartment analyses are given in Tables 3E-19 through 3E-25.

h. Flow remains subcritical between compartments throughout the transients as indicated in Tables 3E-2, 4, 6, 8, 10, 12, 14, 16, and 18.

i. The flow coefficient (C) for a particular geometry is determined as a function of the equivalent head loss coefficient and is expressed as:

$$C = \frac{1}{\sqrt{K_{eff}}}$$

The value of K_{eff} is simply the sum of the head losses for separate parts of the system. For the analyses performed, these head losses are defined as follows:

Entrance loss or contraction - This loss is determined as a function of the ratio of the upstream cross-sectional area to the cross-sectional area of the contraction.

Exit loss or expansion - This loss is determined as a function of the ratio of the cross-sectional area upstream of the expansion to the cross-sectional area downstream of the expansion. For the final expansion into the downstream compartment, a conservative expansion loss coefficient of 1.0 is generally assumed.

Friction losses - These, although generally very small, are calculated as an fl/d term.

The losses listed above are defined specifically in References 1, 2, and 3. Values of components used in the analyses are listed in Figures 3E-2, 4, 6, 8, 10, 12, 14, 16, and 18.

For a more complete description of flow coefficient determination, see Ref. 1.

3E.4 REFERENCES

1. BN-TOP-4, Rev. 1, "Subcompartment Pressure and Temperature Transient Analysis," October 1977.

2. I. E. Idel Chik, "Handbook of Hydraulic Resistance Coefficients of Local Resistance and Friction," USAEC-TR-6630, 1966.

3. Crane Co., "Flow of Fluids," Technical Paper No. 410, 1969, Engineering Division, New York, N.Y.

4. Moody, F.J., "Maximum Flow Rate of a Single Component, Two-Phase Mixture," Journal of Heat Transfer, ASME Series C, Vol. 87, p. 134, February 1965.

TABLE 3E-1: SUBCOMPARTMENT NODAL DESCRIPTION

6" DBB-57 (MS to RCIC) Line Break in RCIC Pump Room

VOLUME NO.	VOLUME (ft^3)	INITIAL CONDITIONS			BREAK CONDITIONS				CALCULATED PEAK PRESSURE (PSIG)	DESIGN PEAK PRESS. DIFF. (PSID)
		TEMP. (F)	PRESS. (PSIA)	HUMID (%)	BREAK LOC. VOL. NO.	BREAK LINE	BREAK AREA (ft^2)	BREAK TYPE		
1	117293.0	105.0	14.7	90	1	6" DBB-57 (MS to RCIC)	0.165	Circumferential	1.7	4.02
2	10034.0	105.0	14.7	90					1.6	4.92
3	4846.0	105.0	14.7	90					1.4	3.02
4	920.0	105.0	14.7	90					0.2	1.0
5	313.0	105.0	14.7	90					1.6	5.12
6	308.0	105.0	14.7	90					1.1	5.12
7	41549.0	125.0	14.7	90					0.6	15.39
8	1258.0	105.0	14.7	90					0.6	15.4
9	14287.0	125.0	14.7	90					0.6	11.0
10	6348.0	105.0	14.7	90					0	3.2
11	Atmos.	105.0	14.7	90					--	--

TABLE 3E-2: SUBCOMPARTMENT VENT PATH DESCRIPTION

6" DBB-57 (MS to RCIC) Line Break in RCIC Pump Room

FROM VOL. NODE NO.	TO VOL. NODE NO.	DESCRIPTION OF VENT PATH FLOW CHOKED/UNCHOKED	AREA (ft^2)	L/A	HEAD LOSS (K)			
					FRICTION K (Fl/d)	EXPANSION (K)	CONTRACTION (K)	TOTAL
1	2	unchoked	68.0	0.058	----------orifice----------			
1	3	unchoked	16.0	1.172	0.01	1.0	0.43	1.44
1	5	unchoked	20.6	0.193	0.021	1.0	0.43	1.451
2	3	unchoked	28.0	0.944	0.0	1.0	0.42	1.42
3	4	unchoked	42.5	1.206	0.006	1.0	0.41	1.416
4	10	unchoked	36.0	0.163	0.007	1.0	0.40	1.407
4	11	unchoked	75.0	0.105	0.005	1.0	0.02	1.025
5	6	unchoked	14.4	2.535	0.10	1.0	0.32	1.42
6	7	unchoked	8.75	2.437	0.16	1.0	---	1.16
7	8	unchoked	70.0	0.190	0.029	1.0	0.42	1.449
7	9	unchoked	382.0	0.060	0.037	1.0	0.39	1.427
9	10	unchoked	352.0	0.075	0.003	1.0	0.34	1.343
10	11	unchoked	517.5	0.015	0.002	1.0	0.02	1.022

TABLE 3E-3: SUBCOMPARTMENT NODAL DESCRIPTION

12" DBB-68 (RHR "B" to Feedwater) Line Break in Penetration Room or 12" DBB-72 (RHR "A" to Feedwater) - Similar due to Symmetry

VOLUME NO.	VOLUME (ft^3)	INITIAL CONDITIONS			BREAK CONDITIONS				CALCULATED PEAK PRESS. (PSIG)	DESIGN PEAK PRESS. DIFF. (PSID)
		TEMP. (F)	PRESS. (PSIA)	HUMID. (%)	BREAK LOC. VOL. NO.	BREAK LINE	BREAK AREA (ft^2)	BREAK TYPE		
1	10034.0	105.0	14.7	90	1	12"DBB-68 (RHR"B" to FW)	0.668	circumfer-ential	8.06	8.06
2	17293.0	105.0	14.7	90					6.73	6.73
3	1546.0	105.0	14.7	90		OR			0	--
4	4846.0	105.0	14.7	90		12"DBB-72 (RHR"A" to FW)			4.82	4.82
5	313.0	105.0	14.7	90					6.10	6.10
6	308.0	105.0	14.7	90					4.72	4.72
7	41549.0	125.0	14.7	90					1.02	1.02
8	1258.0	105.0	14.7	90					1.03	1.03
9	14287.0	125.0	14.7	90					1.00	1.00
10	6348.0	105.0	14.7	90					.31	.31
11	1942.0	105.0	14.7	90					0	0
12	4846.0	105.0	14.7	90					4.82	4.82
13	920.0	105.0	14.7	90					.90	.90
14	920.0	105.0	14.7	90					.90	.90
15	Atmos.	105.0	14.7	90					0	--

TABLE 3E-4: SUBCOMPARTMENT VENT PATH DESCRIPTION

12" DBB-68 (RHR "B" to Feedwater) Line Break in Penetration Room or 12" DBB-72 (RHR "A" to Feedwater) - Similar due to Symmetry

FROM VOL. NODE NO.	TO VOL. NODE NO.	DESCRIPTION OF VENT PATH FLOW CHOKED/ UNCHOKED	AREA (ft^2)	L/A	HEAD LOSS (K)			
					FRICTION K (Fl/d)	EXPANSION (K)	CONTRACTION (K)	TOTAL
1	2	unchoked	68.0	0.058	orifice -			
1	3	unchoked	21.75	0.504	0.009	1.0	0.43	1.439
1	4	unchoked	28.0	0.944	---	1.0	0.42	1.42
1	8	unchoked	21.75	0.171	0.009	1.0	0.43	1.439
1	11	unchoked	21.75	0.503	0.009	1.0	0.43	1.439
1	12	unchoked	28.0	0.944	---	1.0	0.42	1.42
2	4	unchoked	16.0	1.172	0.01	1.0	0.43	1.44
2	5	unchoked	20.6	0.193	0.021	1.0	0.43	1.451
2	12	unchoked	16.0	1.172	0.01	1.0	0.43	1.44
3	4	unchoked	28.0	1.050	---	1.0	0.40	1.40
4	14	unchoked	42.5	1.206	0.006	1.0	0.41	1.416
5	6	unchoked	14.4	2.535	0.10	1.0	0.32	1.42
6	7	unchoked	8.75	2.437	0.16	1.0	--	1.16
7	8	unchoked	70.0	0.190	0.029	1.0	0.42	1.449
7	9	unchoked	382.0	0.060	0.037	1.0	0.39	1.427
9	10	unchoked	352.0	0.075	0.003	1.0	0.34	1.343
10	13	unchoked	36.0	0.163	0.007	1.0	0.40	1.407
10	14	unchoked	36.0	0.163	0.007	1.0	0.40	1.407
10	15	unchoked	517.5	0.015	0.002	1.0	0.02	1.022
11	12	unchoked	28.0	1.033	---	1.0	0.40	1.40
12	13	unchoked	42.5	1.206	0.006	1.0	0.41	1.146

TABLE 3E-4: SUBCOMPARTMENT VENT PATH DESCRIPTION (CONTINUED)

12" DBB-68 (RHR "B" to Feedwater) Line Break in Penetration Room or 12" DBB-72 (RHR "A" to Feedwater) - Similar due to Symmetry

FROM VOL. NODE NO.	TO VOL. NODE NO.	DESCRIPTION OF VENT PATH FLOW CHOKED/ UNCHOKED	AREA (ft^2)	L/A	HEAD LOSS (K)			
					FRICTION K (Fl/d)	EXPANSION (K)	CONTRACTION (K)	TOTAL
13	15	unchoked	75.0	0.105	0.005	1.0	0.02	1.025
14	15	unchoked	75.0	0.105	0.005	1.0	0.02	1.025

TABLE 3E-5: SUBCOMPARTMENT NODAL DESCRIPTION

**12" DBB-68 (RHR "B" to Feedwater) Line Break in RHR Pump Room "B"
or 12" DBB-72 (RHR "A" to Feedwater) Line Break in RHR Pump Room "A" - Similar due to Symmetry**

VOLUME NO.	VOLUME (ft^3)	INITIAL CONDITIONS			BREAK CONDITIONS				CALCULATED PEAK PRESS. (PSIG)	DESIGN PEAK PRESS. DIFF. (PSID)
		TEMP. (F)	PRESS. (PSIA)	HUMID. (%)	BREAK LOC. VOL. NO.	BREAK LINE	BREAK AREA (ft^2)	BREAK TYPE		
1	13114.0	105.0	14.7	90	1	12"DBB-68 (RHR"B" to FW) OR 12"DBB-72 (RHR"A" to FW)	0.668	circumfer-ential	3.36	3.36
2	19189.0	105.0	14.7	90					3.31	3.31
3	4846.0	105.0	14.7	90					1.42	3.02
4	4750.0	105.0	14.7	90					2.19	2.19
5	3547.0	105.0	14.7	90					2.10	2.10
6	7329.0	105.0	14.7	90					1.88	1.88
7	9653.0	105.0	14.7	90					1.50	1.65
8	15088.0	105.0	14.7	90					1.71	3.04
9	6512.0	105.0	14.7	90					1.06	1.06
10	920.0	105.0	14.7	90					0.2	1.0
11	6348.0	105.0	14.7	90					0	3.2
12	Atmos.	105.0	14.7	90					--	--

TABLE 3E-6: SUBCOMPARTMENT VENT PATH DESCRIPTION

**12" DBB-68 (RHR "B" to Feedwater) Line Break in RHR Pump Room "B"
or 12" DBB-72 (RHR "A" to Feedwater) Line Break in RHR Pump Room "A" - Similar due to Symmetry**

FROM VOL. NODE NO.	TO VOL. NODE NO.	DESCRIPTION OF VENT PATH FLOW CHOKED/ UNCHOKED	AREA (ft^2)	L/A	HEAD LOSS (K)			
					FRICTION K (Fl/d)	EXPANSION (K)	CONTRACTION (K)	TOTAL
1	2	unchoked	657.8	0.030	orifice -			
1	6	unchoked	54.0	0.208	0.008	1.0	0.41	1.418
2	3	unchoked	32.0	0.647	0.01	1.0	0.43	1.44
2	4	unchoked	45.3	0.254	0.008	1.0	0.41	1.418
2	5	unchoked	32.7	0.351	0.008	1.0	0.41	1.418
3	10	unchoked	42.5	1.206	0.006	1.0	0.41	1.416
4	5	unchoked	209.3	0.084	orifice -			
5	6	unchoked	251.1	0.098	orifice -			
6	7	unchoked	341.5	0.060	orifice -			
7	8	unchoked	75.0	0.149	0.007	1.0	0.40	1.407
7	9	unchoked	176.0	0.137	0.028	1.0	0.32	1.348
9	12	unchoked	184.0	0.116	0.003	1.0	0.38	1.383
10	11	unchoked	36.0	0.163	0.007	1.0	0.40	1.407
10	12	unchoked	75.0	0.105	0.005	1.0	0.02	1.025
11	12	unchoked	517.5	0.015	0.002	1.0	0.02	1.022

TABLE 3E-7: DELETED

TABLE 3E-8: DELETED

TABLE 3E-9: SUBCOMPARTMENT NODAL DESCRIPTION

4" DBZ-1 (RWCU Pump Suction) Line Break in RWCU Pump Room

VOLUME NO.	VOLUME (ft^3)	INITIAL CONDITIONS			BREAK CONDITIONS				CALCULATED PEAK PRESS. (PSIG)	DESIGN PEAK PRESS. DIFF. (PSID)
		TEMP. (F)	PRESS. (PSIA)	HUMID. (%)	BREAK LOC. VOL. NO.	BREAK LINE	BREAK AREA (ft^2)	BREAK TYPE		
1	1546.0	105.0	14.7	90	1	4" DBZ-1 (RWCU Pump Suction)	0.160*	circumfer-ential	2.65	4.91
2	4846.0	105.0	14.7	90					1.7	3.02
3	920.0	105.0	14.7	90					.2	1.0
4	6348.0	105.0	14.7	90					0	3.2
5	Atmos.	105.0	14.7	90					---	---

*Flow from upstream side of break is restricted by flow limiter in 6" line (flow area = 0.028 ft^2)

TABLE 3E-10: SUBCOMPARTMENT VENT PATH DESCRIPTION

4" DBZ-1 (RWCU Pump Suction) Line Break in RWCU Pump Room

FROM VOL. NODE NO.	TO VOL. NODE NO.	DESCRIPTION OF VENT PATH FLOW CHOKED/ UNCHOKED	AREA (ft^2)	L/A	HEAD LOSS (K)			
					FRICTION K (Fl/d)	EXPANSION (K)	CONTRACTION (K)	TOTAL
1	2	unchoked	28.0	1.050	---	1.0	0.40	1.40
2	3	unchoked	42.5	1.206	0.006	1.0	0.41	1.416
3	4	unchoked	36.0	0.163	0.007	1.0	0.40	1.407
3	5	unchoked	75.0	0.105	0.005	1.0	0.02	1.025
4	5	unchoked	517.5	0.015	0.002	1.0	0.02	1.022

TABLE 3E-11: SUBCOMPARTMENT NODAL DESCRIPTION

8" DBC-1 (RWCU Pump Suction) Line Break in RWCU Pipe Chase

VOLUME NO.	VOLUME (ft^3)	INITIAL CONDITIONS			BREAK CONDITIONS				CALCULATED PEAK PRESS. (PSIG)	DESIGN PEAK PRESS. DIFF. (PSID)
		TEMP. (F)	PRESS. (PSIA)	HUMID (%)	BREAK LOC. VOL. NO.	BREAK LINE	BREAK AREA (FT^2)	BREAK TYPE		
1	1258.0	105.0	14.7	90	1	8" DBC-1(RWCU Pump Suction)	0.634*	Circumferential	1.4	15.40
2	41857.0	125.0	14.7	90					1.0	15.39
3	17606.0	105.0	14.7	90					.5	4.02
4	10034.0	105.0	14.7	90					.5	4.92
5	4846.0	105.0	14.7	90					0	3.02
6	920.0	105.0	14.7	90					0	1.0
7	14287.0	125.0	14.7	90					1.0	11.0
8	6348.0	105.0	14.7	90					.3	3.2
9	Atmos.	105.0	14.7	90					---	---

*Flow from upstream side of break is restricted by flow limiter in 6" line (flow area = 0.028 ft^2).

Flow from downstream side is restricted by flow limiter in 4" line (flow area = 0.02 ft^2)

TABLE 3E-12: SUBCOMPARTMENT VENT PATH DESCRIPTION

8" DBC-1 (RWCU Pump Suction) Line Break in RWCU Pipe Chase

FROM VOL. NODE NO.	TO VOL. NODE NO.	DESCRIPTION OF VENT PATH FLOW CHOKED/ UNCHOKED	AREA (ft^2)	L/A	HEAD LOSS (K)			
					FRICTION K (Fl/d)	EXPANSION (K)	CONTRACTION (K)	TOTAL
1	2	unchoked	70.0	0.190	0.029	1.0	---	1.029
2	3	unchoked	8.75	5.16	0.28	1.0	1.18	2.46
2	7	unchoked	382.0	0.060	0.037	1.0	0.39	1.427
3	4	unchoked	68.0	0.058	----------orifice-			----------
3	5	unchoked	16.0	1.172	0.01	1.0	0.43	1.44
4	5	unchoked	28.0	0.944	---	1.0	0.42	1.42
5	6	unchoked	42.5	1.206	0.006	1.0	0.41	4.416
6	8	unchoked	36.0	0.163	0.007	1.0	0.40	1.407
6	9	unchoked	75.0	0.105	0.005	1.0	0.02	1.025
7	8	unchoked	352.0	0.075	0.003	1.0	0.34	1.343
8	9	unchoked	517.5	0.015	0.002	1.0	0.02	1.022

TABLE 3E-13: SUBCOMPARTMENT NODAL DESCRIPTION

8" DBC-1 (RWCU Pump Suction) Line Break in RWCU Pipe Chase

VOLUME NO.	VOLUME (ft^3)	INITIAL CONDITIONS			BREAK CONDITIONS				CALCULATED PEAK PRESS. (PSIG)	DESIGN PEAK PRESS. DIFF. (PSID)
		TEMP. (F)	PRESS. (PSIA)	HUMID (%)	BREAK LOC. VOL. NO.	BREAK LINE	BREAK AREA (ft^2)	BREAK TYPE		
1	308.0	105.0	14.7	90	1	6" DBB-57(MS to RCIC)	0.165	Circumferential	10.2	10.2
2	313.0	105.0	14.7	90					2.0	5.12
3	17293.0	105.0	14.7	90					1.3	4.02
4	10034.0	105.0	14.7	90					1.3	4.92
5	4846.0	105.0	14.7	90					1.2	3.02
6	920.0	105.0	14.7	90					.2	1.0
7	41549.0	125.0	14.7	90					1.1	15.39
8	1258.0	105.0	14.7	90					1.1	15.40
9	14287.0	125.0	14.7	90					1.0	11.00
10	6348.0	105.0	14.7	90					.9	3.2
11	Atmos.	105.0	14.7	90					---	---

TABLE 3E-14: SUBCOMPARTMENT VENT PATH DESCRIPTION

6" DBC-57 (MS to RCIC) Line Break in RCIC Pipe Chase

FROM VOL. NODE NO.	TO VOL. NODE NO.	DESCRIPTION OF VENT PATH FLOW CHOKED/ UNCHOKED	AREA (ft^2)	L/A	HEAD LOSS (K)			
					FRICTION K (Fl/d)	EXPANSION (K)	CONTRACTION (K)	TOTAL
1	2	unchoked	14.4	2.535	0.10	1.0	---	1.10
2	7	unchoked	8.75	2.437	0.16	1.0	---	1.16
2	3	unchoked	20.6	0.193	0.021	1.0	0.20	1.221
3	4	unchoked	68.0	0.058	orifice-			
3	5	unchoked	16.0	1.172	0.01	1.0	0.43	1.44
4	5	unchoked	28.0	0.944	---	1.0	0.42	1.42
5	6	unchoked	42.5	1.206	0.006	1.0	0.41	1.416
6	10	unchoked	36.0	0.163	0.007	1.0	0.40	1.407
6	11	unchoked	75.0	0.105	0.005	1.0	0.02	1.025
7	8	unchoked	70.0	0.190	0.029	1.0	0.42	1.449
8	9	unchoked	832.0	0.060	0.037	1.0	0.39	1.427
9	10	unchoked	352.0	0.075	0.003	1.0	0.34	1.343
10	11	unchoked	517.5	0.015	0.002	1.0	0.02	1.022

TABLE 3E-15: SUBCOMPARTMENT NODAL DESCRIPTION

28" Main Steam Line Slot Break in Auxiliary Building Steam Tunnel

VOLUME NO.	VOLUME (ft^3)	INITIAL CONDITIONS			BREAK CONDITIONS				CALCULATED PEAK PRESS. (PSIG)	DESIGN PEAK PRESS. DIFF. (PSID)
		TEMP. (F)	PRESS. (PSIA)	HUMID (%)	BREAK LOC. VOL. NO.	BREAK LINE	BREAK AREA (ft^2)	BREAK TYPE		
1	41549.0	125.0	14.7	90	1	28" DBD-56(Main Steam)	3.45	longitud-inal	15.25	15.39
2	1258.0	105.0	14.7	90					15.49	15.49
3	308.0	105.0	14.7	90					5.12	5.12
4	313.0	105.0	14.7	90					2.42	5.12
5	17293.0	105.0	14.7	90					1.40	4.02
6	10034.0	105.0	14.7	90					1.36	4.92
7	4846.0	105.0	14.7	90					1.24	3.02
8	920.0	105.0	14.7	90					0.21	1.0
9	14287.0	125.0	14.7	90					10.86	11.0
10	6348.0	105.0	14.7	90					3.08	3.2
11	Atmos.	105.0	14.7	90					---	---

TABLE 3E-16: SUBCOMPARTMENT VENT PATH DESCRIPTION

28" Main Steam Line Break in Auxiliary Building Steam Tunnel

FROM VOL. NODE NO.	TO VOL. NODE NO.	DESCRIPTION OF VENT PATH FLOW CHOKED/ UNCHOKED	AREA (ft^2)	L/A	HEAD LOSS (K)			
					FRICTION K (Fl/d)	EXPANSION (K)	CONTRACTION (K)	TOTAL
1	2	unchoked	70.0	0.192	0.029	1.0	0.42	1.449
1	3	unchoked	8.75	2.499	0.16	1.0	0.43	1.59
1	9	unchoked	382.0	0.060	0.037	1.0	0.39	1.427
3	4	unchoked	14.4	2.535	0.10	1.0	---	1.10
4	5	unchoked	20.6	0.193	0.021	1.0	0.20	1.221
5	6	unchoked	68.0	0.058	----------orifice-			----------
5	7	unchoked	16.0	1.172	0.01	1.0	0.43	1.44
6	7	unchoked	28.0	0.944	---	1.0	0.42	1.42
7	8	unchoked	42.5	1.206	0.006	1.0	0.41	1.416
8	10	unchoked	36.0	0.163	0.007	1.0	0.40	1.407
8	11	unchoked	75.0	0.105	0.005	1.0	0.02	1.025
9	10	unchoked	352.0	0.075	0.003	1.0	0.34	1.343
10	11	unchoked	517.5	0.015	0.002	1.0	0.02	1.022

TABLE 3E-17: SUBCOMPARTMENT NODAL DESCRIPTION

28" Main Steam Line Slot Break in Turbine Building Steam Tunnel

VOLUME NO.	VOLUME (FT^3)	INITIAL CONDITIONS			BREAK CONDITIONS				CALCULATED PEAK PRESS. (PSIG)	DESIGN PEAK PRESS. DIFF. (PSID)
		TEMP. (F)	PRESS. (PSIA)	HUMID (%)	BREAK LOC. VOL. NO.	BREAK LINE	BREAK AREA (ft^2)	BREAK TYPE		
1	$3.09x10^4$	154.0	14.7	90	1	28" DBD-56(Main Steam)	3.45	circum-ferential	6.3	5*
2	$1.78x10^4$	154.0	14.7	90					1.65	
3	$3.92x10^4$	154.0	14.7	90					3.5	
4	$3.99x10^4$	154.0	14.7	90					1.47	
5	$1.99x10^4$	154.0	14.7	90					0.6	
6	$6x10^6$	154.0	14.7	90					---	

*The design peak applies to CMU walls in steam tunnel only.
See Table 3E-15 for auxiliary building turbine building steam tunnel interface wall design pressure differential.
The remaining subcompartments are nonsafety-related.

TABLE 3E-18: SUBCOMPARTMENT VENT PATH DESCRIPTION

28" Main Steam Line Break in Turbine Building Steam Tunnel

FROM VOL. NODE NO.	TO VOL. NODE NO.	DESCRIPTION OF VENT PATH FLOW CHOKED/ UNCHOKED	AREA (FT^2)	L/A	HEAD LOSS (K)			
					FRICTION K (FL/D)	EXPANSION (K)	CONTRACTION (K)	TOTAL
1	2	unchoked	399.21	0.048	0.027	1.0	0.38	1.407
1	3	unchoked	805.13	0.022	0.024	1.0	0.07	1.094
1	4	unchoked	259.9	0.246	0.13	1.0	0.07	1.20
2	6	unchoked	582.59	0.0083	----------orifice----------			
2	6	unchoked	771.77		0.02	1.0	---	1.02
3	6	unchoked	861.62	0.016	----------orifice----------			
4	5	unchoked	21.0	1.401	---	1.0	0.43	1.43
4	6	unchoked	549.84	0.066	0.02	1.0	---	1.02

TABLE 3E-19: BLOWDOWN

6" DBB-57 (MS to RCIC) Line Breaks in RCIC Pump Room or Pipe Chase*

Time (sec)	Lbs mass/sec	Btu/lb mass	Btu/sec
0.0	346.5	1189.9	4.123×10^5
33.0	346.5	1189.9	4.123×10^5
33.01	0.0	0.0	0.0

* For this case, mass-energy release is constant and undiminished through the time of peak pressure occurrence.

TABLE 3E-20: BLOWDOWN

12" DBB-68 or 72 (MS to RHR HX) Line Breaks in Penetration Room or RHR Pump Room*

Time (sec)	Lbs mass/sec	Btu/lb mass	Btu/sec
0.0	6078.8	397.69	2.417×10^6
120.0	6078.8	397.69	2.417×10^6
120.0	0.0	0.0	0.0

* For this case, mass-energy release is constant and undiminished through the time of peak pressure occurrence.

TABLE 3E-21: DELETED

TABLE 3E-22: BLOWDOWN

4" DBZ-1 (RWCU Pump Suction) Line Break in RWCU Pump Room

Time (sec)	Lbs mass/sec	Btu/lb mass	Btu/sec
0.0	626.5	470.0	2.945×10^5
48.5	626.5	470.0	2.945×10^5
50.0	0.0	470.0	0

TABLE 3E-23: BLOWDOWN

8" DBC-1 (RWCU Pump Suction) Line Break in RWCU Pipe Chase

Time (sec)	Lbs mass/sec	Btu/lb mass	Btu/sec
0.0	626.5	470.0	2.945×10^5
48.5	626.5	470.0	2.945×10^5
50.0	0.0	470.0	0

TABLE 3E-24: BLOWDOWN

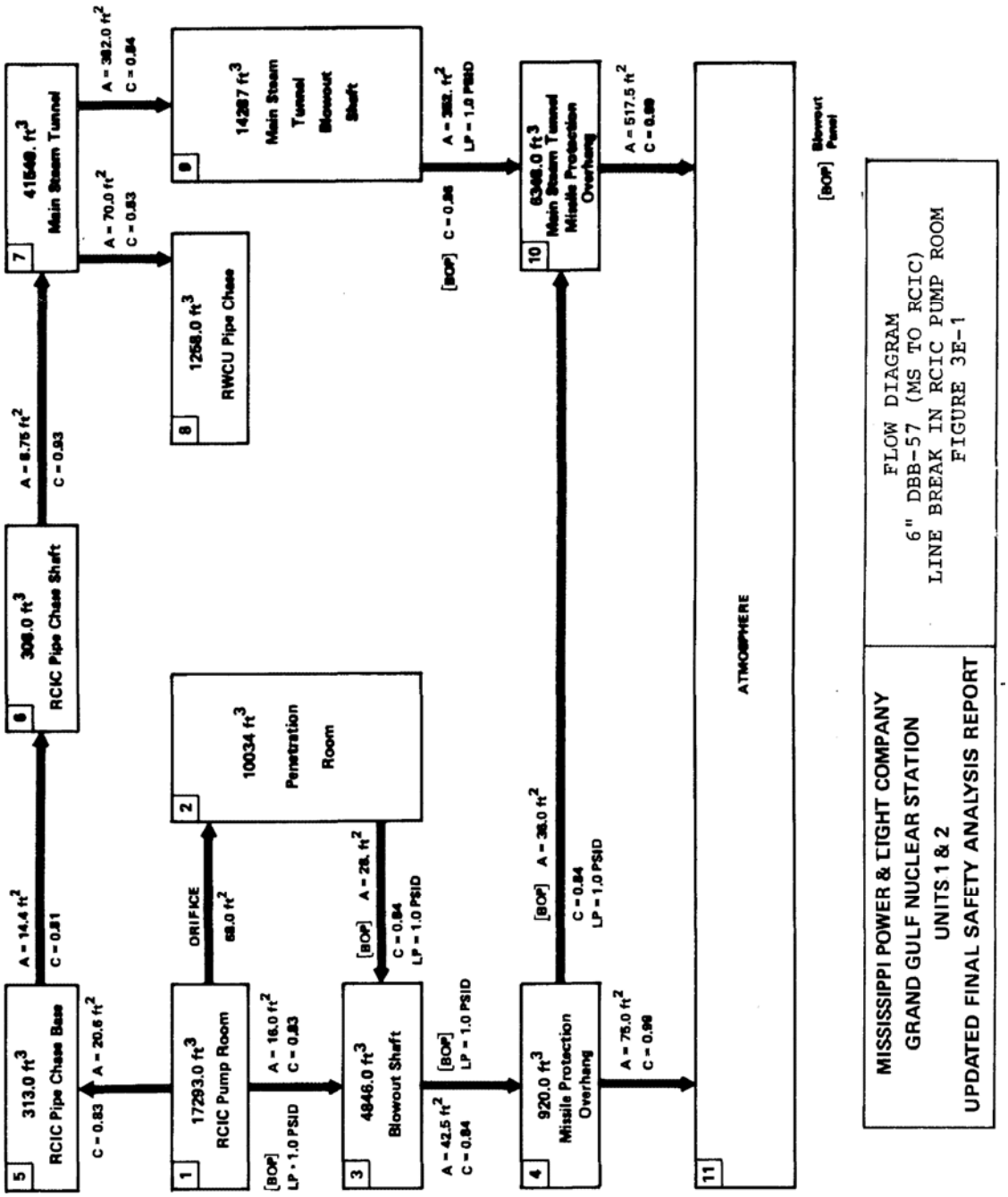
28" DBD-56 Main Steam Line Slot Break

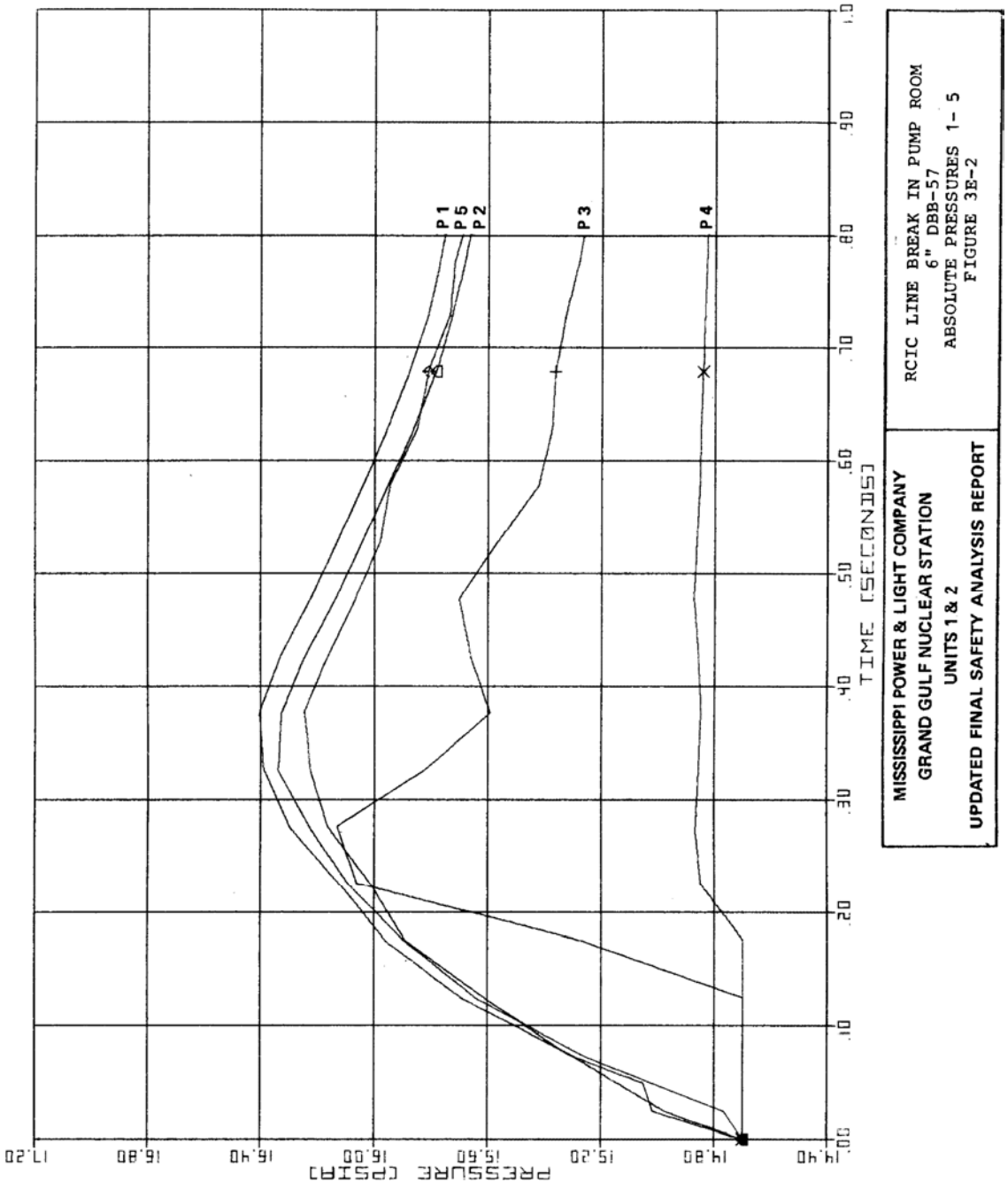
Time (sec)	Lbs mass/sec	Btu/lb mass	Btu/sec
0.0	0.0	1189.9	0.0
0.0	7623.0	1189.9	9.071×10^6
1.0	7623.0	1189.9	9.071×10^6
1.0	25316.0	594.4	1.505×10^6
3.5	25316.0	594.4	1.505×10^7
5.5	0.0	594.4	0.0

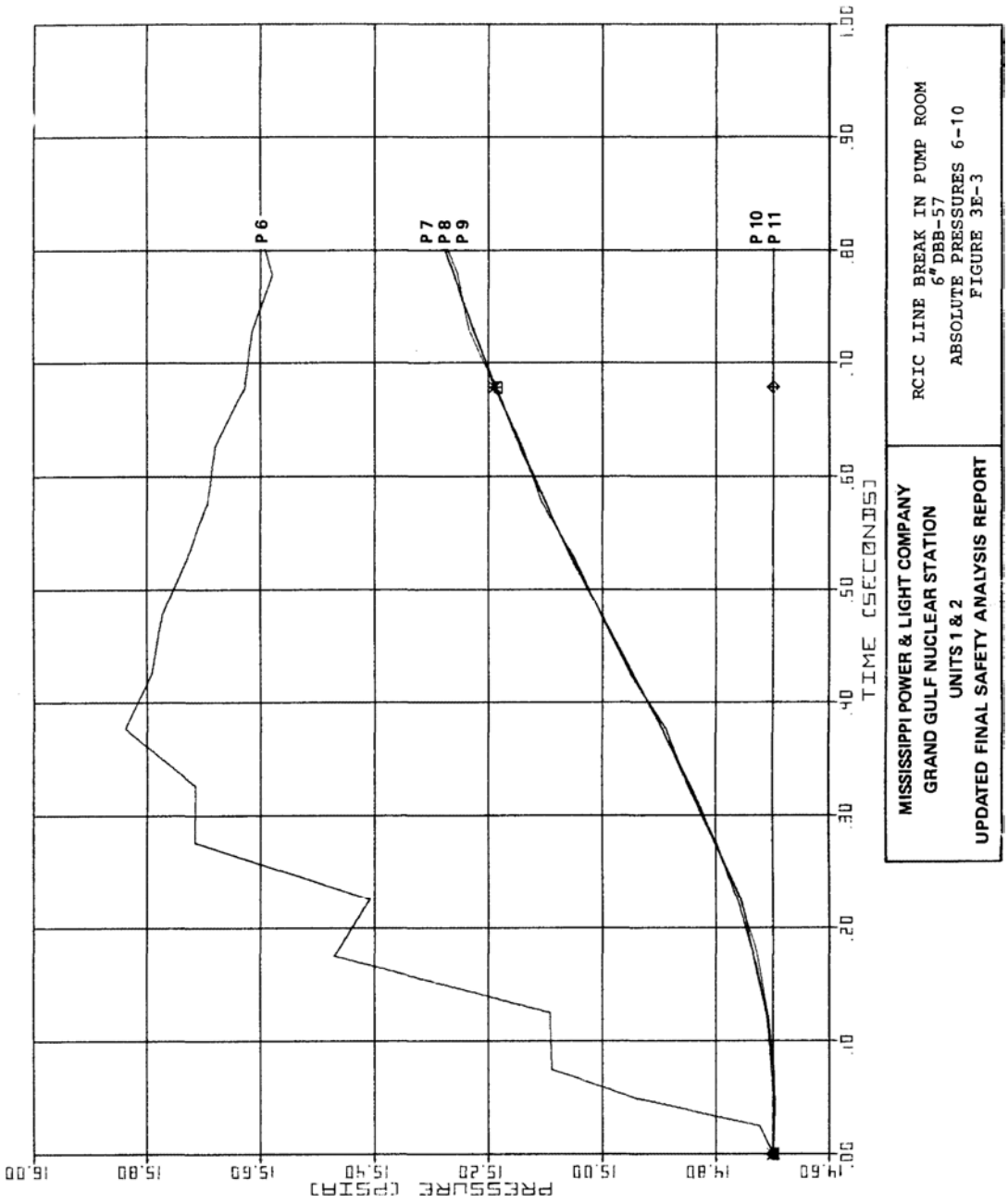
TABLE 3E-25: BLOWDOWN

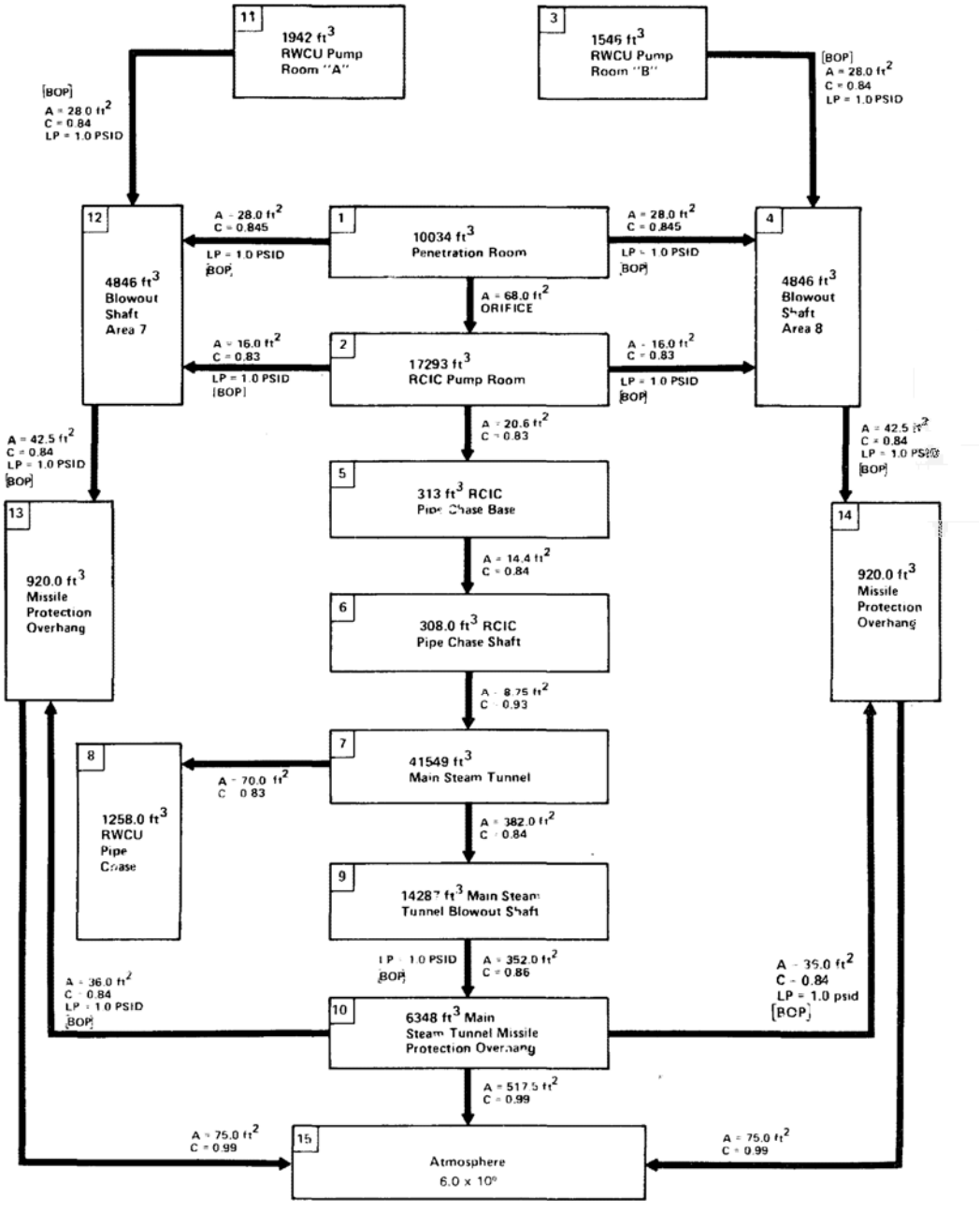
28" DBD-56 Main Steam Line Circumferential Break

Time (sec)	Lbs mass/sec	Btu/lb mass	Btu/sec
0.0	0.0	1189.9	0.0
0.0	15000.0	1189.9	1.785×10^7
1.0	15000.0	1189.9	1.785×10^7
1.0	37000.0	684.9	2.534×10^7
1.65	37000.0	684.9	2.534×10^7
1.65	31300.0	592.9	1.856×10^7
3.90	31300.0	592.9	1.856×10^7
4.10	27300.0	599.2	1.166×10^7
4.50	19600.0	595.7	1.168×10^7
5.0	10000.0	601.2	6.012×10^6
5.25	5100.0	603.9	3.080×10^6
5.5	0.0	603.9	0.0


5
313.0 ft3
RCIC Pipe Chase Base
C = 0.83
A = 20.6 ft2
A = 14.4 ft2
C = 0.81
6
306.0 ft3
RCIC Pipe Chase Shaft
A = 8.76 ft2
C = 0.83
7
41548. ft3
Main Steam Tunnel
A = 70.0 ft2
C = 0.83
A = 382.0 ft2
C = 0.84
8
1258.0 ft3
RWCU Pipe Chase
9
14287 ft3
Main Steam Tunnel Blowout Shaft
A = 382. ft2
LP = 1.0 PSID
[BOP] C = 0.86
10
6348.0 ft3
Main Steam Tunnel Missile Protection Overhang
A = 517.5 ft2
C = 0.80
11
ATMOSPHERE
[BOP] Blowout Panel
1
17293.0 ft3
RCIC Pump Room
[BOP] LP = 1.0 PSID
A = 18.0 ft2
C = 0.83
ORIFICE
68.0 ft2
2
10034 ft3
Penetration Room
3
4946.0 ft3
Blowout Shaft
[BOP] A = 28. ft2
C = 0.84
LP = 1.0 PSID
A = 42.5 ft2
C = 0.84
[BOP] LP = 1.0 PSID
4
920.0 ft3
Missile Protection Overhang
[BOP] A = 38.0 ft2
C = 0.84
LP = 1.0 PSID
A = 75.0 ft2
C = 0.98
MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT
FLOW DIAGRAM
6" DBB-57 (MS TO RCIC)
LINE BREAK IN RCIC PUMP ROOM
FIGURE 3E-1


P1
P5
P2
P3
P4
PRESSURE (PSIA)
TIME (SECONDS)
14.40
14.80
15.20
15.60
16.00
16.40
16.80
17.20
.00
.10
.20
.30
.40
.50
.60
.70
.80
.90
1.0
MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT
RCIC LINE BREAK IN PUMP ROOM
6" DBB-57
ABSOLUTE PRESSURES 1- 5
FIGURE 3E-2

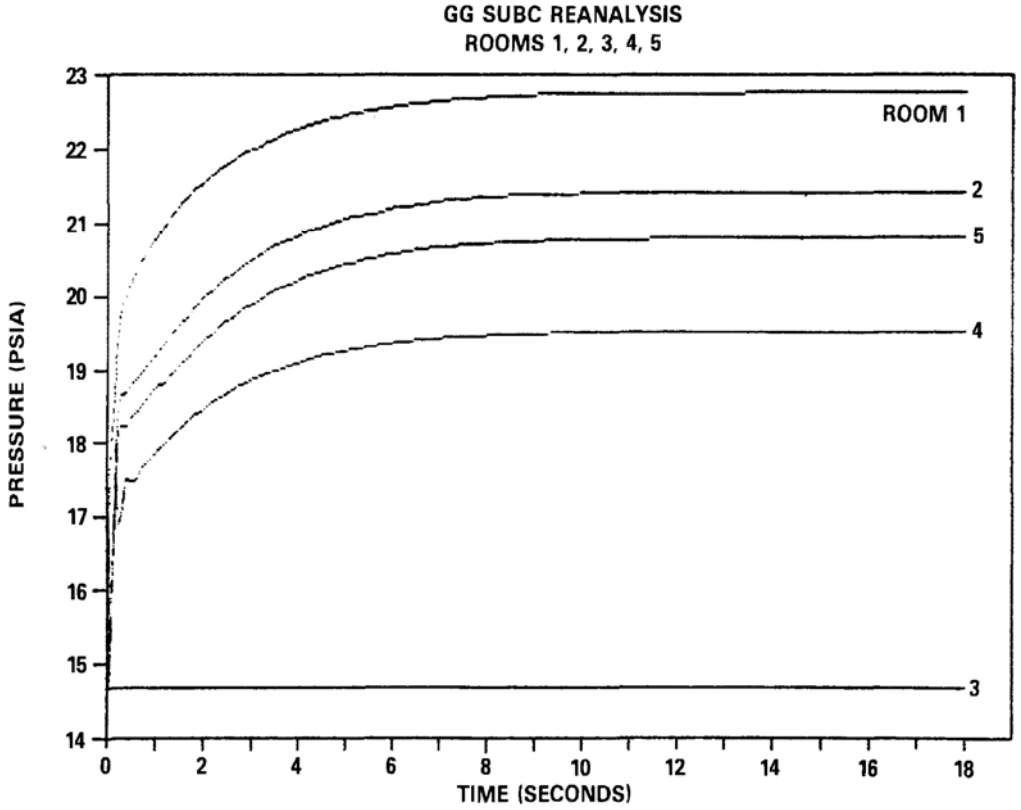

P6
P7
P8
P9
P10
P11
PRESSURE (PSIA)
TIME (SECONDS)
RCIC LINE BREAK IN PUMP ROOM
6" DBB-57
ABSOLUTE PRESSURES 6-10
FIGURE 3E-3
MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT



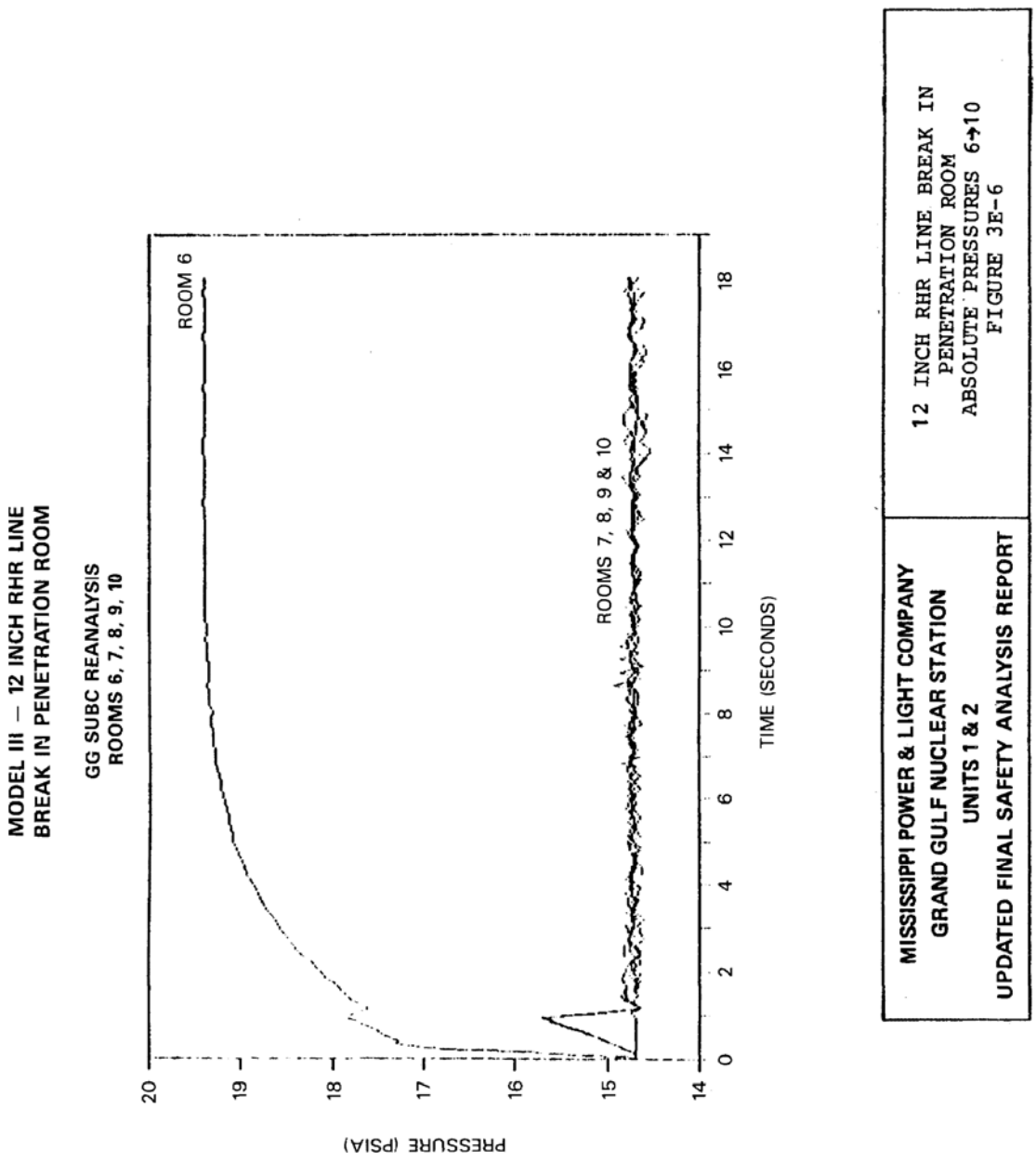
[BOP] : BLOWOUT PANEL

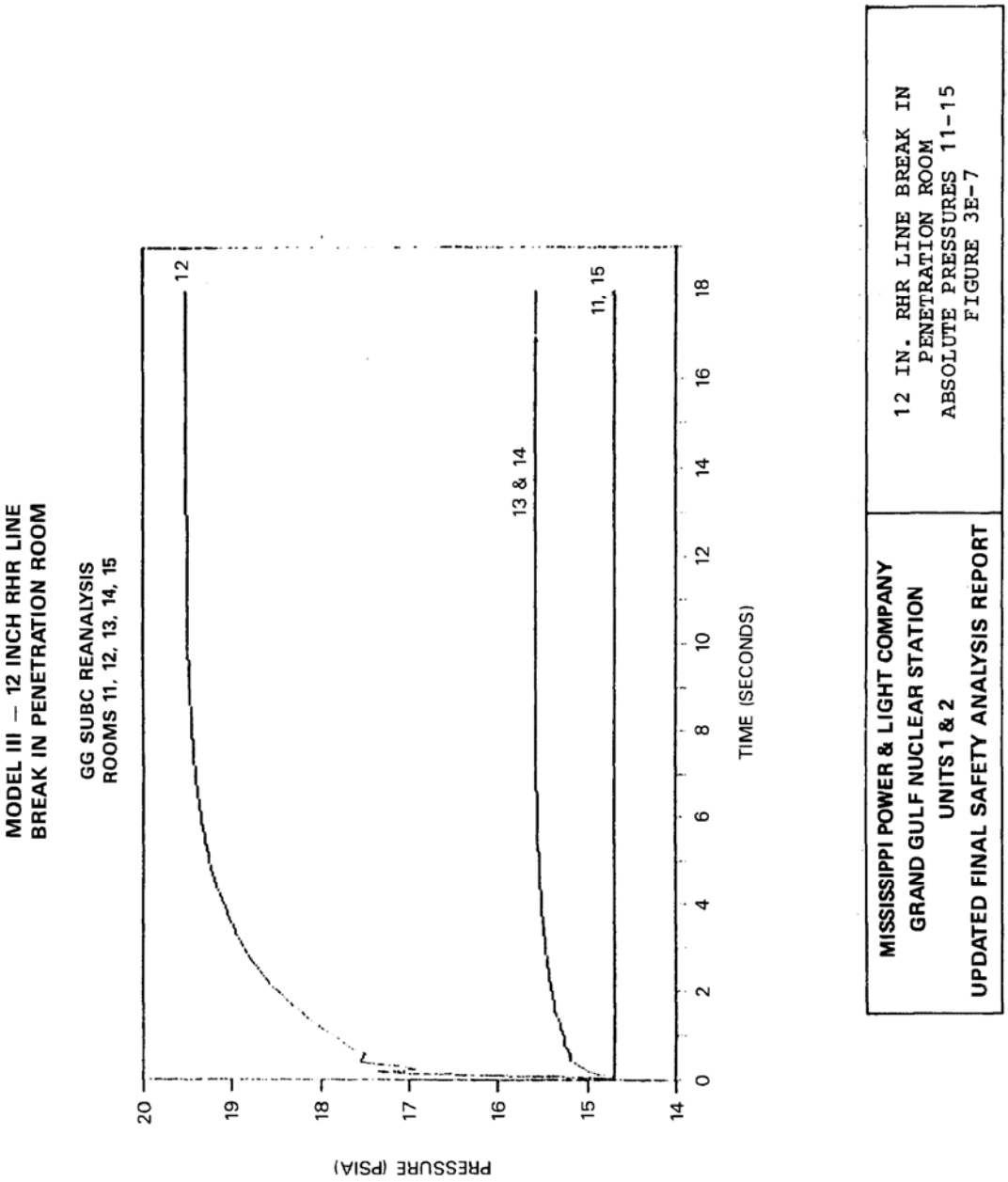
MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT	FLOW DIAGRAM LINE BREAK IN PENETRATION ROOM 12" DBB-68 (RHR "B" TO FW) FIGURE 3E-4

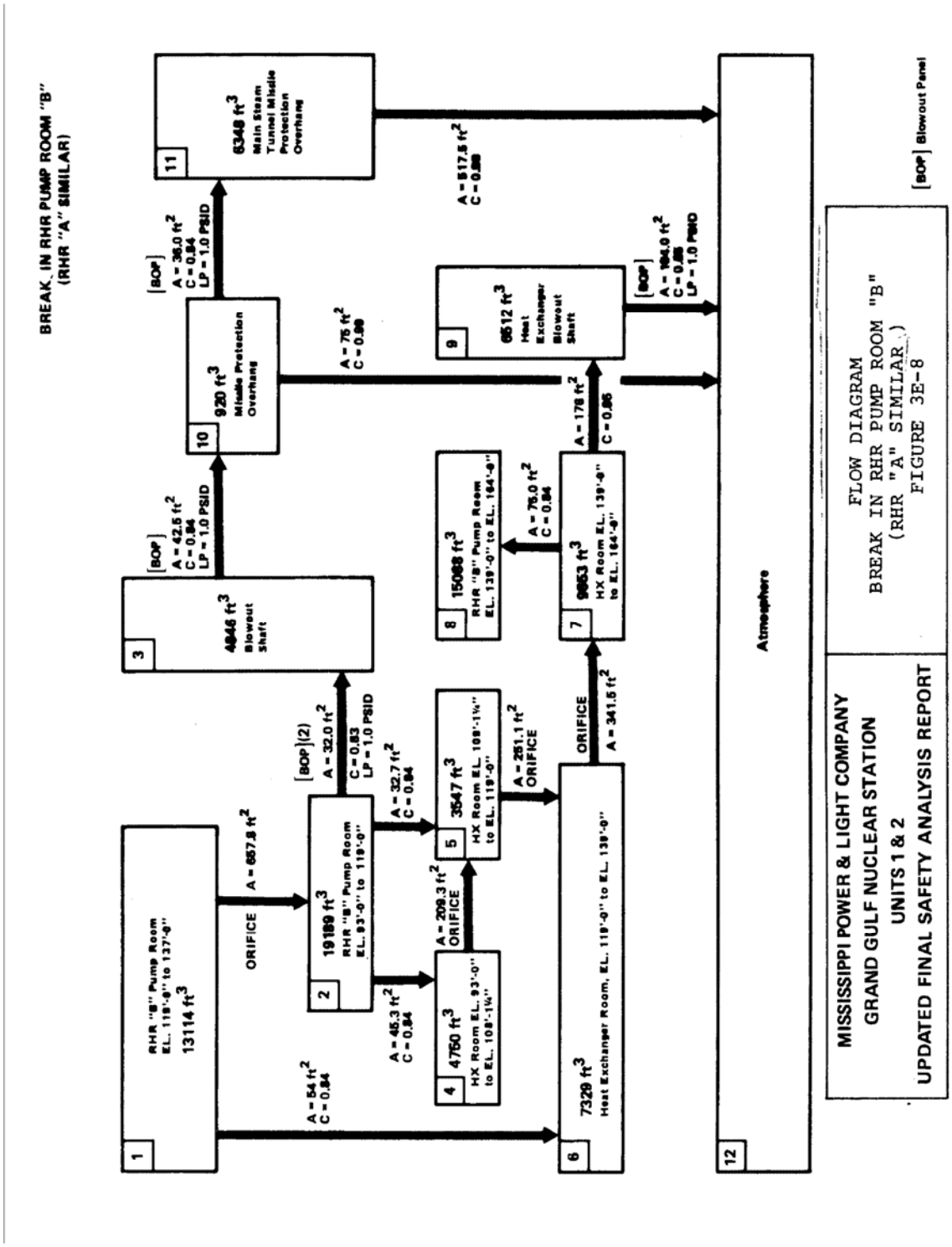
MODEL III. 12 INCH RHR LINE
BREAK IN PENETRATION ROOM

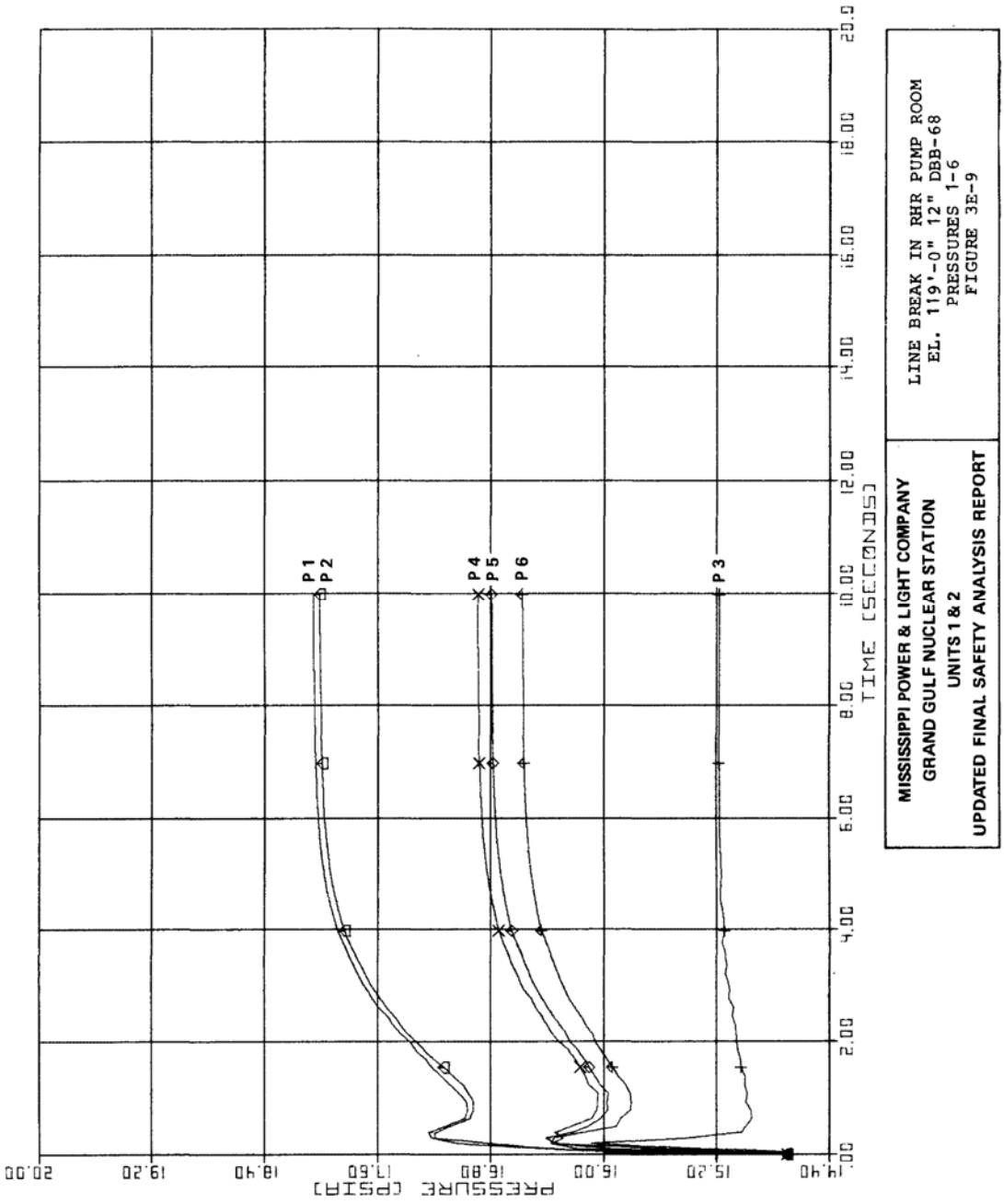


MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT	12 INCH RHR LINE BREAK IN PENETRATION ROOM ABSOLUTE PRESSURES 1 –5 FIGURE 3E-5


MODEL III — 12 INCH RHR LINE
BREAK IN PENETRATION ROOM
GG SUBC REANALYSIS
ROOMS 6, 7, 8, 9, 10
ROOM 6
ROOMS 7, 8, 9 & 10
PRESSURE (PSIA)
TIME (SECONDS)
20
19
18
17
16
15
14
0
2
4
6
8
10
12
14
16
18
MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT
12 INCH RHR LINE BREAK IN
PENETRATION ROOM
ABSOLUTE PRESSURES 6→10
FIGURE 3E-6


MODEL III — 12 INCH RHR LINE
BREAK IN PENETRATION ROOM
GG SUBC REANALYSIS
ROOMS 11, 12, 13, 14, 15
12
13 & 14
11, 15
PRESSURE (PSIA)
TIME (SECONDS)
20
19
18
17
16
15
14
0
2
4
6
8
10
12
14
16
18
MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT
12 IN. RHR LINE BREAK IN
PENETRATION ROOM
ABSOLUTE PRESSURES 11-15
FIGURE 3E-7


BREAK IN RHR PUMP ROOM "B"
(RHR "A" SIMILAR)
1
RHR "B" Pump Room
EL. 119'-0" to 137'-0"
13114 ft3
2
19189 ft3
RHR "B" Pump Room
EL. 93'-0" to 119'-0"
3
4846 ft3
Blowout Shaft
4
4750 ft3
HX Room EL. 93'-0" to EL. 108'-1¼"
5
3547 ft3
HX Room EL. 108'-1¼" to EL. 119'-0"
6
7329 ft3
Heat Exchanger Room, EL. 119'-0" to EL. 139'-0"
7
9853 ft3
HX Room EL. 139'-0" to EL. 164'-0"
8
15088 ft3
RHR "B" Pump Room EL. 139'-0" to EL. 164'-0"
9
6512 ft3
Heat Exchanger Blowout Shaft
10
920 ft3
Missile Protection Overhang
11
6348 ft3
Main Steam Tunnel Missile Protection Overhang
12
Atmosphere
ORIFICE
A = 657.8 ft2
A = 54 ft2
C = 0.84
A = 45.3 ft2
C = 0.84
A = 209.3 ft2
ORIFICE
A = 251.1 ft2
ORIFICE
A = 32.7 ft2
C = 0.84
[BOP](2)
A = 32.0 ft2
C = 0.83
LP = 1.0 PSID
[BOP]
A = 42.5 ft2
C = 0.84
LP = 1.0 PSID
[BOP]
A = 36.0 ft2
C = 0.84
LP = 1.0 PSID
A = 75 ft2
C = 0.86
A = 517.5 ft2
C = 0.86
ORIFICE
A = 341.5 ft2
A = 76.0 ft2
C = 0.84
A = 178 ft2
C = 0.86
[BOP]
A = 164.0 ft2
C = 0.86
LP = 1.0 PSID
FLOW DIAGRAM
BREAK IN RHR PUMP ROOM "B"
(RHR "A" SIMILAR)
FIGURE 3E-8
MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT
[BOP] Blowout Panel


PRESSURE (PSIA)
14.40
15.20
16.00
16.80
17.60
18.40
19.20
20.00
TIME (SECONDS)
0.00
2.00
4.00
6.00
8.00
10.00
12.00
14.00
16.00
18.00
20.0
P1
P2
P3
P4
P5
P6
MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT
LINE BREAK IN RHR PUMP ROOM
EL. 119'-0" 12" DBB-68
PRESSURES 1-6
FIGURE 3E-9

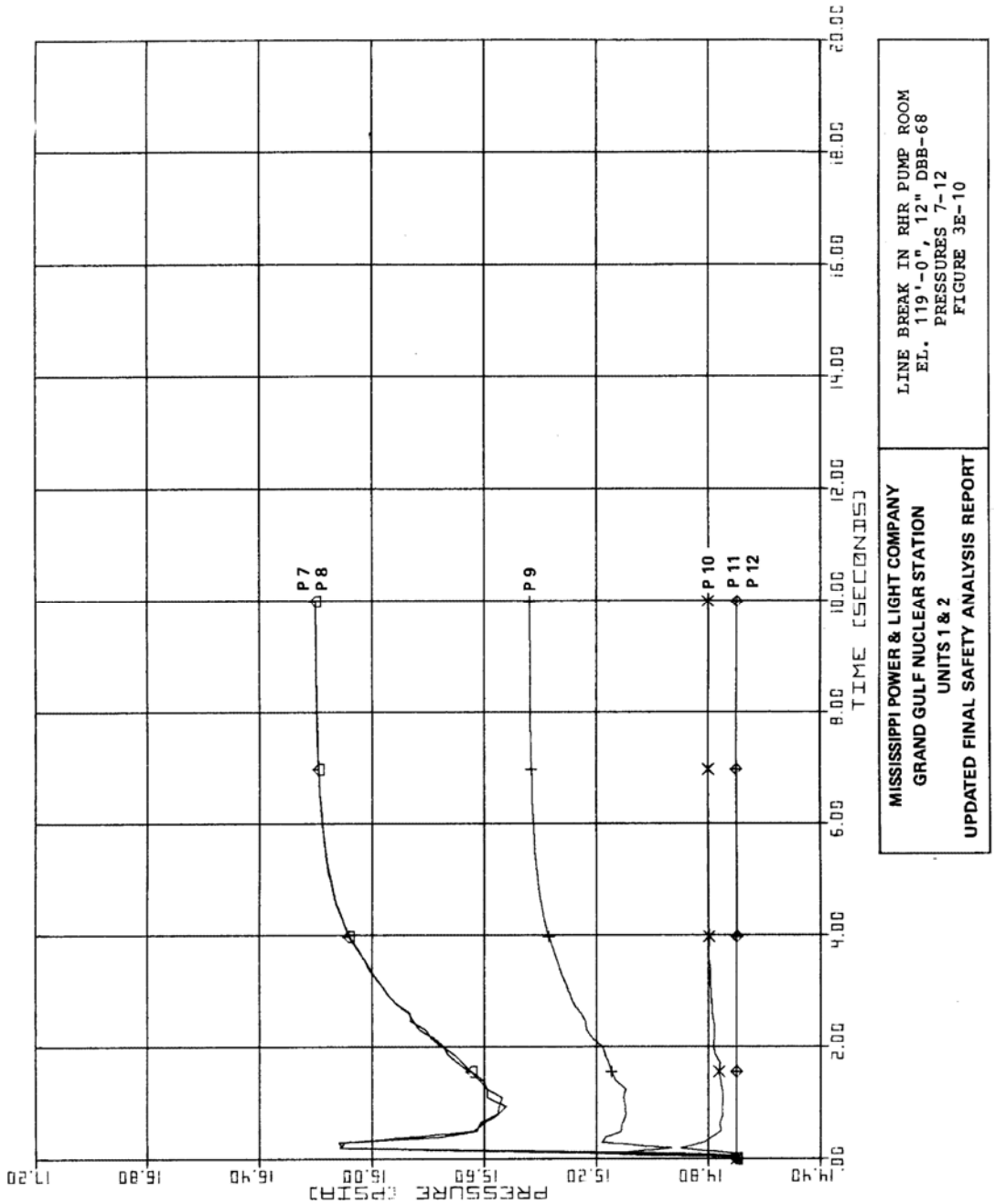

PRESSURE (PSIA)
14.40
14.80
15.20
15.60
16.00
16.40
16.80
17.20
TIME (SECONDS)
0.00
2.00
4.00
6.00
8.00
10.00
12.00
14.00
16.00
18.00
20.00
P 7
P 8
P 9
P 10
P 11
P 12
MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT
LINE BREAK IN RHR PUMP ROOM
EL. 119'-0", 12" DBB-68
PRESSURES 7-12
FIGURE 3E-10

Figure 3E-11

Deleted

Figure 3E-12

Deleted

Figure 3E-13

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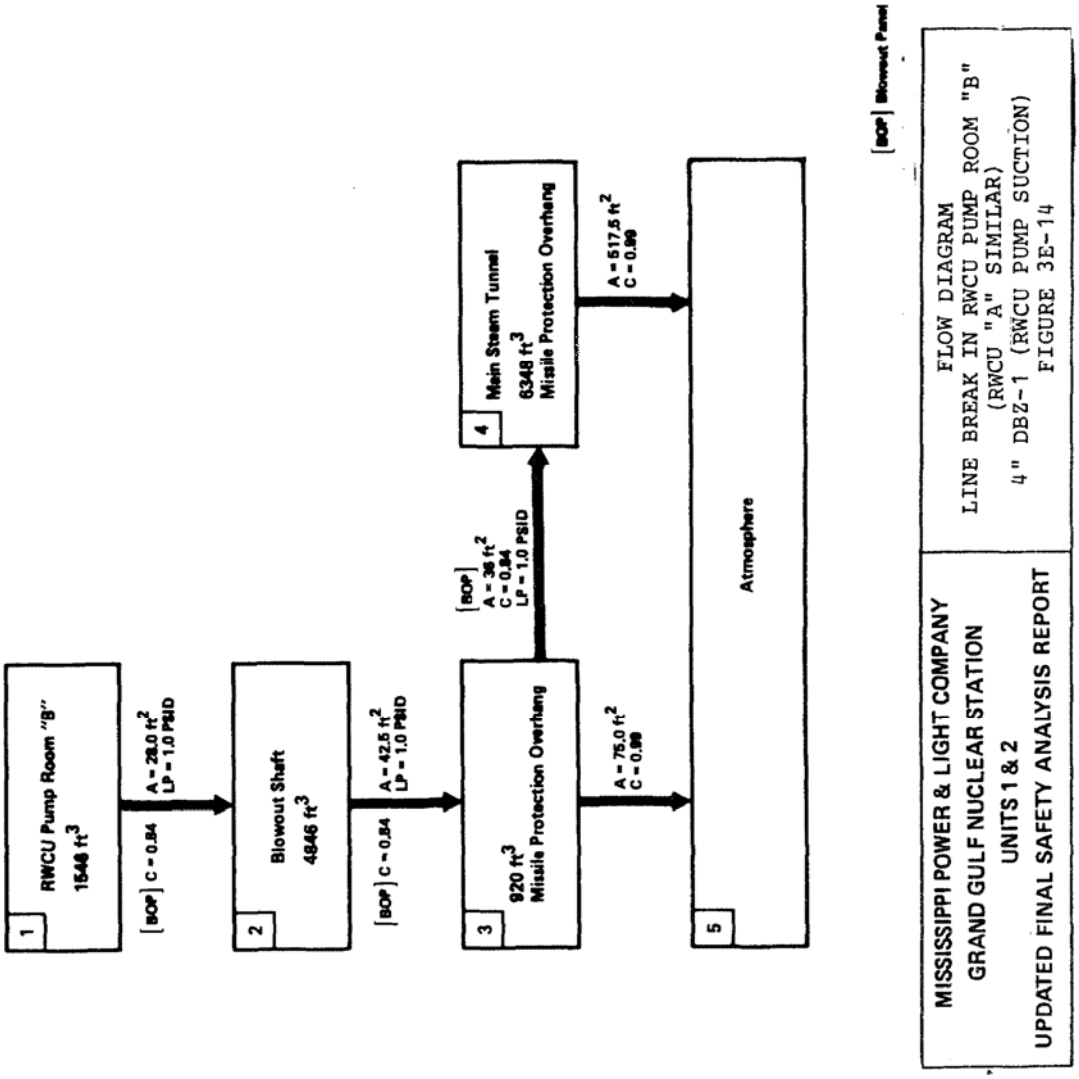
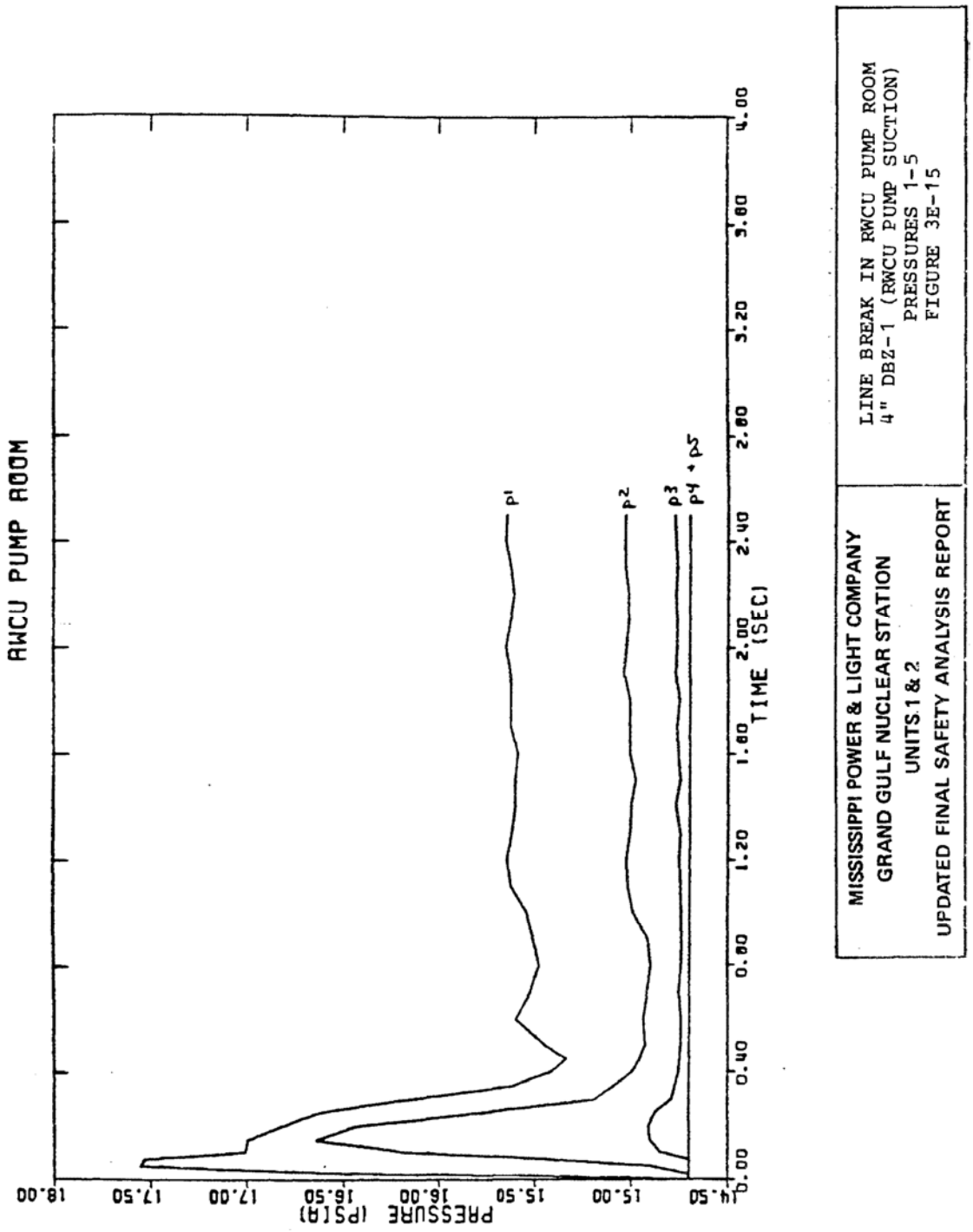
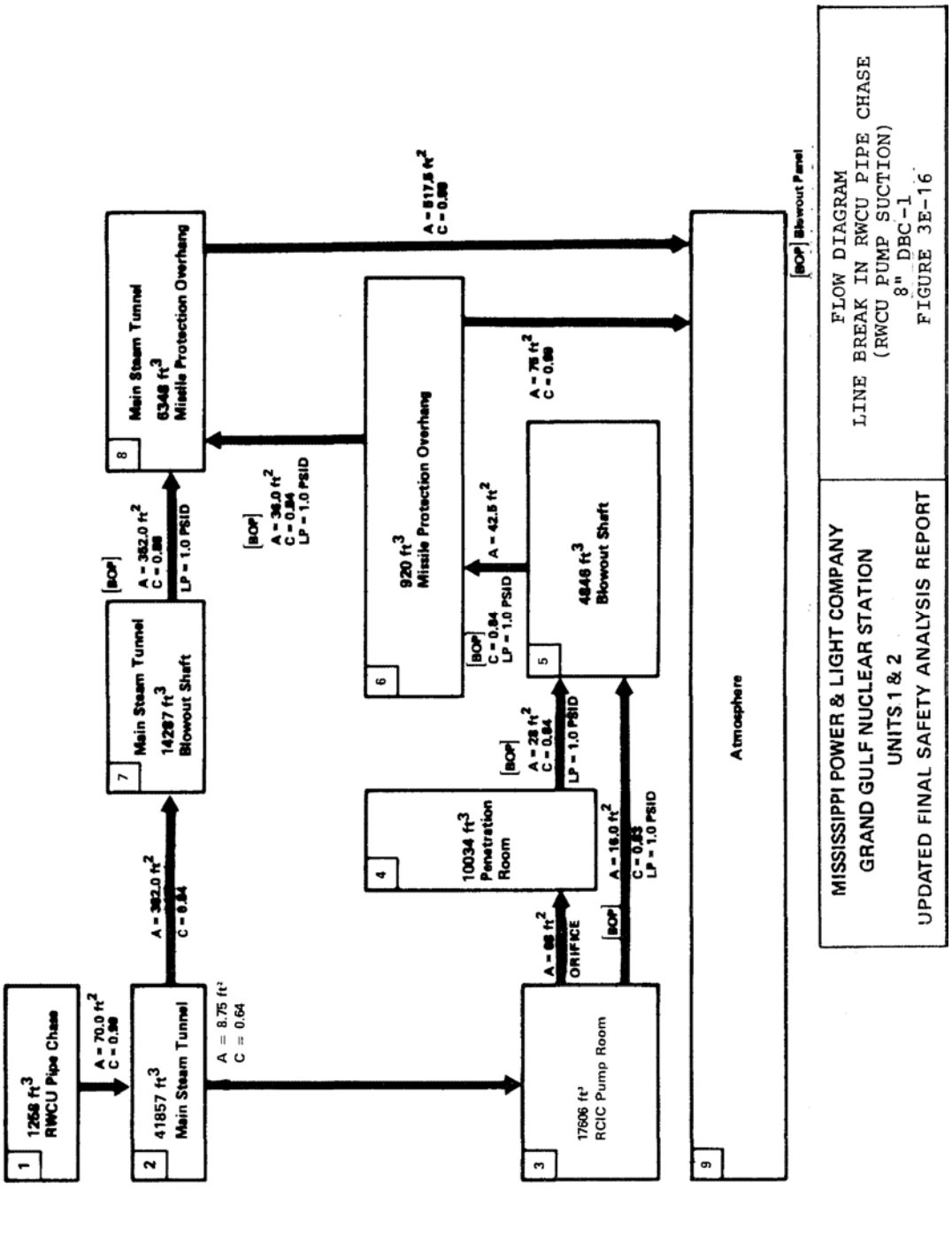
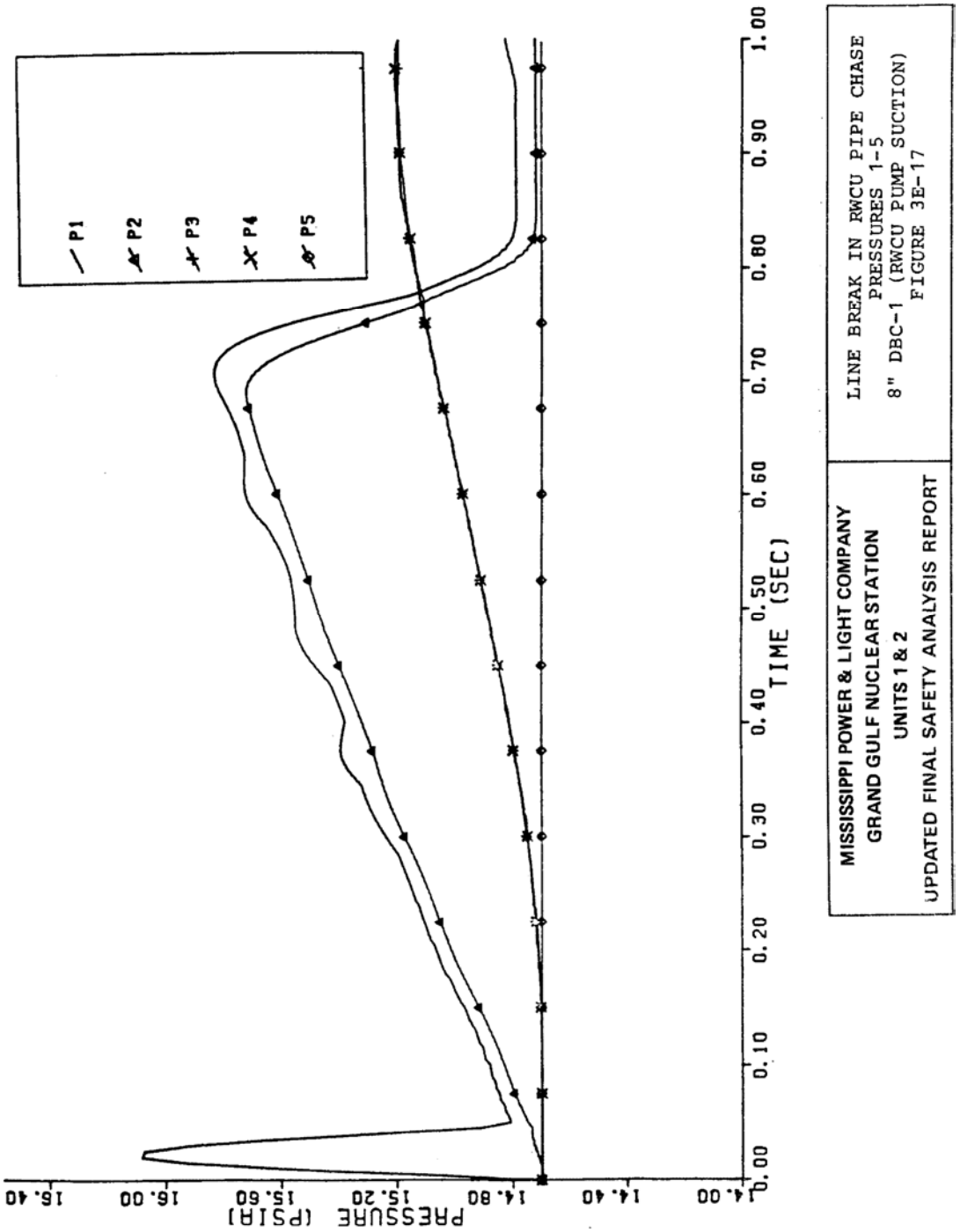
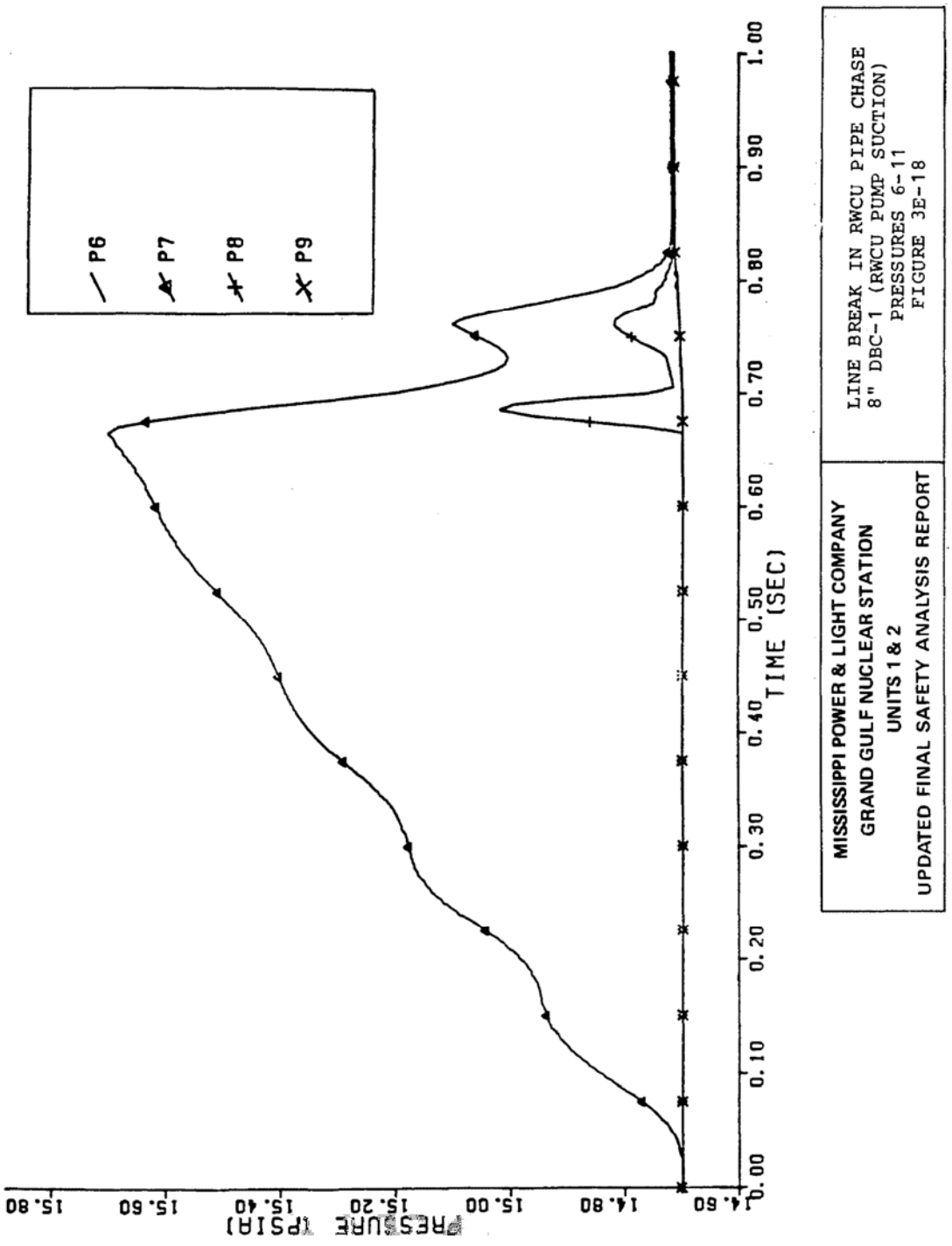
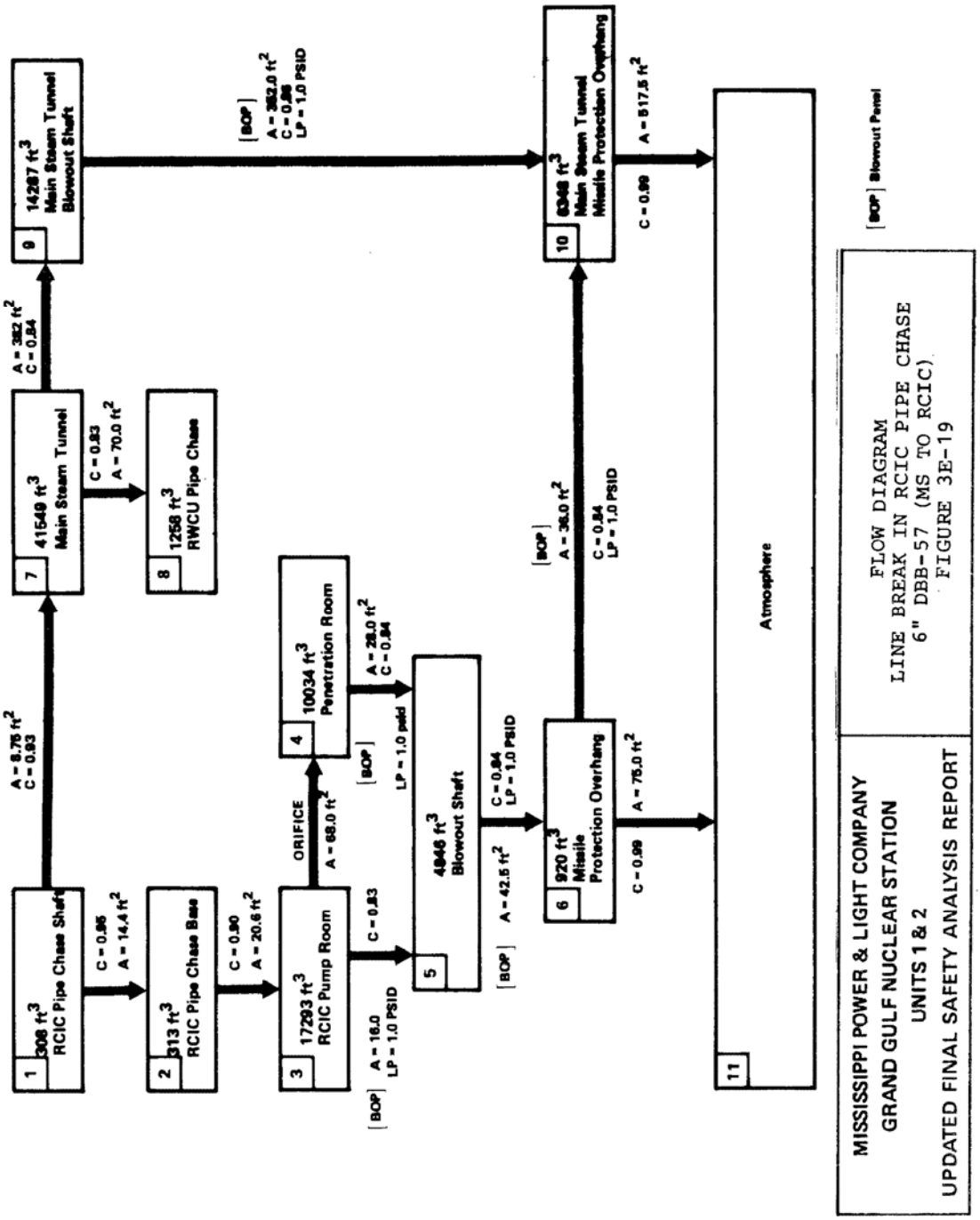

1
RWCU Pump Room "B"
1546 ft3
[BOP] C = 0.84
A = 28.0 ft2
LP = 1.0 PSID
2
Blowout Shaft
4846 ft3
[BOP] C = 0.84
A = 42.5 ft2
LP = 1.0 PSID
3
920 ft3
Missile Protection Overhang
[BOP]
A = 35 ft2
C = 0.84
LP = 1.0 PSID
4
Main Steam Tunnel
6348 ft3
Missile Protection Overhang
A = 517.5 ft2
C = 0.99
A = 75.0 ft2
C = 0.99
5
Atmosphere
[BOP] Blowout Panel
MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT
FLOW DIAGRAM
LINE BREAK IN RWCU PUMP ROOM "B"
(RWCU "A" SIMILAR)
4" DBZ-1 (RWCU PUMP SUCTION)
FIGURE 3E-14

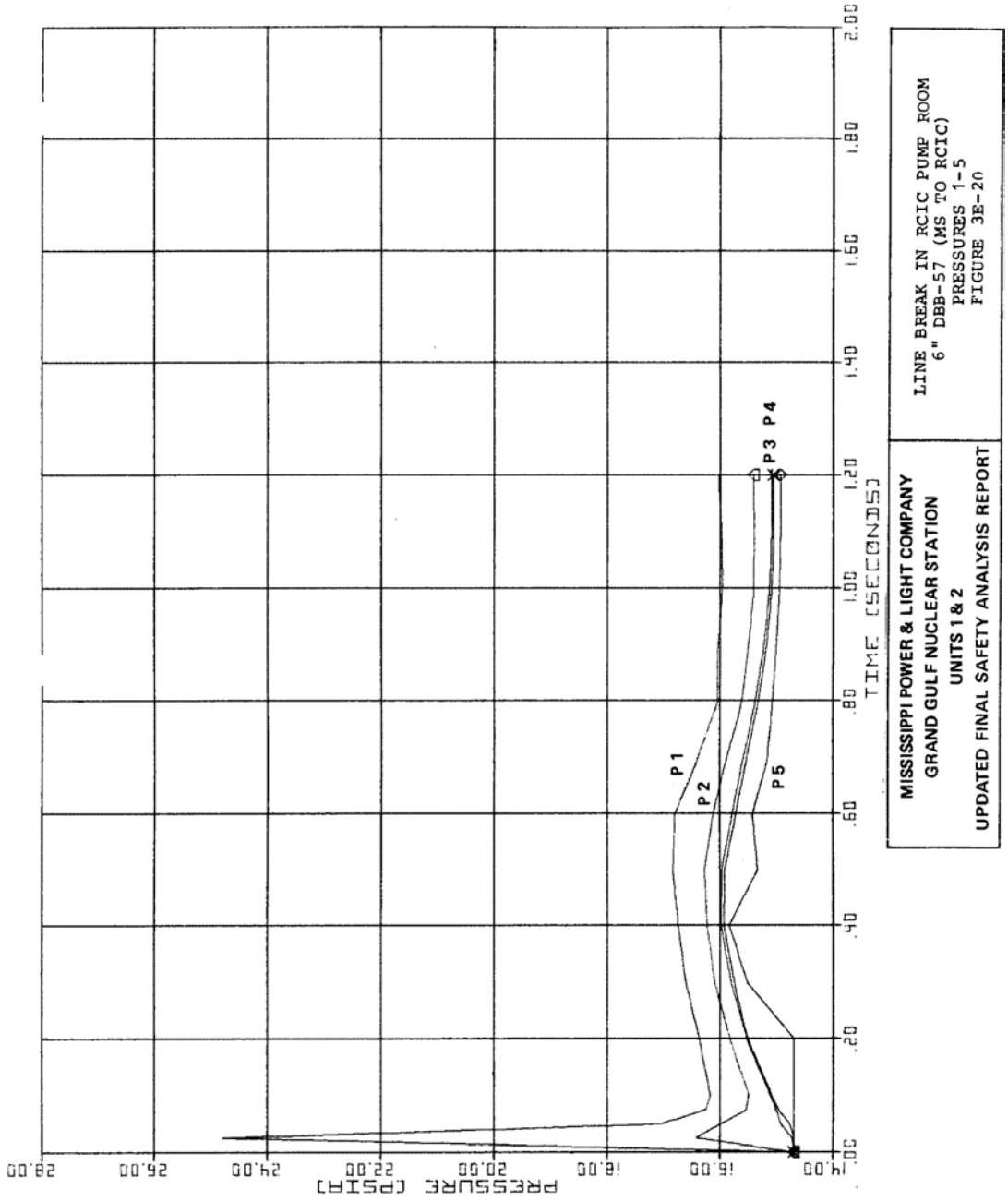

RWCU PUMP ROOM
PRESSURE (PSIA)
TIME (SEC)
P1
P2
P3
P4 + P5
14.50
15.00
15.50
16.00
16.50
17.00
17.50
18.00
0.00
0.40
0.80
1.20
1.60
2.00
2.40
2.80
3.20
3.60
4.00
MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT
LINE BREAK IN RWCU PUMP ROOM
4" DBZ-1 (RWCU PUMP SUCTION)
PRESSURES 1-5
FIGURE 3E-15

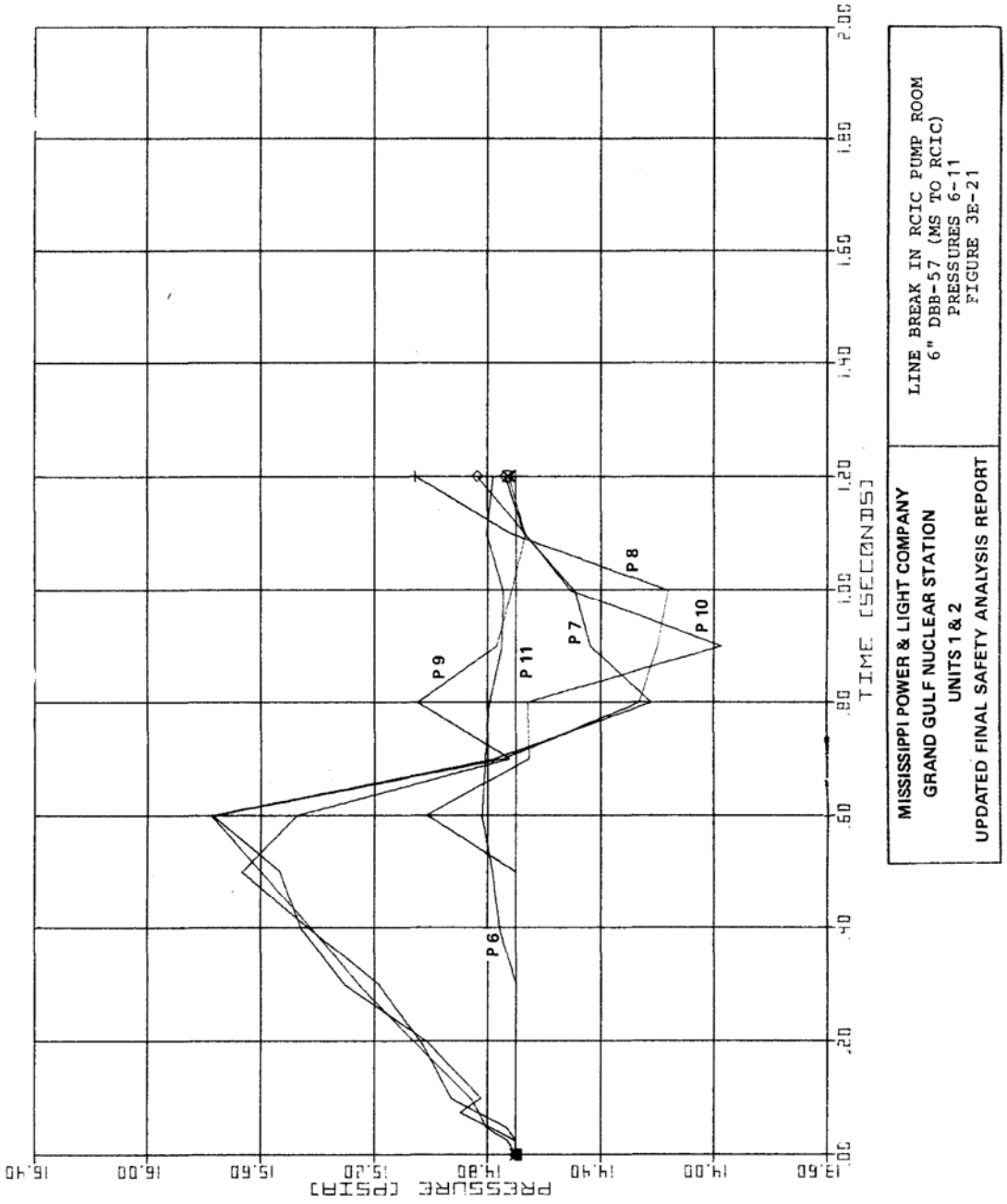

1
1266 ft3
RWCU Pipe Chase
A = 70.0 ft2
C = 0.80
2
41857 ft3
Main Steam Tunnel
A = 8.75 ft2
C = 0.64
A = 382.0 ft2
C = 0.84
7
Main Steam Tunnel
14287 ft3
Blowout Shaft
[BOP]
A = 382.0 ft2
C = 0.80
LP = 1.0 PSID
8
Main Steam Tunnel
6348 ft3
Missile Protection Overhang
A = 517.8 ft2
C = 0.80
[BOP]
A = 36.0 ft2
C = 0.84
LP = 1.0 PSID
6
920 ft3
Missile Protection Overhang
A = 76 ft2
C = 0.80
[BOP]
C = 0.84
LP = 1.0 PSID
A = 42.5 ft2
5
4846 ft3
Blowout Shaft
4
10034 ft3
Penetration
Room
[BOP]
A = 28 ft2
C = 0.84
LP = 1.0 PSID
A = 16.0 ft2
C = 0.84
LP = 1.0 PSID
A = 66 ft2
ORIFICE
[BOP]
3
17606 ft3
RCIC Pump Room
9
Atmosphere
[BOP] Blowout Panel
FLOW DIAGRAM
LINE BREAK IN RWCU PIPE CHASE
(RWCU PUMP SUCTION)
8" DBC-1
FIGURE 3E-16
MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT

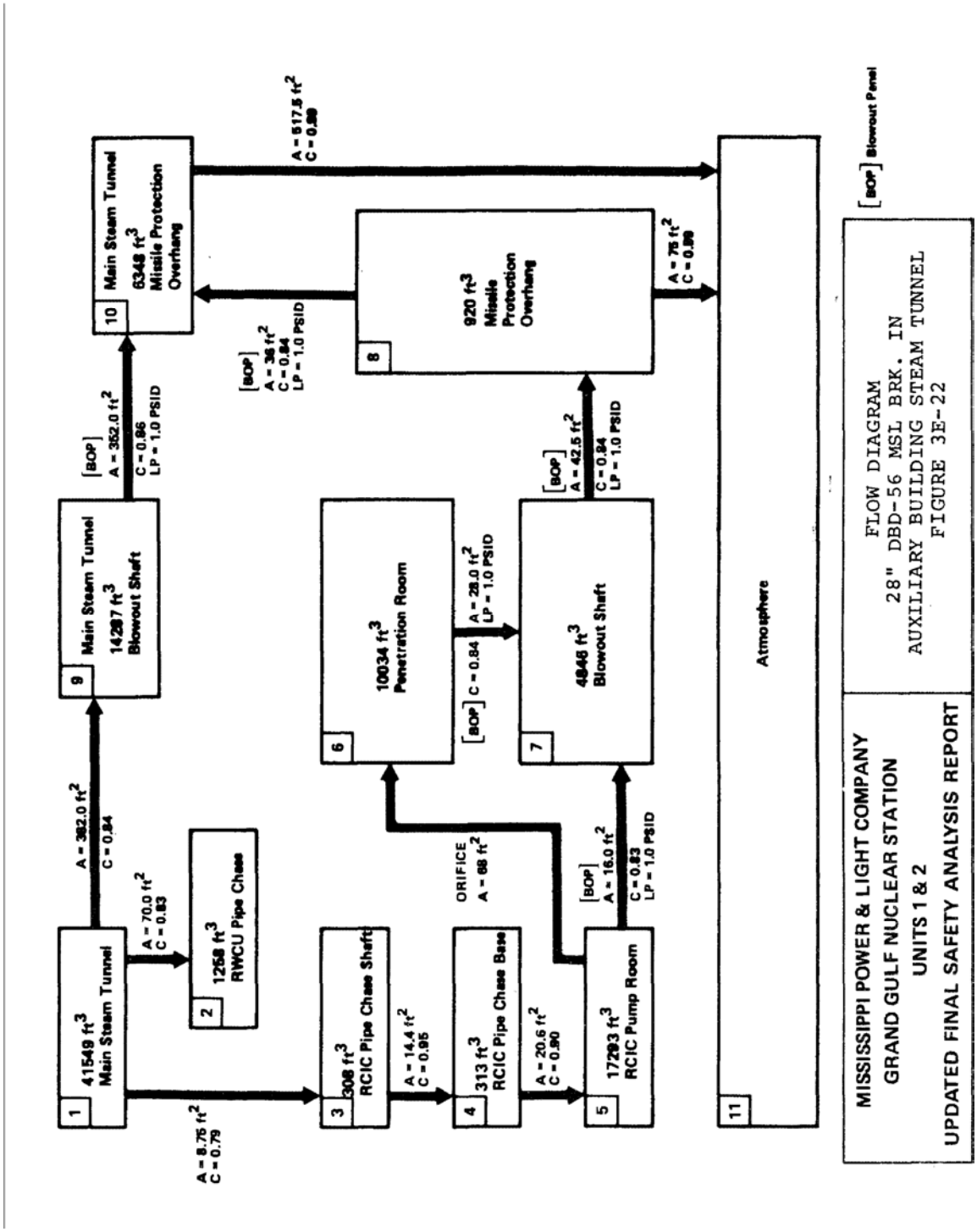

P1
P2
P3
P4
P5
PRESSURE (PSIA)
14.00
14.40
14.80
15.20
15.60
16.00
16.40
TIME (SEC)
0.00
0.10
0.20
0.30
0.40
0.50
0.60
0.70
0.80
0.90
1.00
MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT
LINE BREAK IN RWCU PIPE CHASE
PRESSURES 1-5
8" DBC-1 (RWCU PUMP SUCTION)
FIGURE 3E-17

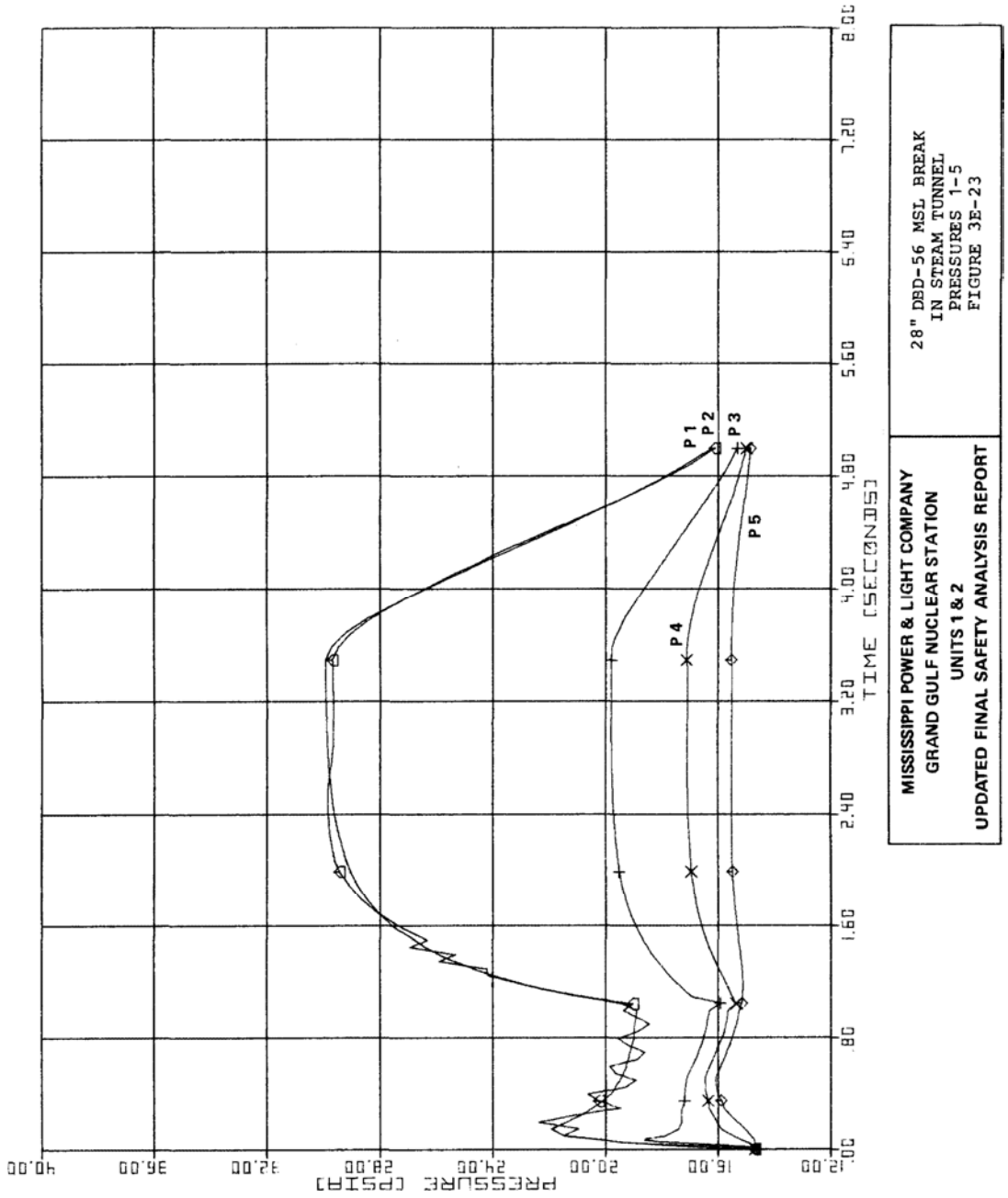

P6
P7
P8
P9
PRESSURE (PSIA)
14.60
14.80
15.00
15.20
15.40
15.60
15.80
TIME (SEC)
0.00
0.10
0.20
0.30
0.40
0.50
0.60
0.70
0.80
0.90
1.00
MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT
LINE BREAK IN RWCU PIPE CHASE
8" DBC-1 (RWCU PUMP SUCTION)
PRESSURES 6-11
FIGURE 3E-18

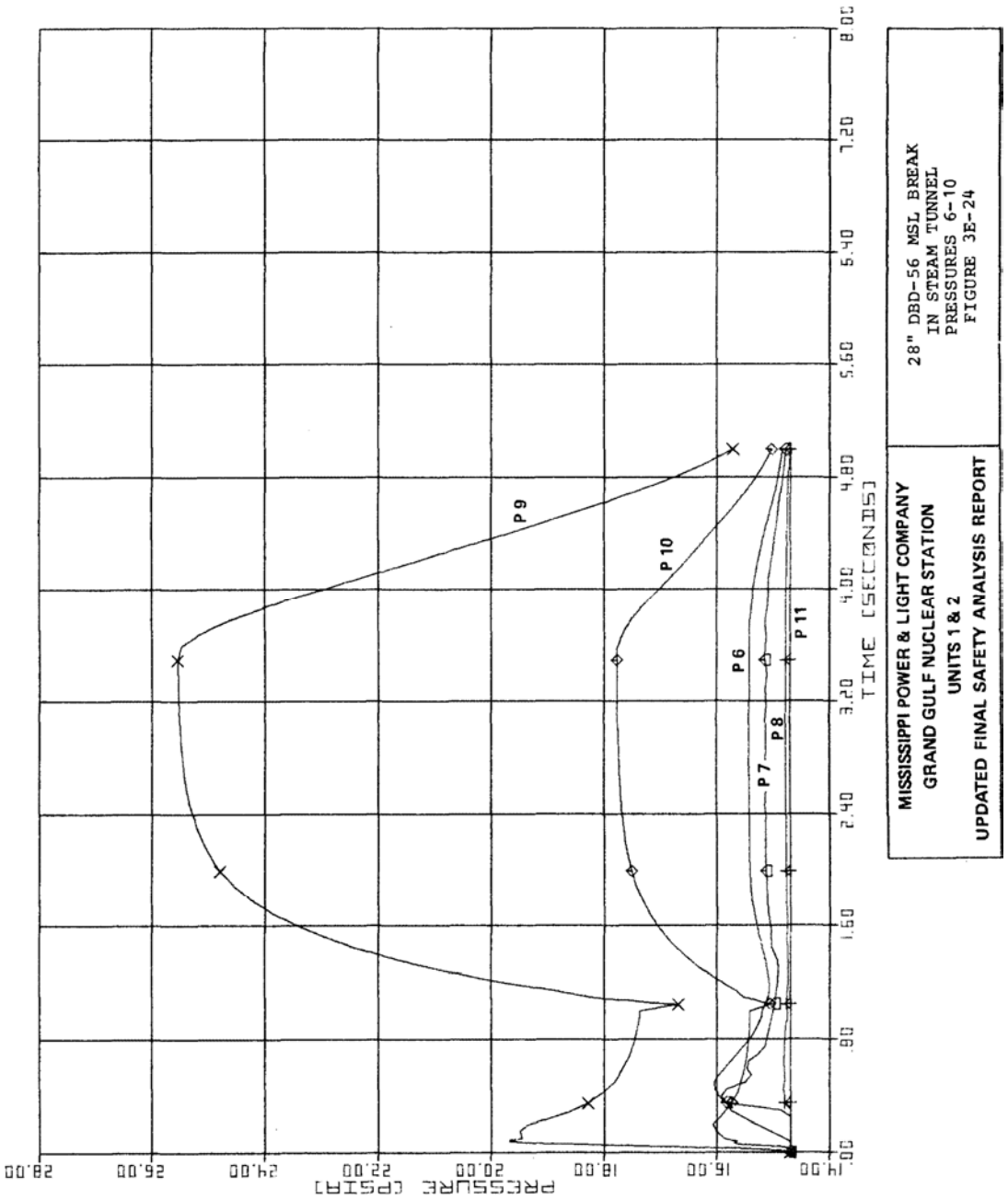

1 308 ft³ RCIC Pipe Chase Shaft
2 313 ft³ RCIC Pipe Chase Base
3 17293 ft³ RCIC Pump Room
4 10034 ft³ Penetration Room
5 4846 ft³ Blowout Shaft
6 920 ft³ Missile Protection Overhang
7 41549 ft³ Main Steam Tunnel
8 1258 ft³ RWCU Pipe Chase
9 14287 ft³ Main Steam Tunnel Blowout Shaft
10 6348 ft³ Main Steam Tunnel Missile Protection Overhang
11 Atmosphere
C = 0.95 A = 14.4 ft²
C = 0.90 A = 20.6 ft²
A = 8.76 ft² C = 0.83
C = 0.83 A = 70.0 ft²
A = 382 ft² C = 0.84
[BOP] A = 353.0 ft² C = 0.86 LP = 1.0 PSID
C = 0.99 A = 817.5 ft²
[BOP] A = 36.0 ft²
C = 0.84 LP = 1.0 PSID
A = 78.0 ft²
C = 0.99
C = 0.84 LP = 1.0 PSID
A = 42.5 ft²
[BOP] LP = 1.0 psid
A = 28.0 ft² C = 0.84
ORIFICE A = 68.0 ft²
C = 0.83
[BOP] A = 16.0 LP = 1.0 PSID
[BOP] Blowout Panel
FLOW DIAGRAM
LINE BREAK IN RCIC PIPE CHASE
6" DBB-57 (MS TO RCIC)
FIGURE 3E-19
MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT

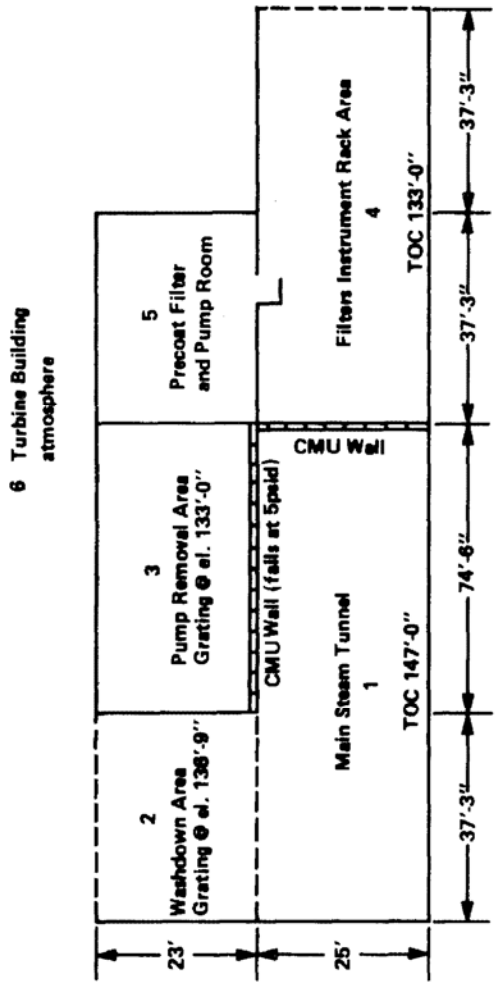

PRESSURE (PSIA)
14.00
16.00
18.00
20.00
22.00
24.00
26.00
28.00
TIME (SECONDS)
.00
.20
.40
.60
.80
1.00
1.20
1.40
1.60
1.80
2.00
P1
P2
P3 P4
P5
MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT
LINE BREAK IN RCIC PUMP ROOM
6" DBB-57 (MS TO RCIC)
PRESSURES 1-5
FIGURE 3E-20


PRESSURE (PSIA)
TIME (SECONDS)
P 6
P 7
P 8
P 9
P 10
P 11
LINE BREAK IN RCIC PUMP ROOM
6" DBB-57 (MS TO RCIC)
PRESSURES 6-11
FIGURE 3E-21
MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT


1
41549 ft3
Main Steam Tunnel
2
1258 ft3
RWCU Pipe Chase
3
308 ft3
RCIC Pipe Chase Shaft
4
313 ft3
RCIC Pipe Chase Base
5
17293 ft3
RCIC Pump Room
6
10034 ft3
Penetration Room
7
4846 ft3
Blowout Shaft
8
920 ft3
Missile Protection Overhang
9
Main Steam Tunnel
14287 ft3
Blowout Shaft
10
Main Steam Tunnel
6348 ft3
Missile Protection Overhang
11
Atmosphere
A = 382.0 ft2
C = 0.84
A = 70.0 ft2
C = 0.83
A = 8.76 ft2
C = 0.79
A = 14.4 ft2
C = 0.95
A = 20.6 ft2
C = 0.90
ORIFICE
A = 68 ft2
[BOP]
A = 16.0 ft2
C = 0.83
LP = 1.0 PSID
[BOP] C = 0.84
A = 28.0 ft2
LP = 1.0 PSID
[BOP]
A = 42.5 ft2
C = 0.84
LP = 1.0 PSID
[BOP]
A = 352.0 ft2
C = 0.86
LP = 1.0 PSID
[BOP]
A = 36 ft2
C = 0.84
LP = 1.0 PSID
A = 517.5 ft2
C = 0.89
A = 75 ft2
C = 0.89
[BOP] Blowout Panel
MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT
FLOW DIAGRAM
28" DBD-56 MSL BRK. IN
AUXILIARY BUILDING STEAM TUNNEL
FIGURE 3E-22

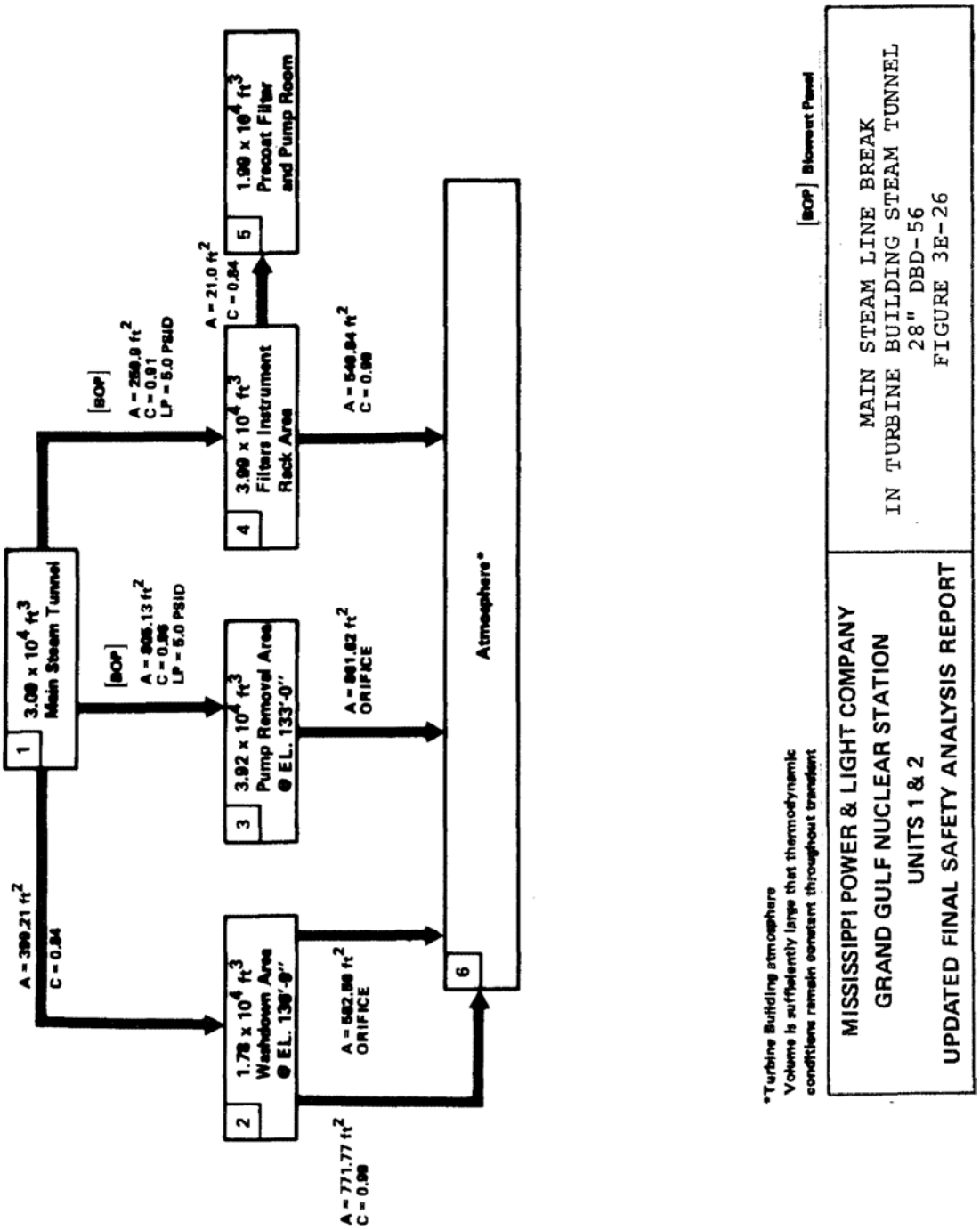

PRESSURE (PSIA)
TIME (SECONDS)
P1
P2
P3
P4
P5
MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT
28" DBD-56 MSL BREAK
IN STEAM TUNNEL
PRESSURES 1-5
FIGURE 3E-23

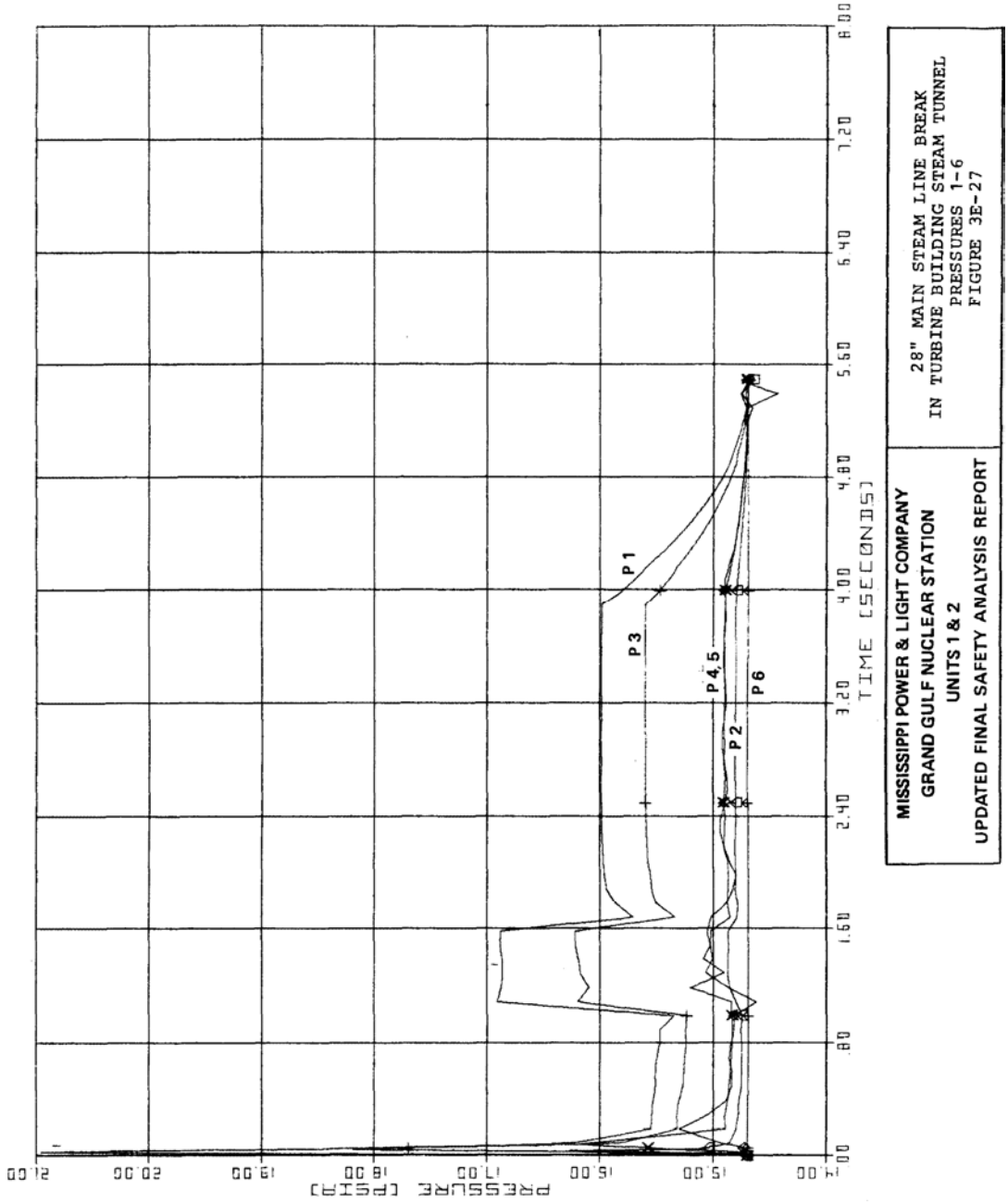

PRESSURE (PSIA)
TIME (SECONDS)
P 9
P 10
P 11
P 6
P 8
P 7
MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT
28" DBD-56 MSL BREAK
IN STEAM TUNNEL
PRESSURES 6-10
FIGURE 3E-24



NODALIZATION SCHEME
28" MAIN STEAM LINE BREAK IN
TURBINE BUILDING
FIGURE 3E-25

MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT


1
3.00 x 10^4 ft^3
Main Steam Tunnel
2
1.78 x 10^4 ft^3
Washdown Area
EL. 136'-0"
3
3.92 x 10^4 ft^3
Pump Removal Area
EL. 133'-0"
4
3.99 x 10^4 ft^3
Filters Instrument
Rack Area
5
1.99 x 10^4 ft^3
Precoat Filter
and Pump Room
6
Atmosphere*
A = 399.21 ft^2
C = 0.84
[BOP]
A = 866.13 ft^2
C = 0.86
LP = 5.0 PSID
[BOP]
A = 259.9 ft^2
C = 0.91
LP = 5.0 PSID
A = 21.0 ft^2
C = 0.84
A = 540.84 ft^2
C = 0.98
A = 881.62 ft^2
ORIFICE
A = 562.89 ft^2
ORIFICE
A = 771.77 ft^2
C = 0.98
[BOP] Blowout Panel
*Turbine Building atmosphere
Volume is sufficiently large that thermodynamic
conditions remain constant throughout transient
MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT
MAIN STEAM LINE BREAK
IN TURBINE BUILDING STEAM TUNNEL
28" DBD-56
FIGURE 3E-26


PRESSURE (PSIA)
TIME (SECONDS)
P1
P3
P4,5
P6
P2
MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT
28" MAIN STEAM LINE BREAK
IN TURBINE BUILDING STEAM TUNNEL
PRESSURES 1-6
FIGURE 3E-27

APPENDIX 3F COMPUTER PROGRAMS USED IN STRESS AND VIBRATION ANALYSIS OF EQUIPMENT

Computer programs used in stress and vibration analysis of equipment supplied by the General Electric Company can be divided into two categories. The first group of programs computes the response of structural systems. This includes the displacement, velocity, acceleration, stress, natural frequency, mode shape, and other physical quantities that are required to define equipment performance and to satisfy design codes.

The second group of programs is solely for data preparation and manipulation. They are used either to prepare input to the first group of programs or to process their output for a comprehensive comparison with design-code requirements.

This section of the report presents a description of the first group of programs, which are listed below in alphabetical order:

1. ANSYS
2. ASHSD
3. CREEP/PLAST
4. Dynamic Analysis of Piping Systems
5. GASP
6. General Matrix Manipulation Program
7. SAMIS
8. Heat Analysis Program
9. MEL
10. NASTRAN
11. OMP
12. Plate, Panel, and Space Structure Analysis Program
13. SAP
14. SHELL 5 and SHELL 9

15. Shell Analysis Program

16. STARDYNE

17. TIGER-5

18. Time Dependent Pipe Force

19. Pipe Dynamic Analysis Program

These computer programs are maintained either by General Electric or by outside computer program developers. In either case, an effort is made to control the quality of the programs and the computed results. For each program, one or more engineers are assigned. His duties are:

1. To keep abreast of the capability, the software contents, and the theory of the program.

2. To run test cases and maintain the reliability of the program.

3. To advise users on the proper usage of the program and the correct interpretation of computed results.

All necessary modifications are coordinated and verified by the responsible engineers. Thus users' confusion over the changes is avoided and the high reliability of these programs is maintained.

3F.1 ANSYS

Program Description

ANSYS is a general-purpose finite element computer program designed to solve a variety of problems in engineering analysis.

The ANSYS program features the following capabilities:

a. Structural analysis including static elastic, plastic and creep, dynamic, seismic and dynamic plastic, and large deflection and stability analysis.

b. One-dimensional fluid flow analyses.

c. Transient heat transfer analysis including conduction, convection, and radiation with direct input to thermal-stress analyses.

d. An extensive finite element library, including gaps, friction interfaces, springs, cables (tension only), direct interfaces (compression only), curved elbows, etc. Many of the elements contain complete plastic, creep, and swelling capabilities.

e. Plotting - Geometry plotting is available for all elements in the ANSYS library, including isometric and perspective views of three-D structures.

f. Restart Capability - The ANSYS program has restart capability for several analyses types. An option is also available for saving the stiffness matrix once it is calculated for the structure, and using it for other loading conditions.

Program Version and Computer

Two current versions of the program operating on the UNIVAC 1108 and on the CDC 6600 computers are used.

History of Use

[HISTORICAL INFORMATION] [The ANSYS program has been used for productive analyses in this country since early 1970. Users now number over 200 include the nuclear, pressure vessels and piping, mining, structures, bridge, chemical, and automotive industries, as well as many consulting firms.]

Extent of Application

Although the program has been extensively used by many users in this country, its application in GEH at present is limited to its nonlinear transient dynamic and thermal stress capabilities, of which both have been verified in GE/NED. Its application in GE/NED is expected to widen, when additional capabilities are verified by GEH engineers.

Test Problems

Results of several test problems solved using ANSYS are presented in an ASME publication (Reference 1.1). GEH engineers have re-analyzed those problems using the current version of ANSYS, and found nearly identical results. Verification of the ANSYS results using those of other programs and analyses are also available in Reference 1.1.

Reference

1.1 I. S. Tuba and W. B. Wright (Editors) “Pressure Vessel and Piping - 1972 Computer Programs Verification” ASME, New York, 1972.

3F.2 ASHSD

Program Description

ASHSD is a finite element program for the dynamic analysis of complex axisymmetric structures subject to asymmetric loads. The load cases which can be handled are dead load, arbitrary static load, arbitrary dynamic load, and horizontal or vertical earthquake acceleration of the base of the model. Dynamic solutions are solved through the time domain either by direct integration or mode superposition by a numerical step-by-step integration procedure.

Structures may be modeled as either shell elements discretized as a series of conical frustrums or solid elements of revolution with either quadrilateral or triangular cross-sections.

Time history loads are given at discrete time intervals with circumferential variations given as Fourier Components.

Program Version and Computer

[HISTORICAL INFORMATION] [The original version developed by S. Gosh and E. L. Wilson of the University of California at Berkeley in 1969 (Ref. 2.1) is being used in GE/NED. The version operates on the CDC 6600 computer.

History of Use

The program has been successfully used for seismic studies in GE/NED since 1971. Several other companies in the nuclear industry have also applied the program to their analyses.]

Extent of Application

The program is mainly used to perform seismic studies of axisymmetrical structures. The program's special elements simplifies the analysis which otherwise must employ three-dimensional elements. The cost for a three-dimensional analysis is prohibitive even in the case of a simple structure.

Test Problems

Results of two sample problems are presented in Figures 2.1 -2.6 with comparison to known solutions to the same problems.

References

2.1 S. Gosh and E. L. Wilson, "Dynamic Stress Analysis of Axisymmetric Structures under Arbitrary Loading," Earthq. Eng. Research Lab., University of California, Berkeley, 1969.

2.2 D. S. Perelman, R. A. Parmelie, and Seng-Lip Lee, "Seismic Response of Single-Story Interaction Systems," Journal of the Structural Division, ASCE, Vol. 94, No. ST 11, Nov. 1968.

2.3 S. C. Tang, "Response of a Finite Tube to Moving Pressure," Journal of the Engineering Mechanics Division, ASCE, Vol. 93, No. EM 3, June 1967.

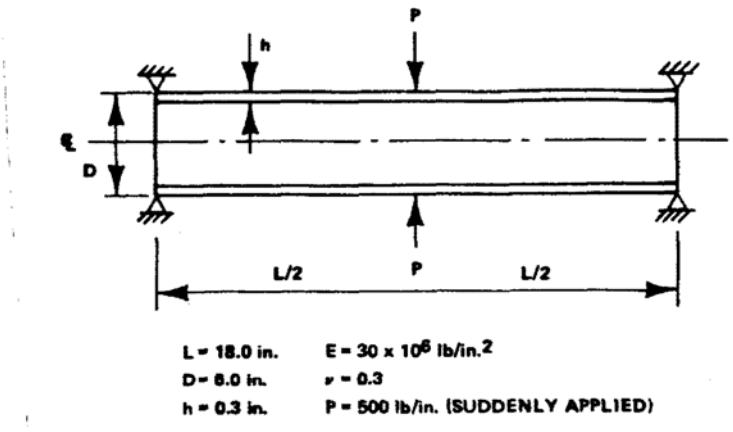


Figure 2.1. Suddenly Applied Ring (Line) Load

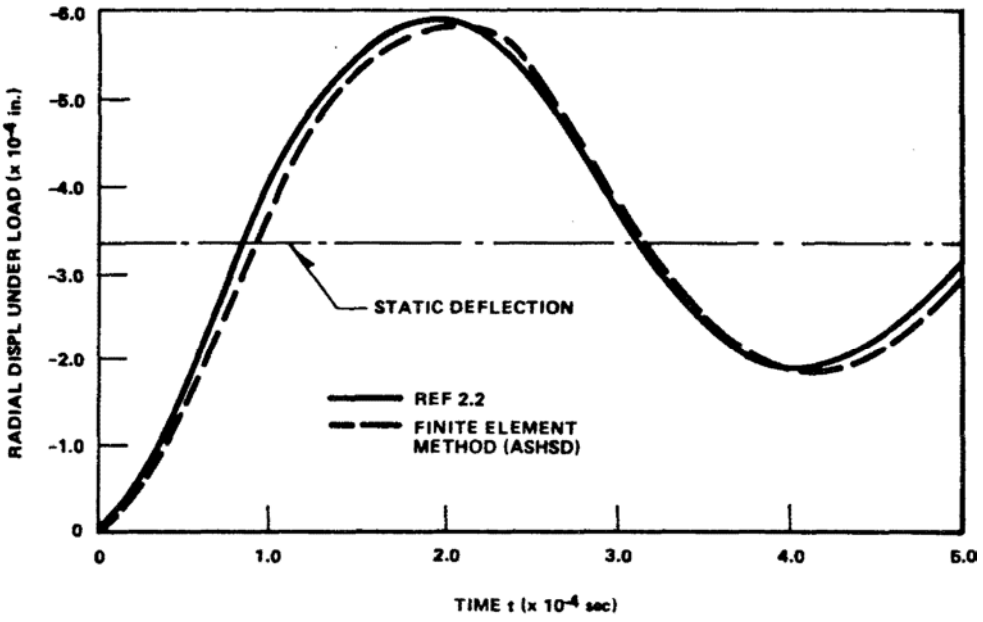


Figure 2.2. Radial Displacement Under Loads vs. Time
Suddenly Applied Ring Load on Cylindrical Tube

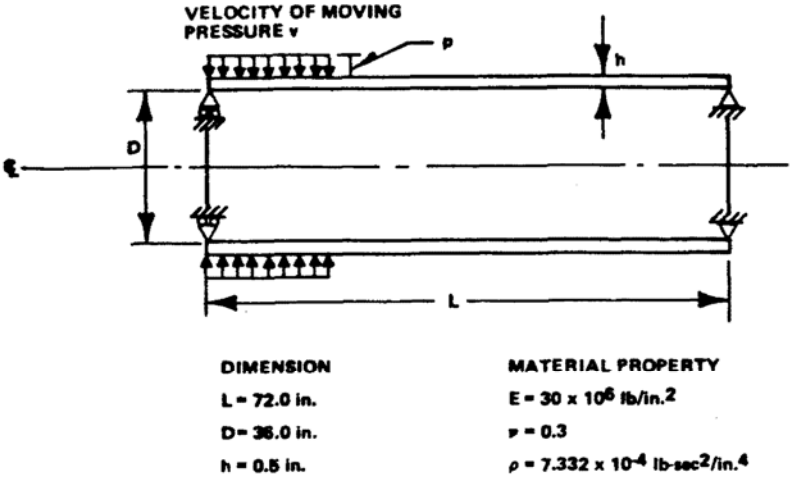


Figure 2.3. Moving Pressure Load

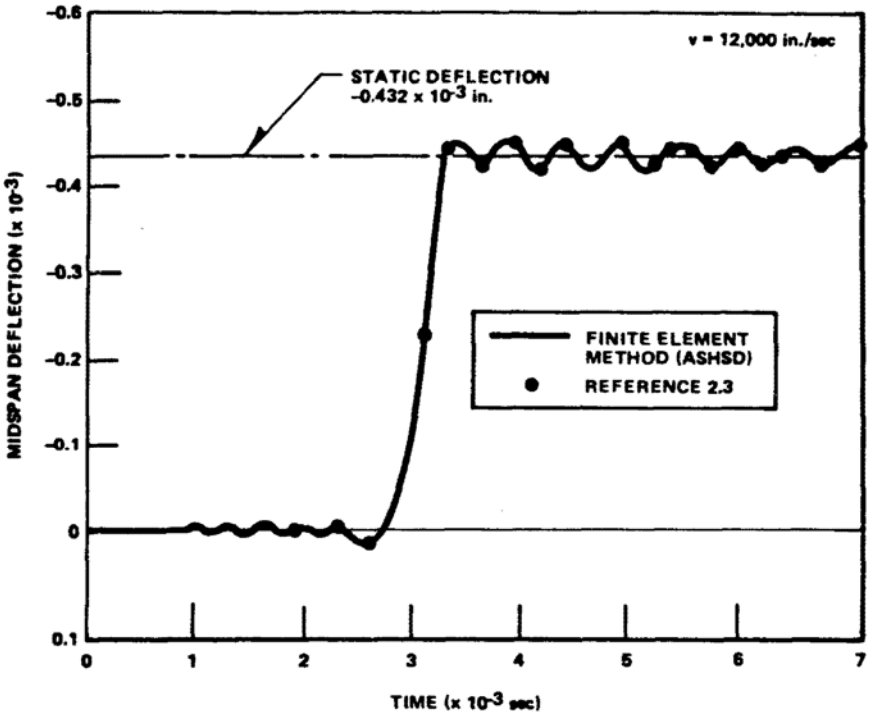


Figure 2.4. Response at Midspan for v = 12,000 in/sec.

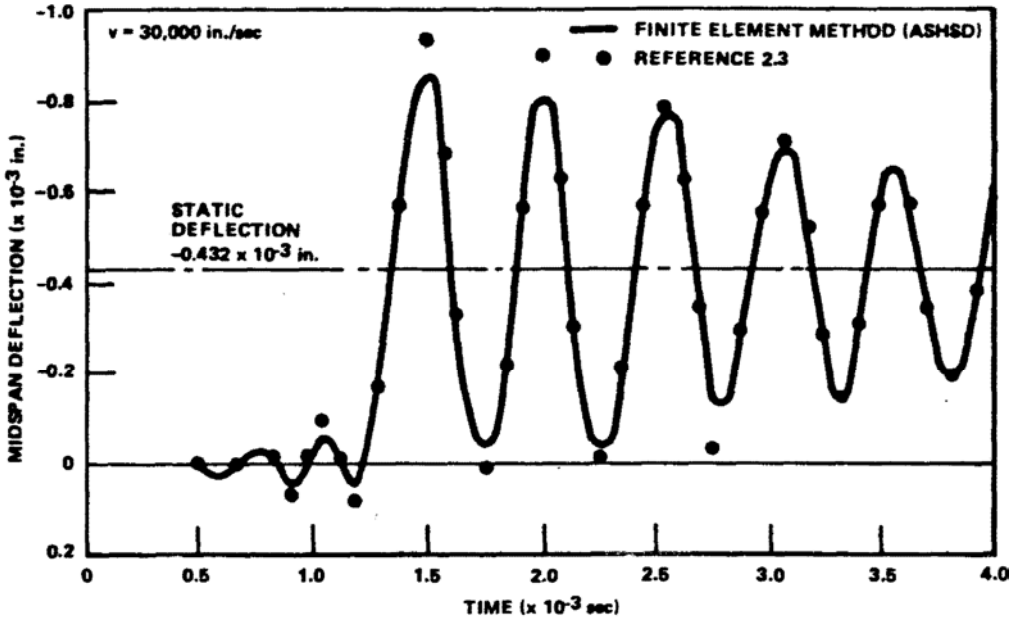


Figure 2.5. Response at Midspan for v = 30,000 in/sec.

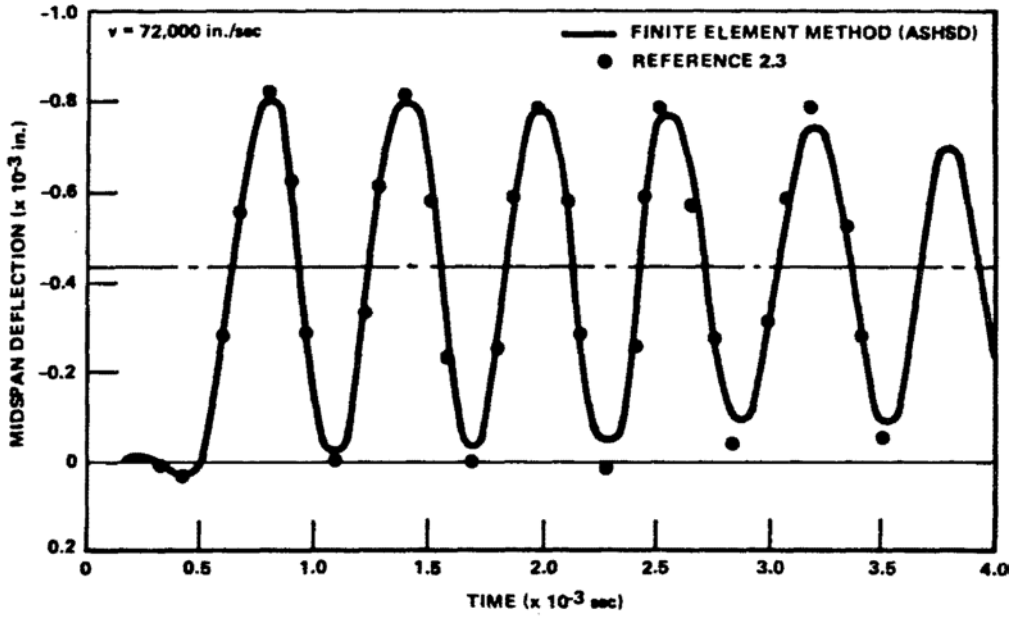


Figure 2.6. Response at Midspan for v = 72,000 in/sec.

3F.3 CREEP/PLAST

Program Description

CREEP/PLAST is a finite element program used for the analysis of two-dimensional (plane and axisymmetric) problems under conditions of creep and plasticity. The creep formulation is based on the memory theory of creep in which the constitutive relations are cast in the form of hereditary integrals. The material creep properties are built into the program and they represent annealed 304 stainless steel. Any other creep properties can be included if required. The plasticity treatment is based on kinematic hardening and the von Mises yield criterion. The hardening modulus can be constant or a function of strain. The elements are triangular with linear displacement approximations. The program can be used for elastic-plastic analysis with or without the presence of creep. It can also be used for creep analysis without the presence of instantaneous plasticity. A detailed description of the theory is given in Reference 3.1.

Program Version and Computer Used

The current version maintained by the original developer, Y. R. Rashid of GE/NED is used. The program operates on the UNIVAC 1108 computer.

History of Use

[HISTORICAL INFORMATION] [The original program was completed under the sponsorship of the AEC in 1971 (Ref. 3.1). Since its completion, it has been successfully applied to the analysis of various reactor components.

Results of a number of test problems have been verified by experimental and other numerical results.]

Extent of Application

The application of the program in GE/NED is mainly for analyses involving plasticity and creep at elevated temperatures. It is extensively used in fuel component analysis.

Test Problems

The analysis of three creep experiments under multi-step loading histories is shown in Figures 3.1 - 3.3. The memory theory solutions are compared with strain-hardening theory solutions and experimental data obtained by Oak Ridge National Laboratory (ORNL) (Ref. 3.2). The creep data are given in equations (4.12) to (4.15) of Reference 3.2. The agreement between the two calculation methods is very good. The difference in the two methods, as far as this particular problem stance, the memory theory solution is independent of the size of the time step when the stress state does not vary continuously with time. The intermediate points in the memory theory solution, as shown in the figures, were obtained to permit the drawing of a curve. In contrast, however, the strain-hardening theory requires sufficiently small time steps to permit the accumulated effective creep strain which enters into the flow rule explicitly, to be calculated accurately.

Another example shown in Figures 3.4 and 3.5 is a thick-walled cylinder under a two-step internal pressure loading as described in the illustration. Both illustrations show the comparison between the memory and hardening theories. The strain-hardening solutions were obtained on the basis of the auxiliary hardening rules described in Reference 3.2 by ORNL. The memory theory solution in Figure 3.4 did not include plasticity and therefore can be compared directly with ORNL analysis. As shown in Figure 3.5, however, plasticity was allowed to occur. The proportional limit for the material at the stated temperature is 10,060 psi. The effect of plasticity is, therefore, greater during the negative pressure phase. Comparing the memory solutions in Figures 3.4 and 3.5 it appears that the slower rate of relaxation, which occurred as a result of developing smaller peak stress, resulted in a higher final stress. The time steps in the memory theory solution ranged from 2 hours to 5,000 hours, resulting in a total of 42 time steps for the entire 60,000 hour history.

References

3.1 Y. R. Rashid, “Theory Report for Creep-Plast Computer Program, “GEAP-10546, AEC Research and Development Report, January 1972.

3.2 Greenstreet, W. L.; Corum, J. M.,; Pugh, C. E.,; and Liu, K. C., "Currently Recommended Constitutive Equations for Inelastic Design Analysis of FFTF Components," Oak Ridge National Laboratory, October 15, 1971 (ORNL-TM-3602).

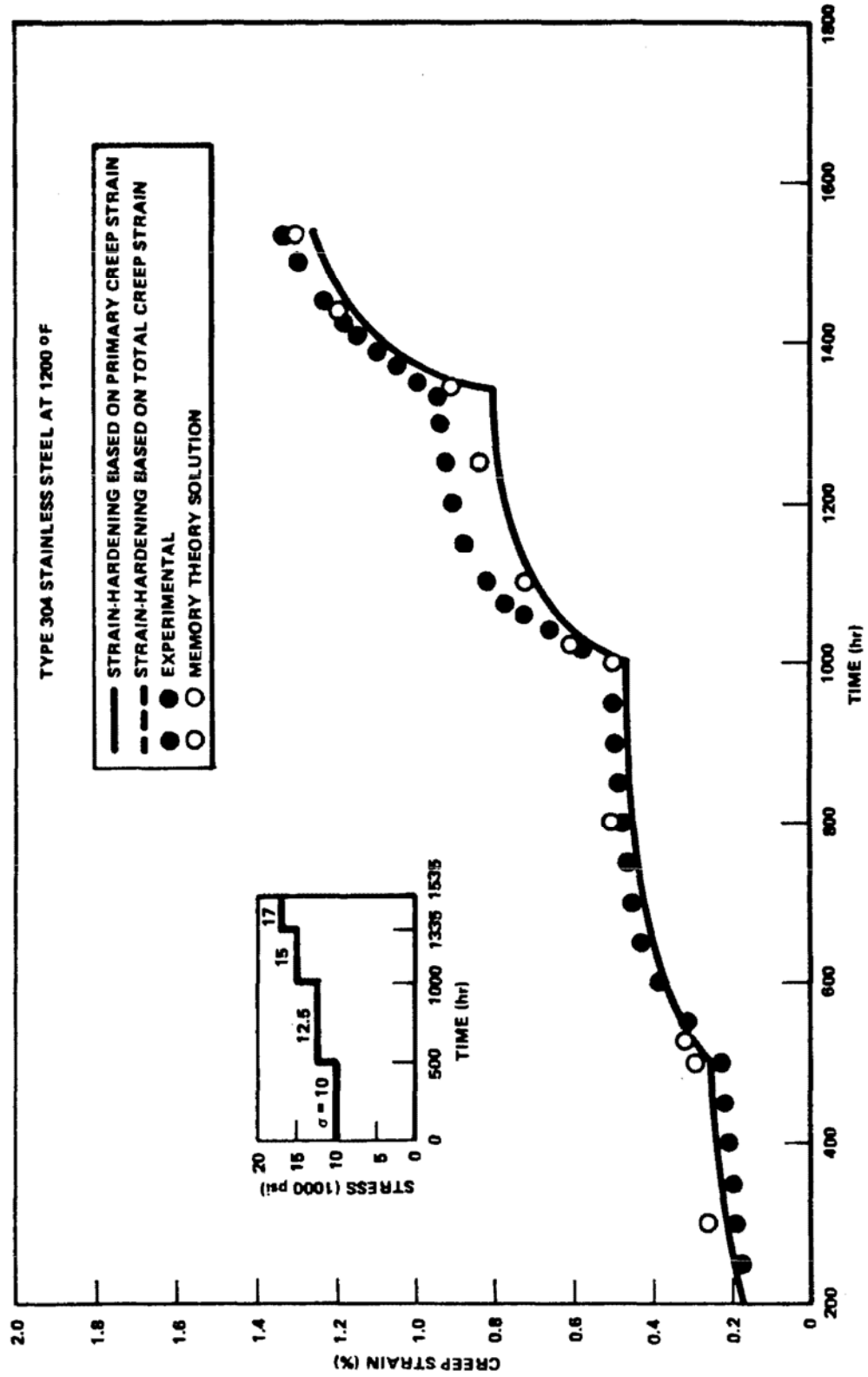


Figure 3.1. Step-Load Creep Test of Type-304 Stainless Steel at 1200 F

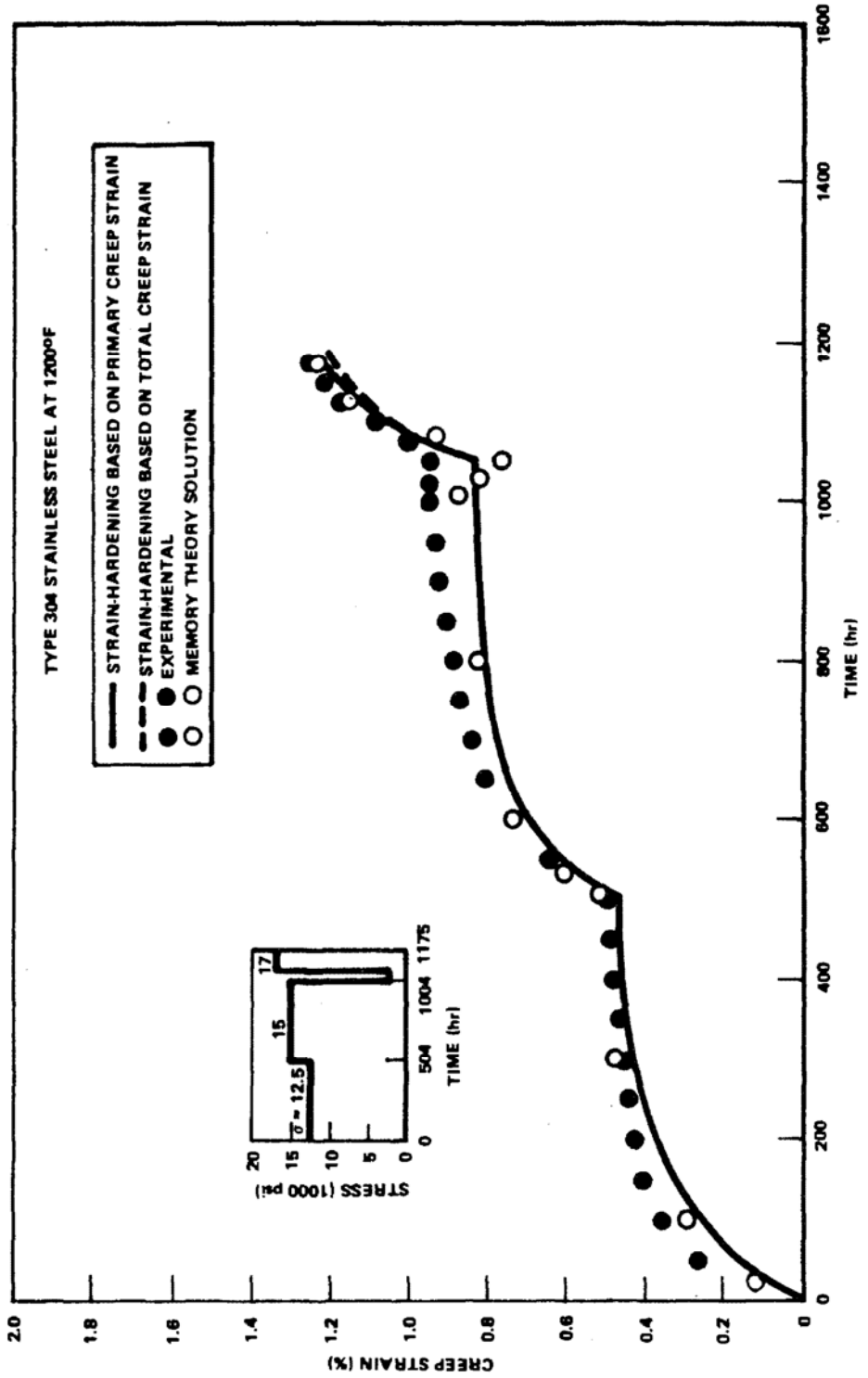


Figure 3.2. Step-Load Creep Test of Type-304 Stainless Steel at 1200 F

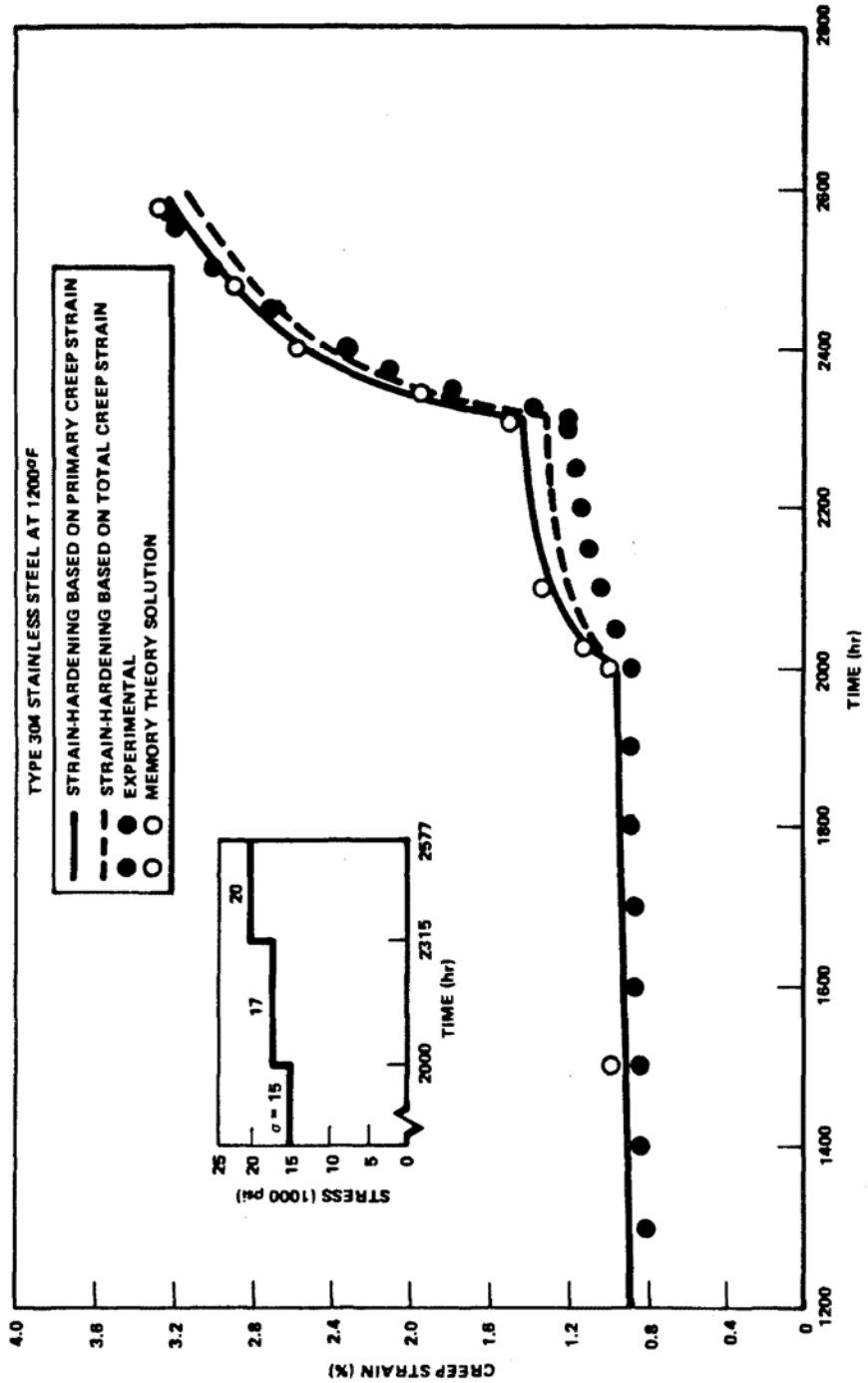


Figure 3.3. Step Load Creep Test of Type-304 Stainless Steel at 1200 F

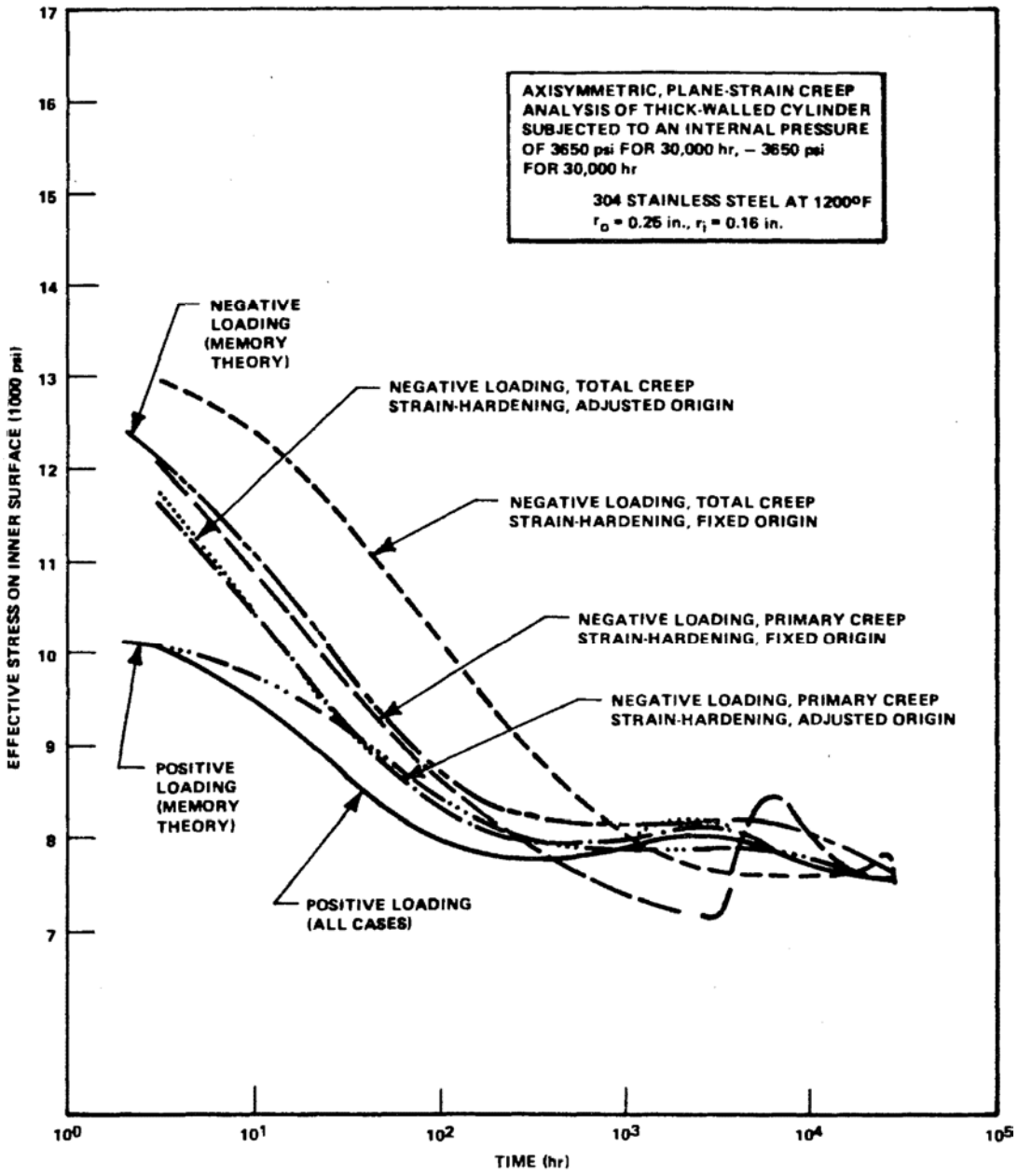


Figure 3.4. Predicted Response of Pressurized Thick-Walled Cylinder (Elastic Creep Analysis)

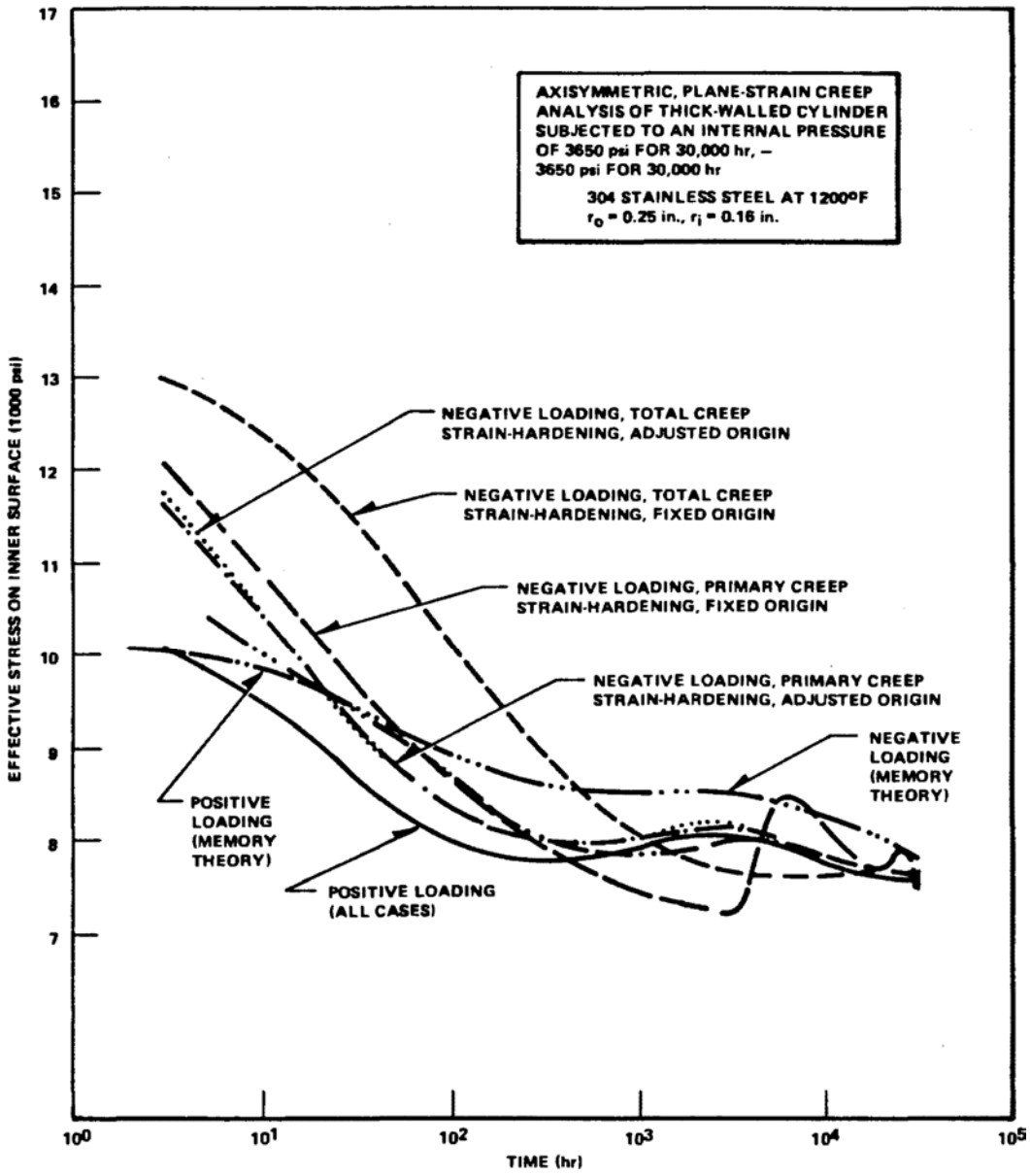


Figure 3.5. Predicted Response of Pressurized Thick-Walled Cylinder (Elastic-Plastic-Creep Analysis)

3F.4 DYNAMIC ANALYSIS OF PIPING SYSTEMS

Program Description

Dynamic Analysis of Piping Systems is a computer program for dynamic and seismic analyses of piping systems based on the computer code SAMIS (Structural Analysis and Matrix Interpretive System). SAMIS, which is described in Section 7 of this appendix, is a general program for matrix manipulation. To perform desired structural analyses, a user of SAMIS must direct the flow of calculations using a series of pseudo instructions. To minimize this effort, the Dynamic Analysis of Piping Systems Program has several packages of these pseudo instructions preassembled for dynamic and seismic analyses of piping systems. The computer program, operating on Honeywell H-6000 machines, can compute the natural frequencies, mode shapes, modal forces, modal displacements, local stresses, and displacements of a given piping system subject to a response spectrum input. The response spectrum technique is used in which the result is obtained as the square-root-of-the-sum-of-the-squares of modal values.

Program Version and Computer

The original version developed by D. Shen of GE/NED in 1972 is being used on the Honeywell 6000 computer (Ref. 4.1).

History of Use

[HISTORICAL INFROMATION] [The program has been in production status since 1972.]

Extent of Application

The program has been used as a special tool for dynamic analysis of reactor piping systems.

Test Problems

The reader is to refer to Section 7 of this appendix (SAMIS) for verification of the matrix manipulation packages used in this piping analysis program.

Reference

4.1 D. T. Shen, "DAPS - A Program for Dynamic Analysis of Piping System," GE/NED Document No. 383 HA721, October 1972.

3F.5 GASP

Program Description

GASP is a finite element program for the stress analysis of axisymmetric or plane two-dimensional geometries. The element representations can be either quadrilateral or triangular. Axisymmetric or plane structural loads can be input at nodal points. Displacements, temperatures, pressure loads, and axial inertia can be accommodated. Effective plastic stress and strain distributions can be calculated using a bilinear stress-strain relationship by means of an iterative convergence procedure.

Program Version and Computer

[HISTORICAL INFORMATION] [The GE version, originally obtained from the developer Professor E. L. Wilson, operates on the Honeywell 6000 computer.

History of Use

The program was developed by E. L. Wilson in 1965 (Ref. 5.1). The present version in GE/NED has been in operation since 1967.]

Extent of Application

The application of GASP in GE/NED is mainly for elastic analysis of axisymmetric and plane structures under thermal and pressure loads. [HISTORICAL INFORMATION] [The GE version has been extensively tested and used by engineers in the company.]

Test Problems

The ASME computer-program-verification problem 19 (Ref. 5.2) was solved using the triangular elements of GASP and ANSYS (Figure 5.1). The results of both solutions are very close to each other and they agree with a closed-form solution given in Reference 5.2 (Figure 5.2).

References

5.1 E.L. Wilson, "A Digital Computer Program for the Finite Element Analysis of Solids With Non-linear Material Properties," Aerojet General Technical Memo No. 23, July 1965.

5.2 I.S. Tuba and W. B. Wright (Editors), "Pressure Vessel and Piping - 1972 Computer Program Verification," ASME, New York, 1972.

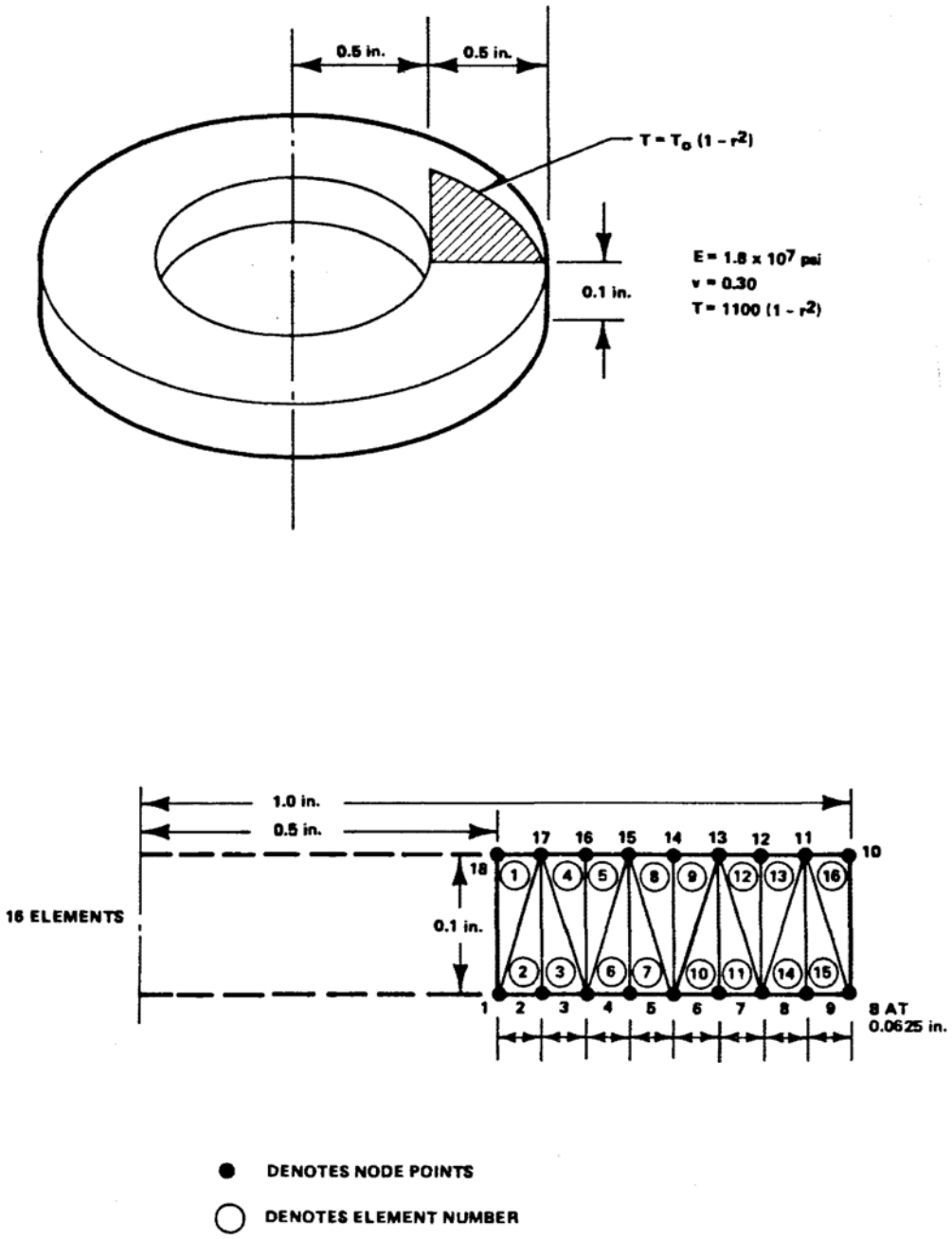


Figure 5.1. ASME Computer Program Verification Problem No. 19 and Finite Element Model

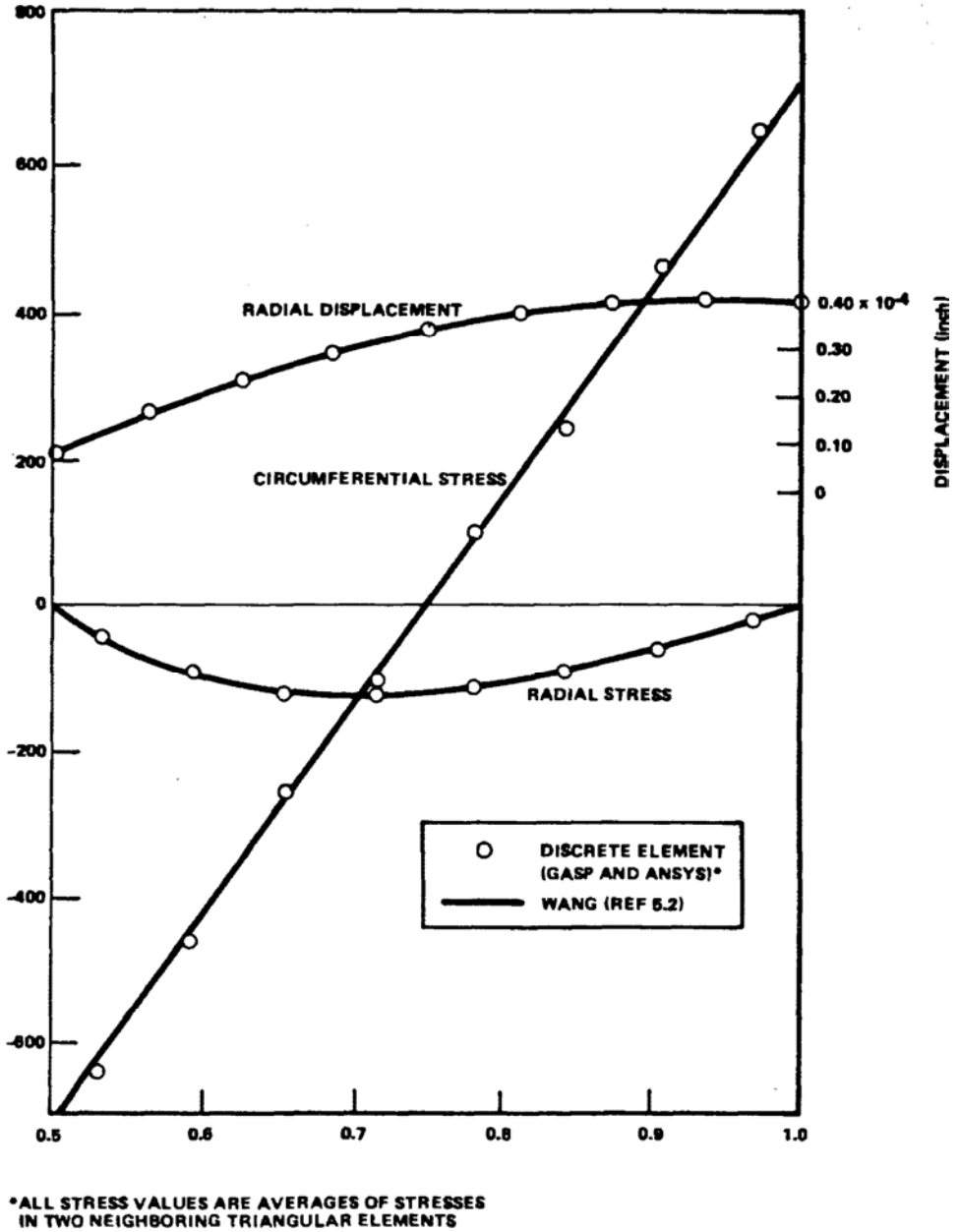


Figure 5.2 Comparison of Analytical Results for ASME Problem 19

3F.6 GENERAL MATRIX MANIPULATION PROGRAM

Program Description

General Matrix Manipulation Program is a general matrix manipulation program capable of performing the majority of standard matrix operations. There presently are 41 operation commands in the program. A maximum of nine full 60 x 60 matrices and 6-60 element vectors may be stored in-core at any one time. Also available for scratch and storage are up to a maximum of three tapes. This latest version of the program includes subroutines for calculating earthquake, or other forcing functions, and response of a lumped mass structure, either by time-history or spectral response methods. The most used features are the eigenvalue, eigenvector subroutine and response subroutine. The response is calculated for a system subjected to any piecewise linear forcing function.

Program Version and Computer

The current version of the program being used in GE/NED was obtained from the originator, the General Electric Knolls Atomic Power Laboratory, in June, 1969. It was converted from the CDC to GE computer. The program is now installed on the Honeywell 6600 computer which is essentially a modification of the original GE computer.

History of Use

[HISTORICAL INFORMATION] [The program was originally written in the General Electric Knolls Atomic Power Laboratory for the solution of vibration problems. In 1969, it was converted and modified by the General Electric Nuclear Energy Division to use on its GE/Honeywell computer for the solution of seismic problems.

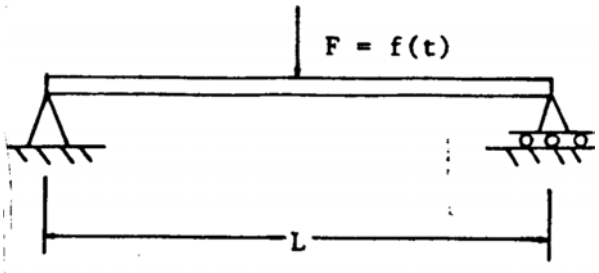
Extent of Application

Since its installation in GE/NED in 1969, the General Matrix Manipulation Program has been constantly used to solve seismic problems involving small lumped-mass systems of less than 80 degrees of freedom. Because of its limitation on problem size, the program is being replaced by SAMIS (Section 7).]

Test Problems

To evaluate its capability, the computer program has been used to solve the following sample problem. The satisfactory agreement between the lumped-mass numerical solution and the continuous-system theoretical solution indicates the reliability of the General Matrix Manipulation Program.

Consider a simply supported beam with a suddenly applied load at its center as the problem.



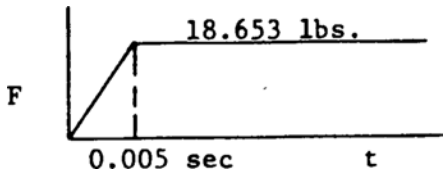
L = 100 in.

E = 30 x 10^6 psi

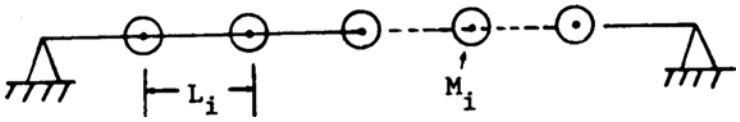
I = 2.5907 $in.^4$

A = 9.8696 $in.^2$

ρ = 0.3 $lb/in.^3$



For this problem the beam is modeled as a five lumped-mass system, or



all lengths are equal

$L_i = 1/6 = 16.667$ in.

all masses are equal

$$M_i = \frac{M}{6} = \frac{A\ P}{6g} = \frac{(9.8686)\ (100)\ (0.3)}{6.\ (386)} = 0.01278$$

A comparison of the calculated natural frequencies of the lumped-mass system compared with the continuous system shows very good agreement.

Beam Natural Frequencies

Mode	GEMϕP Lumped-Mass Frequency, Hz	Continuous System Frequency, Hz	GEMϕP Error, %
1	5.00	5.00	0.0
2	19.98	20.00	0.1
3	44.66	45.00	0.8
4	77.33	80.00	3.3
5	110.59	125.00	11.5

The maximum displacement response of the beam is calculated considering zero damping and a cutoff frequency of 45 Hz. With this cutoff frequency, the calculated response included the effect of modes 1, 2, and 3. The calculated maximum response for this second case was 0.997 in., compared with the theoretical value of 1.00 in. This indicates an error of only 0.32 percent.

3F.7 SAMIS

Program Description

The SAMIS program is well designed to solve problems involving matrix algebra with particular emphasis on structural applications (Refs. 7.1, 7.2, and 7.3). The user has control over the flow of the calculations through the use of pseudo instructions. Execution of the program is performed in two phases - the generation phase and the manipulative phase. Input data defining the idealization of a structure is read and stiffness, stress, and load coefficient matrices are generated for elements available to the user. The program has two fundamental and widely used finite elements incorporated. A triangular flat plate element, called a Facet, is available for idealization of plate and shell structures and a straight beam element is available for idealization of frames and trusses and plate/shell structure stiffener representation.

The element formulation and analyses are based on the finite element matrix displacement method. The triangular plate and beam elements are capable of resisting stretching, shearing, bending, and twisting stresses. In the second phase of execution, the generated or input matrices are manipulated according to the rules of matrix algebra as directed by the user.

The program is written in modular form making it easy to add new modules without major reprogramming of subroutines. This facilitates adding to the structural element library other elements to extend idealization capability. Those structural problems consisting of elements that cannot be adequately idealized by triangular plate or beam elements may have their stiffness coefficients submitted directly as input matrices.

Program Version and Computer

[HISTORICAL INFORMATION] [The SAMIS version, now operating on the Honeywell 6000 computer of GE/NED, was obtained from the developer, Philco Corporation, Western Development Laboratory (WDL) via the General Electric Space Division. A considerable amount of modification was made to the input and output of the original version to suit the analysis need of this division of GE. Both spectrum and time-history analyses can be performed using the GE/NED version.

History of Use

The SAMIS program was developed by the Philco Corporation, Western Development Laboratory (WDL) under contract to and in association with the Jet Propulsion Laboratory in 1966. The program was first used in the General Electric Company in 1967 and in this division of GE in 1970.]

Extent of Application

The current GE/NED version of SAMIS has been extensively used in the analysis of reactor components' response to seismic loadings since 1970. Results of test problems were found to agree closely with theoretical results of the same problem.

Test Problem

Clamped Square Plate Frequency Study

Consider a square plate having all four of its edges clamped. Figure 7.1 shows a quarter panel of the plate.

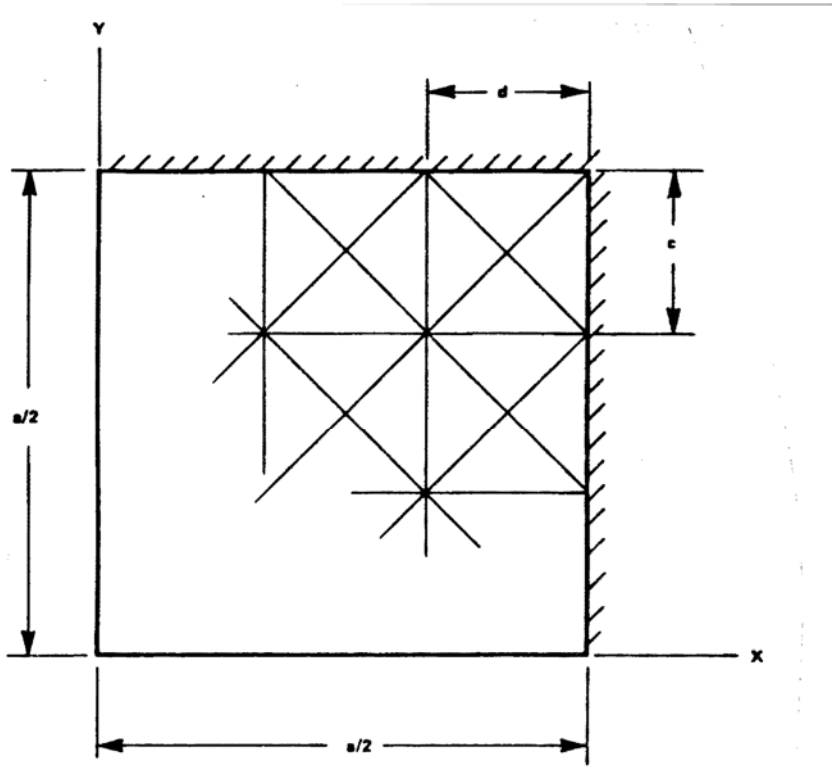


Figure 7.1. Clamped Square Plate Quarter Panel

The particular plate dimensions and material properties used in the analysis were as follows:

$a = 10"$

$t = 0.05"$

$E = 30 \times 10^6$ lbs/in^2

$\nu = 0.03$

$\rho = 7.26 \times 10^{-4}$ lbs-sec^2/in^4

For all grids used in the study, the triangular elements in any one grid were all uniform size except as noted.

A total of four grid models were analyzed for the clamped plate, each based on selected value of a/c = a/d or c = d for the square plate models. Each case will be referred to by number. In the table below are listed four cases and the grid size function for each case.

Case No.	a/c, a/d
1	4
2	5
3	6
4	7

For each case, three sets of modes were computed: symmetric modes, antisymmetric modes, and mixed modes (symmetric about X and antisymmetric about Y). The fourth set of modes was not required since the plate is square and the remaining mixed mode set (symmetric about Y and antisymmetric about X) would only be duplicate frequencies of the first mixed mode set.

Table 7.1 is a list of computed frequencies for the clamped square plate models. The first six frequencies are listed for each case along with the percent variation from the theoretical frequencies. The table shows, as would be expected, successive improvement in computed frequencies as the grid is refined. The results for case number 4 show rather good agreement for all six frequency values; the largest difference being in the third and sixth frequency values which differ from the exact frequency by 12.6% and 13.3%, respectively. A number of other models could be

analyzed using larger grid size functions; however, it is doubtful that the improvement in accuracy would be as marked as in the initial four models for the six frequencies.

References

7.1 Melosh, R. J., et al, "Structural Analysis and Matrix Interpretive System (SAMIS) Program Report Revision 1," Jet Propulsion Laboratory, Pasadena, California, Technical Memorandum 33-307, December 15, 1966.

7.2 Melosh, R. J., and Christiansen, H. N., "Structural Analysis and Matrix Interpretive System (SAMIS) Program: Technical Report," Jet Propulsion Laboratory, Pasadena, California, Technical Memorandum 33-311, November 1, 1966.

7.3 Lang, T. E., "Structural Analysis and Matrix Interpretive System (SAMIS) User Report," Jet Propulsion Laboratory, Pasadena, California, Technical Memorandum 33-305, March 1, 1967.

TABLE 7.1: Clamped Square Plate Theoretical and Computed Frequency Comparison

Case No.	f_1 *	% Diff.	f_2	% Diff.	f_3	% Diff.	f_4	% Diff.	f_5	% Diff.	f_6	% Diff.
Theory	176.5	X	359.9	X	530.9	X	645.5	X	648.6	X	810.0	X
1	205.6	16.5	438.0	21.7	817.4	54.0	727.7	12.7	769.3	18.6	1476.0	82.2
2	190.7	8.0	409.1	13.8	696.5	31.2	672.7	4.2	702.3	8.3	1076.1	32.9
3	183.8	4.1	391.2	8.7	600.2	13.5	705.1	9.2	719.2	10.9	999.0	23.3
4	183.8	3.8	380.7	5.8	597.9	12.6	657.3	1.8	670.4	3.4	917.4	13.3

* Frequency values f_1 - f_6 are in cps

3F.8 HEAT ANALYSIS PROGRAM

Program Description

The Heat Analysis Program is a two-dimensional and axisymmetric transient nonlinear temperature analysis program. An unconditionally stable numerical integration scheme is combined with an iteration procedure to compute the temperature distribution within the body subjected to arbitrary time and temperature dependent boundary conditions.

This program utilizes the finite element method. Included in the analysis are the three basic forms of heat transfer (conduction, radiation, and convection), as well as internal heat generation. In addition, cooling pipe boundary conditions are also treated. The output includes temperature of all the nodal points for the time instants required by the user. The program can handle multitransient temperature input.

Program Version and Computer

[HISTORICAL INFORMATION] [The current version of the program is an improvement of the program NOHEAT developed by I. Farhoomand and Professor E. L. Wilson of the University of California at Berkeley. The program operates on the Honeywell 6000 computer (Ref. 8.1).

History of Use

The program was developed in 1971 and installed in General Electric Honeywell computer by one of its original developers, I. Farhoomand, in 1972. After extensive testing it was finally put in production status in 1973. A number of heat transfer problems related to the reactor pedestal have been satisfactorily solved using the program.

Extent of Application

The program is replacing TIGER-5 (Section 17) in the analysis of axisymmetric heat transfer problems in reactor components.] The program uses the finite element formulation which is compatible with other finite element stress-analysis computer programs. Such compatibility simplifies the connection of the two analyses and minimizes human error.

Test Problems

Problem 1. One-dimensional temperature response of a plate with insulated back face after a sudden change in the external front surface temperature.

The prescribed heat-input boundary condition is useful in problems of aerodynamics, nuclear reactor power plants, and similar problems. The temperature response of a plate caused by a sudden change in external front surface temperature has been solved by the direct integration technique (Ref. 8.2).

Figure 8.1 shows the response in a non-dimensional form. The following variables are used in the description of the figure.

T_o = the initial temperature of the figure

T_e = the boundary temperature

α = thermal diffusivity of body material = k/c

k = conductance

ρ = density

c = specific heat

t = time

L = plate thickness

T = temperature of the body at time t

A linear finite element analysis was conducted using T = 0, T_e = 100, k = .0006, ρ = .3, c = .1, L = 100. The finite element mesh is shown in Figure 8-2. The result of the finite element analysis (dark circles in Figure 8-1 indicates excellent agreement with the result of the analytical solution.

Problem 2. Temperature response of the front face of a plate with insulated back face after sudden exposure to constant-temperature radiation.

This example demonstrates the accuracy of the finite element as compared with an analytical solution of a radiation heat transfer problem. Consider a plate of thickness L with finite conductivity

k, specific heat c and density ρ. The back face of the plate is insulated and the front face is subjected to a constant radiation heat flow with sink temperature (T_s = 0°R). The time variation of the temperature of the front surface was computed in a non-dimensional form by the application of an approximate analytical technique (Ref. 8.2). Figure 8.3 presents the temperature response for a particular value of parameter M. The variables used to construct Figure 8.3 are:

$\sigma = 17.3 \times 10^{-10}$ Btu/hr ft^4/°R

F_e = shape factor for the plate surface

F_A = shape factor for the sink

T_o = initial temperature of the plate (°R)

T_e = surface temperature of the plate (°R)

Several finite element analyses with the same time increment but different numbers of iterations within each cycle were conducted. The finite element mesh layout is shown in Figure 8.4. In the analyses:

$F_A = F_e = 1$, $T_o = 500$, $M = 1$, $\alpha = k/\rho c = 0.02$ and $T_s = 0$°R

Comparison of the finite element solutions with the analytical solution in Figure 8.4 indicates that the incremental approach without iteration does not converge to the "true" solution. However, if one iteration is made in each time increment, the finite element result becomes nearly identical to the analytical result.

References

8.1 I. Farhoomand and E. L. Wilson, "Non-Linear Heat Transfer Analysis of Axisymmetric Solids," SESM Report SESM71-6, University of California, Berkeley, California 1971.

8.2 P. J. Schneider, "Temperature Response Charts," John Wiley & Sons, Inc., New York, 1963.

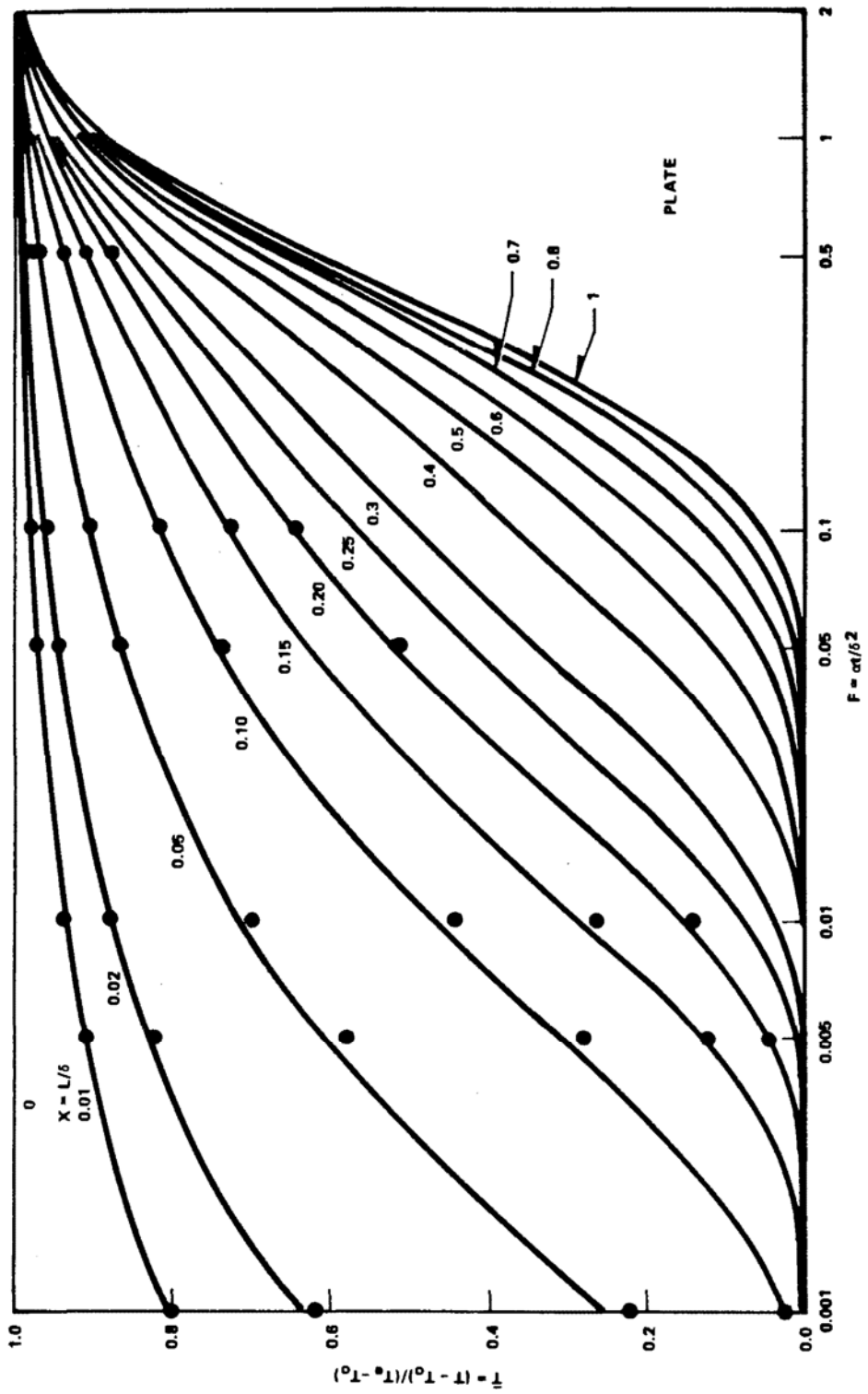


Figure 8.1. Temperature Response of Plate Through the Thickness

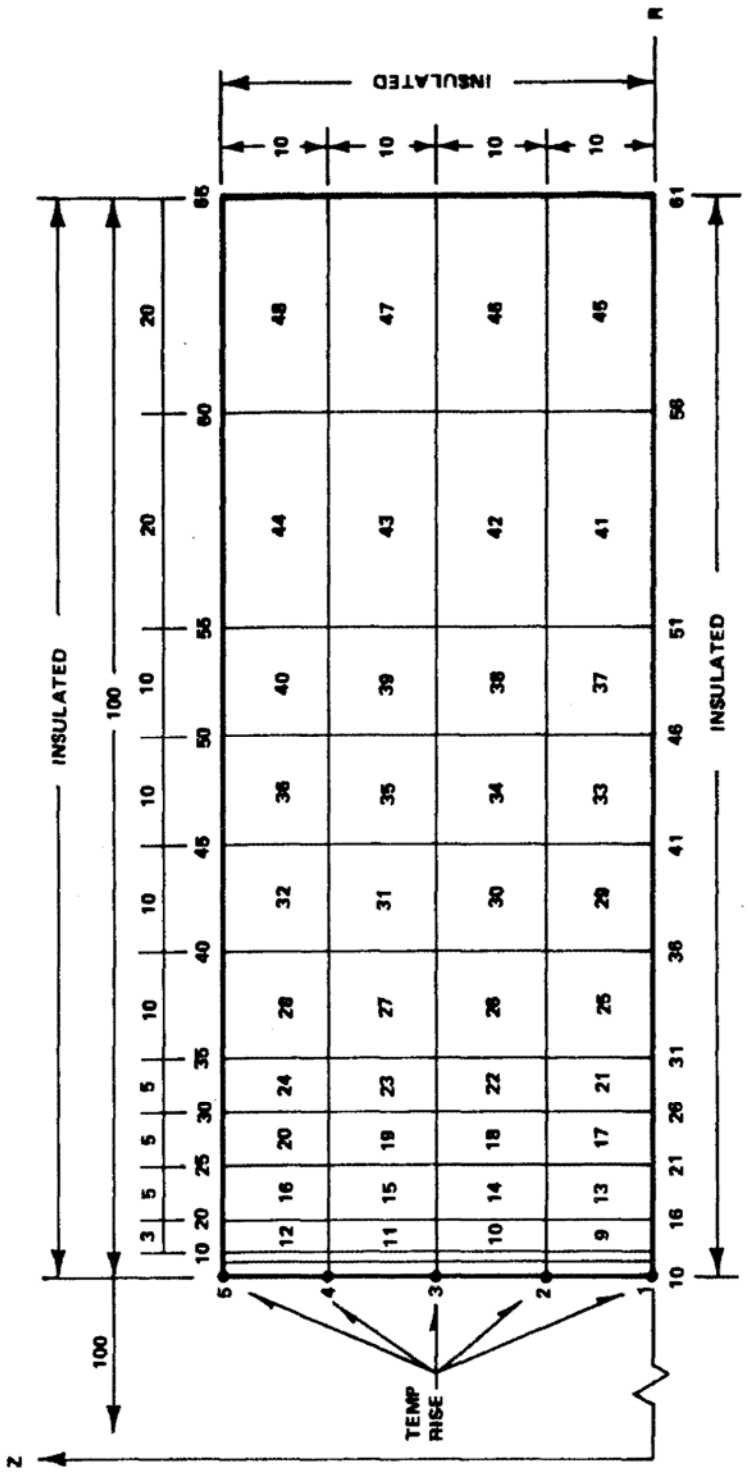


Figure 8.2. Finite Element Mesh Layout

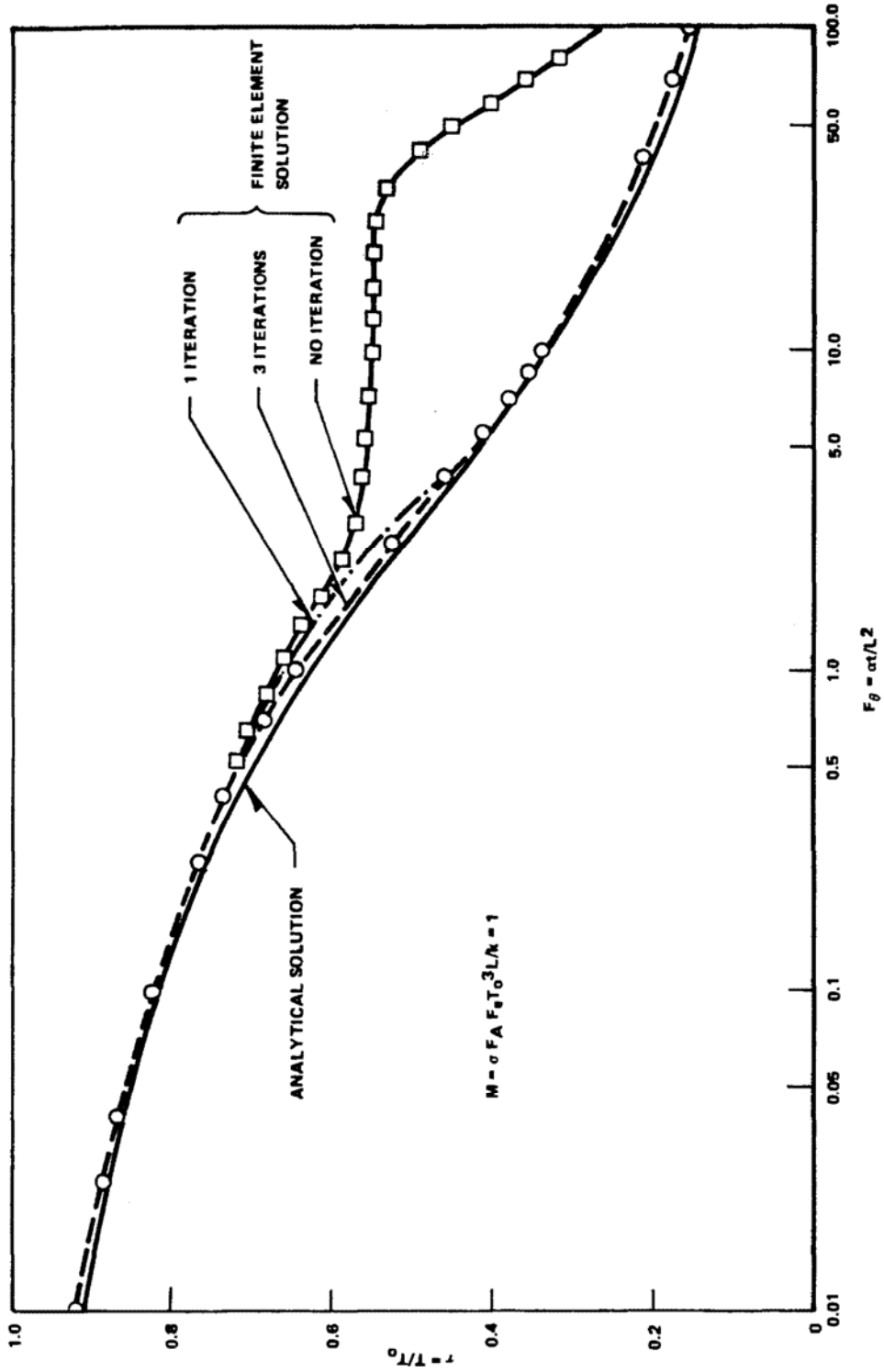


Figure 8.3. Temperature Response to a Plate Subjected to Radiation

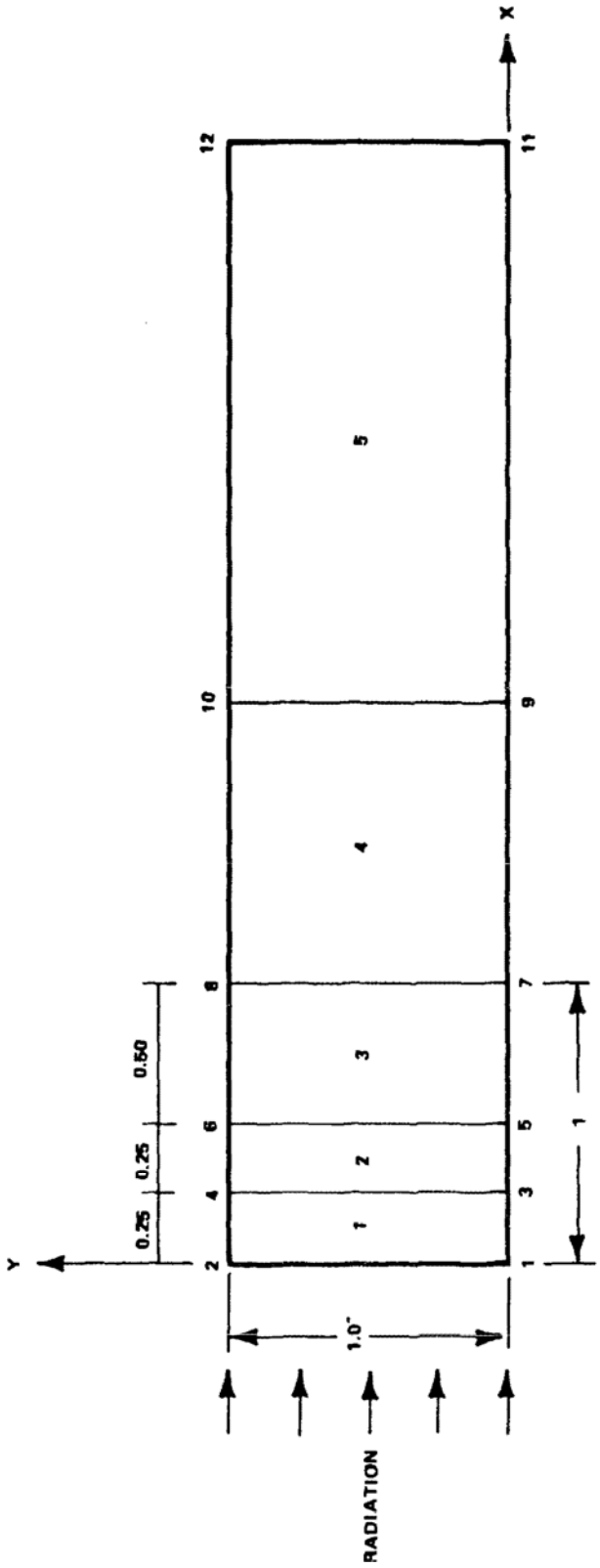


Figure 8.4. Finite Element Mesh Layout

3F.9 MEL

Program Description

MEL is a computer program using finite-element matrix analysis methods to analyze the flexibility of multiple branch and closed-loop piping systems subject to thermal, uniform, and concentrated loadings, and is also applicable on cryogenic piping systems. All computations are performed in accordance with the requirements of ASA B31.1-1955, American Standard Code for Pressure Piping. As an incidental feature, the program is also able to solve certain structural problems.

Program Version and Computer

[HISTORICAL INFORMATION] [The GEH program is an expansion and elaboration of MEL-21P, a piping flexibility calculation computer program developed jointly by the Mare Island Naval Shipyard Design Division and the Los Alamos Scientific Laboratory Engineering Department during the 1960s. The GE version operates on the Honeywell 6000 computer.

History of Use

The original program, one of the better known programs in the piping area, has been extensively used and verified through comparisons with results from model tests and other piping flexibility analysis programs. The GE version was first put on the Honeywell machine in 1968. It has since been verified with results of the original MEL-21P.] The current GEH version matches in capability the one existing in the public domain. MEL, however, is gradually being replaced by SAP in GEH.

3F.10 NASTRAN

Program Description

NASTRAN is a general or multipurpose finite element program specifically designed for, but not limited to, large problems with many degrees of freedom. The program is extremely versatile and includes the following features.

a. 1 and 2D elastic elements (bar, rod, shear and twist panels, plate elements, conical shell element, axisymmetric shell element, scalar elements including springs, dampers, and mass).

b. Applied loads: static, frequency dependent, time dependent.

c. Solutions available:

   1. Static
   2. Normal mode
   3. Buckling
   4. Direct complex eigenvalue
   5. Direct frequency and random response
   6. Direct transient response
   7. Modal complex eigenvalue analysis
   8. Modal frequency and random response
   9. Modal transient response
   10. Direct matrix abstraction (user-oriented language providing user with flexibility for manipulation not contained in rigid formats)

d. Plotting - structure plotting and X-Y deformed and undeformed structure; transient deformation.

e. Restart capability from "check point" old program tape.

Program Version and Computer

The current CDC version (level 15) of NASTRAN is being used by GEH.

History of Use

[HISTORICAL INFORMATION] [NASTRAN, an acronym of NASA Structural Analysis, is a general structural program developed by the agency for the analysis of space vehicles. Its development, which began in the early 1960s, is being continuously supported by this agency and its application is being widened to other industries such as the building, automobile, and nuclear power industries.

Extent of Application

The program has been widely used by many companies in this country. In the GEH Nuclear Energy Division, the application of NASTRAN is mainly confined to the vibration analysis of reactor components.]

Test Problems

Vibration of a plate

This problem demonstrates the solution for natural frequencies of a large-order problem. The structural problem consists of a square plate with hinged supports on all boundaries. The 10 x 20 model, as shown in Figure 10.1, uses one-half of the structure and symmetric boundary constraints on the mid-line in order to reduce the order of the problem and the bandwidth by one-half. The 20 x 40 model is the same dimension with a finer mesh.

Table 10.1 lists the NASTRAN and theoretical natural frequencies as defined in Reference 10.1. Figures 10.2 and 10.3 are comparisons of the first two mode shapes.

Additional test problems can be found in Reference 10.2.

TABLE 10.1: Natural Frequency Comparisons, cps.

Mode No.	Theoretical	NASTRAN 10 x 20	NASTRAN 20 x 40
1	.9069	.9056	.9066
2	2.2672	2.2634	
3	4.5345	4.5329	

Eigenvalue extraction data:

Method:	Inverse power
Region of interest:	$.89 \leq f \leq 1.0$
Number of desired roots:	3
Number of estimated roots:	1

References

10.1 W. F. Stokey, "Vibration of Systems Having Distributed Mass and Elasticity," Chapter 7, Shock and Vibration Handbook, **C.** M. Harris and C. E. Crede (Editors), McGraw-Hill, 1961.

10.2 "NASTRAN Demonstration Problem Manual (Level 15)", NASA SP-224(01), June 1972.

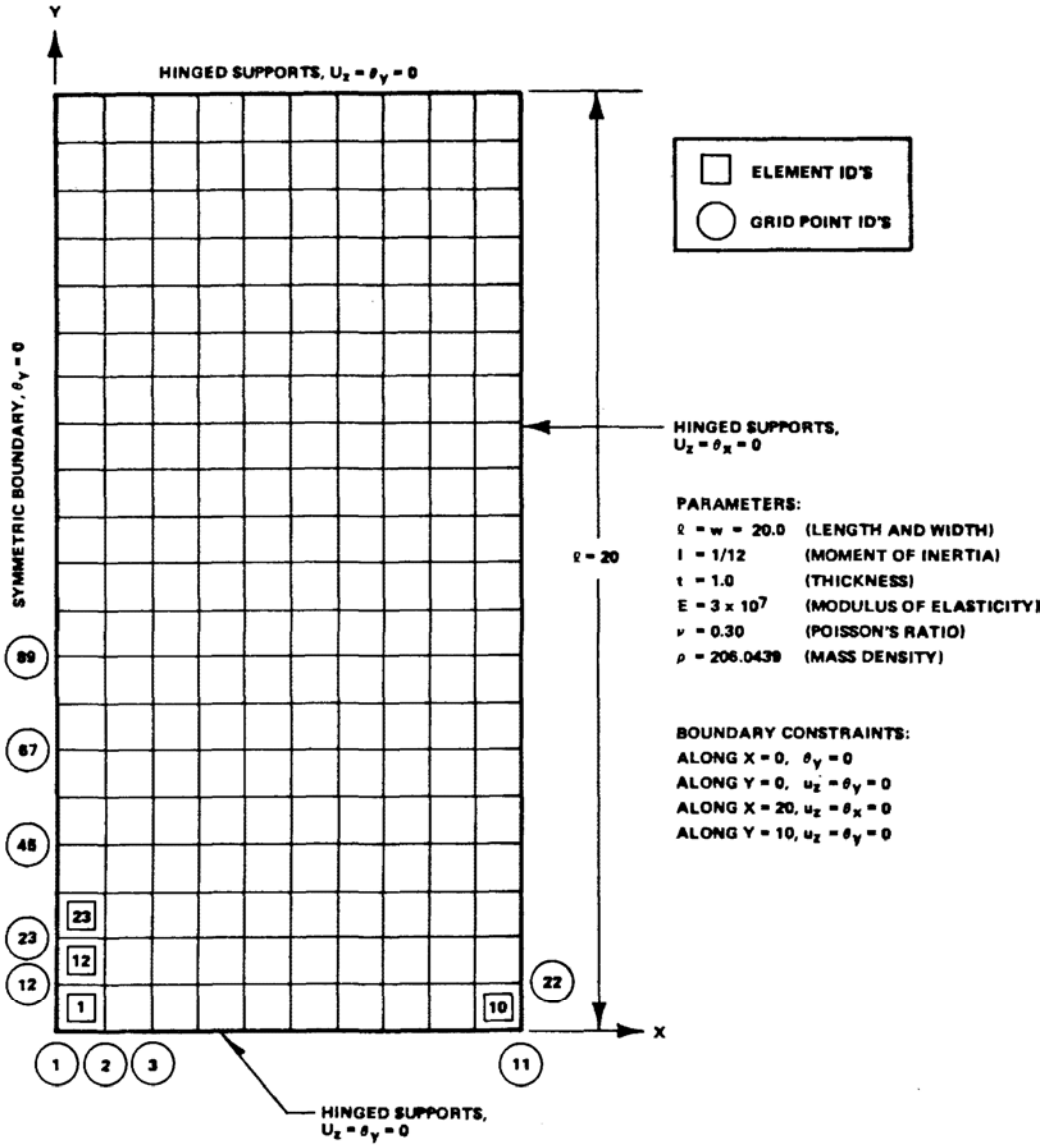


Figure 10.1 10 x 20 Half Plate Model

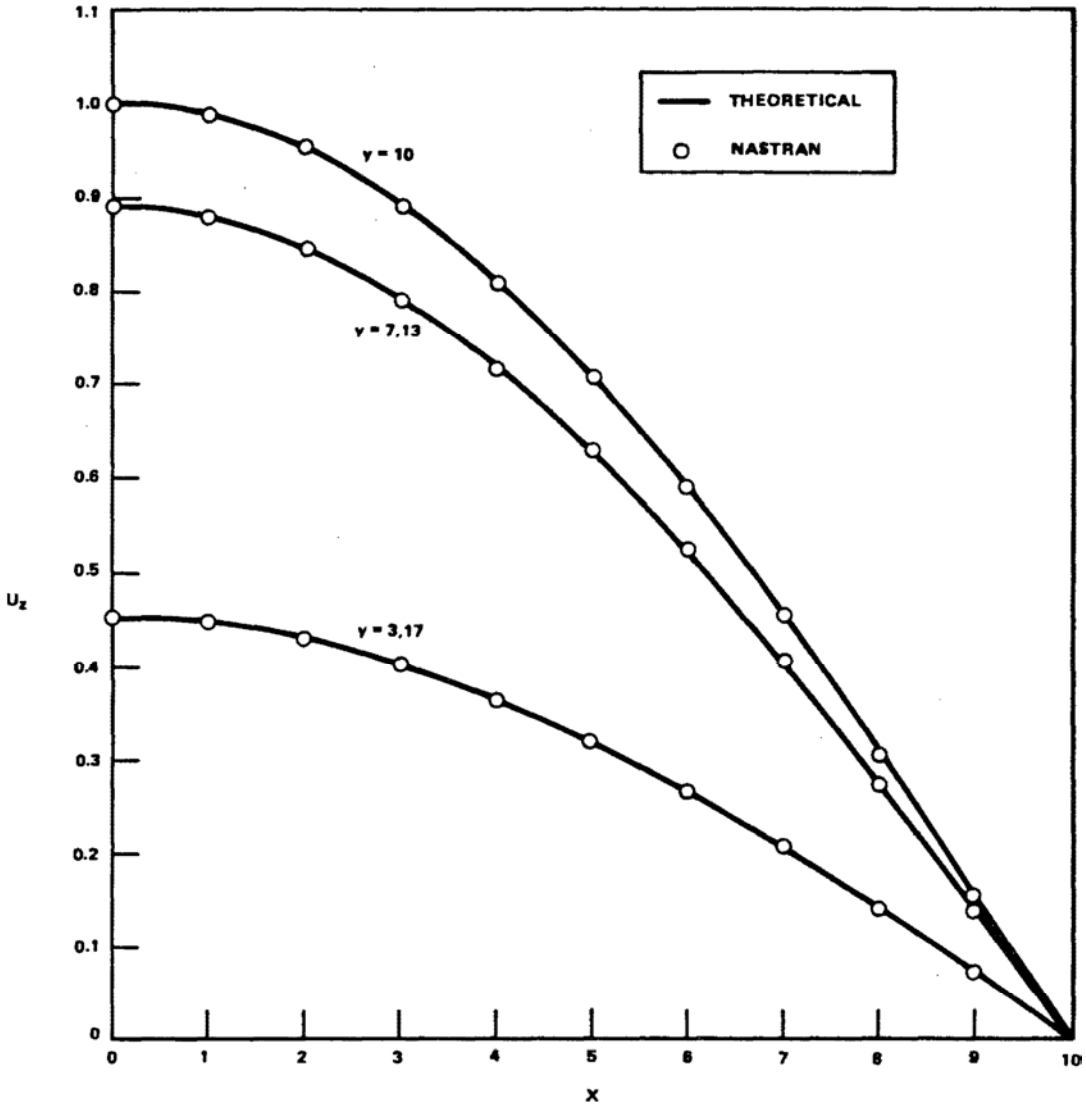


Figure 10.2 Comparison of Displacements, First Mode

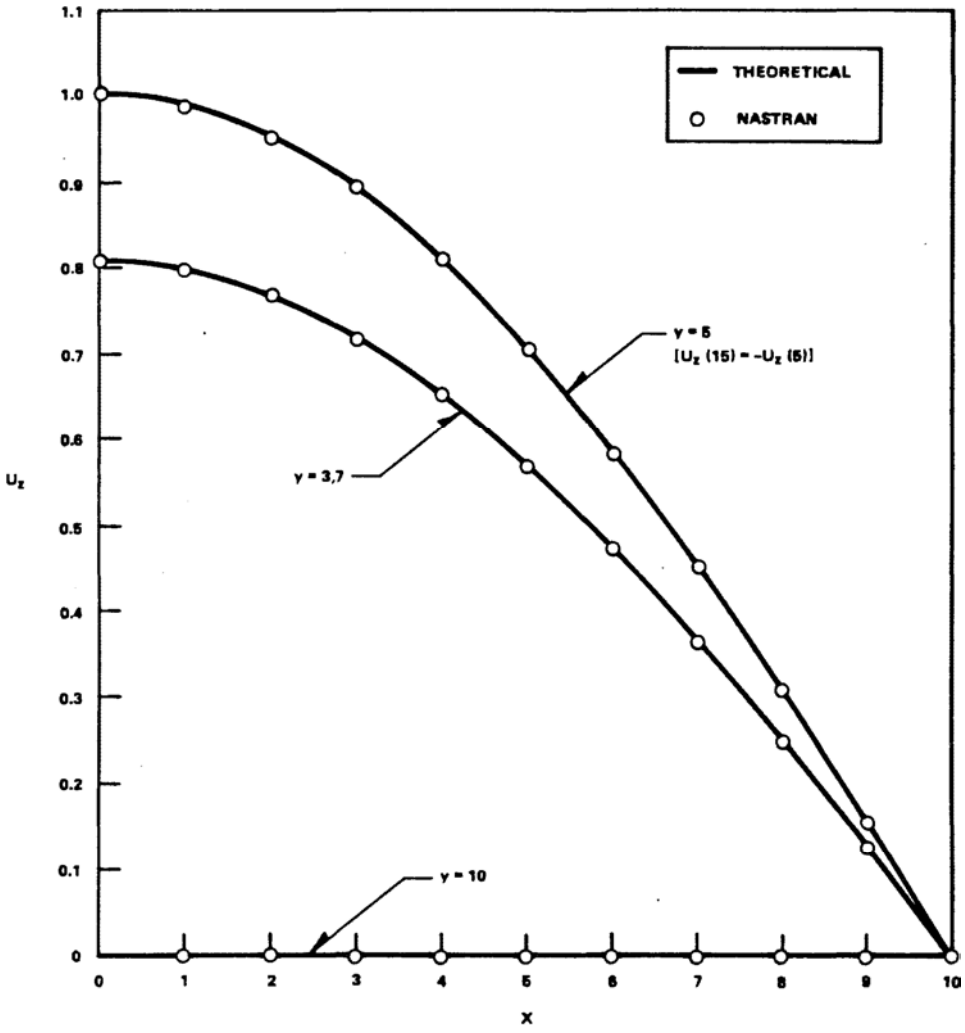


Figure 10.3 Comparison of Displacements, Second Mode

3F.11 OMP (KALNINS)

Program Description

The program calculates the displacement and stresses in thin-walled elastic shells of revolution subjected to static edge, surface, and/or temperature loads with arbitrary spatial distribution. It is possible to include up to three symmetrical branches. In addition, the shell wall may consist of four layers of different orthotropic materials, and the thickness of each layer and the elastic properties of these layers may vary along the meridian.

Basically, the program solves the H. Reissner-Neissner equations by reducing these equations to eight ordinary differential equations. Since the program is based on classical shell theory, it has the same limitations.

Program Version and Computer

[HISTORICAL INFORMATION] [The version being used in GE/NED was obtained from the original developer through the Space Division of General Electric in Valley Forge, Pennsylvania.] The current version operates on the Honeywell 6000 computer.

History of Use

[HISTORICAL INFORMATION] [The program was developed by Professor A. Kalnins of Lehigh University in 1964 (Ref. 11.1 and 11.2) and acquired by the General Electric Nuclear Energy Division in 1966. In 1967, the GE version was also adopted by the Space Division. Since that time the version has been frequently used to analyze space vehicles as well as reactor components. Meanwhile, the original program has also gained acceptance outside of GEH.

Extent of Use

In the GEH Nuclear Energy Division, OMP has been applied to reactor vessel, jet pump, and nozzle analysis. Satisfactory results were obtained.]

Test Problems

A number of test cases were run on the Honeywell computer to check program options and validity of solution. A few cases are presented on the following pages. These include:

a. A cylinder with uniform edge load (Figure 11.1)

b. A flat plate under uniform load (Figure 11.2)

c. A shallow cap under uniform pressure (Figure 11.3)

d. A cylinder with varying edge moments (Figure 11.4)

In the test cases, OMP provides excellent agreement with known solutions. This agreement, however, is not a true indication of the accuracy of the program for all problems encountered. It is probable that some problems will be inconsistent with the assumptions involved in the derivation of the governing equations.

References

11.1 A. Kalnins, "Analysis of Shells of Revolution Subjected to Symmetrical and Non-symmetrical Loads," Jour. Applied Mech., September 1964.

11.2 A. Kalnins, "On Free and Forced Vibration of Rotational Symmetric Layered Shells," Jour. Applied Mech., December 1965.

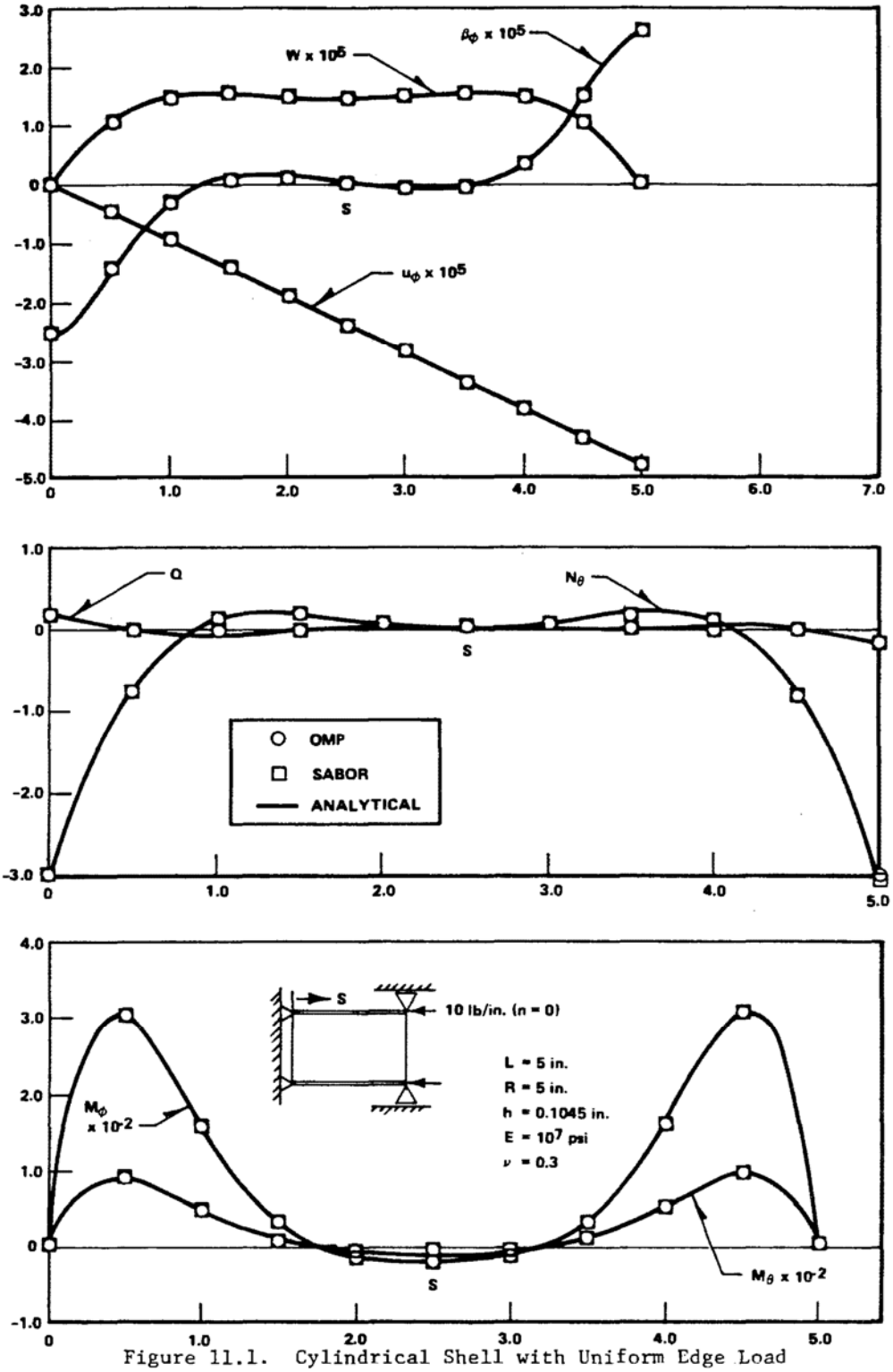


Figure 11.1. Cylindrical Shell with Uniform Edge Load

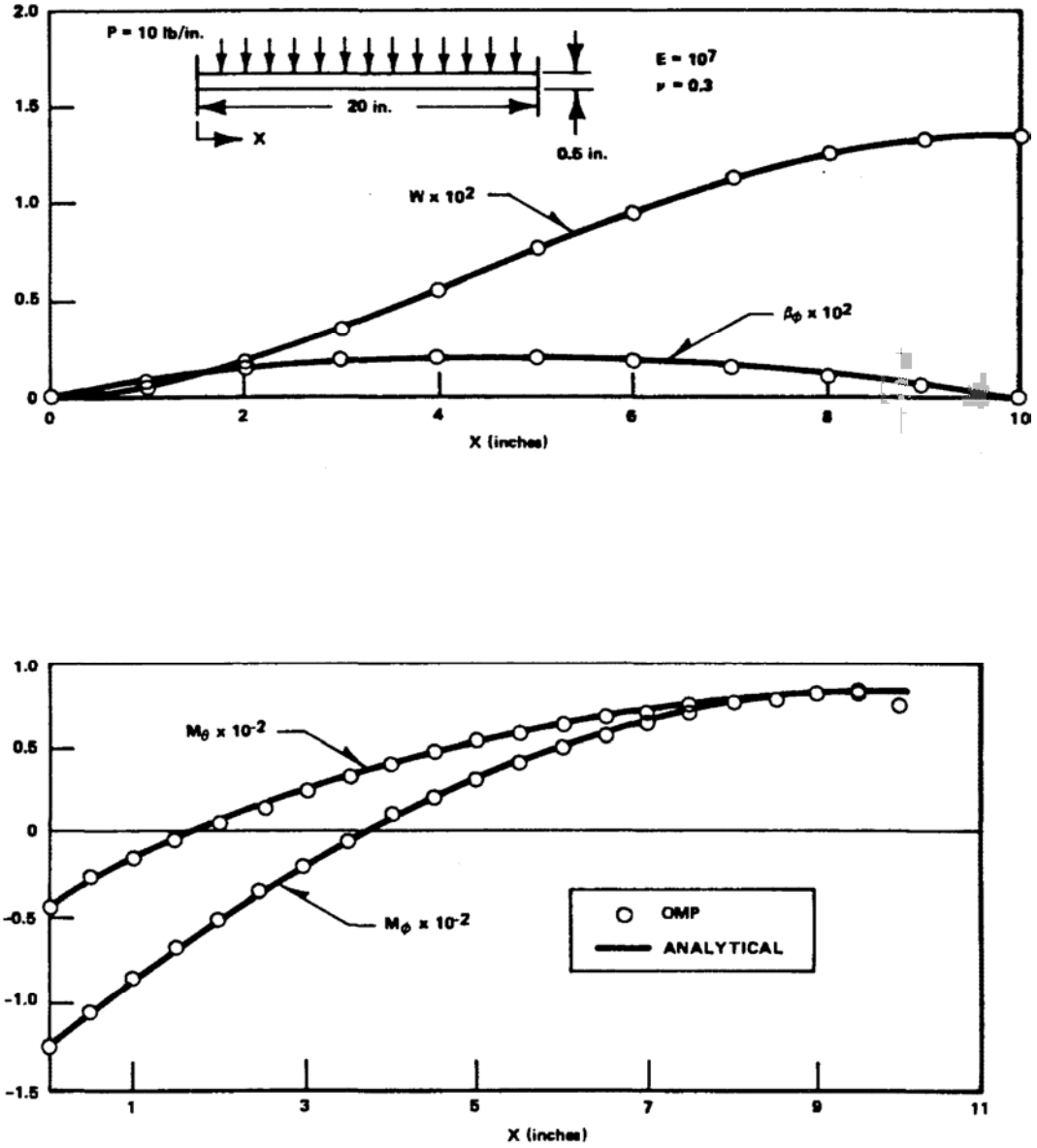


Figure 11.2. Flat Plate Under Uniform Load

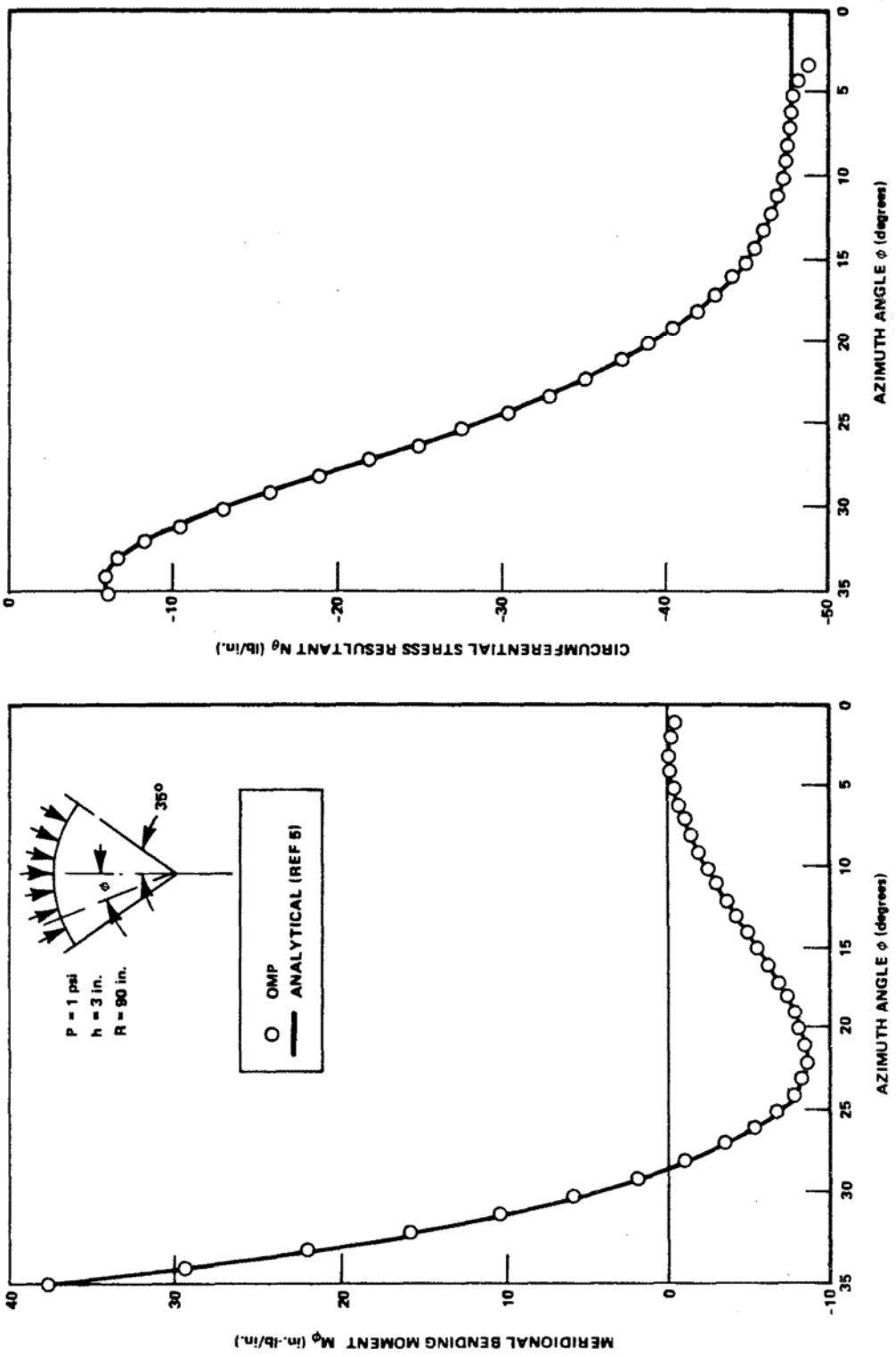


Figure 11.3. Spherical Cap Under External Pressure

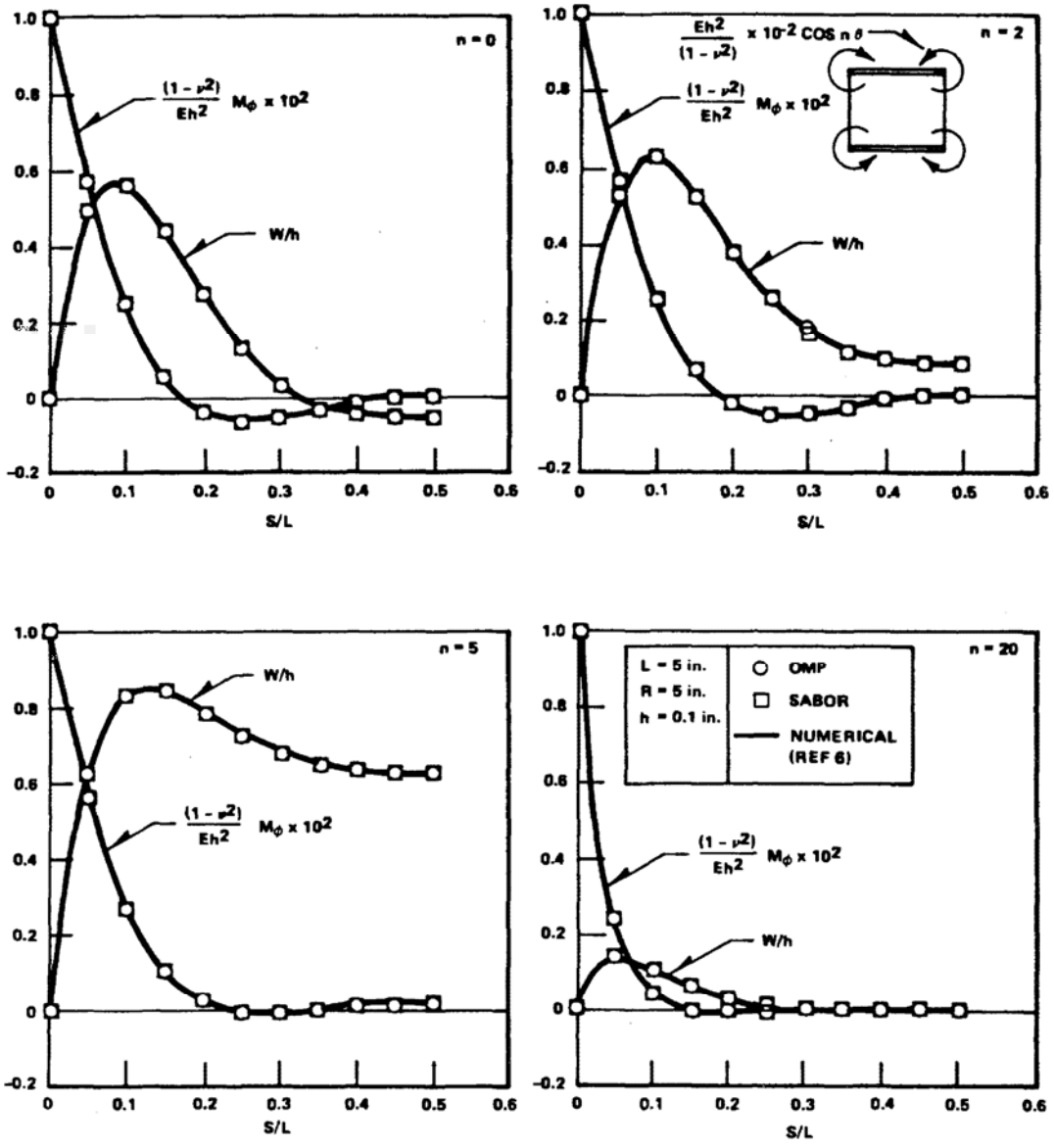


Figure 11.4. Cylinder with Various Harmonics of Edge Bending Moment

3F.12 PLATE, PANEL, AND SPACE STRUCTURAL ANALYSIS PROGRAM

Program Description

The program, proprietary of the General Electric Co., is an outgrowth of the PAPA (Plate and Panel Analysis) program originally developed by L. Beitch in the early 1960s. The program is based on the principle of the finite element method. Governing matrix equations are formed in terms of joint displacements using a "stiffness-influence-coefficient" concept originally proposed by L. Beitch (Ref. 12.1). The program offers curved beam, plate, and shell elements. It can handle mechanical and thermal loads in a static analysis and predict natural frequencies and mode shapes in a dynamic analysis.

Program Version and Computer

The current version maintained by the original developer, L. Beitch, of the General Electric Aircraft Engine Division in Evandale, Ohio is used. The program operates on the Honeywell 6000 computer.

History of Use

[HISTORICAL INFORMATION] [Since its development in the early 1960s, the program has been successfully applied to a wide variety of jet-engine structural problems, many of which involve extremely complex geometries. The use of the program in the Nuclear Energy Division also started shortly after its development.

Extent of Application

Besides the Jet Engine and Nuclear Energy Divisions, the Missile and Space Division, the Appliance Division, and the Turbine Division of General Electric have also applied the program to a wide range of engineering problems. The Nuclear Energy Division used it mainly for piping and jet pump analyses. Recently, most of its work load has been shifted to the computer code SAP (Section 13).]

Test Problems

The examples to be given here are relatively simple and do not really reflect the complete generality of the method. PAPA has been used, for example, on hollow jet-engine buckets, fabricated combustion liners and other complicated panel structures including odd shapes and complex curvatures. It is now being applied to large pipe fittings, such as tees, wyes, elbows, etc. Discussion of applications such as this requires a report in itself to explain modeling techniques, interpretations of results, etc. The samples presented here are meant only to lend credibility to the approach.

Vibration

The critical frequencies for a square plate with all edges simply supported was checked with the agreement for the first two modes shown in Figure 12.1. A fairly rough grid was used here. Each element was a 1 x 1 inch square. Correlation of higher modes was just as good. Mode shapes shown in the figure were also obtained from the computer run. Theoretical frequencies were computed from methods given in Reference 12.2.

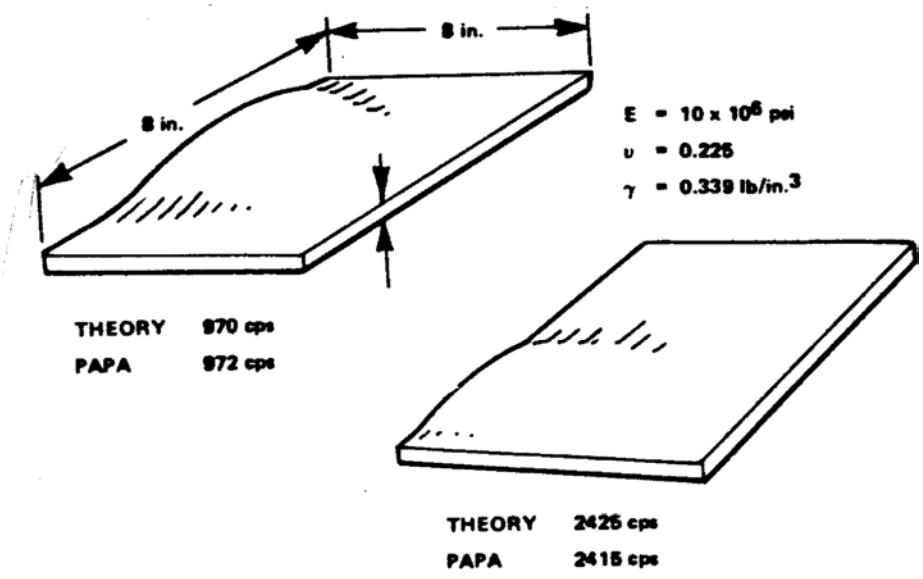


Figure 12.1. Flat Plate Vibration

Curved Plates

Two cylindrical cases are used here to demonstrate the correctness of the curved element. The first of these is a cylinder under constant internal pressure of 100 psi and with simply supported edges. Theoretical calculations were made using the equations found in Reference 12.3. Young's modulus was taken as 10×10^6 psi and Poisson's ratio as 0.30. The radius of the cylinder was 5 inches and its length 16 inches. To get the effects of a short cylinder, a thickness of 1.321 inches was used. The grid for the portion of the cylinder analyzed is shown in Figure 12.2. Again excellent agreement was obtained as shown in Figure 12.2.

To get the interaction of the various loads, the same cylinder was analyzed under a pressure distributed sinusoidally in the circumferential direction and changing linearly in the axial direction, i.e.,

$$p = 100 \left|\frac{z}{\theta}\right| \sin\theta$$

To get a correlation here, another numerical method for the solution of shells of revolution was used. This method solves the differential equations numerically. To get an idea of its correctness, it is also compared with the first case above and is shown in Figure 12.2. To help orient the exactness of the two numerical methods (Shell and PAPA), the Shell program uses about twice the number of points as PAPA for the finite parts used to represent the continuum. Note that its agreement with theory is not as good as PAPA. The comparison of this Shell program with PAPA for the sinusoidal pressure case is shown in Figure 12.3. Although the results here are not conclusive for all the parameters, due to the comparison of two approximate methods, indications of the proper trend are definitely present.

References

12.1 Beitch, L., “Shell Structures Solved Numerically by Using a Network of Partial Panels”, AIAA J., Vol. 5, No. 3, March, 1967.

12.2 Timoshenko, S., “Vibration Problems in Engineering”, D. Van Nostrand Co., 3rd Edition, January, 1955.

12.3 Timoshenko, S., "Theory of Plates and Shells", McGraw-Hill Book Company, 1940.

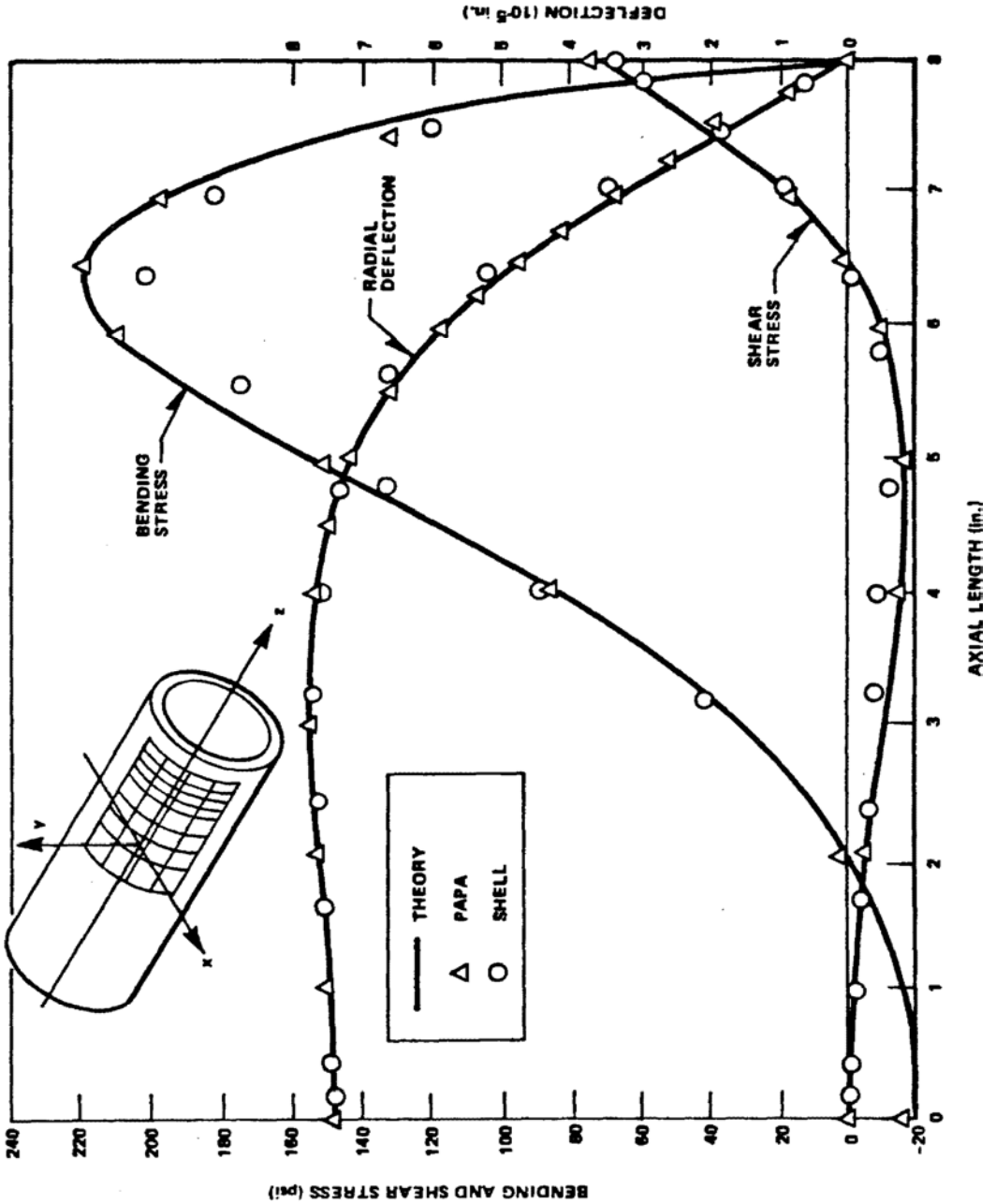


Figure 12.2. Pressure on a Cylinder

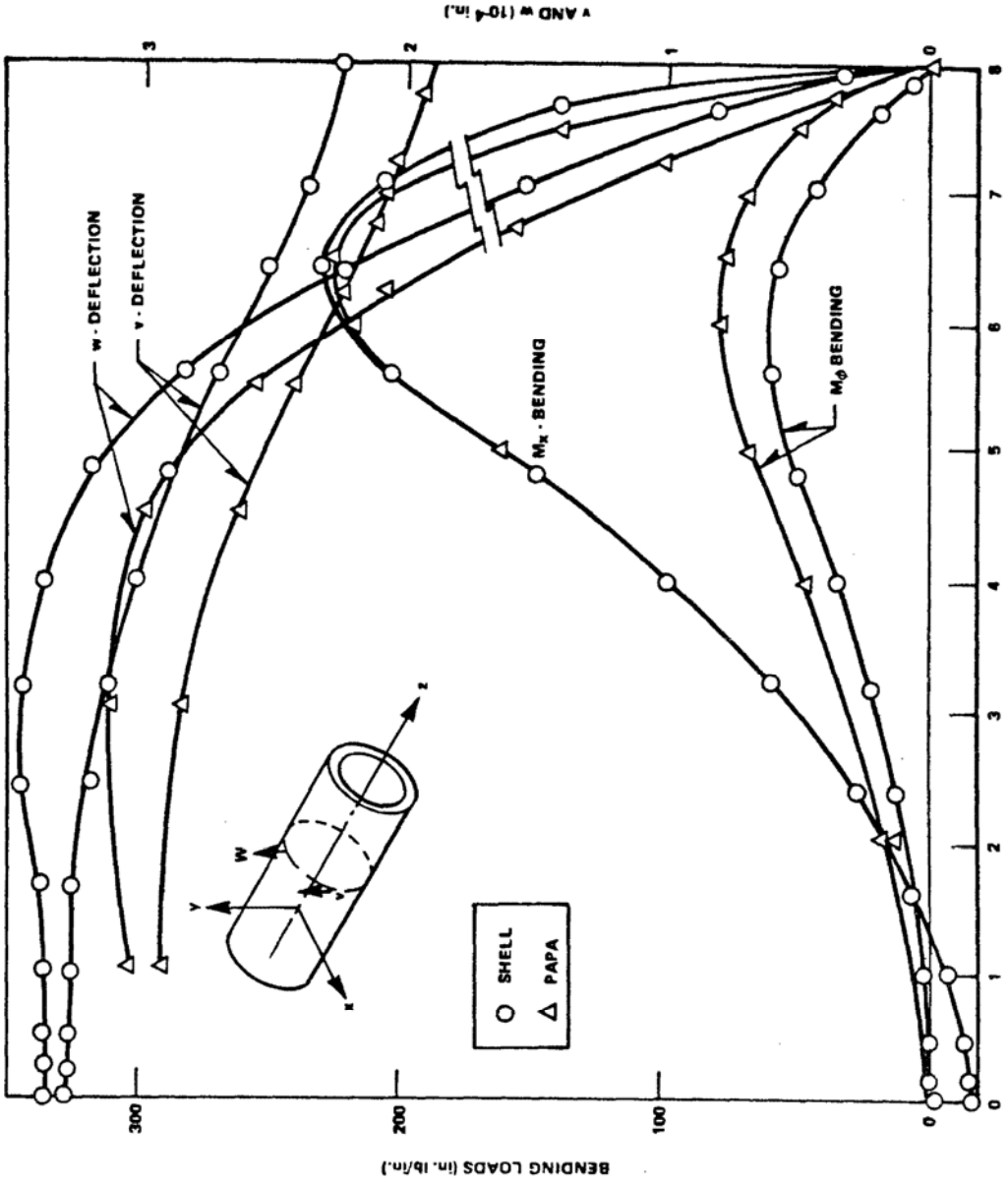


Figure 12.3. Variable Pressure on a Cylinder

3F.13 SAP

Program Description

SAP is a general Structural Analysis Program for static and dynamic analysis of linear elastic complex structures. The finite element displacement method is used to solve for the displacements and compute the stresses of each element of the structure. The structure can be composed of an unlimited number of three-dimensional truss, beam, plate, shell, solid, plane strain-plane stress, brick, thick shell, spring, and axisymmetric elements. The program can treat thermal and various forms of mechanical loading as well as internal element loadings. The dynamic analysis includes mode superposition, time-history, and response spectrum analyses. Earthquake type of loading as well as time varying pressure can be treated. The program is very versatile and efficient in solving large and complex structural systems. The output contains displacements of each nodal point as well as stresses at the surface of each element.

Program Version and Computer

The program being used in GE/NED is equivalent to the version currently released by the NISEE/Computer Applications of the University of California at Berkeley (Ref. 13.1). The GE version however has been converted to double precision from the CDC 600 to the Honeywell 6000 computer.

History of Use

[HISTORICAL INFORMATION] [The development of SAP dates back to 1968, when the first static version was developed and used by Professor E. L. Wilson of the University of California, Berkeley. Since then, there have been many improvements and additions to the program such as the dynamic analysis capability and piping elements.

Extent of Application

Since its development, the program has been used by many companies in this country. It is considered to be one of the most efficient general-finite-element programs. Beginning in 1972, the General Electric Nuclear Energy Division has been applying it to piping

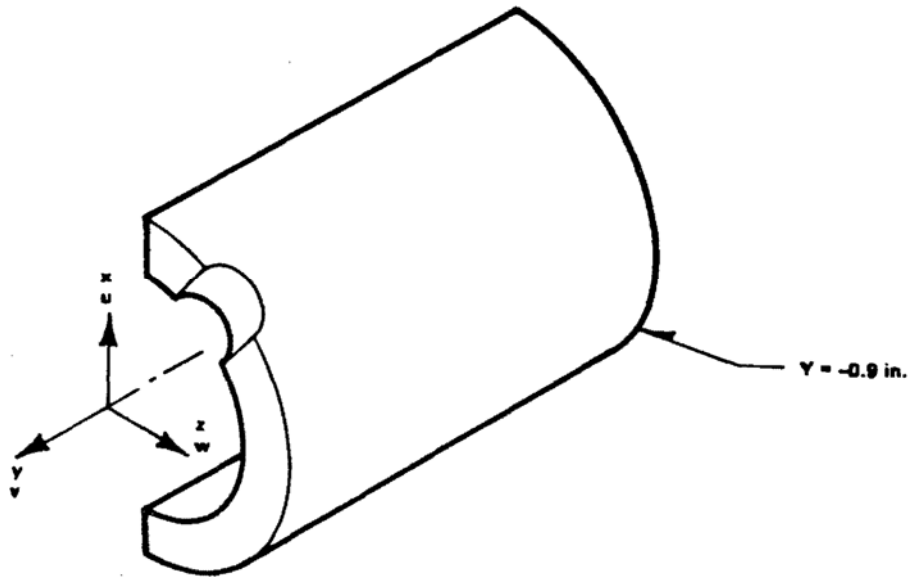
and thermal-stress analyses. Other divisions of General Electric have also applied SAP to the analyses of turbines, space vehicles, and composite materials.]

Test Problems

Two test problems are shown herein. The results of the first problem are contained in Figures 13.1 and 13.2. Those of the second problem are presented in Figures 13.3 and 13.4 and in Table 13.1. Additional test results are shown in Section 5 of this document.

Reference

13.1 J. Bathe, E. L. Wilson, and F. E. Peterson, "SAP IV - A Structural Analysis Program for Static and Dynamic Response of Linear Systems," Earthquake Engineering Research Center Report No. EERC 73-11, University of California, Berkeley, June, 1973.



OUTER RADIUS = R = 0.4375 in.
INNER RADIUS = 0.2656 in.
HOLE RADIUS = 0.125 in.
BOUNDARY CONDITIONS

Y = 0	U = 0 W = 0
Z = 0	U = 0 V = 0
Y = -0.9	U = 0.0042 (Z/R) W = 0.0042 (X/R)

Figure 13.1. Tube Test Case

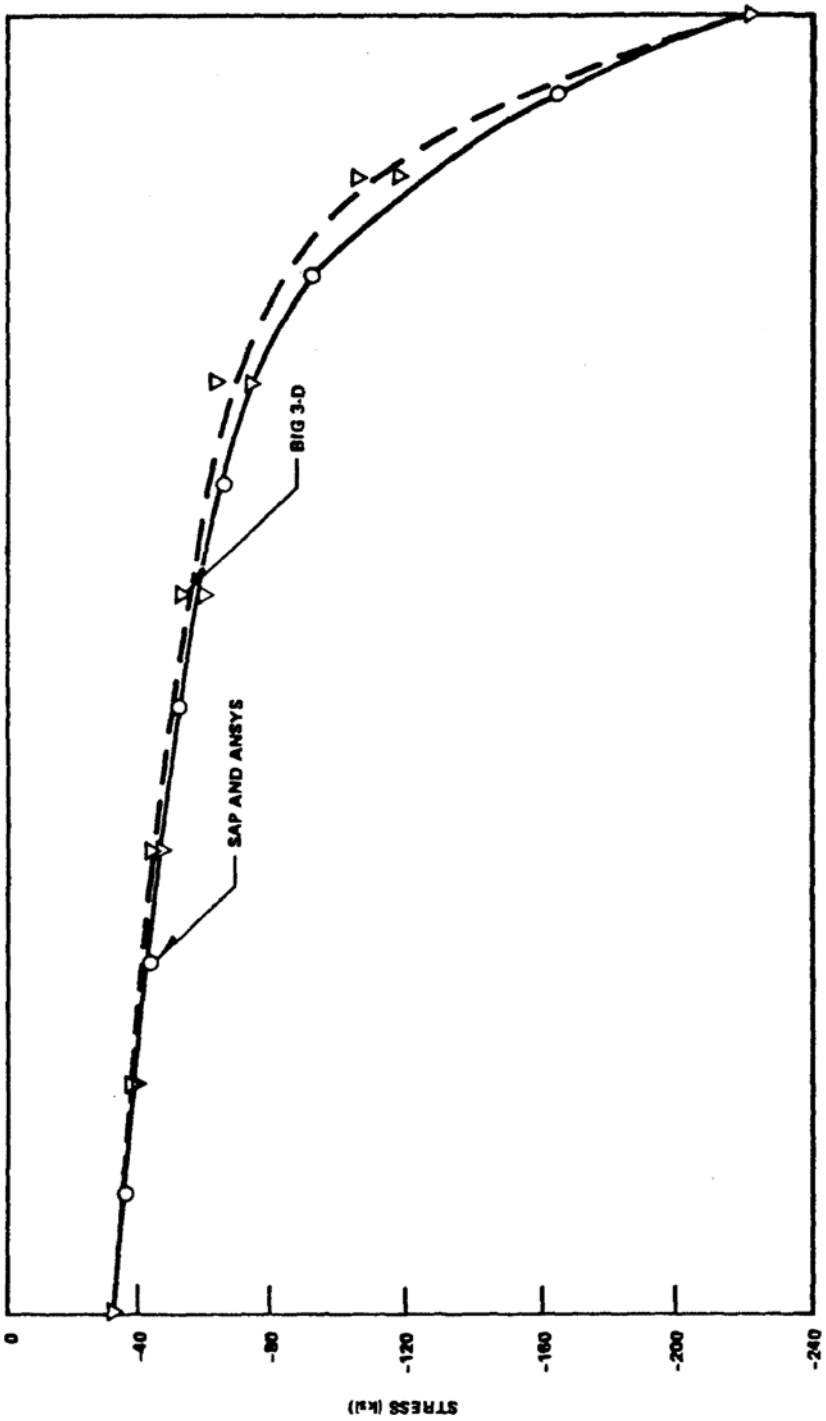


Figure 13.2. Extrapolation of the Minimum Principal Stress to the Edge of the Hole

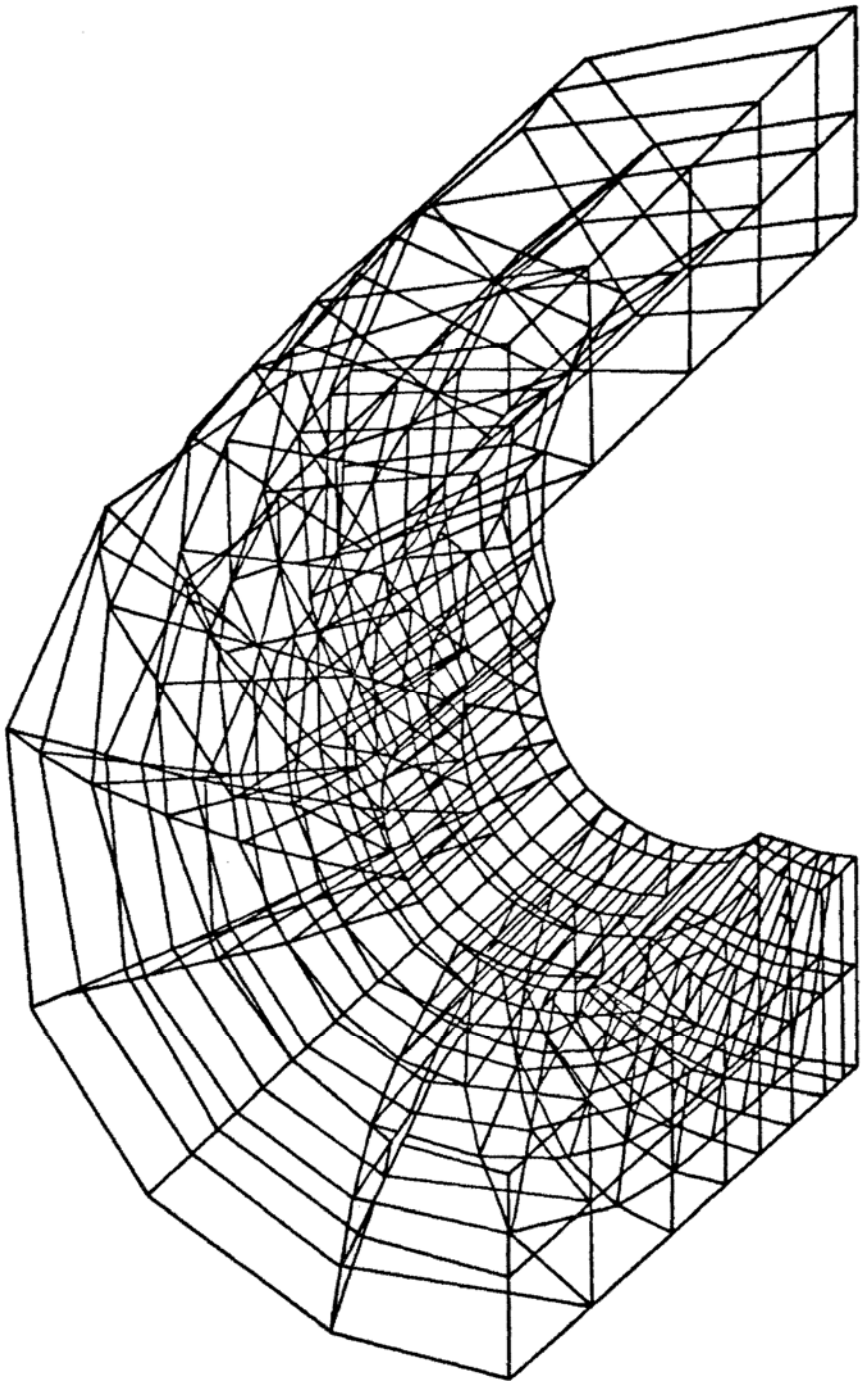


Figure 13.3. Finite Element Model

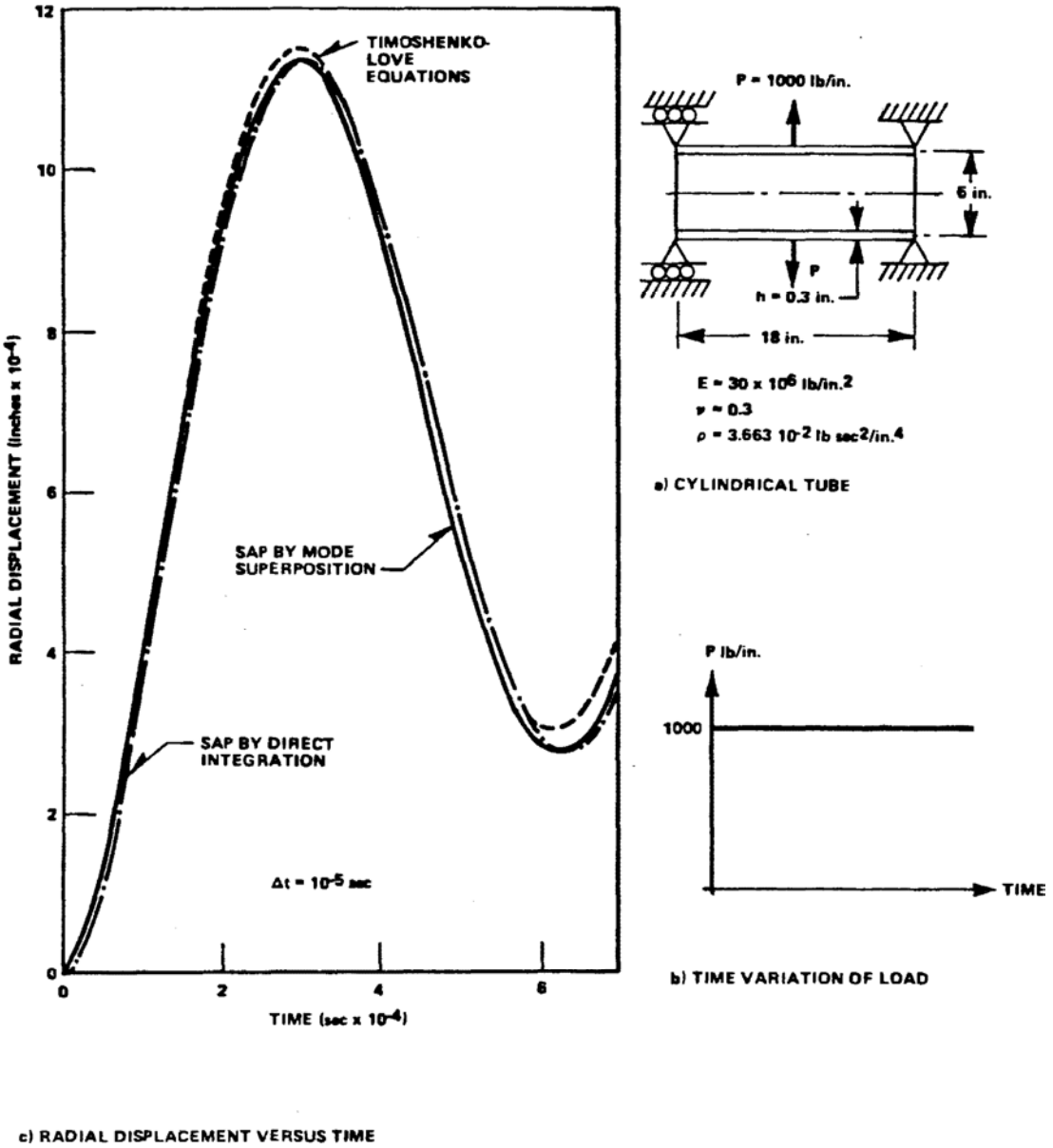


Figure 13.4. Response History Analysis of Cylindrical Tube

TABLE 13.1: Comparison of the Calculated Stress Components for the Most Highly Stressed Element (No. 43)

Stress Component (ksi)					
Solid Sap		-28.3	-108.0	-50.8	-9.2	-164.0
ANSYS		-28.3	-108.2	-51.0	-9.3	-164.0
MARC		-29.1	-104.3	-50.5	N.C.	N.C.
Big-3D	Bottom	-27.1	-108.6	-53.3	-9.4	-165.6
	Top	-27.2	-107.1	-46.9	-7.2	-162.9
	Back	-13.0	-66.2	-36.0	2.3	-106.3
	Front	-41.3	-149.5	-64.2	-13.6	-222.9
	Left	-28.7	-107.4	-51.0	-9.4	-164.2
	Right	-25.5	-208.3	-49.2	-7.2	-164.3
	Avg	-27.1	-107.9	-50.1	-7.4	-164.4

3F.14 SHELL 5 and SHELL 9

Program Description

Shell 5 and Shell 9 are two finite shell element programs used to analyze smoothly curved thin shell structures with any distribution of elastic material properties, boundary constraints, and mechanical thermal and displacement loading conditions. The basic element is triangular whose membrane displacement fields are linear polynomial functions and whose bending displacement field is a cubic polynomial function (Ref. 14.1). Five degrees of freedom (3 displacements and 2 bending rotations) are obtained at each nodal point. Shell 9 is an improvement of Shell 5. It includes a more accurate shell element with 9 degrees of freedom at each node. Output displacements and stresses are in a local (tangent) surface coordinate system.

Due to the approximation of element membrane displacements by linear functions, the in-plane rotation about the surface normal is neglected. Therefore, the only rotations considered are due to bending of the shell cross section and application of the method is not recommended for shell intersection (or discontinuous surface) problems where in-plane rotation, can be significant.

Program Version and Computer

[HISTORICAL INFORMATION] [A copy of the source deck of Shell 5 is maintained in GE/NED by Y. R. Rashid, one of the originators of the program. Shell 9 is a proprietary computer program of the Gulf Atomic Incorporated. Both operate on the UNIVAC 1108 computer.

History of Use

Shell 5 and Shell 9 are programs developed by the Gulf General Atomic Incorporated (Ref. 14.2) in 1969. The program has been in production status at Gulf General Atomic, General Electric, and at other major computer operating systems since 1970.

Extent of Application

Shell 5 has been used at General Electric to analyze the reactor shroud support and torus. Satisfactory results were obtained.]

Test Problems

Two examples showing comparisons of solutions obtained by the present method with a solution based on a simpler model, an exact solution, and experimental data are presented in Figures 14.1 through 14.5. Figure 14.2 shows the radial displacement for the line-loaded cylinder shown in Figure 14.1 compared with the exact solution (Ref. 14.3) and a solution obtained for the element model of Reference 14.1 with linear membrane and cubic bending displacement approximations. Symmetry permitted the analysis of one slice, and in order to show the dependence of the solution on the mesh density, the circumferential angle of the slice was varied which is equivalent to varying the number of elements. It should be pointed out that this comparison does not constitute a complete study and other examples may show different convergence behavior. The differences between the two models shown by this example warrant further study.

Figure 14.3 shows the finite-element idealization of a nozzle-to-cylinder intersection problem that is under experimental investigation at the Oak Ridge National Laboratory. Because of symmetry about the longitudinal plane, only half of the structure is analyzed for internal pressure. The idealization consisted of 640 modal points and 1,168 elements. Comparisons of the analysis with the experimental data in the vicinity of the right-angle intersection for the longitudinal plane are shown in Figures 14.4 and 14.5. Figure 14.4 shows the axial and circumferential outside surface stresses for the nozzle. Figure 14.5 shows the axial and circumferential outside surface stresses for the cylinder. The analytical results compare favorably with the distribution and the magnitude of experimental stresses (obtained from strain gauge results).

References

14.1 W. Clough and C. P. Johnson, “A Finite Element Approximation for the Analysis of Thin Shells,” Intl. J. Solid Structures, Vol. 4, 1968.

14.2 Computer Program for the Structural Analysis of Arbitrary Three-Dimensional Thin Shells, Gulf General Atomic Inc., Report No. GA-9952.

14.3 Timoshenko and S. Woinowsky-Krieger, “Theory of Plates and Shells,” McGraw-Hill Book Company, New York, 1959.

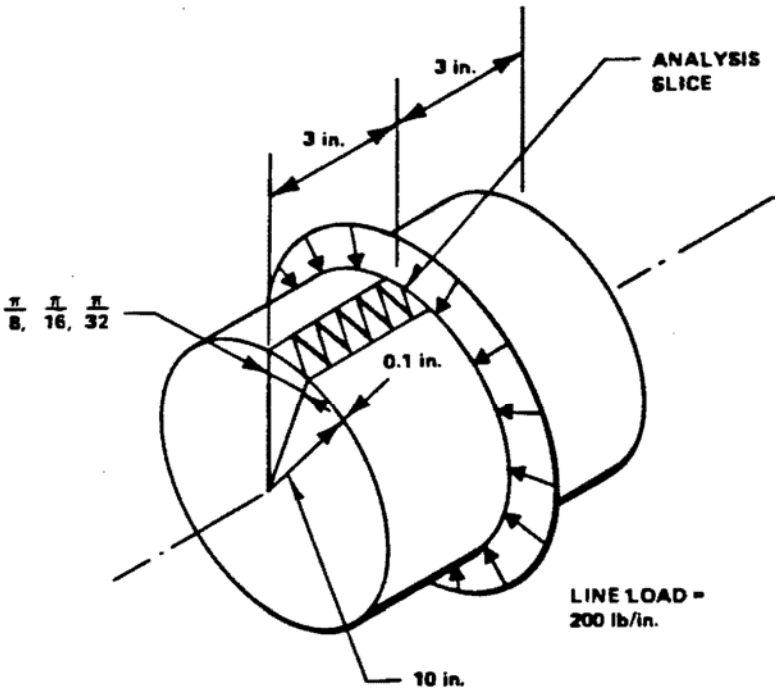


Figure 14.1. Line Loaded Cylinder

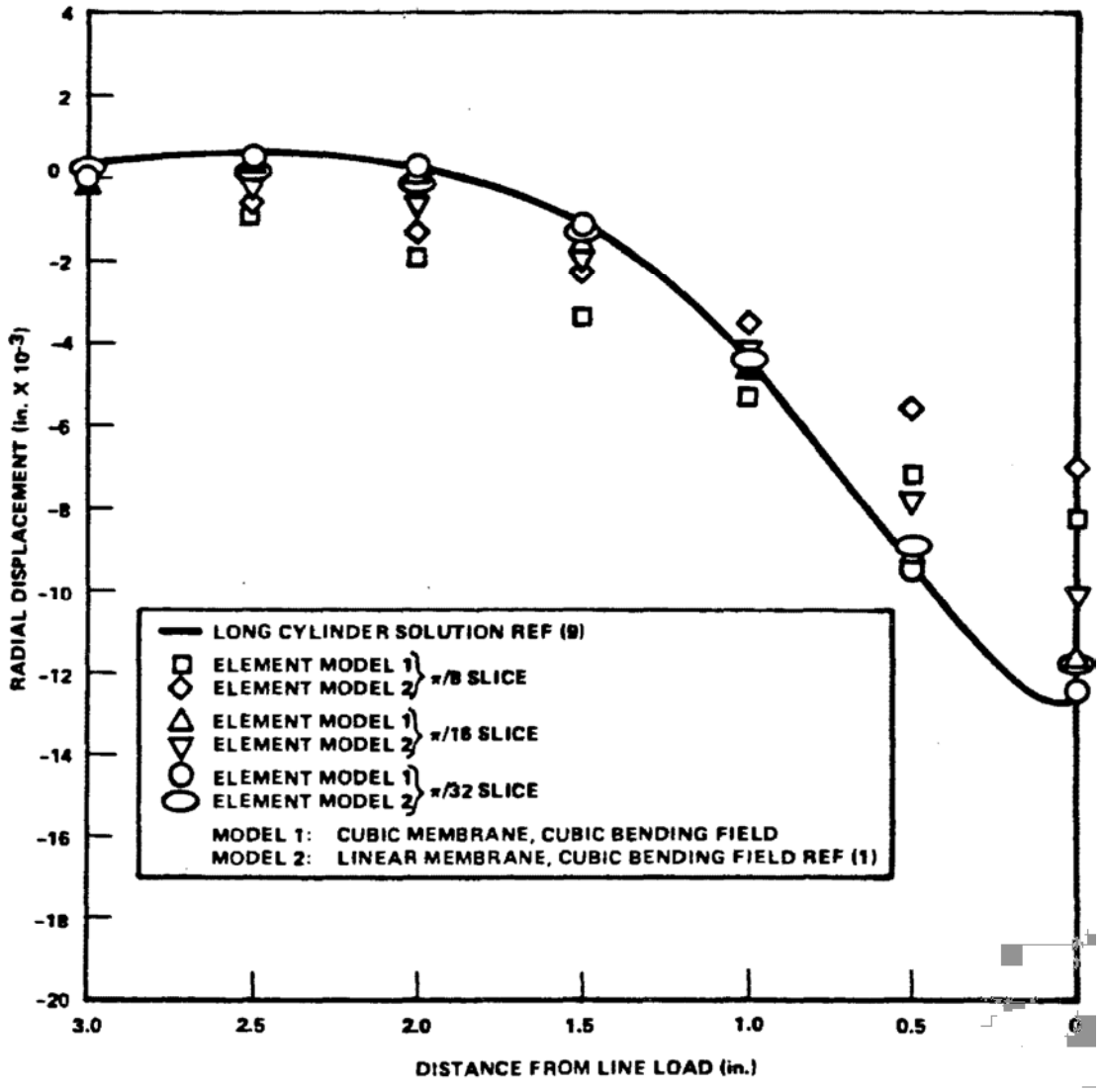


Figure 14.2. Radial Displacement for Line-Loaded Cylinder

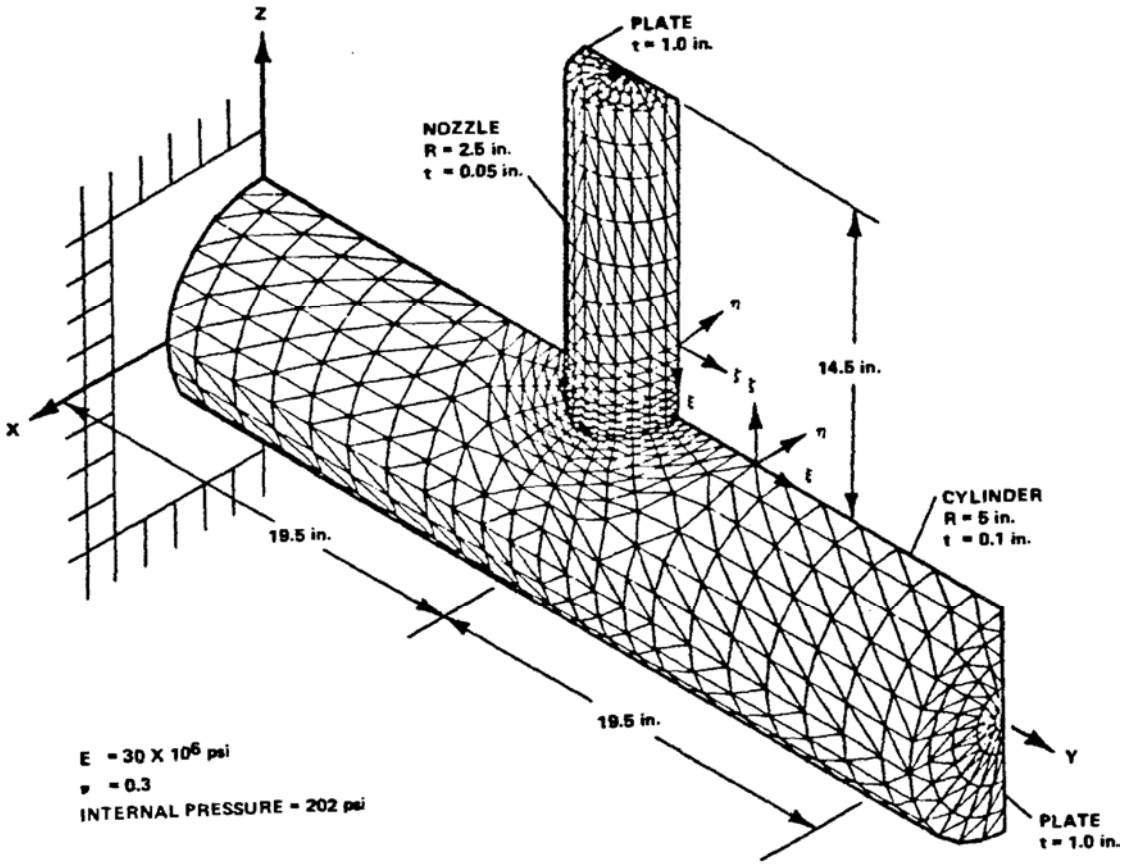


Figure 14.3. Nozzle-to-Cylinder Intersection

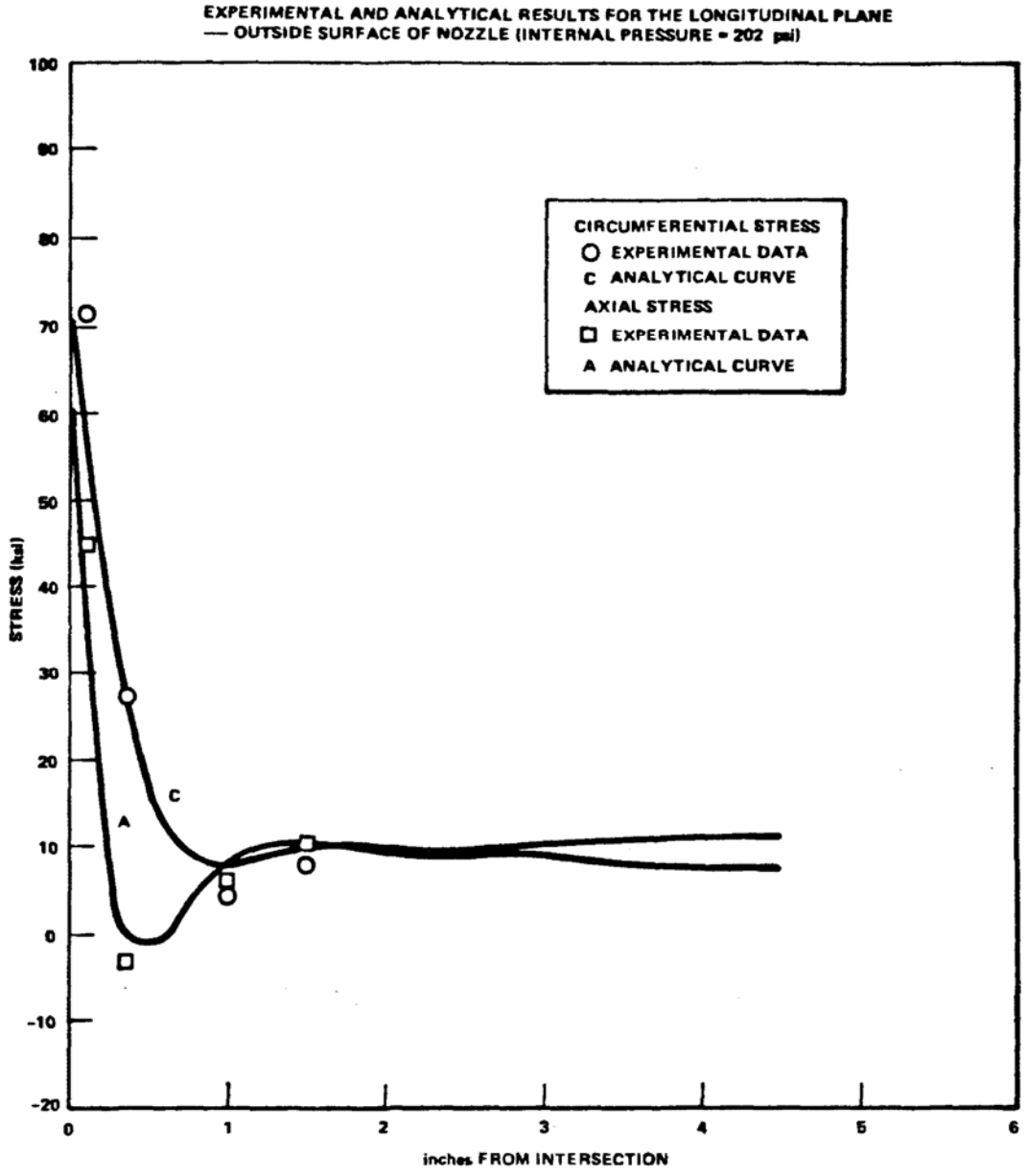


Figure 14.4. Nozzle Stresses

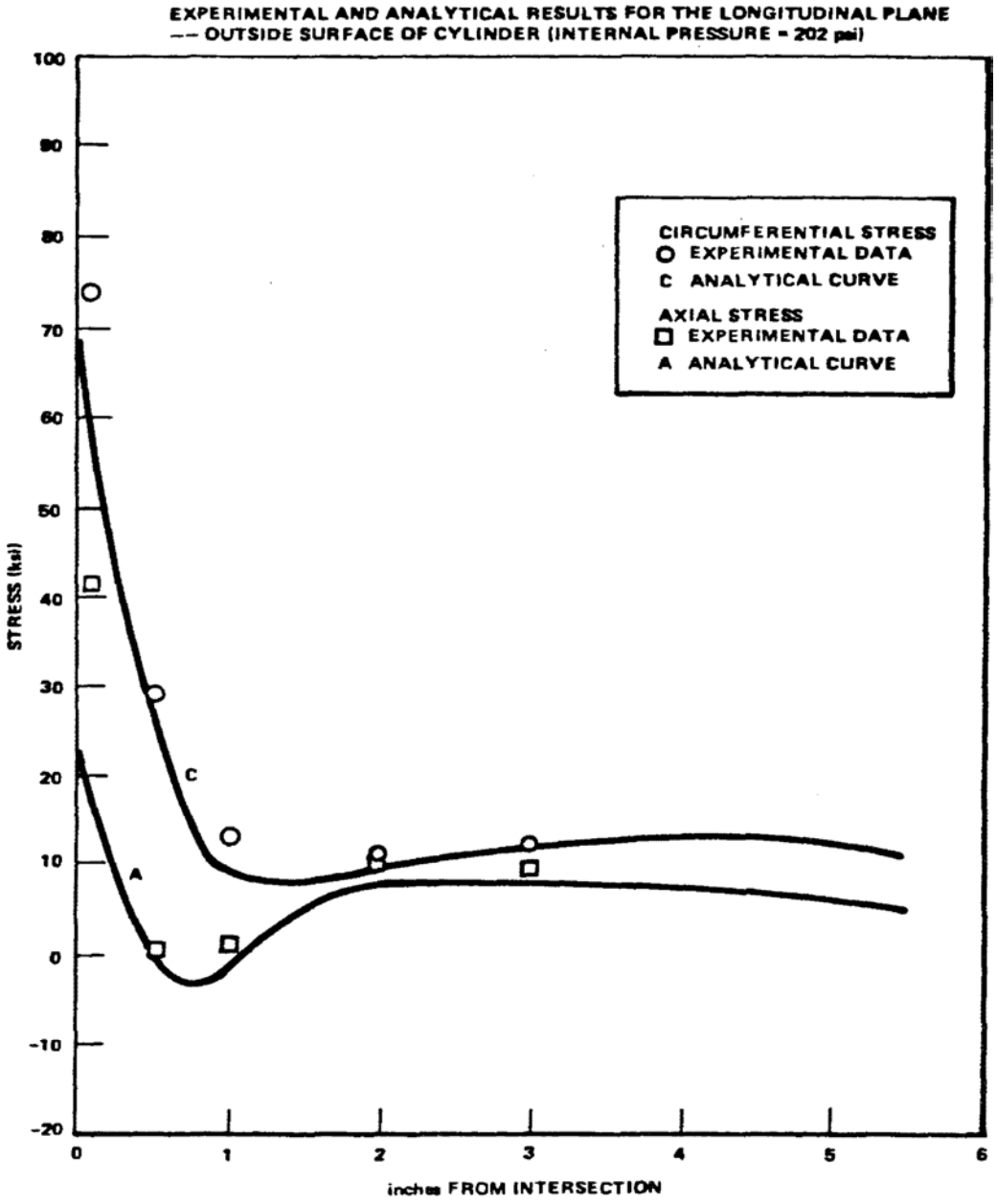


Figure 14.5. Cylinder Stresses

3F.15 SHELL ANALYSIS PROGRAM

Program Description

Shell Analysis Program is a General Electric code which determines the loads, deformations, and stresses of axisymmetric shells of revolution (cylinders, cones, discs, toroids, and rings) for axisymmetric thermal boundary and surface load conditions. Thin shell theory is inherent in the solution of E. Peissner's differential equations for each shell's influence coefficients. Surface loading capability includes pressure, average temperature, and linear through-wall gradients; the latter two may be linearly varied over the shell meridian. The theoretical limitations of this program are the same as those of classical theory.

Program Version and Computer

The current version maintained by the General Electric Jet Engine Division at Evandale, Ohio is being used on the Honeywell 6000 computer in GE/NED.

History of Use

[HISTORICAL INFORMATION] [The initial version of the Shell Analysis Program was completed by the Jet Engine Division in 1961. Since then, a considerable amount of modification and addition has been made to accommodate its broadening area of application. Its application in the Nuclear Energy Division has a history longer than 10 years.

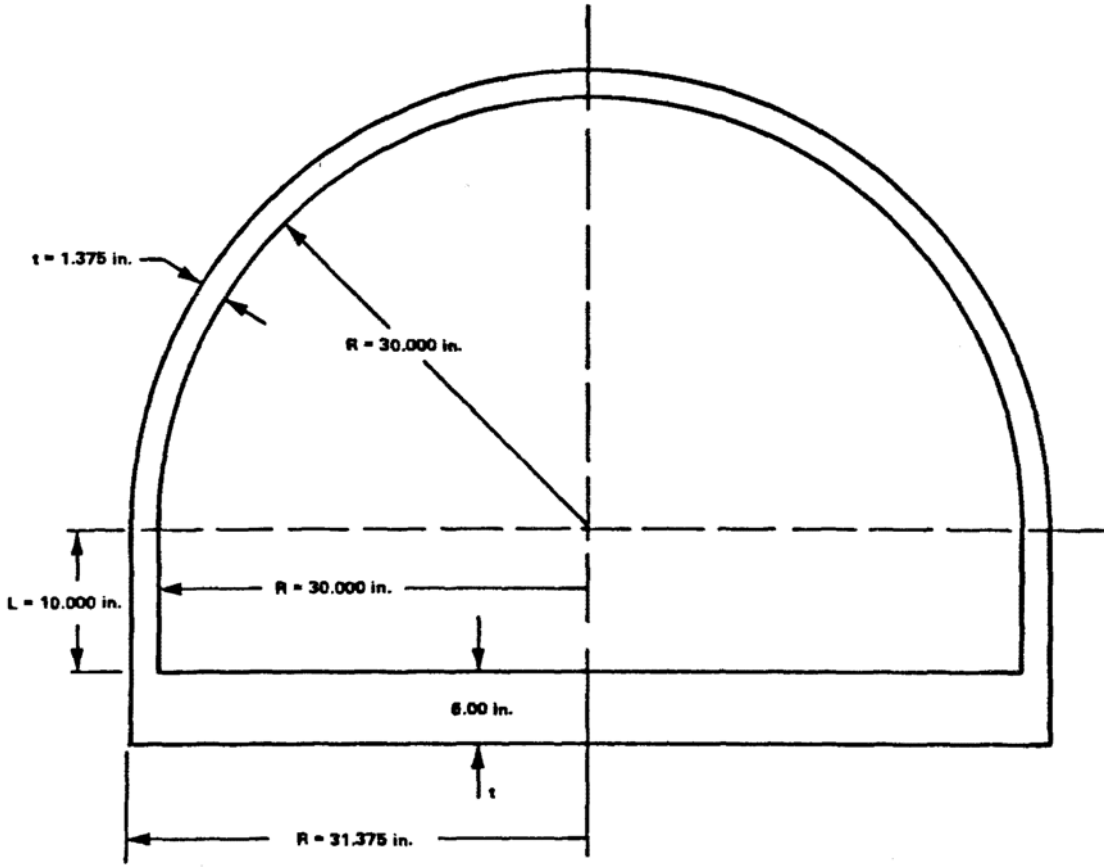
Extent of Application

The program has been used to analyze jet engine, space vehicle, and nuclear reactor components. Because of its efficiency and economy, in addition to reliability, it has been one of the main shell analysis programs in the Nuclear Energy Division of GEH.]

Test Problems

The program (code name MULTISHELL) has been used to analyze the pressure vessel specified in Article 1-7 of ASME Section III. The program results are compared with those from other Shell programs under three loadings: internal pressure, axial temperature gradient, and linear radial temperature gradient. It was found

that the thin shell theory programs (OMP, SOR, and MULTISHELL) were within 4% of thin-shell theoretical results for the pressure loading and within 1% of each other for the other two loadings. The thick shell theory program (SEAL-SHELL-2) was within 7% of theory for the pressure loading, and within 9% of the thin shell codes on the other two loadings. Detail results are presented in Figures 15.1 - 15.9.



MATERIAL	SAE 302-B CARBON STEEL
YOUNG'S MODULUS	29×10^6 psi
POISSON'S RATIO	0.3
COEFFICIENT OF THERMAL EXPANSION	6.6×10^{-3} (°F)$^{-1}$

Figure 15.1 Pressure Vessel Analyzed

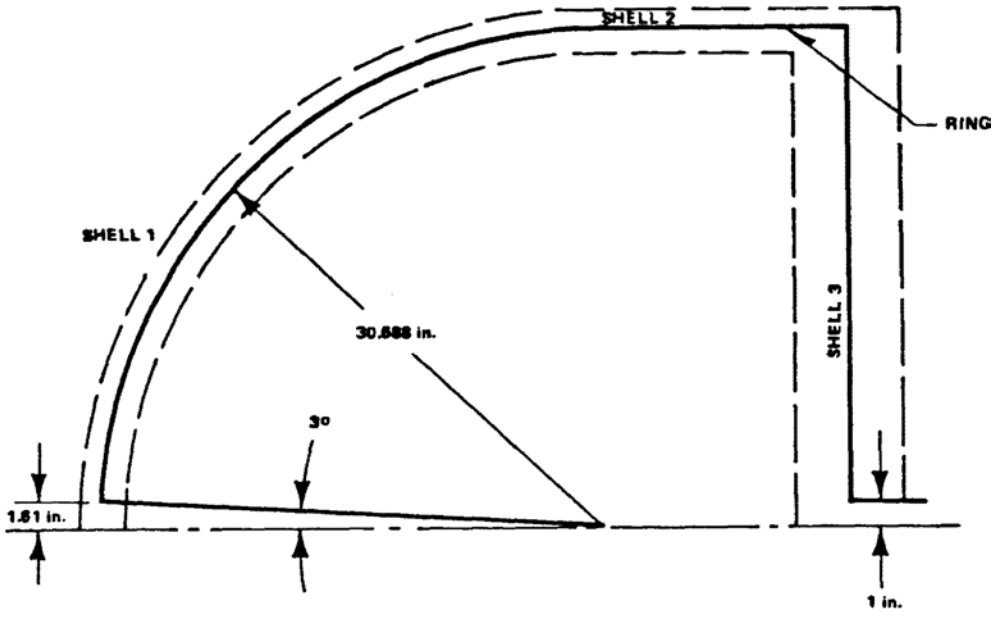


Figure 15.2 MULTISHELL Model

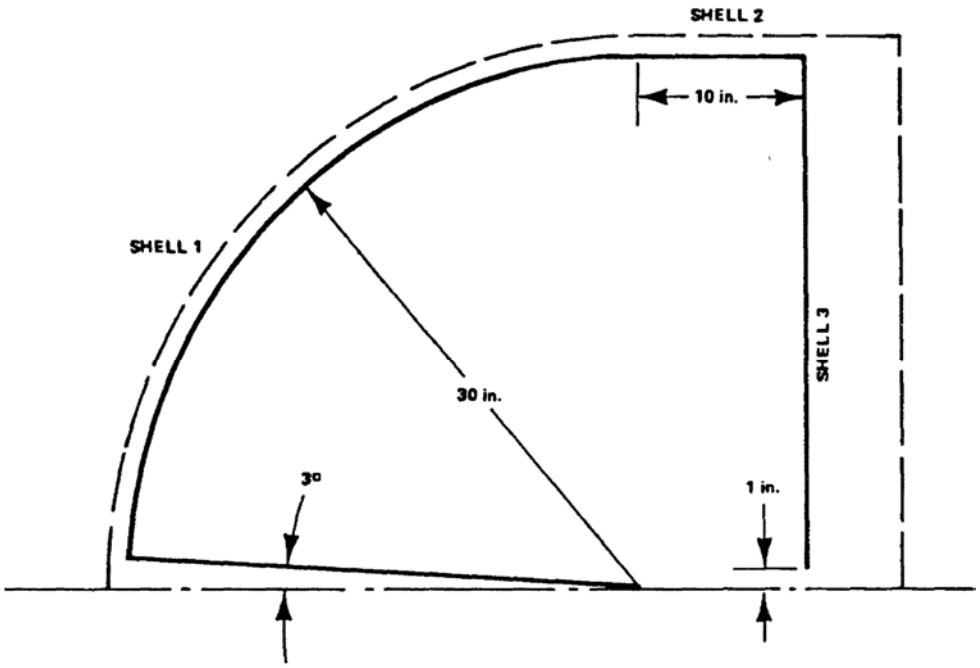


Figure 15.3 OMP Model

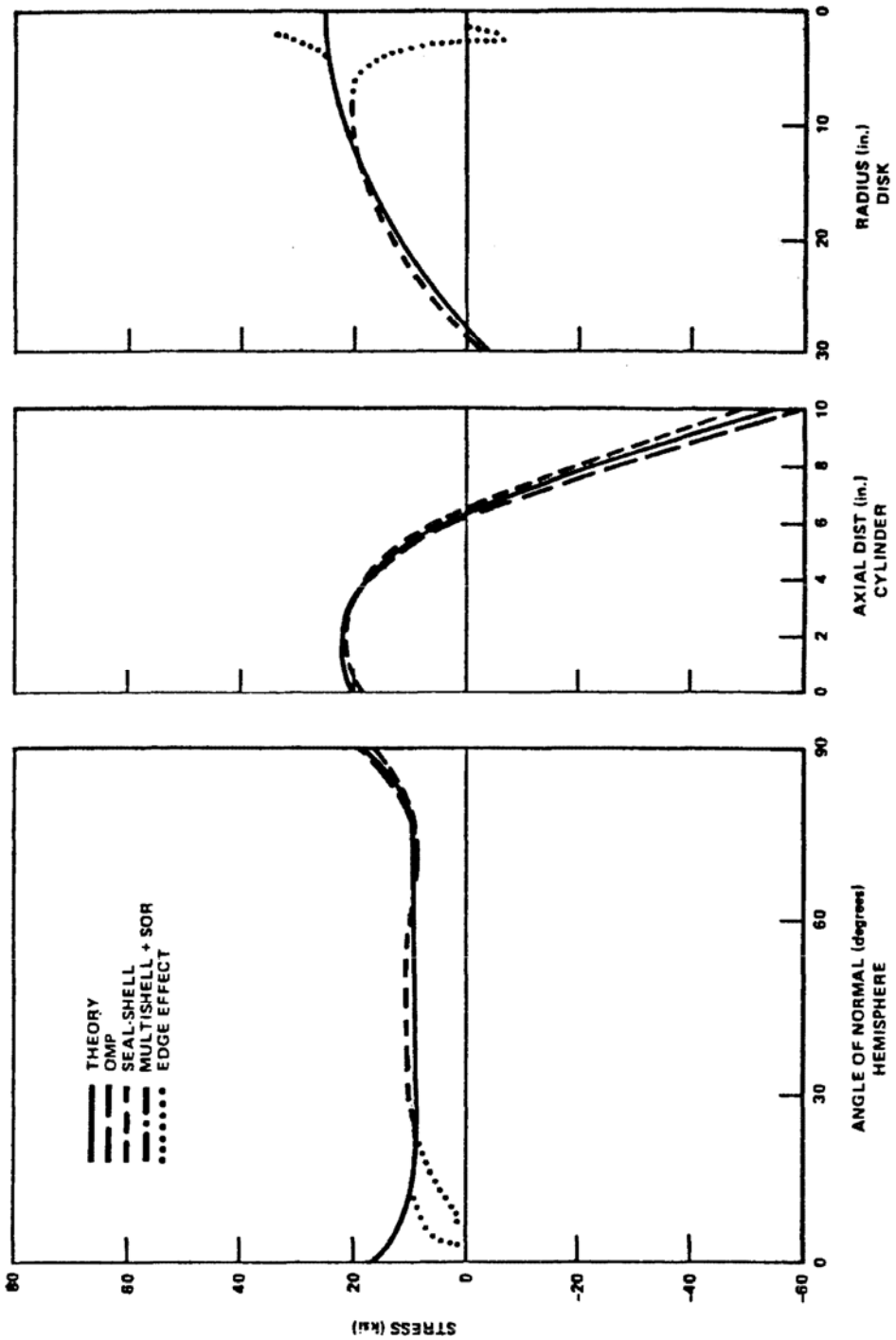


Figure 15.4 950 psi Internal Pressure Axial Stress - Outer Surface

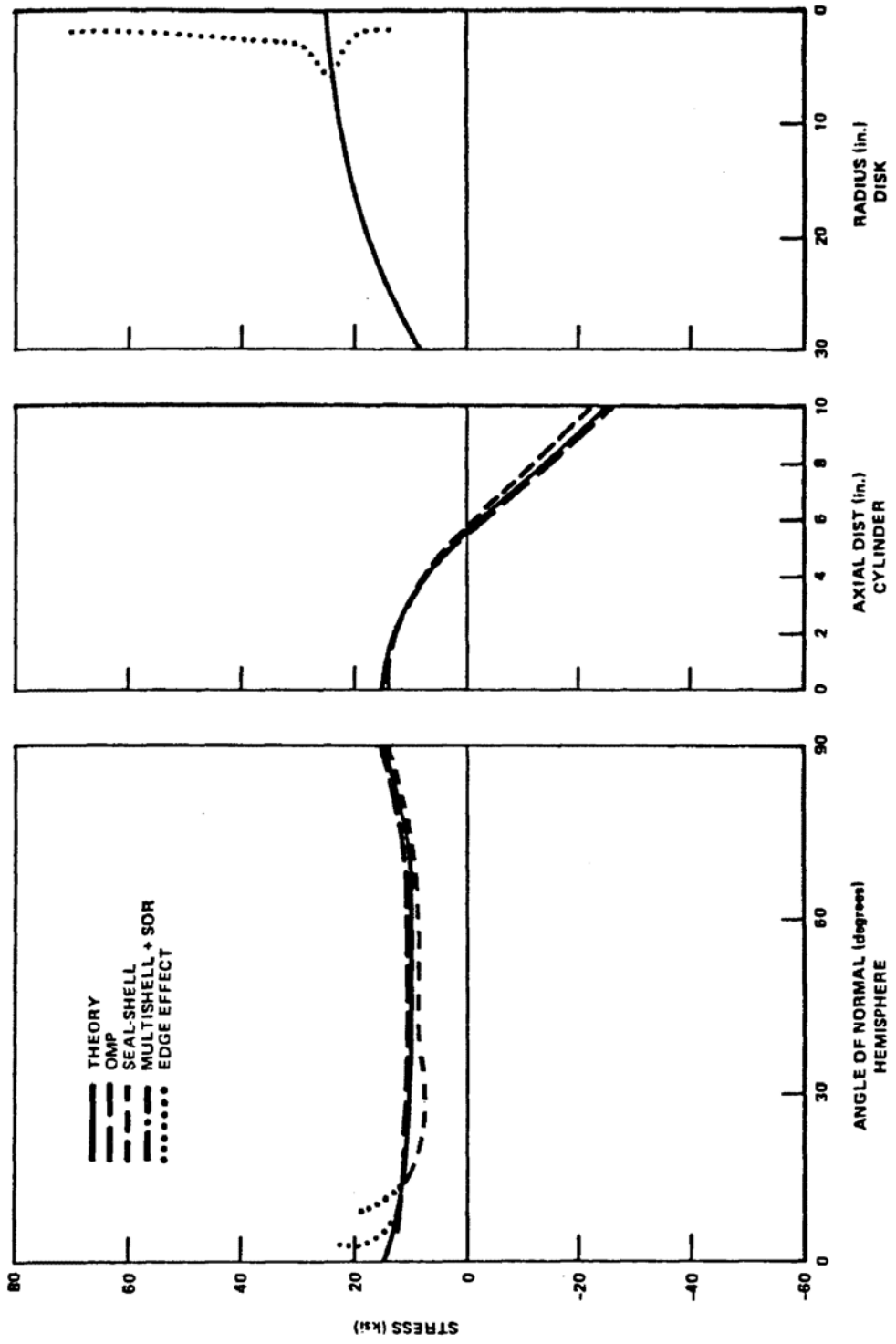


Figure 15.5 950 psi Internal Pressure Hoop Stress - Outer Surface

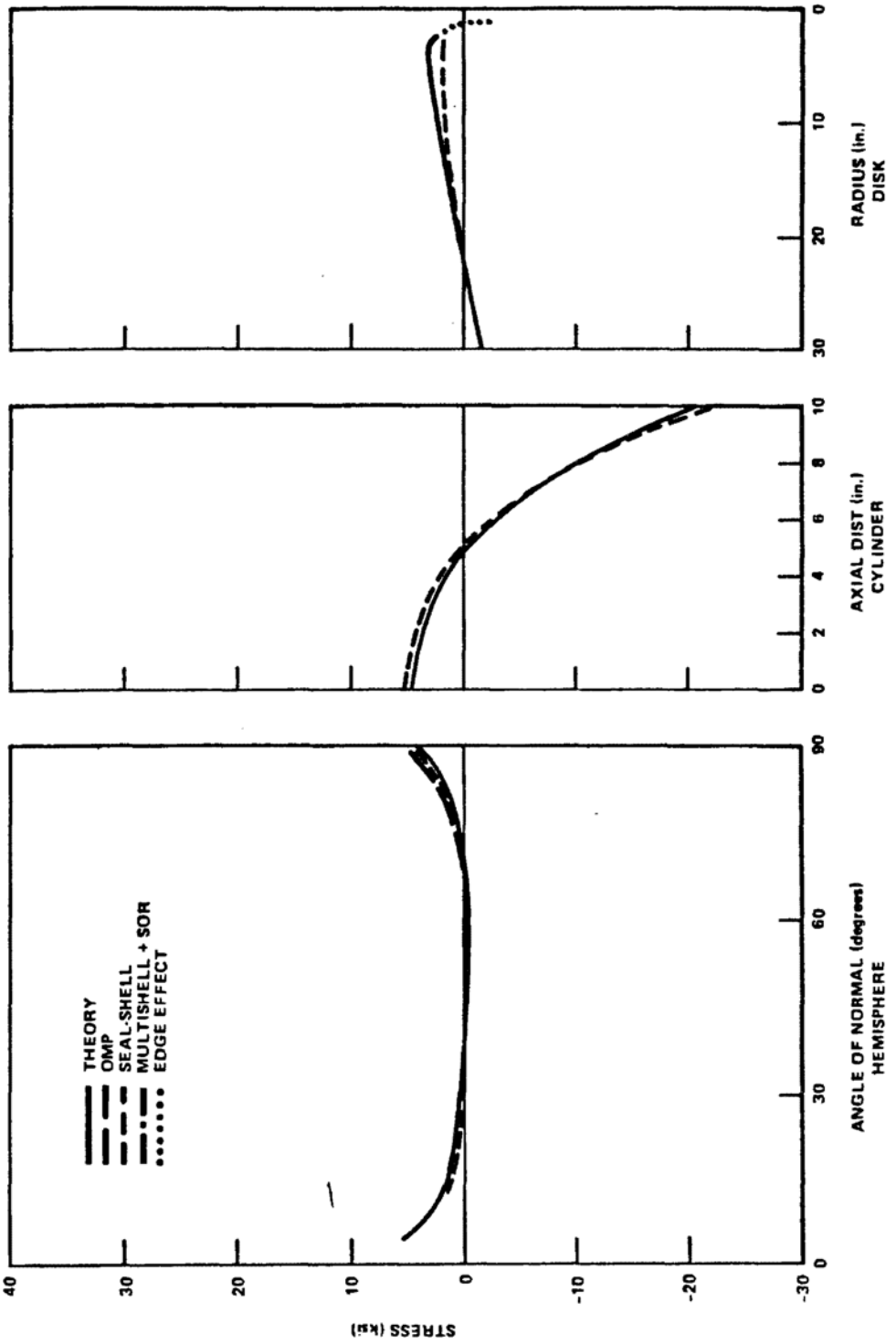


Figure 5.6 Axial Temperature Gradient Axial Stress - Outer Surface

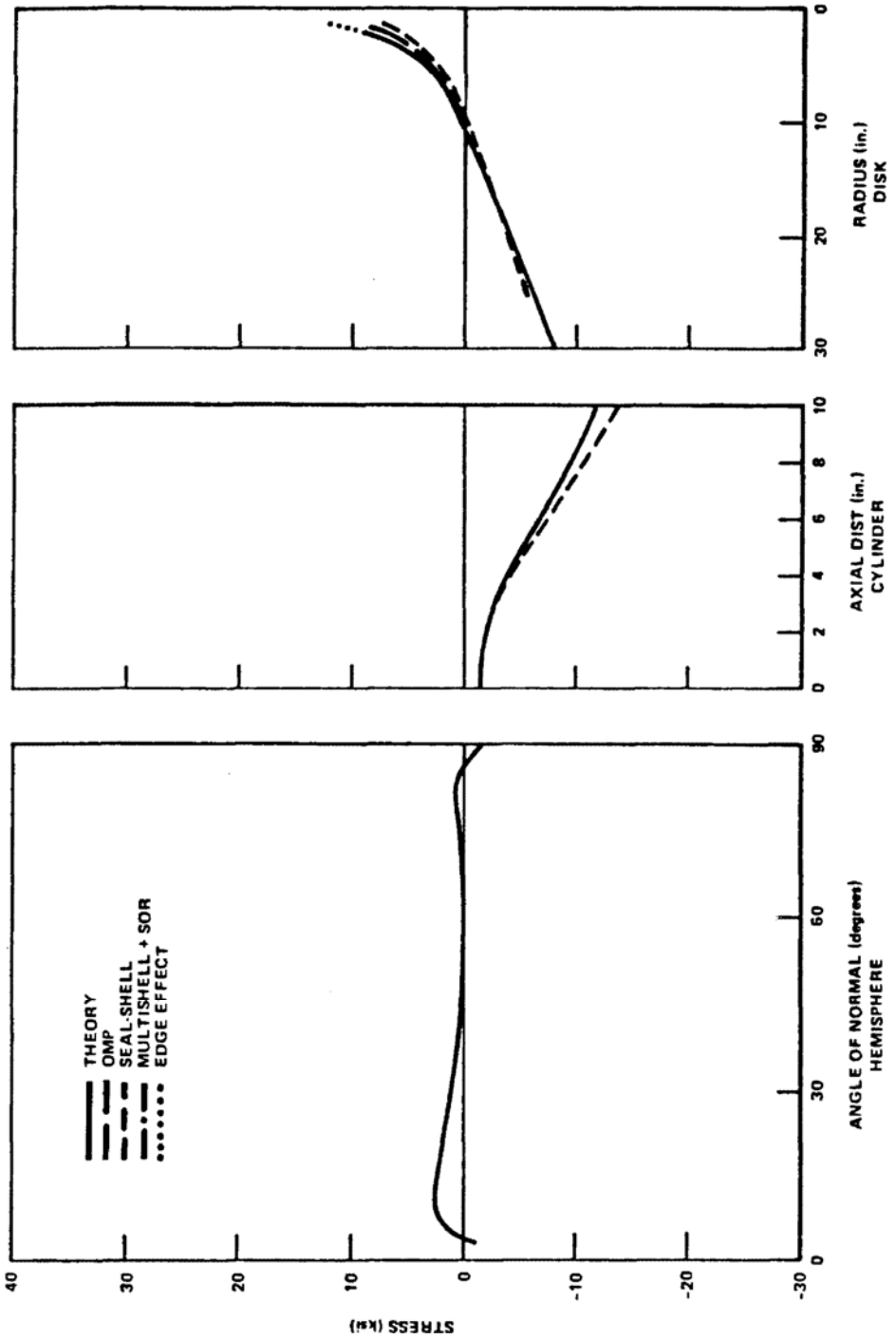


Figure 15.7 Axial Temperature Gradient Hoop Stress - Outer Surface

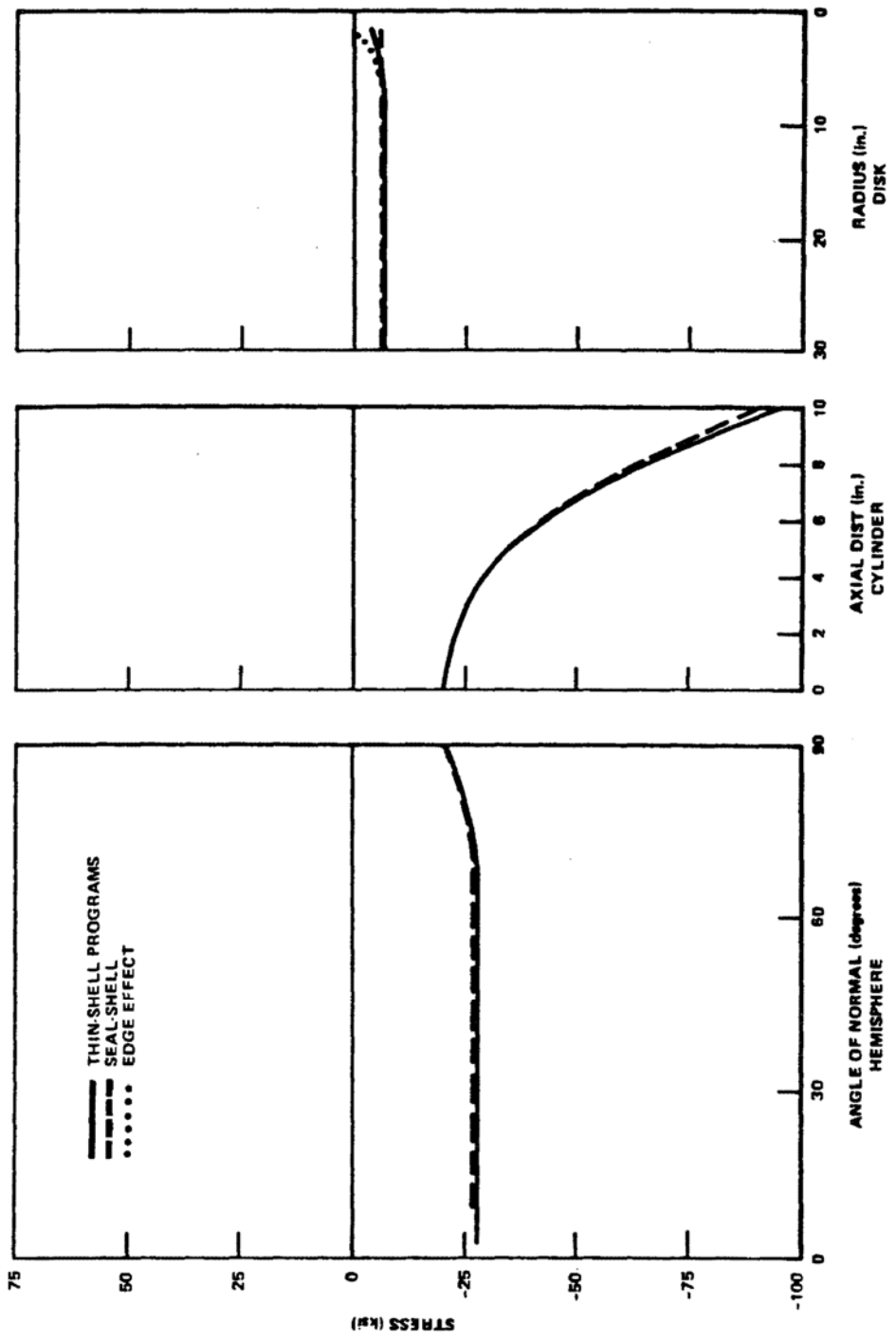


Figure 15.8 Linear Radial Temperature Gradient Axial Stress - Outer Surface

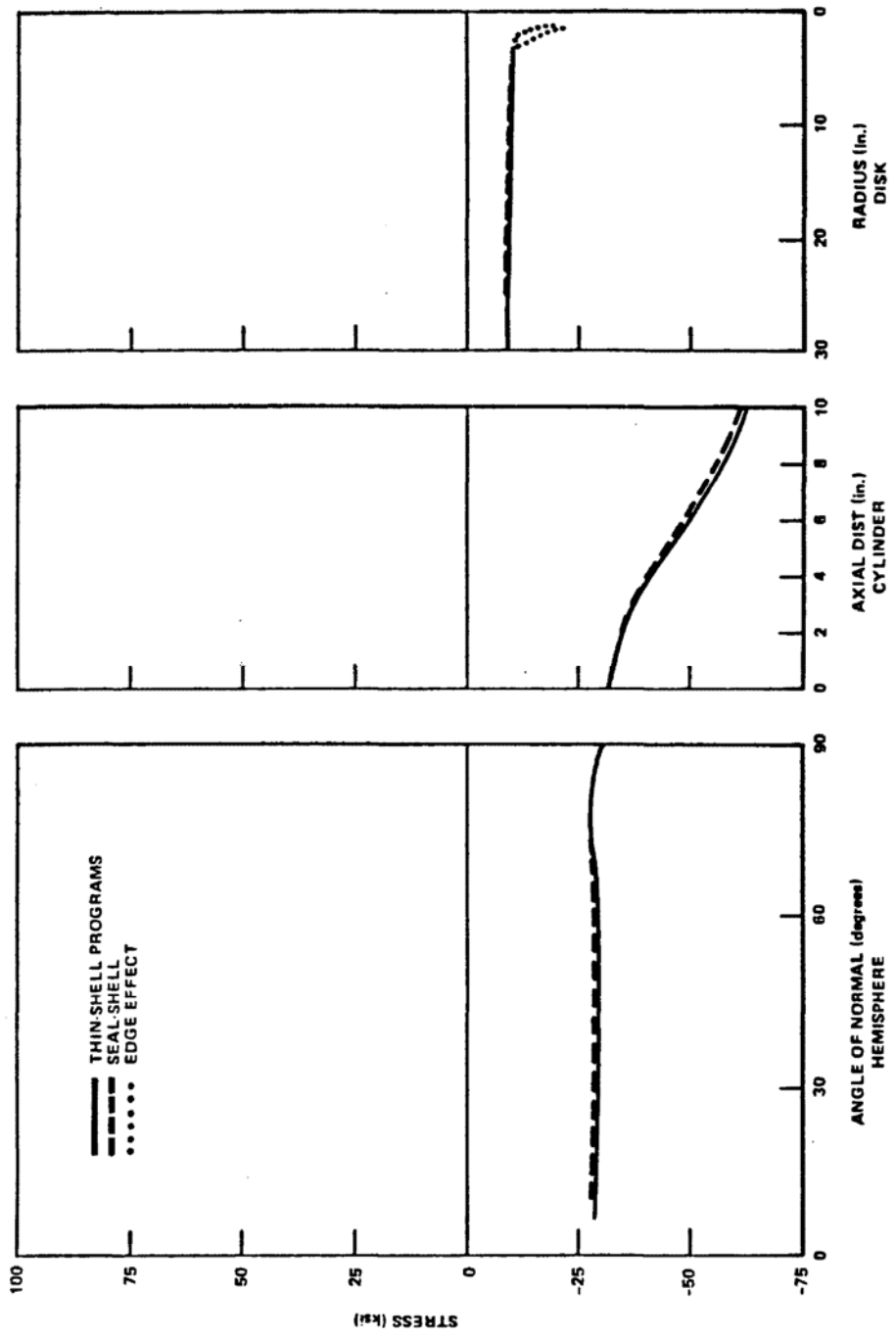


Figure 15.9 Linear Radial Temperature Gradient Hoop Stress - Outer Surface

3F.16 STARDYNE

Program Description

STARDYNE is a static and dynamic structural analysis system of computer programs which analyze linear elastic structural models. The STARDYNE finite element program can be used to evaluate a wide variety of static and dynamic problems:

I Program Capabilities:

a. The static capability includes the computation of structural deformations and member loads and stresses caused by an arbitrary set of thermal, nodal applied loads, and/or prescribed displacements.

b. Dynamic response analyses can be performed for transient, steady-state harmonic, random, and shock spectra excitation loading conditions. Dynamic response results can be presented as structural deformations (displacements, velocities, or accelerations) and/or internal member loads/stresses.

c. The program generates data to plot the deformed and undeformed structure as well as transient deformations.

II Available Finite Modeling Elements:

a. Beam elements with shear stiffness in 3-D space

b. Thick and thin triangular plate elements

c. Quadrilateral plate element

d. Infinitely rigid members

e. Springs

f. Hexahedron (cube) solid element

g. Tetrahedron solid element

Program Version and Computer

The proprietary version maintained by the developer on the CDC 6600 computer is used by GEH.

History of Use

[HISTORICAL INFORMATION] [STARDYNE is a well-known program developed by Mechanics Research Institute in the 1960s. It has been continuously checked and maintained by MRI. Users of this program include people from the United States and from Europe.

GEH's usage of the program began in 1972.

Extent of Application

GEH has used the program for vibration analysis of reactor sparger and for static analyses of the shroud head and attachments.]

Test Problems

Two test problems are presented in Figures 16.1 - 16.6. Results from STARDYNE compare favorably with the theoretical results.

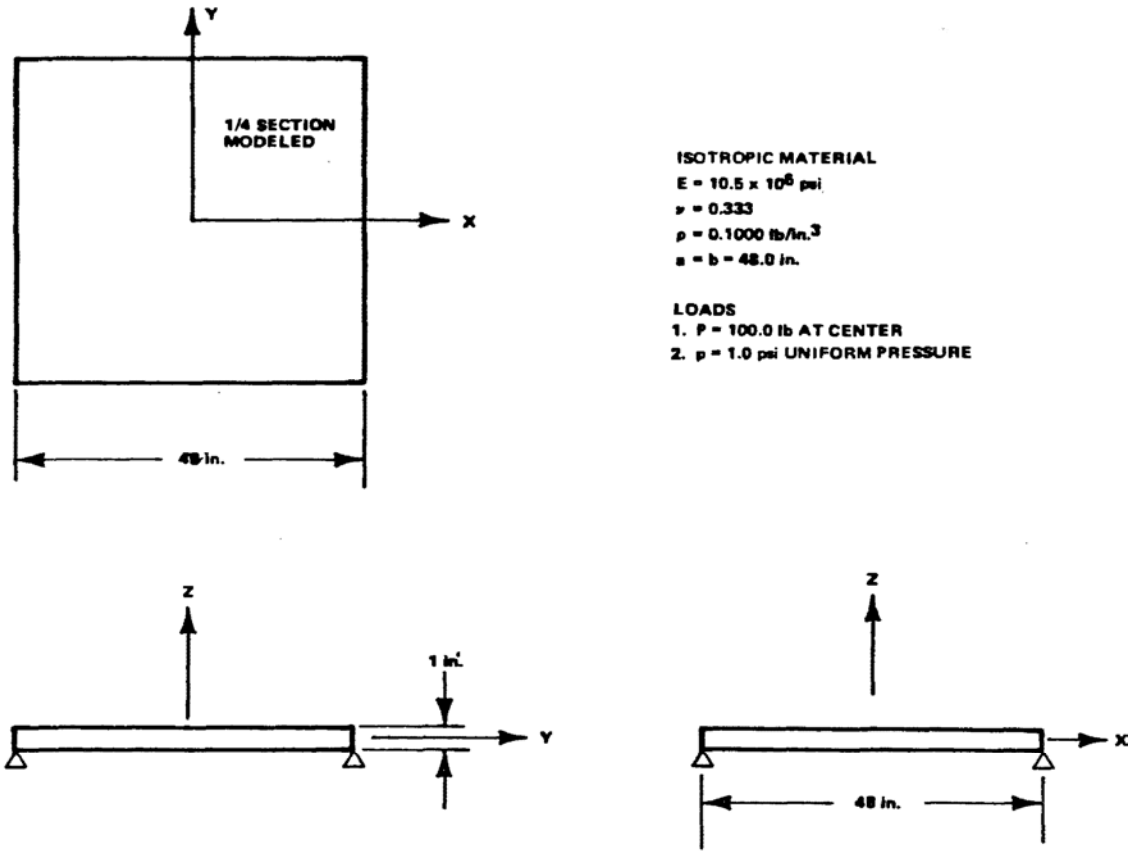


Figure 16.1. Simply Supported Square Plate

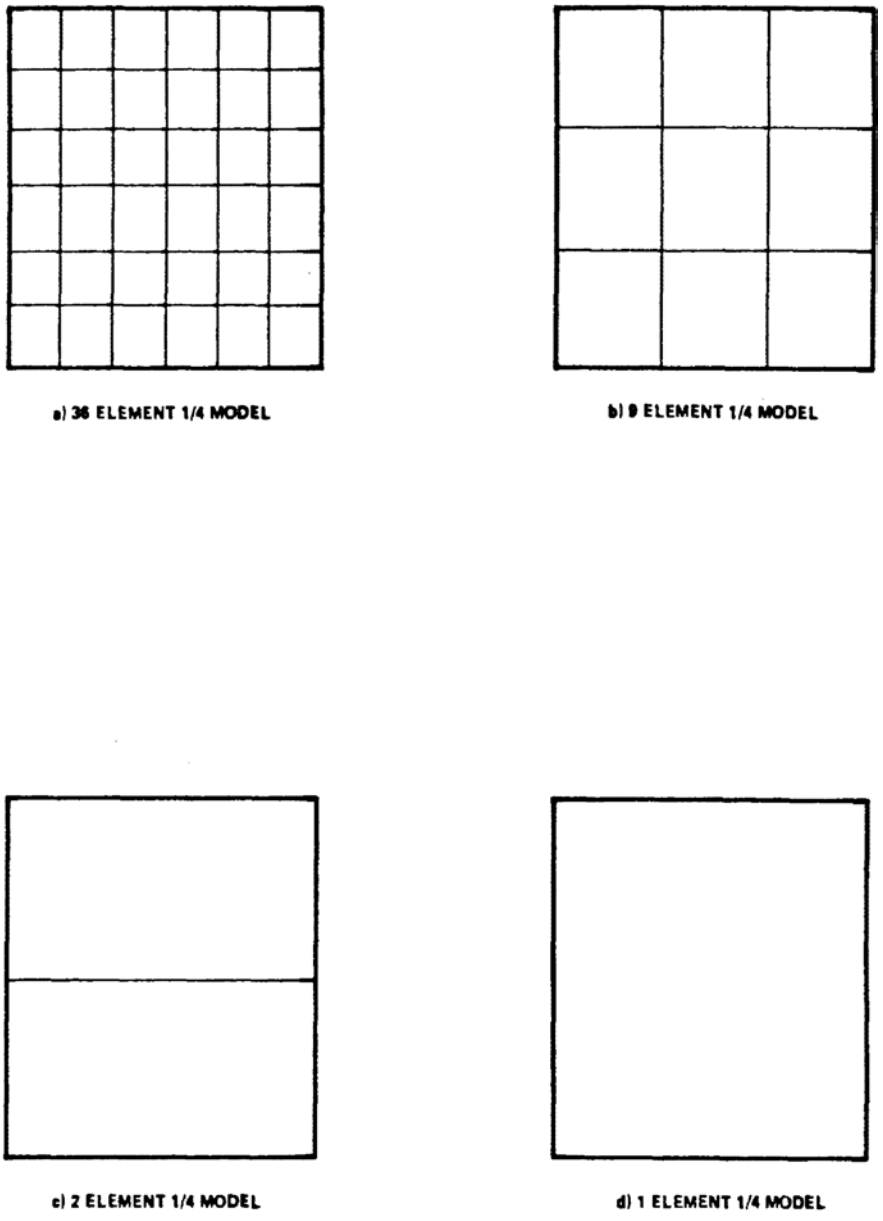


Figure 16.2. Quadrilateral Plate Element Models

MODE NO.	*1 THEORETICAL FREQUENCY (cps)	MRI/STARDYNE FREQUENCY, cps *2		
		No. of Elements Per 1/4 Model		
		1	9	36
1	84.08	82.21 (97.8)	84.64 (100.7)	84.23 (100.2)
2	420.42	-	421.18 (100.1)	421.59 (100.3)
3	756.75	-	759.97 (100.4)	768.25 (101.5)
4	1093.09	-	966.07 (88.4)	1093.27 (100.0)

*1 See Reference [3.5]

*2 Numbers in parenthesis are the percentage comparisons of the finite element solution to the theoretical solution.

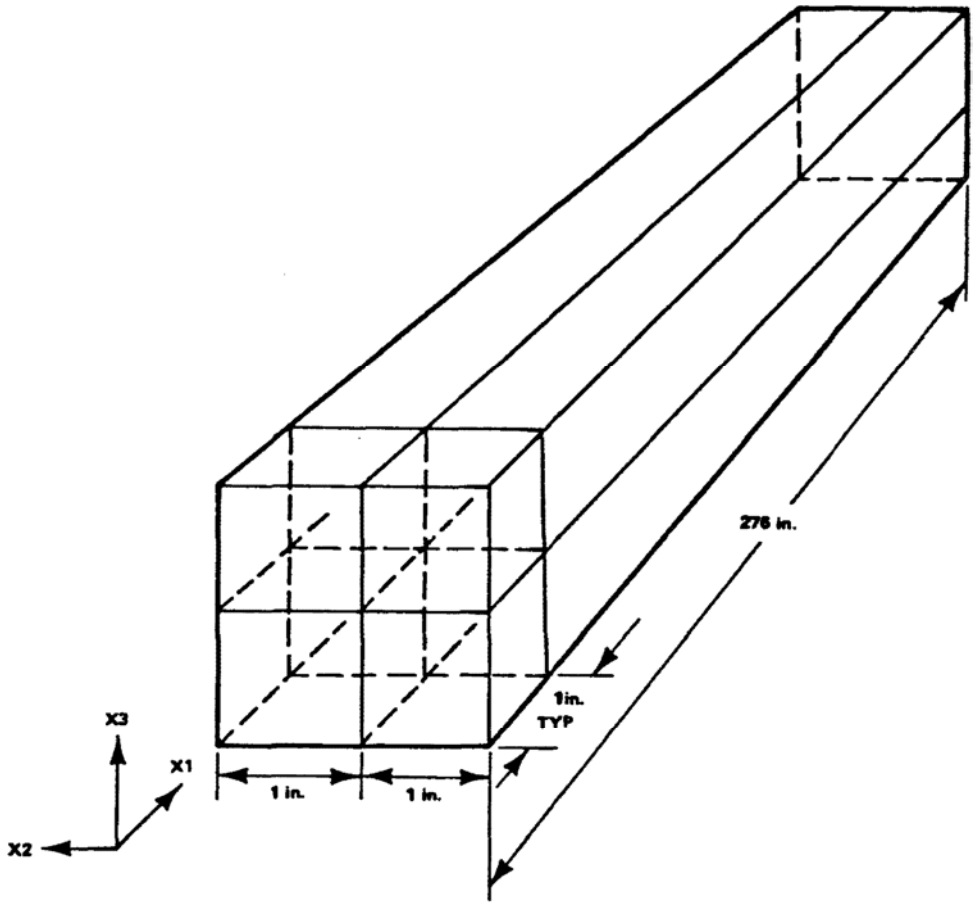
Figure 16.3. Simply Supported Square Plate Frequency Values (Quadrilateral Plate Elements)

LOAD CASE	*1 THEORETICAL DEFLECTION	MRI/STARDYNE SOLUTIONS *2			
		No. of Elements Per 1/4 Model			
		1	2	9	36
		(Displacements in inches x 10^{-2})			
Concentrated Load	.271580	.251430 (92.6)	.252219 (92.9)	.265684 (97.8)	.269891 (99.4)
Uniform Pressure Load	2.18997	2.27319 (104.0)	2.17961 (99.5)	2.16303 (98.8)	2.18395 (99.7)

*1 See References [3.1] and [3.2]

*2 Numbers in parenthesis are the percentage comparisons of the finite element solution to the theoretical solution.

Figure 16.4. Simply Supported Square Plate
(Quadrilateral Plate Elements)



2493 NODES (14958 DOF)
1104 CUBE (HEXAHEDRON) ELEMENTS

DENSITY	= 1 lb/in.^{3}
E	= 1.0 x 10^6 psi
A	= 4 in.^{2}
I	= 1.333 in.^{4}
α	= 1.0 x 10^{-6} in./deg F
W_T	= WEIGHT, TOTAL = 1104 lb
W	= WEIGHT PER inch = 4 lb
L	= 276 inches

Figure 16.5. Cantilever Beam Modeled with Hexahedron Elements

LOADING TYPE	*1 THEORETICAL DEFLECTION (in)	*2 STARDYNE DEFLECTION (in)
100°F Change in Temperature	.02760	.02763 (100.1)
Deflecting Under its own Weight	.03809	.03806 (99.9)
.01G Lateral Acceleration	21.7604	21.7364 (99.9)

*1 See Reference [3.3]

*2 Numbers in parenthesis are the percentage comparisons of the finite element solution to the theoretical solution.

Figure 16.6. Large Hexahedron Model

3F.17 TIGER-5

Program Description

This program utilizes an electrical network analogy to determine the transient and steady-state temperature capabilities of a given geometry. The finite difference representation of the three-dimensional equations of heat transfer are repeatedly solved for small time increments. Linear mathematics are used to solve the mesh network for every time interval. Included in the analysis are the three basic forms of heat transfer (conduction, radiation, and convection), as well as internal heat generation. The output includes temperatures and surface heat flux for points representing the surface of the structure. The program always employs a solution time increment which guarantees stability of the solution.

Program Version and Computer

The current version of TIGER in GE/NED was originally obtained from the General Electric Knoll Atomic Power Laboratory. This version of program operates on the Honeywell 6000 computer.

History of Use

[HISTORICAL INFORMATION] [TIGER-5 is a well-known and well-checked program. The development of the program dates back to 1959 (Ref. 17.1). The current version in GE/NED has been in production status since 1965.

Extent of Application

The program had been used for the analysis of most reactor components in General Electric before the Heat Analysis Program described in Section 8 was developed. The latter program is now used for all two dimensional analyses.]

Test Problems

A comparison of the results from TIGER-5 and from the Heat Analysis Program has been made using a test problem shown in Figure 17.1. Results are shown in Figure 17.2.

Reference

17.1 P. Bray and S. J. McCracken, "TIGER II - An IBM-704 Digital Computer Program: Temperatures from Internal Generation Rates," KAPL 2044, AEC Research and Development Report, 1959.

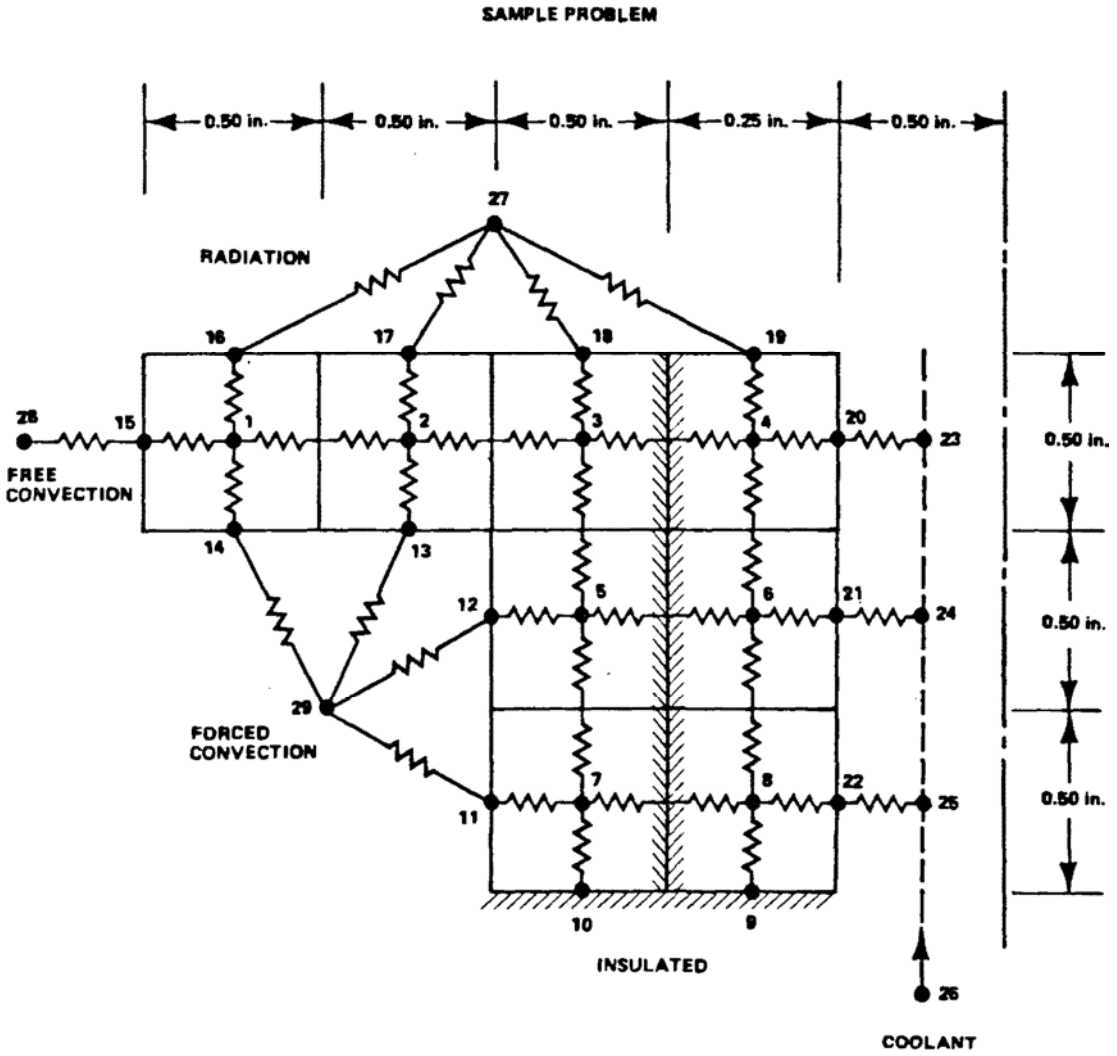


Figure 17.1. TGRV Model

Material Properties

Material No.	C_p (BTU/#-°F)	ρ (#/IN^3)	K (BTU/HR-FT-°F)
1	0.080	0.25	20.0
2	0.120	0.30	8.0

Initial Heat Generation

Node Number	Nominal Power Density
	(BTU/HR-IN^3)
1	100.0
2	100.0
3	100.0
4	300.0
5	100.0
6	300.0
7	100.0
8	300.0

Power Fraction vs. Time

Time (min)	(P/Po)
0.00	1.0
1.0	2.0
1.01	2.0
2.00	3.0
5.00	3.0

Figure 17.2. Material Properties and Heat Generation of Test Problem

Surface Node	Sink Node	Heat Transfer Mechanism	Initial Values	h = f(t)
9-10		Insulated		No
11-14	29	Forced Convection	h = 100 BTU/HR-FT^2-°F	No
15	28	Free Convection	h = 25 $\Delta T^{.25}$	No
16-19	27	Radiation	= F = 1.0	No
20	23	Forced Convection	h = 5000	Yes
21	24	Forced Convection	h = 5000	Yes
22	25	Forced Convection	h = 5000	Yes

Boundary Node	Temp. (°F)	T = f(t)
26	500	Yes
27	100	No
28	100	No
29	100	No

Time Min.	T °F
0.1	500
1.0	500
1.01	500
2.0	300
5.0	300

Time (Min)	h (BTU/HR-FT^2-°F)
0.0	5000
1.0	5000
1.01	2500
2.0	2500
5.0	2500

Coolant Nodes

T °F	C_p (BTU/ #-°F)	ρ (Density) (#/FT^3)
200	1.06	60.0
300	1.08	57.00
400	1.13	53.0
500	1.25	48.0
600	1.56	42.5

Time (Min)	W (Flow Rate) (#/HR)
0.0	50.0
1.0	50.0
1.01	25.0
2.00	25.0
5.00	25.0

Figure 17.3. Boundary Conditions of Test Problem

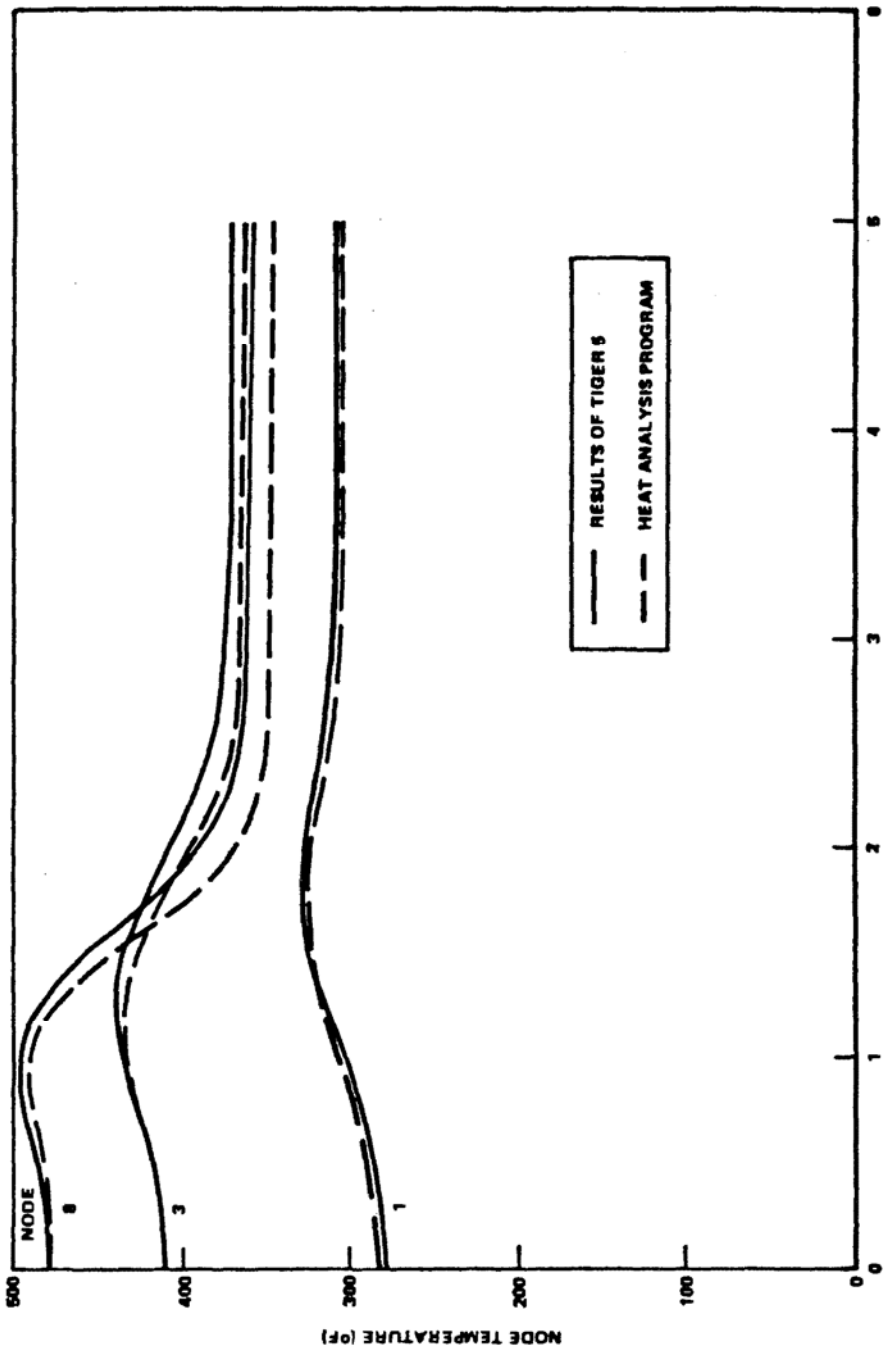


Figure 17.4. Comparison of Results from TIGER-5 and the Heat Analysis Program

3F.18 TIME DEPENDENT PIPE FORCES

Program Description

Time Dependent Pipe Forces is a computer program to calculate time- and space-dependent pipe reaction forces caused by either sudden blowdown or flow stoppage in a fluid piping system. The analytical model is a uniform pipe with arbitrary bends and friction, a pressure source at one end and a sudden opening or closing valve at the other end. A homogeneous mixture of fluid and ideal gas flows in the system. The method of characteristics is used to obtain fluid-mechanical properties, which then are employed to predict associated pipe loads. The computer program currently can accommodate up to 15 pipe segments. Reaction forces are obtained either as single axial loads applied to each straight segment anywhere between two adjacent elbows or as axial loads applied at the valve and other points coincident with each elbow.

Program Version and Capability

[HISTORICAL INFORMATION] [The computer program was developed by F. Moody of the General Electric Nuclear Energy Division. The version being used was installed on the Honeywell 6000 computer at the beginning of 1973.

History of Use

The history of use of the program is relatively short. However, its analytical model is based on proven fluid theories and its programming has been verified using cases with known solutions (Ref. 18.1).

Extent of Use

The program will be used to calculate reactor piping forces caused by hypothetical blowdowns.]

Test Problems

Figure 18.1 shows that the present program closely predicts the reaction data of Hanson (Ref. 18.2) for blowdown of 60 F water from a two-segment pipe, initiated by a rupture disk at one end. Effects of pipe area change were included in the analysis.

References

18.1 Moody, F. J. "Time-Dependent Pipe Forces Caused by Blowdown and Flow Stoppage," ASME paper No. 73-FE-23 presented at the Applied Mechanics and Fluids Engineering Conference, Atlanta, GA, June 1973.

18.2 Hanson, G. H., "Subcooled Blowdown Forces on Reactor-System Components: Calculational Methods and Experimental Confirmation," Idaho Nuclear Corporation Report IN-1354, June 1970.

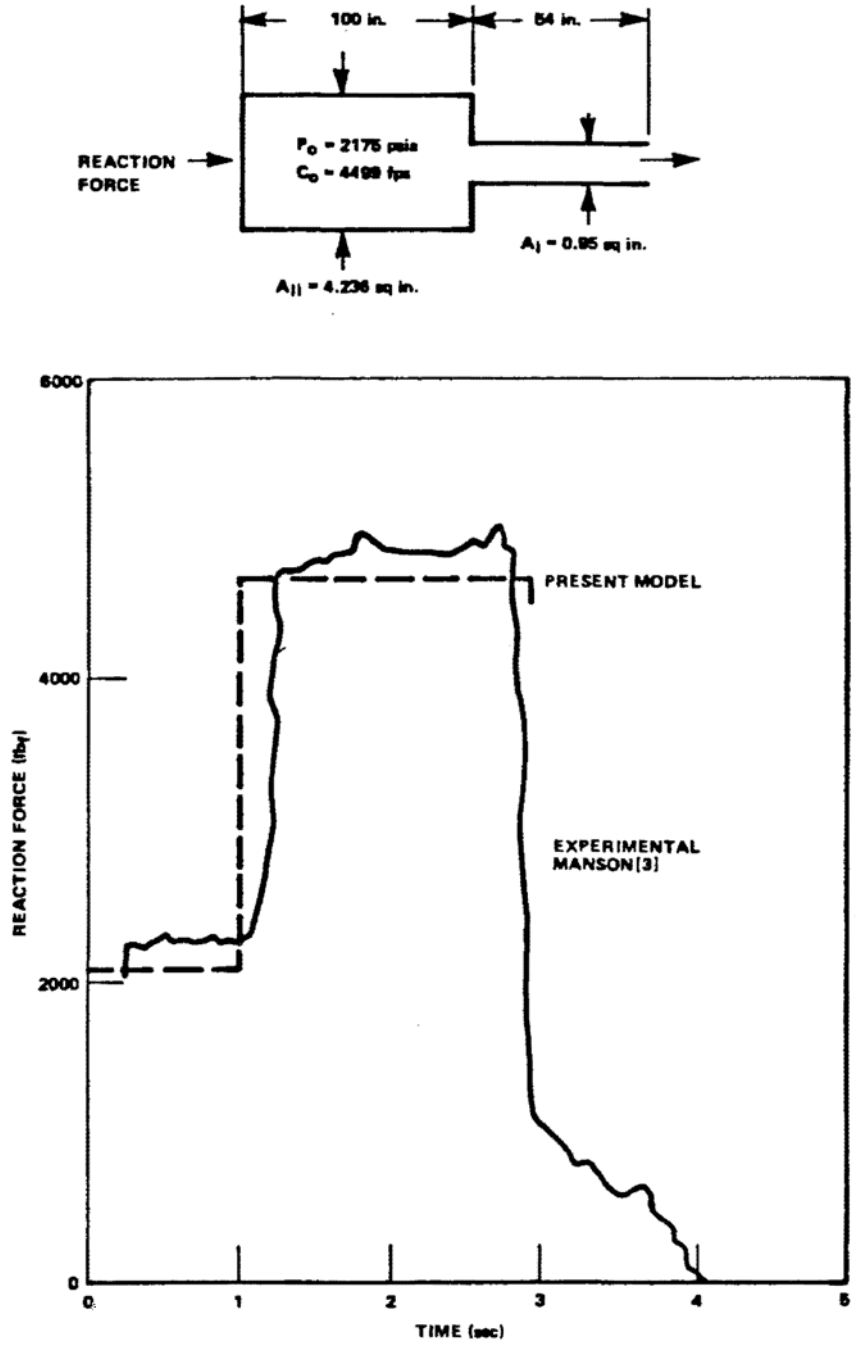


Figure 18.1. Cold Water Wave Force Comparison

3F.19 PIPE DYNAMIC ANALYSIS PROGRAM (PDA)

Program Description

PDA is a computer program used to determine the response of a pipe subjected to the thrust force occurring after a pipe break. The program treats the situation in terms of a generic pipe break configuration, which involves a straight, uniform pipe fixed at one end and subjected to a time-dependent thrust force at the other end. A typical restraint used to reduce the resulting deformation is also included at a location between the two ends. Nonlinear and time-independent stress-strain relations are used for the pipe and the restraint. Similar to the popular plastic-hinge concept, bending of the pipe is assumed to occur only at the fixed end and at the location supported by the restraint. Shear deformation is also neglected. The pipe bending moment-deflection (or rotation) relation used for these locations is obtained from a static nonlinear cantilever beam analysis. Using the moment-rotation relation, nonlinear equations of motion of the pipe are formulated using an energy consideration and the equations are numerically integrated in small time steps to yield time-history information of the deformed pipe.

Program Version and Computer

[HISTORICAL INFORMATION] [The original version developed by L. D. Steinert of the Nuclear Energy Division of General Electric in 1972 is currently used on the company's Honeywell 6000 computer (Ref. 19.1).

History of Use

Since its development in 1972, the program has been applied to a number of pipe break studies. Its prediction accuracy was examined in a combined analytical and experimental study completed in April 1973 by the Detroit Edison Company and the General Electric Company in connection with the design of the Enrico Fermi Power Plant Unit No. 2. The analytical results predicted using the PDA program agree reasonably well with the experimental data. Moreover, because of the conservatism used in the analysis, the analytical prediction of restraint strain and maximum pipe deflection is always on the conservative side. This

fact renders the computer program a desirable analysis tool for design purposes. Some of the results of this study are presented in the next section.]

Test Problems

Problem 1 - Comparison with Hand Calculations

The geometry and loading of the test problem is defined in Figure 19-1. Hand calculations of the deflections and forces can be made easily and are given as follows:

a. Total Pipe Deflection

The total pipe tip deflection is obtained by equating the input energy (W_F) to the strain-energy of the restraint (W_R).

$$W_F = \int_0^{\delta_{P(total)}} F \; d\delta_P = F \; \delta_{P(total)}$$

$$W_R = \int_0^{\delta_{R(total)}} F_R d\delta_R = \int_0^{\delta_R} K(\delta_R)^n \; d\delta_R = \frac{K(\delta_{R(total)})^{n+1}}{n+1}$$

Using $\delta_R = \delta_P - c$, the equation becomes an equation for δ_P (total pipe deflection):

$$F \; \delta_P = \frac{K(\delta_P - c)^{n+1}}{n+1}$$

Substituting the given values of F, K, c, and n, the numerical value of P can be found to be 1.915 in.

b. Total Restraint Deflection

$$\delta_R = \delta_P - c = 1.915 - 1.0$$

$$\delta_R = 0.915 \text{ in.}$$

c. Maximum Force on the Restraint

$$R_R = K \; \delta_R^{\,n} = 747{,}000(.915)^{.38}$$

R_R = 722,205 lb

d. Energy Absorbed by the Restraint

$$W_R = \frac{K(\delta_R)^{n+1}}{n+1} = \frac{747{,}000(.915)^{1.38}}{1.38}$$

W_R = 478,853 lb-in. (= 39,904 lb-ft.)

e. Velocity of Pipe at Impact with Restraint

W_{Fc} = Energy input through clearance = Fc

KE_p = Kinetic energy of pipe = 1/2 $I_p\dot{\theta}^2$

$$I_p = \int_0^L y^2\,dm = \int_0^L y^2\rho\,dy = \frac{\rho L^3}{3}$$

by conservation of energy,

$KE_p = W_{Fc}$

$$\dot{\theta} = \sqrt{2\frac{W_{Fc}}{I_p}} \text{ (Angular velocity of pipe)}$$

$$\rho = 397\frac{lb}{ft} \text{ (includes weight of water and weight of pipe)}$$

$$I_p = \frac{397\,lb}{3\,ft}(35)^3\ ft^3 = 56.7\times10^5$$

$$\dot{\theta} = \left[\frac{2(20833)(32.2)}{56.7\times10^5}\right]^{1/2} = .4863\ \frac{radians}{sec}$$

V_p = Velocity of the pipe end

$V_p = \dot{\theta}L = .4863(35\ ft) = 17.02\ ft/sec$

(small angles assumed $\rightarrow \cos\theta \approx 1$)

f. Impact Time

For conservation of momentum

$$I_p\dot{\theta} = FL\ t_I$$

$$t_I = \frac{I_p\dot{\theta}}{FL} = \frac{(56.7\times10^5)\ (.4863)}{(25.0\times10^4)\ (35)\ (32.2)}$$

t_I = 0.0098 sec.

Numerical results were also obtained from the computer code PDA, using the following input for the program variables:

CONST 1 = 0 (Restr. Elastic Const.)
CONST 2 = 747,000 lb/in.^{n} (Plastic Const.)
CLR = 1 in. (Initial Clearance)
DELFMX = 36 in. (Assumed Maximum Pipe Deflection)
FBLOWI = 250,000 lb (Initial Blowdown)
FBRAT 2 = 1
FBRAT 3 = 1 { Constants that modify FBLOWI }

IBUILT= = 0 (→Pinned-Joint)
ISTRES = 0 (→Load-Deflection Curve for Restr.)
KSPRNG= = 0 (→Structure is fixed, no Deflection)
NFIRST = 1
NFINAL = 1 { One Restraint to begin with, and only runs one case }

PIPEID = 21.83 in.
PIPEPS = .01 (Dummy number for pipe strain)
PIPEOD = 24.0 in.
PIPEWT = 397 lb/ft
PIPK = 93526 Load-Deflection constants for the pipe-not used because of pinned end
PIPN = .3183

PTMASS = 0 (No mass assumed at end of pipe)
RESEPS = 3.0 in. (Assumed maximum allowable restr. defl.)

RESTRF = 0 (Friction factor for restr. - not used here)
RESTRL = 35.0 ft. (Location of restr. from fixed end)
RESTRN = 0.38 (Exponent n for restr. load-deflection)
THIRDL = 0 (Pipe fixed at one end only)
TIME1 = 0.10 sec.

TIME2 = 0.10 sec. { Used with FBRAT2 and FBRAT3 respectively to modify FBLOW1 }

TOTLEN = 35.0 (Total pipe length)

X = 0 (Elastic deflection of restraint - not used here)
IFLAT = 1 (Modification of pipe bending behavior - not used here)
LOADOP = 0 (Option for forcing function)

The PDA results are compared with the hand calculations in Table 19.1. In agreement with other observations, this comparison shows PDA analysis to be generally conservative.

Problem 2 - Comparison with Experimental Results

A series of eight tests has been made in the pipe restraint test program mentioned earlier in the "History of Use" section. The tests, conducted in connection with the design of the Enrico Fermi Power Plant Unit No. 2, use the specimen and support system shown in Figure 19-2. The pipe is made of A106-B carbon steel which has an assumed stress-strain relation $\sigma = 81{,}800\ \varepsilon^{0.20}$ psi.

During the course of the test program certain parameters were varied to investigate the dynamic behavior of the pipe restraint system. The arrangement of the pipe test section and restraints is shown schematically in Figure 19-2. This figure also defines the parameters which were varied:

1. L2, the distance from the center line of a restraint assembly to the line of action of the thrust force.

2. CL, the initial hot clearance between the pipe and the restraint.

3. N, the number of restraint bars used (variable between 1 and 4).

4. θ, the angular orientation of the thrust force with respect to the restraint axis of symmetry (0 degrees in Figure 19-2).

5. Test section pipe schedule (4-inch Schedule 40, and 4-inch Schedule 80).

A pressure vessel was used to generate the required thrust force under the condition of a saturated water blowdown. A typical set of test results is compared in Table 19.2 with that predicted using PDA for the same thrust force. The PDA predictions for pipe deflection and restraint strain are always on the conservative side. PDA predictions on the time history of restraint force and mass acceleration are compared with the experimental results in Figures 19-3 and 19-4. The agreement between the predicted and measured results is generally good at the beginning of the impact.

References

19.1 Steinert, L. D. "PDS - Pipe Dynamic Analysis Program for Pipe Rupture Movement," General Electric Report No. NEDE-10813, 73NED27 Class II, March 1973.

19.2 G. A. Esswein, et al. "Pipe Restraint Testing Program Conducted in Connection with the Design of the Enrico Fermi Power Plant Unit No. 2" General Electric Report No. NEDE-10811, 73NED26, Class II, April 1973.

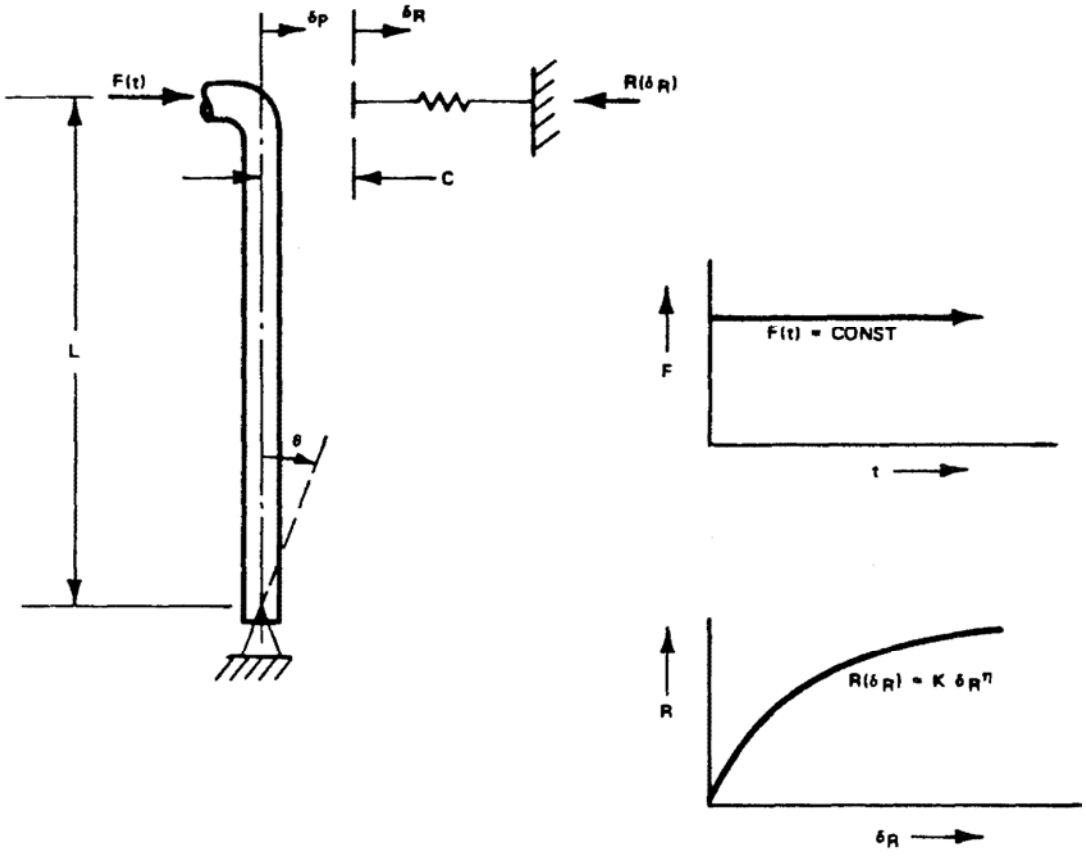
TABLE 19.1: Comparison of Hand Calculations with PDA Output

	Hand Calculations (Exact Solution)	PDA (Numerical Solution)
Total Pipe Deflection (inches)	1.915	1.923
Total Restraint Deflection (inches)	0.915	0.924
Force on Restraint (lb)	722,205	724,896
Energy Absorbed by Restraint (lb-ft)	39,904	40,447
Impact Velocity (ft/sec)	17.02	17.02
Impact Time (sec)	0.0098	0.0091

TABLE 19.2: Predicted Versus Measured Test Results

Test Description	Pipe Design Point Test
Date Performed	10/31/72
Test No.	8
Vessel Water Temp., °F	536
Saturation Pressure, psia	933
Pipe Size and Schedule	4" Sch. 40
Initial Hot Clearance, Inches	4-7/16
Overhand Length, Inches (L_2)	11
Number of Restraint Modules	2

Test Parameters	PDA Predictions	Measured Results
Thrust Force Used	7200	7200
Restraint Defl. (inches)	1.01	0.75
Restraint Strain (percent)	17.5	18.8 14.1
Pipe End Defl. (inches)	8.2	5.2
Pipe Strain, Hinge 2 (percent)	25.1	1.9
Max. Rest. Force (pounds)	23640	26200
Initial Accel. (g's)	22.8	28
Final Accel. (g's)	-69.3	-66
Impact Time (seconds)	0.0305	0.032
Stopping Time (seconds)	0.0474	0.040
Pipe Strain, Hinge 1 (percent)	1.09	
Moment, Hinge 1 (ft-lbs)	11,100	14,000



PIPE DIMENSIONS:

OD = 24 in.
ID = 21.83 in.
L = 35 ft
WEIGHT (INCLUDING WATER) = 397 lb/ft

F(t) = CONSTANT = 250,000 lb
K = 747,000 lb/(in.)^n
n = 0.38
c = INITIAL CLEARANCE = 1 in.

Figure 19-1 Test Problem 1

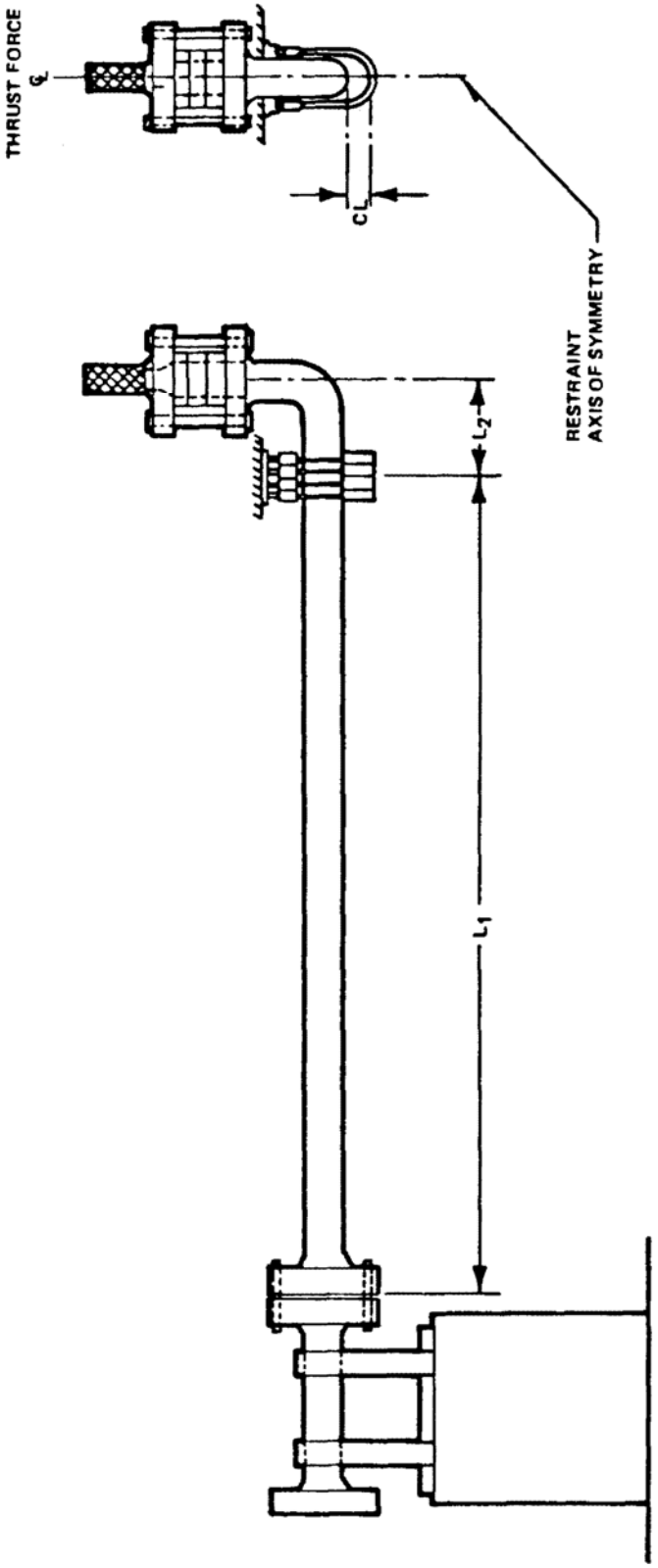


Figure 19-2 Typical Arrangement of Pipe and Restraint

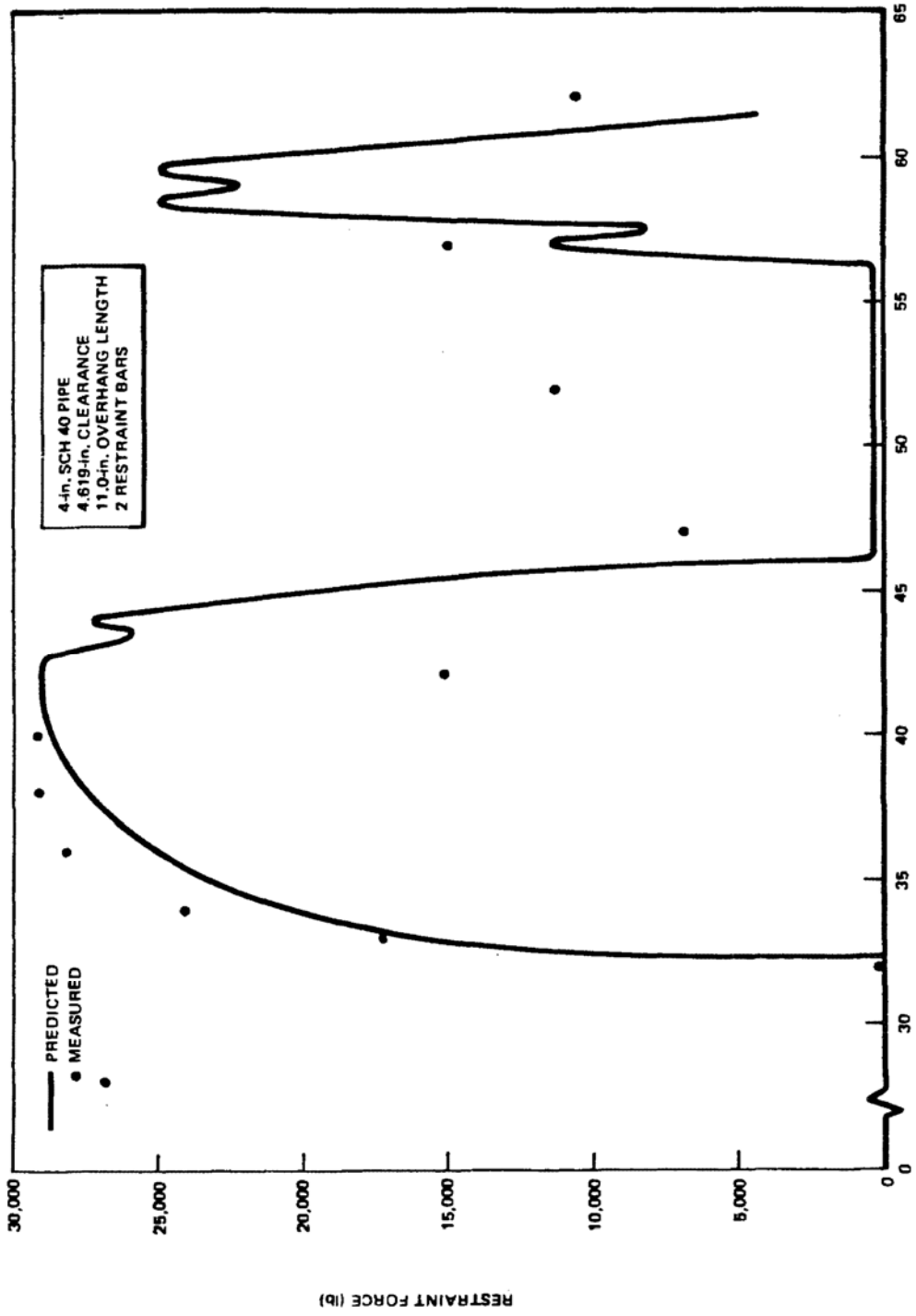


Figure 19-3 Restraint Force versus Time - Test 8

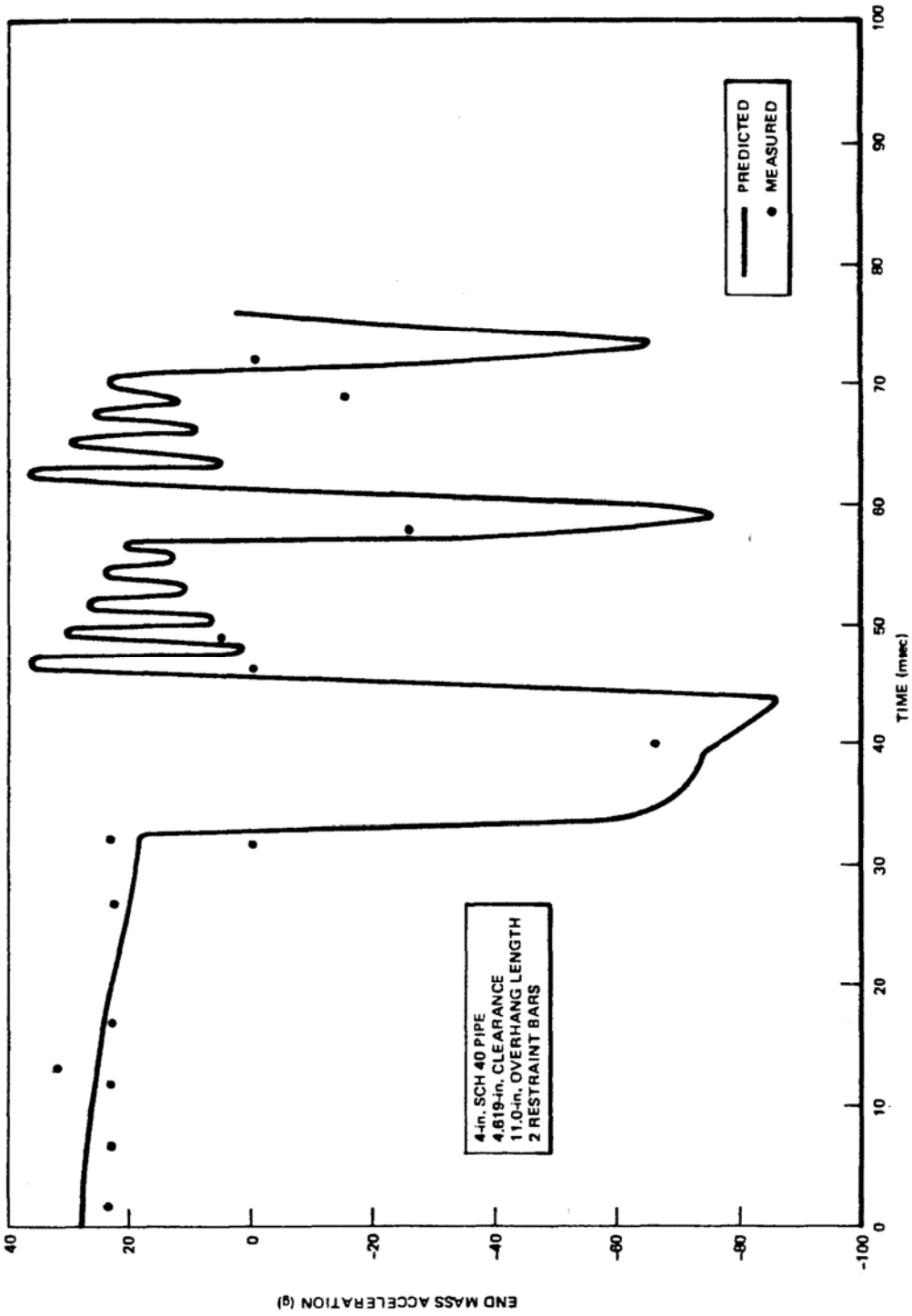


Figure 19-4 End Mass Acceleration versus Time - Test 8

APPENDIX 3G NRC REGULATORY REQUIREMENTS REVIEW COMMITTEE CATEGORY 2, 3, AND 4 MATTERS

The Office of Nuclear Reactor Regulation (NRR) has established a Regulatory Requirements Review Committee (RRRC) which reviews proposed changes to the regulatory requirements issued by the NRC staff and recommends a course of action to the Office of NRR. The course of action includes an implementation schedule.

The RRRC has developed a categorization system to aid in the uniform implementation of new and revised regulatory staff concerns. The system includes four categories (1 through 4) which correspond to the RRRC evaluation of the need for application of the regulatory concerns to new and ongoing license applications. The four categories are defined as follows:

Category 1: Matters to be applied in accordance with the implementation section of the published guide. The RRRC considers it necessary to forward fit (on new applications) the requirements of these matters.

Category 2: Matters for which applicability is to be determined on a case-by-case basis. The NRC staff will give further consideration to the need for backfitting certain identified items.

Category 3: Matters for which the NRC staff considers conformance necessary, either by direct implementation or by implementation of an acceptable alternative.

Category 4: Matters of concern to the NRC staff which have not been reviewed by the RRRC and subsequently categorized as Category 1, 2, or 3 matters.

During the January 9, 1979, meeting with the NRC in which the Grand Gulf licensing review schedule was discussed, numerous recommendations were made by the NRC staff to help speed the review process. One of the recommendations was to address, in the FSAR, all applicable NRC Category 2, 3, and 4 matters. These matters are addressed in this appendix.

CATEGORY 2 MATTERS

Document Number	Revision	Date	Remarks
R.G. 1.27	2	1/76	Addressed in FSAR Appendix 3A
R.G. 1.52	1	7/76	Addressed in FSAR Appendix 3A
R.G. 1.59	2	8/77	Addressed in FSAR Appendix 3A
R.G. 1.63	2	7/78	As applicable to Grand Gulf, R.G. 1.63, Revision 0, 10/73, and Revision 3, 2/87* are addressed in FSAR Appendix 3A
R.G. 1.91	1	2/78	Section D of the guide indicates that implementation of this guide is not applicable to Grand Gulf; however, the subject of this guide is addressed in FSAR subsection 2.2.3.1.1
R.G. 1.102	1	9/76	Addressed in FSAR Appendix 3A
R.G. 1.105	1	11/76	Not addressed in the FSAR. Section D of the guide indicates that implementation of this guide is not applicable to Grand Gulf.
R.G. 1.108	1	8/77	Addressed in FSAR Appendix 3A
R.G. 1.115	1	7/77	Addressed in FSAR Appendix 3A
R.G. 1.117	1	4/78	Addressed in FSAR Appendix 3A

*Note: Regulatory Guide 1.63 February 1987 applies only to fiber optic penetration assemblies.

Document Number	Revision	Date	Remarks
R.G. 1.124	1	1/78	Not addressed in the FSAR. Section D of the guide indicates that implementation of this guide is not applicable to Grand Gulf.
R.G. 1.130	0	7/77	Not addressed in the FSAR. Section D of the guide indicates that implementation of this guide is not applicable to Grand Gulf.
R.G. 1.137	0	1/78	Section D of the guide indicates that implementation of this guide is not applicable to Grand Gulf; however, the subject of this guide is addressed in the response to NRC Question 040.44.
R.G. 8.8	2	3/77	Addressed in FSAR Appendix 3A.
BTP ASB 9.5-1	1	2/77	The subject of this BTP is addressed in FSAR Appendix 9A.
BTP MTEB 5-7	0	4/77	The subject of MTEB 5-7 (NUREG-0313) is addressed in FSAR subsection 5.2.3.5.
R.G. 1.141	0	4/78	Not addressed in the FSAR. Section D of the guide indicates that implementation of this guide is not applicable to Grand Gulf.

CATEGORY 3 MATTERS

Document Number	Revision	Date	Remarks
R.G. 1.56	1	7/78	As applicable to Grand Gulf, R.G. 1.56, Revision 0, 6/73, is addressed in FSAR Appendix 3A.
R.G. 1.68.2	1	7/78	As applicable to Grand Gulf, R.G. 1.68.2, Revision 0, 1/77, is addressed in FSAR Appendix 3A.
R.G. 1.97	1	8/77	The subject of this guide is discussed in FSAR Section 7.5
R.G. 1.99	1	4/77	Addressed in FSAR Appendix 3A
R.G. 1.101	1	3/77	Addressed in FSAR Appendix 3A
R.G. 1.114	1	11/76	Addressed in FSAR Appendix 3A
R.G. 1.121	0	8/76	Addressed in FSAR Appendix 3A
R.G. 1.127	1	3/78	As applicable to Grand Gulf, R.G. 1.127, Revision 0, 4/77, is addressed in FSAR Appendix 3A.
BTP RSB 5-1	1	1/78	The subject of this BTP is addressed in FSAR subsections 5.2.2.4.1 and 5.4.7.1.5, and in the response to NRC Question 212.20.
BTP RSB 5-2	0	3/78	The subject of this BTP is not applicable to Grand Gulf.

CATEGORY 4 MATTERS

A. Regulatory Guides

Document Number	Revision	Date	Remarks
R.G. 1.12	1	4/74	Addressed in FSAR Appendix 3A
R.G. 1.13	1	12/75	Addressed in FSAR Appendix 3A
R.G. 1.14	1	8/75	Addressed in FSAR Appendix 3A
R.G. 1.75	1	1/75	Addressed in FSAR Appendix 3A
R.G. 1.76	0	4/74	Addressed in FSAR Appendix 3A
R.G. 1.79	1	9/75	Addressed in FSAR Appendix 3A
R.G. 1.80	0	6/74	Addressed in FSAR Appendix 3A
R.G. 1.82	0	6/74	Addressed in FSAR Appendix 3A
R.G. 1.83	1	7/75	Addressed in FSAR Appendix 3A
R.G. 1.89	0	11/74	Addressed in FSAR Appendix 3A
R.G. 1.93	0	12/74	Addressed in FSAR Appendix 3A
R.G. 1.104	0	2/76	Addressed in FSAR Appendix 3A

B. Branch Technical Positions

Document Number	Title	Remarks
BTP MTEB 5-3	Monitoring of Secondary Side Water Chemistry in PWR Steam Generators	The subject of this BTP is not applicable to Grand Gulf.
BTP CSB 6-1	Minimum Containment Pressure Model for PWR ECCS Performance Evaluation	The subject of this BTP is not applicable to Grand Gulf.
BTP CSB 6-2	Control of Combustible Gas Concentraions in Containment Following a Loss-of-Coolant Accident	The subject of this BTP is addressed in FSAR subsections 6.2.5 and 7.5.1.2.8.3, Tables 3.2-1 and 3.11-7, and in responses to NRC Questions 021.33 and 311.11.
BTP CSB 6-3	Determination of Bypass Leakage Path in Dual Containment Plants	The subject of this BTP is addressed in subsections 6.2.3 and 15.6.5.5.3, and Table 6.2-42. Grand Gulf does not conduct local leakage rate testing of secondary containment isolaion valves, similar to types B and C testng; successful SGTS testing verifies leak tightness of all secondary containment penetrations.
BTP CSB 6-4	Containment Purging During Normal Plant Operations	The subject of this BTP is addressed in subsection 6.2.4.3.3.

B. Branch Technical Positions (Continued)

Document Number	Title	Remarks
BTP ASB 9.1	Overhead Handling Systems for Nuclear Power Plants	The subject of this BTP is addressed in subsections 15.7.5 and 9.1.2.3.3.
BTP ASB 10.1	Design Guidelines for Auxiliary Feedwater System Pump Drive and Power Supply Diversity for PWRs	The subject of this BTP is not applicable to Grand Gulf.

C. SRP Criteria

SRP Implementation Section/Date	Title	Remarks
3.5.3/11-24-75	Procedures for Composite Section Local Damage Prediction (SRP Section 3.5.3, Par. II.1.c)	The subject of this SRP section is addressed in FSAR subsection 3.5.3.
3.7.1/11-24-75	Development of Design Time History for Soil-Structure Interaction Analysis (SRP Section 3.7.1, Par. II.2)	The subject of this SRP section is addressed in FSAR subsection 3.7.1
3.7.3/11-24-75	Procedures for Seismic Subsystem Analysis (SRP Section 3.7.3, Par. II)	The subject of this SRP section is addresses in FSAR subsection 3.7.3.
3.8.1/11-24-75	Design and Construction of Concrete Containment (SRP Section 3.8.1 Par. II)	The design of the containment structure is addressed in FSAR subsection 3.8.1. The load combinations used meet or exceed ACI 349 Title No. 69-2 and SRP criteria.
3.8.2/11-24-75	Design and Construction of Steel Containments (SRP Section 3.8.2, Par. II)	The subject of this SRP section is not applicable to Grand Gulf.

C. SRP Criteria (Continued)

SRP Implementation Section/Date	Title	Remarks
3.8.3/11-24-75	Structural Design Criteria for Category I Structures Inside Containment (SRP Section 3.8.3, Par. II)	The subject of this SRP section is addressed in FSAR Section 5.4 and subsection 3.8.3 for internal structures and component supports, respectively. The design Par. II) meets or exceeds the load combinations of ACI 349 Title No. 69-2, ACI 318 and SRP criteria.
3.8.4/11-24-75	Structural Design Criteria for Other Seismic Category I Structures (SRP Section 3.8.4, Par. II)	The subject of this SRP section is addressed in FSAR subsection 3.8.4. The design meets or exceeds the load combinations of ACI 318 and SRP criteria.
3.8.5/11-24-75	Structural Design Criteria for Foundations (SRP Section 3.8.5, Par. II)	The subject of this SRP section is addressed in FSAR subsection 3.8.5.
3.7/11-24-75 11.2 11.3 11.4	Seismic Design Requirements for Radwaste Systems and Their Housing Structures (SRP Section 11.2, BTP ETSB 11-1, Par. B.v.)	The subjects of these SRP sections are addressed in FSAR subsections 3.8.4.1.2.2, 11.2.1.2, 11.3.1.3, and 11.4.1.2

C. SRP Criteria (Continued)

SRP Implementation Section/Date	Title	Remarks
3.3.2/11-24-75	Tornado Load Effect Combinations (SRP Section 3.3.2, Par. II.2.d)	The subject of this SRP section is addressed in FSAR subsection 3.3.2. The load combinations are in accordance with Bechtel Topical Report BC-TOP-3A, Revision 3, and this SRP section.
3.4.2/11-24-75	Dynamic Effects of Wave Action (SRP Section 3.4.2, Par. II)	The subject of this SRP section is not applicable to Grand Gulf.
10.4.7/10-1-75	Water Hammer for Steam Generators with Preheaters (SRP Section 10.4.7, Par. I.2.b)	The subject of this SRP section is not applicable to Grand Gulf.
4.4/11-24-75	Thermal-Hydraulic Stability (SRP Section 4.4, Par. II.5)	The subject of this SRP is addressed in FSAR subsection 4.4.4.6.
5.2.5/11-24-75	Intersystem Leakage Detection (SRP Section 5.2.5, Par. II.4) and R.G. 1.45	The subject of this SRP section is addressed in FSAR subsections 5.2.5, 7.6.1.4. and 7.6.2.4.2.1. The requirements of R.G. 1.45, 5/73, are addressed in FSAR Appendix 3A.

C. SRP Criteria (Continued)

SRP Implementation Section/Date	Title	Remarks
3.2.2/11-24-75	Main Steam Isolation Valve Leakage Control System (SRP Section 10.3 Par. III.3 and BTP RSB 3.2)	The subject of this SRP section is addressed in FSAR subsection 6.7.1.1.2 (indicating compliance with R.G. 1.26, R.G. 1.29, and R.G. 1.96).
3.5.3/12-1-76	Ductility of Reinforced Concrete and Steel Structural Elements Subjected to Impactive or Impulsive Loads	The subject of this SRP section is addressed in FSAR subsection 3.5.3.
3.7.1/8-1-76	Response Spectra in Vertical Direction	The subject of this SRP section is addressed in FSAR subsection 3.7.1.1.2 and Appendix 3A (R.G. 1.60). The design earthquake in the vertical direction is taken as two-thirds of that established for the horizontal direction.
3.8.1/4-1-76 3.8.2	BWR Mark III Containment Pool Dynamics	The subjects of these SRP sections are addressed in FSAR subsections 3.8.1.3 and 3.8.1.4.2 for load combinations and suppression pool liner, respectively.

C. SRP Criteria (Continued)

SRP Implementation Section/Date	Title	Remarks
3.8.4/9-1-76	Air Blast Loads	The subject of this SRP section is addressed in FSAR Section 3.8. Air blast loads from transportation, as addressed in FSAR subsection 2.2.3.1.1, are less than the external pressure design capabilities as addressed in FSAR Section 3.8.
3.5.3/10-1-76	Tornado Missile	The subject of this SRP Impact section is addressed in FSAR subsection 3.5.3.1.
6.3/6-1-77	Passive Failure During Long-Term Cooling Following LOCA	The subject of this SRP section is addressed in response to NRC Questions 211.74 and 212.22.
6.3/9-1-77	Control Room Position Indication of Manual (Handwheel) Valves in the ECCS	The subject of this SRP section is addressed in FSAR subsection 7.5.1.3.
15.1.5/4-1-77	Long-Term Recovery from Steamline Break: Operator Action to Prevent Over-pressurization	The subject of this SRP section is not applicable to Grand Gulf.
5.4.6/12-1-77 5.4.7 6.3	Pump Operability Requirements	The subjects of these SRP sections are addressed in FSAR
3.5.1/3-28-78	Gravity Missiles, Vessel Seal Ring Missiles Inside Containment	The subject of this SRP section is addressed in FSAR subsections 3.5.1 and 9.1.4.2

C. SRP Criteria (Continued)

SRP Implementation Section/Date	Title	Remarks
4.4/1-1-77	Core Thermal-Hydraulic Analysis	The subject of this SRP section is addressed in FSAR subsection 4.2.3.1 and FSAR Section 4.4.
8.3/1-1-78	Degraded Grid Voltage Conditions	The subject of this SRP section is addressed in FSAR subsections 8.3.1.1.2.2 and 8.3.1.1.3.
6.2.1.2/6-1-76	Asymmetric Loads on Components Located Within Containment Sub-compartments	The subject of this SRP section is addressed in FSAR subsection 6.2.1.2.
6.2.6/9-1-77	Containment Leak Testing Program	The subject of this SRP section is addressed in FSAR subsection 6.2.6 and FSAR Table 6.2-49.

C. SRP Criteria (Continued)

SRP Implementation Section/Date	Title	Remarks
6.2.1.4/1-1-77	Containment Response Due to Main Steam Line Break and Failure of MSLIV to Close	The subject of this SRP section, as applicable to Grand Gulf, is addressed in FSAR Section 3.11 and subsection 6.2.1. Tables 6.2-3 through 6.2-5 indicate DBA temperatures and pressures. Figures 6.2-2 and 6.2-5 depict maximum containment pressures vs. time. Figures 6.2-3 and 6.2-7 depict maximum tempera- tures vs. time. Further, Figures 3.8-11 through 3.8-15 depict thermal gradients across the containment boundary concrete, and Figures 3.8-68 through 3.8-72 depict thermal gradients across the drywell wall and roof.
3.6.1/11-1-77 3.6.2	Main Steam and Feedwater Pipe Failures	The subjects of these SRP sections are addressed in FSAR Sections 3.6 and 3.11 and in FSAR Appendix 3C.
9.2.2/1-1-77	Design Requirements for Cooling Water to Reactor Coolant Pumps	The subject of this SRP section is addressed in FSAR subsections 9.2.2 and 5.5.2.1.
10.4.7/8-1-76	Design Guidelines for Water Hammer in Steam Generators with Top Feedring Design	The subject of this SRP section is not applicaable to Grand Gulf.

C. SRP Criteria (Continued)

SRP Implementation Section/Date	Title	Remarks
3.11/1-1-76	Environmental Control Systems for Safety-Related Equipment	The subject of this SRP section is addressed in FSAR Sections 3.11 and 9.4, and in the response to NRC Question 031.035.