3.1 CONFORMANCE WITH NRC GENERAL DESIGN CRITERIA

3.1.1 SUMMARY DESCRIPTION

This section contains an evaluation of the design bases of the Susquehanna Steam Electric Station Units 1 and 2 as measured against the NRC General Design Criteria for Nuclear Power Plants, Appendix A of 10CFR50.

3.1.2 CRITERION CONFORMANCE

3.1.2.1 Overall Requirements (Group I)

3.1.2.1.1 Quality Standards and Records (Criterion 1)

Criterion

Structures, systems, and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency, and shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety function. A quality assurance program shall be established and implemented in order to provide adequate assurance that these structures, systems, and components will satisfactorily perform their 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.

Design Conformance

Structures, systems, and components important to safety are listed in Table 3.2-1.

The construction quality assurance program and operational quality assurance program are described in Appendix D of the PSAR and Chapter 17 of the FSAR, respectively, and are applied to the documents which are maintained to demonstrate that all the requirements of the quality assurance program are being satisfied. The 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 licenses.

The Quality Assurance programs developed by PP&L and its contractors satisfy the requirements of General Design Criterion 1.

For further discussion, see the following sections:

1)	Principal Design Criteria	1.2.1
2)	Plant Description	1.2.2
3)	Classification of Structures, Components, and Systems	3.2

3.1.2.1.2 Design Basis for Protection Against Natural Phenomena (Criterion 2)

Criterion

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

Design Conformance

All safety related structures, systems, and components are protected from or designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, and floods without loss of capability to perform their safety function. The natural phenomena and their magnitude are selected in accordance with their probability of occurrence at the Susquehanna SES site. The designs are based upon the most severe of the natural phenomena recorded for the site, with an appropriate margin to account for uncertainties in the historical data. The natural phenomena postulated in the design are 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, 3.5, 3.7, and 3.8. Combinations of natural phenomena and plant-originated accidents that are considered in the design are identified in Sections 3.8, 3.9, 3.10, and 3.11.

The design bases for protection against natural phenomena are in accordance with General Design Criterion 2.

3.1.2.1.3 Fire Protection (Criterion 3)

Criterion

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

Design Conformance

The plant is designed to minimize the occurrence of fire. Plant arrangement allows for isolation of known fire hazards. Nonflammable materials are used to the greatest extent practical to hinder the creation and subsequent spread of fire. Automatic and manual fire protection systems are provided throughout the plant (refer to the Fire Protection Review Report).

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

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

A fire protection evaluation, including a fire hazards analysis, has been performed on the fire protection program for Susquehanna SES Units 1 and 2. Results of this evaluation may be found in the Fire Protection Review Report.

3.1.2.1.4 Environmental and Dynamic Effects Design Bases (Criterion 4)

Criterion

Structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents. These structures, systems, and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit. However, dynamic effects associated with postulated pipe ruptures in nuclear power units may be excluded from the design basis when analysis reviewed and approved by the commission demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping.

Design Conformance

All safety related structures, systems, and equipment are protected from, or designed to withstand, the effects of and are compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including a LOCA, assuming that non-related 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 outside the plant.

The electrical equipment instrumentation and associated cables of the protection and engineered safety features systems which are located inside the containment are discussed in the sections listed below indicating the design requirements in terms of the time which each must survive the extreme environmental conditions following a loss-of-coolant accident.

Environmental and missile design bases are in accordance with General Design Criterion 4.

For further discussion, see the following sections:

1)	Meteorology	2.3
2)	Hydrology	2.4
3)	Geology and Seismology	2.5
4)	Classification of Structures, Components and Systems	3.2
5)	Wind and Tornado Design Criteria	3.3
6)	Water Level Design Criteria	3.4
7)	Missile Protection Criteria	3.5
8)	Criteria for Protection Against Dynamic Effects Associated with a Postulated Rupture of Piping	3.6
9)	Seismic Design	3.7
10)	Design of Category I Structures	3.8
11)	Mechanical Systems and Components	3.9
12)	Seismic Qualification of Seismic Category I Instrumentation and Electrical Equipment	3.10
13)	Environmental Design of Mechanical and Electrical Equipment	3.11
14)	Integrity of Reactor Coolant Pressure Boundary	5.2
15)	Engineered Safety Features	6.0
16)	Instrumentation and Controls	7.0
17)	Electric Power	8.0

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

Criterion

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

Design Conformance

Although Susquehanna SES Units 1 and 2 share certain structures, systems, and components, sharing them does not significantly impair performance of their safety functions.

The following safety related structures are shared between both units:

Control Structure

Diesel Generator Buildings

ESSW Pumphouse

Spray Pond

Spent Fuel Pools

The safety related structures are designed to remain functional during and following the most severe natural phenomena. Therefore sharing these structures will not impair their ability to perform their safety functions.

Seismic Category I structures which house safety related systems and equipment are discussed in Section 3.8.

The shared systems which are important to safety are discussed below; a more detailed discussion may be found in the referenced Subsections:

a)	Emergency Service Water System (ESWS)	9.2.5
b)	Residual Heat Removal Service Water (RHRSW)	9.2.6
c)	Ultimate Heat Sink (Spray Pond)	3.8.4 & 9.2.7
d)	Diesel Generators	8.3.1.4
e)	Offsite Power Supplies	8.2
f)	Unit 1 AC Distribution System	8.3.1
g)	Residual Heat Removal (Fuel Pool Cooling Mode)	5.4.7.1.1.6

Emergency Service Water System (ESWS)

The ESWS is designed to:

a) Supply cooling water to the RHR pump room unit coolers and the motor bearing oil cooler of each RHR pump during all modes of operation of the RHR system.

b) Supply cooling water to all the aligned diesel generator heat exchangers, except the governor oil coolers, during emergency operation or diesel testing, whenever the diesel generators are required to operate.

c) Supply cooling water to the room coolers for the core spray pumps, the high pressure coolant injection (HPCI) pumps and the reactor core isolation cooling (RCIC) pumps to support operation of these systems.

d) Supply cooling water to the control structure chiller and the Unit 2 emergency switchgear cooling condensing unit during emergency operation.

e) During a seismic event, ESWS can also supply water to the spent fuel pools to makeup for evaporative losses as needed to support the RHR fuel pool cooling mode, should the normal makeup source be unavailable.

f) Supply cooling water to the non-safety related reactor building closed cooling water heat exchanger (RBCCW) and turbine building closed cooling water heat exchanger (TBCCW), within the limitations described in Section 9.2.5 of the FSAR.

The ESW system starts automatically after the diesel generators receive their start initiation signal. The ESW system can also be started manually from either the main control room or from either of the two remote shutdown panels located in Units 1 and 2. The system consists of two loops each of which is designed to supply 100 percent of the ESW cooling requirements to both units and the common emergency diesel generators simultaneously. The system has sufficient redundancy so that a single failure of any active component, assuming a loss of offsite power, cannot impair the capability of the system to perform its safety related functions.

For additional discussion, see Subsection 9.2.5.

Residual Heat Removal Service Water System (RHRSW)

The RHRSW System is designed to supply cooling water to the RHR heat exchangers of both units. The system provides a reliable source of cooling water for all operating modes of the RHR system, including heat removal under post-accident conditions, RHR fuel pool cooling following a seismic event and also to provide water to flood the reactor core or the primary containment after an accident, should it be necessary.

The RHRSW pumps are located in the ESSW pumphouse with the ESW pumps. The ESSW pumphouse and the RHRSW system are designed as Seismic Category 1. Each redundant loop of RHRSW provides cooling to one RHR heat exchanger in each unit. The system is designed so that no single failure will prevent it from achieving its safety function.

The RHRSW is a manually operated system. This system can be operated from the control room, or in the event the control room becomes uninhabitable, from the remote shutdown panel in Unit 1 (Loop B) Reactor Building or Unit 2 (Loop A) Reactor Building.

For additional information, see Subsection 9.2.6

Ultimate Heat Sink (Spray Pond)

The ultimate heat sink provides cooling water to support operation of the ESW and RHRSW systems during system testing, during a normal shutdown and during accident conditions. The ultimate heat sink is capable of providing sufficient cooling water without makeup to the spray pond for at least 30 days to permit simultaneous safe shutdown and cooldown of both reactor units and maintain them in a safe shutdown condition. The spray pond is capable of providing enough cooling water without makeup, for a design basis LOCA in one unit with the simultaneous shutdown of the other unit, for 30 days while assuming a concurrent SSE, single failure and loss of offsite power.

The ultimate heat sink consists of a concrete lined spray pond containing approximately 25 million gallons of water and an ESSW intake structure housing four RHRSW pumps and four ESW pumps which pump the water from the pond through their respective loops and back to the pond through a network of sprays located in the pond. The spray pond is concrete lined and is designed in accordance with seismic category 1 requirements.

For additional information, see Subsections 3.8.4.1 and 9.2.7.

Diesel Generators

Diesel Generators A, B, C and D are housed in a Seismic Category I structure. They are separated from each other by concrete walls which provide missile protection. Additionally, a spare diesel generator (Diesel Generator 'E') is provided which can be manually realigned as a replacement for any one of the other four diesel generators. Thus, any one of the other diesel generators (A, B, C or D) can be removed from service for extended maintenance and the Diesel Generator 'E' can be substituted so that there are four operable diesel generators. Diesel Generator 'E' is housed in its own Seismic Category I structure which also provides missile protection. Loss of one of the four aligned diesel generators will not impair the capability to safely shutdown both units, since this can be done with three diesel generators. For additional discussion, see Subsection 8.3.1.4.

For descriptions of the Diesel Generator Fuel Oil System, Cooling Water System, Air Starting System, Lube Oil System, and the Intake and Exhaust Systems see Subsections 9.5.4, 9.5.5, 9.5.6, 9.5.7, and 9.5.8 respectively.

For missile protection see Subsection 3.5. Separation is discussed in Sections 3.12 and 8.3.

Offsite Power Supplies

The two preferred offsite power supplies are shared by both units. The capacity of each offsite power supply is sufficient to operate the engineered safety features of one unit and safe shutdown loads of the other unit.

For additional discussion, see Section 8.2

Unit 1 AC Distribution System

The Unit 1 AC Distribution System is a shared system between both units, since the common equipment (Emergency Service Water, Standby Gas Treatment System, Control Structure HVAC, etc.) is energized only from the Unit 1 AC Distribution System. There are no Unit 2 specific loads energized from the Unit 1 AC Distribution System. The capacity of the Unit 1 AC Distribution System is sufficient to operate the engineered safety features on one unit and the safe shutdown loads of the other unit.

Residual Heat Removal (Fuel Pool Cooling Mode)

With the Spent Fuel Pools cross-tied, one unit's RHR system can be used to cool stored spent fuel in both spent fuel pools. In the cross-tied configuration, the RHRFPC mode of one unit will draw suction from that unit's skimmer surge tank and return the cooled flow to the bottom of the unit's fuel pool. No direct flow to or from the opposite unit's fuel pool will be accomplished. With the pools cross-tied and RHRFPC in operation on one of the units, adequate cooling of both pools will be achieved. For further discussions see Subsections 5.4.7.1.1.6, 5.4.7.1.4, 9.1.3.1, and 9.1.3.3.

3.1.2.2 Protection by Multiple Fission Product Barriers (Group II)

3.1.2.2.1 Reactor Design (Criterion 10)

Criterion

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

Design Conformance

The reactor core 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. Trip set points are selected on operating experience and by the safety design basis. There is no case in which the trip set points allow the core to exceed the thermal-hydraulic safety limits. Power for the reactor protection system is supplied by two independent high inertia AC power supplies which override short duration disturbances in the power system. Alternate power is available to each reactor protection system 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 the minimum critical power ratio (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 ensure 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:

1)	Principal Design Criteria	1.2.1
2)	Plant Description	1.2.2
3)	Fuel Mechanical Design	4.2
4)	Nuclear Design	4.3
5)	Thermal and Hydraulic Design	4.4
6)	Reactor Recirculation System	5.4.1
7)	Reactor Core Isolation Cooling System	5.4.6
8)	Residual Heat Removal System	5.4.7
9)	Accident Analysis	15.0

3.1.2.2.2 Reactor Inherent Protection (Criterion 11)

Criterion

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

Design Conformance

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, and (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 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 (BWR) has an inherently large moderator-to-Doppler coefficient ratio that permits use of coolant flow rate for load following.

In a BWR, 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 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, and

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 –0.04 (Δ k/k)/(Δ P/P) at the beginning of life and about -0.3(Δ k/k)/(Δ 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:

1)	Principal Design Criteria	1.2.1
2)	Nuclear Design	4.3
3)	Thermal and Hydraulic Design	4.4

3.1.2.2.3 Suppression of Reactor Power Oscillations (Criterion 12)

Criterion

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

Design Conformance

The LaSalle instability event described in NRC Information Notice 88-39 demonstrated that reactor instability events have the potential to violate the MCPR safety limit.

The Oscillation Power Range Monitors (OPRM) provide a detection and suppression function for reactor thermal-hydraulic instabilities as described in 10CFR50 Appendix A, Criteria 10 and 12; BWROG reports NEDO-31960-A, NEDO-31960-A Supplement 1, and NEDO-32465-A; Additional OPRM detection and suppression descriptions are outlined in NEDC-32410P-A and NEDC-32410P-A Supplement 1 . The OPRMs monitor local groups of adjacent LPRMs in "cells" as defined in NEDO-32465-A. The OPRM RPS trip function will scram the reactor when there is a reactor core thermal-hydraulic instability to insure that the MCPR Safety Limit is not violated for anticipated instability events.

3.1.2.2.4 Instrumentation and Control (Criterion 13)

Criterion

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

Design Conformance

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 intermediate range monitor (IRM) monitors neutron flux from the upper portion of the source range monitor (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 average power range monitors also provide post accident neutron flux information.

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

To provide protection against the consequences of accidents involving the release of radioactive materials from the fuel and reactor coolant pressure boundary, the containment and 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 corresponds, 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. Flow integrators and recorders are used to determine the leakage flow pumped from the drain sumps. The unidentified leakage rate as established in Chapter 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 to the reactor protection system and containment and reactor vessel isolation control system 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. These instrumentation and controls meet the requirements of Criterion 13.

For further discussion, see the following sections:

1)	Principal Design Criteria	1.2.1
2)	Reactivity Control System	4.1
3)	Reactor Coolant Pressure Boundary Leakage Detection System	5.2
4)	Main Steamline Isolation Valves	5.4
5)	Containment System	6.2
6)	Reactor Protection System	7.2
7)	Primary Containment and Reactor Vessel Isolation Control System	7.3
8)	Neutron Monitoring System	7.6
9)	Reactor Vessel - Instrumentation and Control	7.5
10)	Process Computer System	7.5
11)	Reactor Manual Control System	7.7
12)	Recirculation Flow Control System	7.7

3.1.2.2.5 Reactor Coolant Pressure Boundary (Criterion 14)

Criterion

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

Design Conformance

All NSSS components within the reactor coolant pressure boundary (RCPB) are classified as Quality Group A or ASME Code Class 1 as applicable in compliance with the codes and standards rule section 50.55a of 10 CFR 50, or as a minimum, are classified Quality Group B if the components meet the exclusion requirements 10 CFR Part 50.55a.

The piping and equipment pressure parts within the RCPB through the outer isolation valve(s) are designed, fabricated, erected, and tested to provide a high degree of integrity throughout the plant lifetime. Section 3.2 classifies systems and components within the RCPB as Quality Group A or B. 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 RCPB, the fracture toughness properties and the operating temperature of ferritic materials are controlled to ensure adequate toughness. Subsection 5.2.3 describes the methods utilized to control toughness properties. Materials are impact tested in accordance with ASME Boiler and Pressure Vessel Code, Section III, where applicable. Where RCPB piping penetrates the primary containment, the fracture toughness temperature requirements of the RCPB materials apply.

Piping and equipment pressure parts of the RCPB 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 RCPB prior to and after its assembly and erection. Leakage testing and surveillance is accomplished as described in Criterion 30 design conformance.

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

For further discussion, see the following sections:

1)	Principal Design Criteria	1.2.1
2)	Design Criteria - Structures, Components, Equipment, and Systems	3.1
3)	Overpressurization Protection	5.2
4)	Reactor Vessel and Appurtenances	5.3
5)	Reactor Recirculation System	5.4
6)	Accident Analysis	15.0
7)	Quality Assurance Program	17.0

3.1.2.2.6 Reactor Coolant System Design (Criterion 15)

Criterion

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

Design Conformance

The reactor coolant system consists of the reactor vessel and appurtenances, the reactor recirculation system, the nuclear system pressure relief system, the main steamlines, the reactor core isolation cooling (RCIC) system, and the residual heat removal (RHR) system. These systems are designed, fabricated, erected, and tested to stringent quality requirements and appropriate codes and standards, which ensure high integrity of the RCPB throughout the plant lifetime. The reactor coolant system is designed and fabricated to meet the requirements of the ASME 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 ensure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including anticipated operational occurrences. As described in Subsection 3.1.2.2.4, instrumentation is provided to monitor essential variables to ensure that they are within prescribed operating limits. If the monitored variables exceed their predetermined settings, the auxiliary, control, and protection systems automatically respond to maintain the variables and systems within allowable design limits.

An example of the integrated protective action scheme, which provides sufficient margin to ensure that the design conditions of the RCPB are not exceeded, is the automatic initiation of the nuclear system pressure relief system upon receipt of an overpressure signal. To accomplish overpressure protection, a number of pressure-operated relief valves are provided to 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 LOCA 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 (ECCS) to supply enough cooling water to adequately cool the core. Similarly, other auxiliary, control, and protection systems provide assurance that the design conditions of the RCPB are not exceeded during any conditions of normal operation, including anticipated operational occurrences.

The application of appropriate codes and standards and high quality requirements to the reactor coolant system and the design features of its associated auxiliary, control, and protection systems, ensure that the requirements of Criterion 15 are satisfied. For further discussion, see the following sections:

1)	Principal Design Criteria	1.2.1
2)	Design Criteria - Structures, Components, Equipment, and Systems	3.1
3)	Overpressurization Protection	5.2
4)	Reactor Coolant Pressure Boundary Leakage Detection System	5.2
5)	Reactor Vessel	5.3
6)	Reactor Recirculation System	5.4
7)	Accident Analysis	15.0

3.1.2.2.7 Containment Design (Criterion 16)

Criterion

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.

Design Conformance

The primary containment system, which includes the drywell and suppression chamber, is designed, fabricated, and erected to accommodate, without failure, the pressures and temperatures resulting from the double-ended rupture or equivalent failure of any coolant pipe within the primary containment. The reactor building encompassing the primary containment provides secondary containment. The two containment systems and their associated safety systems are designed and maintained so that offsite doses, which could result from postulated design basis accidents, remain below the guideline values stated in 10CFR50.67 when calculated by the methods of Regulatory Guide 1.183 (July 2000). (Refer to Section 3.13.1 for Regulatory Guide 1.183 compliance.) Sections 6.2 and 15.1 have detailed information which demonstrates compliance with Criterion 16.

3.1.2.2.8 Electric Power Systems (Criterion 17)

Criterion

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

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

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

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

Design Conformance

Two offsite power transmission systems and four onsite standby diesel generators (A, B, C and D) with their associated battery systems are provided. Either of the two offsite transmission power systems or any three of the four onsite standby diesel generator systems have sufficient capability to operate safety related equipment for cooling the reactor core and maintaining primary containment integrity and other vital functions in the event of a postulated accident in one unit with a safe shutdown of the other unit.

Additionally, a fifth diesel generator 'E' with its associated battery system is provided as a replacement, and has the capability of supplying the emergency loading for any one of the other four diesel generators (A, B, C or D). Diesel generator 'E' must be manually aligned to replace any one of the other four diesel generators in the event of a failure.

The two independent offsite power systems supply electric power to the onsite power distribution system via the 230 kV transmission grid. Each of the offsite power sources is supplied from a transmission line which terminates in switchyards (or Substations) not common to the other transmission line. The two transmission lines are on separate rights-of-way. These two transmission circuits are physically independent and are designed to minimize the possibility of their simultaneous failure under operating and postulated accident and environment conditions.

Each offsite power source can supply all Engineered Safety Feature (ESF) buses through the associated transformers. Power is available to the ESF buses from their preferred offsite power source during normal operation and from the alternate offsite power source if the preferred power is unavailable. Each diesel generator (A, B, C, or D) supplies standby power to one of the four ESF buses in each unit. Loss of both offsite power sources to an ESF bus results in automatic starting and connection of the associated diesel generator (A, B, C, or D) within 10 seconds. Loads are progressively and sequentially added to avoid generator instabilities.

There are four independent AC load groups provided to assure independence and redundancy of equipment function. These meet the safety requirements assuming a single failure since any three of the four load groups have sufficient capacity to supply the minimum loads required to safely shut down the unit. Independent routing of the preferred and alternate offsite power source circuits to the ESF buses are provided to meet the single failure safety requirements.

For each of the four AC load groups there is an independent 125 V battery which furnishes DC control power for the corresponding load group. The four load groups are subgrouped to form two divisions to meet the design basis of one out of two ESF load requirements. For each of the two AC divisions there is an independent 250 V battery that supplies DC load power for the corresponding division.

The reactor protection system is powered from the two independent high inertia AC power supplies which override short duration disturbances in the power system.

The power systems as designed meet the requirements of Criterion 17.

For further discussion, see the following sections:

1)	General Plant Description	1.2
2)	Seismic Qualification Design of Seismic Category I Instrumentation and Electrical Equipment	3.10
3)	Environmental Design of Mechanical and Electrical Equipment	3.11
4)	Offsite Power System	8.2
5)	Onsite A-C Power Systems	8.3
6)	Onsite D-C Power Systems	8.3

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

Criterion

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

Design Conformance

The onsite power systems, consisting of the standby diesel generators with their associated switchgear assemblies and battery systems that supply power to safety related equipment, are designed and arranged for periodic testing of each system independently. During refueling shutdowns, a test is conducted to prove the operability of the automatic starting and load sequencing capability of the standby diesel generators. The testing procedure simulates a loss of bus voltage or a safety injection signal to start each standby diesel generator and connect it to its bus. The normal loading sequence is carried out.

Full load testing of each standby diesel generator can be performed while the plant is at power by manually starting each standby generator and by manual synchronization to the normal power supply.

These tests prove the operability of the electric power systems under conditions as close to design as practical, to assess the continuity of these systems and condition of the components.

Inspection and testing of electric power systems, described in Chapters 8 and 16, conform with Criterion 18.

3.1.2.2.10 Control Room (Criterion 19)

Criterion

A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents. Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident.

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

Design Conformance

A control room is provided and equipped to operate the plant safely under normal and accident conditions. Control room shielding and ventilation are designed to permit operator occupancy of the control room for the duration of a design basis accident (DBA). The Criterion 19 dose limit to an individual in the control room has been revised in accordance with 10CFR50.67 and will not exceed 5 Rem TEDE under all accident conditions.

A remote shutdown panel for each unit is located in each reactor building, with equipment, controls, and instrumentation, provided to bring each reactor to hot standby or a cold shutdown in a safe manner. The remote shutdown panels and adjacent controls are located in areas that are physically isolated from the control room so that any event causing the main control room to become inaccessible would have no effect on the availability of the remote shutdown panels and adjacent controls. Also, equipment, controls, and instrumentation are located throughout the units to provide capability for a subsequent cold shutdown through the use of suitable procedures. The main control room and the remote shutdown panels conform with Criterion 19. Ventilation of the main control room is described in Section 9.4, and habitability of the main control room is described in Section 6.4. Remote shutdown is discussed in Subsection 7.4.1.4.

3.1.2.3 Protection and Reactivity Control Systems (Group III)

3.1.2.3.1 Protection System Functions (Criterion 20)

Criterion

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

Design Conformance

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 RCPB barrier. Fuel damage is prevented by initiation of an automatic reactor shutdown if monitored nuclear system variables exceed pre-established limits during anticipated operational occurrences. 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 steamline 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 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 ECCS are initiated automatically to limit the extent of fuel damage following a LOCA. Other systems automatically isolate the reactor vessel or the primary containment to prevent the release of significant amounts of radioactive materials from the fuel and the RCPB. The controls and instrumentation for the ECCS and the isolation systems are initiated automatically when monitored variables exceed preselected operational limits.

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

For further discussion, see the following sections:

1)	Principal Design Criteria	1.2.1
2)	Reactivity Control Mechanical Design	4.1
3)	Control Rod Drive Housing Supports	4.5
4)	Overpressurization Protection	5.2
5)	Main Steam Line Isolation Valves	5.4
6)	Emergency Core Cooling System	6.3
7)	Reactor Protection System	7.2
8)	Primary Containment and Reactor Vessel Isolation Control System	7.3
9)	Emergency Core Cooling Systems – Instrumentation and Control	7.3
10)	Neutron Monitoring System	7.6
11)	Process Radiation Monitoring System	11.5
12)	Reactor Coolant Pressure Boundary Leakage Detection System - Instrumentation and Controls	7.6
13)	Accident Analysis	15.0

3.1.2.3.2 Protection System Reliability and Testability (Criterion 21)

Criterion

The protection system shall be designed for high functional reliability and in-service testability commensurate with the safety functions to be performed. Redundancy and independence designed into the protection system shall be sufficient to assure that, (1) no single failure results in loss of the protection function, and (2) removal from service of any component or channel does not result in loss of the required minimum redundancy unless the acceptable reliability of operation of the protection system can be otherwise demonstrated. The protection system shall be designed to permit periodic testing of its functioning when the reactor is in operation, including a capability to test channels independently to determine failures and losses of redundancy that may have occurred. see Section 8.2.

Design Conformance

Reactor protection trip 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 ensures that when a scram trip point is exceeded, there is a high scram probability. However, should a scram not occur, other monitored components will scram the reactor if their trip points are exceeded. There is sufficient electrical and physical separation between channels and between trip logics monitoring the same variable to prevent environmental factors, electrical transients, and physical events from impairing the ability of the system to respond correctly.

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

The reactor protection (trip) system initiates an automatic reactor shutdown if the monitored plant variables exceed pre-established limits. Each trip system has two trip channels. An automatic or manual trip in either or both trip channels constitutes a trip system trip. A scram results when both trip systems have tripped. This logic scheme is called a one-out-of-two taken twice arrangement. The reactor protection (trip) system can be tested during reactor operation. Manual scram testing is performed by operating one of the four manual scram controls. Two manual scram controls are associated with each trip system, one in each trip channel. Operating one manual scram control tests one trip channel and 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 indicators and in-core neutron detectors are used to verify control rod movement. Each control rod can be withdrawn one notch and then reinserted to the original position without significantly perturbing the nuclear system 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 steamline 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 one at a time. During refueling operation, valve leakage rates can be determined.

RHR 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 reactor recirculation loops. The low pressure coolant injection (LPCI) mode can be tested after reactor shutdown.

Each active component of the ECCS provided to operate in a design basis accident (DBA) is designed to be operable for test purposes during normal operation of the nuclear system.

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

For further discussion, see the following sections:

1)	Principal Design Criteria	1.2.1
2)	Reactivity Control System	4.1
3)	Main Steamline Isolation Valves	5.4
4)	Residual Heat Removal System	5.4
5)	Containment Systems	6.2
6)	Emergency Core Cooling Systems	6.3
7)	Reactor Protection System	7.2
8)	Engineered Safety Feature Systems	7.3
9)	Accident Analysis	15.0

3.1.2.3.3 Protection System Independence (Criterion 22)

Criterion

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

Design Conformance

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 except beneath the reactor vessel as stated in Section 8.1.6.1 (Regulatory Guide 1.75 (1/75), Part 15). No other wiring is run in these wireways. 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 is 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 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 ensures 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:

1)	Principal Design Criteria	1.2.1
2)	Main Steamline Isolation Valves	5.4
3)	Residual Heat Removal System	5.4
4)	Emergency Core Cooling Systems	6.3
5)	Reactor Protection System	7.2
6)	Engineered Safety Feature Systems	7.3
9)	Accident Analysis	15.0

3.1.2.3.4 Protection System Failure Modes (Criterion 23)

Criterion

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

Design Conformance

The 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.

This criterion does not apply to the Alternate Rod Injection (ARI) System. A failure of a single component can prevent the ARI system from completing its function of initiating control rod injection. Failure of the ARI system or any of its components can not prevent the RPS trip system from performing its safety related function.

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:

1)	Principal Design Criteria	1.2.1
2)	Emergency Core Cooling Systems	6.3
3)	Reactor Protection System	7.2
4)	Engineered Safety Feature Systems	7.3

3.1.2.3.5 Separation of Protection and Control Systems (Criterion 24)

Criterion

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

Design Conformance

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.

The primary containment 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 on process liquid and gas lines that may serve as discharge routes for radioactive materials. Four instrumentation channels are used to prevent an inadvertent scram and isolation as a result of instrumentation malfunctions. The output trip signals from each channel are combined in such a way that two channels must signal high radiation to initiate scram and main steam isolation.

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

For further discussion, see the following sections:

1)	Principal Design Criteria	1.2.1
2)	Emergency Core Cooling System	6.3
3)	Reactor Protection System	7.2
4)	Engineered Safety Feature Systems	7.3

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

Criterion

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

Design Conformance

The reactor protection system provides protection against the onset and consequences of conditions that threaten the integrity of the fuel barrier and the RCPB. 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 protection system shall function.

The reactor manual control system is designed so that no single failure can negate the effectiveness of a reactor scram. The circuitry for the manual control system is 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 ensures that specified acceptable fuel limits are not exceeded for any single malfunction of the reactivity control systems as specified in Criterion 25.

For further discussion, see the following sections:

1)	Principal Design Criteria	1.2.1
2)	Reactivity Control System	4.1
3)	Nuclear Design	4.3
4)	Thermal and Hydraulic Design	4.4
5)	Reactor Protection System	7.2
6)	Reactor Manual Control System	7.7
7)	Accident Analysis	15.0

3.1.2.3.7 Reactivity Control System Redundancy and Capability (Criterion 26)

Criterion

Two independent reactivity control systems of different design principles shall be provided. One of the systems shall use control rods, preferably including a positive means for inserting the rods, and shall be capable of 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.

Design Conformance

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 (B_4C) powder only or B_4C and hafnium as neutron absorbing material. 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 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. 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 (Gd2O3) 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.I.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:

1)	Principal Design Criteria	1.2.1
2)	Reactivity Control System	4.1
3)	Engineered Safety Feature System	7.3
4)	Standby Liquid Control System - Instrumentation and Control	7.4
5)	Reactor Manual Control System	7.7

3.1.2.3.8 Combined Reactivity Control Systems Capability (Criterion 27)

Criterion

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

Design Conformance

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. Anticipated Transients without scram are discussed in Section 15.8. 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:

1)	Principal Design Criteria	1.2.1
2)	Reactivity Control System	4.1
3)	Nuclear Design	4.3
4)	Thermal and Hydraulic Design	4.4
5)	Reactor Protection System	7.2
6)	Reactor Manual Control System	7.7
7)	Accident Analysis	15.0

3.1.2.3.9 Reactivity Limits (Criterion 28)

Criterion

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

Design Conformance

The control rod system design incorporates appropriate limits on the potential amount and rate of reactivity increase. Control rod withdrawal sequences and patterns are selected to achieve optimum core performance and low individual rod worths. The rod worth minimizer system prevents withdrawal other than by the preselected rod withdrawal pattern. The rod worth minimizer 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 safety feature protects against a high reactivity insertion rate by limiting the control rod velocity to less than or equal to 3.11 fps. Normal rod movement is limited to 6 in. increments and the rod withdrawal rate is limited through the hydraulic valve to 3 in./sec.

The accident analysis (Chapter 15) evaluates the postulated reactivity accidents as well as abnormal operational transients. Analyses are included for rod dropout, steamline 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 results in damage to the RCPB. 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:

1)	Principal Design Criteria	1.2.1
2)	Control Rod Drive Systems	3.9.4
3)	Reactor Core Support Structures and Internals Mechanical Design	4.2
4)	Reactivity Control System	4.1
5)	Nuclear Design	4.3
6)	Control Rod Drive Housing Supports	4.5
7)	Overpressurization Protection	5.2
8)	Reactor Vessel and Appurtenances	5.3
9)	Main Steam Line Flow Restrictor	5.4
10)	Main Steam Line Isolation Valves	5.4
11)	Process Computer System	7.5
12)	Accident Analysis	15.0

3.1.2.3.10 Protection Against Anticipated Operational Occurrences (Criterion 29)

Criterion

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

Design Conformance

The high functional reliability of the protection and reactivity control systems is achieved through the combination of logic arrangement, redundancy, physical and electrical independence, functional separation, fail-safe design, and in-service testability. These design features are discussed in detail in Subsections 3.1.2.3.2, 3.1.2.3.3, 3.1.2.3.4, 3.1.2.3.5 and 3.1.2.3.7.

An extremely high reliability of timely response to anticipated operational occurrences is maintained by a thorough program of in-service 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 steamline isolation valves, RHR pumps, 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 in-service 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:

1)	Principal Design Criteria	1.2.1
2)	Main Steam Line Isolation Valves	5.4
3)	Residual Heat Removal System	5.4
4)	Containment Systems	6.2
5)	Emergency Core Cooling Systems	6.3
6)	Reactor Protection System	7.2
7)	Engineered Safety Feature Systems	7.3
8)	Accident Analysis	15.0

3.1.2.4 Fluid Systems (Group IV)

3.1.2.4.1 Quality of Reactor Coolant Pressure Boundary (Criterion 30)

Criterion

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

Design Conformance

By utilizing conservative design practices and detailed quality control procedures, the pressure retaining components of the RCPB are designed and fabricated to retain their integrity during normal and postulated accident conditions. Accordingly, components that comprise the RCPB are designed, fabricated, erected, and tested in accordance with recognized industry codes and standards listed in Chapter 5. Furthermore, product and process planning is provided as described in Chapter 17 (operation phase) and Appendix D of the PSAR (construction phase) to ensure conformance with the applicable codes and standards, and to retain appropriate documented evidence verifying compliance. Because the subject matter of this criterion deals with aspects of the RCPB, further discussion on this subject is treated in the response to Subsection 3.1.2.2.5.

Means are provided for detecting reactor coolant leakage. The leak detection system consists of sensors and instruments to detect, annunciate, and in some cases, isolate the RCPB from potentially hazardous leaks before predetermined limits are exceeded. Small leaks are detected by temperature and pressure changes, increased frequency of sump pump operation, and by measuring fission product concentration. 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 a loss of feedwater supply, makeup capabilities are provided by the RCIC system. While the RCIC system provides protection from small leaks, the ECCS network provides protection for the complete range of discharges from ruptured pipes. Thus, protection is provided for the full spectrum of possible discharges.

The RCPB and the leak detection system are designed to meet the requirements of Criterion 30.

For further discussion, see the following sections:

1)	Principal Design Criteria	1.2.1
2)	Design Criteria - Structure, Components, Equipment, and Systems	3.1
3)	Overpressurization Protection	5.2
4)	Reactor Coolant Pressure Boundary Leakage Detection System	5.2
5)	Reactor Vessel and Appurtenances	5.3
6)	Reactor Recirculation System	5.4
7)	Reactor Vessel - Instrumentation and Control	7.3
8)	Reactor Coolant Pressure Boundary Leakage Detection System - Instrumentation and Control	7.6
9)	Quality Control System	17.0

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

Criterion

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

Design Conformance

Brittle fracture control of pressure retaining ferritic materials is provided to ensure protection against nonductile fracture. To minimize the possibility of brittle fracture failure of the reactor pressure vessel, the reactor pressure vessel is designed to meet the requirements of ASME Code, Section III, Appendix G, which consider 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 1017 nvt with neutrons 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 upon the reactor vessel wall. This annular volume contains the core shroud, jet pump assemblies, and reactor coolant. Assuming plant operation at rated power and 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 RCPB is designed, maintained, and tested such that adequate assurance is provided that the boundary will behave in a nonbrittle manner throughout the life of the plant. Therefore, the RCPB is in conformance with Criterion 31.

For further discussion, see the following sections:

1)	Design Criteria - Structures, Components, Equipment, and Systems	3.1
2)	Material Considerations	5.2
3)	Reactor Vessel and Appurtenances	5.3

3.1.2.4.3 Inspection of Reactor Coolant Pressure Boundary (Criterion 32)

Criterion

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

Design Conformance

The reactor pressure vessel design and engineering effort includes provisions for in-service inspection. Removable plugs in the reactor shield and/or removable panels in the insulation provide access for examination of the vessel and its appurtenances. Also, removable insulation is provided on the reactor coolant system safety relief valves, recirculation system, and on the main steam and feedwater systems extending out to and including the first isolation valve outside the containment. Inspection of the RCPB is in accordance with the ASME Boiler and Pressure Vessel Code, Section XI. Subsection 5.2.4 defines the in-service inspection plan, access provisions, and areas of restricted access.

The reactor recirculation piping and main steam piping are hydrostatically tested, with the reactor pressure vessel at a test pressure that is in accordance with Section III of the ASME Code.

Vessel material surveillance samples are located within the reactor pressure vessel. The program includes specimens of the base metal, weld metal, and heat affected zone metal.

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

For further discussion, see the following sections:

1)	Design Criteria - Structures, Components, Equipment, and Systems	3.1
2)	Reactor Coolant Pressure Boundary Leakage Detection System	5.2
3)	In-service Inspection	5.2.4
4)	Reactor Vessel and Appurtenances	5.3
5)	Reactor Recirculation System	5.4

3.1.2.4.4 Reactor Coolant Makeup (Criterion 33)

Criterion

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

Design Conformance

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

For further discussion, see the following sections:

1)	Reactor Coolant Pressure Boundary Leakage Detection Systems	5.2
2)	Emergency Core Cooling System	6.3
3)	Reactor Vessel - Instrumentation and Control	7.3
4)	Makeup Demineralizer System	9.2
5)	Condensate Storage and Transfer System	9.2

3.1.2.4.5 Residual Heat Removal (Criterion 34)

Criterion

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

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

Design Conformance

RHR 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.

Major RHR system equipment consists of two heat exchangers and four main system 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.

The RHR system is designed for four modes of operation:

a) Shutdown cooling

b) Suppression pool cooling (also containment spray)

c) Low pressure coolant injection.

d) Fuel Pool Cooling

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. However, in the event of a loss of offsite power, all normal AC power and auxiliary onsite power will be interrupted.

The plant auxiliary buses supplying power to engineered safety features and reactor protection systems and auxiliaries required for safe shutdown are connected by appropriate switching to the four aligned 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.

Four standby diesel generators (A, B, C, and D) and a spare diesel generator (E), which can be manually realigned as a replacement for any one of the other four diesel generators are provided. These diesel generators 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. The standby diesel generator system is highly reliable. Any three aligned diesel generators are adequate to start and carry the essential loads required for a safe and orderly shutdown.

The RHR system is adequate to remove residual heat from the reactor core to ensure fuel and RCPB design limits are not exceeded. Two RHR cooling loops are designed to provide the normal RHR shutdown cooling (SDC) function. When operating in this mode, both of the SDC loops take suction from the reactor vessel via the reactor recirculation system (RRS) Loop "B" suction piping. Either loop is capable of bringing the reactor to a safe shutdown condition. In the event of a loss of the normal SDC suction flow path from the RRS "B" Loop, an alternate SDC function of RHR can be aligned to bring the unit to safe shutdown. Refer to Section 5.4 of the FSAR for additional information.

Use of RHR in the Fuel Pool Cooling mode will not adversely impact the ability of RHR to perform reactor core cooling functions as discussed in Subsections 5.4.7.1.1.6, 5.4.7.2.6c, 9.1.3.1c and 9.1.3.3. Redundant onsite electric power systems are provided. The design of the RHR system, including its power supply, meets the requirements of Criterion 34.

For further discussion, see the following sections:

1)	Residual Heat Removal System	5.4
2)	Emergency Core Cooling Systems	6.3
3)	Emergency Core Cooling Systems - Instrumentation and Control	7.3
4)	Auxiliary Power System	8.3
5)	Standby AC Power Supply and Distribution	8.3
6)	ESW and RHRSW	9.2
7)	Accident Analysis	15.0

3.1.2.4.6 Emergency Core Cooling (Criterion 35)

Criterion

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

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

Design Conformance

The Emergency Core Cooling Systems (ECCS) consist of the following:

a) High Pressure Coolant Injection (HPCI) System

b) Automatic Depressurization System (ADS)

c) Core Spray (CS) System

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

The ECCS are designed to limit fuel cladding temperature over the complete spectrum of design break sizes in the RCPB, including a complete and sudden circumferential rupture of the largest pipe connected to the reactor vessel.

The HPCI system consists of a steam turbine, a constant-flow pump, system piping, valves, controls and instrumentation. The HPCI system is provided to ensure that the reactor core is adequately cooled to prevent excessive fuel clad temperatures for breaks in the nuclear system that do not result in rapid depressurization of the reactor vessel. 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 CS 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.

Each of two Core Spray Systems consists of two centrifugal pumps that can be powered by normal auxiliary power or the standby a-c power system; a spray sparger in the reactor vessel, 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 and low reactor vessel pressure, the core spray system automatically sprays water onto the top of the fuel assemblies in time and at a sufficient flow rate to cool the core and prevent excessive fuel temperature. The LPCI system starts from the same signals which initiate the CS 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 and low reactor vessel pressure, 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 operates to lower the reactor vessel pressure.

Results of the performance of the ECCS for the entire spectrum of line breaks are discussed in Section 6.3. Peak cladding temperatures are below the 2200°F design basis.

Also provided in Section 6.3 is an analysis to show that the ECCS conform to 10CFR50, Appendix K. This analysis shows complete compliance with the final acceptance criteria with the following results:

a) Peak clad temperatures are below the 2200°F NRC acceptability limit,

b) The amount of fuel cladding reacting with steam is below the 1 percent acceptability limit,

c) The clad temperature transient is terminated while core geometry is amenable to cooling, and

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

The redundancy and capability of the onsite electrical power systems for the ECCS are represented in Subsection 3.1.2.4.5.

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

For further discussion, see the following sections:

1)	Residual Heat Removal System	5.4
2)	Suppression Pool	6.2
3)	Emergency Core Cooling Systems	6.3
4)	Emergency Core Cooling Systems - Instrumentation and Control	7.3
5)	Auxiliary Power Systems	8.3
6)	Standby AC Power Supply and Distribution	8.3
7)	ESW and RHRSW Systems	9.2
8)	Accident Analysis	15.0

3.1.2.4.7 Inspection of Emergency Core Cooling System (Criterion 36)

Criterion

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

Design Conformance

The ECCS discussed in Subsection 3.1.2.4.6 include in-service inspection considerations. The spray spargers within the vessel are accessible for inspection during each refueling outage. The primary shield wall and RPV insulation allow access for examination of nozzles. Removable insulation is provided on the ECCS piping out to and including the first isolation valve outside the primary containment. Inspection of the ECCS is in accordance with the intent of Section XI of the ASME Code. Section 5.2 defines the in-service 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 that are part of the RCPB are designed to specifications for in-service inspection, to detect defects that might affect the cooling performance. Particular attention will be given to the reactor nozzles, CS, and feedwater spargers. The design of the reactor vessel and internals for in-service inspection, and the plant testing and inspection program ensures that the requirements of Criterion 36 will be met.

For further discussion, see the following sections:

1)	Reactor Core Support Structures and Internals Mechanical Design	4.2
2)	In-service Inspection Program (RCPB)	5.2
3)	Reactor Vessel and Appurtenances	5.3
4)	Emergency Core Cooling Systems	6.3
5)	In-service Inspection of Class 2 and 3 Components	6.6

3.1.2.4.8 Testing of Emergency Core Cooling System (Criterion 37)

Criterion

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

Design Conformance

The ECCS consists of the HPCI system, ADS, LPCI mode of the RHR system, and CS system. Each of these systems is provided with sufficient test connections and isolation valves to permit appropriate periodic pressure testing to ensure the structural and leaktight integrity of its components.

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

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

All the ECCS will be tested 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 Subsection 3.1.2.4.15. It is concluded that the requirements of Criterion 37 are met.

For further discussion, see the following sections:

1)	In-service Testing of Pumps and Valves	3.9
2)	Overpressurization Protection	5.2
3)	ECCS Inspection and Testing	6.3
4)	ECCS - Instrumentation and Control	7.3
5)	Standby AC Power System	8.3
6)	Technical Specifications	16.0

3.1.2.4.9 Containment Heat Removal (Criterion 38)

Criterion

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

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

Design Conformance

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

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

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

Either or both RHR heat exchangers can be manually activated to remove energy from the containment. The redundancy and capability of the offsite and onsite electrical power systems for the residual heat removal system are presented in Criterion 34 design conformance.

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

For further discussion, see the following sections:

1)	Residual Heat Removal System	5.4
2)	Containment Systems	6.2
3)	Emergency Core Cooling Systems	6.3
4)	Emergency Core Cooling Systems Control and Instrumentation	7.3
5)	Auxiliary Power System	8.3
6)	Standby AC Power Supply and Distribution	8.3
7)	ESW and RHRSW Systems	9.2
8)	Accident Analysis	15.0

3.1.2.4.10 Inspection of Containment Heat Removal System (Criterion 39)

Criterion

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

Design Conformance

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

The pressure suppression pool is designed to permit appropriate periodic inspection. Space is provided for inspection and maintenance.

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:

1)	Residual Heat Removal System	5.4
2)	Containment Systems	6.2
3)	Emergency Core Cooling Systems	6.3
4)	ESW and RHRSW Systems	9.2

3.1.2.4.11 Testing of Containment Heat Removal System (Criterion 40)

Criterion

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

Design Conformance

The containment heat removal function is accomplished by the containment cooling mode of the RHR system.

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 containment cooling mode is not automatically initiated, but operation of the components is periodically verified. The operation of associated cooling water systems is discussed in Subsection 9.2.5 and 9.2.6. It is concluded that the requirements of Criterion 40 are met.

3.1.2.4.12 Containment Atmosphere Cleanup (Criterion 41)

Criterion

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

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

Design Conformance

Fission products, hydrogen, oxygen, and other substances released from the reactor are contained within the primary containment. Leakage from the primary containment during normal plant operation enters the reactor building (secondary containment). This leakage is discharged from the reactor building through the exhaust system during normal operation. Leakage from the primary containment following the LOCA is limited by the Standby Gas Treatment System (SGTS) (Subsection 6.5.1) and the Main Steam Isolation Valve - Leakage Isolated Condenser Treatment Method (Section 6.7) such that the dose guidelines of 10CFR50.67 are not exceeded. Leakage from primary containment which bypasses secondary containment is maintained within the dose analysis limits as discussed in Subsection 6.2.3.2.3. An air recirculation system is provided to cool and mix the drywell atmosphere during normal operation, and mix the drywell air following a LOCA. The containment atmosphere is also inerted during normal plant operation.

The air recirculation system has sufficient redundancy to be able to withstand a single failure and is operable from either onsite or offsite power.

The SGTS system has redundancy and will meet the single failure criteria imposed by Regulatory Guide 1.52, Design, Testing, and Maintenance Criteria for Engineering-Safety-Feature Atmosphere Cleanup system Air Filtration and Adsorption Units of Light-Water-Nuclear Cooled Power Plants, Revision 1 with either onsite or offsite power.

3.1.2.4.13 Inspection of Containment Atmosphere Cleanup Systems (Criterion 42)

Criterion

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

Design Conformance

The SGTS, post accident recombiner, and purge systems are designed to permit appropriate periodic inspection of the important components (Subsections 6.5.1 and 6.2.5, respectively).

3.1.2.4.14 Testing of Containment Atmosphere Cleanup Systems (Criterion 43)

Criterion

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

Design Conformance

The SGTS, post accident recombiner, and purge systems are designed to permit periodic pressure and functional testing of their components (Subsections 6.5.1 and 6.2.5, respectively).

3.1.2.4.15 Cooling Water (Criterion 44)

Criterion

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

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

Design Conformance

The emergency safeguard service water system, which comprises both the Emergency Service Water system and the Residual Heat Removal Service Water system, provides cooling water for the removal of excess heat from structures, systems, and components which are necessary to maintain safety during all abnormal and accident conditions. These include the standby diesel generators, the RHR pump motor bearing oil coolers, the core spray pump room unit coolers, RCIC pump room unit coolers, the HPCI pump room unit coolers, the RHR heat exchangers, RHR pump room unit coolers, Unit 2 DX Unit, and the control structure chiller. It also provides water to the RHR pump motor bearing oil coolers and above mentioned room unit coolers during a Seismic Event to support operation of the RHR Fuel Pool Cooling (RHR FPC) mode. Make-up water to the Spent Fuel Pool (SFP) is provided during a seismic event in order to make up for evaporative losses and filling of the SFP in support of RHRFPC. RHRSW provides the cooling water to the RHR heat exchangers for the RHRFPC mode.

The engineered safeguard service water system is designed to Seismic Category I requirements. Redundant safety related components served by the engineered safeguard service water system are supplied through redundant supply headers and returned through redundant discharge or return lines. Electric power for operation of redundant safety related components of this system is supplied from separate independent offsite and redundant onsite standby power sources. No single failure renders these systems incapable of performing their safety functions.

Referenced Subsections are as follows:

1)	AC Power Systems	8.3.1
2)	Emergency Service Water System	9.2.5
3)	RHR Service Water System	9.2.6
4)	Ultimate Heat Sink	9.2.7

3.1.2.4.16 Inspection of Cooling Water System (Criterion 45)

Criterion

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

Design Conformance

The engineered safeguard service water systems (ESW and RHRSW Systems) are designed to permit appropriate periodic inspection in order to ensure the integrity of system components.

Referenced Subsections are as follows:

1)	Emergency Service Water System	9.2.5
2)	RHR Service Water System	9.2.6

3.1.2.4.17 Testing of Cooling Water System (Criterion 46)

Criterion

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

Design Conformance

The emergency safeguard service water system is in operation during normal shutdown. The system is tested once per month when the diesel generators are tested. These systems are designed to the extent practicable to permit demonstration of operability of the systems as required for operation during a LOCA or a loss of offsite power.

Referenced Subsections are as follows:

1)	Emergency Service Water System	9.2.5
2)	RHR Service Water System	9.2.6

3.1.2.5 Reactor Containment (Group V)

3.1.2.5.1 Containment Design Basis (Criterion 50)

Criterion

The reactor containment structure, including access openings, penetrations, and the containment heat removal system, shall be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any loss-of-coolant accident. This margin shall reflect consideration of (1) the effects of potential energy sources which have not been included in the determination of the peak conditions, such as energy in steam generators and, as required by 10CFR50.44, energy from metal-water and other chemical reactions that may result from degradation, but not total failure, of emergency core cooling functioning, (2) the limited experience and experimental data available for defining accident phenomena and containment responses, and (3) the conservatism of the calculational model and input parameters.

Design Conformance

The primary containment structure, including access openings, penetrations and the containment heat removal system, is designed so that the containment structure and its internal compartments can withstand, without exceeding the design leakage rate, the peak accident pressure and temperature that could occur during any postulated LOCA. Sections 3.8 and 6.2 have detailed information that demonstrates compliance with Criterion 50.

3.1.2.5.2 Fracture Prevention of Containment Pressure Boundary (Criterion 51)

Criterion

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

Design Conformance

The primary containment boundary is designed to the load combination shown in Section 3.8, which covers the operational, testing, and postulated accident conditions. Each condition results in a stress level that is related to its corresponding temperature and is the basis for comparison with the allowable limits.

The ferritic steel used for the primary containment boundary is specified so that the toughness of the material meets the above established conditions. Adequate toughness at 0°F or lower has been verified by drop weight tear testing or by Charpy V-notch testing to demonstrate minimum energy absorption of ASME III, Table N-421. This will ensure nonbrittle behavior and minimize the probability of a rapidly propagating fracture under the above established conditions.

The weld procedure qualification ensures that the toughness of the weld metal and heat affected zones follow the same criteria as for the base metal.
Since the primary containment is located within the reactor building the possibility of brittle fracture of ferritic material under low temperature is considerably reduced.

Additional information on compliance with GDC 51 has been provided in letters from Mr. N. W. Curtis to Mr. A. Schwencer (NRC) dated June 16 and July 16, 1981.

3.1.2.5.3 Capability for Containment Leakage Rate Testing (Criterion 52)

Criterion

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

Design Conformance

The primary containment structure and related equipment, which are subjected to containment test conditions, are designed so that periodic integrated leakage rate testing, as described in Subsection 6.2.6, can be conducted at containment design pressure.

3.1.2.5.4 Provisions for Containment Testing and Inspection (Criterion 53)

Criterion

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

Design Conformance

The primary containment is designed to permit appropriate periodic inspection of all penetrations. The design includes provisions for periodic testing at containment design pressure of the leaktightness of all electrical penetrations, the drywell head and access hatches, as described in Subsection 6.2.6. The process line penetrations are of welded steel construction without expansion bellows, gaskets, or sealing compounds and are an integral part of the construction. They are tested during the containment integrated leak rate tests. Separate leak tests of the process line penetrations are therefore not considered necessary.
The above design provisions, in conjunction with the leakage monitoring system as described in Subsection 6.2.6, allows appropriate surveillance of the leaktight conditions inside the primary containment.

3.1.2.5.5 Piping Systems Penetrating Containment (Criterion 54)

Criterion

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

Design Conformance

Piping systems penetrating the primary containment are provided with isolation valves. The only exception is the penetration for instrument piping associated with the containment pressure monitors. Compliance for these instrument lines is discussed in Subsection 6.2.4.3.5. Provisions, as described in Subsection 6.2.1, are made to permit leakage testing of the isolation valves. Isolation valves are discussed in Sections 7.3 and 6.2.4.

By increased temperature, radiation, and/or drain sump flow, major leaks in the pipes are located. Isolation signals are discussed in Section 7.3.

3.1.2.5.6 Reactor Coolant Pressure Boundary Penetrating Containment (Criterion 55)

Criterion

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

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

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

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

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

Isolation valves outside containment shall be located as close to 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 in-service inspection, protection against more severe natural phenomena, and additional isolation valves and containment, shall include consideration of the population density, use characteristics, and physical characteristics of the site environs.

Design Conformance

The reactor coolant pressure boundary (as defined in 10CFR50, Section 50.2) consists of the reactor pressure vessel, pressure retaining appurtenances attached to the vessel, valves and pipes which extend from the reactor pressure vessel up to and including the outermost containment isolation valve. The lines of the reactor coolant pressure boundary which penetrate the containment have suitable isolation valves capable of isolating the containment thereby precluding any significant release of radioactivity. Similarly for lines which do not penetrate the containment but which form a portion of the reactor coolant pressure boundary, the design ensures that isolation of the reactor coolant pressure boundary can be achieved.

The design of the isolation systems detailed in the sections listed below meets the requirements of Criterion 55.

For further discussion, see the following sections:

1)	Integrity of Reactor Coolant Pressure Boundary	5.2
2)	Containment Isolation Systems	6.2
3)	Instrumentation and Controls	7.0
4)	Accident Analysis	15.0
5)	Technical Specifications	16.0

3.1.2.5.7 Primary Containment Isolation (Criterion 56)

Criterion

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

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

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

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

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

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

Design Conformance

The system-by-system conformance to the requirements of Criterion 56 is presented in Subsection 6.2.4.

3.1.2.5.8 Closed System Isolation Valves (Criterion 57)

Criterion

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

Design Conformance

The system-by-system conformance to the requirements of Criterion 57 is presented in Subsection 6.2.4.

3.1.2.6 Fuel and Radioactivity Control (Group VI)

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

Criterion

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

Design Conformance

In all cases, the design for radioactivity control is (a) on the basis of the requirements of 10CFR20, 10CFR50, and applicable regulations for normal operations and for any transient situation that might reasonably be anticipated to occur and (b) on the basis of 10CFR50.67 dosage level guidelines for potential accidents of exceedingly low probability of occurrences. All releases are expected to be reported consistent with Regulatory Guide 1.21. (Refer to Section 3.13.1 for Regulatory Guide 1.21 compliance.)

The activity level of waste gas effluents is substantially reduced by differential holdup of noble gases from the offgas system in charcoal decay beds and filtration of particulates before release at the plant exhaust duct.

Control of liquid waste effluents is maintained by batch processing of all liquids, sampling before discharge, and controlled rate of release. Liquid effluents are monitored for radioactivity and rate of flow. Radioactive liquid waste system tankage and evaporator capacity are sufficient to handle any expected transient in the processing of liquid waste volume.

Solid wastes are prepared for offsite disposal by approved procedures. Solid wastes are prepared for shipment by placement in shielded and reinforced containers which meet applicable NRC and Department of Transportation requirements (Section 11.5).

The reference sections are:

1)	Liquid Waste System	11.2
2)	Gaseous Waste Systems	11.3
3)	Process and Effluent Radiological Monitoring System	11.5
4)	Solid Waste System	11.4
5)	Accidents Analysis	15.0

3.1.2.6.2 Fuel Storage and Handling and Radioactivity Control (Criterion 61

Criterion

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

Design Conformance

New Fuel Storage

New fuel is placed in dry storage in the new fuel storage vault that is located inside the reactor building. The storage vault within the reactor building provides adequate shielding for radiation protection. Storage racks preclude accidental criticality (see Subsection 3.1.2.6.3). The new fuel storage racks do not require any special inspection and testing for nuclear safety purposes. However, the racks are accessible for periodic inspection.

Spent Fuel Handling and Storage

Irradiated fuel is stored submerged in the spent fuel storage pool located in the reactor building. Fuel pool water is circulated through the fuel pool cooling and cleanup system to maintain fuel pool water temperature, purity, water clarity, and water level. Storage racks preclude accidental criticality (see Subsection 3.1.2.6.3).

Reliable decay heat removal is provided by the fuel pool cooling and cleanup system. The pool water is circulated through the system with suction taken from the pool and is discharged through diffusers at the bottom of the fuel pool. Pool water temperature is maintained below 125°F when removing the Maximum Normal Heat Load (MNHL) from the pool with the service water temperature at its maximum design value. The RHR system with its substantially larger heat removal capacity can be used as a backup for fuel pool cooling when heat loads larger than the capability of the Fuel Pool Cooling System(s) are in the Spent Fuel Pool(s).

RHR also provides reliable decay heat removal to the spent fuel pool(s) in the event the normal fuel pool cooling system is lost due to a seismic event. Operation of RHR Fuel Pool Cooling (RHRFPC) mode will provide Seismic Category I, Class 1E cooling to the spent fuel pool(s) so that boiling of the Spent Fuel Pool(s) does not occur as a result of a seismic event. ESW provides Seismic Category I, Class 1E make-up in support of RHRFPC.

High and low level switches indicate pool water level changes in the main control room. Fission product concentration in the pool water is minimized by use of the filters and demineralizer. This minimizes the release from the pool to the reactor building.

The reactor building ventilation system and the secondary containment are designed to limit the release of radioactive materials to the environs and ensure that offsite doses are less than the limiting values specified in 10CFR50.67 during operation and all accident conditions.

No special tests of the fuel pool cooling and cleanup system are required, because at least one pump and heat exchanger are continuously in operation while fuel is stored in the pool. Duplicate units are operated periodically to handle high heat loads or to replace a unit for servicing. Routine visual inspection of the system components, instrumentation, and trouble alarms are adequate to verify system operability. Testing of the RHRFPC mode is accomplished through routine testing of the pumps and heat exchangers in support of other modes of RHR. The valves supporting the RHRFPC mode are routinely stroked to confirm proper operation of the valves for their RHRFPC mission.

Independent Spent Fuel Storage Facility

An additional on site spent fuel storage facility is provided for storage requirements in excess of the capacity of the Spent Fuel Storage Pools. The Independent Spent Fuel Storage Installation (ISFSI) is designed, constructed, and licensed in accordance with the requirements of 10CFR72. The ISFSI is the Transnuclear West NUHOMS® Dry Storage System as described in Section 11.7. Handling of spent fuel stored at the ISFSI is in the Reactor Building and is designed to preclude criticality and to maintain adequate shielding and cooling for spent fuel.

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 dilution. Liquid wastes are also evaporated and sludge is accumulated for disposal as solid radwaste. Wet solid wastes are solidified and packaged in steel liners and high integrity containers. Dry solid radwastes are compressed and packaged in steel drums. Gaseous radwastes are monitored, processed, recorded, and controlled, and released such that radiation doses to persons outside the controlled area are below those allowed by applicable regulations.

Accessible portions of the spent fuel pool area and radwaste building have sufficient shielding to maintain dose rates within the limits set forth in 10CFR20 and 10CFR50. The radwaste building is designed to preclude accidental release of radioactive materials to the environs above those allowed by the applicable regulations.

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

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

For further discussion, see the following sections:

1)	Residual Heat Removal System	5.4
2)	Containment Systems	6.2
3)	New Fuel Storage	9.1
4)	Spent Fuel Storage	9.1
5)	Fuel Pool Cooling and Cleanup System	9.1
6)	Air Conditioning, Heating, Cooling and Ventilation Systems	9.4
7)	Radioactive Waste Management	11.0
8)	Radiation Protection	12.0
9)	Independent Spent Fuel Storage Installation (ISFSI)	11.7

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

Criterion

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

Design Conformance

Appropriate plant fuel handling and storage facilities are provided to preclude accidental criticality for new and spent fuel. Criticality in the new fuel storage vault is prevented by the geometrically safe configuration of the storage rack. Criticality in the spent fuel pool is prevented by poison cans containing Boral slabs between adjacent fuel assemblies. The new and spent fuel racks are Seismic Category I structures.

The dry storage of spent fuel in a Dry Shielded Canister (DSC) in a Horizontal Storage Module (HSMs) at the Independent Spent Fuel Storage Installation (ISFSI) meets the requirements of 10CFR72.124, i.e., nuclear criticality safety criteria.

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 secondary containment) are designed to prevent an accidental critical array, even if the vault becomes flooded or subjected to seismic loadings. The center to center new fuel assembly spacing limits the effective multiplication factor (k-eff) of the array to less than or equal to 0.95 for dry or fully flooded conditions.

Spent fuel is stored under water in the spent fuel storage pool and is stored dry at the ISFSI. New fuel can be stored in the spent fuel pool in a dry or wet condition. The top loading racks which store spent and new fuel assemblies, are designed and arranged to ensure subcriticality in the storage pool racks. 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.

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 and spent fuel storage, the design of fuel handling systems and the poison control method of the spent fuel storage racks precludes accidental criticality in accordance with Criterion 62.

For further discussion, see the following sections:

1)	Refueling Interlocks	7.6
2)	New Fuel Storage Racks	9.1
3)	Spent Fuel Storage Racks	9.1
4)	Independent Spent Fuel Storage Installation (ISFSI)	11.7

3.1.2.6.4 Monitoring Fuel and Waste Storage (Criterion 63)

Criterion

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

Design Conformance

Appropriate systems have been provided to meet the requirements of this criterion. A malfunction of the fuel pool cooling and cleanup system that could result in loss of residual heat removal capability and excessive radiation levels is alarmed in the main control room. Alarmed conditions include high/low fuel pool level and high fuel pool temperature. The refueling floor ventilation exhaust radiation monitoring system detects abnormal amounts of radioactivity and initiates appropriate action to control the release of radioactive material to the environs.

The dry storage of spent fuel in a Dry Shielded Canister (DSC) in a Horizontal Storage Modules (HSMs) at the Independent Spent Fuel Storage Installation (ISFSI) meets the requirements of 10CFR72.125, i.e., radiological protection criteria and 10CFR72.126, i.e., criteria for spent fuel, high-level radioactive waste and other radioactive waste storage and handling.

Area radiation and sump levels are monitored and alarmed to give indication of conditions that may result in excessive radiation levels in radioactive waste system areas. These systems satisfy the requirements of Criterion 63.
For further discussion, see the following sections:

1)	Fuel Storage and Handling	9.1
2)	Liquid Waste Systems	11.2
3)	Gaseous Waste Systems	11.3
4)	Solid Waste Systems	11.4
5)	Process Radiation Monitoring	11.5
6)	Low Level Radwaste Holding Facility (LLRWHF)	11.6
7)	Independent Spent Fuel Storage Installation (ISFSI)	11.7

3.1.2.6.5 Monitoring Radioactivity Releases (Criterion 64)

Criterion

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

Design Conformance

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

a) Liquid discharge to the discharge pipe

b) Reactor building ventilation

c) Radwaste building ventilation

d) Turbine building ventilation

e) SGTS vent

The drywell atmosphere is continuously monitored during normal and transient operations, using a continuous airborne radioactivity monitoring system (Section 12.3). In the event of an accident, samples of drywell atmosphere are obtained from the drywell air sample vacuum pump line to provide data on existing airborne radioactivity concentrations inside the drywell. The areas contiguous to the secondary containment, such as the turbine building, are monitored by ventilation air sample particulate and gas monitors. Radioactivity levels in the normal plant effluent discharge paths and in the environs are continuously monitored during normal and accident conditions by the various radiation monitoring systems (Sections 12.3 and 11.4) and by the offsite radiological monitoring programs.

The Radioactive Effluent Release Report covering the operation of the unit during the previous year shall be submitted prior to May 1 of each year in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the ODCM and Process Control Program and in conformance with 10 CFR 50.36a and 10 CFR Part 50, Appendix I, Section IV.B.1.

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

1)	Reactor Coolant Pressure Boundary Leakage Detection System	5.2
2)	Containment and Reactor Vessel Isolation Control System	7.3
3)	Radioactive Waste Management	11.0
4)	Airborne Radioactivity Monitoring	12.3

3.2 CLASSIFICATION OF STRUCTURES, COMPONENTS, AND SYSTEMS

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

3.2.1 Seismic Classification

General Design Criterion 2 of Appendix A to 10CFR50 and Appendix A to 10CFR100 require that nuclear power plant structures, systems, and components important to safety be designed to withstand the effects of earthquakes without loss of capability to perform their safety function. NRC Regulatory Guide 1.29 (Rev. 2, 2/76) provides additional guidance and defines Seismic Category I structures, components, and systems as those 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 condition, or

(3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the guideline exposures of 10CFR 50.67.

Plant structures, systems, and components, including their foundations and supports, designed to remain functional in the event of a Safe Shutdown Earthquake are designated as Seismic Category I, as indicated in Table 3.2-1. Class 1E electric equipment is Seismic Category I equipment. Seismic classification of systems instrumentation is discussed in Chapter 7.

All Seismic Category I structures, systems, and components are analyzed under the loading conditions of the SSE and OBE. Since the two earthquakes vary in intensity, the design of Seismic Category I structures, components, equipment, and systems to resist each earthquake and other loads will be based on levels of material stress or load factors, whichever is applicable, and will yield margins of safety appropriate for each earthquake. The margin of safety provided for Safety Class structures, components, equipment, and systems for the SSE will be sufficiently large to assure that their design functions are not jeopardized.

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

The Operating Basis Earthquake as defined in 10 CFR 100, Appendix A, is not incorporated as a part of the seismic classification scheme.

The seismic classification indicated in Table 3.2-1 meets the requirements of NRC Regulatory Guide 1.29 except as otherwise noted in the table. Where only portions of systems are identified as Seismic Category I on this table, the boundaries of the Seismic Category I portions of the system are shown on the piping and instrument diagrams in appropriate sections of this report.

3.2.2 System Quality Group Classifications

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

(1) prevent or mitigate the consequences of accidents and malfunctions originating within the reactor coolant pressure boundary,

(2) permit shutdown of the reactor and maintain it in the safe shutdown conditions, and

(3) contain radioactive material.

A tabulation of quality group classification for each component so defined is shown in Table 3.2-1 under the heading, "Quality Group Classification." Figure 3.2-1 is a diagram which depicts the relative locations of these components along with their quality group classification. Interfaces between components of different classifications are indicated on the system piping and instrumentation diagrams which are found in the pertinent section of the FSAR.

System Quality Group Classifications and design and fabrication requirements as indicated in Tables 3.2-1, 3.2-2, 3.2-3, and 3.2-4 meet the requirements of Regulatory Guide 1.26 (Rev. 3, 2/76) except as noted.

3.2.2.1 Quality Group D (Augmented)

Certain portions of the radwaste system meet the additional requirements of Quality Group D (Augmented) as defined in the NRC Branch Technical Position ETSB 11-1 (Rev. 1), parts B.IV and B.VI. Portions of the radwaste system meeting the requirements of Quality Group D (Augmented) may be determined from notes on the appropriate figures in Chapter 11.

3.2.3 System Safety Classifications

Structures, systems, and components are classified as Safety Class I, Safety Class 2, Safety Class 3, or Other in accordance with the importance to nuclear safety. Equipment is assigned a specific safety class, 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.

The safety classes are defined in this section 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 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.

3.2.3.1 Safety Class 1

3.2.3.1.1 Definition of Safety Class 1

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

3.2.3.2 Safety Class 2

3.2.3.2.1 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:

(1) inserting negative reactivity to shut down the reactor,

(2) preventing rapid insertion of positive reactivity,

(3) maintaining core geometry appropriate to all plant process conditions,

(4) providing emergency core cooling,

(5) providing and maintaining containment,

(6) removing residual heat from the reactor and reactor core, and

(7) storing spent fuel.

Safety Class 2 includes the following:

(1) Reactor protection system and Alternate Rod Injection system.

(2) Those components of the control rod system which are necessary to render the reactor subcritical.

(3) Systems or components which restrict the rate of insertion of positive reactivity.

(4) 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.

(5) 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.

(6) Emergency core cooling systems.

(7) Primary containment.

(8) Reactor building (secondary containment)

(9) Post-accident containment heat removal systems.

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

(11) 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.

(12) Spent fuel storage racks and spent fuel pool.

(13) Electrical and instrument auxiliaries necessary to operation of the above.

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

3.2.3.3 Safety Class 3

3.2.3.3.1 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

(1) 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.

(2) Which provide or support any safety system function. Safety Class 3 includes the following:

   a. Service water systems required for the purpose of:

      1. Removal of decay heat from the reactor

      2. Emergency core cooling

      3. Post-accident heat removal from the suppression pool

4. Providing cooling water needed for the functioning of emergency systems.

b. Fuel supply for the onsite emergency electrical system.

c. Emergency equipment area cooling.

d. Compressed gas or hydraulic systems required to support control or operation of safety systems.

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

3.2.3.4 Other Structures, Systems, and Components

3.2.3.4.1 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."

3.2.3.4.2 Design Requirements for Other Structures, Systems, and Components

The design requirements for equipment classified as "other" are 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.

3.2.4 Quality Assurance

Structures, systems, and components whose safety functions require conformance to the quality assurance requirement of 10CFR50, Appendix B, are summarized in Table 3.2-1 under the heading, "Quality Assurance Requirements." The Operational Quality Assurance Program is described in Chapter 17.

3.2.5 Correlation of Safety Classes with Industry Codes

The design of plant equipment will be 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 are summarized in Table 3.2-5.

TABLE 3.2-1

SSES DESIGN CRITERIA SUMMARY

Principal Components (34*)	FSAR Section	Source Of Supply (1)*	Location (2)*	Quality Group Classification (3)*	Safety Class (4)*	Principal Construction Codes and Standards (5)*	Seismic Category (6)*	Quality Assurance Requirement (7)*	Comments *
Reactor System	4.5		C						
Reactor vessel		GE	C	A	1	III-A	I	Y	
Reactor vessel support skirt		GE	C	NA	1	III-A	I	Y	
Reactor vessel appurtenances, pressure retaining portions		GE	C	A	1	III-A	I	Y	
CRD housing supports		GE	C	NA	2	X	I	Y	
Reactor internal structures, engineered safety features		GE	C	NA	2	X	I	Y	
Reactor internal structures, other		GE	C	N/A	Other	X	N/A	N	
Control rods		GE	C	N/A	2	X	I	Y	
Control rod drives		GE	C	N/A	2	III-2	I	Y	
Core support structure		GE	C	N/A	2	III-1	I	Y	
Power range detector hardware - pressure retaining portions		GE	C	A	1	III-1	I	Y	
Fuel assemblies		AREVA	C	N/A	2	X	I	Y	

TABLE 3.2-1

SSES DESIGN CRITERIA SUMMARY

Principal Components (34*)	FSAR Section	Source Of Supply (1)*	Location (2)*	Quality Group Classification (3)*	Safety Class (4)*	Principal Construction Codes and Standards (5)*	Seismic Category (6)*	Quality Assurance Requirement (7)*	Comments *
Nuclear Boiler System	4.5								
Vessels, level instrumentation condensing chambers		GE	C	A	1	III-1	I	Y	10
Vessels, air accumulators		P	C	C	3	III-3	I	Y	
Air supply check valves, piping downstream of air supply check valve		P	C	C	3	III-3	I	Y	
Piping, relief valve discharge		P	C	C	3	III-3	I	Y	
Piping, main steam, within outermost Isolation valve		GE	C	A	1	III-1	I	Y	
Pipe supports, main steam		P	C	NA	1	III-1	I	Y	
Pipe restraints, main steam		P	C	NA	1	X	I	Y	
Piping, other within outermost isolation valves		P	C	A	1	III-1	I	Y	10
Piping, instrumentation beyond outermost isolation valves		P	R,T	B	2	III-2	Note 20	N	
Safety/relief valves		GE	C	A	1	III-1	I	Y	
Valves, main steam isolation valves		GE	C,R	A	1	III-1	I	Y	
Quenchers and quencher supports		P	C	C	3	III-3	I	Y	
Valves, other, isolation valves within primary containment		P	C	A	1	III-1	I	Y	10
Feedwater piping inside isolation valves		P	C	A	1	III-1	I	Y	
Valves, instrumentation beyond outermost isolation valves		P	C,R,T	B	2	111-2	I	Y	5
Mechanical modules, instrumentation with safety function		GE	C	NA	2	X	I	Y	
Electrical modules with safety function		GE	C	NA	2	IEEE-279/323	I	Y	
Cable, with safety function		P	C	NA	2	IEEE-279/323/383	NA	Y	15

TABLE 3.2-1

SSES DESIGN CRITERIA SUMMARY

Principal Components (34*)	FSAR Section	Source Of Supply (1)*	Location (2)*	Quality Group Classification (3)*	Safety Class (4)*	Principal Construction Codes and Standards (5)*	Seismic Category (6)*	Quality Assurance Requirement (7)*	Comments *
Recirculation System	5								
Piping		GE	C	A	1	III-1	I	Y	10
Piping suspension, recirculation line		GE	C	NA	1	III-1	I	Y	61
Pipe restraints, recirculation line		GE	C	NA	2	X	NA	Y	
Pumps		GE	C	A	1	III-1	I	Y	
Valves		GE	C	A	1	III-1	I	Y	10
Pump Motors		GE	C	NA	2	NEMA/NEC	I	N	
Electrical modules, with safety function		GE/P	C	NA	2	IEEE-279/323	I	Y	
Cable with safety function		P	C,R	NA	2	IEEE-279/323/383	NA	Y	15
Piping		P	T	D	Other	B31.1.0	NA	N	
CRD Hydraulic System	4								
Valves, scram discharge volume lines		P/GE	R	B	2	III-2	I	Y	10
Valves, insert and withdraw lines		P/GE	R	B	2	III-2	I	Y	35
Valves, other		P	R	D	Other	B31.1.0	NA	N	
Piping, scram discharge volume lines		P	R,C	B	2	III-2	I	Y	
Piping, insert and withdraw lines		P	C,R	B	2	III-2	I	Y	
Piping, other		P	R	D	Other	B31.1.0	NA	N	50
Hydraulic control unit		GE	R	NA	2	NA	I	Y	12
Electrical modules, with safety function		GE	R	NA	2	IEEE-279/323	I	Y	
Cable, with safety function		P	C,R	NA	2	IEEE-279/323/383	NA	Y	15
ENGINEERED SAFETY FEATURES									
Standby Liquid Control System	9.3.5								
Standby liquid control tank		GE	R	B	2	API 650	I	Y	66
Pump		GE	R	B	2	NP&V-II	I	Y	
Pump motor		GE	R	NA	2	X	I	Y	
Valves, explosive		GE	R	B	2	NP&V-II	I	Y	
Valves, isolation and within		P	C,R	A	1	III-1	I	Y	10
Valves, beyond isolation valves		P	R	B	2	III-2	I	Y	10
Piping, within isolation valves		P	C	A	1	III-1	I	Y	10
Piping, beyond isolation valves		P	R	B	2	III-2	I	Y	10
Electrical modules, with safety function		GE	R	NA	2	IEEE-279/323	I	Y	
Cable, with safety function		P	R	NA	2	IEEE-279/323/383	NA	Y	15

TABLE 3.2-1

SSES DESIGN CRITERIA SUMMARY

Principal Components (34*)	FSAR Section	Source Of Supply (1)*	Location (2)*	Quality Group Classification (3)*	Safety Class (4)*	Principal Construction Codes and Standards (5)*	Seismic Category (6)*	Quality Assurance Requirement (7)*	Comments *
RHR System	5.4.7								
Heat exchangers, primary side		GE	R	B	2	III-2	I	Y	
Heat exchangers, secondary side		GE	R	C	3	III-3	I	Y	
Piping, within outermost containment isolation valves		P	C	A	1	III-1	I	Y	10
Piping, beyond outermost containment isolation valves		P	R	B	2	III-2	I	Y	10
Containment spray line piping within isolation valve		P	C,R	B	2	III-2	I	Y	
Containment spray line piping beyond isolation valve		P	R	B	2	III-2	I	Y	
Pumps		GE	R	B	2	NP&V-II	I	Y	
Pump motors		GE	R	NA	2	NEMA/NEC	I	Y	
Reactor vessel head spray line piping inside second isolation valve		P	C	A	1	III-1	I	Y	
Reactor vessel head spray line piping beyond second isolation valve		P	R	B	2	III-2	I	Y	
Valves, isolation LPCI line		P	C,R	A	1	III-1	I	Y	
Valves, isolation, other		P	C,R	B	2	III-2	I	Y	10
Valves, beyond isolation valves		P	R	B	2	III-2	I	Y	10
Mechanical modules		GE	R	NA	2		I	Y	
Electrical modules, with safety function		GE	R	NA	2	IEEE-279/323	I	Y	
Cable, with safety function		P	C,R	NA	2	IEEE-279/323/383	NA	Y	15
Core Spray	6.3								
Piping, within outermost isolation valves		P	C	A	1	III-1	I	Y	10
Piping, beyond outermost isolation valves		P	R,C	B	2	III-2	I	Y	10
Pumps		GE	R	B	2	NP&V-II	I	Y	
Pump motors		GE	R	NA	2	NEMA/NEC	I	Y	
Valves, containment isolation and within containment		P	C	A	1	III-1	I	Y	10
Valves, beyond outermost containment isolation valves		P	R	B	2	III-2	I	Y	10
Electrical modules with safety function		GE	R	NA	2	IEEE-279/323	I	Y	
Cable, with safety function		P	R	NA	2	IEEE-279/323/383	NA	Y	15

TABLE 3.2-1

SSES DESIGN CRITERIA SUMMARY

Principal Components (34*)	FSAR Section	Source Of Supply (1)*	Location (2)*	Quality Group Classification (3)*	Safety Class (4)*	Principal Construction Codes and Standards (5)*	Seismic Category (6)*	Quality Assurance Requirement (7)*	Comments *
High Pressure Coolant Injection	6.3								
Piping, and valves within outermost containment isolation valve (turbine inlet steam line and instrument lines only)		P	C	A	1	III-1	I	Y	10,28
Piping and valves within outermost containment isolation valves (other than above)		P	C	B	2	III-2	I	Y	10,28
Piping, return test line to condensate storage tank beyond second isolation valve		P	R,O	D	Other	B31.1.0	NA	N	
Piping, beyond outermost containment isolation valve, other		P	R	B	2	III-2	I	Y	10
Pumps		GE	R	B	2	NP&V-II	I	Y	
HPCI turbine		GE	R	NA	2	X	I	Y	11,38
Valves, beyond isolation valves, motor operated		P	R	B	2	III-2	I	Y	10
Valves, other		P	R	B	2	III-2	I	Y	10
Electrical modules, with safety function		GE	R	NA	2	IEEE-279/323	I	Y	
Cable with safety function		P	R	NA	2	IEEE-279/323/383	NA	Y	15

TABLE 3.2-1

SSES DESIGN CRITERIA SUMMARY

Principal Components (34*)	FSAR Section	Source Of Supply (1)*	Location (2)*	Quality Group Classification (3)*	Safety Class (4)*	Principal Construction Codes and Standards (5)*	Seismic Category (6)*	Quality Assurance Requirement (7)*	Comments *
RCIC System	5.4.6								
Piping, and valves within outermost containment							-	-	
isolation valves (turbine inlet steam line and									
instrument lines only)		P	C,R	A		III-1			
Piping and valves within outermost containment					1		I	Y	10,28
isolation valves (other than above)		P	C,R	B		III-2			
Piping, and valves beyond outermost					2		I	Y	10,28
containment isolation valves		P	R	B		III-2			
(except for “other” shown below)					2		I	Y	10
Piping, and valves: Other; return test line to									
condensate storage tank beyond second									
isolation valve; vacuum pump discharge from									
vacuum pump to check valve F028;									
condensate pump discharge to valve for									
F049; all leakoff piping from RCIC governor		P	O,R	D		B31.1.0			
valve; gland exhaust piping from RCIC turbine		GE	R	D	Other	B31.1.0	NA	N	
RCIC barometric condenser					Other		NA	N	
RCIC condensate pump and condenser vacuum		GE	R	D		X			
pump		GE	R	B	Other	NP&V-II	NA	N	
Pumps		GE	R	NA	2	X	I	Y	
RCIC turbine		GE	R	NA	2	IEEE-279/323	I	Y	11,38
Electrical modules, with safety function		P	R,C	NA	2	IEEE-279/323/383	I	Y	
Cable, with safety function					2		NA	Y	15

TABLE 3.2-1

SSES DESIGN CRITERIA SUMMARY

Principal Components (34*)	FSAR Section	Source Of Supply (1)*	Location (2)*	Quality Group Classification (3)*	Safety Class (4)*	Principal Construction Codes and Standards (5)*	Seismic Category (6)*	Quality Assurance Requirement (7)*	Comments *
FUEL STORAGE AND HANDLING **Storage Equipment**	9.1 9.1.1, 9.1.2, 9.1.4								
New fuel storage racks		GE	R	NA	2	AWS D1.1	I	Y	
Spent fuel storage racks (includes storage of control rods, control rod guide tubes, defective fuel storage containers, out-of core sipping containers and channels)		P	R	NA	2	AISI/AA	I	Y	
Control Rod Storage Hangers (includes control rod blades)		P	R	NA	2	AISC	I	Y	
Channel storage racks		GE	R	NA	Other	AWS DI.I	NA	N	
In vessel racks		GE	R	NA	Other	AWS DI.I	I	Y	
Defective fuel storage containers		GE	R	NA	3	AWS DI.I	NA	Y	
Independent Spent Fuel Storage Facility (ISFSI)									
Horizontal Storage Modules		TNW	ISFSI	NA	Other	ACI 349 ACI 318	I	Y	62
Dry Shielded Canisters		TNW	ISFSI	NA	Other	ASME	I	Y	62,63
Fuel Servicing Equipment	9.1.4								
Fuel preparation machine		GE	R	NA	3	X	I	Y	
New fuel inspection stand		GE	R	NA	Other	X	NA	N	
General purpose grapple		GE	R	NA	2	X	I	Y	
Irradiated fuel shipping cask		NA	R	NA	Other	49CFR 173.393, 49CFR 173.396	I	Y	45
Jib cranes		P	R	NA	Other	CMAA 70/B30.10	NA	N	
Railway bay unloading crane		P	R	NA	Other	CMAA 70/B30.10	NA	N	

TABLE 3.2-1

SSES DESIGN CRITERIA SUMMARY

Principal Components (34*)	FSAR Section	Source Of Supply (1)*	Location (2)*	Quality Group Classification (3)*	Safety Class (4)*	Principal Construction Codes and Standards (5)*	Seismic Category (6)*	Quality Assurance Requirement (7)*	Comments *
Reactor Vessel Servicing Equipment	9.1.4								
Main Steam Line Plugs (REM*Light Model)		GEH	R	NA	Other	AA	I	Y	48, 58
Dryer & separator sling [Supplied with Nuclear System]		GE	R	NA	Other	X	NA	Y	
RPV head strongback/carousel		GE	R	NA	Other	X	NA	Y	53
Service platform		GE	R	NA	Other	X	NA	N	73
Control rod grapple		GE	R	NA	Other	X	NA	Y	
Reactor building crane		P	R	NA	Other	CMAA 70/B30.20	I	Y	23
Main Steam Line Plugs (Spring Disk Model) [Wetlift]		NA	R	NA	Other	X	I	Y	57,58
MSL Plugs Restraint Ring [Wetlift]		NA	R	NA	Other	X	I	Y	57
Watertight Hook Box [Wetlift]		NA	R	NA	Other	X	NA	Y	59
Rigid Pole Handling System [Wetlift]		NA	R	NA	Other	X	NA	Y	60
Refuel Floor Auxiliary Platform (RFAP)		GE	R	NA	Other	CMAA-74	NA	Y	23
Jet Pump Plugs		NA	R	NA	Other	X	L	Y	69
360 Degree Refuel Work Platform		GE	R	NA	Other	AISC	NA	Y	23
Refueling Equipment	9.1.4								
Refueling platforms		GE	R	NA	2	X	I	Y	23
Fuel grapples		GE	R	NA	Other	X	NA	N	
Under Reactor Vessel Service Equipment	9.1.4								
Equipment handling platform		GE	C	NA	Other	X	NA	N	
CRD handling equipment		NES	C	NA	Other	X	NA	N	

TABLE 3.2-1

SSES DESIGN CRITERIA SUMMARY

Principal Components (34*)	FSAR Section	Source Of Supply (1)*	Location (2)*	Quality Group Classification (3)*	Safety Class (4)*	Principal Construction Codes and Standards (5)*	Seismic Category (6)*	Quality Assurance Requirement (7)*	Comments *
Fuel Pool Cooling & Cleanup System	9.1.3								
Heat Exchangers		P	R	C	Other	III-3, TEMA C	NA	N	
Pumps		P	R	C	Other	III-3	NA	N	
Skimmer surge tanks		P	R	C	3	III-3	I	Y	
Filter demineralizer vessels		P	R	D	Other	VIII-1	NA	N	19,31
Resin and precoat tanks		P	R	D	Other	API-650	NA	N	
Cooling loop piping and valves downstream of Valve 1-53-001, 2-53-001		P	R	C	Other	III-3	NA	N	46,55
RHR intertie piping and valves		P	R	C	3	III-3	I	Y	
Emergency service water makeup piping and valves		P	R	C	3	III-3	I	Y	
Other piping and valves		P	R	D	Other	B31.1.0	NA	N	19,31,56
Cooling loop piping upstream of Valve 1-53-001, 2-53-001 from skimmer surge tank		P	R	C	3	III-3	I	Y	
RADIOACTIVE WASTE MANAGEMENT	11								
Liquid Waste Management Systems	11.2								
Centrifugal pumps		P	R/RW/T	D	Other	III-3**	NA	N	31,22
Atmospheric tanks		P	RW/T	D	Other	VIII-1/III-3	NA	N	31,22
Filter vessel		P	RW	D	Other	VIII-1	NA	N	31,22
Demineralizer vessel		P	RW	D	Other	III-3**	NA	N	31,22
Evaporator, complete system		P	RW	D	Other	III-3**/MA B	NA	N	31,22
Laundry drain filter		P	RW	D	Other	VIII-1	NA	N	
Liquid and chemical waste piping and valves		P	R/RW/T	D	Other	B31.1	NA	N	31,22
Laundry drain waste and auxiliary piping and valves		P	RW	D	Other	B31.1	NA	N	

**These items were constructed to the ASME Code but are not required to be maintained to this code per NRC Branch Technical Position ETSB 11-1.

TABLE 3.2-1

SSES DESIGN CRITERIA SUMMARY

Principal Components (34*)	FSAR Section	Source Of Supply (1)*	Location (2)*	Quality Group Classification (3)*	Safety Class (4)*	Principal Construction Codes and Standards (5)*	Seismic Category (6)*	Quality Assurance Requirement (7)*	Comments *
Offgas System	11.3								
Heat exchangers		P		D	Other		NA	N	22,31
Recombiner Condenser-Unit 2 & Common			T			VIII-1			50
Recombiner Condenser-Unit 1			T			III/VIII-1			
Recombiner Preheater			T			III-3			
Motive Steam Jet Condenser			T			III-3			
Condensate Cooler			T			VIII-1			
Charcoal Treatment Inlet Precooler			RW			VIII-1			
Chiller			T			VIII-1			
Piping		P	T,RW	D	Other	B31.1.0	NA	N	10,22,31
Valves, flow control		P	T,RW	D	Other	B31.1.0	NA	N	22,31
Valves, other		P	T,RW	D	Other	B31.1.0	NA	N	10,22,31
Motors		P	RW	NA	Other	NEMA-MG1	NA	N	22
HEPA filters		P	RW	D	Other	VIII-1	NA	N	22,31
Pressure vessels		P		D	Other		NA	N	22,31
Recombiner Vessel-Unit 1, 2 & Common			T	D		VIII-3			
Motive Steam Jet Ejector			T			VIII-1			
Charcoal Guard Bed			RW			VIII-1			
Charcoal Adsorber Vessels			RW			VIII-1			

TABLE 3.2-1

SSES DESIGN CRITERIA SUMMARY

Principal Components (34*)	FSAR Section	Source Of Supply (1)*	Location (2)*	Quality Group Classification (3)*	Safety Class (4)*	Principal Construction Codes and Standards (5)*	Seismic Category (6)*	Quality Assurance Requirement (7)*	Comments *
Solid Waste Management System	11.4								
Centrifugal pumps		P	RW	D	Other	III-3	NA	N	19,31,22
Regeneration waste transfer pumps		P	T	D	Other	Manuf. Standard	NA	N	22
Solidification system pumps		P	RW	D	Other	Manuf. Standard	NA	N	22
Filter demineralizer backwash tanks		P	R	D	Other	III-3	NA	N	22
Phase separators		P	RW	D	Other	VIII-1	NA	N	31,22
Regen. Waste surge tanks		P	T	D	Other	VIII-1	NA	N	22
Waste mixing tanks		P	RW	D	Other	VIII-1	NA	N	22
Waste containers, HSA		PL	RW	NA	Other	D1.1,D1.1	NA	N	
Waste containers, LSA		PL	RW	NA	Other	D1.1	NA	N	
Solid radwaste collecting piping and valves		P	R/T/RW	D	Other	B31.1	NA	N	31,22
Solidification system piping and valves		P	RW	D	Other	B31.1	NA	N	22
Aux. piping and valves		P	R/T/RW	D	Other	B31.1	NA	N	
Backwash tank drain lines		P	R/T/RW	D	Other	B31.1	NA	N	22
Reactor Water Cleanup System	5.4.8								
Filter demineralizer vessels		GE	R	C	Other	III-3	NA	N	
Regenerative and nonregenerative heat exchangers		GE	R	C	Other	III-3	NA	N	
Piping and valves within reactor coolant pressure boundary (RCPB)		P	R,C	A	1	III-1	I	Y	26 10
RWCU Recirc Pumps		GE	R	C	Other	III-3	NA	N	
Piping and valves beyond outermost containment isolation valve up to valves F104, F042, F034, F035		P	R	C	Other	III-3	NA	N	
Piping and valves beyond valves F104 and F042 to feedwater system		P	R	B	2	III-2	I	Y	
Piping and valves beyond F034 and F035		P	R	D	Other	B31.1.0	NA	N	
Mechanical modules		GE	R	NA	Other	X	NA	N	

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SSES DESIGN CRITERIA SUMMARY

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WATER SYSTEMS									
RHR Service Water and Spray Pond System	9.2.6								
Cross connect piping to RHR system, within second automatic isolation valve		P	R	B	2	III-2	I	Y	
Piping and valves, chemical treatment makeup water, blowdown		P	O	D	Other	B31.1.0	NA	N	
Piping, other		P	O,R,SW	C	3	III-3	I	Y	
RHR SW Pumps		P	SW	C	3	III-3	I	Y	
Pump motors		P	SW	NA	3	IEEE-323/344	I	Y	
Valves, isolation		P	C,R	B	2	III-2	I	Y	
Valves, other		P	O,R,SW	C	3	III-3	I	Y	
Electrical modules, with safety function		P	O,R,SW	NA	3	IEEE-279/323	I	Y	15
Cable, with safety function		P	O,R,SW	NA	3	IEEE-279/323/383	NA	Y	
Heat exchangers		P	R	C	3	III-3/TEMA C	I	Y	
Piping drain pumps		P	O	NA	Other	NA	NA	N	
Emergency Service Water System	9.2.5								
Piping up to RHR SW system		P,GH	O,G,R, T,CS, EG,SW	C	3	III-3	1	Y	
Piping supports in Diesel Generator 'E' building		GH	EG	C	3	III-3	I	Y	
Pumps		P	SW	C	3	III-3	I	Y	
Pump Motors		P	SW	NA	3	IEEE-323/344	I	Y	
Valves		P,GH	O,G,R, T,CS, SW,EG	C	3	III-3	I	Y	
Electrical modules with safety function		P	O,G,R,T, CS,SW	NA	3	IEE-279/323	I	Y	
Cable, with safety function		P	O,G,R, SW,T,CS	NA	3	IEEE-279/323/383	NA	Y	15
Heat exchangers		P,GH	R,T,G, CS,EG	C	3	III-3	I	Y	52

TABLE 3.2-1

SSES DESIGN CRITERIA SUMMARY

Principal Components (34*)	FSAR Section	Source Of Supply (1)*	Location (2)*	Quality Group Classification (3)*	Safety Class (4)*	Principal Construction Codes and Standards (5)*	Seismic Category (6)*	Quality Assurance Requirement (7)*	Comments *
Reactor Building Closed Cooling Water System	9.2.2								
Piping and valves forming part of containment boundary		P	R,C	B	2	III-2	I	Y	
Piping and valves, other		P	R,C,T	D	Other	B31.1.0	NA	N	
Tanks		P	R	D	Other	VIII-1	NA	N	
Heat exchangers		P	R	D	Other	VIII-1/TEMA C	NA	N	
Pumps		P	R	D	Other	Hyd.I	NA	N	24
Plant Service Water System	9.2.1								
Piping and valves forming part of the SW/ESW Interface		P	R,C	C	3	III-3	I	Y	
Piping and valves, other		P	R,C,T	D	Other	B31.1.0	NA	N	
Heat Exchangers		P	CT	D	Other	VIII-1/TEMA C	NA	N	
Pumps		P	CW	D	Other	Hyd.I	NA	N	24
Turbine Building Closed Cooling Water System	9.2.3								
Piping and valves		P	T	D	Other	B31.1.0	NA	N	
Heat exchangers		P	T	D	Other	VIII-1/TEMA C	NA	N	
Tanks		P	T	D	Other	VIII-1	NA	N	
Pumps		P	T	D	Other	Hyd.I	NA	N	24
Circulating Water System	9.2								
Piping		P	O,T,CW	D	Other	AWWA-C201	NA	N	
Condenser		P	T	D	Other	HEI	NA	N	
Pumps		P	CW	NA	Other	VIII-1/Hyd.I	NA	N	24
Valves		P	T,CW	D	Other	AWWA-C201 &	NA	N	
Cooling tower		P	O	NA	Other	C504	NA	N	
Piping, Non-pipe Class		P	O,T,CW	NA	Other	NONE	NA	N	
Valves, Non-pipe Class		P	T,CW	NA	Other	AWWA	NA	N	
						AWWA			

TABLE 3.2-1

SSES DESIGN CRITERIA SUMMARY

Principal Components (34*)	FSAR Section	Source Of Supply (1)*	Location (2)*	Quality Group Classification (3)*	Safety Class (4)*	Principal Construction Codes and Standards (5)*	Seismic Category (6)*	Quality Assurance Requirement (7)*	Comments *
Diesel Generator 'A-D' Systems	9.5.4,9.5.5, 9.5.6, 9.5.7, 9.5.8								
Diesel Generator		P	G	NA	2	IEEE-387	I	Y	
Heat Exchangers, Jacket Water, Intercoolers and Lube Oil		P	G	C	3	III-3/TEMA C	I	Y	54
Engine Mounted Piping and Valves for Fuel Oil, Lube Oil, Jacket Water, Intake/Exhaust, Starting Air Systems		P	G	NA	Other	X	I	Y	
Filter Housings		P	G	C	3	VIII/B31.1.0	I	Y	
Starting Air System Piping and Valves From Air Receiver Inlet Check Valves to Engine Skid		P	G	C	3	III-3	I	Y	
Other Starting Air Piping and Valves Upstream of the Air Receiver Inlet Check Valves		P	G	D	Other	B31.1.0	NA	N	
Air Dryer Piping and Components		P	G	NA	Other	NA	NA	N	
Air receivers		P	G	C	3	III-3	I	Y	
Air Compressors		P	G	D	Other	NA	NA	N	
Fuel Oil Storage Tanks		P	O	C	3	III-3	I	Y	
Fuel Oil Day Tanks		P	G	C	3	III-3	I	Y	
Fuel Oil System Piping and Valves, Auxiliary Skid and Transfer System (except vent lines and portion of fill lines)		P	G, O	C	3	III-3	I	Y	
Fuel Oil Transfer Pump		P	O	C	3	III-3	I	Y	
Fuel Oil Transfer Pump Motor		P	O	NA	3	IEEE-323/344	I	Y	
Jacket Water System Piping and Valves		P	G	D	Other	X	I	Y	
Jacket Water Heater		P	G	NA	Other	NA	I	Y	
Jacket Water Circulating Pump		P	G	D	Other	Hyd. I	I	Y	24
Air Intake and Exhaust Piping System (except Mufflers, Filers, Manifolds and Expansion Joints)		P	G	C	3	III-3	I	Y	
Lube Oil System Piping and Valves		P	G	D	Other	X	I	Y	
Lube Oil Circulating Pump		P	G	D	Other	Hyd. I	NA	N	24
Dirty Lube Oil Drain Tank		P	G	NA	Other	None	NA	N	
Lube Oil Heater		P	G	NA	Other	None	NA	N	
Electrical Modules with Safety Functions		P	G	NA	3	IEEE-279	I	Y	
Cable with Safety Functions		P	G	NA	3	IEEE-279/323/383	NA	Y	15

TABLE 3.2-1

SSES DESIGN CRITERIA SUMMARY

Principal Components (34*)	FSAR Section	Source Of Supply (1)*	Location (2)*	Quality Group Classification (3)*	Safety Class (4)*	Principal Construction Codes and Standards (5)*	Seismic Category (6)*	Quality Assurance Requirement (7)*	Comments *
Diesel Generator 'E' Systems	9.5.4,-5,-.6,-.7,-.8								
Diesel Generator		GH	EG	N/A	2	DEMA	I	Y	
Diesel Engine intercoolers		GH	EG	C	3	III-3	I	Y	
Engine mounted piping and valves for lube oil, jacket water, fuel oil, intake/exhaust, starting air systems required to perform a safety function		GH	EG	N/A	Other	X	I	Y	
Intake/Exhaust piping and expansion joints		GH	EG	C	3	III-3	I	Y	
Auxiliary skid mounted piping, valves, filters and strainers		GH	EG	C	3	III-3	I	Y	
Jacket water, lube oil, fuel oil motor driven pumps		GH	EG	C	3	III-3	I	Y	
Jacket water and lube oil pump motor		GH	EG	N/A	Other	IEEE-323,-344	I	Y	
Fuel oil pump motor		GH	EG	N/A	3	IEEE-323,-344	I	Y	
Jacket water Stand Pipe		GH	EG	C	3	III-3	I	Y	
Jacket water, lube oil, fuel oil, heat exchangers		GH	EG	C	3	III-3	I	Y	
Jacket water and lube oil heaters		GH	EG	N/A	Other	IEEE-323,-344	I	Y	
Fuel oil transfer system piping and valves (except vent line and portion of fill line)		GH	EG,O	C	3	III-3	I	Y	
Fuel oil transfer pump		GH	EG	C	3	III-3	I	Y	
Fuel oil transfer pump motor		GH	EG	N/A	3	IEEE-323,-344	I	Y	
Fuel oil day tank		GH	EG	C	3	III-3	I	Y	
Fuel oil storage tank		GH	O	C	3	III-3	I	Y	
Fuel oil transfer system strainer		GH	EG	C	3	III-3	I	Y	
Electrical modules with safety function		GH	EG	N/A	3	IEEE-323,-344	I	Y	
Cable, with safety functions		GH	EG,O	N/A	3	IEEE-383	I	Y	
Air receiver skid piping and valves		GH	EG	C	3	III-3	I	Y	
Air receivers		GH	EG	C	3	III-3	I	Y	
Air Compressors		GH	EG	D	Other	NA	NA	N	
Engine mounted equipment required to perform a safety function		GH	EG	N/A	3	DEMA	I	Y	
Engine mounted equipment and valves not required to perform safety function		GH	EG	N/A	Other	DEMA	N/A	N	

TABLE 3.2-1

SSES DESIGN CRITERIA SUMMARY

Principal Components (34*)	FSAR Section	Source Of Supply (1)*	Location (2)*	Quality Group Classification (3)*	Safety Class (4)*	Principal Construction Codes and Standards (5)*	Seismic Category (6)*	Quality Assurance Requirement (7)*	Comments *
HEATING, VENTILATING & AIR CONDITIONING SYSTEMS									
Control Structure	9.4.1								
Control Structure Emergency Outside Air Supply System CSEOASS or CREOASS									
Motors		P	CS	NA	3	IEEE323/344	I	Y	16
Fans		P	CS	NA	3	AMCA	I	Y	16
Prefilters		P	CS	NA	3	UL CLASS I	I	Y	16
Electric Heaters		P	CS	NA	3	UL-1096	I	Y	16
HEPA Filters		P	CS	NA	3	MIL-F-51068C (or ASME AG-1-1997)[71] MIL-F-51079A (or ASME AG-1-1997)[71] UL-586	I	Y	16
Adsorber Units		P	CS	NA	3	AACC CS-8	I	Y	16
Ductwork		P	CS	NA	3	AISI, AAWS	I	Y	16
Dampers		P	CS	NA	3	AMCA	I	Y	16
Control Room & Computer Room HVAC									
Motors		P	CS	NA	3	NEMA MG1 IEEE-344/323	I	Y	
Instrumentation		P	CS	NA		IEEE-279/323	I	Y	
Fans		P	CS	NA	3	AMCA	I	Y	
Prefilters		P	CS	NA	3	UL Class I	I	Y	
HEPA filters		P	CS	NA	3	MIL-F-51068C (or ASME AG-1-1997)[71]	I	Y	
		P	CS	NA	3	MIL-F-51079 (or ASME AG-1-1997)[71]	I	Y	
Adsorber units		P	CS	NA	3	AACC CS-8 ANSI N509-80 Table 5-1	I	Y	
Dampers, isolation		P	CS	NA	3	AMCA	I	Y	
Dampers, flow distribution		P	CS	NA	3	AMCA	I	Y	
Ductwork		P	CS		3	AISI, AWS	I	Y	
Coils, cooling		P	CS	C	3	ARI	I	Y	
Electric heating coils						NEC,NEMA			

TABLE 3.2-1

SSES DESIGN CRITERIA SUMMARY

Principal Components (34*)	FSAR Section	Source Of Supply (1)*	Location (2)*	Quality Group Classification (3)*	Safety Class (4)*	Principal Construction Codes and Standards (5)*	Seismic Category (6)*	Quality Assurance Requirement (7)*	Comments *
Relay Room, Cable Spreading, Battery Room HVAC, and HVAC Equipment Room	9.4.1								
Motors		P	CS	NA	3	NEMA MG1 IEEE-344/323	I	Y	
Fans		P	CS	NA	3	AMCA	I	Y	
Prefilters		P	CS	NA	3	UL Class 1	I	Y	
Coils, heating, electric		P	CS	NA	3	NEC, NEMA	I	Y	
Coils, cooling		P	CS	NA	3	ARI	I	Y	
Dampers		P	CS	NA	3	AMCA	I	Y	
Ductwork		P	CS	NA	3	AISI	I	Y	
Piping & valves		P	CS	C	3	B31.1	I	Y	
Instrumentation		P	CS	NA	Other	IEEE-279/323	I	Y	
SGTS Equipment Room H&V	9.4.1.1.5								
Motors		P	CS	NA	3	NEMA MG1 IEEE-344/323	I	Y	
Fans		P	CS	NA	3	AMCA	I	Y	
Heaters, electric		P	CS	NA	3	NEC 424 NFPA 90A & 90B	I	Y	
Dampers		P	CS	NA	3	AMCA	I	Y	
Ductwork		P	CS	NA	3	AISI,AWS	I	Y	
Instrumentation		P	CS	NA	3	IEEE-279/323			

TABLE 3.2-1

SSES DESIGN CRITERIA SUMMARY

Principal Components (34*)	FSAR Section	Source Of Supply (1)*	Location (2)*	Quality Group Classification (3)*	Safety Class (4)*	Principal Construction Codes and Standards (5)*	Seismic Category (6)*	Quality Assurance Requirement (7)*	Comments *
REACTOR BUILDING									
Reactor Building HVAC (Zone I and Zone II)	9.4								
Includes Steam Tunnel Cooling, U2 Elec. Eq. Room H&V and U1 Remote Shutdown Room Ventilation									
Motors		P	R	NA	Other	NEMA MG 1	NA	N	
Fans		P	R	NA	Other	AMCA	NA	N	
Prefilters		P	R	NA	Other	UL Class I	NA	N	
HEPA Filters						MIL-F-51068C, (or ASME AG-1-1997)[71] MIL-F-51079 (or ASME AG-1-1997)[71]			
Adsorber Units		P	R	NA	Other	AACC CS-8 RDT M-16-1T	NA	N	
Coils, Coiling – Chilled & Service Water		P	R	NA	Other	ARI	NA	N	
Coils, Heating		P	R	NA	Other	NEC, NEMA	NA	Y	
Dampers, Isolation, & Ductwork Connected to RB Recirculation System		P	R	NA	3	AMCA, SMACNA, AISI, AWS	I	Y	
Dampers, Other		P	R	NA	Other	AMCA	NA	N	
Ductwork – Other		P	R	NA	Other	SMACNA, AISI, AWS	NA	N	
Piping Connected to SGTS		P	R	C	3	NFPC	I	Y	
Remainder		P	R	D	Other	B31.1	NA	N	
Also see Plant Chilled Water System									
ECCS and RCIC Pump Rooms	9.4.2								
Motors		P	R	NA	3	IEEE-323/344	I	Y	
Fans		P	R	NA	3	AMCA	I	Y	
Filters		P	R	NA	3	NA	I	Y	
Coils, cooling		P	R	NA	3	ARI	I	Y	
Piping and valves		P	R	C	3	III-3	I	Y	

TABLE 3.2-1

SSES DESIGN CRITERIA SUMMARY

Principal Components (34*)	FSAR Section	Source Of Supply (1)*	Location (2)*	Quality Group Classification (3)*	Safety Class (4)*	Principal Construction Codes and Standards (5)*	Seismic Category (6)*	Quality Assurance Requirement (7)*	Comments *
Emergency SWGR and Load Center Rooms	9.4								
Motors		P	R	NA	3	NEMA MG1	I	Y	
						IEEE-344/323			
Fans		P	R	NA	3	AMCA	I	Y	
Prefilters		P	R	NA	3	UL Class 1	I	Y	
Coils, cooling U1-CSCW, U2-DX & condenser		P	R	C	3	III-3	I	Y	
Coils Cooling – RBCW, Both Units									
		P	R	NA	3	ARI	I	Y	
Dampers		P	R	NA	3	AMCA	I	Y	
Ductwork		P	R	NA	3	AISI,AWS	I	Y	
Piping & Valves,		P	R	C	3	III-3	I	Y	
Unit 1-CSCW, Unit 2-Refrigeration		P	R	NA	Other	IEEE-279/323	I	Y	
Instrumentation									
Also See Plant Chilled Water System									

TABLE 3.2-1

SSES DESIGN CRITERIA SUMMARY

Principal Components (34*)	FSAR Section	Source Of Supply (1)*	Location (2)*	Quality Group Classification (3)*	Safety Class (4)*	Principal Construction Codes and Standards (5)*	Seismic Category (6)*	Quality Assurance Requirement (7)*	Comments *
Refueling Floor HVAC (Zone III)-Both Units	9.4.6								
Motors		P	R	NA	Other	NEMA MG1	NA	N	
Fans		P	R	NA	Other	AMCA	NA	N	
Prefilters		P	R	NA	Other	UL Class 1	NA	N	
HEPA filters		P	R	NA	Other	MIL-F-51079	NA	N	
						(or ASME AG-1-1997)[71]			
		P	R	NA	Other	MIL-F-51068C	NA	N	
						(or ASME AG-1-1997)[71]			
Adsorber units		P	R	NA	Other	RDT M-16-1T	NA	N	
		R	R	NA	3	AACC CS-8	I	Y	
Coils, Cooling (RBCW) & Heating						ARI			
Damper-Isolation and Ductwork Connected to		P	R	NA	Other	AMCR,SMACNA	NA	N	
RB Recirculation System		P	R	NA	Other	AISI,AWS	NA	N	
Ductwork - Other		P	R	NA	Other	SMACNA/AISI	NA	N	
Dampers - Other						AMCA			
Piping & Valves						B31.1			
Also See Plant Chilled Water System									
Drywell Atmosphere Recirculation and Cooling System	9.4.5								
Motors		P	C	NA	Other	IEEE-334/ NEMA MG1	I	Y	65
Fans		P	C	NA	Other	AMCA 210	I	Y	65
Coils, cooling		P	C	NA	Other	ARI	I	Y	65
Ductwork		P	C	NA	Other	AISI,AWS	I	Y	65
Dampers		P	C	NA	3	AMCA	I	Y	65
Piping and valves		P	C	NA	Other	B31.1	NA	N	
Combustible Gas Control System									
Hydrogen recombiners inside containment		P	C	NA	2	IEEE-279	I	Y[74]	
Primary Containment Atmosphere monitoring						IEEE-344			
system (PCAMS)		P	C,R	B,D	2	III-2	I	Y	10, 41
Piping valves forming Containment Penetration						IEEE-344			69
Boundary		P	C,R	B	2	III-2	I	Y	

TABLE 3.2-1

SSES DESIGN CRITERIA SUMMARY

Principal Components (34*)	FSAR Section	Source Of Supply (1)*	Location (2)*	Quality Group Classification (3)*	Safety Class (4)*	Principal Construction Codes and Standards (5)*	Seismic Category (6)*	Quality Assurance Requirement (7)*	Comments *
Standby Gas Treatment & RB Recirculation System									
Motors		P	CS	NA	3	IEEE-323/344	I	Y	16
Fans		P	CS	NA	3	AMCA	I	Y	16
Prefilters		P	CS	NA	3	UL Class 1	I	Y	16
Demisters		P	CS	NA	3	MSAR 71-45	I	Y	16
HEPA filters		P	CS	NA	3	MIL-F-51079 (or ASME AG-1-1997)[71]	I	Y	16
		P	CS	NA	3	MIL-F-51068C (or ASME AG-1-1997)[71]	I	Y	16
Adsorber units		P	CS	NA	3	AACC CS-8	I	Y	16
		P	CS	NA	3	ANSI N509-80 Table 5-1	I	Y	16
Ductwork		P	CS	C	3	AISI,AWS	I	Y	16
Dampers		P	CS	C	3	AMCA	I	Y	16
Piping		P	CS	NA	3	NFPC	I	Y	
Valves		P	CS	NA	3	B31.1	I	Y	
Electric heaters						NEMA & NEC			
Control panels						NEMA, IEEE 323			
Radwaste Building HVAC	9.4.3								
Motors		P	RW	NA	Other	NEMA MG1	NA	N	
Fans		P	RW	NA	Other	AMCA	NA	N	
Prefilters		P	RW	NA	Other	UL Class 1	NA	N	
HEPA filters		P	RW	NA	Other	MIL-F-51079A (or ASME AG-1-1997)[71]	NA	N	
		P	RW	NA	Other	MIL-F-51068C	NA	N	
		P	RW	NA	Other	(or ASME AG-1-1997)[71]	NA	N	
Coils, cooling & heating						ARI & UL			
Adsorber units		P	RW	NA	Other	MIL-C-17605	NA	N	
		P	RW	NA	Other	RDT M-16-1T	NA	N	
Ductwork		P	RW	NA	Other	SMACNA	NA	N	
Dampers						AMCA			
Electric heating coil						NEC			

TABLE 3.2-1

SSES DESIGN CRITERIA SUMMARY

Principal Components (34*)	FSAR Section	Source Of Supply (1)*	Location (2)*	Quality Group Classification (3)*	Safety Class (4)*	Principal Construction Codes and Standards (5)*	Seismic Category (6)*	Quality Assurance Requirement (7)*	Comments *
Diesel Generator Buildings HVAC	9.4.7								
Motors		P,GH	G,EG	NA	3	NEMA MG-1 IEEE344	I	Y	
Fans		P,GH	G,EG	NA	3	AMCA	I	Y	
Ductwork		P,GH	G, EG	NA	3	AISI,AWS	I	Y	
Dampers		P,GH	G,EG	NA	3	AMCA	I	Y	
Turbine Building HVAC	9.4.4								
Motors		P	T	NA	Other	NEMA MG1	NA	N	
Fans		P	T	NA	Other	AMCA	NA	N	
Filters		P	T	NA	Other	NA	NA	N	
Coils, cooling		P	T	NA	Other	ARI	NA	N	
Ductwork		P	T	NA	Other	SMACNA	NA	N	
Dampers		P	T	NA	Other	AMCA	NA	N	
Electric heating coil		P	T	NA	Other	NEC,NEMA	NA	N	
Emergency Service Water Pumphouse Ventilation	9.4.8								
Motors		P	SW	NA	3	NEMA MG1 IEEE344	I	Y	
Fans		P	SW	NA	3	AMCA	I	Y	
Ductwork		P	SW	NA	3	AISI,AWS	I	Y	
Dampers		P	SW	NA	3	AMCA	I	Y	

TABLE 3.2-1

SSES DESIGN CRITERIA SUMMARY

Principal Components (34*)	FSAR Section	Source Of Supply (1)*	Location (2)*	Quality Group Classification (3)*	Safety Class (4)*	Principal Construction Codes and Standards (5)*	Seismic Category (6)*	Quality Assurance Requirement (7)*	Comments *
Administration Building HVAC									
Motors		P	O	NA	Other	NEMA MG1	NA	N	
Fans		P	O	NA	Other	AMCA	NA	N	
Prefilters		P	O	NA	Other	UL Class 1	NA	N	
Dampers		P	O	NA	Other	AMCA	NA	N	
Coils, cooling		P	O	NA	Other	ARI	NA	N	
Coils, heating		P	O	NA	Other	NEC, NEMA	NA	N	
Ductwork		P	O	NA	Other	SMACNA	NA	N	
Main Steam and Power Conversion System	10.3								
Main Steam System									
Main steam piping to turbine stop valves and branch line piping up to and including first valve.		P	R,T	B	2	III-2	NA	N	20
Main Steam piping from and including the turbine stop valve to turbine HP casing and branch line piping up to and including first valve.		P	T	D	Other	B31.1.0	NA	N	9,18,33
Steam piping and valves, other		P	T	D	Other	B31.1.0	NA	N	
Main Condenser Evacuation System	10.4.2								
Piping and components		P	T,RW	D	Other	B31.1.0	NA	N	
Heat exchangers		P	T	D	Other	VIII-1	NA	N	
Air ejectors		P	T	D	Other	B31.1.0	NA	N	

TABLE 3.2-1

SSES DESIGN CRITERIA SUMMARY

Principal Components (34*)	FSAR Section	Source Of Supply (1)*	Location (2)*	Quality Group Classification (3)*	Safety Class (4)*	Principal Construction Codes and Standards (5)*	Seismic Category (6)*	Quality Assurance Requirement (7)*	Comments *
Condensate and Feedwater System	10.4.7								
Reactor feedwater piping and valves, RPV to outermost isolation valve		P	C,R	A	1	III-1/III-2	I	Y	32
Reactor feedwater, piping and valves, other		P	R,T	D	Other	B31.1.0	NA	N	
Steam piping to feedwater pump turbine		P	T	D	Other	B31.1.0	NA	N	
Crossover (low pressure) piping		P	T	D	Other	B31.1.0	NA	N	
Bypass (high pressure) piping, downstream of first isolation valve									
Condensate piping and valves		P	T	D	Other	B31.1.0	NA	N	
Heat exchangers		P	T	D	Other	VIII-1/TEMA C	NA	N	
Pressure Vessels		P	T	D	Other	VIII-I	NA	N	
Pumps, feedwater and condensate		P	T	NA	Other	Hyd.I	NA	N	24
Condensate Cleanup System	10.4.6								
Piping and valves		P	T	D	Other	B31.1.0	NA	N	
Pressure vessels		P	T	D	Other	VIII-1	NA	N	
Condensate Storage and Transfer System	9.2.10								
Tanks		P	O	D	Other	D100	NA	N	
Piping and valves		P	RW,O,T,R	D	Other	B31.1.0	NA	N	
Pumps		P	T	D	Other	Hyd.I	NA	N	24
Turbine Gland Sealing System	10.4.3								
Steam seal evaporator (SSE)		P	T	D	Other	VIII-1	NA	N	
Steam Packing Exhauster		P	T	D	Other	X	NA	N	
Piping and valves		P	T	D	Other	B31.1.0/X	NA	N	
SSE Drain Tank		P	T	D	Other	VIII-1	NA	N	

TABLE 3.2-1

SSES DESIGN CRITERIA SUMMARY

Principal Components (34*)	FSAR Section	Source Of Supply (1)*	Location (2)*	Quality Group Classification (3)*	Safety Class (4)*	Principal Construction Codes and Standards (5)*	Seismic Category (6)*	Quality Assurance Requirement (7)*	Comments *
Auxiliary Steam System	10.4.11								
Auxiliary boilers		P	T	D	Other	I	NA	N	
Piping and valves		P	T	D	Other	B31.1.0	NA	N	
Main Chlorination System	9.2.8								
Pumps		P	CA	D	Other	Hyd.I	NA	N	24
Motors		P	CA	NA	Other	NEMA MG1	NA	N	
Piping and valves		P	CA	D	Other	B31.1.0	NA	N	
Lube Oil System	10.2								
Batch oil tank		P	O	D	Other	VIII-1	NA	N	
Reservoirs		P	T	D	Other	API-620	NA	N	
Pumps		P	T	D	Other	VIII/Hyd.I	NA	N	24
Motors		P	T	NA	Other	NEMA MG1	NA	N	
Conditioners		P	T	NA	Other	NA	NA	N	
Heat Exchangers		P	T	D	Other	VIII/TEMA C	NA	N	
Piping and valves		P	T	D	Other	B31.1.0	NA	N	
Instrumentation and Control Systems									
Reactor Instrumentation	7.2								
Reactor Protection System All portions that must operate to control and safety shut down the reactor to a hot shutdown condition (Electronic modules)		GE	C,R,T	NA	2	IEEE-279	I	Y	
Cable with safety function		P	C,R,T	NA	2	IEEE-279/383	NA	Y	15
Alternate Rod Injection All portions that must operate to control and safely shut down the reactor to a hot shutdown condition (Electronic modules) Cable with safety function		P	R	NA	2	10CFR50.62	NA	Y	47

TABLE 3.2-1

SSES DESIGN CRITERIA SUMMARY

Principal Components (34*)	FSAR Section	Source Of Supply (1)*	Location (2)*	Quality Group Classification (3)*	Safety Class (4)*	Principal Construction Codes and Standards (5)*	Seismic Category (6)*	Quality Assurance Requirement (7)*	Comments *
Neutron Monitoring System									
Guide Tubes, TIP (from Ball/Shear valve assembly through penetration to first connection)		GE	C,R	B	2	III-2	I	Y	
Guide Tubes, TIP (remainder of tube after first connection)		GE	C,R	B	2	III-2	NA	Y	
Valves, isolation, TIP subsystem		GE	C,R	B	2	III-2	I	Y	
Electrical modules, IRM and APRM		GE	C,R	NA	2	IEEE-279	I	Y	
Cable, IRM and APRM, with safety function		P	C,R	NA	2	IEEE-279/383	NA	Y	15
Non-Nuclear Instrumentation									
All portions that input to the reactor protection system		GE	C,R	NA	2	IEEE-279	I	Y	
All portions that input to the engineered safety feature actuation system		P/GE	C,R	NA	2	IIII-279	I	Y	
Engineered Safety Features Actuation System	7.3								
All portions		GE	C,R	NA	2	IEEE-279	I	Y	
Engineered Safety Features Systems (controls and instrumentation required for safety associated with each actuated system)	7.3								
Emergency core cooling system		GE	C,R	NA	2	IEEE-279	I	Y	
Containment isolation system		P	C,R	NA	2	IEEE-279	I	Y	
Containment purge systems (pressure boundary only)		P	C,R	NA	2	IEEE-279	I	Y	
Emergency diesel generator systems		P,GH	G,EG	NA	2	IEEE-279	I	Y	
Main steam line break detection system			C,R,T	NA	2	IEEE-279	I	Y	

TABLE 3.2-1

SSES DESIGN CRITERIA SUMMARY

Principal Components (34*)	FSAR Section	Source Of Supply (1)*	Location (2)*	Quality Group Classification (3)*	Safety Class (4)*	Principal Construction Codes and Standards (5)*	Seismic Category (6)*	Quality Assurance Requirement (7)*	Comments *
Controls and Instrumentation Associated with Safe Shutdown Systems	7.4								
PCAMS		P	C,R	B,D	2	IEEE-279	I	Y	
Instrumentation Associated with Other Systems Required for Safety	7.6								
Spent fuel pooling cooling system		P	R	NA	2	IEEE-279	I	Y	
Fuel handling area ventilation isolation system		P	R	NA	2	IEEE-279	I	Y	
Control room panels		P	CS	NA	2	IEEE-279	I	Y	
Local instrument racks associated with safety related equipment		P	ALL	NA	2	IEEE-279	I	Y	
Instrumentation Associated with Systems Not Required for Safety	7.7								
Seismic Instrumentation		P	ALL	NA	Other	NA	I	Y	
Area radiation monitoring		P	ALL	NA	Other	NA	NA	N	
Leak Detection Instrumentation									
Temperature elements		GE	C,R,T		2	IEEE-323	I	Y	39
Differential temperature switch		GE	C,R		2	IEEE-323	I	Y	39
Differential flow indicator		GE	CS		2	IEEE-323	I	Y	39
Pressure switch		GE	C,R		2	IEEE-323	I	Y	39
Differential pressure indicator switch		GE	CS		2	IEEE-323	I	Y	39
Differential flow summer		GE	CS		2	IEEE-323	I	Y	39
Process Radiation Monitors									
Electrical modules, main steam line and reactor building ventilation monitor		GE	R	NA	2	IEEE-323	I	Y	
Cable, main steam line and reactor building ventilation monitors		P	R	NA	2	IEEE-279/323/383	NA	Y	15

TABLE 3.2-1

SSES DESIGN CRITERIA SUMMARY

Principal Components (34*)	FSAR Section	Source Of Supply (1)*	Location (2)*	Quality Group Classification (3)*	Safety Class (4)*	Principal Construction Codes and Standards (5)*	Seismic Category (6)*	Quality Assurance Requirement (7)*	Comments *
ELECTRIC SYSTEMS	8								
Engineered Safety Features AC Equipment									
4.16 kV switchgear	8.3	P,GH	R,G,EG	NA	2	IEEE-308/323/344	I	Y	
480 V load centers		P,GH	R,EG	NA	2	IEEE-308/323/344	I	Y	
480 V motor control centers		P,GH	R,G,EG	NA	2	IEEE-308/323/344	I	Y	
Engineered Safety Features DC Equipment	8.3								
125 V and 250 V station batteries and racks, battery chargers		P,GH	CS,EG	NA	2	IEEE-308/323/344	I	Y	
125 V switchgear and distribution panels		P,GH	CS,EG	NA	2	IEEE-308/323/344	I	Y	
120 V Vital AC System Equipment	8.3								
Static inverters		P	CS	NA	2	IEEE-308/323/344	NA	Y	
120 V distribution panels		P	CS,R,EG	NA	2	IEEE-308/323/344	I	Y	
Electric Cables for ESF Equipment	8.3								
5 kV power cables		P	ALL	NA	2	IEEE-323/383	NA	Y	15
600 V power cables		P	ALL	NA	2	IEEE-323/383	NA	Y	15
Control and instrumentation cables		P	ALL	NA	2	IEEE-323/383	NA	Y	15

TABLE 3.2-1

SSES DESIGN CRITERIA SUMMARY

Principal Components (34*)	FSAR Section	Source Of Supply (1)*	Location (2)*	Quality Group Classification (3)*	Safety Class (4)*	Principal Construction Codes and Standards (5)*	Seismic Category (6)*	Quality Assurance Requirement (7)*	Comments *
Miscellaneous Electrical	8								
Primary containment building electrical penetration assemblies		P	C	NA	2	IEEE-317/344/383	I	Y	
Conduit supports, safety related		P	ALL	NA	2	IEEE-344	I	Y	15
Tray supports, safety related		P	ALL	NA	2	IEEE-344	I	Y	15
Emergency lighting systems		P	ALL	NA	2	IEEE-344	I**	Y	
Emergency communications systems		P	ALL	NA	Other	NONE	NA	N	
AUXILIARY SYSTEMS									
Compressed Air and Instrument Gas Systems	9.3.1								
Compressors		P,PL	T,R,I,RW	NA	Other	NONE	NA	N	
Pressure Vessels, for safety related equipment		P	C,R	C	3	III-3	I	Y	
Pressure vessels, not for safety related equipment		P,PL	ALL	D	Other	VIII-1	NA	N	
Piping and valves forming part of containment boundary		P	C,R	B	2	III-2	I	Y	
Piping and valves, safety related		P	C,R	C	3	III-3	I	Y	
Piping and valves, other		P	ALL	D	Other	B31.1.0	NA	N	
Nitrogen storage bottles		P	R	NA	Other	DOT	I	N	64
Piping and supports – Diesel Generator 'E' Building		GH	EG	D	Other	B31.1	I	N	49

TABLE 3.2-1

SSES DESIGN CRITERIA SUMMARY

Principal Components (34*)	FSAR Section	Source Of Supply (1)*	Location (2)*	Quality Group Classification (3)*	Safety Class (4)*	Principal Construction Codes and Standards (5)*	Seismic Category (6)*	Quality Assurance Requirement (7)*	Comments *
Sampling Systems	9.3.2								
Sample coolers		P, PL	R,T,RW	D	Other	VIII-1 TEMA C	NA	N	
Piping and valves on III-1 systems		P	C	A	1	III-1	I	Y	10
Piping and valves on III-2 systems		P	R	B	2	III-2	I	Y	10
Piping and valves on III-3 systems		P	R	C	3	III-3	NA	Y	68
Piping and valves, other systems		P	R,T,RW	D	Other	B31.1.0	NA	N	10
Piping and valves, containment penetration, isolation		P	C	A	1	III-1	I	Y	10
Fire Protection System	9.5.1								
Tanks		P	O	D	Other	API-650/D100	NA	N	
Pumps, piping and water system components		P	ALL	NA	Other	NFPA/NEPIA	NA	N	
Gas system components (CO and Halon 1301)		P	CS	NA	Other	NFPA/NEPIA	NA	N	
Fire and smoke detection and alarm system		P	ALL	NA	Other	NFPA/NEPIA	NA	N	
Piping and supports – Diesel Generator 'E' Building		GH	EG	NA	Other	NFPA/NEPIA	I	N	49
General External Hydrogen System									
Vessels		P	T	D	Other	VIII-1	NA	N	
Piping		P	T	D	Other	B31.1.0	NA	N	
Valves		P	T	D	Other	B31.1.0	NA	N	
Nitrogen System									
Vessels		P	O	D	Other	VIII-1	NA	N	
Piping		P	R,O,RW	D	Other	B31.1.0	NA	N	
Valves		P	O	D	Other	B31.1.0	NA	N	

TABLE 3.2-1

SSES DESIGN CRITERIA SUMMARY

Principal Components (34*)	FSAR Section	Source Of Supply (1)*	Location (2)*	Quality Group Classification (3)*	Safety Class (4)*	Principal Construction Codes and Standards (5)*	Seismic Category (6)*	Quality Assurance Requirement (7)*	Comments *
Reactor Building Chilled Water System	9.2.12.2								
Chillers		P	R	D	Other	X/B9.1	NA	N	
Chilled Water Heat Exchangers		P	R	D	Other	VIII/ TEMA C	NA	N	
Pumps		P	R	D	Other	VIII/ Hyd.I	NA	N	24
Piping		P	R	D	Other	B31.1	NA	N	
Valves		P	R	B	2	III-2	I	Y	
Isolation, Chilled Water to Primary Containment									
Remainder		P	R	D	Other	B31.1	NA	N	
Turbine Building Chilled Water System	**9.2.12.3**								
Chillers		P	T	D	Other	X/B9.1	NA	N	
Chilled Water Heat Exchangers		P	T	D	Other	VIII/TEMA C	NA	N	
Pumps		P	T	D	Other	VIII/ Hyd.I	NA	N	24
Piping		P	T	D	Other	B31.1	NA	N	
Valves		P	T	D	Other	B31.1	NA	N	
Radwaste Building Chilled Water System	**9.2.12.4**								
Chillers		P	RW	D	Other	X/B9.1	NA	N	
Chilled Water Heat Exchangers		P	RW	D	Other	VIII/TEMA C	NA	N	
Pumps		P	RW	D	Other	VIII/Hyd.I	NA	N	24
Piping		P	RW	D	Other	B31.1	NA	N	
Valves		P	RW	D	Other	B31.1	NA	N	

* Refer to the general notes at the end of this table.

TABLE 3.2-1

SSES DESIGN CRITERIA SUMMARY

Principal Components (34*)	FSAR Section	Source Of Supply (1)*	Location (2)*	Quality Group Classification (3)*	Safety Class (4)*	Principal Construction Codes and Standards (5)*	Seismic Category (6)*	Quality Assurance Requirement (7)*	Comments *
Control Structure Chilled Water System	**9.2.12.1**								
Centrifugal Water Chillers - (Except Condenser)		P	CS	D	3	VIII	I	Y	
Centrifugal Water Chillers - Condenser		P	CS	C	3	III-3	I	Y	
Heat exchangers		P	CS	D	3	VIII-1/TEMA C	I	Y	
Pumps		P	CS	D	3	VIII-1/L Hyd.I	I	Y	
Motors		P	CS	NA	3	IEEE-323/344	I	Y	
Piping		P	CS	D	3	B31.1	I	Y	
Valves		P	CS	D	3	B31.1	I	Y	
Equipment and Floor Drains	9.3.3								
Piping, radioactive		P	ALL	D	Other	B31.1.0	NA	N	
Piping, nonradioactive		P	ALL	D	Other	B31.1.0	NA	N	
Piping & valves, containment penetrating isolation		P	R,C	B	2	III-2	I	Y	
Piping and supports in Diesel Generator 'E' Building – nonradioactive		GH	EG	NA	Other	B31.1	I	N	49
Demineralized Water Makeup System	9.2.9								
Tanks		P	CW	D	Other	VIII-1	NA	N	
Pumps		P	CW	D	Other	B31.1.0/Hyd.I	NA	N	24
Motors		P	CW	NA	Other	NEMA MG1	NA	N	
Piping and Valves		P	ALL	D	Other	B31.1.0	NA	N	
Piping and supports – Diesel Generator 'E' Building		GH	EG	D	Other	B31.1	I	N	49

TABLE 3.2-1

SSES DESIGN CRITERIA SUMMARY

Principal Components (34*)	FSAR Section	Source Of Supply (1)*	Location (2)*	Quality Group Classification (3)*	Safety Class (4)*	Principal Construction Codes and Standards (5)*	Seismic Category (6)*	Quality Assurance Requirement (7)*	Comments *
Buildings									
Reactor Building		P	R	B	2	ACI/AISC	I	Y	
Pressure resistant doors		P	R	B	2	ASTM/AWS AISC	NA	Y	
Watertight door		P	R	B	2	ASTM/AWS	NA	Y	
R.B. Equipment door		P	R	B	2	ASTM/AWS	NA	Y	
Primary Containment		P	C	B	2	ACI/AISC/III	I	Y	27,30
Access hatches/locks/doors		P	C	B	2	III-MC	I	Y	
Liner plate		P	C	B	2	III-MC	I	Y	
Penetration assemblies		P	C	B	2	III-MC	I	Y	29
Vacuum relief valves		P	C	B	2	III-2	I	Y	
Downcomers		P	C	B	2	III-2	I	Y	44
Downcomer Bracing		P	C	B	2	AISC	I	Y	
Diesel Generator ‘A-D’ Building		P	G	NA	2	ACI/AISC	I	Y	
Control structure		P	CS	NA	2	ACI/AISC	I	Y	
Radwaste and offgas building		P	RW	NA	Other	ACI/AISC	NA	N	22
Turbine building		P	T	NA	Other	ACI/AISC	NA	N	21
Administration Building		P	0	NA	Other	ACI/AISC	NA	N	
Circulating water pump house		P	0	NA	Other	ACI/AISC	NA	N	
ESSW pumphouse		P	0	NA	3	ACI/AISC	I	Y	
Low Level Radwaste Holding Facility		P	0	NA	Other	ACI/AISC /UBC	NA	N	
Diesel Generator ‘E’ Building		GH	EG	NA	2	ACI/AISC	I	Y	

TABLE 3.2-1

SSES DESIGN CRITERIA SUMMARY

Principal Components (34*)	FSAR Section	Source Of Supply (1)*	Location (2)*	Quality Group Classification (3)*	Safety Class (4)*	Principal Construction Codes and Standards (5)*	Seismic Category (6)*	Quality Assurance Requirement (7)*	Comments *
Structures									
Roof Scuppers and Parapet Openings		P	R,CS,G	NA	2	ACI/AISC	NA	Y	
Spray pond & Emergency Spillway		P	O	NA	3	ACI	I	Y	
Condensate storage tank		P	O	D	Other	D100	NA	N	
Spent fuel pool, Rxwell,Dryer-Sep.Pool&Cask Pit		P	R	NA	2	ACI/AISC	I	Y	
Spent fuel pool liner		P	R	NA	2	ACI/AISC	I	Y	
Refueling water storage tank		P	O	D	Other	D100	NA	N	
Pipe Whip Restraints		P	R,C	NA	3	AISC	I	Y	
Missile Barriers for safety related equipment		P	C,R,CS, SW,G	NA	Other	ACI/AISC	I	Y	
Biological shielding within Primary containment, Reactor Building and Control Building		P	C,R,CS	NA	Other	ACI/AISC	I	Y	42
Safety related masonry walls		P	R,G,CS	NA	Other	ACI/UBC	I	Y	
New Fuel Storage Vault		P	R	NA	2	ACI/AISC	I	Y	

TABLE 3.2-1

SSES DESIGN CRITERIA SUMMARY

Principal Components (34*)	FSAR Section	Source Of Supply (1)*	Location (2)*	Quality Group Classification (3)*	Safety Class (4)*	Principal Construction Codes and Standards (5)*	Seismic Category (6)*	Quality Assurance Requirement (7)*	Comments *
Post Accident Monitoring	7.6								
SRV position indication system		P	R	NA	2	344	I	Y	
Noble gas effluent radiological monitor		PL	T	NA	NA	ANSI N13.1	NA	N	
Continuous samples of plant effluents for radioactive iodine & particulates		PL	T	NA	NA	ANSI N13.1	NA	N	
Containment hi-range radiation monitor		P	R	NA	2	323/344	I	Y	
Containment pressure monitor		P	R	NA	2	323/344	I	Y	
Containment Suppression pool water level instr.		P	R	NA	2	323/344	I	Y	
Containment H_2/O_2 monitor system		P	R	NA	2	323/344	I	Y	70
Hydrogen Water Chemistry System	9.5.9								
Tanks		NA	O	NA	NA	VIII	NA	N	67
Gas System Components		NA	O, T	NA	NA	B31.1	NA	N	
Piping		NA	O, T	NA	NA	B31.1	NA	N	
Passive Zinc Injection System	9.5.10								
Vessel		NA	T	NA	NA	VIII-1	NA	N	
Piping and valves		NA	T	NA	NA	B31.1	NA	N	

TABLE 3.2-1

SSES DESIGN CRITERIA SUMMARY

General Notes and Comments

1) GE = General Electric

GEH = General Electric - Hitachi

PL = Pennsylvania Power & Light

P = Bechtel as agents for Pennsylvania Power & Light

GH = Gibbs and Hill (Architect/Engineer) and Dravo Constructors, Inc. as agents for Pennsylvania Power & Light

AREVA= AREVA NP, INC. (for reload fuel) Formerly Framatome ANP, formally SPC)

TNW = Transnucléaire West

NA = Not Applicable, see comments

2) Location

C Part of or within primary containment

R Reactor Building

T Turbine Building

CS Control Structure

RW Radwaste and Offgas Building

G Diesel Generator 'A - D' Building

EG Diesel Generator ‘E' Building

TABLE 3.2-1

SSES DESIGN CRITERIA SUMMARY

I	Intake Structure
A	Administration Building
CW	Circulating Water Pumphouse
SW	Engineering Safeguards Service Water (ESSW) Pumphouse
CA	Chlorine and Acid Storage Building
ISFSI	Independent Spent Fuel Storage Installation
O	Outdoors, Onsite

3) A,B,C,D - Quality group classification as defined in Regulatory Guide 1.26. The equipment shall be constructed in accordance with codes listed in Tables 3.2-2, 3.2-3, and 3.2-4.

NA - Not applicable to quality group classification

4) 1,2,3, other = safety classes defined in ANSI-N212 and Section 3.2.3.

NA - Not applicable to safety classification

5) Where shown this supplements information in Tables 3.2/2, 3.2/3, and 3.2/4. Notations for principle construction codes:

I	ASME Boiler and Pressure Vessel Code, Section I
III	1,2,3, NA, NF, NG, MC = ASME Boiler and Pressure Vessel Code Section III, Class 1,2,3 or MC, or subsection NA, NF or NG
VIII-1	ASME Boiler and Pressure Vessel Code, Section VIII, Div. 1

TABLE 3.2-1

SSES DESIGN CRITERIA SUMMARY

NP&V-II	ASME Nuclear Pressure & Valve Code, Class II
API-650	American Petroleum Institute, Welded Steel Tanks for Oil Storage
API-620	American Petroleum Institute, Recommended Rules for Design and Construction of Large, Welded, Low- Pressure Storage Tanks
B9.1	ANSI B9.1, Safety Code for Mechanical Refrigeration
B31.1.0	ANSI B31.1.0, Code for Pressure Piping
SMACNA	Sheet Metal & Air Conditioning Contractors National Assoc., Inc.
HEI	Heat Exchange Institute
TEMA C	Tubular Exchanger Manufacturers Assoc., Class C
HYD.I	Hydraulic Institute
AISC	American Institute of Steel Construction
AISI	American Iron and Steel Institute, "Specification for the Design of Coldformed Steel Structural Members", 1968, "Design of Light Gage Cold-Formed Stainless Steel Structural Members", 1968
ACI	American Concrete Institute
AMCA	AMCA 210 "Test Codes for Air Moving Devices" AMCA 211 A "AMCA Certified Ratings Program for Air Performance"
AWS D1.1	American Welding Society, Structural Welding Code
AWWA	American Water Works Association

TABLE 3.2-1

SSES DESIGN CRITERIA SUMMARY

CS-8T	American Association for Contamination Control, AACC CS-8T, "Tentative Standard for High-efficiency Gas Phase Adsorber Cells" July, 1972
DEMA	Diesel Engine Manufacturer Association, "Standard Practices for Stationary Diesel and Gas Engines", 1971
D100	American Waterworks Association, AWWA-D100 "Standard for Steel Tanks Standpipes, Reservoirs and Elevated Tanks for Water Storage"
NEC	National Electrical Code
NEMA	National Electrical Manufacturer's Association
NEMA MG1	National Electrical Manufacturers' Association, NEMA-MG-1, 1971 "Motors and Generators"
NEMA SM22	National Electrical Manufacturers' Association, NEMA-SM-22, 1970, "Single Stage Steam Turbine for Mechanical Drive Service"
IEEE-279	IEEE-279, Criteria for Protection Systems for Nuclear Power Generating Stations - 1971.
IEEE-308	IEEE-308, Standard Criteria for Class IE Electric Systems for Nuclear Power Generating Stations 1974
IEEE-317	IEEE-317, Standard for Electrical Penetration Assemblies in Containment Structures for Nuclear Fueled Power Generating Stations - 1972
IEEE-323	IEEE-323, General Guide for Qualifying Class IE Electric Equipment for Nuclear Power Generating Stations - 1974
IEEE-344	IEEE-344, Guide for Seismic Qualification of Class IE Electric Equipment for Nuclear Power Generating Stations - 1971 (1975 version used for the Diesel Generator ‘E' Facility)
IEEE-383	Type Test of Class IE Electrical Cables, Field Splices, and Connections for Nuclear Power Generating Stations-1975

TABLE 3.2-1

SSES DESIGN CRITERIA SUMMARY

IEEE-387	IEEE-387, Criteria for Diesel Generator Units applied as Standby Power Supplies for Nuclear Power Generating Stations - 1972
HSI-306	Health and Safety Information, USAEC, Revised Minimal Specification for the High Efficiency Particulate Air Filter. Issue No. 306
NFPA	National Fire Protection Association
NEPIA	Nuclear Energy Property Insurance Association
ARI	Air Conditioning and Refrigeration Institute
DOT	Department of Transportation – Title 49, Section 178.37, Specification 3AA
D1.1	See AWS-D1.1 above
UBC	Uniform Building Code
NA	None Applicable
x	Manufacturer's Standards
AA	Aluminum Association Standard for Aluminum Structures

6) I - The equipment shall be constructed in accordance with the seismic requirements for the Safe Shutdown Earthquake, as described in Section 3.7.

NA - The seismic requirements for the Safe Shutdown Earthquake are not applicable to the equipment or structure.

7) Y - Requires compliance with the requirements of 10CFR50, Appendix B in accordance with the quality assurance program described in Chapter 17.

N - Not within the scope of 10CFR50, Appendix B.

TABLE 3.2-1

SSES DESIGN CRITERIA SUMMARY

8) This note has been intentionally left blank.

9) The following qualification shall be met with respect to the certification requirements:

1. The manufacturer of the turbine stop valves, turbine control valves, turbine bypass valves, and main steam leads from turbine control valve to HP turbine casing shall use quality control procedures equivalent to those defined in General Electric Publication GEZ/4982A, "General Electric Large Steam Turbine-Generator Quality Control Program".

2. A certification shall be obtained from the manufacturer of these valves and steam leads that the quality control program so defined has been accomplished.

10) 1. Instrument and sampling piping from the point where they connect to the process boundary and through the process shutoff (root) valve(s), isolation valve(s), and excess flow check valve, when provided, will be of the same classification as the system to which they connect.

2. See Figure 3.2-2 for instrument line classifications.

3. Other instrument lines:

a) Those connected to special equipment or Group D system pressure boundaries and utilized to actuate safety systems will be Group C from the system pressure boundary through the process shutoff valve(s) to the sensing instrumentation.

b) Those connected to Group B and Group C systems and not utilized to actuate safety systems will be of Group D classification except for those Group C systems by GE utilizing capillary (filled and sealed) instrument lines.

c) Those connected to Group D systems and not utilized to actuate safety systems will be of Group D classification.

4. For sample lines connected to the Reactor Recirculation System, the sample line shall be Group A through the penetration to the outboard containment isolation valve and Group D from the isolation valve to the shutoff valve outside the sample station.

TABLE 3.2-1

SSES DESIGN CRITERIA SUMMARY

11) The HPCI and RCIC turbines do not fall within the applicable design codes. To ensure that the turbine is fabricated to the standards commensurate with their safety and performance requirements, General Electric has established specific design requirements for this component.

12) The hydraulic control unit (HCU) is a General Electric factory assembled, engineered module of valves, tubing, piping, and stored water which 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, while providing rapid insertion for reactor scram.

Although the hydraulic control 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 Group A, B, C, and D pressure integrity quality levels clearly apply at all levels to the interfaces between the HCU and the connecting conventional piping components (eg, pipe nipples, fittings, simple hand valves, etc.), it is considered that they do not apply to the specialty parts (eg, 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, (4) all work is done per written procedures.

The following examples are typical of the problems associated with codes designed to control field assembled components when applied to the design and production of factory fabricated specialty components:

1. The HCU nitrogen gas bottle is a punch forging which is mechanically joined to the accumulator. It stores the energy required to scram a drive at low vessel pressures. It has been code stamped since its introduction in 1966, although its size exempts it from mandatory stamping. It is constructed of a material listed by ASME B&PV Code Section VIII which was selected for its strength and formability.

2. The scram accumulator is joined to the HCU by a split flange joint chosen for its compact design to facilitate both assembly and maintenance. Both the design and construction conform to ANSI B31.1.0 Power Piping Code. This joint, which requires a design pressure of 1750 psig, has been proof tested to 10,000 psi.

TABLE 3.2-1

SSES DESIGN CRITERIA SUMMARY

3. The accumulator nitrogen shutoff valve is a 6,000 psi cartridge valve whose copper alloy material is listed by ASME B&PV Code Section VIII. The valve was chosen for this service partly because it is qualified by the U.S. Navy for submarine service.

4. The directional control valves are solenoid pilot operated valves which are subplate mounted on the HCU. The valve has a body specially designed for the HCU, but the operating parts are identical to a commercial valve with a proven history of satisfactory service. The pressure containing parts are stainless steel alloys chosen for service, fabrication and magnetic properties. The manufacturer cannot substitute a code material for that used for the solenoid core tube.

The foregoing examples are not meant to justify one pressure integrity quality level or another, but to demonstrate the codes and standards invoked by those quality levels are not strictly applicable to special equipment and part designs. Group D Classification is generally applicable, supplemented by the QC techniques described above. Thus, the Hydraulic Control Unit shall be classified as "Special Equipment".

13) This Note Has Been Deleted.

14) This Note Has Been Deleted.

15) The trays and supports for safety related cables meet Seismic Category I and 10CFR50, Appendix B requirements, except in the turbine building. All Class IE and affiliated circuits, including RPS circuits located in a non-Seismic Category I structure (i.e. Turbine Building) are contained within Class IE, Seismic Category I raceways although they are supported from a non-Seismic Category I structure. (See Subsection 3.7b.2.8 for seismic information about the turbine building).

16) AEC Regulatory Guide 1.52, June 1973, suggests various industry standards and codes for this equipment. These references were used for system design, with exceptions as noted in section.

17) AMCA Publication 211A, "AMCA Certified Ratings Program for Air Performance" or AMCA Standard 210, "Test Codes for Air Moving Devices" can be used for blower design purposes.

18) This section of steam piping was seismically analyzed to ensure that it will not fail under loadings normally associated with an SSE.

TABLE 3.2-1

SSES DESIGN CRITERIA SUMMARY

19) All or part of this component is constructed to a more stringent code or standard than indicated.

20) The MSS from its outer isolation valve up to and including the turbine stop valve and all branch lines 2-1/2 in. in diameter and larger, up to and including the first valve (including their restraints) shall be designed by the use of an appropriate dynamic seismic-system analysis to withstand the Operating Bases Earthquake (OBE) and Safe Shutdown Earthquake design loads in combination with other appropriate loads, within the limits specified for Class 2 pipe in the ASME, Section III Code. The mathematical model for the dynamic seismic analyses of the MSS and branch line piping shall include the turbine stop valves and piping beyond the stop valves including the piping to the turbine casing. The dynamic input loads for design of the MSS shall be derived from a time history model analysis (or an equivalent method) of the reactor and applicable portions of the turbine building. An elastic multi-degree-of-freedom system analysis shall be used to determine the input to the MSS. The stress allowable and associated deformation limits for piping shall be in accordance with the ASME Section III Class 2 requirements for the OBE and SSE loading combinations. The MSS supporting structures (those portions of the turbine building) shall be such that the MSS and its supports can maintain their integrity.

21) The power conversion system structures may be constructed in accordance with applicable codes for steam power plants. Those portions of the turbine building interacting with the main steam lines and branch lines are analyzed to show that system integrity is maintained for the main steam lines and branch lines during the SSE.

22) The lower quality group classification, associated construction codes and seismic category are appropriate for this system as a result of analysis per regulatory guides 1.26 and 1.29. The loss of effluent from system components was analyzed to demonstrate that the site boundary dose would not exceed .5 Rem. The classifications indicated in the table are considered justified for the aforementioned doses.

23) These components and associated supporting structures must be designed to retain structural integrity during and after the SSE but do not have to retain operability for protection of public safety. The basic requirement is prevention of structural collapse and damage to equipment and structures required for protection of the public safety and health.

24) There is no established standard for commercial pumps. ASME Section VIII, Division 1 and ANSI B31.1.0 Power Piping represent related, available standards which, while intended for other applications, are used for guidance and recommendations in determining quality group D pump allowable stresses, steel casting quality factors, wall thicknesses, materials compatibility and specifications, temperature pressure environment restrictions, fittings, flanges, gaskets, and bolting, installation procedures, etc.

TABLE 3.2-1

SSES DESIGN CRITERIA SUMMARY

25) This Note Has Been Deleted.

26) The shell side of the nonregenerative heat exchanger was constructed in accordance with ASME Section VIII, Division I. The regenerative and nonregenerative heat exchangers were also constructed to TEMA Class R requirements.

27) The containment spray ring header and connecting piping extending from the containment isolation valve meets all of the requirements of Group B except that hydrostatic testing is not required.

28) The HPCI and RCIC turbine exhaust lines extending from the containment isolating valve to the suppression pool meets all of the requirements of Group B except that hydrostatic testing of this portion of the piping is not required.

29) Piping which penetrates the containment, thus acting as an extension of the containment pressure boundary meets the requirements of Group B or higher. This requirement extends from the first pipe weld on the inside of the penetration to and involving the first isolation valve outside the containment.

30) Reinforced concrete primary containment, including drywell head, hatches, vent pipes, penetrations and spare penetrations are in accordance with Pennsylvania Special Certification. Personnel locks are in accordance with ASME Code Section III, Subsection NE, 1971 Edition, up to and including Addenda of Summer, 1972.

31) Systems and components so designated conform to Quality Group D (Augmented) as defined in NRC Branch Technical Position ETSB 11-1 (Rev. 1) Parts B. IV and B.VI. The Gaseous Radwaste System also conforms to the seismic requirements defined in NRC BTP ETSB-11-1 (Rev. 1) Part B. II. a (3).

32) The feedwater lines from the reactor vessel through the third isolation valve are part of the reactor coolant pressure boundary. The classification of the feedwater line from the reactor vessel through the second isolation valve is Group A. The classification of the feedwater line from the second isolation valve through the third valve is Group B. These classifications are in accordance with Regulatory Guide 1.26 Revision 3, February 1976. Beyond the third valve the classification is Group D.

33) 1. The main steam leads from the turbine control valve to the turbine casing meets all of the requirements of Group D plus the addition of the following requirements:

TABLE 3.2-1

SSES DESIGN CRITERIA SUMMARY

a. All longitudinal and circumferential butt weld joints are 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 are at least equivalent to those specified in ANSI B31.1.0 Power Piping Code.

b. All fillet and socket welds are examined by either magnetic particle or liquid penetrant methods. All structural attachment welds to pressure retaining materials are examined by either magnetic particle or liquid penetrant methods. Examination procedures and acceptance standards are at least equivalent to those specified in ANSI B31.1.0 Power Piping Code.

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

<u>OR</u>

2. The manufacturer of the main leads utilized quality control procedures equivalent to those defined for main steam leads in the General Electric Publication GEZ-4982, "General Electric Large Steam Turbine-Generator Quality Control Program".

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

34) This Note Has Been Deleted.

35) The control rod drive insert and withdraw lines from the drive flange, up to and including the first valve on the hydraulic control unit shall be Safety Class 2.

36) These Notes Have Been Deleted.

37) This Note Has Been Deleted.

38) The turbine does not fall within the applicable design codes. To ensure 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:

TABLE 3.2-1

SSES DESIGN CRITERIA SUMMARY

a. All welding shall be qualified in accordance with Section IX, ASME Boiler and Pressure Vessel Code,

b. All pressure-containing castings and fabrications shall be hydrotested in 1.5 X design pressure,

c. All high-pressure castings shall be radiographed according to:

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

d. As-cast surfaces shall be magnetic particle or liquid penetrant tested according to ASME, Section III, Paragraph N-232.4 or N-323.3,

e. Wheel and shaft forgings shall be ultrasonically tested according to ASTM A-388,

f. Butt-welds shall be radiographed according to ASME, Section III, Paragraph N624, and magnetic particle or liquid penetrant tested according to ASME Section III, Paragraph N626 or N627 respectively,

g. Notification to be made on major repairs, and records maintained thereof, and

h. Record system and traceability according to ASME Boiler and Pressure Code Section III, Appendix IX, Paragraph IX 225.

39) These safety grade instruments provide signals for alarms and/or isolation in the following areas and are collected into this table in one area for ease of identification. Systems: Nuclear Boiler; RHR; RCIC; HPCI; RWCU.

40) This note has been intentionally left blank.

41) Sample piping and isolation valves are quality group B. Because the analyzers are isolated from containment atmosphere on accident conditions, the piping in the analyzers is quality group D. Isolation is manually removed to allow monitoring.

42) Reactor shield wall concrete is a non-structural element (see subsection 3.8.3.1.3) and is therefore non-Category I. Shield wall concrete, because of concrete placement, is non-safety related.

TABLE 3.2-1

SSES DESIGN CRITERIA SUMMARY

43) Code Case 1481-1 has been used because the design temperature of the piping involved is greater than 700°F. ASME Sec. III Appendix Table 1-7.2 only gives allowable stress data up to 700°F. The use of this code case allows stress analysis to be done using stress values in accordance with stress tables of ASME Sec. VIII Division I.

44) ASME Boiler and Pressure Vessel Code, Division 1 Section III Subsection NC has been used for design and fabrication of the downcomers.

45) Shipping casks will not be bought. They will be rented from the shipper.

46) Portions of embedded fuel pool piping are B31.1.

47) Seismically qualified for operating basis earthquakes.

48) The main steam line plugs are supplied by GE-Hitachi and have integral installation tools. The plugs are designed to withstand a design pressure of 60 psig from the steam line side and 16 psig from the vessel side. The plugs also have cable lanyards designed to prevent a dropped plug from reaching the upper core support plate during installation or removal of the plugs. The main steam line plugs are considered as safety-related components and the cable and installation tool are classified as non-quality.

49) All non-safety related piping inside the diesel generator E building has been seismically supported to satisfy Seismic Category 1 requirements in order to eliminate potential safety impact item concerns.

50) Table notations do not reflect seismic island design. For a further description on seismic island reference FSAR Subsection 6.2.3.2.3.1.

51) The Unit #1 offgas recombiner condenser is a dual code vessel. The shell is ASME Section VIII and the bonnet, tubes, and tube sheet are Section III. Section III is in excess of ESTB11-1 requirement but is remaining as Section III due to the inability of the shell supplier to re-stamp the entire condenser as Section VIII.

52) The Reactor and Turbine Building Closed Cooling Water System Heat Exchangers are presented separately.

TABLE 3.2-1

SSES DESIGN CRITERIA SUMMARY

53) For the strongback/carousel with integral nut-rack, compliance with the requirements of 10CFR50 Appendix B, (refer to column "Quality Assurance Requirements" and Note 7) is required only for the strongback components which are load-bearing during the RPV head lift. All other components are not within the scope of 10CFR50 Appendix B.

54) The diesel generator jacket water coolers (OE507B and OE507D) utilize an ASME Section VIII replacement tube bundle in accordance with the guidance of NRC Generic Letter 89-09.

55) The following manually operated valves provide a fillable volume for use of the RHRFPC mode.

The following manually operated valves, which are in the seismically analyzed sections of pipe, require a capability to be closed following a seismic event. These valves have been analyzed to demonstrate that they will be capable of closure following a seismic event:

Spent Fuel Pool to 153018A/B (253018A/B), Fuel Pool Gate Drain to 153038 (253038), and Reactor Well Diffuser to 15303OA/B (25303OA/B).

The following manually operated valves, which are in seismically analyzed sections of pipe, have a post seismic event function to remain in the closed position:

Reactor Well Drain to 153031 (253031), Reactor Well Drain to 153032 (253032), Reactor Well Drain to 153062 (253062), Dryer Separator Pool Drain to 153040 (253040), Dryer Separator Pool Drain to 153041 (253041), Cask Pit Gate Drain to 153050 (2503050), Cask Pit Drain to 153054 (253054), Cask Pit Drain to 053084 & 253800, and Cask Pit Diffuser to 053025.

56) The portions of piping between the surge tank up to and including Valves HV15308 (25308), 153076 (253076), and 153064A/B (253064A/B) have been analyzed to show that they will remain intact following a seismic event. These valves have been analyzed to demonstrate that they will be capable of closure (or remaining closed) following a seismic event. Closure of these valves is necessary to provide a fillable volume for use of the RHRFPC mode. The Skimmer Surge Tank Drain Line Valves, 153065A (253065A), are normally closed and assumed to remain closed during a seismic event.

57) Refuel Floor Wetlift System: The Main Steam Line (MSL) Plugs (Disk Spring Model) are supplied by Preferred Engineering. The MSL Plugs are designed to withstand a design pressure of 50 psig. The MSL Plugs Restraint Ring supplied by Preferred Engineering provides a mechanical means to prevent ejection of the MSL Plugs while moving fuel during 45.4 psig Local Leak Rate Test (LLRT) of Main Steam Isolation Valve (MSIV) and during 22.5 psig back pressurization LLRT of MSIV.

TABLE 3.2-1

SSES DESIGN CRITERIA SUMMARY

58) Qualified for Safe Shutdown Earthquake (SSE).

59) Refuel Floor Wetlift System: The Watertight Hook Box is supplied by Preferred Engineering for use with the Dryer and Separator Sling.

60) Refuel Floor Wetlift System: The Rigid Pole Handling System is supplied by ABB Combustion Engineering for use on the Unit 1 or 2 Refueling Platforms.

61) ASME Section III – NB-3674 "Design of Pipe Supporting Elements" states that supporting elements, including hangers, anchors, and sliding components shall be designed in accordance with NF-3600. (Pending completion of Subsection NF, supporting elements shall be designed in accordance with the requirements of ANSI B31.7-1969).

ANSI B31.7 and MSS-SP-58 (included by reference in ANSI B31.7) were the principal design codes for the GE portion of the suspension system.

62) The Horizontal Storage Modules and Dry Shielded Canisters are designed in accordance with 10CFR72. These components are designated as "Important to Safety".

63) The Dry Shielded Canister (DSC) is designed to meet the intent of ASME Section III, Subsection NB and the DSC Basket is designed to meet the intent of the ASME Section III, Subsections NF and NG, however the DSC is not a code vessel. Utilization of this ASME criterion meets or exceeds the requirements of 10CFR72.

64) Bottles conform to Department of Transportation (DOT) Standards, Title 49, Section 178.37, Specification 3AA. These bottles and associated connection assemblies are not available as Seismic Category I components. However, the bottles are mounted in Seismic Category I racks and are connected to Seismic Category I gas distribution piping.

65) Seismic Category "I" and Quality Assurance Requirement "Y" applies to the safety related subsystems (Motors, Fans, Cooling Coils, Ductwork and Dampers) of Drywell Unit Coolers 1V414A/B, 1V416A/B and the Recirculation Fans 1V418A/B. The Seismic Category for all other subsystems of Drywell Unit Coolers is "safety impact" type. The Quality Assurance requirements for all other subsystems of Drywell Unit Coolers is "N".

TABLE 3.2-1

SSES DESIGN CRITERIA SUMMARY

66) The SLC System Storage Tanks were purchased before Article NC-3800 on atmospheric tanks was included in the ASME Section III, Class 2 code. The tanks were designed and fabricated to API-650 and supplemental ASME Section III, Class C testing and examination requirements and therefore, meet Quality Group B requirements.

67) Hyrodgen Water Chemistry System: The hydrogen and oxygen storage tanks and associated equipment are located south of the Unit 2 turbine building, outside of the plant security boundary. The storage facility is owned, operated and maintained by a commercial gas supply vendor.

68) ASME Section III, Class 3 sample piping consists of those sample lines connected to the RWCU and FPCC Systems. These portions of the RWCU and FPCC Systems are design and constructed as ASME Section III, Class 3, yet are not Seismic Category I.

69) This section does not apply to the H_2O_2 Analyzers. See Post Accident Monitoring for the design criteria for the H_2O_2 Analyzers.

70) The design of the H_2O_2 Analyzer closed system outside primary containment is in accordance with the design requirements for such systems specified in USNRC Standard Review Plan 6.2.4 (September 1975), Containment Isolation Provisions, paragraph II.3.e., except as follows. The boundary valves between the H_2O_2 Analyzer and Post Accident Sampling System (i.e.,SV-1(2)2361, SV-1(2)2365, SV-1(2)2366, SV-1(2)2368, & SV-1(2)2369) are not electrical Class 1E. See Figure 3.2-2, requirements for "instruments which are open to containment and form a containment pressure boundary "for additional guidance regarding piping/tubing classification.

71) The Jet Pump Plugs are supplied by Preferred Engineering. The Jet Pump Plugs are designed to withstand a design pressure of 100 psi.

72) The referenced military standards (MIL-F-51068C and MIL-F-51079A) have been deleted, but represent acceptable standards for installed (or previously purchased) HEPA filters. New HEPA filters will meet the standards presented in ASME AG-1-1997.

73) The Service Platform is not used and has been eliminated.

74) Note the hydrogen recombiners are not credited in the accident analysis and do not perform a safety function but the equipment is currently maintained safety related.

TABLE 3.2-2

SUMMARY OF CODES AND STANDARDS FOR COMPONENTS OF WATER-COOLED NUCLEAR POWER UNITS SUPPLIED BY AE (ORDERED PRIOR TO JULY 1, 1971 WITH THE EXCEPTIONS OF THOSE COMPONENTS LOCATED INSIDE THE RCPB, AND THE REACTOR PRESSURE VESSEL)

	CODE CLASSIFICATIONS			
COMPONENT	**GROUP A**	**GROUP B**	**GROUP C**	**GROUP D**
Pressure Vessels	ASME Boiler and Pressure Vessel Code, Section III, Class A. See Footnote (2)	ASME Boiler and Pressure Vessel Code, Section III, Class C. See Footnote (2)	ASME Boiler and Pressure Vessel Code, Section VIII, Division 1	ASME Boiler and Pressure Vessel Code, Section VIII, Division 1 or Equivalent
0-15 Psig Storage Tanks	—	API-620 with NDT Examination	API-620 with NDT Examination	API-620 or Equivalent
Atmospheric Storage Tanks	—	Applicable Storage Tank Codes such as API-650, AWWAD100 or ANSI B 96.1 with NDT Examination	Applicable Storage Tank Codes such as API-650 AWWAD100 or ANSI B 96.1 with NDT Examination	API-650, AWWAD100 or ANSI B 96.1 or Equivalent
Piping	ANSI B 31.7, Class 1. See Footnote (3)	ANSI B 31.7, Class II. See Footnote (3)	ANSI B 31.7, Class III. See Footnote (3)	ANSI B 31.1.0 or Equivalent
Pumps and Valves	Draft ASME Code for Pumps and Valves Class I. See Footnote (1) & (4)	Draft ASME Code for Pumps and Valves Class II. See Footnote (1) & (4)	Draft ASME Code for Pumps and Valves Class III. See Footnote (4)	Valves - ANSI B 31.1.0 or Equivalent Pump - Draft ASME Code for Pumps Valves Class III or Equivalent

(1) All pressure-retaining cast parts are 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 are at least equivalent to those specified in the applicable class in the code.

(2) 1968 Edition including Addenda through Summer 1970.

(3) 1969 Edition and Addenda.

(4) November 1968 Edition and March 1970 Addenda.

TABLE 3.2-3

SUMMARY OF CODES AND STANDARDS FOR COMPONENTS OF WATER-COOLED NUCLEAR POWER UNITS SUPPLIED BY AE ORDERED AFTER JULY 1, 1971

CODE CLASSIFICATIONS

COMPONENT	GROUP A[1]	GROUP B[2]	GROUP C[3]	GROUP D[4]
Pressure Vessels	ASME Boiler and Pressure Vessel Code, Section III, Nuclear Power Plant Components – CLASS 1	ASME Boiler and Pressure Vessel Code, Section III, Nuclear Power Plant Components – CLASS 2	ASME Boiler and Pressure Vessel Code, Section III, Nuclear Power Plant Components – CLASS 3	ASME Boiler and Pressure Vessel Code, Section VIII, Division 1
Piping	As above[5][12][14][15][17][20]	As above[6][11][14][18][20]	As above [7][14][19][20]	ANSI B31.1 Power Piping[20]
Pipe Supports	As above	As above[11][13]	As above[11][13]	ANSI B31.1
Pumps	As above	As above	As above	Manufacturer's Standards
Valves	As above	As above	As above	ANSI B31.1
0-15 psig Storage Tanks	---	As above[8]	As above[8]	AP-620 or ASME Boiler and Pressure Vessel Code Section VIII, Division 1
Atmospheric Storage Tanks	---	As above[8]	As above[8][9][10]	API-650, AWWA D 100, ANSI B 96.1, or ASME Boiler and Pressure Vessel Code Section VIII, Division 1

[1][2][3] Components ordered after July 1, 1971 comply with the Codes and Standards in effect at the date of award of the order, except that Group A, B and C components ordered between July 1, 1971 and July 1, 1972 also comply with the following paragraphs of the ASME Boiler and Pressure Vessel Code, Section III, Winter, 1971 Addenda as applicable: (1) NB-2510, NB-2541, NB-2553, NB-2561, (2) NC-2510, NC-2571, (3) ND-2510, ND-2571.

[4] Certain portions of the radwaste systems meet the additional requirements of Quality Group D (Augmented) as defined in NRC Branch Technical Position ETSB 11-1, Parts B.IV and B.VI.

[5][6][7] For installation of ASME items, ASME Section III, 1971 Edition with Addenda through the Winter of 1972 shall apply. ASME material shall meet the requirements of ASME Section II, 1971 Edition through the Winter 1972 Addenda or any later Edition or Addenda. Any additional ASME Section III material requirements of Subsection 2000, 1971 Edition through the Winter 1972 Addenda, shall apply. For postweld heat treatment, Paragraphs NB-4600, NC-4600 and ND-4600 of ASME Section III, 1974 Edition, Summer 1976 Addenda are used.

For the installation of attachments to piping systems after testing, paragraphs NB-4436, NC-4436, and ND-4436 of ASME Section III, 1974 Edition, Summer 1976 Addenda are used.

For attachments to piping systems, Paragraphs NB-4433, NC-4433 and ND-4433 of ASME Section III, 1977 Edition, Summer 1979 Addenda are used.

For Code Nameplates, Stamping, and Data Reports, paragraphs NCA-8210, NCA-8220, NCA-8230, NCA-8300, NCA-8414, NCA-8415, NCA-8416, NCA-8417, NCA-8418, and NCA-8420 of ASME Section III, 1977 Edition, Winter 1977 Addenda are used.

TABLE 3.2-3 (Continued)

SUMMARY OF CODES AND STANDARDS FOR COMPONENTS OF WATER-COOLED NUCLEAR POWER UNITS SUPPLIED BY AE ORDERED AFTER JULY 1, 1971

(8)	Orders for Nuclear Storage Tanks were placed after December 31, 1971.
(9)	Atmospheric Storage Tanks fabricated to Group C requirements may be used in a Group D or Group D (Augmented) system.
(10)	The Diesel ‘E’ Fuel Oil Storage Tank Complies with ASME B&PV Code Section III, 1971 Edition, Winter 1972 Addenda. The A-D Diesel Generator Fuel Oil Storage Tanks comply with the ASME Boiler and Pressure Vessel Code, Section III, 1974 Edition, Winter 1975 Addenda as applicable.
(11)	Control Rod Drive Hydraulic System (CRD) piping and supports are constructed in accordance with ASME Section III, 1974 Edition with Addenda through Winter 1975 except as permitted by NA-1140(f) of ASME III as follows. Materials conform with ASME Section III, 1974 Edition, with Addenda through Winter 1975, or any later Edition of Addenda. ASME Section III, 1977 Edition, with Addenda through Winter 1977, Subsection NF, Paragraph NF-2610, shall apply to piping system support.
(12)	1” and smaller Nuclear Class 1 Piping is designed in accordance with the rules for Nuclear Class 2 piping per ASME Section III, 1974 Edition, Summer 1975 Addenda, Paragraph NB3630.
(13)	Allowable stresses for pipe supports for Nuclear Class 1, 2 and 3 piping shall be in accordance with ANSI Power Piping Code B31.1, 1973.
(14)	For the design of ASME flanges, ASME Section III, 1977 Edition with addenda through summer 1979 is used.
(15)	For the design of Nuclear Class 1, 1” branch connections, ASME Section III, 1977 Edition with Addenda through Summer 1979 is used.
(16)	Code case N316, approved for use at Susquehanna SES by the NRC on 2/17/82, is used in the Bechtel design of small pipe and CRD small pipe.
(17)	For the evaluation of Nuclear Class 1 piping components for snubber elimination or other piping modifications, ASME Section III, 1977 edition with addenda through summer of 1979 may be applied.
(18)	For the evaluation of Nuclear Class 2 piping components for snubber elimination or other piping modifications, ASME Section III, 1980 edition with addenda through winter of 1981 may be applied.
(19)	For the evaluation of Nuclear Class 3 piping components for snubber elimination or other piping modifications, ASME Section III, 1983 edition with addenda through summer of 1984 may be applied.
(20)	For the evaluation of ASME piping components or ANSI piping components which are analyzed for Seismic Category I requirements, Code Case N-411 may be applied for Snubber Elimination or other piping modifications/evaluations.

TABLE 3.2-4

CODE GROUP DESIGNATIONS - INDUSTRY CODES AND STANDARDS
FOR MECHANICAL COMPONENTS SUPPLIED BY THE NSSS VENDOR
(SEE NOTE a)

Group Classification	ASME III Code Classes		Components Ordered on or after Jan. 1, 1970 to July 1, 1971	Components Ordered on or after July 1, 1971
	1968 Ed.	1971 Ed.		
A	A	1	ASME III, 1 NA & NB Subsections TEMA C	ASME III, 1 NA & NB Subsection TEMA C note (d)
B	B*,C	2,MC*	ASME III, B* C ANSI B31.7 II NP & VC, TEMA C TANKS	ASME III, 2 & MC*, NA & NC Subsections NA & NE Subsections TEMA C TANKS NA, NC Note(d)
C	-	3	ASME VIII, Div. 1 ANSI B31.7, III NP & VC, III TEMA C TANKS	ASME III, 3 NA & ND Subsections TEMA C TANKS NA, ND Note (d)
D	-	-	ASME VIII, Div. 1 ANSI B31.1.0 TEMA C TANKS (b) Note (c)	ASME VIII, Div. 1 ANSI B31.1.0 TEMA C TANKS (b) Note (c)

* Metal containment vessel (as applicable) and extensions of containment only. Future addenda will include concrete containment vessels under ASME Section III, Divisions 2, at which time the requirements of this division shall also be met.

NOTES:

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

(b) Class D tanks shall be designed, constructed, and tested to meet the intent of API Standards 620/650, AWWA Standard D100, or ANSI B96.1 Standard for Aluminum Tanks.

(c) For pumps classified 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.

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

TABLE 3.2-5

SUMMARY OF SAFETY CLASS DESIGN REQUIREMENTS (MINIMUM)

Design Requirements	Safety Class			
	1	2	3	Other
Quality Group Classification[1]	A	B	C	D
Quality Assurance Requirement[2]	B	B	B	N/A
Seismic Category[3]	I	I	I	N/A

(1) The equipment shall be constructed in accordance with the indicated code group listed in Table 3.2-1 and defined in Tables 3.2-2, 3.2-3, and 3.2-4.

(2) B - The equipment shall be constructed in accordance with the quality assurance requirements of 10CFR50, Appendix B.

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

(3) I - The equipment for these safety classes shall be 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.

Security-Related Information
Figure Withheld Under 10 CFR 2.390

SUSQUEHANNA STEAM ELECTRIC STATION
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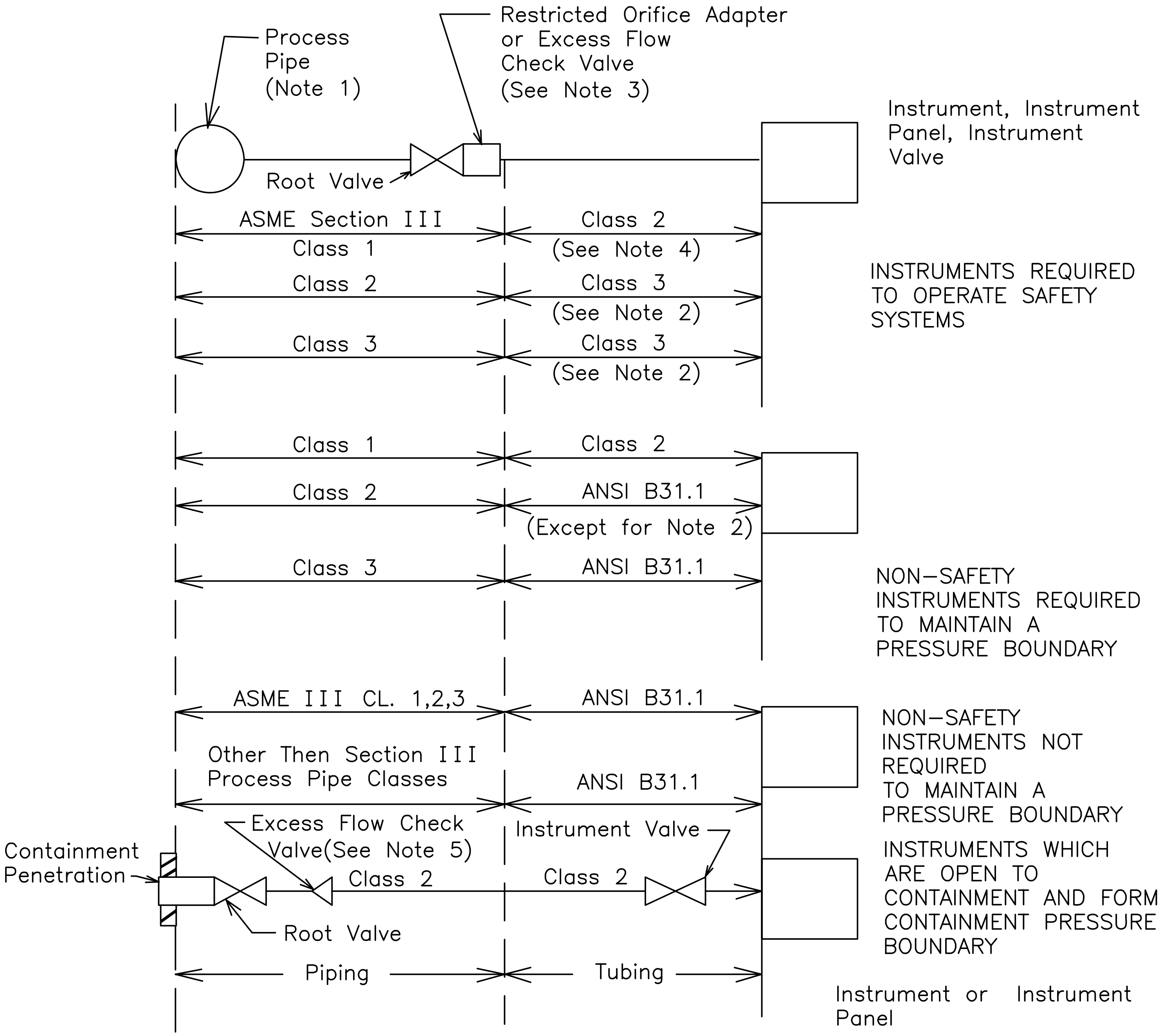
FINAL SAFETY ANALYSIS REPORT

CODE CLASSIFICATION
OF
PIPING AND VALVES

FIGURE 3.2-1, Rev. 53

Auto-Cad Figure Fsar 3_2_1.dwg



Notes:
1) Class for instrument lines from pipe to root valve and adaper is same as process pipe class.
2) Class 2 shall be required on lines that can contain reactor coolant or are radiation Class V and are outside contaiment.
3) A reducing adapter at the root valve serves as a restriction orifice.
4) Most GE shutoff instrument valves are B31.1 not Class 2.
5) Any automatic valve equivalent to an excess flow check valve may be used as an isolation valve for this type of line.

FSAR REV.65

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MINIMUM INSTRUMENT LINE
CLASSFICATIONS

FIGURE 3.2-2, Rev. 48

Auto-Cad Figure Fsar 3_2_2.dwg

3.3 WIND AND TORNADO LOADINGS

3.3.1 WIND LOADINGS

All exposed structures are designed for wind loading.

3.3.1.1 Design Wind Velocity

The design wind velocity for all structures is 80 mph at 30 ft above ground for a 100-year recurrence interval. The design wind velocity is based on Figure 5 of Reference 3.3-1. (References are listed in Subsection 3.3.3.)

The vertical velocity distribution is based on Table 1(a) of Reference 3.3-2. The velocity distribution is tabulated in Table 3.3-1.

A gust factor of 1.1, as given in Reference 3.3-2, is used.

3.3.1.2 Determination of Applied Forces

The procedure used to transform the wind velocity into an effective pressure applied to exposed surfaces of structures is as described in Reference 3.3-2 and is summarized as follows:

The dynamic pressure is given by:

$q = 0.002558\ V^2$ where,

q = Dynamic pressure in psf

V = Wind velocity in mph (design wind velocity x gust factor).

The local pressure at any point on the surface of a building is equal to:

$q \times Cp$

Where

Cp = Pressure coefficient.

The total pressure on a building is equal to:

$q \times C_D$

Where,

C_D = Shape coefficient.

The Susquehanna SES structures have sloping roofs with a pitch less than 20 degrees. The following are values for Cp and C_D. (See Reference 3.3-2, p. 1151 and Figure 7.)

Cp for windward wall = 0.8 (pressure)

Cp for leeward wall = -0.5 (suction)

Cp for windward slope = 0

Cp for leeward slope = -0.6 (suction)

C_D = 1.3 (pressure).

Wind loads on structures are tabulated in Table 3.3-1.

Exposed tanks are designed to resist a minimum wind load of 30 psf on the vertical projection, based on Reference 3.3-3. For cylindrical tanks, wind is considered acting on six-tenths of the vertical projection. No increases in allowable working stresses are permitted for these structures for loading conditions involving wind.

3.3.2 TORNADO LOADINGS

Table 3.3-2 lists the systems that are protected against tornadoes and the enclosures which provide this protection. This table is based on NRC Regulatory Guide 1.117 (Reference 3.3-4).

3.3.2.1 Applicable Design Parameters

The following design parameters are used for the design of tornado-resistant structures and are based on Reference 3.3-5:

a) Dynamic Wind Loading

Tangential speed: 300 mph
Translational speed: 60 mph

These speeds apply to all tornado-resistant structures except the Diesel Generator 'E' Building where a tangential speed of 290 mph and a translational speed of 70 mph are used.

b) Pressure Differential Between the Inside and Outside of a Building

A pressure drop of 3 psi is applied. A rate of 1 psi per second is used for all tornado-resistant structures except the Diesel Generator 'E' building where a rate of 2 psi per second is used.

c) Tornado-Generated Missiles

These are discussed in Subsection 3.5.1.4.

3.3.2.2 Determination of Forces on Structures

The following procedures are used to transform the tornado loadings into effective loads on structures:

a) Dynamic Wind Loading

A procedure the same as the one utilized to transform the wind velocity into an effective pressure, as described in Subsection 3.3.1.2, is used with the following exceptions:

1) Velocity and velocity pressure are assumed not to vary with height.
2) The gust factor is taken as unity.

As shown in Figure 5 of Reference 3.3-5, and as explained therein, the equivalent uniform tornado wind velocity on the building due to a tangential component of 300 mph and a translational component of 60 mph is 220 mph. The pressure loads are calculated on the basis of a uniform 300 mph wind velocity for all tornado-resistant structures except the Diesel Generator 'E' Building where they are calculated using a 360 mph wind velocity. The pressure loads are as follows:

	For All Tornado-Resistant Structures Except the Diesel Generator 'E' Bldg.	For the Diesel Generator 'E' Bldg.
Windward pressure on walls:	185 psf	266 psf
Leeward suction on walls:	115 psf	166 psf
Total design pressure:	300 psf	432 psf
Suction (uplift) on roof:	140 psf	199 psf

"The turbine building is designed to resist the tornado loading assuming 2/3 of the metal siding and the roof deck being blown away. However, all the frames are designed for the full tornado loading. The metal siding and the roof deck of all structures are not designed to resist full tornado loading."

b) Differential Pressure Loading

Differential pressure loading is calculated using the following pressure-time function:

The differential pressure is assumed to vary from zero to 3 psi, remain at 3 psi for 2 seconds and then return to zero. A rate of 1 psi per second is used for all tornado-resistant structures except the Diesel Generator 'E' building where a rate of 2 psi per second is used.

Blowout panels are used as necessary on safety-related structures to minimize differential pressure.

c) Tornado-Generated Missiles

Tornado-generated missiles used in the design of the tornado-resistant structures are given in Table 3.5-4 except those missiles used in the design of the Diesel Generator 'E' Building which are given in Table 3.5-4a. The barrier design procedures are described in Subsection 3.5.3.

Loadings a), b), and c) are combined in the following manner to obtain the total tornado loading:

(i) W' = Ww
(ii) W' = Wp
(iii) W' = Wm
(iv) W' = Ww+0.5Wp
(v) W' = Ww+Wm
(vi) W' = Ww+0.5Wp+Wm

Where,

W' = Total tornado load
Ww = Tornado wind load
Wp = Tornado differential pressure load, and
Wm = Tornado missile load

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

Structures not designed for tornado loads are checked to ensure that during a tornado they will not generate missiles that have more severe effects than those listed in Table 3.5-4. The modes of failure of these structures are analyzed to verify that they will not collapse on safety related structures.

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3.3.3 REFERENCES

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3.3-3. "Steel Tanks, Standpipes, Reservoir, and Elevated Tanks for Water Storage," AWWA Standard, D100-73.

3.3-4. "Tornado Design Classification," US NRC Regulatory Guide 1.117, (June 1976).

3.3-5. J. A. Dunlap and Karl Wiedner, "Nuclear Power Plant Tornado Design Considerations," Journal of the Power Division, ASCE, (March 1971).

3.3-6 "Design Basis Tornado For Nuclear Power Plants," US NRC Regulatory Guide 1.76, (April 1974).

TABLE 3.3-1

WIND LOADS ON STRUCTURES

Height Zone	Basic Wind Velocity	Dynamic Pressure	Wall Load Windward Pressure	Wall Load Leeward Suction	Total Design Pressure	Roof Load Suction
(ft)	(mph)	q(psf)	0.8q	0.5q	1.3q	.6q
0-50	80	20	16	10	26	12
50-150	95	30	24	15	39	18
150-400	110	40	32	20	52	24
400-700	120	45	36	23	59	27

TABLE 3.3-2

TORNADO WIND PROTECTED SYSTEMS AND TORNADO RESISTANT ENCLOSURES

	Protected System	Tornado Resistant Enclosure
1.	Reactor coolant pressure boundary	Reactor Building
2.	Reactor core and reactor vessel internals	Reactor Building
3.	Systems or portions of systems required for	
	a) Reactor shutdown	Reactor Building
	b) Residual Heat Removal	Reactor Building
	c) Cooling the spent fuel storage pool	Reactor Building
	d) Makeup water for primary system	Reactor Building
	e) Systems necessary to support service water, cooling water source, and component cooling	ESSW Pumphouse and Reactor Building
4.	Reactivity control systems	Reactor Building and Control Building
5.	Control room	Control Building
6.	Monitoring, actuating, and operating systems important to safety	Reactor Building and Control Building
7.	Electric and mechanical devices and circuitry between the process sensors and the input terminals of the actuator systems involved in generating signals that initiate protective action	Reactor Building, Diesel Generator Buildings, and ESSW Pumphouse

TABLE 3.3-2 (Continued)

	Protected System	Tornado Resistant Enclosure
8.	Long-term emergency core cooling system	Reactor Building, Diesel Generator Buildings, and ESSW Pumphouse
9.	Class 1E electric systems	All Seismic Category I structures

3.4 WATER LEVEL (FLOOD) DESIGN

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3.5 MISSILE PROTECTION

Where possible, the Seismic Category I and safety-related structures, equipment, and systems are protected from missiles generated by internal rotating or pressurized equipment through basic station component arrangement so that, if equipment failure occurs, the missile does not cause the failure of these structures, equipment, or systems. Where it is impossible to provide protection through plant layout, suitable physical barriers were provided to isolate the credible missile or to shield the critical system or component. Also, redundant Seismic Category I components are suitably protected so that a single missile cannot simultaneously damage a critical system component and its backup system. Table 3.2-1 provides a tabulation of safety-related structures, systems, and components, along with their applicable seismic category and quality group classification.

Section 3.12 - Separation Criteria for Safety-Related Mechanical and Electrical Equipment provides a detailed discussion of protection from missiles, such as equipment separation and redundancy, to preclude damage to the systems necessary to achieve and maintain a safe plant shutdown.

3.5.1 MISSILE SELECTION AND DESCRIPTION

3.5.1.1 Internally Generated Missiles (Outside Primary Containment)

There are two general sources of postulated missiles outside the primary containment:

a) Rotating component failure missiles

b) Pressurized component failure missiles

3.5.1.1.1 Rotating Component Failure Missiles

The systems located outside the primary containment have been examined to identify and classify potential missiles. The basic approach is to ensure design adequacy against generation of missiles, rather than to allow missile formation and then containing their effects.

Catastrophic failure of rotating equipment, such as pumps, fans, and compressors leading to the generation of missiles, is not considered credible. Massive and rapid failure of these components is incredible because of the conservative design, material characteristics, inspections, quality control during fabrication and erection, and prudent operation as applied to the particular component. The analysis of turbine missiles is discussed in Section 3.5.1.3.

It has been concluded that large, massive rotating components, such as the various ECCS pumps and motors, fans, and compressors outside the primary containment, do not have sufficient energy to move the masses of their rotating parts through the housings in which they are contained.

Similarly, it is concluded that the HPCI and RCIC turbines cannot generate missiles. Overspeed tripping devices ensure that the HPCI and RCIC turbines will not reach runaway speed where component failure could take place.

However, even with this conservative design, the RCIC and HPCI turbines are located in separate compartments so that any turbine missile will affect only one division of equipment.

This is also true for other large rotating safety-related equipment, such as pumps, fans, and compressors. Redundant equipment is normally located in different areas of the plant or separated by walls, so that a single missile from a rotating mass will not damage both redundant systems.

3.5.1.1.2 Pressurized Component Failure Missiles

The following potential internal missile donors from pressurized equipment were investigated:

a) High Energy Piping

Pressurized components in systems where service temperature exceeds 200°F or service pressure exceeds 275 psig were evaluated as to their potential for becoming missiles. Pipe whip restraints were provided at possible breakpoints of these high energy lines, which may impact on safety-related equipment or structures (see Section 3.6).

Additional attention has been given to ensure that safety relief valves and valve headers are not credible missiles. All SRV headers are restrained in accordance with the pipe whip criteria described in Section 3.6 to ensure that in the event of a circumferential type break of the header, no missile would result.

The safety relief valves are attached to welded, Schedule 160 sweepolet fittings on the headers. The design of this attachment includes all dynamic loads that may be associated with the SRV discharge.

The SRV header is designed and built to the conservative requirements of the ASME Section III, Class 1, Code and as such is subject to the ASME Section XI Inservice Inspection requirements. This inspection plus the RCPB leak detection capability would provide early indication of any possible failure in this area.

Therefore, it is concluded that the likelihood of missiles from high energy piping, which may impact on safety-related equipment, is remote.

b) Valve Bonnets

Valves of ANSI 900 psig rating and above, constructed in accordance with Section III of the ASME Boiler and Pressure Vessel Code, 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.

The bonnet bolts preload the pressure seal gasket so the valve will be sealed when it is not under pressure. When pressurized, the valve is sealed by process fluid pressure and the bonnet bolts are under no load. All ASME III Class I, 900 # bonnet-seal type valves were analyzed per ASME B & PV Code, Section III. Standard calculation pressure used in these analyses was given by Figure NB-3545.1-2 for weld-end valves.

Using the typical pressure seal valve shown in Figures 3.5-9 and 3.5-10 as an example, the total thrust load on the retaining ring and valve body was calculated. The results are listed in Table 3.5-7. The results show both the retaining ring and valve body meet the NB-3227 requirement while using a calculation pressure which is much higher than the normal operating pressure of the valve.

The majority of valves inside containment have massive valve operators which are supported by the yoke. For these valves, the valve operators act as an additional limitation to the yoke becoming a missile.

For a yoke clamp to fail, one would have to assume that the retaining ring fails completely and instantaneously so that the bonnet could strike the yoke. The yoke is normally under no load and complete failure of the yoke clamp is not considered credible.

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

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

c) Valve Stems

Valve stems are not considered 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 will be effectively restrained by the valve operators.

d) Temperature Detectors

Temperature or other detectors installed on piping or in wells are evaluated as potential missiles if a single circumferential weld would cause their ejection. This is highly improbable, since a complete and sudden failure of a circumferential weld is needed for a detector to become a missile. In addition, because of the spatial separation of redundant safety-related equipment, a small missile such as a detector, assuming the circumferential weld fails completely, is not likely to hit redundant safety-related equipment.

e) Nuts and Bolts

Nuts, bolts, nut and bolt combinations, and nut and stud combinations have little stored energy and thus are of no concern as potential missiles.

f) Blind Flanges

Bolted blind flanges are not considered credible missiles because of the extremely unlikely occurrence of all bolts experiencing simultaneous complete severance failure as discussed in (b) above.

g) Safety Relief Valve and Main Steam Isolation Valve Accumulators

Pressurized ASME III vessels such as SRV and MSIV accumulators are not considered credible missiles. These accumulators are operated at a maximum pressure and temperature of 150 psig and 150°F. These vessels have low stresses and operate in the "moderate energy" range and therefore, any failures would be a slow type and not of concern for missile generation.

3.5.1.2 Internally Generated Missiles (Inside Containment)

There are three general sources of postulated missiles inside the primary containment:

a) Rotating component failure missiles

b) Pressurized component failure missiles

c) Gravitationally generated missiles

3.5.1.2.1 Rotating Component Failure Missiles

The most significant pieces of rotating equipment in the primary containment are the recirculation pumps and motors. GE Licensing Topical Report NEDO-10677, submitted to the NRC contained a discussion of the potential overspeed of a recirculation pump due to LOCA blowdown flow past the pump impeller and the possible results of such overspeed. That report also presents a decoupler concept to protect the pump motor under such conditions.

In a letter to the NRC dated November 6, 1975, GE wrote that an analytical study has shown that a decoupling device is not needed, and that the NEDO-10677 report should be rescinded.

The following results were outlined in the GE letter to the NRC:

a) If a break were to occur in the pump discharge pipe, either a guillotine or longitudinal break, the maximum calculated resultant pump speed would be 110 percent of rated. In this analysis, the flow choking at the volume diffuser inlet area in the pump casing determined the differential feed and volumetric flow rate used to predict pump speed during blowdown. Longitudinal breaks up to one pipe cross-sectional area were considered.

b) For a longitudinal break in the pump suction pipe, the maximum calculated pump speed in the reverse direction would be 140 percent of rated. This speed does not result in mechanical motor damage. Longitudinal breaks up to one pipe cross-sectional area were considered.

c) For a guillotine suction pipe break the maximum calculated pump speed in the reverse direction is 710 percent of rated, which is a destructive overspeed of the motor. However, the initial torque for this event is 40 times the rated motor torque and this is sufficient to decouple the motor from the pump by mechanical failure of the pump to motor shaft. Mechanical failure is calculated to occur at 5 to 10 times the rated motor torque with or without a decoupler device in the drive train. Thus, an inherent self-decoupling would exist for this case.

On November 19, 1976, the NRC wrote GE a letter stating that applicants must file a formal application for amendment of their construction permit or operating license before they would be released from their commitment to installed the decoupler.

The letter also stated that "any such application to delete the decoupler from a boiling water reactor design must include a thorough safety evaluation setting for the reasons why a recirculation pump decoupler is no longer necessary."

GE has completed such a safety analysis report on a generic basis, in a letter from E.A. Hughes (GE) to R.C. DeYoung (NRC), January 18, 1977, "GE Recirculation Pump Potential Overspeed."

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

(1) Low Energy Missiles (Kinetic energy less than 1,000 ft-lbs):

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

(2) Medium Energy Missiles (kinetic energy less than 20,000 ft-lbs):

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

(3) The Motor as a Potential Missile:

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

It is concluded that the other rotating components inside the containment such as fans and chillers do not have sufficient energy to move the masses of their rotating parts through the housings in which they are contained.

In addition, redundant safety-related components are located in different areas of the containment, so that a rotating component failure missile will not damage both redundant components.

3.5.1.2.2 Pressurized Component Failure Missiles

A discussion of the potential for missile generation from the failure of pressurized components, e.g. valve stems, valve bonnets, and temperature element assemblies, is presented in Subsection 3.5.1.1.2. That discussion is also applicable to pressurized components inside containment.

3.5.1.2.3 Gravitationally Generated Missiles

Components necessary for the operation and safety of the reactor are designed to remain in place and functioning during all design basis conditions. Equipment which is not necessary for operation, startup testing, or safety is removed from the containment or seismically supported and secured in place prior to operation to ensure that it will not become a missile during plant operation or during a safe shutdown earthquake. Therefore, during reactor operation and following a LOCA, all equipment inside containment is secured. During maintenance when such equipment is returned to the containment or made operational, administrative and procedural methods will be used to ensure that significant damage is not caused to safety equipment even when the reactor is in the shutdown condition.

3.5.1.3 Turbine Missiles

An analysis was performed to evaluate the probability of damage from postulated turbine missiles to safety-related components. The probability of unacceptable damage due to turbine missiles (P4) has been calculated to be less than 1.00 E-7 per unit per year (see reference 3.5-20)

The NRC has established in NUREG 1048, Appendix U (reference 3.5-19) an acceptable methodology for establishing maintenance and inspection schedules for specific turbine systems including the original General Electric main turbines installed at Susquehanna. As a result of a retrofit of the main turbines with Siemens turbines, the missile probability analysis outlined in Reference 3.5-20 has been applied. This methodology also supports and maintains the established maintenance and inspection program outlined in Section 10.2.3.6 for the installed turbine.

The turbine inspection program frequencies implemented in Section 10.2.3.6 are supported by the probabilistic approach outlined in references 3.5-19 and 3.5-20. This approach shifts emphasis in the turbine missile damage calculations from the strike and damage portion to the missile generation portion. Turbine missile damage is a product of these two factors.

By managing turbine reliability through maintenance and inspection, the probability of generating a turbine missile can be determined.

The intent of the maintenance and inspection program is to ensure that the probability of generating a turbine missile (PI) is maintained to less than 1.00 E-5 per unit per year for an unfavorably oriented turbine with respect to the reactor building. Susquehanna's turbines are unfavorably oriented. The analysis supporting the program takes into account specific turbine wheel operating conditions, material properties, periodic maintenance and inspection results, and related system operating conditions. As a result, the main turbine maintenance and inspection program can facilitate evaluations of the effects of changes in parameters used as inputs to determining the probability of generating a turbine missile. Should any of these parameters change, the frequency changes to the maintenance and inspection program can be determined and adjusted accordingly. With this method, effects from changes to input parameters can be evaluated. Table 3.5-10, Turbine System Reliability Criteria reflects the recommendations from Table U.1 in reference 3.5-19 for an unfavorably oriented main turbine. By managing the probability of generating a missile to less than 1.00 E-5 (PI), the overall probability of turbine damage (P4) is maintained at less than or equal to 1.00 E-7 per unit per year.

Schedules for future inspection of low pressure turbine rotors with shrunk-on-disks will be based on this probabilistic approach and the analysis established in reference 3.5-19.

3.5.1.3.1 Turbine Placement and Orientation

The safety-related structures are those in which a single strike by a postulated turbine missile could result in a loss of the capability to function in a manner necessary to meet the requirements of 10CFR100.

At Susquehanna SES, these are the reactor buildings, diesel generator buildings, the control structure, and the ESSW pumphouse.

3.5.1.3.2 Missile Identification and Characteristics – Unit 1

The turbines at Susquehanna are manufactured by Siemens. Each unit consists of a tandem compound, six-flow, non-reheat, 1800 rpm turbine, directly connected to a synchronous generator.

Siemens has performed an analysis (Reference 3.5-20) to determine the characteristics of the missiles that can be expected as a result of a turbine burst. The most significant cause of a turbine missile is a burst-type failure of one or more bladed disks of an LP rotor. Relatively massive and strong turbine casings (Reference 3.5-20) would contain failures of other rotors including the HP and generator rotor.

3.5.1.3.3 Probability Analysis

The probability of turbine missile damage is expressed as:

P4 = P1 x (P2 x P3) (Eq. 3.5-1)

where:

P4 = probability of unacceptable turbine missile damage, per year

P1 = probability of a turbine failure resulting in the ejection of a missile, per year

P2 = probability that a missile will strike a barrier that houses a critical plant component, given that a missile has been ejected from the turbine, and

P3 = probability that a missile will spall the struck barrier, thus damaging an essential critical plant component, given that a missile has been ejected from the turbine and has struck the barrier.

P1, P2 and P3 are evaluated using a methodology the NRC has established in NUREG 1048, Appendix U (reference 3.5-19).

This methodology ensures that the probability of generating a turbine missile (PI) is maintained to less than 1.00 E-5 per unit year for an unfavorably oriented turbine with respect to the reactor building. Susquehanna's turbines are unfavorably oriented.

The value for P2 x P3 is assigned 1.00 E-2 for an unfavorably oriented turbine. NRC experience and simple estimates based on gross plant layouts formed the basis for this value (reference 3.5-19),

The P4 is obtained by multiplying P2x P3 by P1. Since P2 x P3 has been assigned 1.00 E-2 and P1 is less than 1.00 E-5, the limit for P4 is 1.00 E-7.

3.5.1.4 Missiles Generated by Natural Phenomena

Only tornado-generated missiles are considered. Table 3.5-4 lists the missiles considered in the design. Table 3.5.4a lists the missles considered in the design of the Diesel Generator 'E' Building. The structures designed for tornado-generated missiles are listed in Table 3.3-2.

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END – HISTORICAL INFORMATION

Based on the low event probability, aircraft hazards are eliminated as a design basis concern for Susquehanna SES.

3.5.2 SYSTEMS TO BE PROTECTED

3.5.2.1 Missile Protection Design Philosophy

Systems that are reviewed for missile protection are listed in Subsection 3.12.2.

For internally generated missiles, protection is provided through basic station component arrangement so that, if equipment failure occurs, the missile does not cause the failure of a Seismic Category I structure or any safety-related system. Where it is impossible to provide protection through station layout, suitable physical barriers are provided whose function is either to isolate the missile or to shield the critical system or component. In addition, redundant Seismic Category I components are suitably protected so that a single missile cannot simultaneously damage a critical component and its backup system.

3.5.2.2 Structures Designed to Withstand Missile Effects

Seismic Category I structures are designed to withstand postulated external or internal missiles which may impact them. Table 3.3-2 is a list of the structures designed to withstand external tornado-generated missiles, and the safety-related equipment which they protect. The missiles are listed in Table 3.5-4 for all tornado-resistant structures except the Diesel Generator 'E' Building. Table 3.5-4a lists the missiles used in the design of the Diesel Generator 'E' Building.

An investigation of the capability of plant safety-related structures, systems, and components has shown that exterior walls and roofs of Class I structures housing safety-related systems and components are adequate to withstand the 1-inch steel rod and the utility pole listed in Table 3.5-4.

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3.5.3 BARRIER DESIGN PROCEDURES

The structure and barriers are designed in accordance with the procedures detailed in Reference 3.5-5. The procedures include:

a) Prediction of local damage (penetration, perforation, and spalling) in the impact area including estimation of the depth of penetration

b) Estimation of barrier thickness required to prevent perforation

c) Prediction of the overall structural response of the barrier and portions thereof to missile impact.

The use of a ductility ratio higher than 10 but less than the allowables given in Reference 3.5.5 will be governed by the following conditions:

(1) Reinforced concrete barriers

The allowable displacement of reinforced concrete flexure members can be based on an upper limit for plastic hinge rotation r_θ as follows:

$$r_\theta = 0.0065 \frac{d}{c} \leq 0.07$$

where

d = distance from compression face to centroid of tensile steel reinforcement (inch)

c = distance from compression face to the neutral axis at ultimate strength (inch)

This condition is given in section C.3.5 of Appendix C and commentary to Appendix C of ACI 349-76. The design of the diesel Generator 'E' Building is based on ACI 349-80.

(2) Steel barriers

To insure the ability of a steel beam to sustain fully plastic behavior and thus to possess the assumed ductility at plastic hinge formation, it is necessary that the elements of the beam section meet minimum thickness requirements sufficient to prevent local buckling failure.

The conditions to preclude local buckling as given in AISC Manual are satisfied.

3.5.4 REFERENCES

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3.5-17 U.S. Nuclear Regulatory Commission, "Standard Review Plan 3.5.1.4, Rev. 2," NUREG-0800, (July, 1981).

3.5-18 U.S. Nuclear Regulatory Commission, "Standard Review Plan 3.5.3, Rev. 1," NUREG-0800, (July, 1981).

3.5-19 NUREG-1048, Supplement No. 6, Safety Evaluation Report Related to the Operation of Hope Creek Generating Station, Appendix U, Probability of Missile Generation in General Electric Nuclear Turbines

3.5-20 EC-093-1023, Turbine Missile Probability Analyses for Susquehanna Unit 1 & 2.

TABLE 3.5-1

THIS TABLE HAS BEEN DELETED

Table Rev. 42

TABLE 3.5-2

THIS TABLE HAS BEEN DELETED

Table Rev. 42

TABLE 3.5-3

THIS TABLE HAS BEEN DELETED

TABLE 3.5-4

TORNADO-GENERATED MISSILE PARAMETERS
FOR ALL TORNADO-RESISTANT STRUCTURES
EXCEPT THE DIESEL GENERATOR 'E' BUILDING

Missile	Weight (lb)	Velocity (mph)
Wood plank, 4 in. x 12 in. x 12 ft, traveling end-on	108	300
Steel pipe, 3 in. dia., Schedule 40, 10 ft long, traveling end-on	76	100
Automobile flying through the air at not more than 25 ft above the ground and having contact area of 20 sq ft.	4000	50
Steel rod 1-inch diameter x 3 feet long	8	216
Utility pole 13-1/2 inch diameter, 35 feet long acting not more than 30 feet above the ground	1490	144

NOTE:

The vertical velocities will be considered equal to 80% of the horizontal velocities mentioned above.

Security-Related Information
Text Withheld Under 10 CFR 2.390

Security-Related Information
Text Withheld Under 10 CFR 2.390

Security-Related Information
Text Withheld Under 10 CFR 2.390

TABLE 3.5-7

CALCULATED STRESS FOR
BONNET-SEAL TYPE VALVES

Bearing Stress		
Zone[3]	Calculated Stress	Stress Limit
b-c d-e	17.05 ksi 19.54 ksi	28.3 ksi 30.7 ksi
Shearing Stress		
Zone	Calculated Stress	Stress Limit
a-b c-f	7.60 ksi 10.83 ksi	11.34 ksi 12.3 ksi

Note:

(1) Above results are based on calculation pressure - 2425 psi.

(2) Valve design pressure = 1500 psi

(3) Refer to Figure 3.5-10.

TABLE 3.5-8

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TABLE 3.5-9

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Table Rev. 1

TABLE 3.5-10
TURBINE SYSTEM RELIABILITY CRITERIA

CRITERION	PROBABILITY. YR^{-1} UNFAVORABLY ORIENTED TURBINE	REQUIRED ACTION
(A)	$P_1 < 10^{-5}$	This is the general, minimum reliability requirement for loading the turbine and bringing the system on-line.
(B)	$10^{-5} < P_1 < 10^{-4}$	If this condition is reached during operation, the turbine may be kept in service until the next scheduled outage, at which time the licensee is to take action to reduce P_1 to meet the appropriate A criterion (above) before returning the turbine to service.
(C)	$10^{-4} < P_1 < 10^{-3}$	If this condition is reached during operation, the turbine is to be isolated from the steam supply within 60 days, at which time the licensee is to take action to reduce P_1 to meet the appropriate (A) criterion (above) before returning the turbine to service.
(D)	$10^{-3} < P_1$	If this condition is reached at any time during operation, the turbine is to be isolated from the steam supply within 6 days, at which time the licensee is to take action to reduce P_1 to meet the appropriate (A) criterion (above) before returning the turbine to service.

TABLE 3.5-4a

TORNADO-GENERATED MISSILE PARAMETERS FOR DIESEL GENERATOR 'E' BUILDING

	Missile	Weight (lb)	Impact Velocity (fps)
A)	Wood plank, 4 in. x 12 in. x 12 ft, traveling end-on	108	440
B)	Steel pipe, 3 in. dia., Schedule 40, 10 ft long, traveling end-on	72	147
C)	Steel Pipe, 6 in. dia. Schedule 40, 15 ft. long	285	170
D)	Steel 12 in. diameter Schedule 40, 15 ft. long	750	155
E)	Steel rod 1-inch dia. x 3 ft. long	8	317
F)	Automobile flying through the air at not more than 25 ft. above the ground and having contact area of 20 sq. ft.	4000	195
G)	Utility pole 13.5 in. dia, 35 ft. long	1490	211

Note:

The vertical velocities will be considered equal to 80 percent of the horizontal velocities mentioned above.

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Figure Deleted

FIGURE 3.5-1, Rev. 49

AutoCAD Figure 3_5_1.doc

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Figure Deleted

FIGURE 3.5-2, Rev. 49

AutoCAD Figure 3_5_2.doc

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FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Figure Deleted FIGURE 3.5-3, Rev. 49

AutoCAD Figure 3_5_3.doc

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Figure Deleted

FIGURE 3.5-4, Rev. 49

AutoCAD Figure 3_5_4.doc

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FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Figure Deleted FIGURE 3.5-5, Rev. 49

AutoCAD Figure 3_5_5.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
A-17, Sh. 1

FSAR REV. 65

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Figure 3.5-6 replaced by dwg.
A-17, Sh. 1

FIGURE 3.5-6, Rev. 55

AutoCAD Figure 3_5_6.doc

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REPLACED BY DWG.
A-21, Sh. 1

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Figure 3.5-7 replaced by dwg.
A-21, Sh. 1

FIGURE 3.5-7, Rev. 55

AutoCAD Figure 3_5_7.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
A-5, Sh. 1

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Figure 3.5-8 replaced by dwg.
A-5, Sh. 1

FIGURE 3.5-8, Rev. 48

AutoCAD Figure 3_5_8.doc

1. Valve design and manufacture in accordance with ASME Boiler and Pressure Vessel Code, Section III, Nuclear Power Plant Components, Class 1 with addenda up to and including the Winter 1972 Addenda, Bechtel Design Specification 8856-P-17 and Code Cases 1334-2, 1516-1, 1534 and 1535-2.

2. Hydrostatic Tests
   a. Shell in accordance with ASME Code Paragraph NB-6000 to 1830 psig. Test duration - 16 minutes minimum.
   b. Seat leakage in accordance with MSS-SP-61 to 2200 psig. Test duration - 4 minutes minimum. Allowable leakage rate 2 cc/hr./inch of nominal valve size.

3. Nondestructive testing and repair of pressure boundary parts shall be in accordance with the applicable paragraphs under Section NB-2000 of Section III and references thereto except as modified by Bechtel Design Specification 8856-P-17

4. Material certification per ASME Code: Para. NB-2130.

5.

Cert. of Tests	Body	Bonnet	Disc	Seat Ring
Radiography	YES		YES	
Ultrasonic		YES		
Mag. Particle				
Liquid Penetrant	YES	YES	YES	
Charpy-V-Notch				
L.P. Hardfacing		YES	YES	YES
Wall Thickness	YES	YES	YES	

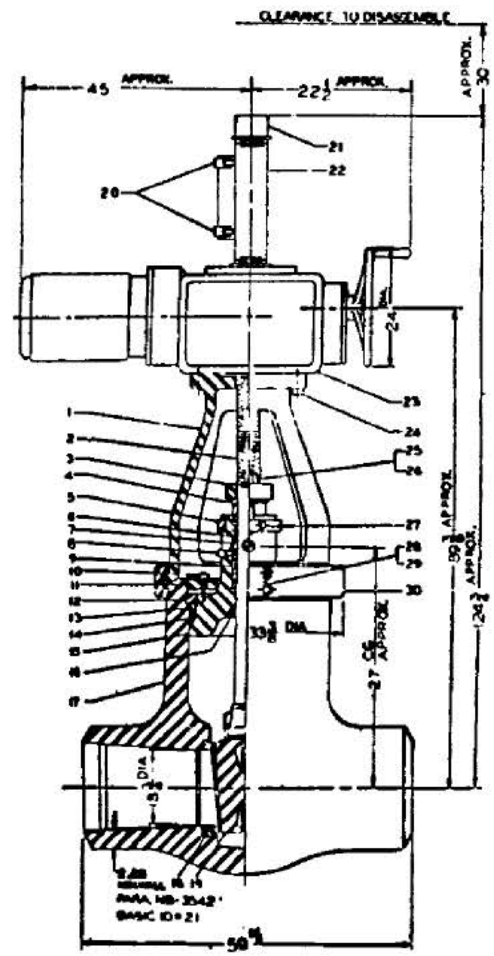
6.

Bechtel Item No.	Valve Mark No.	A.W. No. Serial No.	OPER. POS. NO.	STEM POSITION
1.1	24-DCA-GT-MO-F05A	E5855-1-1	2	HORIZ.
1.1	24-DCA-GT-MO-F05B	E5855-1-2	4	HORIZ.
1.2	24-DCA-GT-MO-F05A	E5855-15-1	2	HORIZ.
1.2	24-DCA-GT-MO-F05B	E5855-15-2	4	HORIZ.

7. Valve assembly weight 12,850 lbs. with operator.
8. Motor operator weight 1950 lbs.
9. Cv = 34,000 gpm
10. Limitorque Motor Operator Dwg. 02-408-0532-3
11. Limitorque Wiring Diagram Dwg. 15-477-5424-3

PARTS LIST

Part No.	Qty.	Description	Material
1	1	Yoke	A216-WCB
2	1	Stem	A564-630-H75
3	1	Follower Flange	A36
4	1	Follower Gland	A276-316
5	1	Gland Clamp	A36
6	1 Set	Upper Packing	John Crane 1871
7	1	Lantern Gland	A276-316
8	1	Pipe Plug	316 STAINLESS STEEL
9	1 Set	Lower Packing	John Crane 1871
10	12	Bonnet Cap Screws	Commercial Steel
11	1	Bonnet Clamp Ring	A182-F316
12	1	Gasket Retaining Ring	SA182-F316
13	1	Spacer Ring	A240-316
14	1	Pressure Seal Gasket	304L STAINLESS
15	1	Bonnet	SA182-F316
16	1	Stop Ring w/Stellite	SA240-316
17	1	Weld Ends Body	SA351-CF8M
18	2	Seat Rings w/Stellite	SA182-F316
19	1	Disc w/Stellite	SA351-CF3M
20	2	POSITION SWITCHES	NAMCO D2400X
21	1	Pipe Cap	Commercial Iron
22	1	Stem Protector	Commercial Steel
23	1	Limitorque	Commercial
24	8	Limitorque Cap Screws	A574-67
25	2	Follower BOLTS	A193-B7
26	2	Follower Nuts	A194-2H
27	8	Gland Cap Screws	A574-67
28	4	Yoke Clamp Studs	A193-B7
29	8	Yoke Clamp Nuts	A194-2H
30	1	Yoke Clamp	A36



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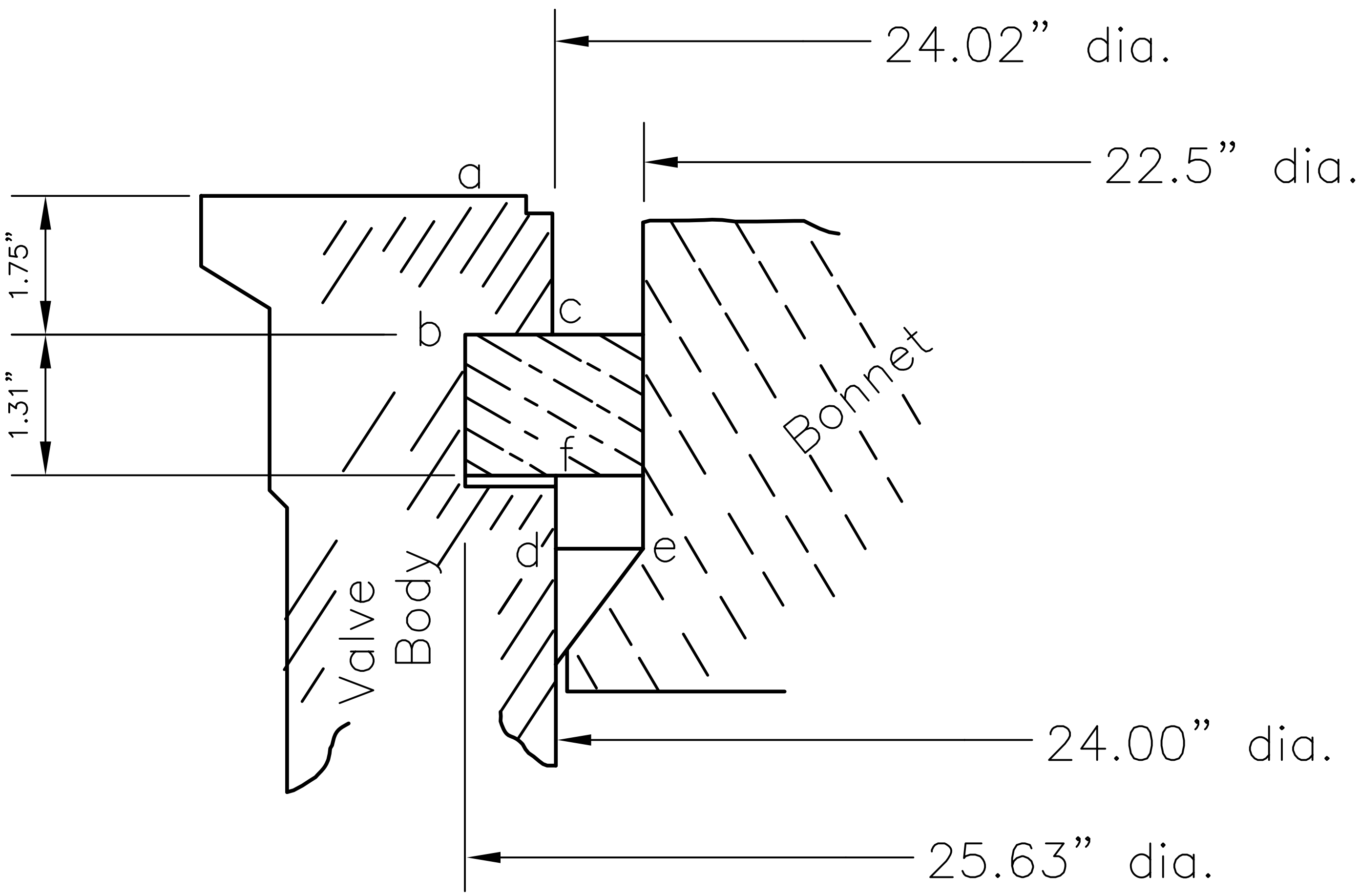
TYPICAL 900#
BONNET SEAL
TYPE VALVE

FIGURE 3.5-9, Rev. 47

Auto-Cad Figure Fsar 3_5_9.dwg



FSAR REV.65

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RETAINING RING DESIGN
FOR 900# BONNET-
SEAL TYPE VALVE

FIGURE 3.5-10, Rev. 47

Auto-Cad Figure Fsar 3_5_10.dwg

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

This section describes protection against dynamic effects associated with postulated rupture of piping both inside and outside containment.

3.6.1 POSTULATED PIPING FAILURES IN FLUID SYSTEMS

3.6.1.1 Design Bases

The underlined terms in this section are defined in Subsection 3.6.3. The ASME Boiler and Pressure Vessel Code, 1971 and the Addenda through Winter 1972 are referred to as the Code in the text.

In the design of nuclear safety systems, it is necessary to ensure that all components which are required for the safe shutdown of the plant will not fail as a result of a failure in a high energy or a moderate energy piping system. The separation criteria and a listing of separation techniques, for nuclear safety systems are given in Subsections 3.12.2.1 and 3.12.2.2, respectively.

Pipe breaks are postulated to occur in all high energy fluid system piping (or portion of system) in accordance with the criteria stated in Subsection 3.6.2.

Pipe cracks are postulated to occur in all moderate energy fluid system piping in accordance with the criteria stated in Subsection 3.6.2.1.2.

The failure of piping containing high energy fluid may lead to damage of surrounding systems and equipment. The effects of such a failure including pipe whip, fluid jet impingement, flooding, compartment pressurization, and environmental effects require special consideration to ensure the following:

a) The ability to shut down the reactor safely and maintain it in a safe shutdown condition

b) Containment integrity

c) A pipe break which is not a loss of reactor coolant must not cause a loss of reactor coolant

d) Resultant doses are below the guideline values of 10CFR 50.67.

A design basis for Susquehanna SES is that a postulated pipe break inside the containment (up to and including a rupture of the recirculation piping), in conjunction with the SSE, plus a single failure will not prevent the plant from accomplishing the above. For outside the containment, the single failure is qualified per NRC Branch Technical Position APCSB BTP 3-1, paragraph B.3.b.3. No credit is taken for non-seismic system in plant shutdown following a SSE, with the exception of the components and piping systems described in Subsection 6.7.

Components which are required to operate for a safe shutdown of the plant are protected from the below listed effects of postulated pipe failures, unless it can be demonstrated that the function of the safety equipment is not impaired.

Pipe Whip

Pipe whip is assumed to be one consequence of a guillotine failure of a high energy pipe. Cracks in moderate energy systems do not cause pipe whip. Pipe whip is an unrestrained pipe movement of either end of the ruptured pipe in any direction about a plastic hinge formed at the nearest pipe whip restraint.

A whipping pipe is assumed to rupture an impacted pipe of smaller nominal pipe size, and of the same nominal size with smaller wall thickness. A whipping pipe is assumed to have sufficient energy to cause the failure of impacted electrical cable ways and instrumentation unless the equipment can be shown to be sufficiently strengthened or protected. High energy piping is located away from the essential safety system wherever practical. Otherwise, pipe whip restraints are located on the piping to prevent pipe whip.

Jet Impingement

Jet impingement loads (due to pipe failures) on equipment and safety systems are considered. Protection against the jet impingement is provided either by separation by adding additional supports, or by the addition of barriers and enclosures.

Environmental

Pipe failures in high and moderate energy lines will release fluid which can increase temperature, pressure, and humidity in the vicinity of the pipe failure and also in remote areas that communicate with the local atmosphere. Safety equipment required after the pipe failure may be exposed to abnormal conditions which can degrade the capability of the equipment to perform its function.

Safety related equipment is qualified to meet the postulated environmental conditions.

Piping systems whose failure might generate hazardous environmental condition are located in compartments which are capable of being isolated from required safety systems. Isolation, where necessary, of compartments which enclose high energy lines is provided by maintaining normally closed accessways and providing automatic isolation of other communication paths, such as ventilation ductwork. Compartments are designed to withstand internal pressurization or are provided with vent capability to the atmosphere.

Pressure rise analysis and verification of structural adequacy of enclosures used to provide protection are discussed in Subsections 6.2.3, 3.8.1, 3.8.2, 3.8.3, and 3.8.4. Transportation of a steam environment which could affect the habitability of the control room has been discussed in Subsection 6.4.2.4.

An additional environmental consequence of pipe failure is radiation. Safety equipment is designed to tolerate the integrated exposure resulting from normal plant operations. Safety equipment inside the containment is designed for the additional exposure resulting from a DBA.

Water Spray

Water alone is a hazard to certain equipment, particularly electric equipment. Safety equipment is protected from water sprays by barriers or by enclosure of the equipment. In most cases, spatial separation is adequate to prevent spray from reaching the equipment.

Flooding

Any significant failure of a steam or fluid system may result in flooding. The flooding rate and the total fluid volume released are computed based on the break configuration, the service of the system, and the time required to isolate the system.

For compartments containing safety equipment, design features are provided to permit rapid detection and isolation of flooding due to major line breaks, except where it can be demonstrated that flooding will not affect the performance of the equipment or its redundant counterpart.

Because of the high degree of separation in Susquehanna SES, failure of ECCS equipment due to flooding will always be limited to one division of equipment. All non-safety grade piping in ECCS and other safety-related areas whose failure could potentially reduce the function of a safety-related plant feature to an unacceptable safety level, have been analyzed for the effects of a seismic event. This analysis is performed consistent with Regulatory Guide 1.29 paragraph C.2. A single failure is postulated and system availability is consistent with BTP APCSB 3-1, B.3.b.

If the initiating event is a break in a primary reactor coolant line, with subsequent leakage in an ECCS equipment room, isolation of the ECCS equipment room is required to prevent the depletion of the suppression pool inventory thus ensuring that long-term cooling capacity is adequate. This is discussed in Subsection 6.3.6.

If the initiating event is a pipe break in an ECCS equipment room, isolation is not required for long-term cooling adequacy. However, the room will be isolated and the equipment in the room declared inoperative, consistent with the requirements of the Technical Specifications.

Refer to Section 3.4 for additional information regarding flood protection from postulated piping failures inside the Reactor Building.

3.6.1.2 Description

A listing of high energy fluid system piping is provided in Table 3.6-1. A listing of moderate energy fluid piping systems is provided in Table 3.6-1a. Proximity of the essential systems and components in relation to the high and moderate energy fluid system piping is reviewed and the essential systems and components are either relocated to achieve separation, protection against the effects of pipe failure is provided, or it can be shown that the effects of pipe failure could be withstood. Tables 3.6-2 and 3.6-3 show those safety components in close proximity to high energy fluid system piping which required jet impingement protection. The method used to protect each component is also shown.

Some of the high energy fluid system piping is separated from all essential systems and components. These piping systems are listed in Table 3.6-1 designated by Note 1.

Descriptions of some typical high energy fluid system piping are provided below.

3.6.1.2.1 Main Steam System

Separation

The main steamlines inside the containment are routed, wherever possible, away from safety related equipment. Two steamlines, A and B, are connected to the north side of the reactor vessel and the other two steamlines, C and D, are connected to the opposite side of the vessel.

To avoid failing all the main steam ADS safety relief valves by a single rupture of a main steam pipe, the ADS valves are divided so that three ADS valves are connected to the A and B lines, and the remaining ADS valves are connected to the C and D lines.

To avoid failing both the RCIC and HPCI steam supply lines by a single rupture of a main steam pipe, the RCIC is connected to main steamline C and the HPCI is connected to main steamline B.

Besides those areas identified in Tables 3.6-2 and 3.6-3, the main steam lines are separated by adequate distance from safety-related components. The following design features have been incorporated into the main steamlines to ensure the core cooling capability over the entire range of operating and postulated accident conditions.

1) A flow restrictor (venturi) is located in each main steamline just upstream of the inboard isolation valve. The purpose is to limit the flow of steam and therefore the loss of reactor coolant from the reactor vessel in the event of a postulated break in this line, outside the primary containment.

2) The safety/relief valves protect the main steamlines from abnormal pressure. The safety/relief valves, besides protecting the steamlines against over-pressurization, precipitate the initiation of the LPCI mode of the RHR system for smaller pipeline breaks by rapidly depressurizing the reactor vessel.

3) Separate main steam loops supply high pressure steam to run the turbine driven pumps of RCIC and HPCI systems. Should steam power not be available to drive the reactor feedwater pumps during shutdown, part of the residual steam will be used to drive the turbine in the RCIC system which supplies makeup water to the reactor from the condensate storage tanks or the suppression pool.

In addition, the following design features are incorporated into the design to ensure isolation valve operability and the leaktight integrity of the containment:

a) The piping between the containment isolation valves is designed to meet "no break" criteria stress limits of Subsection 3.6.2.1.1.

b) Moment limiting restraints are placed upstream of inside containment isolation valves and downstream of outside containment isolation valves for HPCI, RCIC, feedwater outside containment, main steam drain, main steam and RWCU pipes.

c) Plate barriers are provided to protect the inboard main steam isolation valve operators from a high energy pipe break of the feedwater lines. In addition, the main steam isolation valve limit switch support brackets are reinforced to address the jet impingement effects from a high energy break of the recirculation nozzles. The actuator springs are capable of closing the main steam isolation valves under jet impingement conditions without pneumatic assist, see Section 5.4.5.2.

<u>Pipe Break Locations and Pipe Whip Restraints</u>

The postulated pipe break locations and the type of the break are determined based on the criteria given in Subsection 3.6.2. Figures 3.6-1A, 3.6-1B, 3.6-1C and 3.6-1D shows the locations of postulated pipe breaks and pipe whip restraints. The main steamlines are restrained inside the primary containment to prevent the main steam pipe whip. The main steamlines in the turbine building are separated from essential systems and components.

Verification of the Safe Shutdown of the Plant

1) The routing of main steam piping, locations of pipe whip restraints, and the protective measures described in Table 3.6-2 ensure that the emergency core cooling systems are not adversely affected by a postulated pipe break in the main steam lines.

2) Subsequent to any postulated pipe break in the main steamlines, containment isolation is achieved by closure of either or both of the isolation valves and the safe shutdown of the plant is accomplished by the emergency core cooling systems.

3.6.1.2.2 Feedwater System

Separation

The feedwater spargers are connected to opposite sides of the reactor vessel. The sparger restricts the rate of loss of reactor water level in the event of a postulated pipe break inside the primary containment. The spargers are then connected into two feedwater loops, which run in parallel through the primary containment and which are reasonably separated from safety related components.

Pipe whip restraints are provided inside the primary containment to prevent one feedwater pipe from damaging the other as a result of pipe break.

The two feedwater lines extend outside the primary containment before they connect to a common header. Restraints are also provided outside the containment. Bumpers are provided at the end of the header, before it enters the turbine building.

The HPCI return line taps into one of the feedwater lines and the RCIC taps into the other. The RWCU return line taps into both feedwater lines.

The feedwater piping in the turbine building is separated from essential systems and components.

The following design features have been incorporated into the design of feedwater lines to ensure isolation valve operability and the leaktight integrity of the containment:

a) The piping between the containment isolation valves is designed to meet the "no break" criteria stress limits of Subsection 3.6.2.

b) Moment limiting restraints are placed upstream of two outside containment isolation valves to protect against the pipe break beyond the restraints. Inside containment check valves on one feedwater line are protected against the pipe break postulated in the other feedwater line. The containment penetration flued head is designed to withstand pipe break loads.

Pipe Break Locations and Pipe Whip Restraints

The postulated pipe break locations and the type of the break are determined based on the criteria given in Subsection 3.6.2. The Figure 3.6-2 shows the locations of postulated pipe breaks and pipe whip restraints. The feedwater lines are restrained inside the primary containment to prevent pipe whip.

Verification of Safe Shutdown of the Plant

1) For any postulated pipe break in the feedwater piping inside or outside the primary containment, isolation of the reactor and the containment from the external environment is provided by the two containment isolation valves located outside primary containment. The outermost containment isolation valve provides positive closure by virtue of being a stop check valve.

   For a feedwater line break inside containment, the operability of the containment valve inside containment is not credited as providing containment isolation, as described in Section 6.2.4, for this event.

2) The two feedwater lines are restrained to prevent pipe whip damage. The HPCI, which is a high pressure emergency core cooling system, taps into one feedwater loop while the RCIC taps into the other loop to ensure adequate core cooling. In addition to the HPCI and the RCIC, the ADS relief valves and the RHR system, which are not in this area, are also available for core cooling.

3.6.1.2.3 High Pressure Coolant Injection (HPCI) System

Separation

The steam supply line to the HPCI turbine taps off main steamline B, inside the primary containment. The piping is routed, wherever possible, away from safety related components which are required for the safe shutdown of the plant.

For small line pipe breaks, the HPCI system functions as a redundant system to the combination of ADS and LPCI (mode of RHR) system. The routing of this pipe ensures that any postulated pipe break does not disable the ADS function.

Proximity of the essential systems and components in relation to the HPCI lines were reviewed and the findings listed in Tables 3.6-2 and 3.6-3.

Pipe Break Locations and Pipe Whip Restraints

The postulated pipe break locations and the type of the break are determined based on the criteria given in Subsection 3.6.2.

Figure 3.6-3 shows the locations of postulated pipe breaks and pipe whip restraints. Pipe whip restraints are provided for the high energy portions of this system.

Verification of the Safe Shutdown of the Plant

Postulated pipe breaks in this line affect neither the primary containment integrity nor the systems required to bring the reactor to a safe shutdown condition.

If a pipe break does occur in the HPCI line, the reactor and the primary containment are isolated from the external environment by isolation valves. The other emergency core cooling systems, ADS and LPCI (RHR), would be used to bring the reactor to a safe shutdown.

3.6.1.2.4 Reactor Core Isolation Cooling (RCIC) System

Separation

The steam supply line to the RCIC turbine taps off main steamline C, inside the primary containment. The piping is routed, wherever possible, away from safety related components which are required for the safe shutdown of the plant. Proximity of the essential systems and components in relation to the RCIC lines were reviewed and the findings listed in Tables 3.6-2 and 3.6-3.

Pipe Break Locations and Pipe Whip Restraints

The postulated pipe break locations and the types of breaks are determined based on the criteria given in Subsection 3.6.2.

Figure 3.6-4 shows the location of postulated pipe breaks and pipe whip restraints. Pipe whip restraints are provided for the high energy portions of this system.

Verification of the Safe Shutdown of the Plant

Postulated pipe breaks in this line neither affect the primary containment integrity nor the systems required to bring the reactor to a safe shutdown condition.

If a pipe break does occur in the RCIC line, the reactor and the primary containment are isolated from the external environment by isolation valves. Shutdown of the plant is achieved by the emergency core cooling systems.

3.6.1.3 Safety Evaluation

The analysis of postulated line failure and the resulting addition of restraint features into the design has ensured that failure in any single high energy fluid system piping in the plant will not result in unacceptable damage to any other safety-related system or component.

3.6.2 DETERMINATION OF PIPE FAILURE LOCATIONS AND DYNAMIC EFFECTS ASSOCIATED WITH THE POSTULATED PIPING FAILURE

Information concerning break and crack location criteria and methods of analysis is presented in this section. 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.

The analyses to determine the postulated break and crack locations are based on the original plant life of 40 years. The locations determined by those analyses and the criteria specified in this section are identified in Tables 3.6-6 through 3.6-15. A fatigue monitoring program tracks the fatigue (cumulative usage) at all critical piping locations. When a monitored location exceeds the cumulative usage predicted by the original design fatigue analysis, the affected piping system is evaluated to determine if any additional break and crack locations must be postulated. Any new locations that are postulated are accommodated by appropriate pipe break restraints, barriers, and shields.

3.6.2.1 Criteria Used To Determine Pipe Break and Crack Locations and Their Configurations

Pipe failures are postulated in high and moderate energy fluid systems piping that are not separated from essential systems and components based on the criteria given in this section. The types of failures considered at those locations are also discussed in this section.

3.6.2.1.1 High Energy Fluid System Piping Other than Recirculation System Piping

Fluid System Piping Between Containment Isolation Valves

Pipe breaks are not postulated in these portions of high energy fluid system piping provided the following additional design requirements are met:

1) The following design stress and fatigue limits are satisfied:

For ASME Code, Section III Class 1 piping:

a) The primary plus secondary stress intensity range, Sn, calculated for normal and upset conditions by equation (10) of Paragraph NB-3653, does not exceed 2.4 Sm. Or,

b) The range of stress intensity, Sn, calculated for normal and upset conditions by equation (10) does not exceed 3.0 Sm, and the cumulative fatigue usage factor associated with normal, upset and testing conditions is less than 0.10. Or,

c) The range of stress intensity, Sn, calculated for normal and upset conditions by equation (10) exceeds 3.0 Sm, but the stress intensity ranges computed by equations (12) and (13) are less than 2.4 Sm. In addition, the fatigue usage factor associated with normal, upset and testing conditions is less than 0.10. And,

d) The loading resulting from a postulated pipe break beyond these portions of the piping

    1) Does not cause the primary stress intensity, as calculated by equation (9) of Paragraph NB-3652 to exceed 2.25 Sm.

2) A plastic hinge is not formed and the operability of the isolation valve is assured.

For ASME Code, Section III Class 2 and 3 piping:

e) The maximum stress ranges as calculated by the sum of equations (9) and (10) in Paragraph NC-3652, for normal and upset conditions does not exceed $0.8(1.2S_h + S_A)$.

f) The maximum stress, as calculated by equation (9) in Paragraph NC-3652 under the loadings resulting from a postulated piping failure of fluid system piping beyond these portions of piping does not exceed 1.8S. Higher stresses are allowed provided that the valve operability is not impaired.

2) The piping is restrained reasonably close to the valve, such that occurrence of a pipe break inside or outside containment beyond these restraints will impair neither operability of the valve nor the integrity of the containment penetration. Terminal ends of the piping runs extending beyond these portions of high energy piping are considered to originate at a point adjacent to these restraints.

3) Welded pipe support attachments to those portions of piping penetrating containment are avoided to eliminate stress concentrations.

4) The number of piping circumferential and longitudinal welds and branch connections is minimized.

5) The length of piping run is minimized, consistent with requirements to keep stress levels low and provide access for in-service inspection.

6) The design at points of pipe fixity, e.g., pipe anchors or welded connections at containment penetrations, do not require welding directly to the outer surface of the piping (e.g., flued, integrally forged pipe fittings are acceptable designs), except where such welds are 100 percent volumetrically examinable in service and a detailed stress analysis is performed to demonstrate compliance with the limits of 1) above.

7) To the extent described in Subsection 6.6.8, the in-service examination completed during each inspection interval will provide 100 percent volumetric examination of circumferential and longitudinal pipe welds within these portions of piping. See paragraphs 5.2.4.7 and 6.6.8.

<u>Fluid System Piping Other Than That Between Containment Isolation Valves</u>

Pipe breaks are postulated to occur at terminal ends, and at all intermediate break locations determined by one of the following two criteria:

a) At each location of potential high stress such as pipe fittings (elbows, tees, reducers, etc.), valves, flanges, and welded attachments

b) At each location where the following stress and fatigue limits are not met:

For ASME Code, Section III, Class 1 Piping under normal and upset conditions,

1) The primary plus secondary stress intensity range, Sn, as calculated by equation (10) of Paragraph NB-3653, does not exceed 2.4 Sm, or

2) The stress intensity range, Sn, as calculated by equation (10) of Paragraph NB-3653 exceeds 2.4 Sm, but is less than 3.0 Sm, and the cumulative fatigue usage factor is less than 0.10, or

3) The stress intensity range, Sn, calculated by equation (10) exceeds 3.0 Sm, but the ranges of stresses computed by equations (12) and (13) of subparagraph NB-3653 are less than 2.4 Sm and the fatigue usage factor is less than 0.10.

For ASME Code, Section III, Class 1 piping:

4) In the event that at least two intermediate pipe break locations cannot be determined by the above stress and fatigue usage criteria, a minimum of two locations of highest stress as calculated by equation (10) in Paragraph NB-3653, and which are separated by a change of direction in the pipe run, are selected.

For ASME Code, Section III, Class 2 and 3 piping:

5) The maximum range of stress, as calculated by the sum of equations (9) and (10) in Paragraph NC-3652, for normal and upset plant conditions, does not exceed 0.8 (1.2 S_h + S_A).

6) If two intermediate break locations cannot be determined by the above stress and fatigue usage criteria, a minimum of two locations of highest stress, as calculated by the sum of equations (9) and (10) in Paragraph NC-3652, and which are separated by a change in direction of the pipe run, are selected.

For piping not designed to seismic Category I standards:

7) Criteria for ASME Code, Class 2 and 3 piping was used if all necessary analyses are made. Otherwise, longitudinal and circumferential breaks in non-Category I piping are postulated in accordance with (a) above. All breaks or cracks were assumed to occur at the worst location. Only one pipe break at a time is postulated to occur concurrent with the SSE.

For all classes of pipe:

8) When the above stress and fatigue criteria result in less than two intermediate break locations, a minimum of two separated locations are chosen based on highest stress. Where the piping consists of a straight run without fittings, welded attachments, or valves, a minimum of one location is chosen. The two locations chosen are with at least 10 percent difference in stress, or separated by a change of direction of pipe run if stress differs by less than 10 percent.

For high energy piping in the Reactor Building, shown in Figures 3.6-17-1, 3.6-17-2 and 3.6-17-3, pipe breaks are postulated to occur at terminal ends, and at all intermediate break locations determined by criterion "a" above. Alternatively, criterion “b” may be used if intermediate breaks become too numerous and/or it becomes necessary to minimize the number of whip restraints required. Both circumferential and longitudinal breaks are postulated at each of the intermediate break locations, whereas only circumferential breaks are postulated at the terminal ends. Additionally, NRC Generic Letter 87-11 is used on a case-by-case basis for identifying high energy pipe breaks.

Protection in the areas shown on Figures 3.6-17-1, 3.6-17-2 and 3.6-17-3, is a combination of separation, barriers, and pipe whip restraints. The CRD system high energy piping as noted on Figure 3.6-17-1 required no restraints due to separation and barrier location.

3.6.2.1.2 Moderate Energy Fluid System Piping Other than Recirculation Piping System

1) Through-wall leakage crack locations are postulated in moderate energy piping located in areas containing systems important to safety. Orientation of the crack is such as to result in the most adverse water spray and flooding conditions.

2) Through-wall leakage crack locations are postulated in fluid system piping located within, or outside and adjacent to, protective structures designed to protect essential systems and components except in seismic Category I systems where exempted by (3), (4), or where the maximum stress range in these portions of Class 2 or 3 piping or non-nuclear piping as calculated by the sum of equations (9) and (10) in Paragraph NC-3652 is less than $0.4(1.2S_h + S_A)$, or where the maximum stress intensity range of Class I piping, as calculated by equation (9) of NB-3652, is less than 0.6 Sm.

The cracks are postulated to occur individually at locations that result in the maximum effects from fluid spraying and flooding, with the consequent hazards on environmental conditions developed.

3) No through-wall leakage crack locations are postulated in moderate energy piping systems in areas where high energy piping system break locations are postulated, except where a postulated leakage crack in the moderate energy fluid system piping results in more severe environmental conditions than the break in proximate high energy fluid system piping.

4) Through-wall cracks are not postulated in portions of seismic Category I moderate energy piping between containment isolation valves, provided they meet the requirements of Subarticle NE-1120 of the Code and they are designed so that the maximum stress, for ASME Code, Section III, Class I piping, as calculated by equation (9) of Paragraph NB-3652, does not exceed 0.6 Sm, and the maximum stress range for Class 2, 3 or non-nuclear piping, as calculated by the sum of equations (9) and (10) of Paragraph NC-3652, does not exceed $0.4\ (1.2S_h + S_A)$.

5) For moderate energy piping not designed to seismic Category I standards, through-wall leakage cracks are postulated at locations that result in maximum effects from fluid spray and flooding.

3.6.2.1.3 Types of Breaks and Leakage Cracks in Fluid System Piping Other than Recirculation Piping System

Circumferential Pipe Break

A circumferential break is assumed to result in severance of a high energy pipe, perpendicular to the pipe axis, and separation amounting to at least a one-diameter lateral displacement of the ruptured piping section unless physically limited by piping restraints, structural members, or piping stiffness.

Circumferential breaks are postulated in high energy fluid system piping of nominal pipe size greater than 1 in. at the locations determined by the criteria given in Subsection 3.6.2.1.1, except where it can be shown that the maximum stress is in the circumferential direction and is at least 1.5 times the longitudinal stress, in which case only a longitudinal break is postulated.

Longitudinal Pipe Break

A longitudinal pipe break is an axial split parallel to pipe axis without pipe severance. Break opening area is assumed to be equal to the effective cross-sectional flow area of the pipe at the break location and length of the break is assumed to be twice the inside diameter of the pipe. The orientation of the break is assumed to be such that the jet reaction force causes out-of-plane bending of the piping configuration.

Longitudinal pipe breaks are postulated in high energy fluid system piping of nominal size 4 in. and larger at the break locations determined by the criteria given in Subsection 3.6.2.1.1 with the following exceptions:

1) Longitudinal pipe breaks are not postulated at

   a) Terminal ends provided the piping at the terminal ends contains no longitudinal pipe welds

   b) Intermediate break location where the criterion for a minimum number of break locations must be satisfied

   c) Where it can be shown that the maximum stress is in longitudinal direction and is at least 1.5 times the circumferential stress. In this case only circumferential break needs to be postulated.

Through Wall Leakage Cracks

A through-wall leakage crack is a crack opening in a moderate energy pipe assumed as a circular orifice of cross-sectional flow area equal to one-half the pipe inside diameter times one-half the pipe wall thickness. The crack may occur at any orientation about the circumference of the pipe and is postulated to occur in moderate energy piping larger than 1 in. nominal pipe diameter.

3.6.2.1.4 Criteria for Recirculation System Piping (NSSS Supply)

3.6.2.1.4.1 Definition of High Energy Fluid System

See Subsection 3.6.3.

3.6.2.1.4.2 Definition of Moderate Energy Fluid System

There are no moderate energy lines in the recirculation system piping.

3.6.2.1.4.3 Postulated Pipe Breaks

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 system only.

A high-energy piping system break is not postulated to be simultaneous 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.

3.6.2.1.4.4 Exemptions from Pipe Whip Protection Requirements

Protection from pipe whip need not be provided if any one of the following conditions exist:

(1) Piping which is classified as moderate energy piping.

(2) 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.

(3) 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 will be considered as insufficient to rupture an impacted pipe of equal or greater nominal pipe size and equal or heavier wall thickness.

All other effects from pipe breaks, such as jet impingement, pressure, temperature, humidity, wetting of all exposed equipment and flooding have been considered for those breaks exempted by the above criteria.

3.6.2.1.4.5 Location for Postulated Pipe Breaks

Postulated pipe break locations are selected in accordance with Regulatory Guide 1.46, NRC Branch Technical Position APCSB 3-1, Appendix B and as expanded in NRC Branch Technical Position MEB 3-1. For ASME Section III, Class 1 piping systems which are classified as high energy, the postulated break locations are:

(1) The terminal ends of the pressurized portions of the run.

(2) 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:

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

(b) The stress ranges, as calculated by Equations (12) or (13) of the Code, exceed 2.4 Sm or if the cumulative usage factor exceeds 0.1 when equation (10) exceeds 3 Sm.

(3) In the event that two or more intermediate locations cannot be determined by stress or usage factor limits, a total of two intermediate locations shall be identified on a reasonable basis (a) for each piping run or branch run.

(a) Reasonable basis shall be one or more of the following:

(1) Fitting locations

(2) Highest stress or usage factor locations

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.

3.6.2.1.4.6 Types of Breaks to be Postulated in Fluid System Piping

The following types of breaks are postulated in high energy fluid system piping:

(1) No breaks need be postulated in piping having a nominal diameter less than or equal to one inch.

(2) Circumferential breaks are postulated only in piping exceeding a one inch nominal pipe diameter.

(3) Longitudinal splits are postulated only in piping having a nominal diameter, equal to or greater than 4 inches.

(4) Circumferential breaks are to be assumed at all terminal ends and at intermediate locations identified by the criteria in Subsection 3.6.2.1.4.5. At each of the intermediate postulated break locations identified to exceed the stress and usage factor limits of the criteria in Subsection 3.6.2.1.4.5 either a circumferential or a longitudinal break, or both, shall be postulated per the following:

a. Circumferential breaks shall be postulated at fitting joints and;

b. Longitudinal breaks shall be 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.

c. Consideration shall be 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 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 shall be considered.

d. At intermediate locations chosen to satisfy the minimum break location criteria, only circumferential breaks shall be postulated.

(5) For design purposes, a longitudinal break area shall be assumed to be the equivalent of one circumferential pipe area unless analytical methods representing test results can conservatively reduce forces based on a mechanistic approach.

(6) For both longitudinal and circumferential breaks, after assessing the contribution of upstream piping flexibilities, pipe whipping is 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.

(7) For a circumferential break, the dynamic force of the jet discharge at the break location will be 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 shall be used, as applicable, in the reduction of the jet discharge.

3.6.2.1.5 High Energy Fluid Systems With and Without Sufficient Capacity to Develop a Jet Stream

Some of the high energy fluid system piping do not have any flow during plant normal and upset operating conditions. These lines have either a check valve or a normally closed valve in the system. Only that portion of the piping between the RPV and the check valve of the normally closed valve, is considered to be high energy system.

For a postulated pipe break in the high energy portion of the system, that portion of the piping towards the normally closed valve, is considered not to have fluid energy reservoir with sufficient capacity to cause pipe whip. Table 3.6-13 lists these systems and the reservoirs with and without sufficient capacity to develop a jet stream.

3.6.2.2 Analytical Models to Define Forcing Functions and Response Models

3.6.2.2.1 For Piping Other than Recirculation Piping System

Analyses are performed for the pipe failure postulated in Subsection 3.6.2.1. Analysis of jet thrust forces which result in the event of a pipe rupture are described in Section 2.2 of Reference 3.6-2. Fluid jet impingement forces are discussed in Section 2.3 of Reference 3.6-2. Impulsive loading and impact combined with impulsive loading are described in Sections 3.2 and 3.3 respectively of Reference 3.6-2. Alternatively, nonlinear time history dynamic analyses are performed. The forcing function used in piping dynamic analysis is obtained using Reference 3.6-1 and Reference 3.6-7. A typical forcing function and the piping system model used for the dynamic response analysis is provided on Figure 3.6-12 and Figure 3.6-12a.

A typical piping system model used in the dynamic analysis is provided on Figure 3.6-11A.

Protection against the pipe whip is accomplished by restraining the motion of the pipe after pipe break. The pipe whip restraints are designed with energy absorbing components, i.e., crushable honeycomb, in the direction of the pipe whip. Crushable honeycomb limits the reaction load in the whip restraint in most cases to about 80% of the design yield load for the restraint and absorbs the energy to greatly reduce the tendency of the pipe to rebound after impact.[1]

When the required energy absorption is too great to be entirely accomplished by the honeycomb, the plastic deformation capability of the whip restraint itself is taken into account. The structural steel whip restraint is permitted to have plastic deformation that results in ductility ratio no greater than 20.[2] For structural steel subjected to shock and impact loading, ductility ratio of 20 is an acceptable practice (Reference 3.6-9). Reference 3.6-8 was used in determining the response of the piping system under pipe break loads.

The criteria for the dynamic analyses are as follows:

1) An analysis of the piping system is performed for each longitudinal and circumferential postulated rupture at the break locations determined in accordance with the criteria of Subsection 3.6.2.1.

2) The loading condition of a piping system prior to postulated rupture in terms of internal pressure, temperature, and stress state is that condition associated with reactor operating at 100 percent power.

3) For a circumferential rupture, pipe whip dynamic analyses are performed only for that end (or ends) of the pipe or branch that is connected to a contained fluid energy reservoir having sufficient capacity to develop a jet stream.

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[1] Energy absorption capacity of the honeycomb associated with crushing up to 60% of its original height is used in the design calculations. The load deflection curve in this region is relatively flat.

[2] Ductility ratio is defined as plastic strain (deformation) divided by the strain (deformation, at yield strength of the material.

4) Dynamic analytic methods used for calculating the piping and piping/restraint system response to the pipe break forces adequately account for the effects of:

   a) Translational masses (and rotational masses for major components) and stiffness properties of the piping system, restraint system, major components, and support walls

   b) Transient forcing function(s) acting on the piping system

   c) Elastic and inelastic deformation of piping and/or restraint

   d) The design clearance between the pipe and the restraint.

5) A 10 percent increase of minimum specified design yield strength (Sy) is used to account for strain rate effects in inelastic nonlinear analyses.

Figures 3.6-1A to 3.6-8E show the pipe break locations and pipe break restraint locations and Tables 3.6-6a, 3.6-6b, 3.6-6c, 3.6-6d, 3.6-6e, 3.6-6f, 3.6-6g, 3.6-6h, 3.6-7, 3.6-7a, 3.6-8, 3.6-8a, 3.6-9, 3.6-9a, 3.6-10, 3.6-10a, 3.6-11, 3.6-11a, 3.6-12a, 3.6-12a.1, 3.6-12a.2, 3.6-12a.3, 3.6-12a.4, 3.6-12a.5, 3.6-12a.6, 3.6-12a.7, 3.6-12b, 3.6-12b.1, 3.6-12b.2, 3.6-12b.3, 3.6-12b.4, 3.6-12c.1, 3.6-12d.1, 3.6-12d.3, 3.6-12e.1, 3.6-12e.2, and 3.6-12e.3. 3.6-13 show the summary of the analysis of main steam, feedwater water, HPCI, RCIC, CORE SPRAY, RHR SUPPLY and RHR Return Lines, Head Vent Line, Head Spray, STANDBY LIQUID CONTROL, and MSIV Drain Lines.

These figures and tables indicate the breaks for which dynamic analysis was performed and the type of the break assumed.

3.6.2.2.2 Analytic Methods to Define Blowdown Forcing Functions and Response Models for Recirculation Piping System (NSSS Supply)

3.6.2.2.2.1 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 recirculation piping system are presented in the following sections.

The criteria used for calculation of fluid blowdown forcing functions includes:

(1) Circumferential breaks are 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 is demonstrated by the inelastic pipe whip analysis (Subsection 3.6.2.2.2.2).

(2) The dynamic force of the jet discharge at the break location are 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 are taken into account, as applicable, in the reduction of jet discharge.

(3) A rise time not exceeding one millisecond is used for the initial pulse.

Blowdown forcing functions are determined by either of two methods given in (1) and (2) below:

(1) 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 3.6-6. These are simply described as follows:

a. 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).

b. The waves cause various sections of the pipe to be loaded with time-dependent forces. It is 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 is assumed to travel from the break at a speed equal to the local speed of sound within the fluid. Wave reflections will 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 are used as boundary conditions. The blowdown thrust causes a reaction force perpendicular to the pipe break.

c. 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).

d. 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:

$$F = \left[(P - P_a) + \frac{\rho u^2}{2g} \right] A$$

Where

F	=	Blowdown Force
P	=	Pressure at exit plane
P_a	=	Ambient pressure
u	=	Velocity at exit plane
ρ	=	Density at exit plane
A	=	Area of break
g	=	Gravitational constant

e. 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 3.6-3. 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.

(2) The following is an alternate method for calculating blowdown forcing functions.

The computer code RELAP3 (Ref. 3.6-4) 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:

$$\frac{T}{A_e} = P_E - P_{00} + \frac{G_E^2 \, \overline{V}_E}{g_c}$$

$$R = -\frac{T}{A_e} \times A_{te}$$

Where

$\frac{T}{A_e}$	=	Thrust Per unit Break Area – Lb/ft^2
P_E	=	Exit Pressure – Lb/ft^2
P_{00}	=	Receiver Pressure – Lb/ft^2
G_E	=	Exit Mass Flux – $Lb/Sec\text{-}ft^2$
$\overline{V}_E$	=	Exit Specific Volume – ft^3/Lb
g_c	=	Gravitational Constant – 32.174 $Ft\text{-}Lb_m/sec^2\text{-}Lb_f$
R	=	Reaction Force on the Pipe – Lb
A_{te}	=	Effective Target Area – ft^2

3.6.2.2.2.2 Pipe Whip Dynamic Response Analysis for Recirculation Piping System

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 3.6.2.2.2.1. Analytical methods used to account for this loading are discussed below.

The criteria used for performing the pipe whip dynamic response analyses include:

(1) A pipe whip analysis is performed for each postulated pipe break. However, a given analysis can 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.

(2) The analysis includes the dynamic response of the pipe in question, and the pipe whip restraints which transmit loading to the structures.

(3) The analytical model adequately represents the mass/inertia and stiffness properties of the system.

(4) Pipe whipping is assumed to occur in the plane defined by the piping geometry and configuration, and to cause pipe movement in the direction of the jet reaction.

(5) Piping within the broken loop is no longer considered part of the RCPB. Plastic deformation in the pipe is considered as a potential energy absorber. Limits of strain are imposed which are similar to strain levels allowed in restraint plastic members. Piping systems are designed so that plastic instability does not occur in the pipe at the design dynamic and static loads unless damage studies are performed which show the consequences does not result in direct damage to any essential system or component.

(6) Components such as vessel safe ends and valves which are attached to the broken piping system and do not serve a safety function or whose failure would not further escalate the consequences of the accident, are not designed to meet ASME Code imposed limits for essential components under faulted loading. However, if these components are 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 ensure operability if required will be met.

The pipe whip analysis was performed using the PDA computer program (Reference 3.6-5). 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 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.

A comprehensive verification has been performed to demonstrate the conservatisms inherent in the PDA pipe whip computer program and the analytical methods utilized. This is described in

Reference 3.6-6. 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 3.6-6. 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 GE. This approximation is demonstrated in Figure 3.6-15. 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 GE analysis. The deflection limit used by NSC is the design deflection at one-half of the ultimate uniform strain for the GE restraint design. The restraint properties used for both analyses are provided in Table 3.6-4.

A comparison of the NSC analysis with the PDA analysis, as presented in Table 3.6-5 and Figure 3.6-16, 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% 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.

3.6.2.3 Dynamic Analysis Methods to Verify Integrity and Operability

3.6.2.3.1 For Piping Other than Recirculation Piping System

Pipe whip restraints and compensating struts are used to control pipe whipping during a postulated rupture of the pipe. Barriers are used to protect components against jet impingement.

Compensating struts are mechanical snubbers used to perform the following functions:

a. Permit unrestrained thermal motion of the pipe.

b. Restrain pipe motion under seismic and other dynamic loads, and

c. Resist sustained loads resulting from a pipe break.

The pipe whip restraints used to protect the mechanical components are designed either as a part of the normal restraint system or as independent restraints. The independent restraints are designed solely to control movement of the pipe following pipe break and function only during pipe break. A typical pipe whip restraint of this type is shown in Figure 3.6-10.

Pipe whip restraints are placed near the isolation valves whose operability is required. These pipe whip restraints are an integral part of the normal pipe support system and are designed to pipe break loads. A typical pipe whip restraint arrangement to protect the isolation valve is shown in Figure 3.6-11.

A time-history dynamic analysis of the piping near isolation valves is performed for the pipe break loads and stresses in the pipe and loads on the restraints are determined. The stress in the pipe at the isolation valve is maintained below yield strength of the material to ensure valve operability. Since the section modulus of the valve is much greater than that of the pipe, the stress in the valve body would be below yield strength of the valve. Therefore, the deformations in the valve body would be small and would be in the elastic range such that binding of the valve internals cannot occur.

3.6.2.3.1.1 Design Loading Combinations

The design loading combinations applied in the design of the restraints for equipment and piping are categorized with respect to the plant operating conditions which are identified as normal, upset, emergency, and faulted as described in Table 3.9-1.

3.6.2.3.1.2 Design Stress Limits

Integral Restraints - when restraints for equipment piping are designed as an integral part of the normal support system, the design loading combinations for normal, upset, emergency, and faulted conditions are applicable. In evaluating the supports and restraints for ASME, Section III, Classes 1, 2, and 3 piping, the design stress limits applied in evaluating loading combinations for normal, upset, emergency, and faulted (except for pipe rupture) conditions are those given in Table 3.9-11. After rupture of the supported pipe occurs, the piping system is no longer within the jurisdiction of ASME Section III because the pressure boundary has been breached. The restraints are evaluated for pipe rupture loads as described in Subsection 3.6.2.2.1.

Independent Restraints - when restraints are designed solely to control movement following a postulated pipe rupture and to function independently of the normal support system, only the design pipe rupture loads are applicable.

To ensure that restraints do function independently of the normal support system, the motions of the intact pipe due to all normal and upset plant conditions and the vibratory motion of the SSE are calculated and used to specify a minimum clearance between the pipe and the restraint. Wherever possible, gaps between pipes and restraints are maximized to avoid possible contact during plant operation. Where a particular location requires minimizing a gap, special features are provided to permit adjustment of the gap size during hot functional testing.

The restraints are evaluated for the pipe rupture loads as described in Subsection 3.6.2.2.1.

3.6.2.3.2 Dynamic Analysis Methods to Verify Integrity and Operability for Recirculation Piping System (NSSS Supply)

3.6.2.3.2.1 Jet Impingement Analyses and Effects on Safety Related Components Resulting from Postulated Ruptures of the Recirculation Piping System

The methods used to evaluate the jet effects resulting from the postulated breaks of recirculation piping are same as those discussed in Subsection 3.6.2.3.1.

3.6.2.3.2.2 Pipe Whip Effects Following a Postulated Rupture of the Recirculation System Piping

Pipe whip (displacement) effects on safety related structures, systems and components can be placed in two categories: (I) pipe displacement effects on components (nozzles, valves, tees, etc.) which are in the same piping run that the break occurred in; and (2) pipe whip or controlled displacements onto external components such as building structure, other piping systems, cable trays and conduits, etc.

(1) Pipe displacement effects on components in same piping run.

a. The criteria which is used for determining the effects of pipe displacements on in-line components is as follows:

(i) Components such as vessel safe ends and valves which are attached to the broken piping system and do not serve a safety function or whose failure would not further escalate the consequences of the accident, need not be designed to meet ASME Code Section III imposed limits for essential components under faulted loading.

(ii) If these components are 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, will be met.

b. The methods used to calculate the pipe whip loads on piping components in the same run as the postulated break are described in Subsection 3.6.2.2.2.2.

(2) Pipe displacement effects on structures, other systems and components.

a. Pipe displacement effects on structures are as follows:

The drywell floor and the reactor pedestal support pipe whip restraints for the 28 in. diameter recirculation loop piping. A description of the loading on these structures due to a postulated rupture of a 28 in. diameter recirculation loop pipe is given in Subsection 3.8.3.3.2.1.

The reactor shield wall supports pipe whip restraints for the 12 in. and 22 in. diameter recirculation loop piping. The equivalent static loads on the reactor shield wall due to a postulated rupture of a recirculation loop pipe are specified by G.E. and are as follows:

Pipe Diameter (in.)	Equivalent Static Load (kips)
12	270
22	630

3.6.2.3.2.3 Loading Combinations and Design Criteria for Recirculation Piping 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 other unbroken pipe, its restraints, and structure to which the restraint is attached, are analyzed and designed accordingly.

The pipe whip restraint devices designed, tested, fabricated and installed by GE for the recirculation loop piping, utilize energy absorbing wire rope cable restraints. The wire rope cable restraint uses a low clearance design with a frame attached to a support and carbon steel wire ropes restraining the pipe. The low clearances between the cable restraints and the pipe prevent the pipe from building up a large amount of kinetic energy. Thus, the cables have to absorb only a limited quantity of energy, and resist large forces. A conceptual sketch for the restraints is shown in Figure 3.6-13. However, the restraints do have some clearance between them and the process pipe to allow for installation of some normal pipe insulation, and thermal movements during plant operation.

The specific design objectives for the restraints are:

(1) 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.

(2) 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.

(3) The restraints should provide minimum hindrance to in-service inspection of the process piping.

For the purposes of design, the pipe whip restraints are designed for the following dynamic loads:

(1) Blowdown thrust of the pipe section that impacts the restraint;

(2) Dynamic inertia loads of the moving pipe section which is accelerated by the blowdown thrust and subsequent impact on the restraint;

(3) Design characteristics of the pipe whip restraints are included and verified by the pipe whip dynamic analysis described in Subsection 3.6.2.2.2.2; and

(4) Since the pipe whip restraints are not contacted during normal plant operation, the postulated pipe rupture event is the only design loading condition.

As previously described, the recirculation loop pipe whip restraints are composed of two parts, the cable and the restraint frame. Both parts of the restraining device function as load carry members, and will deflect under load. The load configurations for a cable restraint are shown in

Figure 3.6-13. The components of the restraints are categorized as Type I and II, as described below:

Type I - radial load-carrying members - these members composed of cables will absorb energy loaded in the direction perpendicular to the restraint base by elastic, and plastic deformations (Figure 3.6-13 Item a)

Type II - tangential load-carry members - these members composed of restraint frames will absorb energy loaded in the direction parallel to the base by plastic deformation. (Figure 3.6-13 Item b)

Each of these components is constructed of a different material in order to fulfill different design objectives. The design requirements and design limits for each component are therefore different. They are specified as below:

(1) Type I - Carbon steel wire ropes.

For carbon steel wire ropes, the maximum acceptable load was

- 90 percent of the load carrying capacity of the cable in the restraint configuration. This limit takes into consideration efficiency reduction experienced when a cable is wrapped around a pipe. This means that the design load is limited to about 75 percent of a minimum certified load carrying capacity of the cable in tension.

(2) Type 2 - Restraint Frames

Design limits for the ASTM A36 restraint frames is as follows:

(i) Design Load

The load bearing member is primarily a cantilever beam with an extra support (the diagonal plate) at approximately midspan. At loads approaching the plastic moment capability of the beam, the plastic hinge forms at section determined from an elastic structural analysis. The maximum design load and the ultimate load are calculated based on plastic moment capability, Mp, of this section, with the diagonal plate stressed uniformly at the minimum ultimate stress.

(ii) Design Deflection

The design and ultimate deflection are calculated assuming the beam remains straight and rotates about a point on the upper surface of the beam. The maximum design deflection at the load point is calculated assuming the diagonal plate undergoes 10 percent elongation. The ultimate deflection of the beam is based on a 20 percent ultimate elongation of the diagonal plate.

3.6.2.4 Guard Pipe Assembly Design Criteria

Guard pipe assembly design is not used in this plant.

3.6.2.5 Material To Be Submitted for Operating License Review

3.6.2.5.1 For Piping Other than Recirculation Piping System

The following paragraphs indicate how the criteria for protection against dynamic effects associated with postulated piping features are implemented.

1) The criteria given in Subsection 3.6.2.1 have been adhered to in locating the pipe failure locations and type of the failure. These locations are shown on Figures 3.6-1 to 3.6-8e.

2) Protective devices such as pipe whip restraints and the barriers are used. A typical pipe break restraint is shown in Figure 3.6-10. The in-service inspection requirements are implemented as discussed in Section 6.6.

3) Analytical methods to analyze the effects of pipe break are discussed in Subsections 3.6.2.2 and 3.6.2.3. Summary of the results are shown on Tables 3.6-6a, 3.6-6b, 3.6-6c, 3.6-6d, 3.6-6e, 3.6-6f, 3.6-6g, 3.6-6h, 3.6-7, 3.6-7a, 3.6-8, 3.6-8a, 3.6-9, 3.6-9a, 3.6-10, 3.6-10a, 3.6-11, 3.6-11a, 3.6-12a, 3.6-12a.1, 3.6-12a.2, 3.6-12a.3, 3.6-12a.4, 3.6-12a.5, 3.6-12a.6, 3.6-12a.7, 3.6-12b, 3.6-12b.1, 3.6-13.

4) All safety related systems and components have been protected from the effects of pipe whip and their design intended function will not be impaired to an unacceptable level.

3.6.2.5.2 Implementation of Criteria for Pipe Break and Crack Location and Orientation for Recirculation Piping System (NSSS Supply)

3.6.2.5.2.I 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 3.6.2.I.4. The postulated pipe break locations and types selected in accordance with these criteria are shown in Figure 3.6-14. Conformance with these criteria is demonstrated in Tables 3.6-14 and 3.6-15.

3.6.2.5.2.2 Implementation of Special Protection Criteria

The pipe whip restraints provided for the recirculation piping system are also shown in Figure 3.6-14. Using the analysis methods of Subsection 3.6.2.2.2.2, this system of restraints has been found to prevent unrestrained pipe whip resulting from a postulated rupture at any of the identified break locations.

3.6.2.5.2.3 Jet Effects for Postulated Ruptures of Recirculation System Piping

Jet effects from postulated breaks in the recirculation piping have been reviewed and modifications made as part of the jet impingement review program.

3.6.3 DEFINITIONS

Essential Systems and Components - Systems and components required to shut down the reactor and mitigate the consequences of a postulated piping failure, without offsite power.

High-Energy Fluid Systems - Fluid systems that, during normal plant conditions, are either in operation or maintained pressurized under conditions where either or both of the following are met:

a) Maximum operating temperature exceeds 200°F

b) Maximum operating pressure exceeds 275 psig

Moderate-Energy Fluid Systems - Fluid systems that, during normal plant conditions, are either in operation or maintained pressurized (above atmospheric pressure) under conditions where both of the following are met:

a) Maximum operating temperature is 200°F or less

b) Maximum operating pressure is 275 psig or less.

A system that operates within pressure-temperature conditions specified for a high energy fluid system, for less than 2 percent of the time the system operates as a moderate energy fluid system, is considered a moderate energy fluid system.

Normal Plant Conditions - Plant operating conditions during reactor startup, operation at power, or reactor cooldown to cold shutdown condition, but excluding test modes.

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

Sh and Sa - Allowable stresses at maximum (hot) temperature and allowable stress range for thermal expansion, respectively, as defined in Article NC-3600 of the ASME Code, Section III.

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

Single Active Component Failure - Malfunction or loss of function of a component of electrical or fluid systems. The failure of an active component of a fluid system is considered to be a loss of component function as a result of mechanical, hydraulic, pneumatic, or electrical malfunction, but not the loss of component structural integrity. The direct consequences of a single active component failure are considered to be part of the single failure.

Terminal Ends - Extremities of piping runs that connect to structures, components (e.g., vessels, pumps, valves), or pipe anchors that act as rigid constraints to piping thermal expansion. A branch connection to a main piping run is a terminal end of the branch run, except when all three of the following conditions are in effect:

1) The branch nominal size is at least half that of the main run;

2) The intersection is not rigidly constrained to the building structure; and

3) The branch and main runs are included together in the same piping stress analysis model.

In piping runs which are maintained pressurized during normal plant conditions for only a portion of the run (i.e., up to the first normally closed valve), a terminal end of such runs is the piping connection to this closed valve.

3.6.4 REFERENCES

3.6-1 F.J. Moody, Fluid Reactions and Impingement Loads, Structural Design of Nuclear Plant Facilities, Vol. 1, (1973).

3.6-2 "Design for Pipe Break Effects," BN-TOP-2A, Bechtel Power Corporation.

3.6-3 GE Spec. No. 22A2625 - "System Criteria and Applications for Protection Against the Dynamic Effects of Pipe Break."

3.6-4 Relap 3 - A Computer Program for Reactor Blowdown Analysis IN-1321, issued June 1970, Reactor Technology TID-4500.

3.6-5 GE Report NEDE-10813 - "PDA - Pipe Dynamic Analysis Program for Pipe Rupture Movement." (Proprietary Filing)

3.6-6 Nuclear Services Corporation Report No. GEN-02-02, "Final Report Pipe Rupture Analysis of Recirculation System for 1969 Standard Plant Design."

3.6-7 Relap 4 - A computer program for transient Thermal Hydraulic analysis of Nuclear Reactors and Related Systems, ANCR-NUREG-1335, issued in September, 1976.

3.6-8 PIPRUP - A computer program for pipe Rupture Analysis, developed by nuclear services corporation (1977), Campbell, Ca.

3.6-9 "Design of Structures for Missile Impact," BC-TOP-9A, Revision 2, Bechtel Power Corporation, September 1974.

3.6-10 "COTTAP-4 (Compartment Transient Temperature Analysis Program)," EC-034-1019, Revision 0, Pennsylvania Power and Light Company, November, 1999.

TABLE 3.6-1

HIGH ENERGY FLUID SYSTEM PIPING

P&ID No.	Title	Description
M-101[(1)]	Main Steam	From nuclear boiler to the turbines, high pressure turbines to moisture/separator, to low pressure turbines. Main steam flow sensing line.
M-106[(1)]	Feedwater	From condensate demineralizers to feedwater heaters, to RF pumps, from RFP to nuclear boiler.
M-105[(1)]	Condensate	From condensate pump discharge to steam jet air ejector condenser, to steam packing exhausters, to condensate filters and to condensate demineralizers. From condensate demineralizer to control valves.
M-116[(1)]	Condensate Demineralizer	From condensate filters to condensate demineralizer to drain coolers.
M-141[(2)]	Nuclear Boiler	Main steam lines from reactor vessel to outside containment. From feedwater lines to reactor vessel. Main steam drains from the main steam lines to the condenser. Head vent line from the RPV head to the main steam "A" line.
M-142	Nuclear Boiler Instrumentation	Reactor Pressure Vessel pressure and level sensing lines, jet pump flow sensing lines and core delta ρ sensing lines.
M-143	Reactor Recirculation	Recirculation piping. From CRD to recirc. pump seal. Recirc flow sensing line.
M-144 M-145	Reactor Water Cleanup	From recirculation piping to cleanup pumps, through regenerative and non-regenerative heat exchangers, through cleanup Filter Demineralizer to feedwater.
M-146[(2)] & M-147	Control Rod Drive	From CRD pump discharge, to hydraulic control units, to control rod drives, to reactor vessel.

TABLE 3.6-1

HIGH ENERGY FLUID SYSTEM PIPING

P&ID No.	Title	Description
M-148	Standby Liquid Control	From isolation valve inside containment to reactor vessel.
M-149 & M-150	Reactor Core Isolation	From main steam to RCIC turbine stop valve, and drain pot. From RCIC Injection valve to feedwater line.
M-151	Residual Heat Removal	From recirc. piping to RHR inboard isolation valves, reactor vessel head spray, RHR flow sensing line.
M-152	Core Spray	From reactor vessel to inboard isolation valves.
M-155	High Pressure Coolant Injection	From main steam line to HPCI turbine stop valve, and drain pot. From HPCI Injection valve to feedwater line.

(1) High energy fluid system piping on these P&ID's are located in the Turbine Building. The components located in the Turbine Building (including safety related components) are not the essential systems and components that are required to operate to achieve safe shutdown following a high energy fluid system pipe break in the Turbine Building.

(2) High energy fluid system piping on these P&ID's are partially located in the Turbine Building.

Table Rev. 54

TABLE 3.6-2

Security-Related Information
Text Withheld Under 10 CFR 2.390

TABLE 3.6-2

Security-Related Information
Text Withheld Under 10 CFR 2.390

TABLE 3.6-3

Security-Related Information
Text Withheld Under 10 CFR 2.390

TABLE 3.6-4

RESTRAINT DATA

General Restraint Data for 1 Bar of a Restraint

$F = C_2 (\Delta restraint)^n$

Where $\Delta restraint = \delta pipe$ - Total clearance

Pipe Size (In)	Rest Load Direction	C_2	n	Limit ΔRestraint	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	5.084
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 3.6-5

COMPARISON OF PDA AND NSC CODE

Break Indent (Figure 3.6-15)	Restraint Indent (Figure 3.6-15)	No. of Bars		Load (kips)		Restraint Deflection (in.)		% of Design Restraint Deflection		Pipe Deflection (in.)	
		PDA	NSC	PDA	NSC	PDA	NSC	PDA	NSC	PDS	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 %	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
$RC4_{LL}$	RHR3	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
$PC6A_V$	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 RCR7	4 6	4 6	599.0 895.0	0 0	8.28 8.16	0 0	112.46% 110.76%	0 0	26.76 29.316	8.39
$RC9C_V$	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
PC13	RCR10	4	4	668.0	478.4	5.87	3.66	93.5 %	58.39%	13.37	10.44
PC16	RCR11	4	4	687.4	518.4	6.59	4.38	105 %	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 3.6-7

SUMMARY OF STRESS IN HIGH ENERGY
ASME CLASS 1 PIPING

MAIN STEAM LINE INSIDE CONTAINMENT
UNIT 2 – LINE “D”

Loop “A”/Loop “B” Node	Stress (ksi) (Eq. 10)	Cumulative Factor	Usage Pipe Break Stress Limit (ksi) (2.4 Sm)	Remarks
125 (tapered transition joint)	56.999	.6134	42.48	A
122 (elbow)	53.188	.1755	42.48	C
118 (elbow)	41.663	.1647	42.48	C
80 (tapered transition joint)	67.405	.9087	42.48	A
82 (elbow)	61.362	.1924	42.48	C
85 (elbow)	55.739	.1656	42.48	C
75 (tee)	115.915	.9489	42.48	C
40 (tapered transition joint)	72.600	.8174	42.48	A
42 (elbow)	65.367	.1708	42.48	C
45 (elbow)	59.112	.1773	42.48	C
35 (tee)	106.302	.6353	42.48	C
25 (tapered transition joint)	61.194	.1240	42.48	C
20 (tapered transition joint)	49.151	.0868	42.48	C**

Table 3.6-7

SUMMARY OF STRESS IN HIGH ENERGY
ASME CLASS 1 PIPING

MAIN STEAM LINE INSIDE CONTAINMENT
UNIT 2 – LINE “D”

Loop “A”/Loop “B” Node	Stress (ksi) (Eq. 10)	Cumulative Factor	Usage Pipe Break Stress Limit (ksi) (2.4 Sm)	Remarks
12 (tapered transition joint)	50.854	.0974	42.48	C**
10 (tapered transition joint)	50.307	.0932	42.48	A

NOTES:

A. Terminal End Break
B. Breakers determined by “Minimum Break Locations” Criteria
C. Breaks determined by Stress Requirement
D. See Figure 3.6-2 for Node Locations
* Highest values of either Loop “A” or “B”.
** These locations can be considered as arbitrary breaks based on criteria given in Section 3.6.2, however, original break classification is retained above.

Table 3.6-8

SUMMARY OF STRESS IN HIGH ENERGY
ASME CLASS 1 PIPING

HPCI STEAM SUPPLY LINE INSIDE CONTAINMENT
UNIT 1

Node	Stress (ksi) (Eq. 10)	Cumulative Factor	Usage Pipe Break Stress Limit (ksi) (2.4 Sm)	Remarks
405 (tapered transition)	45.754	.1209	42.10	A
411 (elbow)	66.674	.0836	42.10	B
425 (elbow)	50.053	.0292	42.10	B
431 (butt weld)	17.964	.0009	42.10	A

NOTES:

A. Terminal End Break
B. Breaks determined by “Minimum Break Location” Criteria
C. Breaks determined by Stress Requirement
D. See Figure 3.6-3 for Node Locations

Table 3.6-9

SUMMARY OF STRESS IN HIGH ENERGY
ASME CLASS 1 PIPING

RCIC STEAM SUPPLY LINE INSIDE CONTAINMENT
UNIT 1

Node	Stress (ksi) (Eq. 10)	Cumulative Factor	Usage Pipe Break Stress Limit (ksi) (2.4 Sm)	Remarks
643 (tapered transition joint)	78.272	.3638	42.10	A
644 (elbow)	56.466	.0081	42.10	B
652 (elbow)	39.408	.0139	42.10	C*
671 (butt weld)	15.679	.0002	42.10	A

NOTES:

A. Terminal End Break
B. Breaks determined by "Minimum Break Location" Criteria
C. Breaks determined by Stress Requirement
D. See Figure 3.6-4 for Node Locations
* These locations can be considered as arbitrary breaks based on criteria given in Section 3.6.2, however, original break classification is retained above.

TABLE 3.6-10

SUMMARY OF STRESS IN HIGH ENERGY ASME CLASS 1 PIPING

CORE SPRAY LINE INSIDE CONTAINMENT*
UNIT 1

Node	Stress (ksi) (Eq. 10)	Cumulative Factor	Usage Pipe Break Stress Limit (ksi) (2.4 Sm)	Remarks
10 (Tapered Transition Joint))	63.84	0.0843	40.02	A
15 (Reducer)	65.04	0.0831	40.02	C**
20 (Elbow Beginning - Butt Weld)	59.13	0.0209	40.02	C**
25 (Tapered Transition Joint)	41.14	0.0018	40.02	A

NOTES:

A. Terminal End Break
B. Breaks determined by "Minimum Break Location" Criteria
C. Breaks determined by Stress Requirement
D. See Figure 3.6-5 for Node Locations
* Highest values of either Loop "A" or "B".
** These locations can be considered as arbitrary breaks based on criteria given in Section 3.6.2; however, original break classification is retained above.

TABLE 3.6-11

SUMMARY OF STRESS IN HIGH ENERGY ASME CLASS 1 PIPING

RHR SUPPLY LINE INSIDE CONTAINMENT
UNIT 1

Node	Stress (ksi) (Eq. 10)	Cumulative Factor	Usage Pipe Break Stress Limit (ksi) (2.4 Sm)	Remarks
636 (butt weld)	33.52	.0008	40.20	A
636 (elbow end)	33.44	.0162	40.20	C
639 (tapered transition joint)	52.16	.0555	40.20	C
645 (tapered transition joint)	53.40	.0696	40.20	C
648 (tapered transition joint)	51.50	.0515	40.20	A

NOTES:

A. Terminal End Break
B. Breaks determined by "Minimum Break Location" Criteria
C. Breaks determined by Stress Requirement
D. See Figure 3.6-6 for Node Locations

TABLE 3.6-13

HIGH ENERGY FLUID SYSTEMS

WITH AND WITHOUT SUFFICIENT CAPACITY TO DEVELOP A JET STREAM

System	Reservoir Without Sufficient Capacity To Develop Jet Stream	Reservoir With Sufficient Capacity To Develop Jet Stream
Core Spray Inside Containment.	Between the postulated pipe break and the normally closed valve (Check Valve).	Between the postulated pipe break and the RPV.
Steam Supply To HPCI Turbine, Outside Containment.	Between the postulated pipe break and the normally closed valve HV1F001.	Between the postulated pipe and the RPV.
Steam Supply To RCIC Turbine, Outside Containment.	Between the postulated pipe break and the normally closed valve HV F045.	Between the postulated pipe break and the RPV.
Other High Energy Systems.	None.	All.

Table 3.6-14

SUMMARY OF STRESS IN HIGH ENERGY
ASME CLASS 1 PIPING

RECIRCULATION PIPING SYSTEM – LOOP “A”
UNIT 1

Node	Stress (ksi) (Eq. 10)	Cumulative Factor	Usage Pipe Break Stress Limit (ksi) (2.4 Sm)	Remarks
S1 (tapered transition joint)	46.53	.0061	40.02	A
F1 (sweepolet)	63.42*	.144*	40.02	C
F3 (sweepolet)	63.42*	.144*	40.02	C
F5 (cross)	69.85*	.199*	40.02	C
F7 (sweepolet)	63.42*	.144*	40.02	C
F9 (sweepolet)	63.42*	.144*	40.02	C
F2 (tapered transition joint)	46.93*	.039*	40.02	A
F4 (tapered transition joint)	46.93*	.039*	40.02	A
F6 (tapered transition joint)	46.93*	.039*	40.02	A
F8 (tapered transition joint)	46.93*	.039*	40.02	A
F10 (tapered transition joint)	46.93*	.039*	40.02	A

NOTES:

A. Terminal End Break
B. Breaks determined by “Minimum Break Location” Criteria
C. Breaks determined by Stress Requirement
D. See Figure 3.6-14 for Node Locations
* Envelope Value Represents Maximum Values at Similar Components and/or Node Locations in Both Loops.

Table 3.6-14

SUMMARY OF STRESS IN HIGH ENERGY
ASME CLASS 1 PIPING

RECIRCULATION PIPING SYSTEM – LOOP “B”
UNIT 1

S1 (tapered transition joint)	49.77	.0073	40.02	A
F1 (sweepolet)	63.42*	.144*	40.02	C
F3 (sweepolet)	63.42*	.144*	40.02	C
F5 (cross)	74.46	.199	40.02	C
F7 (sweepolet)	63.46	.144	40.02	C
F9 (sweepolet)	63.42*	.144	40.02	C
F2 (tapered transition joint)	46.93*	.039*	40.02	A
F4 (tapered transition joint)	46.93*	.039*	40.02	A
F6 (tapered transition joint)	46.93*	.039*	40.02	A
F8 (tapered transition joint)	46.93*	.039*	40.02	A
F10 (tapered transition joint)	46.93*	.039*	40.02	A
S2LL (tee)	74.99	.192	40.02	C

NOTES:

A. Terminal End Break
B. Breaks determined by “Minimum Break Location” Criteria
C. Breaks determined by Stress Requirement
D. See Figure 3.6-14 for Node Locations
* Envelope Value Represents Maximum Values at Similar Components and/or Node Locations in Both Loops.

Table 3.6-15

SUMMARY OF STRESS IN HIGH ENERGY
ASME CLASS 1 PIPING

RECIRCULATION PIPING SYSTEM – LOOP “A”
UNIT 2

Node	Stress (ksi) (Eq. 10)	Cumulative Factor	Usage Pipe Break Stress Limit (ksi) (2.4 Sm)	Remarks
425 (tapered transition joint)	46.63	.006	40.02	A
15 (sweepolet)	63.42*	.144*	40.02	C
45 (sweepolet)	63.88	.144*	40.02	C
70 (cross)	73.53	.199*	40.02	C
95 (sweepolet)	63.42	.144*	40.02	C
125 (sweepolet)	63.42*	.144*	40.02	C
155 (tapered transition joint)	46.93*	.039*	40.02	A
175 (tapered transition joint)	46.93*	.039*	40.02	A
200 (tapered transition joint)	46.93*	.039*	40.02	A
220 (tapered transition joint)	46.93*	.039*	40.02	A
240 (tapered transition joint)	46.93*	.039*	40.02	A

NOTES:

A. Terminal End Break
B. Breaks determined by “Minimum Break Location” Criteria
C. Breaks determined by Stress Requirement
D. See Figure 3.6-14 for Node Locations
* Envelope Value Represents Maximum Values at Similar Components and/or Node Locations in Both Loops

Table 3.6-15

SUMMARY OF STRESS IN HIGH ENERGY
ASME CLASS 1 PIPING

RECIRCULATION PIPING SYSTEM – LOOP “B”
UNIT 2

135 (sweepolet)	63.42*	.144*	40.02	C
105 (cross)	72.75	.199*	40.02	C
55 (sweepolet)	63.42	.144*	40.02	C
641 (tapered transition joint)	46.93*	.039*	40.02	A
626 (tapered transition joint)	46.93*	.039*	40.02	A
195 (tapered transition joint)	46.93*	.039*	40.02	A
59 (tapered transition joint)	46.93*	.039*	40.02	A
19 (tapered transition joint)	46.93*	.039*	40.02	A
400 (tee)	64.16	.0576	40.02	C
15 (sweepolet)	63.42*	.144*	40.02	C
425 (tapered transition joint)	46.49	.0069	40.02	A
150 (sweepolet)	63.42*	.144*	40.02	C

NOTES:

A. Terminal End Break
B. Breaks determined by “Minimum Break Location” Criteria
C. Breaks determined by Stress Requirement
D. See Figure 3.6-14 for Node Locations

* Envelope Value Represents Maximum Values at Similar Components and/or Node Locations in Both Loops.

TABLE 3.6-1a

Security-Related Information
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TABLE 3.6-6a

SUMMARY OF STRESS IN HIGH ENERGY
ASME CLASS 1 PIPING

MAIN STEAM LINE INSIDE CONTAINMENT
UNIT 1 - LINE "A"

Node	Stress (ksi) (Eq. 10)	Cumulative Factor	Usage Pipe Break Stress Limit (ksi) (2.4 Sm)	Remarks
8 (butt weld)	18.59	.0015	43.4	A
541 (sweepolet)	62.80	.2111	43.4	C
542 (sweepolet)	59.76	.0740	43.4	C
543 (sweepolet)	63.32	.1076	43.4	C
544 (sweepolet)	57.19	.0621	43.4	C
545 (sweepolet)	56.17	.0577	43.4	C
27 (tapered transition joint)	30.57	.0111	43.4	A

NOTES:

A. Terminal End Break
B. Breaks determined by "Minimum Break Location" Criteria
C. Breaks determined by Stress Requirement
D. See Figure 3.6-1A for Node Locations

Table 3.6-6b

SUMMARY OF STRESS IN HIGH ENERGY
ASME CLASS 1 PIPING

MAIN STEAM LINE INSIDE CONTAINMENT
UNIT 1 – LINE “B”

Node	Stress (ksi) (Eq. 10)	Cumulative Factor	Usage Pipe Break Stress Limit (ksi) (2.4 Sm)	Remarks
402 (tapered transition joint)	29.119	.0098	43.4	A
441 (sweepolet)	59.481	.0672	43.4	C
442 (sweepolet)	60.690	.0757	43.4	C
443 (sweepolet)	61.386	.0309	43.4	C
257 (butt weld)	19.783	.0016	43.4	A

NOTES:

A. Terminal End Break
B. Breaks determined by “Minimum Break Location” Criteria
C. Breaks determined by Stress Requirement
D. See Figure 3.6-1B for Node Locations

Table 3.6-6c

SUMMARY OF STRESS IN HIGH ENERGY
ASME CLASS 1 PIPING

MAIN STEAM LINE INSIDE CONTAINMENT
UNIT 1 – LINE "C"

Node	Stress (ksi) (Eq. 10)	Cumulative Factor	Usage Pipe Break Stress Limit (ksi) (2.4 Sm)	Remarks
508 (butt weld)	17.886	.0014	43.4	A
528 (sweepolet)	56.121	.0524	43.4	C
536 (sweepolet)	61.449	.0772	43.4	C
620 (sweepolet)	64.425	.0648	43.4	C
660 (tapered transition joint)	28.867	.0093	43.4	A

NOTES:
A. Terminal End Break
B. Breaks determined by "Minimum Break Location" Criteria
C. Breaks determined by Stress Requirement
D. See Figure 3.6-1C for Node Locations

Table 3.6-6d

SUMMARY OF STRESS IN HIGH ENERGY
ASME CLASS 1 PIPING

MAIN STEAM LINE INSIDE CONTAINMENT
UNIT 1 – LINE “D”

Node	Stress (ksi) (Eq. 10)	Cumulative Factor	Usage Pipe Break Stress Limit (ksi) (2.4 Sm)	Remarks
762 (butt weld)	18.267	.0015	43.4	A
782 (sweepolet)	68.242	.4155	43.4	C
850 (sweepolet)	63.177	.2956	43.4	C
860 (sweepolet)	59.512	.1696	43.4	C
870 (sweepolet)	57.465	.0948	43.4	C
880 (sweepolet)	55.616	.1348	43.4	C
778 (tapered transition joint)	30.683	.0117	43.4	A

NOTES:

A. Terminal End Break
B. Breaks determined by “Minimum Break Location” Criteria
C. Breaks determined by Stress Requirement
D. See Figure 3.6-1D for Node Locations

Table 3.6-6e

SUMMARY OF STRESS IN HIGH ENERGY
ASME CLASS 1 PIPING

MAIN STEAM LINE INSIDE CONTAINMENT
UNIT 2 – LINE “A”

Node	Stress (ksi) (Eq. 10)	Cumulative Factor	Usage Pipe Break Stress Limit (ksi) (2.4 Sm)	Remarks
8 (butt weld)	19.238	.0016	43.40	A
590 (sweepolet)	64.090	.1969	43.40	C
690 (sweepolet)	60.334	.1168	43.40	C
790 (sweepolet)	65.379	.1286	43.40	C
890 (sweepolet)	58.256	.0561	43.40	C
990 (sweepolet)	56.421	.0607	43.40	C
27 (tapered transition joint)	30.693	.0112	43.40	A

NOTES:

A. Terminal End Break
B. Breaks determined by “Minimum Break Location” Criteria
C. Breaks determined by Stress Requirement
D. See Figure 3.6-1A for Node Locations

Table 3.6-6f

SUMMARY OF STRESS IN HIGH ENERGY
ASME CLASS 1 PIPING

MAIN STEAM LINE INSIDE CONTAINMENT
UNIT 2 – LINE “B”

Node	Stress (ksi) (Eq. 10)	Cumulative Factor	Usage Pipe Break Stress Limit (ksi) (2.4 Sm)	Remarks
108 (tapered transition joint)	28.353	.0034	43.40	A
930 (sweepolet)	65.257	.4452	43.40	C
920 (sweepolet)	62.391	.1790	43.40	C
910 (sweepolet)	58.614	.0958	43.40	C
20 (butt weld)	19.960	.0014	43.40	A

NOTES:

A. Terminal End Break
B. Breaks determined by “Minimum Break Location” Criteria
C. Breaks determined by Stress Requirement
D. See Figure 3.6-1B for Node Locations

Table 3.6-6g

SUMMARY OF STRESS IN HIGH ENERGY
ASME CLASS 1 PIPING

MAIN STEAM LINE INSIDE CONTAINMENT
UNIT 2 – LINE “C”

Node	Stress (ksi) (Eq. 10)	Cumulative Factor	Usage Pipe Break Stress Limit (ksi) (2.4 Sm)	Remarks
508 (butt weld)	17.636	.0013	43.40	A
528 (sweepolet)	55.447	.0624	43.40	C
536 (sweepolet)	61.708	.0831	43.40	C
701 (sweepolet)	58.315	.0583	43.40	C
750 (tapered transition joint)	29.933	.0046	43.40	A

NOTES:

A. Terminal End Break
B. Breaks determined by “Minimum Break Location” Criteria
C. Breaks determined by Stress Requirement
D. See Figure 3.6-1C for Node Locations

Table 3.6-6h

SUMMARY OF STRESS IN HIGH ENERGY
ASME CLASS 1 PIPING

MAIN STEAM LINE INSIDE CONTAINMENT
UNIT 2 – LINE "D"

Node	Stress (ksi) (Eq. 10)	Cumulative Factor	Usage Pipe Break Stress Limit (ksi) (2.4 Sm)	Remarks
25 (butt weld)	18.724	.0015	43.40	A
75 (sweetpolet)	72.150	.6341	43.40	C
90 (sweetpolet)	65.942	.3880	43.40	C
105 (sweetpolet)	62.784	.2499	43.40	C
110 (sweetpolet)	56.323	.1737	43.40	C
120 (sweetpolet)	57.233	.1774	43.40	C
210 (tapered transition joint)	31.635	.0324	43.40	A

NOTES:

A. Terminal End Break
B. Breaks determined by "Minimum Break Location" Criteria
C. Breaks determined by Stress Requirement
D. See Figure 3.6-1D for Node Locations

Table 3.6-7a

SUMMARY OF STRESS IN HIGH ENERGY
ASME CLASS 1 PIPING

FEEDWATER LINE INSIDE CONTAINMENT*
UNIT 2

Loop “A”/Loop “B” Node	Stress (ksi) (Eq. 10)	Cumulative Factor	Usage Pipe Break Stress Limit (ksi) (2.4 Sm)	Remarks
450/125 (tapered transition joint)	55.682	.5767	42.48	A
443/122 (elbow)	52.425	.1754	42.48	C
425/118 (elbow)	40.867	.1658	42.48	C
202/80 (tapered transition joint)	70.954	.8845	42.48	A
200/82 (elbow)	63.855	.1850	42.48	C
192/86 (elbow)	48.417	.1653	42.48	C
75/75 (tee)	107.204	.9594	42.48	C
370/40 (tapered transition joint)	67.737	.9769	42.48	A
360/42 (elbow)	52.935	.1681	42.48	C
345/46 (elbow)	58.002	.1781	42.48	C
50/35 (tee)	87.447	.5927	42.48	C
35/25 (tapered transition joint)	61.124	.1027	42.48	C**
30/20 (tapered transition joint)	48.564	.0836	42.48	C**
20/12 (tapered transition joint)	50.389	.0934	42.48	C**

Table 3.6-7a

SUMMARY OF STRESS IN HIGH ENERGY
ASME CLASS 1 PIPING

FEEDWATER LINE INSIDE CONTAINMENT*
UNIT 2

Loop “A”/Loop “B” Node	Stress (ksi) (Eq. 10)	Cumulative Factor	Usage Pipe Break Stress Limit (ksi) (2.4 Sm)	Remarks
15/10 (tapered transition joint)	50.508	.0939	42.48	A**

NOTES:

A. Terminal End Break
B. Breaks determined by “Minimum Break Location” Criteria
C. Breaks determined by Stress Requirement
D. See Figure 3.6-2 for Node Locations
* Highest values of either Loop “A” or “B”.
** These locations can be considered as arbitrary breaks based on criteria given in Section 3.6.2; however, original break classification is retained above.

Table 3.6-8a

SUMMARY OF STRESS IN HIGH ENERGY
ASME CLASS 1 PIPING

HPCI STEAM SUPPLY LINE INSIDE CONTAINMENT
UNIT 2

Node	Stress (ksi) (Eq. 10)	Cumulative Factor	Usage Pipe Break Stress Limit (ksi) (2.4 Sm)	Remarks
304 (tapered transition joint)	57.878	.1835	42.10	A
310 (elbow)	66.600	.0911	42.10	B
327 (elbow)	60.766	.0823	42.10	B
341 (butt weld)	21.087	.0015	42.10	A

NOTES:

A. Terminal End Break
B. Breaks determined by “Minimum Break Location” Criteria
C. Breaks determined by Stress Requirement
D. See Figure 3.6-3 for Node Locations

Table 3.6-9a

SUMMARY OF STRESS IN HIGH ENERGY
ASME CLASS 1 PIPING

RCIC STEAM SUPPLY LINE INSIDE CONTAINMENT
UNIT 2

Node	Stress (ksi) (Eq. 10)	Cumulative Factor	Usage Pipe Break Stress Limit (ksi) (2.4 Sm)	Remarks
643 (tapered transition joint)	75.972	.4869	42.10	A
644 (elbow)	57.592	.0077	42.10	B
652 (elbow)	64.182	.1046	42.10	C
676 (butt weld)	13.258	.0000	42.10	A

NOTES:

A. Terminal End Break
B. Breaks determined by “Minimum Break Location” Criteria
C. Breaks determined by Stress Requirement
D. See Figure 3.6-4 for Node Locations

TABLE 3.6-10a

SUMMARY OF STRESS IN HIGH ENERGY ASME CLASS 1 PIPING

CORE SPRAY LINE INSIDE CONTAINMENT
UNIT 2

Node*	Stress (ksi) (Eq. 10)	Cumulative Factor	Usage Pipe Break Stress Limit (ksi) (2.4 Sm)	Remarks
10 (Tapered Transition Joint)	64.11	0.0912	40.02	A
15 (Reducer)	67.09	0.1093	40.02	C
20 (Elbow)	57.09	0.0210	40.02	C**
25 (Tapered Transition Joint)	40.29	0.0015	40.02	A

NOTES:

A. Terminal End Break
B. Breaks determined by "Minimum Break Location" Criteria
C. Breaks determined by Stress Requirement
D. See Figure 3.6-5 for Node Locations
* Highest values of either Loop "A" or "B".
** These locations can be considered as arbitrary breaks based on criteria given in Section 3.6.2; however, original break classification is retained above.

TABLE 3.6-11a

SUMMARY OF STRESS IN HIGH ENERGY ASME CLASS 1 PIPING

RHR SUPPLY LINE INSIDE CONTAINMENT

UNIT 2

Node	Stress (ksi) (Eq. 10)	Cumulative Factor	Usage Pipe Break Stress Limit (ksi) (2.4 Sm)	Remarks
435 (butt weld)	33.39	.008	40.20	A
435 (end of elbow)	52.22	.0156	40.20	C*
445 (tapered transition joint)	50.75	.0466	40.20	C*
465 (tapered transition joint)	53.56	.0779	40.20	C
485 (tapered transition joint)	51.53	.052	40.20	A

NOTES:

A. Terminal End Break
B. Breaks determined by "Minimum Break Location" Criteria
C. Breaks determined by Stress Requirement
D. See Figure 3.6-6 for Node Locations
* These locations can be considered as arbitrary breaks based on criteria given in Section 3.6.2; however, original break classification is retained above.

Table 3.6-12a

SUMMARY OF STRESS IN HIGH ENERGY
ASME CLASS 1 PIPING

RHR RETURN LINE INSIDE CONTAINMENT
UNIT 1 – LOOP "A"

Node	Stress (ksi) (Eq. 10)	Cumulative Factor	Usage Pipe Break Stress Limit (ksi) (2.4 Sm)	Remarks
125 (tapered transition joint)	54.04	.114	40.20	A
132 (tapered transition joint)	48.15	.068*	40.20	B
143 (tapered transition joint)	48.52	.066*	40.20	A

NOTES:

A. Terminal End Break
B. Breaks determined **by** "Minimum Break Location" Criteria
C. Breaks determined **by** Stress Requirement
D. See Figure 3.6-7 for Node Locations

Envelope Value Represents Maximum Values at Similar Components And/Or Node Locations in Both Loops.

Table 3.6-12b

SUMMARY OF STRESS IN HIGH ENERGY
ASME CLASS 1 PIPING

RHR RETURN LINE INSIDE CONTAINMENT
UNIT 1 – LOOP "B"

Node	Stress (ksi) (Eq. 10)	Cumulative Factor	Usage Pipe Break Stress Limit (ksi)	Remarks
684 (tapered transition ioint)	52.82	.088	40.20	A
690 (tapered transition joint)	50.01	.068*	40.20	B
698 (tapered transition joint)	49.13	.066*	40.20	A

NOTES:

A. Terminal End Break
B. Breaks determined **by** "Minimum Break Location" Criteria
C. Breaks determined **by** Stress Requirement
D. See Figure 3.6-8 for Node Locations

Envelope Value Represents Maximum Values at Similar Components And/Or Node Locations in Both Loops.

TABLE 3.6-12c

SUMMARY OF STRESS IN HIGH ENERGY ASME CLASS 1 PIPING

HEAD SPRAY LINE INSIDE CONTAINMENT
UNIT 1

Node	Stress (ksi) (Eq. 10)	Cumulative Factor	Usage Pipe Break Stress Limit (ksi) (2.4 Sm)	Remarks
5 (taper transition joint)	69.34	.4376	39.76	A
12 (taper transition joint)	58.71	.1162	39.76	C
14 (taper transition joint)	56.12	.0798	39.76	C*
15 (taper transition joint)	52.65	.0482	39.76	A

NOTES:

A. Terminal End Break
B. Breaks determined by "Minimum Break Location" Criteria
C. Breaks determined by Stress Requirement
D. See Figure 3.6-8C for Node Locations
* These locations can be considered as arbitrary breaks based on criteria given in Section 3.6.2, however, original break classification is retained above.

Table 3.6-12a.1

SUMMARY OF STRESS IN HIGH ENERGY
ASME CLASS 1 PIPING

REACTOR WATER CLEAN UP LINE INSIDE CONTAINMENT
UNIT 1

Node	Stress (ksi) (Eq. 10)	Cumulative Factor	Usage Pipe Break Stress Limit (ksi) (2.4 Sm)	Remarks
173 (butt weld)	42.23	.0123	42.864	C*
222 (tapered transition joint)	41.41	.0052	34.27	A
318 (tapered transition joint)	36.68	.0039	34.27	A
802 (butt weld)	51.18	.0687	34.27	A
808 (reducer)	54.38	.0251	34.27	C
822 (socket weld)	23.65	.0002	34.27	C
804 (tee)	77.98	.6140	34.27	C
842 (elbow)	60.16	.0272	34.27	C

NOTES:

A. Terminal End Break
B. Breaks determined by “Minimum Break Location” Criteria
C. Breaks determined by Stress Requirement
D. See Figure 3.6-8A.1 for Node Locations
* These locations can be considered as arbitrary breaks based on criteria given in Section 3.6.2, however, original break classification is retained above.

Table 3.6-12a.2

SUMMARY OF STRESS IN HIGH ENERGY
ASME CLASS 1 PIPING

REACTOR WATER CLEAN UP LINE INSIDE CONTAINMENT
UNIT 1

Node	Stress (ksi) (Eq. 10)	Cumulative Factor	Usage Pipe Break Stress Limit (ksi) (2.4 Sm)	Remarks
5 (butt weld)	10.52	.0053	42.864	A
59 (tee)	38.78	.0158	34.27	C*
504 (butt weld)	40.04	.0365	34.27	C*
61 (butt weld)	43.11	.0208	34.27	C*
710 (socket weld)	37.12	.0026	34.27	A
153 (curb)	13.64	.0012	42.864	A
80 (tee)	47.96	.0486	42.864	C*

NOTES:

A. Terminal End Break
B. Breaks determined by “Minimum Break Location” Criteria
C. Breaks determined by Stress Requirement
D. See Figure 3.6-8A.1,3-6-8A.2, 3.6-8A.3 for Node Locations
* These locations can be considered as arbitrary breaks based on criteria given in Section 3.6.2, however, original break classification is retained above.

Table 3.6-12a.3

SUMMARY OF STRESS IN HIGH ENERGY
ASME CLASS 1 PIPING

REACTOR WATER CLEAN UP LINE INSIDE CONTAINMENT
UNIT 1

Node	Stress (ksi) (Eq. 10)	Cumulative Factor	Usage Pipe Break Stress Limit (ksi) (2.4 Sm)	Remarks
846 (elbow)	65.08	.0468	34.27	C
301 (butt weld)	40.36	.0040	34.27	A
320 (elbow)	48.08	.0013	34.27	C*
330 (elbow)	47.33	.0012	34.27	C*
335 (tee)	53.43	.0278	34.27	C*
340 (reducer)	49.09	.0072	34.27	C
352 (socket weld)	31.22	.0007	34.27	A
850 (elbow)	53.78	.0090	34.27	C

NOTES:

A. Terminal End Break
B. Breaks determined by “Minimum Break Location” Criteria
C. Breaks determined by Stress Requirement
D. See Figure 3.6-8A.1, for Node Locations
* These locations can be considered as arbitrary breaks based on criteria given in Section 3.6.2, however, original break classification is retained above.

Table 3.6-12a.4

SUMMARY OF STRESS IN HIGH ENERGY
ASME CLASS 1 PIPING

REACTOR WATER CLEAN-UP LINE INSIDE CONTAINMENT
UNIT 2

Node	Stress (ksi) (Eq. 10)	Cumulative Factor	Usage Pipe Break Stress Limit (ksi) (2.4 Sm)	Remarks
432 (butt weld)	40.71	.0043	34.27	A
435 (tee)	55.31	.0283	34.27	C
441 (reducer)	51.76	.0118	34.27	C
460 (socket weld)	27.01	.0002	34.27	A
510 (elbow)	47.28	.0012	34.27	C*
535 (elbow)	46.20	.0010	34.27	C*
551 (tapered transition joint)	64.07	.4768	34.27	A
610 (butt weld)	51.57	.0917	34.27	A
615 (tee)	86.601	.794	34.27	C
705 (elbow)	64.52	.1002	34.27	C
695 (elbow)	59.63	.0226	34.27	C
715 (elbow)	54.54	.0074	34.27	C

NOTES:

A. Terminal End Break
B. Breaks determined by “Minimum Break Location” Criteria
C. Breaks determined by Stress Requirement
D. See Figure 3.6-8A.1, for Node Locations
* These locations can be considered arbitrary breaks based on criteria given in Section 3.6.2; however, original break classification is retained above.

Table 3.6-12a.5

SUMMARY OF STRESS IN HIGH ENERGY
ASME CLASS 1 PIPING

REACTOR WATER CLEAN-UP LINE INSIDE CONTAINMENT
UNIT 2

Node	Stress (ksi) (Eq. 10)	Cumulative Factor	Usage Pipe Break Stress Limit (ksi) (2.4 Sm)	Remarks
735 (tapered transition)	72.27	.9975	34.27	A
619 (reducer)	72.67	.6331	34.27	C
635 (socket weld)	21.83	.0001	34.27	C

NOTES:

A. Terminal End Break
B. Breaks determined by “Minimum Break Location” Criteria
C. Breaks determined by Stress Requirement
D. See Figure 3.6-8A.1 for Node Locations

Table 3.6-12a.6

SUMMARY OF STRESS IN HIGH ENERGY
ASME CLASS 1 PIPING

RHR RETURN LINE INSIDE CONTAINMENT
UNIT 2 – LOOP "A"

Node	Stress (ksi) (Eq. 10)	Cumulative Factor	Usage Pipe Break Stress Limit (ksi)	Remarks
640 (tapered transition joint)	54.31	,1179	40.20	A
655 (tapered transition joint)	49.30	.068*	40.20	B
680 (tapered transition joint)	49.49	.066*	40.20	A

NOTES:

A. Terminal End Break
B. Breaks determined **by** "Minimum Break Location" Criteria
C. Breaks determined **by** Stress Requirement
D. See Figure 3.6-7 for Node Locations

Envelope Value Represents Maximum Values at Similar Components And/Or Node Locations in Both Loops.

Table 3.6-12a.7

SUMMARY OF STRESS IN HIGH ENERGY ASME CLASS 1 PIPING

REACTOR WATER CLEAN-UP LINE INSIDE CONTAINMENT UNIT 2

Node	Stress (ksi) (Eq. 10)	Cumulative Factor	Usage Pipe Break Stress Limit (ksi) (2.4 Sm)	Remarks
248 (butt weld)	38.33	0.0059	42.864	B
5 (butt weld)	23.90	0.0104	42.864	A
102 (tree)	48.81	0.0623	34.27	B
101 (butt weld)	42.46	0.0385	34.27	B
515 (socket weld)	24.07	0.0003	34.27	A
154 (curve end)	13.88	0.0013	42.864	A
235 (tee)	46.50	0.0460	42.864	B
55 (tee)	48.51	0.0291	34.27	C*
281 (reducer)	42.75	0.0092	42.864	B

NOTES:

A. Terminal End Break
B. Breaks determined by “Minimum Break Location” Criteria
C. Breaks determined by Stress Requirement
D. See Figure 3.6-8A.1,3-6-8A.4, 3.6-8A.5 for Node Locations
* These locations can be considered arbitrary breaks based on criteria given in Section 3.6.2; however, original break classification is retained above.

TABLE 3.6-12b.1

SUMMARY OF STRESS IN HIGH ENERGY ASME CLASS 1 PIPING

HEAD VENT LINE INSIDE CONTAINMENT
UNIT 1

Node	Stress (ksi) (Eq. 10)	Cumulative Factor	Usage Pipe Break Stress Limit (ksi) (2.4 Sm)	Remarks
117 (red)	22.97	0.0	33.60	A
145 (taper transition joint)	27.89	.0025	42.10	A
254 (socket weld)	36.77	.1239	42.10	C
256 (socket weld)	30.85	.0824	42.10	C*
260 (socket weld)	28.94	.0666	42.10	A
358 (socket weld)	23.79	.0504	42.10	C*
365 (straight pipe)	13.64	0.0	42.10	A
408 (socket weld)	51.34	.0565	42.10	C*

NOTES:

A. Terminal End Break
B. Breaks determined by "Minimum Break Location" Criteria
C. Breaks determined by Stress Requirement
D. See Figure 3.6-8B for Node Locations
* These locations can be considered as arbitrary breaks based on criteria given in Section 3.6.2, however, original break classification is retained above.

TABLE 3.6-12b.2

SUMMARY OF STRESS IN HIGH ENERGY ASME CLASS 1 PIPING

HEAD VENT LINE INSIDE CONTAINMENT
UNIT 1

Node	Stress (ksi) (Eq. 10)	Cumulative Factor	Usage Pipe Break Stress Limit (ksi) (2.4 Sm)	Remarks
410 (socket weld)	55.96	.0846	42.10	C*
420 (socket weld)	54.09	.0682	42.10	A
613 (reducer)	24.31	.0001	33.60	A

NOTES:

A. Terminal End Break
B. Breaks determined by "Minimum Break Location" Criteria
C. Breaks determined by Stress Requirement
D. See Figure 3.6-8B for Node Locations
* These locations can be considered as arbitrary breaks based on criteria given in Section 3.6.2, however, original break classification is retained above.

Table 3.6-12b.3

SUMMARY OF STRESS IN HIGH ENERGY
ASME CLASS 1 PIPING

RHR RETURN LINE INSIDE CONTAINMENT
UNIT 2 – LOOP "B"

Node	Stress (ksi) (Eq. 10)	Cumulative Factor	Usage Pipe Break Stress Limit (ksi)	Remarks
815 (tapered transition joint)	53.28	,0941	40.20	A
840 (tapered transition joint)	49.73	.068*	40.20	B
860 (tapered transition joint)	49.08	.066*	40.20	A

NOTES:

A. Terminal End Break
B. Breaks determined **by** "Minimum Break Location" Criteria
C. Breaks determined **by** Stress Requirement
D. See Figure 3.6-8 for Node Locations

Envelope Value Represents Maximum Values at Similar Components And/Or Node Locations in Both Loops.

TABLE 3.6-12b.4

SUMMARY OF STRESS IN HIGH ENERGY ASME CLASS 1 PIPING

HEAD VENT LINE INSIDE CONTAINMENT
UNIT 2

Node	Stress (ksi) (Eq. 10)	Cumulative Factor	Usage Pipe Break Stress Limit (ksi) (2.4 Sm)	Remarks
422 (red)	24.87	.0001	33.60	A
7 (tapered transition joint)	28.30	.0028	42.10	A
618 (reducer)	25.08	.0001	33.60	A
260 (socket weld)	17.67	.0098	42.10	A
363 (straight pipe)	21.08	.0001	42.10	A
716 (socket weld)	46.13	.0358	42.10	A
251 (socket weld)	24.33	.0161	42.10	C*
249 (socket weld)	35.70	.0873	42.10	C*
358 (socket weld)	23.45	.1063	42.10	C
708 (socket weld)	47.24	.0370	42.10	C*
712 (socket weld)	51.47	.0420	42.10	C*

NOTES:

A. Terminal End Break
B. Breaks determined by "Minimum Break Location" Criteria
C. Breaks determined by Stress Requirement
D. See Figure 3.6-8B for Node Locations
* These locations can be considered as arbitrary breaks based on criteria given in Section 3.6.2; however, original break classification is retained.

TABLE 3.6-12c.1

SUMMARY OF STRESS IN HIGH ENERGY ASME CLASS 1 PIPING

HEAD SPRAY LINE INSIDE CONTAINMENT

UNIT 2

Node	Stress (ksi) (Eq. 10)	Cumulative Factor	Usage Pipe Break Stress Limit (ksi) (2.4 Sm)	Remarks
7 (tapered transition joint)	87.95	.9195	39.76	A
12 (tapered transition joint)	62.47	.2222	39.76	C
14 (tapered transition joint)	58.66	.1361	39.76	C
30 (tapered transition joint)	54.56	.0672	39.76	A

NOTES:

A. Terminal End Break
B. Breaks determined by "Minimum Break Location" Criteria
C. Breaks determined by Stress Requirement
D. See Figure 3.6-8C for Node Locations

TABLE 3.6-12d.1

SUMMARY OF STRESS IN HIGH ENERGY ASME CLASS 1 PIPING

STANDBY LIQUID CONTROL LINE INSIDE CONTAINMENT
UNIT 1

Node	Stress (ksi) (Eq. 10)	Cumulative Factor	Usage Pipe Break Stress Limit (ksi) (2.4 Sm)	Remarks
32 (socket weld)	37.36	.0087	34.080	B
25 (socket weld)	43.98	.0678	34.080	A
230 (anchor)	10.72	0.0	34.080	A
71 (socket weld)	31.95	.0055	34.080	B
50 (socket weld)	33.37	.0056	34.080	A
192 (socket weld)	37.57	.0075	34.080	B
203 (socket weld)	34.33	.0060	34.080	B

NOTES:

A. Terminal End Break
B. Breaks determined by "Minimum Break Location" Criteria
C. Breaks determined by Stress Requirement
D. See Figure 3.6-8D for Node Locations

TABLE 3.6-12d.2

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TABLE 3.6-12d.3

SUMMARY OF STRESS IN HIGH ENERGY ASME CLASS 1 PIPING

STANDBY LIQUID CONTROL LINE INSIDE CONTAINMENT
UNIT 2

Node	Stress (ksi) (Eq. 10)	Cumulative Factor	Usage Pipe Break Stress Limit (ksi) (2.4 Sm)	Remarks
32 (socket weld)	36.63	.0081	34.080	B
25 (socket weld)	46.68	.1245	34.080	A
230 (anchor)	8.98	0.0	34.080	A
71 (socket weld)	32.76	.0049	34.080	B
50 (socket weld)	33.37	.0056	34.080	A
192 (socket weld)	37.57	.0075	34.080	B
203 (socket weld)	34.33	.0060	34.080	B

NOTES:

A. Terminal End Break
B. Breaks determined by "Minimum Break Location" Criteria
C. Breaks determined by Stress Requirement
D. See Figure 3.6-8D for Node Locations

TABLE 3.6-12e.1

SUMMARY OF STRESS IN HIGH ENERGY ASME CLASS 1 PIPING

MSIV DRAIN LINES INSIDE CONTAINMENT

UNIT 1

Node	Stress (ksi) (Eq. 10)	Cumulative Factor	Usage Pipe Break Stress Limit (ksi) (2.4 Sm)	Remarks
113 (socket weld)	53.05	.0487	42.10	A
112 (elbow)	52.44	.0038	42.10	B
35 (tee)	27.23	.0039	42.10	A
41 (tee)	34.33	.0059	42.10	B
43 (elbow)	34.50	.0008	42.10	B
66 (elbow)	43.83	.0066	42.10	B
51 (socket weld)	45.84	.0337	42.10	A

NOTES:

A. Terminal End Break
B. Breaks determined by "Minimum Break Location" Criteria
C. Breaks determined by Stress Requirement
D. See Figure 3.6-8E for Node Locations

TABLE 3.6-12e.2

SUMMARY OF STRESS IN HIGH ENERGY ASME CLASS 1 PIPING

MSIV DRAIN LINES INSIDE CONTAINMENT
UNIT 1

Node	Stress (ksi) (Eq. 10)	Cumulative Factor	Usage Pipe Break Stress Limit (ksi) (2.4 Sm)	Remarks
67 (socket weld)	40.52	.0157	42.10	A
93 (elbow)	35.42	.0067	42.10	B
95 (elbow)	36.03	.0071	42.10	B
99 (socket weld)	42.08	.0298	42.10	A
106 (elbow)	37.99	.0012	42.10	B

NOTES:

A. Terminal End Break
B. Breaks determined by "Minimum Break Location" Criteria
C. Breaks determined by Stress Requirement
D. See Figure 3.6-8E for Node Locations

TABLE 3.6-12e.3

SUMMARY OF STRESS IN HIGH ENERGY ASME CLASS 1 PIPING

MSIV DRAIN LINES INSIDE CONTAINMENT
UNIT 2

Node	Stress (ksi) (Eq. 10)	Cumulative Factor	Usage Pipe Break Stress Limit (ksi) (2.4 Sm)	Remarks
113 (socket weld)	53.05	.0487	42.10	A
112 (elbow)	52.44	.0038	42.10	B
106 (elbow)	37.99	.0012	42.10	B
93 (elbow)	35.42	.0067	42.10	B
95 (elbow)	36.03	.0071	42.10	B
99 (socket weld)	42.08	.0298	42.10	A
35 (tee)	27.23	.0039	42.10	A
41 (tee)	34.33	.0059	42.10	B
43 (elbow)	34.50	.0008	42.10	B
51 (socket weld)	45.84	.0337	42.10	A
66 (elbow)	43.83	.0066	42.10	B
67 (socket weld)	40.52	.0157	42.10	A

NOTES:

A. Terminal End Break
B. Breaks determined by "Minimum Break Location" Criteria
C. Breaks determined by Stress Requirement
D. See Figure 3.6-8E for Node Locations

Security-Related Information
Figure Withheld Under 10 CFR 2.390

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
FEEDWATER SYSTEM
FIGURE 3.6-2, Rev. 54

Auto-Cad Figure Fsar 3_6_2.dwg

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Figure Withheld Under 10 CFR 2.390

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
HPCI STEAM SUPPLY
FIGURE 3.6-3, Rev. 54

Auto-Cad Figure Fsar 3_6_3.dwg

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SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

RCIC STEAM SUPPLY

FIGURE 3.6-4, Rev. 54

Auto-Cad Figure Fsar 3_6_4.dwg

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SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

CORE SPRAY

FIGURE 3.6-5, Rev. 54

Auto-Cad Figure Fsar 3_6_5.dwg

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SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

RHR SUPPLY

FIGURE 3.6-6, Rev. 54

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SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

RHR RETURN LOOP 'A'

FIGURE 3.6-7, Rev. 54

Auto-Cad Figure Fsar 3_6_7.dwg

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Figure Withheld Under 10 CFR 2.390

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

RHR RETURN LOOP 'B'

FIGURE 3.6-8, Rev. 54

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SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
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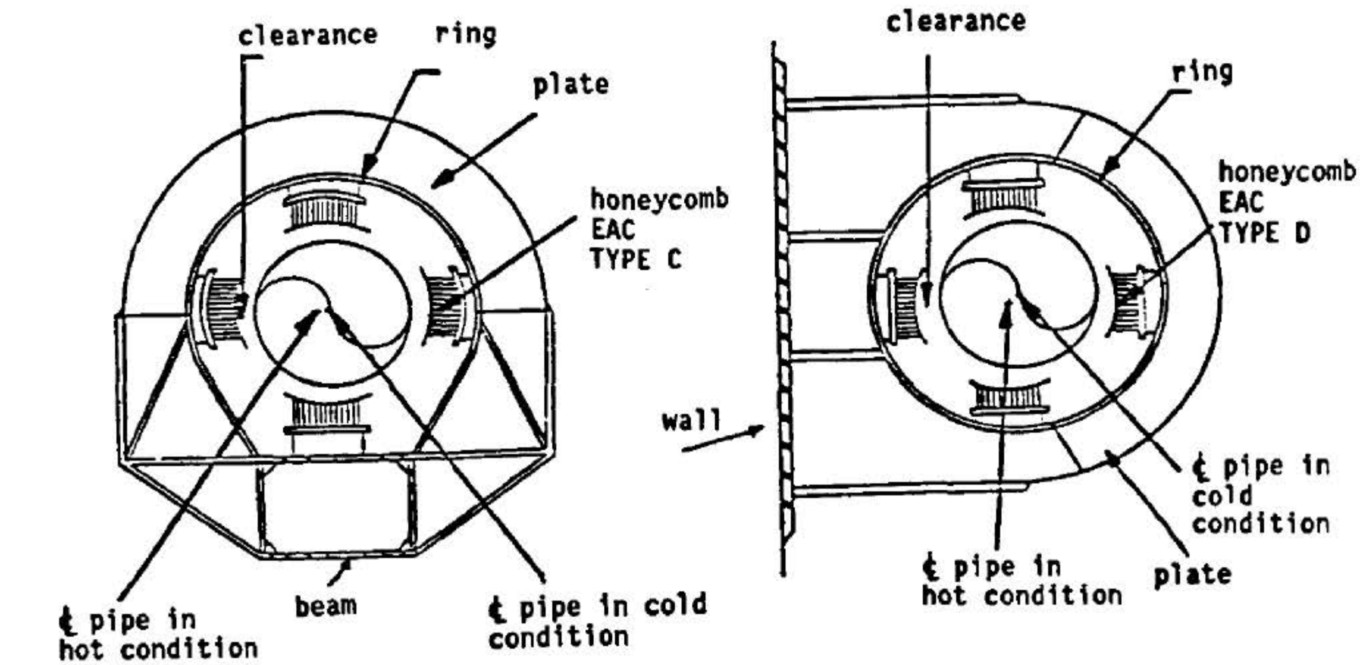
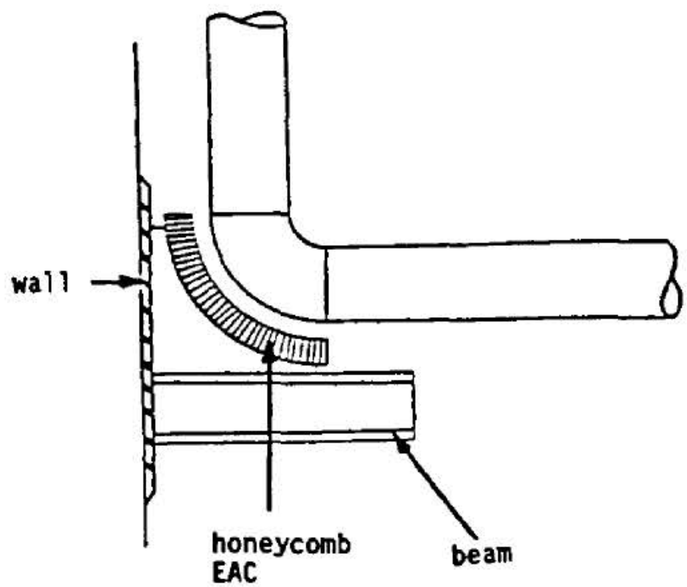
FIGURE 3.6-9, Rev. 47

AutoCAD Figure 3_6_9.doc



a) Pipe whip restraints with honeycomb (EAC) and clearance all around the pipe



b) Bumpers

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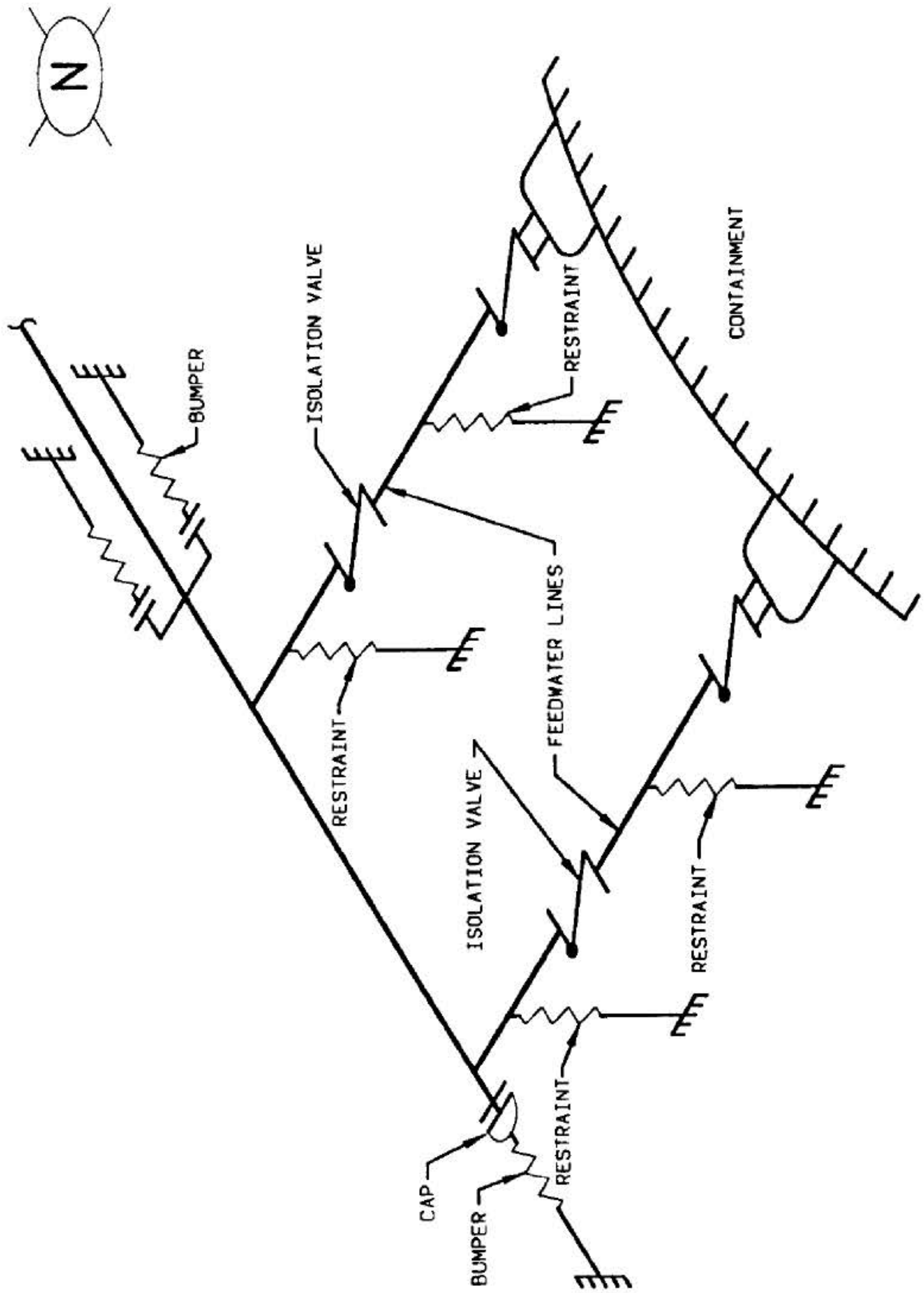
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

TYPICAL PIPE
WHIP RESTRAINTS

FIGURE 3.6-10, Rev. 47

Auto-Cad Figure Fsar 3_6_10.dwg



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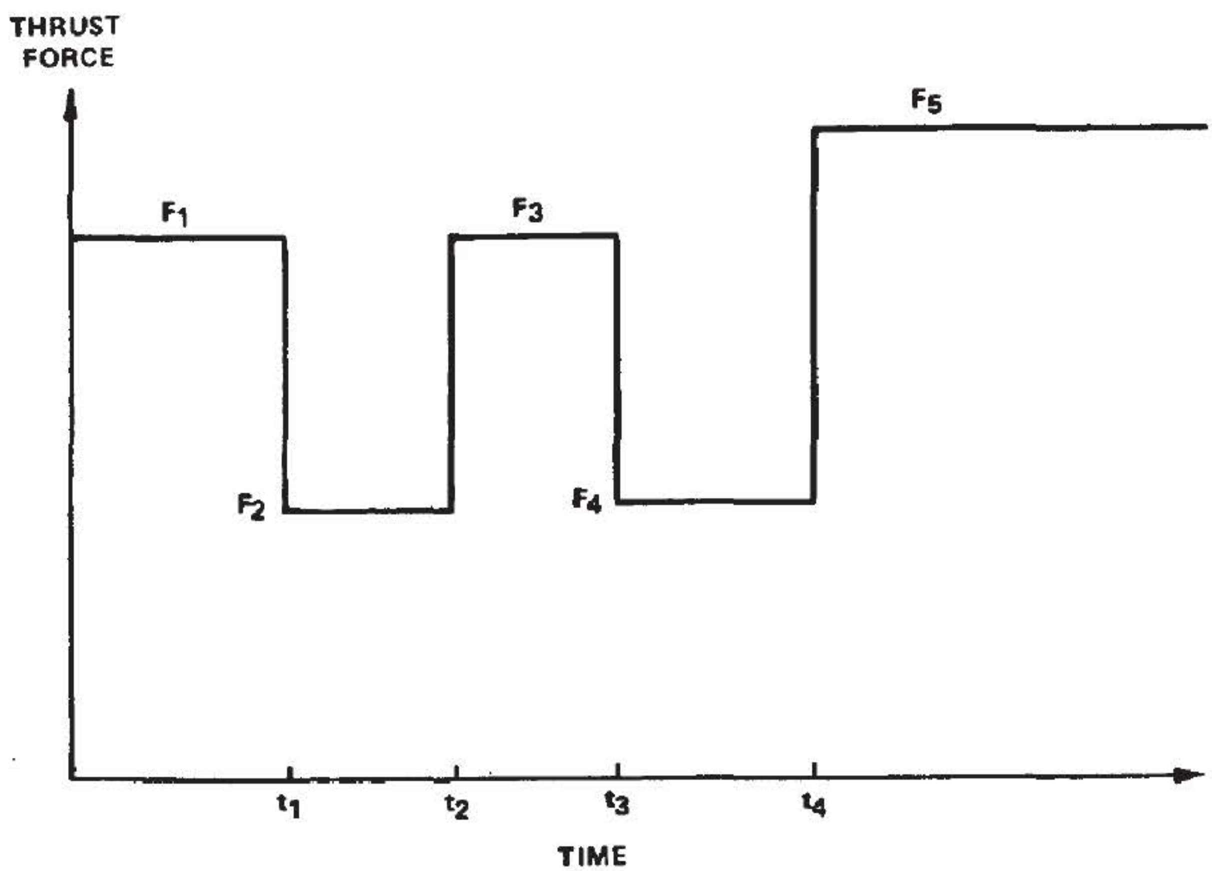
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

PIPE WHIP RESTRAINT
ARRANGEMENT TO PROTECT
FEEDWATER OUTSIDE
CONTAINMENT ISOLATION VALVES

FIGURE 3.6-11, Rev. 47

Auto-Cad Figure Fsar 3_6_11.dwg



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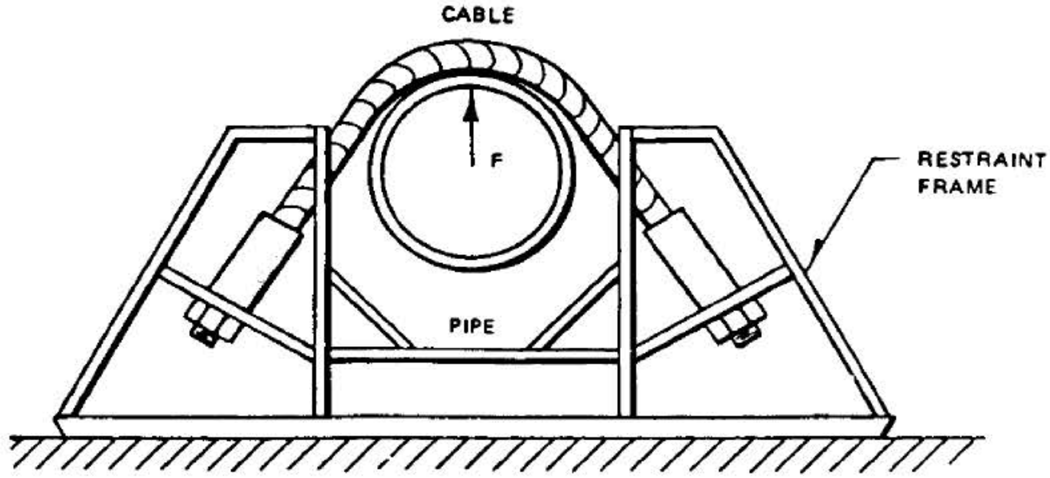
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

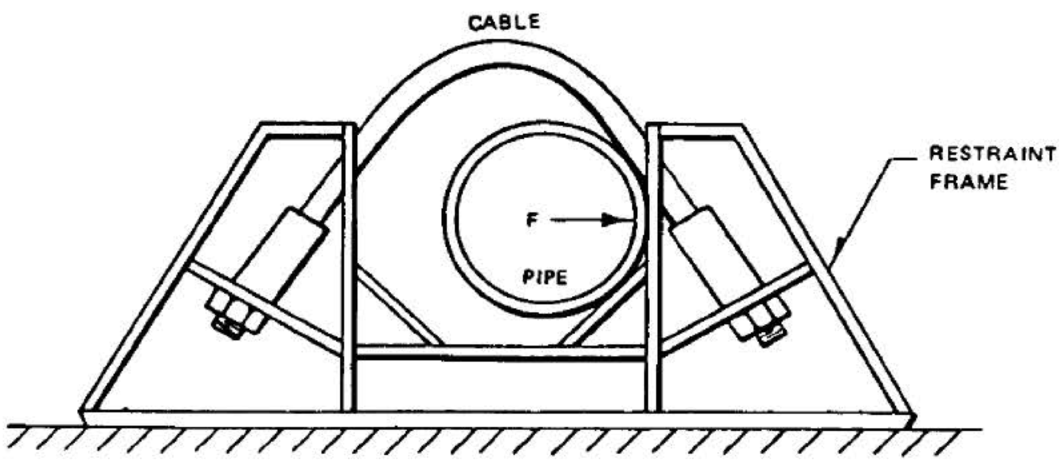
FORCING FUNCTIONS MODEL
ASSOCIATED WITH PIPE WHIP
DYNAMIC ANALYSIS

FIGURE 3.6-12, Rev. 47

Auto-Cad Figure Fsar 3_6_12.dwg



(a) LOAD APPLIED PERPENDICULAR TO RESTRAINT BASE AGAINST CABLES



(b) LOAD APPLIED PARALLEL TO FRAME BASE AGAINST ONE SIDE OF RESTRAINT FRAME

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SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

TYPICAL PIPE WHIP
RESTRAINT CONFIGURATION

FIGURE 3.6-13, Rev. 47

Auto-Cad Figure Fsar 3_6_13.dwg

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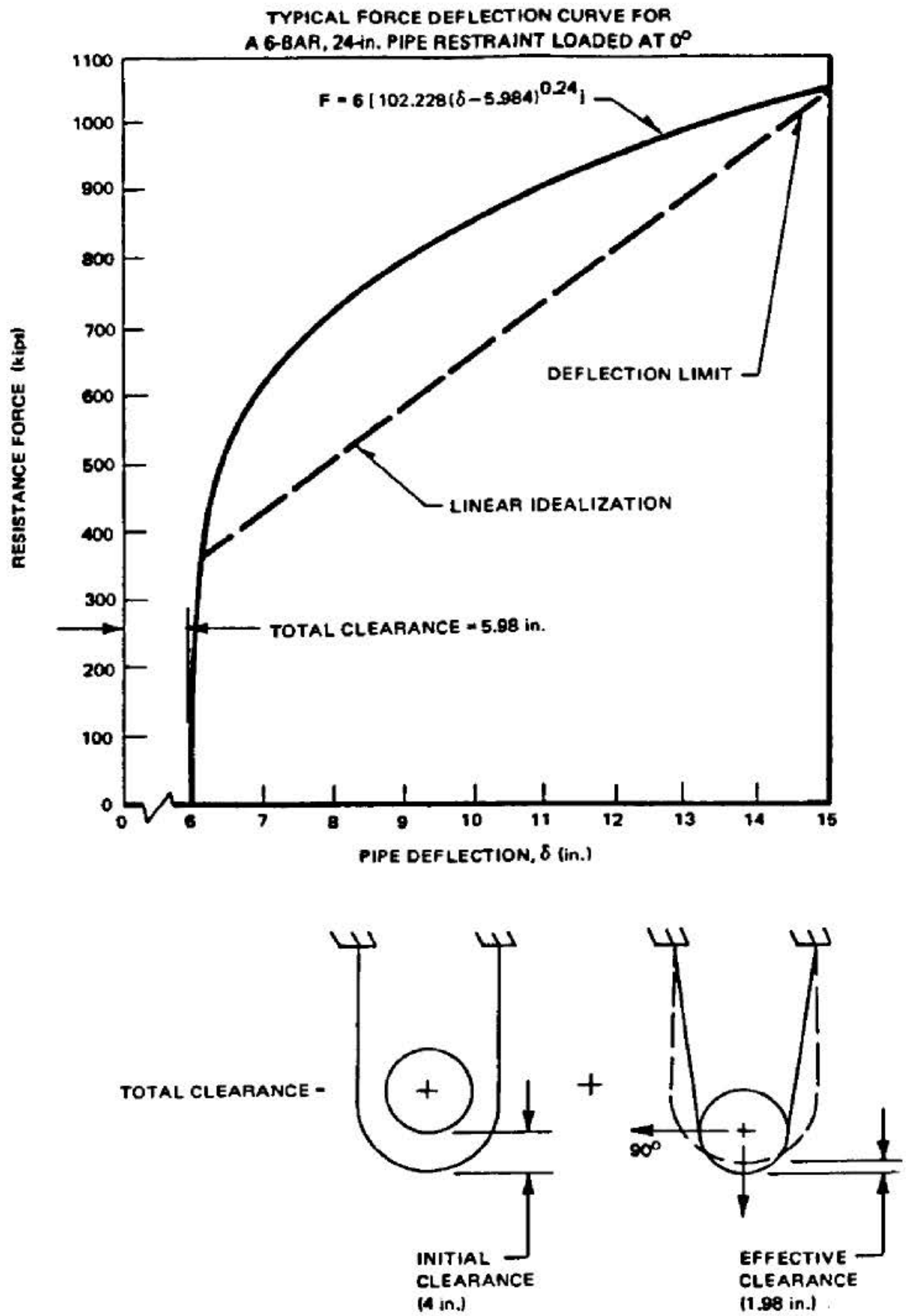
SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
RECIRCULATION SYSTEM POSTULATED BREAK LOCATIONS AND RESTRAINT LOCATIONS (LOOP A AND B SAME, UNLESS OTHERWISE SPECIFIED) FIGURE 3.6-14, Rev. 50

Auto-Cad Figure Fsar 3_6_14.dwg



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SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

TYPICAL RESTRAINT FORCE
DEFLECTION CURVE

FIGURE 3.6-15, Rev. 47

Auto-Cad Figure Fsar 3_6_15.dwg

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Figure Withheld Under 10 CFR 2.390

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
BREAK LOCATIONS AND RESTRAINTS ANALYZED PDA VERIFICATION PROGRAM FIGURE 3.6-16, Rev. 47

Auto-Cad Figure Fsar 3_6_16.dwg

Security-Related Information
Figure Withheld Under 10 CFR 2.390

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
MAIN STEAM LINE 'A'
FIGURE 3.6-1A, Rev. 54

Auto-Cad Figure Fsar 3_6_1A.dwg

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SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

MAIN STEAM LINE 'B'

IGURE 3.6-1B, Rev. 54

Auto-Cad Figure Fsar 3_6_1B.dwg

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Figure Withheld Under 10 CFR 2.390

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

MAIN STEAM LINE 'C'

FIGURE 3.6-1C, Rev. 54

Auto-Cad Figure Fsar 3_6_1C.dwg

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SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

MAIN STEAM LINE 'D'

FIGURE 3.6-1D, Rev. 54

Auto-Cad Figure Fsar 3_6_1D.dwg

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SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

REACTOR VESSEL HEAD VENT

FIGURE 3.6-8B, Rev. 54

Auto-Cad Figure Fsar 3_6_8B.dwg

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SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
HEAD SPRAY
FIGURE 3.6-8C, Rev. 54

Auto-Cad Figure Fsar 3_6_8C.dwg

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SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
STANDBY LIQUID CONTROL FIGURE 3.6-8D, Rev. 54

Auto-Cad Figure Fsar 3_6_8D.dwg

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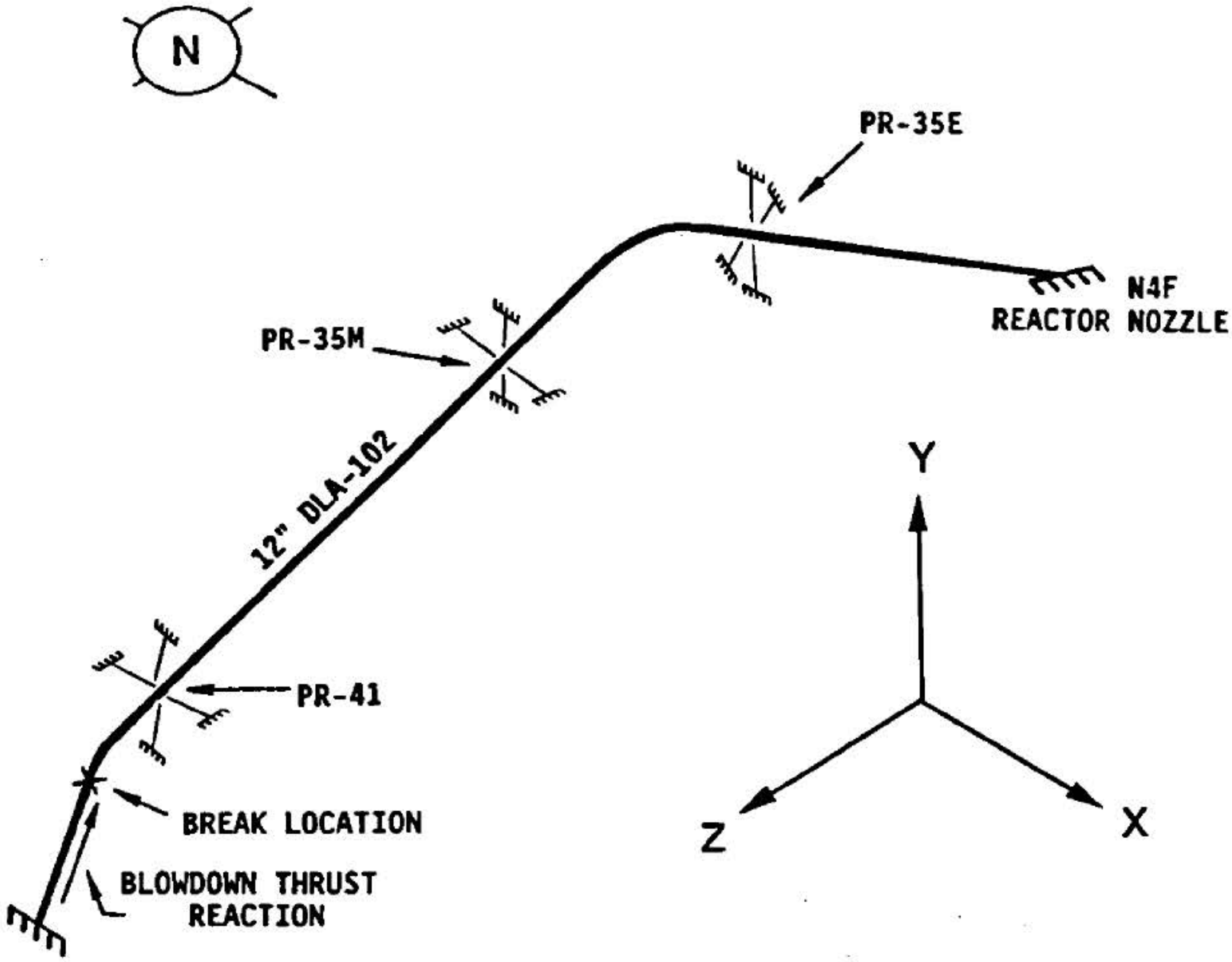
Figure Withheld Under 10 CFR 2.390

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

MSIV DRAINS

FIGURE 3.6-8E, Rev. 54

Auto-Cad Figure Fsar 3_6_8E.dwg



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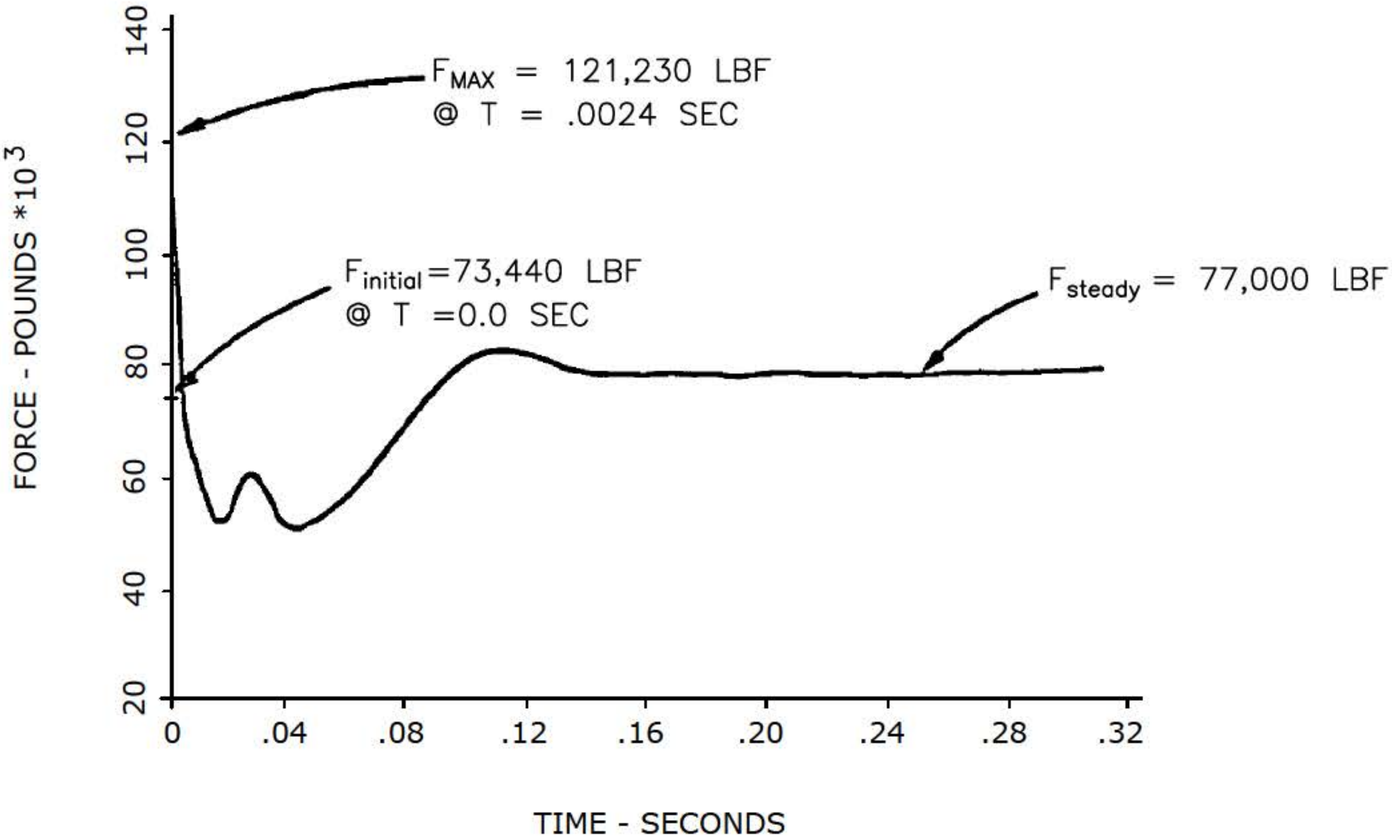
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

MAIN FEEDWATER LINE
PIPERUP MATHEMATICAL MODEL

FIGURE 3.6-11A, Rev. 47

Auto-Cad Figure Fsar 3_6_11A.dwg



HPCI INSIDE CONTATINMENT PIPE BREAK FORCING FUNCITON REACTOR SIDE

FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

FORCING FUNCTIONS MODEL
ASSOCIATED WITH PIPE WHIP
DYNAMIC ANALYSIS

FIGURE 3.6-12A, Rev. 47

Auto-Cad Figure Fsar 3_6_12A.dwg

Security-Related Information
Figure Withheld Under 10 CFR 2.390

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

PIPE BREAK PROTECTION
FOR HIGH ENERGY PIPING
IN THE REACTOR BUILDING

FIGURE 3.6-17-1, Rev. 54

Auto-Cad Figure Fsar 3_6_17_1.dwg

Security-Related Information
Figure Withheld Under 10 CFR 2.390

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

PIPE BREAK PROTECTION
FOR HIGH ENERGY PIPING
IN THE REACTOR BUILDING

FIGURE 3.6-17-2, Rev. 54

Auto-Cad Figure Fsar 3_6_17_2.dwg

Security-Related Information
Figure Withheld Under 10 CFR 2.390

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

PIPE BREAK PROTECTION FOR
HIGH ENERGY PIPING IN THE
REACTOR BUILDING

FIGURE 3.6-17-3, Rev. 47

Auto-Cad Figure Fsar 3_6_17_3.dwg

Security-Related Information
Figure Withheld Under 10 CFR 2.390

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
REACTOR WATER CLEANUP
FIGURE 3.6-8A.1, Rev. 54

Auto-Cad Figure Fsar 3_6_8A1.dwg

Security-Related Information
Figure Withheld Under 10 CFR 2.390

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

UNIT 1
REACTOR WATER CLEANUP

FIGURE 3.6-8A.2, Rev. 55

Auto-Cad Figure Fsar 3_6_8A2.dwg

Security-Related Information
Figure Withheld Under 10 CFR 2.390

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
UNIT 2 REACTOR WATER CLEANUP FIGURE 3.6-8A.3, Rev. 56

Auto-Cad Figure Fsar 3_6_8A3.dwg

Security-Related Information
Figure Withheld Under 10 CFR 2.390

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
UNIT 1 REACTOR WATER CLEANUP FIGURE 3.6-8A.4, Rev. 55

Auto-Cad Figure Fsar 3_6_8A4.dwg

Security-Related Information
Figure Withheld Under 10 CFR 2.390

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
UNIT 1 REACTOR WATER CLEANUP FIGURE 3.6-8A.5, Rev. 55

Auto-Cad Figure Fsar 3_6_8A5.dwg

APPENDIX 3.6A

PIPE BREAK OUTSIDE CONTAINMENT
SUMMARY OF ANALYSIS AND RESULTS

PART I - ANALYSIS FOR SPACES OTHER THAN MAIN STEAM TUNNEL

In addition to the analysis provided in Table 3.6-3, compartments containing high energy lines were analyzed to determine the peak pressures and temperatures that might result from breaks in these lines. For the HPCI, RCIC and RWCU pipe breaks outside primary containment, a concurrent LOOP and single failure is assumed to occur, which is consistent with the response time testing (ATT) assumption. The analysis was done, in part, to verify structural integrity. Duration of the blowdown was not a factor in the pressure transient since adequate vent area was provided, and pressure peaked quickly then declined to a lower steady state value. The blowout panels are designed to release at design pressure of approximately 0.5 psig. The structures and safe shutdown equipment are adequate to withstand the peak pressures and temperatures indicated by the analysis.

The valves which would be used to terminate the blowdown are indicated. In general, however, it is unnecessary to qualify equipment for the pipe break environment because the safeguards systems are separated into compartments which are vented directly to the atmosphere and high energy breaks affect only a single space. The plant can be safely shutdown using equipment not affected by the high energy line break.

The following information for each compartment was utilized with the analytical techniques described in Reference 3.6-10 of the FSAR to determine the pressures and temperatures resulting from high energy line breaks outside containment.

ANALYSIS FOR HPCI PENETRATION ROOM (UNIT 1)

Pipe Break Data

Location: HPCI Penetration Room (I-202, I-204, I-205)
Line Identification/Size: DBB-114/10"

Isolation Valve Designation and Location: HV-E41-1F003 located in the HPCI Penetration Room

Blowdown Data:

t (sec)	m (lbm/sec)	h (BTU/lbm)
0.0	2074.	1190.2
0.1	2074.	1190.2
0.1	1501.	1190.2
0.18	1501.	1190.2
0.18	464.	1190.2
13.0	464.	1190.2
63.0	0	1190.2

Compartment Volume: 87,680. ft^3

Vent Area: 69.6 ft^2 (3 circular panels, each with a flow area of 23.2 ft^2)
Vent Loss Coefficient: 1.95
L/A: 0.97 ft^{-1}

<u>Results (BDIDs Closed)</u>:

Peak Pressure:	1.95 psig
Peak Temperature:	295.6°F

<u>Results (BDIDs Open, HPCI Steam Supply Break)</u>:

Peak HPCI Penetration Room Pressure:	1.84 psig
Peak HPCI Penetration Room Temperature:	294.4ºF
Peak RBCCW Heat Exchanger Area (I-203) Pressure:	0.38 psig
Peak RBCCW Heat Exchanger Area (I-203) Temperature:	105.0ºF
Peak 683' Equipment Area (I-200) Pressure:	0.50 psig
Peak 683' Equipment Area (I-200) Temperature:	106.6ºF

<u>Results (BDIDs Open, RCIC Steam Supply Break, 4"-DBB-109)</u>:

Note: Break is isolated by isolation valve HV-E51-1F008

Peak HPCI Penetration Room Pressure:	1.26 psig
Peak HPCI Penetration Room Temperature:	151.0ºF
Peak RBCCW Heat Exchanger Area (I-203) Pressure:	1.09 psig
Peak RBCCW Heat Exchanger Area (I-203) Temperature:	112.4ºF
Peak 683' Equipment Area (I-200) Pressure:	1.25 psig
Peak 683' Equipment Area (I-200) Temperature:	114.0ºF

<u>ANALYSIS FOR HPCI PUMP ROOM (UNIT 1)</u>

Pipe Break Data

Location:	HPCI Pump Room (I-11)
Line Identification/Size:	DBB-114/10"

Isolation Valve Designation and Location: HV-E41-1F003 located in the HPCI Penetration Room

Blowdown Data:

t (sec)	m (lbm/sec)	h (BTU/lbm)
0.0	2074.	1190.2
0.076	2074.	1190.2
0.076	1037.	1190.2
.218	1037.	1190.2
.218	314.	1190.2
13.0	314.	1190.2
63.0	0	1190.2

Compartment Volume: 27,883 ft^3

Vent Area: 60 sq ft
Vent Loss Coefficient: 2.63
L/A: 0.39 ft^{-1}

Results (Duct Closed):

Peak Pressure:	3.55 psig
Peak Temperature:	303.3°F

Results (Duct Open**)**:

Peak HPCI Pump Room Pressure:	3.30 psig
Peak HPCI Pump Room Temperature:	303.1ºF
Peak 670' General Access Area (I-102) Pressure:	0.67 psig
Peak 670' General Access Area (I-102) Temperature:	108.4ºF
Peak "B" Core Spray Room (I-10) Pressure:	0.40 psig
Peak "B" Core Spray Room (I-10) Temperature:	109.0ºF

ANALYSIS FOR RCIC PUMP ROOM (UNIT 1)

Pipe Break Data

Location:	RCIC Pump Room (I-12)
Line Identification/Size:	DBB-109/4"

Isolation Valve Designation and Location: HV-E51-1F008 located in the HPCI Penetration Room

Blowdown Data:

t (sec)	m (lbm/sec)	h (BTU/lbm)
0.0	314.0	1190.2
0.021	314.0	1190.2
0.021	157.0	1190.2
0.278	157.0	1190.2
0.278	30.1	1190.2
13.0	30.1	1190.2
33.0	0	1190.2

Compartment Volume: 18,129 ft^3

Vent Area: 46.0 sq ft
Vent Loss Coefficient: 2.67
L/A: 0.43 ft^{-1}

Results (BDIDS Closed):

Peak Pressure: 1.17psig
Peak Temperature: 220.0°F

Results (BDIDs Open):

Peak RCIC Pump Room Pressure: 0.99 psig
Peak RCIC Pump Room Temperature: 219.8ºF
Peak 670' General Access Area (I-102) Pressure: 0.09 psig
Peak 670' General Access Area (I-102) Temperature: 101.1ºF

ANALYSIS FOR RHR ROOM A (UNIT 1)

Pipe Break Data

Location: RHR Room A (I-14)
Line Identification/Size: HBB-110/24"

Isolation Valve Designation and Location: HV-E11-1F008 located in the HPCI Penetration Room

Compartment Volume: 48,554 cu ft

Vent Area: 85 sq ft

Results: Peak Pressure: 0.93 psig
Peak Temperature: 215.12°F

ANALYSIS FOR RHR ROOM B (UNIT 1)

Pipe Break Data

Location: RHR Room B (I-13)
Line Identification/Size: HBB-110/24"

Isolation Valve Designation and Location: HV-E11-1F008 located in the HPCI Penetration Room

Compartment Volume: 60,000 cu ft

Vent Area: 85 sq ft

Results: Peak Pressure: 0.93 psig
Peak Temperature: 215.12°F

Due to the removal of the steam condensing mode of RHR, the only high energy piping which would cause room pressurization during normal plant operation in both RHR Pump Rooms is during the initial stages of the shutdown cooling mode of RHR. Per BTP MEB 3-1, when RHR is placed in shutdown cooling, the piping is classified as a moderate-energy fluid system and only a pipe crack (not break) is postulated.

ANALYSIS FOR REACTOR WATER CLEANUP SYSTEM (RWCS) PENETRATION ROOM, PUMP ROOMS, AND HEAT EXCHANGER ROOMS (UNIT 1)

Pipe Break Data

Various break locations were analyzed to determine the maximum pressure and temperature which develop in each room.

Isolation Valve Designation and Location: HV-G33-F004 in RWCS Penetration Room

Blowdown Data:

Penetration Room (I-501)

t (sec)	m (lbm/sec)	h (BTU/lbm)
0.00	4570.0	518.5
0.07	4570.0	518.5
0.07	3030.0	518.5
0.14	3030.0	518.5
0.14	1155.0	518.5
0.84	1155.0	518.5
0.84	410.0	518.5
20.00	410.0	518.5
50.00	0.0	518.5

Pump Rooms (I-502,503)

t (sec)	m (lbm/sec)	h (BTU/lbm)
0.00	1990.0	518.5
0.10	1990.0	518.5
0.10	1640.0	518.5
0.21	1640.0	518.5
0.21	1055.0	518.5
1.10	1055.0	518.5
1.10	410.0	518.5
20.00	410.0	518.5
50.00	0.0	518.5

Heat Exchanger Rooms (I-504,505)

t (sec)	m (lbm/sec)	h (BTU/lbm)
0.00	2420.0	518.5
0.08	2420.0	518.5
0.08	1855.0	518.5
0.30	1855.0	518.5
0.30	1055.0	518.5
0.50	1055.0	518.5
0.50	410.0	518.5
20.00	410.0	518.5
50.00	0.0	518.5

Compartment Volumes:

Arch. Room No.	Volume (Cu. Ft.)
I-501	6940
I-502 & 503	6350
I-504 & 505	12229

Intercompartment Flow Path Data:

Flow Path	Area (Ft^2)	Loss Coefficient	L/A (ft^{-1})
I-501 to ATM	46.4 (2 circular panels, each with a flow area of 23.2 ft^{-2})	1.81	1.25
I-501 to I-503	60.0	1.00	0.1181
I-503 to I-504	60.0	1.00	0.0749

Results

Architectural Room Number	Peak Pressure (psig)	Peak Temperature (ºF)
I-500	0.21	102.7
I-501	3.76	215.1
I-502	2.73	213.1
I-503	2.73	213.1
I-504	3.18	213.0
I-505	3.18	213.0

Note: To provide a bounding case, a larger enthalpy condition was coupled with a larger mass flow rate. A break in the RWCU Heat Exchanger Room with the BDIDs open results in the most severe environment in the 749' general access area (I-500); therefore, the results for this area are presented. All other values are the result of breaks with the BDIDs in the closed position.

Analysis for Compartment Pressurization in Unit 2 is identical to Unit 1, with the exception of breaks in the HPCI and RCIC Rooms. These analyses are presented below.

ANALYSIS FOR RCIC PUMP ROOM (UNIT 2)

Pipe Break Data

Location: RCIC Pump Room (II-12)
Line Identification/Size: DBB-209/4"

Isolation Valve Designation and Location: HV-E51-2F008 located in HCPI Penetration Room

Blowdown Data:

t (sec)	m (lbm/sec)	h (BTU/lbm)
0	314.0	1190.2
0.021	314.0	1190.2
0.021	157.0	1190.2
0.278	157.0	1190.2
0.278	30.1	1190.2
13.0	30.1	1190.2
33.0	0.0	1190.2

Compartment Volumes:

RCIC 18,129 ft^3
HPCI 27.883 ft^3
Tunnel 3,312 cu ft

Flow Path	Area (Ft^2)	Loss Coefficient	L/A (ft^{-1})
RCIC to Tunnel	25	0.88	0.341
Tunnel to HPCI	72	0.50	0.3551
Tunnel to ATM	45	5.33	0.3914

Results (BDIDs Closed):

Room	Peak Pressure (PSIG)	Peak Temperature. (°F)
RCIC	1.56	218.3
HPCI	1.50	112.8
Tunnel	1.63	195.5

Results (BDIDs Open):

Room	Peak Pressure (psig)	Peak Temperature (ºF)
RCIC	1.27	215.7
HPCI	1.27	113.3
Connecting Tunnel	1.40	185.2
670' General Access Area (II-102)	1.26	117.0

Note: A break in the RCIC pump room results in a change in environment to the HPCI pump room via connection of the tunnel to both rooms. Therefore, peak pressures are shown for all three compartments.

ANALYSIS FOR HPCI PUMP ROOM (UNIT 2)

Pipe Break Data

Location: HPCI Pump Room (II-11)
Line Identification/Size: DBB-214/10"

Isolation Valve Designation and Location: HV-E41-2F003 located in the HPCI Penetration Room

Blowdown Data:

t (sec)	m (lbm/sec)	h (BTU/lbm)
0	2074.	1190.2
0.06	2074.	1190.2
0.06	1037.	1190.2
0.223	1037.	1190.2
0.223	308.	1190.2
13.0	308.	1190.2
63.0	0	1190.2

Compartment Volumes:

HPCI	27.883 ft^3
RCIC	18,129 ft^3
Tunnel	3,312 cu ft

Flow Path	Area (Ft^2)	Loss Coefficient	L/A (ft^{--1})
HPCI to Tunnel	72	0.50	0.3551
Tunnel to RCIC	25	0.88	0.341
Tunnel to ATM	45	5.33	0.3914

Results (BDIDs Closed):

Room	**Peak Pressure (psig)**	**Peak Temperature (°F)**
HPCI	3.71	304.2
RCIC	3.29	133.3
Tunnel	3.50	304.7

Results (BDIDs Open):

Room	Peak Pressure (psig)	Peak Temperature (ºF)
RCIC	2.59	128.5
HPCI	3.16	303.6
Connecting Tunnel	2.97	303.6
670' General Access Area (II-102)	1.39	120.3

Note: A break in the HPCI pump room results in a change in environment to the RCIC pump room via connection of the tunnel to both rooms. Therefore, peak pressures are shown for all three compartments.

PART II - ANALYSIS OF MAIN STEAM LINE BREAKS IN THE MAIN STEAM LINE TUNNEL

Subcompartment differential pressure analysis was performed for the Reactor and Turbine Building main steamline tunnel. The blowout panels in the reactor building steam tunnel are designed to release at design pressure of approximately 0.5 psig. Two break locations were chosen to render the design of each portion of the tunnel (viz. - Reactor and Turbine Building sides) conservative. They are:

Case A. MSLB in the Reactor Building.
(24" DBB-103 at the elbow on El. 719'-8")

Case B. MSLB in the Turbine Building.
(24" DBB-103 at El. 719'-6", 1st elbow in the Turbine Building)

The pressure and temperature response of these areas to the postulated pipe breaks are predicted using COTTAP 4 for the Reactor Building Main Steam Tunnel and the analytical model described in Appendix 6B with the changes described below for the Turbine Building Main Steam Tunnel. COTTAP 4 uses a similar analytical model as the model discussed in this section and Appendix 6B. Any differences between COTTAP 4 and the models presented in this section and Appendix 6B were reviewed and determined to have an insignificant or conservative effect on the peak pressures and peak temperatures. The Appendix 6B model ignores "momentum effects" within a subcompartment. For most cases considered, this is justified as the momentum effects are insignificant relative to the absolute pressure peaks. However, momentum effects are important to conservatively predicting pressures resulting from the main steam tunnel case. Therefore, for this study, the momentum equation

$$\frac{\partial}{\partial t}(\rho\vec{u})+\vec{\nabla}(\rho\vec{u}\vec{u})=-\vec{\nabla}p+\vec{\nabla}\vec{\vec{\tau}}+\rho\vec{g}$$

is "one-dimensionalized" and solved in the following manner:

$$\left(\frac{1}{g_c\,A(x)}\right)\frac{\partial}{\partial t}\left[A(x)\;G(x,t)\right]=-\left(\frac{1}{g_c\,A(x)}\right)\frac{\partial}{\partial x}\left[\frac{A(x)\,G^2(x,t)}{\rho(x,t)}\right]-\frac{\partial \mathrm{p(x,t)}}{\partial \mathrm{x}}-\frac{1}{A(x)}\frac{\partial F(x,t)}{\partial x} \quad (1)$$

Where G = Δv

Where the F(x,t) term includes shear forces and non-one-dimensional momentum change effects. Its integral over a flow path is evaluated by means of empirically determined flow coefficients (see Appendix 6B).

Equation (1) is now integrated from midpoint to midpoint of two adjoining compartments assuming uncompressible flow, but with a uniquely determined fluid density. The density of the flow mixture is evaluated in a way which assures that, as flow approaches steady state conditions, the density and the computed mass flux approach the values obtained from the compressible steady state equations in Appendix 6B.

Using this assumption and integrating term by term, we obtain:

First term:

$$\frac{1}{g_c}\int_{x_1}^{x_2}\frac{1}{\mathrm{A(x)}}\frac{\partial}{\partial t}\left[A(x)\,G(x,t)\right]dx=\frac{1}{\mathrm{g_c}}\frac{\partial}{\partial t}W(t)\int_{x_1}^{x_2}\frac{\mathrm{dx}}{\mathrm{A(x)}}=\frac{1}{\mathrm{g_c}}\frac{\mathrm{dW(t)}}{\mathrm{dt}}\sum_{\mathrm{i}}\left(\mathrm{L_i}\big/\mathrm{A_i}\right) \quad (1a)$$

Where the integral of (dx/A(x)) is evaluated sequentially for constant area segments between X_1 and X_2. Li thus represents the length of segment i.

Second term:

$$-\frac{1}{g_c}\int_{x_1}^{x_2}\frac{1}{\mathrm{A(x)}}\frac{\partial}{\partial x}\left[\frac{A(x)\,G^2(x,t)}{\rho(x,t)}\right]dx=-\frac{W^2(t)}{g_c\rho}\int_{x_1}^{x_2}\left(\frac{1}{A(x)}\right)\frac{\mathrm{d}}{\mathrm{dx}}\left(\frac{1}{\mathrm{A(x)}}\right)\mathrm{dx}=-\frac{W^2(t)}{2g_c\rho}\left[\frac{1}{A_2^2}-\frac{1}{A_1^2}\right] \quad (1b)$$

Where the Δ in the above expression remains to be defined.

Third term:

$$-\int_{x_1}^{x_2}\frac{\partial p(x,t)}{\partial x}dx=-\left[P_2-P_1\right] \quad (1c)$$

It should be noted that the above pressures are static values and to match the units of Equation (1) are, at this point, given in terms of $1\mathrm{b/ft}^2$.

Fourth term:

$$-\int_{x_1}^{x_2}\frac{1}{A(x)}\frac{\partial F(x,t)}{\partial x}dx = -Ki\frac{V_T^2}{2g_c}\rho \qquad (1d)$$

Where i = +1 if W ≥ 0 and i = -1 if W < 0.

The above equation is not really a proper integration, but just a replacement of this integral by the appropriate empirical correlation. The coefficient K is a properly summed coefficient for the flow path from x_1 to x_2 and can include friction terms. The velocity $V^2{}_T$ depends on the empirical correlation used, but is usually taken as the "throat" velocity. This is assumed to be the case, then Equation (1d) becomes:

$$-\frac{KiV_T^2}{2g_c}\rho\left(\frac{\rho A_T^2}{\rho A_T^2}\right) = -Ki\frac{W^2(t)}{2g_c A_T^2\rho} \qquad (1e)$$

Where $A^2{}_T$ is the junction flow area.

Before collecting all the integrated terms, it is preferable to convert the static pressures of Equation 1c into stagnation pressures.

$$P^*_{stat(i)} = P^*_{stag(i)} - \frac{\rho V(i)^2}{2g_c} = \mathrm{P}^*_{\mathrm{stag(i)}} - \frac{W^2(t)}{2g_c\rho A_i^2} \qquad (1f)$$

Summing the expressions obtained by Equations (1b) to (1e) and using (1f) we get:

$$\frac{1}{g_c}\frac{dW(t)}{dt}\sum_i\left(\frac{L_i}{A_i}\right) = \mathrm{P}_1^* - \mathrm{P}_2^* - \frac{\mathrm{Ki}\mathrm{W}^2(t)}{2g_c\rho A_T^2} \qquad (1g)$$

Where the starred pressures imply stagnation values.

Now the flow rate of the previous time step is used to evaluate a finite-difference approximation of the time derivative:

$$\frac{dW(t)}{dt} = \frac{W(t) - W(t - \Delta t)}{\Delta t} \qquad (2)$$

In a given time interval, W(t-Dt) is known, thus Equation (1g) is a quadratic in W(t). Writing it in the customary quadratic form we have:

$$\frac{Ki}{2\rho g_c A_T^2} W^2(t) + \frac{\sum_i \left(L_i / A_i \right)}{g_c \Delta t} W(t) - \left\{ \frac{\sum_i \left(L_i / A_i \right)}{g_c \Delta t} W(t-\Delta t) + \mathrm{P}_1^* - P_2^* \right\} = 0 \qquad (3)$$

and substituting the compressible flow equation for W. The resulting ratio is:

$$\frac{\rho}{\rho_2} = \left(\frac{k}{k-1} \right) \left[\frac{1}{1 - \frac{P_2}{P_1}} \right] \left[\left(\frac{P_2}{P_1} \right)^{1/k} - \left(\frac{P_2}{P_1} \right) \right] \qquad (4)$$

In the limit as $(P_2/P_1) \rightarrow 1$, Equation 4 approaches a value of one as required and the P_2/P_1 ratio stays below one for all other values of p_2/p_1 and for all positive k. Δ is thus smaller than the arithmetic mean of the densities and smaller than the downstream density itself. This assures a conservatively minimized flow rate for a given pressure gradient. This also holds true when the inertial effects (time dependent momentum equation) are included. Table 3.6A-1 shows representative mass flux values calculated by density Δ_2, and the proper compressible flow compatible density Δ is used. As seen for all cases, the use of r results in minimum and thus conservative flow rates.

The calculational sequence can now be summarized.

1. After compartment state functions have been obtained, a first estimate of W(t) is evaluated using the compressible flow equation.

2. The estimate of W(t) is used in Equation 3b to evaluate the fluid density.

3. Utilizing the flow rate from the previous time step and the calculated Δ, Equation (3) is solved to obtain W(t).

During each time step, the junction flow rate is chosen as the smaller of the flow rate resulting from the one-dimensional momentum equation or the flow rate resulting from the selected steady state compressible flow correlation. (Appendix 6B.)

Schematic drawings showing the nodalization of the steam tunnel for Case A and Case B are given in Figures 3.6A-1 and 3.6A-5, respectively. Blow out panel locations are shown in Figure 3.6A-2. Volumes, flow areas, flow coefficients, L/A's and blowdown rates for the models are presented in Tables 3.6A-2 through 3.6A-6. As indicated in Figure 3.6A-1, for Case A, the main steam tunnel is subdivided into a total of eighteen volumes to model the effect of obstructions such as pipe restraints and blow out panels. For Case B, in Figure 3.6A-5, a ten volume model is used since the one-way blowout panels completely block the flow path to reactor building side, leaving it unpressurized. The overall flow diagrams for both Cases A and B are presented in Figures 3.6A-3 and 3.6A-6.

The blowdown data for the postulated double end guillotine mainsteam line break is shown in Table 3.6A-4. This blowdown is done in a way similar to ANS 176 standard (draft, now known as ANSI/ANS 58.2-1980), as discussed below, but system friction is accounted for to reduce the calculated mass and energy releases to reasonable levels while maintaining a degree of conservatism. Other criteria are addressed as follows:

1. Full double-end break area Moody flow for steam blowdown immediately after pipe break.

2. Choking Moody flow occurs first at the break, then moves up to choke at flow restrictors.

3. Frictional loss of valves is not included.

4. Level swell (4% quality blowdown) occurs at 1 sec.

5. Steam isolation valves close in 5 seconds with a 0.6 second instrument and signal delay time. A linear ramp in flow area is used to model this closure.

The computational method of this double-end guillotine mainsteam line break is shown in Fig. 3.6A-8.

In Figure 3.6A-8, flow from the RPV to the break location is "forward flow," while the flow from the turbine to the break location is "reverse flow."

Let L_1 = The distance from flow restrictors to break location.

L_2 = The distance from reactor pressure vessel nozzle to the flow restrictors.

L_3 = The distance from flow restrictor to the turbine crosstie.

L_4 = The shortest distance from the MSL crosstie back to the break location.

(A) Calculation of mass and energy release rates from the forward direction.

let

A_p	=	The cross-sectional flow area of the break, ft^2.
A_v	=	The throat area of the flow restrictor, ft^2.
P_o	=	No-load system pressure, PSIA.
X	=	Steam quality.
h	=	Enthalpy of fluid, BTU/lbm.
N	=	Number of lines.
c	=	Sonic speed for steam.
f	=	Frictional factor.
D	=	Diameter of the pipe system.

1. At $O \le T \le L_1/C$ sec.

$$\dot{W}_{1F} = \left(G_{M1}A_p - \dot{W}_{2F}\right)\left(1 - T/\left(L_1/C\right)\right) + \dot{W}_{2F}$$

Where G_{M1} = Moody specific flowrate (lbm/sec*ft^2) based on P_0 = 1050 PSIA and h = 1190.0 BTU/lbm.

This ramp-down in flow rate simulates the increasing system resistance downstream of the decompression wave front.

2. $At = L_1 / C \le T \le \frac{2*(1.1*L_1)}{0.9*C}\sec$ (Time for choking at flow restrictor)

$$\dot{W}_{2F} = G_{M2} * A_p$$

Where G_{M2} = Moody specific flow rate based on p = 1050 psia and h = 1190 BTU/lbm with $\frac{fl_1}{D}$.

3. $At\,\frac{2*(1.1*L_1)}{0.9*C} \le T \le 1.0\sec$ (Time for level swell)

$W_{3f} = G_{M3} * A_v$

Where G_{M3} = Moody specific flow rate based on p = 1050 psia and h = 1190 BTU/lbm with $\frac{fl_1}{D}$.

(B) Calculation of mass and energy release rates from the reverse direction.

1. At $O \le T \le L_4/C$ sec.

$$W_{1R} = (G_{M1} * A_p - W_{2R})(1 - T/(L_4/C)) + W_{2R}$$

This ramp-down in flow rate simulates the increasing system resistance downstream of the decompression wave front.

2. $At\,L_4 / C \le T \le \frac{2*(L_3 + L_4)}{C}\sec$ (Time for choking at the flow restrictors)

$$\dot{W}_{2R} = G_{M2R} * A_V * N$$

Where G_{M2R} = Moody specific flow rate based on h = 1190 BTU/lbm with $f\,\frac{(L_3 + L_4)}{D}$

3. $At\,\frac{2*(L_3 + L_4)}{C} \le T \le 1.00\sec$ (Time for level swell)

$$\dot{W}_{3R} = \dot{W}_{3R}(\text{A LINE}) + \dot{W}_{3R}(\text{B LINE}) + \dot{W}_{3R}(\text{C LINE})$$
$$= \text{A}_\text{V}\left[G_{M3R}(A) + G_{M3R}(B) + G_{M3R}(C)\right]$$

Where G_{M3R} (A), G_{M3R} (B) and G_{M3R} (C) are the Moody specific flow rates for lines A, B, C based on P_o = 1050 PSIA and h = 1190 BTU/lbm with fL_2/D for each line.

(C) Calculation of mass and energy release rates from the swell phenomenon.

1. At 1.0 ≤ T ≤ 4.35 sec. (Time for choking at the valve)

$$\dot{W}_S = \dot{W}_S(A) + \dot{W}_S(B) + \dot{W}_S(C) + \dot{W}_S(D)$$
$$A_V\left[G_{MA}(A) + G_{MS}(B) + G_{MS}(C) + G_{MS}(D)\right]$$

Where G_{M2} (A), G_{M2} (B), G_{M2} (C), G_{M2} (D) are the Moody specific flow rates for lines A, B, C, D based on h =572 BTU/LBM (4% quality) and fL_2/D for each line.

2. At T = 5.6 sec. (Time for valve completely closed)

$$\dot{W}_3 = 0.0\,\text{lbm/sec}$$

(D) Calculation of the total mass and energy release rates.

The total flow rate is obtained by adding up the forward flow and reverse flow at each time sequence by superpositioning of the two curves (forward and reverse). Then after 1.0 second, the total flow rate will be just the flow rate calculated from swell on section (C).

The pressure transients of this analysis for Cases A (with BDIDs closed) and B are plotted in Figures 3.6A-4 and 3.6A-7. It can be seen that the maximum pressure for Case A in the Reactor Building is 23.1 PSIA and for Case B in the Turbine Building is 37.1 PSIA. The peak temperature for Case A is 303.0°F and for Case B is 325.0°F. For Case A in which the BDIDs are open, the peak pressure in the reactor building steam tunnel is 23.0 psia and the peak temperature is 303.0ºF. The open BDIDs will allow the transport of the reactor building main steam tunnel environment to the 719' elevation general access area and the valve access area on elevation 749'. The peak pressure for the 719' general access area is 15.0 psia and the peak temperature for this area is 104.5ºF, the peak pressure in the valve access area on elevation 749' is 15.2 psia and the peak temperature is 111.7º.

The following essential equipment is located with the steam tunnels on Susquehanna SES:

Main Steam Isolation Valves (MSIV's) and Piping
Feedwater Check Valves and Piping
HPCI Piping
RCIC Piping
Leak Detection Instrumentation

Pipe breaks in the remaining portion of the main steam piping between the reactor building and the turbine building will not impact essential equipment since breaks in these areas are completely vented to the turbine building.

Waterflooding in either the turbine building or reactor building portion of the tunnel will drain to the turbine building without damage to the structure.

All of the terms in the coefficients of Equation 3 can be evaluated except for the as yet undefined fluid density, ρ. As stated in the assumptions, ρ will be evaluated in such a way that, under steady state conditions, Equation (3) and the compressible flow equations of Appendix 6B will yield identical results for W(t). Under steady state conditions W(t) = W(t-Δt) and Equation (3) reduces to:

$$\frac{K}{2g_c \rho A_T^2} W^2 - \Delta\rho^* = 0 \qquad (3a)$$

which yields

$$\rho = \frac{W^2 K}{2g_c A_T^2 \Delta\rho^*} \qquad (3b)$$

where the W^2 can be obtained from the steady state compressible flow equations in Appendix 6B.

Under steady state conditions, the above value of ρ which is used in the momentum equation has a straightforward definition -- it is the density which has to be used in the steady state incompressible flow equation in order to reproduce correct steady state compressible flow rates. To achieve this, the density includes an implied correction factor which compensates for the energy required in compressible flow to accelerate the expanding fluid. Because of this correction, ρ will, in fact, be smaller than the downstream density, ρ_2, calculated by the isentropic expansion relationship. This can be shown by dividing Equation (3b) by

$$\rho_2 = \rho_1 \left(\frac{P_2}{P_1} \right)^{1/k} \qquad (4)$$

TABLE 3.6A-1

COMPARISON OF FLOW RATES COMPUTED FROM THE TIME DEPENDENT MOMENTUM EQUATION

	$\frac{G_m(i-1)}{G_{comp}}$	$G_{mi}(\rho)$	$G_{mi}(\rho_{au})$	$G_{mi}(p_c)$	$\frac{G_{mi}(p_{au})}{G_{mi}(p)}$
$k = 1.08$	1.0 (Steady State)	43.44	44.69	44.69	1.029
	.5 (Flow Acceleration)	24.10	24.50	24.31	1.017
	1.2 (Flow Deceleration)	50.86	52.55	51.76	1.033
$\frac{P_2}{P_1} = .6$					
$k = .5$	1.0	28.94	31.12	30.14	1.076
	.5	16.75	17.18	17.18	1.025
	1.2	33.55	36.44	35.13	1.086
$k = 1.2$	1.0	45.01	6.17	45.63	1.026
	.5	24.98	25.26	25.09	1.015
	1.2	52.73	54.29	53.56	1.030

ρ = Compressible flow mean density (Equation 4m)
ρ_1 = Upstream node density
ρ_2 = Downstream node density
ρ_{au} = $\rho_1 + \rho_2$

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TABLE 3.6A-2

CASE A
STEAM TUNNEL COMPARTMENT VOLUMES

NODE	VOL (FT^3)	NODE	VOL (FT^3)
1	7326.7	10	1.0E15
2	1.0E15	11	10922.3
3	9911.9	12	27723.5
4	11148.8	13	5911.8
5	1.0E15	14	6803.9
6	900000.	15	2183.1
7	54000.	16	13994.1
8	54000.	17	1911.3
9	2.3E6	18	1932.6

TABLE 3.6A-3

STEAM TUNNEL FLOW AREAS, COEFFICIENTS AND L/A

PATH	AREA (FT^2)	LOSS COEFFICIENTS	L/A (FT^1)
1-3	614.7	0.13	.0151
1-11	125.0	0.60	.0480
3-4	612.7	0.25	.0146
4-12	459.0	0.27	.0717
6-7	390.0	2.19	.0415
6-8	390.0	2.19	.0398
7-9	980.0	2.19	.0623
8-9	980.0	2.19	.0722
9-10	6000.0	1.87	.0087
11-13	111.6	0.27	.3380
11-14	52.5	1.30	.0239
12-2	420.0	1.50	.0032
13-14	98.5	0.77	.0318
13-15	110.1	0.28	.1580
14-5	140.0	1.50	.0090
15-14	35.1	1.25	.0617
15-17	132.5	0.14	.0810
16-6	300.0	0.56	.0313
17-14	33.3	1.25	.0643
17-18	108.5	0.30	.0781
18-14	32.7	1.25	.0652
18-16	137.7	0.65	.0711

TABLE 3.6A-4

MASS FLOW RATES FOR CASE A

T (SEC)	M (lbm/SEC)	h (BTU/lbm)
0.000	10376.0	1190.0
0.051	7710.6	1190.0
0.125	4067.6	1190.0
0.131	3956.0	1190.0
0.590	3956.0	1190.0
0.590	4670.0	1190.0
1.000	4670.0	1190.0
1.000	16948.0	572.0
4.350	16948.0	572.0
4.500	14914.2	572.0
5.000	8135.0	572.0
5.600	0.0	572.0

MASS FLOW RATES FOR CASE B

T (SEC)	M (lbm/SEC)	h (BTU/lbm)
0.000	11852.0	1190.0
0.045	8681.5	1190.0
0.111	6907.6	1190.0
.0130	3499.0	1190.0
0.630	3499.0	1190.0
0.630	4142.0	1190.0
1.000	4142.0	1190.0
1.000	16948.0	572.0
4.350	16948.0	572.0
4.500	14340.6	572.0
5.000	7822.1	572.0
5.600	0.0	572.0

TABLE 3.6A-5

CASE B
STEAM TUNNEL COMPARTMENT VOLUMES

NODE	VOL (FT^3)	NODE	VOL (FT^3)
1	10922.3	6	900000.
2	5913.7	7	54000.
3	6227.0	8	54000.
4	13994.1	9	2.3E6
5	6803.9	10	1.7E9

TABLE 3.6A-6

STEAM TUNNEL FLOW AREAS, COEFFICIENTS AND L/A FOR CASE B

PATH	AREA (FT^2)	COEFFICIENTS	L/A (FT^{-1})
1-2	111.6	.89	.338
1-5	52.5	.66	.0239
2-3	110.1	.89	.236
2-5	98.5	.70	.0318
3-4	137.7	.78	.0711
3-5	101.1	.70	.0252
4-6	300.00	.80	.0313
5-10	210.0	.87	.009
6-7	390.0	.56	.0415
6-8	390.0	.56	.0398
7-9	980.0	.56	.0623
8-9	980.0	.56	.0722
9-10	6000.0	.59	.0087

Security-Related Information
Figure Withheld Under 10 CFR 2.390

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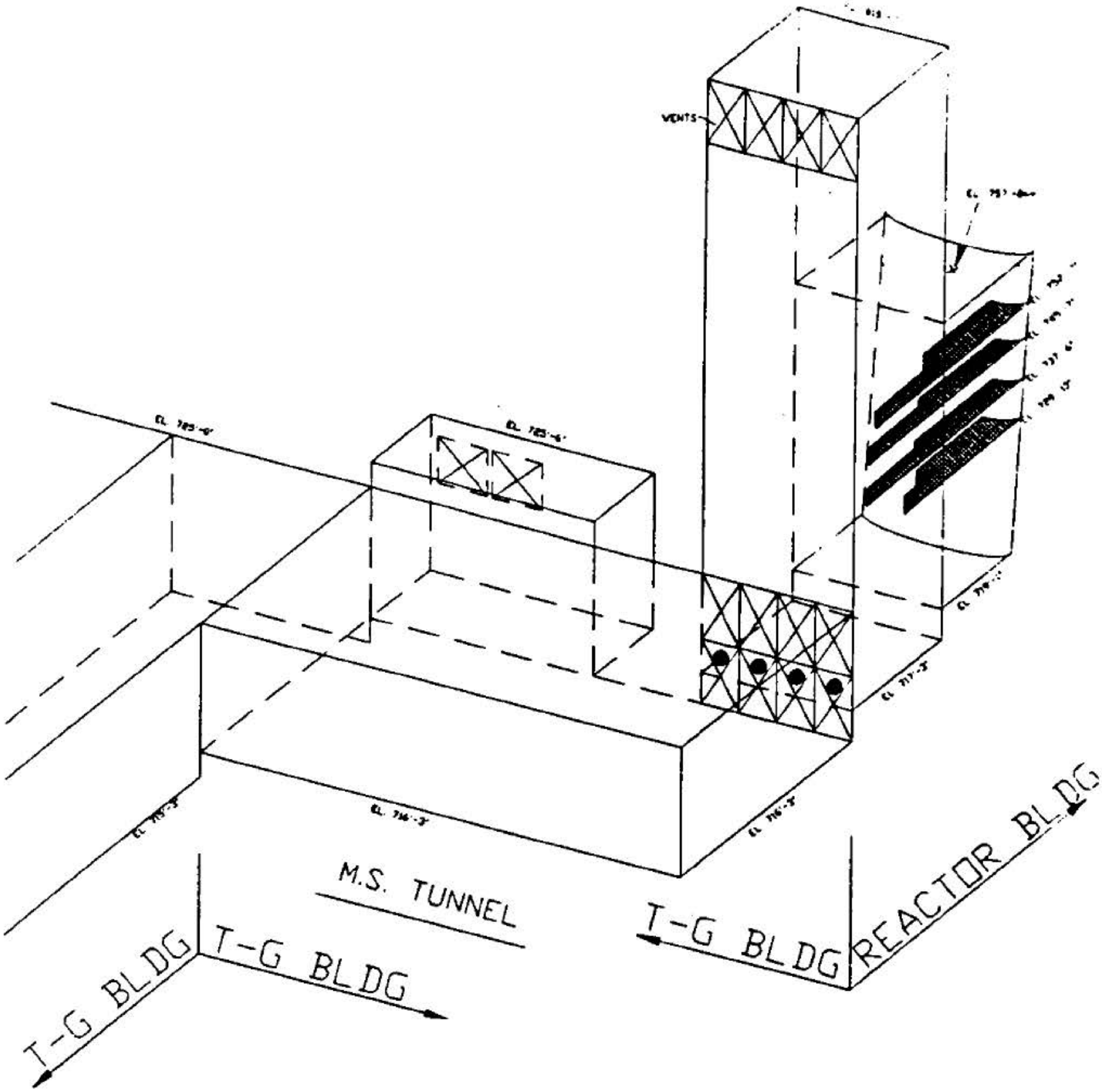
CASE A MSLB IN
REACTOR BUILDING

FIGURE 3.6A-1, Rev. 48

Auto-Cad Figure Fsar 3_6A_1.dwg



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SUSQUEHANNA STEAM ELECTRIC STATION
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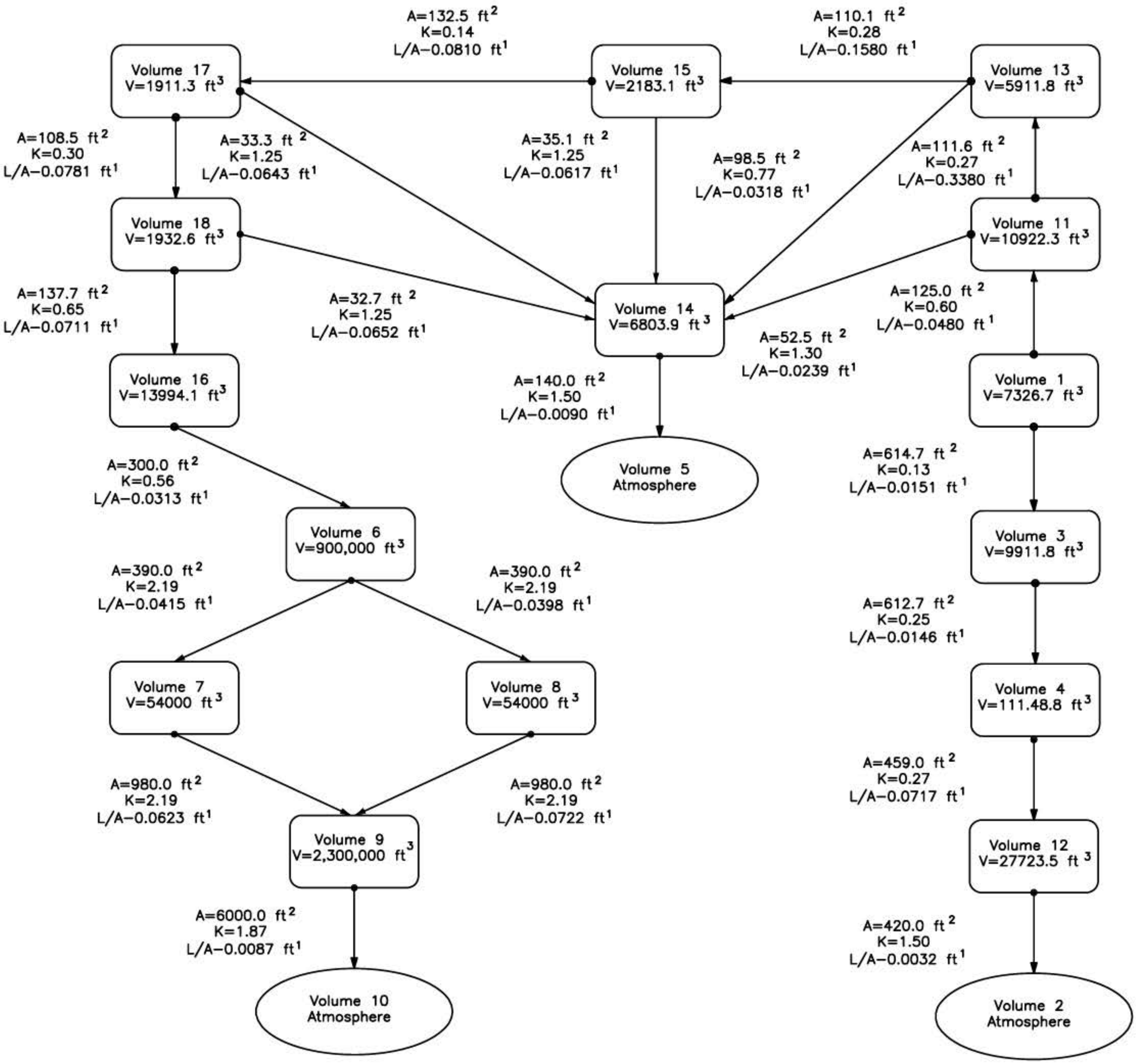
PANEL AND PLATFORM
LOCATIONS

FIGURE 3.6A-2, Rev. 47

Auto-Cad Figure Fsar 3_6A_2.dwg



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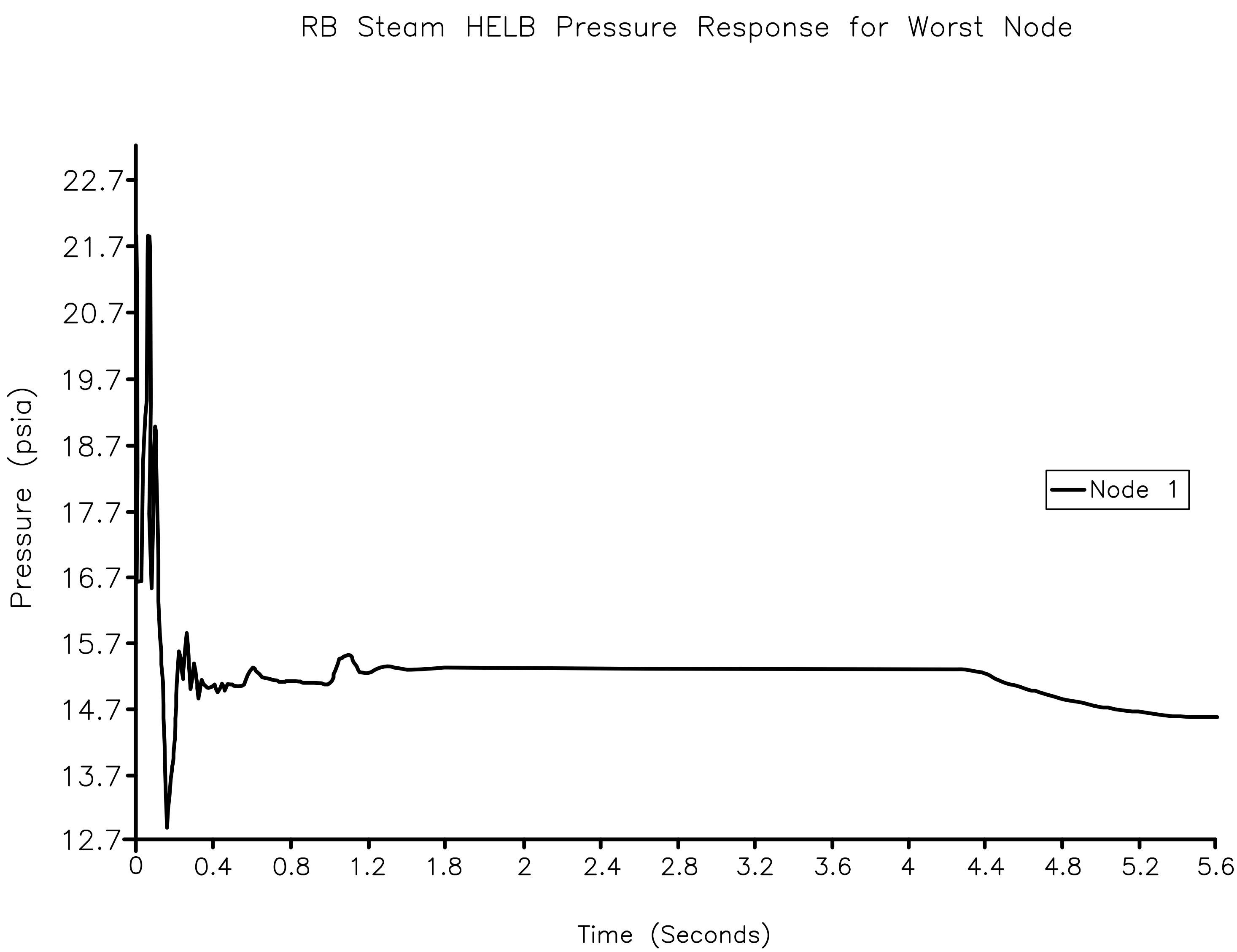
CASE A VOLUME FLOWS

FIGURE 3.6A-3, Rev. 48

Auto-Cad Figure Fsar 3_6A_3.dwg



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CASE A PRESSURE TRANSIENT

FIGURE 3.6A-4, Rev. 48

Auto-Cad Figure Fsar 3_6A_4.dwg

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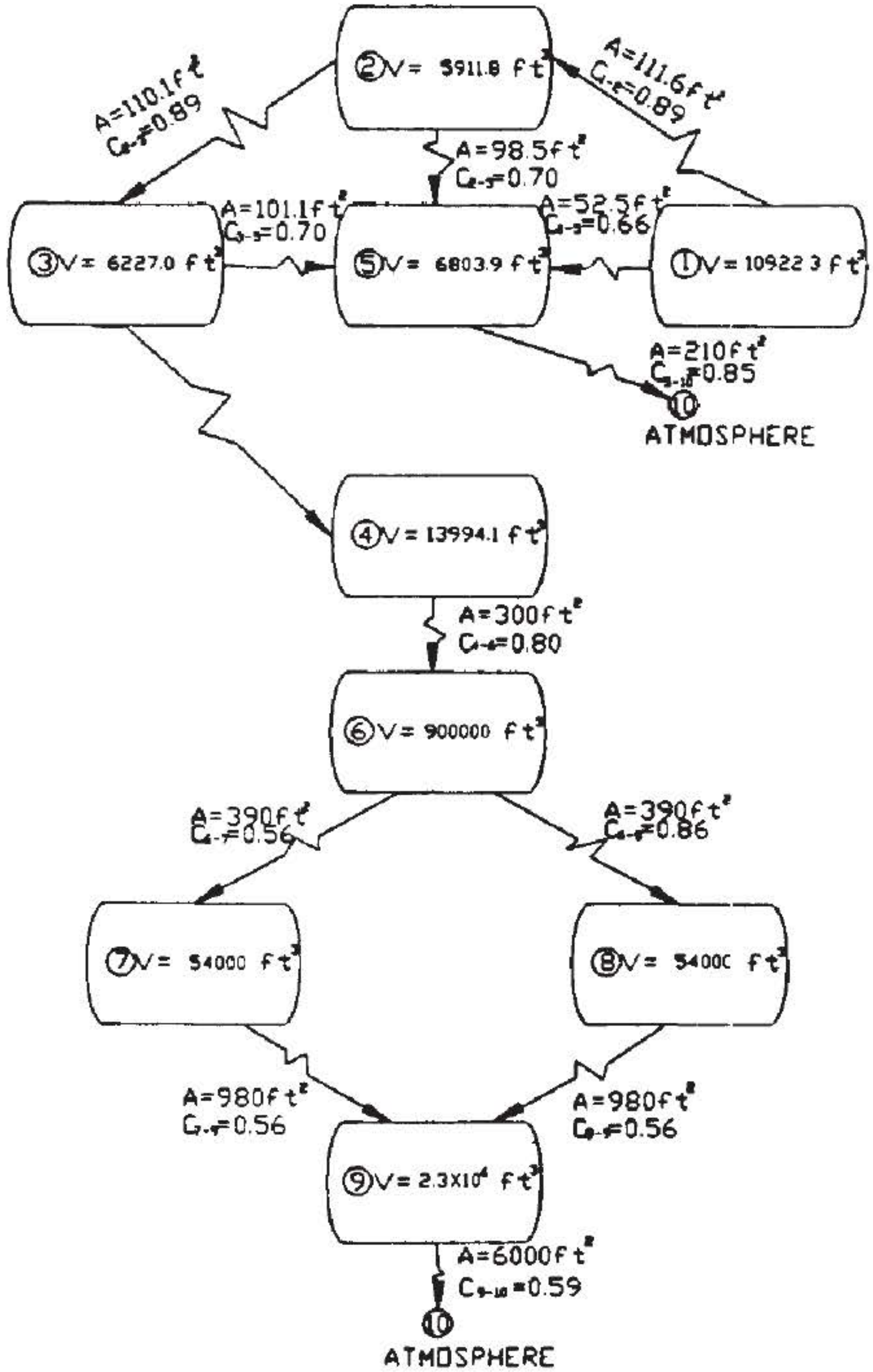
CASE B MSLB IN
TURBINE BUILDING

FIGURE 3.6A-5, Rev. 47

Auto-Cad Figure Fsar 3_6A_5.dwg



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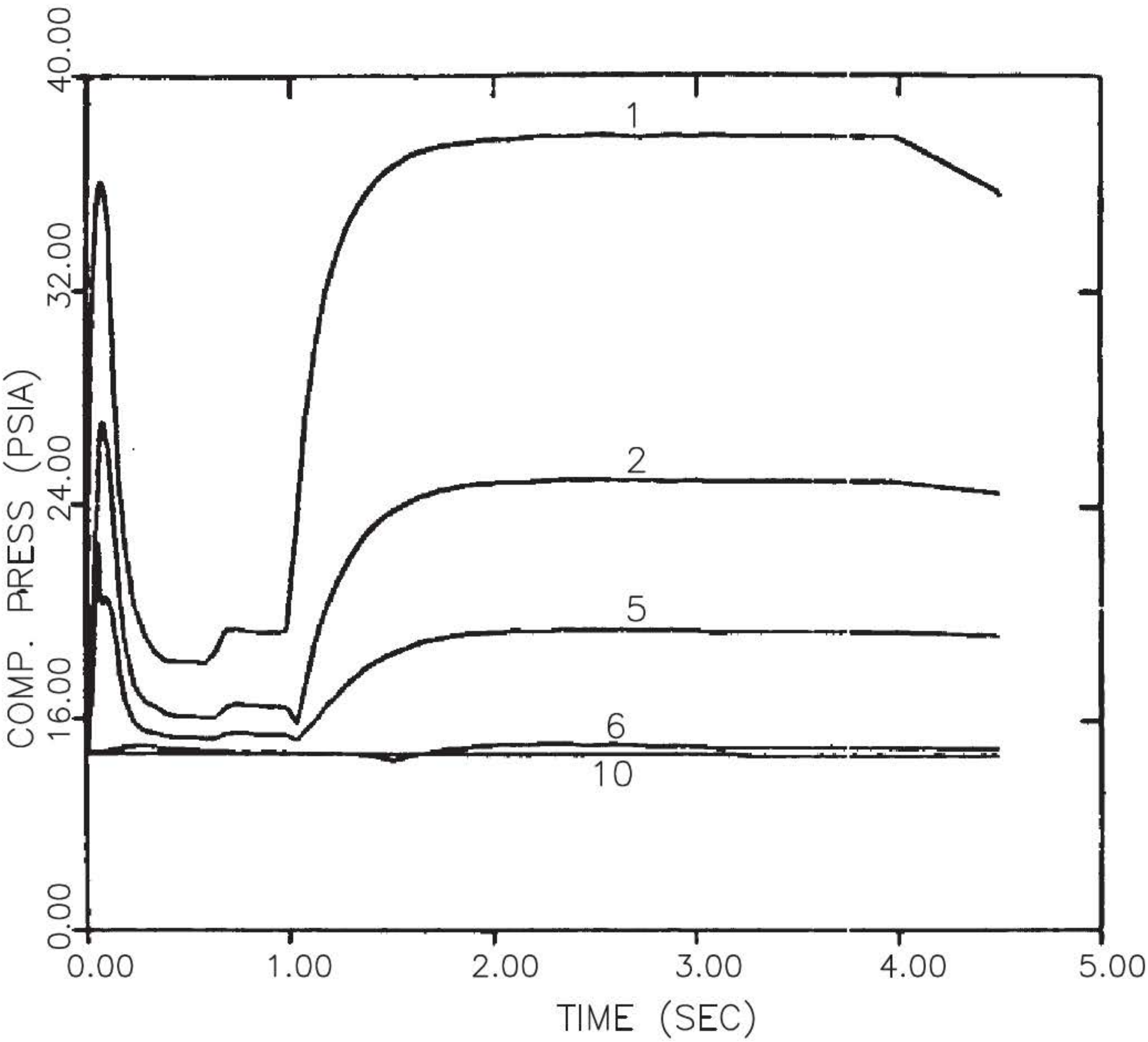
CASE B VOLUME FLOWS

FIGURE 3.6A-6, Rev. 47

Auto-Cad Figure Fsar 3_6A_6.dwg



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SUSQUEHANNA STEAM ELECTRIC STATION
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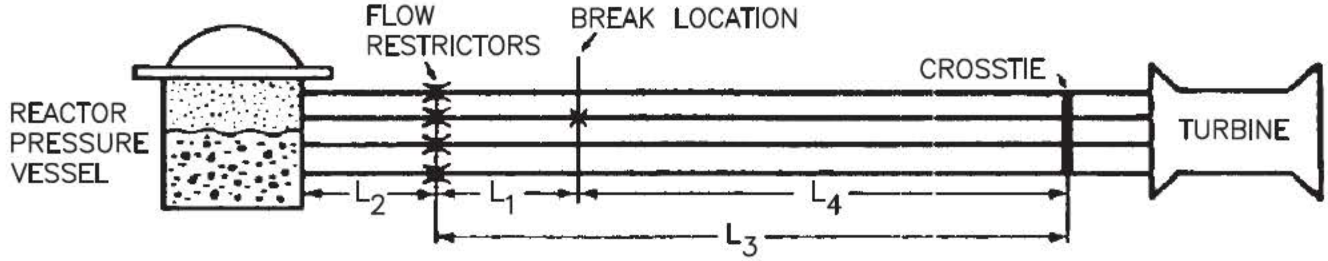
FINAL SAFETY ANALYSIS REPORT

CASE B PRESSURE TRANSIENT

FIGURE 3.6A-7, Rev. 47

Auto-Cad Figure Fsar 3_6A_7.dwg



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SUSQUEHANNA STEAM ELECTRIC STATION
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MODEL FOR DOUBLE-ENDED
GUILLOTINE

FIGURE 3.6A-8, Rev. 47

Auto-Cad Figure Fsar 3_6A_8.dwg

3.7a SEISMIC DESIGN

All systems and equipment of the NSSS are defined as either Seismic Category I or Non-Seismic Category I. The requirements for Seismic Category I classification are given in Section 3.2 along with a list of systems, components, and equipment which are so categorized.

All systems, components, and equipment related to plant safety are designed to withstand the potential safe shutdown earthquake and operating bases earthquakes.

The "Safe Shutdown Earthquake" is that earthquake which is based upon an evaluation of the maximum earthquake potential considering the regional and local geology, and seismology and specific characteristics of local subsurface material. It is that earthquake which produces the maximum vibratory ground motion for which Seismic Category I systems and components are designed to remain functional. These systems and components are those necessary to ensure:

(1) The integrity of the reactor coolant pressure boundary.

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

(3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the guidelines exposures of 10CFR 50.67.

The "Operating Basis Earthquake" is that earthquake which, considering the regional and local geology, and seismology and specific characteristics of local subsurface material, 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 these 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.

The seismic design of systems, components, and structures within the nuclear steam supply system (NSSS) scope of responsibility is presented in the following pages. The information presented in this section is intended to add to the information presented in Section 3.7b in order to better differentiate responsibilities in the seismic design of Susquehanna SES. As a result, not all subsections have a response but rather refer back to the corresponding subsection in Section 3.7b.

3.7a.1 SEISMIC INPUT

3.7a.1.1 Design Response Spectra

This subsection is covered in Subsection 3.7b.1.1.

3.7a.1.2 Design Time History

This subsection is covered in Subsection 3.7b.1.2.

3.7a.1.3 Critical Damping Values

The damping factors indicated in Table 3.7a-1 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. The values given in Table 3.7a-1 are less than or equal to those given in Regulatory Guide 1.61 and therefore are generally more conservative. See Note 1 on Table 3.7a-1 which describes the uses of higher damping values for piping systems.

3.7a.1.4 Supporting Media for Seismic Category I Structures

This subsection is covered in Subsection 3.7b.1.4.

3.7a.2 SEISMIC SYSTEM ANALYSIS

3.7a.2.1 Seismic Analysis Methods

Analysis of Seismic Category I NSSS systems and components is accomplished 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 are analyzed statically by using 1.5 times the peak acceleration of the required response spectra. In some cases, dynamic testing of equipment is used for seismic qualification.

The time history analyses involve the solution of the equations of the dynamic equilibrium (Subsection 3.7a.2.1.1) by means of the methods discussed in Subsection 3.7a.2.1.2. 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 3.7a.2.1.1) by the method discussed in Subsection 3.7a.2.1.3.

3.7a.2.1.1 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:

$$[M]\{\ddot{u}(t)\}+[C]\{\breve{u}(t)\}+[K]\{u(t)\}=\{P(t)\} \qquad \text{Eq. 3.7a - 1}$$

where:

u(t)	=	time dependent displacement of non-support points relative to the supports
$\breve{u}$ (t)	=	time dependent velocity of non-support points relative to the supports
ü(t)	=	time dependent acceleration of non-support points relative to the supports
[M]	=	diagonal matrix of lumped masses
[C]	=	damping matrix
[K]	=	stiffness matrix
P(t)	=	time dependent inertial forces acting as non-support points.

3.7a.2.1.2 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

$$[M]\{\ddot{u}(t)\} + [K]\{u(t)\} = \{0\} \qquad \text{Eq. 3.7a-2}$$

Since the free oscillations are assumed to be harmonic, the displacements can be written as

$$\{u(t)\} = \{\phi\} e^{iwt} \qquad \text{Eq. 3.7a-3}$$

where:

$\{\phi\}$	=	column matrix of the amplitude of displacements $\{u\}$
w	=	circular frequency of oscillation
t	=	time

Substituting Equation 3.7a-3 and its derivatives in Equation 3.7a-2 and noting that e^{iwt} is not necessarily zero for all values of t yields

$$[-o2\,[M] + [K]]\{\phi\} = \{0\} \qquad \text{Eq. 3.7a-4}$$

Equation 3.7a-4 is the classical algebraic eigen value problem wherein the eigen values are the frequencies of vibrations and the eigen vectors are the mode shapes, $\{\phi_i\}$.

3.7a.2.1.3 Analysis by Response Spectrum

As an alternative to the step-by-step mode superposition method described in Subsection 3.7a.2.1.2, 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 3.7a.3.7.

The total seismic structural 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 is based on the method described in Subsection 3.7b.2.6.

3.7a.2.1.4 Support Displacements in Multi-Supported Structure

The Multi-Support dynamic analysis was not used during the original design of Susquehanna SES nor was this type of analysis a requirement of the construction permit. Other analytical methods are used to demonstrate the integrity of multi-supported structures during a postulated seismic event (for structures, see Subsection 3.7b.2.1). However, independent support motion analysis is used in conjunction with Regulatory Guide 1.61 damping as one of the acceptable alternative analytical methods during snubber elimination.

3.7a.2.1.5 Dynamic Analysis of Seismic Category I Structures, Systems, and Components

Time-History Techniques or the Response Spectrum Technique are used for the dynamic analysis of Seismic Category I structures, systems, and components which are sensitive to dynamic seismic events.

3.7a.2.1.5.1 Dynamic Analysis of Piping Systems

Each pipeline is idealized as a mathematical model consisting of lumped masses connected by elastic members. The stiffness matrix for the piping system is determined using the elastic properties of the pipe. This includes the effects of torsional, bending, shear, and axial deformations as well as change in stiffness due to curved members. Next the dynamic response of the system is calculated by using the response spectrum method of analysis.

The relative displacement between anchors is determined from the dynamic analysis of the structures. The results of the relative anchor point displacement are used for a static analysis to determine the additional stresses due to relative anchor point displacements.

3.7a.2.1.5.2 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 supported at more than two points located at different elevations in the building, the response spectra for the most severe support point or spectra that envelope the response spectra of all support points is chosen as the design spectra.

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 the following subsection.

3.7a.2.1.5.2.1 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 point 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 SRSS (Square Root of Sum of the Square) Method. Since the maximum displacement for different modes do 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.

3.7a.2.1.6 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 reflect the following:

(1) 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.

(2) Test data from previously tested comparable equipment which, under similar conditions, has been subjected to dynamic loads equal to or greater than those specified.

(3) Actual testing of equipment in accordance with one of the methods described in Sections 3.9 and 3.10.

3.7a.2.2 Natural Frequencies and Response Loads

This subsection is covered in Subsection 3.7b.2.2.

3.7a.2.3 Procedure Used for Modeling

3.7a.2.3.1 Modeling Techniques for Seismic Category I Structures, Systems, and Components

An important step in the seismic analysis of Seismic Category I systems or structures is the procedure used for modeling. The systems or structures are represented by lumped masses, springs and dashpots idealizing the inertial, stiffness, and damping properties of the system. The details of the mathematical models are determined by the complexity of the actual structures and the information required for the analysis.

For information about modeling non-NSSS Seismic Category I structures, systems or components, see Subsections 3.7b.2.3 and 3.7b.3.3.

3.7a.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 models shown in Figure 3.7A-1 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. The effects of both bending and shear are included. 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 results 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 will serve 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 3.7a-1. 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 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 contribution of rotary inertia 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.

3.7a.2.3.3 Comparison of Responses

The comparison between the calculated maximum seismic loads and the allowable loads in the RPV and internals is given in Table 3.7a-2.

3.7a.2.4 Soil Structure Interaction

This subsection is covered in Subsection 3.7b.2.4.

3.7a.2.5 Development of Floor Response Spectra

This subsection is covered in Subsection 3.7b.2.5.

3.7a.2.6 Three Components of Earthquake Motion

This subsection is covered in Subsection 3.7b.2.6

3.7a.2.7 Combination of Modal Responses

This subsection is covered in Subsection 3.7b.2.7.

3.7a.2.8 Interaction of Non-Category I Structures with Seismic Category I Structures

This subsection is covered in Subsection 3.7b.2.8.

3.7a.2.9 Effects of Parameter Variations on Floor Response Spectra

This subsection is covered in Subsection 3.7b.2.9.

3.7a.2.10 Use of Constant Vertical Static Factors

This subsection is covered in Subsection 3.7b.2.10.

3.7a.2.11 Methods Used to Account for Torsional Effects

This subsection is covered under Subsection 3.7b.2.11.

3.7a.2.12 Comparison of Responses

This subsection is covered under Subsection 3.7b.2.12.

3.7a.2.13 Methods for Seismic Analysis of Dams

This subsection is covered under Subsection 3.7b.2.13.

3.7a.2.14 Determination of Seismic Category I Structure Overturning Moments

This subsection is covered under Subsection 3.7b.2.14.

3.7a.2.15 Analysis Procedure for Damping

In a linear dynamic analysis, the procedure utilized to properly account for damping in different elements of a coupled system model is as follows:

(1) 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.

(2) Perform a modal analysis of the linear system model. This will result in a modal matrix (ϕ) normalized such that $\phi_i^T K \phi_i = W_{i,}^2$ = W2i, where K is the stiffness matrix, W_i the circular natural frequency of mode i and ϕ_i^T is the transpose ϕ, which is a column vector of ϕ corresponding to the mode shape of mode i. Matrix ϕ contains all translational and rotational coordinates.

(3) Using the strain energy of the individual components as a weighting function, the following equation can be derived to obtain a suitable damping ratio (B_i) for the i^{th} mode.

$$B_j \frac{\sum_{j=1}^{N}\left[\phi_i^T B_j K_j \phi_i\right]}{W_i^2}$$

where

N	=	Total number of structural elements
ϕi	=	Mode shape for mode i (ϕ transpose)
B_j	=	Percent damping associated with element j

K_j = Stiffness contribution of element j

W_i = Circular natural frequency of mode i

3.7a.3 SEISMIC SUBSYSTEM ANALYSIS

3.7a.3.1 Seismic Analysis Methods See Subsection 3.7a.2.1

3.7a.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-reactor dynamic model was excited by three different recorded time histories - May 18, 1940, El Centro NS component, 29.4 sec; 1952, Taft N69°W component, 30 sec; and March 1957, Golden Gate S80°E component, 13.2 sec. The model response was truncated such that the response of three different frequency bandwidths could be studied, 0^+-10 Hz, 10-20 Hz, and 20-50 Hz. This was done to give an 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 as given in Table 3.7a-3 was formed.

Independent of earthquake or component frequency, 99.5% of the stress reversals occur below 75% of the maximum stress level, and 95% of the reversals lie below 50% of the maximum stress level. This relationship is graphically shown in Figure 3.7A-2.

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 are found from Table 3.7a-3 according to the frequency range within which the fundamental frequency lies.

(c) For fatigue evaluation, one-half percent (0.005) of these cycles are conservatively assumed to be at the peak load 4.5% (0.045%) at three-quarter peak. The remainder of the cycles will 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 operating life of a plant. Fatigue evaluation due to the SSE is not necessary since it is a faulted condition and thus not required by ASME Section III.

The OBE is an upset condition and therefore, must be included in fatigue evaluations according to ASME Section III. Investigation of seismic histories for many plants show 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% of the proposed SSE intensity, will occur. Therefore, the probability of even an OBE is extremely low. To cover the combined effects of these earthquakes and the cumulative effects of even lesser earthquakes, one OBE intensity earthquake with 10 peak stress cycles is postulated for fatigue evaluation.

3.7a.3.3 Procedure Used for Modeling

3.7a.3.3.1 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 member, representing the physical properties of each segment. The pipe lengths between mass points will be 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.

3.7a.3.3.2 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:

(1) 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 as significant if the corresponding natural frequencies are less than 33 Hz and the stress calculated from these modes are greater than 10% of the total stresses obtained from lower modes.

(2) 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.

(3) 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.

(4) 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.

3.7a.3.3.3 Location of Supports and Restraints

The location of seismic supports and restraints for Seismic Category I piping and piping systems components is selected to satisfy the following two conditions:

(1) The location selected must furnish the required response to control stress and/or strain within allowable limits.

(2) Adequate building strength for attachment of the components must be available.

3.7a.3.4 Basis of Selection of Frequencies

All frequencies in the range of 0.25 to 33 Hz are considered in the analysis and testing of structures, systems, and components. The frequency range of between 0.25 Hz and 33 Hz covers the range of the broad band response spectrum used in the design. 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.

3.7a.3.5 Use of Equivalent Static Load Method of Analysis

This subsection is covered under Subsection 3.7b.3.5.

3.7a.3.6 Three Components of Earthquake Motion

3.7a.3.6.1 Response Spectrum Method

The use of three components of earthquake motion was not a design basis requirement of the construction permit for this plant. The total seismic response is predicted by combining the response calculated from analyses due to one horizontal and one vertical seismic input. For this case, where the response spectrum method of seismic analysis is used, the basis for continuing the loads from the two analyses is given below:

(1) The peak responses of the different modes for the same earthquake excitations do not occur at the same time.

(2) The peak responses of a particular move due to earthquake excitations from different directions do not occur at the same time.

(3) The peak stresses 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, the two translation components of earthquake excitations are combined by summing the absolute sum of all responses of interest (e.g., strain, displacement stress, moment, shear, etc.) from seismic motion, the one horizontal (x or z) and one vertical direction (y), i.e., |x+y| or |y+z|. The design is made for the larger of the two sums |x+y| or |y+z|.

3.7a.3.6.2 Time History Method

The algebraic sum of contributions (to displacements, loads, stresses, etc.) due to the two earthquake components are calculated for each natural mode for each time interval of analysis. The time interval should be less than or equal to 0.2 of the smallest period of interest. The maximum of the algebraically summed values (displacements, loads, stresses) over all time intervals are the design displacements, accelerations, loads, or stresses.

The above method demonstrates the integrity of the Seismic Category I subsystems.

3.7a.3.7 Combination of Modal Responses

When the response spectra method of modal analysis is used, all modes are combined by the square root of the sum of the squares (SRSS) method. When the response spectra method of modal analysis is used for snubber elimination or other piping modifications, modal combinations shall be in accordance with Regulatory Guide 1.92 whenever Code Case N-411 or Regulatory Guide 1.61 is invoked for damping values.

3.7a.3.8 Analytical Procedure for Piping

The analytical procedures for piping analysis have been described in Subsection 3.7a.2.1.5.1.

3.7a.3.9 Multiply Supported Equipment Components with Distinct Inputs

The procedure and criteria for analysis has been described in Subsection 3.7a.2.1.5.2.

3.7a.3.10 Use of Constant Vertical Static Factors

This subsection is covered under Subsection 3.7b.3.10.

3.7a.3.11 Torsional Effects of Eccentric Masses

Torsional effects of eccentric masses are discussed in Subsection 3.7a.3.3.1.

3.7a.3.12 Buried Seismic Category I Piping Systems and Tunnels

This subsection is covered under Subsection 3.7b.3.12.

3.7a.3.13 Interaction of Other Piping with Seismic Category I Piping

When other piping is attached to Seismic Category I piping, the other piping is analytically simulated in a manner that does not 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.

3.7a.3.14 Seismic Analysis for Reactor Internals

The modeling of RPV internals has been discussed in Subsection 3.7a.2.3.2. The damping values are given in Table 3.7a-1. A comparison of responses is shown in Table 3.7a-2.

3.7a.3.15 Analysis Procedures for Damping

Analysis procedures for damping have been discussed in Subsection 3.7a.2.15.

3.7a.4 SEISMIC INSTRUMENTATION

This subsection is covered under Subsection 3.7b.4.

3.7a.5 REFERENCES

3.7a-1 L. K. Liu, "Seismic Analysis of Boiling Water Reactor," Symposium on Seismic Analysis of Pressure Vessel and Piping Components, First National Congress on Pressure Vessel and Piping, San Francisco, California, May 1971.

TABLE 3.7a-1

CRITICAL DAMPING RATIOS FOR DIFFERENT MATERIALS

Item	Percent Critical Damping	
	OBE Condition	SSE Condition
Reinforced concrete structures	2.0	5.0
Welded structural assemblies (equipment and supports)	1.0	2.0
Bolted or riveted structural assemblies	2.0	3.0
Vital piping systems	0.5	1.0
Drywell-Building (Coupled)	2.0	5.0
Reactor pressure vessel, support skirt, shroud head, separator and guide tubes	2.0	2.0
Control rod drive housings	3.5	3.5
Fuel	7.0	7.0
Steel frame structures	2.0	3.0

Other values may be used if they are indicated to be reliable by experiment or study.

NOTE: For snubber elimination or other piping modifications, damping values per Code Case N-411 or Regulatory Guide 1.61 may be applied. When either Code Case N-411 or Regulatory Guide 1.61 is invoked, modal combination for closely spaced modes per Regulatory Guide 1.92 shall be applied.

Table 3.7a-2

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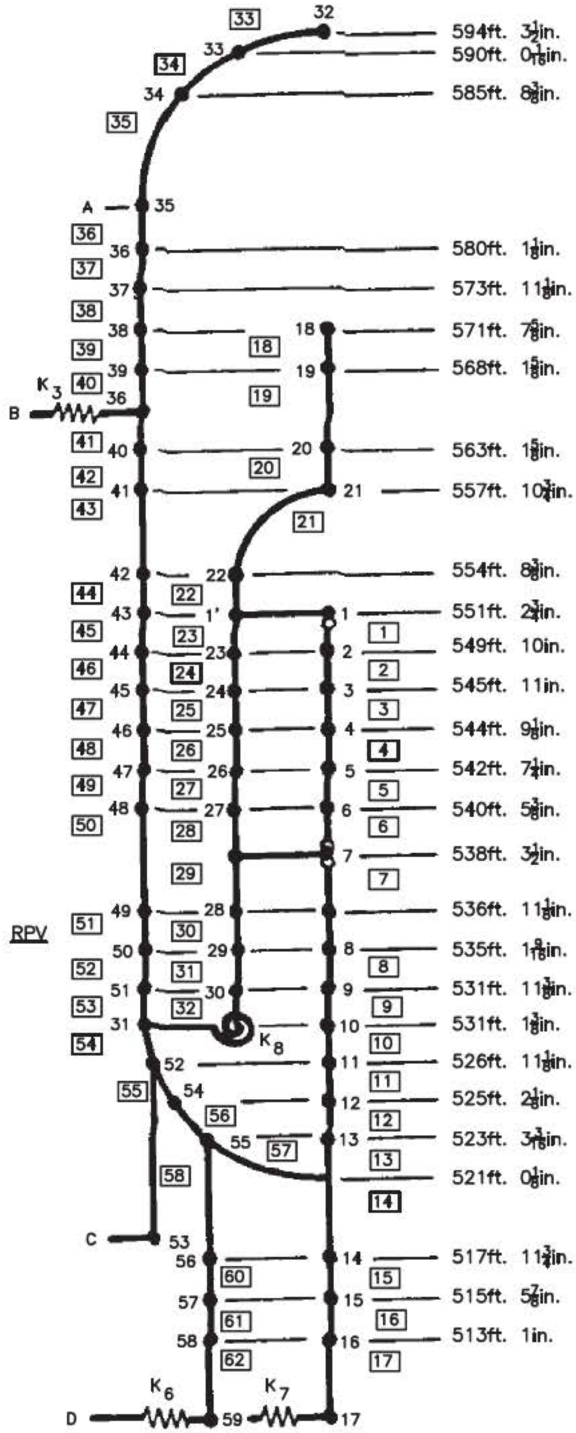
TABLE 3.7a-3

NUMBER OF DYNAMIC RESPONSE CYCLES EXPECTED DURING A SEISMIC EVENT

Frequency Band (Hz)	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



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SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
REACTOR PRESSURE VESSEL AND INTERNAL SEISMIC MODEL FIGURE 3.7A-1, Rev. 47

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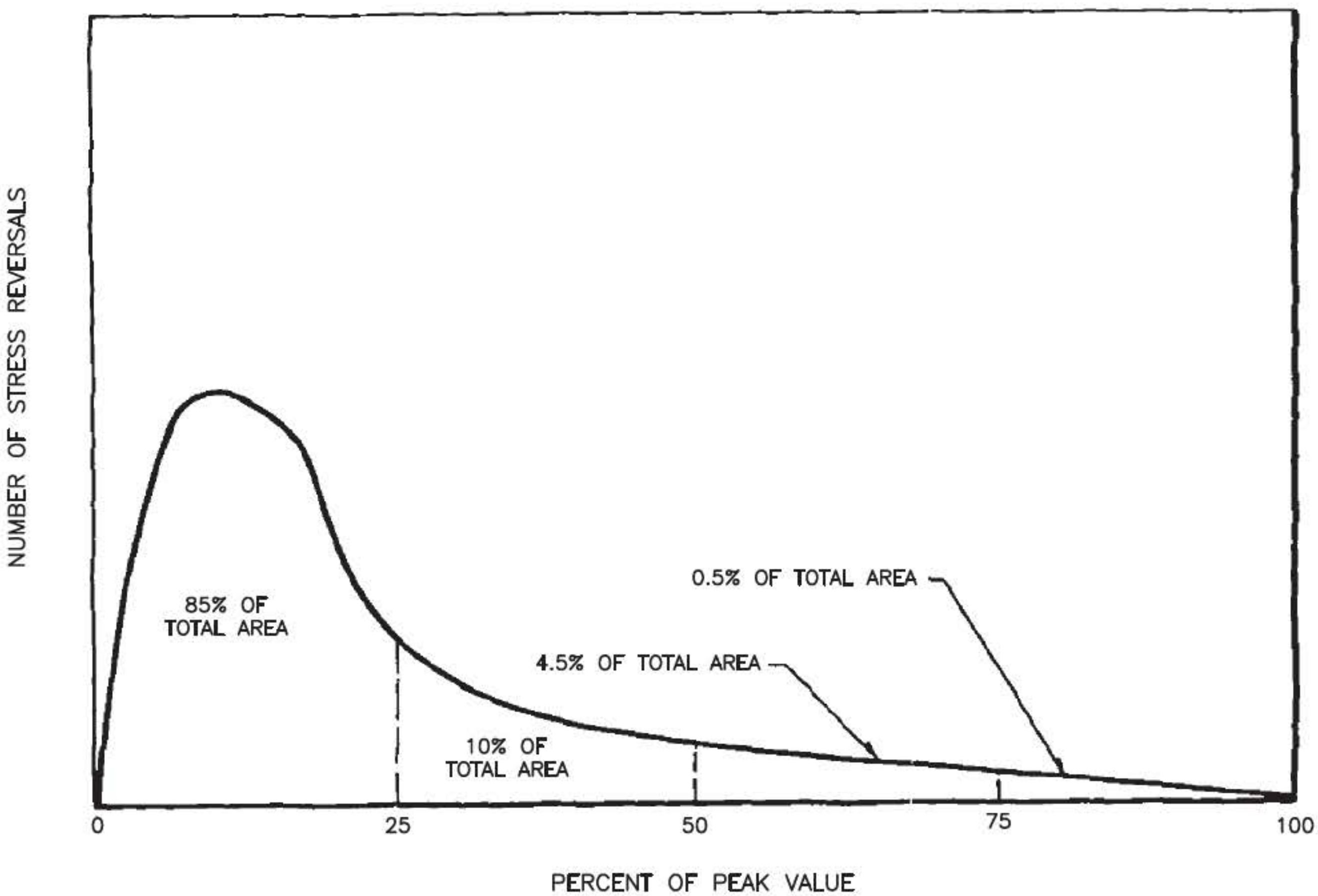

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UNITS 1 & 2
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DENSITY OF STRESS REVERSALS

FIGURE 3.7A-2, Rev. 47

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3.7b SEISMIC DESIGN

This section describes the seismic design requirements and methods used for Susquehanna SES and the seismic design and analysis of non-NSSS equipment. Seismic design of NSSS equipment is described in Section 3.7a.

3.7b.1 SEISMIC INPUT

3.7b.1.1 Design Response Spectra

The site design response spectra for all rock founded structures except the Diesel Generator 'E' Building are illustrated on Figures 3.7B-1 and 3.7B-2 for the horizontal components of the Operating Basis Earthquake (OBE) and Safe Shutdown Earthquake (SSE) respectively. For the Diesel Generator 'E' Building, the horizontal site design response spectra are based on Regulatory Guide 1.60, Rev. 1 and are illustrated on Figures 3.7B-2 and 3.7B-4. The design earthquake is assumed to be the free field motion at the base mat of the structure without the effect of the structure. For all Seismic Category I structures founded on rock the maximum horizontal ground acceleration values are 5 and 10 percent of gravity for OBE and SSE respectively (refer to Subsections 2.5.2.6 and 2.5.2.7). However, Seismic Category I structures founded on soil, and the spray pond have been designed for maximum horizontal ground accelerations of 8 percent (OBE) and 15 percent (SSE) of gravity. The maximum ground displacement is taken proportional to the maximum ground acceleration. The displacement associated with a 1.0 gravity ground acceleration is set at 40 inches for all Seismic Category I structures except the Diesel Generator 'E' Building where it is set at 36 inches.

The base diagram of all design spectra consists of three parts: the maximum ground acceleration line on the left part, the maximum ground displacement line on the right part, and the middle part depends on the maximum pseudo-velocity.

For various damping values, the numerical values of design displacements and accelerations for the horizontal component design response spectra used for all Seismic Category I structures except the Diesel Generator 'E' Building are obtained by multiplying the values of the maximum ground displacement and acceleration by the corresponding factors given in Table 3.7b-1. Table 3.7b-2 provides the amplification factors for the horizontal and vertical design response spectra associated with the Diesel Generator 'E' Building.

The acceleration lines of the design response spectra are drawn parallel to the maximum ground acceleration line between the frequency lines of 6.67 cps (control point B of Figures 3.7B-1 and 3.7B-2) and 2 cps (control point C). The acceleration lines converge at the junction of the maximum ground acceleration line and the 33 cps frequency line (control point A). For frequencies higher than 33 cps, the maximum ground acceleration line represents the design response spectra. The displacement lines are drawn parallel to the maximum ground displacement line. The maximum pseudo-velocity is assumed to be constant. Lines were drawn parallel to the constant velocity lines connecting the acceleration lines at control point C and the displacement lines.

For all Seismic Category I structures except the Diesel Generator 'E' building, the design response spectra values for the vertical component of the earthquake are taken as 2/3 of the corresponding values of the horizontal component of the earthquake.

The site design spectra for all Seismic Category I structures except the Diesel Generator 'E' Building deviate from those suggested in Regulatory Guide 1.60. Figures 3.7B-88 through 3.7B-91 provide comparison of the two. The damping values for the NRC spectra are those specified by Regulatory Guide 1.61 for reinforced concrete structures.

Both the horizontal and vertical site design spectra for the Diesel Generator 'E' Building are based on Regulatory Guide 1.60, Rev. 1. The vertical ground acceleration values are the same as the horizontal ground acceleration values.

3.7b.1.2 Design Time History

A synthetic time history motion for all Seismic Category I structures, except the Diesel Generator 'E' Building, is generated by modifying the actual records of the 1952 Taft earthquake according to the techniques proposed in Reference 3.7b-1. Figure 3.7B-5 shows the normalized synthetic time history motion. The duration of the time history is 20 sec. The time interval of the time history is 0.005 sec.

Figures 3.7B-8 and 3.7B-9 show a comparison of the time history response spectra and the design response spectra for 2, 3, 5, and 7 percent damping values. The spectra are computed at the following frequency values (in cps):

0.2 to 1.0 (increment of 0.05)

1.0 to 10.0 (increment of 0.1)

10.0 to 30.0 (increment of 1.0)

Figure 3.7B-10 shows a comparison of the time history response spectra and the design response spectra for 2 and 5 percent damping values for a frequency range between 0.2 and 1.0 cps, with intervals of 0.0125 cps. All the above figures show that the time history response spectra envelop the design response spectra.

The synthetic time history motions for the Diesel Generator 'E' Building are generated from noise and are not based on actual earthquake recordings. Figures 3.7B-6 and 3.7B-7 show the horizontal and vertical synthetic time history motions, respectively. The duration of these time histories is 25 seconds. The time interval of these time histories is 0.01 seconds.
Figures 3.7B-11 through 3.7B-16 show a comparison of the time history response spectra and the design response spectra for the horizontal and vertical directions at 2, 5 and 7 percent damping values. The spectra are computed at the frequencies suggested in Standard Review Plan 3.7.1, July 1981. Figures 3.7B-11 through 3.7B-16 show that the time history response spectra meet the acceptance criteria described in the referenced Standard Review Plan.

3.7b.1.3 Critical Damping Values (Non-NSSS)

Table 3.7b-3 summarizes the damping values used on Susquehanna SES except for the Diesel Generator 'E' facility. They are expressed as a percentage of critical damping and are based on Reference 3.7b-2. For the Diesel Generator 'E' facility, the damping values are based on Regulatory Guide 1.61, Rev. 0 and are summarized in Table 3.7b-4.

The ESSW pumphouse, piping to the reactor building, the spray pond and the Diesel Generator 'E' fuel tank are some of the Seismic Category I structures and systems founded on soil. The equivalent spring constants and the soil damping coefficients used in the analysis of the ESSW pumphouse are shown in Table 3.7b-5. These values are based on formulae contained in Table 3-2 of Reference 03.7b-3. A lumped representation of soil structure interaction was used.

Soil structure interaction is also considered in the generation of the response spectra for the containment. As in the ESSW pumphouse, a lumped representation of the soil structure interaction is considered. Table 3.7b-5 shows the equivalent spring and damping coefficients used in the containment model.

3.7b.1.4 Supporting Media for Seismic Category I Structures

All Seismic Category I structures, with the exception of ESSW pumphouse, the spray pond, and its pipe supports, the Diesel Generator ‘E’ Fuel Oil Tank, miscellaneous structures and other buried pipes are founded on rock. For the structural analysis of the rock based structures, soil structure interaction is considered to be negligible due to the high stiffness of the rock, which has a modulus of elasticity of approximately 3.0×10^6 psi. However, the response spectra of the containment are derived from a model that considers the flexibility of the rock.

The properties of the rock and soil supporting the ESSW pumphouse are shown in Table 3.7b-6. Discussion of the embedment of structures in soil will be limited to the ESSW pumphouse, since all the other structures are founded on rock.

The ESSW pumphouse is 59 ft high and rests on a 64 ft x 112 ft reinforced concrete mat foundation. The embedment depth of the foundation is 29 ft. The depth of soil below the mat foundation varies from 35 to 60 ft. The soil is predominantly sand, gravel, cobbles, and boulders. Near the surface, the soil is primarily sand and sandy gravel. With increasing depth, the soil changes to more cobbles and boulders. Near bedrock, the soil is mostly cobbles and boulders.

The site geology is discussed in detail in Section 2.5.

3.7b.2 SEISMIC SYSTEM ANALYSIS

Section 3.2 identifies Seismic Category I structures, systems, and components. Seismic Category I structures are considered seismic systems and are discussed here. Seismic Category I systems and components are considered seismic subsystems and are discussed in Subsection 3.7b.3. Seismic systems are analyzed for both the OBE and SSE.

3.7b.2.1 Seismic Analysis Methods

The response spectrum method is used for seismic analysis of Seismic Category I structures. A description of the method is given in Section 4.2.1 of Reference 3.7b-3 for all Seismic Category I structures except the Diesel Generator 'E' Building where it is given in Section 6 of Reference 3.7b-21. Separate lateral and vertical analyses of structures are performed. The responses are then combined to predict the total response of the structure.

A time history analysis of the Seismic Category I structures is done to generate the response spectra at the various mass points of the model.

The mathematical models used for these analyses are lumped mass, stick models. The same models were used for both the response spectrum and time history analyses. The mathematical models of the reactor and control building are shown on Figures 3.7B-19, through 3.7B-21.

For all models, the masses are located at elevations of mass concentrations, such as floors and roofs. However, in the case of the containment which is a structure of continuous mass distribution, masses are lumped at variable intervals ranging from 6.6 feet to 15.7 feet along the containment shell and reactor pedestal. These methods of mass distribution are in accordance with the procedures of Section 3.2 of Reference 3.7b-3 to provide an adequate number of masses. The mathematical models of the containment are shown on Figures 3.7b-17 and 3.7-18.

The reactor and control buildings act as a single structure due to the monolithic construction. The entire reactor and control building structure is shown as a single unit in Figure 3.7B-22. Both the control building and the line 29 wall of the reactor building are connected to the P-line wall, which is common to both the reactor and control buildings. In the east-west direction, the control building and the line 29 wall are considered to respond as a single unit.

The horizontal mathematical models are shown on Figures 3.7B-19 and 3.7B-20. The sticks represent shear walls located at the base mat elevation in the reactor building in the direction of the earthquake motion. In the east-west model (Figure 3.7B-19), the control building is lumped entirely on the line 29 stick. The entire control building is considered to contribute to the stiffness of the line 29 stick. In the North-South direction (Figure 3.7b-20), the control building has its own stick connected to the P-line wall by springs.

The springs between the sticks represent the flexibility of the floor slab connecting each stick. Since these springs act in the direction of the earthquake motion, the model allows relative displacement between sticks. Figure 3.7B-21 shows the vertical earthquake model of the reactor and control buildings. The left stick represents the steel columns. The right stick represents the concrete walls of both the reactor and control buildings. The floors are represented by lumped masses and beam elements with the appropriate stiffness to capture the out of plane flexural vibration. Vertical translational coupling springs are provided to represent the coupling stiffness of the floor slab between the wall and column sticks. Mass numbers 8, 55, and 57 represent the fuel pool girder masses. Mass numbers 34, 35, 41, 43, 44, 46, 53 and 54 represent the floors between the fuel pool girders and columns/walls. Figure 3.7B-23 shows the correlation between the model mass points and the actual structure.

To more accurately determine the dynamic characteristics of the mathematical models the modulus of elasticity for concrete used in the analysis, is determined based on test results of concrete samples obtained from the plant site. The modulus value used is 720,000 ksf for all Seismic Category I structures except the Diesel Generator 'E' building where it was taken to be 518,400 ksf.

The seismic analysis of the Seismic Category I structures considers all modes whose frequencies are less than 33 cps. However, if a structure has only one or two modes with a natural frequency below 33 cps, then the three lowest modes are used. If a structure has three or less degrees of freedom, then all modes are considered in the analysis. For the Diesel Generator 'E' Building and its pedestal, all modes were considered.

The Seismic Category I structures are supported by continuous base mats; therefore, relative displacement of supports is not a consideration.

Nonlinear responses are not considered since the Seismic Category I structures are designed to remain elastic.

3.7b.2.1.1 Flexible Base and Fixed Base Containment Models

The original structural design of the containment was based uupon results obtained from a fixed base model of the containment. The fixed base model used a damping value of 5% of critical damping for all structural modes. The utilization of a fixed base model can be justified since the containment is founded on hard competent rock.

At a later date, a flexible base model of the containment was developed. The flexible base model of the containment is more realistic since it takes into account soil-structure interaction effects. The flexible base containment model used composite modal damping as described in reference 3.7b-3, (BC-TOP-4A, Rev. 3, Appendix D). Analyses were performed using the flexible base model to generate structural response spectra for evaluation of equipment, piping systems, etc.

Both models are fully in accordance with the requirements in Reference 3.7b-3, which has been approved by the NRC. For information regarding the comparison of results from the fixed base and flexible base models, see FSAR Section 3.7b.2.2.1, Revision 46 and previous revisions.

NSSS equipment qualified by GE used loads obtained from the fixed base model. All subsequent structural assessments have used loads derived from the more realistic flexible base model throughout. All future analyses shall use the loads derived from the more realistic flexible model. All remaining discussions regarding the containment presented in the FSAR are for the flexible base model.

3.7b.2.2 Natural Frequencies and Response Loads

The natural frequencies of the containment and the reactor and control building below 33 cps are shown in Tables 3.7b-7 and 3.7b-8 respectively. The first seven frequencies of the reactor and control building in the east-west direction are dependent upon the location of the reactor building cranes.

Some of the significant mode shapes of the containment and the reactor and control building are shown on Figures 3.7B-24 through 3.7B-39. The mode shapes for containment are for the horizontal and vertical directions. The reactor and control building mode shapes are for each of the three principal directions: east-west, north-south, and vertical. As with the frequencies, the first seven mode shapes of the reactor and control building in the east-west direction depend on the location of the cranes. Figures 3.7B-30 through 3.7B-34 show that it is the superstructure of the reactor building that is excited at these low frequencies. The location of the cranes is noted on the figures.

Figures 3.7B-40 through 3.7B-47 show the response displacements and accelerations of the containment for both OBE and SSE. The response of the reactor and control building is shown on Figures 3.7B-48 through 3.7B-59.

Response spectra at critical locations are shown on Figures 3.7B-60 through 3.7B-87. The curves are shown for each of the three principal directions at the damping values used for each design earthquake (see Subsection 3.7b.2.15 for further discussion of damping values). A brief description of the location of each series of curves is provided below with the corresponding figure numbers.

Figures 3.7B-60 through 3.7B-63	RPV Pedestal
Figures 3.7B-64 through 3.7B-69,	Refueling Area
Figures 3.7B-70 through 3.7B-81	Diesel Generator 'A-D' and 'E' Pedestals
Figures 3.7B-82 through 3.7B-87	Operating Floor of ESSW Pumphouse

3.7b.2.3 Procedure Used for Modeling

Seismic systems and subsystems were defined in Subsection 3.7b.2.

All equipment, components, and piping systems are lumped into the supporting structure mass except for the reactor vessel, which is analyzed using a coupled model of the containment structure and the reactor vessel (refer to Figures 3.7B-17 and 3.7B-18). See Section 3.2 of reference 3.7b-3 for the criteria of lumping the equipment, components and piping systems into the supporting structure mass.

Adequacy of the number of masses and degrees of freedom is discussed in Subsection 3.7b.2.1.

Each Seismic Category I structure is considered to be independent because of a gap between adjacent structures. For example, there is a 2 in. horizontal gap between the reactor and control building and the containment above the foundation mat.

To form these gaps rodofoam material (Ref. 3.7b-12) was used. Rodofoam was left in place in the following areas:

(1) Joints where the provided actual gap is 0.5 inch greater than that originally specified on the civil drawings.

(2) Joints where the interaction forces between structures due to presence of rodofoam cause insignificant effect on shear and moment.

3.7b.2.4 Soil Structure Interaction

All Seismic Category I structures, except the ESSW pumphouse and spray pond, are founded on rock. The seismic analysis of these structures is done assuming a fixed base. As stated in Subsection 3.7b.2.1, the containment response spectrum curves are generated from a flexible base model. The rock is assumed to be a homogeneous material comprising an entire elastic half-space. The soil springs and dampers used to represent the effect of the soil are discussed in Subsection 3.7b.1.3.

The ESSW pumphouse is supported by natural soil formation; consequently, soil structure interaction has been considered in the analysis of the pumphouse.Information regarding soil characteristics, foundation embedment, etc., is contained in Subsection 3.7b.1.4. The soil structure interaction analysis is performed using the lumped spring approach. The soil is considered a homogeneous material. The equivalent spring constants and the soil damping coefficients are discussed in Subsection 3.7b.1.3.

The seismic analysis of the spray pond is discussed in Subsection 2.5.5.

3.7b.2.5 Development of Floor Response Spectra

A time history analysis is used to develop the floor response spectra. The mathematical models used for this analysis are discussed in Subsections 3.7b.2.1, 3.7b.2.3, and 3.7b.2.4.

The floor response spectra for all Seismic Category I structures except the Diesel Generator 'E' Building are calculated at the frequencies listed in Table 5-1 of Reference 3.7b-3. For the Diesel Generator 'E' Building, the floor response spectra are calculated at the frequencies recommended in Regulatory Guide 1.122, Rev. 1. Structural frequencies up to 33 cps are used.

3.7b.2.6 Three Components of Earthquake Motion

Independent analyses are done for the vertical and two horizontal (east-west and north-south) directions. For design purposes, the response value used for all Seismic Category I structures except the Diesel Generator 'E' Building is the maximum value obtained by adding the response due to vertical earthquake with the larger value of the response due to one of the horizontal earthquakes by the absolute sum method. For the Diesel Generator 'E' Building, the responses due to three simultaneous orthogonal components of an earthquake are combined by the square root of the sum of the squares method per Regulatory Guide 1.92, Rev. 1.

3.7b.2.7 Combination of Modal Responses

The modal responses, i.e., shears, moments, deflections, accelerations, and inertia forces, are combined by either the sum of the absolute values method or by the square root of the sum of the squares method. When the latter method is used in all Seismic Category I structures except the Diesel Generator 'E' Building, the absolute values of closely spaced modes for each group are added first and then combined with the other modes or groups of closely spaced modes by the square root of the sum of the squares method. Two consecutive modes are defined as closely spaced when their frequencies differ from each other by 0.5 cps or less.

The definition for closely spaced modes was established for the Susquehanna Project in November, 1974 (Reference Question C-11 of PSAR Amendment #16.) It can be seen from Table 3.7b-7 that the natural frequencies of the containment are so widely spaced that they are not closely spaced modes based on the SRP definition for closely spaced modes. For the reactor and control buildings (see Table 3.7b-8) where frequencies are not widely spaced, the model responses are combined by the absolute sum method.

For the Diesel Generator 'E' Building, the total response is obtained by combining the absolute values of all closely spaced modal responses with the square root of the sum of squares of the remaining modal responses. Two consecutive modes are defined as closely spaced when their frequencies differ from each other by 10 percent or less (reference: Regulatory Guide 1-92).

3.7b.2.8 Interaction of Non-Category I Structures with Seismic Category I Structures

Non-Category I structures that are close to Seismic Category I structures, the turbine and radwaste buildings, have been designed to withstand an SSE. Dynamic analyses of these structures were done by the response spectrum method.

The remaining non-Category I structures were designed for seismic loads according to the UBC (Ref. 3.7b-4). The collapse of any of these remaining non-Category I structures will not cause the failure of a Seismic Category I structure.

Structural separations have been provided to ensure that interaction between Category I and non-Category I structures does not occur. The minimum separation at any point is maintained at one and a half times the absolute sum of the predicted maximum displacements of the two structures.

The rodofoam material that was used to form the separation gaps was left in place in some areas as mentioned in Section 3.7b.2.3.

3.7b.2.9 Effects of Parameter Variations on Floor Response Spectra

To account for variations in the structural frequencies owing to uncertainties in the material properties of the structure and to approximations in the modeling techniques used in the seismic analysis, the computed floor response spectra are smoothed and peaks associated with each of the structural frequencies are broadened. The parameters, which are considered variable, are the masses, the modulus of elasticity of the material, and the cross-sectional properties of the members. In addition, variation in the structural frequency is also taken into account because the base of the structures may not be fully fixed as assumed in the analysis.

Let

nf	=	Natural frequency of the building at a peak value of the floor response spectra
Δnf	=	Total variation in nf
Δnf_m	=	Variation in nf due to variation in the mass
Δnf_e	=	Variation in nf due to variation in the modulus of elasticity of the material
Δnf_s	=	Variation in nf due to variation in the cross-sectional properties of the members

A factor of 0.05 is used to account for the decrease in nf due to the possibility that the base of the structures may not be fully fixed.

Since it is highly improbable that the maximum variations in the individual parameters would occur simultaneously, Δ nf is determined by the square root of the sum of the squares of the individual variations as follows:

The maximum increase in nf is given by:

$$+\Delta nf = [(\Delta nf_m)^2 + (\Delta nf_e)^2 + (\Delta nf_s)^2]\ 2$$

$$-\Delta nf = [(\Delta nf_m)^2 + (\Delta nf_e)^2 + (\Delta nf_s)^2 + (0.05)_2]\ 2$$

For all Seismic Category I structures, except the Diesel Generator 'E' Building, the following values of $\pm\Delta nf$ are used:

$+\Delta nf = 0.12$ nf

$-\Delta nf = -0.14$ nf

For the Diesel Generator 'E' Building, the computed floor response spectra were smoothed and peak width associated with each structural frequency was increased by $\pm$15 percent.

3.7b.2.10 Use of Constant Vertical Static Factors

Constant vertical static factors are not used in the seismic design of Seismic Category I structures. The methodology used for the vertical seismic analysis is similar to the horizontal analysis.

3.7b.2.11 Methods Used To Account for Torsional Effects

Torsional effects for the diesel generator buildings and ESSW pumphouse are accounted as follows:

A static analysis was done to account for torsion on the Diesel Generator 'A-D' Building and ESSW pumphouse. For the ESSW pumphouse the eccentricity was determined by the distance between the center of mass and the center of rigidity of the structure. The inertia force from the response spectrum analysis was applied at the center of mass. The resulting torsional moment is equal to the inertial force multiplied by the eccentricity. The shear forces due to the torsional moment were then distributed to the walls. The torsional shear forces are distributed according to the method described in Section 3.4 of Reference 3.7b-5.

In the Diesel Generator 'A-D' Building, torsion is considered due to the eccentricity caused by the difference in rigidities of the east and west shear walls. The torsional shear forces are assumed to be taken entirely by east and west walls only.

For the Diesel Generator 'E' Building, the torsional effects due to its asymmetry are accounted for by lumping the floor masses at their respective center of gravity in the mathematical model of the building discussed in Section 3.7b.2.1. The stiffness matrix is calculated at these mass points and thus reflects the actual asymmetrical building configuration including the various wall openings. To account for accidental torsion, an additional torsional moment, produced by an eccentricity of $\pm$5 percent of the maximum building dimension, is added to the gross torsional moment obtained from the dynamic analysis of the above mathematical model. The mathematical model of the diesel generator ‘E’ building is shown in Figure 3.7B-95.

Torsional effects are negligible for the containment because of the symmetry of the structure.

The reactor/control building is modeled for horizontal dynamic analysis as multiple sticks coupled by springs representing the shear stiffness of the floor slabs. Each stick represents a major structural shear wall. The mass and stiffness distribution of the structural walls is such that torsional effects are properly represented in the dynamic analysis.

Torsional effects for the Diesel Generator 'A-D' Building, ESSW pumphouse, and reactor/control building are also discussed in response to NRC questions 130.21 and 130.22.

3.7b.2.11.1 Torsional Analysis of Diesel Generator Building A-D and ESSW Pumphouse

During the dynamic analysis state, the inertia force at each mass was considered to be applied at the center of mass. However, since the center of rigidity does not coincide with the center of mass, there is torsion. The inertia force obtained from the dynamic analysis was used by multiplying it with the eccentricity (the distance between the center of mass and the center of rigidity) to obtain the torsional moment. This moment was then distributed to the structural walls for assessment.

A minimum eccentricity of 5% was considered.

(i) The eccentricities of these structures were calculated.

(ii) The structures were represented by fixed base 3-D stick models with structural masses properly lumped at the calculated eccentricities, as shown in Figures 3.7B-93 and 3.7B-94.

(iii) Modal frequency analyses of the 3-D stick models were performed to determine the structure frequencies.

(iv) The frequencies determined are then compared with the corresponding frequencies associated with the fixed base models having zero eccentricities.

The results of comparison for the ESSW Pumphouse is shown on Table 3.7b-9 and for the Diesel Generator Building is shown on Table 3.7b-10. These results indicate that there are insignificant shifts in the structural frequencies by including the eccentricities in the dynamic analysis.

From the results of this study, it is concluded that the structures modeled by lumped stick models without the inclusion of eccentricities in the dynamic analysis is adequate for the prediction of desired structural responses.

Table 3.7b-11 shows the comparisons of torsional moments for SSE obtained from the studies made using 3-D stick model with the torsional moments used in the original analysis. Evaluation of the comparisons is shown as follows:

(1) Torsional moment used in the original design of ESSW Pumphouse is higher than the torsional moments computed from 3-D stick model results. Therefore, the original design is adequate.

(2) Torsional moments used in the original design of Diesel Generator building are lower than the torsional moments computed from the 3-D stick results. However, the stresses computed from the higher torsional moments result in a maximum shear stress of 16 psi which gives a maximum total shear stress of 74 psi due to torsion and direct shear, compared to an allowable of 126 psi. Thus, the original design of the diesel generator building is adequate.

3.7b.2.11.2 Torsional Analysis of the Reactor/Control Building

The torsional effect in the reactor/control building was considered in the dynamic analysis. Units 1 and 2 were considered simultaneously.

In the N-S direction, the eccentricity is larger than 5%. The N-S dynamic model presented on Figure 3.7B-20 consists of three sticks at each floor and the stiffness distribution of the structural walls are such that proper representation of the eccentricity is obtained. Therefore, the torsional effect is properly accounted for in the dynamic analysis. The computed dynamic member forces and modal point responses were used for the assessment of structure and equipment.

In the E-W direction (see seismic model on Figure 3.7B-19), the eccentricity is less than 5%. However, a minimum eccentricity of 5% was considered by redistributing the masses. This was done for the assessment of walls.

3.7b.2.12 Comparison of Responses

Figures 3.7B-8 through 3.7B-10 (applicable for all Seismic Category I structures except the Diesel Generator 'E' Building) show that the response spectra of the time history envelop the design response spectra at all frequencies. The time history has been used to generate response spectra in the structures but has not been used to calculate forces in the structures. Response in typical Category I Structures, obtained from the response spectrum analysis compare closely with those obtained from time history analysis based on studies comparing displacements and accelerations obtained by the two methods, however there is some variation. Both methods are acceptable per Regulatory Guide 1.92 and Regulatory Guide 1.122.

The corresponding comparisons of the time history response spectra to the design response spectra for the Diesel Generator 'E' Building are provided in Figures 3.7B-11 through 3.7B-13 for the horizontal direction and Figures 3.7B-14 through 3.7B-16 for the vertical direction.

3.7b.2.13 Methods for Seismic Analysis of Dams

Dams are not provided on Susquehanna SES.

3.7b.2.14 Determination of Seismic Category I Structure Overturning Moments

For all Seismic Category I structures, except the Diesel Generator 'E' Building, the overturning moment is the sum of the moments at the base of each stick of the mathematical model. For each stick, the moment at the base is determined by combining the modal overturning moments. The moments are combined by the methods described in Subsection 3.7b.2.7. For the Diesel Generator 'E' Building, the total accelerations at each floor elevation, due to an earthquake component resulting from the modal combination described in Subsection 3.7b.2.7, are used to compute the overturning moment.

The components of the earthquake motion used are the same as those discussed in Subsection 3.7b.2.6.

Subsection 3.8.5 discusses the factor of safety against overturning for several loadings, which include seismic loads.

3.7b.2.15 Analysis Procedure for Damping

All Seismic Category I structures except the Diesel Generator 'E' Building consist of reinforced concrete and welded/bolted structural steel. Damping values for these materials are shown in Table 3.7b-3. However, in the seismic analysis of the structures, (except the Diesel Generator 'E' Building), damping values of 2 and 5 percent are used for OBE and SSE respectively for reinforced concrete, as well as welded/bolted structural steel. Therefore, analysis of composite modal damping is not necessary.

The Diesel Generator 'E' Building is constructed solely out of reinforced concrete. As shown in Table 3.7b-4, damping values of 4 and 7 percent are used for OBE and SSE, respectively.

All Seismic Category I structures except the ESSW pumphouse and spray pond and its pipe supports are founded on rock. Consequently, soil damping values are calculated for the ESSW pumphouse as described in Appendix D of Reference 3.7b-3.

The interaction damping values for the time history analysis of the containment are also calculated by the method described in Appendix D of Reference 3.7b-3.

3.7b.3 SEISMIC SUBSYSTEM ANALYSIS

As explained in Subsection 3.7b.2, this section discusses the seismic analysis of subsystems, i.e., equipment, piping, Class IE cable trays and supports for Seismic Category I HVAC ducts and cable trays.

3.7b.3.1 Seismic Analysis Methods

3.7b.3.1.1 Equipment

Seismic qualification of equipment is performed by using one of the following methods:

a) Analysis

b) Dynamic testing

c) Combination of analysis and dynamic testing

3.7b.3.1.1.1 Analysis

Seismic qualification of equipment is performed by analysis when the equipment can be adequately represented by a model and the analysis can determine its structural and functional adequacy. The analysis can either be an equivalent static analysis or a dynamic analysis.

Equivalent static analysis is described in Subsection 3.7b.3.5.

Dynamic analysis can be classified into three cases according to the relative rigidity of the equipment based on the magnitude of the fundamental natural frequency. Dynamic Analysis refer to Seismic Loads only, a discussion of the Hydrodynamic Load can be found in DBD046, Sections 2.2.1.2 and 2.2.1.3.

For structurally simple equipment, which can be represented by one degree of freedom, the dynamic load consists of a static load obtained as the equipment mass multiplied by the acceleration corresponding to the equipment's natural frequency. If the fundamental frequency is not known, the peak acceleration from the response spectra is taken.

For rigid equipment having a fundamental frequency greater than 33 Hz, the dynamic load consists of a static load obtained as the equipment's mass multiplied by the acceleration corresponding to 33Hz.

For structurally complex equipment, which cannot be classified as structurally simple or rigid, the equipment is idealized by a mathematical model and dynamic analysis is performed using standard analytical procedures. An alternative method used for verifying structural integrity of members physically similar to beams and columns is the static coefficient method. In this method no determination of natural frequency is made. Dynamic forces are calculated as product of the mass and peak acceleration of response spectra multiplied by a static coefficient of 1.5.

Equipment damping values used are given in Tables 3.7b-3 and 3.7b-4.

3.7b.3.1.1.2 Dynamic Testing

Dynamic testing is performed when analysis is insufficient to determine either the structural or functional adequacy of the equipment or both. Typical test methods used are as follows:

a) Single frequency sine beat test

b) Single frequency dwell test

c) Multifrequency test

All seismic qualification tests subject the equipment to excitation for at least 30 seconds.

3.7b.3.1.1.3 Combination of Analysis and Dynamic Testing

Certain equipment is qualified by a combination of analysis and dynamic testing.

3.7b.3.1.2 Piping Systems

BP-TOP-1, Rev. 3 (Ref. 3.7b-6) describes the methods used for seismic analysis of piping systems found in all Seismic Category I structures, except the Diesel Generator 'E' Building. Reference 3.7b-6 is followed on Susquehanna SES with the following exceptions:

In seismic analysis the modal responses are combined by SRSS and lower damping values than specified in Reference 3.7b-6 are used. For snubber elimination or other piping modifications, the combination of modal responses for closely spaced modes shall be in accordance with Regulatory Guide 1.92 whenever Regulatory Guide 1.61 or Code Case N-411 are used.

See Subsection 3.7b.3.7.

AEG-502, Rev. 0 (Ref. 3.7b-14) describes the methods used for seismic analysis of piping systems found in the Diesel Generator 'E' Building.

3.7b.3.1.3 Class IE Cable Trays

Cable trays are seismically qualified by one of two methods:

A. Capacity Evaluation Method which consists of the following:

   a) Calculation of the fundamental frequency of the cable tray based on the tray properties obtained from static tests

   b) Seismic load computation based upon the tray frequency, the possible support frequencies and the design spectra

   c) Calculation of the tray allowable capacity

   d) Evaluation of the tray capacity by interaction formulae

B. Static Analysis Method which consists of the following:

   a) Determine the maximum tray capacity in the two lateral directions by test

   b) Determine the maximum tray longitudinal capacity by analysis

   c) Calculate the maximum tray load by the equivalent static load method (discussed in Subsection 3.7b.3.5)

   d) Evaluation of the tray capacity by interaction formulae

3.7b.3.1.4 Supports for Seismic Category I HVAC Ducts

The supports of HVAC ducts are analyzed by the response spectrum method or by the equivalent static load method (discussed in Subsection 3.7b.3.5).

3.7b.3.1.5 Concrete Block Masonry Structures (Blockwalls)

The dynamic analysis of safety related concrete masonry blockwalls in Class I structures is performed by the response spectrum method. Response spectrum for the lower floor has been used for vertical motion and for walls, cantilevered from the floor. For horizontal motion, the acceleration of the lower floor or average of the lower and upper floor, whichever is greater, is used in determining inertia loads. Frequency calculations for blockwalls supporting class I attachments or located in areas of class I equipment are based on either cracked section, partially cracked section, or uncracked section properties; whichever represents the condition based upon the calculated loads.

Partially cracked section analysis is based on the following AC1 318 (Ref. 10A of Table 3.8-1) formula:

$I_e \quad = \quad (M_{cr}/M_a)^3 \, I_g + (1 - (M_{cr}/M_a)^3) \, I_{cr}$

where,

I_e = effective moment of inertia of cracked Section

I_{cr} = moment of inertia of cracked Section

M_a = bending moment applied to the blockwall

I_g = Gross section moment of inertia (uncracked)

M_{cr} = cracking bending moment $= \dfrac{fr \;\; I_g}{Y_t}$

fr = modulus of rupture for masonry = 50 psi

modulus of rupture for concrete = $6\sqrt{f'_c \text{ psi}}$

Yt = distance from centroid axis of gross section to the extreme fiber in tension.

For assessing the effects of frequency variations on the responses, the variable items such as boundary conditions, mass, modulus of elasticity, cracking moment are considered. Damping values used are in accordance with Table 3.7b-3. The response of attachments to blockwalls is determined as described in Subsection 3.7b.3.1.1.1.

The three components of earthquake motion are combined in accordance with Subsection 3.7b.2.6.

3.7b.3.1.6 Supports of Seismic Category I Electrical Raceway Systems

This section defines the procedures used for the design of the supports of electrical raceway systems, i.e., cable tray, conduit, and wireway gutter systems, subject to the seismic and other applicable loads. The raceway support system usually consists of raceways, horizontal and vertical support members and lateral and longitudinal bracing members.

3.7b.3.1.6.1 Loading Combinations

The adequacy of raceway systems (except for cable tray supports installed during construction of the Diesel Generator 'E' facility) to withstand seismic and other applicable static loads is determined according to the loading combinations and allowable responses given below:

Equation	Condition	Load Combination	Allowable Response
1	Normal	D + L + SRV	F - See note 4
2	Normal/Severe	D + L + E	See Notes 2 & 4
(Equation 2 applies only to connections for fatigue considerations)			
3	Abnormal/Extreme	D + E' + SRV + LOCA	See Notes 2, 3, & 4

NOTES: 1. For notations, see Table 3.8-2.

2. The following equation is applicable for bending in overhead connections:

$$\frac{5n_{EQ}}{N_{OBE}} + \frac{n_{EQ}}{N_{SSE}} \leq 1.0$$

where:

n_{EQ} = Total number of load/stress cycles per earthquake.

N_{OBE} = Allowable number of load/stress cycles per OBE event.

N_{SSE} = Allowable number of load/stress cycles per SSE event.

3. The following criteria are used for checking the members. In no case shall the allowable stress exceed 0.90F in bending, 0.85F in axial tension or compression, and 0.50F in shear. Where the design is governed by requirements of stability (local or lateral buckling), the actual stress shall not exceed 1.5F.

4. Allowable shear and normal loads in connections are determined from the manufacturers' data or from code allowable stresses whichever is applicable. The allowable values are increased 50% for load combination equation 3.

The loading combinations and the allowable stresses for the design of cable tray supports installed during construction of the Diesel Generator 'E' facility are as follows:

Equation	Condition	Load Combination	Allowable Response
1	Normal	D + L	F
2	Normal/Severe	D + E	F
3	Abnormal/Extreme	D + E'	1.6F
The definition of terms D, L, E and E' are as per Table 3.8-2.			

3.7b.3.1.6.2 Analytical Techniques

One of three methods of analysis is used. Method 1 is a simplified method of analysis that determines the fundamental frequency of braced supports using two dimensional analysis. Frequencies are determined in each of three principal directions. Then loads are determined by taking the spectral accelerations multiplied by the mass; and stresses are determined from static analysis. All members and connections are checked using stress criteria.

Method 2 uses a three dimensional computer analysis and includes springs to represent joint stiffness. Response spectrum analyses are done to determine stresses and deformations. The number of stress cycles is determined by multiplying the time of maximum earthquake motion by the natural frequency of the system. The allowable number of cycles is taken from Reference 3.7b-8 for the joint rotations calculated. Only overhead connections are checked for fatigue since the test results (ref. 3.7b-8, pg. 7-19) demonstrate that failures occur only in overhead connections.

The basis for the design criteria and analysis method 2 is the "Cable Tray and Conduit Raceway Test Program" (references 3.7b-7 through 3.7-10).

Method 3 uses the equivalent static load method of analysis (as described in Subsection 3.7b.3.5). In this method, the acceleration response is assumed to be the peak of the response spectrum at the damping values described in Subsection 3.7b.3.1.6.3. Stresses are determined from static analysis. All members and connections are checked using stress criteria.

3.7b.3.I.6.3 Damping

A maximum damping of 7% of the critical is used for the design of all raceway systems. The test program demonstrates that for cable tray systems damping is, in general, much higher than 7%. Reference 3.7b-7 recommends using 20% but values up to 50% are reported. The recommended damping values, developed from the test program and based on lower bound values, are shown in Figure 3.7B-92. Damping is amplitude dependent, i.e., it increases with increasing amplitude of input motion. For conduit systems the damping increases with increasing amplitude, but is much lower than for cable tray systems. This 7% is a realistic value for input motion exceeding 0.1g for conduit systems. Wireway gutters were not tested; however, the manner in which they are constructed - with more bolted connections and more cables than conduit - provides more damping mechanisms that are present in conduit systems so that 7% is a conservatively low damping value.

3.7b.3.1.6.4 Operating Basis Earthquake (OBE)

Except for cable tray supports installed during construction of the Diesel Generator 'E' facility, the OBE is considered in the load combinations only for the overhead connections which are checked for fatigue. The OBE stresses are not checked during design for two reasons: first, raceway systems do not fail in a brittle or catastrophic mode as demonstrated by the test program in which such failures did not occur and the electrical systems were able to continue to function in all cases. Thus, there is no need to limit the OBE stresses to the low levels usually used to preclude such failures. Second, the OBE stresses will always be less than the SSE stresses as demonstrated below.

In all cases the ZPA values are high enough to use 7% damping based on Figure 3.7B-92 since they all exceed 0.1g. A comparison of response spectra for corresponding damping values demonstrates that for all response spectra the OBE acceleration values are less than the corresponding SSE acceleration values. (See References 3.7b-8 and 3.7b-10) Thus, the OBE acceleration response and stresses are below the SSE acceleration response and stresses.

3.7b.3.2 Determination of Number of Earthquake Cycles

In general, the design of the equipment is not fatigue controlled because the equipment is elastic and the number of cycles in an earthquake is low.

Equipment that is qualified by analysis is designed to remain elastic during the earthquake. Any fatigue effects in tested equipment are accounted for by performing extended duration test on selected specimens. Consequently, the number of cycles of the earthquake has been accounted for.

In order to conduct a fatigue evaluation for nuclear Class I piping, the number of cycles for a given load set is obtained. This is done by considering ten maximum stress cycles per earthquake and five OBE's and one SSE to occur within the life of the plant.

3.7b.3.3 Procedure Used for Modeling

The models are developed to represent the equipment. Two or three dimensional models are used depending on the complexity of the equipment. The boundary conditions are modeled to reflect the in-plant mounting conditions. The equipment is represented by lumped mass models. Massless elastic members are used to connect the masses.

Supports for HVAC ducts are modeled as two or three dimensional (depending upon support complexity), lumped mass models. The masses are lumped at the center or at the corners of the ducts. The cable tray support analytical techniques are discussed in Subsection 3.7b.3.1.6.2.

Sections 2.0 and 3.0 of Reference 3.7b-6 discuss the techniques and procedures used to model piping other than the buried type.

3.7b.3.4 Basis for Selection of Frequencies

The natural frequencies of components are calculated. If the natural frequency of the component falls within the broadened peak of the response spectrum curve, then it is designed to withstand the peak acceleration.

3.7b.3.5 Use of Equivalent Static Load Method of Analysis

The equivalent static load method of analysis is used when the natural frequency of the equipment is not determined. If the equipment can be adequately represented by a single degree of freedom system, then the applied inertia load is equal to the mass of the equipment multiplied by the peak value of the response spectrum curve. If the equipment requires more than one degree of freedom for an adequate representation, then a factor of 1.5 is applied to the peak of the response spectrum curve.

Section 2.3.2 and Appendix D of Reference 3.7b-6 discuss the use of equivalent static load method of analysis as applicable to piping.

3.7b.3.6 Three Components of Earthquake Motion

For equipment, raceway, and HVAC duct supports, the three spatial components of the earthquake are combined by one of the following methods:

a. Absolute Sum

   Independent analyses are done for the vertical and two horizontal (east-west and north-south) directions. For design purposes, the response value used is the maximum value obtained by adding the response due to vertical earthquake with the larger value of the response due to one of the horizontal earthquakes by the absolute sum method.

b. Square Root of the Sum of the Squares

   Stress levels produced by the three individual accelerations (caused by the three spatial components of the earthquake) are combined by the square root of the sum of the squares method.

The criteria used for combining the results of horizontal and vertical seismic responses for piping systems are described in Section 5.1 of Reference 3.7b-6.

3.7b.3.7 Combination of Modal Responses

The modal responses of equipment (except the equipment in the Diesel Generator 'E' Building) are combined by the square root of the sum of the squares method. The absolute values of two closely spaced modes are added first before combining with the other modes by the square root of the sum of the squares method. Two consecutive modes are defined as closely spaced when their frequencies differ from each other by 10 percent or less. For equipment located in the Diesel Generator 'E' Building, the modal responses are combined using the criteria presented in Regulatory Guide 1.92, Rev. 1.

Procedures given in Regulatory Guide 1.92 for combining modal responses, when closely-spaced modes are present, are not complied with in the seismic response spectra analysis for piping, except for piping within the Diesel Generator 'E' Building and as noted below. All modal responses are combined by square root of sum of squares (SRSS) in the response spectra method of modal analysis for seismic loading (OBE and SSE). Seismic response spectra used in the piping analysis corresponds to conservative damping values of 1/2% for OBE and 1% for SSE. For snubber elimination or other piping modifications, Regulatory Guide 1.92 is complied with in the seismic response spectra analysis of piping components for combining modal responses of closely spaced modes whenever Regulatory Guide 1.61 or Code Case N-411 damping values are used. The damping values used for the Diesel Generator 'E' facility are shown in Table 3.7b-4.

The procedures used in evaluating the piping system for hydrodynamic loads (SRV and LOCA) by response spectra method is in compliance with Regulatory Guide 1.92. The modal responses in this case are combined in accordance with section 5.2 of BP-TOP-1, Rev. 3, which has been accepted by the NRC staff, per the letter dated September 29, 1976, from Karl Kniel, Chief Light Water Reactors Branch No. 2, Division of Project Management to Burton L. Lex, Bechtel Power Corporation.

The criteria used for piping systems are described in Sections 5.1 and 5.2 of Reference 3.7b-6

3.7b.3.8 Analytical Procedures for Piping

The design criteria and the analytical procedures applicable to piping systems are as described in Section 2.0 of Reference 3.7b-6. The methods used to consider differential piping support movements at different support points are as described in Section 4.0 of Reference 3.7b-6.

3.7b.3.9 Multiple Supported Equipment and Components with Distinct Inputs

For cable trays and ducts whose supports have two distinct inputs, a response spectrum curve (or maximum acceleration) is used that envelops the curves (or accelerations) at the two locations. Section 4.0 of Reference 3.7b-6 discusses the methods used for the analysis of multiple supported piping systems.

3.7b.3.10 Use of Constant Vertical Static Factors

Constant vertical static factors are not used in the seismic design of subsystems.

3.7b.3.11 Torsional Effects of Eccentric Masses

The torsional effects of valves and other eccentric masses are considered in the seismic analysis of piping by the techniques discussed in Section 3.2 of Reference 3.7b-6.

3.7b.3.12 Buried Seismic Category I Piping Systems and Tunnels

Buried Seismic Category I piping has been analyzed and designed for seismic effects in accordance with Section 6.0 of Reference 3.7b-3, and Reference 3.7b-13 for the Diesel Generator 'E' facility.

The majority of the anticipated settlement due to static loading of the ESSW Pumphouse will have occurred prior to connecting the piping to the building. During a SSE event, the differential settlement between the pumphouse and the surrounding soil which supports the piping, will be less than one inch (see Subsection 2.5.4.7 for further discussion of settlements). This movement will be accommodated by the piping without exceeding code allowable stresses.

Tunnels on the Susquehanna SES are non-Seismic Category I.

3.7b.3.13 Interaction of other Piping with Seismic Category I Piping

The techniques used to consider the interaction of Seismic Category I piping with non-Seismic Category I piping are in Section 3.4 of Reference 3.7b-6. All piping in the Diesel Generator 'E' Building was analyzed to Seismic Category I requirements.

3.7b.3.14 Seismic Analysis for Reactor Internals

This subsection is covered under Subsection 3.7a.3.14.

3.7b.3.15 Analysis Procedure for Damping

In general, a single damping value, as shown in Table 3.7b-3, is used for the analysis of Seismic Category I subsystems. The critical damping value related to electrical raceway system is discussed in Subsection 3.7b.3.I.6.3.

For a structural system, located in the Diesel Generator 'E' Building and consisting of various components having different damping materials, composite modal damping is computed in accordance with Sheet 3.7.2.11, equation (4) of the Standard Review Plan.

3.7b.4 SEISMIC INSTRUMENTATION

3.7b.4.1 Comparison with NRC Regulatory Guide 1.12, Rev 1.

Unit 1 and Unit 2 containments are assumed to respond identically to a given earthquake. This is considered to be a reasonable assumption, since both are identically designed and built and founded on rock. For this reason, instrumentation redundancy between units was not employed; identical seismic instrumentation was not, in general, installed in both units. Foundation interaction was assumed to be negligible due to the high stiffness of the rock.

Equipment required by Regulatory Guide 1.12 for a Safe Shutdown Earthquake maximum ground acceleration of less than 0.3g was implemented. The characteristics of the seismic instrumentation specified for Susquehanna exceed the range, frequency and other performance requirements of Regulatory Guide 1.12. The equipment is shown on Dwg M-157, Sh. 2.

3.7b.4.1.1 Triaxial Time - History Accelerographs

Required:
1) one at the containment foundation
2) one on the containment structure

Actual:
1) Unit 1 containment foundation
2) Unit 1 containment structure, 74 feet directly above item 1).
3) Unit 2 containment foundation
4) ESSW pumphouse floor
5) Unit 1 reactor* boiler equipment
6) Unit 1 reactor building floor, near RHR pumps
7) Free field, near the Security Control Center. This unit is a combination, self-contained sensor-trigger-recorder. It is included even though not required by Regulatory Guide 1.12.
8) Standalone free field, near Secondary Alarm Station. This unit is a combination, self-contained sensor-trigger-recorder. It is included even though not required by Regulatory Guide 1.12.

3.7b.4.1.2 Triaxial Seismic Switches

Required: 1) Containment foundation

Actual: 1) Unit 1 containment foundation
2) Unit 2 containment foundation
3) ESSW pumphouse floor

3.7b.4.1.3 Triaxial Response Recorders

Required:
1) Containment foundation, with immediate control room indication
2) Nuclear boiler equipment or piping supports
3) Seismic Category I equipment supports or piping support outside the containment.
4) The foundation of a Seismic Category I structure where the response is different from that of the containment structure.

Actual:
1) Unit 1 containment foundation, with immediate control room indication.
2) Unit 1 reactor* equipment.
3) Floor mounting, near Unit 1 RHR pumps.
4) ESSW pumphouse floor, with immediate control room indication.
5) Unit 1 containment structure.
6) Unit 2 containment foundation, with immediate control room indication.

3.7b.4.2 Description of Instrumentation

The seismic instrumentation consists of tri-axial acceleration sensors, time history recorders, alarm module, and a computer for performing an automatic frequency domain comparison to OBE and SSE design limits. Each sensor is continuously monitored and a common trigger to activate recording for all sensors is activated if the signal from at least two trigger sensors exceeds a threshold concurrently for any axis.

The requirement that two trigger sensors exceed a threshold concurrently provides the system with the capability to distinguish a seismic event from a non seismic, local event.

The recorders are configured to capture pre-trigger and a post-trigger data to ensure the event is captured in its entirety. Data is recorded on non-volatile memory, which can store data from numerous trigger events. Upon completion of recording, the computer software downloads data from the recorders associated with locations used for OBE and SSE comparison and performs automatic analysis of this data (download and analysis typically completed within 5 minutes). If

* The actual location of this instrument is on the outside of the biological shield wall. It is located in the optimum location for measuring the input motion experienced by the reactor pressure vessel after properly taking into account accessibility for servicing, and functionality due to radiation levels.

the analysis determines the event is possibly seismic in nature an automatic comparison to OBE and SSE limits it performed and the results are indicated to the operator. The system performs self-diagnostics including computer failure monitoring which if not completed successfully will activate the fail safe trouble annunciator. Sound the seismic monitoring system external power be lost, an uninterruptable power supply is included which will run the system for greater than 25 minutes required by Regulatory Guide 1.12.

3.7b.4.3 Control Room Operator Notification

Activation of the common trigger for recording of all sensor locations is annunciated at the control room (OC653 panel) and also at the Seismic Warning Panel (OC696). Activation of the system trouble condition is annunciated at the control room (OC653 panel) and also at the Seismic Warning Panel (OC696) OBE or SSE exceeded is indicated at the Seismic Warning Panel (OC696) only.

3.7b.4.4 Comparison of Measured and Predicted Responses

The operator is provided with a procedure and predicted response curves, by which action to continue operation or shut down may be decided. The plant will be shut down following an earthquake if the vibratory ground motion exceeds that of the OBE. Operation will not resume until it has been determined through detailed inspections and analyses that no damage has been sustained.

3.7b.5 REFERENCES

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3.7b-2 N.M. Newmark, "Design Criteria for Nuclear Reactors Subject to Earthquake Hazards," Proc IAEA Panel on A Seismic Design and Testing of Nuclear Facilities, Japan Earthquake Engineering Promotion Society, Tokyo, Japan (1967).

3.7b-3 "Seismic Analyses of Structures and Equipment for Nuclear Power Plants," BC-TOP-4A, Rev 3, Bechtel Power Corporation, San Francisco, California (November 1974).

3.7b-4 Uniform Building Code (UBC), by International Conference of Building Officials, Whittier, California, 1970 Edition.

3.7b-5 A.T. Derecho, D.M. Schultz, and M. Fintel, "Analysis and Design of Small Reinforced Concrete Buildings for Earthquake Forces," Portland Cement Association (1974).

3.7b-6 "Seismic Analysis of Piping Systems," BP-TOP-1, Rev 3, Bechtel Power Corporation, San Francisco, California (January 1976).

3.7b-7 "Development of Analysis and Design Techniques from Dynamic Testing of Electrical Raceway Support Systems," Technical Report, July, 1979, Bechtel Power Corporation.

3.7b-8 “Cable Tray and Conduit Raceway Seismic Test Program-Release 4," Test Report #1053-21.1-4, Volumes 1 and 2, December 15, 1978, ANCO Engineers, Inc.

3.7b-9 "Hatago, P.Y., Reimer, G.S., "Dynamic Testing of Electrical Raceway Support Systems for Economical Nuclear Power Plant Installations," presented at the February 4-9, 1979 IEEE-PES.

3.7b-10 "Cable Tray and Conduit Raceway Seismic Test Program-Release 4," Addendum to Test Report #1053-21.1-4, Volume 3, May 1980, ANCI Engineers, Inc.

3.7b-11 Cable Tray Qualification Data for the Susquehanna Steam Electric Station Units 1 and 2. Specification 8856-E-132, November 29, 1976, Husky Products, Inc.

3.7b-12 Rodofoam II manufactured by W. R. Grace & Co. or equivalent equal.

3.7b-13 M. A. Igbal and E. C. Goodling, "Seismic Design of Buried Pipes," presented at the 2nd ASCE Specialty Conference on Structural Design of Nuclear Plant Facilities at New Orleans, Louisiana, December, 1975.

3.7b-14 "Seismic Analysis of Piping Systems in Nuclear Power Plants," AEG-502, Rev. 0, Gibbs and Hill, Inc., New York, New York (June 1981).

3.7b-15 "Design Response Spectra for Seismic Design of Nuclear Power Plants," US NRC Regulatory Guide 1.60 Rev. 1 (December 1973).

3.7b-16 "Damping Values for Seismic Design of Nuclear Power Plants," US NRC Regulatory Guide 1.61 (October, 1973).

3.7b-17 "Combining Modal Responses and Spatial Components in Seismic Response Analysis," US NRC Regulatory Guide 1.92, Rev. 1 (February 1976).

3.7b-18 "Development of Floor Design Response Spectra for Seismic Design of Floor-Supported Equipment or Components," US NRC Regulatory Guide 1.122, Rev. 1 (February 1978).

3.7b-19 "Standard Review Plan 3.7.1," US NRC NUREG-0800 (July 1981).

3.7b-20 "Standard Review Plan 3.7.2," US NRC NUREG-0800 (July 1981).

3.7b-21 "Diesel Generator 'E' Building Seismic Analysis," Calculation Number SE-DB-1C, Rev. 1.

TABLE 3.7b-1

AMPLIFICATION FACTORS FOR GROUND SPECTRA*

Percent of Critical Damping	Acceleration	Displacement
0.0	5.2	2.0
0.5	4.7	1.8
1.0	4.2	1.6
2.0	3.5	1.5
3.0	3.0	1.2
5.0	2.1	1.1
7.0	1.5	1.0

* For all seismic Category 1 structures except the Diesel Generator 'E' Building

TABLE 3.7b-2

AMPLIFICATION FACTORS FOR DIESEL GENERATOR 'E' BUILDING'S GROUND SPECTRA

HORIZONTAL DESIGN RESPONSE SPECTRA

Percent of Critical Damping	Amplification Factors for Control Points			
	Acceleration		Displacement	
	A(33Hz)	B(9Hz)	C(2.5Hz)	D(0.25Hz)
0.5	1.0	4.96	5.95	3.20
2	1.0	3.54	4.25	2.50
4	1.0	2.92	3.50	2.20
5	1.0	2.61	3.13	2.05
7	1.0	2.27	2.72	1.88

VERTICAL DESIGN RESPONSE SPECTRA

Percent of Critical Damping	Amplification Factors for Control Points			
	Acceleration		Displacement	
	A(33Hz)	B(9Hz)	C(3.5Hz)	D(0.25Hz)
0.5	1.0	4.96	5.67	2.13
2	1.0	3.54	4.05	1.67
4	1.0	2.92	3.34	1.47
5	1.0	2.61	2.98	1.37
7	1.0	2.27	2.59	1.25

TABLE 3.7b-3

DAMPING VALUES FOR NON-NSSS MATERIALS*
(PERCENT OF CRITICAL DAMPING)

Structure of Component	OBE	SSE
Welded steel structures	2	5
Bolted steel structures	3	5
Reinforced concrete structures	2	5
Concrete masonry structures		
Uncracked	2	2
Partially Cracked	4	7
Cracked	4	7
Piping systems	0.5	1
Equipment	0.5	1

*Notes

1. For seismic design of all non-NSSS safety related structures, piping systems and equipment, except those associated with the Diesel Generator 'E' Facility
2. Higher damping values are used if justified.
3. For snubber elimination or other piping modifications, damping values per Code Case N-411 or Regulatory Guide 1.61 may be applied to piping systems. When either Code Case N-411 or Regulatory Guide 1.61 is invoked, modal combinations for closely spaced modes per Regulatory Guide 1.92 shall be applied.

TABLE 3.7b-4

DAMPING VALUES FOR DIESEL GENERATOR 'E' FACILITY
(Percent of Critical Damping)

Structure or Component[3]	Operating Basis Earthquake (OBE)[1]	Safe Shutdown Earthquake (SSE)
Equipment and large-diameter piping systems [2,4], pipe diameter greater than 12 in	2	3
Small-diameter piping systems [4], diameter equal to or less than 12 in	1	2
Welded steel structures	2	4
Bolted steel structures	4	7
Reinforced concrete structures	4	7

1 In the dynamic analysis of active components as defined in U.S. NRC Regulatory Guide 1.48, these values should be used for the SSE.

2 Includes both material and structural damping. If the piping system consists of only one or two spans with little structural damping, use values for small-diameter piping.

3 If the maximum combined stresses due to static, seismic, and other dynamic loading are significantly lower than the yield stress and 1/2 yield stress for SSE and OBE, respectively, in any structure or component, damping values lower than those specified above should be used for that structure or component to avoid underestimating the amplitude of vibrations or dynamic stresses.

4 Damping values per Code Case N-411 may be applied to piping systems.

TABLE 3.7b-5

STRUCTURE FOUNDATION INTERACTION COEFFICIENTS

Structure	Motion	Equivalent Spring Constant	Equivalent Damping Coefficient
ESSW Pumphouse	Transitional	EW 1.97+6 k/ft(1) NS 1.97+6 k/ft	3.31+4 k-sec/ft 3.31+4 k-sec/ft
	Rocking	EW 6.1+9 kft/rad NS 2.94+9 kft/rad	3.77+7 k-ft-sec/rad 2.10+7 k-ft-sec/rad
	Vertical	1.81+6 k/ft	5.22+4 k-sec/ft
Containment	Translational	4.07+7 k/ft	1.89+5 k-sec/ft
	Rocking	7.96+10 k-ft/rad	6.16+7 k-ft-sec/rad
	Vertical	4.78+7 k/ft	3.27+5 k-sec/ft

(1) $1.97+6 = 1.97 \times 10^6$

TABLE 3.7b-6

PROPERTIES OF FOUNDATION MEDIA FOR CONTAINMENT AND ESSW PUMPHOUSE

	Containment (rock)	ESSW Pumphouse (soil)
Density (pcf)	140	130
Shear modulus (psi)	$1.15\ (10^5)$	$6.1\ (10^4)$
Shear wave velocity (fps)	6200	1480

TABLE 3.7b-7

NATURAL FREQUENCIES OF CONTAINMENT BELOW 33 CPS*

Mode No.	Frequency (CPS)	
	Horizontal	Vertical
1	4.99	16.19
2	8.01	20.95
3	16.12	38.24
4	19.83	
5	23.89	

* The frequency of 38.8 cps is included in the table because three modes are used in the vertical analysis.

TABLE 3.7b-8

NATURAL FREQUENCIES OF THE REACTOR AND CONTROL BUILDING BELOW 33 CPS

Mode No.	Frequency (CPS) E-W	N-S	Vertical
1	2.23	3.92	4.43
2	2.51	4.53	6.21
3	3.49	4.72	6.80
4	4.31	5.98	7.50
5	4.77	12.0	7.85
6	6.14	12.5	7.99
7	6.23	13.5	8.89
8	11.26	14.0	9.20
9	11.33	16.7	9.56
10	11.96	22.6	9.88
11	12.81	23.0	10.17
12	13.17	23.6	10.96
13	17.81	28.2	11.01
14	21.74	29.8	11.09
15	21.95		11.58
16	23.19		11.80
17	24.37		14.24
18	25.31		14.33
19	26.22		15.53
20	26.91		16.14
21	27.87		19.71
22	28.65		20.76
23	30.65		21.36
24	30.81		23.66
25			26.18
26			26.75
27			27.77
28			29.86
29			30.11
30			32.58

TABLE 3.7b-9

ESSW PUMPHOUSE : FREQUENCIES WITH AND WITHOUT ECCENTRICITIES

(SEE FIGURE 3.7B-125)

Mode #	Frequencies (cps)	
	With Eccentricity	**Without Eccentricity**
1	13.93	13.94
2	18.05	18.06
3	28.94	28.97
4	38.83	40.01

TABLE 3.7b-10

DIESEL GENERATOR A-D BUILDING
FREQUENCIES WITH AND WITHOUT ECCENTRICITIES

(SEE FIGURE 3.7B-126)

Mode #	Frequencies (cps)	
	With Eccentricity	Without Eccentricity
1	8.86	8.96
2	9.65	9.71
3	22.56	23.42
4	31.69	32.04
5	33.45	33.66

TABLE 3.7b-11

COMPARISONS OF TORSIONAL MOMENTS
BETWEEN ORIGINAL DESIGN AND THE VALUES
COMPUTED FROM THE RESULTS OF 3-D STICK MODEL

Building	Torsional Moment (k.ft.)	
	Original Design	3-D Stick Model
ESSW Pumphouse	24,440	11,780
Diesel Generator A-D Building El. 677'-0"	29,420	46,400
Diesel Generator A-D Building El. 710'-9"	23,450	34,900

Table 3.7b-12

This Table Has Been Deleted

Table 3.7b-13

This Table Has Been Deleted

Table 3.7b-1a

This Table Has Been Renumbered to 3.7b-2

Table 3.7b-2a

This Table Has Been Renumbered to 3.7b-4

THIS FIGURE IS A
DUPLICATION OF
2.5-28

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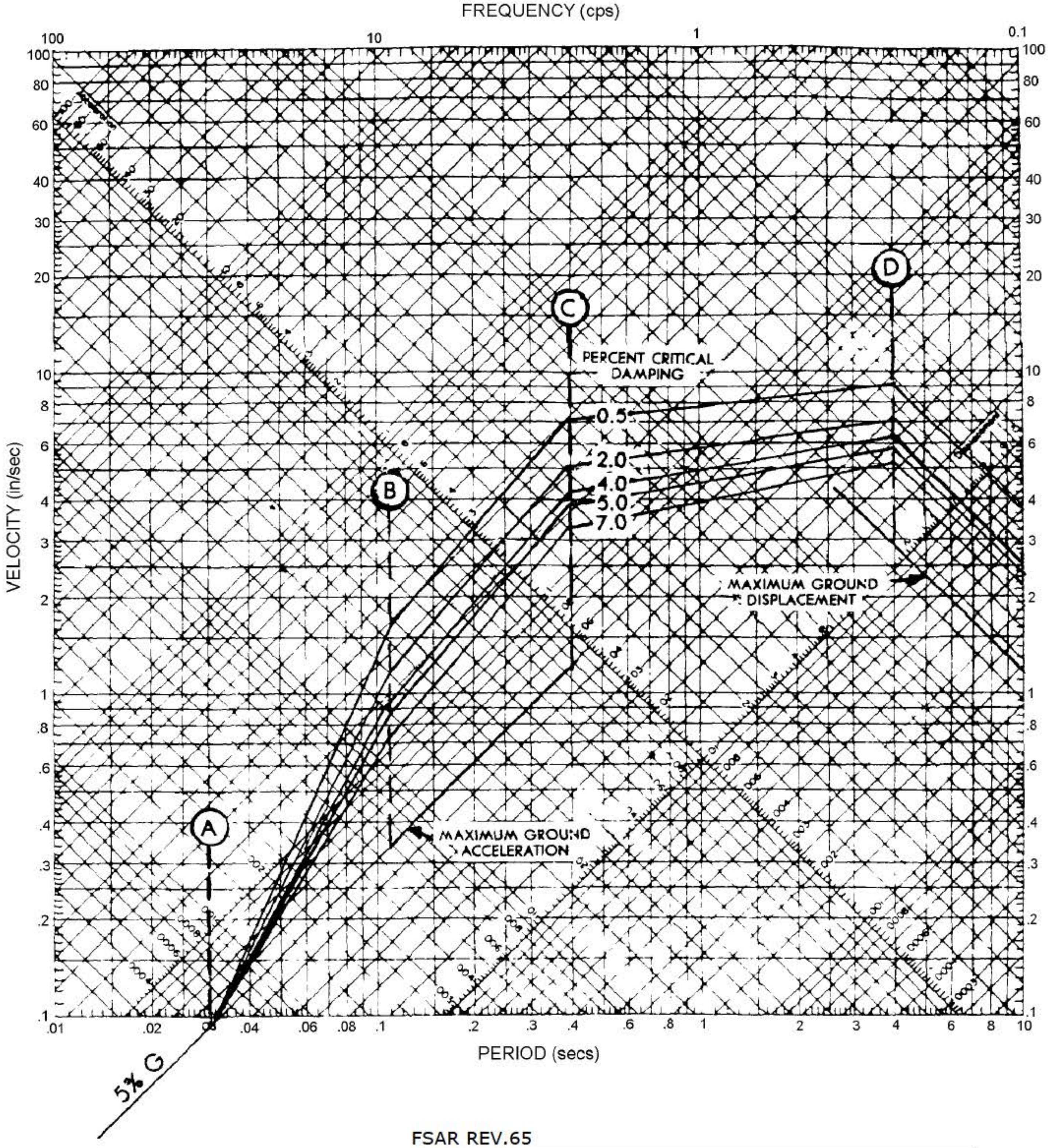
SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Figure duplicate to Figure 2.5-28 FIGURE 3.7B-1, Rev. 54

AutoCAD Figure 3_7B_1.doc



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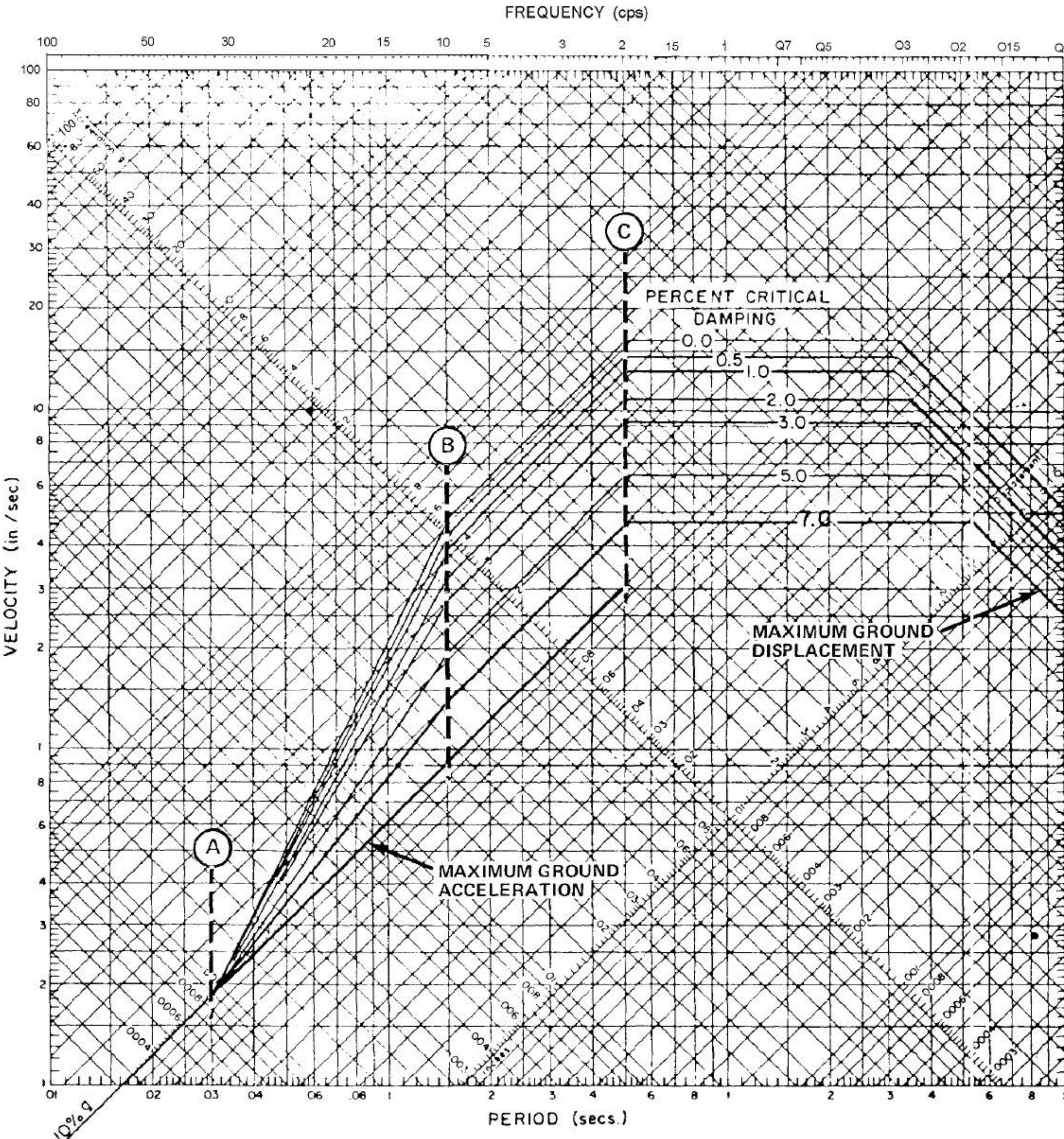
FINAL SAFETY ANALYSIS REPORT

DIESEL GENERATOR 'E' BUILDING
DESIGN RESPONSE SPECTRA
OPERATING BASIS EARTHQUAKE
HORIZONTAL COMPONENT

FIGURE 3.7B-2, Rev. 55

Auto-Cad Figure Fsar 3_7B_2.dwg



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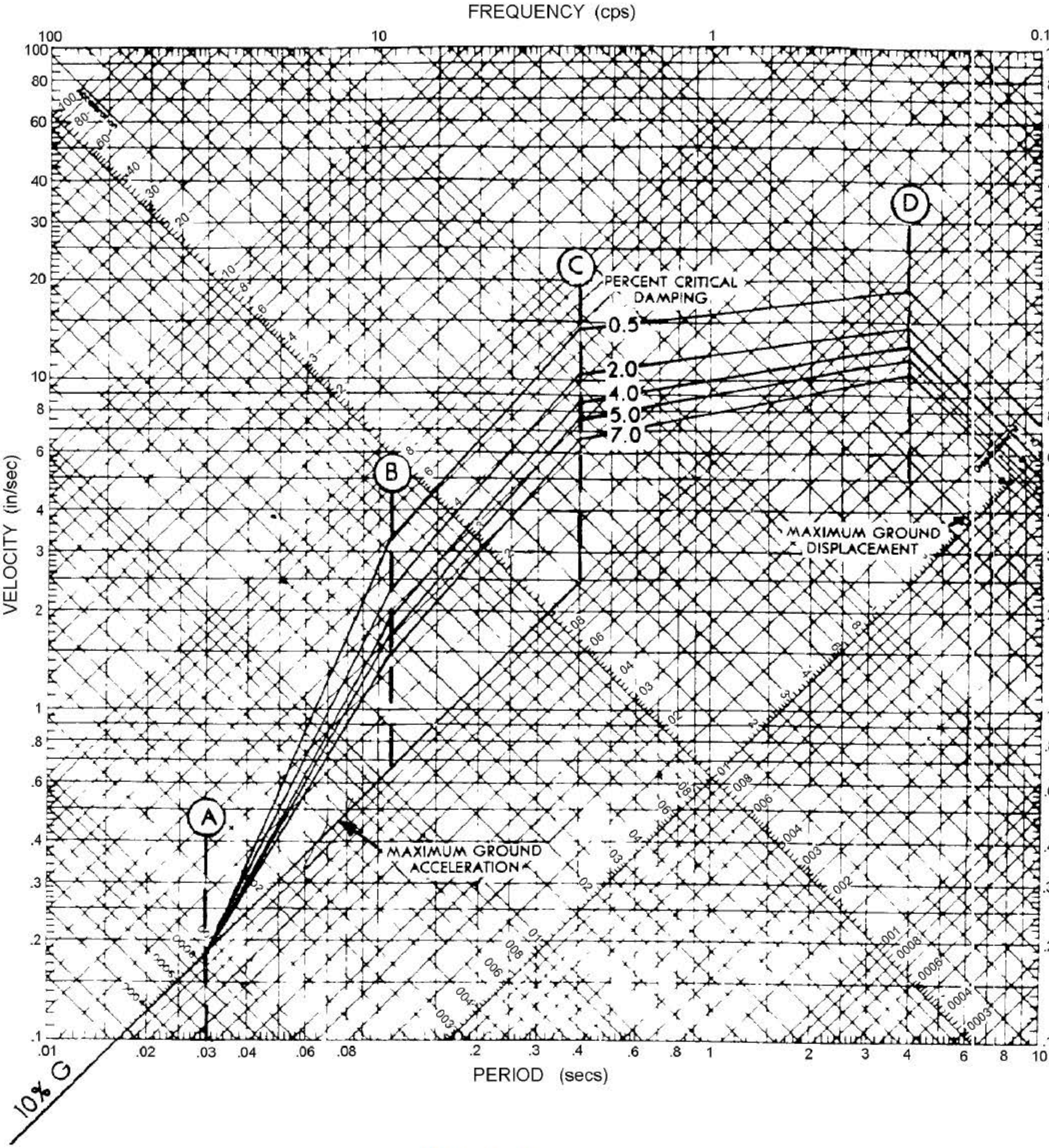
DESIGN RESPONSE SPECTRA
SAFE SHUTDOWN EARTHQUAKE
HORIZONTAL COMPONENT

FIGURE 3.7B-3, Rev. 55

Auto-Cad Figure Fsar 3_7B_3.dwg



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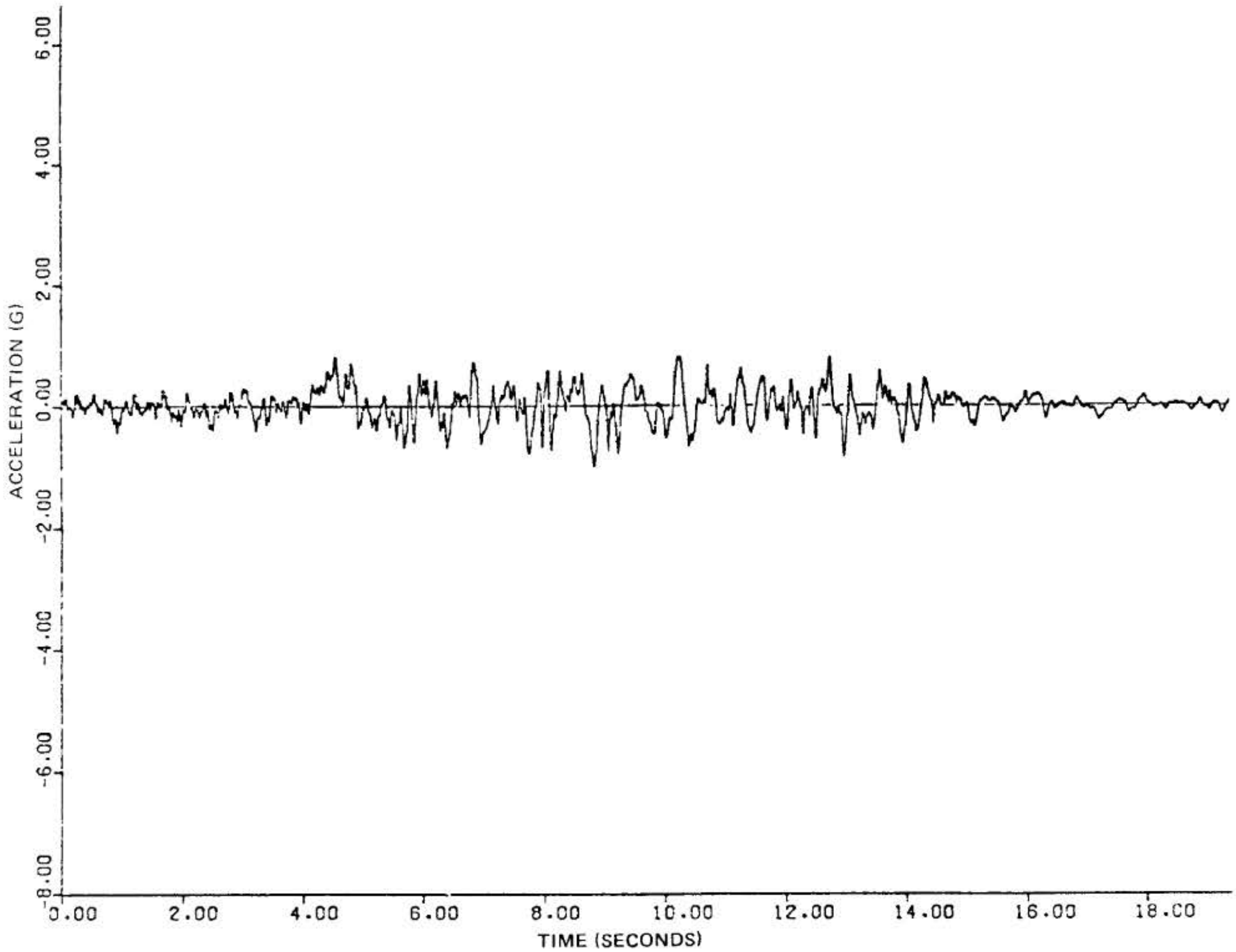
FINAL SAFETY ANALYSIS REPORT

DIESEL GENERATOR 'E' BUILDING'S
DESIGN RESPONSE SPECTRA
SAFE SHUTDOWN EARTHQUAKE
HORIZONTAL COMPONENT

FIGURE 3.7B-4, Rev. 55

Auto-Cad Figure Fsar 3_7B_4.dwg



* FOR ALL SEISMIC CATEGORY I STRUCTURES EXCEPT THE DIESEL GENERATOR 'E' BUILDING.

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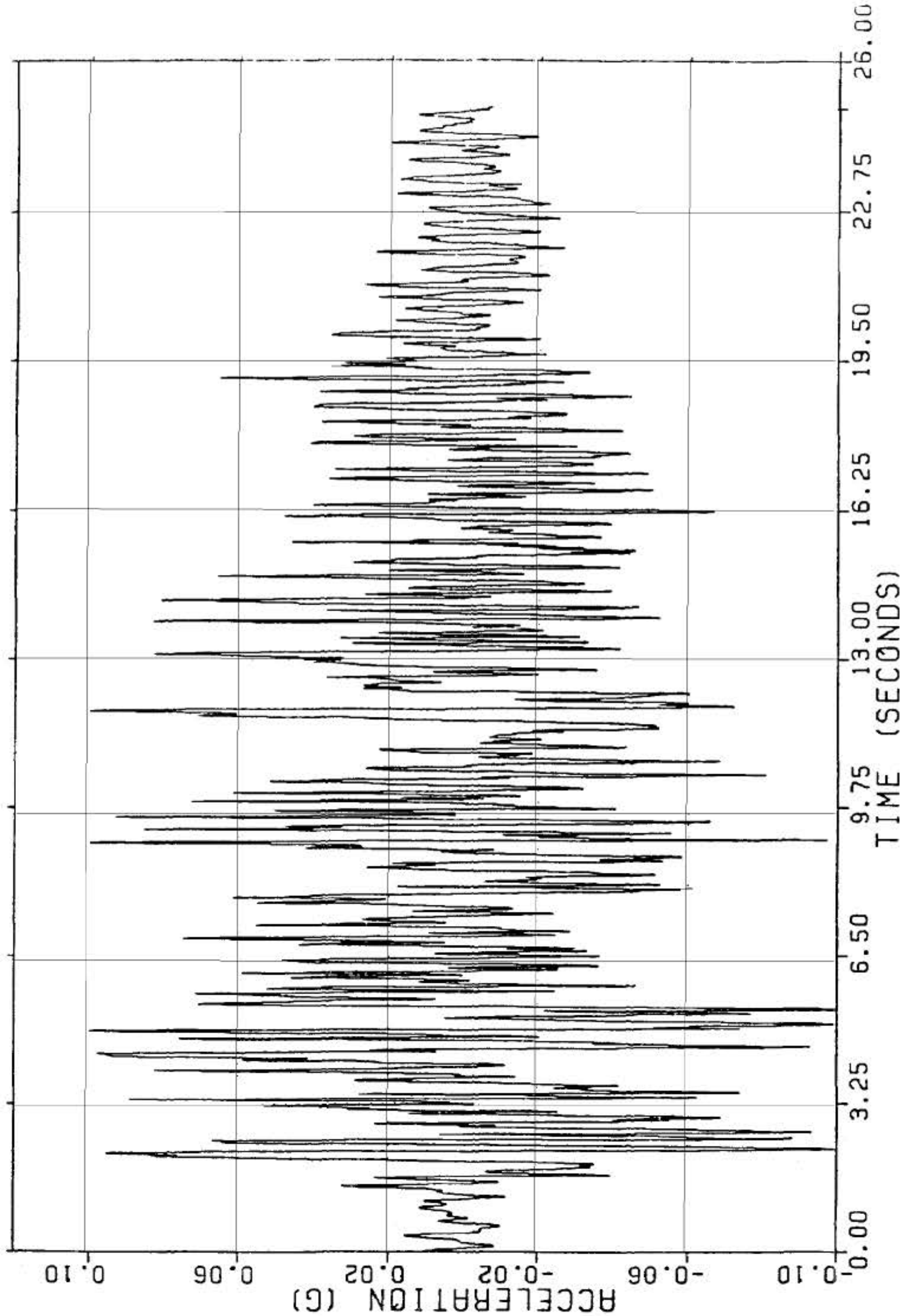
SYNTHETIC TIME HISTORY*
NORMALIZED TO 1G

FIGURE 3.7B-5, Rev. 55

Auto-Cad Figure Fsar 3_7B_5.dwg



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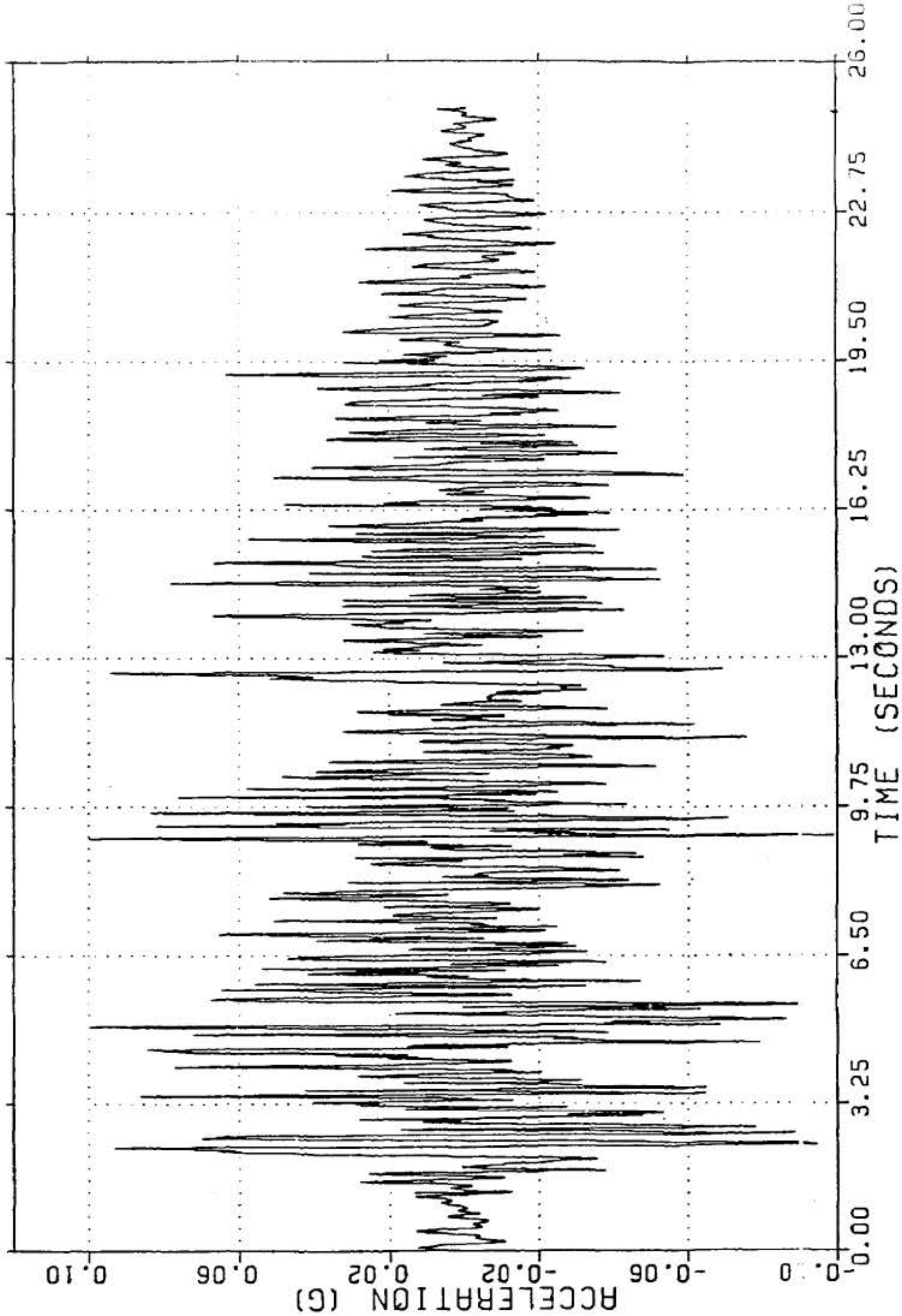
DIESEL GENERATOR 'E' BUILDING
HORIZONTAL SYNTHETIC TIME
HISTORY NORMALIZED TO 0.1G

FIGURE 3.7B-6, Rev. 55

Auto-Cad Figure Fsar 3_7B_6.dwg



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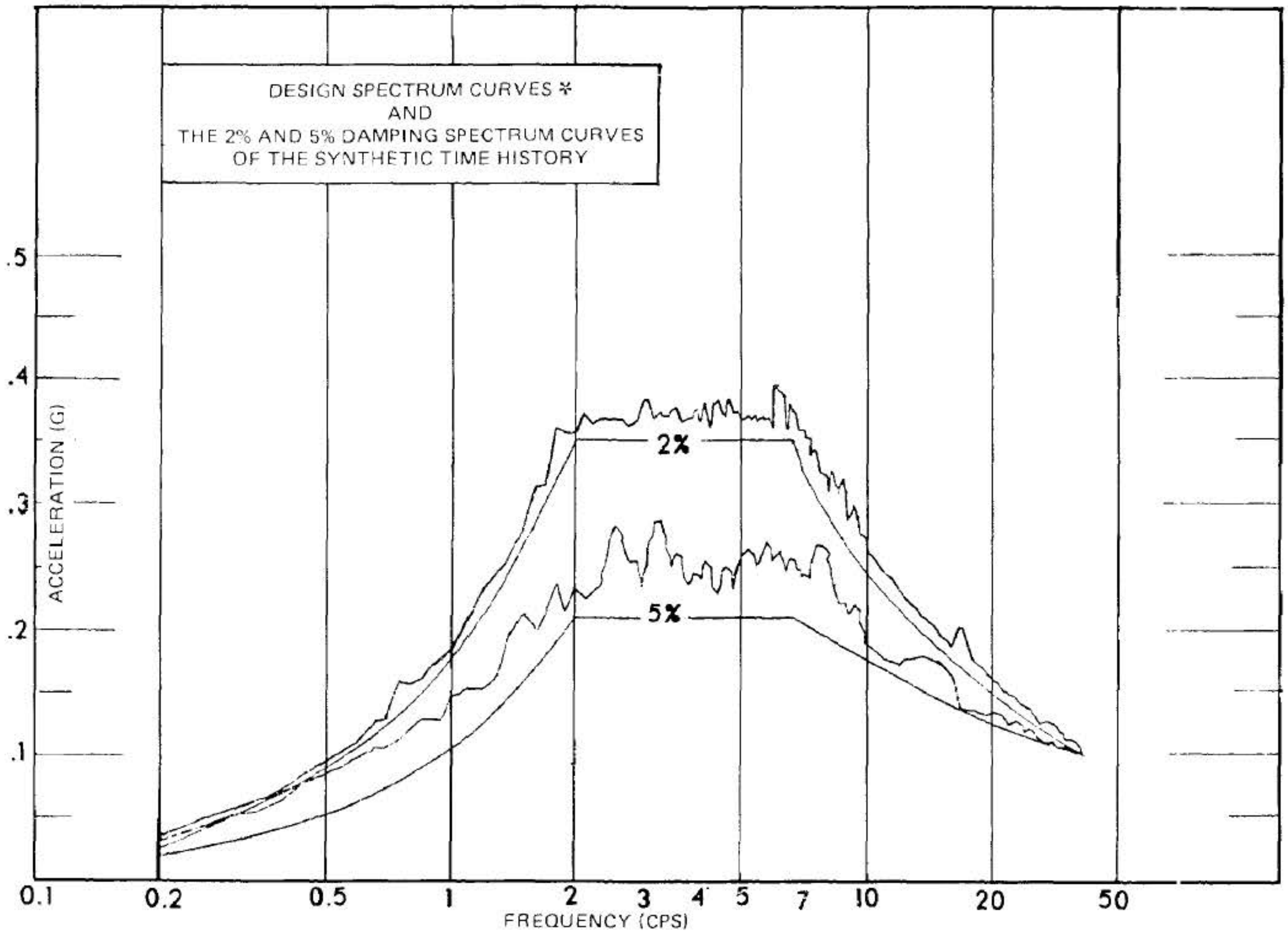
DIESEL GENERATOR 'E' BUILDING
VERTICAL SYNTHETIC TIME
HISTORY NORMALIZED TO 0.1G

FIGURE 3.7B-7, Rev. 55

Auto-Cad Figure Fsar 3_7B_7.dwg



* SSE HORIZONTAL COMPONENT FOR ALL ROCK FOUNDED SEISMIC CATEGORY I STRUCTURES EXCEPT THE DIESEL GENERATOR 'E' BUILDING.

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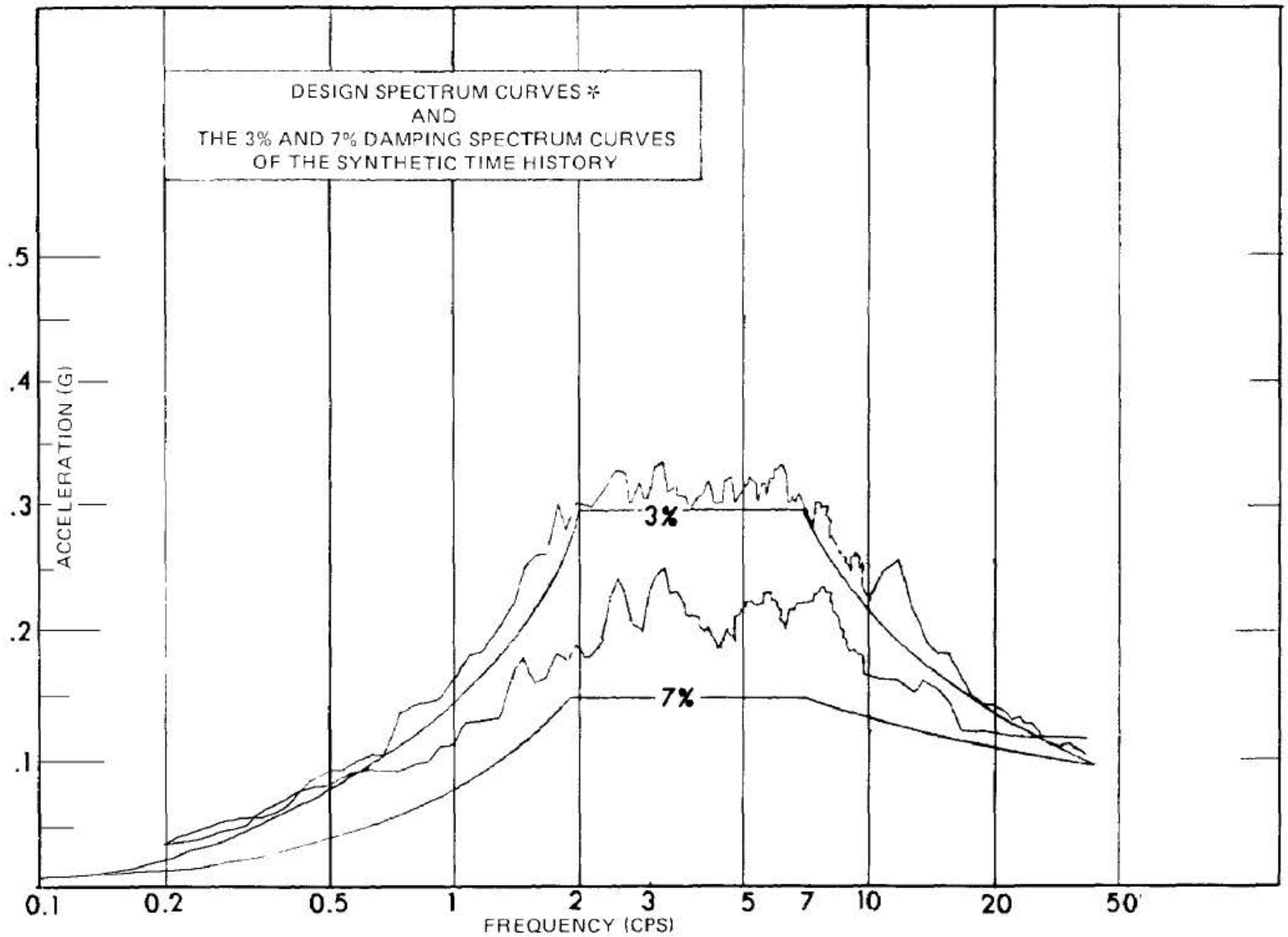
COMPARISON OF TIME HISTORY
RESPONSE SPECTRA AND
DESIGN RESPONSE SPECTRA
2% AND 5% DAMPING (0.2-30 CPS)

FIGURE 3.7B-8, Rev. 55

Auto-Cad Figure Fsar 3_7B_8.dwg



* SSE HORIZONTAL COMPONENT FOR ALL ROCK FOUNDED SEISMIC CATEGORY I STRUCTURES EXCEPT THE DIESEL GENERATOR 'E' BUILDING.

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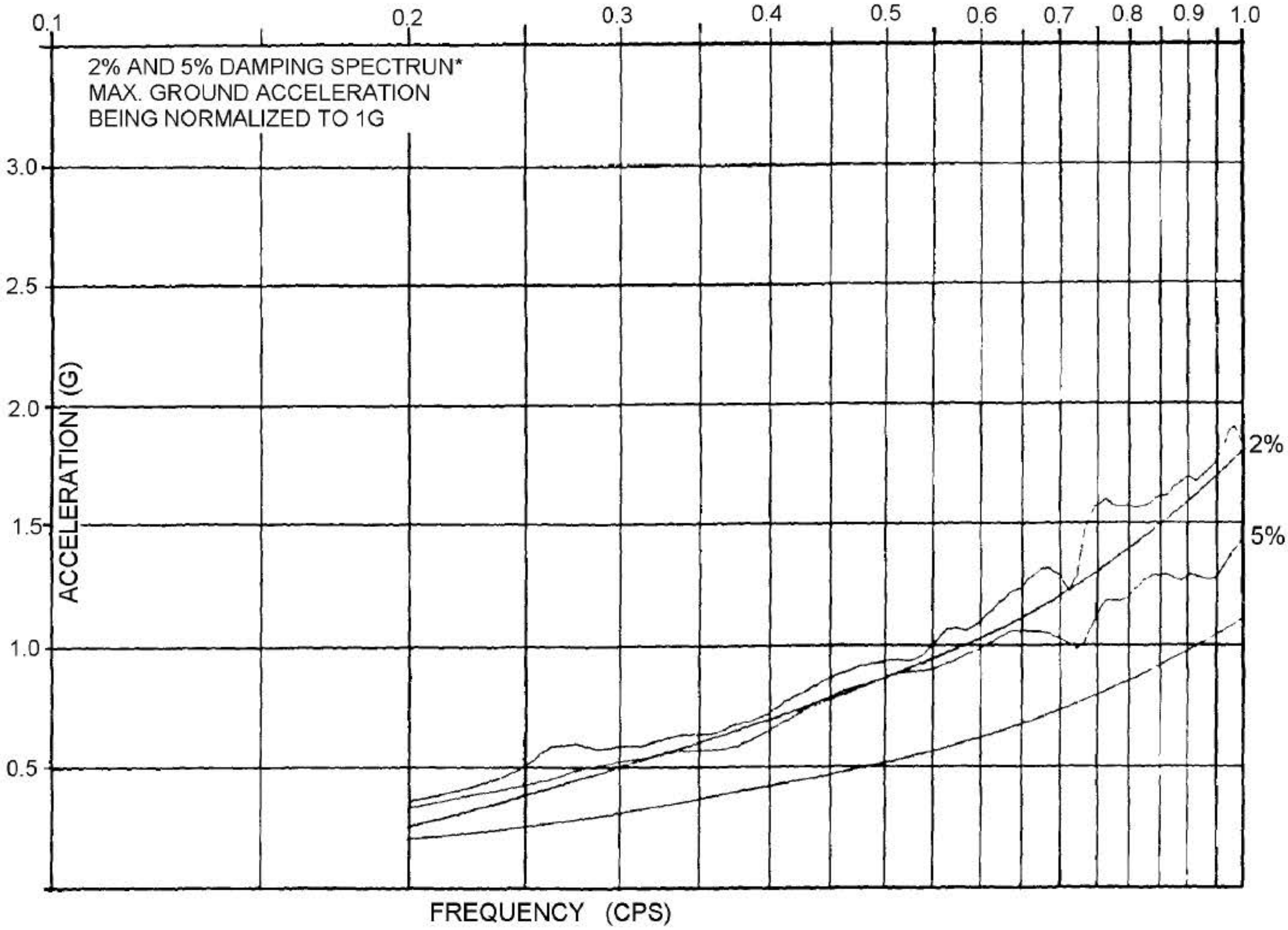
COMPARISON OF TIME HISTORY
RESPONSE SPECTRA AND
DESIGN RESPONSE SPECTRA
3% AND 7% DAMPING (0.2-30 CPS)

FIGURE 3.7B-9, Rev. 55

Auto-Cad Figure Fsar 3_7B_9.dwg



* SSE HORIZONTAL COMPONENT FOR ALL ROCK FOUNDED SEISMIC CATEGORY I STRUCTURES EXCEPT THE DIESEL GENERATOR 'E' BUILDING.

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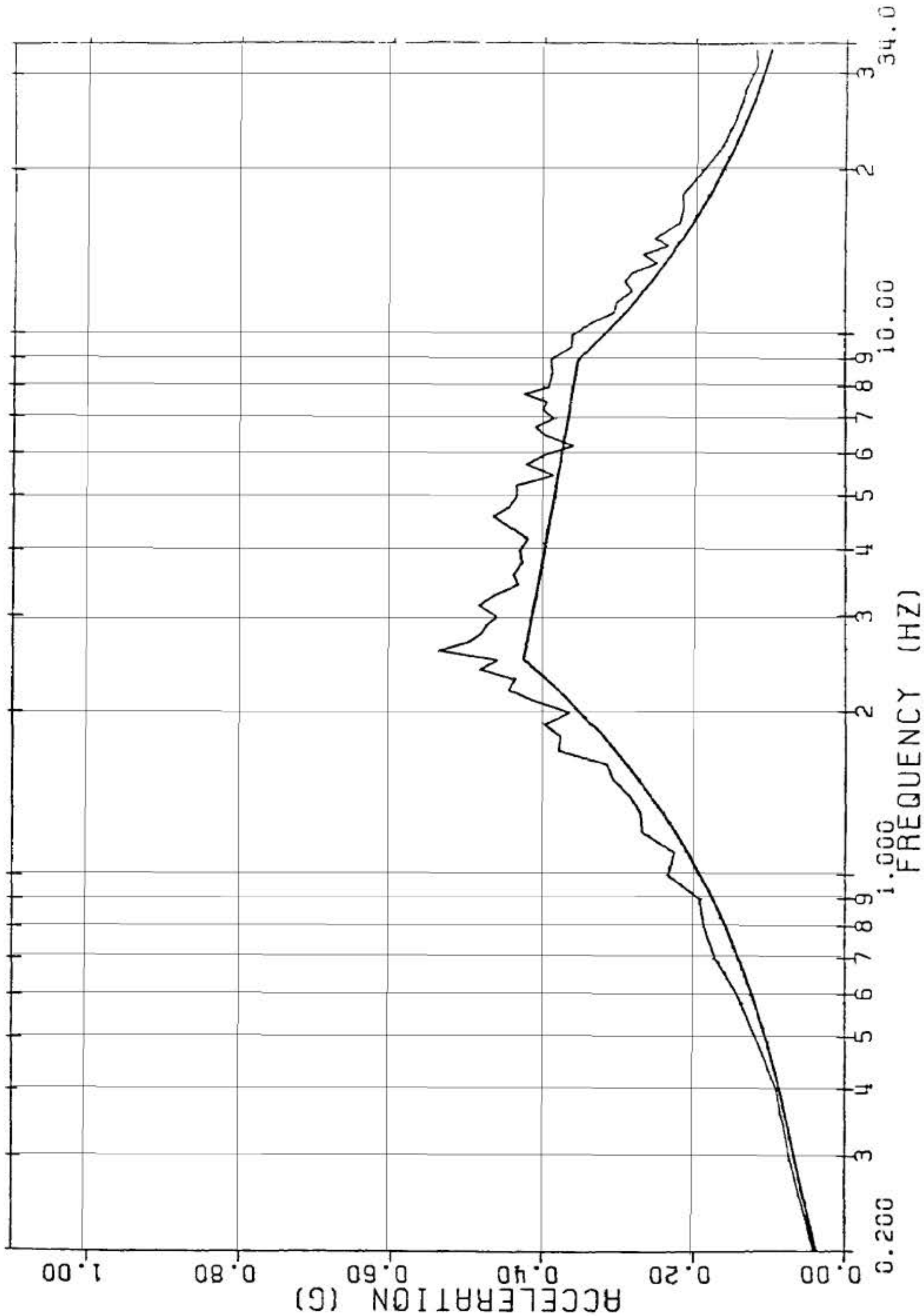
COMPARISON OF TIME HISTORY
RESPONSE SPECTA AND
DESIGN RESPONSE SPECTRA-
2% AND 5% DAMPING (0.2-1.0 CPS)

FIGURE 3.7B-10, Rev. 55

Auto-Cad Figure Fsar 3_7B_10.dwg



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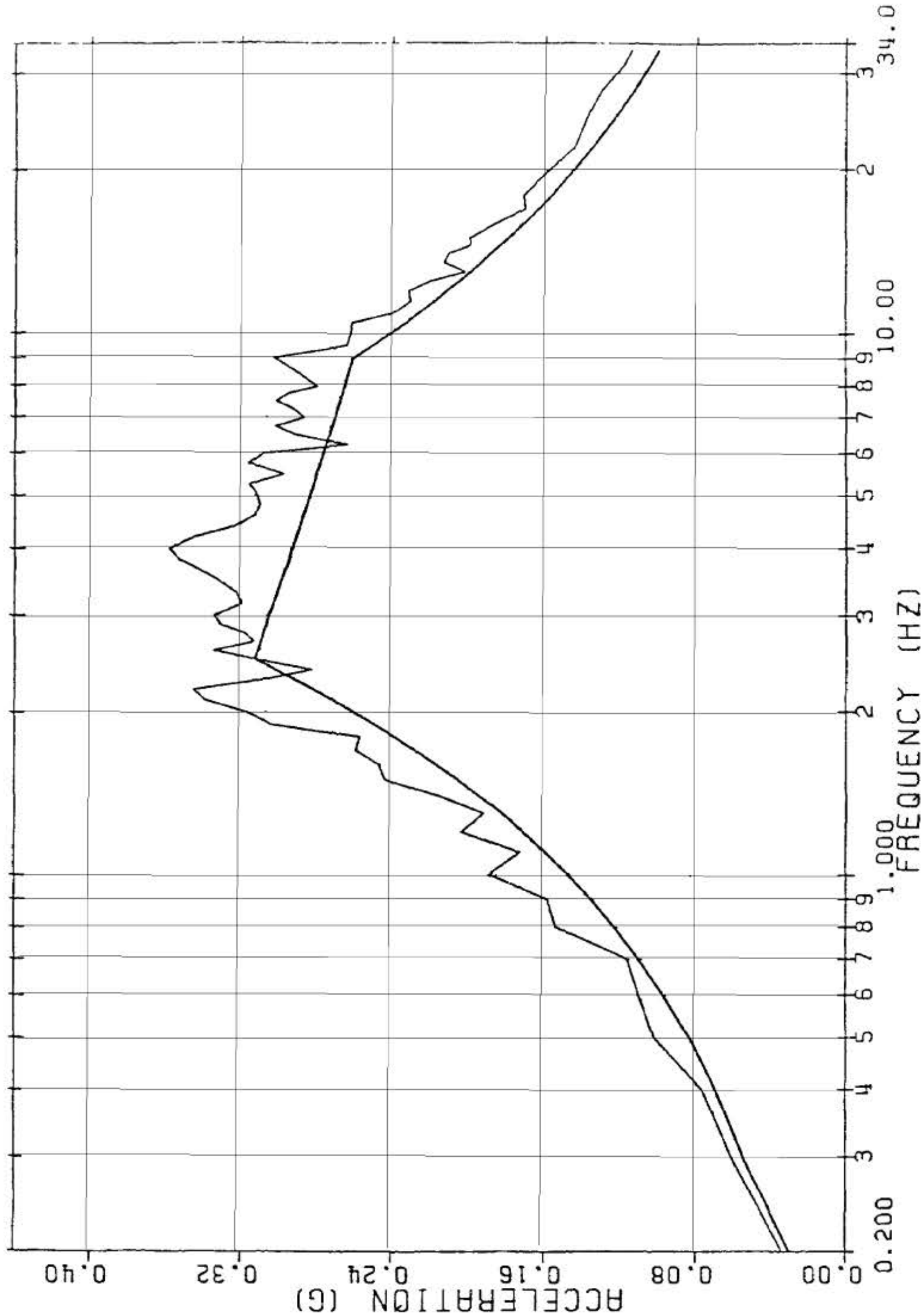
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

DIESEL GENERATOR 'E' BUILDING
COMPARISON OF HORIZONTAL TIME
HISTORY RESPONSE SPECTRUM AND
HORIZONTAL DESIGN RESPONSE
SPECTRUM 2% DAMPING

FIGURE 3.7B-11, Rev. 55

Auto-Cad Figure Fsar 3_7B_11.dwg



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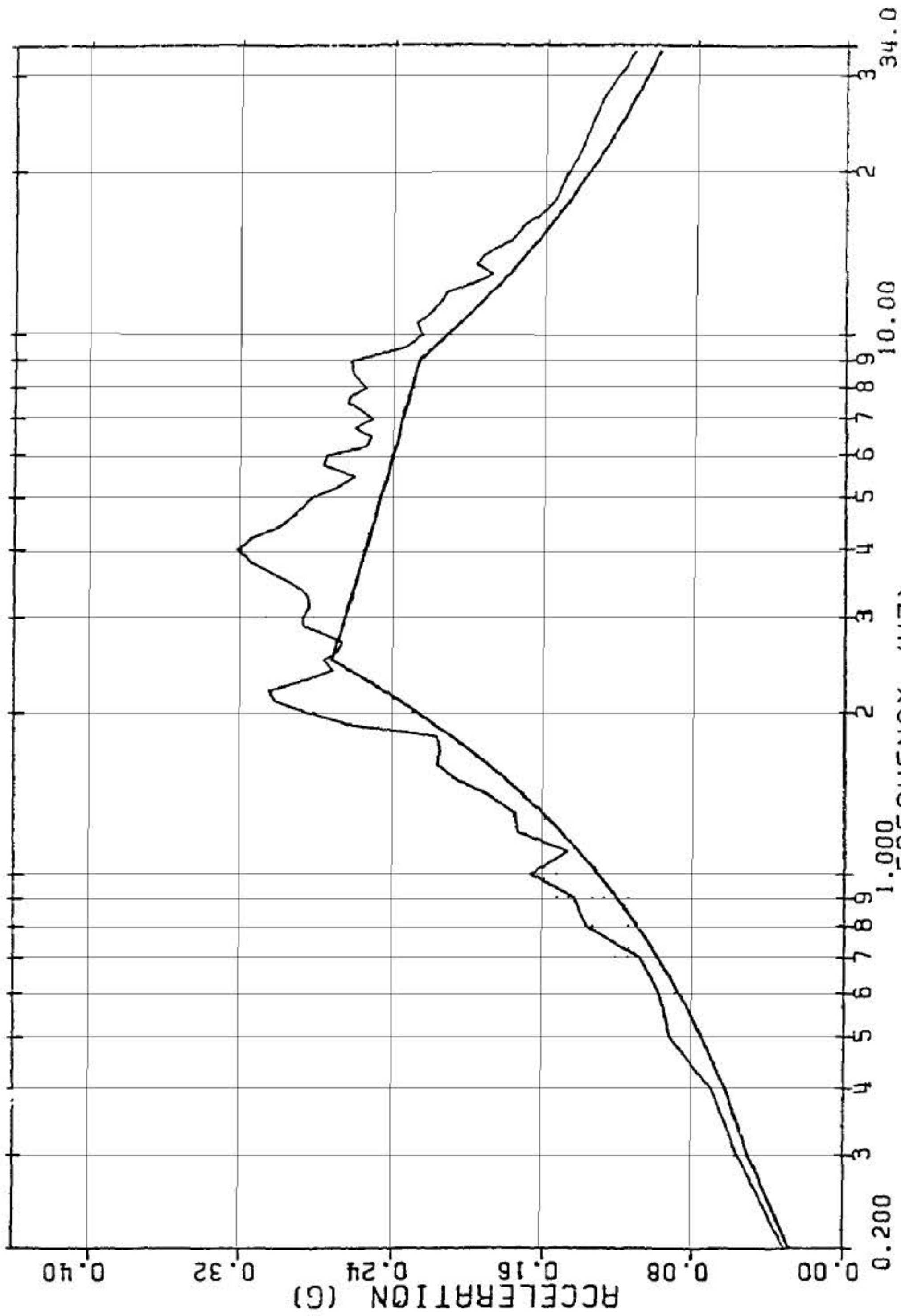
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

DIESEL GENERATOR 'E' BUILDING
COMPARISON OF HORIZONTAL TIME
HISTORY RESPONSE SPECTRUM AND
HORIZONTAL DESIGN RESPONSE
SPECTRUM 5% DAMPING

FIGURE 3.7B-12, Rev. 55

Auto-Cad Figure Fsar 3_7B_12.dwg



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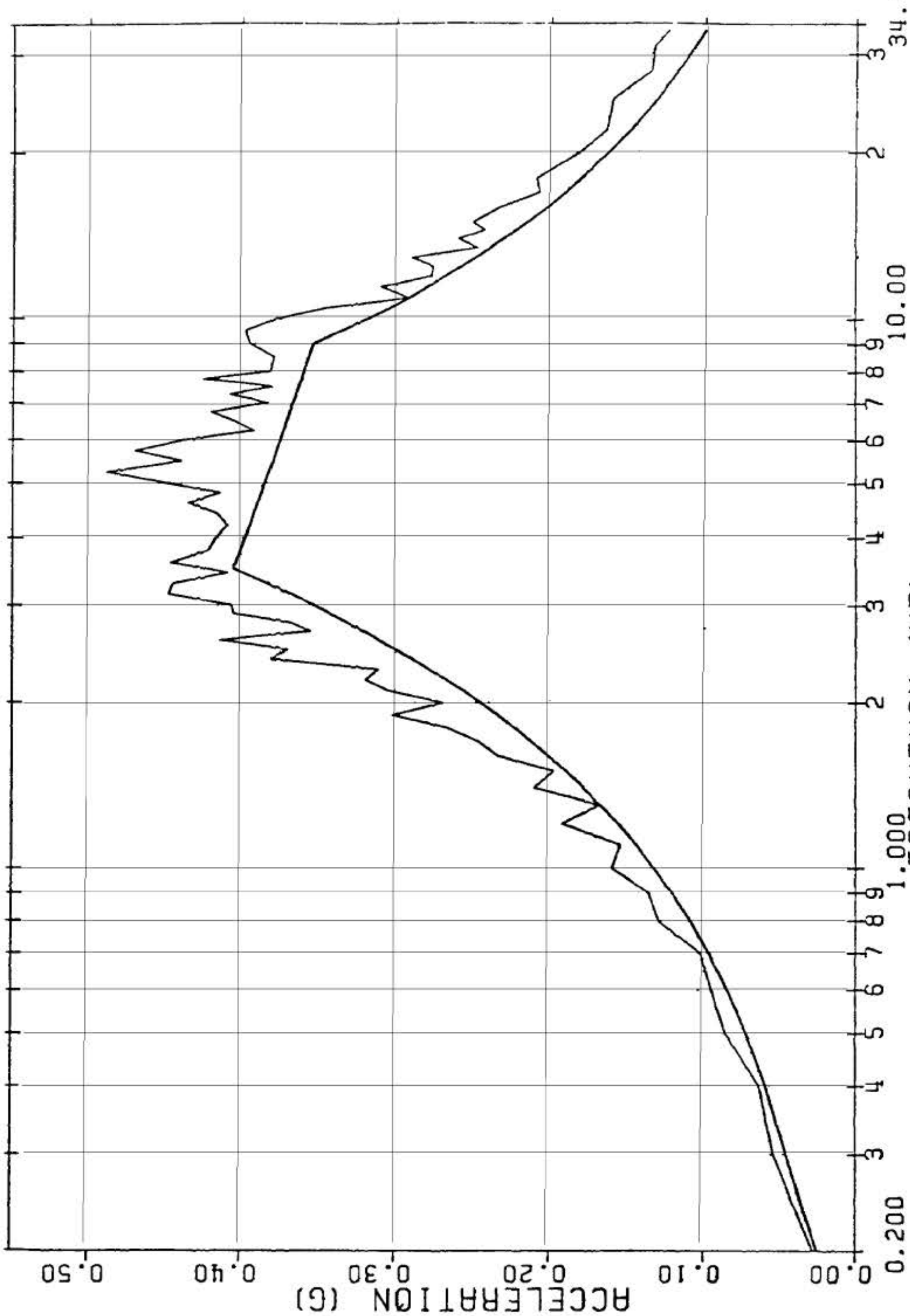
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

DIESEL GENERATOR 'E' BUILDING
COMPARISON OF HORIZONTAL TIME
HISTORY RESPONSE SPECTRUM AND
HORIZONTAL DESIGN RESPONSE
SPECTRUM 7% DAMPING

FIGURE 3.7B-13, Rev. 55

Auto-Cad Figure Fsar 3_7B_13.dwg



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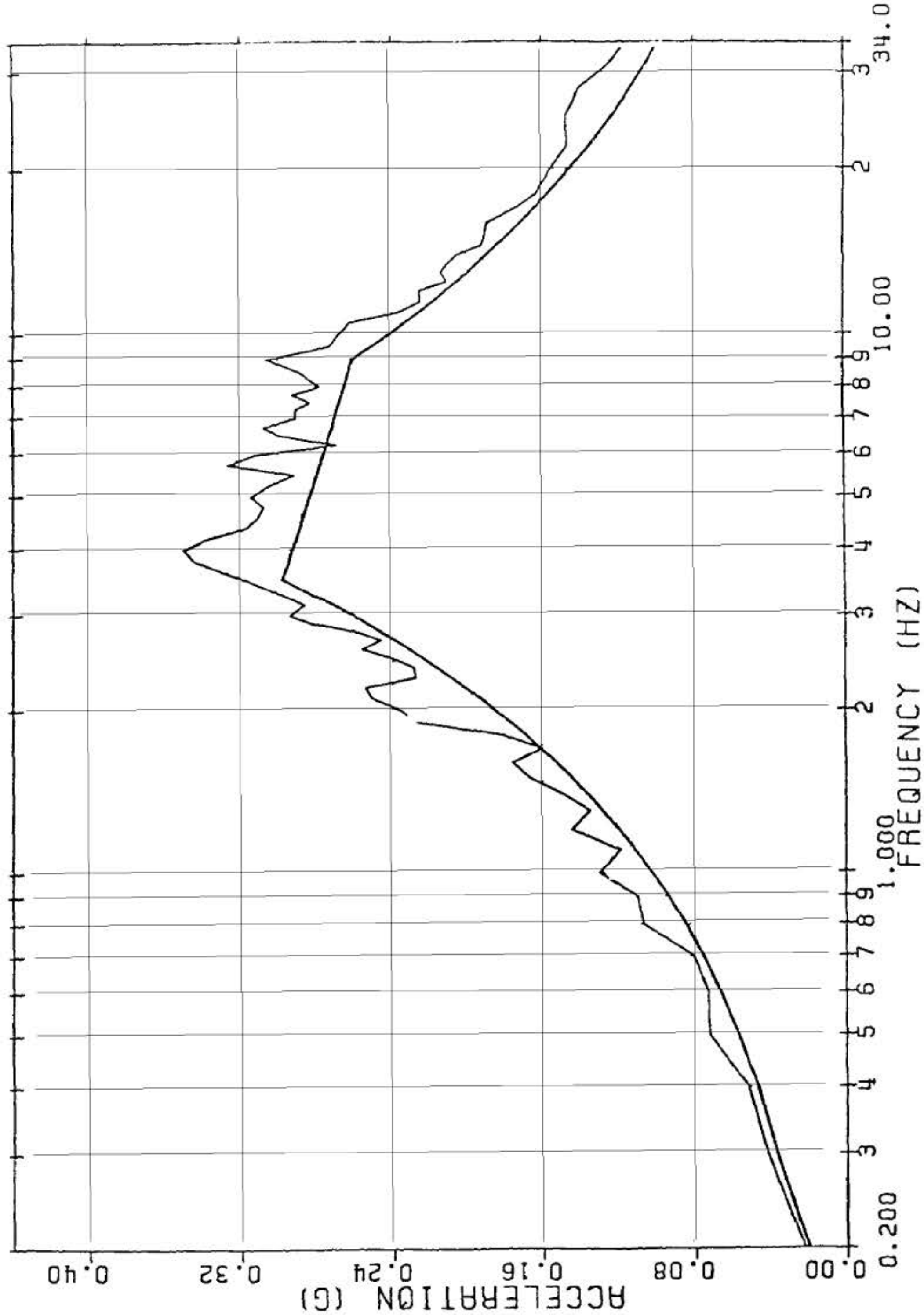
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

DIESEL GENERATOR 'E' BUILDING
COMPARISON OF VERTICAL TIME
HISTORY RESPONSE SPECTRUM AND
VERTICAL DESIGN RESPONSE
SPECTRUM 2% DAMPING

FIGURE 3.7B-14, Rev. 55

Auto-Cad Figure Fsar 3_7B_14.dwg



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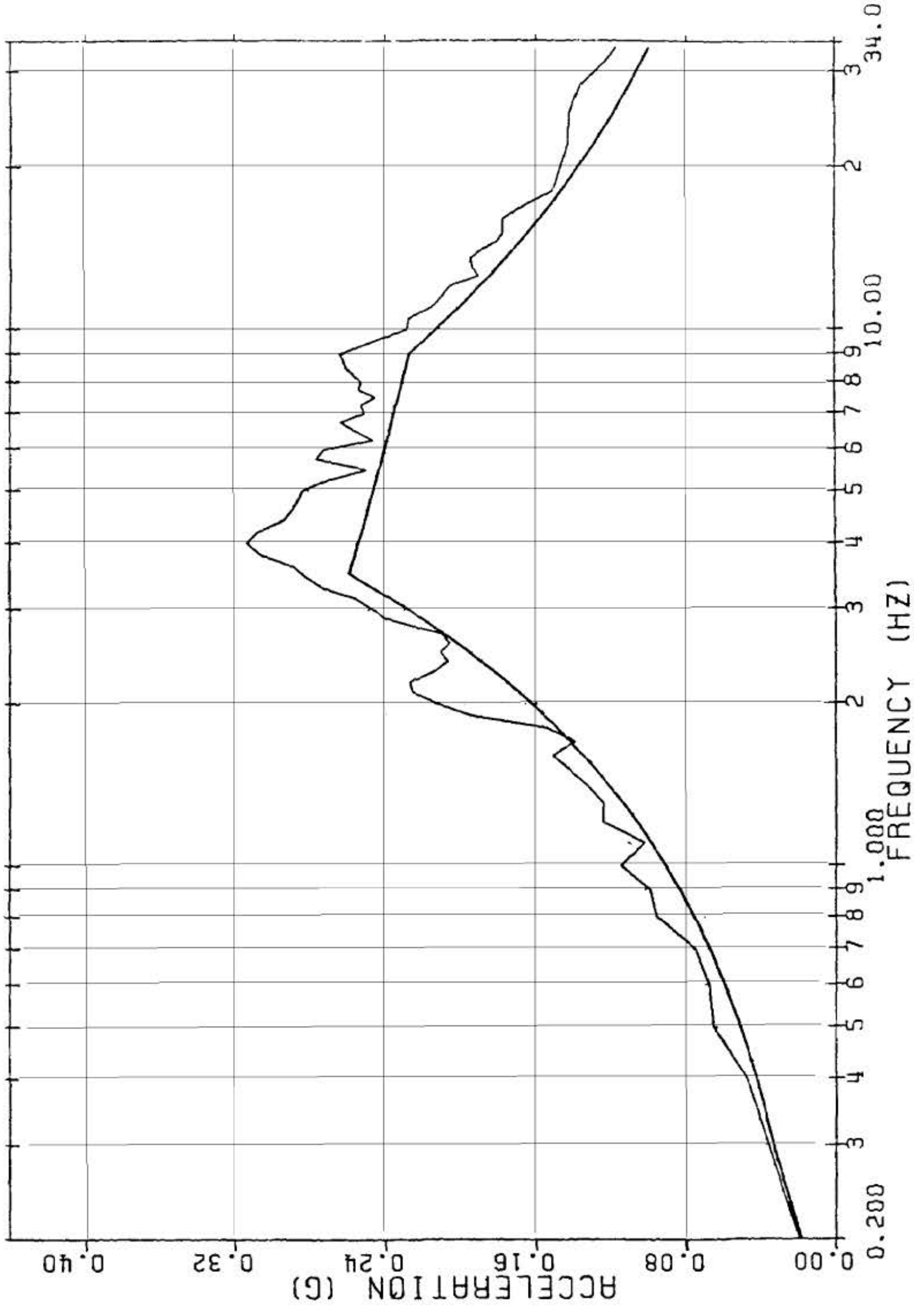
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

DIESEL GENERATOR 'E' BUILDING
COMPARISON OF VERTICAL TIME
HISTORY RESPONSE SPECTRUM AND
VERTICAL DESIGN RESPONSE
SPECTRUM 5% DAMPING

FIGURE 3.7B-15, Rev. 55

Auto-Cad Figure Fsar 3_7B_15.dwg



FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

DIESEL GENERATOR 'E' BUILDING
COMPARISON OF VERTICAL TIME
HISTORY RESPONSE SPECTRUM AND
VERTICAL DESIGN RESPONSE
SPECTRUM 7% DAMPING

FIGURE 3.7B-16, Rev. 55

Auto-Cad Figure Fsar 3_7B_16.dwg

Security-Related Information
Figure Withheld Under 10 CFR 2.390

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

HORIZONTAL SEISMIC MODEL
OF CONTAINMENT WITH
FLEXIBLE BASE

FIGURE 3.7B-17, Rev. 55

Auto-Cad Figure Fsar 3_7B_17.dwg

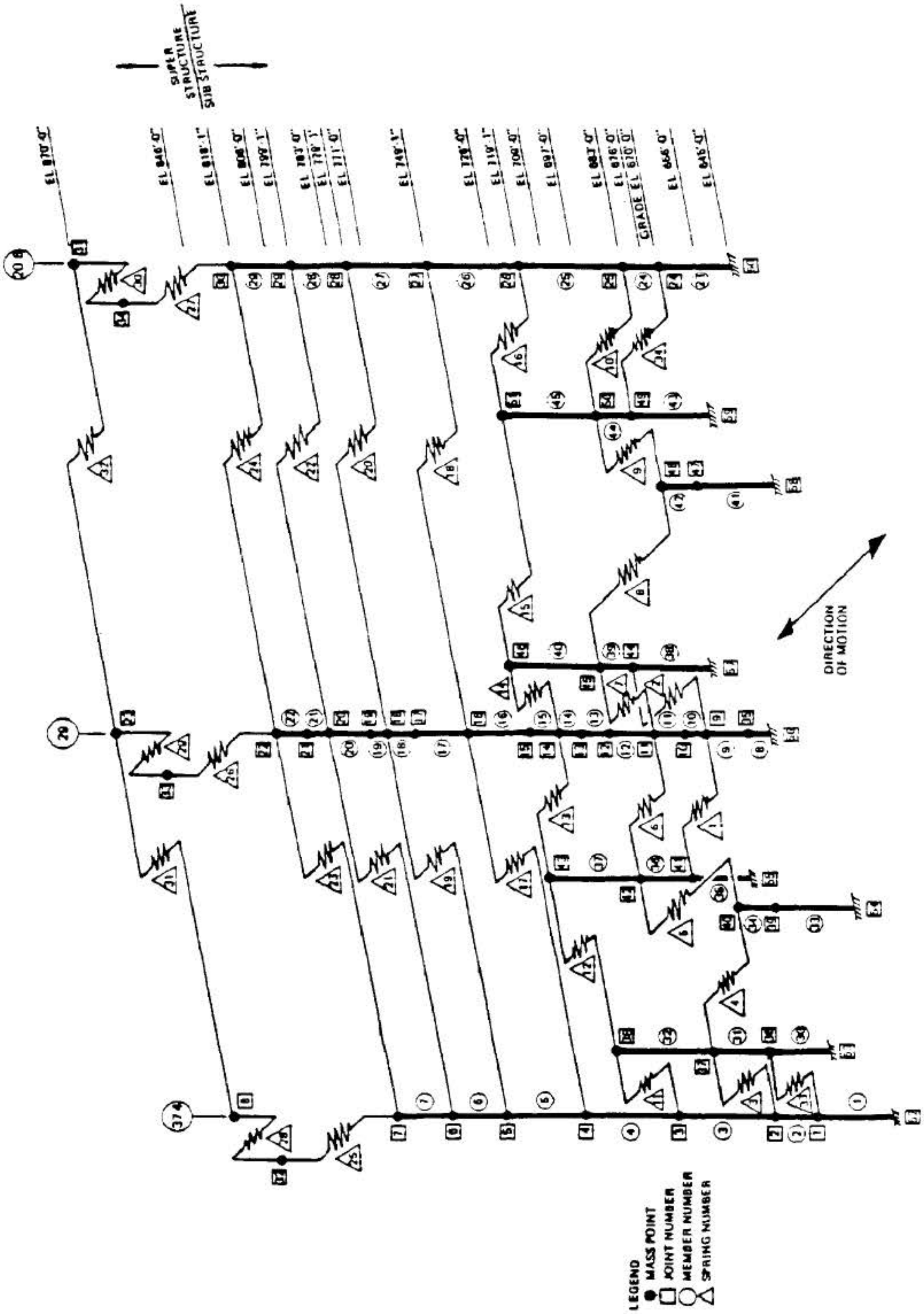
Security-Related Information
Figure Withheld Under 10 CFR 2.390

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

VETICAL SEISMIC
MODEL OF CONTAINMENT
WITH FLEXIBLE BASE

FIGURE 3.7B-18, Rev. 55

Auto-Cad Figure Fsar 3_7B_18.dwg



FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

E-W SEISMIC MODEL OF
REACTOR AND CONTROL BUILDING

FIGURE 3.7B-19, Rev. 55

Auto-Cad Figure Fsar 3_7B_19.dwg

THIS FIGURE HAS BEEN
RENUMBERED TO 3.7B-2

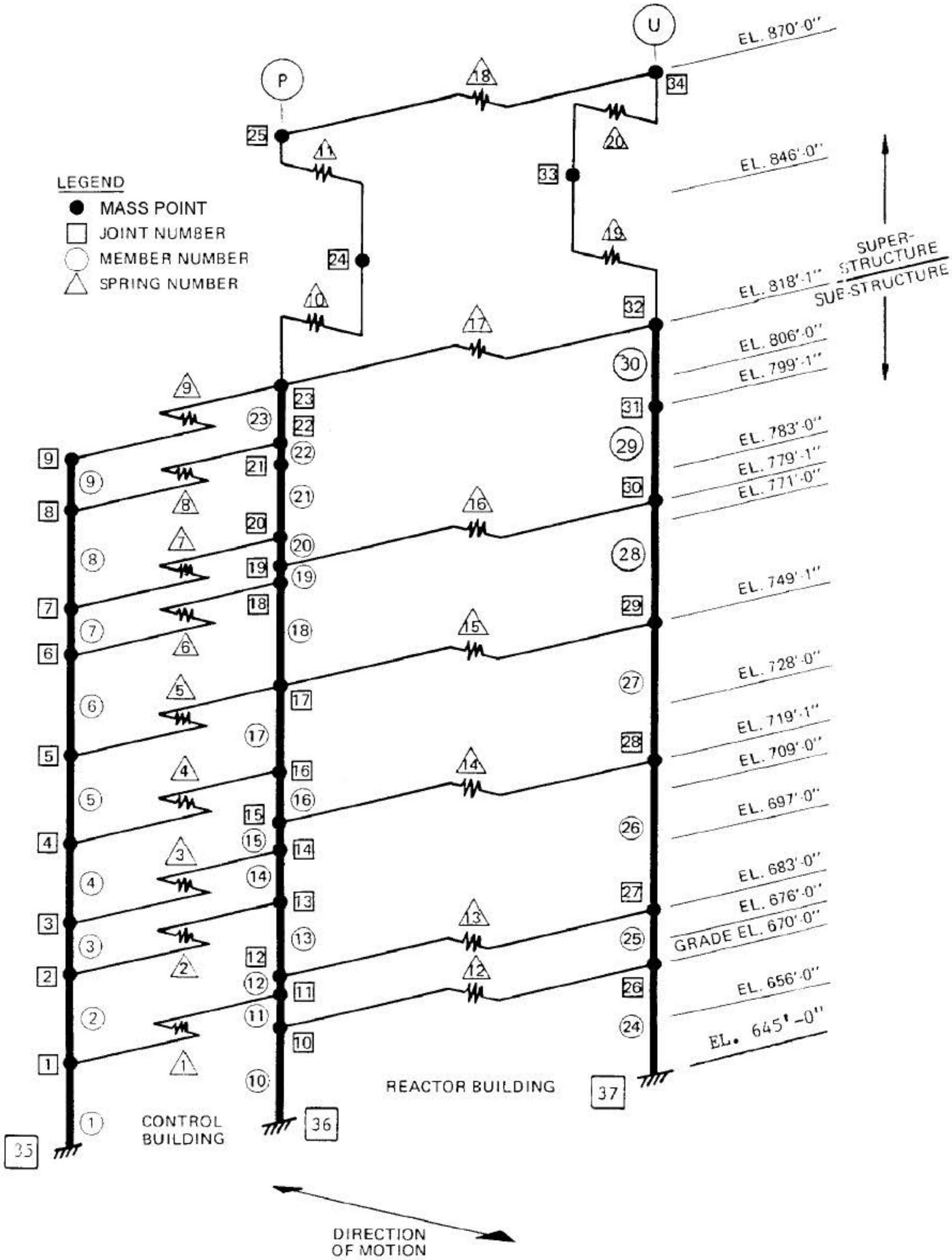
FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

Figure renumbered from 3.7B-1A to 3.7B-2

FIGURE 3.7B-1A, Rev. 55

AutoCAD Figure 3_7B_1A.doc



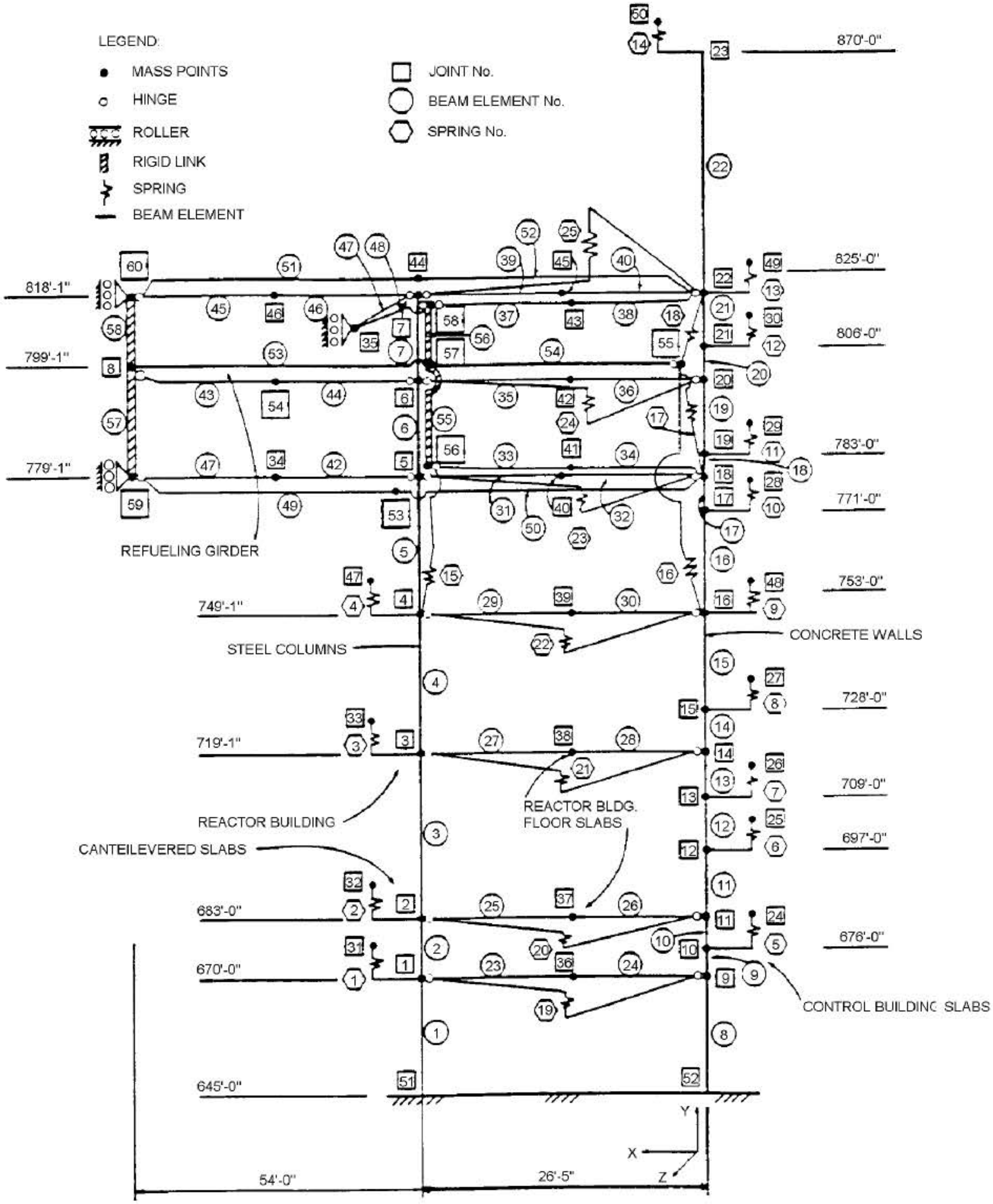
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

N-S SEISMIC MODEL OF
REACTOR AND CONTROL BUILDING

FIGURE 3.7B-20, Rev. 55

Auto-Cad Figure Fsar 3_7B_20.dwg



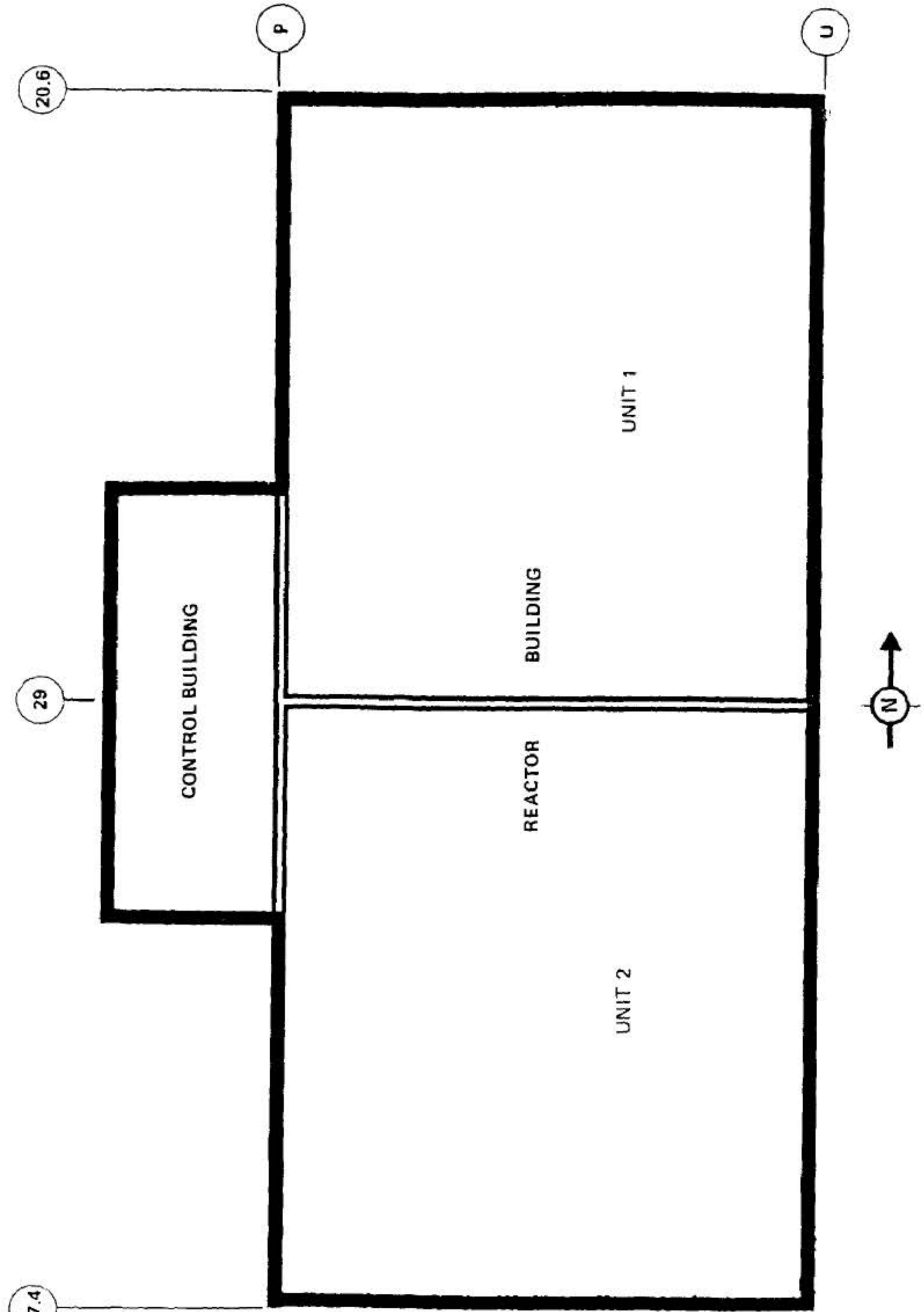
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

VERTICAL SEISMIC MODEL OF
REACTOR AND CONTROL BUILDING

FIGURE 3.7B-21, Rev. 55

Auto-Cad Figure Fsar 3_7B_21.dwg



FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

PLAN VIEW OF
REACTOR AND CONTROL BUILDING

FIGURE 3.7B-22, Rev. 55

Auto-Cad Figure Fsar 3_7B_22.dwg

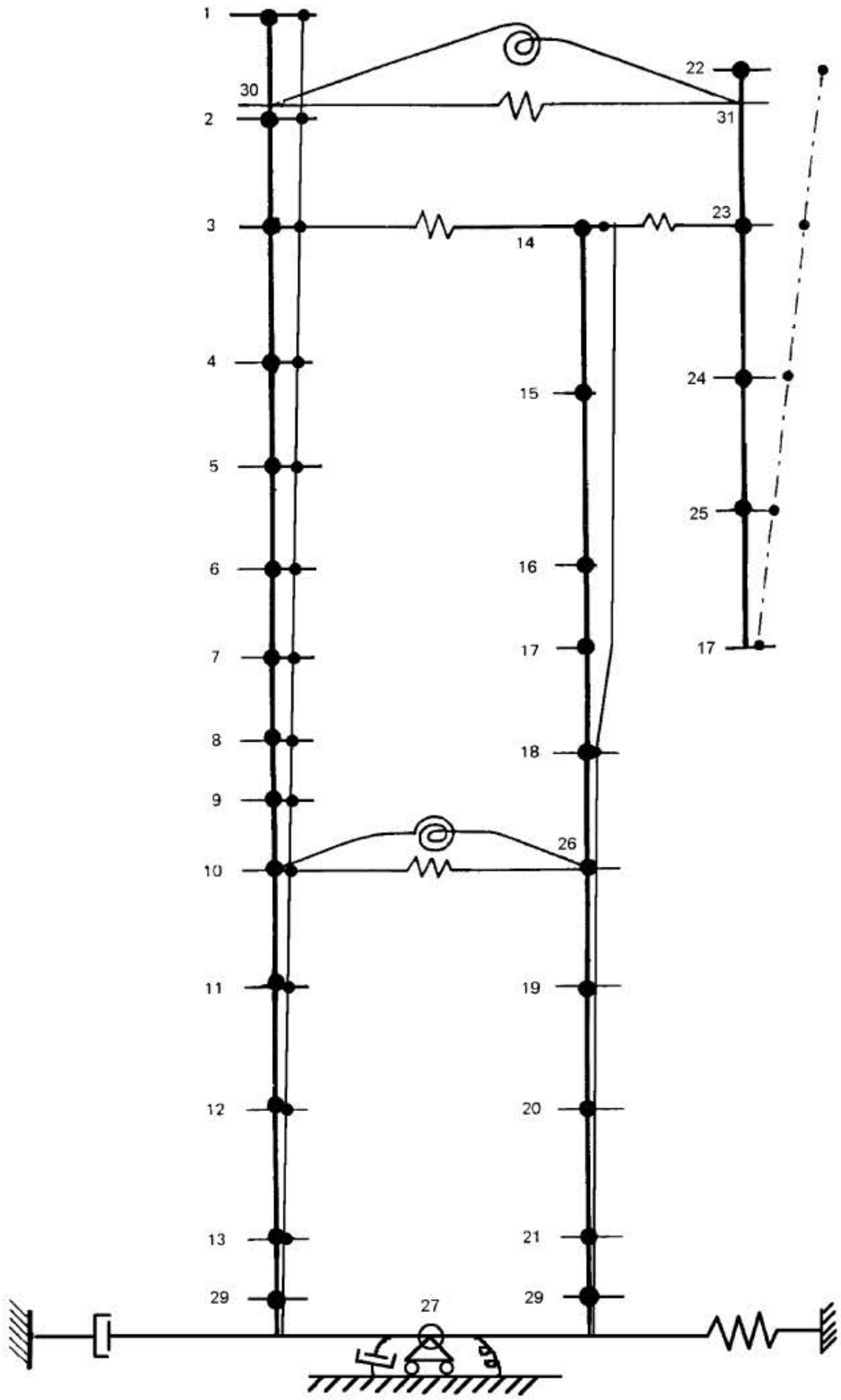
Security-Related Information
Figure Withheld Under 10 CFR 2.390

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

CORRELATION OF VERTICAL
SEISMIC MODEL MASSPOINTS OF
THE PHYSICAL STRUCTURE

FIGURE 3.7B-23, Rev. 55

Auto-Cad Figure Fsar 3_7B_23.dwg



FREQUENCY = 4.99 CPS

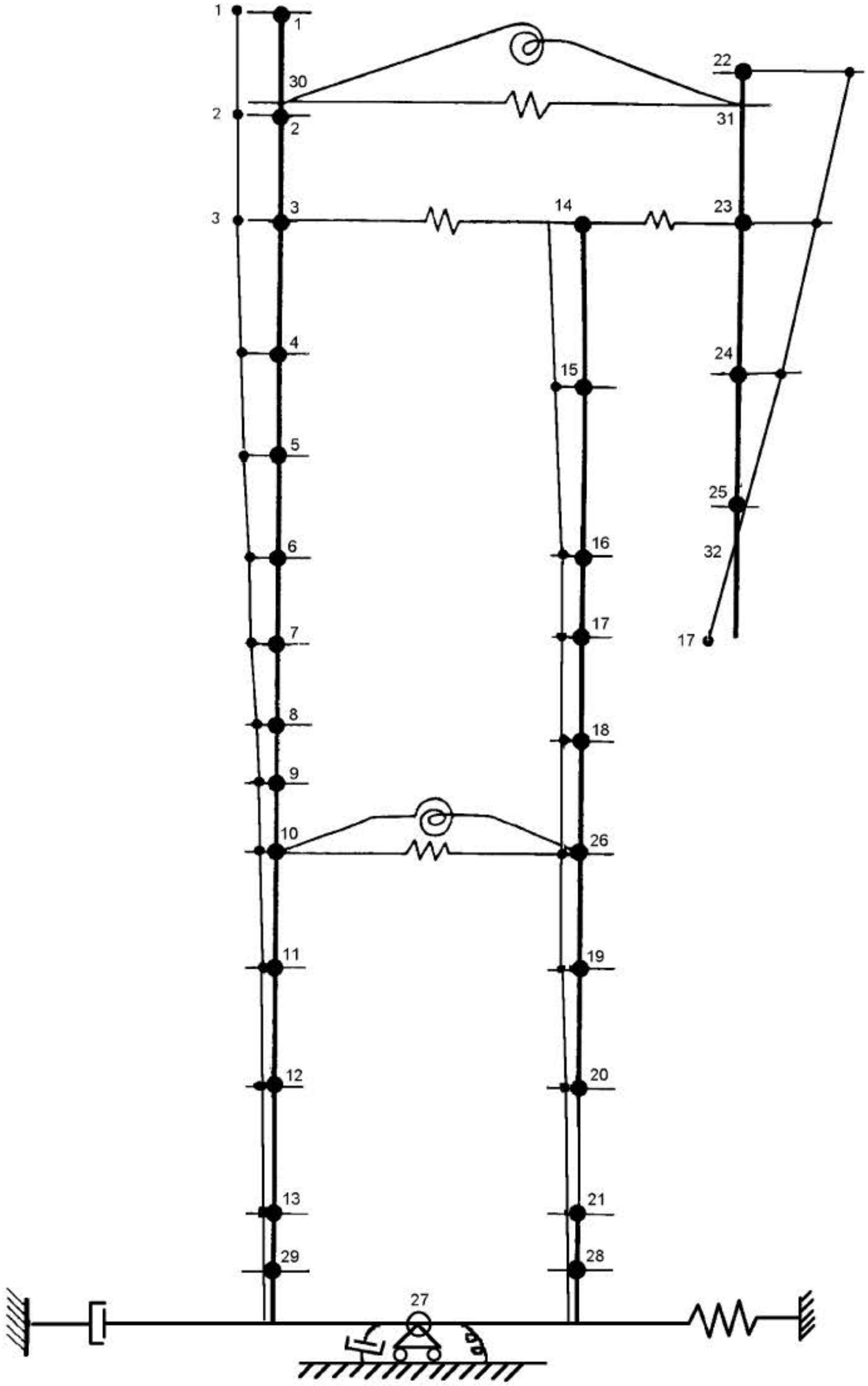
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

CONTAINMENT
HORIZONTAL MODE SHAPES
MODE 1

FIGURE 3.7B-24, Rev. 55

Auto-Cad Figure Fsar 3_7B_24.dwg



FREQUENCY = 8.77 CPS

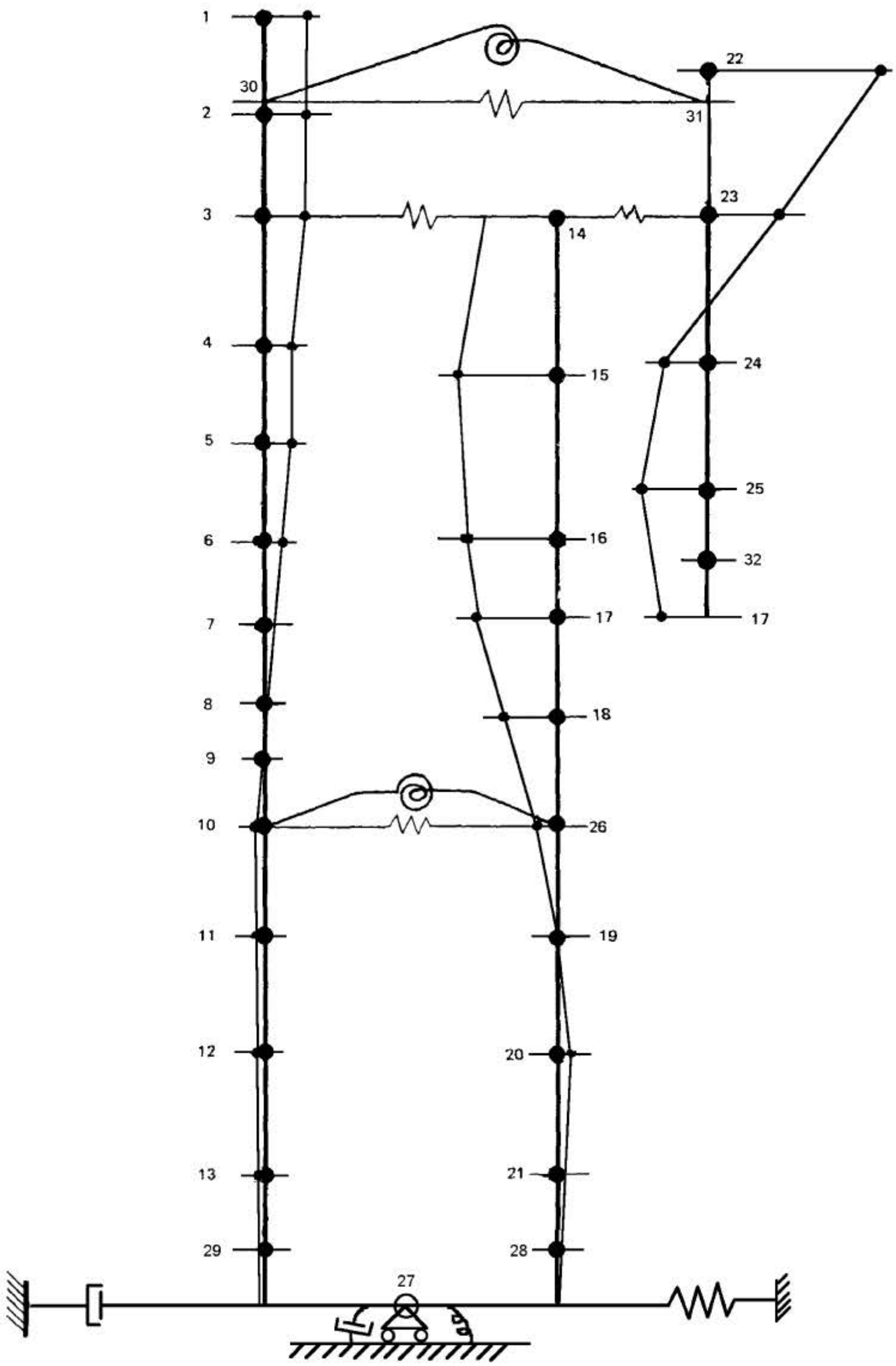
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

CONTAINMENT
HORIZONTAL MODE SHAPES
MODE 2

FIGURE 3.7B-25, Rev. 55

Auto-Cad Figure Fsar 3_7B_25.dwg



FREQUENCY = 16.12 CPS

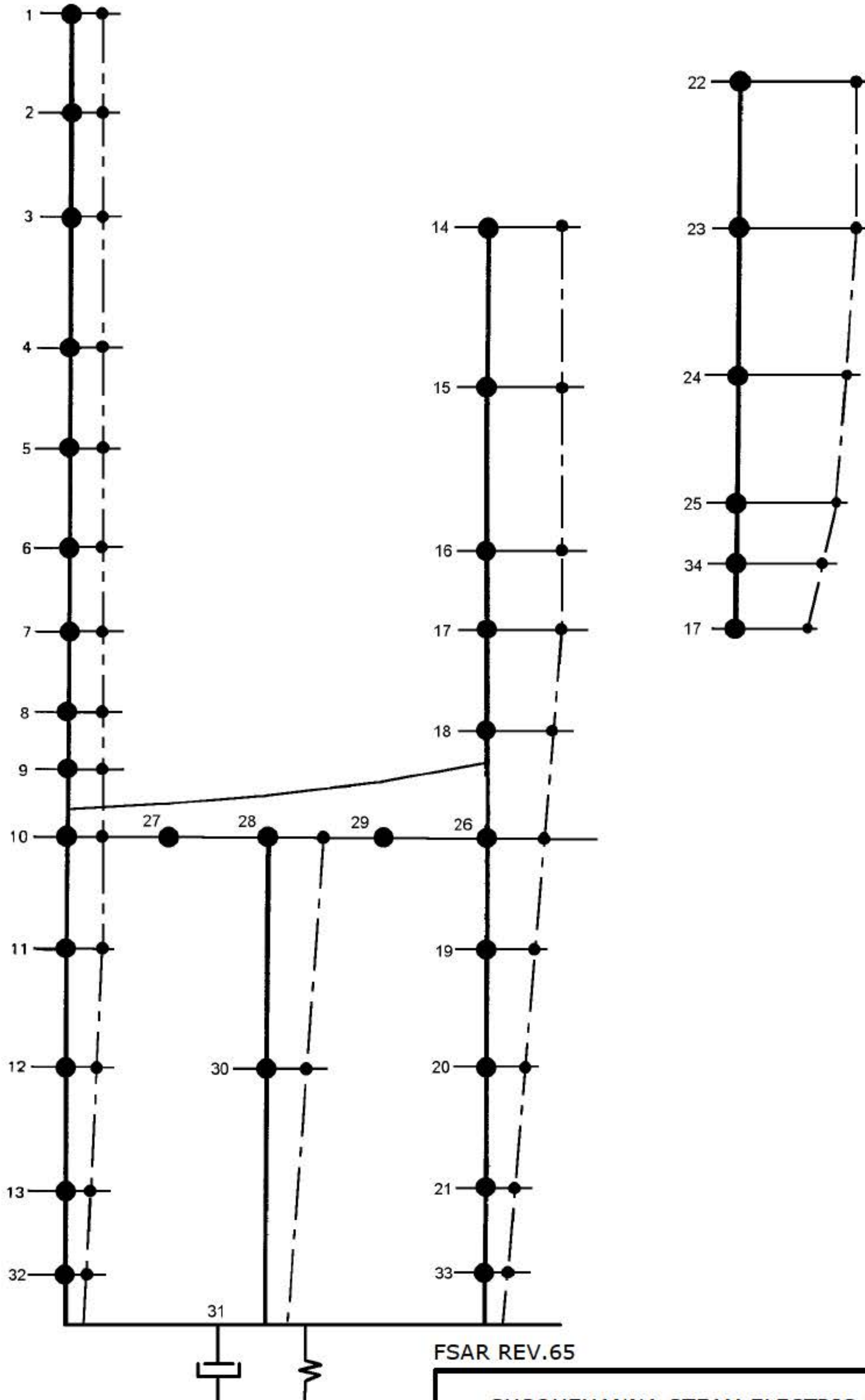
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

CONTAINMENT
HORIZONTAL MODE SHAPES
MODE 4

FIGURE 3.7B-26, Rev. 55

Auto-Cad Figure Fsar 3_7B_26.dwg



FREQUENCY = 16.19 CPS

FSAR REV.65

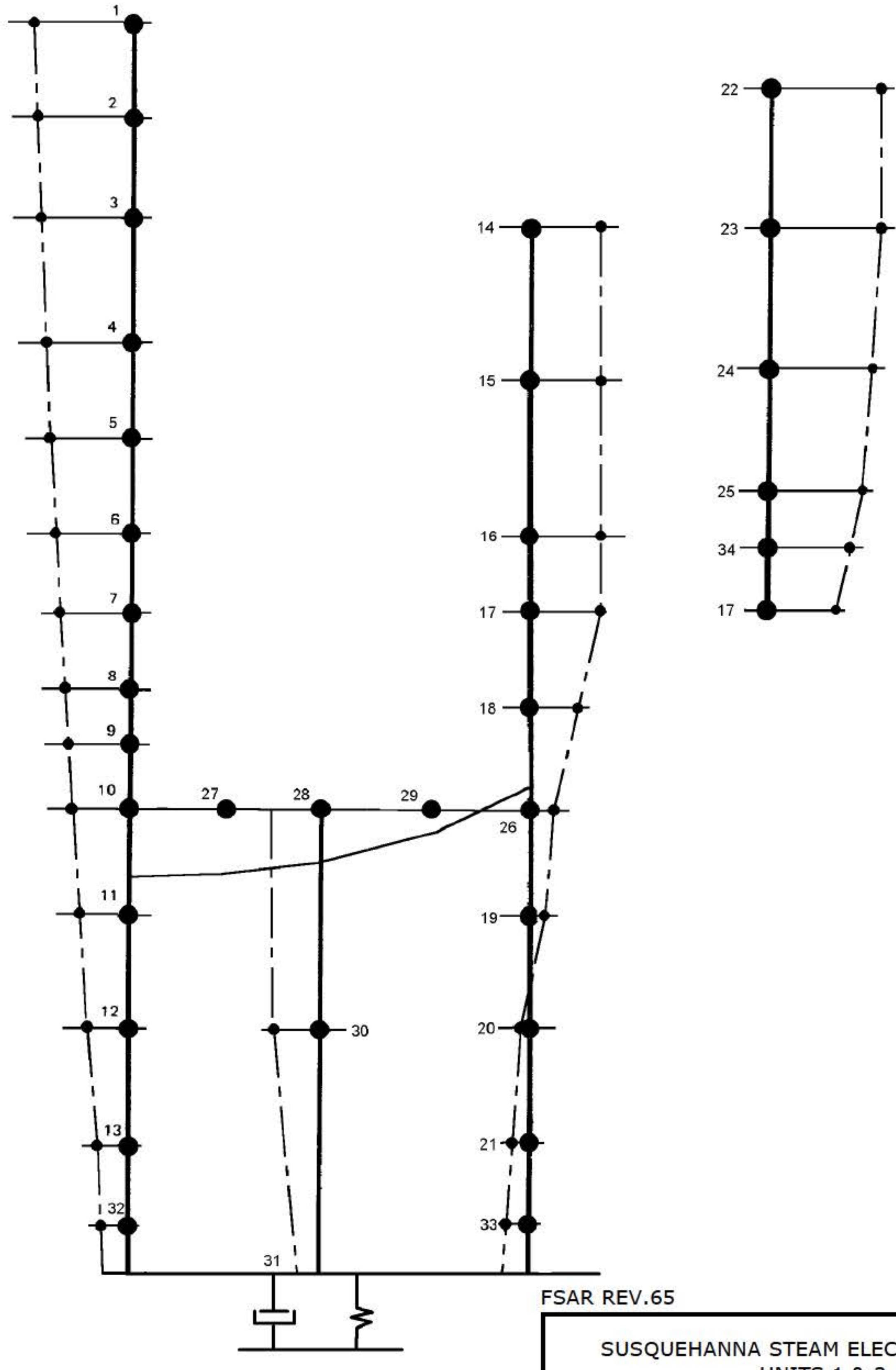
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

CONTAINMENT
VERTICAL MODE SHAPES
MODE 1

FIGURE 3.7B-27, Rev. 55

Auto-Cad Figure Fsar 3_7B_27.dwg

NOTE:
FOR CLARITY OF ILLUSTRATION, THE DISPLACEMENT VALUES ASSOCIATED WITH THE NODES ORIENTATED ALONG THE VERTICAL MEMBERS ARE DISPLAYED GRAPHICALLY IN THE HORIZONTAL DIRECTION. THE DISPLACEMENTS ARE IN THE VERTICAL DIRECTION.



FREQUENCY = 20.95 CPS

NOTE:
FOR CLARITY OF ILLUSTRATION, THE DISPLACEMENT VALUES ASSOCIATED WITH THE NODES ORIENTATED ALONG THE VERTICAL MEMBERS ARE DISPLAYED GRAPHICALLY IN THE HORIZONTAL DIRECTION. THE DISPLACEMENTS ARE IN THE VERTICAL DIRECTION.

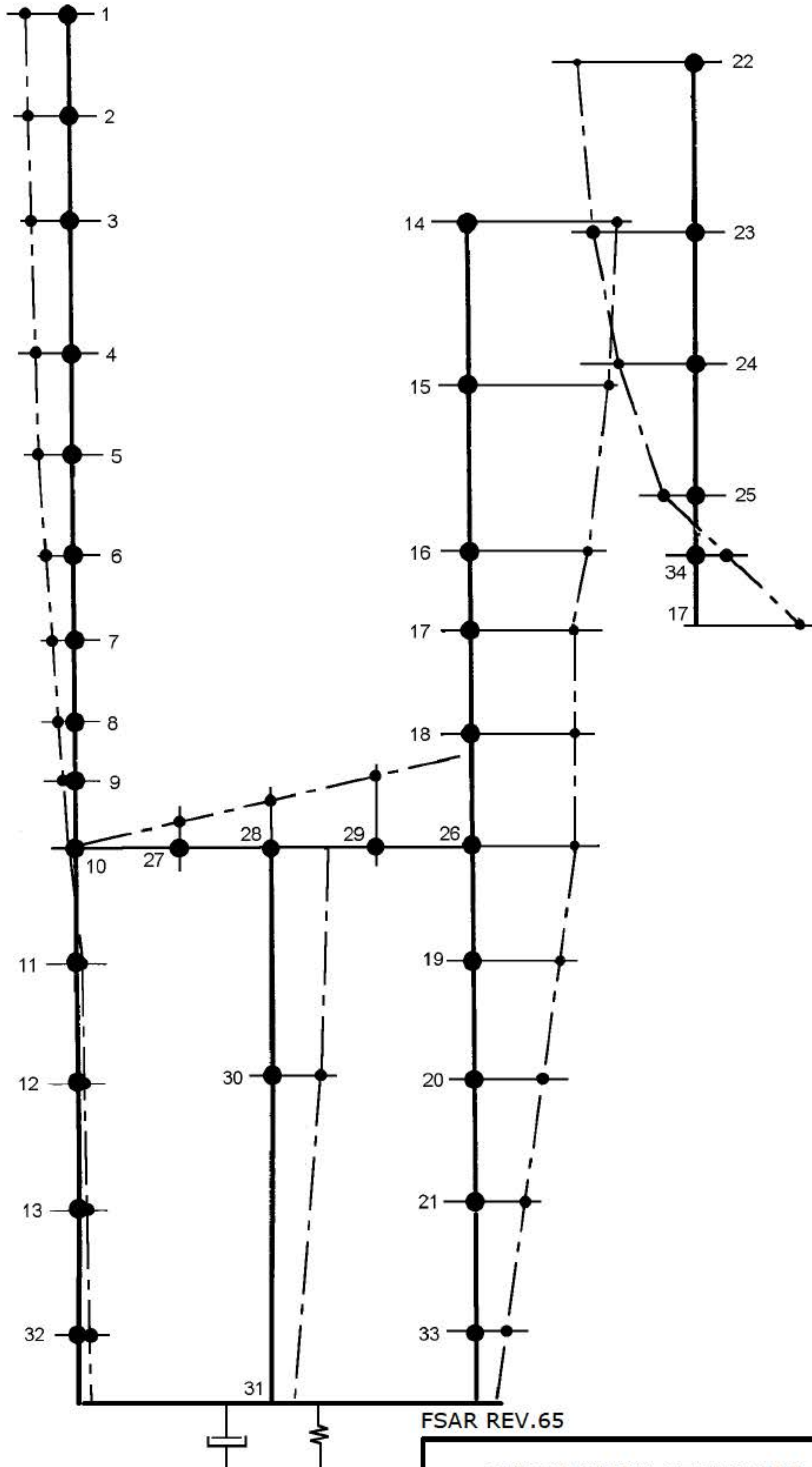
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

CONTAINMENT
VERTICAL MODE SHAPES
MODE 2

FIGURE 3.7B-28, Rev. 55

Auto-Cad Figure Fsar 3_7B_28.dwg



FREQUENCY = 38.24

NOTE:
FOR CLARITY OF ILLUSTRATION, THE DISPLACEMENT VALUES ASSOCIATED WITH THE NODES ORIENTATED ALONG THE VERTICAL MEMBERS ARE DISPLAYED GRAPHICALLY IN THE HORIZONTAL DIRECTION. THE DISPLACEMENTS ARE IN THE VERTICAL DIRECTION.

FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

CONTAINMENT
VERTICAL MODE SHAPES
MODE 3

FIGURE 3.7B-29, Rev. 55

Auto-Cad Figure Fsar 3_7B_29.dwg

THIS FIGURE HAS BEEN
RENUMBERED TO 3.7B-4

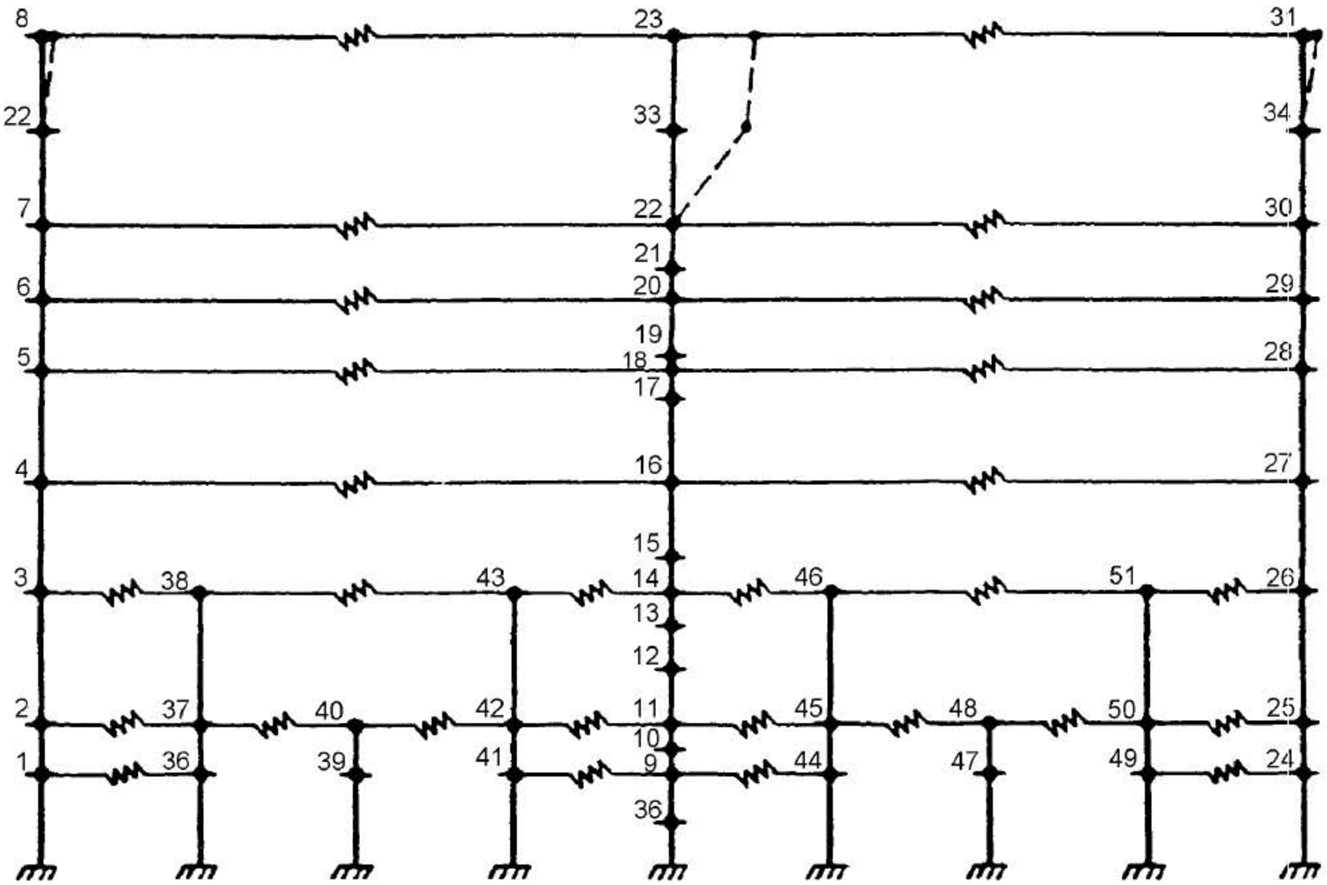
FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

Figure renumbered from 3.7B-2A to 3.7B-4

FIGURE 3.7B-2A, Rev. 55

AutoCAD Figure 3_7B_2A.doc



NOTE: CRANES ARE LOCATED
AT MASS POINTS 32 AND 33

FREQUENCY = 2.23 CPS

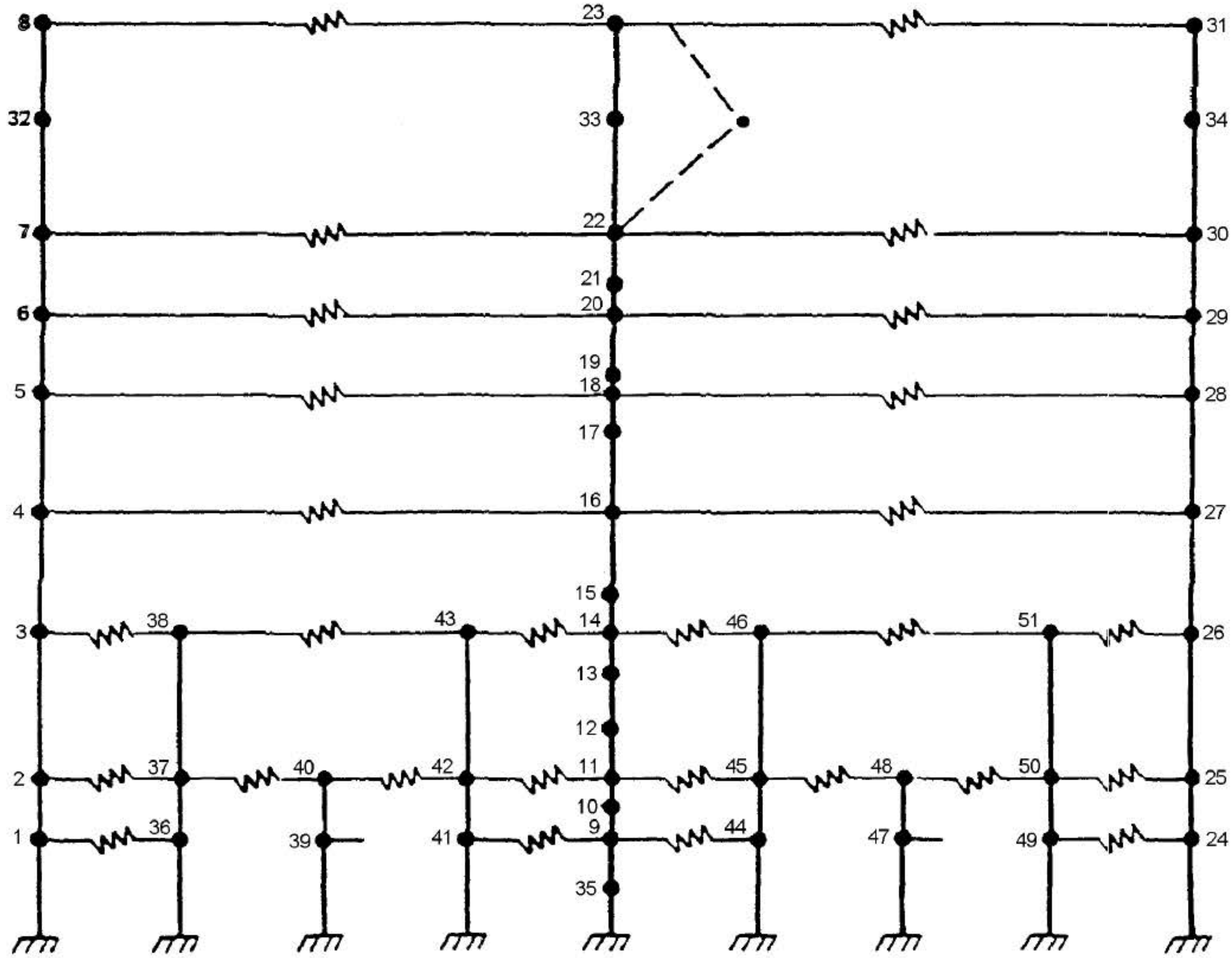
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

REACTOR AND CONTROL BUILDING
E-W MODE SHAPES - MODE 1
(CRANES AT POINTS 32 AND 33)

FIGURE 3.7B-30, Rev. 55

Auto-Cad Figure Fsar 3_7B_30.dwg



NOTE: CRANES ARE LOCATED AT MASS POINTS 32 AND 33

FREQUENCY = 2.51 CPS

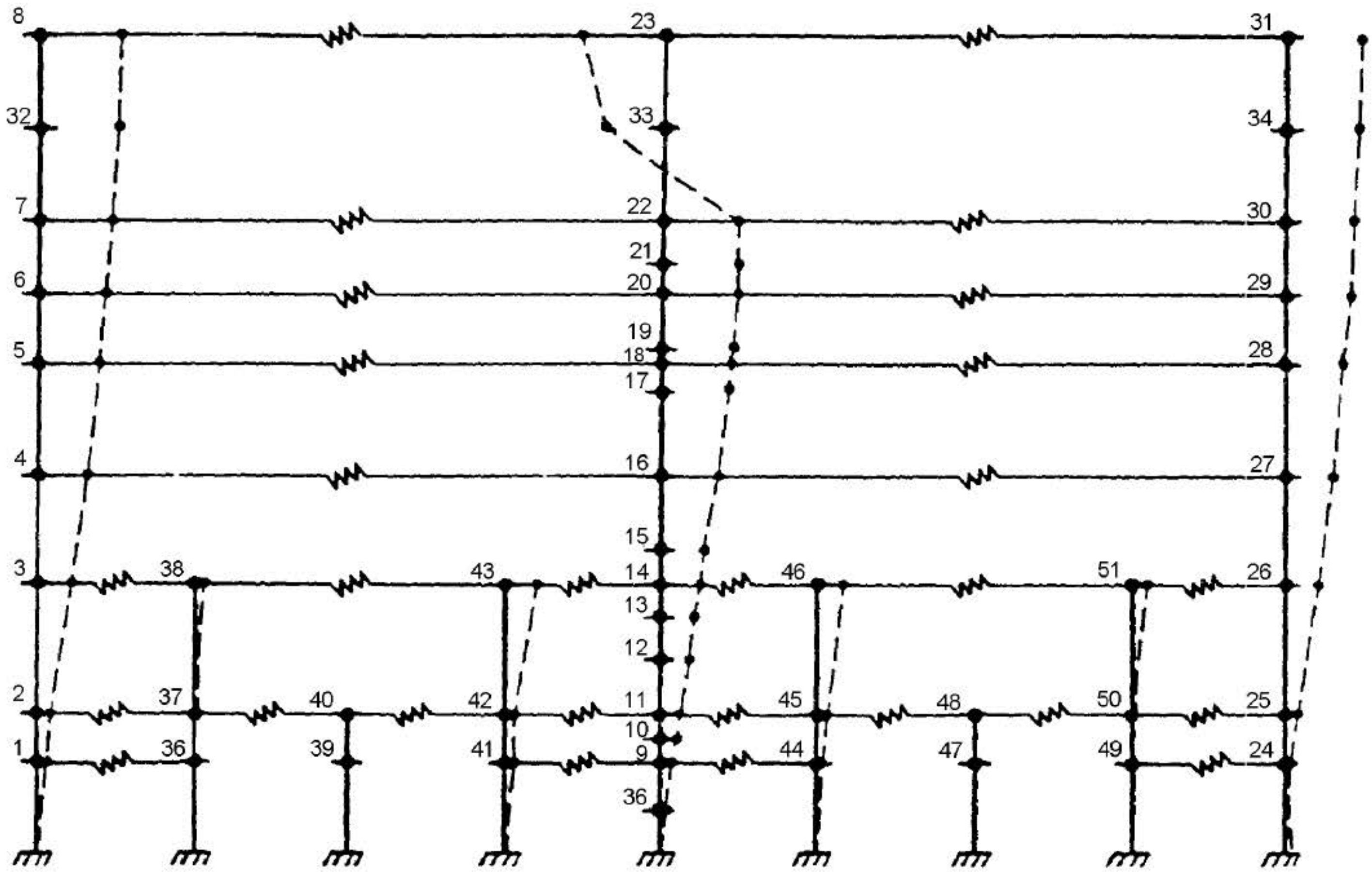
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

REACTOR AND CONTROL BUILDING
E-W MODE SHAPES
MODE 2

FIGURE 3.7B-31, Rev. 55

Auto-Cad Figure Fsar 3_7B_31.dwg



NOTE: CRANES ARE LOCATED
AT MASS POINTS 32 AND 33

FREQUENCY = 3.49 CPS

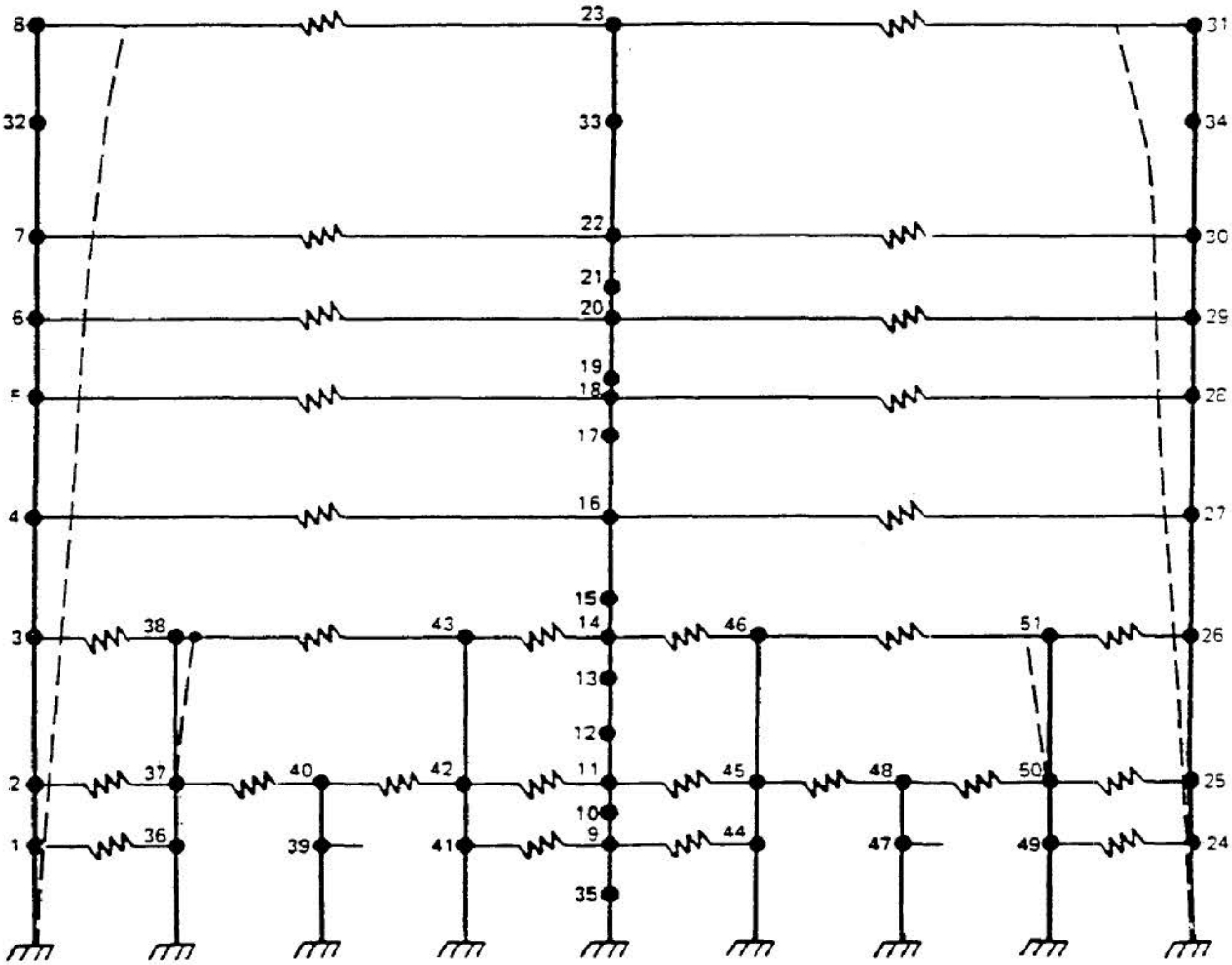
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

REACTOR AND CONTROL BUILDING
E-W MODE SHAPES - MODE 3
(CRANES AT POINTS 32 AND 33)

FIGURE 3.7B-32, Rev. 55

Auto-Cad Figure Fsar 3_7B_32.dwg



NOTE: CRANES ARE LOCATED AT
MASS POINTS 32 AND 33

FREQUENCY = 4.31 CPS

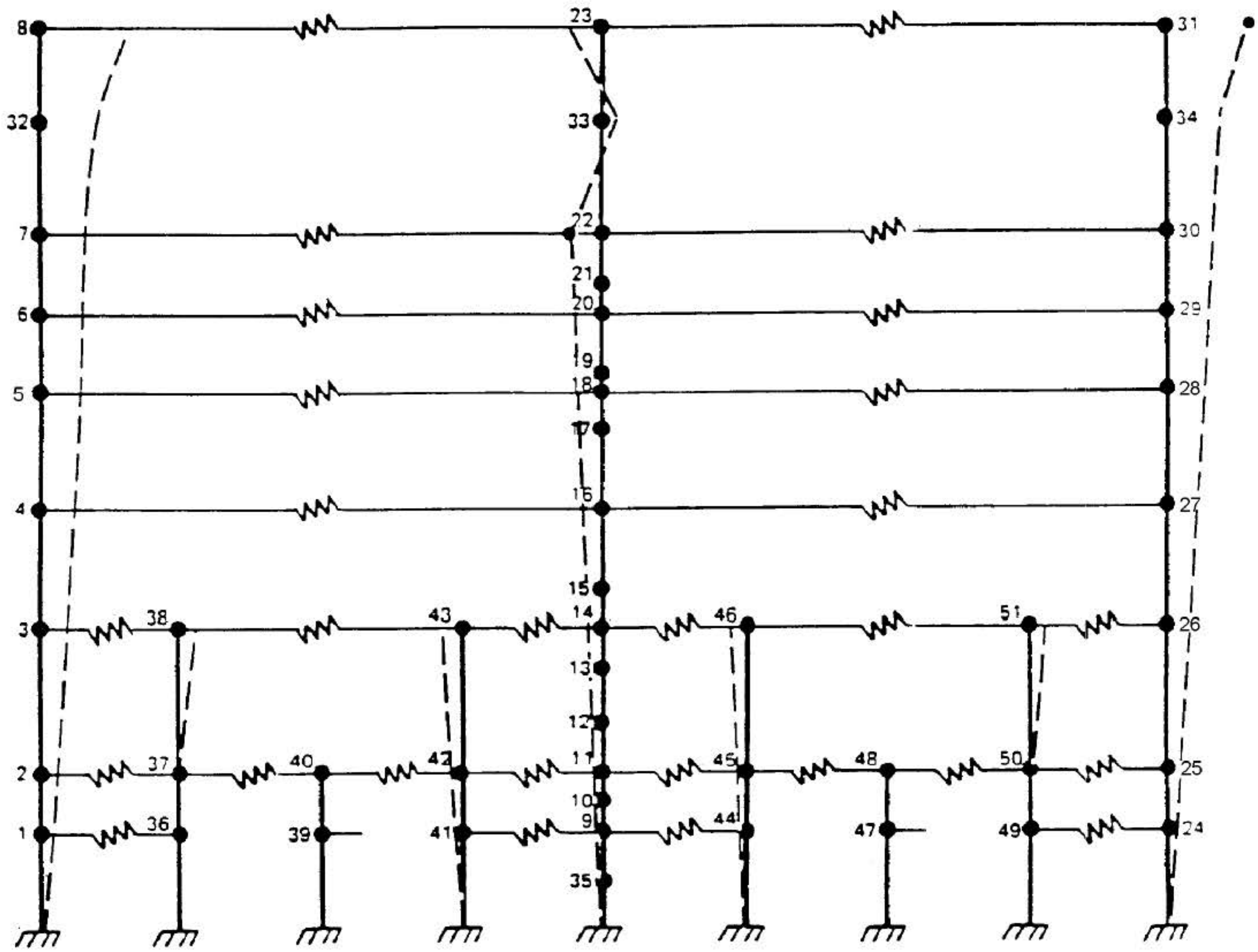
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

REACTOR AND CONTROL BUILDING
E-W MODE SHAPES
MODE 4

FIGURE 3.7B-33, Rev. 55

Auto-Cad Figure Fsar 3_7B_33.dwg



NOTE: CRANES ARE LOCATED AT MASS POINTS 32 AND 33

FREQUENCY = 4.77 CPS

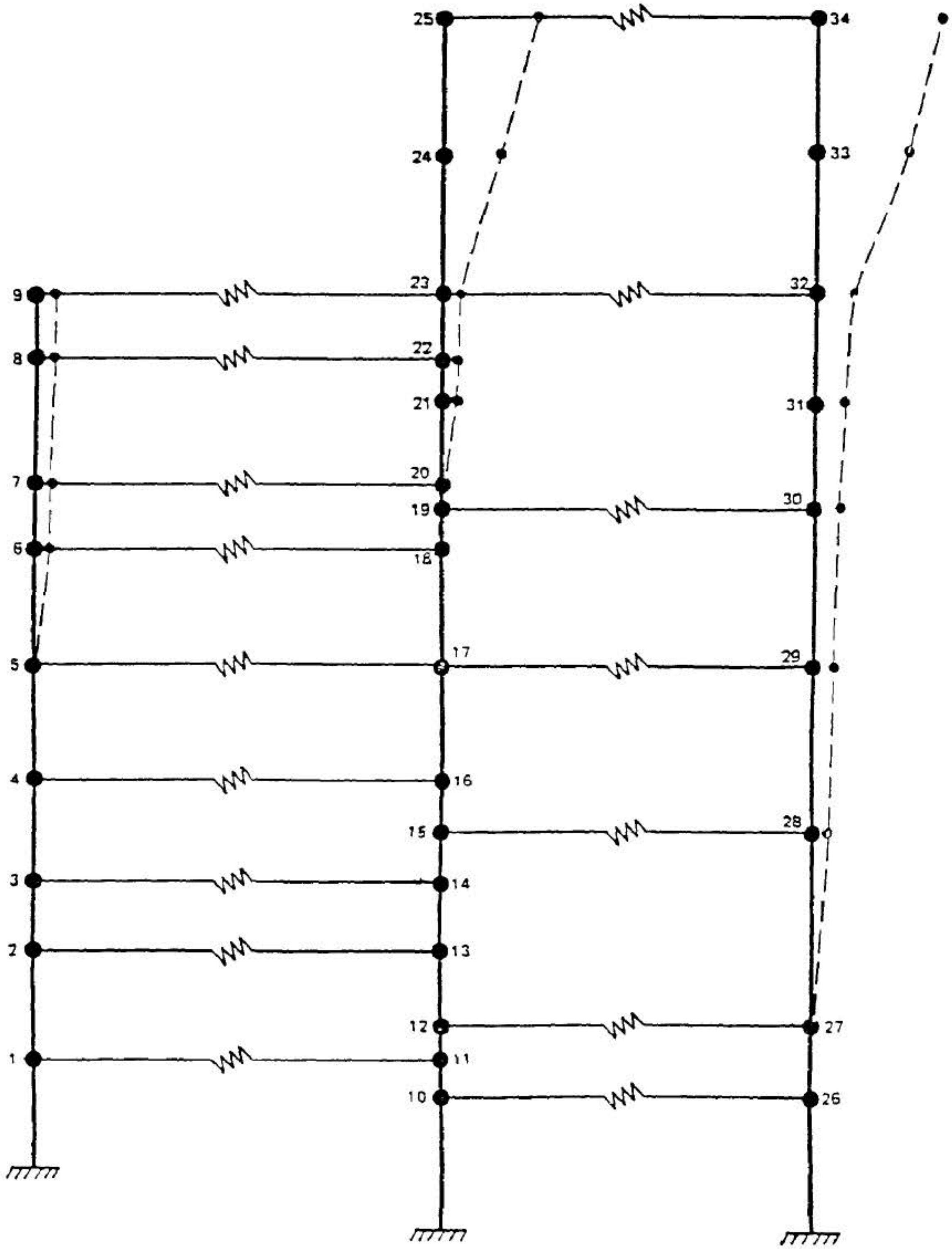
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

REACTOR AND CONTROL BUILDING
E-W MODE SHAPES
MODE 5

FIGURE 3.7B-34, Rev. 55

Auto-Cad Figure Fsar 3_7B_34.dwg



FSAR REV.65

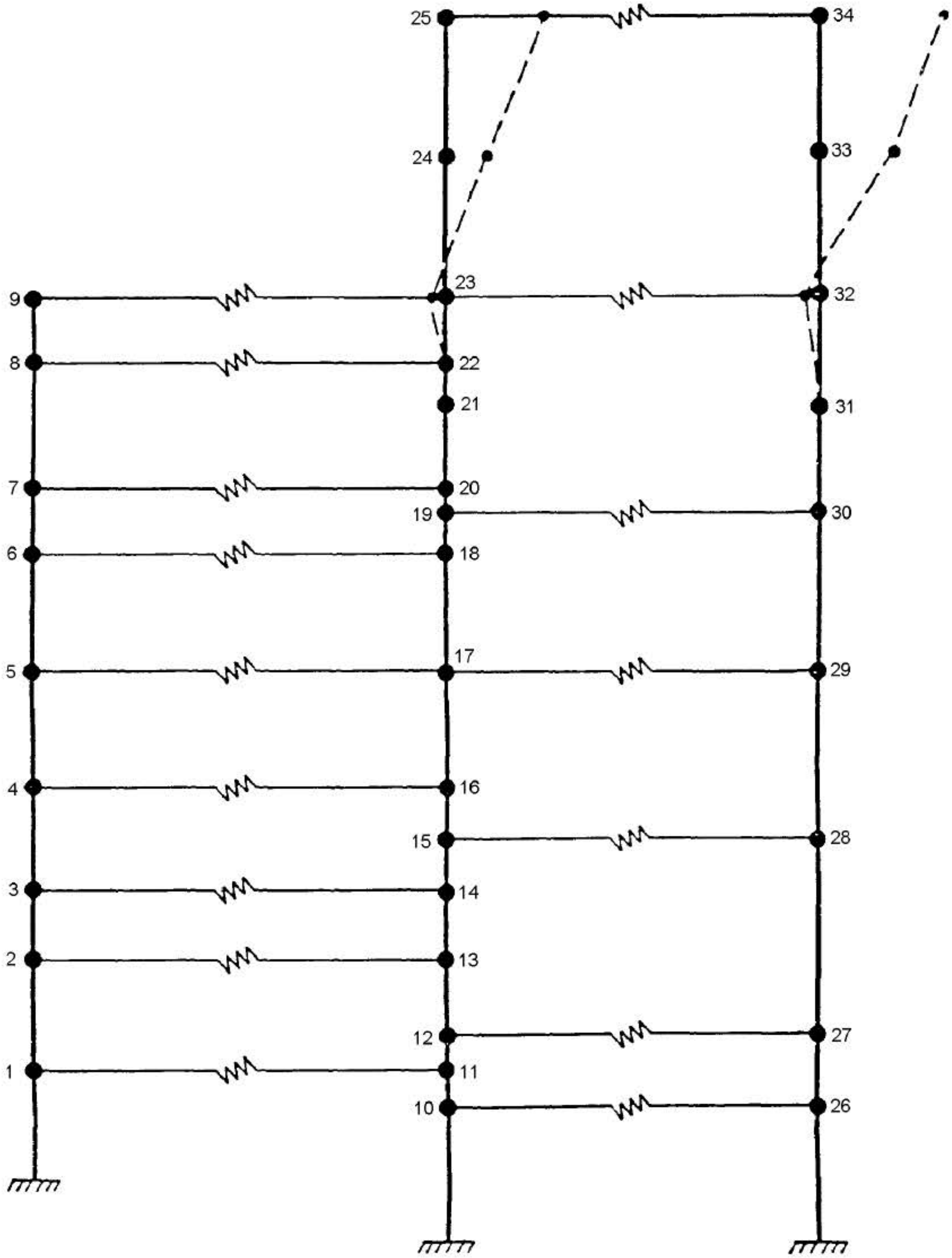
FREQUENCY = 3.92 CPS

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

REACTOR AND CONTROL BUILDING
N-S MODE SHAPES
MODE 1

FIGURE 3.7B-35, Rev. 55

Auto-Cad Figure Fsar 3_7B_35.dwg



FREQUENCY = 4.53 CPS

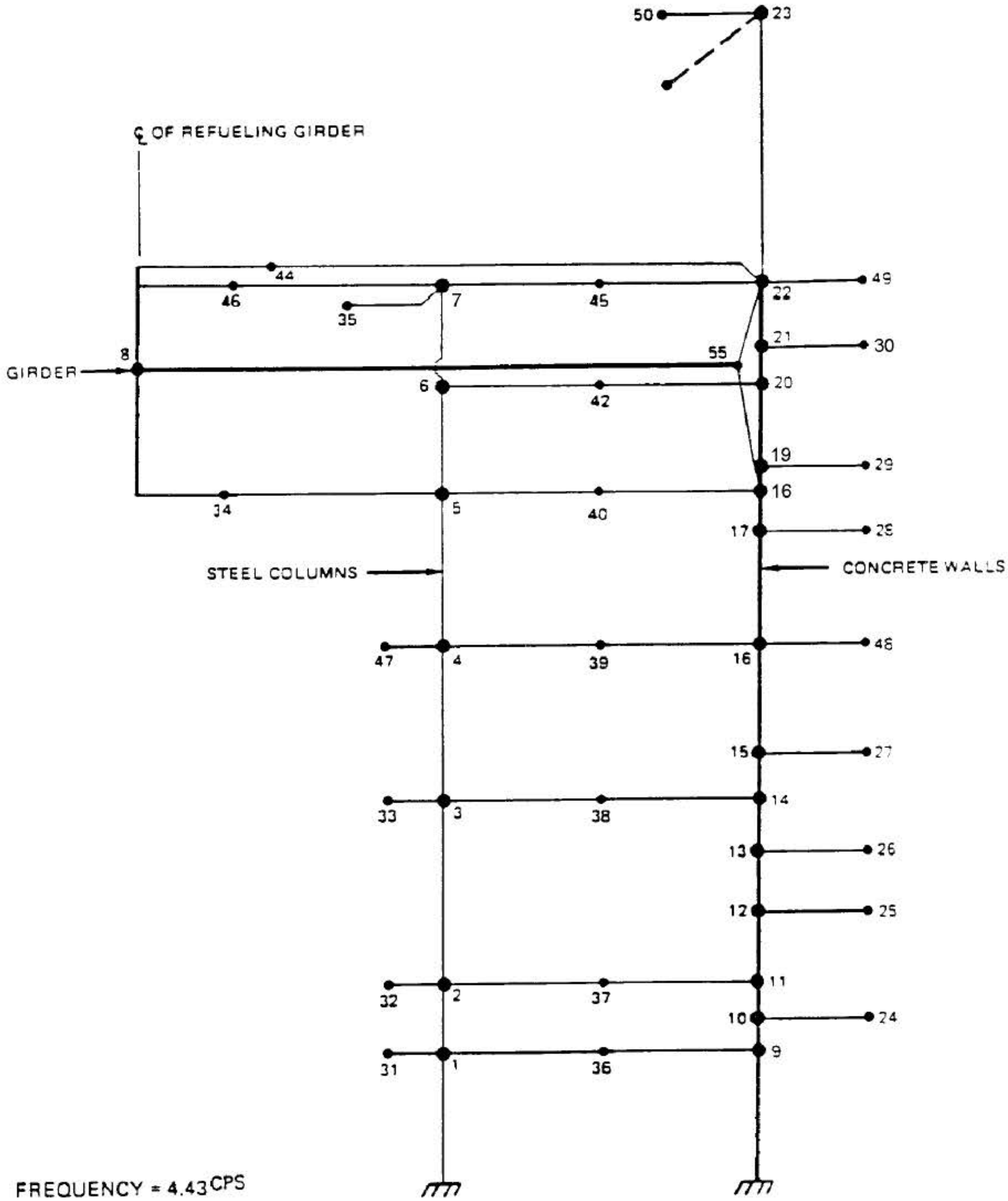
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

REACTOR AND CONTROL BUILDING
N-S MODE SHAPES
MODE 3

FIGURE 3.7B-36, Rev. 55

Auto-Cad Figure Fsar 3_7B_36.dwg



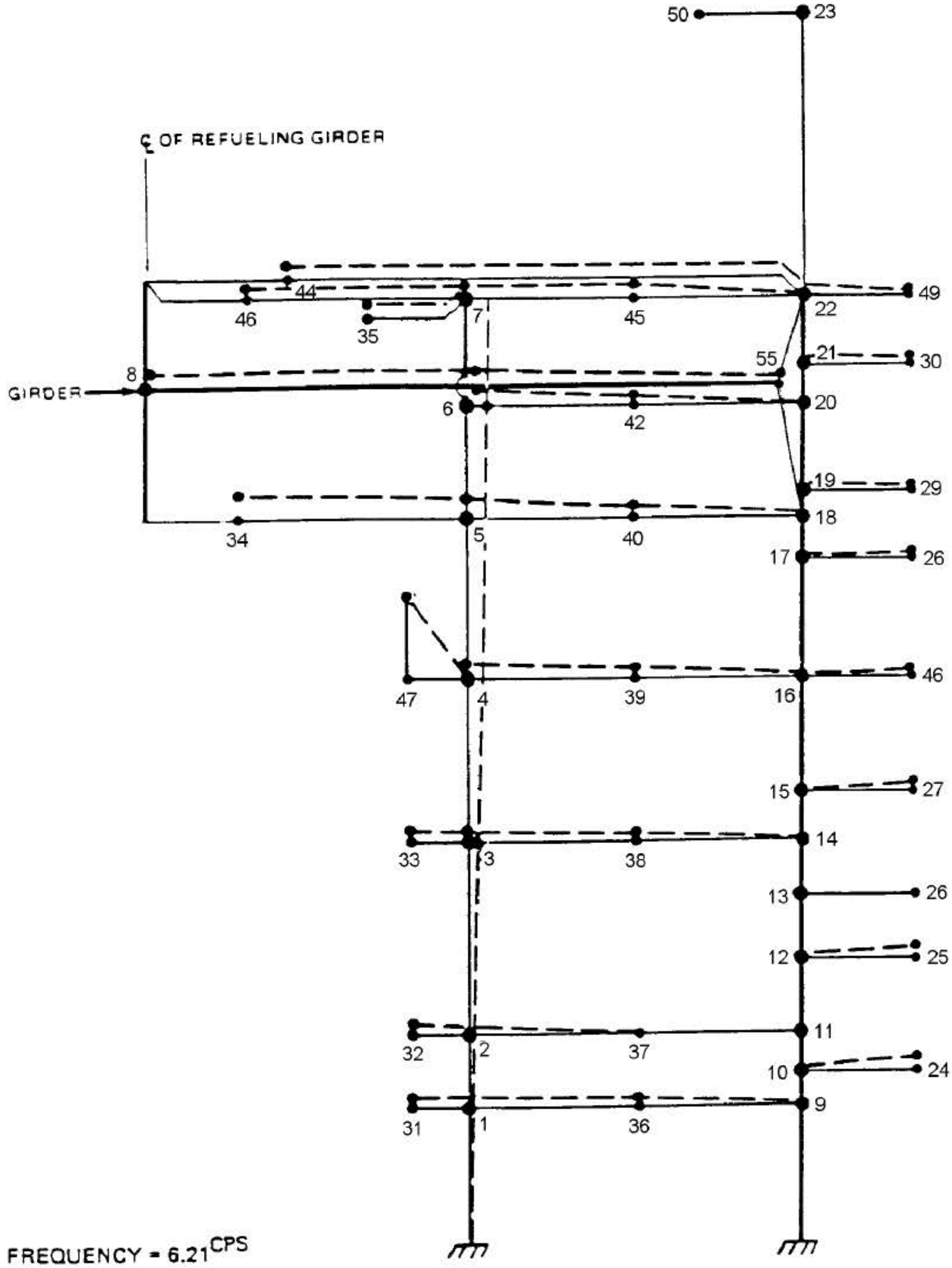
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

REACTOR AND CONTROL BUILDING
VERTICAL MODE SHAPES
MODE 1

FIGURE 3.7B-37, Rev. 55

Auto-Cad Figure Fsar 3_7B_37.dwg



NOTE:
FOR CLARITY OF ILLUSTRATION, THE DISPLACEMENT VALUES ASSOCIATED WITH THE NODES ORIENTATED ALONG THE VERTICAL MEMBERS ARE DISPLAYED GRAPHICALLY IN THE HORIZONTAL DIRECTION. THE DISPLACEMENTS ARE IN THE VERTICAL DIRECTION.

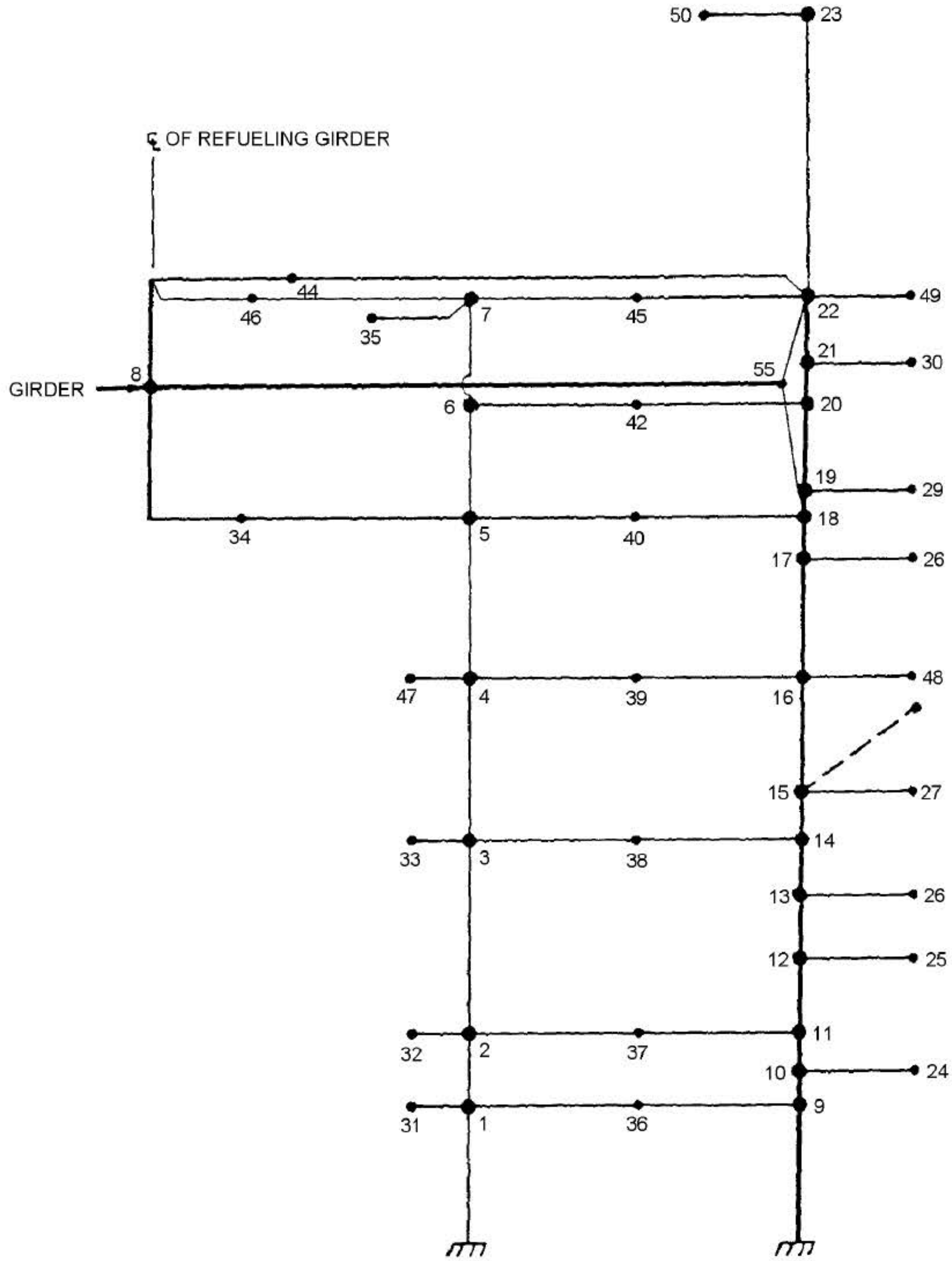
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

REACTOR AND CONTROL BUILDING
VERTICAL MODE SHAPES
MODE 2

FIGURE 3.7B-38, Rev. 55

Auto-Cad Figure Fsar 3_7B_38.dwg



FREQUENCY = 6.80 CPS

FSAR REV.65

NOTE:
FOR CLARITY OF ILLUSTRATION, THE DISPLACEMENT VALUES ASSOCIATED WITH THE NODES ORIENTATED ALONG THE VERTICAL MEMBERS ARE DISPLAYED GRAPHICALLY IN THE HORIZONTAL DIRECTION. THE DISPLACEMENTS ARE IN THE VERTICAL DIRECTION.

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
REACTOR AND CONTROL BUILDING VERTICAL MODE SHAPES MODE 3 FIGURE 3.7B-39, Rev. 55

Auto-Cad Figure Fsar 3_7B_39.dwg

THIS FIGURE HAS BEEN
RENUMBERED TO 3.7B-6

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Figure renumbered from 3.7B-3A to 3.7B-6 FIGURE 3.7B-3A, Rev. 55

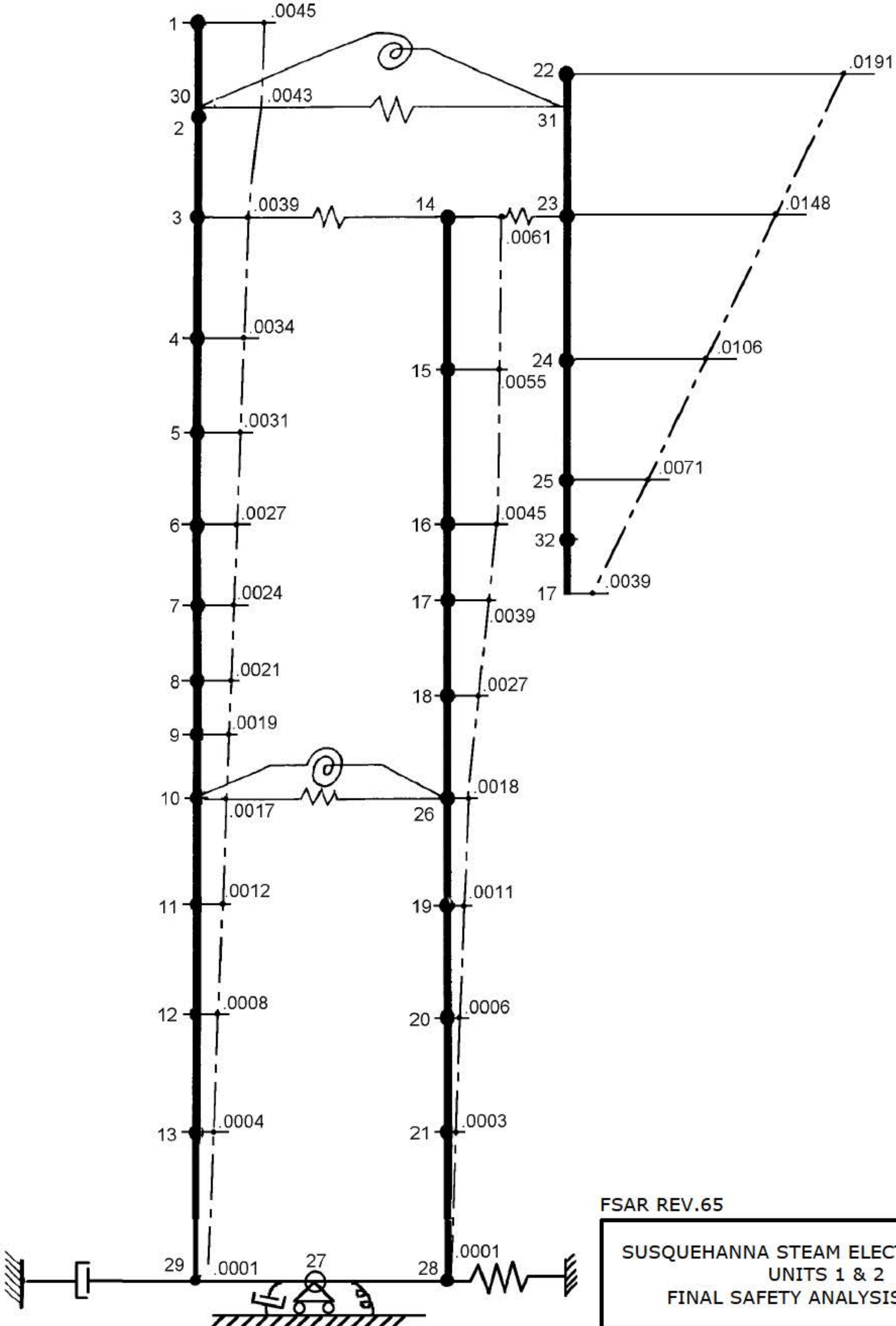
AutoCAD Figure 3_7B_3A.doc

THIS FIGURE HAS BEEN
RENUMBERED TO 3.7B-7

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Figure renumbered from 3.7B-3B to 3.7B-7 FIGURE 3.7B-3B, Rev. 55

AutoCAD Figure 3_7B_3B.doc



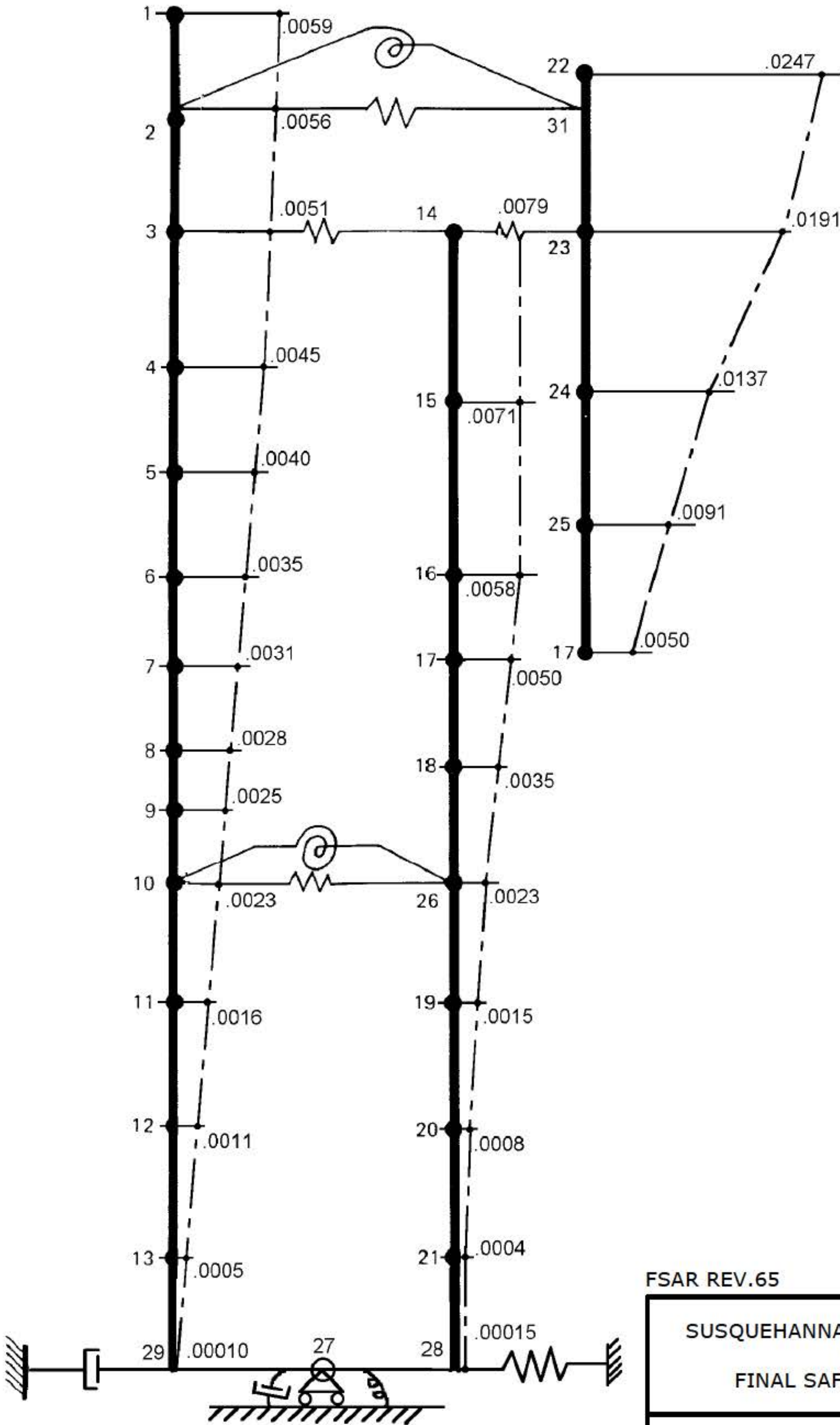
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

CONTAINMENT
HORIZONTAL REPLACEMENTS
OBE

FIGURE 3.7B-40, Rev. 55

Auto-Cad Figure Fsar 3_7B_40.dwg



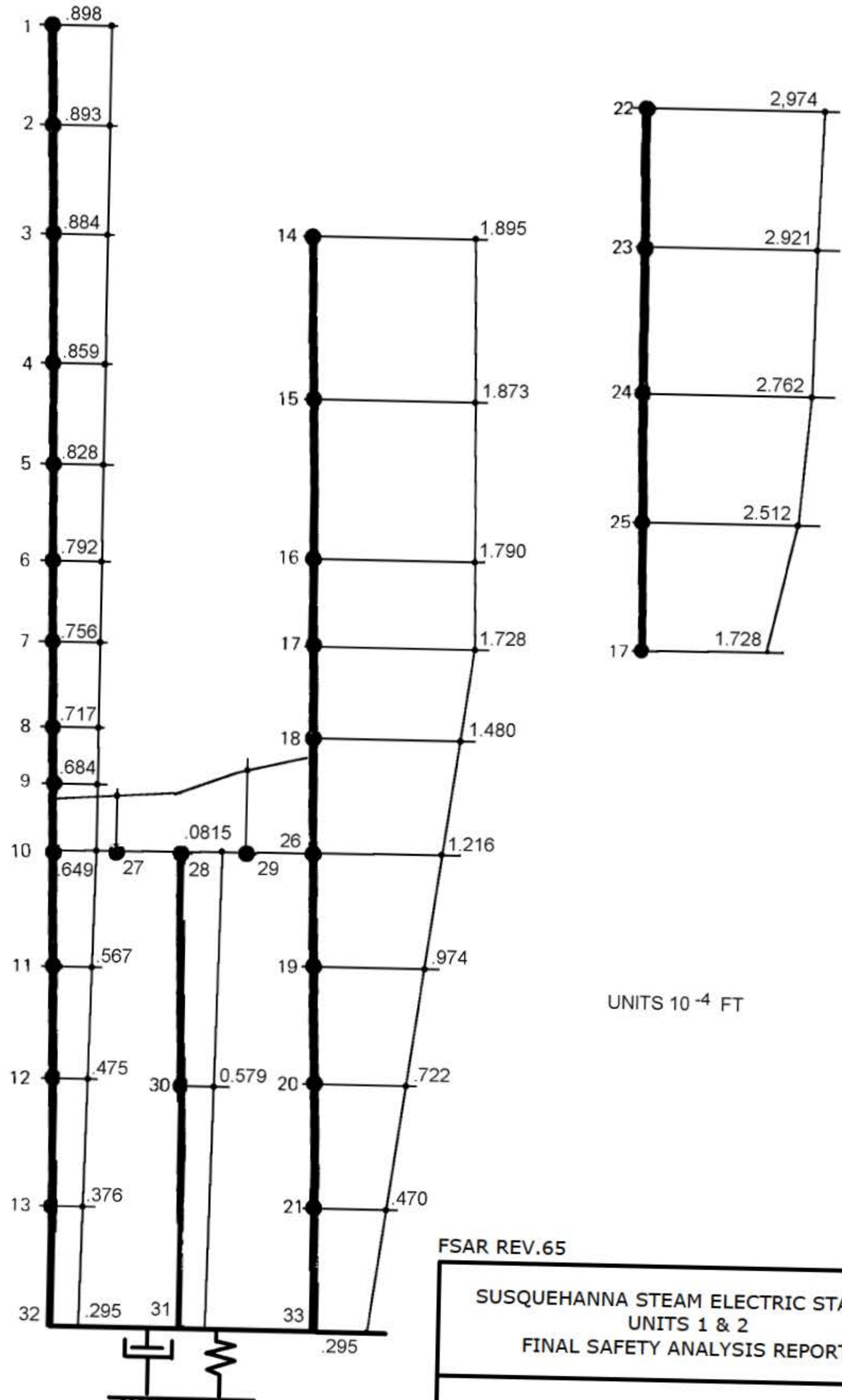
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

CONTAINMENT
HORIZONTAL DISPALCEMENTS
SSE

FIGURE 3.7B-41, Rev. 55

Auto-Cad Figure Fsar 3_7B_41.dwg

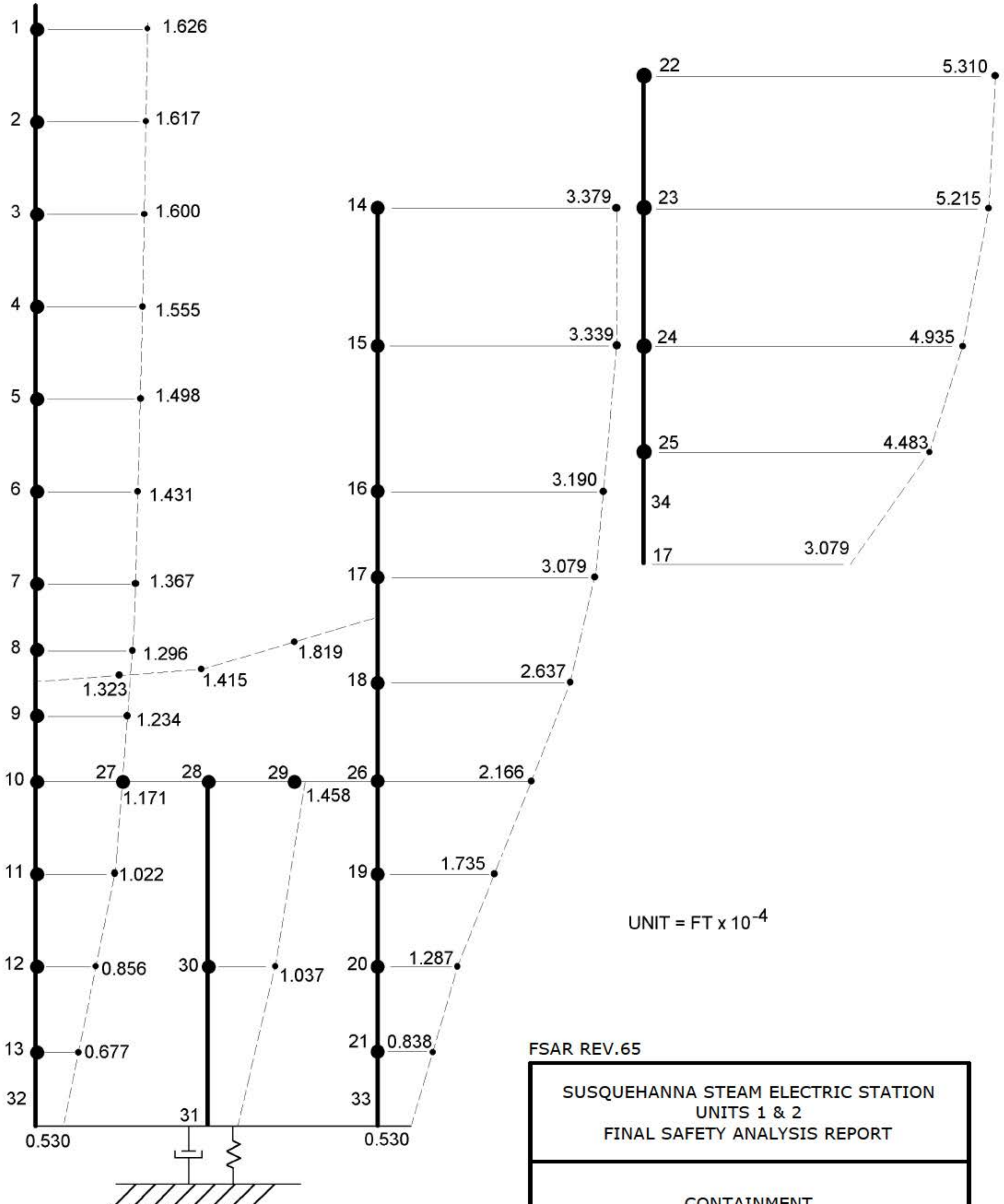


NOTE:
FOR CLARITY OF ILLUSTRATION, THE DISPLACEMENT VALUES ASSOCIATED WITH THE NODES ORIENTATED ALONG THE VERTICAL MEMBERS ARE DISPLAYED GRAPHICALLY IN THE HORIZONTAL DIRECTION. THE DISPLACEMENTS ARE IN THE VERTICAL DIRECTION.

FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
CONTAINMENT VERTICAL DISPALCEMENTS OBE
FIGURE 3.7B-42, Rev. 55

Auto-Cad Figure Fsar 3_7B_42.dwg



NOTE: For clarity of illustration, the displacement values associated with the nodes orientated along the vertical members are displayed graphically in the horizontal direction. The displacements are in the vertical direction.

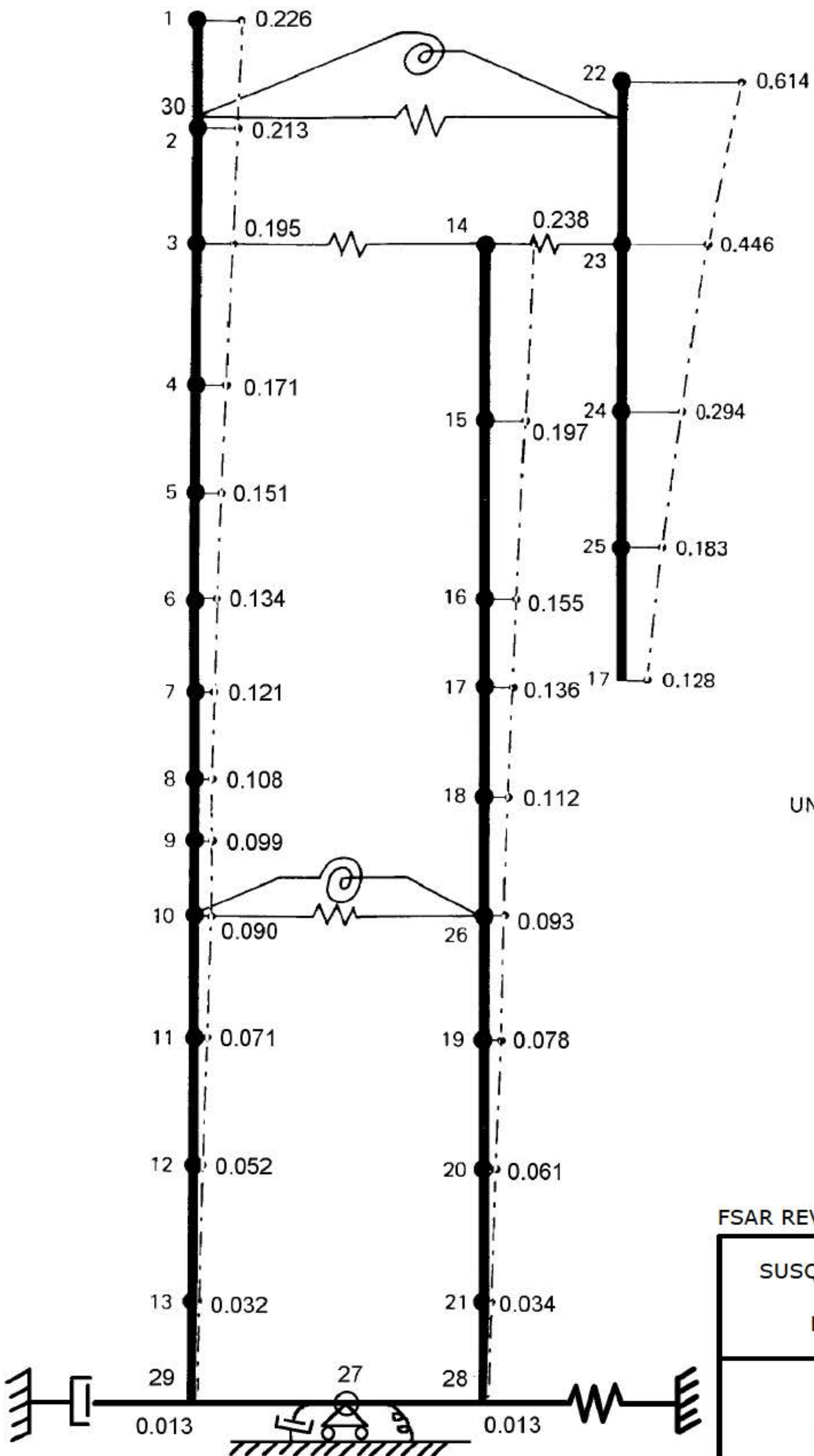
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

CONTAINMENT
VERTICAL DISPALCEMENTS
SSE

FIGURE 3.7B-43, Rev. 55

Auto-Cad Figure Fsar 3_7B_43.dwg



UNITS: G's

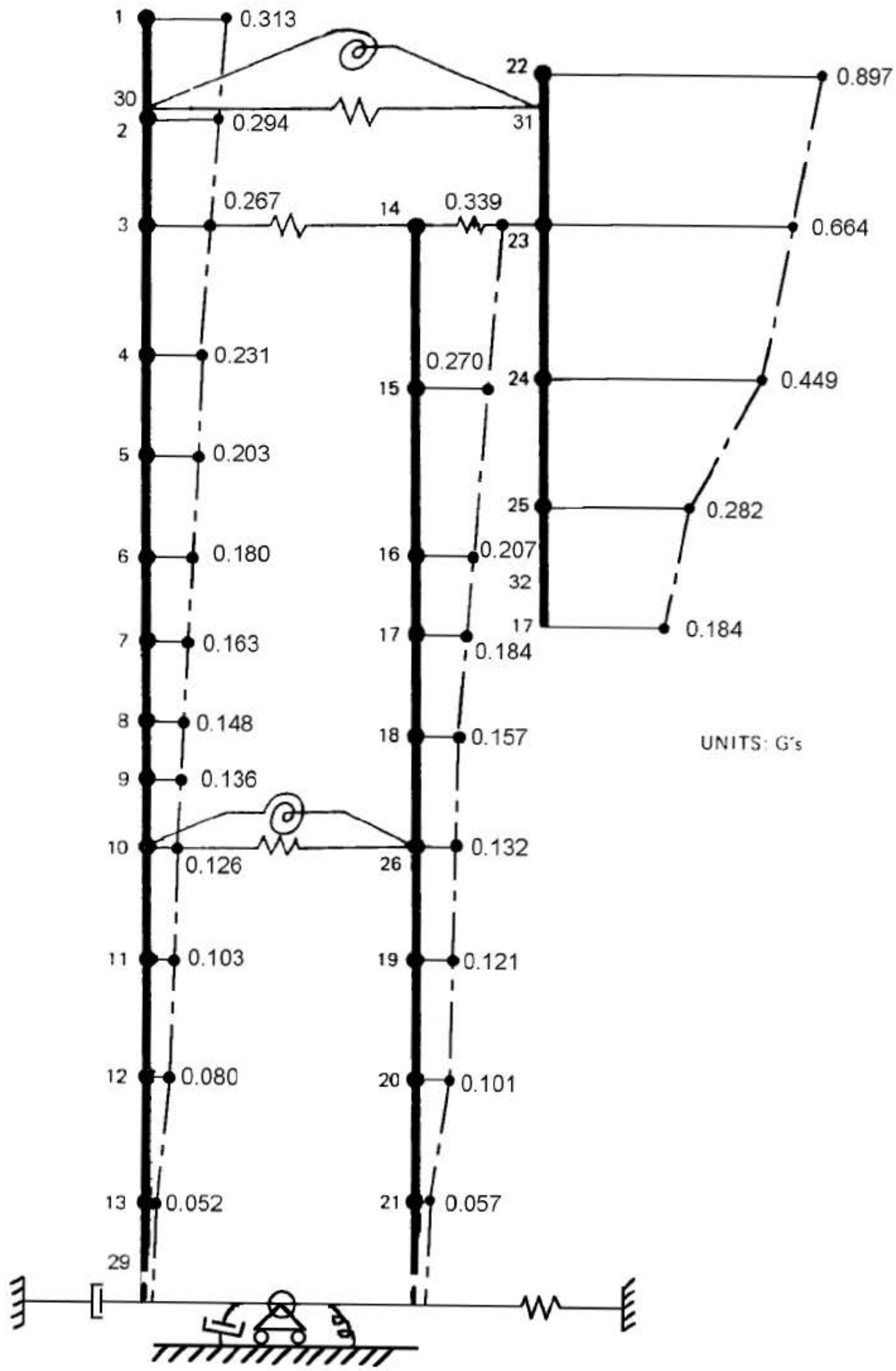
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

CONTAINMENT
HORIZONTAL ACCELERATIONS
OBE

FIGURE 3.7B-44, Rev. 55

Auto-Cad Figure Fsar 3_7B_44.dwg



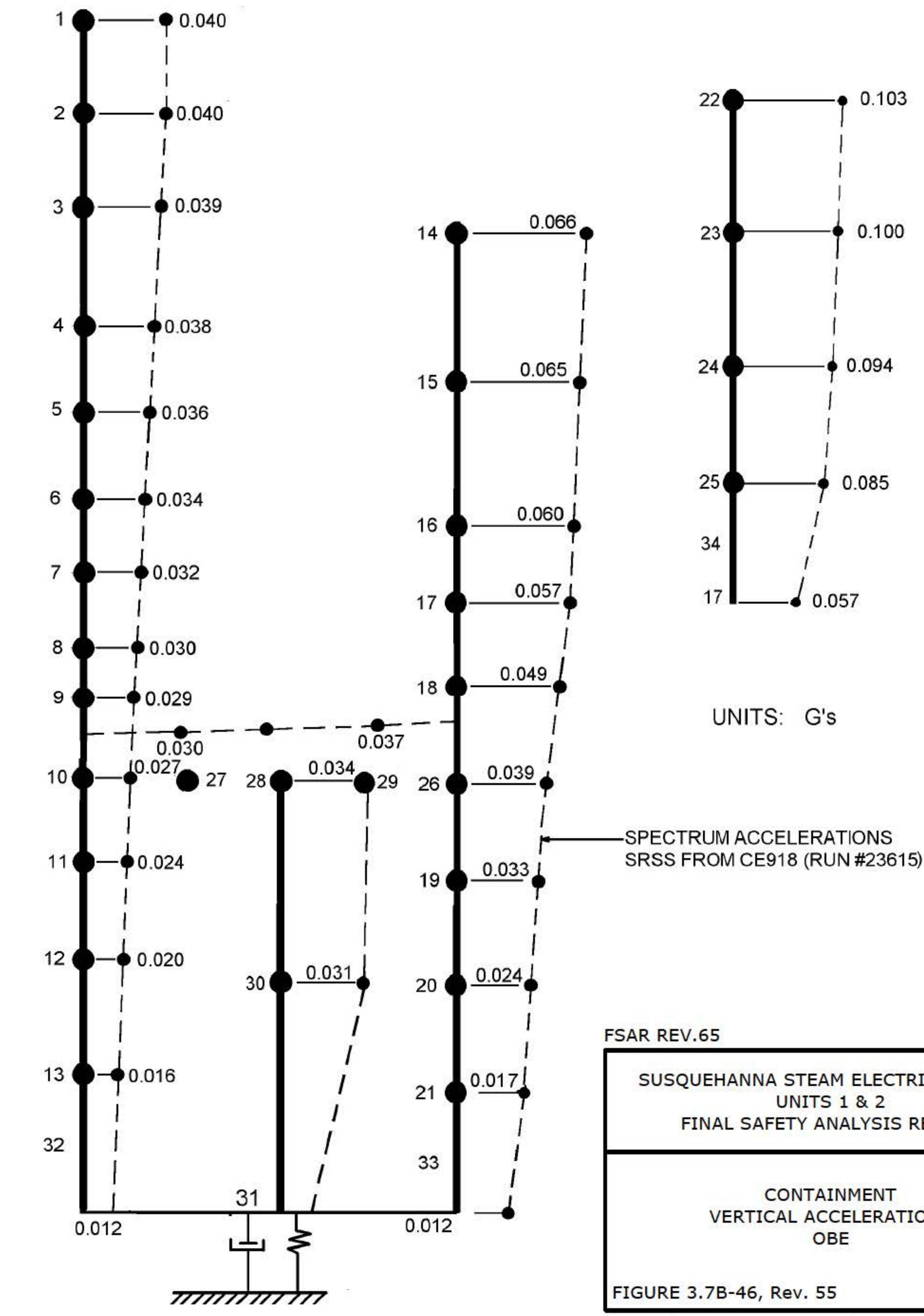
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

CONTAINMENT
HORIZONTAL ACCELERATIONS
SSE

FIGURE 3.7B-45, Rev. 55

Auto-Cad Figure Fsar 3_7B_45.dwg



FSAR REV.65

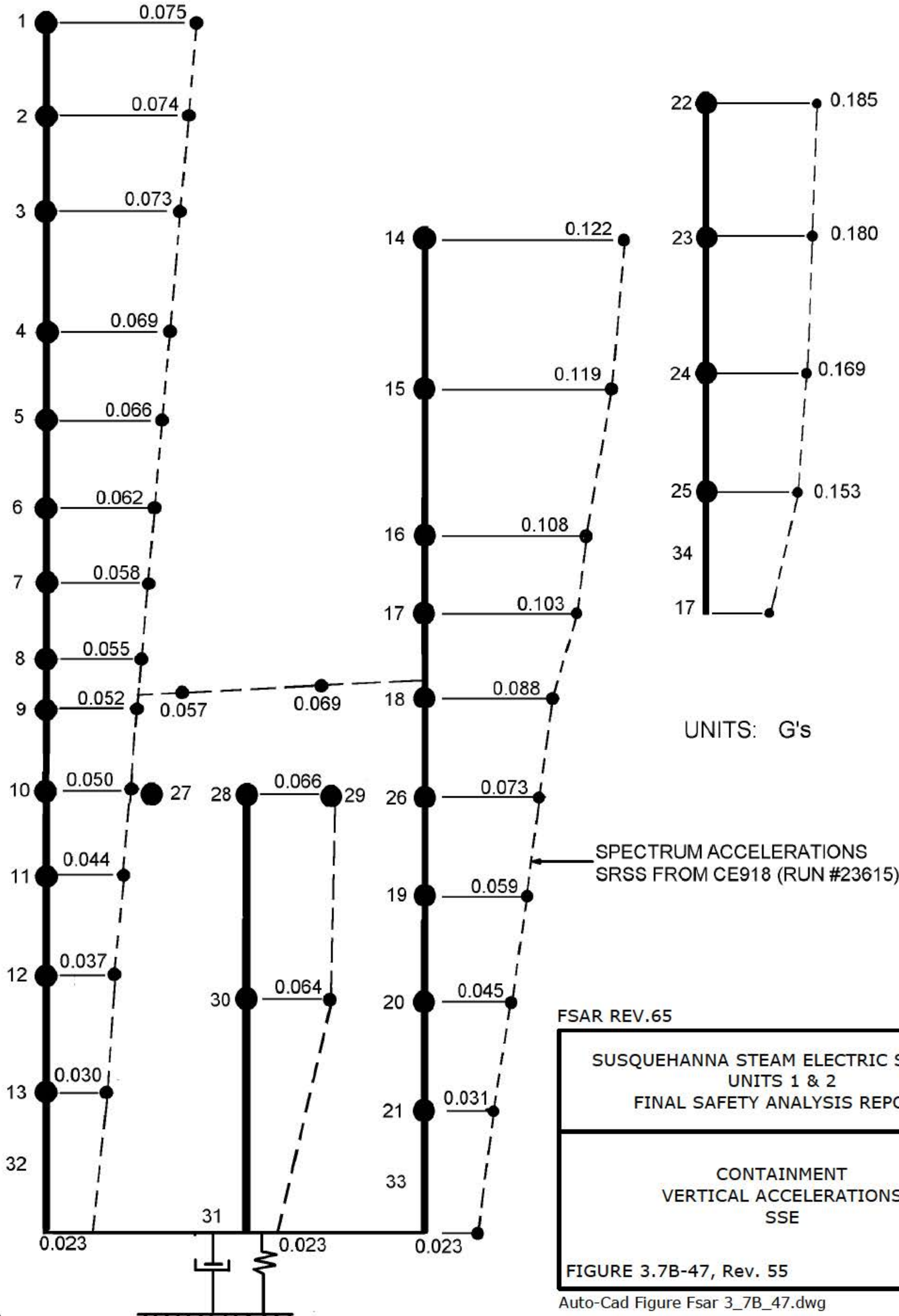
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

CONTAINMENT
VERTICAL ACCELERATIONS
OBE

FIGURE 3.7B-46, Rev. 55

Auto-Cad Figure Fsar 3_7B_46.dwg

Note;
For clarity of illustration, the acceleration values associated with the nodes oriented along the vertical members are displayed graphically in the horizontal direction. The accelerations are in the vertical direction.



FSAR REV.65

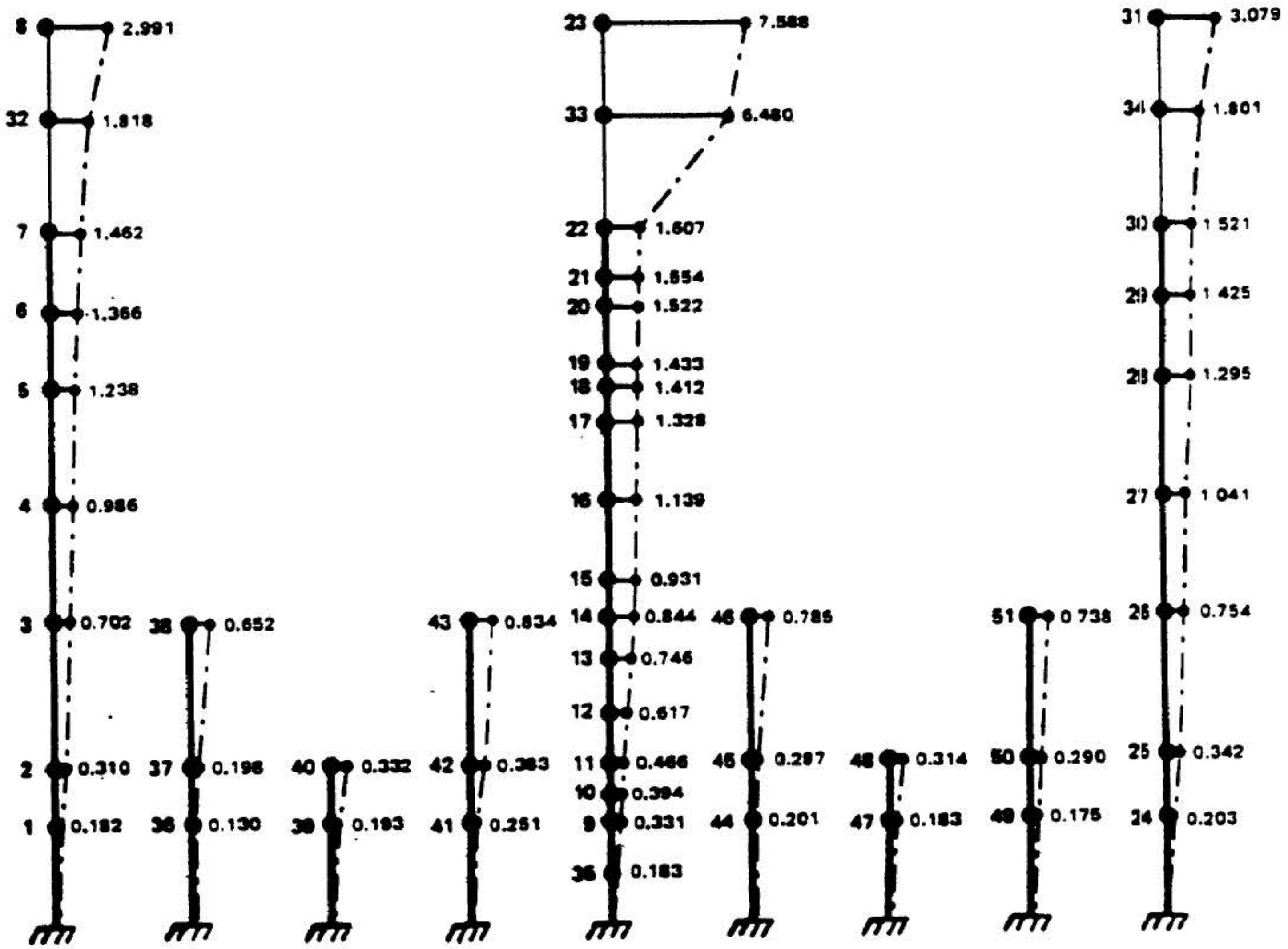
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

CONTAINMENT
VERTICAL ACCELERATIONS
SSE

FIGURE 3.7B-47, Rev. 55

Auto-Cad Figure Fsar 3_7B_47.dwg

Note;
For clarity of illustration, the acceleration values associated with the nodes oriented along the vertical members are displayed graphically in the horizontal direction. The accelerations are in the vertical direction.



NOTE: CRANES ARE LOCATED AT MASS POINTS 32 AND 33

UNITS: 10^{-2} FT

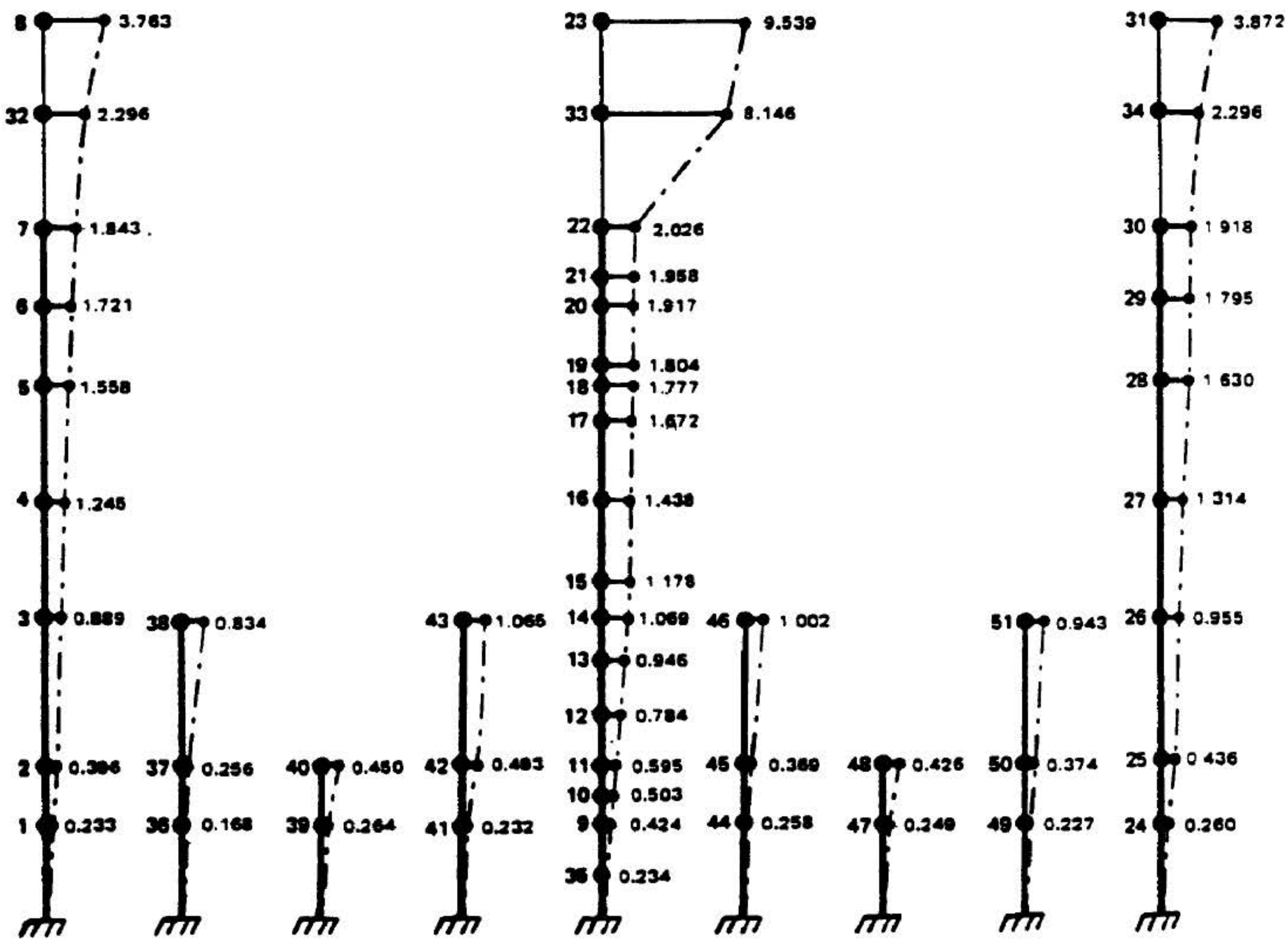
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

REACTOR AND CONTROL BUILDING
E-W DISPLACEMENTS
OBE

FIGURE 3.7B-48, Rev. 55

Auto-Cad Figure Fsar 3_7B_48.dwg



NOTE: CRANES ARE LOCATED AT MASS POINTS 32 AND 33

UNITS: 10^{-2} FT

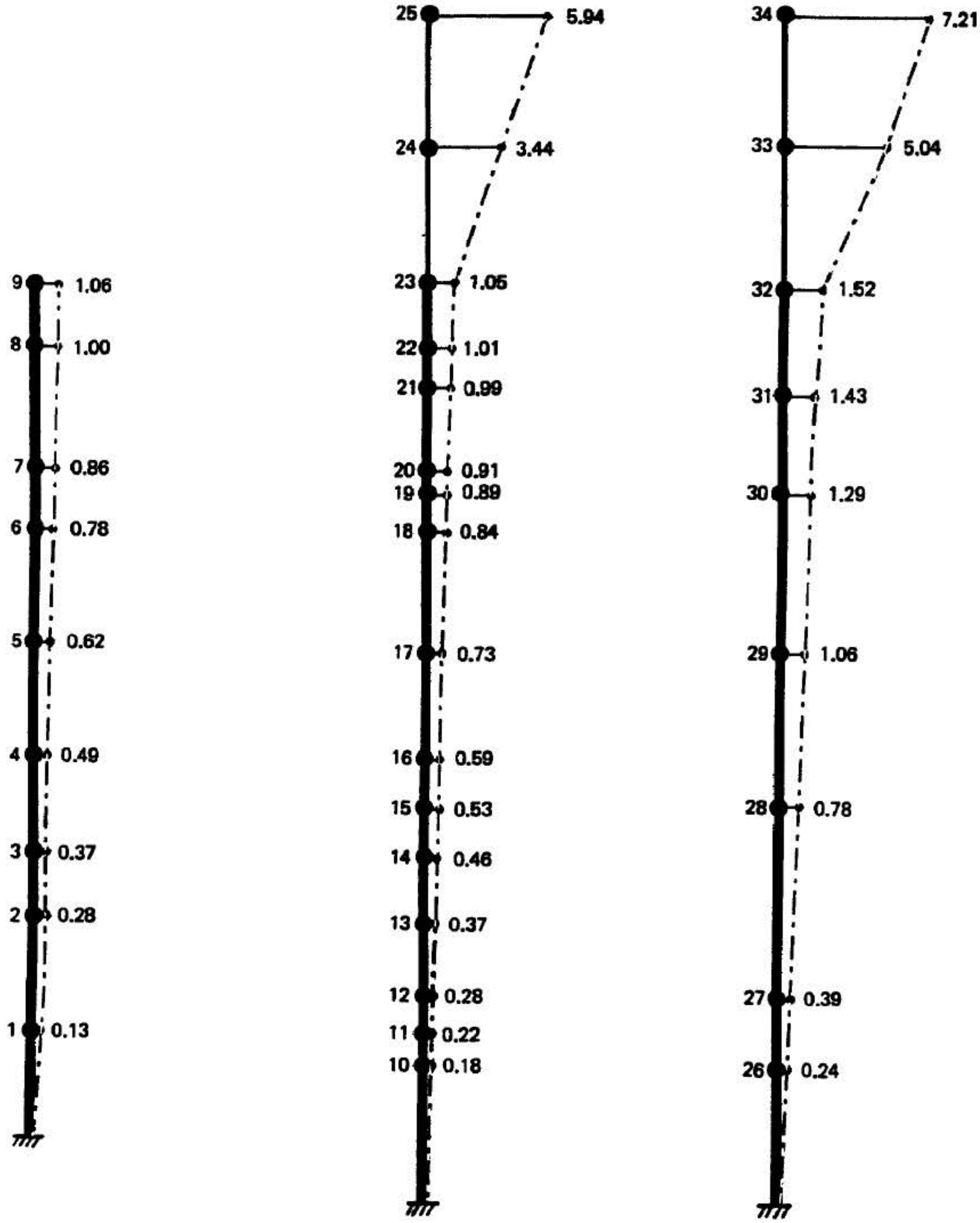
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

REACTOR AND CONTROL BUILDING
E-W DISPLACEMENTS
SSE

FIGURE 3.7B-49, Rev. 55

Auto-Cad Figure Fsar 3_7B_49.dwg



FSAR REV.65

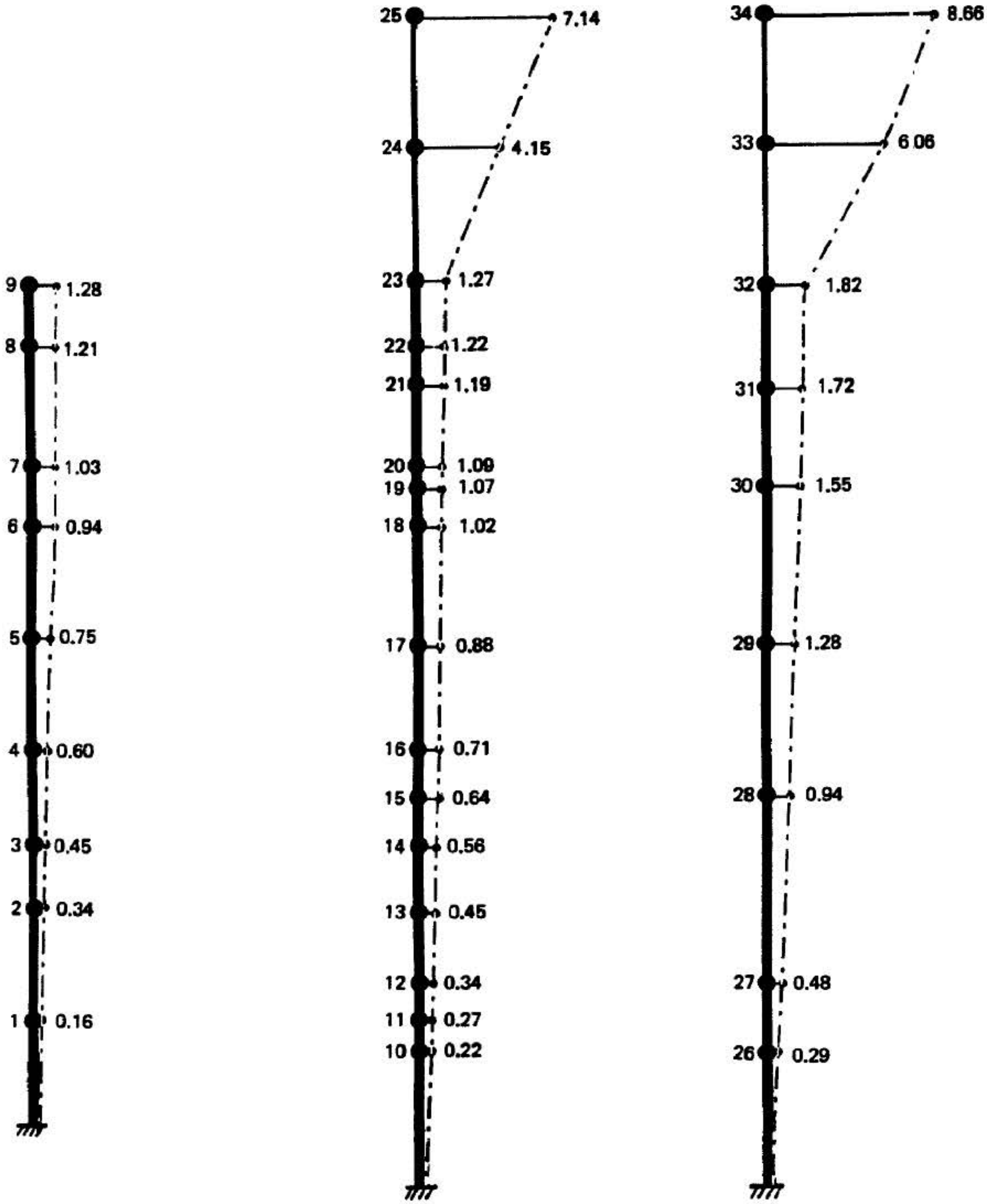
UNITS: 10^{-2} FT

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

REACTOR AND CONTROL BUILDING
N-S DISPLACEMENTS
OBE

FIGURE 3.7B-50, Rev. 55

Auto-Cad Figure Fsar 3_7B_50.dwg



FSAR REV.65

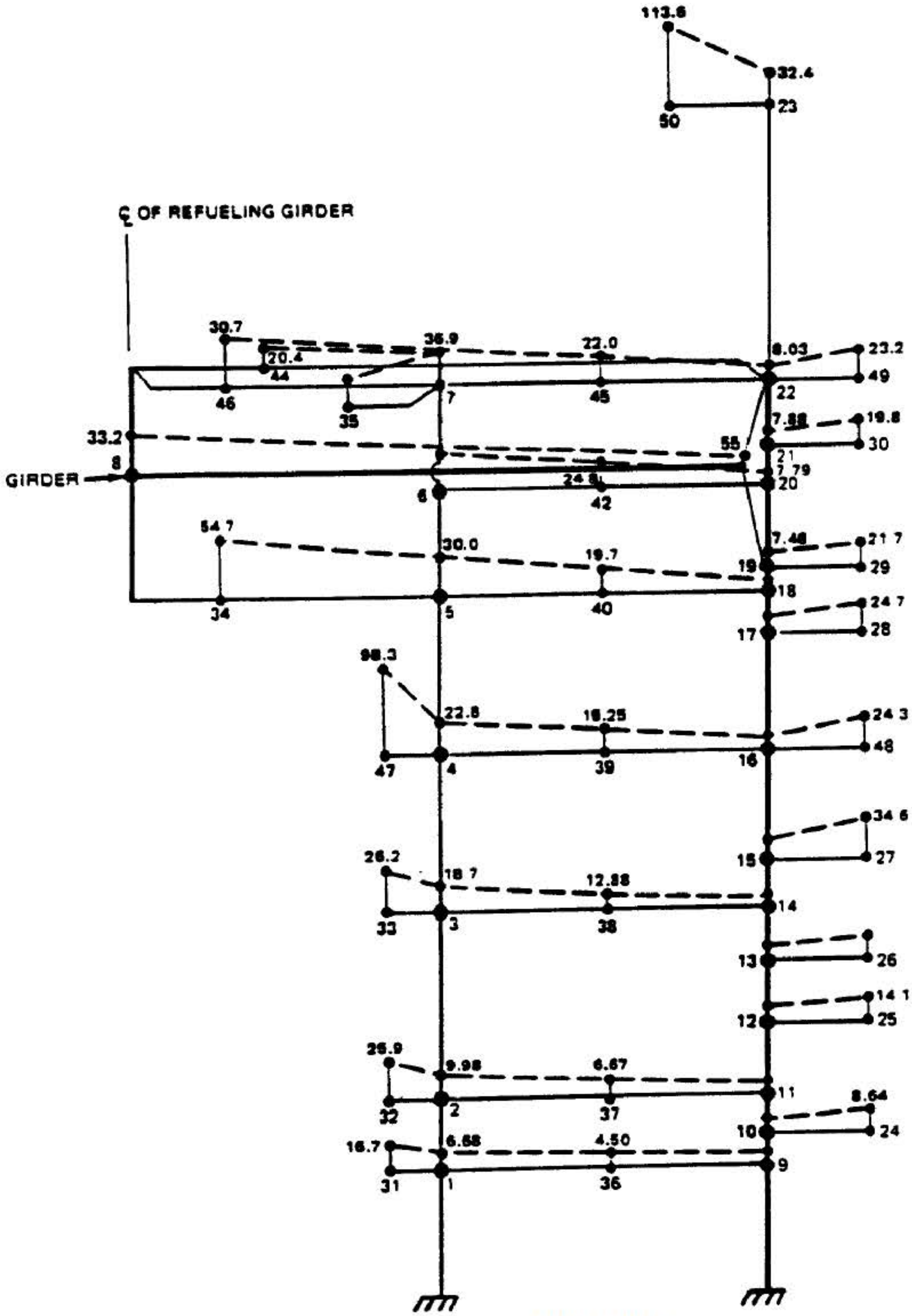
UNITS: 10^{-2} FT

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

REACTOR AND CONTROL BUILDING
N-S DISPLACEMENTS
SSE

FIGURE 3.7B-51, Rev. 55

Auto-Cad Figure Fsar 3_7B_51.dwg



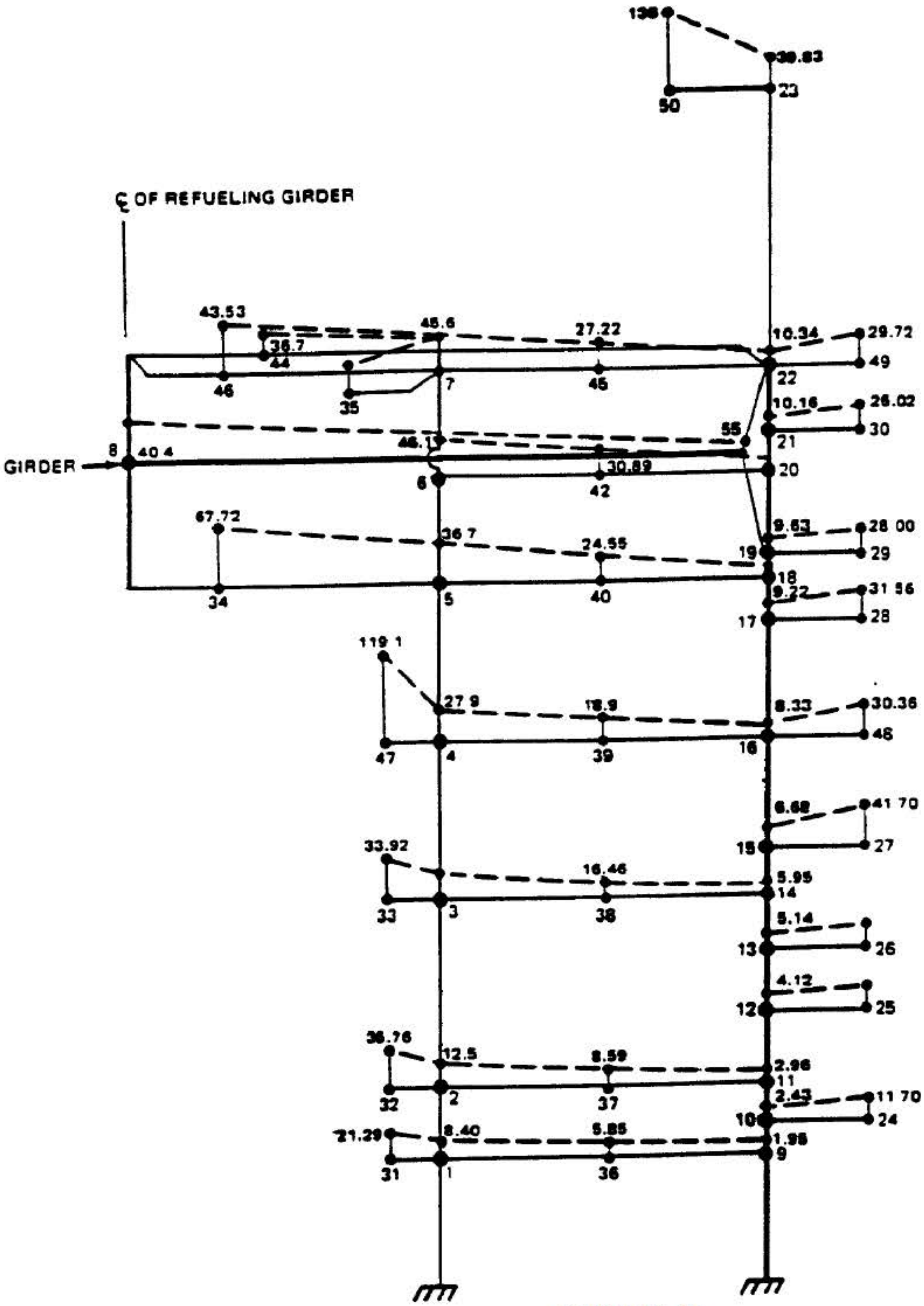
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

REACTOR AND CONTROL BUILDING
VERTICAL DISPLACEMENTS
OBE

FIGURE 3.7B-52, Rev. 55

Auto-Cad Figure Fsar 3_7B_52.dwg



UNITS: 10^{-4} FT

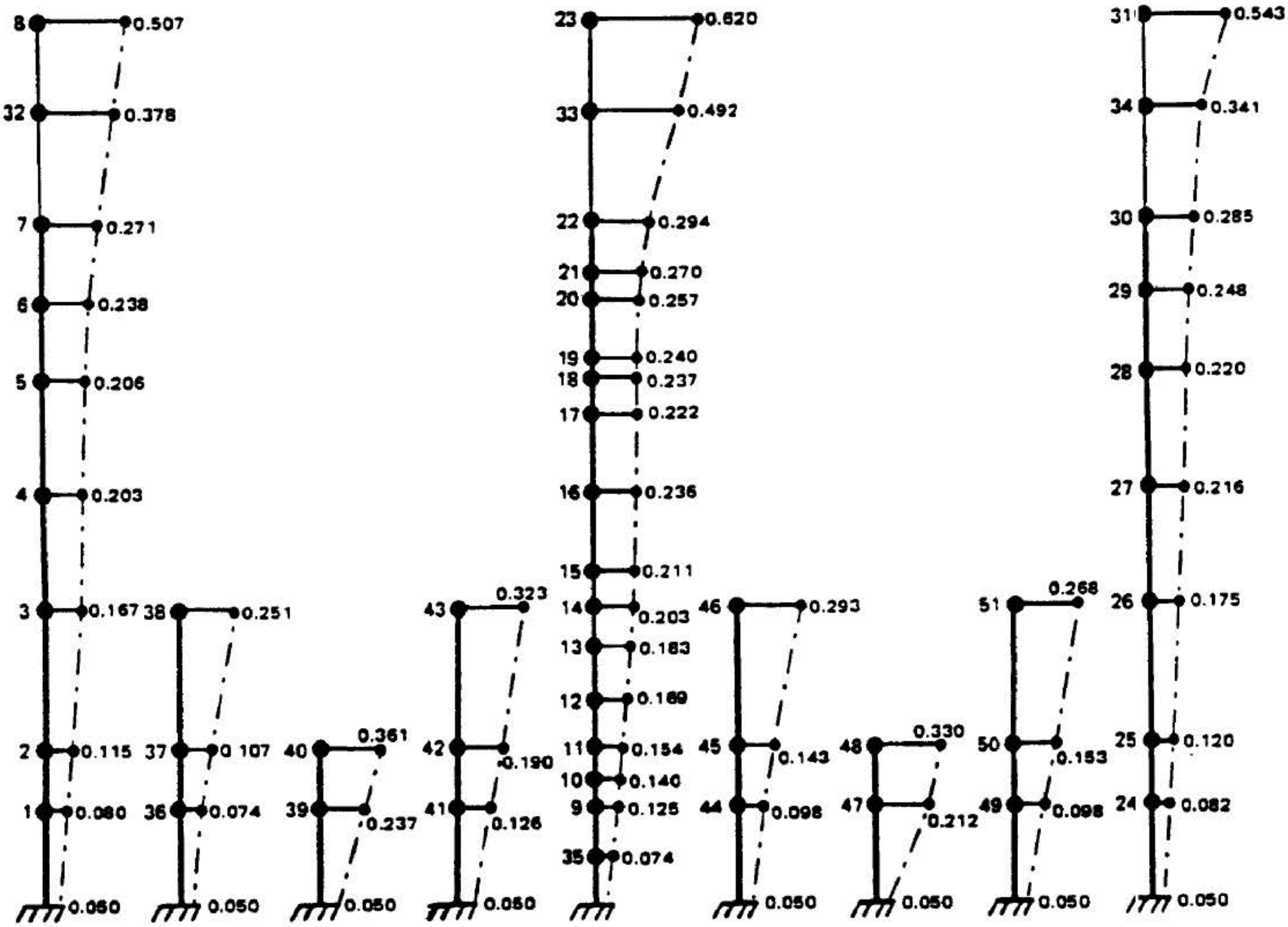
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

REACTOR AND CONTROL BUILDING
VERTICAL DISPLACEMENTS
SSE

FIGURE 3.7B-53, Rev. 55

Auto-Cad Figure Fsar 3_7B_53.dwg



NOTE: CRANES ARE LOCATED AT MASS POINTS 32 AND 33

UNITS: G's

ϕBE RESPONSES

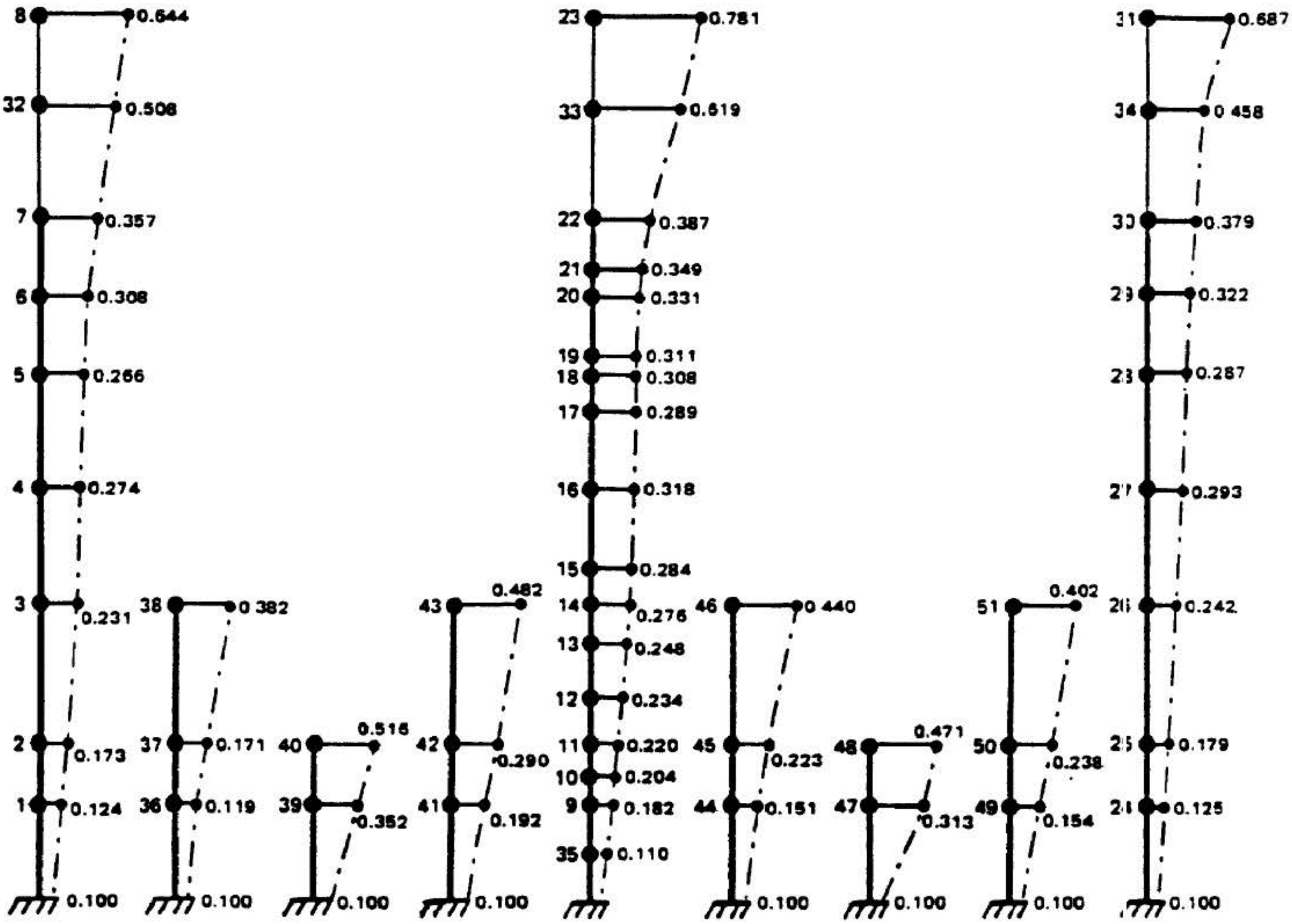
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

REACTOR AND CONTROL BUILDING
E-W ACCELERATIONS
OBE

FIGURE 3.7B-54, Rev. 55

Auto-Cad Figure Fsar 3_7B_54.dwg



NOTE: CRANES ARE LOCATED AT
MASS POINTS 32 AND 33

UNITS = G's

SSE RESPONSES

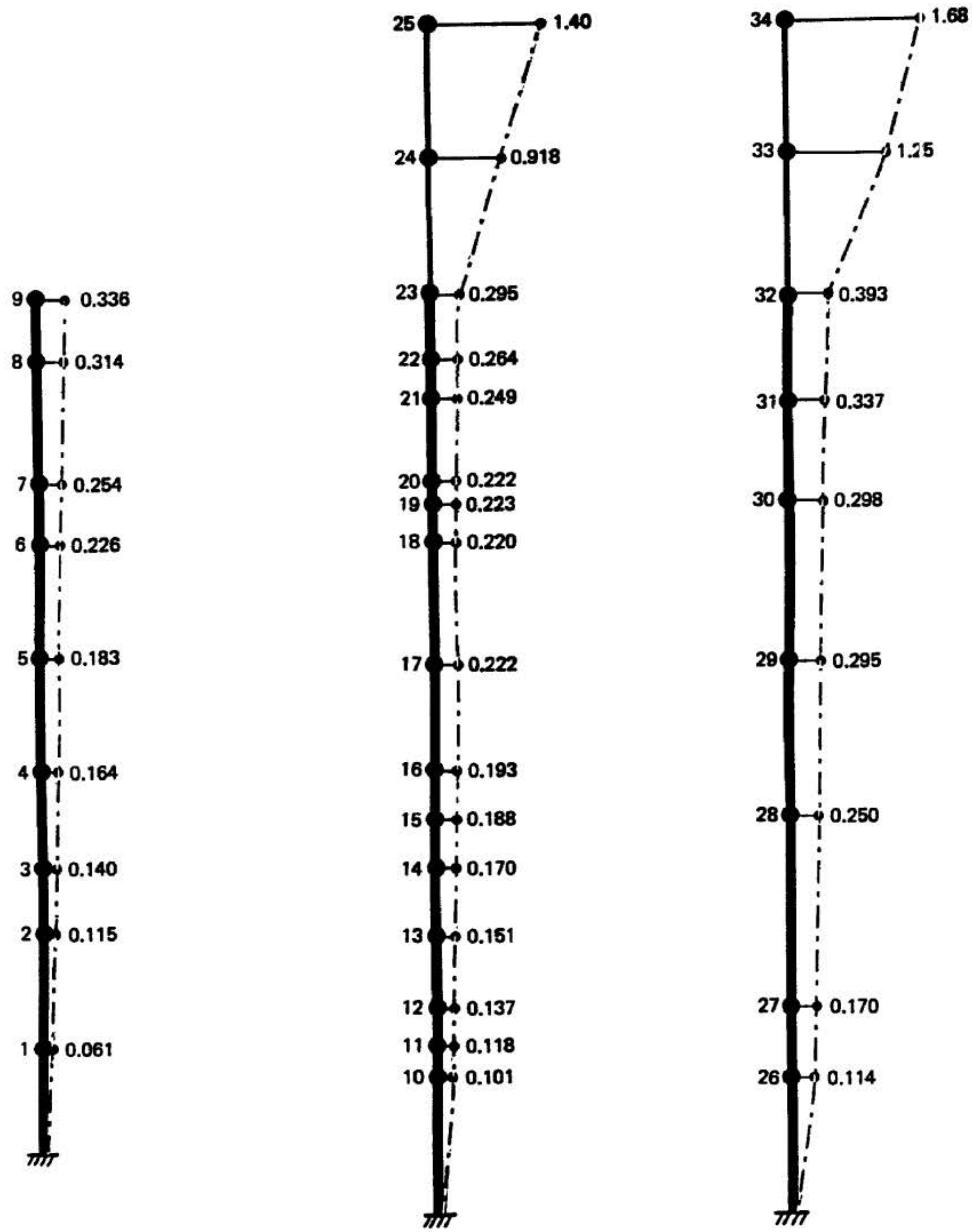
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

REACTOR AND CONTROL BUILDING
E-W ACCELERATIONS
SSE

FIGURE 3.7B-55, Rev. 55

Auto-Cad Figure Fsar 3_7B_55.dwg



UNITS: G's

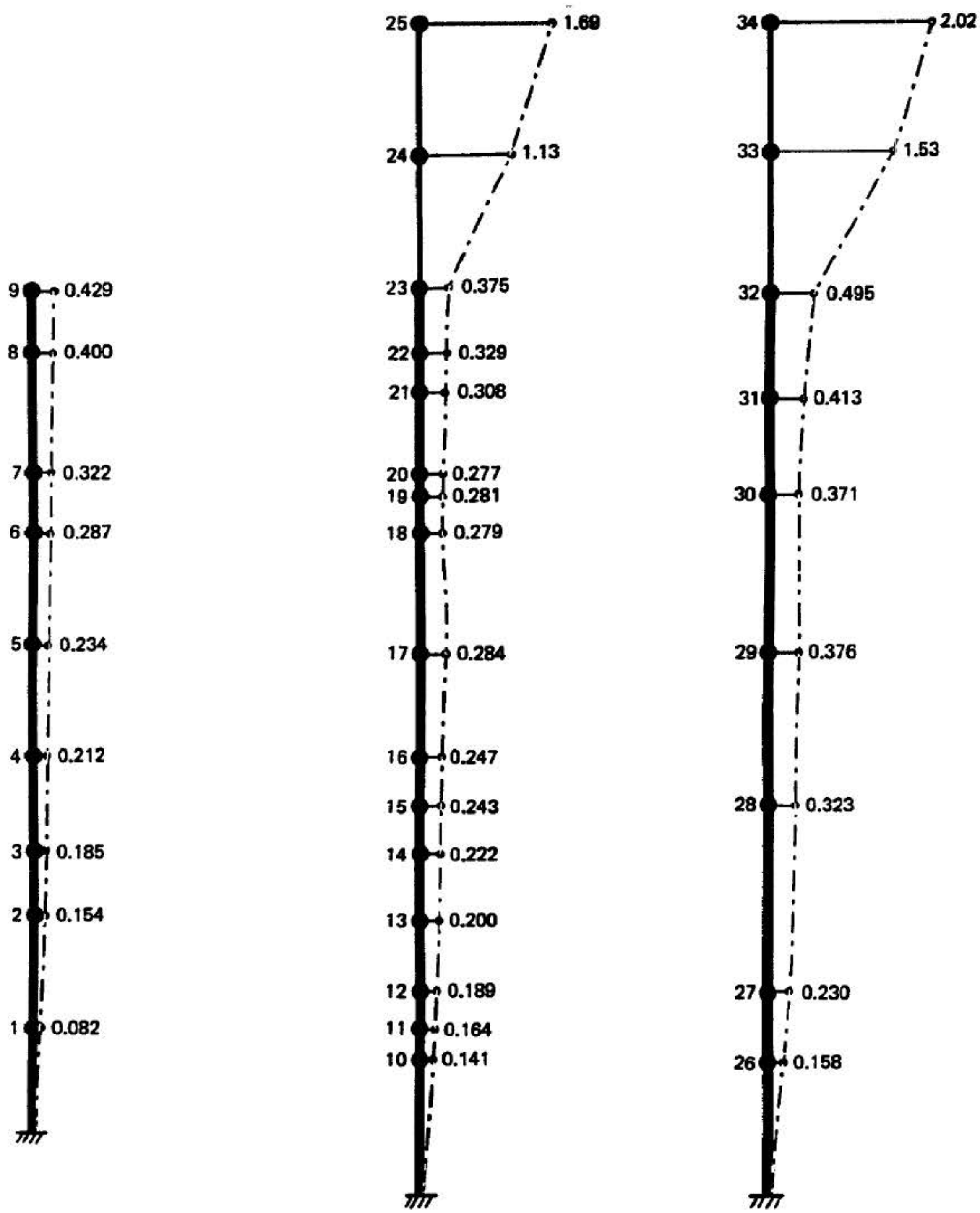
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

REACTOR AND CONTROL BUILDING
N-S ACCELERATIONS
OBE

FIGURE 3.7B-56, Rev. 55

Auto-Cad Figure Fsar 3_7B_56.dwg



UNITS: G's

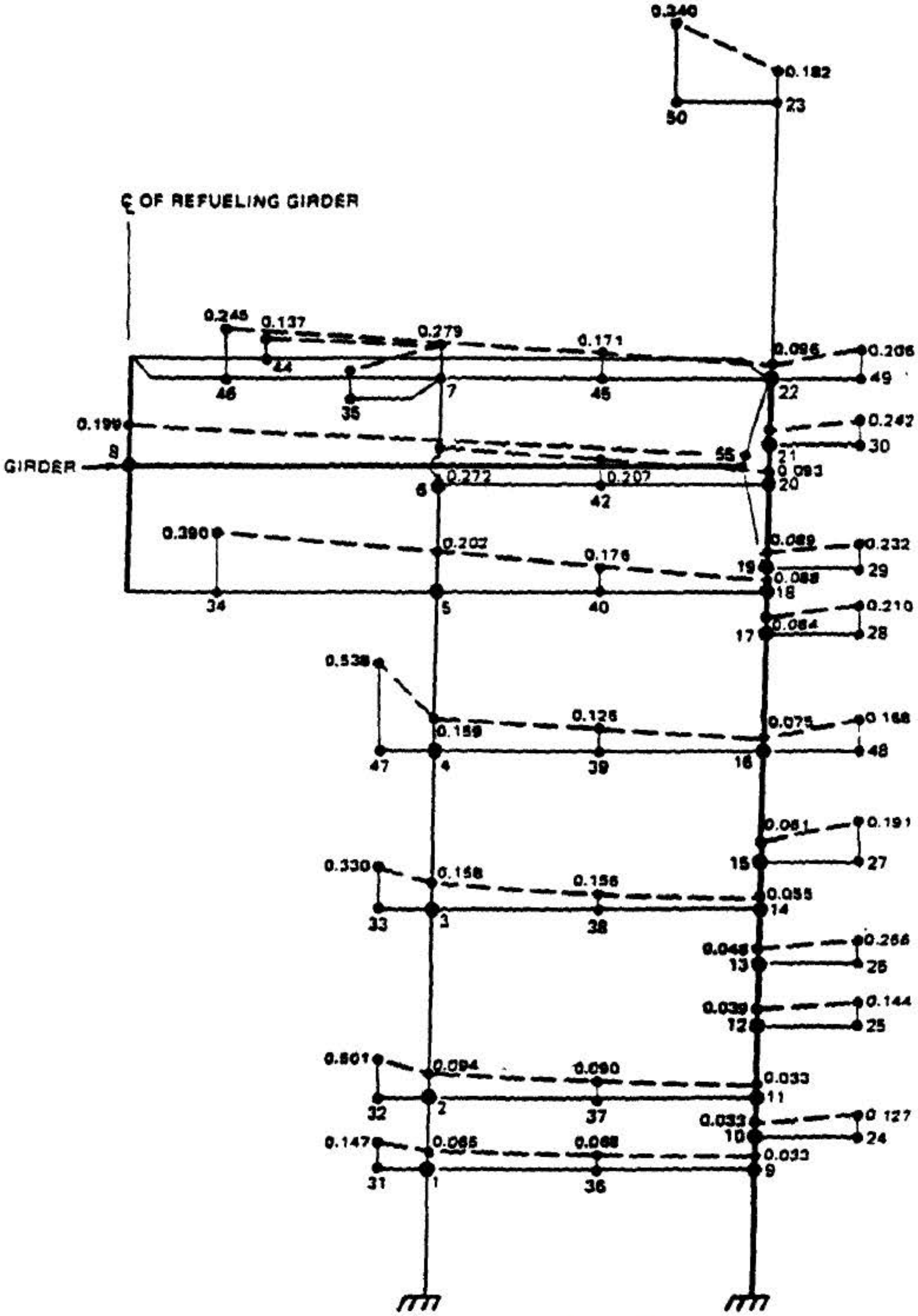
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

REACTOR AND CONTROL BUILDING
N-S ACCELERATIONS
SSE

FIGURE 3.7B-57, Rev. 55

Auto-Cad Figure Fsar 3_7B_57.dwg



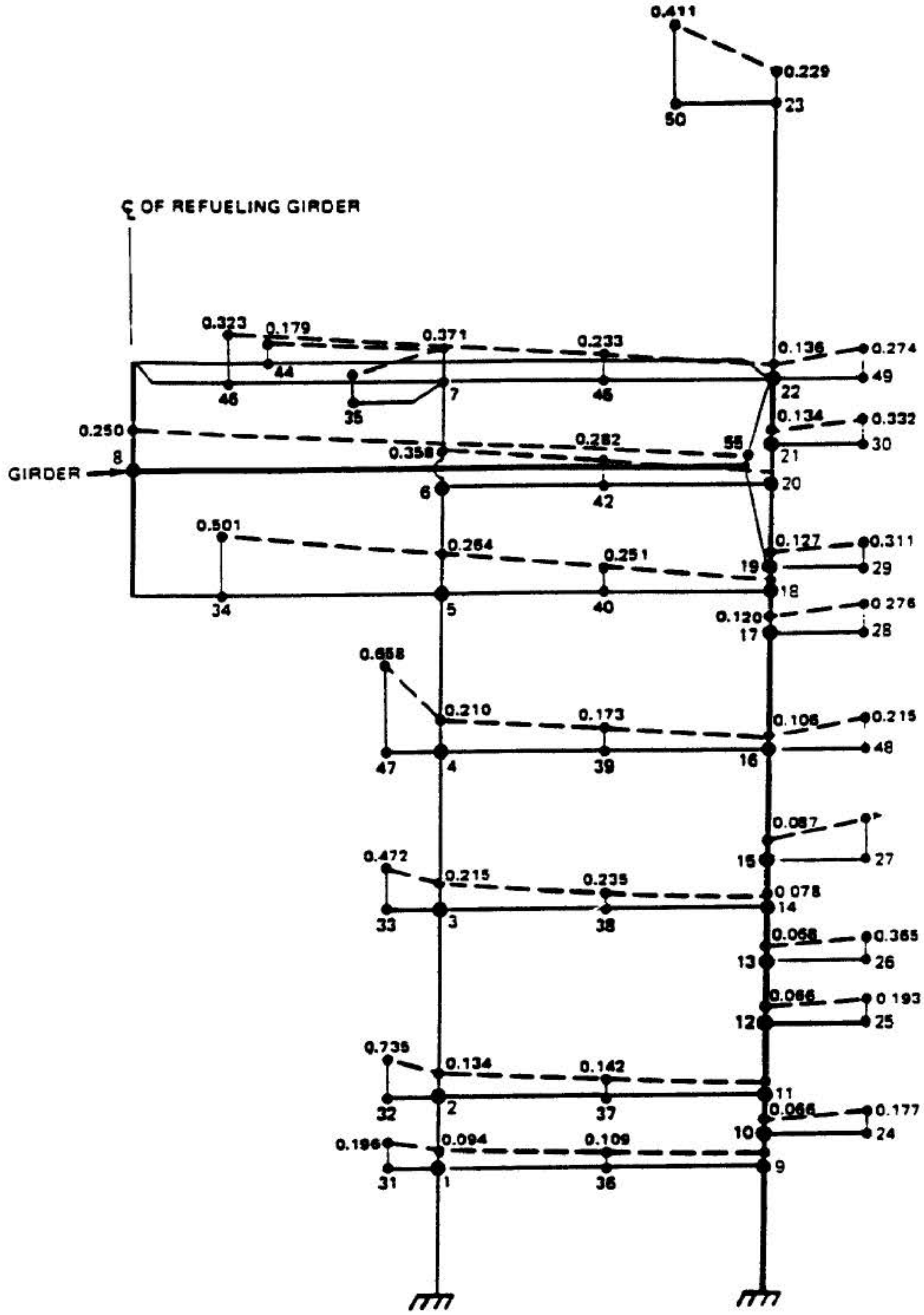
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

REACTOR AND CONTROL BUILDING
VERTICAL ACCELERATION
OBE

FIGURE 3.7B-58, Rev. 55

Auto-Cad Figure Fsar 3_7B_58.dwg



UNITS: G's

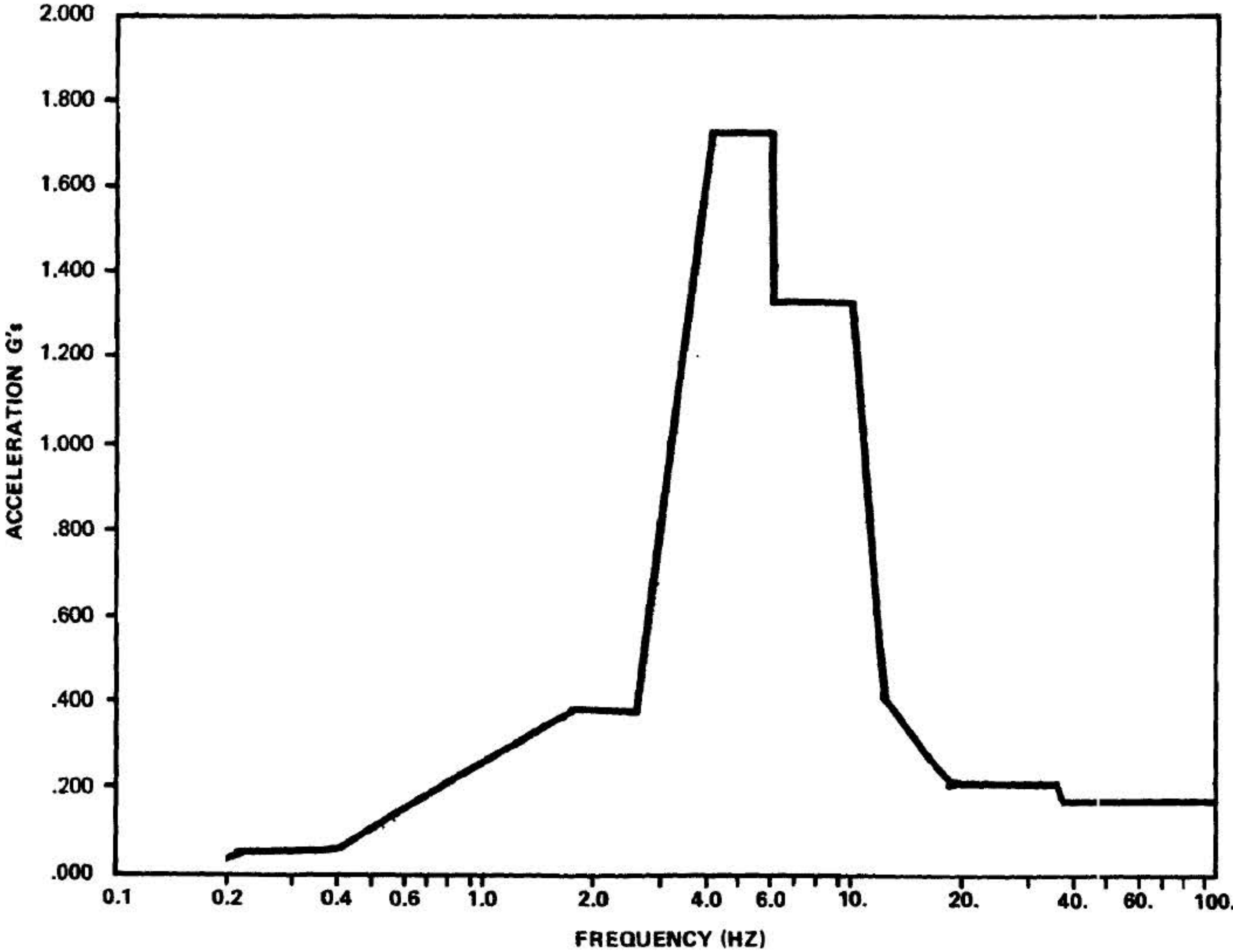
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

REACTOR AND CONTROL BUILDING
VERTICAL ACCELERATIONS
SSE

FIGURE 3.7B-59, Rev. 55

Auto-Cad Figure Fsar 3_7B_59.dwg



LOCATION: RPV PEDESTAL
DIRECTION: HORIZONTAL
EARTHQUAKE: OBE
DAMPING: 0.005

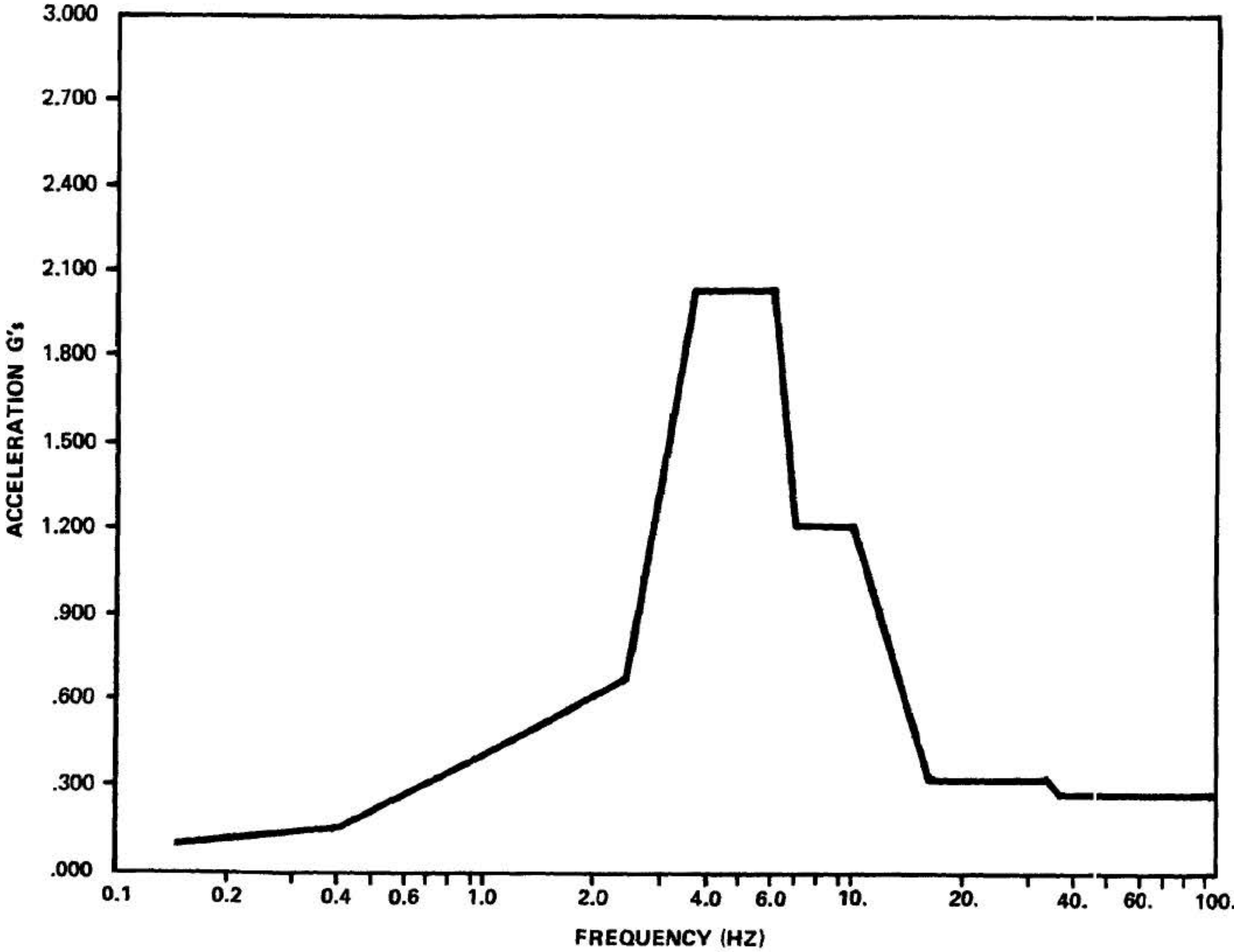
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

RESPONSE SPECTRUM
AT RPV PEDESTAL
HORIZONTAL OBE

FIGURE 3.7B-60, Rev. 55

Auto-Cad Figure Fsar 3_7B_60.dwg



LOCATION: RPV PEDESTAL
DIRECTION: HORIZONTAL
EARTHQUAKE: SSE
DAMPING: 0.010

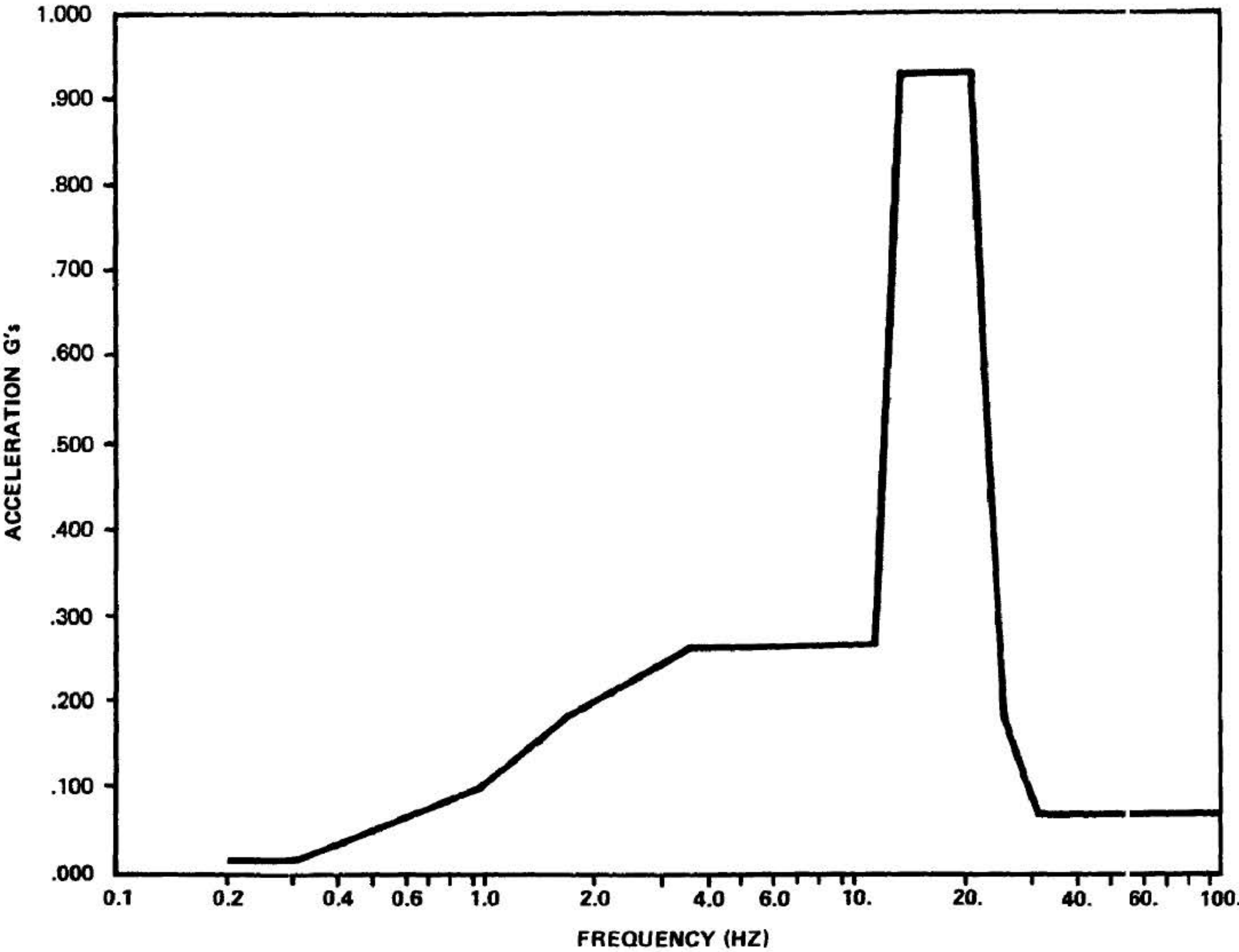
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

RESPONSE SPECTRUM
AT RPV PEDESTAL
HORIZONTAL SSE

FIGURE 3.7B-61, Rev. 55

Auto-Cad Figure Fsar 3_7B_61.dwg



LOCATION: RPV PEDESTAL
DIRECTION: VERTICAL
EARTHQUAKE: OBE
DAMPING: 0.005

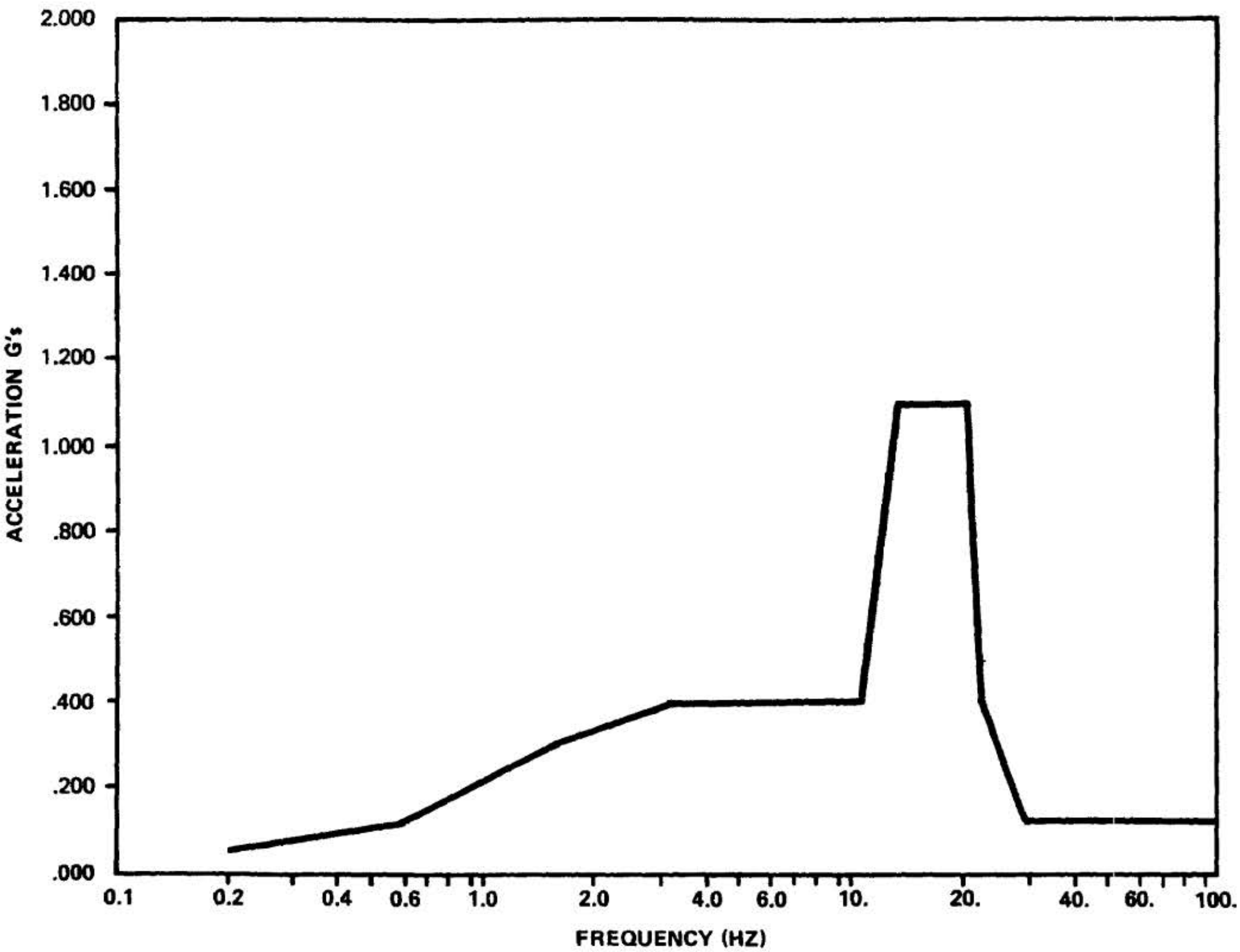
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

RESPONSE SPECTRUM
AT RPV PEDESTAL
VERTICAL OBE

FIGURE 3.7B-62, Rev. 55

Auto-Cad Figure Fsar 3_7B_62.dwg



LOCATION: RPV PEDESTAL
DIRECTION: VERTICAL
EARTHQUAKE: SSE
DAMPING: 0.010

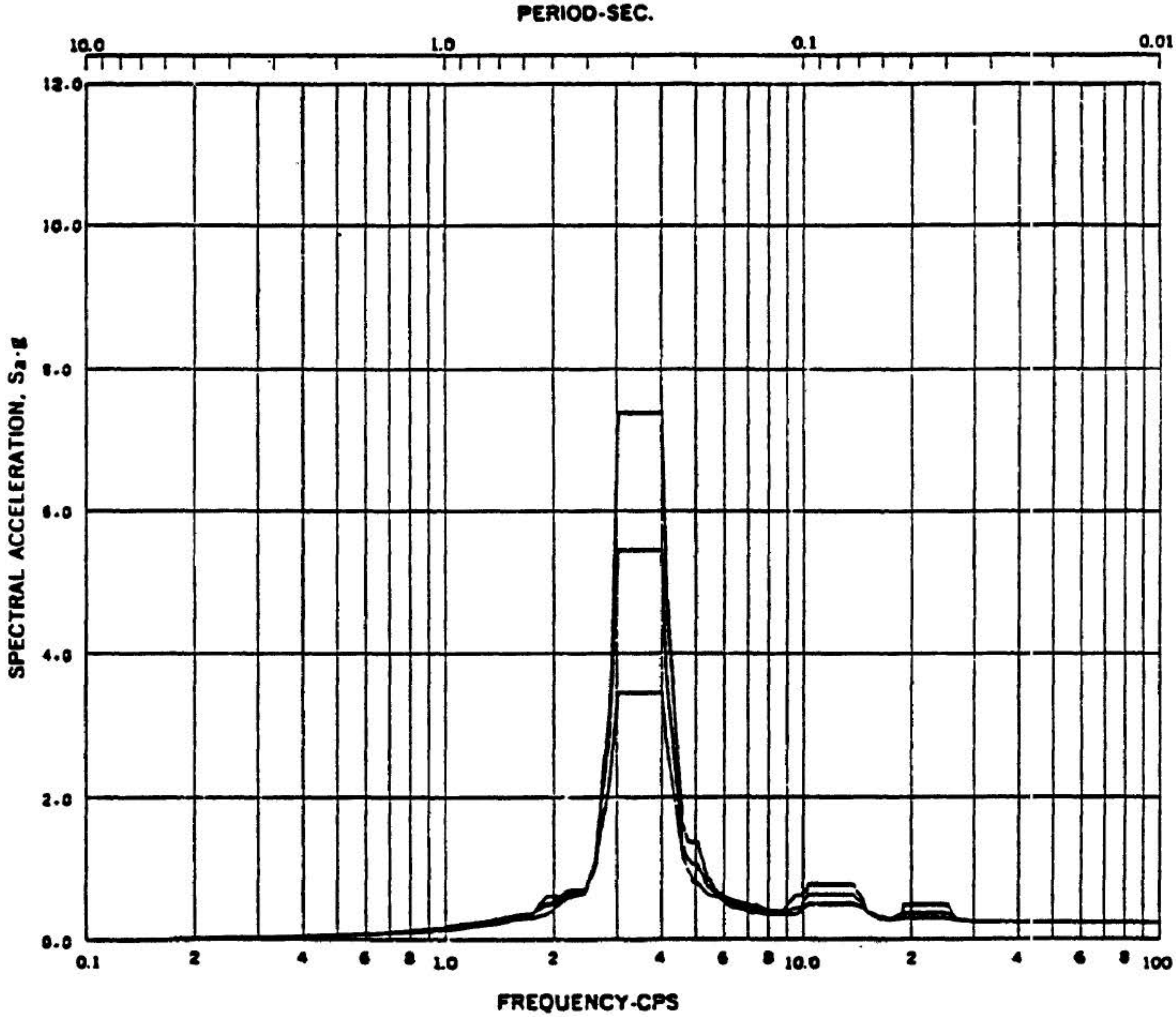
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

RESPONSE SPECTRUM
AT RPV PEDESTAL
VERTICAL SSE

FIGURE 3.7B-63, Rev. 55

Auto-Cad Figure Fsar 3_7B_63.dwg



ACCELERATION SPECTRA FOR REACTOR/CONTROL BLDG

LOAD CASE SUSQUEHANNA OBE

NODE 22, DIRECTION EW, ELEV 818'-1

DAMPING 0.005.0.010.0.02

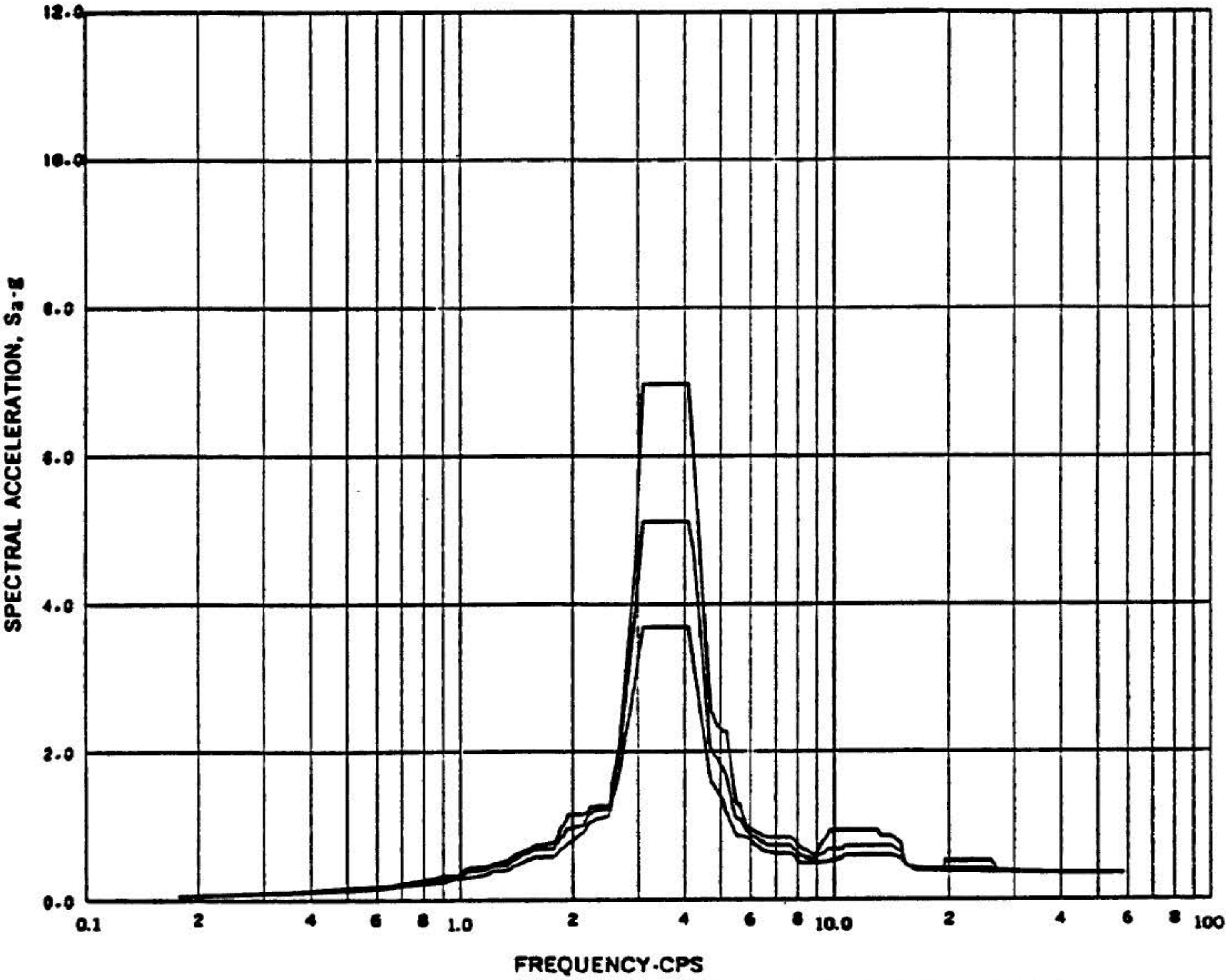
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

RESPONSE SPECTRUM
AT REFUELING AREA
E-W OBE

FIGURE 3.7B-64, Rev. 55

Auto-Cad Figure Fsar 3_7B_64.dwg



ACCELERATION SPECTRA FOR REACTOR/CONTROL BLDG

LOAD CASE SUSQUEHANNA SSE

NODE 22, DIRECTION EW, ELEV 818'-1

DAMPING 0.005.0.010.0.02

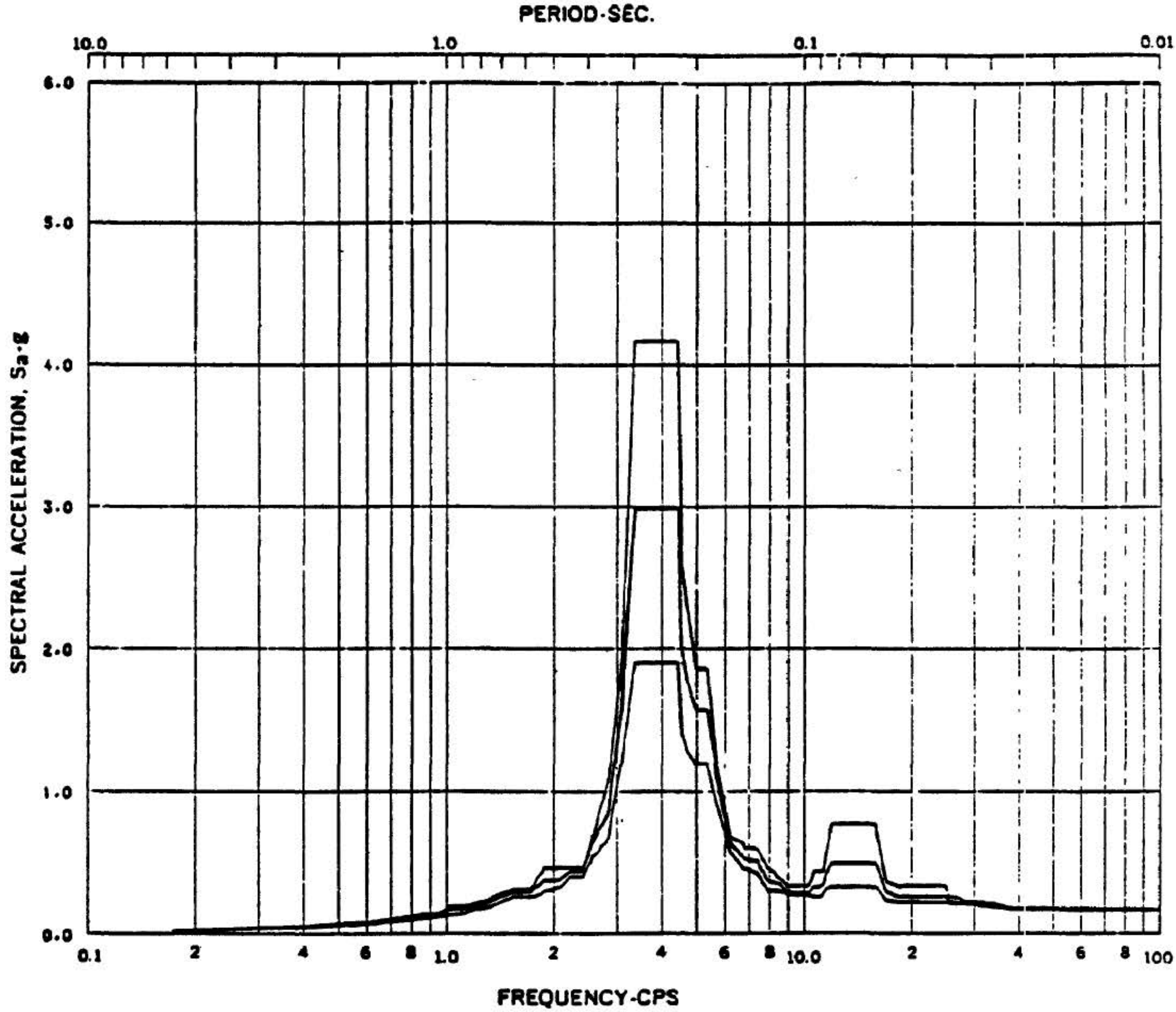
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

RESPONSE SPECTRUM
AT REFUELING AREA
E-W SSE

FIGURE 3.7B-65, Rev. 55

Auto-Cad Figure Fsar 3_7B_65.dwg



ACCELERATION SPECTRA FOR REACTOR/CONTROL BLDG

LOAD CASE SUSQUEHANNA NS OBE

NODE 23, DIRECTION NS, ELEV 818'-1

DAMPING 0.005,0.01,0.02

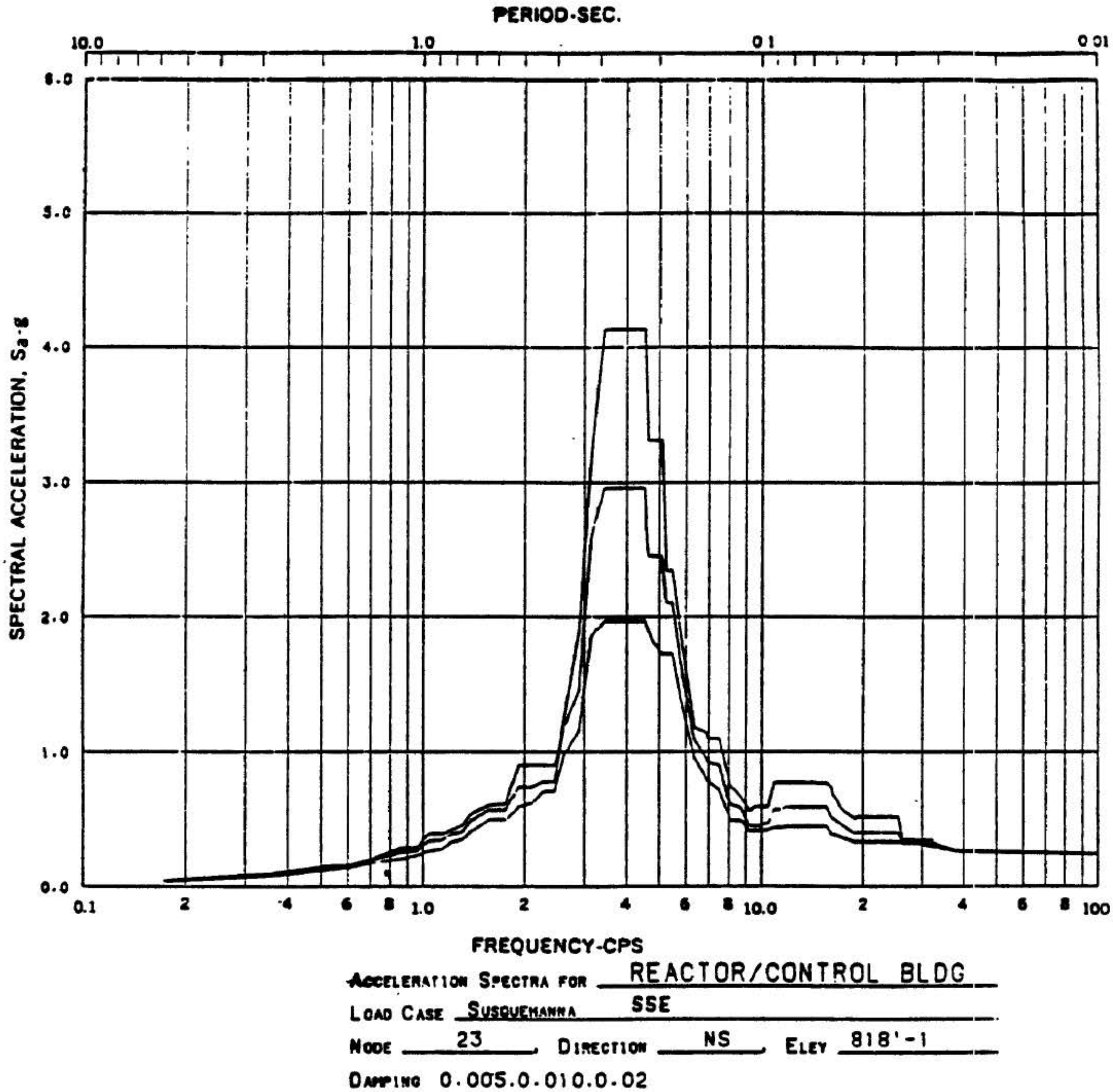
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

RESPONSE SPECTRUM
AT REFUELING AREA
N-S OBE

FIGURE 3.7B-66, Rev. 55

Auto-Cad Figure Fsar 3_7B_66.dwg



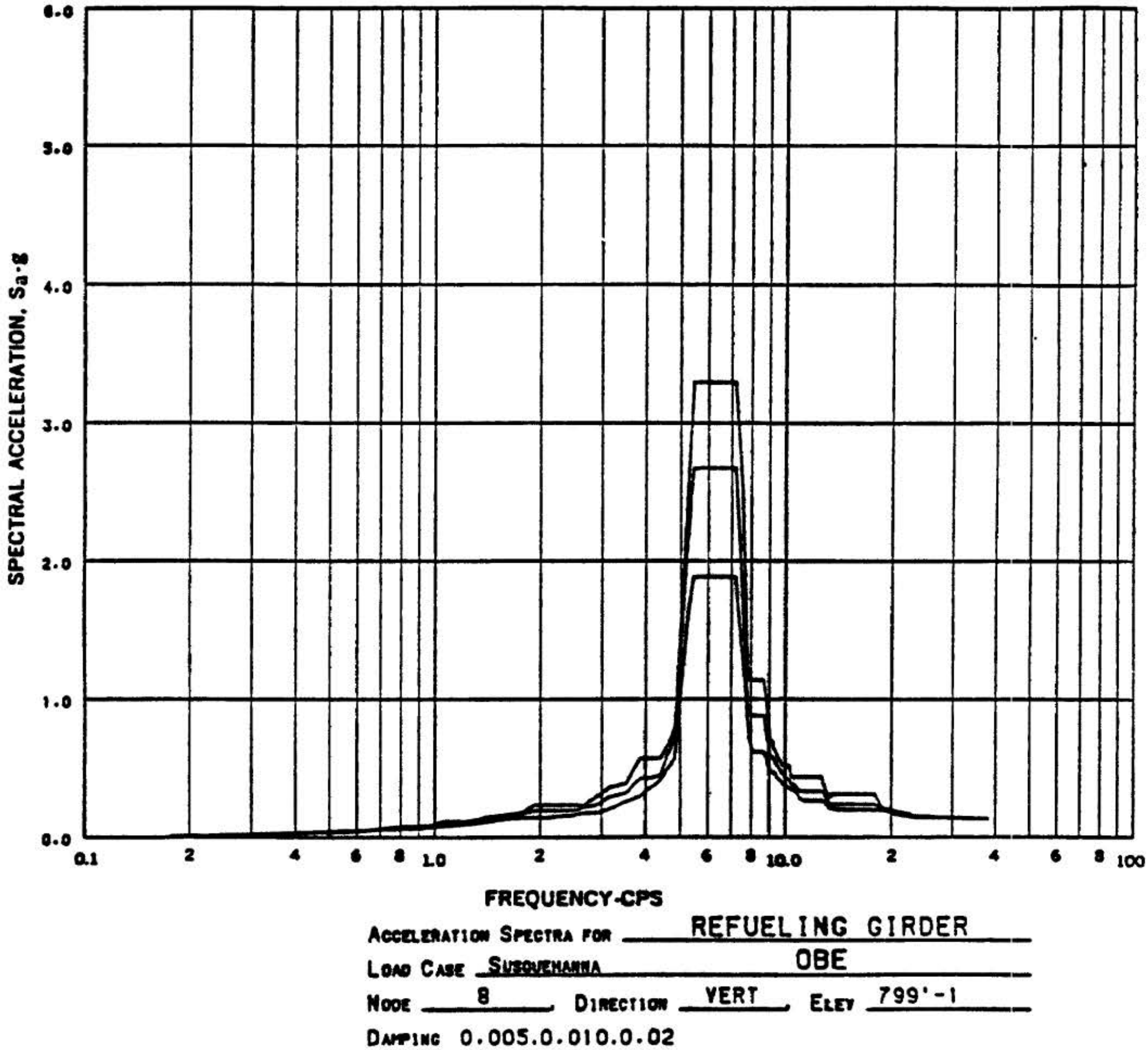
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

RESPONSE SPECTRUM
AT REFUELING AREA
N-S SSE

FIGURE 3.7B-67, Rev. 55

Auto-Cad Figure Fsar 3_7B_67.dwg



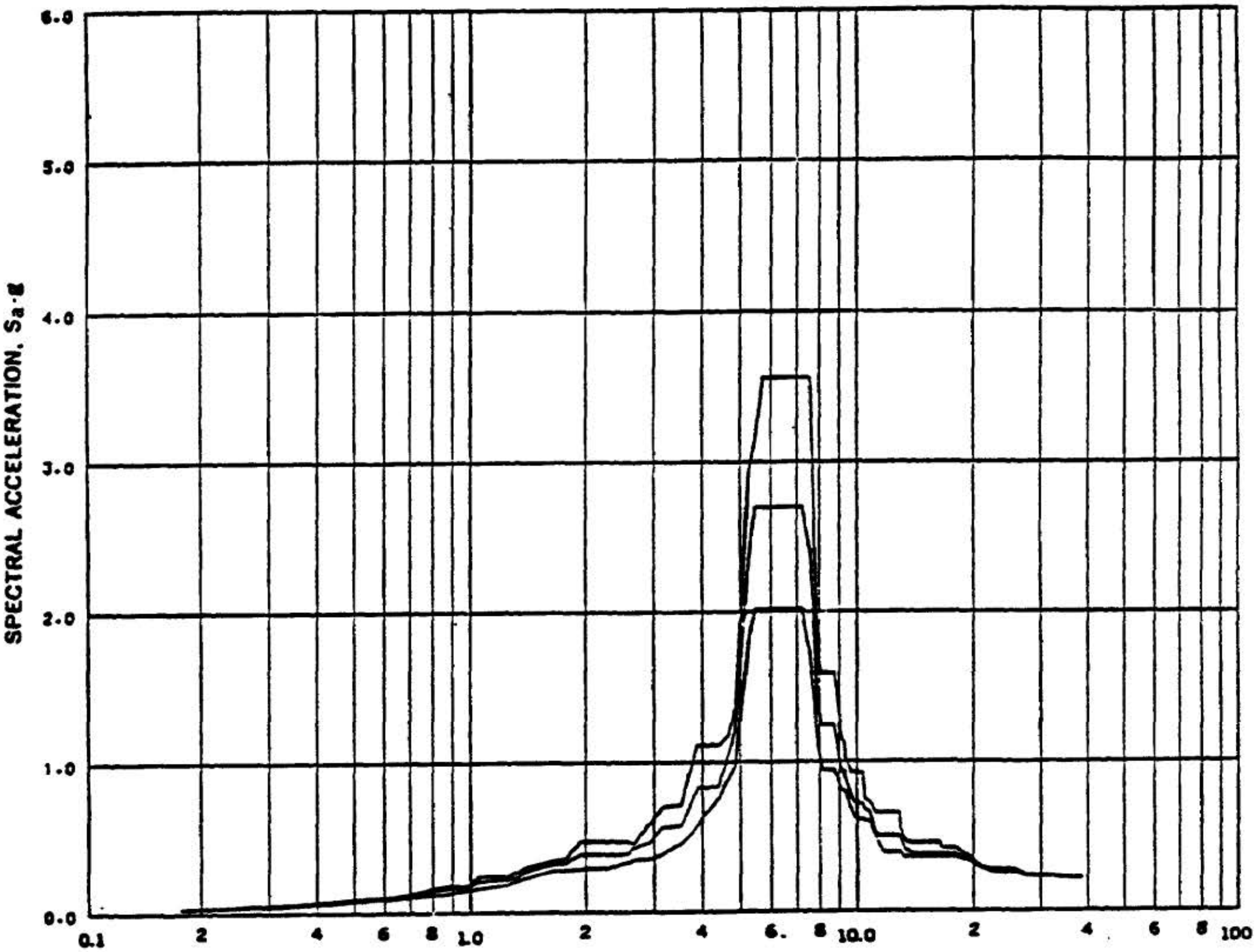
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

RESPONSE SPECTRUM
AT REFUELING AREA
VERTICAL OBE

FIGURE 3.7B-68, Rev. 55

Auto-Cad Figure Fsar 3_7B_68.dwg



ACCELERATION SPECTRA FOR REFUELING GIRDER

LOAD CASE SUSQUEHANNA SSE

NODE 8, DIRECTION VERT, ELEV 799'-1

DAMPING 0.005.0.010.0.02

FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

RESPONSE SPECTRUM
AT REFUELING AREA
VERTICAL SSE

FIGURE 3.7B-69, Rev. 55

Auto-Cad Figure Fsar 3_7B_69.dwg

THIS FIGURE HAS BEEN
RENUMBERED TO 3.7B-11

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Figure renumbered from 3.7B-6A to 3.7B-11 FIGURE 3.7B-6A, Rev. 55

AutoCAD Figure 3_7B_6A.doc

THIS FIGURE HAS BEEN
RENUMBERED TO 3.7B-12

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

Figure renumbered from 3.7B-6B to 3.7B-12

FIGURE 3.7B-6B, Rev. 55

AutoCAD Figure 3_7B_6B.doc

THIS FIGURE HAS BEEN
RENUMBERED TO 3.7B-13

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Figure renumbered from 3.7B-6C to 3.7B-13 FIGURE 3.7B-6C, Rev. 55

AutoCAD Figure 3_7B_6C.doc

THIS FIGURE HAS BEEN
RENUMBERED TO 3.7B-14

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Figure renumbered from 3.7B-6D to 3.7B-14 FIGURE 3.7B-6D, Rev. 55

AutoCAD Figure 3_7B_6D.doc

THIS FIGURE HAS BEEN
RENUMBERED TO 3.7B-15

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Figure renumbered from 3.7B-6E to 3.7B-15 FIGURE 3.7B-6E, Rev. 55

AutoCAD Figure 3_7B_6E.doc

THIS FIGURE HAS BEEN
RENUMBERED TO 3.7B-16

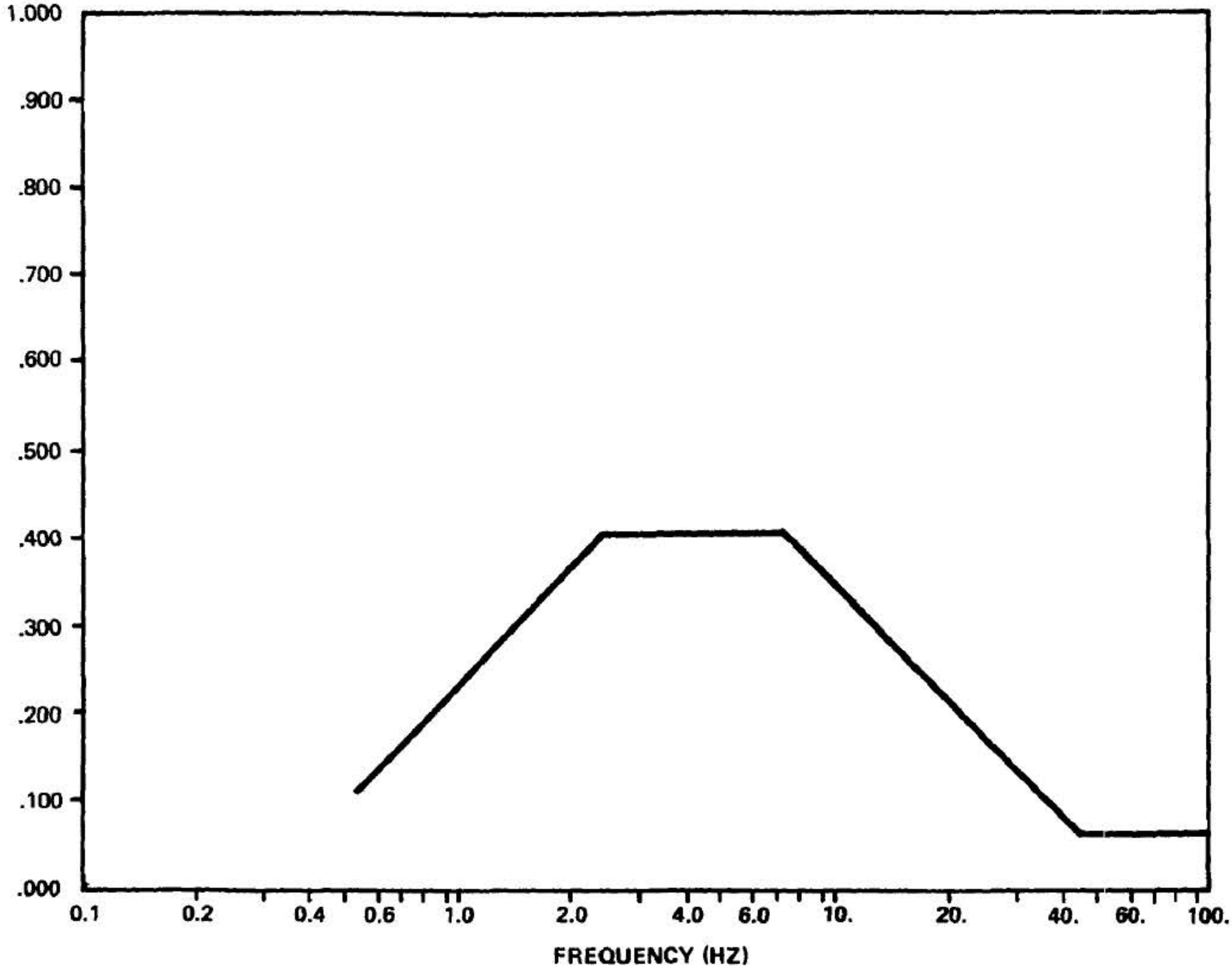
FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

Figure renumbered from 3.7B-6F to 3.7B-16

FIGURE 3.7B-6F, Rev. 55

AutoCAD Figure 3_7B_6F.doc



LOCATION: TOP OF PEDESTAL FOR DIESELS 'A-D'
DIRECTION: E-W
EARTHQUAKE: OBE
DAMPING: 0.005

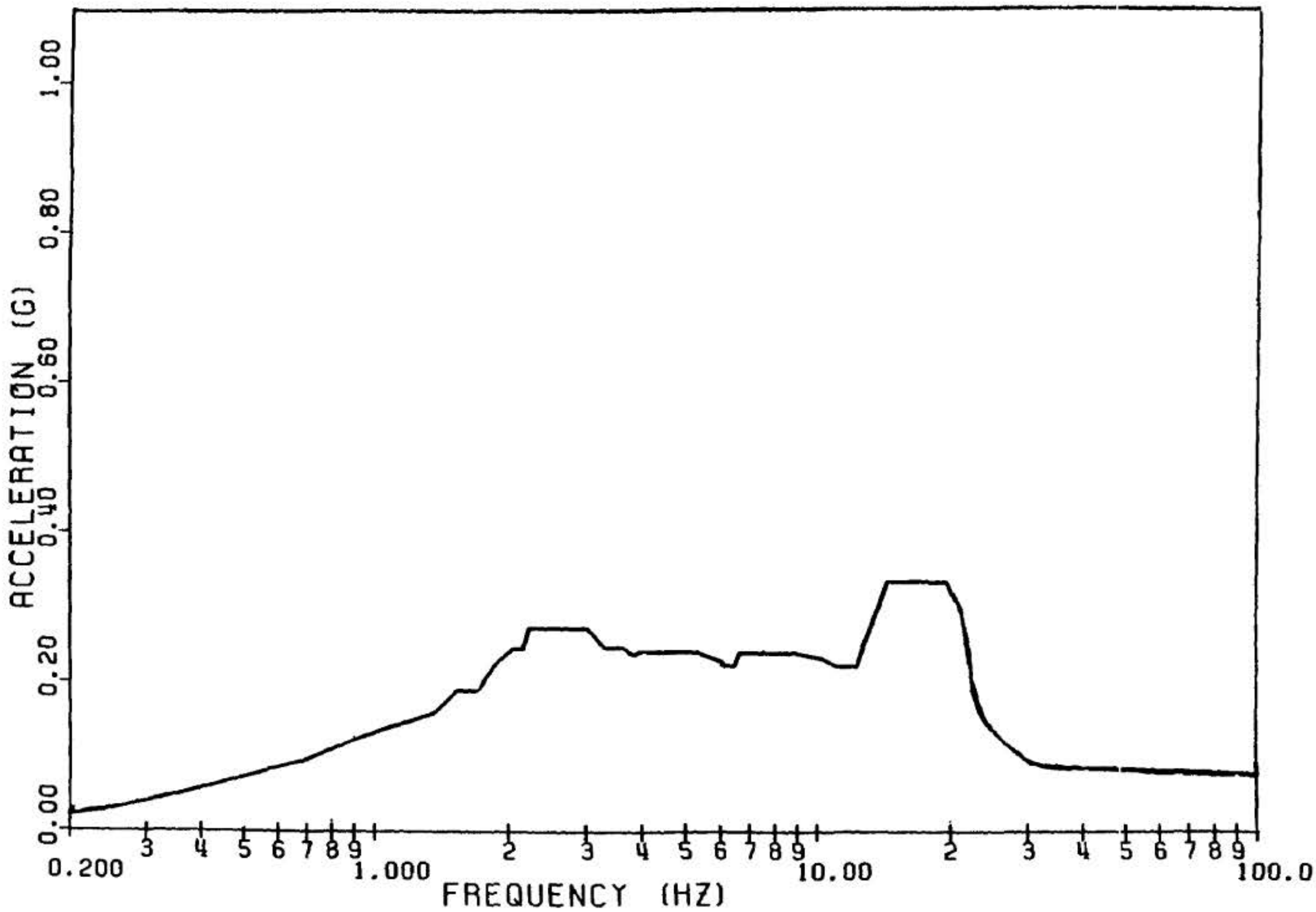
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

RESPONSE SPECTRUM
AT TOP OF PEDESTAL
(DIESELS 'A-D') E-W OBE

FIGURE 3.7B-70, Rev. 55

Auto-Cad Figure Fsar 3_7B_70.dwg



LOCATION: TOP OF PEDESTAL FOR DIESEL E
DIRECTION: E-W
EARTHQUAKE: OBE
DAMPING: 0.020

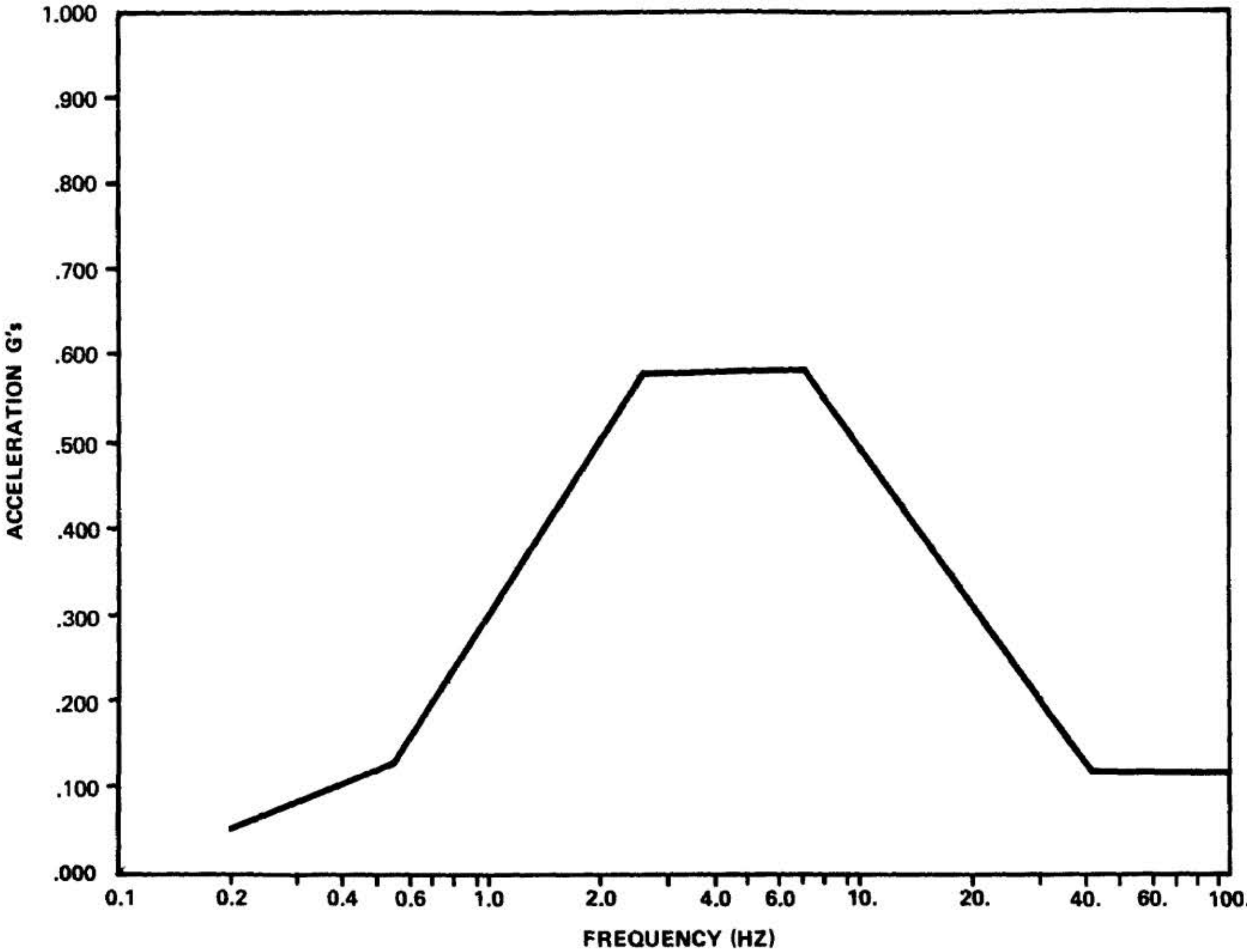
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

RESPONSE SPECTRUM
AT TOP OF PEDESTAL
FOR DIESEL 'E'
E-W OBE

FIGURE 3.7B-71, Rev. 55

Auto-Cad Figure Fsar 3_7B_71.dwg



LOCATION: TOP OF PEDESTAL FOR DIESELS 'A-D'
DIRECTION: E-W
EARTHQUAKE: SSE
DAMPING: 0.010

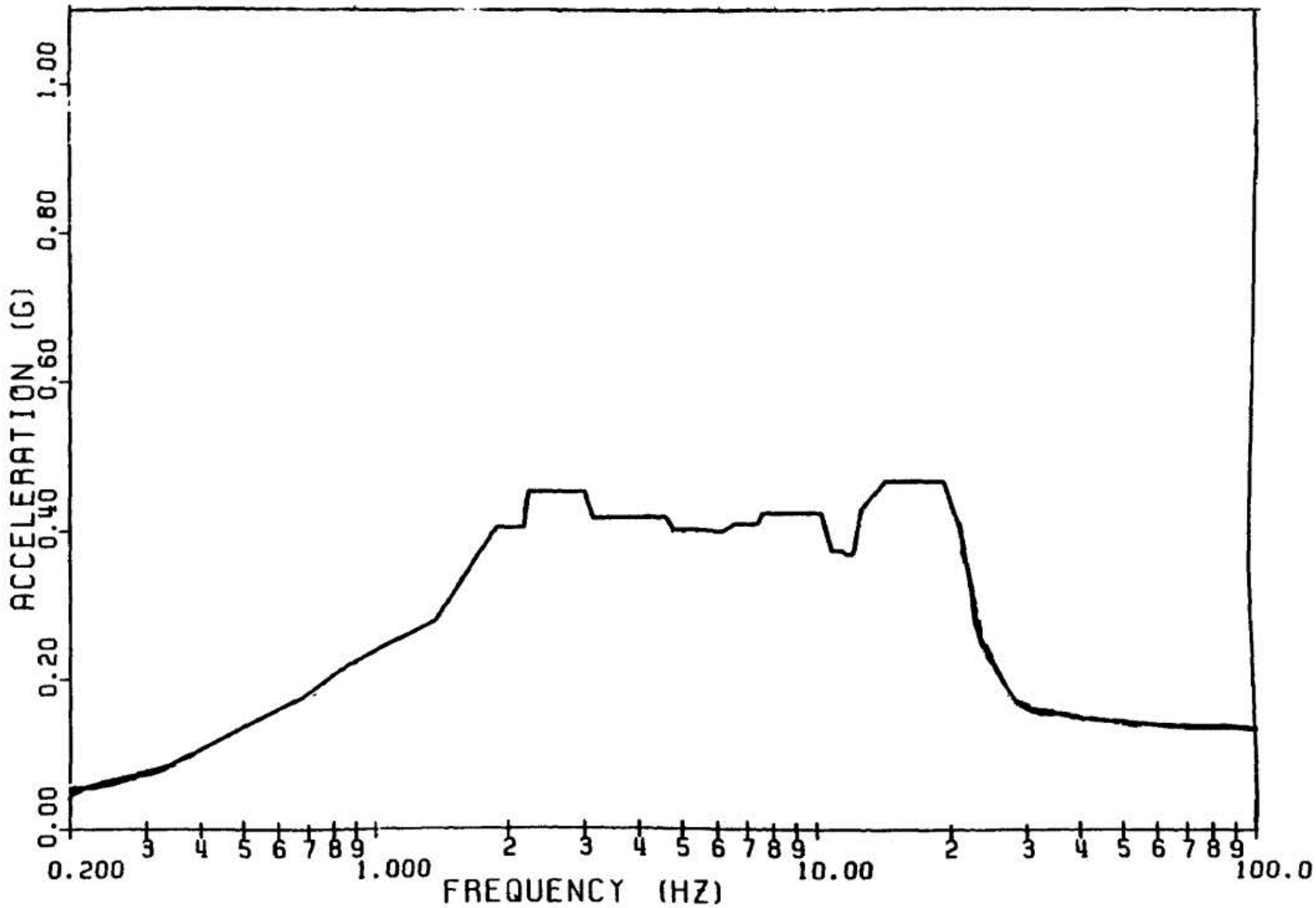
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

RESPONSE SPECTRUM
AT TOP OF PEDESTAL (DIESEL 'A-D')
E-W SSE

FIGURE 3.7B-72, Rev. 55

Auto-Cad Figure Fsar 3_7B_72.dwg



LOCATION: TOP OF PEDESTAL FOR DIESEL E
DIRECTION: E-W
EARTHQUAKE: SSE
DAMPING: 0.030

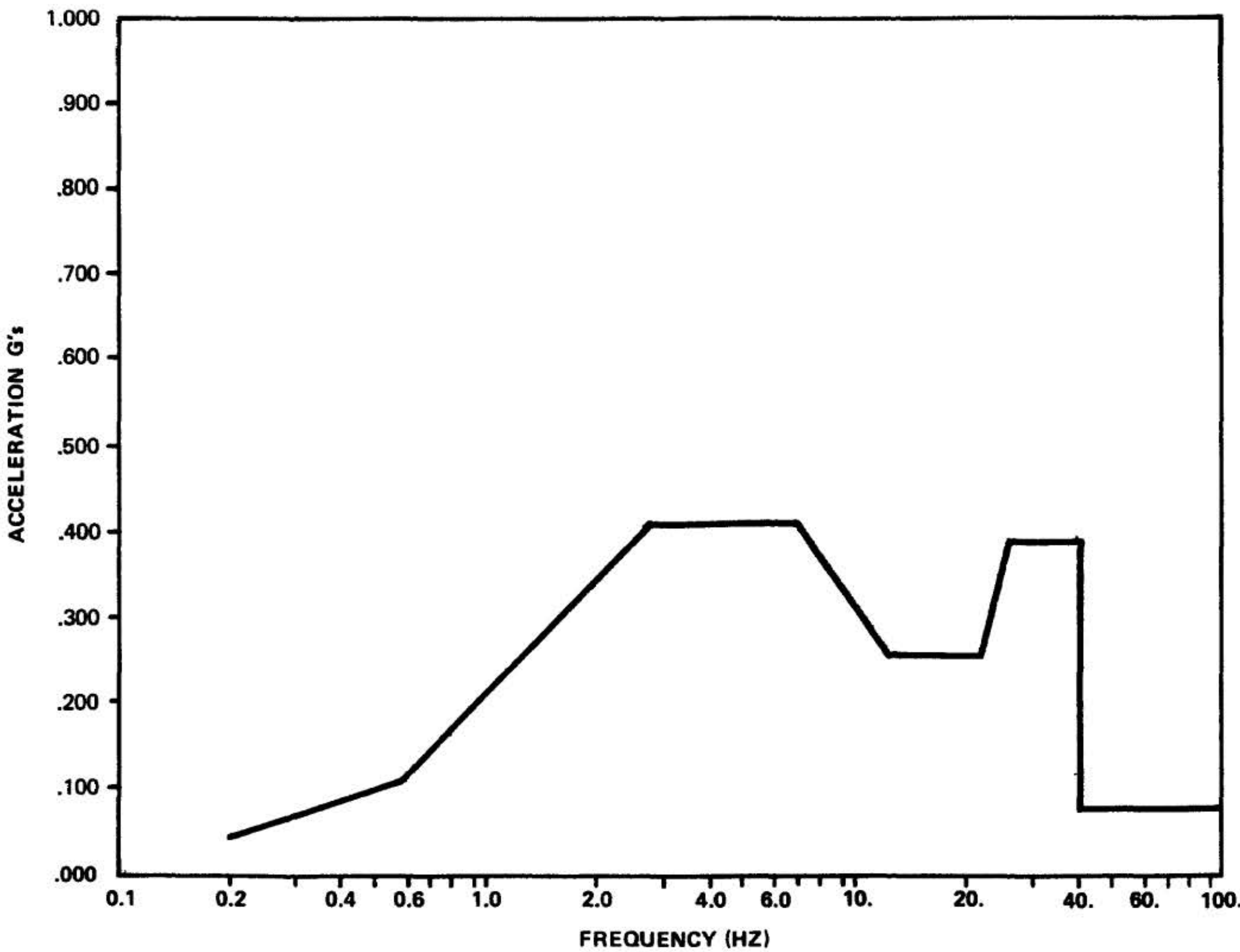
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

RESPONSE SPECTRUM
AT TOP OF PEDESTAL FOR DIESEL 'E'
E-W SSE

FIGURE 3.7B-73, Rev. 55

Auto-Cad Figure Fsar 3_7B_73.dwg



LOCATION: TOP OF PEDESTAL FOR DIESEL 'A-D'
DIRECTION: N-S
EARTHQUAKE: OBE
DAMPING: 0.005

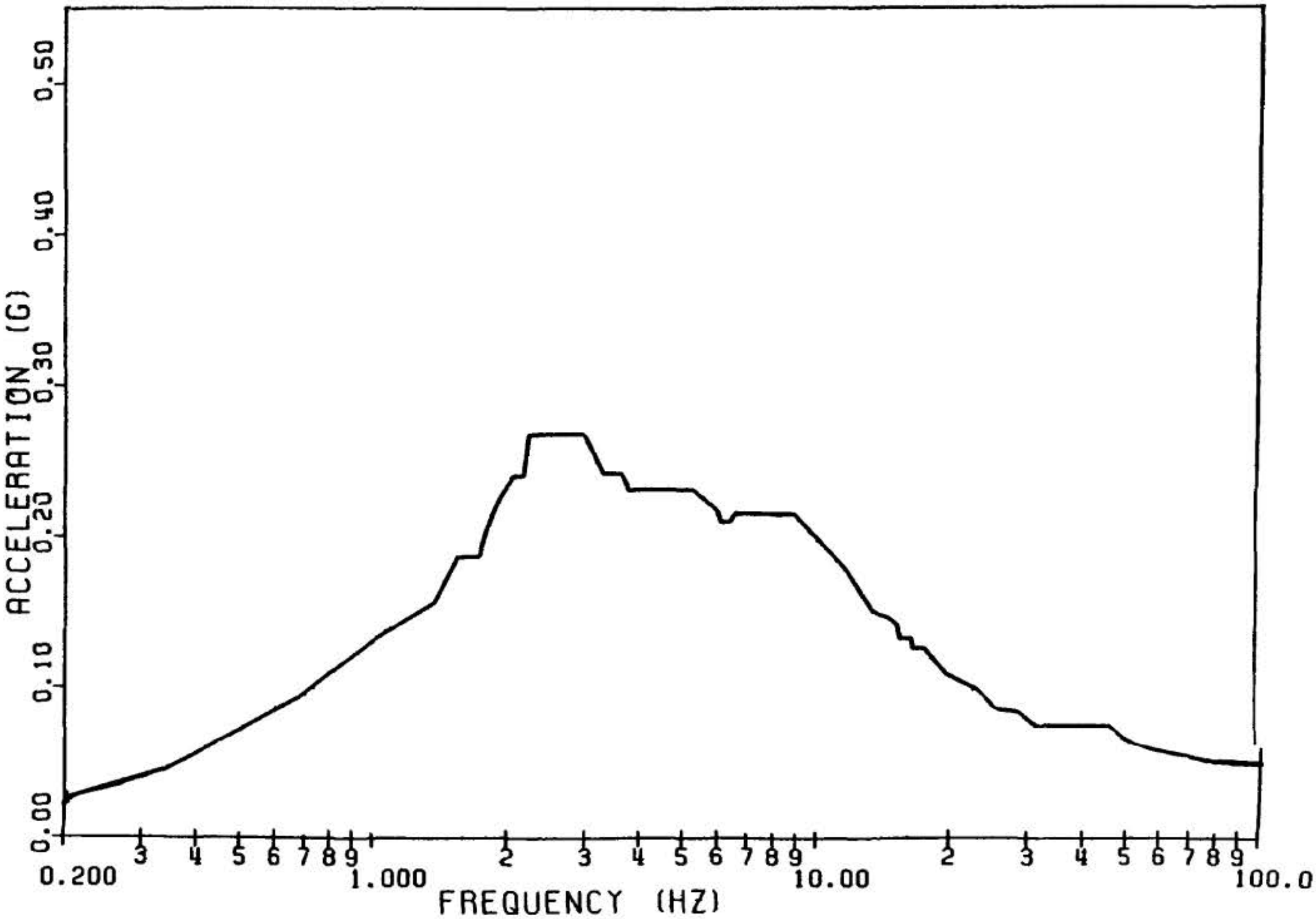
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

RESPONSE SPECTRUM
AT TOP OF PEDESTAL (DIESEL 'A-D')
E-W OBE

FIGURE 3.7B-74, Rev. 55

Auto-Cad Figure Fsar 3_7B_74.dwg



LOCATION: TOP OF PEDESTAL FOR DIESEL E
DIRECTION: N-S
EARTHQUAKE: OBE
DAMPING: 0.020

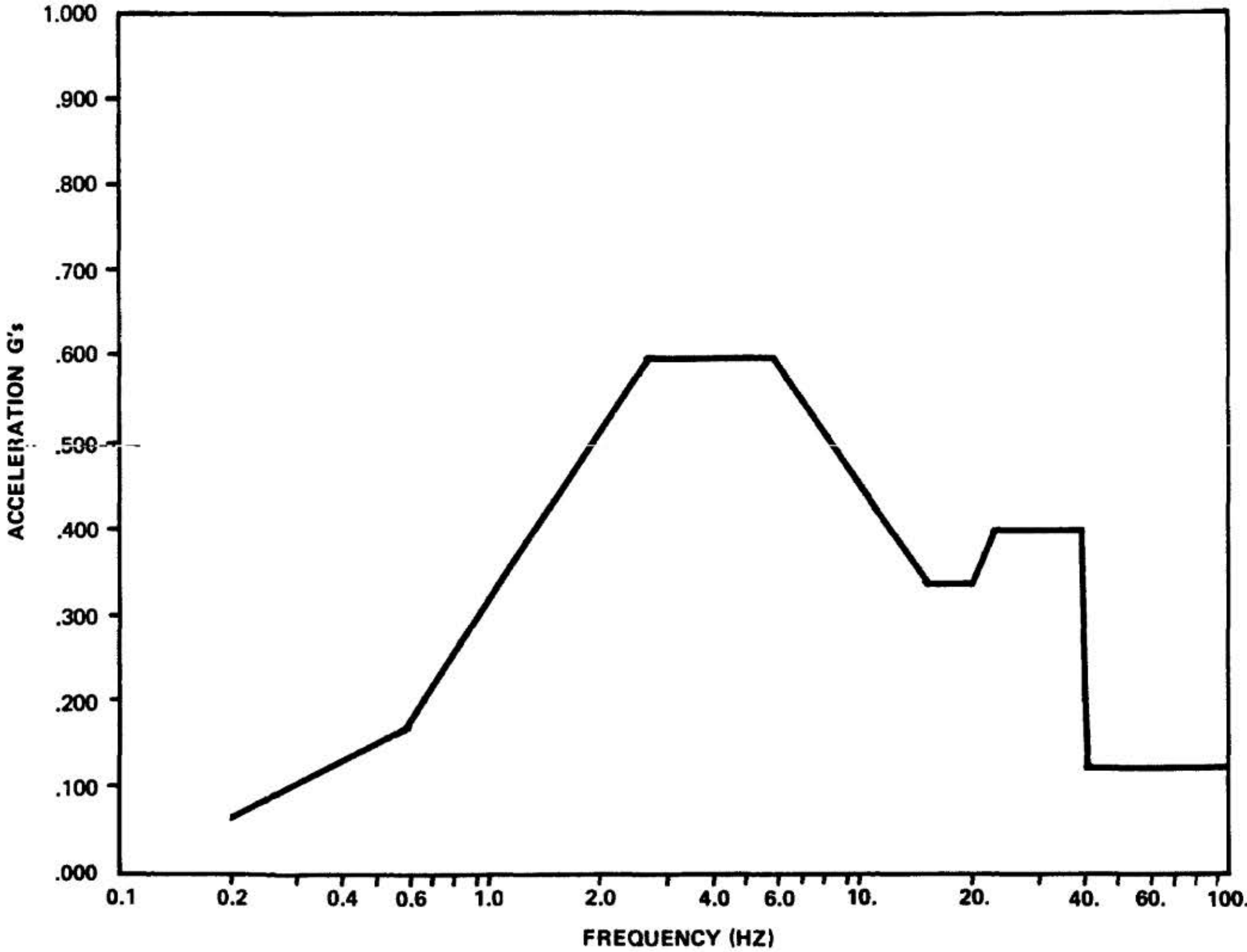
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

RESPONSE SPECTRUM
AT TOP OF PEDESTAL FOR DIESEL 'E'
N-S OBE

FIGURE 3.7B-75, Rev. 55

Auto-Cad Figure Fsar 3_7B_75.dwg



LOCATION: TOP OF PEDESTAL FOR DIESEL 'A-D'
DIRECTION: N-S
EARTHQUAKE: SSE
DAMPING: 0.010

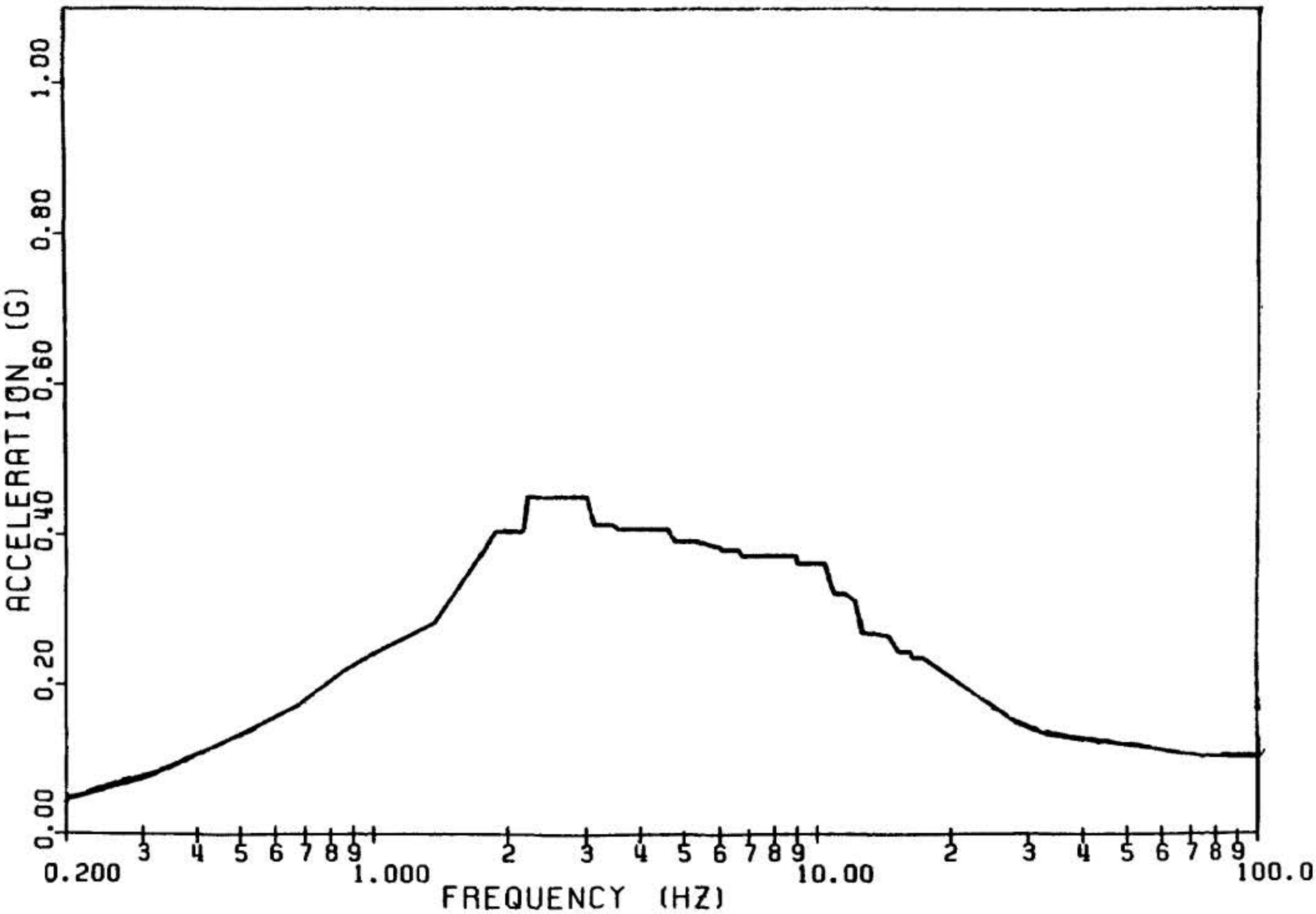
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

RESPONSE SPECTRUM
AT TOP OF PEDESTAL (DIESELS 'A-D')
N-S SSE

FIGURE 3.7B-76, Rev. 55

Auto-Cad Figure Fsar 3_7B_76.dwg

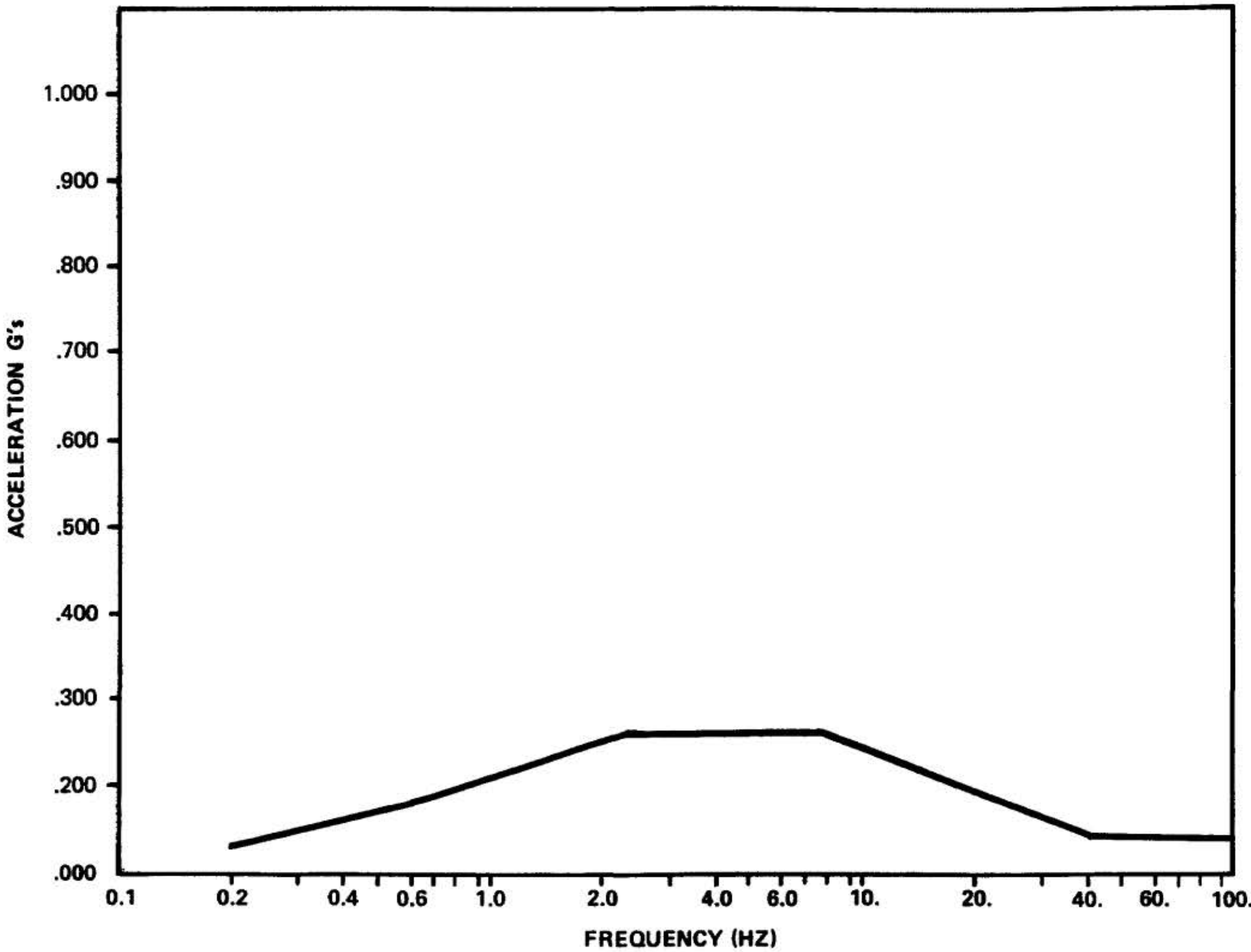


LOCATION: TOP OF PEDESTAL FOR DIESEL E
DIRECTION: N-S
EARTHQUAKE: SSE
DAMPING: 0.030

FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
RESPONSE SPECTRUM AT TOP OF PEDESTAL FOR DIESEL 'E' N-S SSE
FIGURE 3.7B-77, Rev. 55

Auto-Cad Figure Fsar 3_7B_77.dwg



LOCATION: TOP OF PEDESTAL FOR DIESEL 'A-D'
DIRECTION: VERTICAL
EARTHQUAKE: OBE
DAMPING: 0.005

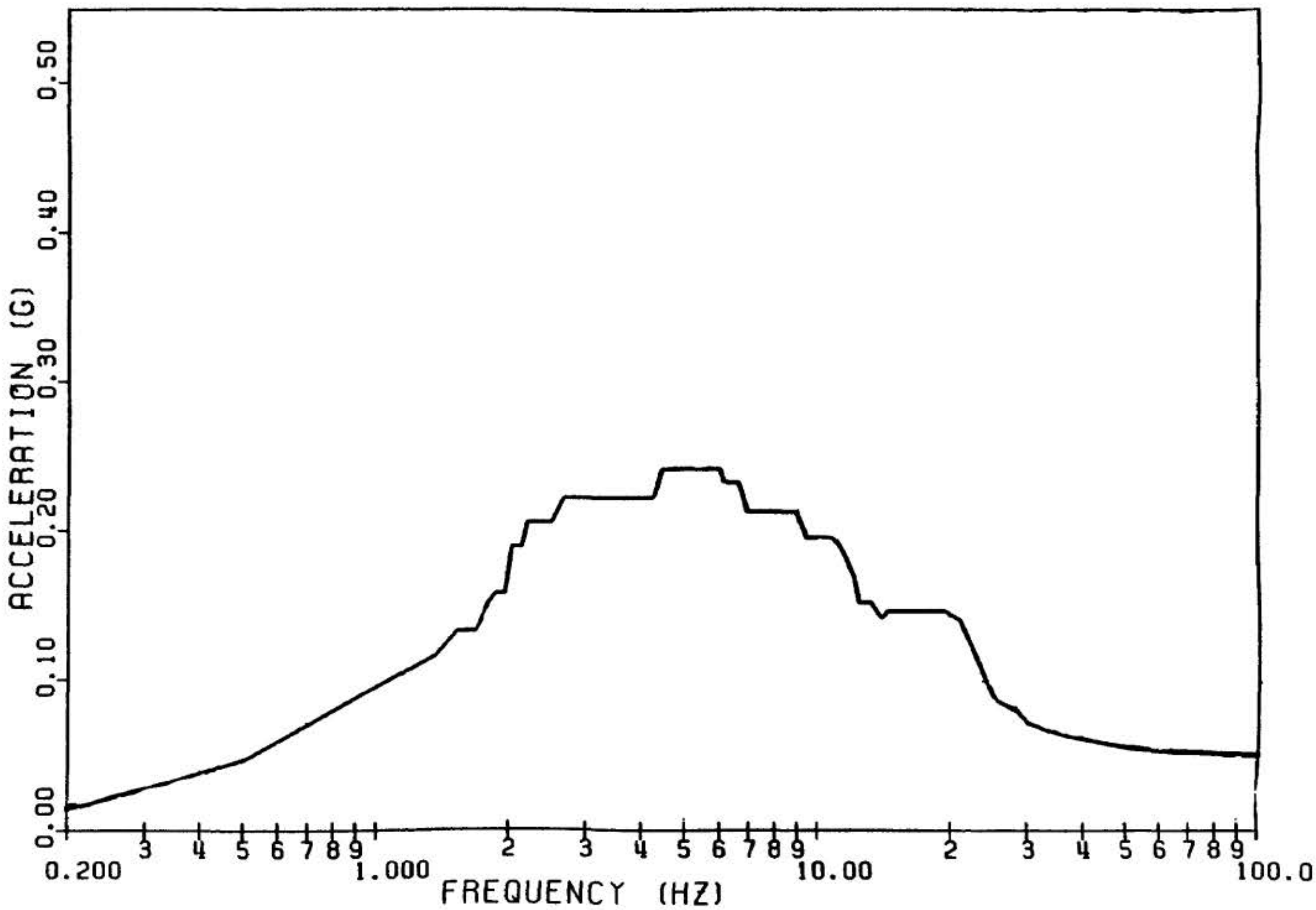
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

RESPONSE SPECTRUM
AT TOP OF PEDESTAL (DIESEL 'A-D')
VERTICAL OBE

FIGURE 3.7B-78, Rev. 55

Auto-Cad Figure Fsar 3_7B_78.dwg



LOCATION: TOP OF PEDESTAL FOR DIESEL E
DIRECTION: VERTICAL
EARTHQUAKE: OBE
DAMPING: 0.020

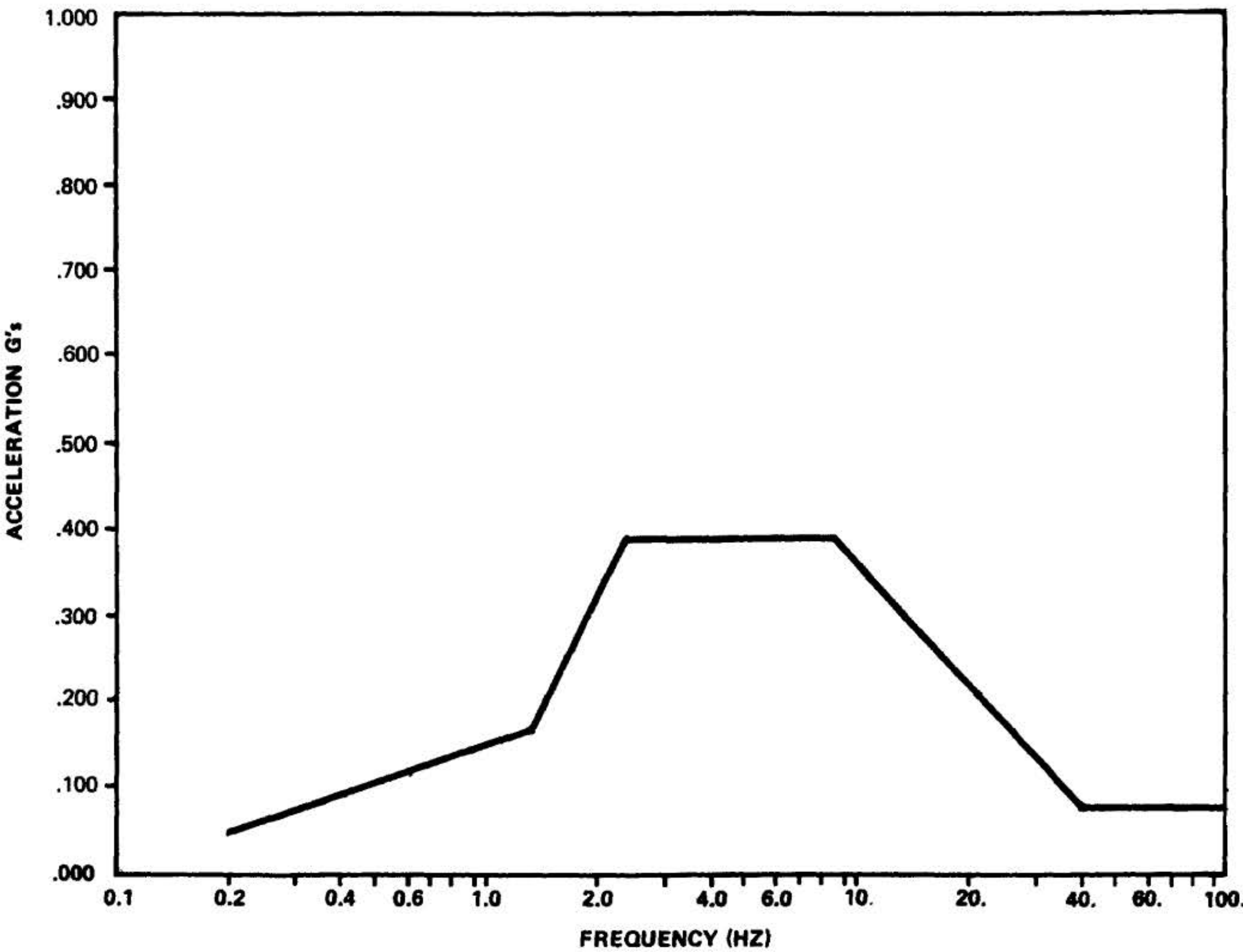
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

RESPONSE SPECTRUM
AT TOP OF PEDESTAL FOR DIESEL 'E'
VERTICAL OBE

FIGURE 3.7B-79, Rev. 55

Auto-Cad Figure Fsar 3_7B_79.dwg



LOCATION: TOP OF PEDESTAL FOR DIESEL 'A-D'
DIRECTION: VERTICAL
EARTHQUAKE: SSE
DAMPING: 0.010

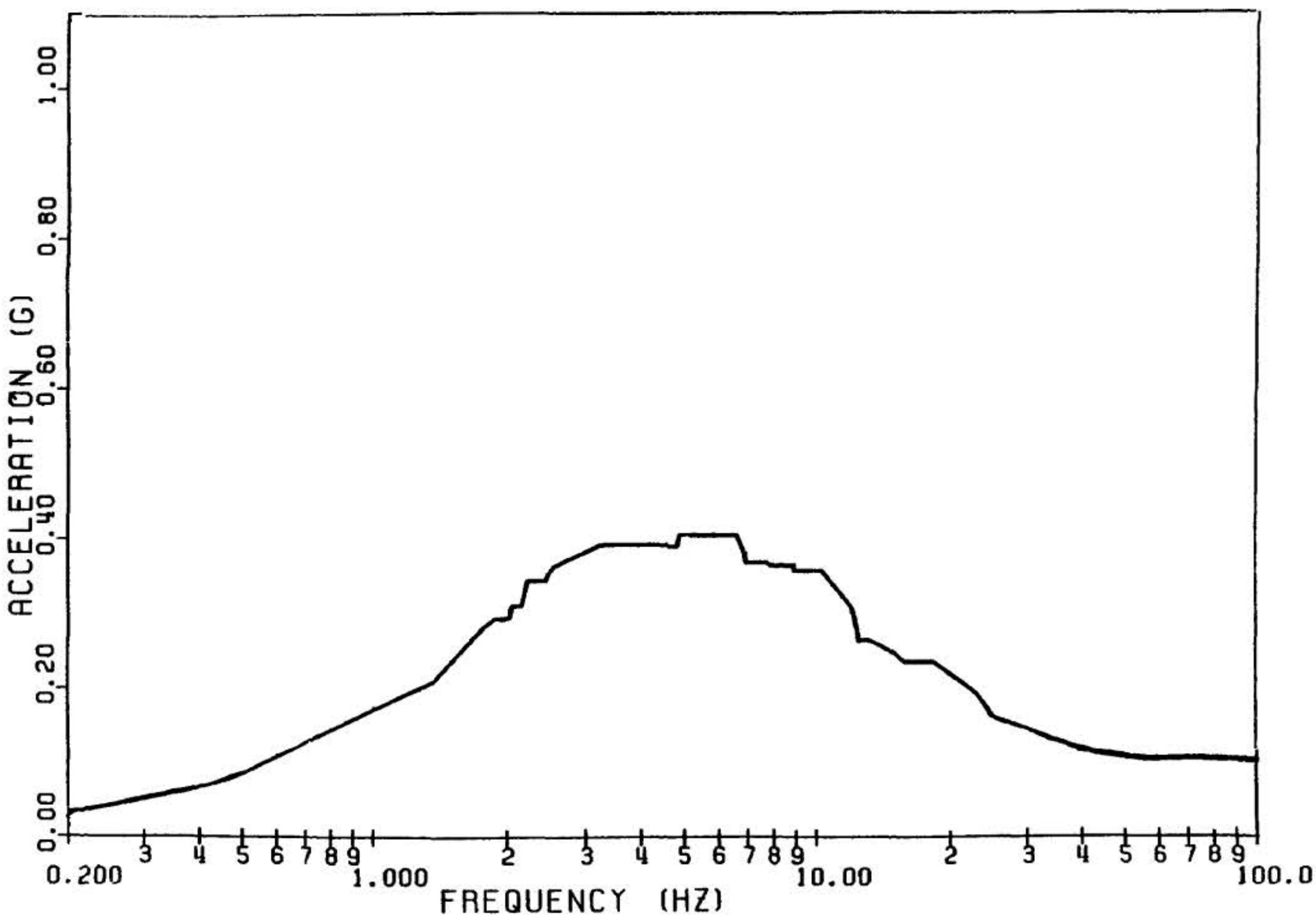
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

RESPONSE SPECTRUM
AT TOP OF PEDESTAL (DIESEL 'A-D')
VERTICAL SSE

FIGURE 3.7B-80, Rev. 55

Auto-Cad Figure Fsar 3_7B_80.dwg



LOCATION: TOP OF PEDESTAL FOR DIESEL E
DIRECTION: VERTICAL
EARTHQUAKE: SSE
DAMPING: 0.030

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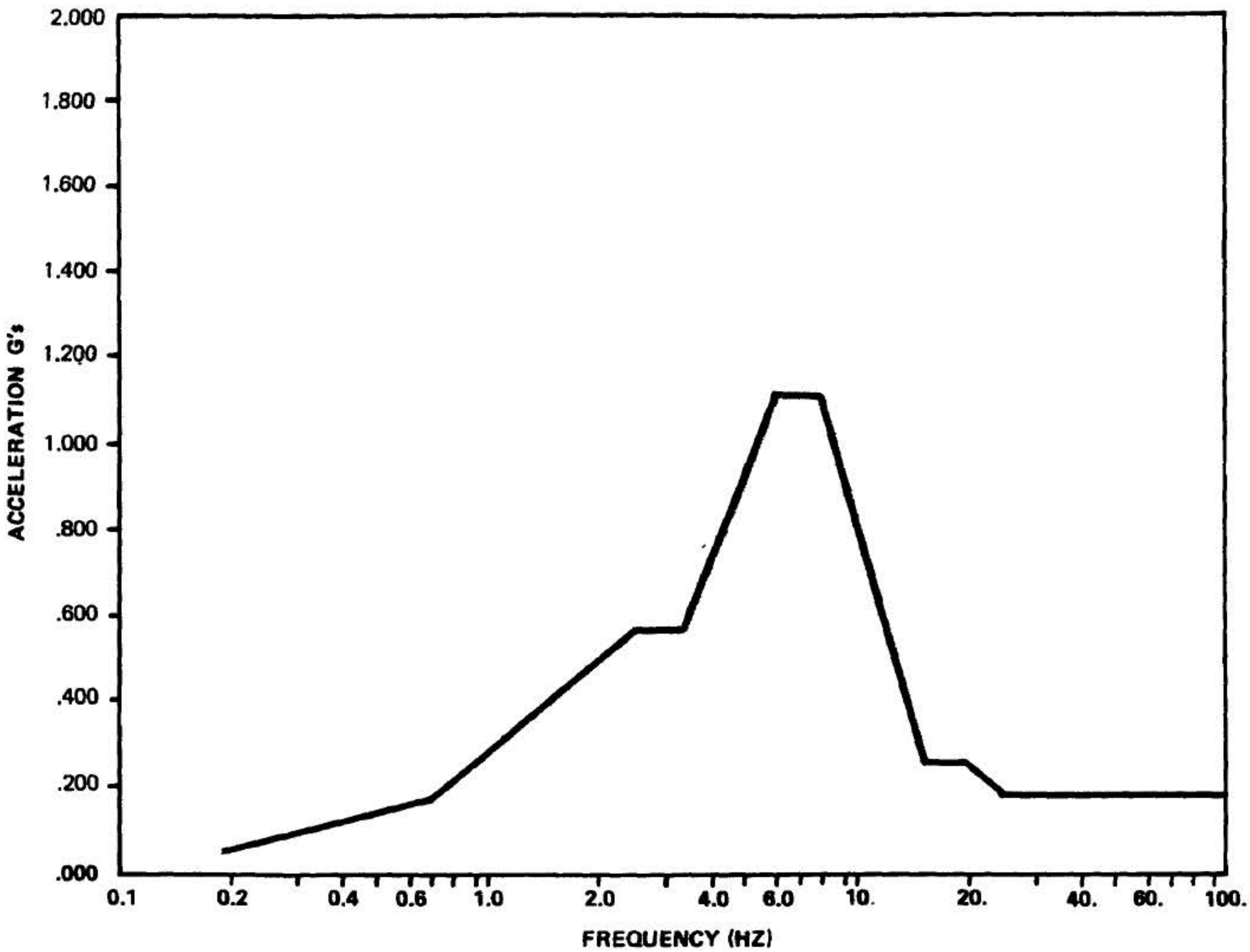
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

RESPONSE SPECTRUM
AT TOP OF PEDESTAL
FOR DIESEL GENERATOR 'E'
VERTICAL SSE

FIGURE 3.7B-81, Rev. 55

Auto-Cad Figure Fsar 3_7B_81.dwg



LOCATION: ESSW PUMPHOUSE (OPERATING FLOOR)
DIRECTION: E-W
EARTHQUAKE: OBE
DAMPING: 0.005

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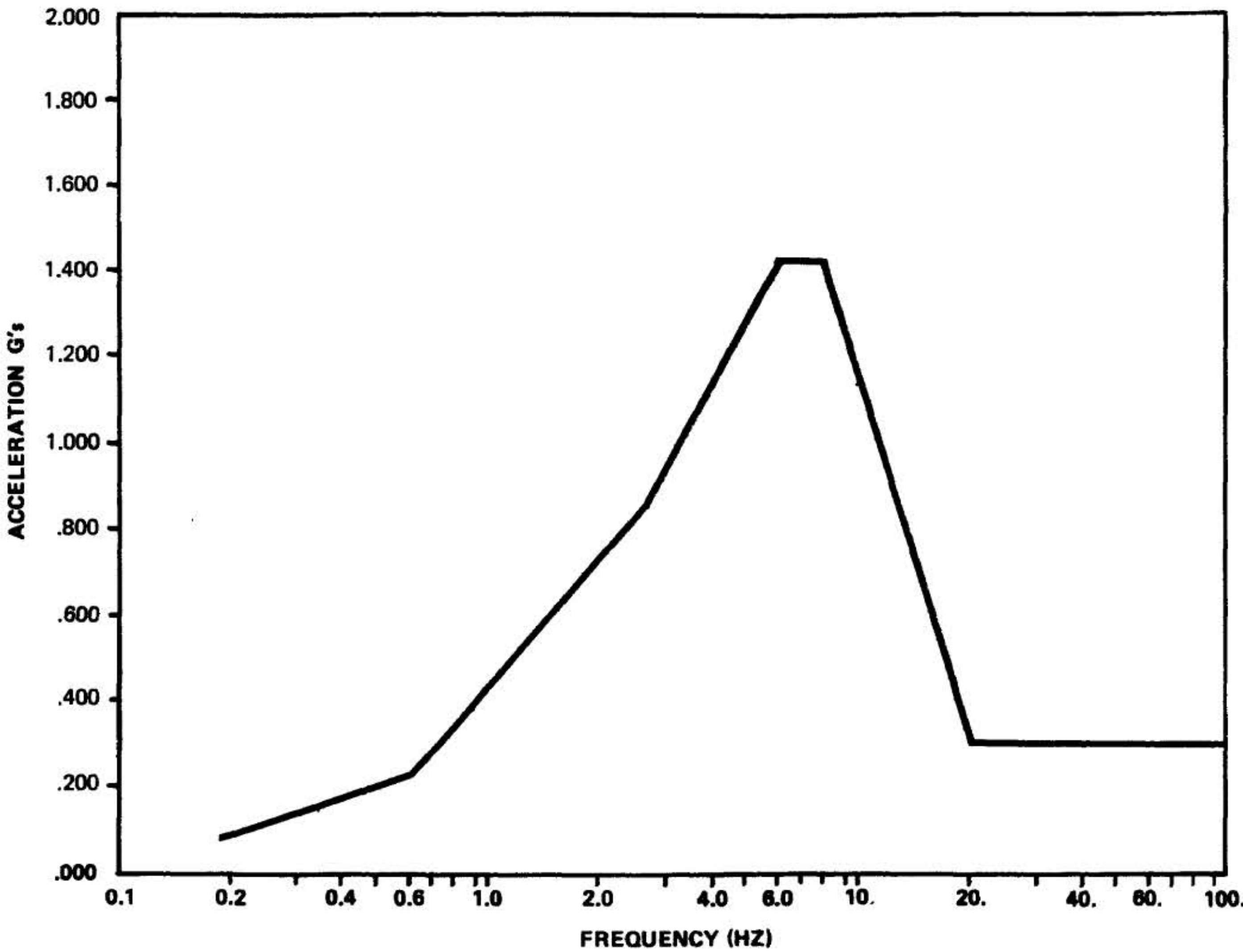
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

RESPONSE SPECTRUM
AT OPERATING FLOOR OF ESSW
PUMPHOUSE
E-W OBE

FIGURE 3.7B-82, Rev. 55

Auto-Cad Figure Fsar 3_7B_82.dwg



LOCATION: ESSW PUMPHOUSE (OPERATING FLOOR)
DIRECTION: E-W
EARTHQUAKE: SSE
DAMPING: 0.010

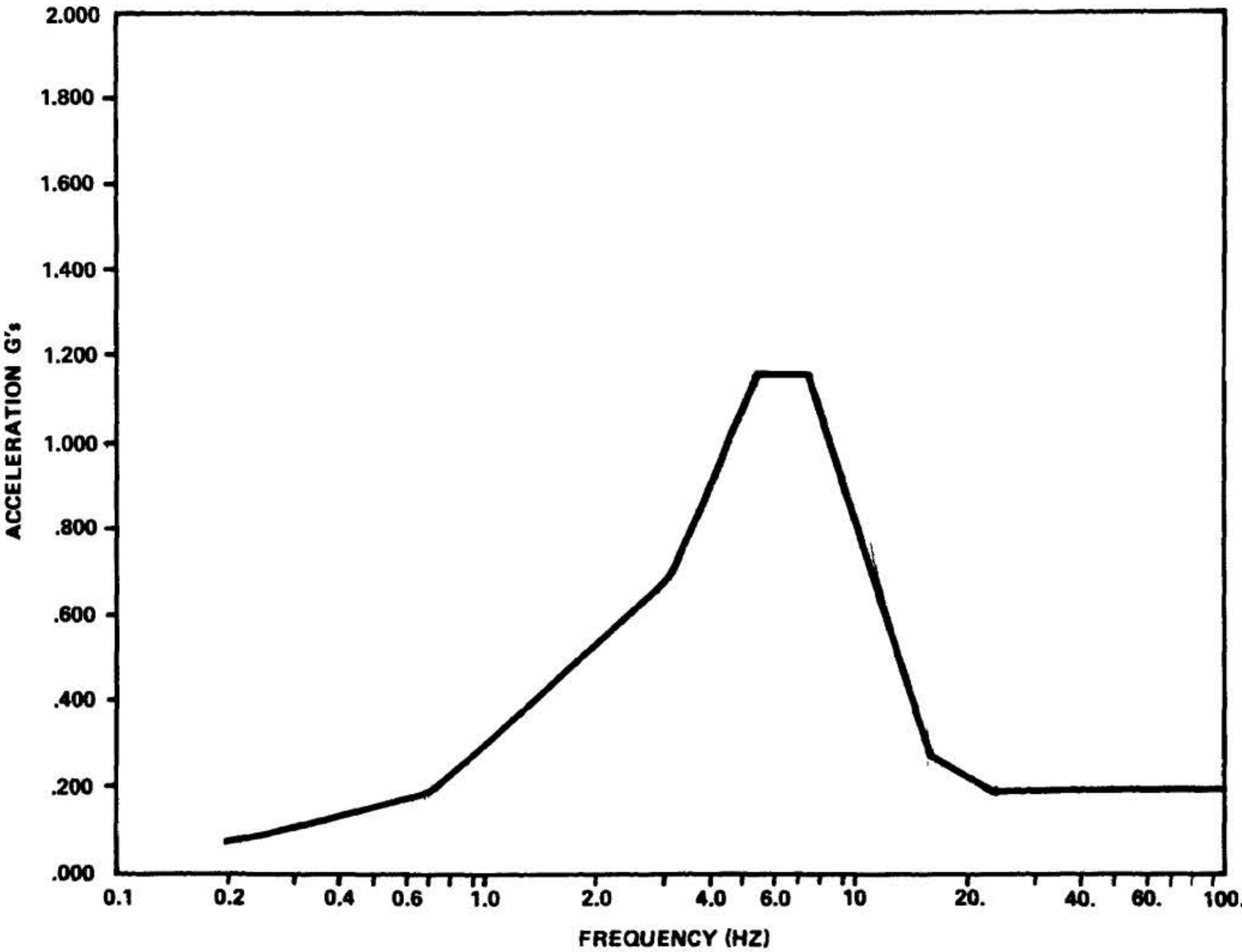
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

RESPONSE SPECTRUM
AT OPERATING FLOOR OF ESSW
PUMPHOUSE
E-W SSE

FIGURE 3.7B-83, Rev. 55

Auto-Cad Figure Fsar 3_7B_83.dwg

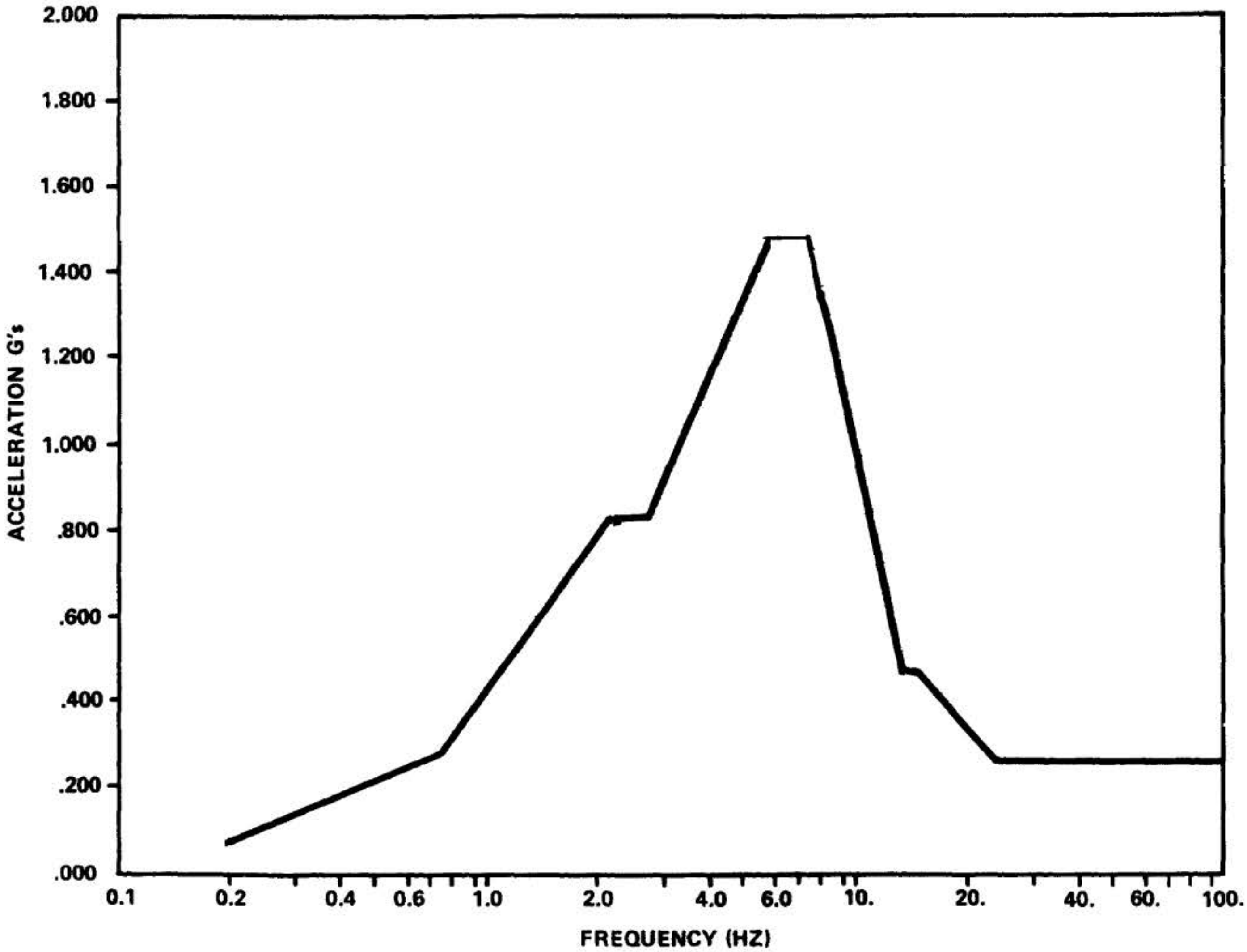


LOCATION: ESSW PUMPHOUSE (OPERATING FLOOR)
DIRECTION: N-S
EARTHQUAKE: OBE
DAMPING: 0.005

FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
RESPONSE SPECTRUM AT OPERATING FLOOR OF ESSW PUMPHOUSE N-S OBE
FIGURE 3.7B-84, Rev. 55

Auto-Cad Figure Fsar 3_7B_84.dwg



LOCATION: ESSW PUMPHOUSE (OPERATING FLOOR)
DIRECTION: N-S
EARTHQUAKE: SSE
DAMPING: 0.010

FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
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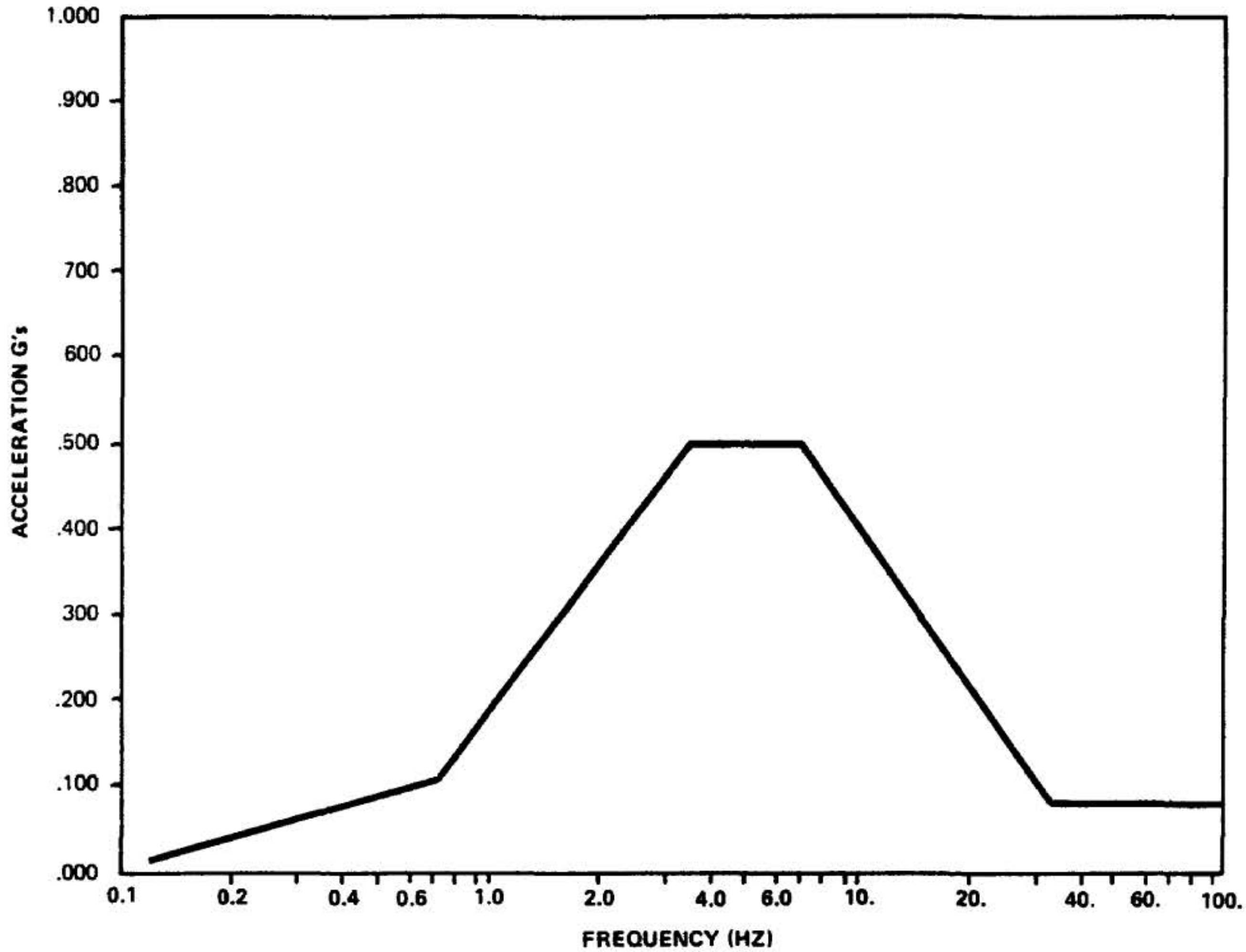
RESPONSE SPECTRUM
AT OPERATING FLOOR OF ESSW
PUMPHOUSE
N-S SSE

FIGURE 3.7B-85, Rev. 55

Auto-Cad Figure Fsar 3_7B_85.dwg



LOCATION: ESSW PUMPHOUSE (OPERATING FLOOR)
DIRECTION: VERTICAL
EARTHQUAKE: OBE
DAMPING. 0.005

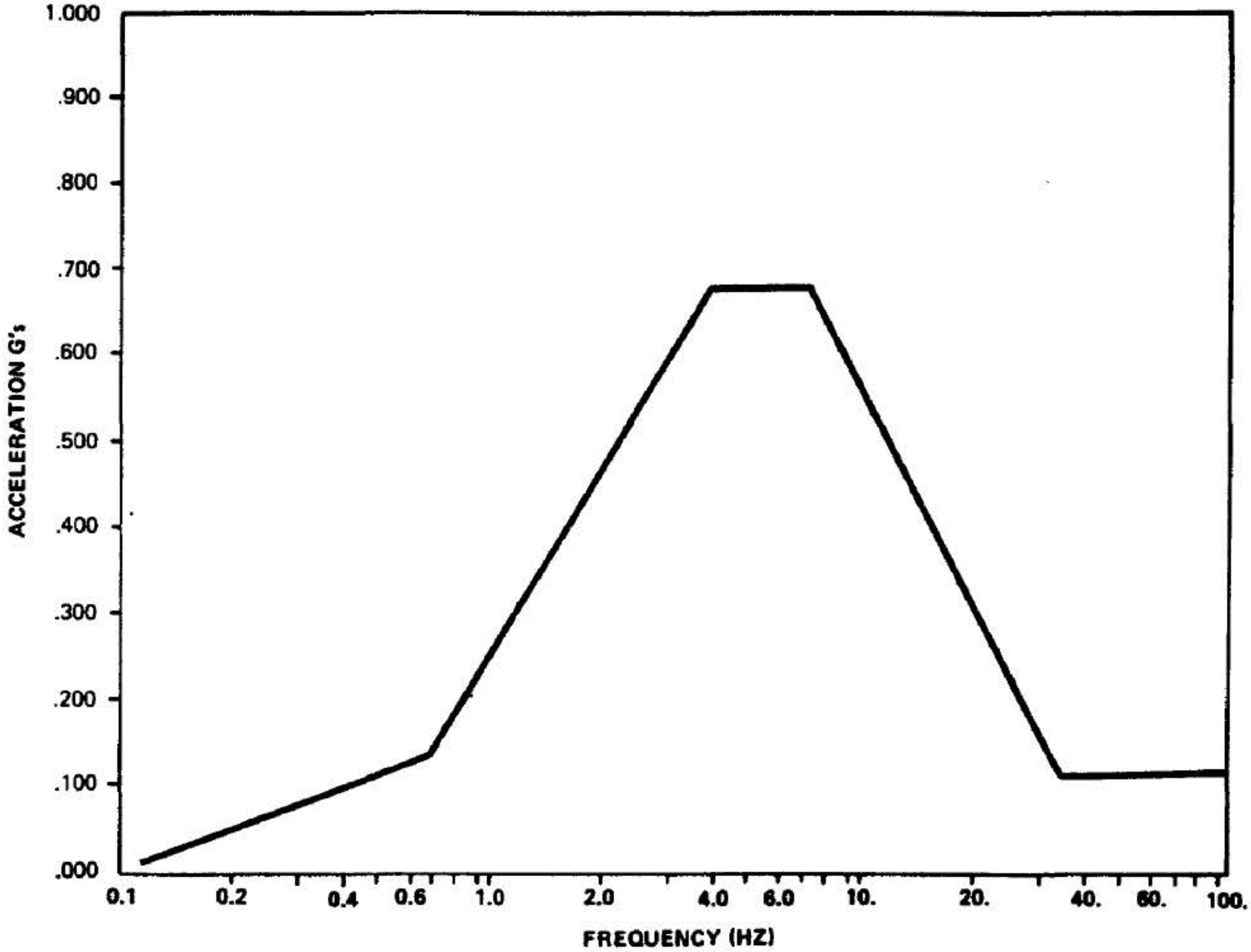
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

RESPONSE SPECTRUM
AT OPERATING FLOOR OF ESSW
PUMPHOUSE
VERTICAL OBE

FIGURE 3.7B-86, Rev. 55

Auto-Cad Figure Fsar 3_7B_86.dwg



LOCATION: ESSW PUMPHOUSE (OPERATING FLOOR)
DIRECTION: VERTICAL
EARTHQUAKE: SSE
DAMPING: 0.010

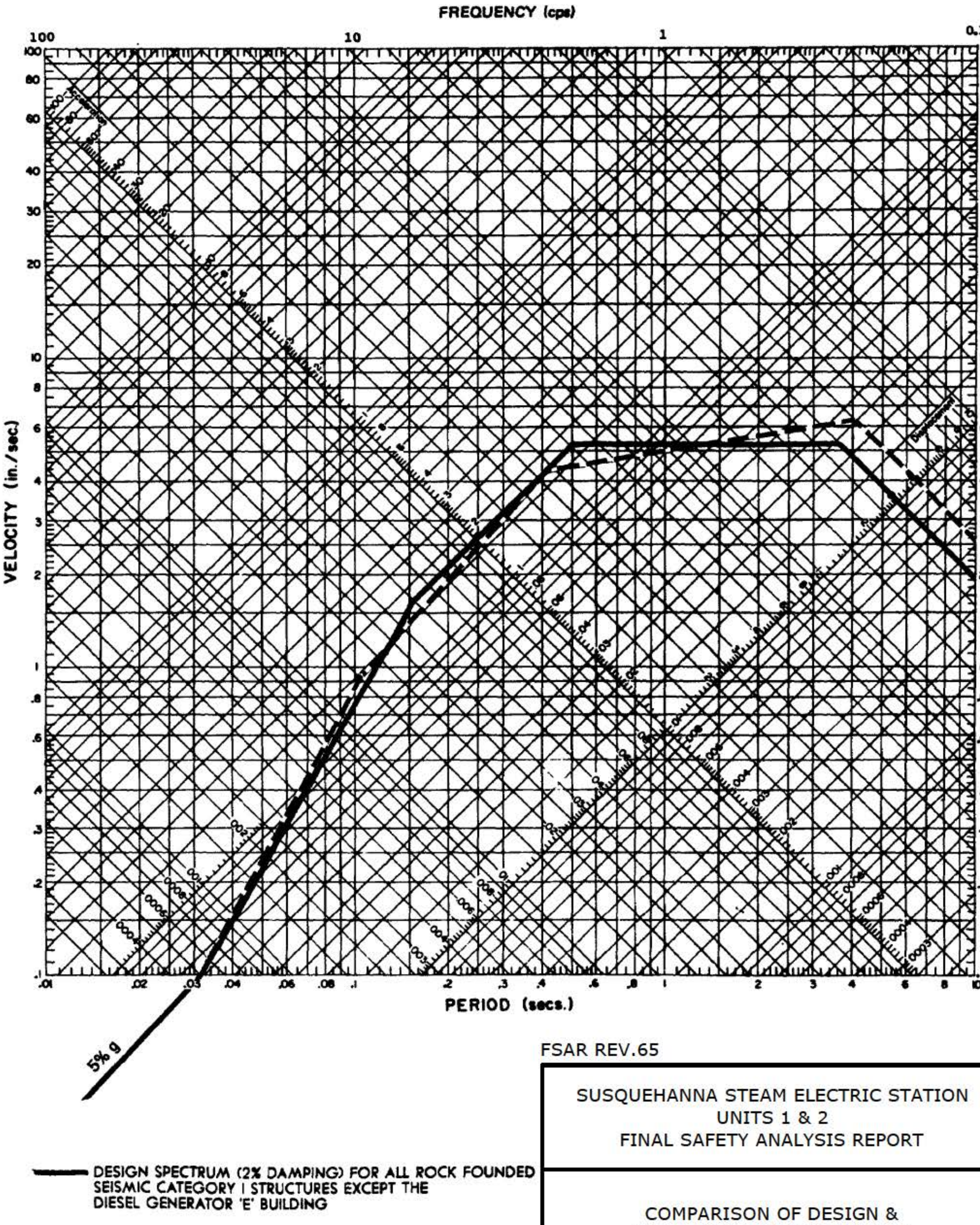
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

RESPONSE SPECTRUM
AT OPERATING FLOOR OF ESSW
PUMPHOUSE
VERTICAL SSE

FIGURE 3.7B-87, Rev. 55

Auto-Cad Figure Fsar 3_7B_87.dwg



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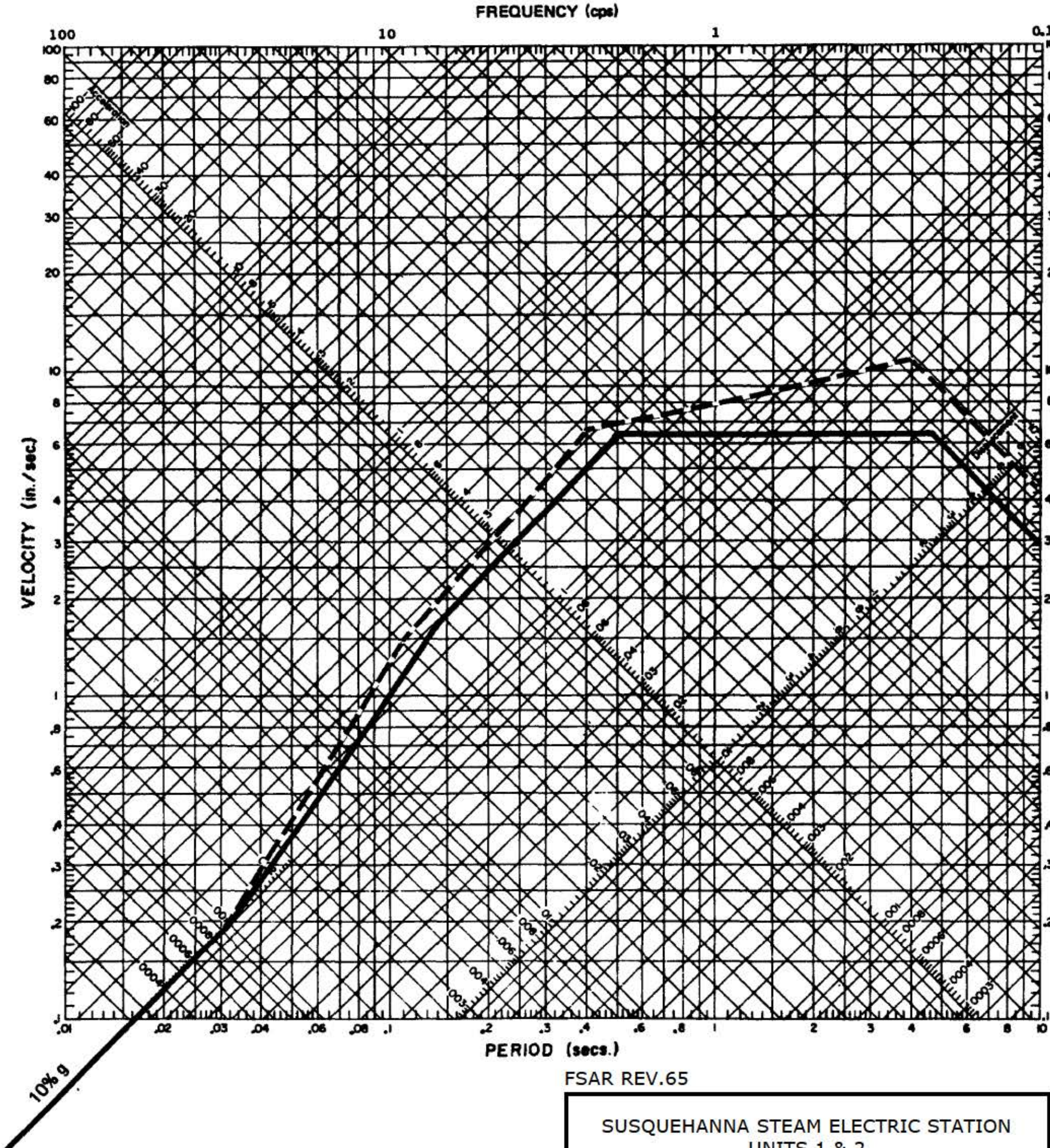
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

COMPARISON OF DESIGN &
R.G. 1.60 RESPONSE SPECTRA
HORIZONTAL OBE

FIGURE 3.7B-88, Rev. 55

Auto-Cad Figure Fsar 3_7B_88.dwg



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SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

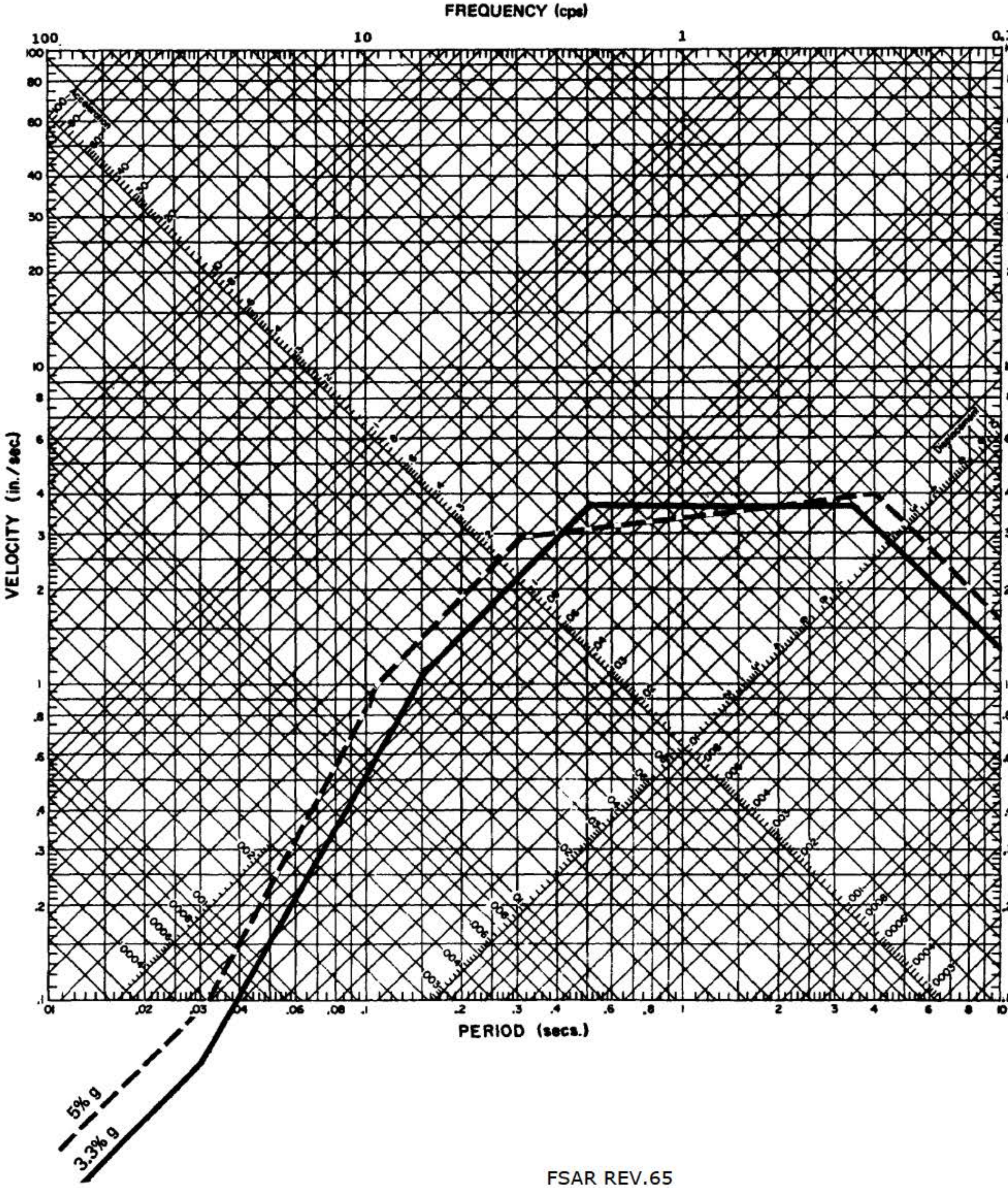
COMPARISON OF DESIGN &
R.G. 1.60 RESPONSE SPECTRA
HORIZONTAL SSE

FIGURE 3.7B-89, Rev. 55

Auto-Cad Figure Fsar 3_7B_89.dwg

DESIGN SPECTRUM (5% DAMPING) FOR ALL ROCK FOUNDED SEISMIC CATEGORY I STRUCTURES EXCEPT THE DIESEL GENERATOR 'E' BUILDING

R.G. 1.60 SPECTRUM (7% DAMPING) USED FOR THE DESIGN OF THE DIESEL GENERATOR 'E' BUILDING



FSAR REV.65

DESIGN SPECTRUM (2% DAMPING) FOR ALL ROCK FOUNDED SEISMIC CATEGORY I STRUCTURES EXCEPT THE DIESEL GENERATOR 'E' BUILDING

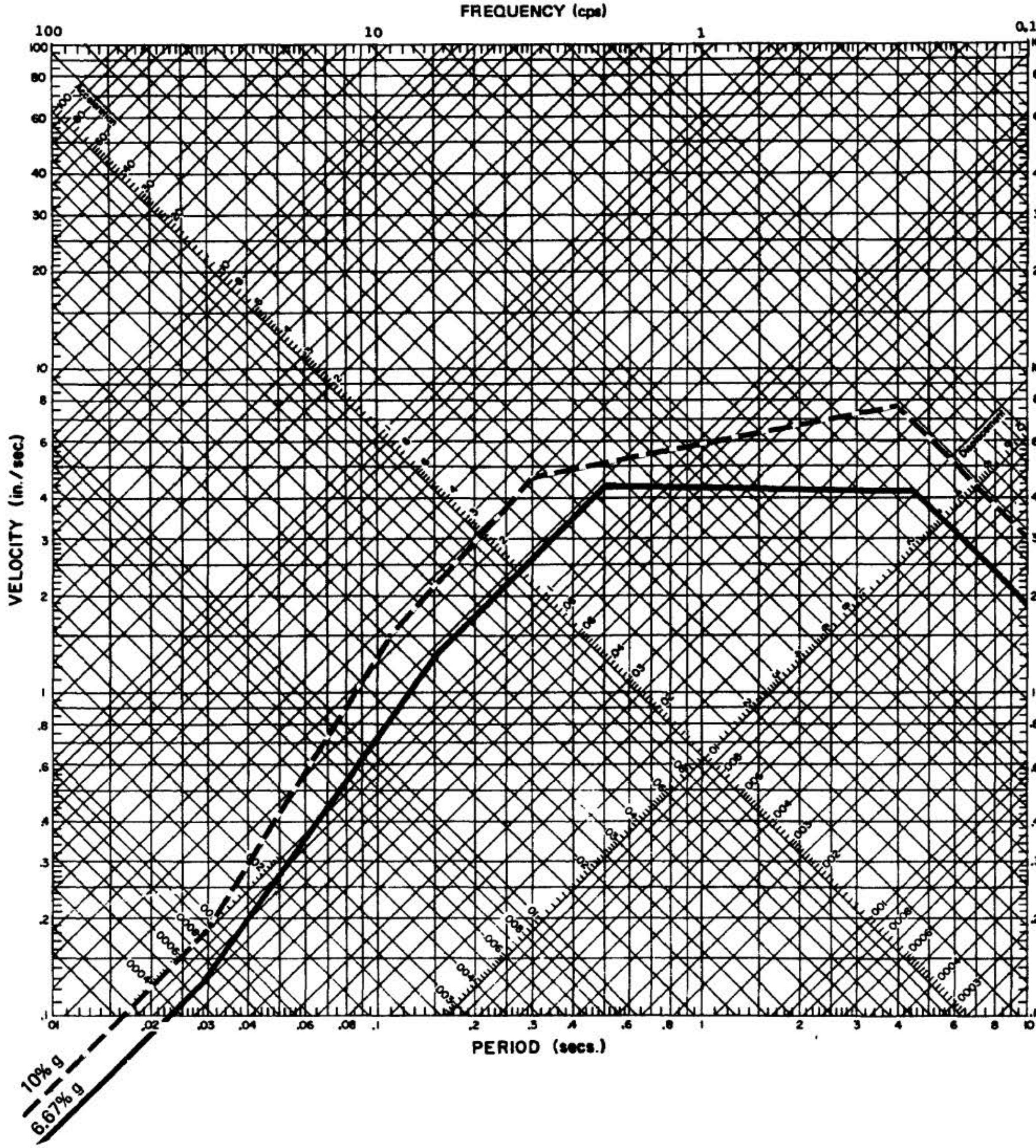
R.G. 1.60 SPECTRUM (4% DAMPING) USED FOR THE DESIGN OF THE DIESEL GENERATOR 'E' BUILDING

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

COMPARISON OF DESIGN &
R.G. 1.60 RESPONSE SPECTRA
VERTICAL OBE

FIGURE 3.7B-90, Rev. 55

Auto-Cad Figure Fsar 3_7B_90.dwg



DESIGN SPECTRUM (5% DAMPING) FOR ALL ROCK FOUNDED SEISMIC CATEGORY I STRUCTURES EXCEPT THE DIESEL GENERATOR 'E' BUILDING

R.G. 1.60 SPECTRUM (7% DAMPING) USED FOR THE DESIGN OF THE DIESEL GENERATOR 'E' BUILDING

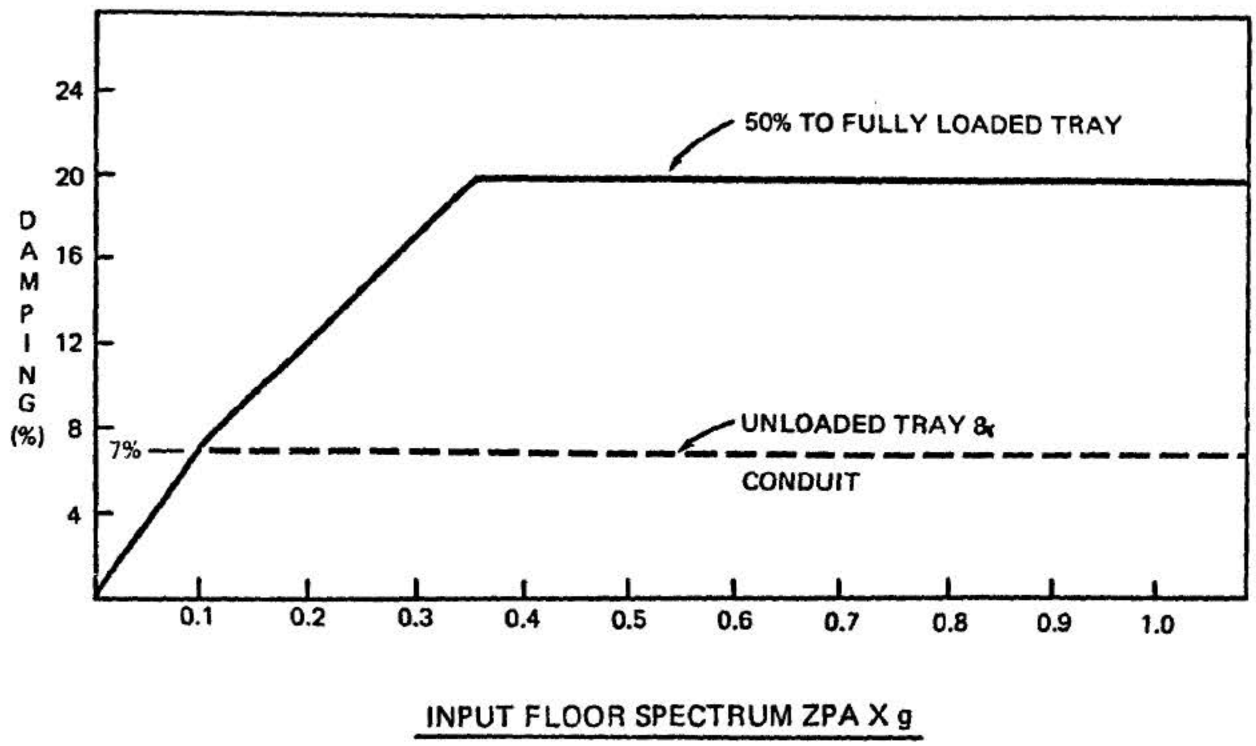
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

COMPARISON OF DESIGN &
R.G. 1.60 RESPONSE SPECTRA
VERTICAL SSE

FIGURE 3.7B-91, Rev. 55

Auto-Cad Figure Fsar 3_7B_91.dwg



SOURCE: REF. 3.7b-7

FSAR REV.65

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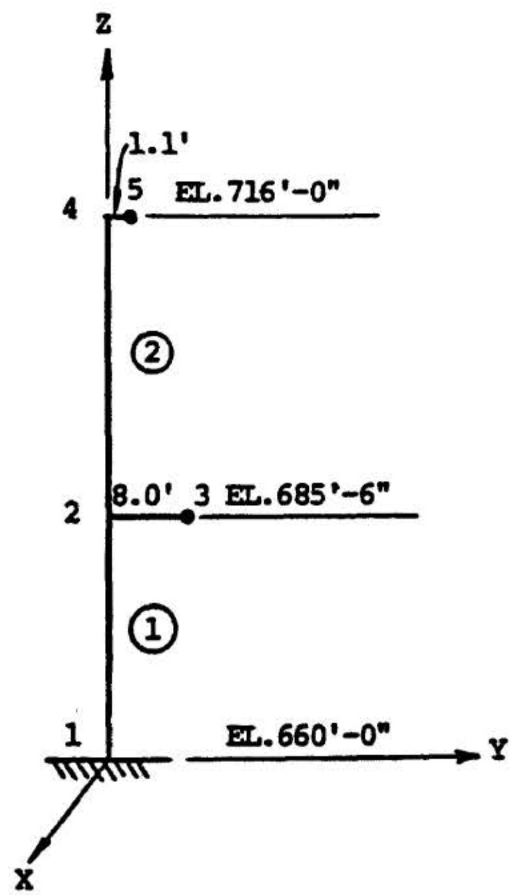
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

DAMPING V/S ZPA FOR
RACEWAY SYSTEM

FIGURE 3.7B-92, Rev. 55

Auto-Cad Figure Fsar 3_7B_92.dwg



COORDINATES			
NODES	X	Y	Z
1	0.0	0.0	660.0
2	0.0	0.0	685.5
3	0.0	8.0	685.5
4	0.0	0.0	716.0
5	0.0	1.1	716.0

MASSES AT NODES 3 AND 5

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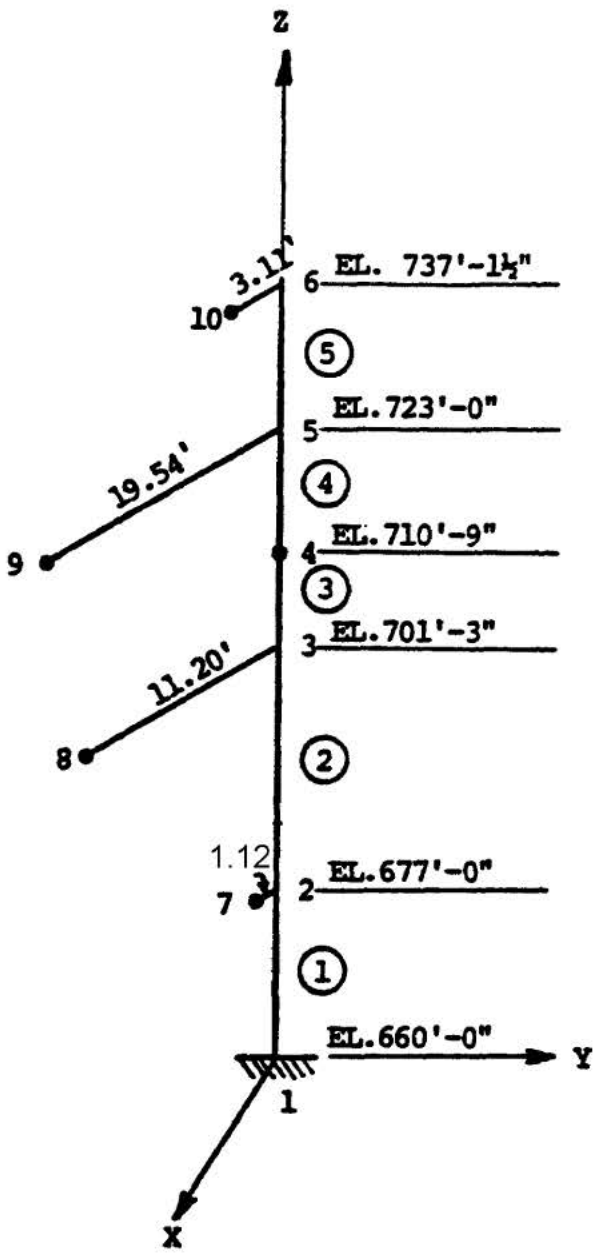
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

ESSW
PUMPHOUSE
3-D STICK MODEL

FIGURE 3.7B-93, Rev. 55

Auto-Cad Figure Fsar 3_7B_93.dwg



MASSES AT NODES 4,8,9,10

COORDINATES			
NODES	X	Y	Z
1	0.0	0.0	660.0
2	0.0	0.0	677.0
3	0.0	0.0	701.3
4	0.0	0.0	710.8
5	0.0	0.0	723.0
6	0.0	0.0	737.1
7	0.5	-1.0	677.0
8	8.9	-6.8	701.3
9	19.5	-1.3	723.0
10	0.3	-3.1	737.1

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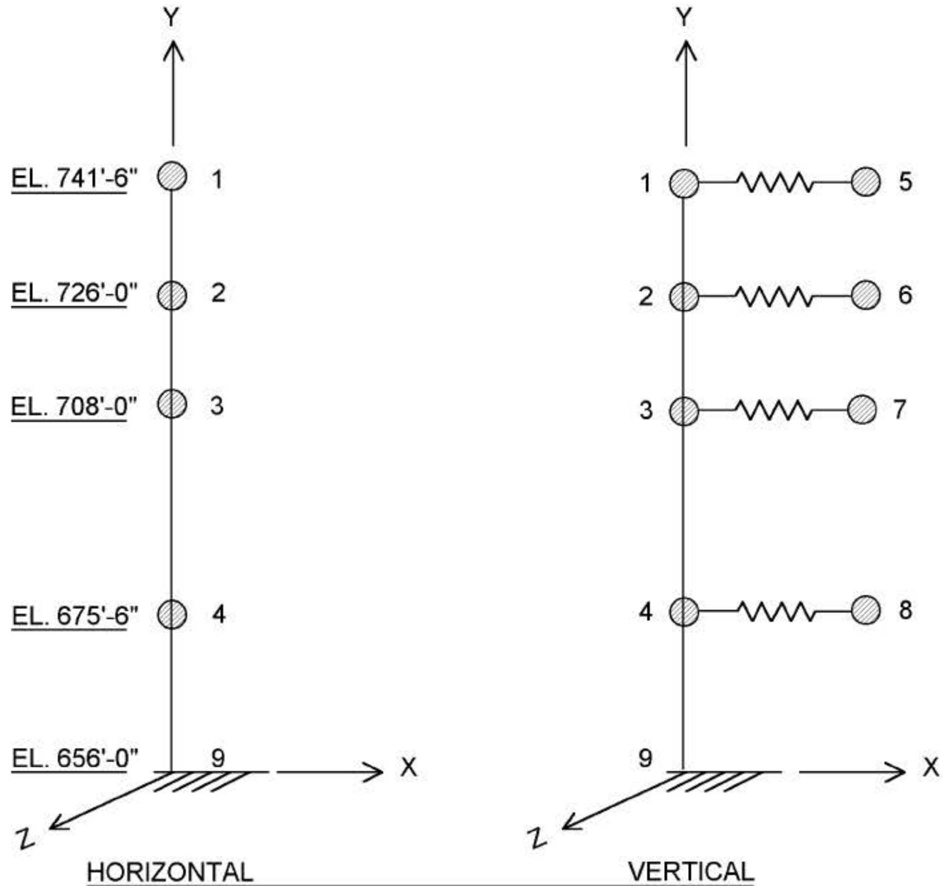
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

DIESEL GENERATOR
'A-D' BUILDING
3D STICK MODEL

FIGURE 3.7B-94, Rev. 56

Auto-Cad Figure Fsar 3_7B_94.dwg



NODES	COORDINATES (ft)		
	X	Y	Z
1.5	25.4	741.5	0
2.6	2.0	726.0	0
3.7	0	708.0	0
4.8	0.6	675.5	-1.0
9	0	656.8	0

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SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

DIESEL GENERATOR
'E' BUILDING
SEISMIC MODELS

FIGURE 3.7B-95, Rev. 55

Auto-Cad Figure Fsar 3_7B_95.dwg

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SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
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FIGURE 3.7B-96, Rev. 55

AutoCAD Figure 3_7B_96.doc

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SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
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FIGURE 3.7B-97, Rev. 55

AutoCAD Figure 3_7B_97.doc

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SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
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AutoCAD Figure 3_7B_98.doc

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SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
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AutoCAD Figure 3_7B_99.doc

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SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

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FIGURE 3.7B-100, Rev. 55

AutoCAD Figure 3_7B_100.doc

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SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
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FIGURE 3.7B-101, Rev. 55

AutoCAD Figure 3_7B_101.doc

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SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
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FIGURE 3.7B-102, Rev. 55

AutoCAD Figure 3_7B_102.doc

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SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

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FIGURE 3.7B-103, Rev. 55

AutoCAD Figure 3_7B_103.doc

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UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

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FIGURE 3.7B-104, Rev. 55

AutoCAD Figure 3_7B_104.doc

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UNITS 1 & 2
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FIGURE 3.7B-105, Rev. 55

AutoCAD Figure 3_7B_105.doc

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UNITS 1 & 2
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FIGURE 3.7B-106, Rev. 55

AutoCAD Figure 3_7B_106.doc

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UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

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FIGURE 3.7B-107, Rev. 55

AutoCAD Figure 3_7B_107.doc

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UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

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FIGURE 3.7B-108, Rev. 55

AutoCAD Figure 3_7B_108.doc

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UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

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FIGURE 3.7B-109, Rev. 55

AutoCAD Figure 3_7B_109.doc

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SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
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AutoCAD Figure 3_7B_110.doc

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SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

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FIGURE 3.7B-111, Rev. 55

AutoCAD Figure 3_7B_111.doc

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SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
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AutoCAD Figure 3_7B_112.doc

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SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

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FIGURE 3.7B-113, Rev. 55

AutoCAD Figure 3_7B_113.doc

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UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

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FIGURE 3.7B-114, Rev. 55

AutoCAD Figure 3_7B_114.doc

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UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

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FIGURE 3.7B-115, Rev. 55

AutoCAD Figure 3_7B_115.doc

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SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

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FIGURE 3.7B-116, Rev. 55

AutoCAD Figure 3_7B_116.doc

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SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
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FIGURE 3.7B-117, Rev. 55

AutoCAD Figure 3_7B_117.doc

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SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

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FIGURE 3.7B-118, Rev. 55

AutoCAD Figure 3_7B_118.doc

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SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

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FIGURE 3.7B-119, Rev. 55

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SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

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FIGURE 3.7B-120, Rev. 55

AutoCAD Figure 3_7B_120.doc

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SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

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FIGURE 3.7B-121, Rev. 55

AutoCAD Figure 3_7B_121.doc

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SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

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FIGURE 3.7B-122, Rev. 55

AutoCAD Figure 3_7B_122.doc

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SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

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FIGURE 3.7B-123, Rev. 55

AutoCAD Figure 3_7B_123.doc

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SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

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FIGURE 3.7B-124, Rev. 55

AutoCAD Figure 3_7B_124.doc

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SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

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FIGURE 3.7B-125, Rev. 55

AutoCAD Figure 3_7B_125.doc

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SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

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FIGURE 3.7B-126, Rev. 55

AutoCAD Figure 3_7B_126.doc

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RENUMBERED TO 3.7B-71

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

Figure renumbered from 3.7B-90A to 3.7B-71

FIGURE 3.7B-90A, Rev. 55

AutoCAD Figure 3_7B_90A.doc

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RENUMBERED TO 3.7B-73

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Figure renumbered from 3.7B-91A to 3.7B-73 FIGURE 3.7B-91A, Rev. 55

AutoCAD Figure 3_7B_91A.doc

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RENUMBERED TO 3.7B-75

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

Figure renumbered from 3.7B-92A to 3.7B-75

FIGURE 3.7B-92A, Rev. 55

AutoCAD Figure 3_7B_92A.doc

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RENUMBERED TO 3.7B-77

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

Figure renumbered from 3.7B-93A to 3.7B-77

FIGURE 3.7B-93A, Rev. 55

AutoCAD Figure 3_7B_93A.doc

THIS FIGURE HAS BEEN
RENUMBERED TO 3.7B-79

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

Figure renumbered from 3.7B-94A to 3.7B-79

FIGURE 3.7B-94A, Rev. 55

AutoCAD Figure 3_7B_94A.doc

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FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
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FIGURE 3.7B-95A, Rev. 55

AutoCAD Figure 3_7B_95A.doc

3.8 DESIGN OF CATEGORY I STRUCTURES

3.8.1 CONCRETE CONTAINMENT

The Susquehanna primary containments Units 1 and 2 are boiling water reactor, Mark II (over/under) types.

3.8.1.1 Description of the Containment

3.8.1.1.1 General

The primary containment is an enclosure for the reactor vessel, the reactor coolant recirculation loops, and other branch connections of the reactor coolant system. Essential elements of the primary containment are the drywell, the pressure suppression chamber that stores a large volume of water, the drywell floor that separates the drywell and the suppression chamber, the connecting vent pipe system between the drywell and the suppression chamber, isolation valves, the vacuum relief system, and the containment cooling systems and other service equipment.

The primary containment (as shown in Dwgs. C-331, Sh. 1, C-371, Sh. 2, C-1932, Sh. 3, C-1932, Sh. 4, and C-1932, Sh. 5) is in the form of a truncated cone over a cylindrical section, with the drywell in the upper conical section and the suppression chamber in the lower cylindrical section. These two sections comprise a structurally integrated reinforced concrete pressure vessel, lined with welded steel plate and provided with a steel domed head for closure at the top of the drywell. Connection of the drywell head to the top of the drywell wall is shown on Figure 3.8-9. The drywell floor is a reinforced concrete slab structurally connected to the containment wall as shown on Dwg. C-284, Sh. 1.

The primary containment is structurally separated from the surrounding reactor building except at the base foundation slabs where a cold joint between the two adjoining foundation slabs is provided.

3.8.1.1.1.1 Dimensions

The dimensions of the primary containment are as follows:

a) Inside Diameter

   1) Suppression chamber - 88 ft. 0 in.
   2) Base of drywell - 86 ft. 3 in.
   3) Top of drywell - 36 ft. 4 1/2 in.

b) Height

   1) Suppression chamber - 52 ft. 6 in.
   2) Drywell - 87 ft. 9 in.

c) Thickness

1) Base foundation slab - 7 ft. 9 in.

2) Containment wall - 6 ft. 0 in.

3.8.1.1.2 Base Foundation Slab

The containment base foundation slab is a 7 ft. 9 in. thick reinforced concrete mat. The top of the base foundation slab is lined with a carbon steel liner plate.

3.8.1.1.2.1 Reinforcement

The base foundation slab is reinforced with #18, Grade 60 rebar at top and bottom faces. The average rebar spacing is 18 in. Shear reinforcement consists of #8 and #9 vertical and inclined ties. Mechanical ("Cadweld") splices are used for splicing all main reinforcing bars. Dwg. C-332, Sh. 1 and C-333, Sh. 1 shows plan and section views of reinforcement.

3.8.1.1.2.2 Liner Plate and Anchorages

The steel liner plate is 1/4 in. thick and is anchored to the concrete slab by structural steel beams embedded in the concrete and welded to the plate. See Dwg. C-281, Sh. 1 for details of the liner plate and anchorages. All liner plate weld seams less than l/2 inch thick are provided with a leak chase system.

3.8.1.1.2.3 Pedestal and Suppression Chamber Column Base Liner Anchorages

Dwgs. C-281, Sh. 1 and C-370, Sh. 1 show the base foundation slab liner anchorages for the reactor pedestal and the suppression chamber columns, respectively. For the pedestal anchorage, B-series "Cadweld" sleeves are welded to the top and bottom surfaces of the thickened base liner to permit anchorage of the pedestal vertical rebar into the base foundation slab. Metal studs are welded to the top and bottom surfaces of the thickened base liner in order to transfer radial and tangential shear forces from the pedestal to the base foundation slab. For the suppression chamber column anchorage, pipe caps are welded to the thickened base liner, where the column anchor bolts penetrate the base liner, to ensure the leak-tight integrity of the base liner.

3.8.1.1.3 Containment Wall

The containment wall is a 6 ft. 0 in. thick reinforced concrete wall. The inside surface of the containment wall is lined with a carbon steel liner plate.

3.8.1.1.3.1 Reinforcement

The containment wall is reinforced with #18, Grade 60 rebar at inner and outer faces. The inner rebar curtain consists of two meridional layers and one hoop layer. The outer rebar curtain consists of one meridional layer, two hoop layers and two helical layers. Shear reinforcement consists of #6 horizontal and inclined ties. Mechanical ("Cadweld") splices are used for splicing all main reinforcing bars. Dwgs. C-334, Sh. 1, C-335, Sh. 1, C-336, Sh. 1, C-337, Sh. 1, C-338, Sh. 1, C-351, Sh. 1, C-352, Sh. 1, C-353, Sh. 1, C-354, Sh. 1, C-355, Sh. 1, C-356, Sh. 1, C-357, Sh. 1, C-358, Sh. 1, C-359, Sh. 1, C-360, Sh.1, C-393, Sh. 1, C-394, Sh. 1, C-395, Sh. 1, C-396, Sh. 1, C-397, Sh. 1, C-398, Sh. 1, C-399, Sh. 1, and C-400, Sh. 1 show section and developed elevation views of suppression chamber and drywell wall reinforcement, respectively.

3.8.1.1.3.2 Liner Plate and Anchorages

The steel liner plate is 1/4 in. thick and is anchored to the concrete wall by structural tee vertical stiffeners spaced horizontally every 2 ft. Horizontal plate stiffeners and horizontal structural channels spaced vertically every 5 ft. provide additional stiffening. See Dwgs. C-282, Sh. 1, and C-285, Sh. 1 for details of the liner plate and anchorages.

Around the containment liner plate penetrations, the liner is reinforced in accordance with ASME Boiler and Pressure Vessel Code, Section III, 1971 Edition. See Subsection 3.8.1.1.3.3 for a further description of penetrations.

Loads from internal containment attachments such as beam seats and pipe restraints are transferred directly into the containment concrete wall. This is accomplished by thickening the liner plate and attaching to it structural weldments to transfer to the concrete any type of load without relying on the liner plate or its anchorages. Where internal containment attachment loads are large, the structural weldments penetrate the liner plate rather than being welded to opposite sides of the liner plate. This was done to eliminate the possibility of lamellar tearing. Where internal containment attachment loads are small, e.g., pipe hangers, HVAC duct supports, electrical raceway supports, etc., the load is transferred by means of the liner plate into the anchorages which are embedded in the containment concrete. No additional structural weldments are provided for these small attachments, since the liner plate and anchorages are capable of supporting such loads. See Subsection 3.8.1.1.3.4 for a further description of internal containment attachments.

3.8.1.1.3.3 Penetrations

General

Services and communications between the inside and outside of the containment are performed through penetrations. Basic penetration types include the drywell head, access hatches (equipment hatches, personnel lock, suppression chamber access hatches, CRD removal hatch), pipe penetrations, and electrical penetrations. Penetrations consist of a pipe with a plate flange welded to it. The plate flange is embedded in the concrete wall and provides an anchorage for the penetration to resist normal operating and accident pipe reaction loads. The pipe is also welded to the containment liner plate to provide a leak-tight penetration.

Meridional and hoop reinforcement are bent around typical penetrations as shown on Dwgs. C-288, Sh. 1, C-287, Sh. 1, C-283, Sh. 1, and Figures 3.8-20-1 and 3.8-20-2. Additional local reinforcement in the hoop and diagonal directions is added at all large penetrations as shown on Dwgs. C-288, Sh. 1, C-287, Sh. 1, C-283, Sh. 1, and Figures 3.8-20-1 and 3.8-20-2. Local thickening of the containment wall at penetrations is generally not required. See Subsection 3.8.2.1.5 for a further description of penetrations.

Pipe Penetrations

Details of typical pipe penetrations are shown on Dwgs. C-288, Sh. 1, C-287, Sh. 1, and C-283, Sh. 1. There are two basic types of pipe penetrations. For piping systems containing high temperature steam or water, a sleeved penetration is furnished, thereby providing an air gap between the containment concrete wall and the hot pipe. This air gap is large enough to maintain the concrete temperature in the area of the penetration below 200°F. A flued head outside the containment connects the process pipe to the pipe sleeve. For piping systems containing low temperature water, an unsleeved penetration is furnished. For this type of penetration, the process pipe is welded directly to the pipe penetration.

Electrical Penetrations

Figure 3.8-20-1 and 3.8-20-2 shows a typical electrical penetration assembly used to extend electrical conductors through the containment. The assembly is sized to be inserted in the 12 in., Schedule 80 penetration nozzles that are furnished as part of the containment. The penetrations are hermetically sealed and provide for leak testing at design pressure.

Equipment Hatches and Personnel Lock

Two 12 ft. 2 in. I.D. equipment hatches are furnished in the drywell wall. One of these equipment hatches includes an 8 ft. 7 in. I.D. personnel lock. Dwg. C-351, Sh. 1, C-352, Sh. 1, C-353, Sh. 1, C-354, Sh. 1, C-355, Sh. 1, C-356, Sh. 1, C-357, Sh. 1, C-358, Sh. 1, C-359, Sh. 1, C-360, Sh. 1, C-393, Sh. 1, C-394, Sh. 1, C-395, Sh. 1, C-396, Sh. 1, C-397, Sh. 1, C-398, Sh. 1, C-399, Sh. 1, and C-400, Sh. 1 shows details of reinforcement around the equipment hatches. Additional meridional, hoop, helical, and shear reinforcement is provided to account for local stress concentrations at the opening. The shell is thickened at the equipment hatches to accommodate the additional rebars.

Drywell Head Assembly

The drywell head lower flange assembly is anchored to the top of the drywell wall by one-third (108) of the total number of meridional reinforcing bars in the inner curtain as shown on Figure 3.8-9.

Suppression Chamber Access Hatches

Two 6 ft. 0 in. I.D. access hatches are furnished in the suppression chamber wall. Figure 3.8-15-2 shows a detail of reinforcement around the suppression chamber access hatches. Additional local reinforcement in the meridional, hoop, and diagonal directions is added as shown on Dwg. C-335, Sh. 1.

3.8.1.1.3.4 Internal Containment Attachments

Drywell Floor Embedments

The drywell floor is attached to the containment wall by a structural weldment at the junction of the two structural components shown on Dwg. C-284, Sh. 1. Radial force and bending moment carried by the drywell floor main reinforcement is transferred to the containment wall by cadwelding the drywell floor rebar to the top and bottom flanges of the structural weldment. The top and bottom flanges of the structural weldment penetrate the thickened containment liner plate and are embedded deeply into the containment concrete wall. Flexural shear in the drywell floor is transferred to the containment wall through the web of the structural weldment, which is welded to opposite sides of the containment liner plate.

Beam Seat Embedments

Beam seats are provided to support the drywell platforms. A typical beam seat embedment is shown on Dwg. C-286, Sh. 1.

Pipe Restraint Embedments

Pipe restraints are provided to prevent pipe whip for all high energy piping systems. Typical pipe restraint embedments are shown on Dwg. C-291, Sh. 1.

Seismic Truss Embedments

The seismic truss provides lateral support for the reactor vessel. A typical seismic truss embedment in the drywell wall is shown on Dwg. C-286, Sh. 1.

Snubber Embedments

Snubbers dampen the vibratory motion of piping systems due to seismic or any other dynamic loading. A typical snubber embedment in the drywell wall is shown on Dwg. C-278, Sh. 1.

3.8.1.1.3.5 External Containment Attachments

There are no major external structural attachments. A 2 in. wide separation gap is provided between the containment and the surrounding reactor building to prevent interaction of the two structures. The only place where the containment is in contact with the reactor building is at the base foundation slabs where a cold joint between the two adjoining foundation slabs is provided.

3.8.1.1.3.6 Steel Components Not Backed by Structural Concrete

A description of steel portions of the containment that are not backed by concrete, such as the drywell head, equipment hatches, personnel lock, suppression chamber access hatches, CRD removal hatch, and piping and electrical penetrations, is given in Subsection 3.8.2.

3.8.1.2 Applicable Codes, Standards, and Specifications

The codes, standards, and specifications used in the design and construction of the containment are listed in Table 3.8-1 and given a reference number.
The reference numbers for the concrete containment are 10A, 12A, 1C, 2C, 3C, 6C and 2K.

The reference numbers for the liner plate and anchorages are 4C, 1H, 1J and 1K.

3.8.1.3 Loads and Loading Combinations

3.8.1.3.1 General

Table 3.8-2 lists the loading combinations used for the design and analysis of the containment. The loading combinations are in compliance with those given in Reference 12A of Table 3.8-1. The loading combinations shown in Table 3.8-2 do not include the hydrodynamic loads.

The containment has also been analyzed and designed for hydrodynamic loads from main steam safety/relief valve discharge and LOCA. For a definition of these loads and loading combinations including hydrodynamic loads, refer to GE's "Mark II Containment Dynamic Forcing Functions Information Report" (NEDO-21061), and the "Susquehanna Plant Design Assessment Report."

3.8.1.3.2 Description of Loads

Normal Loads: Those loads operation and shutdown, including dead loads, live loads, thermal loads due to operating temperature, and other permanent loads contributing stress such encountered during normal plant as hydrostatic loads. Dead and live loads are described in Subsection 3.8.1.3.2.1 and 3.8.I.3.2.2, respectively.

Severe Environmental Loads: Those loads sustained during severe environmental conditions, including those induced by the operating basis earthquake (OBE) and the design basis wind. Loads due to OBE are discussed in Section 3.7 and Subsection 3.8.1.3.2.6. Wind loads are discussed in Section 3.3.

Extreme Environmental Loads: Those loads sustained during extreme environmental conditions, including those induced by the safe shutdown earthquake (SSE) and the design basis tornado. Loads due to SSE are discussed in Section 3.7 and Subsection 3.8.1.3.2.6. Tornado loads are discussed in Section 3.3.

Abnormal Loads: Those loads sustained during abnormal plant conditions. Such abnormal plant conditions include the postulated rupture of high-energy piping. Loads induced by such an accident include elevated temperatures and pressures within or across compartments, and jet impingement and impact forces associated with such ruptures. Loads due to postulated rupture of piping are discussed in Section 3.6.

3.8.1.3.2.1 Dead Load

Dead load includes the weight of the structure plus any other permanent loads contributing stress, such as hydrostatic loads.

3.8.1.3.2.2 Live Load

Live load includes those loads expected to be present when the plant is operating, such as movable equipment, piping, cables, and lateral earth pressure.

3.8.1.3.2.3 Design Basis Accident Pressure Load

The design basis accident (DBA) is defined as a loss of coolant accident (LOCA) that produces the largest containment pressure. Transients resulting from the design basis accident are presented in Subsection 6.2.1 and serve as the basis for the containment internal design pressure of 53 psig.

3.8.1.3.2.4 Thermal Loads

The temperature gradients through the containment wall are shown on Figure 3.8-24 for the operating and the postulated design accident conditions. The design accident temperature gradient shown on Figure 3.8-24 occurs five minutes after LOCA. This transient temperature gradient is used for the design of the containment since it produces the largest stresses in the structure.

Thermal effects anticipated at the time of the structural acceptance test are insignificant because changes in temperature inside and outside the containment during the Unit 1 structural acceptance test were small. Therefore, thermal effects at the time of the structural acceptance test are insignificant.

3.8.1.3.2.5 Wind and Tornado Loads

Tornado depressurization load has an insignificant effect on the containment since the pressure value is much less than the DBA LOCA pressure. See Section 3.3 for a description of wind and tornado loads.

3.8.1.3.2.6 Seismic Loads

a) Loads from the Operating Basis Earthquake result from ground surface horizontal acceleration of 0.05 g, and vertical ground surface acceleration of 0.033 g, acting simultaneously.

b) Loads from the Safe Shutdown Earthquake result from ground surface horizontal acceleration of 0.10 g, and vertical ground surface acceleration of 0.067 g, acting simultaneously.

3.8.1.3.2.7 External Pressure Load

The containment shell is designed to withstand an external pressure of 5 psi differential.

3.8.1.3.2.8 Missile and Pipe Rupture Loads

The containment wall is designed to withstand the missile and pipe rupture loads due to a postulated rupture of a 26 in. diameter main steam pipe, which produces the largest loads on the containment wall. These loads include the effects of jet impingement, pipe whip, and pipe reaction. An equivalent static load of 1000 kips is considered. This load includes an appropriate dynamic load factor to account for the dynamic nature of the load. See Section 3.6 for a further discussion of postulated pipe rupture loads.

3.8.1.4 Design and Analysis Procedures

3.8.1.4.1 General

This subsection describes the procedures used for the design and analysis of the containment. The description does not include the effects of hydrodynamic loads from main steam safety/relief valve discharge and LOCA. For a description of the design and analysis procedures that consider the effects of hydrodynamic loads refer to GE's "Mark II Containment Dynamic Forcing Functions Report" (NEDO-21061) and the "Susquehanna Plant Design Assessment Report."

The analysis procedure consists of two parts. First, the uncracked forces, moments, and shears for both axisymmetric and non-axisymmetric loads are determined. Axisymmetric loads are dead load, live load, design accident pressure load, vertical seismic load, and operating and design accident thermal loads. Non-axisymmetric loads are horizontal seismic load and localized missile and pipe rupture load. The second part consists of taking into account the expected cracking of the concrete and determining the concrete and reinforcing steel stresses and strains. The liner plate is not considered to be a load resisting element for the containment wall or the base foundation slab.

The 3D/SAP computer program (Appendix 3.8A) is used to determine the uncracked forces, moments, and shears due to axisymmetric loads. The operating and design accident temperature gradients are computed using ME 620 computer program (Appendix 3.8A). For transient loads such as design accident pressure and thermal loads, the most critical combination of these loads is considered.

The forces, moments, and shears in the uncracked structure due to seismic loads are determined per Bechtel Topical Report BC-TOP-4-A (Ref. 2K of Table 3.8-1). The effect of variations in the values of structural and foundation parameters on the modal frequencies is considered. See Section 3.7 for a description of the containment seismic analysis. The 3D/SAP program is used to analyze the containment for non-axisymmetric loads due to missile and postulated pipe rupture.

The CECAP computer program (Appendix 3.8A) is used to determine the extent of concrete cracking and the concrete and rebar stresses and strains. The input data for the CECAP program consists of the uncracked forces, moments, and shears calculated by the 3D/SAP and seismic analysis programs. The CECAP program models a single element of unit height, unit width, and depth equal to the thickness of the wall or slab. The program assumes isotropic, linear elastic material properties and uses an iterative technique to obtain stresses considering their redistribution due to cracking. The program determines the redistribution of thermal stresses due to the relieving effect of concrete cracking.

3.8.1.4.2 Containment Wall

Figure 3.8-25 shows the 3D/SAP finite element model used to analyze the containment wall for axisymmetric loads. A 10 degree wedge of the containment is modeled using solid finite elements having linear elastic, isotropic material properties. The model includes the containment wall, base foundation slab, drywell floor, reactor pedestal and the foundation material. Boundary conditions are imposed on the analytical model by specifying nodal point forces or displacements. Referring to Figure 3.8-25, the nodal points lying along Boundary A are allowed to move within the X-Z plane, and Boundary B within the X-Y plane. Points along Boundary C are prevented from moving in the radial direction and points along Boundary D are prevented from moving in the hoop direction. Nodal forces, moments, and shears are applied to Boundaries E and F to account for reaction loads from the drywell head and reactor vessel and reactor shield wall, respectively.

Figure 3.8-26 shows the 3D/SAP finite element model used to analyze the drywell wall for non-axisymmetric missile and pipe rupture loads. A 180 degree half model of the drywell wall consisting of linear elastic, isotropic, solid finite elements is used. Referring to Figure 3.8-26, the nodal points lying along Boundary A are allowed to move within the X-Z plane. Points along Boundary B are prevented from moving in the vertical and radial directions. Nodal forces, moments, and shears are applied to Boundary C to account for reaction loads from the drywell head.

Tangential shears caused by seismic loads are totally resisted by helical reinforcing bars and concrete. No tangential shear is taken by the concrete. The tangential shear is considered as diagonal tension and compression components. The helical reinforcing bars resist diagonal tension and the concrete resists diagonal compression. In calculating the reinforcing steel requirement, the helical reinforcement is designed to resist stresses due to design accident pressure and thermal loads as well as tangential shears caused by seismic loads.

3.8.1.4.3 Base Foundation Slab

Figure 3.8-27 shows the 3D/SAP finite element model used to analyze the base foundation slab. A 180 degree half model of the base foundation slab consisting of linear elastic, isotropic, solid finite elements is used. The model includes the base foundation slab, a portion of the containment wall and the foundation material. Referring to Figure 3.8-27, the nodal points lying along Boundary A are allowed to move within the X-Z plane, and Boundary B within the X-Y plane. Points along Boundary C are prevented from moving in the radial direction. Axisymmetric forces, moments, and shears calculated using the 3D/SAP containment model and seismically-induced, tangential shears are applied to Boundary D. The height of the model is chosen so that the overturning moment caused by the tangential shear is the same as the overturning moment determined by the seismic analysis. In order to be able to consider uplifting of the base foundation slab from its foundation, a thin layer of foundation material is provided immediately beneath the foundation slab. If the computer output indicates tension in any of these thin foundation elements, the modulus of elasticity of these elements is reduced to almost zero. Then a second computer run is made and any additional uplift is identified. Further iterations and modifications of foundation material properties are made until the complete extent of uplift is determined. Uplift does not result in overstressing the containment foundation.

3.8.1.4.4 Analysis of Areas Around Equipment Hatches

Figure 3.8-28 shows the 3D/SAP finite element model used to analyze the areas of the containment wall around the equipment hatches. A 60 degree wedge of the containment wall is modeled using solid finite elements having linear elastic, isotropic material properties. To reduce the size of the analytical model, Boundary A follows the vertical plane of symmetry of the equipment hatch. The points delineating the outermost boundaries of the model are located at a sufficient distance from the opening so that the behavior of the model along the boundaries is compatible with that of the undisturbed shell. Referring to Figure 3.8-28, the nodal points lying along Boundary A are allowed to move within the X-Z plane, and Boundary B within the X-Y plane. Points along Boundary C are prevented from moving in the hoop direction. Axisymmetric forces, moments, and shears calculated using the 3D/SAP containment model are applied to Boundary D. Seismic loads calculated by the seismic analysis are applied locally to the elements. Seismically induced, tangential shears around the equipment hatches are resisted by helical reinforcing bars and concrete in compression.

3.8.1.4.5 Liner Plate and Anchorages

The design and analysis of the liner plate and anchorages is per Bechtel Topical Report BC-TOP-1 (Ref. 1K of Table 3.8-1). The analysis of the liner plate and anchorages for small attachment loads is done using membrane theory for the liner plate and the theory of beams on elastic foundations for the anchorages.

3.8.1.5 Structural Acceptance Criteria

3.8.1.5.1 Reinforced Concrete

3.8.1.5.1.1 Working Stress

The preoperational testing condition listed in Table 3.8-2 is designed according to the stress limitations of ACI 318, Section 8.10 except that the maximum permissible tensile stress for reinforcement shall be 0.5 Fy. This criterion conforms to Reference 12A of Table 3.8-1.

Since the temporary construction live load on the containment during and after construction is small, it did not govern the containment design.

The containment was not analyzed for a "normal/extreme environmental" load combination. However, the containment was analyzed for a "normal/severe" load combination with a load factor of 1.425 on the OBE load. Since the SSE loads for Susquehanna SES exceed the OBE loads by only approximately 35%, the "normal/severe" load combination that was considered is more critical than a "normal/extreme" load combination with a load factor of 1.0 on the SSE load. Therefore, the "normal/extreme" load combination was not investigated. Also, the "abnormal/extreme" condition is critical compared to the "normal/extreme" condition.

Table 3.8-2 used working stress criteria for the "preoperational testing" condition. As discussed above, the "construction" load combination was not considered as it did not govern the design. For the "normal/severe" load combination, Table 3.8-2 used ultimate strength design (USD) as opposed to Table CC-3200-1 of the ASME Code which uses working stress design (WSD). A comparison of the OBE load factors and the allowable reinforcing steel tensile stresses is given below.

		OBE Load Factor	Allowable Reinforcing Steel Tensile Stress	
USD		1.425	0.9	fy
WSD		1.0	0.67	fy*
Ratio	USD / WSD	1.425	1.343	WSD

*Includes a 33% increase per Subsection CC-3422.1 of the ASME "Proposed Standard Code for Concrete Reactor Vessels and Containments," April 1973 edition since load combination includes temperature loads.

A comparison of allowable concrete compressive stresses is unnecessary since concrete compressive stresses are low and do not govern the design.

Since the USD load combination uses a 42.5% higher seismic load but allows only 34.3% higher reinforcing steel stress, the USD load combination is slightly more conservative than the WSD load combination.

3.8.1.5.1.2 Strength Method

The factored load combinations listed in Table 3.8-2 are designed according to the strength method of ACI 318. The following allowable stresses are used:

a) Concrete

   1) Compression - 0.85 f'c

   2) Tension - not permitted

   3) Radial shear - ACI 318-71 (Chapter 11)

   4) Tangential shear - not permitted

b) Reinforcing Steel

   1) Tension - 0.90 Fy

   2) Compression - 0.90 Fy

The allowables are defined as:

f'c	=	Specified compressive strength of concrete
Fy	=	Specified yield strength of reinforcing steel

3.8.1.5.2 Liner Plate and Anchorages

The allowable strain in the liner plate due to design basis accident thermal load is 0.5 percent. This value is based on ASME Code, Section III (Ref. 1J of Table 3.8-1), Figure I-9.I which permits an allowable strain of approximately 2 percent for 10 cycles. Since the graph in Figure I-9.I does not extend below 10 cycles, 10 cycles are conservatively used for the DBA instead of one cycle.

The liner plate and anchorages are also used to support small loads from pipe hangers, HVAC duct supports, electrical raceway supports, etc. For this condition, the following allowable stresses are used:

Loading Condition	Allowable Membrane Tensile Stress Due to Mechanical Loads
Normal	0.6 Fy
Abnormal	0.9 Fy

The allowables are defined as:

Fy = Specified yield strength of liner plate.

The allowable forces on the liner plate anchorages are in accordance with Bechtel Topical Report BC-TOP-1 (Ref. 1K of Table 3.8-1).

3.8.1.6 Materials, Quality Control, and Special Construction Techniques

3.8.1.6.1 Concrete Containment

The concrete and reinforcing steel materials for the containment are discussed in Appendix 3.8B. Concrete design compressive strengths are given in Table 3.8-11.

3.8.1.6.2 Liner Plate, Anchorages, and Attachments

3.8.1.6.2.1 Materials

Liner plate materials conform to the requirements of the standard specifications listed below:

Item	Specification
Liner plate (less than 1/2 in. thick)	ASTM A 285, Grade A
Liner plate (1/2 in. thick or thicker)	ASME SA-516, Grade 60 or 70 conforming to the requirements of ASME Boiler and Pressure Vessel Code (ASME B&PV Code), 1971 Edition with Addenda through Summer 1972, Section III, Article NE-2000
Anchorages and attachments other than pipe restraints	ASTM A36
Pipe restraint attachments	ASTM A441

3.8.1.6.2.2 Welding

Liner plate and structural steel welding conform to the applicable portions of Part UW of Section VIII of the ASME B&PV Code. Specifically, Paragraph UW-26 through UW-38 inclusive apply in their entirety. The welding of liner plate butt welds and attachments that penetrate the liner plate is performed by either the shielded metal arc or the automatic submerged arc process. The minimum number of individual weld layers for welds that must maintain leak-tightness is two. Welders and weld procedures are qualified in accordance with either Section IX of the ASME Code or AWS D1.1.

3.8.1.6.2.3 Materials Testing

Liner plate material 3/4 in. thick or over is impact tested at 0^oF or below as required by the ASME Code. Liner plate or attachment material subjected to transverse tensile stress is vacuum degassed and ultrasonically tested in accordance with ASME Code, Section III, NB-2530 and conforms to the requirements of Article NE-2000 of Section III.

3.8.1.6.2.4 Nondestructive Examination of Liner Plate Seam Welds

Nondestructive examination of liner plate welds is performed in accordance with Regulatory Guide 1.19, Revision 1 except that for leak chase testing, the leak chase pressure is 115 percent of design pressure instead of 100 percent of design pressure, and the pressure is held for 15 minutes instead of two hours. This exception is considered justifiable since any significant leakage (i.e., any pressure decay in excess of the rated accuracy of the pressure gage) will be determined within 15 minutes.

Spot radiographic examination is performed for all radiographable liner plate seam welds. Radiography is performed in accordance with Section V, Article 3 of the ASME Code. Personnel performing radiographic examinations are qualified in accordance with the Society for Non-Destructive Testing's Recommended Practice No. SNT-TC-1A, Supplement A, plus any additional requirements of the ASME Code, Section V. Acceptance standards are in accordance with Paragraph UW-51, of Section VIII, Division 1 of the ASME Code. The first 10 ft. of weld for each welder and welding position is 100 percent radiographed. Thereafter, one 12 in. long radiograph is taken for each welder and weld position in each additional 50 ft. increment of weld. A minimum of 2 percent of all liner seam welds are examined by radiography. For nonradiographable welds, the length of weld needed to meet the 2 percent requirement is accounted for by additional radiographs of that length for the accessible welds.

Where nonradiographable weld joints are used, the entire length of weld is magnetic particle examined. All magnetic particle examinations conform to the ASME Code, Section V. Personnel performing magnetic particle examinations are qualified in accordance with SNT-TC-1A plus any additional requirements of the ASME Code, Section V. Acceptance standards are in accordance with the ASME Code, Section VIII, Division 1, Appendix VI. The vacuum box soap bubble test is performed on all accessible liner plate weld seams. A 5 psi minimum pressure differential is maintained for a minimum time of 20 seconds. The leak detecting solution is continuously observed for bubbles that indicate leaks. If a leak is detected, the defective weld is repaired and reinspected by vacuum box testing.

Welds that are inaccessible for vacuum box testing are 100 percent liquid penetrant tested. Liquid penetrant examinations conform to the ASME Code, Section V. Personnel performing liquid penetrant examinations are qualified in accordance with SNT-TC-1A plus any additional requirements of the ASME Code, Section V. Acceptance standards conform to the ASME Code, Section VIII, Division 1, Appendix VIII.

A leak chase system is provided on liner plate seam welds less than 1/2 in. thick on the base foundation slab liner plate and on that portion of the suppression chamber wall liner plate that is below the suppression pool water level. This system will allow periodic leak testing of welds that are submerged in the suppression pool. It also provides a secondary leak-tight barrier at the liner plate weld seams. Following installation of the leak chase system, the leak chase system is pressurized to 63 psig. The pressure is monitored by valving off the air supply and measuring any pressure decay with a pressure gage. Any pressure decay in excess of the rated accuracy of the pressure gage within 15 minutes is cause for rejection of that portion of the liner plate seam welds and the leak chase system. Any leaks are repaired, and following repair, the affected portion of the leak chase system is retested.

3.8.1.6.2.5 Quality Control

Quality control requirements are discussed in Appendix D and amendments to the PSAR for the construction phase.

3.8.1.6.2.6 Erection Tolerances

The specified erection tolerances for the liner plate are as follows:

a) The slope of any 10 ft. section of cylindrical liner plate, referred to true vertical, does not exceed 1:180. The deviation from theoretical slope of any 10 ft. section of conical liner plate, measured within a vertical plane, does not exceed 1:120.

b) The cylindrical shell is plumb within 1/400 of the height. The vertical axis of the conical shell, as established at the top and bottom of the conical section, is plumb within 1/400 of the height.

c) The radial dimension to any point on the liner plate does not vary from the design radius by more than $\pm$1 in., and at any given elevation the maximum diameter minus the minimum diameter shall not exceed 4 in., except that there is a radial tolerance of $\pm$2 in. for local out-of-roundness. Radial measurements are taken at 24 locations spaced equally around the containment at any elevation. Local out-of-roundness tolerance is used for not more than two measurements at any given elevation and is not used at adjacent measurements.

d) Plates joined by butt welding are matched accurately and retained in position during the welding operation. Misalignment in completed joints shall not exceed the requirements of Paragraph UW-33 of Section VIII, Division 1 of the ASME Code.

e) The levelness of anchorages placed in the base foundation slab is within -1/4 in. of the theoretical elevation over the entire area, plus a local tolerance of -1/8 in. in any 30 ft. length.

Actual deviations from the above were handled in accordance with the procedures covered in Subsection 3.8.1.6.2.5.

3.8.1.7 Testing and In-service Surveillance Requirements

3.8.1.7.1 Preoperational Testing

3.8.1.7.1.1 Structural Acceptance Test

This subsection briefly describes the Unit 1 containment structural acceptance test. For a more detailed description, refer to the "SSES, Unit 1 Containment Structure, Structural Integrity Test Report."

The Unit 1 containment structural acceptance test was performed after completion of the containment structure but prior to installation of piping and equipment. The reactor vessel was installed at the time of the test and the suppression chamber was filled with water to the normal level. The Unit 2 containment structural acceptance test will be performed after completion of the containment including all piping and equipment. The Unit 1 test was a prototype test and, therefore, internal concrete strains were measured. The Unit 2 test will be a non-prototype test and, therefore, internal concrete strains will not be measured.

The Unit 1 test was done and the Unit 2 test will be done in accordance with Regulatory Guide 1.18, Revision 1, except for the following:

a) A continuous increase in containment pressure, rather than incremental pressure increases, was used. This is considered justifiable since data observations at each pressure level were made rapidly. Rapidly is defined as requiring a time interval for the data point sample sufficiently short so that the change in pressure during the observation would cause a change in structural response of less than five percent of the total anticipated change. Also, the maximum rate of pressurization was limited to 3 psi/hr to ensure that the structure would respond to the pressure load without any time lag.

b) The distribution of measuring points for monitoring radial deflections was selected so that the as-built condition could be considered in the assessment of the general shell response. In general, the locations of measuring points for radial deflections was in agreement with Regulatory Guide 1.18, Figure B, except point 1. Point 1 was provided at a distance of two times the wall thickness (12 ft) above the base mat. This variation was made to properly predict the containment behavior near the base mat to wall connection. If point 1 was provided at a height of three times the wall thickness (18 ft.), it would be located close to point 2 (suppression chamber wall mid-height is 26 ft.) and would not yield any additional behavior pattern of the containment.

c) Some of the strain gage instrumentation was farther from the equipment hatch than 0.5 times the wall thickness (3 ft.) as required by Regulatory Guide 1.18, Paragraph C.5. This was required in order to clear reinforcement and is considered justifiable since the intent of the Regulatory Guide, i.e., to demonstrate the structural integrity of the containment, was met.

d) Tangential deflections of the containment wall adjacent to the equipment hatch were not measured because the predicted values of tangential deflection were small and it would have been difficult to obtain fixed reference points for measurement of local tangential deflections.

e) Triaxial concrete strain measurements were not used to evaluate the concrete strain distribution because the measured strain values could not be properly interpreted. The difficulty in interpreting the data was due to the large size of the strain gages relative to the wall thickness. The concrete strain was evaluated using linear strain measurements in the meridional and hoop directions.

f) Humidity inside the containment was not measured during the test since it does not affect the response of the structure.

The containment was pneumatically pressurized to 1.15 times the design accident pressure as shown on Figure 3.8-29. The drywell floor was tested to 1.15 times the design downward differential pressure.

Structural measurements were taken at peak pressure and peak differential pressure as well as at intermediate stages. Measured structural data include the following:

1) Radial and vertical deflections of the containment

2) Internal concrete strains

3) External concrete surface cracks.

The above data were measured for the containment and for the largest opening which are the two equipment hatches. Since the areas of the containment wall around the equipment hatches are of identical design, only one of the hatches was instrumented. See Figures 3.8-30 and 3.8-31 for the locations of deflection measuring devices for the containment and the equipment hatch, respectively. See Dwg. C-384, Sh. 1 for the location of strain gage instrumentation for the containment and the equipment hatch, respectively. Strain gages were located within the walls and slabs at the rebar layers in the direction of the main reinforcement. An inspection of external concrete surface cracks was performed at six locations. Each crack inspection area was at least 40 sq. ft. Dwg. C-387, Sh. 1 shows the locations of the crack mapping areas.

Deflections and strains were calculated prior to the test. A 15 percent margin was added to the calculated values of deflection and strain to arrive at the predicted values. The FINEL computer program (Appendix 3.8A) was used to calculate the deflections and strains for the containment. The program performs a finite element, static analysis of axisymmetric structures with axisymmetric loading. Special material properties that can be considered include bilinearity in compression and bilinearity or cracking in tension. Figure 3.8-34 shows a vertical section through the model. Points along Boundary A are prevented from moving in the vertical direction and points along Boundary B are prevented from moving in the radial direction. Concrete, reinforcing steel, and liner plate materials are included in the model. The SUPERB computer program (Appendix 3.8A) was used to calculate the predicted deflections and strains for the equipment hatch. Figure 3.8-35 shows the analytical model of the equipment hatch. Shell elements are used to represent the containment wall around the equipment hatch and the drywell floor. Points along Boundary A are allowed to move within the X-Z plane, and Boundary B within the X-Y plane. Points along Boundary C are prevented from moving in the hoop direction, and points along Boundary D are prevented from moving in the radial direction. Nodal forces, moments, and shears are applied to Boundary E to account for the reaction loads from the upper portion of the drywell wall.

Deflections and strains measured during the test were less than or equal to the predicted values at all critical locations. Thus, the design of the containment provides an adequate safety margin against internal pressure. Figure 3.8-36 shows a comparison between measured and predicted deflections for the containment at peak pressure. Figure 3.8-37 shows a comparison between measured and predicted deflections for the equipment hatch at peak pressure. The maximum strain occurs at mid-height of the suppression chamber wall. Figures 3.8-38, 3.8-39, 3.8-40, 3.8-41 and 3.8-42 compare measured and predicted strains at this location. Very little concrete cracking was observed. Figure 3.8-43 shows the cracks mapped at mid-height of the drywell wall where the greatest amount of concrete surface cracks were observed.

3.8.1.7.1.2 Leak Rate Testing

Preoperational leak rate testing is discussed in Subsection 6.2.6.

3.8.1.7.2 In-service Leak Rate Testing

In-service leak rate testing is discussed in Subsection 6.2.6.

3.8.2 ASME CLASS MC STEEL COMPONENTS OF THE CONTAINMENT

This subsection pertains to the ASME Class MC steel components of the concrete containment that form a portion of the containment pressure boundary and are not backed by structural concrete. These components include the drywell head assembly, the equipment hatches and personnel lock, the suppression chamber access hatches, the CRD removal hatch, and piping and electrical penetrations.

3.8.2.1 Description of the ASME Class MC Components

3.8.2.1.1 Drywell Head Assembly

The drywell head provides a removable closure at the top of the containment for reactor access during the refueling operation. The drywell head assembly consists of a 2:1 hemi-ellipsoidal head and a cylindrical lower flange. The lower flange is supported on the top of the drywell wall as shown on Figure 3.8-9. The head is made of 1-1/2 in. thick plate and is secured with 80 2-3/4 in. diameter bolts at the 4 in. thick mating flange. Double rubber gaskets are provided at the head-to-lower flange connection to permit local leakage testing of the gaskets. The inside diameter (ID) of the drywell head at the mating flange is 37 ft. 7-1/2 in.

A 24 in. diameter double-gasketed manhole is provided in the drywell head.

Figure 3.8-44 shows details of the drywell head assembly.

3.8.2.1.2 Equipment Hatches and Personnel Lock

Two 12 ft. 2 in. ID equipment hatches are furnished in the drywell wall to permit the transfer of equipment and components into and out of the drywell. One hatch is furnished with a double-gasketed flange and a bolted dished door. The other hatch is furnished with a double-gasketed flange and a bolted personnel lock. The personnel lock is an 8 ft. 7 in. ID cylindrical pressure vessel with inner and outer flat bulkheads. Interlocked, double-gasketed doors are furnished in each bulkhead. A quick-acting, equalizing valve vents the personnel lock to the drywell to equalize the pressure in the two systems when the doors are opened and then closed. The two doors in the personnel lock are mechanically interlocked to prevent them from being opened simultaneously and to ensure that one door is closed before the opposite door can be opened. The personnel lock has an ASME Code N-stamp. See Dwg. C-287, Sh. 1 for details of the equipment hatch and the equipment hatch with personnel lock, respectively.

3.8.2.1.3 Suppression Chamber Access Hatches

Two 6 ft. 0 in. ID access hatches are furnished in the suppression chamber wall to permit personnel access and the transfer of equipment and components into and out of the suppression chamber. Each hatch is furnished with a double-gasketed flange and a bolted flat cover. See Dwg. C-283, Sh. 1 for details of the suppression chamber access hatches.

3.8.2.1.4 CRD Removal Hatch

One 3 ft. 0 in. ID CRD removal hatch is furnished in the drywell wall to permit transfer of the control rod drive assemblies into and out of the drywell. The hatch is furnished with a double-gasketed flange and a bolted flat cover. See Dwg. C-288, Sh. 1 for details of the CRD removal hatch.

3.8.2.1.5 Penetrations

The entire length of any penetration sleeve is considered an MC component and, as such, is designed in accordance with Subsection NE of the ASME B&PV Code, Section III. See Subsection 3.8.1.1.3.3 for a description of the containment penetrations. Dwgs. C-288, Sh. 1 and C-283, Sh. 1 and Figure 3.8-20 show details of typical pipe and electrical penetrations, respectively.

3.8.2.2 Applicable Codes, Standards, and Specifications

The codes, standards, and specifications used in the design and construction of the containment are listed in Table 3.8-1 and given a reference number.

The reference numbers for the ASME Class MC components are 7C, 1H, 1J, and 1K.

3.8.2.3 Loads and Loading Combinations

3.8.2.3.1 General

Table 3.8-3 lists the loading combinations used for the design and analysis of the ASME Class MC components. The loading combinations comply with Regulatory Guide 1.57. The loading combinations shown in Table 3.8-3 do not include the hydrodynamic loads.

The ASME Class MC components have also been analyzed for hydrodynamic loads from main steam safety/relief valve discharge and LOCA. For a definition of loads and loading combinations including hydrodynamic loads, refer to GE's "Mark II Containment Dynamic Forcing Functions Information Report" (NEDO-21061), and the "Susquehanna Plant Design Assessment Report."

The loading combinations given in Table 3.8-3 are not in agreement with those of SRP Section 3.8.2.II.3.b. Table 3.8-3 does, however, base allowable stresses on Subsection NB of the ASME code. Table 3.8-3a compares FSAR and SRP load combinations and allowable stresses for ASME Class MC components. The principal material for the MC components is SA-516, Grade 70. The allowable stresses listed in Table 3.8-3a are based on the following values:

Sm = 19.3 ksi

Sy = 38.0 ksi for $T \leq 100°F$

29.4 ksi for $T = 550°F$ (local steam/water jet temperature)

Su = 70.0 ksi (minimum) for T $\leq$ 100°F

Assume Su at T = 550°F

$$= 70.0 \text{ ksi} \times \frac{29.4 \text{ ksi}}{38.0 \text{ ksi}} = 54.2 \text{ ksi}$$

3.8.2.3.2 Description of Loads

3.8.2.3.2.1 Dead and Live Load

For a description of dead and live load, see Subsections 3.8.1.3.2.1 and 3.8.1.3.2.2, respectively.

3.8.2.3.2.2 Design Basis Accident Pressure Load

The MC components are designed for a containment design basis accident internal pressure of 53 psig. The personnel lock is also designed for a design basis accident internal pressure of 53 psig.

3.8.2.3.2.3 External Pressure Load

The MC components are designed for a containment external pressure of 5 psi differential.

3.8.2.3.2.4 Thermal Loads

The operating and postulated design accident temperatures for the MC components are as follows:

	Temperature (F°)	
Condition	Drywell	Suppression Chamber
Operating	135	90
Design Accident	340	220

Thermal cycles used in design are as follows:

a) Startup and shutdown - 500 cycles, 105°F range

b) Design Basis Accident - 1 cycle, 220°F range.

3.8.2.3.2.5 Seismic Loads

3.8.2.3.2.5.1 Design Basis Loads

The MC components are designed for acceleration values, which are calculated using methods described in Bechtel Topical Report BC-TOP-4-A (Ref. 2K of Table 3.8-1).

The following acceleration values are used for the design of the drywell head assembly:

a) - 1.5g horizontal, ±0.6g vertical

The following acceleration values are used for the design of all other class MC components:

a) For equipment hatches, personnel lock, control rod drive removal hatch.

E	=	1.10 g horizontal, ± 0.65 g vertical
E'	=	0.75 g horizontal, ± 0.54 g vertical
LOCA	=	1.68 g horizontal, ± 1.06 g vertical
SRV	=	0.26 g horizontal, ± 0.70 g vertical

b) Suppression chamber hatches and all other components in the suppression chamber.

E	=	0.43 g horizontal, ± 0.38 g vertical
E'	=	0.38 g horizontal, ± 0.31 g vertical
LOCA	=	4.5 g horizontal, ± 0.51 g vertical
SRV	=	1.3 g horizontal, ± 0.28 g vertical

3.8.2.3.2.6 Missile and Pipe Rupture Loads

The drywell head assembly is designed for a local pipe rupture load of 48,000 lb. uniformly distributed over a circular area of 0.56 sq. ft. at any location on the drywell head. This load is due to the postulated rupture of the 6 in. diameter reactor vessel head spray pipe, which produces the largest load on the drywell head.

The equipment hatches are designed for a pipe rupture load of 1,200,000 lb. uniformly distributed over a circular area of 12 ft. diameter.

The CRD removal hatch is designed for a pipe rupture load of 160,000 lb. uniformly distributed over a circular area of 3 ft. diameter.

The loads on the equipment hatches and the CRD removal hatch are due to the rupture of a 28 in. diameter recirculation loop outlet pipe, which produces the largest load on the components.

The above values of static load include an appropriate dynamic load factor to account for the dynamic nature of the load. See Section 3.6 for a further discussion of pipe rupture loads.

3.8.2.4 Design and Analysis Procedures

3.8.2.4.1 Drywell Head Assembly

The analysis of the drywell head assembly is done using the thin shell computer program E0781 (Appendix 3.8A). This program calculates the stresses and displacements in thin-walled, elastic shells of revolution when subjected to static edge, surface, and/or temperature loads with an arbitrary distribution over the surface of the shell.

The drywell head assembly is divided into two analytical models. Figure 3.8-45 shows the drywell head model and the lower flange model. Displacement compatibility of the two models at the mating flange surface is maintained in the analysis. Boundary conditions are imposed on the analytical models by specifying boundary forces or displacements. Referring to Figure 3.8-45, the translation and rotation of the top of the drywell wall are imposed as boundary conditions to Boundary A. Boundary forces applied to Boundary B are calculated in accordance with thin shell theory.

3.8.2.4.2 Access Hatches

Access hatches, including the equipment hatches, personnel lock, suppression chamber access hatches and CRD removal hatch, are designed as pressure retaining components. The portions of the sleeves not backed by concrete are designed and analyzed according to the provisions of Section III, Subsection NE of the ASME B&PV Code.

At the junction of the hatch cover to the flange on the sleeve, where local bending and secondary stresses occur, the computer program E0119 (Appendix 3.8A) is used for analysis. This program is also used for the analysis of the flat head covers.

3.8.2.4.3 Pipe and Electrical Penetrations

For nuclear Class I flued head penetrations, the stress calculations are performed according to the requirements of Article NB-3200 of the ASME B&PV Code, Section III for design, normal and upset, emergency, and faulted conditions. Nuclear Class II flued head penetrations are designed for the most severe condition which is the faulted condition. The stress calculations are performed using acceptable simplified equations or finite element computer program.

For Class IE electrical cable penetrations, the procedures used in design and analysis are in compliance with Subsection NE of the ASME Code, Section III, Division 1. The stress calculations were performed using acceptable simplified equations shown in Appendix A-5000 of the ASME Code, Section III.

3.8.2.5 Structural Acceptance Criteria

Table 3.8-3 lists the allowable stress criteria used for the design and analysis of the ASME Class MC components. The criteria comply with Regulatory Guide 1.57 except that the Code addendum (Summer 1973) applicable to the Regulatory Guide is subsequent to the Code addendum used for the design of the MC components (Summer 1972).

3.8.2.6 Materials, Quality Control, and Special Construction Techniques

3.8.2.6.1 Materials

3.8.2.6.1.1 General

All carbon steel materials conform to the requirements of Article NE-2000, Materials, Section III of the ASME B&PV Code, 1971 Edition, with addenda through Summer 1972. Stainless steel materials for the CRD supply and return pipe penetrations conform to the requirements of Subsection NC of Section III of the ASME B&PV Code, 1971 Edition, with addenda through Summer 1972.

3.8.2.6.1.2 Drywell Head Assembly

Item	Specification
Drywell head and lower flange	SA-516, Grade 70, normalized
Bolts	SA-320, Grade L43
Nuts	SA-194, Grade 7

3.8.2.6.1.3 Access Hatches

Item	Specification
Sleeve and cover	SA-516, Grade 60 or 70, normalized
Bolts	SA-193, Grade B7
Nuts	SA-194, Grade 7

3.8.2.6.1.4 Penetrations

Item	Specification
Carbon steel sleeves	SA-333, Grade 1 or SA-516, Grade 60 normalized
Carbon steel caps for spare penetrations	SA-234, Grade WPB
Stainless steel sleeves for CRD supply	SA-312, Grade TP 304
Stainless steel fittings for CRD supply and return penetrations	SA-182, Grade F 304

3.8.2.6.2 Welding

Welding conforms to the requirements of Subsection NE, Section III, ASME B&PV Code, except all welding of the CRD supply and return penetrations conforms to the requirements of Subsection NC of Section III of the ASME B&PV Code. All pressure boundary welds are full penetration welds of double welded, bevel type. Welders and weld procedures are qualified in accordance with either Section IX of the ASME Code or AWS D1.1.

Penetrations, access hatches, and the drywell head flange are post-weld heat treated in accordance with Article NE-4000 of Section III of the ASME Code. Penetrations are preassembled into the liner plate sections and post-weld heat treated as complete subassemblies.

3.8.2.6.3 Materials Testing

Impact testing as required by the ASME Code is performed at 0°F or below.

3.8.2.6.4 Nondestructive Examination of Welds

All welds between penetrations and liner plate, access hatches and liner plate, and pressure retaining welds not backed by concrete are examined in accordance with Article NE-5000 of Section III of the ASME Code. Nondestructive examination complies with Regulatory Guide 1.19.

3.8.2.6.5 Quality Control

Quality control requirements for the construction phase are discussed in Appendix D and amendments to the PSAR.

3.8.2.6.6 Erection Tolerances

The specified erection tolerances for ASME Class MC steel components of the containment are as follows:

a) Suppression chamber penetrations are within 1 in. of their design elevations and circumferential locations.

b) Drywell penetrations are within 1 in. of their design circumferential locations. Critical penetrations, such as main steam, feedwater, core spray, etc., are within 1 in. of their design elevations. All other drywell penetrations vary from within 1 in. of design elevations for penetrations near the base of the drywell wall to within 2 in. of design elevations for penetrations near the top of the drywell wall.

c) Alignments of penetrations are within 1 degree of the design alignments.

d) The average elevation of the mating flange between the drywell head and the lower flange is within 3 in. of the design elevation. The mating flange is within 1/2 in. of level.

Actual deviations from the above were handled in accordance with procedures covered in Subsection 3.8.2.6.5.

3.8.2.7 Testing and In-service Inspection Requirements

3.8.2.7.1 Preoperational Testing

3.8.2.7.1.1 Structural Acceptance Test

The drywell head assembly, equipment hatches, suppression chamber access hatches, CRD removal hatch, and pipe and electrical penetrations are pneumatically tested to 1.15 times the design accident pressure during the containment structural acceptance test. See Subsection 3.8.1.7.1.1 for a description of the structural acceptance tests.

The personnel lock is pneumatically tested to 1.25 times the design accident pressure, following shop fabrication and following field erection, to verify its structural integrity.

The CRD supply and return pipe penetrations are hydrotested to 1.25 times the design pressure of 1750 psig following field erection in accordance with the ASME Code, Section III, Subsection NC.

3.8.2.7.1.2 Leak Rate Testing

Leaktightness of the containment Class MC components that are pressure retaining is verified during the integrated leak rate test. See Subsection 6.2.6 for a description of the containment integrated leak rate test.

The personnel lock is leak rate tested to 100 percent of the design accident pressure following shop fabrication and following field erection. The maximum allowable leak rate is 0.2 percent of the weight of the contained air in 24 hours when measured at ambient temperature and test pressure.

3.8.2.7.2 In-service Leak Rate Testing

In-service leak rate testing is discussed in Subsection 6.2.6.

3.8.3 CONTAINMENT INTERNAL STRUCTURES

3.8.3.1 Description of the Internal Structures

The internal structures of the containment perform the following major functions:

a) Support and shield the reactor vessel

b) Support piping and equipment

c) Form the pressure suppression boundary.

The containment internal structures are constructed of reinforced concrete and structural steel. The containment internal structures include the following:

a) Drywell floor

b) Reactor pedestal

c) Reactor shield wall

d) Suppression chamber columns

e) Drywell platforms

f) Seismic truss

g) Reactor steam supply system supports

Dwgs. C-331, Sh. 1, C-371, Sh. 2, C-1932, Sh. 3, C-1932, Sh. 4, and C1932, Sh. 5 show an overview of the containment including the internal structures.

3.8.3.1.1 Drywell Floor

The drywell floor serves as a barrier between the drywell and suppression chamber. It is a reinforced concrete circular slab with an outside diameter of 88 ft. 0 in. and a thickness of 3 ft. 6 in. See Dwg. C-348, Sh. 1, C-349, Sh. 1, and C-350, Sh. 1 for details of the drywell floor reinforcement.

The drywell floor is supported by the reactor pedestal, the containment wall, and 12 steel columns. The connection of the drywell floor to the containment wall is shown on Dwg. C-284, Sh. 1. The drywell floor is penetrated by 87 24-in. diameter vent pipes. Additional reinforcement is furnished at vent pipe penetrations. A 1/4 in. thick carbon steel liner plate is provided on top of the drywell floor and anchored to it. The liner plate prevents bypass of the vent pipes during LOCA. Refer to Subsection 6.2.1 for a description of the bypass leakage requirements. The liner plate also provides support for attachments such as pipe hangers. Loads from these attachments are transferred by means of the liner plate into the anchorages which are embedded in the drywell floor concrete. Dwg. C-293, Sh. 1 shows the drywell floor liner plate and anchorage system.

3.8.3.1.2 Reactor Pedestal

The reactor pedestal is a 82 ft. high, upright cylindrical reinforced concrete shell that rests on the containment base foundation slab and supports the drywell floor, reactor vessel, and reactor shield wall as well as drywell platforms, pipe restraints, and recirculation pumps. The connection of the reactor pedestal to the base foundation slab is shown on Dwg. C-281, Sh. 1. The reactor pedestal below the drywell floor has a 19 ft. 7 in. inside diameter and a 5 ft. 1 in. wall thickness. The reactor pedestal above the drywell floor has a 20 ft. 3 in. inside diameter and a 4 ft. 5 in. wall thickness. The thickness at the top of the pedestal is increased to 5 ft. 4 in., where it supports the reactor vessel and the reactor shield wall. See Dwgs. C-340, Sh. 1 and C-341, Sh. 1 for details of reinforcement. Openings are provided in the reactor pedestal to permit flow of air and suppression pool water into and out of the pedestal cavity. Additional reinforcement is furnished at openings. A 1/4 in. thick carbon steel form plate is provided on the inside and outside surfaces of the reactor pedestal below the drywell floor. This plate acts as a concrete form during construction and preserves the water quality of the suppression pool by preventing the leaching of chemicals from the reactor pedestal concrete into the suppression pool.

3.8.3.1.3 Reactor Shield Wall

The reactor shield wall is a 49 ft. high upright cylindrical shell which rests on the top of the reactor pedestal and provides primary radiation shielding as well as supports for pipe restraints and drywell platforms. The reactor shield wall is constructed of inner and outer carbon steel plates and unreinforced concrete between the two plates. See Dwg. C-376, Sh. 1 for details of the reactor shield wall. The reactor shield wall has a 25 ft. 7 in. inside diameter and a 1 ft. 9 in. wall thickness. The outer steel plate is 1-1/2 in. thick and is designed to withstand any local pipe restraint and drywell platform attachment loads. The inner steel plate is 1/2 in. thick and is designed to act with the outer plate to withstand local and non-localized loads. The inner and outer plates are connected with steel bars spaced on 2 ft. 6 in. centers. The annular space between the inner and outer plates is filled with unreinforced concrete. The concrete is used for radiation shielding only and is not relied upon as a structural element. Normal density concrete is used in the top and bottom portions of the reactor shield wall. High density concrete is used at the mid-height of the reactor shield wall opposite the reactor core for additional radiation shielding. The reactor shield wall is connected to the top of the reactor pedestal by 48 2 in. diameter, high strength anchor bolts as shown on Dwg. C-344, Sh. 1, and C-377, Sh. 1. The seismic truss and seismic stabilizer, which provide lateral support to the reactor vessel, are attached to the top of the reactor shield wall. Penetrations with hinged doors or removable plugs are provided in the reactor shield wall to facilitate piping connections to the reactor vessel and to provide access for in-service inspection. The wall thicknesses of penetration sleeves are large enough to prevent local stress concentrations in the inner and outer plates.

3.8.3.1.4 Suppression Chamber Columns

Twelve hollow steel pipe columns are furnished to support the drywell floor. Each column is 52 ft. 6 in. long, 42 in. outside diameter, with a 1-1/4 in. wall thickness as shown on Dwg. C-370, Sh. 1. The columns are connected to the base foundation slab at the bottom and to the drywell floor at the top with embedded anchor bolts. Dwg. C-370, Sh. 1 shows the connection to the base foundation slab.

3.8.3.1.5 Drywell Platforms

Platforms are furnished at five elevations in the drywell to provide access and support to electrical and mechanical components. The platforms consist of structural steel framing with steel grating. Builtup box shapes are used for beams that must resist biaxial bending. Beams that span between the pedestal or shield and the containment wall are provided with sliding connections at one end. Thus, no thermal axial loads are developed in the beams and no radial loads are imposed on the pedestal, shield, or containment wall. See Dwgs. C-362, Sh. 1, C-363, Sh. 1, C-364, Sh. 1, C-365, Sh. 1, and C-367, Sh. 1 for details of the drywell platforms.

3.8.3.1.6 Seismic Truss and Seismic Stabilizer

The seismic truss and the seismic stabilizer provide lateral support for the reactor vessel during earthquake and pipe rupture loading. The seismic truss spans between the containment wall and the reactor shield wall, and the seismic stabilizer spans between the reactor shield wall and the reactor vessel. For a description of the seismic stabilizer, see Section 3.9. The seismic truss is shaped like an eight pointed star and is fabricated of steel plates. See Dwg. C-380, Sh. 1 for details of the seismic truss. Dwg. C-286, Sh. 1 shows the connection of the seismic truss to the containment wall. This connection is designed to allow vertical and radial movement of the seismic truss relative to the containment wall but to prevent tangential movement.

3.8.3.1.7 Reactor Steam Supply System Supports

The steam supply system piping and pumps are supported by hangers, which in turn are supported by the reactor pedestal, reactor shield, and drywell platforms. A description of these supports is given in Section 3.9. In addition, the reactor vessel itself is supported on the reactor pedestal by 120, 31/4 in. diameter, high strength anchor bolts as shown on Dwg. C-344, Sh. 1, and C-377, Sh. 1. The reactor vessel is supported laterally by the seismic truss and seismic stabilizer as discussed in Subsection 3.8.3.1.6.

3.8.3.2 Applicable Codes, Standards, and Specifications

The codes, standards, and specifications used in the design and construction of the containment internal structures are listed in Table 3.8-1 and given a reference number.

The reference numbers for the drywell floor are 10A, 12A, 1C, 2C, 3C, 6C, and 2K.

The reference numbers for the drywell floor liner plate and anchorages are 4C, 1H, 1J, and 1K.

The reference numbers for the reactor pedestal are 7A, 10A, 12A, 1C, 2C, 3C, 6C, and 2K.

The reference numbers for the reactor shield wall are 1B, 6C, 1H, and 2K.

The reference numbers for the suppression chamber columns are 1H, 2H, 3H, 1J, and 2K.

The reference numbers for the drywell platforms and seismic truss are 1B, 1H, 2H, 3H and 2K.

3.8.3.3 Loads and Loading Combinations

3.8.3.3.1 General

Tables 3.8-2, 3.8-2a and 3.8-4, 3.8-5, 3.8-6 and 3.8-7 list the loading combinations used for the design and analysis of the containment internal structures. The loading combinations shown in these tables do not include hydrodynamic loads.

The internal structures have also been analyzed for hydrodynamic loads from main steam safety/relief valve discharge and LOCA. For a definition of loads and loading combinations including hydrodynamic loads, refer to GE's "Mark II Containment Dynamic Forcing Functions Information Report" (NEDO-21061) and the "Susquehanna Plant Design Assessment Report."

3.8.3.3.2 Drywell Floor and Reactor Pedestal

Table 3.8-2 lists the loading combinations used for the design of the drywell floor. The loading combinations are in compliance with those given in Reference 12A of Table 3.8-1.

Table 3.8-2a lists the loading combinations used for the design of the reactor pedestal. The loading combinations are in compliance with those given in SRP Section 3.8.3.II.3.

3.8.3.3.2.1 Description of Loads

Dead Load, Live Load, and Seismic Loads

For a description of dead load, live load, and seismic loads, see Subsections 3.8.1.3.2.1, 3.8.1.3.2.2 and 3.8.1.3.2.6, respectively.

Design Basis Accident Pressure Load

The drywell floor and the reactor pedestal are designed for the following pressures:

a) Maximum pressure: 53 psig in the drywell and the suppression chamber

b) Maximum differential pressure: 28 psig (53 psig in the drywell and 25 psig in the suppression chamber).

Thermal Loads

The temperature gradients through the drywell floor and the reactor pedestal are shown on Figure 3.8-58 for the operating and the postulated design accident condition. The design accident temperature gradients shown on Figure 3.8-58 occur five minutes after LOCA. These transient temperature gradients are used for the design of the drywell floor and the reactor pedestal because they produce the largest stresses in the structure.

Thermal effects anticipated at the time of the structural acceptance test are insignificant since changes in temperature inside and outside the containment during the test will be small.

Missile and Pipe Rupture Loads

The drywell floor and the reactor pedestal are designed to withstand the missile and pipe rupture loads due to a postulated rupture of a 28 in. diameter recirculation loop pipe, which produces the largest loads on the structures. These loads include the effects of jet impingement, pipe whip, and pipe reaction. An equivalent static load of 1030 kips is considered. This load includes an appropriate dynamic load factor to account for the dynamic nature of the load. See Section 3.6 for a further discussion of postulated pipe rupture loads.

3.8.3.3.3 Reactor Shield Wall

The reactor shield wall is designed using the elastic working stress design methods of AISC, "Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings," dated 1969, Part 1. Table 3.8-4 lists the load combination used for the design of the reactor shield wall. Since this loading condition combines the design basis accident loads with the maximum seismic loads, it is the most severe loading condition and other, less severe load combinations are not considered.

3.8.3.3.3.1 Description of Loads

Dead Load, Live Load, and Seismic Loads

For a description of dead load, live load, and seismic loads, see Subsections 3.8.1.3.2.1, 3.8.1.3.2.2 and 3.8.1.3.2.6, respectively.

Design Basis Accident Pressure Load

The reactor shield wall is designed for internal pressure due to a postulated pipe rupture at the connection of the pipe to the reactor vessel nozzle safe end. The following two pressure conditions are considered:

a) Maximum unbalanced pressure: pressure condition shortly after pipe break, which produces the largest lateral load on the reactor shield wall, as shown in Figure 6A-3b.

b) Maximum uniform pressure: 70 psig internal pressure.

Thermal Loads

The temperature gradients through the reactor shield wall are shown on Figure 3.8-59 for the operating and the postulated design accident conditions. The design accident temperature gradient shown on Figure 3.8-59 occurs five minutes after LOCA. This transient temperature gradient is used for the design of the reactor shield wall since it produces the largest stresses in the structure.

Missile and Pipe Rupture Loads

The reactor shield wall is designed to withstand the missile and pipe rupture loads due to a postulated rupture of any high energy pipe that penetrates the reactor shield wall and connects to the reactor vessel, such as recirculation and feedwater pipes. These loads include the effects of jet impingement, pipe whip, and pipe reaction. Equivalent static loads are considered, which include an appropriate dynamic load factor to account for the dynamic nature of the load. See Section 3.6 for a further discussion of postulated pipe rupture loads.

3.8.3.3.4 Suppression Chamber Columns

The suppression chamber columns are designed using the plastic design methods of AISC, "Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings," dated 1969, Part 2. Table 3.8-5 lists the load combinations used for the design of the suppression chamber columns. The columns are designed to resist the reaction loads from the drywell floor for the LOCA conditions. Subsection 3.8.3.3.2 includes a description of the loads for the drywell floor. The abnormal loading conditions govern the design since they include the design basis accident pressure load, which is the critical load for columns.

3.8.3.3.5 Drywell Platforms

The drywell platforms are designed using working stress design methods except for the pipe restraints supported on the platforms. The pipe restraints are designed to undergo local inelastic deformations due to postulated pipe rupture loads. However, there is no loss of function of the pipe restraints since they will restrain any postulated pipe whip. The built-up box beams that support the pipe restraints are designed to withstand all postulated pipe rupture loads. Design accident pressure and operating and design accident thermal loads do not affect the design of the drywell platforms. For the design of box beams, seismic loads due to dead weight of the beams may be neglected since these loads are insignificant relative to the pipe rupture loads. For the design of the framing beams, seismic loads due to dead weight of the beams are small and may be neglected since these beams are laterally braced by other framing beams and by the grating. The uniform design live load for the grating and framing beams is 200 psf. The live load for the framing beams also includes the gravity load, thermal reaction load, and seismic SSE reaction load of all piping and equipment supported on the beams. Table 3.8-6 lists the load combinations used to design the drywell platforms. Pressure, thermal and seismic loads are not considered since they are not critical.

3.8.3.3.6 Seismic Truss

The seismic truss is designed using working stress design methods. It is designed primarily for lateral seismic loads. However, it is also designed for jet impingement loads due to the postulated rupture of a 26 in. diameter main steam pipe. Design accident pressure and operating and design accident thermal loads do not affect the design of the seismic truss. Table 3.8-7 lists the load combination used to design the seismic truss. Pressure and thermal loads are not considered since they are not critical.

3.8.3.4 Design and Analysis Procedures

This section describes the procedures used for the design and analysis of the containment internal structures. The description does not include the effects of hydrodynamic loads from main steam safety/relief valve discharge and LOCA. For a description of the design and analysis procedures that consider the effects of hydrodynamic loads, refer to GE's "Mark II Containment Dynamic Forcing Functions Report" (NEDO-21061) and the "Susquehanna Plant Design Assessment Report."

3.8.3.4.1 Drywell Floor

The design and analysis procedures used for the drywell floor are similar to those used for the containment wall. Used for the analysis are 3D/SAP, CECAP, ME620, and seismic analysis computer programs (Appendix 3.8A). See Subsection 3.8.1.4.1 for a detailed description of the analysis procedures.

Figure 3.8-60 shows the 3D/SAP finite element model used to analyze the drywell floor for all loads other than seismic loads. A 15 degree wedge of the drywell floor is modeled using solid finite elements having linear elastic, isotropic material properties. One vertical boundary plane goes through a suppression chamber column and the other is halfway between two columns. The model includes the drywell floor, suppression chamber wall, reactor pedestal below the drywell floor, and a suppression chamber column. Boundary conditions are imposed on the analytical model by specifying nodal point forces or displacements. Referring to Figure 3.8-60, the nodal points lying along Boundary A are allowed to move within the X-Z plane, and Boundary B within the X-Y plane. Points along Boundary C are prevented from moving in the hoop direction. Points along Boundary D are prevented from moving in the radial direction to account for the restraining effect of the inner portion of the drywell floor. Nodal forces, moments, and shears are applied to Boundaries E and F to account for reaction loads from the drywell wall and reactor pedestal above the drywell floor, respectively.

Analytical techniques as described in Bechtel Topical Report BC-TOP-4-A (Ref. 2K of Table 3.8-1) are used to analyze the drywell floor for seismic loads.

3.8.3.4.2 Drywell Floor Liner Plate and Anchorages

The design and analysis of the drywell floor liner plate and anchorages is in accordance with Bechtel Topical Report BC-TOP 1 (Ref. 1K of Table 3.8-1). The analysis of the liner plate and anchorages for attachment loads is done using membrane theory for the liner plate and the theory of beams on elastic foundations for the anchorages.

3.8.3.4.3 Reactor Pedestal

The reactor pedestal is designed for axisymmetric loads using the FINEL computer program (Appendix 3.8A). The program performs a finite element, static analysis of axisymmetric structures with axisymmetric loading. Both concrete and reinforcing steel materials are included in the model. Special material properties include bilinearity in compression and bilinearity or cracking in tension. The operating and design accident temperature gradients are computed using ME620 computer program (Appendix 3.8A). For transient loads such as design accident pressure and thermal loads, the most critical combination of these loads is considered. Figure 3.8-34 shows a vertical section through the FINEL model of the containment used to analyze the reactor pedestal below the drywell floor. Points along Boundary A are prevented from moving in the vertical direction and points along Boundary B are prevented from moving in the radial direction.

Figure 3.8-61 shows the FINEL model used to analyze the reactor pedestal above the drywell floor. The model includes the reactor pedestal above the drywell floor and portions of the reactor vessel and the reactor shield wall. Local thermal effects at the top of the reactor pedestal due to heat input from the reactor vessel are determined by using the ME620 computer program (Appendix 3.8A). Referring to Figure 3.8-61, nodal points along Boundary A are prevented from moving in the vertical and radial directions. Nodal forces, moments, and shears are applied to Boundaries B and C to account for reaction loads from the reactor vessel and the reactor shield wall, respectively.

Non-axisymmetric loads on the reactor pedestal include seismic loads and reactor vessel and reactor shield reaction loads. Seismic forces, moments, and shears are calculated as described in Section 3.7. Vertical forces, horizontal shears, and overturning moments at the base of the reactor shield wall are determined as described in Subsection 3.8.3.4.4. These loads are applied to the top of the reactor pedestal. Concrete and reinforcing steel stresses in the reactor pedestal due to the above loads are calculated using the design methods of ACI 307. ACI 307 includes equations for determining the neutral axis of reinforced concrete cylindrical shells subjected to axial force and overturning moment. The position of the neutral axis satisfies the equilibrium of internal stresses and external forces and moments.

Concrete and reinforcing steel stresses due to axisymmetric and non-axisymmetric loads are combined to determine the total stress. Additional meridional, hoop, and shear reinforcement is provided at the top of the pedestal as shown in Dwg. C-341, Sh. 1 to resist local loads on the pedestal from the reactor vessel and the reactor shield. The seismically-induced tangential shears on the reactor pedestal are considerably less than the seismically-induced tangential shears on the containment wall. Therefore, helical reinforcement is not provided in the reactor pedestal in order to resist tangential shears. Meridional and hoop reinforcement is designed to carry the entire tangential shear by shear friction using the design methods of ACI 318-71.

3.8.3.4.4 Reactor Shield Wall

The reactor shield wall is analyzed in two stages. First, the effect of openings on the behavior of the reactor shield is investigated. This is done to determine whether the shield may be analyzed as an axisymmetric cylindrical shell without openings or whether the openings cause local stress concentrations. Loads considered for this analysis are design accident pressure and postulated pipe rupture loads. The EASE computer program (Appendix 3.8A) is used for this analysis. Figure 3.8-62 shows the finite element model. A full 360 degree section of the reactor shield wall is modeled using plate elements having linear elastic, isotropic material properties. One 64 in. diameter recirculation outlet penetration and two adjacent 48 in. diameter recirculation inlet penetrations are included in the model. Smaller finite elements are used in the area of the openings to obtain an accurate stress gradient. Referring to Figure 3.8-62, points along Boundary A are prevented from moving in the vertical and radial directions. Boundary B is a free edge. The results of this analysis confirm that there are no significant local stress concentrations in the shield around the openings. This is due to the stiffening of the shell that is provided by the thick-walled penetration sleeves. Therefore, the use of an axisymmetric analytical model without openings is justified.

The second stage analyzes the reactor shield wall as an axisymmetric shell. For axisymmetric loads, which include dead load and design accident thermal load, the FINEL computer program is used. The most critical temperature gradient as determined by the ME620 computer program (Appendix 3.8A) is considered. The FINEL program performs a finite element, static analysis of axisymmetric structures with axisymmetric loading. For non-axisymmetric loads, which include design accident pressure load, seismic load, and pipe rupture load, the ASHSD computer program (Appendix 3.8A) is used. The ASHSD program performs an elastic, finite element, static, or dynamic analysis of axisymmetric structures with non-axisymmetric loading. The distribution of non-axisymmetric load around the shell is approximated by a Fourier series expansion. Figure 3.8-63 shows a vertical section through the model used for FINEL and ASHSD programs. Points along Boundary A are prevented from moving in the vertical and radial directions. For non-axisymmetric loads, Boundary B at the connection of the seismic truss to the containment wall is prevented from moving in the radial direction. Total stresses in the reactor shield wall are determined by summing the axisymmetric and non-axisymmetric stresses.

3.8.3.4.5 Suppression Chamber Columns

Axial force, shear, and moment in the columns due to axisymmetric loads, such as dead load and design accident pressure and thermal loads, are determined using the FINEL computer program (Appendix 3.8A). Figure 3.8-34 shows the FINEL model of the containment used to analyze the suppression chamber columns. A description of the program and the boundary conditions is given in Subsection 3.8.3.4.3. Since the FINEL program can consider only axisymmetric structures, the 12 columns are modeled as an equivalent cylinder having the cross-sectional area and axial stiffness of the columns. Axial force in the columns is calculated from the axial stress determined by the FINEL program. Shear and moment in the columns are calculated from relative displacements of the drywell floor and the base foundation slab determined by the FINEL program.

Axial force, shear, and moment in the columns due to seismic loads are determined using several methods. Axial force in the columns due to horizontal seismic load is determined using the ASHSD program (Appendix 3.8A). Figure 3.8-64 shows the model. Axisymmetric shell and solid finite elements having linear elastic, isotropic material properties are used. Nodal points lying along Boundary A are prevented from moving in the vertical direction and points along Boundary B are prevented from moving in the radial direction. The load applied to the ASHSD model is the seismic horizontal shear and overturning moment for the containment calculated as described in Section 3.7.

Shear and moment in the columns due to horizontal seismic load are determined using the analytical procedures described in Bechtel Topical Report BC-TOP-4-A (Ref. 2K of Table 3.8-1). The lumped mass model of the containment including columns and vent pipes is shown in Figure 3.8-65. Since the vent pipes are laterally braced to the columns, shear and moment are produced in the columns due to seismic motion of the vent pipes.

Axial force in the columns due to vertical seismic load is determined by applying the vertical forces calculated from the containment seismic analysis to the drywell floor at its connections to the containment wall and the reactor pedestal. The vertical force transmitted to the columns through the drywell floor is calculated considering the relative vertical stiffnesses of the containment wall, reactor pedestal, and columns.

The postulated rupture of a 28 in. diameter recirculation loop pipe produces a vertical jet impingement load on the top of the drywell floor and, therefore, produces loads in the columns. Axial force, shear, and moment in the columns due to jet force is calculated by the CE 668 computer program (Appendix 3.8A). The program performs a static, linear elastic analysis of flat slabs of arbitrary dimensions subjected to arbitrary loading. Figure 3.8-66 shows the 180 degree model of the drywell floor. A vertical jet force is applied along the axis of symmetry and the reaction is calculated in the column adjacent to the applied load. Edges of the drywell floor along Boundaries A and B are considered to be fixed supports. Nodal points at the columns are fixed in the plane of the model.

The total axial force, shear, and moment in the columns for all load combinations are determined by summing the results of the separate analyses. Stability of the columns for the most critical load combination is checked using the plastic design methods of AISC, "Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings," dated 1969, Part 2 (Ref. 1H of Table 3.8-1).

3.8.3.4.6 Drywell Platforms

The drywell platforms are designed using conventional elastic design methods which conform to the AISC Specification, 1969, Part 1 (Ref. 1H of Table 3.8-1).

3.8.3.4.7 Seismic Truss

Seismic forces in the seismic truss are calculated using the methods described in Bechtel Topical Report BC-TOP-4-A (Ref. 2K of Table 3.8-1). Axial force, shear force, and moment in the seismic truss due to postulated pipe rupture loads are calculated using moment distribution. Figure 3.8-67 shows the rigid frame model including boundary conditions.

3.8.3.5 Structural Acceptance Criteria

3.8.3.5.1 Reinforced Concrete

The allowable stresses for the reinforced concrete portions of the containment internal structures are the same as the allowable stresses for the reinforced concrete portions of the containment. See Subsection 3.8.1.5.1 for a description.

3.8.3.5.2 Drywell Floor Liner Plate and Anchorages

The structural acceptance criteria for the drywell floor liner plate and anchorages are the same as the structural acceptance criteria for the containment liner plate and anchorages. See Subsection 3.8.1.5.2 for a description.

3.8.3.5.3 Structural Steel

Structural steel portions of the containment internal structures include the reactor shield wall, suppression chamber columns, drywell platforms, and seismic truss. For normal loading conditions, the allowable stresses are in accordance with the AISC Specification (Ref. 1H of Table 3.8-1).

For extreme environmental and abnormal loading conditions, the allowable stresses are as follows:

a) Bending - 0.90 Fy

b) Axial tension or compression - 0.85 Fy except, where allowable stress is governed by requirements of stability (local or lateral buckling), allowable stress shall not exceed 1.5 Fs.

c) Shear - 0.50 Fy

For extreme environmental and abnormal loading conditions, the allowable stress for bolted and welded connections is 1.7 Fs.

The allowables are defined as:

Fs = Allowable stress according to the AISC Specification, Part 1 (Ref. 1H of Table 3.8-1)

Fy = Specified yield strength of structural steel

3.8.3.6 Materials, Quality Control, and Special Construction Techniques

3.8.3.6.1 Concrete Containment Internal Structures

The concrete and reinforcing steel materials for the containment internal structures are discussed in Appendix 3.8B. Concrete design compressive strengths are given in Table 3.8-11.

3.8.3.6.2 Drywell Floor Liner Plate, Anchorages, Attachments

3.8.3.6.2.1 Materials

Liner plate materials conform to the requirements of the standard specifications listed below:

Item	Specification
Liner plate (less than1/2 in. thick)	ASTM A 285, Grade A

Liner plate (1/2 in. thick or thicker)	ASME SA-516, Grade 60 or 70 conforming to the requirements of ASME Boiler and Pressure Vessel Code (ASME B&PV Code), 1971 Edition with Addenda through Summer 1972, Section III, Article NE-2000, Materials
Anchorages and attachments	ASTM A 36

3.8.3.6.2.2 Welding

Welding requirements for the drywell floor liner plate and anchorages are the same as the welding requirements for the containment liner plate and anchorages. See Subsection 3.8.1.6.2.2 for a description of the welding requirements.

3.8.3.6.2.3 Nondestructive Examination of Liner Plate Seam Welds

Nondestructive testing of liner plate welds is performed in accordance with Regulatory Guide 1.19, Revision 1.

Liner plate seam welds are 100 percent magnetic particle examined. Liner plate seam welds are also 100 percent vacuum box soap bubble tested. Welds that are inaccessible for vacuum box testing are 100 percent liquid penetrant tested. Examination procedures, personnel qualification, and acceptance standards are in accordance with Subsection 3.8.1.6.2.4.

3.8.3.6.2.4 Erection Tolerances

The specified levelness of anchorages placed in the drywell floor is within -1-1/4 in. of the theoretical elevation over the entire area, plus a local tolerance of ±1/8 in. in any 30 ft. length. Actual deviations from the above were handled in accordance with quality control procedures covered in Appendix D and amendments to the PSAR.

3.8.3.6.3 Reactor Shield Wall and Seismic Truss

3.8.3.6.3.1 Materials

Item	Specification
Inner and outer plates, seismic truss, pipe restraints, etc.	ASTM A 588, Grade A or B
Internal stiffeners	ASTM A 36
Seismic Truss Male Stabilizer Block	ASME SA 181, Grade II

3.8.3.6.3.2 Welding and Nondestructive Examination of Welds

Welding and nondestructive examination is performed in accordance with AWS D1.1.

3.8.3.6.3.3 Materials Testing

The 1-1/2 in. thick outer plate and other plates subjected to transverse tensile stress are vacuum degassed and ultrasonically tested in accordance with supplementary requirements S-1 and S-8.1, respectively, of ASTM A 20-72a.

3.8.3.6.3.4 Erection Tolerances

The specified erection tolerances for the reactor shield are as follows:

a) The radial dimension from the as-built centerline of the reactor vessel to any point on the reactor shield is within 3/8 in. of the theoretical radius.

b) The top of the reactor shield is set within 1/4 in. of its theoretical elevation.

c) The azimuths of the shield penetrations are within 1/2 in. of the theoretical azimuths.

d) Seismic truss members do not deviate from axial straightness by more than 1/1000 of axial length.

Actual deviations from the above were handled in accordance with procedures covered in Appendix D and amendments to the PSAR.

3.8.3.6.4 Suppression Chamber Columns

3.8.3.6.4.1 Materials

The column shafts, base plates, and top plates are fabricated of ASME SA-516, Grade 70 material.

3.8.3.6.4.2 Welding

Weld procedures and qualifications conform to the provisions of Section IX and Section VIII, Division 1 of the ASME Boiler and Pressure Vessel Code, 1971 Edition with addenda through Summer 1972. All welders are qualified in accordance with Section IX of the ASME Code.

3.8.3.6.4.3 Nondestructive Examination of Welds

Nondestructive examinations conform to Section V of the ASME B&PV Code, 1971 Edition with addenda through Summer 1972. All personnel performing nondestructive examination are qualified in accordance with the American Society for Nondestructive Testing's Recommended

Practice No. SNT-TC-1A and its applicable supplements. Acceptance standards conform to Section VIII, Division 1 of the ASME Code.

3.8.3.6.4.4 Fabrication and Erection Tolerances

The specified fabrication and erection tolerances for suppression chamber columns are as follows:

a) The outside diameter, based on circumferential measurements, does not deviate from the theoretical outside diameter by more than 0.5 percent.

b) Out-of-roundness, defined by the difference between the maximum and minimum diameters related to the theoretical diameter, is in accordance with ASME Code, Section VIII, Division 1, Paragraph UG-80.

c) The finished length does not differ from the theoretical length by more than 1/4 in.

d) The finished column shaft does not deviate from straightness by more than 1/8 in. in 1 ft, with a maximum for the full length of 1/1000 of the total length.

e) Erection tolerances are in accordance with the AISC Specification (Ref. 1H and 2H of Table 3.8-1).

Actual deviations from the above were handled in accordance with procedures covered in Subsection 3.8.3.6.6.

3.8.3.6.5 Drywell Platforms

3.8.3.6.5.1 Materials

Item	Specification (or Engineer Approved Equal)
Box Beams	ASTM A 441
Rolled Shapes	ASTM A 36
Connection Bolts	ASTM A 325

3.8.3.6.5.2 Welding and Nondestructive Examination of Welds

Welding and nondestructive examination is performed in accordance with AWS D1.1.

3.8.3.6.5.3 Erection Tolerances

Erection tolerances for the drywell platforms are in accordance with AISC Specification (Ref. 2H of Table 3.8-1).

3.8.3.6.6 Quality Control

Quality control requirements for construction are discussed in Appendix D and amendments to the PSAR.

3.8.3.7 Testing and In-service Inspection Requirements

3.8.3.7.1 Preoperational Testing

3.8.3.7.1.1 Structural Acceptance Test

The drywell floor is tested to 1.15 times the design downward differential pressure. See Subsection 3.8.1.7.1.1 for a description of the structural acceptance tests.

Deflections and strains of the drywell floor measured during the Unit 1 test were less than the predicted values. Thus, the design of the drywell floor provides an adequate safety margin against internal pressure. Figure 3.8-68 shows a comparison between measured and predicted deflections for the drywell floor at peak differential pressure.

3.8.3.7.1.2 Leak Rate Testing

Preoperational leak rate testing is discussed in Subsection 6.2.6.

3.8.3.7.2 In-service Leak Rate Testing

In-service leak rate testing is discussed in Subsection 6.2.6.

3.8.4 OTHER SEISMIC CATEGORY I STRUCTURES

This section gives information on all Seismic Category I structures except the primary containment and its internals. It also describes non-seismic Category I structures designated with a safety classification of “other”. The structures included in this section are as follows:

Seismic Category I Structures

Reactor Building

Control Building

Diesel Generator 'A-D' Building

Diesel Generator 'E' Building

Engineered Safeguards Service Water Pumphouse

Spray Pond

Non-Seismic Category I, Structures Designated with a Safety Classification of “Other”

Turbine Building

Radwaste Building

The general arrangement of these structures is shown on Dwgs. A-11, Sh. 1, A-12, Sh. 1, A-13, Sh. 1, M-203, Sh. 1, M-204, Sh. 1, A-16, Sh. 1, and A-17, Sh. 1. Figures 3.8-77 and 3.8-78. Dwgs. M-227, Sh. 1, M-237, Sh. 1, M-260, Sh. 1, M-261, Sh. 1, M-5200, Sh. 1, M-5200, Sh. 2, M-284, Sh. 1, C-64, Sh. 1, C-65, Sh. 1, C-66, Sh. 1, and C-67, Sh. 1, M-270, Sh. 1, M-271, Sh. 1, M-272, Sh. 1, M-273, Sh. 1, and M-274, Sh. 1.

3.8.4.1 Description of the Structures

Reactor Building

Refer to Dwgs. A-11, Sh. 1, A-12, Sh. 1, A-13, Sh. 1, M-203, Sh. 1, M-204, Sh. 1, A-16, Sh. 1, A-17, Sh. 1, Figures. 3.8-77, 3.8-78 and Dwg. A-17, Sh. 1.

The reactor building encloses the primary containment, and provides secondary containment when the primary containment is in service during power operation. It also serves as containment during reactor refueling and maintenance operations, when the primary containment is open. It houses the auxiliary systems of the nuclear steam supply system, new fuel storage vaults, the refueling facility, and equipment essential to the safe shutdown of the reactor.

The reactor building, up to and including the operating floor, is of reinforced concrete on a mat foundation. The bearing walls are of reinforced concrete and are designed as shear walls to resist lateral loads. The floors are of reinforced concrete supported by a steel beam and column framing system and are designed as diaphragms to resist lateral load. The framing runs in both east-west and north-south directions, with the exterior ends of the beams supported by either the bearing walls or steel columns. The steel columns are supported by base plates on the mat foundation. The reinforced concrete walls and floors meet structural as well as radiation shielding requirements. Where structurally permissible, concrete block masonry walls are used at certain locations to provide better access for erection and installation of equipment. The block walls also meet the radiation shielding requirements.

The reactor building superstructure above the operating floor is a steel structure. The structural steel framing supports the roof, metal siding, and overhead cranes. The framing consists of a series of rigid frames connected by roof and wall bracing systems. The roof consists of built-up roofing on metal deck.

The refueling facility is located above the containment structure. It consists of spent fuel pool, fuel shipping cask storage pool, steam dryer and separator storage pool, reactor cavity, skimmer surge tank vault, and load center room. The facility is supported by two reinforced concrete girders running north-south, spanning over the containment. The girders are supported at the ends by concrete walls and at intermediate points by steel box columns. A gap is provided between the bottom of the girders and the top of the containment to ensure that loads from the refueling facility are not transferred to the containment. The walls and slabs of the spent fuel pool, the fuel shipping cask storage pool, the reactor cavity, and the steam dryer and separator storage pool are lined on the inside with a stainless steel liner plate. The facility meets the radiation shielding requirements.

The reactor building is separated from the primary containment by a gap, except at the foundation level, where a cold joint is provided between the two mats. A gap is also provided at the interface of the reactor building with the diesel generator and turbine buildings.

Control Building

Refer to Dwgs. A-11, Sh. 1, A-12, Sh. 1, A-13, Sh. 1, M-203, Sh. 1, M-204, Sh. 1, A-16, Sh. 1 and Figure 3.8-77.

The control building houses the control room, the cable spreading rooms, computer and relay room, the battery room, H&V equipment room, off-gas treatment room, and the visitors' gallery for the control room.

The control building is structurally integrated with the reactor building. It is a reinforced concrete structure on a mat foundation. The bearing walls are of reinforced concrete and are designed as shear walls to resist lateral loads. The floors and roof are of reinforced concrete supported by steel beams, and are designed as diaphragms to resist lateral loads. The beams span in the east-west direction and are supported by the bearing walls at the ends. The reinforced concrete walls and floors meet structural as well as radiation shielding requirements. Where structurally permissible, concrete block masonry walls are used at certain locations to provide better access for erection and installation of equipment. The block walls also meet the radiation shielding requirements.

The control building is separated from the turbine building by a gap, except at the foundation level, where a cold joint is provided between the two mats.

Diesel Generator 'A-D' Building

Refer to Dwgs. M-260, Sh. 1 and M-261, Sh. 1.

The diesel generator 'A-D' building houses diesel generators A, B, C and D which are essential for safe shutdown of the plant.

The diesel generators are separated from each other by concrete walls. A concrete overhang on the east side of the building serves as an air intake plenum. A concrete plenum for diesel exhaust is located on the roof.

It is a reinforced concrete structure on a mat foundation. The bearing walls are of reinforced concrete and are designed as shear walls to resist lateral loads. The floors and roof are of reinforced concrete supported by steel beams, and are designed as diaphragms to resist lateral loads. The south side of the building interfaces with the reactor building; there, a reinforced concrete wall is provided from foundation up to the design high water table level and then a steel frame is provided up to the roof. Where structurally permissible, concrete block masonry walls are used at certain locations to provide better access for erection and installation of equipment.

The diesel generators are supported by reinforced concrete pedestals. The pedestals are separated from the operating floor by a gap to allow for their independent vibration.

Diesel Generator 'E' Building

Refer to Dwgs. M-5200, Sh. 1 and M-5200, Sh. 2.

The Diesel Generator 'E' Building houses diesel generator E which is used to replace one of the A-D diesel generators.

Openings for air intake and diesel exhaust are flush with the north and south exterior walls, respectively. Interior plenums are provided for missile protection.

It is a reinforced concrete structure on a mat foundation. The bearing walls are of reinforced concrete and are designed as shear walls to resist lateral loads. The floors and roof are of reinforced concrete and are designed as diaphragms to resist lateral loads. The building is a free-standing detached structure with no other building in the immediate vicinity. Concrete block masonry walls are not used in this building.

Diesel Generator E is supported by a reinforced concrete pedestal. The pedestal is separated from the operating floor by a gap to allow for their independent vibration.

Engineered Safeguards Service Water (ESSW) Pumphouse

Refer to Dwg. M-284, Sh. 1.

The ESSW Pumphouse contains the Emergency Service Water (ESW) and Residual Heat Removal Service Water (RHRSW) pumps and the weir and discharge conduit for the spray pond.

It is a two-story reinforced concrete structure on a mat foundation. The bearing walls are of reinforced concrete and are designed as shear walls to resist lateral loads. The operating floor and roof are of reinforced concrete supported by steel beams and are designed as diaphragms to resist lateral loads. A mezzanine floor composed of grating over steel beams is provided to support the heating and ventilating equipment.

Spray Pond

Refer to Dwgs. C-64, Sh. 1, C-65, Sh. 1, C-66, Sh. 1, and C-67, Sh. 1.

The spray pond is a reservoir, free form in shape, which holds approximately 25 million gal. of water during normal operation. The water surface area is approximately eight acres and has a depth of approximately 10 ft. 6 in. It is designed so that normal operating water is retained in excavation alone, i.e., not by constructed embankments. Embankments are provided to ensure a minimum freeboard of 3 ft. and to direct flood water away from safety related facilities in a controlled manner.

The ESSW pumphouse is located at the southeast corner of the spray pond. A reinforced concrete liner covers the entire spray pond and is integrated with the outer walls of the ESSW pumphouse.

The water level in the pond is controlled by a weir housed in the ESSW pumphouse. During normal operation, excess water is discharged into the Susquehanna river via a conduit from the ESSW pumphouse.

An emergency spillway is provided at the east end of the pond. The only anticipated use of this spillway will be either during a malfunction of the discharge conduit leading out of the ESSW pumphouse or during certain postulated flood conditions. This is discussed in Subsection 2.4.8. The ESSW and RHRSW pipes enter the south side of the pond and traverse to the spray bank areas buried in 18 in. of concrete, provided as missile protection. Concrete columns support the riser pipes in the spray bank areas.

Turbine Building

Refer to Dwgs. A-11, Sh. 1, A-12, Sh. 1, A-13, Sh. 1, M-203, Sh. 1, M-204, Sh. 1, A-16, Sh. 1, Figure 3.8-77, Dwg M-227, Sh. 1, and M-237, Sh. 1.

The turbine building is divided into two units with an expansion joint separating the two units. It houses two in-line turbine generator units and auxiliary equipment including condensers, condensate pumps, moisture separators, air ejectors, feedwater heaters, reactor feed pumps, motor-generator sets for reactor recirculating pumps, recombiners, interconnecting piping and valves, and switchgears.

Two 220-ton overhead cranes are provided above the operating floor for service of both turbine generator units. Two reinforced concrete tunnels, one for each unit, are provided for the off-gas pipelines at the foundation level between the recombiners and the radwaste building. Reinforced concrete tunnels are also provided for the main steam lines below the operating floor from the reactor building to the condenser areas of the turbine generators.

The turbine building rests on a reinforced concrete mat foundation. The superstructure is framed with structural steel and reinforced concrete. Rigid steel frames support the two 220 ton cranes. They also resist all transverse (east-west) lateral loads. Steel bracings resist longitudinal (north-south) lateral loads above the operating floor. Below this level, reinforced concrete shear walls transfer all lateral loads to the foundations.

A seismic separation gap, also serving as an expansion joint, is provided near the center of the building between the two units. Seismic separation gaps are also provided at the interface of turbine building with the reactor, control, and radwaste buildings.

The floors of the turbine building are of reinforced concrete on structural steel beams. They are designed as diaphragms for lateral load transfer to the shear walls. The roof is built-up roofing on metal decking.

Exterior walls are precast reinforced concrete panels except for the upper 30 ft. which are metal siding.

Interior walls required for radiation shielding or fire protection are constructed of reinforced concrete block. These walls are not used as elements of the load resistant system.

The turbine generator units are supported on freestanding reinforced concrete pedestals. The mat foundations for the pedestals are founded on rock at the same level as the base mat for the turbine building. Separation joints are provided between the pedestals and the turbine building floors and walls to prevent transfer of vibration to the building. The operating floor of the building is supported on vibration damping pads at the top edge of the pedestal.

Radwaste Building

Refer to Dwgs. M-270, Sh. 1, M-271, Sh. 1, M-272, Sh. 1, M-273, Sh. 1 and M-274, Sh. 1.

The radwaste building houses systems for receiving, processing, and temporarily storing the radioactive waste products generated during the operation of the plant. It is a reinforced concrete structure on a mat foundation. The bearing walls are of reinforced concrete and are designed as shear walls to resist lateral loads. The floors and roof are of reinforced concrete supported by a beam and column framing system and are designed as diaphragms to resist lateral loads. The columns are supported by base plates on the mat foundation. The reinforced concrete walls and floor meet structural as well as radiation shielding requirements. Where structurally permissible, concrete block masonry walls are used at certain locations to provide better access for erection and installation of equipment. The block walls also meet the radiation shielding requirements.

The radwaste building is separated from the turbine building by a gap.

3.8.4.2 Applicable Codes, Standards, and Specifications

The codes, standards, and specifications used in the design, fabrication, and construction of the structures listed in Subsection 3.8.4 are shown in Table 3.8-1.

3.8.4.3 Loads and Load Combinations

The following loads and load combinations are considered in the design of Seismic Category I structures (other than the containment).

3.8.4.3.1 Description of Loads

For a general description of loads, see Subsection 3.8.1.3.2.

3.8.4.3.2 Load Combinations

Table 3.8-8 describes the load combinations applicable to the reactor building. Tables 3.8-9 and 3.8-9a contain the load combinations applicable to Seismic Category I structures other than the reactor building. Table 3.8-10 describes the load combinations used in the design of the turbine and the radwaste buildings.

3.8.4.4 Design and Analysis Procedures

The structures described in Subsection 3.8.4.1 are designed to maintain elastic behavior under various loads and their combinations. The loads and the load combinations are fully described in Subsection 3.8.4.3. All reinforced concrete components of the structure are designed by the strength method per ACI 318 and ACI 349 (Ref 10A and 12A of Table 3.8-1). All structural steel components are designed by the working stress method per AISC specification (Ref 1H of Table 3.8-1). Determination of wind and tornado loads is described in Section 3.3.

Seismic design of structures is described in Section 3.7. The buildings are analyzed dynamically.

Design of structure for missile protection is covered in Subsection 3.5.3.

Computer programs STRESS and ICES STRUDL-II (Ref 1 and 2, respectively, of Subsection 3.8.4.8) are used to analyze structural steel framing.

The refueling facility of the reactor building is designed based on finite element analysis by use of computer program MRI/STARDYNE 3 (Ref 3 of Subsection 3.8.4.8).

The spray pond is basically a concrete-lined soil structure. Its design is discussed in Subsection 2.5.5.

Concrete masonry blockwalls in all Seismic Category I structures have been analyzed dynamically as described in Section 3.7b.3.I.5. They are designed for out-of-plane and in-plane inertia forces generated by the mass of the blockwall and attachment loads, combined with other loads as described in Tables 3.8-8 and 3.8-9. Walls in the turbine and radwaste buildings have been designed for seismic loads per UBC (Ref. 1L of Table 3.8-1).

3.8.4.5 Structural Acceptance Criteria

Reinforced Concrete

The reinforced concrete structural components are designed by the strength method per ACI 318 and ACI 349 (Ref 10A and 12A of Table 3.8-1) for loads and load combinations described in Subsection 3.8.4.3.

Structural Steel

The structural steel components are designed by the working stress method per AISC specification (Ref 1H of Table 3.8-1) for loads and load combinations described in Section 3.8.4.3. The allowable stresses for different load combinations are indicated therein.

Concrete Block Masonry Walls

All masonry blockwalls are reinforced walls and do not act as shear walls. Masonry blockwalls are designed by the working stress method per UBC (Ref. 1L of Table 3.8-1). The allowable loads per UBC Tables 24-B or 24-H (special inspection) are modified as described in Tables 3.8-8, 3.8-9 and 3.8-12, except as noted below.

For double wythe walls designed as composite sections and having concrete or grout infill thickness of 8 inches or more, the allowable shear or tension between masonry block and infill is 1/1 $\sqrt{f'}$ i.e. 43 psi. However, the actual design stress does not exceed 15 psi. For other double wythe walls, allowable shear/tension stress is assumed to be zero at the interface.

3.8.4.6 Materials, Quality Control, and Special Construction Techniques

3.8.4.6.1 Concrete and Reinforcing Steel

The concrete and reinforcing steel materials are discussed in Appendix 3.8B. Concrete design compressive strengths are given in Table 3.8-11. Materials for concrete block masonry walls are discussed in Appendix 3.8C.

3.8.4.6.2 Structural Steel

3.8.4.6.2.1 Materials

The various structural steel components conform to the following specifications:

Item	Specification (or Engineer Approved Equal)
Beams, girder, and plates	ASTM A36 and ASTM A588
Box columns including base plates and cap plates	ASTM A588
Structural tubing	ASTM A500 and ASTM A501
High strength bolts	ASTM A325 and ASTM A490
Studs	AWS D1.1

3.8.4.6.2.2 Welding and Nondestructive Testing

Welding and nondestructive testing is performed in accordance with either AWS D1.1 (Ref. 1B of Table 3.8-1) or Section IX of the ASME Code (Ref. 1J of Table 3.8-1).

3.8.4.6.2.3 Fabrication and Erection

The fabrication and erection of structural steel conforms to the AISC specification (Ref. 1H, 2H and 3H of Table 3.8-1).

3.8.4.6.2.4 Quality Control

Quality control of structural steel for the construction phase is discussed in Appendix D of the PSAR and amendments to the PSAR.

3.8.4.6.3 Special Construction Techniques

Techniques involved in the construction of Seismic Category I structures are standard construction procedures.

3.8.4.7 Testing and In-service Inspection Requirements

Testing and in-service inspection are not required for Seismic Category I structures (other than the containment).

3.8.4.8 Computer Programs Used in the Design and Analysis of Other Seismic Category I Structures

1) STRESS, Department of Civil Engineering, Massachusetts Institute of Technology

2) ICES STRUDL-II, Department of Civil Engineering, Massachusetts Institute of Technology

3) MRI/STARDYNE (Version 3), Control Data Corporation.

For other computer programs refer to Subsection 2.5.5 and Section 3.7

3.8.5 FOUNDATIONS

This subsection describes foundations for all Seismic Category I structures except the spray pond. The spray pond is basically a soil structure and its design is discussed in Subsection 2.5.5. Descriptions of foundations for non-seismic Category I structures designated with a safety classification of "other" such as the turbine building and the radwaste building, are also included in this section.

3.8.5.1 Description of the Foundations

Typical details of the foundations for various structures are shown on Dwg. C-795, Sh. 1.

Reinforced concrete mat foundations have been provided for all structures. The mats rest on sound rock except the ESSW pumphouse mat is supported by natural soil.

All bearing walls of the structures are rigidly connected to the foundation mat. Where steel columns are provided, they are attached to the mat by base plates and anchor bolts. The bearing walls and the steel columns carry all the vertical loads from the structure to the mat. Horizontal shears due to wind, tornado, and seismic loads are transferred to the shear walls by the roof and floor diaphragms. The shear walls transfer the horizontal shears to the foundation mat and from there to the foundation medium through friction. Also, as shown on Dwg. C-795, Sh. 1, the sides of the base mats of all the structures except the ESSW pumphouse are keyed to the foundation rock all around by poured concrete, which helps in transferring the horizontal shears to the

foundation rock. The edges of the ESSW pumphouse base mat are poured directly against the excavated slopes of the natural soil formation.
A mudmat (unreinforced concrete layer) is provided between the base of the foundation mat and the foundation medium. Except for the ESSW pumphouse, a waterproofing membrane is provided in the mudmat and on the outside face of peripheral subterranean walls. Perforated pipes are provided around the periphery of the buildings to collect groundwater seepage and drain it to the sumps. Waterproofing membrane under the ESSW pumphouse foundation mat is not considered necessary as the predicted groundwater table at the pumphouse site is well below the foundation mat (refer to Subsection 2.5.5).

Peripheral subterranean walls are designed to resist lateral pressures due to backfill, groundwater, and surcharge loads, in addition to dead loads, live loads, and seismic loads.

Containment: The containment foundation is described in Subsection 3.8.1.

Reactor Building and Control Building

The foundation mats of the reactor and control buildings are poured monolithically.

The reactor building foundation mat is approximately 4 ft. 9 in. thick and is reinforced typically with #11 bars at 12 in. centers at top and bottom in both the north-south and east-west directions. The mat surrounds the containment mat, with a cold joint separating the two.

The control building foundation mat is about 2 ft. 6 in. thick and is reinforced typically with #8 bars at 12 in. centers at top and bottom in the north-south direction and #11 bars at 12 in. centers at top and #8 bars at 12 in. centers at bottom in the east-west direction. A cold joint is provided between the control and the turbine building mats.

Diesel Generator Buildings:

The foundation mats of the diesel generator 'A-D' and 'E' buildings are approximately 2 ft. 6 in. thick and 3 ft. 10 in. thick, respectively. The foundation mats are reinforced typically with #9 bars at 12 in. centers at top and bottom in both the north-south and east-west directions. Cold joints are provided between the diesel generator pedestals and the diesel generator building mats.

SSW Pumphouse: The foundation mat of the ESSW pumphouse is about 3 ft. thick and is reinforced typically with #9 bars at 12 in. centers at top and bottom in both the north-south and east-west directions.

Turbine Building: The turbine building mat is approximately 2 ft. 6 in. thick and is reinforced typically with #6 bars at 12 in. centers at top and bottom in both the north-south and east-west directions. A cold joint is provided between the turbine pedestal mat and the turbine building mat.

Radwaste Building: The radwaste building mat is about 3 ft. thick and is reinforced typically with #9 bars at 12 in. centers at top and bottom in both the north-south and east-west directions.

3.8.5.2 Applicable Codes, Standards, and Specifications

The codes, standards, and specifications used in the design, fabrication, and construction of foundations of structures are listed in Table 3.8-1.

3.8.5.3 Loads and Load Combinations

The loads and load combinations used in the design of the containment foundation are described in Subsection 3.8.1.3. The loads and load combinations used in the design of foundations of other Seismic Category I structures are discussed in Subsection 3.8.4.3. In addition, the following load combinations are considered to determine the factors of safety against sliding and overturning due to winds, tornadoes, and seismic loads, and against flotation due to groundwater pressure:

a) D+H+W

b) D+H+W'

c) D+H+E

d) D+H+E'

e) D+F where:

D, W, W', E, and E' are as described in Subsections 3.8.1.3 and 3.8.4.3 and H and F are as follows:

H = Lateral earth pressure
F = Buoyant force due to groundwater pressure.

3.8.5.4 Design and Analysis Procedures

The foundations are generally designed to maintain elastic behavior under different loads and their combinations. The loads and the load combinations are described in Subsection 3.8.5.3. The design and analysis of the reinforced concrete mat foundations have been carried out in accordance with ACI 318. Design and analysis of the reinforced concrete mat foundation was also carried out in accordance with ACI 349 for the Diesel Generator 'E' Building. (Refs 10A and 12A of Table 3.8-1.)

The bearing walls and the steel columns carry all the vertical loads from the structure to the foundation mat. The lateral loads are transferred to the shear walls by the roof and floor diaphragms, which then transmit them to the foundation mat. Determination of overturning moment due to seismic loads is discussed in Subsection 3.7b.2.14.

Except for ESSW pumphouse, settlement of the foundations of Seismic Category I structures is considered negligible as the foundations are supported by sound rock. The settlement of the ESSW pumphouse mat is considered in the design and is discussed in Subsection 2.5.4.

As explained in Subsection 3.8.5.1 and shown in Dwg. C-795, Sh. 1, the sides of the foundation mats (except for the ESSW pumphouse) are keyed to the rock by poured concrete, which resists sliding of the mats. Stability against sliding for the ESSW pumphouse is maintained by the friction on the underside of the basemat and passive resistance of the soil against the edge of the mat.

Detailed description of the foundation rock and soil is contained in Subsections 2.5.4 and 2.5.5. For design purposes, the allowable bearing pressures of rock and soil are 40 and 2.5 tons/sq. ft., respectively. The calculated bearing pressures for loads and load combinations described in Subsection 3.8.5.3 do not exceed these allowable values.

The design and analysis of the containment foundation mat are discussed in detail in Subsection 3.8.1.4.

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 and 3.8.4.5. In addition, for the additional load combinations delineated in Subsection 3.8.5.3, the minimum allowable factors of safety against overturning, sliding, and flotation are as follows:

Minimum Factors of Safety			
Load Combination	Overturning	Sliding	Flotation
a) D+H+W	1.5	1.5	-
b) D+H+W'	1.1	1.1	-
c) D+H+E	1.5	1.5	-
d) D+H+E'	1.1	1.1	-
e) D+F	-	-	1.1

The calculated factors of safety exceed the above minimum factor of safety.

3.8.5.6 Materials, Quality Control, and Special Construction Techniques

The foundations of Seismic Category I structures are constructed of reinforced concrete. The concrete and reinforcing steel materials are discussed in Appendix 3.8B. Concrete design compressive strengths are given in Table 3.8-11. Techniques involved in the construction of these foundations are standard construction procedures.

3.8.5.7 Testing and In-service Inspection Requirements

The containment foundation is load tested during the structural acceptance test as described in Subsection 3.8.1.7. An in-service surveillance program to monitor the settlement of the ESSW pumphouse foundation has been instituted. Detailed discussion of the program is contained in Subsection 2.5.4. Testing and in-service inspection is not necessary for foundations of all other Seismic Category I structures.

TABLE 3.8-1

LIST OF APPLICABLE CODES, STANDARDS, RECOMMENDATIONS, AND SPECIFICATIONS

Reference Number	Designation	Title	Edition*
(A) American Concrete Institute			
1A	ACI 211.1	Recommended Practice for Selecting Proportions for Normal and Heavyweight Concrete	1970
2A	ACI 214	Recommended Practice for Evaluation of Compression Test Results of Field Concrete	1965
3A	ACI 301	Specifications for Structural Concrete for Buildings	1972
4A	ACI 304	Recommended Practice for Measuring, Mixing, Transporting, and Placing Concrete	1973
5A	ACI 305	Recommended Practice for Hot Weather Concreting	1972
6A	ACI 306	Recommended Practice for Cold Weather Concreting	1966 (1972)
7A	ACI 307	Specification for the Design and Construction of Reinforced Concrete Chimneys	1969
8A	ACI 308	Recommended Practice for Curing Concrete	1971
9A	ACI 309	Recommended Practice for Consolidation of Concrete	1972
10A	ACI 318	Building Code Requirements for Reinforced Concrete	1971

* Principal editions used are listed; later editions may be applied for specific cases, such as the diesel generator 'E' building.

TABLE 3.8-1 (Continued)

Reference	Designation	Title Number	Edition*
11A	ACI 347	Recommended Practice for Concrete Formwork	1968
12A	ACI 349	Criteria for Reinforced Concrete Nuclear Power Containment Structures (included in ACI Manual of Standard Practice, Part 2, 1973)	-
13A	ACI SP2	Manual of Concrete Inspection	1975
(B) American Welding Society			
1B	AWS D1.1	Structural Welding Code	1972 (Generally all work) 1975 , 1980, 1981 (Some work after June 1975)
2B	AWS D12.1	Recommended Practice for Welding Reinforcing Steel and Connections in Reinforced Concrete Construction	1961
(C) US Nuclear Regulatory Commission			
1C	RG 1.10	Mechanical (Cadweld) Splices in Reinforcing Bars of Category I Concrete Structures	Revision 1 Jan. 1973
2C	RG 1.15	Testing of Reinforcing Bars for Category I Concrete Structures	Revision 1 Dec. 1972
3C	RG 1.18	Structural Acceptance Test for Concrete Primary Reactor Containments	Revision 1 Dec. 1972

* Principal editions used are listed; later editions may be applied for specific cases, such as the diesel generator 'E' building.

TABLE 3.8-1 (Continued)

Reference Number	Designation	Title	Edition*
4C	RG 1.19	Nondestructive Examination of Primary Containment Liner Welds	Revision 1 Aug. 1972
5C	RG 1.54	Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Power Plants	June 1973
6C	RG 1.55	Concrete Placement in Category I Structures	June 1973
7C	RG 1.57	Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components	June 1973
8C	RG 1.58	Qualification of Nuclear Power Plant Inspection, Examination, and Testing Personnel	Aug. 1973
9C	RG 1.69	Concrete Radiation Shields for Nuclear Power Plants	Dec. 1973
10C	RG 1.94	Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants	Apr. 1975
11C**	RG 1.28	Quality Assurance Program Requirements (Design and Construction)	Feb. 79

* Principal editions used are listed; later editions may be applied for specific cases, such as the diesel generator 'E' building.

** Reference used for the diesel generator 'E' building.

TABLE 3.8-1 (Continued)

Reference Number	Designation	Title	Edition*
12C**	RG 1.60 Rev. 1	Design Response Spectra for Seismic Design of Nuclear Power Plants	Dec. 73
13C**	RG 1.61 Rev. 0	Damping Values for Seismic Design of Nuclear Power Plants	Oct. 73
14C**	RG 1.76 Rev. 0	Design Basis Tornado for Nuclear Power Plants	Apr. 74
15C**	RG 1.92 Rev. 1	Combining Modal Responses and Spatial Components in Seismic Response Analysis	Feb. 76
16C**	RG 1.117 Rev. 1	Tornado Design Classification	Apr. 78
17C**	RG 1.132 Rev. 1	Site Investigations for Foundations of Nuclear Power Plants	Apr. 78
18C**	1.142 Rev. 1	Safety-Related Concrete Structures for Nuclear Power Plants (other than Reactor Vessels and Containments)	Oct. 81
(D) American Society for Testing and Materials			
1D	ASTM A519	Seamless Carbon and Alloy Steel Mechanical Tubing	1971, 1974, 1975
2D	ASTM A615	Deformed and Plain Billet Steel Bars for Concrete Reinforcement	1972, 1974, 1975

* Principal editions used are listed; later editions may be applied for specific cases, such as the diesel generator 'E' building.

** Reference used for the diesel generator 'E' building.

Reference	Designation	Title Number	Edition*
3D	ASTM C29	Unit Weight of Aggregate	1971
4D	ASTM C31	Making and Curing Concrete Test Specimens in the Field	1969
5D	ASTM C33	Concrete Aggregates	1971, 1974
6D	ASTM C39	Compressive Strength of Cylindrical Concrete Specimens	1972
7D	ASTM C40	Organic Impurities in Sands for Concrete	1966, 1973
8D	ASTM C87	Effect of Organic Impurities in Fine Aggregate on Strength of Mortar	1969
9D	ASTM C88	Soundness of Aggregates by Use of Sodium Sulfate or Magnesium Sulfate	1971, 1973
10D	ASTM C94	Ready-Mixed Concrete	1973, 1974
11D	ASTM C109	Compressive Strength of Hydraulic Cement Mortars	1973, 1975
12D	ASTM C117	Materials Finer than No. 200 Sieve in Mineral Aggregates by Washing	1969
13D	ASTM C123	Lightweight Pieces in Aggregate	1969
14D	ASTM C127	Specific Gravity and Absorption of Coarse Aggregate	1968, 1973

* Principal editions used are listed; later editions may be applied for specific cases, such as the diesel generator 'E' building.

TABLE 3.8-1 (Continued)

Reference Number	Designation	Title	Edition*
15D	ASTM C128	Specific Gravity and Absorption of Fine Aggregate	1968, 1973
16D	ASTM C131	Resistance to Abrasion of Small Size Coarse Aggregate by Use of the Los Angeles Machine	1969
17D	ASTM C136	Sieve or Screen Analysis of Fine and Coarse Aggregates	1971
18D	ASTM C138	Unit Weight, Yield, and Air Content of Concrete	1973, 1974, 1975
19D	ASTM C142	Clay Lumps and Friable Particles in Aggregates	1971
20D	ASTM C143	Slump of Portland Cement Concrete	1971, 1974
21D	ASTM C150	Portland Cement	1973, 1974, 1976, 1978, 1980
22D	ASTM C215	Fundamental Transverse, Longitudinal, and Torsional Frequencies of Concrete Specimens	1960
23D	ASTM C231	Air Content of Freshly Mixed Concrete by the Pressure Method	1973, 1974, 1975
24D	ASTM C235	Scratch Hardness of Coarse Aggregate Particles	1968
25D	ASTM C260	Air Entraining Admixtures for Concrete	1973, 1974
26D	ASTM C289	Potential Reactivity of Aggregates	1971
27D	ASTM C295	Petrographic Examination of Aggregates for Concrete	1965

* Principal editions used are listed; later editions may be applied for specific cases, such as the diesel generator 'E' building.

Reference	Designation	Title Number	Edition*
28D	ASTM C311	Sampling and Testing Fly Ash for Use as an Admixture in Portland Cement Concrete	1968
29D	ASTM C330	Lightweight Aggregates for Structural Concrete	1969, 1975
30D	ASTM C469	Static Modulus of Elasticity and Poisson's Ratio of Concrete in Compression	1965
31D	ASTM C494	Chemical Admixtures for Concrete	1971
32D	ASTM C566	Total Moisture Content of Aggregate by Drying	1967
33D	ASTM C618	Fly Ash and Raw or Calcined Natural Pozzolans for Use in Portland Cement Concrete	1973
34D	ASTM C637	Aggregates for Radiation Shielding Concrete	1973
(E)	American Association of State Highway and Transportation Officials		
1E	AASHTO T26	Quality of Water to be Used in Concrete	1970
2E	AASHTO T150	Percentage of Particles of Less Than 1.95 Specific Gravity in Coarse Aggregate	1949

* Principal editions used are listed; later editions may be applied for specific cases, such as the diesel generator 'E' building.

Reference	Designation	Title Number	Edition*
3E	AASHTO T161	Resistance of Concrete Specimens to Rapid Freezing and Thawing in Water	1970
(F) US Army Corps of Engineers			
1F	CRD C36	Test for Thermal Diffusivity of Concrete	1973
2F	CRD C39	Test for Coefficient of Linear Thermal Expansion of Concrete	1955
3F	CRD C119	Test for Flat and Elongated Particles in Coarse Aggregate	1953
4F**	CRD C572	Specification for Polyvinylchloride Waterstop	1974
(G) American National Standards Institute			
1G	ANSI N45.2.5	Supplementary QA Requirements for Installation, Inspection and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants.	1972
2G	ANSI N101.6	Concrete Radiation Shields	1972
3G**	ANSI N45.2	Quality Assurance Program Requirements for Nuclear Facilities	1977

* Principal editions used are listed; later editions may be applied for specific cases, such as the diesel generator 'E' building.

** Reference used for the diesel generator 'E' building.

TABLE 3.8-1 (Continued)

Reference Number	Designation	Title	Edition*
4C**	ANSI N45.2.2	Packaging, Shipping, Receiving, Storage and Handling of Items for Nuclear Power Plants	1978
5C**	ANSI N45.2.6	Qualifications of Inspection, Examination and Testing Personnel for the Construction Phase of Nuclear Power Plants	1978
6C**	ANSI N45.2.9	Requirements for Collection, Storage, and Maintenance of Quality Assurance Records for Nuclear Power Plants	1974
7C**	ANSI N45.2.10	Quality Assurance Terms and Definitions	1973
8C**	ANSI N45.2.11	Quality Assurance Requirements for the Design of Nuclear Power Plants	1974
9C**	ANSI N45.2.12	Requirements for Auditing of Quality Assurance Programs for Nuclear Power Plants	1977
10C**	ANSI N45.2.13	Quality Assurance Requirements for Control of Procurement of Items and Services for Nuclear Power Plants	1976
11C**	ANSI N45.2.23	Qualifications of Quality Assurance Program Audit Personnel for Nuclear Power Plants	1978

* Principal editions used are listed; later editions may be applied for specific cases, such as the diesel generator 'E' building.

** Reference used for the diesel generator 'E' building.

Reference	Designation	Title Number	Edition*
(H) American Institute of Steel Construction			
1H	AISC	Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings and Supplement Nos. 1, 2 and 3	1969
2H	AISC	Code of Standard Practice for Steel Buildings and Bridges	1970 (Some work before) 1972 (Generally all work) 1976 (Some work after Sept. 1976)
3H	AISC	Specification for Structural Joints Using ASTM A325 or A490 Bolts	1966,1972 and 1976
4H	AISC	Specification for the design, fabrication and erection of Structural Steel for buildings	1978 (Some work after July 1977)
(J) American Society of Mechanical Engineers			
1J	ASME	ASME Boiler and Pressure Vessel Code, Sections II, III, V, VIII, and IX	1971 with Addenda through Summer 1972
(K) Bechtel Power Corporation, San Francisco, California, Topical Reports			
1K	BC-TOP-1	Containment Building liner Plate Design Report	Revision 1 Dec. 1972

* Principal editions used are listed; later editions may be applied for specific cases, such as the diesel generator 'E' building.

TABLE 3.8-1 (Continued)

Reference	Designation	Title Number	Edition*
2K	BC-TOP-4-A	Seismic Analyses of Structures and Equipment for Nuclear Power Plants	Revision 3 Nov. 1974
3K	BC-TOP-9A	Design of Structures for Missile Impact	Revision 2 Sept. 1974
(L)	International Conference of Building Officials		
1L	UBC	Uniform Building Code	1973, 1976

* Principal editions used are listed; later editions may be applied for specific cases, such as the diesel generator 'E' building.

TABLE 3.8-2

LOAD COMBINATIONS FOR PRIMARY CONTAINMENT AND DRYWELL FLOOR

Notations:

S = Required capacity of the section based on the working stress design method and the allowable stresses in ACI 318-71, Section 8.10 except that the maximum allowable tensile stress for reinforcement shall be 0.5 Fy, where Fy is specified yield strength of reinforcing steel.

U = Required capacity of the section based on the strength design method described in ACI 318-71.

D = Dead load

L = Live load

T_t = Thermal effects anticipated at time of structural acceptance test.

T_o = Thermal effects during normal operating conditions including temperature gradients and equipment and pipe reactions.

T_a = Added thermal effects (over and above operating thermal effects) which occur during a design accident.

P = Design basis accident pressure load

R = Local force or pressure on structure due to postulated pipe rupture including the effects of steam/water jet impingement, pipe whip, pipe reaction, steam pressurization, and water flooding.

E = Load due to Operating Basis Earthquake.

E' = Load due to Safe Shutdown Earthquake.

B = Hydrostatic loading due to post-LOCA flooding of the primary containment to the reactor core.

P' = Pressure of atmosphere in the primary containment with the containment flooded to the reactor core.

P_v = External Pressure Load

The primary containment and drywell floor are designed for the following load combinations:

Condition

Preoperational Testing $S = 1.0D+1.0L+1.0T_t+1.15P$

TABLE 3.8-2 (Continued)

Normal	$U = 1.4D+1.7L+1.0T_o + 1.0\ P_v$*
Normal/Severe	$U = 0.75(1.4D+1.7L+1.9E)+1.0T_o + 1.0\ P_v$*
Abnormal	$U = 1.05D+1.05L+1.0(T_o+T_a)+1.0R+1.5P$
Abnormal/Severe	$U = 1.05D+1.05L+1.0(T_o+T_a)+1.0R+1.25P+1.25E$
Abnormal/Extreme	$U = 1.0D+1.0L+1.0(T_o+T_a)+1.0R+1.0P+1.0E'$
Abnormal/Severe (Post-LOCA flooding)	$U = 1.05D+1.0B+1.25P'+1.25E$

*This load was not considered along with other loads in the original design. Since P_v is small, relative to other loads, it may be combined with other loads without affecting the design.

The containment liner plate and anchorages are designed for all loads and load combinations listed above except that all load factors are 1.0.

TABLE 3.8-3

LOAD COMBINATIONS AND ALLOWABLE STRESSES FOR ASME CLASS MC COMPONENTS
(For definitions of loads, see Table 3.8-2)

The drywell head assembly, equipment hatches, personnel lock, suppression chamber access hatches, and CRD removal hatch are designed for the following loading combinations and allowable stresses:

Condition		Allowable Stress
Preoperational Testing	$D+L+T_t+1.15P$	1.15 times ASME, Section III, Class MC for "Design Conditions"
Abnormal	$D+L+(T_o+T_a)+P$	ASME, Section III, Class MC for "Design Conditions"
Abnormal/Severe	$D+L+(T_o+T_a)+P+R+E$	ASME, Section III, Figure NB-3224-1, for "Emergency Conditions"
Abnormal/Extreme	$D+L+(T_o+T_a)+P+R+E'$	ASME, Section III, Figure NB-3225-1 for "Faulted Conditions"

The MC components are also designed for external pressure loads according to ASME, Section III, Subsection NE-3133.

The pipe and electrical penetrations are designed for the following load combinations and allowable stresses:

a) The loads used in the design are as follows:

1) Moments and forces transmitted by the piping to the penetration due to thermal expansion, weight, earthquake (including inertial effects and anchor movements) and other dynamic loads.
2) Pressures
3) Thermal transients
4) Number of operating cycles
5) Pipe failure loads for faulted condition

b) The loading combinations are specified in Section 3.9.

c) Stress limits specified in ASME Code, Section III, Article NB-3220 are used as the design criteria for Class I flued heads for design, normal and upset, and emergency condition. The rules contained in ASME Code, Section III, Appendix F are used in evaluating the faulted condition for Class I and II flued heads.

TABLE 3.8-4

LOAD COMBINATION FOR THE REACTOR SHIELD WALL
(For definitions of loads, see Table 3.8-2)

The reactor shield wall is designed for the following loading combination:

Condition

Abnormal/Extreme	$D+L+(T_o+T_a)+R+P+E'$

TABLE 3.8-5

LOAD COMBINATIONS FOR THE SUPPRESSION CHAMBER COLUMNS
(For definitions of loads, see Table 3.8-2)

The suppression chamber columns are designed for the following loading combinations:

Condition	
Normal/Severe	$1.7D+1.7L+1.7E$
Normal/Severe	$1.3(D+L+E+T_o)$
Normal/Extreme*	$D+L+T_o+E'$
Abnormal	$1.05D+1.05L+1.0(T_o+T_a)+1.0R+1.5P$
Abnormal/Severe	$1.05D+1.05L+1.0(T_o+T_a)+1.0R+1.25P+1.25E$
Abnormal/Extreme	$1.0D+1.0L+1.0(T_o+T_a)+1.0R+1.0P+1.0E'$

---

*Allowable stresses	=	90% of the values given in Subsection 3.8.3.5.3 for extreme environmental and abnormal loading conditions.
Section strength required for stability	=	90% of the allowables given in Part 2 of the AISC Specification, 1969 (Ref. 1H of Table 3.8-1).

TABLE 3.8-6

LOAD COMBINATIONS FOR THE DRYWELL PLATFORMS
(For definitions of loads, see Table 3.8-2)

The drywell platforms are designed for the following loading combinations:

Condition	
Normal	D+L
Abnormal	D+L+R

TABLE 3.8-7

LOAD COMBINATION FOR THE SEISMIC TRUSS
(For definitions of loads, see Table 3.8-2)

The seismic truss is designed for the following loading combination:

Condition	
Abnormal/Extreme	D+R+E'

TABLE 3.8-8

LOAD COMBINATIONS APPLICABLE TO REACTOR BUILDING

Notations

W = Wind load
W' = Tornado wind load
Wms= Site proximity missile load (Diesel Generator 'E' Building only)
fs = Calculated stress in structural steel
Fs = Allowable stress for structural steel
Fy = Yield strength of structural steel
H_o = Force on structure due to thermal expansion of pipes under operating conditions
H_a = Force on structure due to thermal expansion of pipes under accident conditions
D_s = Force on blockwall due to story drift under Operating Basis Earthquake Loading
D'_s = Force on blockwall due to story drift under Safe Shutdown Earthquake Loading
S_m = Allowable stress for reinforced concrete masonry per UBC, Table 24-H (special inspection) for global wall analysis; or allowable stress for unreinforced concrete masonry per UBC Table 24-B (special inspection) for local wall analysis as a result of attachments.
f_s = Allowable working stress in tension for reinforcing steel (as specified in UBC).
f_y = Yield strength of reinforcing steel.

For all other notations, see Table 3.8-2.

A. Reinforced Concrete

Normal operating loads:

$$U = 1.4D+1.7L+1.0T_o + 1.25\ H_o$$

Normal operating loads with Severe environmental loads:

$$U = 0.75[1.4D+1.7L+1.7(1.1)E]+1.0T_o + 1.25\ H$$

$$U = 0.75(1.4D+1.7L+1.7W)+1.0T_o + 1.25\ H$$

Where overturning forces cause net tension in the absence of live load, the following load combinations are considered:

$$U = 0.9D+1.3(1.1)E+1.0T_o + 1.25\ H_o$$

$$U = 0.9D+1.3W+1.0T_o + 1.25\ H_o$$

For structural shear walls carrying seismic forces, the following load combination is also considered:

$$U = 1.0D+1.0L+1.8E+1.0T_o + 1.25\ H_o$$

Normal operating loads with Extreme environmental loads:

$$U = 1.0D+1.0L+1.0T_o +1.0W' + 1.0\ H_o$$

Normal operating loads with Abnormal loads:

$$U = 1.05D+1.05L+1.0(T_o+T_a)+1.0R+1.5P + 1.0\ H_o$$

Normal operating loads with Severe environmental and Abnormal loads:

$$U = 1.05D+1.05L+1.0(T_o+T_a)+1.0R+1.25P+1.25E + 1.0\ H_o$$

Where overturning forces cause net tension in the absence of live load, the following load combination is considered:

$$U = 0.95D+1.25E+1.0(T_o+T_a)+1.0R + 1.0\ H_o$$

Normal operating loads with Extreme environmental and Abnormal loads:

$$U = 1.0D+1.0L+1.0(T_o+T_a)+1.0E'+1.0P+1.0R + 1.0\ H_a$$

$$U = 1.0D+1.0L+1.0T_o+1.0E'+1.0R + 1.25\ H_a$$

TABLE 3.8-8 (Continued)

B. Structural Steel

Condition	Load Combination	Allowable Stress Increase
Normal operating loads:	$D + L + T_o + H_o$	Fs
Normal operating loads with Severe environmental loads:	$D + L + T_o + E + H_o$	1.25 Fs
	$D + L + T_o + W + H_o$	1.33 Fs
Normal operating loads with Extreme environmental loads:	$D + L + T_o + W' + H_o$	See note below
Normal operating loads with Extreme environmental and Abnormal loads:	$D+L+R+T_o + E'+P+H_o$	See note below
	$D + L + R + (T_o + T_a) + P + E' + H_a$	See note below

Note: The allowable stress in structural steel does not exceed 0.9 Fy in bending, 0.85 Fy in axial tension or compression, and 0.5 Fy in shear. Where Fs is governed by requirements of stability (local or lateral buckling), fs does not exceed 1.5 Fs.

C. Concrete Masonry Structures (Blockwalls)

Safety related blockwalls in category I structures other than the reactor building are designed for the following load combinations and allowable stress increase. The load combinations apply to out-of-plane loading as well as in-plane loading. Acceptance criteria is in accordance with Section 3.8.4.5.

Condition	Load Combination	Allowable Stress Increase
Normal	$D + L + T_o + H_a$	No increase
Normal/Severe	$D + L + T_o + H_o + E + D_s$	No increase
Normal/Extreme	$D + L + T_o + H_o + W'$	See Table 3.8-12
Abnormal	$D + L + (T_o + T_a) + R + H_a + 1.25P$	See Table 3.8-12
Abnormal/Severe	$D+L+(T_o + T_a)+R+H_a +1.25E+D_s$	See Table 3.8-12
Abnormal/Extreme	$D+L+(T_o +T_a)+R+H_a +E'+D'_s$	See Table 3.8-12

LOAD COMBINATIONS APPLICABLE TO SEISMIC CATEGORY I
STRUCTURES OTHER THAN CONTAINMENT, REACTOR BUILDING
AND DIESEL GENERATOR 'E' BUILDING

Notations: See Tables 3.8-2 and 3.8-8

A. Reinforced Concrete

Normal operating loads:

$$U = 1.4D+1.7L+1.0T_o + 1.25\ H_o$$

Normal operating loads with Severe environmental loads:

$$U = 0.75(1.4D+1.7L+1.7(1.1E))+1.0T_o + 1.25\ H_o$$

$$U = 0.75(1.4D+1.7L+1.7W)+1.0T_o + 1.25\ H_o$$

Where overturning forces cause net tension in the absence of live load, the following load combinations are considered:

$$U = 0.9D+1.3(1.1E)+1.0T_o + 1.25\ H_o$$

$$U = 0.9D+1.3W+1.0T_o + 1.25\ H_o$$

For structural elements carrying mainly seismic forces:

$$U = 1.0D+1.0L+1.8E+1.0T_o + 1.25\ H$$

Normal operating loads with Extreme environmental loads:

$$U = 1.0D+1.0L+1.0W'+1.0T_o + 1.0\ H$$

Normal operating loads with Severe environmental and Abnormal loads:

$$U = 1.05D+1.05L+1.25E+1.0(T_o+T_a)+1.0R + 1.0\ H$$

Where overturning forces cause net tension in the absence of live load, the following load combination is considered:

$$U = 0.95D+1.25E+1.0(T_o+T_a)+1.0R + 1.0\ H$$

TABLE 3.8-9 (Continued)

Normal operating loads with Extreme environmental and Abnormal loads:

$$U = 1.0D+1.0L+1.0E'+1.0T_o +1.0R + 1.25 H_o$$

$$U = 1.0D+1.0L+1.0E'+1.0(T_o+T_a)+1.0R + 1.0 H_a$$

B. Structural Steel

Condition	Load Combination	Allowable Stress
Normal operating loads:	$D+L+T_o+H_o$	Fs
Normal operating loads with Severe environmental loads:	$D+L+T_o+E+H_o$ $D+L+T_o+W+H_o$	1.25 Fs 1.33 Fs
Normal operating loads with Extreme environmental loads:	$D+L+T_o+W'+H_o$	See note below
Normal operating loads with Extreme environmental and Abnormal loads:	$D+L+R+T_o+E'+H_o$ $D+L+R+T_o+T_a+E'+H_a$	See note below See note below

Note: The allowable stress in structural steel does not exceed 0.9 Fy in bending, 0.85 Fy in axial tension or compression, and 0.5 Fy in shear. Where Fs is governed by requirements of stability (local or lateral buckling), fs does not exceed 1.5 Fs.

C. Concrete Masonry Structures (Blockwalls)

Safety related blockwalls in the reactor building are designed for the following load combinations and allowable stress increase. The load combinations apply to out-of-plane loading as well as in-plane loading. Acceptance criteria is in accordance with Section 3.8.4.5.

Condition	Load Combination	Allowable Stress Increase
Normal	$D + L + T_o + H_o$	No increase
Normal/Severe	$D + L + T_o + H_o + E + D_s$	No increase
Normal/Extreme	$D + L + T_o + H_o + W'$	See Table 3.8-12
Abnormal	$D + L + (T_o + T_a) + R + 1.5P + H_o$	See Table 3.8-12
Abnormal/Severe	$D + L + (T_o + T_a) + R + 1.25P + H_a + 1.25E + D_s$	See Table 3.8-12
Abnormal/Extreme	$D + L + (T_o + T_a) + R + P + H_a + D'_s + E'$	See Table 3.8-12

TABLE 3.8-10

LOAD COMBINATIONS APPLICABLE TO TURBINE & RADWASTE BUILDING

Notation: See Tables 3.8-2 and 3.8-8

A. Reinforced Concrete

Normal Operating Loads:

$$U = 1.4D + 1.7L + 1.0\ T_o + 1.25\ H_o$$

Normal Operating Loads with severe environmental loads:

$$U = 0.75(1.4D+1.7L+1.7W)+1.0T_o + 1.25H_o$$

Where overturning forces cause net tension in the absence of live load, the following load combination is considered:

$$U = 0.9D+1.3W+1.0T_o+1.25H_o$$

B. Structural Steel

Condition	Load Combination	Allowable Stress
Normal Operating Loads	$D+L+T_o+H_o$	Fs
Normal Operating Loads with Severe environmental loads	$D+L+T_o+H_o+W$	1.33 Fs

The turbine and radwaste buildings are also designed to prevent collapse under SSE and tornado loadings. The following load combinations are used when SSE and tornado loadings are considered:

Reinforced Concrete

$$U = 1.0D+1.0L+1.0W'+1.0T_o+1.0H_o$$
$$U = 1.0D+1.0L+1.0E'+1.0T_o+1.25H_o$$

Structural Steel

Load Combination	Allowable Stress
$D+L+T_o+W'+H_o$	See note below.
$D+L+T_o+E'+H_o$	See note below.

Note: The Allowable stress in structural steel does not exceed 0.9 Fy in bending, 0.85 Fy in axial tension or compression, and 0.5 Fy in shear. Where fs is governed by requirements of stability (local or lateral buckling), fs does not exceed 1.5 Fs.

TABLE 3.8-11

CONCRETE DESIGN COMPRESSIVE STRENGTHS

Structure	Concrete Design Compressive Strength, f'c (psi)
Turbine generator pedestal	3000
All other Seismic Category I and safety-related, non-Seismic Category I structures and their associated foundation mats including: a) Containment (including its internal structures) b) Reactor Building c) Control Building d) Diesel Generator 'A-D' Building e) Diesel Generator 'E' Building f) ESSW Pumphouse g) Spray Pond h) Turbine Building i) Radwaste Building	4000

TABLE 3.8-12

ALLOWABLE STRESS INCREASE FACTOR FOR MASONRY STRUCTURES

STRESS	INCREASE FACTOR	COMMENT
Axial or flexural compression	1.67	
Bearing	1.67	
Reinforcement stress except shear	1.67	See Note 1
Shear Reinforcement and/or bolts	1.5	
Masonry tension parallel to bed joint	1.5	
Shear carried by masonry	1.0	See Note 2
Masonry tension perpendicular to bed joint		
For reinforced masonry	0	
For unreinforced masonry	-	Not applicable

1) Shall not exceed .90 fy.

2) The actual shear stress carried by masonry is in accordance with masonry walls acceptance criteria in section 3.8.4.5 with no increase factor applied.

TABLE 3.8-2a

Security-Related Information
Text Withheld Under 10 CFR 2.390

TABLE 3.8-3a

COMPARISON OF FSAR AND SRP LOAD COMBINATIONS AND ALLOWABLE STRESSES FOR ASME CLASS MC COMPONENTS

SRP Section 3.8.2.II.3.b Combination No.	Comparative Load Combination from FSAR Table 3.8-3	SRP or FSAR	Comparison of Allowable Stresses (ksi): Primary Stresses: General Memb. P_m	Primary Stresses: Local Memb. P_L	Primary Stresses: Bend. & Local Memb. $P_B + P_L$	Primary and Secondary Stresses	Peak Stresses	Buckling
(1)	Preoperational Testing	SRP	$.9\ S_y = 34.2$	$1.25\ S_y = 47.5$	$1.25\ S_y = 47.5$	$3\ S_m = 57.9$	Consider for fatigue analysis	125% of allowable given by NE-3133
		FSAR	$1.15\ S_m = 22.2$	$1.15 \times 1.5\ S_m = 33.3$	$1.15 \times 1.5\ S_m = 33.3$	$3\ S_m = 57.9$	N/A	N/A
(2) and (3)	-	SRP	$S_m = 19.3$	$1.5\ S_m = 29.0$	$1.5\ S_m = 29.0$	$3\ S_m = 57.9$	Consider for fatigue analysis	Allowable given by NE-3133
		FSAR	-	-	-	-	-	-
-	Abnormal	SRP	-	-	-	-	-	-
		FSAR	$S_m = 19.3$	$1.5\ S_m - 29.0$	$1.5\ S_m = 29.0$	$3\ S_m = 57.9$	N/A	N/A
(4)	Abnormal/Severe	SRP	$S_m = 19.3$	$1.5\ S_m = 29.0$	$1.5\ S_m = 29.0$	N/A	N/A	Allowable given by NE-3133
		FSAR*	$S_y = 29.4$	$1.5\ S_y - 44.1$	$1.5\ S_y = 44.1$	N/A	N/A	N/A
(5)	-	SRP	$S_m = 19.3$	$1.5\ S_m = 29.0$	$1.5\ S_m = 29.0$	N/A	N/A	Allowable given by NE-3133
		FSAR	-	-	-	-	-	Allowable given by NE-3133

TABLE 3.8-3a (cont'd)

SRP Section 3.8.2.II.3.b Combination No.	Comparative Load Combination from FSAR Table 3.8-3	SRP or FSAR	Comparison of Allowable Stresses (ksi): Primary Stresses: General Memb. P_m	Primary Stresses: Local Memb. P_L	Primary Stresses: Bend. & Local Memb. $P_B + P_L$	Primary and Secondary Stresses	Peak Stresses	Buckling
(6)	-	SRP*	$S_y = 29.4$	$1.5\ S_y = 44.1$	$1.5\ S_y = 44.1$	N/A	N/A	120% of allowable given by NE-3133
		FSAR	-	-	-	-	-	-
(7)	-	SRP	$S_y = 29.4$	$1.5\ S_y = 44.1$	$1.5\ S_y = 44.1$	N/A	N/A	120% of allowable given by NE-3133
		FSAR	-	-	-	-	-	Allowable given by NE-3133
(8)	Abnormal/Extreme	SRP*	$S_m = 0.85 \times 0.70 \times S_u = 32.2$	$1.5\ S_m = 48.3$	$1.5\ S_m = 48.3$	N/A	N/A	35% of allowable given by F-1325 of Appendix F
		FSAR*	$S_y = 29.4$	$1.5\ S_y - 44.1$	$1.5\ S_y - 44.1$	N/A	N/A	N/A
(9)	-	SRP	$1.5\ S_m = 29.0$	$1.5\ S_y = 57.0$	$1.5\ S_y = 57.0$	N/A	N/A	120% of allowable given by NE-3133
		FSAR	-	-	-	-	-	-

*Integral and Continuous

SRP Section 3.8.2.II.3.b Combination No.	Comparative Load Combination from FSAR Table 3.8-3	Conclusions
(1)	Preoperational Testing	Since load combinations are identical and FSAR allowable stresses are less than or equal to SRP allowable stresses, FSAR criteria is as conservative as SRP criteria.
(2) and (3)	-	SRP load combinations (2) and (3) need not be considered since FSAR "Abnormal" load combination causes higher actual stresses and considers the same allowable stresses.
-	Abnormal	See Above
(4)	Abnormal/Severe	FSAR load combination includes pipe rupture loads (including effects of steam/water jet impingement, pipe whip, and pipe reaction) and SRP load combination does not include these loads. FSAR allowable stresses are 52% larger than SRP allowable stresses. Pipe rupture loads increase actual stresses by at least 52%. Therefore, FSAR criteria is as conservative as SRP criteria.
(5)	-	Since SRP and FSAR used the same buckling allowable, FSAR is as conservative as SRP.
(6)	-	SRP load combination does not include pipe rupture loads. Since FSAR "Abnormal/Extreme" load combination includes pipe rupture loads and uses the same allowable stresses as SRP load combination, SRP load combination need not be considered.
(7)	-	Since FSAR buckling allowable is less than SRP buckling allowable, FSAR criteria is more conservative than SRP criteria.
(8)	Abnormal/Extreme	Since load combinations are identical and FSAR allowable stresses are less than SRP allowable stresses, FSAR criteria is more conservative than SRP criteria.
(9)	-	Based on the following reasons, SRP load combination (9) is less critical than FSAR "Abnormal/Extreme" load combination and, therefore, need not be considered: 1. SRP allowable stresses for load combination (9) are similar to FSAR allowable stresses for "Abnormal/Extreme" load combination. 2. Hydrostatic pressures due to post-LOCA flooding are less than design basis accident pressure. 3. OBE seimic loads during post-LOCA flooding are similar to SSE seismic loads for FSAR "Abnormal/Extreme" load combination. 4. SRP load combination (9) does not include pipe rupture loads.

TABLE 3.8-9a

LOAD COMBINATIONS APPLICABLE TO DIESEL GENERATOR 'E' BUILDING

(See Tables 3.8-2 and 3.8-8 for definitions of loads and other notations)

The Diesel Generator 'E' Building is designed for the following load combinations:

A. Reinforced Concrete

Service Load Combinations:

a. $U = 1.4D + 1.7L$

b. $U = 1.4D + 1.7L + 1.9E$

c. $U = 1.4D + 1.7L + 1.7W$

d. $U = 1.2D + 1.9E$

e. $U = 1.2D + 1.7W$

Where soil or hydrostatic pressures are present and have been included in L and D, in addition to all the preceding combinations, the requirements of Sections 9.2.4 and 9.2.5 of ACI 318.77 have been satisfied.

Factored Load Combinations:

a. $U = 1.0D + 1.0L + 1.0E'$

b. $U = 1.0D + 1.0L + 1.0W_t$

c. $U = 1.0D + 1.0L + 1.0W_{ms}$

Regarding preceding loads which are variable, the full range of variation has been considered in order to determine the most critical combination of loading.

B. Structural Steel

The following combinations of loadings have been considered in the design of structural steel seismic Category I structures. S is the required section strength based on the elastic design methods and the allowable stresses defined in Part I of American Institute of Steel Construction (AISC) Specification for the Design, Fabrication and Erection of

Structural Steel for Buildings, November, 1978, except that the 33-percent increase in allowable stresses for seismic or wind loadings has not been permitted. In determining the most critical loading condition to be used in design, the absence of a load or loads has been considered as appropriate.

Service Load Combinations

a. $S = D + L$

b. $S = D + L + E$

c. $S = D + L + W$

Factored Load Combinations

a. $1.6S = D+L+E'$

b. $1.6S = D+L+W_t$

c. $1.6S = D+L+W_{ms}$

THIS FIGURE HAS BEEN
REPLACED BY DWG.
C-331, Sh. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

Figure 3.8-1 replaced by dwg.
C-331, Sh. 1

FIGURE 3.8-1, Rev. 48

Figure Fsar 3_8_1.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
C-371, Sh. 2

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Figure 3.8-2 replaced by dwg. C-371, Sh. 2
FIGURE 3.8-2, Rev. 48

AutoCAD Figure 3_8_2.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
C-1932, Sh. 3

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

Figure 3.8-3 replaced by dwg.
C-1932, Sh. 3

FIGURE 3.8-3, Rev. 55

THIS FIGURE HAS BEEN
REPLACED BY DWG.
C-1932, Sh. 4

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SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

Figure 3.8-4 replaced by dwg.
C-1932, Sh. 4

FIGURE 3.8-4, Rev. 55

AutoCAD Figure 3_8_4.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
C-1932, Sh. 5

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SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

Figure 3.8-5 replaced by dwg.
C-1932, Sh. 5

FIGURE 3.8-5, Rev. 55

AutoCAD Figure 3_8_5.doc

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DELETED

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

Figure Deleted

FIGURE 3.8-6, Rev. 54

AutoCAD Figure 3_8_6.doc

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DELETED

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

Figure Deleted

FIGURE 3.8-7, Rev. 54

AutoCAD Figure 3_8_7.doc

THIS FIGURE HAS BEEN
DELETED

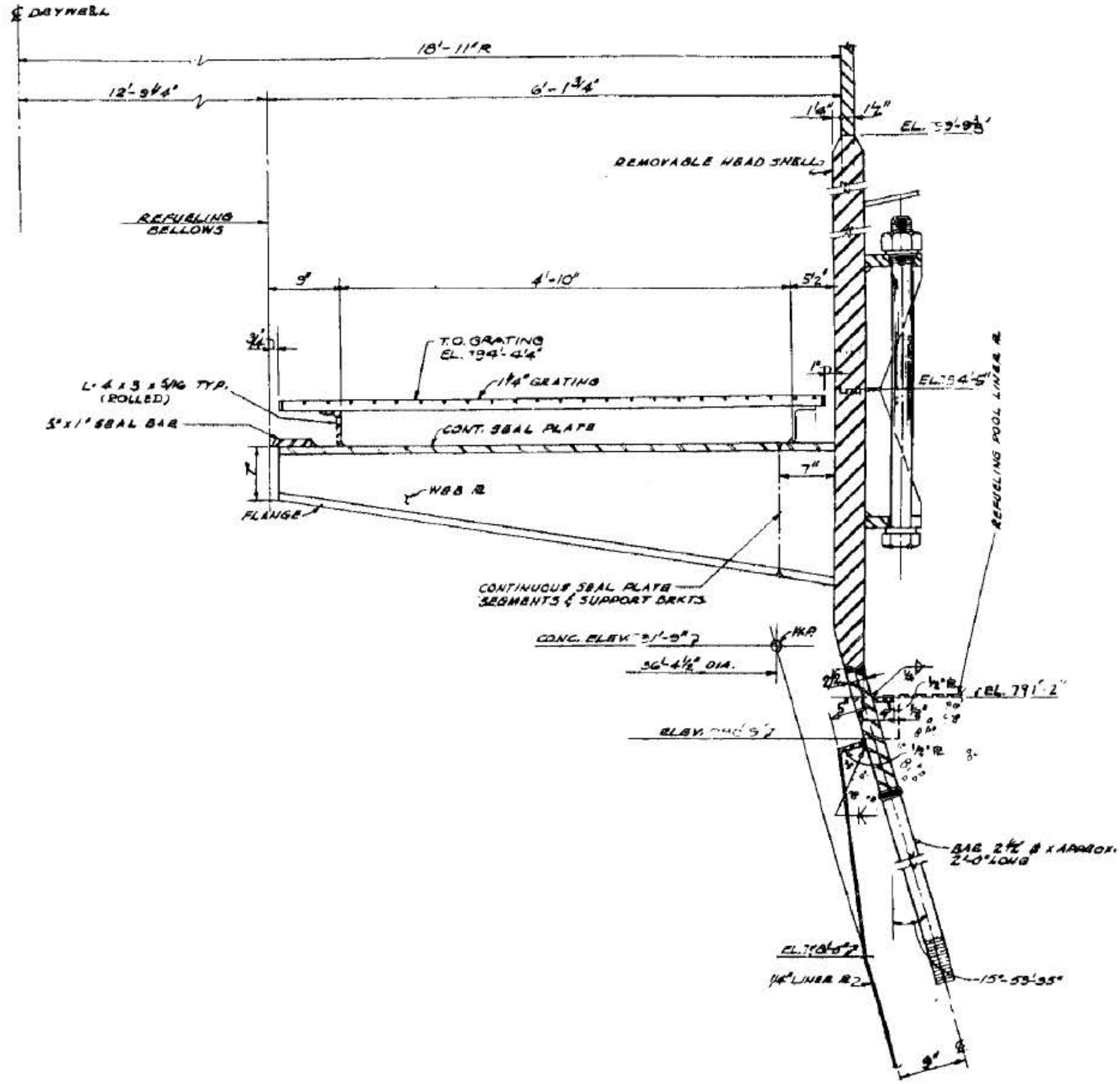
FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

Figure Deleted

FIGURE 3.8-8, Rev. 54

AutoCAD Figure 3_8_8.doc



FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

PRIMARY CONTAINMENT
DRYWELL HEAD CONNECTION

FIGURE 3.8-9, Rev. 47

Auto-Cad Figure Fsar 3_8_9.dwg

THIS FIGURE HAS BEEN
REPLACED BY DWG.
C-284, Sh. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

Figure 3.8-10 replaced by dwg.
C-284, Sh. 1

FIGURE 3.8-10, Rev. 48

Figure Fsar 3_8_10.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
C-281, Sh. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Figure 3.8-12 replaced by dwg. C-281, Sh. 1 FIGURE 3.8-12, Rev. 48

Figure Fsar 3_8_12.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
C-281, Sh. 1

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SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

Figure 3.8-13 replaced by dwg.
C-281, Sh. 1

FIGURE 3.8-13, Rev. 55

THIS FIGURE HAS BEEN
REPLACED BY DWG.
C-370, Sh. 1

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SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

Figure 3.8-14 replaced by dwg.
C-370, Sh. 1

FIGURE 3.8-14, Rev. 48

Figure Fsar 3_8_14.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
C-282, Sh. 1

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SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

Figure 3.8-17 replaced by dwg.
C-282, Sh. 1

FIGURE 3.8-17, Rev. 48

AutoCAD Figure 3_8_17.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
C-285, Sh. 1

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SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

Figure 3.8-18 replaced by dwg.
C-285, Sh. 1

FIGURE 3.8-18, Rev. 48

AutoCAD Figure 3_8_18.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
C-286, Sh. 1

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SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

Figure 3.8-21 replaced by dwg.
C-286, Sh. 1

FIGURE 3.8-21, Rev. 48

AutoCAD Figure 3_8_21.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
C-291, Sh. 1

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SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

Figure 3.8-22 replaced by dwg.
C-291, Sh. 1

FIGURE 3.8-22, Rev. 48

AutoCAD Figure 3_8_22.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
C-278, Sh. 1

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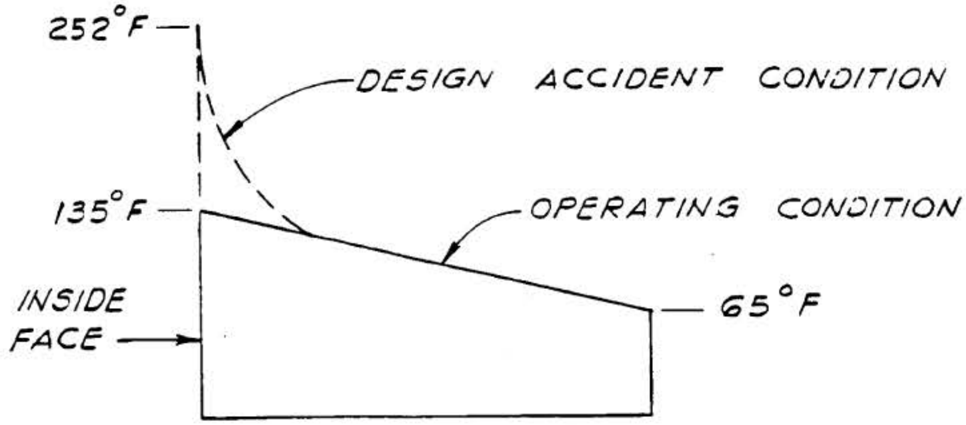
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

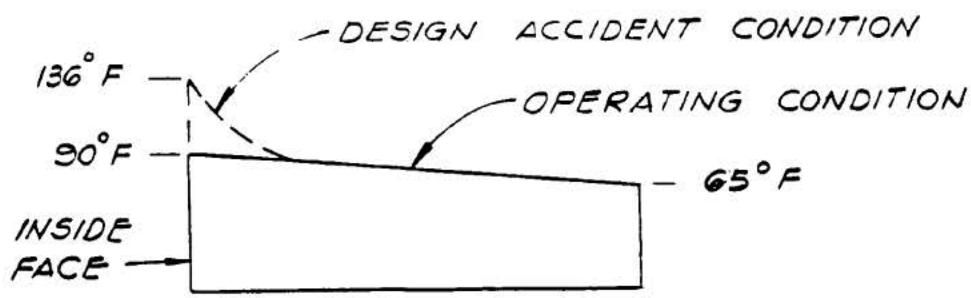
Figure 3.8-23 replaced by dwg.
C-278, Sh. 1

FIGURE 3.8-23, Rev. 55

AutoCAD Figure 3_8_23.doc



DRYWELL WALL



SUPPRESSION CHAMBER WALL

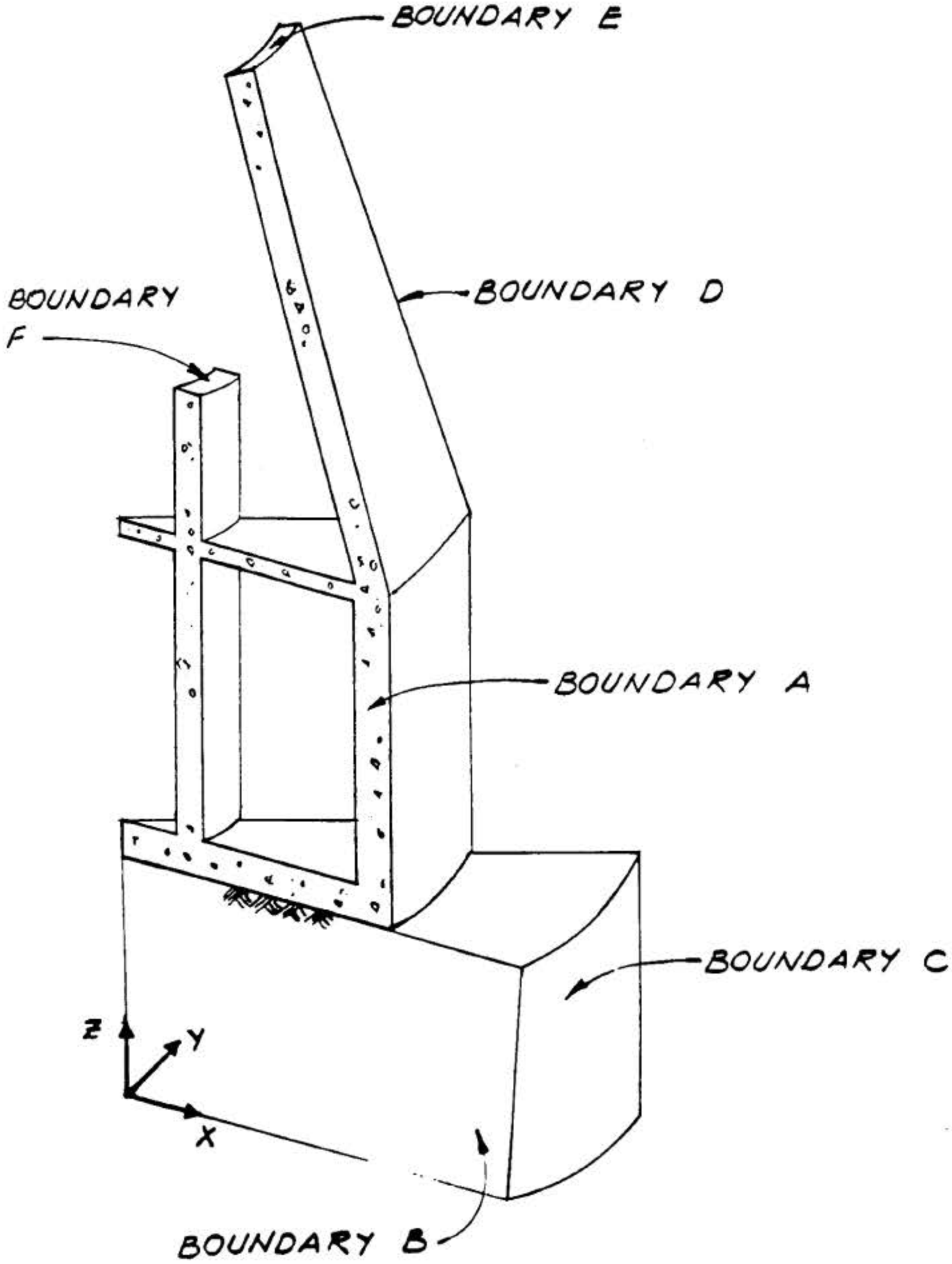
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

CONTAINMENT WALL
TEMPERATURE GRADIENTS

FIGURE 3.8-24, Rev. 47

Auto-Cad Figure Fsar 3_8_24.dwg



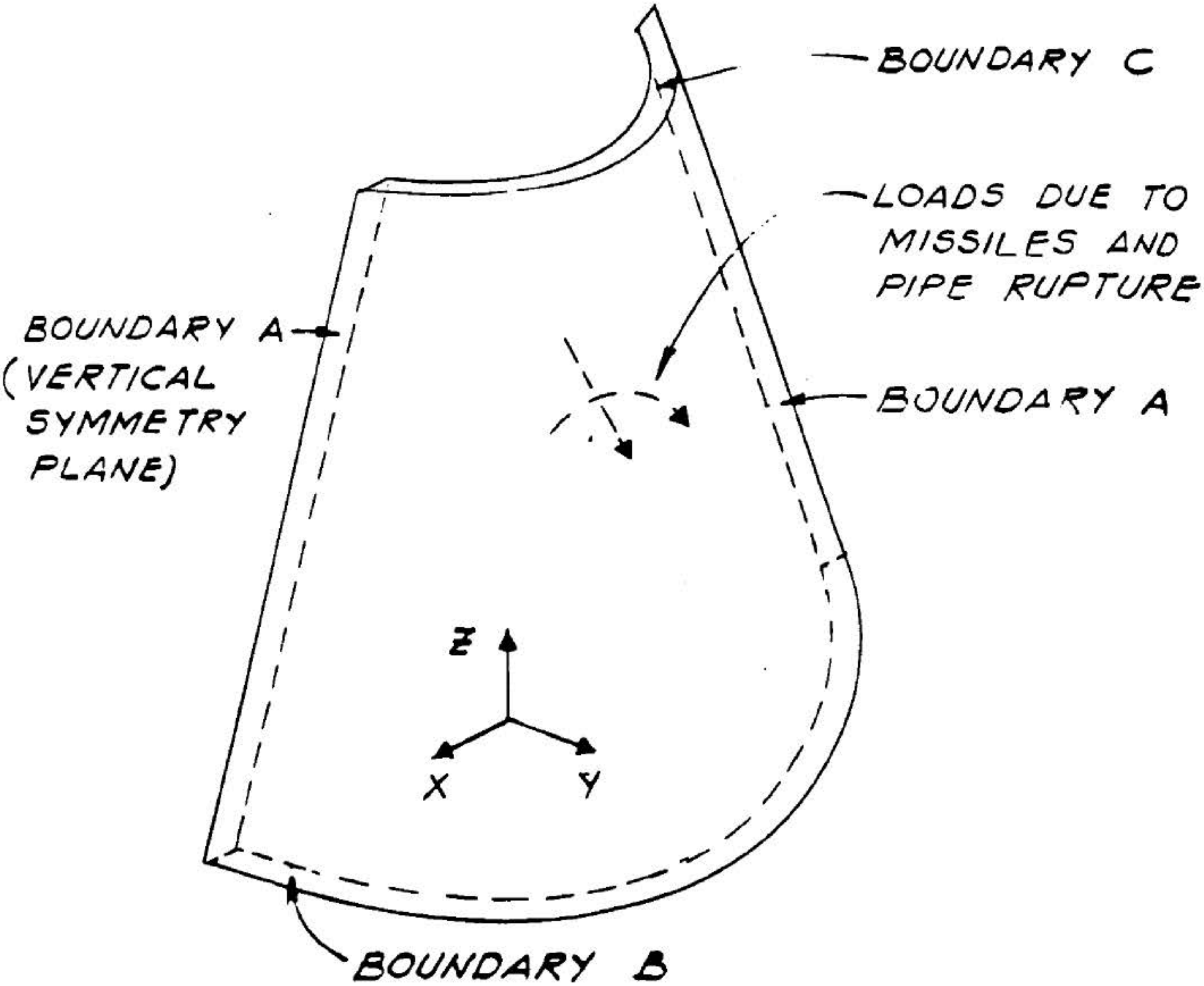
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

CONTAINMENT WALL
ANALYTICAL MODEL FOR
AXISYMMETRIC LOADS

FIGURE 3.8-25, Rev. 47

Auto-Cad Figure Fsar 3_8_25.dwg



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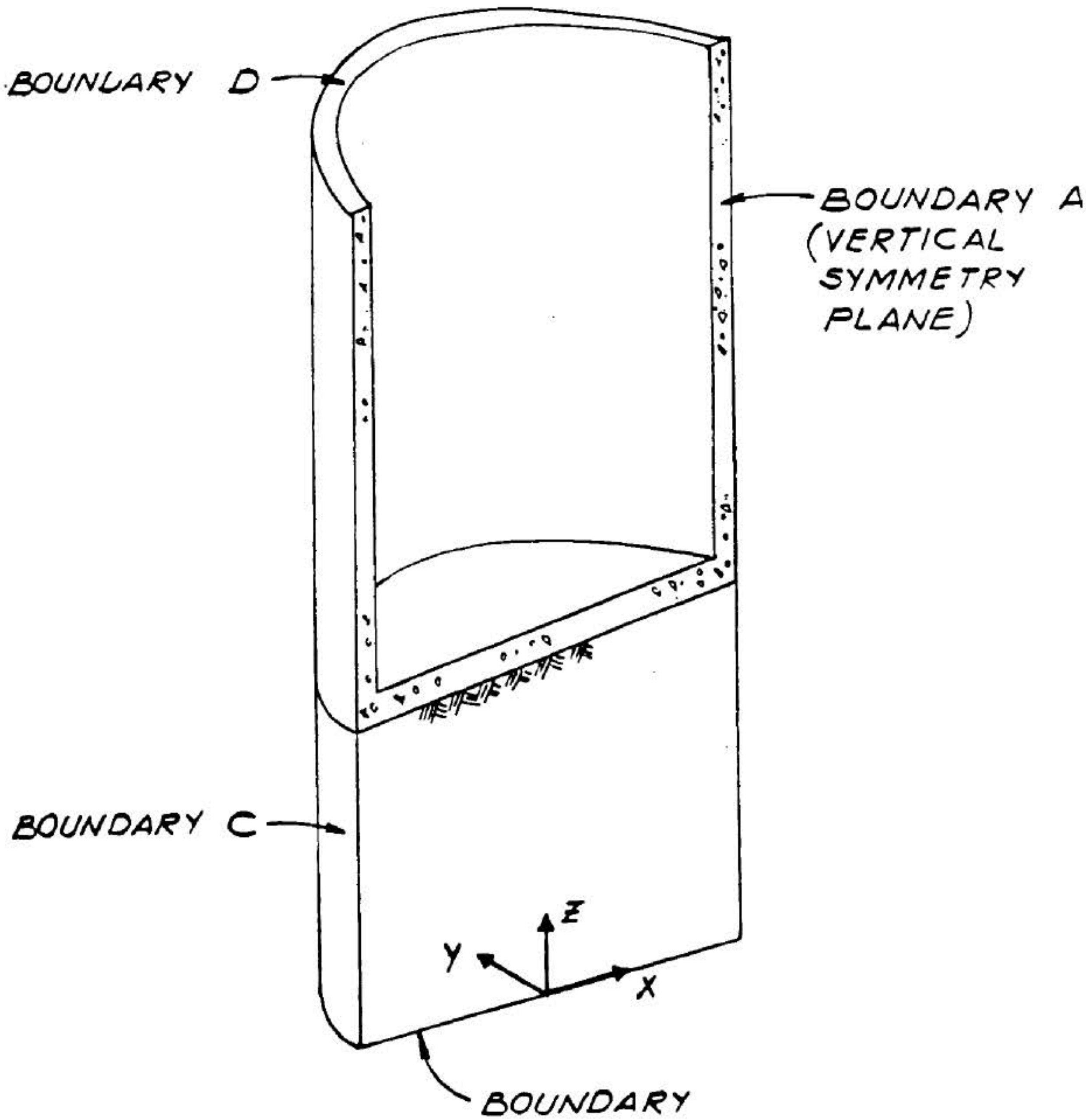
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

DRYWELL WALL MODEL
FOR NON-AXISYMMETRIC MISSILE &
POSTULATED PIPE RUPTURE LOADS

FIGURE 3.8-26, Rev. 47

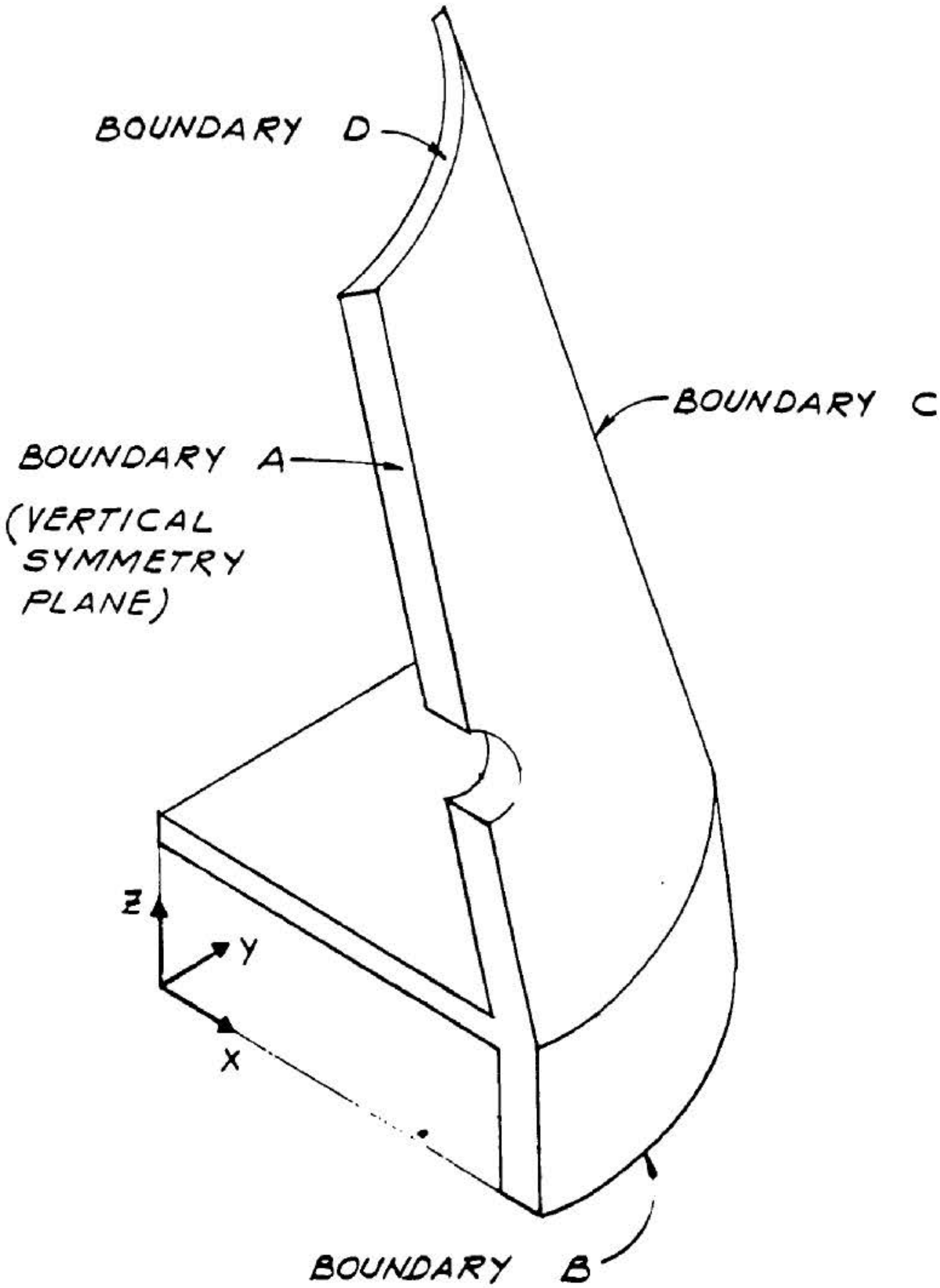
Auto-Cad Figure Fsar 3_8_26.dwg



FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
BASE FOUNDATION SLAB ANALYTICAL MODEL FIGURE 3.8-27, Rev. 47

Auto-Cad Figure Fsar 3_8_27.dwg



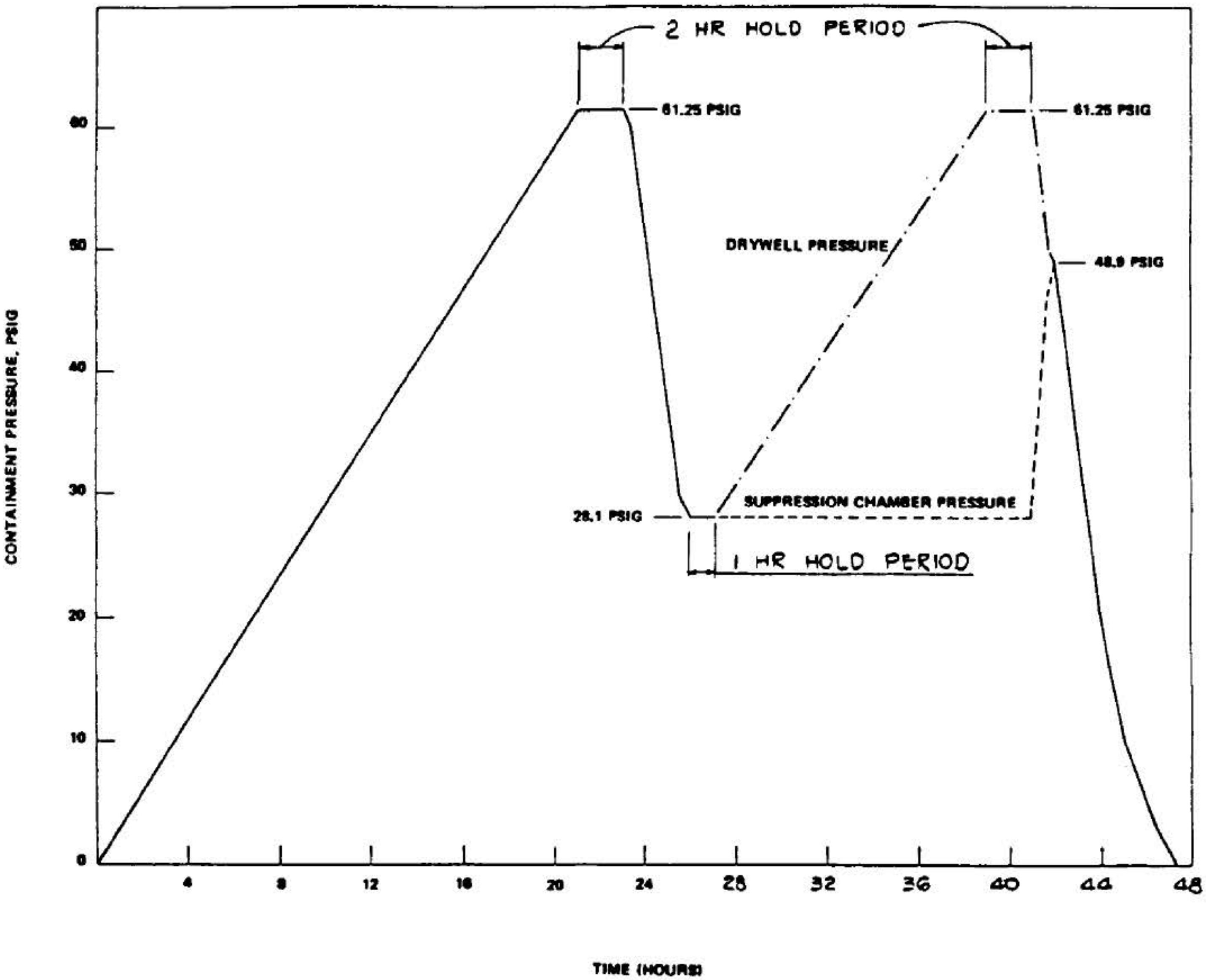
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

EQUIPMENT HATCH
ANALYTICAL MODEL

FIGURE 3.8-28, Rev. 47

Auto-Cad Figure Fsar 3_8_28.dwg



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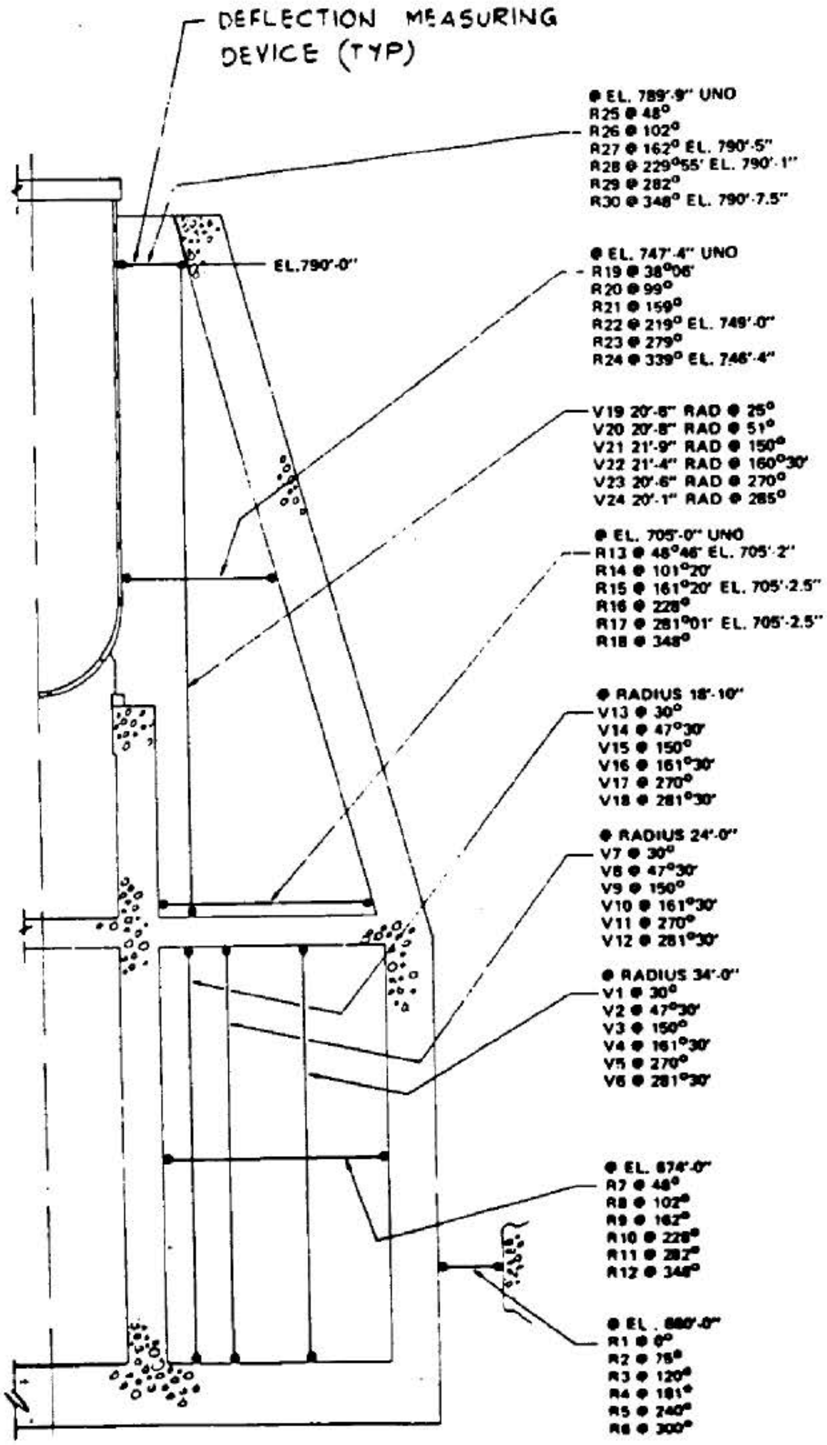
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

STRUCTURAL ACCEPTANCE TEST
PRESSURIZATION SEQUENCE

FIGURE 3.8-29, Rev. 47

Auto-Cad Figure Fsar 3_8_29.dwg



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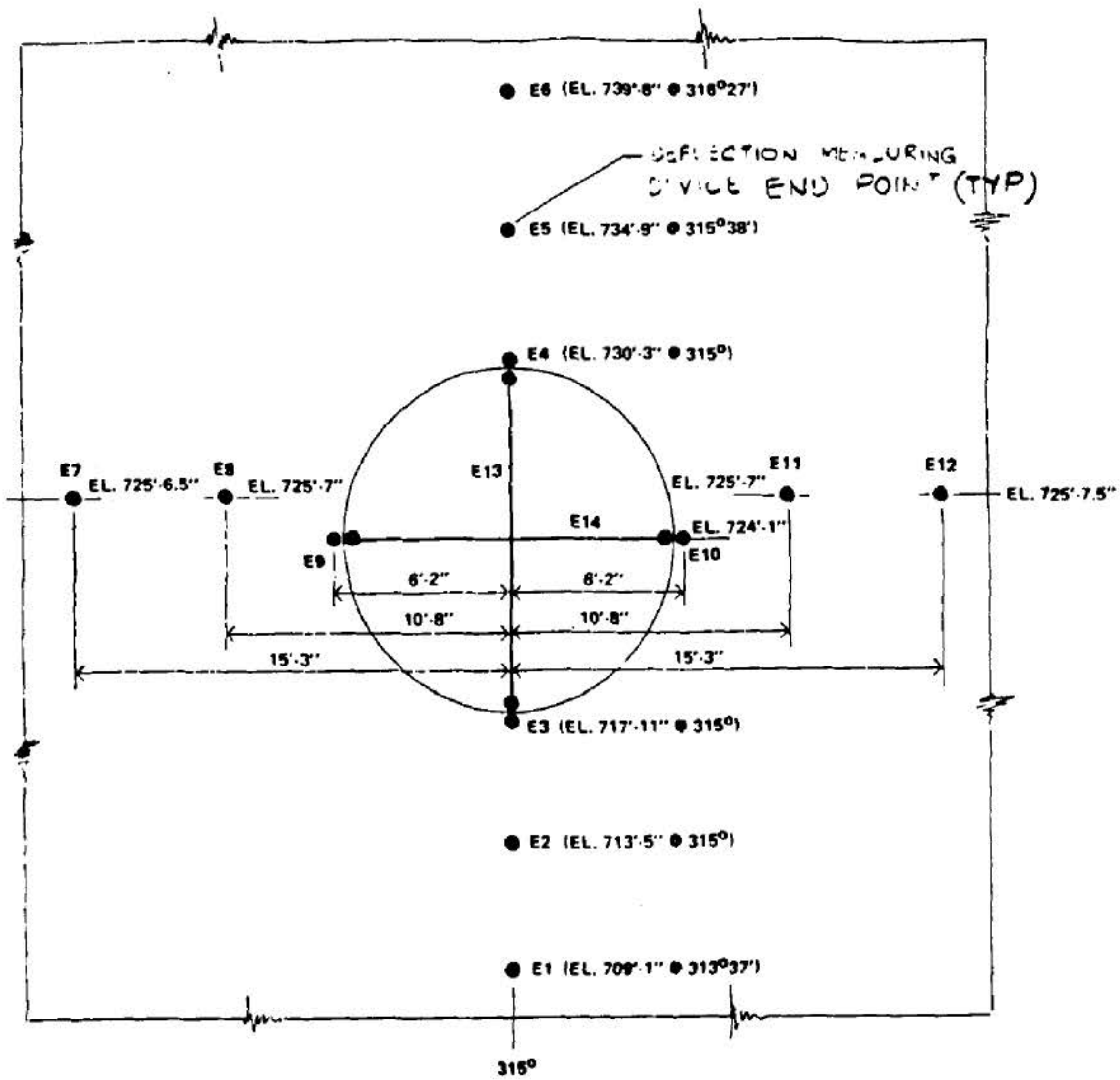
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

STRUCTURAL ACCEPTANCE TEST
LOCATION OF DEFLECTION
MEASURING DEVICES FOR
THE CONTAINMENT

FIGURE 3.8-30, Rev. 47

Auto-Cad Figure Fsar 3_8_30.dwg



E13 & 14 ACROSS HORIZONTAL & VERTICAL OPENING DIAMETERS
(VERTICAL WIRE PARALLEL TO CONE SURFACE)

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SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

STRUCTURAL ACCEPTANCE TEST
LOCATION OF DEFLECTION
MEASURING DEVICES FOR
EQUIPMENT HATCH

FIGURE 3.8-31, Rev. 47

Auto-Cad Figure Fsar 3_8_31.dwg

THIS FIGURE HAS BEEN
REPLACED BY DWG.
C-384, Sh. 1

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SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

Figure 3.8-32 replaced by dwg.
C-384, Sh. 1

FIGURE 3.8-32, Rev. 48

AutoCAD Figure 3_8_32.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
C-387, Sh. 1

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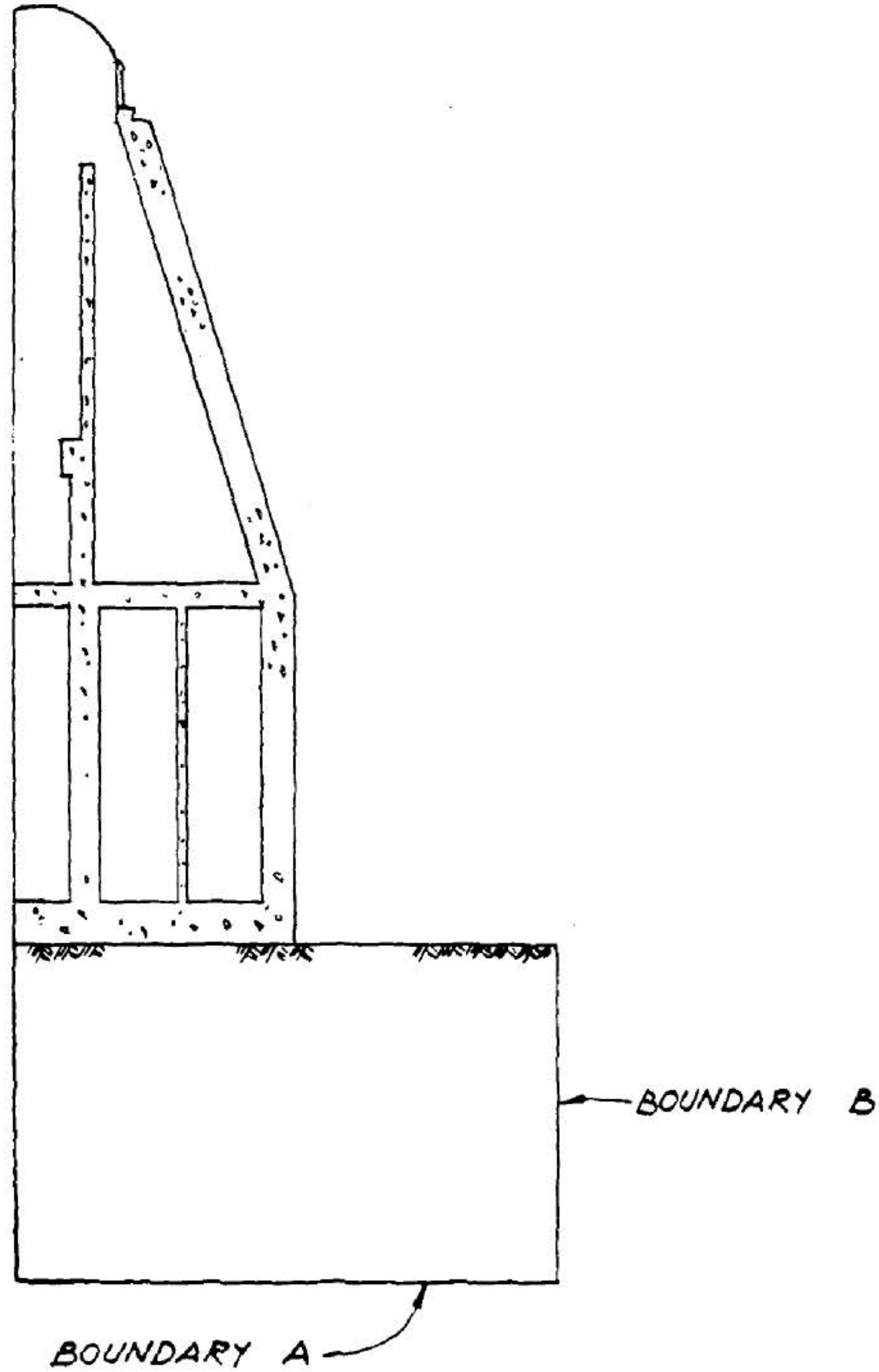
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

Figure 3.8-33 replaced by dwg.
C-387, Sh. 1

FIGURE 3.8-33, Rev. 48

AutoCAD Figure 3_8_33.doc



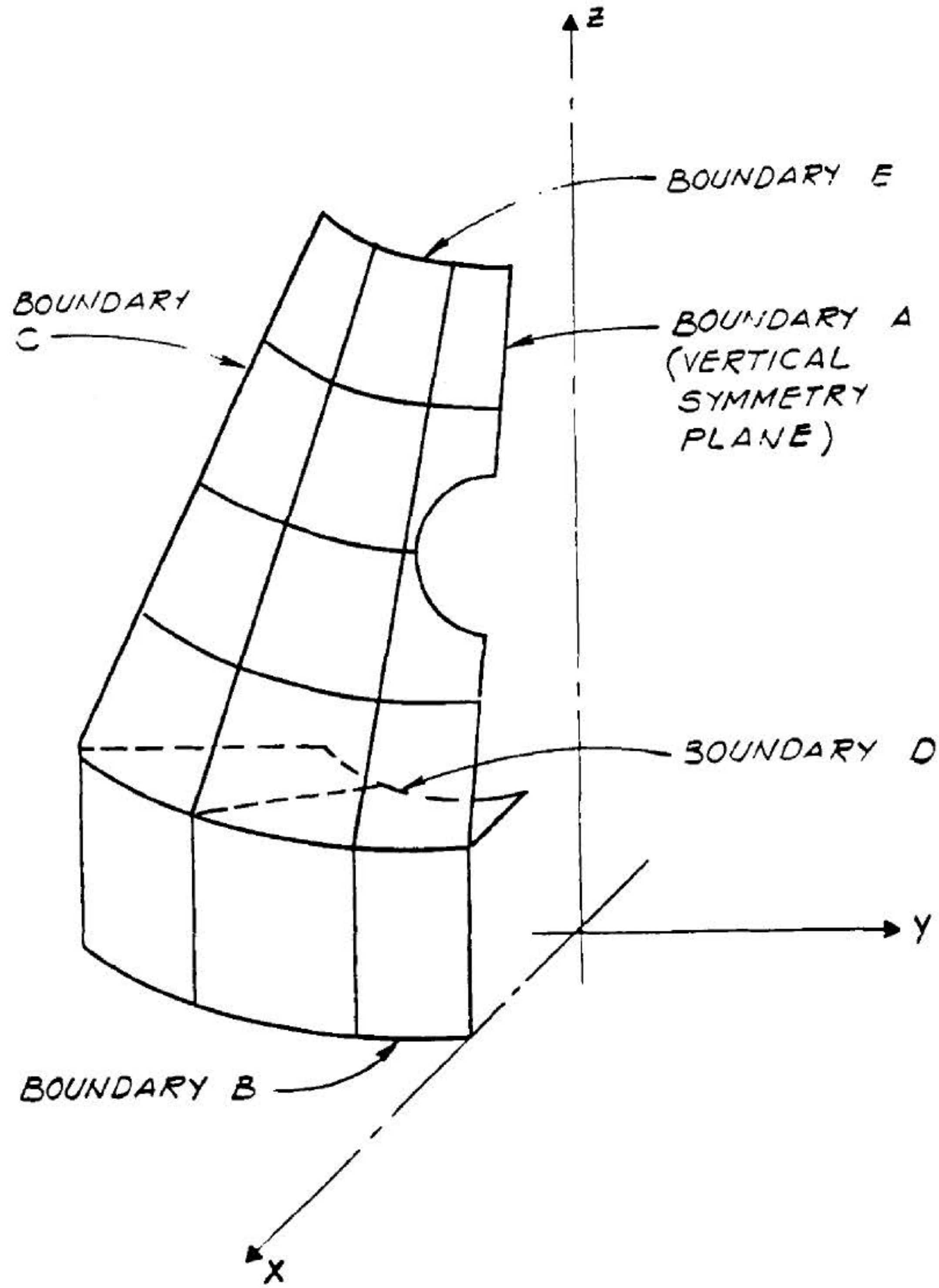
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

STRUCTURAL ACCEPTANCE TEST
CONTAINMENT ANALYTICAL MODEL

FIGURE 3.8-34, Rev. 47

Auto-Cad Figure Fsar 3_8_34.dwg



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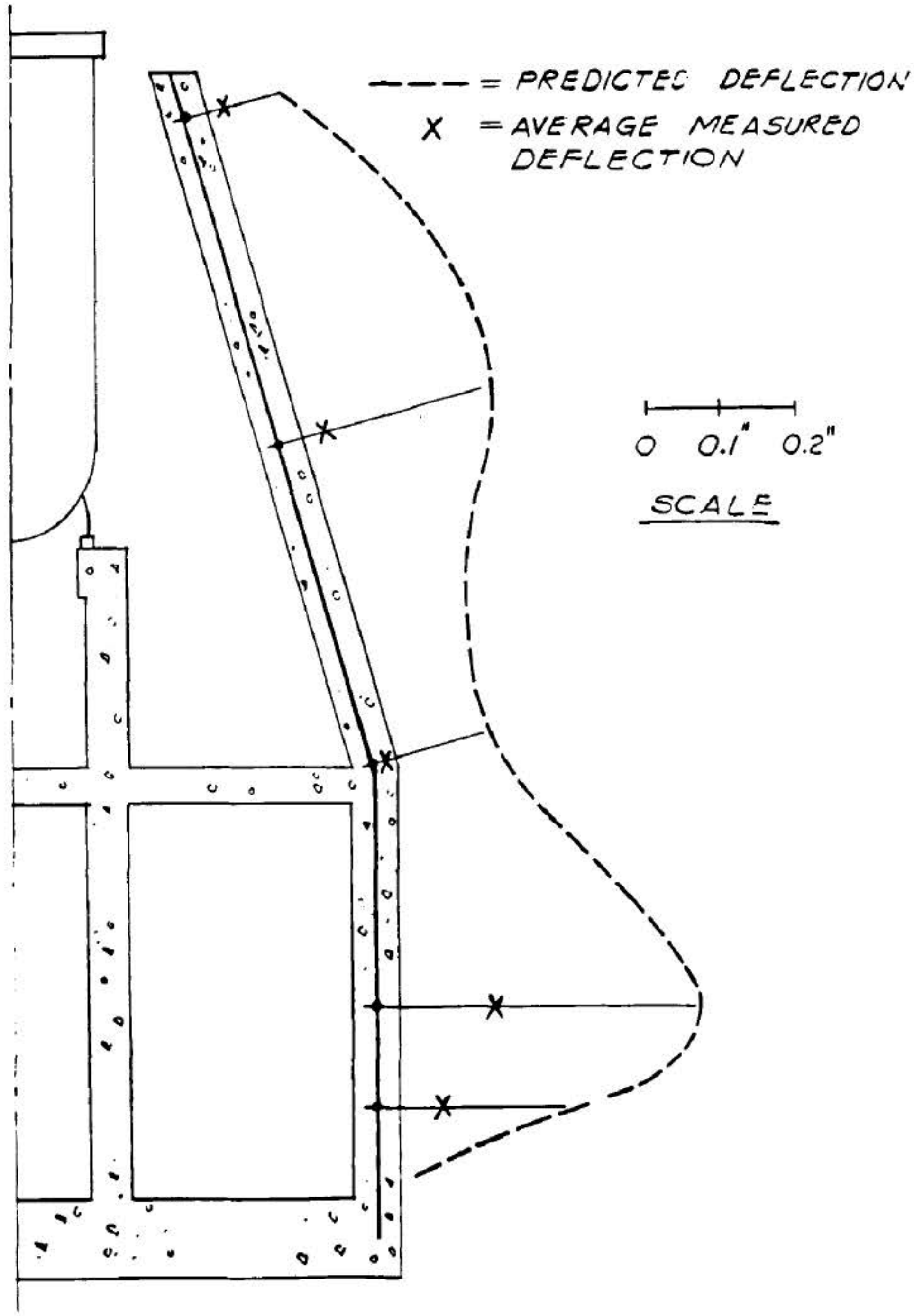
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

STRUCTURAL ACCEPTANCE TEST
EQUIPMENT HATCH
ANALYTICAL MODEL

FIGURE 3.8-35, Rev. 47

Auto-Cad Figure Fsar 3_8_35.dwg



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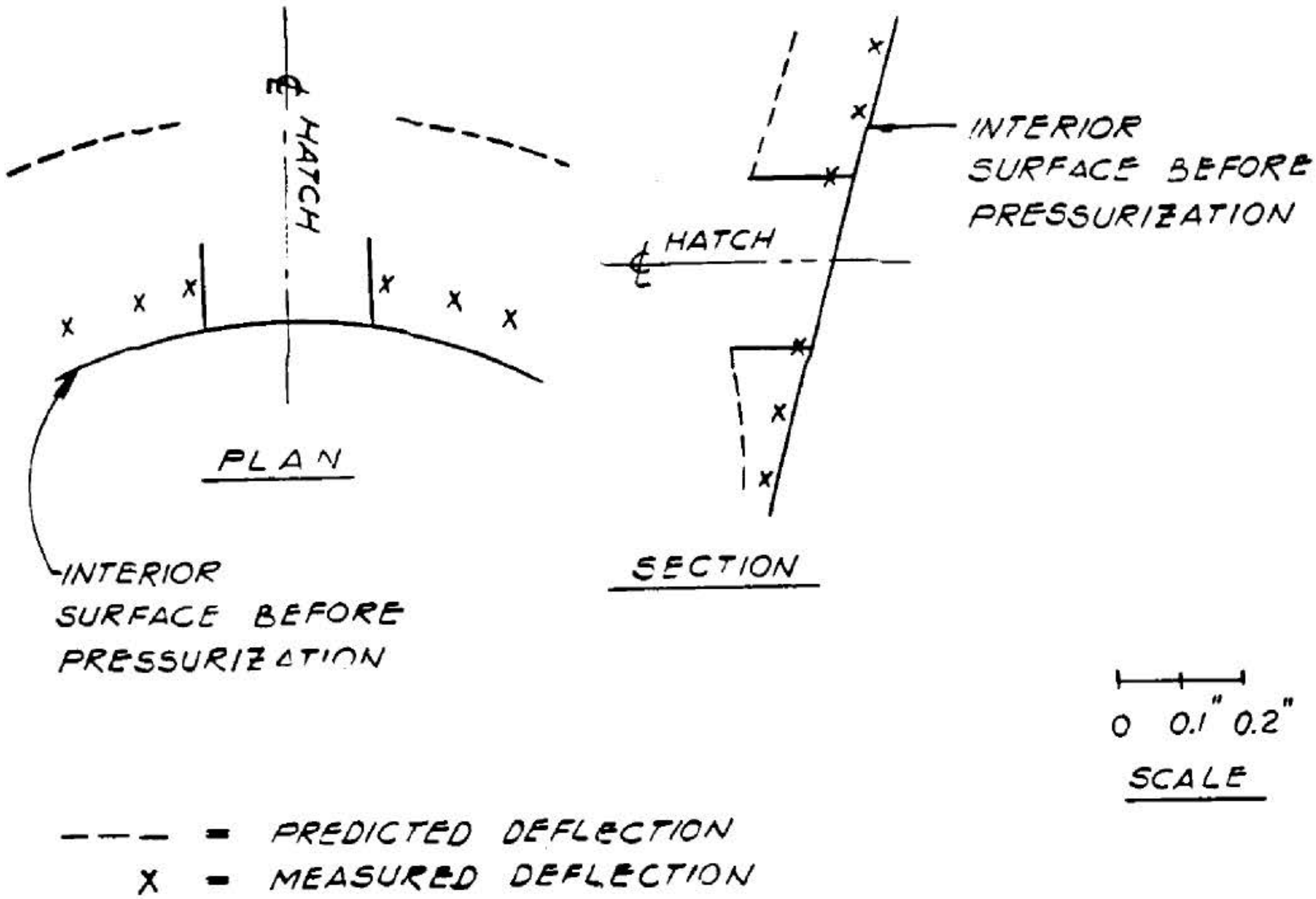
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

STRUCTURAL ACCEPTANCE TEST
COMPARISON OF MEASURED &
PREDICTED DEFLECTIONS
FOR THE CONTAINMENT

FIGURE 3.8-36, Rev. 47

Auto-Cad Figure Fsar 3_8_36.dwg



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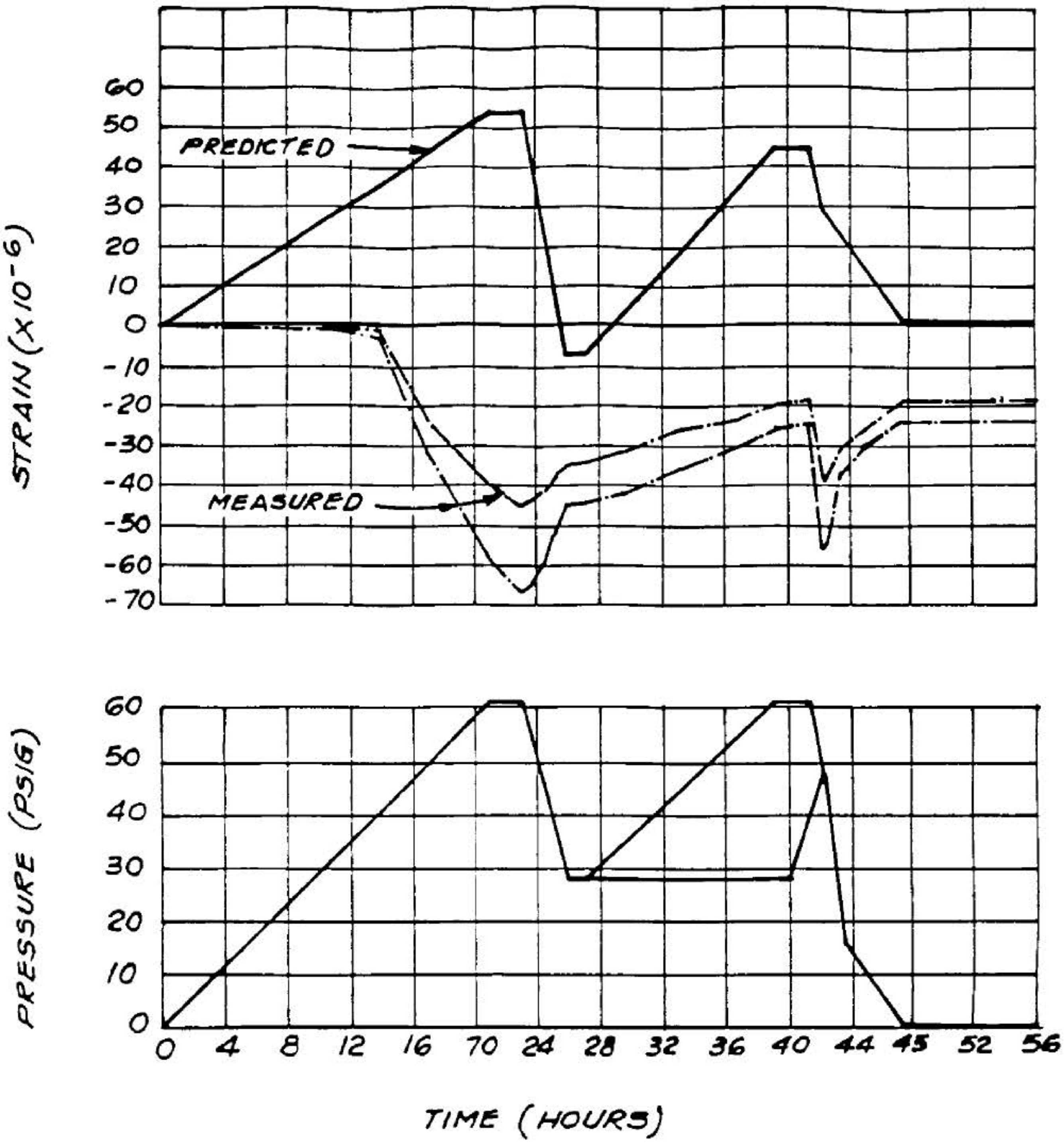
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

STRUCTURAL ACCEPTANCE TEST
COMPARISON OF MEASURED &
PREDICTED DEFLECTION
FOR THE EQUIPMENT HATCH

FIGURE 3.8-37, Rev. 47

Auto-Cad Figure Fsar 3_8_37.dwg



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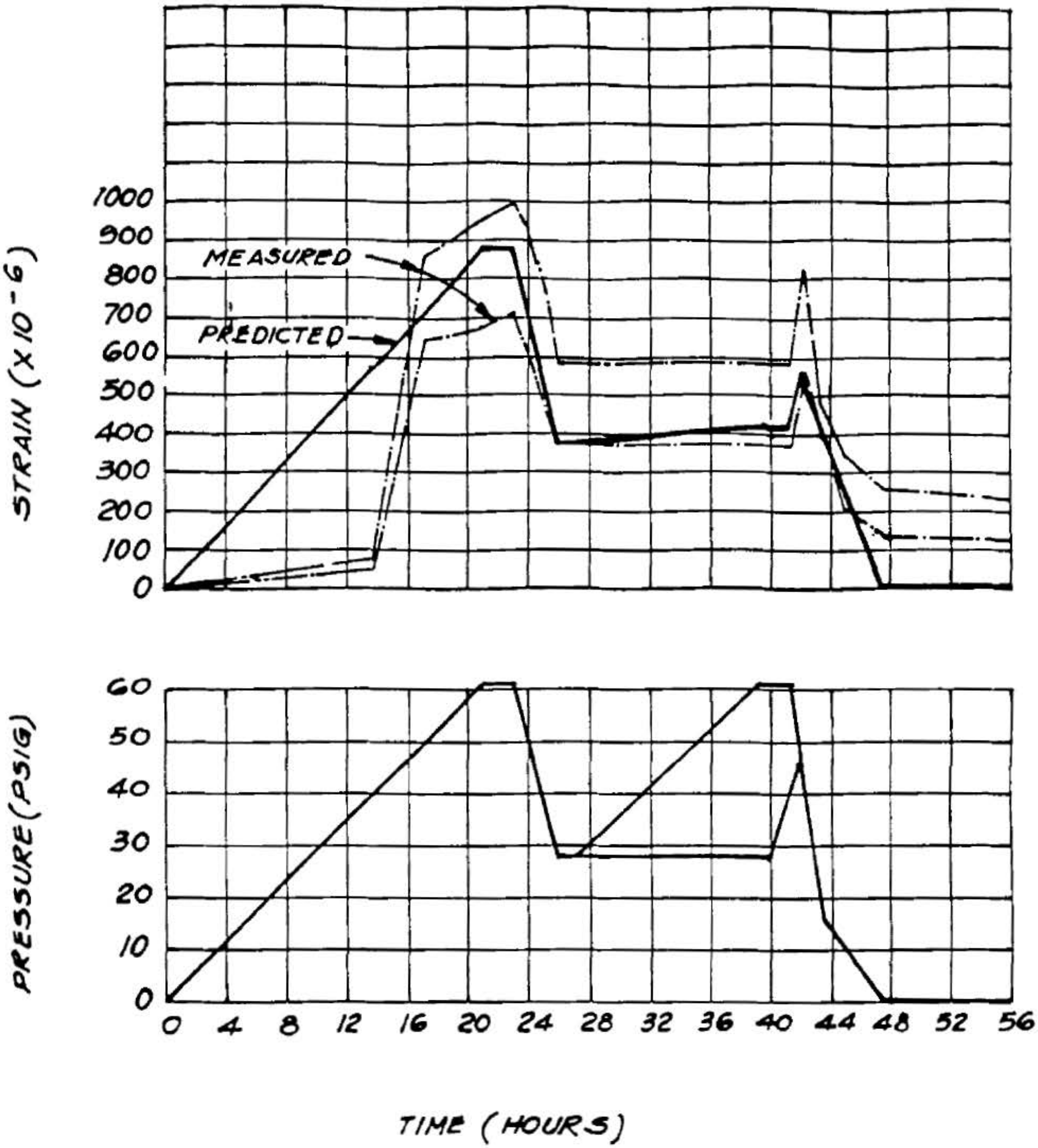
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

STRUCTURAL ACCEPTANCE TEST
INSIDE MERIDIONAL STRAIN AT
MID-HEIGHT OF
SUPPRESSION CHAMBER WALL

FIGURE 3.8-38, Rev. 47

Auto-Cad Figure Fsar 3_8_38.dwg



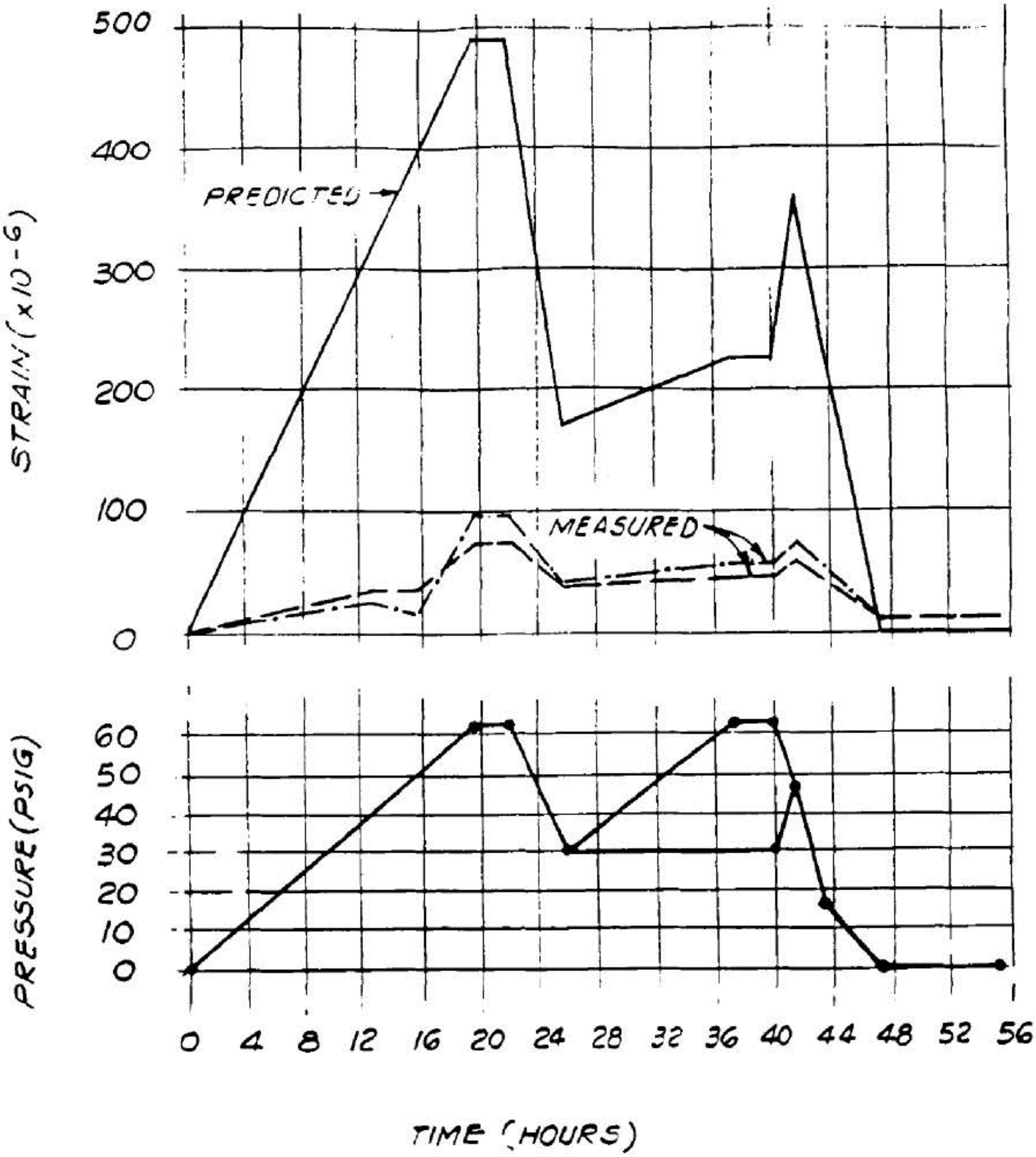
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

STRUCTURAL ACCEPTANCE TEST
INSIDE HOOP STRAIN AT
MID-HEIGHT OF
SUPPRESSION CHAMBER WALL

FIGURE 3.8-39, Rev. 47

Auto-Cad Figure Fsar 3_8_39.dwg



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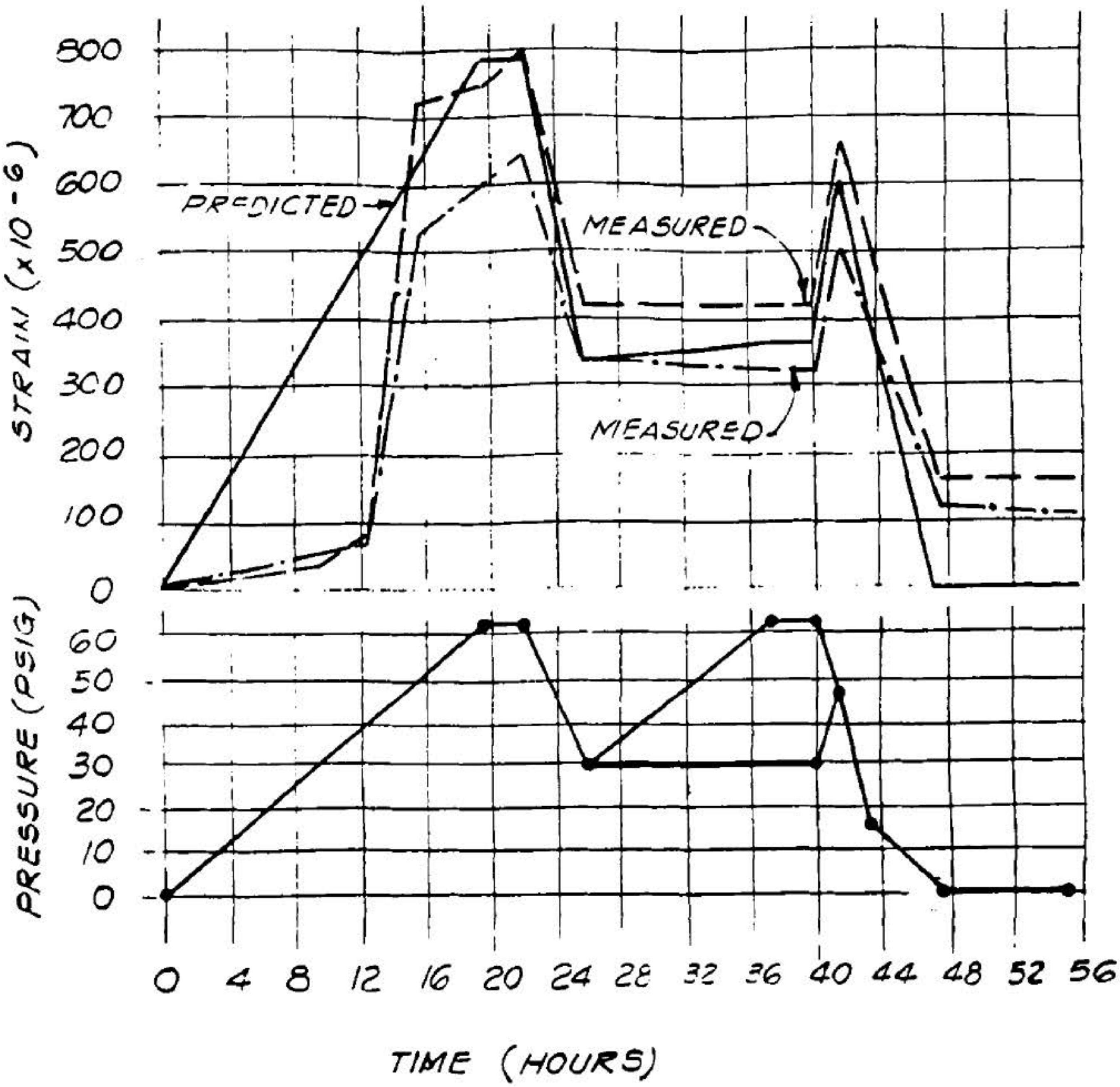
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

STRUCTURAL ACCEPTANCE TEST
OUTSIDE MERIDIONAL STRAIN AT
MID-HEIGHT OF
SUPPRESSION CHAMBER WALL

FIGURE 3.8-40, Rev. 47

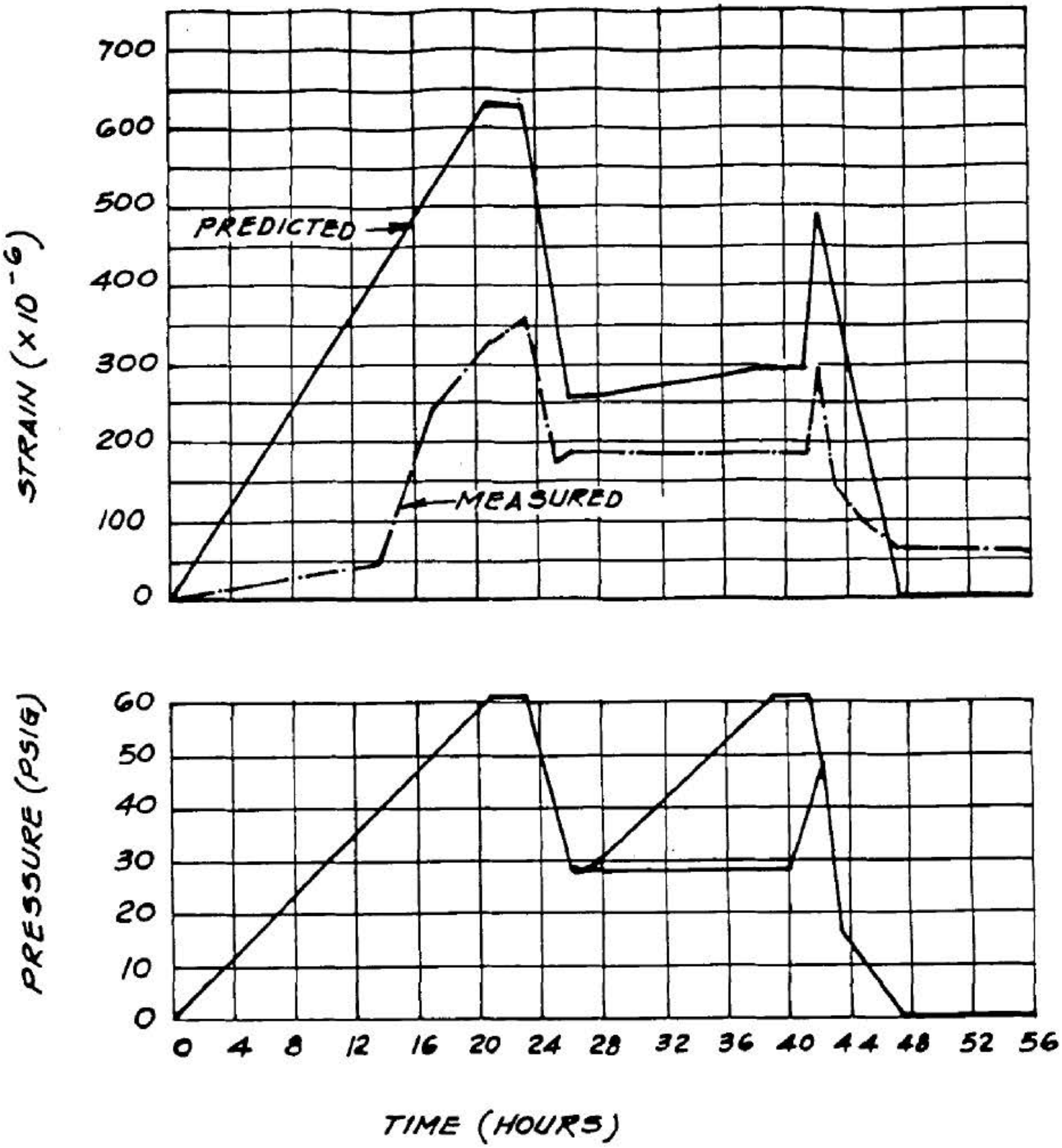
Auto-Cad Figure Fsar 3_8_40.dwg



FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
STRUCTURAL ACCEPTANCE TEST OUTSIDE HOOP STRAIN AT MID-HEIGHT OF SUPPRESSION CHAMBER WALL FIGURE 3.8-41, Rev. 47

Auto-Cad Figure Fsar 3_8_41.dwg



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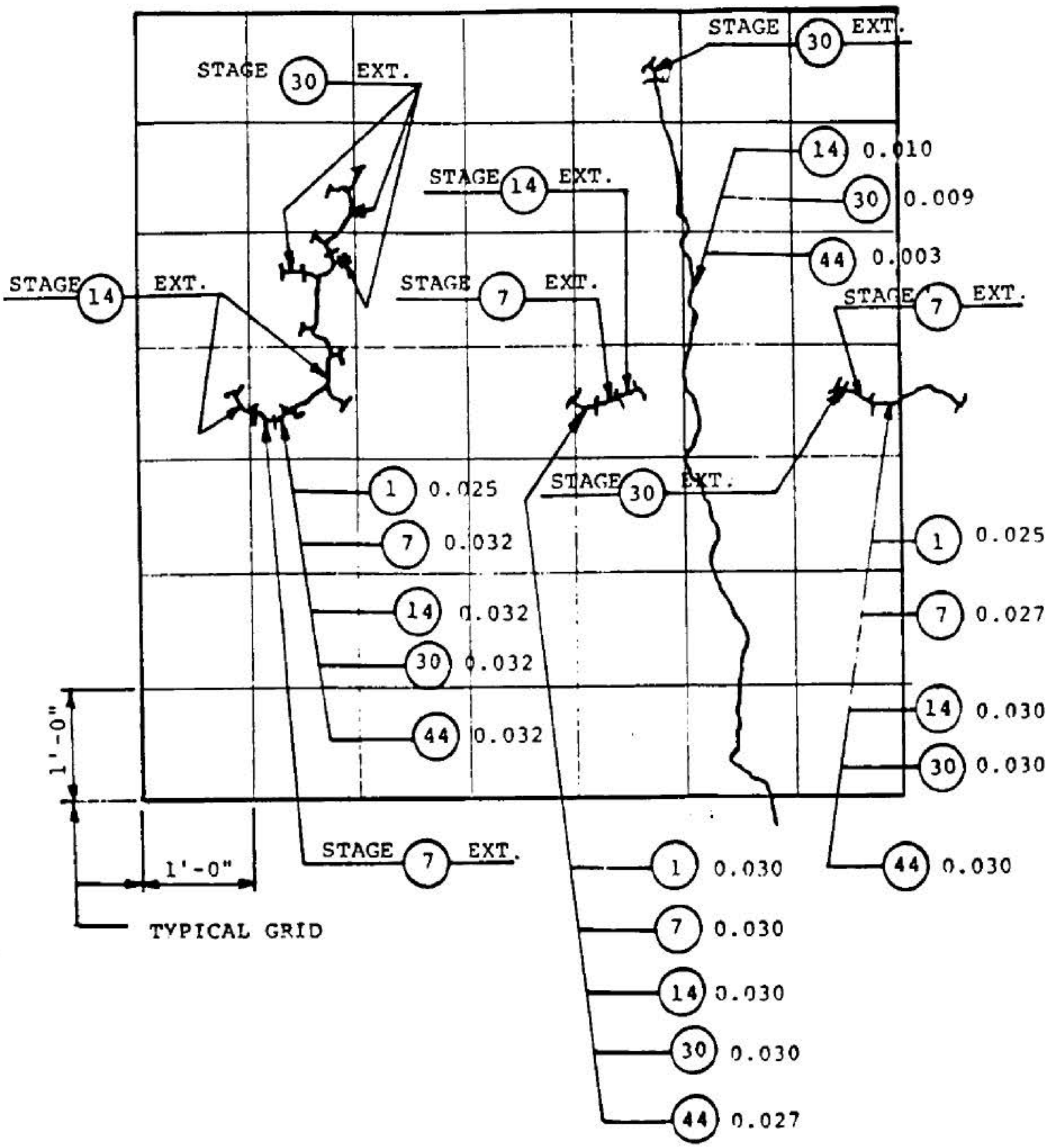
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

STRUCTURAL ACCEPTANCE TEST
OUTSIDE HELICAL STRAIN AT
MID-HEIGHT OF
SUPPRESSION CHAMBER WALL

FIGURE 3.8-42, Rev. 47

Auto-Cad Figure Fsar 3_8_42.dwg



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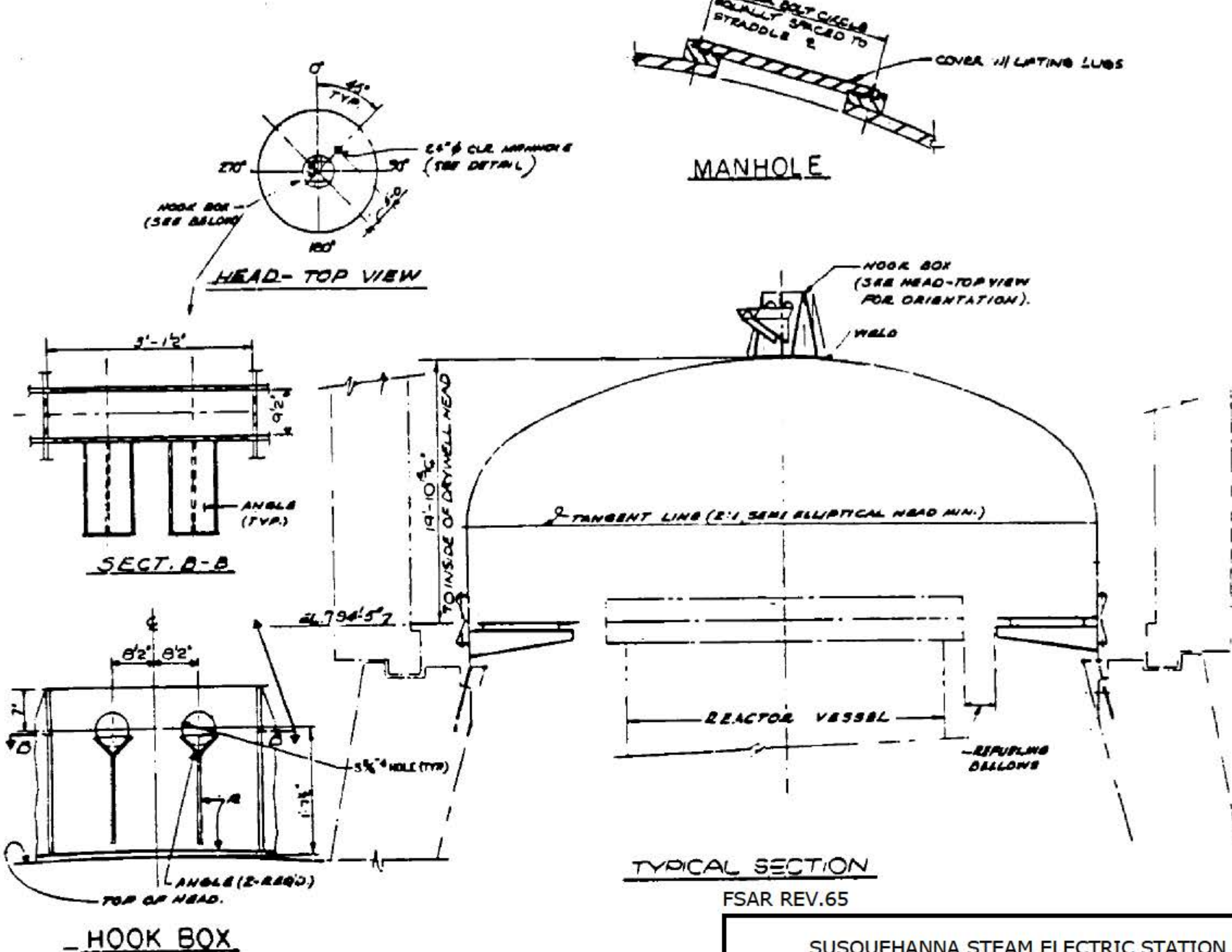
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

STRUCTURAL ACCEPTANCE TEST
EXTERNAL CONCRETE SURFACE
CRACKS AT MID-HEIGHT
OF DRYWELL WALL

FIGURE 3.8-43, Rev. 47

Auto-Cad Figure Fsar 3_8_43.dwg



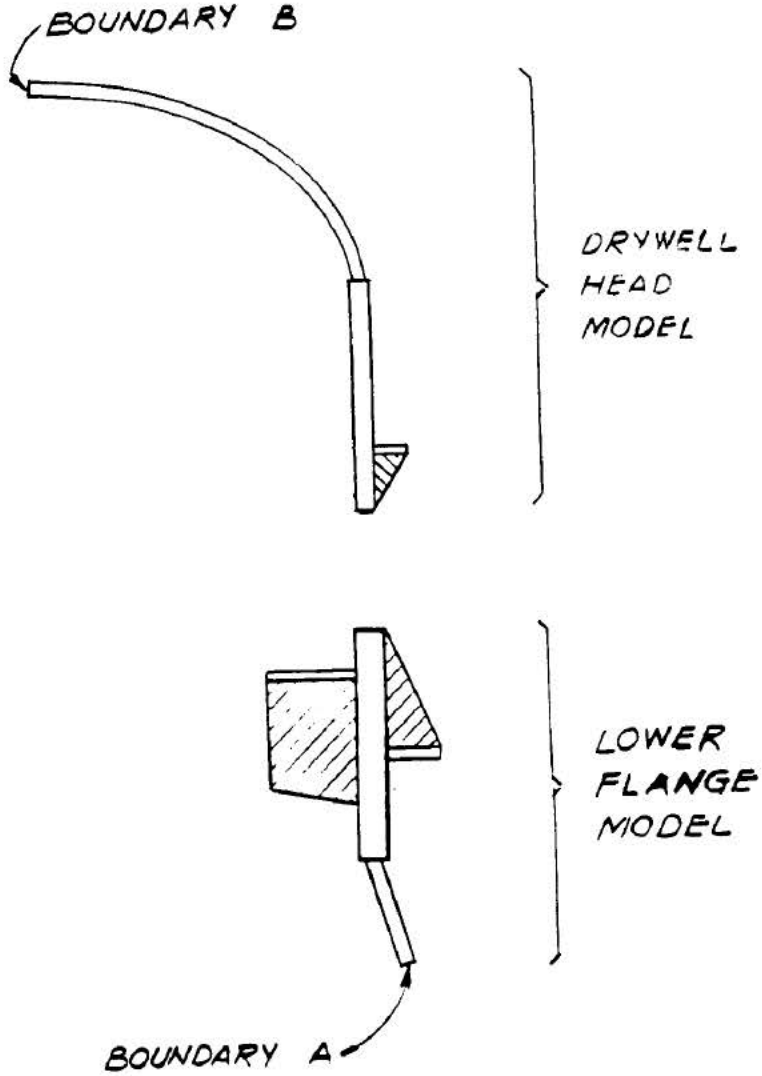
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

DRYWELL HEAD

FIGURE 3.8-44, Rev. 47

Auto-Cad Figure Fsar 3_8_44.dwg



NOTE: CROSS-HATCHED AREAS ARE MODELLED AS ORTHOTROPIC LAYERS.

FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

ANALYTICAL MODEL OF
DRYWELL HEAD ASSEMBLY

FIGURE 3.8-45, Rev. 47

Auto-Cad Figure Fsar 3_8_45.dwg

THIS FIGURE HAS BEEN
REPLACED BY DWG.
C-293, Sh. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

Figure 3.8-47 replaced by dwg.
C-293, Sh. 1

FIGURE 3.8-47, Rev. 48

AutoCAD Figure 3_8_47.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
C-340, Sh. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Figure 3.8-48 replaced by dwg. C-340, Sh. 1 FIGURE 3.8-48, Rev. 48

AutoCAD Figure 3_8_48.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
C-341, Sh. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

Figure 3.8-49 replaced by dwg.
C-341, Sh. 1

FIGURE 3.8-49, Rev. 48

AutoCAD Figure 3_8_49.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
C-376, Sh. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

Figure 3.8-50 replaced by dwg.
C-376, Sh. 1

FIGURE 3.8-50, Rev. 48

THIS FIGURE HAS BEEN
REPLACED BY DWG.
C-362, Sh. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

Figure 3.8-52 replaced by dwg.
C-362, Sh. 1

FIGURE 3.8-52, Rev. 48

AutoCAD Figure 3_8_52.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
C-363, Sh. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

Figure 3.8-53 replaced by dwg.
C-363, Sh. 1

FIGURE 3.8-53, Rev. 55

AutoCAD Figure 3_8_53.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
C-364, Sh. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

Figure 3.8-54 replaced by dwg.
C-364, Sh. 1

FIGURE 3.8-54, Rev. 55

AutoCAD Figure 3_8_54.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
C-365, Sh. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Figure 3.8-55 replaced by dwg. C-365, Sh. 1 FIGURE 3.8-55, Rev. 48

AutoCAD Figure 3_8_55.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
C-367, Sh. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Figure 3.8-56 replaced by dwg. C-367, Sh. 1 FIGURE 3.8-56, Rev. 48

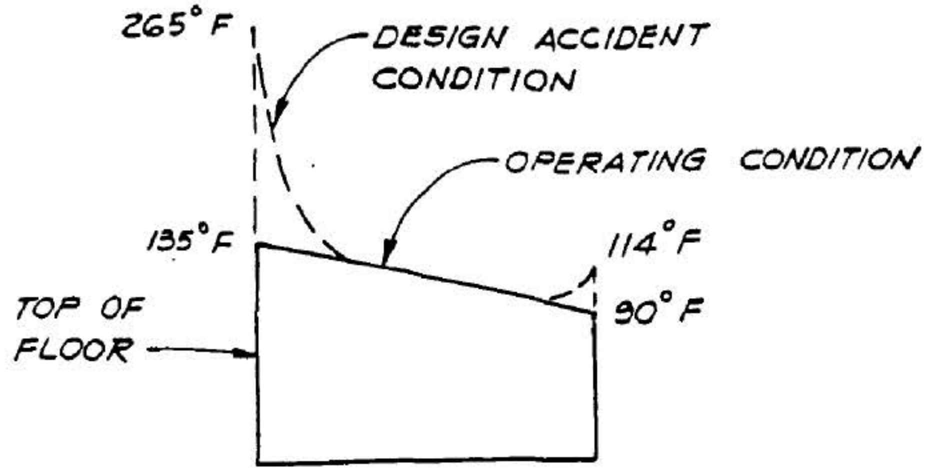
AutoCAD Figure 3_8_56.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
C-380, Sh. 1

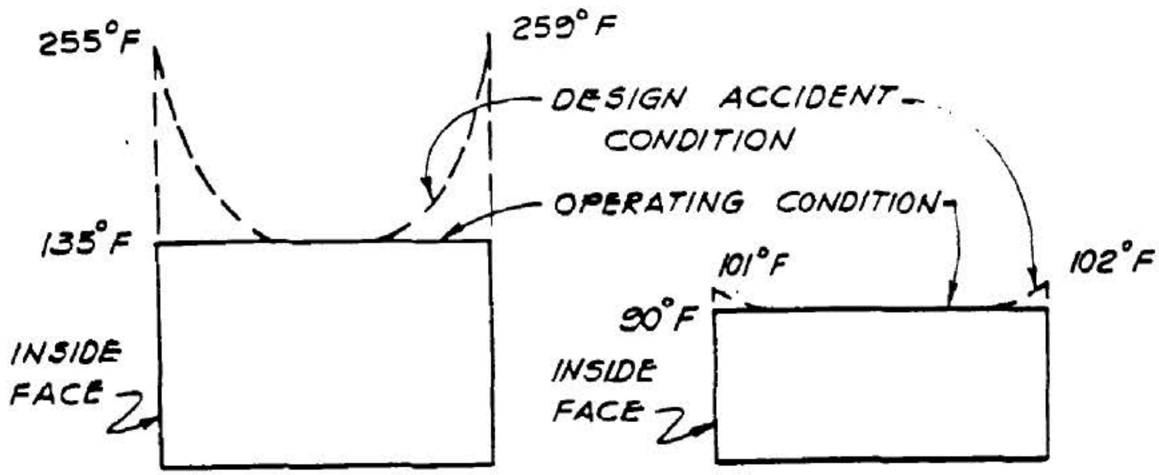
FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Figure 3.8-57 replaced by dwg. C-380, Sh. 1 FIGURE 3.8-57, Rev. 48

AutoCAD Figure 3_8_57.doc



DRYWELL FLOOR



ABOVE DRYWELL FLOOR

BELOW DRYWELL FLOOR

REACTOR PEDESTAL

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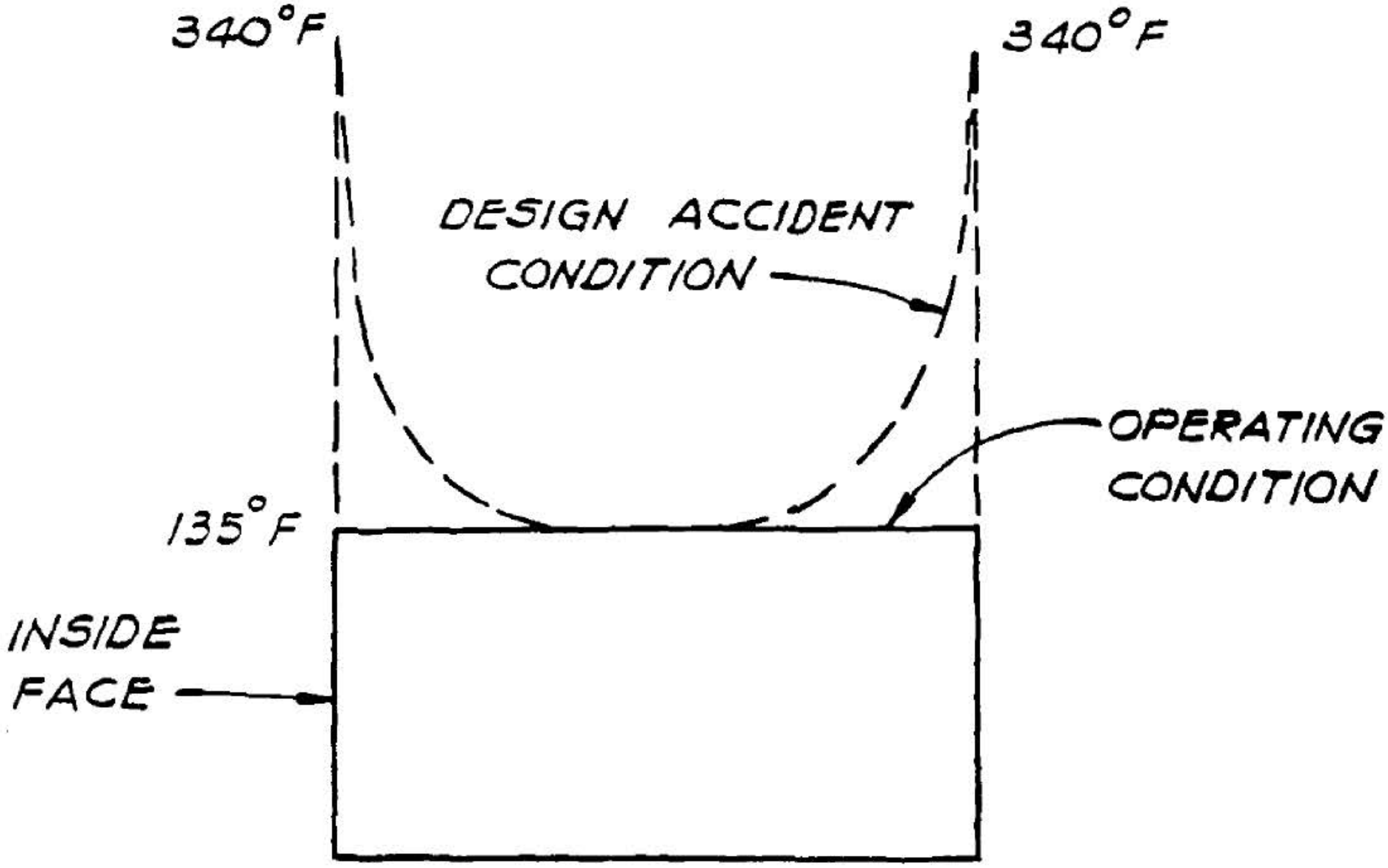
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

TEMPERATURE GRADIENTS
FOR DRYWELL FLOOR
AND REACTOR PEDESTAL

FIGURE 3.8-58, Rev. 47

Auto-Cad Figure Fsar 3_8_58.dwg



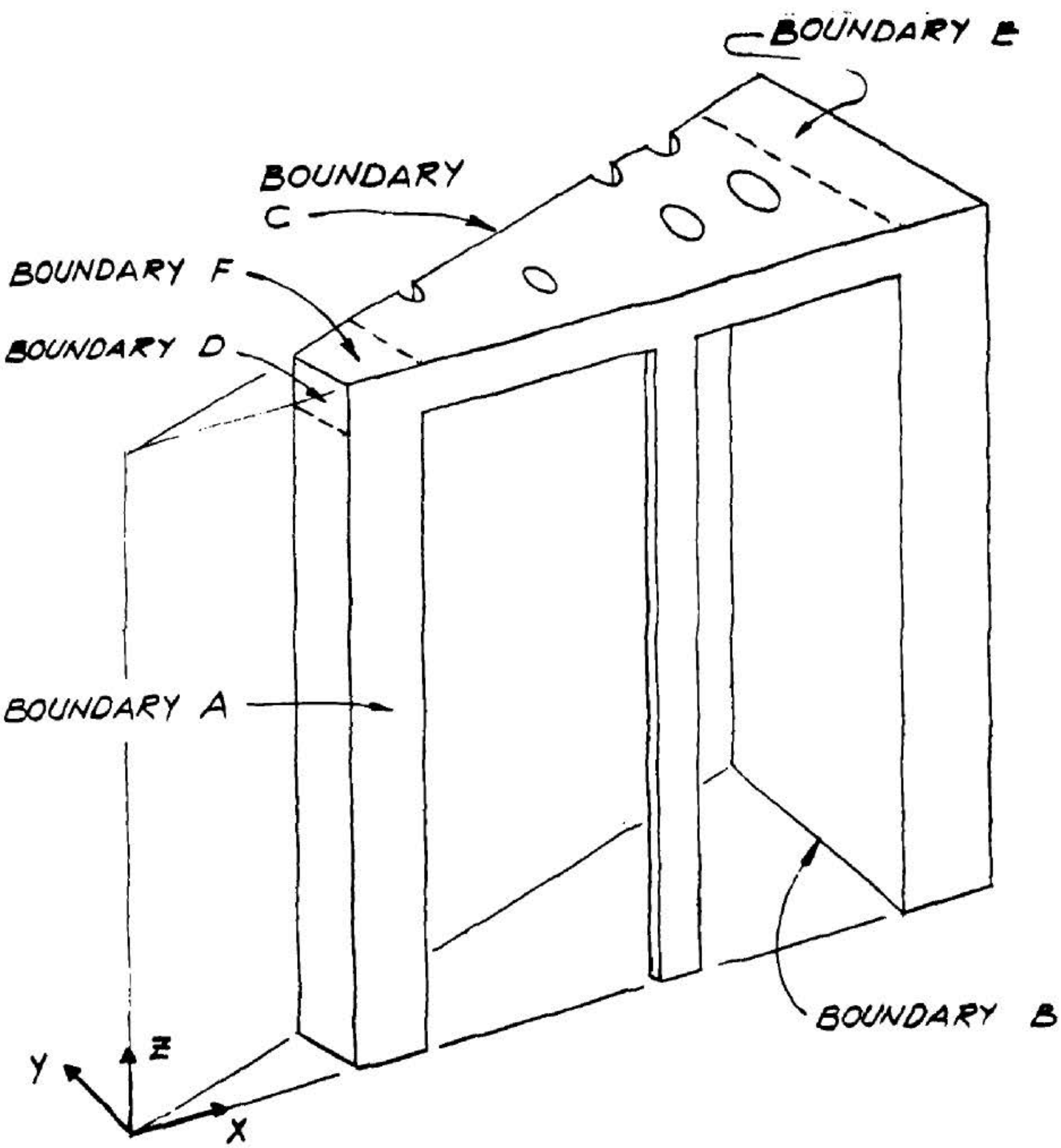
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

REACTOR SHIELD WALL
TEMPERATURE GRADIENTS

FIGURE 3.8-59, Rev. 47

Auto-Cad Figure Fsar 3_8_59.dwg



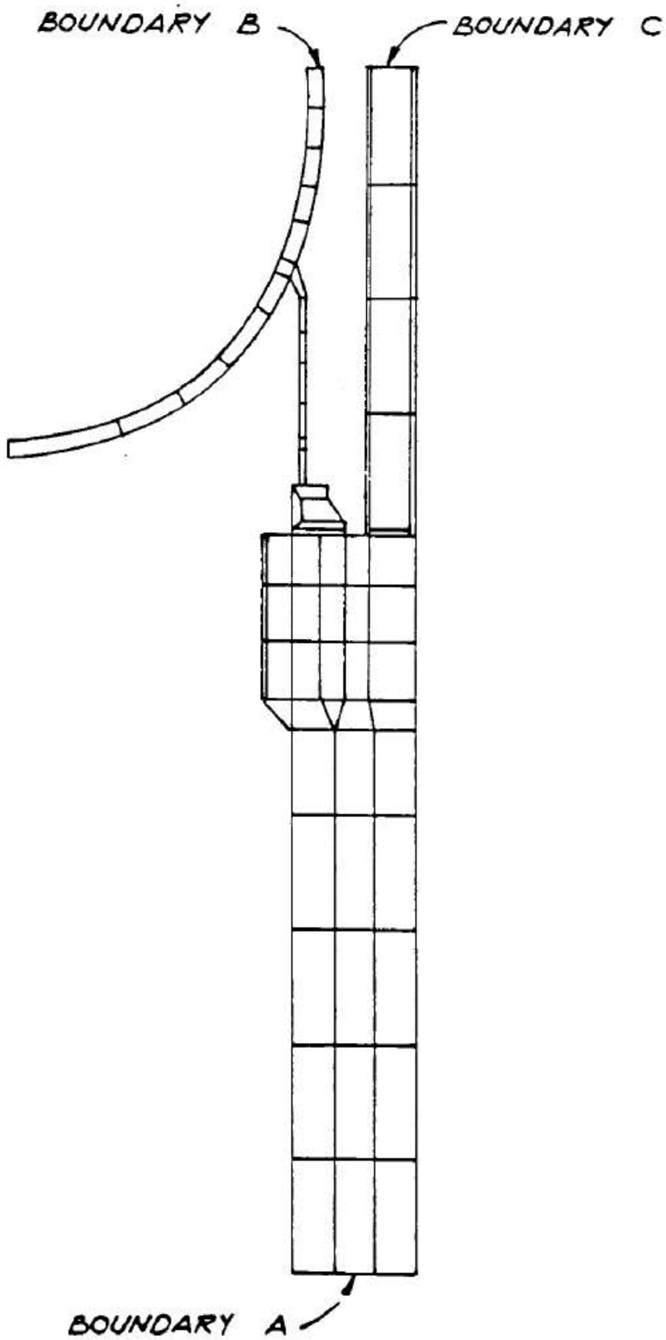
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

DRYWELL FLOOR
ANALYTICAL MODEL

FIGURE 3.8-60, Rev. 47

Auto-Cad Figure Fsar 3_8_60.dwg



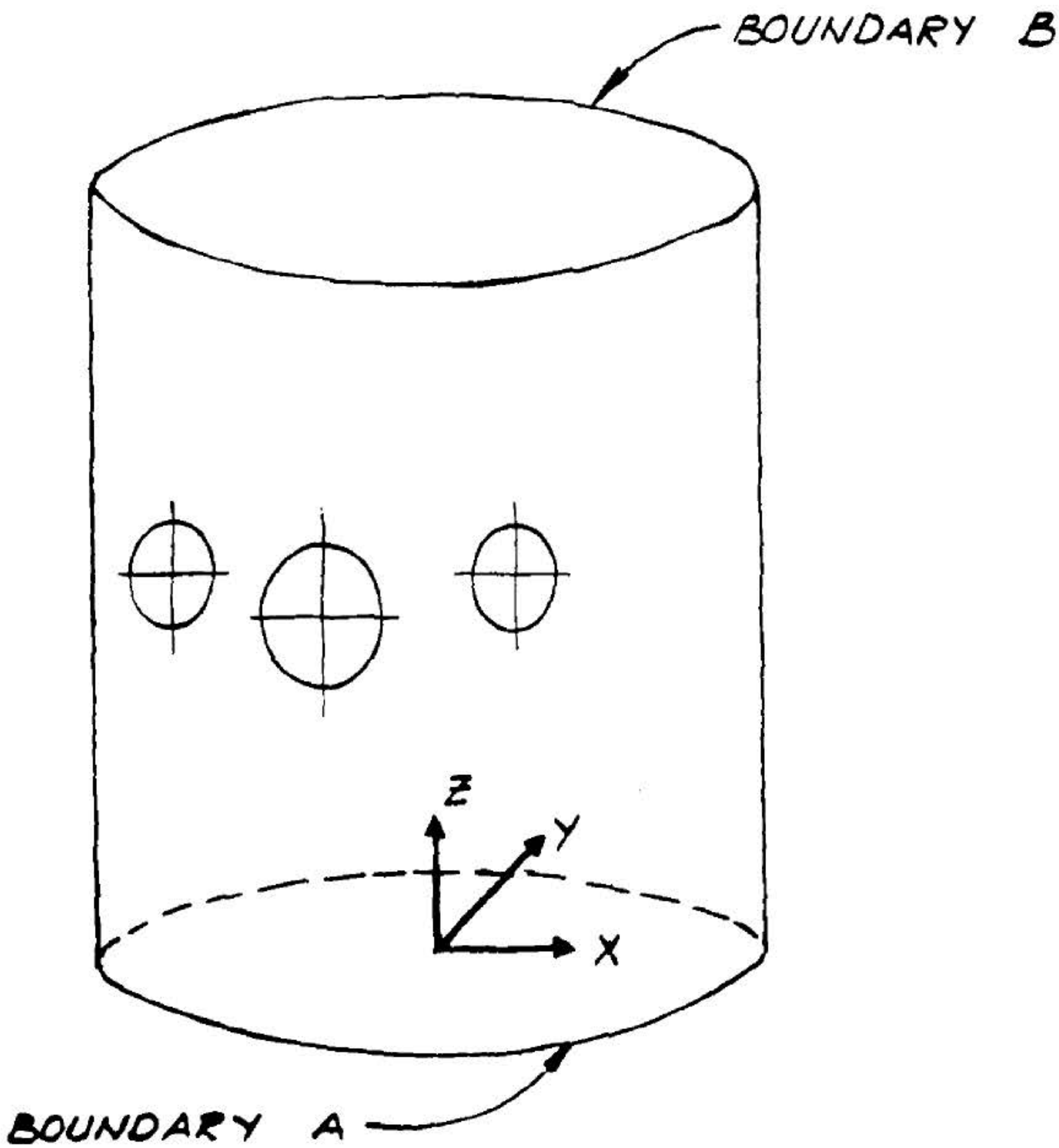
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

ANALYTICAL MODEL
FOR REACTOR PEDESTAL
ABOVE DRYWELL FLOOR

FIGURE 3.8-61, Rev. 47

Auto-Cad Figure Fsar 3_8_61.dwg



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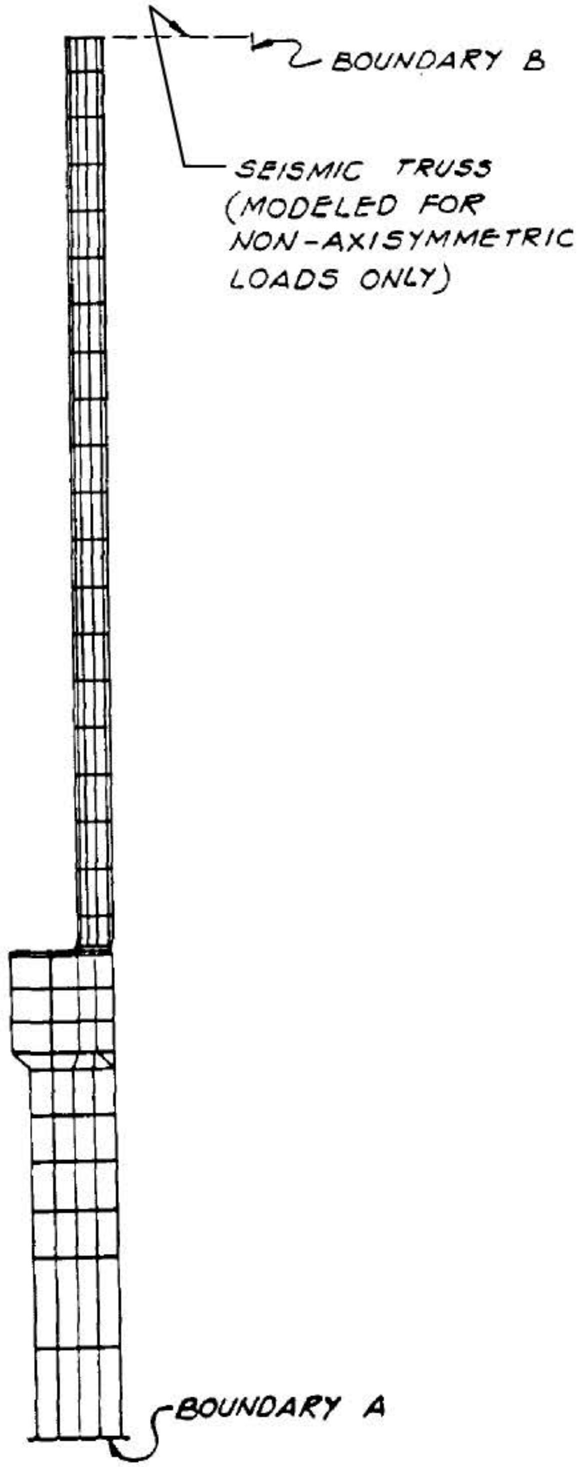
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

REACTOR SHIELD WALL
"EASE" PROGRAM
ANALYTICAL MODEL

FIGURE 3.8-62, Rev. 47

Auto-Cad Figure Fsar 3_8_62.dwg



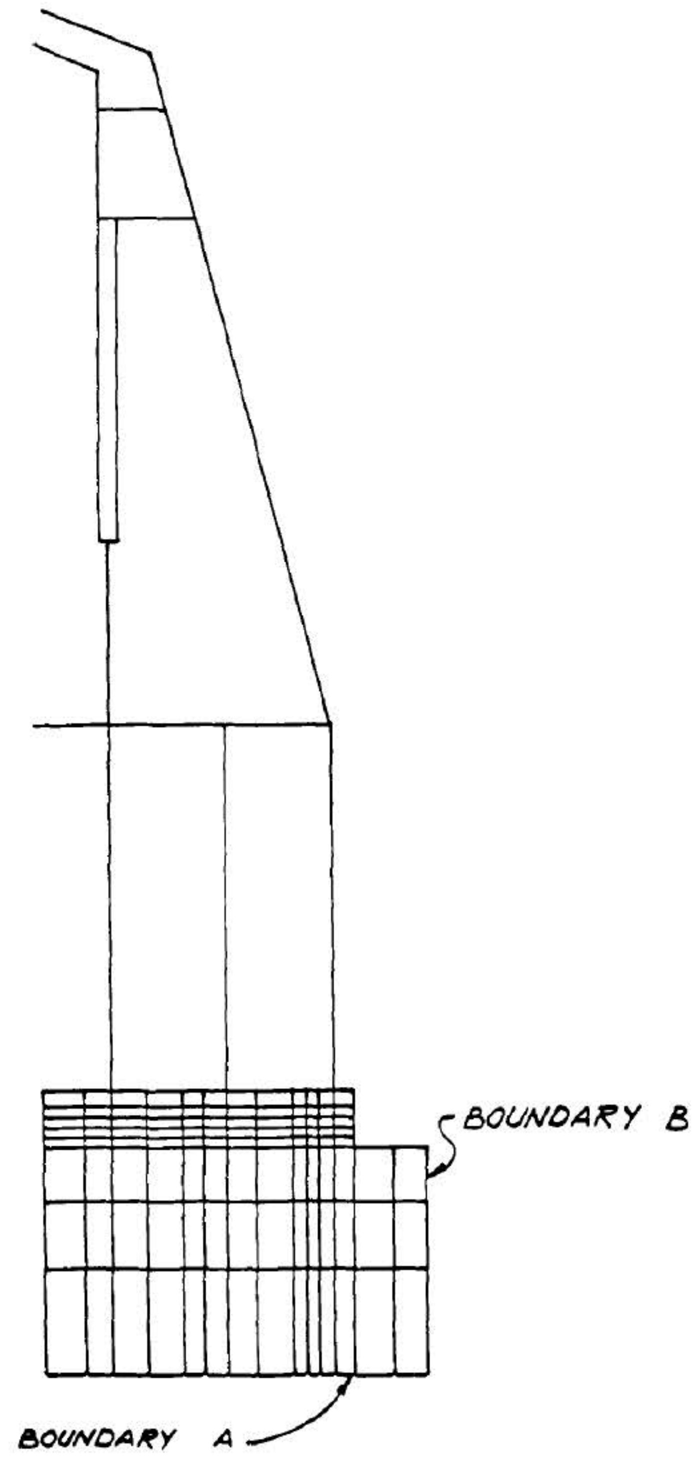
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

REACTOR SHIELD WALL
ANALYTICAL MODEL FOR
"FINEL" AND "ASHSD" PROGRAMS

FIGURE 3.8-63, Rev. 47

Auto-Cad Figure Fsar 3_8_63.dwg



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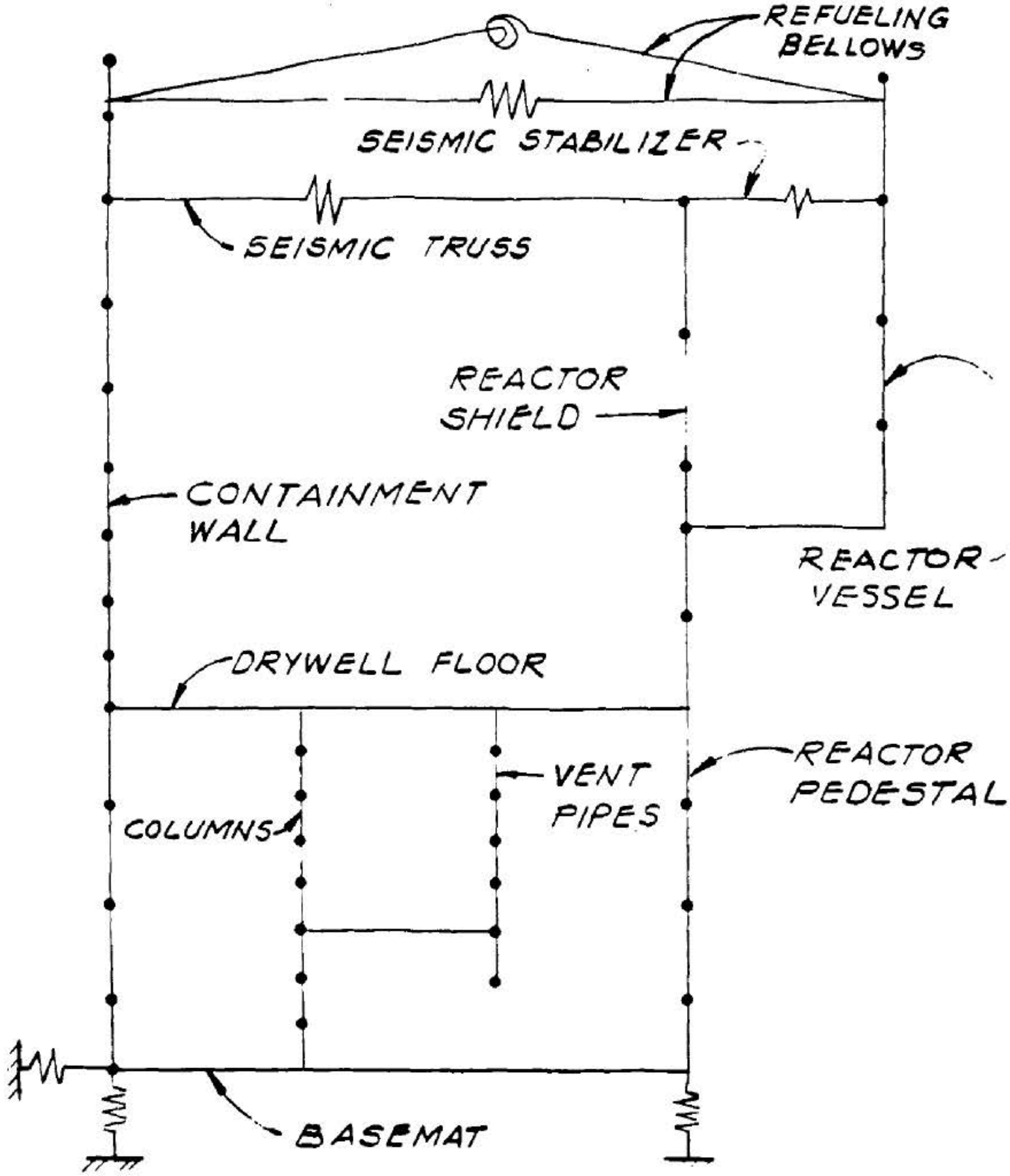
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

SUPPRESSION CHAMBER COLUMNS
"ASHSD" PROGRAM
ANALYTICAL MODEL

FIGURE 3.8-64, Rev. 47

Auto-Cad Figure Fsar 3_8_64.dwg



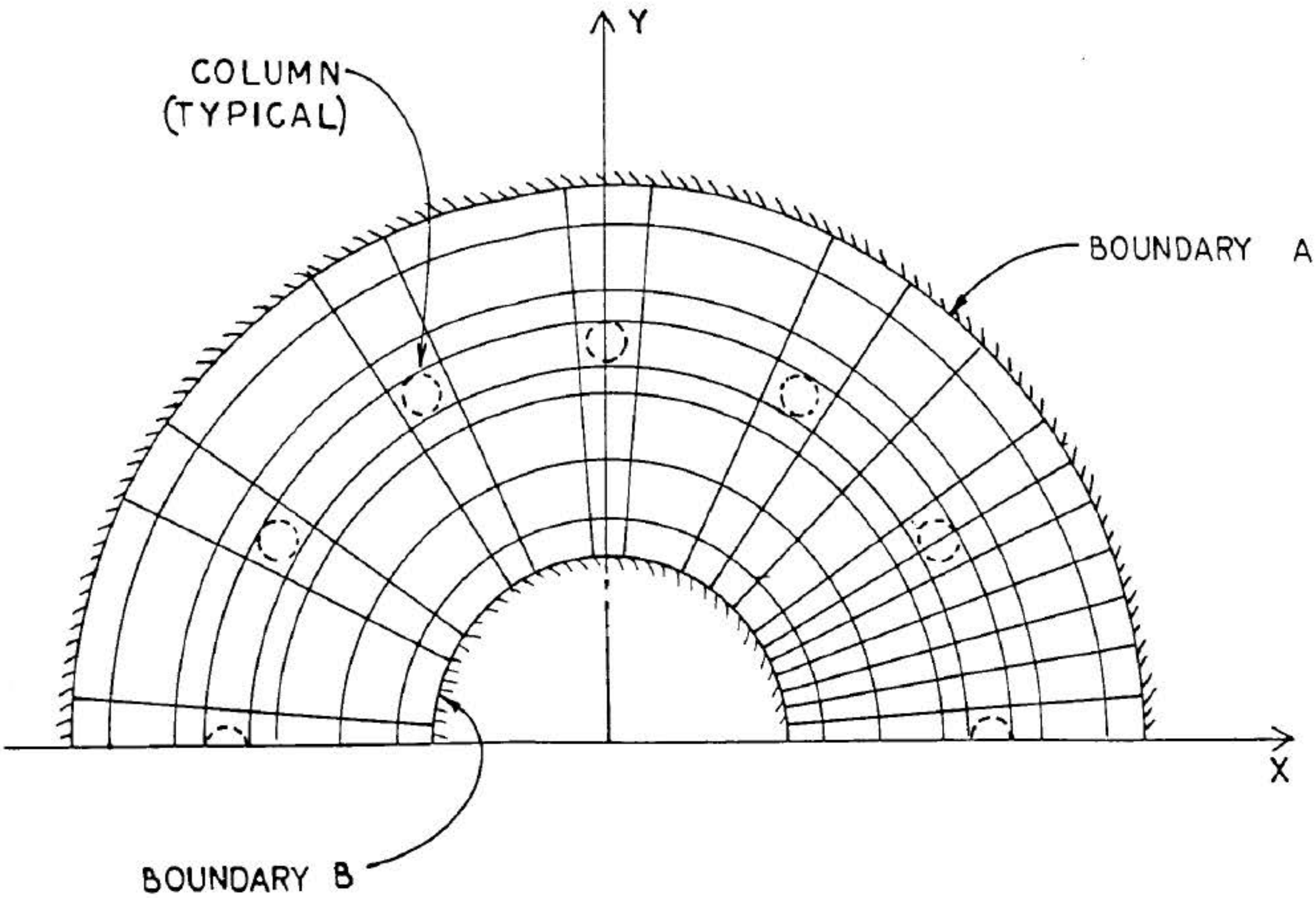
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

SUPPRESSION CHAMBER COLUMNS
SEISMIC MODEL

FIGURE 3.8-65, Rev. 47

Auto-Cad Figure Fsar 3_8_65.dwg



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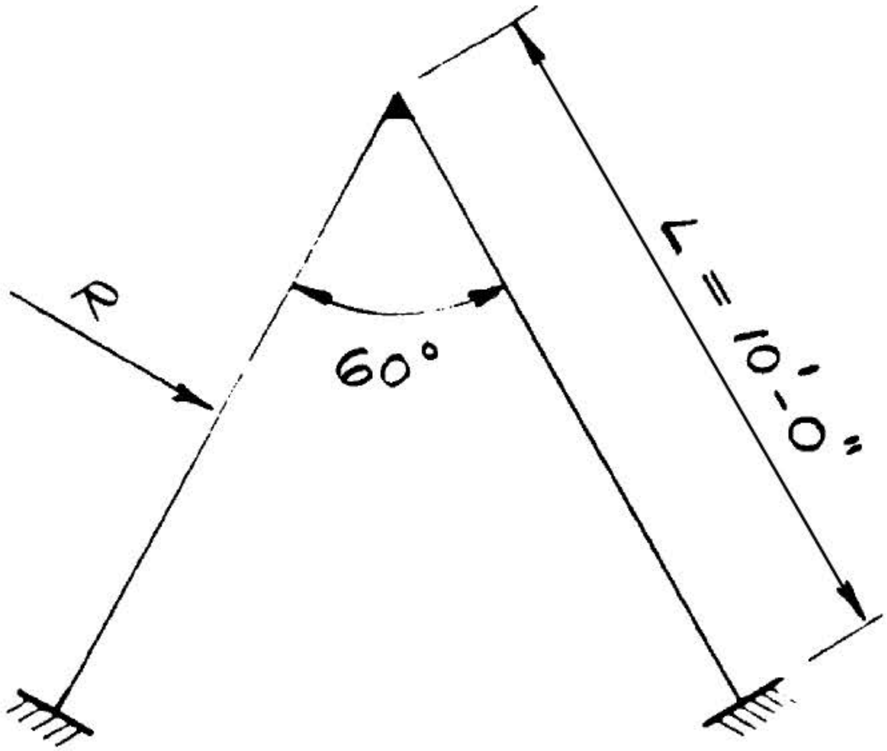
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

SUPPRESSION CHAMBER COLUMNS
"CE 668" PROGRAM
ANALYTICAL MODEL

FIGURE 3.8-66, Rev. 47

Auto-Cad Figure Fsar 3_8_66.dwg



R = PIPE RUPTURE LOAD

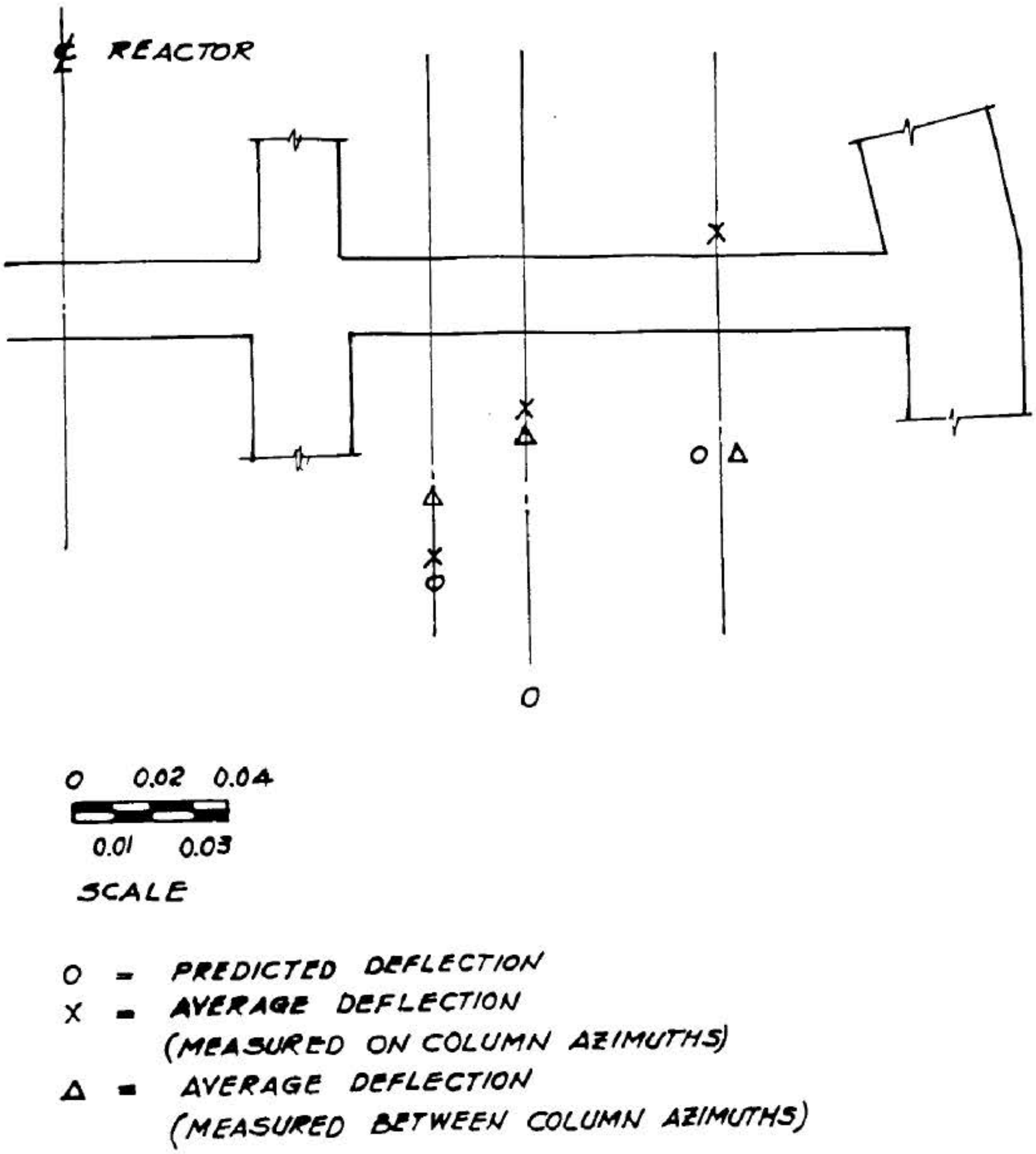
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

SEISMIC TRUSS
ANALYTICAL MODEL

FIGURE 3.8-67, Rev. 47

Auto-Cad Figure Fsar 3_8_67.dwg



FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

STRUCTURAL ACCEPTANCE TEST
COMPARISON OF MEASURED AND
PREDICTED DEFLECTIONS FOR
THE DRYWELL FLOOR

FIGURE 3.8-68, Rev. 47

Auto-Cad Figure Fsar 3_8_68.dwg

THIS FIGURE HAS BEEN
REPLACED BY DWG.
A-11, Sh. 1

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SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
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Figure 3.8-69 replaced by dwg.
A-11, Sh. 1

FIGURE 3.8-69, Rev. 55

AutoCAD Figure 3_8_69.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
A-12, Sh. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

Figure 3.8-70 replaced by dwg.
A-12, Sh. 1

FIGURE 3.8-70, Rev. 56

AutoCAD Figure 3_8_70.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
A-13, Sh. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Figure 3.8-71 replaced by dwg. A-13, Sh. 1 FIGURE 3.8-71, Rev. 55

AutoCAD Figure 3_8_71.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-203, Sh. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

Figure 3.8-72 replaced by dwg.
M-203, Sh. 1

FIGURE 3.8-72, Rev. 55

AutoCAD Figure 3_8_72.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-204, Sh. 1

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SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

Figure 3.8-73 replaced by dwg.
M-204, Sh. 1

FIGURE 3.8-73, Rev. 48

AutoCAD Figure 3_8_73.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
A-16, Sh. 1

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SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

Figure 3.8-74 replaced by dwg.
A-16, Sh. 1

FIGURE 3.8-74, Rev. 55

AutoCAD Figure 3_8_74.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
A-17, Sh. 1

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SUSQUEHANNA STEAM ELECTRIC STATION
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FINAL SAFETY ANALYSIS REPORT

Figure 3.8-75 replaced by dwg.
A-17, Sh. 1

FIGURE 3.8-75, Rev. 55

AutoCAD Figure 3_8_75.doc

THIS FIGURE HAS BEEN
DELETED

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SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

Figure Deleted

FIGURE 3.8-76, Rev. 54

AutoCAD Figure 3_8_76.doc

Security-Related Information
Figure Withheld Under 10 CFR 2.390

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
REACTOR, CONTROL AND TURBINE BUILDING SECTIONS LOOKING NORTH FIGURE 3.8-77, Rev. 54

Auto-Cad Figure Fsar 3_8_77.dwg

Security-Related Information
Figure Withheld Under 10 CFR 2.390

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
REACTOR BUILDING SECTION LOOKING WEST
FIGURE 3.8-78, Rev. 54

Auto-Cad Figure Fsar 3_8_78.dwg

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-227, Sh. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

Figure 3.8-79 replaced by dwg.
M-227, Sh. 1

FIGURE 3.8-79, Rev. 55

AutoCAD Figure 3_8_79.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-237, Sh. 1

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SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

Figure 3.8-80 replaced by dwg.
M-237, Sh. 1

FIGURE 3.8-80, Rev. 55

AutoCAD Figure 3_8_80.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-260, Sh. 1

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SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

Figure 3.8-81 replaced by dwg.
M-260, Sh. 1

FIGURE 3.8-81, Rev. 55

AutoCAD Figure 3_8_81.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-261, Sh. 1

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SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

Figure 3.8-82 replaced by dwg.
M-261, Sh. 1

FIGURE 3.8-82, Rev. 55

AutoCAD Figure 3_8_82.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-5200, Sh. 1

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SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Figure 3.8-83 replaced by dwg. M-5200, Sh. 1 FIGURE 3.8-83, Rev. 55

AutoCAD Figure 3_8_83.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-5200, Sh. 2

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Figure 3.8-84 replaced by dwg. M-5200, Sh. 2 FIGURE 3.8-84, Rev. 55

AutoCAD Figure 3_8_84.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-284, Sh. 1

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SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

Figure 3.8-85 replaced by dwg.
M-284, Sh. 1

FIGURE 3.8-85, Rev. 55

AutoCAD Figure 3_8_85.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
C-64, Sh. 1

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SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

Figure 3.8-86 replaced by dwg.
C-64, Sh. 1

FIGURE 3.8-86, Rev. 48

AutoCAD Figure 3_8_86.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
C-65, Sh. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Figure 3.8-87 replaced by dwg. C-65, Sh. 1 FIGURE 3.8-87, Rev. 48

AutoCAD Figure 3_8_87.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
C-66, Sh. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
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FINAL SAFETY ANALYSIS REPORT

Figure 3.8-88 replaced by dwg.
C-66, Sh. 1

FIGURE 3.8-88, Rev. 48

AutoCAD Figure 3_8_88.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
C-67, Sh. 1

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SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

Figure 3.8-89 replaced by dwg.
C-67, Sh. 1

FIGURE 3.8-89, Rev. 48

AutoCAD Figure 3_8_89.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-270, Sh. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

Figure 3.8-90 replaced by dwg.
M-270, Sh. 1

FIGURE 3.8-90, Rev. 55

AutoCAD Figure 3_8_90.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-271, Sh. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

Figure 3.8-91 replaced by dwg.
M-271, Sh. 1

FIGURE 3.8-91, Rev. 55

AutoCAD Figure 3_8_91.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-272, Sh. 1

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SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Figure 3.8-92 replaced by dwg. M-272, Sh. 1 FIGURE 3.8-92, Rev. 55

AutoCAD Figure 3_8_92.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-273, Sh. 1

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SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
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Figure 3.8-93 replaced by dwg.
M-273, Sh. 1

FIGURE 3.8-93, Rev. 55

AutoCAD Figure 3_8_93.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-274, Sh. 1

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SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

Figure 3.8-94 replaced by dwg.
M-274, Sh. 1

FIGURE 3.8-94, Rev. 55

AutoCAD Figure 3_8_94.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
C-795, Sh. 1

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SUSQUEHANNA STEAM ELECTRIC STATION
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FINAL SAFETY ANALYSIS REPORT

Figure 3.8-95 replaced by dwg.
C-795, Sh. 1

FIGURE 3.8-95, Rev. 48

AutoCAD Figure 3_8_95.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
C-332, Sh. 1

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SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Figure 3.8-11-1 replaced by dwg. C-332, Sh. 1 FIGURE 3.8-11-1, Rev. 49

Figure Fsar 3_8_11_1.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
C-333, Sh. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

Figure 3.8-11-2 replaced by dwg.
C-333, Sh. 1

FIGURE 3.8-11-2, Rev. 49

THIS FIGURE HAS BEEN
REPLACED BY DWG.
C-334, Sh. 1

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SUSQUEHANNA STEAM ELECTRIC STATION
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Figure 3.8-15-1 replaced by dwg.
C-334, Sh. 1

FIGURE 3.8-15-1, Rev. 49

Figure Fsar 3_8_15_1.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
C-335, Sh. 1

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SUSQUEHANNA STEAM ELECTRIC STATION
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Figure 3.8-15-2 replaced by dwg.
C-335, Sh. 1

FIGURE 3.8-15-2, Rev. 49

Figure Fsar 3_8_15_2.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
C-336, Sh. 1

THIS FIGURE HAS BEEN
REPLACED BY DWG.
C-337, Sh. 1

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SUSQUEHANNA STEAM ELECTRIC STATION
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FINAL SAFETY ANALYSIS REPORT

Figure 3.8-15-4 replaced by dwg.
C-337, Sh. 1

FIGURE 3.8-15-4, Rev. 49

Figure Fsar 3_8_15_4.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
C-338, Sh. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
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FINAL SAFETY ANALYSIS REPORT

Figure 3.8-15-5 replaced by dwg.
C-338, Sh. 1

FIGURE 3.8-15-5, Rev. 49

Figure Fsar 3_8_15_5.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
C-351, Sh. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Figure 3.8-16-1 replaced by dwg. C-351, Sh. 1 FIGURE 3.8-16-1, Rev. 49

Figure Fsar 3_8_16_1.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
C-352, Sh. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Figure 3.8-16-2 replaced by dwg. C-352, Sh. 1 FIGURE 3.8-16-2, Rev. 49

Figure Fsar 3_8_16_2.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
C-353, Sh. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

Figure 3.8-16-3 replaced by dwg.
C-353, Sh. 1

FIGURE 3.8-16-3, Rev. 49

Figure Fsar 3_8_16_3.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
C-354, Sh. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

Figure 3.8-16-4 replaced by dwg.
C-354, Sh. 1

FIGURE 3.8-16-4, Rev. 49

Figure Fsar 3_8_16_4.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
C-355, Sh. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Figure 3.8-16-5 replaced by dwg. C-355, Sh. 1 FIGURE 3.8-16-5, Rev. 49

Figure Fsar 3_8_16_5.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
C-356, Sh. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

Figure 3.8-16-6 replaced by dwg.
C-356, Sh. 1

FIGURE 3.8-16-6, Rev. 49

AutoCAD Figure 3_8_16_6.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
C-357, Sh. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

Figure 3.8-16-7 replaced by dwg.
C-357, Sh. 1

FIGURE 3.8-16-7, Rev. 49

AutoCAD Figure 3_8_16_7.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
C-358, Sh. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Figure 3.8-16-8 replaced by dwg. C-358, Sh. 1 FIGURE 3.8-16-8, Rev. 49

AutoCAD Figure 3_8_16_8.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
C-359, Sh. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

Figure 3.8-16-9 replaced by dwg.
C-359, Sh. 1

FIGURE 3.8-16-9, Rev. 49

AutoCAD Figure 3_8_16_9.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
C-288, Sh. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

Figure 3.8-19-1 replaced by dwg.
C-288, Sh. 1

FIGURE 3.8-19-1, Rev. 55

AutoCAD Figure 3_8_19_1.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
C-287, Sh. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

Figure 3.8-19-2 replaced by dwg.
C-287, Sh. 1

FIGURE 3.8-19-2, Rev. 55

AutoCAD Figure 3_8_19_2.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
C-283, Sh. 1

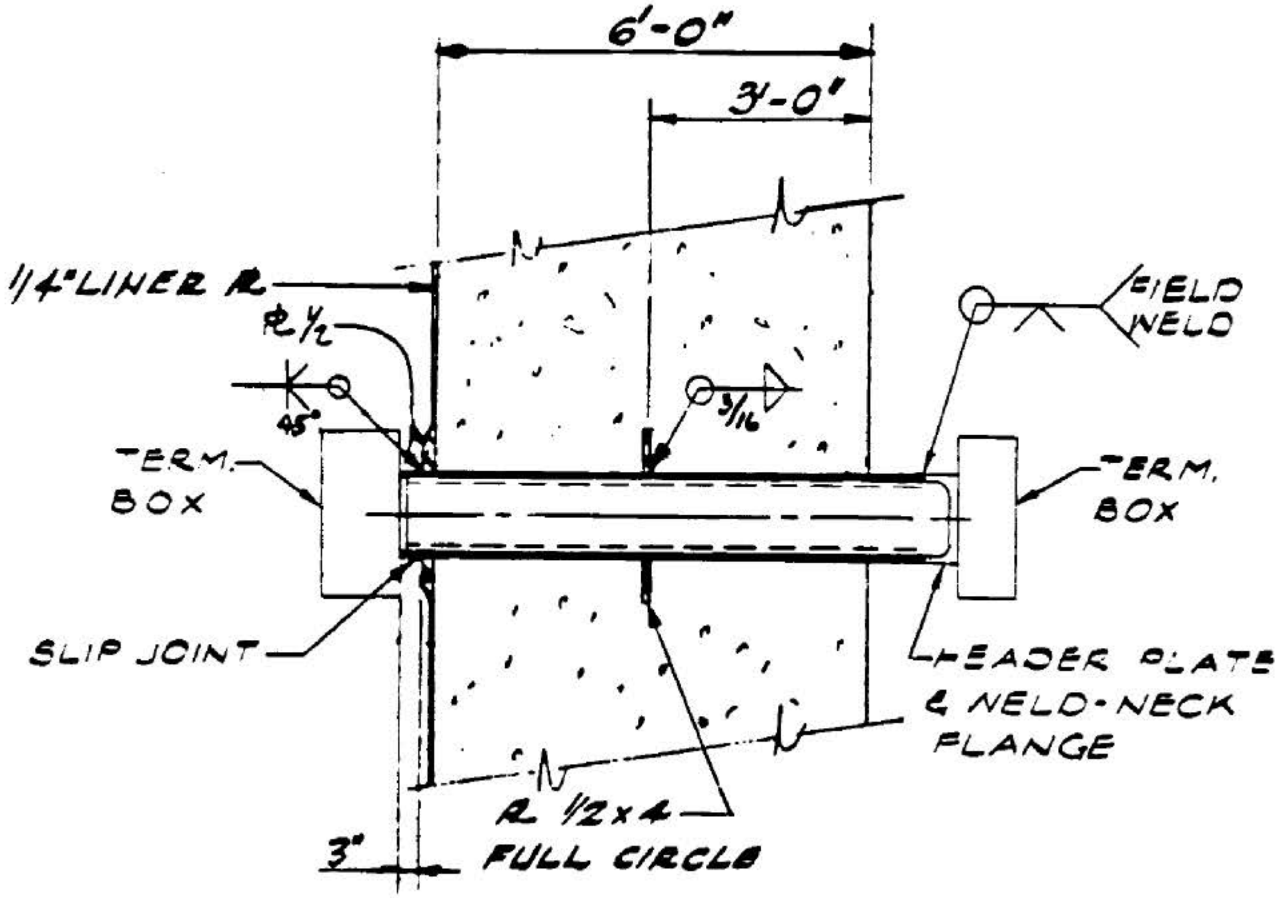
FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

Figure 3.8-19-3 replaced by dwg.
C-283, Sh. 1

FIGURE 3.8-19-3, Rev. 55

AutoCAD Figure 3_8_19_3.doc



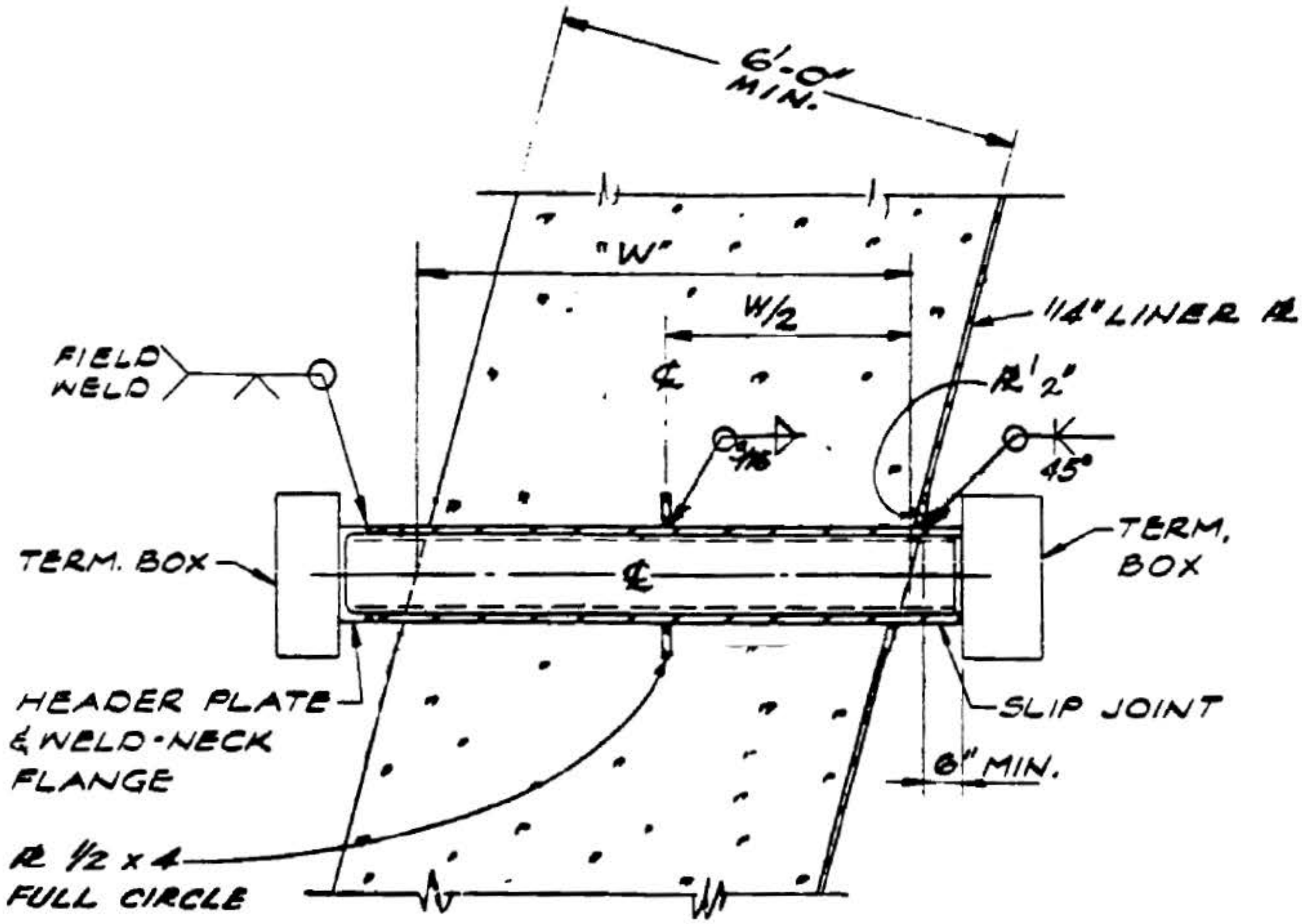
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

SUPPRESSION CHAMBER
ELECTRICAL PENETRATION
DETAILS

FIGURE 3.8-20-1, Rev. 48

Auto-Cad Figure Fsar 3_8_20_1.dwg



FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

DRYWELL
ELECTRICAL
PENETRATION DETAILS

FIGURE 3.8-20-2, Rev. 48

Auto-Cad Figure Fsar 3_8_20_2.dwg

THIS FIGURE HAS BEEN
REPLACED BY DWG.
C-348, Sh. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Figure 3.8-46-1 replaced by dwg. C-348, Sh. 1 FIGURE 3.8-46-1, Rev. 49

AutoCAD Figure 3_8_46_1.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
C-349, Sh. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Figure 3.8-46-2 replaced by dwg. C-349, Sh. 1 FIGURE 3.8-46-2, Rev. 49

AutoCAD Figure 3_8_46_2.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
C-350, Sh. 1

FSAR REV. 65

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Figure 3.8-46-3 replaced by dwg.
C-350, Sh. 1

FIGURE 3.8-46-3, Rev. 49

AutoCAD Figure 3_8_46_3.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
C-344, Sh. 1

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Figure 3.8-51-1 replaced by dwg.
C-344, Sh. 1

FIGURE 3.8-51-1, Rev. 49

AutoCAD Figure 3_8_51_1.doc

THIS FIGURE HAS BEEN
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C-377, Sh. 1

FSAR REV. 65

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Figure 3.8-51-2 replaced by dwg.
C-377, Sh. 1

FIGURE 3.8-51-2, Rev. 49

AutoCAD Figure 3_8_51_2.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
C-360, Sh. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Figure 3.8-16-10 replaced by dwg. C-360, Sh. 1 FIGURE 3.8-16-10, Rev. 49

Figure Fsar 3_8_16_10.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
C-393, Sh. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
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Figure 3.8-16-11 replaced by dwg.
C-393, Sh. 1

FIGURE 3.8-16-11, Rev. 49

THIS FIGURE HAS BEEN
REPLACED BY DWG.
C-394, Sh. 1

FSAR REV. 65

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Figure 3.8-16-12 replaced by dwg.
C-394, Sh. 1

FIGURE 3.8-16-12, Rev. 49

THIS FIGURE HAS BEEN
REPLACED BY DWG.
C-395, Sh. 1

FSAR REV. 65

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Figure 3.8-16-13 replaced by dwg.
C-395, Sh. 1

FIGURE 3.8-16-13, Rev. 49

Figure Fsar 3_8_16_13.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
C-396, Sh. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

Figure 3.8-16-14 replaced by dwg.
C-396, Sh. 1

FIGURE 3.8-16-14, Rev. 49

Figure Fsar 3_8_16_14.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
C-397, Sh. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

Figure 3.8-16-15 replaced by dwg.
C-397, Sh. 1

FIGURE 3.8-16-15, Rev. 49

Figure Fsar 3_8_16_15.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
C-398, Sh. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
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FINAL SAFETY ANALYSIS REPORT

Figure 3.8-16-16 replaced by dwg.
C-398, Sh. 1

FIGURE 3.8-16-16, Rev. 49

Figure Fsar 3_8_16_16.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
C-399, Sh. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
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FINAL SAFETY ANALYSIS REPORT

Figure 3.8-16-17 replaced by dwg.
C-399, Sh. 1

FIGURE 3.8-16-17, Rev. 49

Figure Fsar 3_8_16_17.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
C-400, Sh. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
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Figure 3.8-16-18 replaced by dwg.
C-400, Sh. 1

FIGURE 3.8-16-18, Rev. 49

Figure Fsar 3_8_16_18.doc

APPENDIX 3.8A
Computer Programs

This appendix contains a description of the computer programs used for the structural analysis of all Seismic Category I structures. For each computer program, there is a brief description of the program's theoretical basis, the assumptions and references used in the program, and the extent of the application. Examples of verification procedures are included for each PP&L in-house program.

The computer programs discussed in this section are those programs used for the original plant design. Changes to later versions of these programs or the addition of entirely new computer programs for safety related applications is controlled by procedures under our Operational Quality Assurance Program.

3.8A.1 3D/SAP

3D/SAP is a finite element program used to perform the static analysis of arbitrary, three-dimensional, elastic solids subjected to concentrated or distributed (pressure) loadings thermal expansion and/or arbitrarily directed static body forces. 3D/SAP is a mathematical version of "SAP" (Reference 3.8A-1) which is a general purpose structural analysis computer code.

3D/SAP was developed by the Control Data Corporation and is in the public domain.

3.8A.2 ASHSD

ASHSD (Axisymmetric Shell And Solid) is a special-purpose program which can be used in the elastic, static or dynamic analysis of structural systems capable of being represented as axisymmetric shells and/or solids.

This program is a refinement of the original ASHSD code developed at the University of California at Berkeley. The present program has been highly modified for the special purpose of static and dynamic analysis of nuclear containment structures. The modified program has the following features:

- The code has a shell finite element which uses an interaction stiffness that allows analysis of layered shells.

- Since shell layers may be bonded or unbonded from each other, it is possible to describe concrete shells in their actual geometric form. For example, it is possible to describe liner plate, concrete, reinforcing steel, and post-tensioning steel in their real spatial locations.

- Post-tension forces may be applied to the shell by subjecting only the unbonded post tensioning elements to a pseudothermal loading.

- Isotropic or orthotropic elastic constants are possible for both shell and solid elements. The orthotropic material properties may be used to describe the different stiffness of reinforcing steel in the hoop and meridional directions, for examples.

- Nonuniform thermal gradients through the wall thickness may be imposed.

- Eigenvalues and eigenvectors may be computed by the program.

- Three dynamic response routines are available in the program. They are:

  - Arbitrary dynamic-loading or earthquake-base excitation using an uncoupled (modal) technique.

  - Arbitrary dynamic-loading or earthquake-base excitation using a coupled (direct integration) technique.

  - Response spectrum modal analysis for absolute and square root of the sum of the squares displacements and element stresses.

- The coupled time-history solution has the capability to allow an arbitrary damping matrix.

- The stiffness and mass matrices may be obtained as punched output for input into other programs.

This program allows a useful study of the interaction between a typical nuclear containment structure modeled as an axisymmetric shell and the subsoil modeled as an axisymmetric solid.

This program was verified by comparing the computer results with hand calculations and published references. Three sample problems are presented as examples of verification.

Sample Problem: Closed Cylinder Under Internal Pressure

This problem demonstrated the membrane state of stress in a closed cylinder subjected to a uniformly distributed internal pressure. Hand calculations were used to verify this aspect of the program.

The selected problem was a cylinder with closed ends subjected to internal pressure. Only one half of the cylinder was required in the model because of symmetry. Furthermore, it was assumed that the closed ends were distant from the section being analyzed and they were excluded.

Two models of the cylinder were actually analyzed. One model used the thin shell elements and the other used the axisymmetric solid elements. These models are shown in Figures 3.8A-1 and 3.8A-2 with their key dimensions.

The problem parameters for both test cases are as follows:

Boundary Conditions:

Node 1: Z displacement = 0
θ displacement = 0
Rotation in R-Z plan = 0
(free to move radially)

Node 16: θ displacement = 0
(free to move axially, radially and to rotate about the θ axis)

Numerical Data:

Material: concrete
Modulus of Elasticity = E = 4.031 x 10^6 psi
Thickness = t = 36"
Radius = R = 900"
Poisson's Ratio = υ = 0.17
Pressure = p = 60 psi
Length = L = 1800"
N = 27,000 lb/in (an equivalent node load applied at Node 16)

The theoretical values for the membrane force resultants were calculated to be pR/2 (= 27,000 lb/in) axial force, and pR (= 54,000 lb/in) for the circumferential force (hoop direction).

The results obtained from the ASHSD program are presented in Table 3.8A-1, both for the thin shell and the layered shell models. Analytical computations indicated maximum errors at Node 16 of .4% for the longitudinal force and 3.2% for the circumferential force.

Sample Problem: Cyclindrical Shell Subjected to Internal Pressure and Uniform Temperature Rise

This test example demonstrated the use of a combined static load and thermal load condition. A short circular cylindrical shell clamped at both ends was subjected to an internal pressure and a uniform temperature rise. The theoretical solutions given in Reference 3.8A-2 were used to verify this analysis.

This test used a short cylinder that was clamped at both ends. The cylinder had an internal pressure applied and was subjected to a uniform temperature increase. The general arrangement is shown in Figure 3.8A-3.

Because of symmetry, only one-half of the cylinder was used for the finite element model. This is shown in Figure 3.8A-4 with Node 1 located at the middle of the cylinder. For the purpose of inputting the thermal coefficient of expansion of this isotropic shell, it was required to identify the shell material as orthotropic.

Boundary Conditions:

At center of cylinder, Node 1:	Z displacement = 0 θ displacement = 0 Rotation in the R-Z plane = 0
At end of cylinder, Node 26:	R displacement = 0 Z displacement = 0 θ displacement = 0 (tangential) Rotation in the R-Z plane = 0

Numerical Data

Material: concrete
Modulus of Elasticity = E = 4,030,508 psi
Poisson's Ratio = υ = 0.17
Thermal Coefficient of Expansion = α = 55 x 10^{-7} in/in/°F
Thickness = t = 30"
Radius = R = 600"
Length = L = 1200"
Pressure = p = 60 psi
Temperature = T = 150°F
R/t = 20
L/R = 2

The theoretical results are shown in Figure 3.8A-5. These values were obtained by using the following equations from Reference 3.8A-2:

Axial Moment: $$M_X = 2\mu^2 D_X \left(\frac{pR^2}{Et} + R\alpha T\right)$$

where $$\mu^2 = \left[\frac{3\ (1-\nu^2)}{R^2 t^2}\right] 1/2$$

and $$D_X = \frac{Et^3}{12\,(1-\nu^2)}$$

Normalized length: Ln. = (Z/R) (L/2R)

Figure 3.8A-5 compares the results obtained from the ASHSD program and the theoretical solution. The results of ASHSD agree well with those of the reference.

Sample Problem: Asymmetric Bending of a Cylindrical Shell

The purpose of this test example was to illustrate the use of higher harmonics for asymmetric loading cases. As a comparison to the computer output, results for this problem were taken from B. Budiansky and P. P. Radkowski's Numerical Analysis of Unsymmetric Bending of Shells of Revolution (Reference 3.8A-3).

The cylindrical shell that was analyzed was a short, wide cylinder as shown in Figure 3.8A-6. The finite element idealization of the cylinder and the pertinent data are illustrated in Figure 3.8A-7. At each end of the cylinder, moments of the form $M = M_0 \cos \theta$ were input for harmonics n = 0, 2, 5, 20.

The problem parameters are as follows:

Material: steel
$E = 29 \times 10^6$ psi
$t = 1.25"$
$R = 60.0"$
$\upsilon = 0.3$
$L = 60.0"$
$= L/R = 1$
$R/t = 48$

$$M_0 = \frac{Et^2}{100(1-\upsilon^2)}$$

$= 497939.56$ lb - in/in

The comparison results were taken directly from the reference. Those results were plotted in Figures 3.8A-8-1 and 3.8A-8-2.

The comparison of the computer results to the reference results are shown in Figures 3.8A-8-1 and 3.8A-8-2. (Note that the longitudinal moments and radial displacements are expressed as nondimensional ratios.)

The reference and computer results showed good agreement. This verified the accuracy of the program for this type of analysis.

3.8A.3 CECAP

CECAP computes stresses in a concrete element under thermal and/or non-thermal (real) loads, considering effects of concrete cracking. The element represents a section of a concrete shell or slab, and may include two layers of reinforcing, transverse reinforcing, prestressing tendons, and a liner plate.

CECAP assumes linear stress-strain relationships for steel and concrete in compression. Concrete is assumed to have no tensile strength. The solution is an iterative process, whereby tensile stresses found initially in concrete are relieved (by cracking) and redistributed in the element. Equilibrium of nonthermal loads is preserved. For thermal effects, the element is assumed free to expand inplane, but fixed against rotation. The capability for expansion and cracking generally results in a reduction in thermal stresses from the initial condition.

To verify this program, example problems were analyzed by CECAP and compared with hand calculation solutions. These example problems considered a reinforced concrete beam as shown in Figure 3.8A-9. The problem parameters are as follows:

Concrete modulus of elasticity,	E_c	$= 3 \times 10^6$ psi
Rebar modulus of elasticity,	E_s	$= 30 \times 10^6$ psi
Concrete Poisson's ratio,	ν_c	$= .22$
Concrete coefficient of thermal expansion	α_c	$= 6 \times 10^{-6}$ in/in/°F
Temperature difference	ΔT	$= 100°F$
Rebar coefficient of thermal expansion	α_R	$= \alpha_c$

Three sample problems are presented as examples of verification.

Sample Problem: Beam With a Thermal Moment

The analysis of a reinforced concrete beam subjected to a linear thermal gradient was performed to test the redistribution of thermal stresses due to the relieving effect of concrete cracking. The results were compared with hand calculations.

Figure 3.8A-10 shows the reinforced concrete beam and the corresponding CECAP concrete element used in the analysis. Boundary conditions, geometry, and applied loads are illustrated.

The following illustrates how thermal loads are treated in a cracked section analysis of a reinforced concrete beam. The main assumptions pertaining to thermal boundary conditions are:

(1) The beam is allowed to expand freely axially.

(2) There is no rotation of the initial thermal stress slope.

The beam cross-section and initial thermal stress distribution are shown in Figure 3.8A-11. For T - 100°F, the equivalent thermal moment and concrete and rebar stresses are:

$$M = \Delta T^{\alpha c} E_c bt^2/12 = (100)(6x10^{-6})(3x10^6)(12)(42)^2/12$$
$$= 3{,}175{,}000 \text{ in-lbs}$$

$$\sigma c = \Delta T^{\alpha c} E_c/2 = (100)(6x10^{-6})(3x10^6)/2 = 900 \text{ psi (compression)}$$

$$\sigma c = \frac{(t/2-2)}{t/2}\sigma_c = \frac{(21-2)}{21} 900 = 814 \text{ psi (tension)}$$

The stress diagram used for the cracked section analysis with thermal loading is shown in Figure 3.8A-12. The assumptions of free movement axially and constant thermal stress slope are maintained by a lateral translation of the initial reference axis to a final cracked position.

From force equilibrium: $F_{rebar} + F_{concrete} = 0$

$$\underbrace{1.0\,(814 + \Delta\sigma_c)\ 10}_{F_{rebar}} \underbrace{- 900\,(\frac{42}{2})\,(\frac{12}{2}) + \frac{\Delta\sigma_c(12)}{2}\,[21 + (\frac{900-\Delta\sigma_C}{900})\,21]}_{F_{concrete}} = 0$$

Solving for σ_c,

$\Delta\sigma_c = 582$ psi

Rebar and concrete stresses are:

$f_s = (814+582)10 = 13{,}970$ psi (Tension)

$f_c = 900-582 = 318$ psi (Compression)

Location of cracked neutral axis is:

$$kd = x = (\frac{900-582}{900})\ 21 = 7.42\ in.$$

Self-relieved thermal moment is:

$$M_T = \frac{f_S A_S \ (d-\frac{X}{3})}{12} = \frac{13970(1)\ (40-2.47)}{12} = 43{,}690 \frac{inch\text{-}lb}{inch}$$

The rebar and concrete stresses, self-relieved thermal moment and neutral axis location obtained from the CECAP program are compared with the hand calculations in Table 3.8A-2. It can be seen that the CECAP results compare favorably with the hand calculations.

Sample Problem: Beam With a Real Moment

The analysis of a reinforced concrete beam subjected to a real moment was performed to test the CECAP program for non-thermal moments. The results were compared with hand calculations.

Figure 3.8A-13 shows the loading and geometry for the reinforced concrete beam and the corresponding CECAP concrete element model.

The following illustrates the working stress analysis of reinforced concrete beams. The beam cross-section, stress block, and transformed sections are shown in Figure 3.8A-14. The resultant forces and moment are:

$$C = f_c \quad (kd)(b)/2$$

$$T = A_s \ f_s$$

$$M = Cjd = Tjd$$

Equating the first moments of the compression and tension areas about the neutral axis of the transformed section,

$$kd(b)\frac{(kd)}{2} = nA_s \ (d - kd)$$

which yields

$$kd^2 + 1.67kd - 66.67 = 0$$

Solving for kd;

$$kd = 7.37 \text{ in.}$$

The resultant forces are:

$$C = T = \frac{M}{jd} = \frac{3,175,000}{\left(40 - \frac{7.37}{3}\right)}$$

C = T = 84,570 lb.

Rebar and concrete stresses are:

$$f_s = \frac{T}{A_s} = 84,574 \text{ psi (tension)}$$

$$f_c = \frac{2C}{kdb} = \frac{2(84,574)}{(7.37)(12)} = 1,193 \text{ psi (compression)}$$

Table 3.8A-3 shows a comparison of rebar and concrete stresses and neutral axis locations obtained from the CECAP program and hand calculations. The CECAP results are shown to compare to hand calculations within the force accuracy limits in the program.

Sample Problem: Beam with a Real Moment and a Real Axial Load

This verification problem involves the analysis of a reinforced concrete beam subjected to both a real moment and a real axial compressive load. A hand calculation solution using the equations presented in Reference 3.8A-4 was obtained and compared with the CECAP results.

The loading and geometry for the reinforced concrete beam and corresponding CECAP model are illustrated in Figure 3.8A-15.

The following illustrates the working stress analysis of reinforced concrete beams subjected to both moments and axial compressive loads. The beam cross-section and stress block are shown in Figure 3.8A-16. The analysis uses the equations presented in Reference 3.8A-4, which are simplified to the following:

(1) $$(kd)^3 + 3\left(\frac{M}{N} - \frac{t}{2}\right)(kd)^2 + \frac{6nA_s}{b}\left(d - \frac{t}{2} + \frac{M}{N}\right)(kd) - \frac{6nA_s d}{b}\left(d - \frac{t}{2} + \frac{M}{N}\right) = 0$$

(2) $$f_s = \frac{N}{A_s} \frac{(\frac{M}{N} = \frac{kd}{3} - \frac{t}{2})}{(d - \frac{kd}{3})}$$

(3) $$f_c = \frac{f_s kd}{n(d - kd)} \ for \ \frac{M}{N} \geq t/6$$

Equation (1) becomes:

$$kd^3 + 55.8kd^2 - 293kd = 11720 = 0$$

$$M/N = \frac{317500}{101000} = 31.4 \geq t/6 = \frac{42}{6} = 7$$

Solving the above equations by iteration for kd yields:

kd = 12.7 in.

The resulting rebar and steel stresses are:

$$f_s = \frac{101000}{1.0} \frac{(31.4 + 12.7/3 - 21)}{)40 - 12.7/3)} = 41,320 \ psi \ (Tension)$$

$$f_c = \frac{41320 \ (12.7)}{10 \ (40-12.7)} = 1,922 \ psi \ (Compression)$$

The rebar and concrete stresses and neutral axis location obtained from the CECAP program are compared with the hand calculations in Table 3.8A-4. The results for the two solution methods agree very closely.

3.8A-4 CE 668

This program performs the linear elastic analysis of a plate with arbitrary shape and supports, stiffener beams, and elastic subgrade, under loads normal to the middle plane of the plate.

This program was verified by comparing selected hand calculated values to CE 668 values with the deflections and moments of a rectangular plate for different loading and support conditions.

Sample Problem: Rectangular Plate with a Concentrated Load at the Center

The simply supported rectangular plate, shown in Figure 3.8A-17 was subjected to a concentrated load of 300 lbs. at the center. Because of symmetry only half of the plate was modelled by the finite elements. The boundary conditions were zero displacement with free normal rotation at the simply supported edges and free displacement with zero normal rotation at the symmetry axis. The plate had isotropic structural properties.

The problem parameters are as follows:

Poisson's Ratio	$\upsilon = 0.3$
Young's Modulus	$E = 2.9 \times 10^7$ psi
Thickness	$h = 0.5$ in.
Concentrated Load	$P = 300$ lb.

The formulas for the deflections and moments were taken from Reference 3.8A-5.

a) Deflection

$$@\ center\ \omega = .01695\ \frac{Pa2}{D} = .01695\ \frac{300\ (100)\ 12\ (1-(\ .3)^2)}{(2.9\ x\ 10^7)\quad (.5^3)}$$

$$\omega = .00153\ in.\ @\ Node\ 116$$

b) Moments

The hand calculated values for deflections and moments are compared with the CE 668 values in Table 3.8A-5. The results are very close with the greatest difference being 1.55%.

Sample Problem: Uniform Load on a Rectangular Plate With Various Edge Conditions

The rectangular plate had one edge fixed, one edge free, and two edges simply supported as shown in Figure 3.8A-18. It was subjected to a uniformly distributed load of intensity q = 2.0 psi. Because of symmetry only half of the plate was modelled by finite elements. Boundary conditions were specified according to the appropriate edge support conditions.

$M\chi$: (*for* $b>>$) a

$$@x = 2, y = 0 \quad M\chi = \frac{-P(1+\nu)}{8\pi} \, 1n \, [\frac{1 - \sin \frac{\pi \ \chi}{a}}{1 + \sin \frac{\pi \ \chi}{a}}]$$

$$M\chi = \frac{-300(1.3)}{8\pi} \, 1n \, [\frac{1 - \sin \frac{\pi}{5}}{1 + \sin \frac{\pi}{5}}] = (-15.52)\ (-1.348)$$

$M\chi = 20.92$ 1b - *in.* @ *Node* 113

My : (*for* $b >> a$)

$$@\chi = 6, y + 0 \quad My = \frac{-P(1 + \nu)}{8\pi} \, 1n \, [\frac{1 - \sin \frac{\pi \ \chi}{a}}{1 + \sin \frac{\pi \ \chi}{a}}]$$

$$= \frac{-300(1.3)}{8\pi} \, 1n \, [\frac{1 - \sin \frac{3\pi}{5}]}{1 + \sin \frac{3\pi}{5}}]$$

$$= (-15.52)\ (-3.685)$$

$$My = 57.198 \ lb\text{-}in \ @ \ Node \ 117$$

The problem parameters are as follows:

Poisson's Ratio	$\nu = 0.3$
Young's Modulus	$E = 2.9 \times 10^7$ psi
Thickness	h = 0.2 in.
Load Intensity	q = 2.0 psi

The formulas used to calculate the deflections and moments were taken from Reference 3.8A-5.

a) Deflection

$$@\chi = 15, y = 15\omega = .0582 \ (\frac{gb^4}{D}) = .0582 \ [\frac{2(15)^4(12)\ (1-(.3)^2)}{(2.9 \ x \ 10^7)\ (.2)^3}]$$

$\omega = .277$ *in.* @ *Node* 11

b) Moments

The hand calculated values for the deflection and moments are compared to the CE 668 results in Table 3.8A-6. The results

M_χ:

$@\chi = 15, y = 15$ $\quad M_\chi = .0293\ ga^2 = .0293\ (2)(30)^2$
$M_\chi = 52.74\ in\text{-}lbs\ @\ Node\ 11$

M_y:

$@\chi = 15, y = 0$ $\quad M_y = .319\ gb^2 = .319\ (2)(15)^2$
$M_y = 143.55\ in\text{-}lbs.\ @\ Node\ 121$

agree closely, with the largest difference being 3.4%.

3.8A.5 EASE

EASE (Elastic Analysis for Structural Engineering) performs static analysis of two- and three-dimensional trusses and frames, plane elastic bodies and plate and shell structures. The finite element approach is used with standard linear or beam elements, a plane stress triangular element or a triangular plate bending element. The EASE program accepts thermal loads as well as pressure, gravity, or concentrated loads.

The program output includes joint displacements, beam forces and triangular element stresses and moments.

EASE was developed by the Engineering Analysis Corporation, Redondo Beach, California, in 1969 and is in the public domain. The version currently used by Bechtel is maintained by the Control Data Corporation, Cybernet Service.

3.8A.6 EO119

This program performs an analysis of a bolted flange. Flange dimensions reflect the corroded condition. Symbols, terms, and mathematics are in accordance with Appendix XI of ASME Code Section III. Stress values for both design (operating) and bolt-up conditions are printed. Both allowable and actual stresses are printed out for bolts, longitudinal flange stress, radial flange stress, and tangential flange stress. The shape constants and moments are printed out for information only.

Two program solutions are included in verifying Program EO119. A welding neck flange design and a slip-on flange design have been prepared. Also attached are solutions of the same problems as published in Bulletin 502, Modern Flange Design from Gulf & Western Manufacturing Company (Reference 3.8A-6).

The problem parameters for the two sample problems are as follows:

Design pressure = 400 psi
Design temperature = 500°F
Atmospheric temperature = 75°F
Poisson's ratio = 0.30
Corrosion allowance = 0
Gasket width = 0.75"
Effective gasket width = 0.306"
Gasket Factor = 2.75
Gasket seating strength = 3700 psi

Sample Problem: Welding Neck Flange

Figure 3.8A-19 shows the dimensions of the welding neck flange. Table 3.8A-7 compares the results of EO119 computer program with those published in Reference 3.8A-6. The results compare very closely.

Sample Problem: Slip-on Flange

Figure 3.8A-20 shows the dimensions of the slip-on flange. Table 3.8A-8 compares the results of EO119 computer program with those published in Reference 3.8A-6. The results compare very closely.

3.8A.7 EO781

The Shells of Revolution Program was developed by Aerturs Kalnin while at Yale University. The Mathematics are based on a method of analysis contained in his paper "Analysis of Shells of Revolution Subjected to Symmetrical and Non-Symmetrical Loads" published in the Journal of Applied Mechanics, Vol. 31, September, 1964 (Reference 3.8A-7).

This program calculates the stresses and displacements in thin walled elastic shells of revolution when subjected to static edge, surface, and/or temperature loads with arbitrary distribution over the surface of the shell. The Geometry of the shell must be symmetric, but the shape of the median is arbitrary. It is possible to include up to three branch shells with the main shell in a single model. In addition, the shell wall may consist of different orthotropic materials, and the thickness of each layer and the elastic properties of each layer may vary along with the median.

Program E0781 numerically integrates the eight ordinary first order differential equations of thin shell theory derived by H. Reissner. The equations are derived such that the eight variables are chosen which appear on the boundaries of the axially symmetric shell so that the entire problem can be expressed in these fundamental variables.

Kalnin's program has been altered such that a 4 x 4 force-displacement relation can be used as a boundary condition as an alternative to the usual procedure of specifying forces or displacements. This force-displacement relation can be used to describe the forces at the boundary in terms of displacements at the boundary, or the displacements at the boundary in terms of forces or some compatible combination of the two. In this manner, it is possible to study the behavior of a large complex structure. It is also possible to introduce a "Spring Matrix" at the end of any part of the stress model. This matrix must be expressed in the form, Force = Spring Matrix X Displacement. In addition, to the above changes, the Kalnin's Program has been modified to increase the size of the problem that can be considered and to improve the accuracy of the solution.

This program was verified by comparing the computer results with experimental measurements and published references. Two sample problems are presented as examples of verification.

Sample Problem: Comparison of 2:1 Ellipsoidal and Torispherical Heads Subjected to an Internal Pressure Load

This problem illustrates Program E0781's ability to generate cylindrical, torispherical, and ellipsoidal shapes.

A comparison is made to an experimental investigation of 2:1 ellipsoidal heads subjected to internal pressure (see Reference 3.8A-8).

The problem consists of comparing a 2:1 ellipsoidal head to an equivalent torispherical head subjected to the same uniformly distributed internal pressure. An equivalent torisphere will be defined as one having the same height above the tangent line as the ellipsoid and a minimal L/b ratio (thus having the least possible discontinuity between the torus and the sphere). For the geometry shown in Figure 3.8A-21:

$$(L-b)\ \sin\phi_0 = A-r \quad (1)$$

$$(L-b)\ \cos\phi_0 = L-B \quad (2)$$

Minimizing L/b using (1) and (2):

$$\tan \phi_0 = B/A = 0.5019$$

$$\phi_0 = 26.653°$$

$$L/A = \frac{C \pm \sqrt{C^2 - 2C}}{2}$$

$$C = B/A + A/B = 2.494$$

$$L = \frac{18.19}{2} [2.5 + \sqrt{6.22 - 4.99}] = 32.778"$$

$$b = B [B/A - L/A] + A = 9.13 [.5019 - 1.80198] + 18.19 = 6.32"$$

Note: For purpose of calculation:

A = 18.19"
B = 9.13" from Figure 3.8A-21

Segment lengths used are:

cylinder - $\sqrt{rt} = \sqrt{18.16 (0.31)} = 2.37$

torisphere

5° to 10° - 4 @ 1.25°

10° to 26.567° - 4. @ 4.13°

26.567° to 90° - 6 @ 10.57°

ellipsoid

5° to 10° - 4 @ 1.25°

10° to 30° - 4 @ 5°

30° to 90° - 6 @ 10°

<u>Boundary Conditions</u>:

It will be assumed that at 5° from the pole a membrane state of stress exists in both the ellipsoid and the torisphere:

$$Q = M\phi = 0$$

$$N\phi = \frac{pr}{2 \sin\phi}$$

where r = distance to polé = 32.778"

Q = tranverse shear in direction.
$M\phi$ = moment resultant in ϕ direction.
$N\phi$ = membrane force in ϕ direction.

Letting p = 680 psi

Then for the torisphere:

$N\phi$ = (680/2) (32.778) = 11,144.5 lb/in.

If $N\phi$ = 11,144.5 lb/in., a preliminary run yields Q = 95.202 lb/in., so a new value for $N\phi$ for the torisphere was calculated:

$$\Delta N = \Delta \frac{Q}{\tan \phi}$$

$N\phi$ = 11,144.5 + ΔN = 10056.3 lb/in. and an appropriate membrane state was generated.

For the ellipsoid

$$r = \frac{A \sin \phi}{R}$$

where

$$R = \sqrt{C_1 + (1-C_1)\ \sin^2\phi}$$

$$C_1 = (B/A)^2 = 0.2519$$

$$R = \sqrt{.2519 + .7481\ (0.0871557)^2} = 0.5075$$

$$N\phi = \frac{A\ sin\phi}{R}\ \frac{P}{2\ sin\phi} = \frac{18.19\ (680)}{2\ (0.5075)} = 12{,}185.78\ lb/in.$$

To better compare the heads it seemed desirable to have the longitudinal displacement at the center of the cylinder 0 ("ϕ = 0). So the problem was run twice, the first run yielding the radial displacement, W required for 0 displacement at the center (W = 0.0966").

1. Start W = 0.0966" $N\phi$ = 10,056 lb/in $M\phi$ = N = 0
2. End Q = N = $M\phi$ = 0 $N\phi$ = 12,186 lb/in.

Figure 3.8A-24 shows the analytical model with boundary conditions.

Results

To check the results, first the answers at the boundaries should be examined. It was assumed that there was a membrane state of stress at the boundaries and, therefore, at the edges Q and M must be approximately 0.

	Q (lbs/in)	Mϕ (in. - lbs/in.)
Start	- 0.01027	0.0
End	- 0.0008613	-0.0001487

Also to satisfy equilibrium in the cylinder, $N\phi \simeq 0.5pr$ = 6169 lb/in.

Plots of the hoop force and longitudinal bending from E0781 results compare the ellipsoidal and torispherical heads. Even though the change in radii has been minimized the disturbance at the junction of the sphere and torus is considerable (see Figure 3.8A-25).

Comparison to the experimental ellipsoidal head shows good correlation of stress values. See Figures 3.8A-26 through 3.8A-30 for plots $\nabla\phi$ and $\nabla\theta$ on the inside, outside, and meridian of the head. Deviations are caused by the changes in thickness and the experimental head's variation from a true 2:1 ellipsoidal head.

Sample Problem: Cylindrical Water Tank with Tapered Walls

This problem illustrates Program E0781's capability to analyze a pressure load with one fixed boundary condition and one free boundary condition.

The problem used for this verification is "Shell of Variable Thickness" taken from "Stresses in Shells", by W. Flugge, pp. 289-295 (Reference 3.8A-9).

The problem consists of a tapered shell filled with water. The shell has a radius of 9'-0" and is 12'-0" high. The shell thickness varies from 11" at the bottom to 3" at the top. See Figure 3.8A-31 for location of the Z axis. The length of a segment is 18" ($\sqrt{rt}$).

Taking the weight of water as 62.5 lb/ft^3, the pressure at the bottom of the tank is

$$p = \frac{(12\ ft)\ (62.5\ lb/ft^3)}{144\ in^2/ft^2} = 5.2083psi$$

The pressure at the top is zero. The pressure varies linearly so that only two points are needed in the function generator in order to fully describe the function.

Boundary Conditions

W - displacement normal to surface
$U\phi$ - displacement component in ϕ direction
$B\phi$ - rotation of reference surface in ϕ direction
Q - transverse shear in ϕ direction
$N\phi$ - membrane force in ϕ direction
$M\phi$ - moment resultant in ϕ direction

1. fixed at start $W = U\phi = B\phi = 0$
2. free at end $Q = N\phi = M\phi = 0$

Results

Table 3.8A-9 lists the Program EO781 results and compares them with the theoretical solutions from Reference 3.8A-9 at two locations.

Program EO781 gives a maximum hoop force, $N\theta$ = 346.8 lb/in. = 4160 lb/ft at 54" from the base. This value differs from the theoretical solution of 4180 lb/ft by 0.48%.

Program EO781 gives a maximum moment of the base, $M\phi$ = 1539 in-lb/in. = -1539 ft-lb/ft. This value differs from the theoretical solution of -1470 ft-lb/ft by 4.69%.

3.8A.8 FINEL

This program performs the static analysis of stresses and strains in plane and axisymmetric structures by the finite element method. In this method, the structure is idealized as an assemblage of two-dimensional finite elements of triangular or quadrilateral shapes having arbitrary material properties. Reinforcement of concrete materials is included by adjusting the element material properties. Special emphasis is made on bilinearity in compression and bilinearity or cracking in tension. FINEL computes the displacements of the corners of each element and the stresses and strains within each element.

To verify this program, example problems were analyzed by FINEL and compared to experimental and/or hand calculated solutions. Three sample problems are presented as examples of verification.

Sample Problem: Simply Supported Beam with a Concentrated Load at the Center

The beam shown in Figure 3.8A-32 has been the subject of an experimental and analytical investigation. The purpose of this investigation is to compare results obtained from the FINEL program with those obtained from References 3.8A-13 and 3.8A-14.

The finite element mesh used in Reference 3.8A-14 and in the FINEL analysis are shown in Figures 3.8A-33 and 3.8A-34, respectively. The FINEL analysis required a finer mesh because it used linear displacement elements while Reference 3.8A-14 used quadratic displacement elements.

The material properties of the concrete and reinforcing steel, and the loading history used in the FINEL analysis are given in Tables 3.8A-10 and 3.8A-11, respectively.

This problem was not continued beyond the yield point of the reinforcing steel due to an error in the FINEL program. The stiffness of an element which yielded should have been determined according to:

$$E_{eff} = \frac{T_y + n\ (T - T_y)}{T}\ E_o$$

where,

E_o = initial material stiffness or modulus

T_y = yield stress

T = element stress, in yield direction, at end of previous cycle (< Ty)

n = E_{plast} / E_o ; E_{plast} = plastic stiffness

E_{eff} = effective stiffness, in yield direction, to use in next cycle

A new E_{eff} should be calculated after each cycle. The FINEL program calculated an Eeff only after the first cycle following yielding, (or first cycle in a restart run), and used the value of E for all subsequent cycles in the same computer run. (This error could be overcome by making a series of one cycle restart runs).

The cracking patterns obtained from Reference 3.8A-14 and FINEL are shown in Figures 3.8A-35-1 and 3.8A-35-2. The load-deflection curves from References 3.8A-13 and 3.8A-14 and the FINEL analysis are shown in Figure 3.8A-36. The load deflection curve obtained from the FINEL analysis show very good agreement with the experimental results. The cracked region grows faster in the FINEL analysis and more slowly in Reference 3.8A-14, since the FINEL and Reference 3.8A-14 load-deflection curves show difference gradients (stiffnesses).

The results of analytical, experimental, and FINEL solutions are shown in Figure 3.8A-36. The FINEL analysis agrees well with the experimental results up to the point where the reinforcing steel in the beam yields. After the yield point, the FINEL analysis incorrectly calculated the effective stiffness of elements which have yielded. Therefore, the solution was not valid for further loadings. However, since all reinforcing steel remains elastic for the containment analysis, the FINEL program is verified and restricted for that application.

Sample Problem: <u>Axially Constrained Hollow Cylinder with a Distributed Pressure Loading</u>

This verification involves the response of an axially constrained hollow cylinder to internal pressure. A hand calculated solution yields values of tangential, axial, and radial stresses at various radii from the center of the cylinder, which are then compared to the FINEL values.

The finite element model is illustrated in Figure 3.8A-37. Nodal points are free to move only in the radial direction, representing the conditions of axisymmetry and plane strain.

The problem parameters are as follows:

Poisson's Ratio	υ	$= 0.25$
Young's Modulus	E	$= 4.32 \times 10^5$ ksf
Number of nodal points		$= 22$
Number of elements		$= 10$
Internal Pressure	P	$= 1.0$ ksf

From Reference 3.8A-15, the following equations were used:

hoop or tangential stress, T_θ :

$$T_\theta = p\,\frac{a^2\left(b^2 + r^2\right)}{r^2\left(b^2 - a^2\right)}$$

axial stress, T_Z :

$$T_Z = \frac{p}{z} \frac{a^2}{b^2 - a^2}$$

radial stress, T_R :

$$T_R = p \frac{a^2\left(b^2 - r^2\right)}{r^2\left(b^2 - a^2\right)}$$

where a = 65.0 ft.
b = 68.75 ft.
p = 1.0 ksf
$a \leq r \leq b$

The results from FINEL for tangential, axial, and radial stresses of the hollow cylinder are compared with the hand calculated values in Table 3.8A-12. The results are exactly the same except for one value where there is only 4.17% difference.

Sample Problem: <u>Axially Constrained Hollow Cylinder with a Linear Temperature Gradient</u>

The response of an axially constrained hollow cylinder to a radially varying linear temperature gradient was the problem used for this verification. The tangential, axial, and radial stresses were determined by hand calculations and compared to the FINEL results.

Figure 3.8A-38 illustrates the finite element mesh. The conditions of axisymmetry and plane strain were imposed by using the axisymmetric quadrilateral element and restraining all nodes against axial displacement.

The temperature profile is shown in Figure 3.8A-39.

The problem parameters are as follows:

Poisson's Ratio	ν	$= 0.25$
Young's Modulus	E	$= 4.32 \times 10^5$ ksf
Coefficient of Thermal Expansion	α	$= 6 \times 10^{-6}$ ft/ft/°F
Number of nodal points		= 22
Number of elements		= 10

From References 3.8A-16 and 3.8A-17, the following equations were used:

hoop or tangential stress, δ_θ :

$$\delta_\theta = \frac{\alpha E}{1-\nu}\,\frac{1}{r^2}\,\left[\left(\frac{r^2 + a^2}{b^2 - a^2}\right)\,{}_a\!\int^b Trdr + {}_a\!\int^r Tr'\,dr' - TR^2\right]$$

axial stress, δz :

$$\delta_z = \frac{\alpha E}{1-\nu}\,\frac{1}{r^2}\,\left[\left(\frac{r^2 + a^2}{b^2 - a^2}\right)\,{}_a\!\int^b Trdr - T\right]$$

radial stress, δ_r :

$$\delta_r = \frac{\alpha E}{1-\nu}\,\frac{1}{r^2}\,\left[\left(\frac{r^2 - a^2}{b^2 - a^2}\right)\,{}_b\!\int^a Trdr - {}_a\!\int^r Tr'dr'\right]$$

where: a = 65.0 ft.
b = 68.75 ft.
T = T(r) = temperature above reference (T_{REF} = 100°F)

Expression for the temperature field:

$T(r) = C_2\,r + C_1$

$T(a) = 25 = C_1 + 65.0C_2$

$T(b) = -25 = C_1 + 68.75C_2$

solving,

$$C_2 = \frac{-50}{68.75-65} = -13.33$$

$$C_1 = -25 - 68.75(-13.33) = 891.67$$

then

$$T(r) = -13.33r + 891.67$$

Evaluation of the integral:

$$\int Trdr = \int (-13.33r + 891.67)\ rdr$$

$$= \frac{-13.33r^3}{3} + \frac{891.67r^2}{2} + C$$

$$= -4.44r^3 + 445.83r^2 + C$$

$$\int_a^b Trdr = -4.44(b^3-a^3) + 445.83(b^2-a^2)$$

$$\int_a^r Tr'dr' = -4.44(r^3-a^3) + 445.83(r^2-a^2)$$

The results from FINEL for the tangential, axial, and radial stresses are compared with the values obtained by hand calculations in Table 3.8A-13. The results between the two methods of solution agree very closely.

3.8A.9 ME 620

The heat conduction program, ME 620, is used to determine the temperature distribution, as a function of time, within a plane or axisymmetric solid body subjected to step-function temperature or heat flux inputs. The program is also used for steady-state temperature analysis.

The program utilizes a finite element technique coupled with a step-by-step time integration procedure as described in "Application of the Finite Method to Heat Conduction Analysis" by E. L. Wilson and R. E. Nickell (Reference 3.8A-18).

The program was developed at the University of California, Berkeley, by Professor E. L. Wilson and subsequently modified by Bechtel Corporation to incorporate the save and restart capabilities.

To verify this program, example problems were analyzed by ME 620 and compared with program data. Two sample problems are presented as examples of verification.

Sample Problem: Heat Conduction in a Square Plate with One Edge Quenched

This problem tested the ability of the program to solve the temperature changes in a plane region subjected to conduction boundary conditions. The plate was brought to an equilibrium temperature and one edge was quenched while the other three edges were kept insulated.

A square plate was brought to equilibrium at a given initial temperature, T . Three edges were perfectly insulated while a third edge was suddenly brought to a lower temperature, T . This quench was kept constant for the entire analysis. A temperature time history was then obtained for the corner farthest from the quenched edge.

Figure 3.8A-40 shows the actual plate arrangement, while Figure 3.8A-41 shows a diagram of the finite elements.

The problem parameters are as follows:

Nomenclature

L = length of longest heat flow path
T_o = initial temperature of slab (°F)
T_1 = quenching temperature of edge (°F)

Data:

The plate was 10" x 10" square.

T_o = 100°F
T_1 = 0°F

Diffusivity α = 1.0 in^2/sec (chosen for convenience)
Time increment ΔT - 1 second for numerical solution

At any time t during the transient state, the time factor T (or characteristic time) is given by $T = \alpha t/L^2$. The time to reach steady-state is given when $T = 1.0$, hence the transient time is $t = L^2/\alpha$ = 100 seconds. The results derived from Reference 3.8A-19 are plotted in Figure 3.8A-42.

The temperature variation at point A was plotted in Figure 3.8A-42 according to the results of ME 620 and compared with the theoretical transient change. The curves are seen to agree quite well. Deviations are due to the selected finite element mesh size and to the selected time step for the analysis.

Sample Problem: Heat Conduction in a Surface Quenched Sphere

This problem tested the ability of ME 620 to analyze the temperature distribution in an axisymmetric solid with given temperature boundary conditions. The results of the program analysis were compared to a closed-form solution derived from Reference 3.8A-20.

This problem considered a solid steel sphere (shown in Figure 3.8A-43) that was brought to an equilibrium temperature, and then its surface was suddenly quenched to a lower uniform temperature. The quenching environment was held at a constant temperature. A temperature-time history for three seconds was obtained from the program for all node points. The points used for the comparison were at a radius of 0.2 inches, and only one time period was checked. The finite element model is shown in Figure 3.8A-44. The problem parameters are as follows:

Nomenclature:

L = length of the longest heat flow path (radius of sphere)
T_o = initial temperature of sphere (°F)
T_1 = quenching temperature of outer surface (°F)

Data

Radius of sphere = R = .59 in.

T_o = 1472°F
T_1 = 68°F

Conductivity = 6.02×10^{-4} Btu/in-sec-°F
Diffusivity = α = .0193 in^2/sec
Specific heat = .11 Btu/(lb-°F)
Density = ρ = .284 lb/in^3
Time increment = .2 sec.

At any time, t, during the transient state, the time factor T (or characteristic time) is given by $T = \alpha \ t/L^2$. The time to reach steady-state is given when $T = 1.0$, hence the transient time is $t = L^2/\alpha = 3.0$ seconds. The result from Reference 20 for the temperature at a radius of 0.2 inches at time t = 1.8 seconds; was 933.8°F.

The temperatures from both the program and the reference are shown in Table 3.8A-14. There is an error of 1.1%.

3.8A.10 SUPERB

SUPERB is a general-purpose, isoparametric, finite element computer program. The program determines the displacement and stress characteristics of complex structures subjected to concentrated loads, pressure distributions, enforced displacements, and thermal gradients, as well as the temperature distribution due to steady-state heat transfer. Isoparametric elements with curved boundaries and high-order strain variations permit curved regions and area with high stress concentrations to be accurately represented with a minimum number of elements.

The SUPERB program is a recognized program in the public domain and has had sufficient history of use to justify its application and validity without further demonstration. The version of the program currently used by Bechtel is maintained by the Control Data Corporation, Cybernet Service.

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TABLE 3.8A-1

TABULATION OF MEMBRANE STRESS RESULTANTS FROM THE ASHSD PROGRAM

Node Point	Thin Shell		Layered Shell	
	Longitudinal Force lb/in	Circumferential Force lb/in	Longitudinal Force lb/in	Circumferential Force lb/in
1	27000.	54004.	27000.	54004.
2	27000.	54005.	27000.	54005.
3	27000.	54008.	27000.	54008.
4	27000.	54012.	27000.	54012.
5	27000.	54015.	27000.	54015.
6	27000.	54012.	27000.	54012.
7	27001.	53999.	27001.	53999.
8	27001.	53968.	27001.	53968.
9	27001.	53912.	27001.	53912.
10	27000.	53829.	27000.	53829.
11	26999.	53731.	26999.	53731.
12	26997.	53654.	26997.	53654.
13	26994.	53674.	26994.	53674.
14	26989.	53912.	26989.	53912.
15	26984.	54532.	26984.	54532.
16	27111.	55724.	27111.	55724.

NOTE: Node Point 1 represents the center of the cylinder.

TABLE 3.8A-2

CECAP and Hand Calculation Comparison - Thermal Gradient

	CECAP	HAND CALCULATIONS	% ERROR
f_s	13,150 psi	13,790 psi	5.9
f_c	-331 psi	-318 psi	4.1
k_d	7.55 in	7.42 in	1.8
M_T	43,760 in-lb/in	43,690 in-lb/in	0.2

TABLE 3.8A-3

Comparison of CECAP and Hand Calculation Results - Real Moment

	CECAP	HAND CALCULATIONS	% ERROR
f_s	79,170 psi	84,570 psi	6.4
f_c	-1,845 psi	-1.913 psi	3.6
k_d	7.6 in	7.4 in	2.7

TABLE 3.8A-4

CECAP and Hand Calculation Comparison - Real Moment and Real Compressive Load

	CECAP	HAND CALCULATIONS	% ERROR
f_s	41,620 psi	41,320 psi	0.7
f_c	-1908 psi	-1922 psi	0.7
k_d	12.2 in	12.7 in	3.9

TABLE 3.8A-5

Comparison of Results for the Rectangular Plate with a Concentrated Load at the Center

	Hand Calculations	CE 668
Deflection (in):		
@ Node 116	0.00153	0.00151
Moments (in-lbs):		
M_x @ Node 113	20.92	21.24
M_y @ Node 117	57.198	56.377

TABLE 3.8A-6

Comparison of Results for the Rectangular Plate with Various Edge Conditions

	Hand Calculations	CE 668
Deflection (in):		
@ Node 11	0.277	0.278
Moments (in-lbs):		
M_x @ Node 11	52.74	50.92
M_x @ Node 121	143.55	142.28

TABLE 3.8A-7

Comparison of Stresses for Welding Neck Flange

Stress Component	Allowable Stress (psi)	Actual Stress (psi)	
		E0119	Reference 3.8A-6
A. Design (operating) condition			
Bolts	25000	21801	--
Longitudinal Flange	26250	22856	22865
Radial Flange	17500	10981	10982
Tangential Flange	17500	6799	6800
B. Bolt-up Condition			
Bolts	25000	6077	--
Longitudinal Flange	26250	20278	20288
Radial Flange	17500	9743	9744
Tangential Flange	17500	6032	6033

TABLE 3.8A-8

Comparison of Stresses for Slip-on Flange

Stress Component	Allowable Stress (psi)	Actual Stress (psi)	
		E0119	Reference 3.8A-6
A. Design (operating) condition			
Bolts	25000	20971	--
Longitudinal Flange	26250	21160	21163
Radial Flange	17500	11128	11128
Tangential Flange	17500	13763	13764
B. Bolt-up Condition			
Bolts	25000	5671	--
Longitudinal Flange	26250	15644	15648
Radial Flange	17500	8227	8228
Tangential Flange	17500	10175	10177

TABLE 3.8A-9

Comparison of Final Results for Hoop Force, N_θ and Meridional Moment, M_ϕ

Distance From Base	Program E0781 Results N_θ (lb/in)	M_ϕ (in.-lb) in.	"Stresses in Shells" Solution At Maximums
0.0	5.919×10^{-6}	-1539.0	M_ϕ = -1470 ft-lb/ft
6.0	21.15	-903.9	
12.0	71.29	-440.5	
18.0	134.0	-124.8	
24.0	194.3	71.47	
30.0	253.3	177.1	
36.0	297.2	218.3	
42.0	327.3	192.8	
48.0	343.3	192.8	
54.0	346.8	157.1	N_θ = 4180 lb/ft
60.0	339.6	119.5	
66.0	324.2	85.46	
72.0	303.0	57.80	
78.0	277.9	36.29	
84.0	250.8	23.41	
90.0	222.9	15.00	
96.0	195.1	10.58	
102.0	167.8	8.685	
108.0	141.4	8.075	
114.0	115.9	7.754	
120.0	91.45	7.032	
126.0	68.13	5.584	
132.0	46.29	3.453	
138.0	26.50	1.177	
144.0	94.53	-1.481×10^{-3}	

Table 3.8A-10

Material Properties of the Concrete and Reinforcing Steel Used for FINEL Verification

Property	Concrete	Steel
E	4.3×10^6 psi	29×10^6 psi
ν	.15	.29
T_{yield}	-4820 psi	±44900 psi
E_{yield}	0.	0.
T_{crack}	+546 psi	-----
E_{crack}	1.0 psi	-----
Shear stiffness reduction factor for once cracked concrete	0.5	-----

Table 3.8A-11

Loading History Used for
the FINEL Verification

Load, P	Number of Cycles At Load for Convergence
1 lb.	1
8,700 lb.	4
20,000 lb.	4
28,000 lb.	1
31,200 lb.	4
31,300 lb.	1*

*Reinforcing steel yielded

Table 3.8A-12

Comparison of Stress Results

Element	r,ft.	Tangential Stress,ksf Hand Calculated	Tangential Stress,ksf FINEL	Axial Stress,ksf Hand Calculated	Axial Stress,ksf FINEL	Radial Stress,ksf Hand Calculated	Radial Stress,ksf FINEL
1	65.19	17.79	17.79	4.212	4.212	-0.95	-0.95
2	65.56	17.69	17.69	4.212	4.212	-0.84	-0.84
3	65.94	17.58	17.58	4.212	4.212	-0.73	-0.73
4	66.31	17.48	17.48	4.212	4.212	-0.63	-0.63
5	66.69	17.38	17.38	4.212	4.212	-0.53	-0.53
6	67.06	17.28	17.28	4.212	4.212	-0.43	-0.43
7	67.44	17.18	17.18	4.212	4.212	-0.33	-0.33
8	67.81	17.08	17.08	4.212	4.212	-0.24	-0.23
9	68.19	16.99	16.99	4.212	4.212	-0.14	-0.14
10	68.56	16.89	16.89	4.212	4.212	-0.05	-0.05

Table 3.8A-13

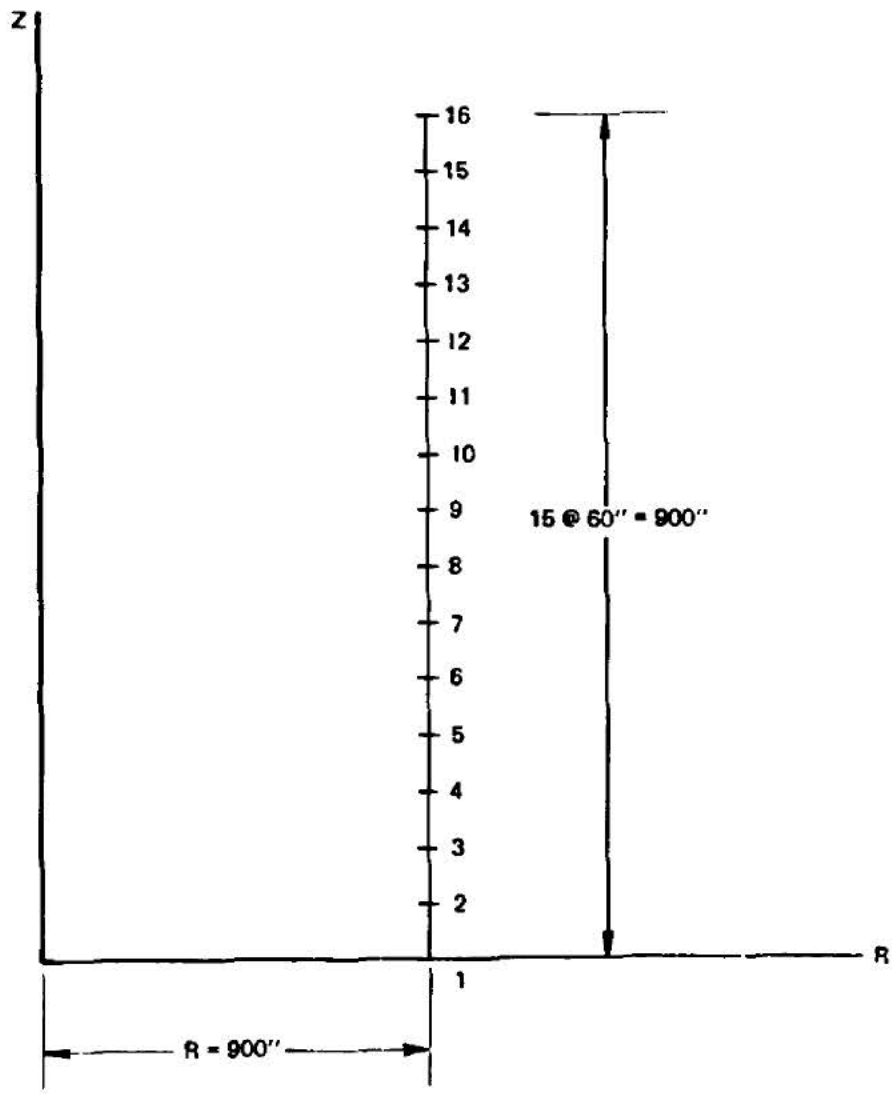
Comparison of Stress Results

Element	r, ft	Tangential Stress, ksf Hand Calculations	Tangential Stress, ksf FINEL	Axial Stress, ksf Hand Calculations	Axial Stress, ksf FINEL	Radial Stress, ksf Hand Calculations	Radial Stress, ksf FINE
1	65.19	-78.34	-78.33	-77.96	-77.96	-0.22	-0.2
2	65.56	-60.67	-60.66	-60.68	-60.68	-0.62	-0.6
3	65.94	-43.10	-43.09	-43.40	-43.40	-0.91	-0.9
4	66.31	-25.63	-25.62	-26.12	-26.12	-1.10	-1.1
5	66.69	- 8.26	- 8.25	- 8.84	- 8.84	-1.19	-1.1
6	67.06	9.01	9.02	8.44	8.44	-1.18	-1.1
7	67.44	26.19	26.20	25.72	25.72	-1.08	-1.0
8	67.81	43.27	43.28	43.00	43.00	-0.88	-0.[illegible]
9	68.19	60.26	60.27	60.28	60.28	-0.59	-0.5[illegible]
10	68.56	77.16	77.17	77.56	77.56	-0.21	-0.21

Table 3.8A-14

Comparison of Results

	ME 620	Reference 3.8A-20
Temp. 1.8 seconds after quenching	923.4	933.8°F



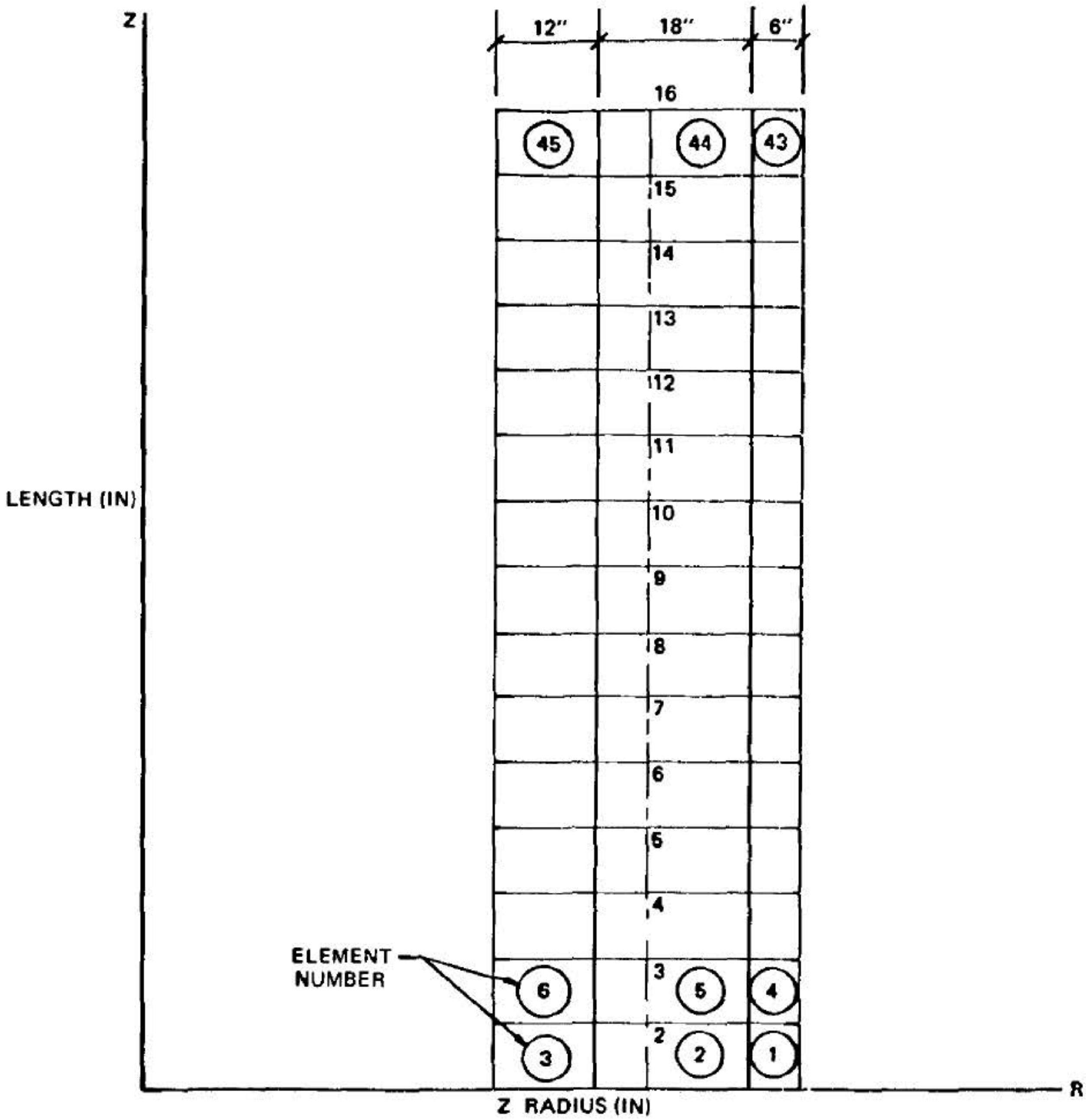
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

THIN SHELL CYLINDER

FIGURE 3.8A-1, Rev. 47

Auto-Cad Figure Fsar 3_8A_1.dwg



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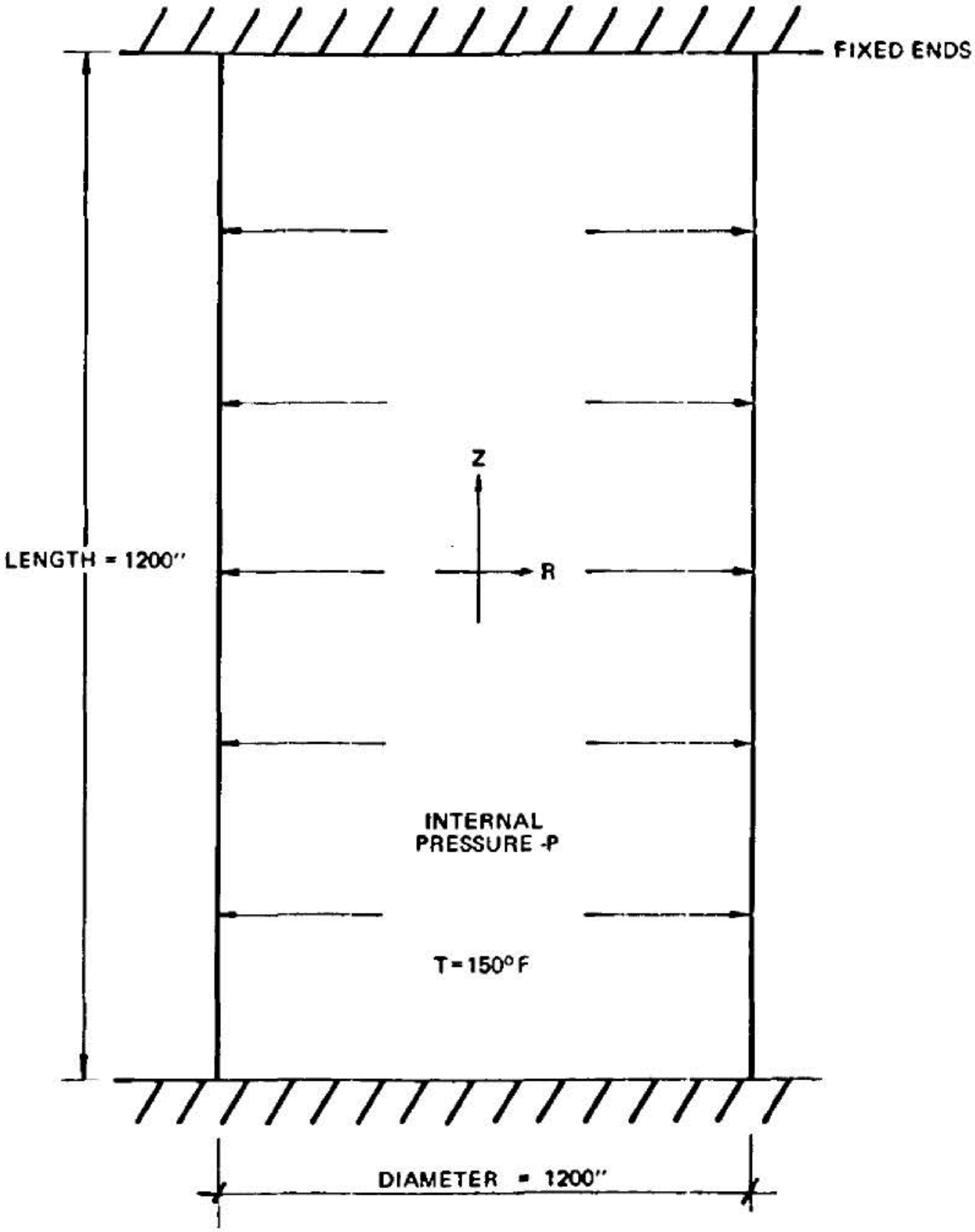
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

LAYERED CYLINDER

FIGURE 3.8A-2, Rev. 47

Auto-Cad Figure Fsar 3_8A_2.dwg



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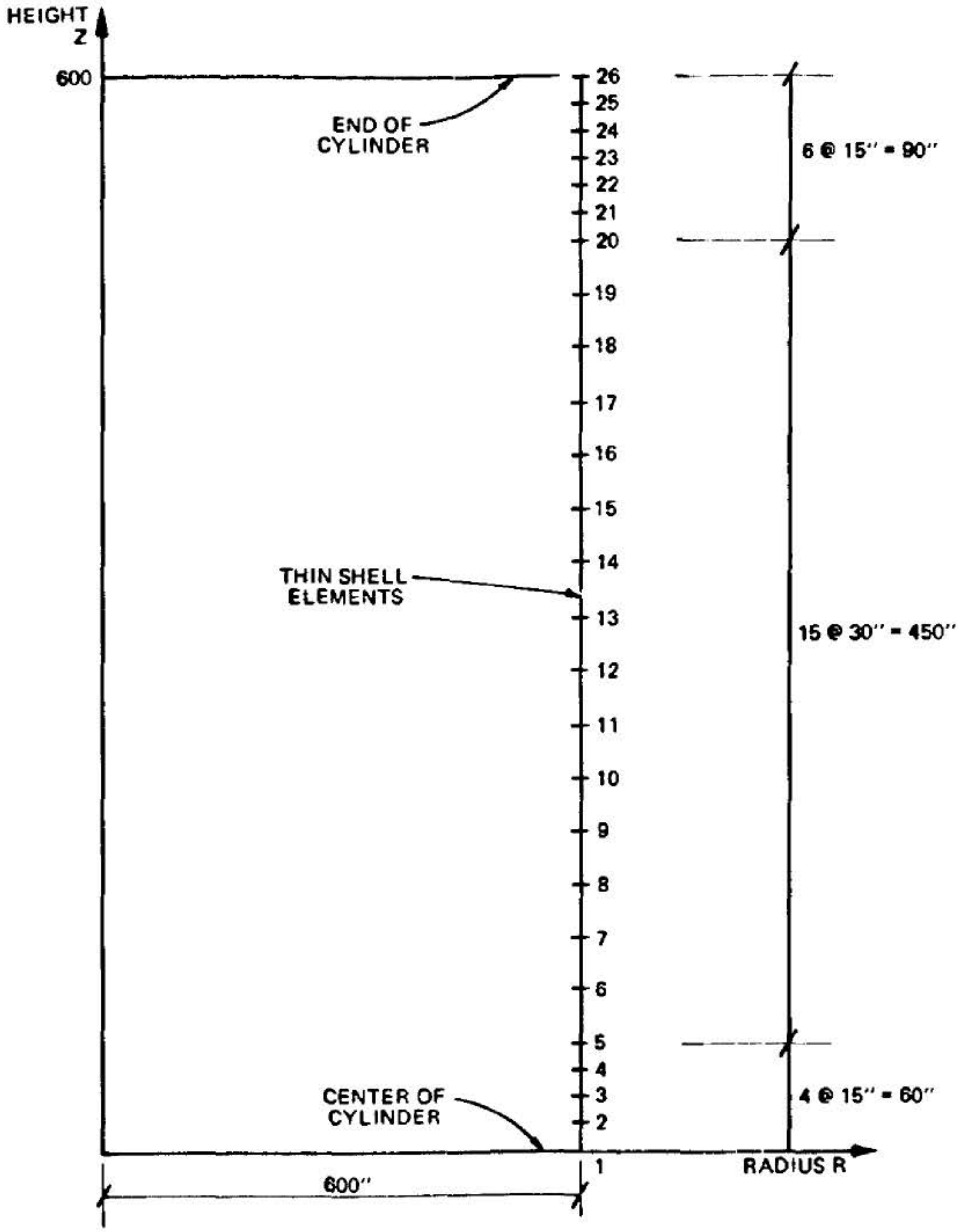
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

GENERAL LAYOUT OF CYLINDER

FIGURE 3.8A-3, Rev. 47

Auto-Cad Figure Fsar 3_8A_3.dwg



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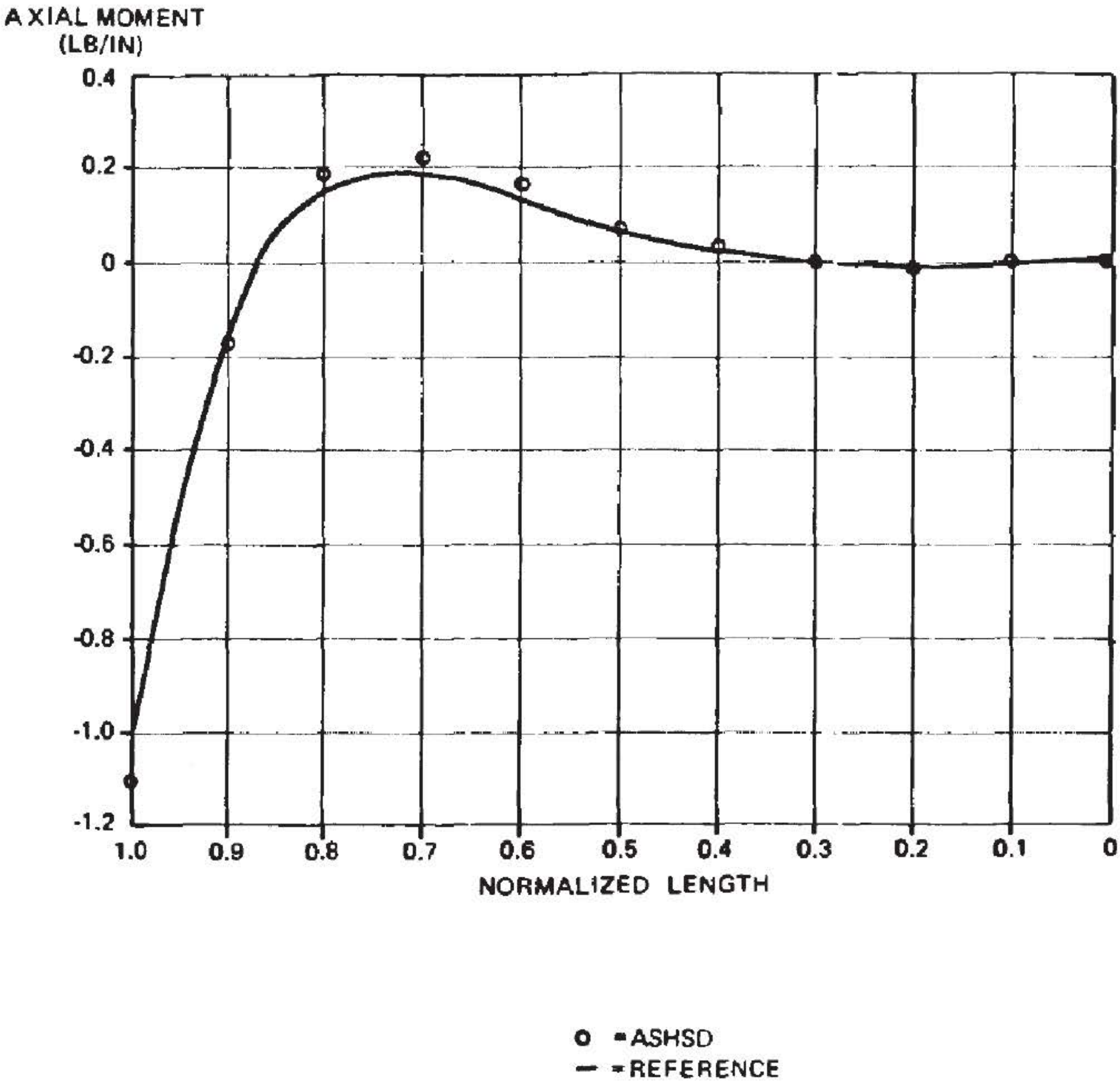
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

FINITE ELEMENT MODEL

FIGURE 3.8A-4, Rev. 47

Auto-Cad Figure Fsar 3_8A_4.dwg



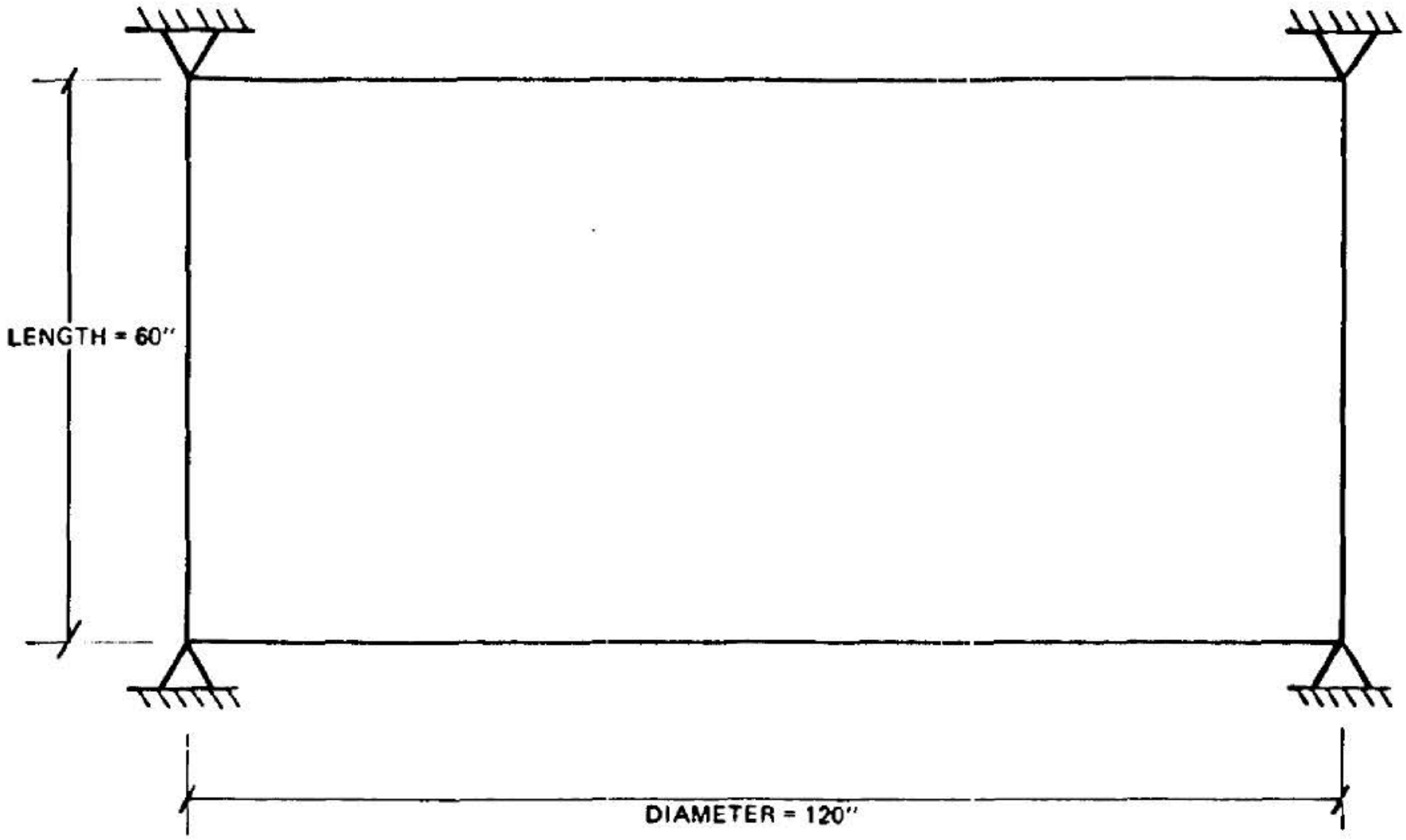
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

AXIAL MOMENT

FIGURE 3.8A-5, Rev. 47

Auto-Cad Figure Fsar 3_8A_5.dwg



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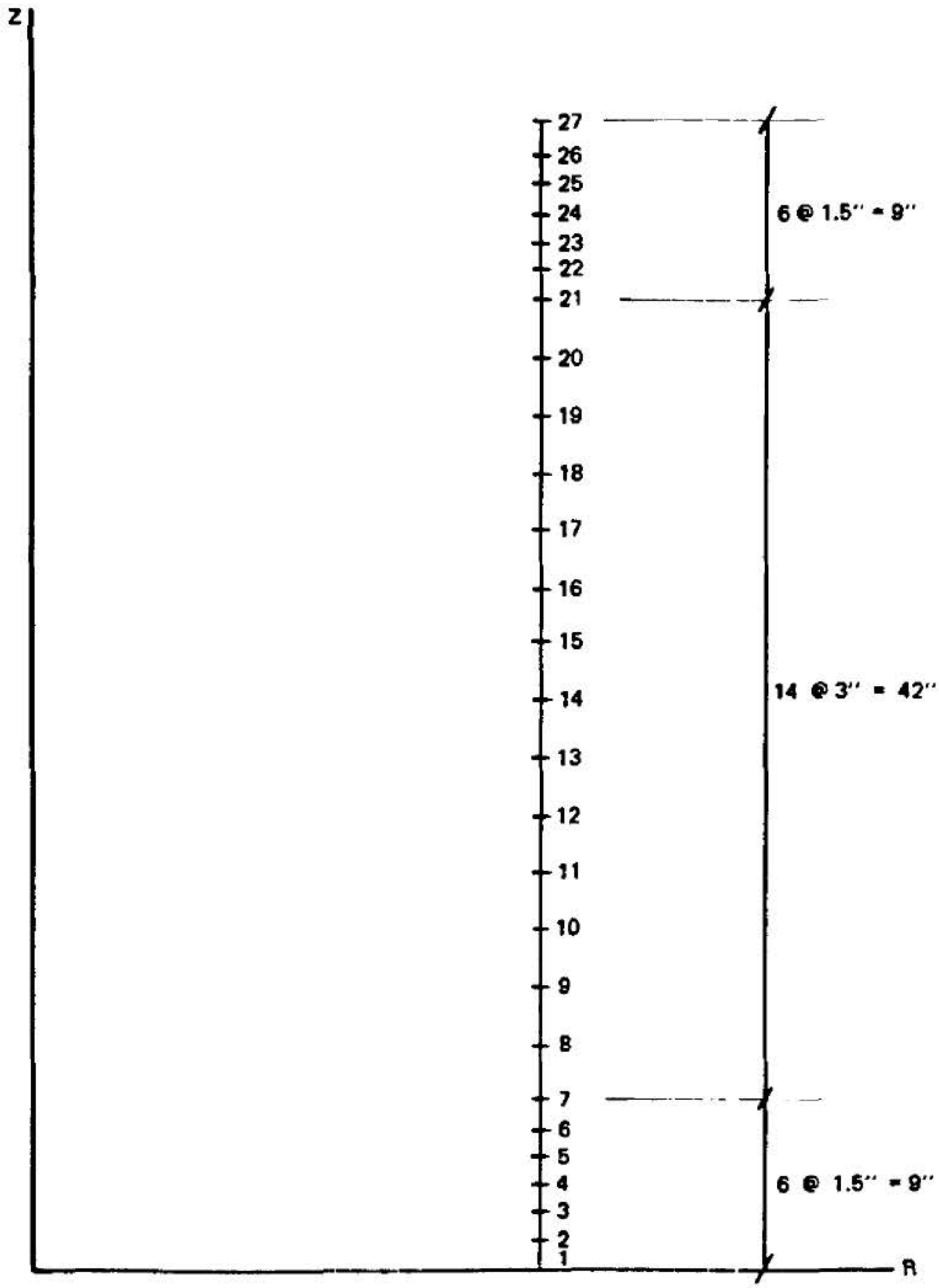
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

CYLINDER WITH HINGE ENDS

FIGURE 3.8A-6, Rev. 47

Auto-Cad Figure Fsar 3_8A_6.dwg



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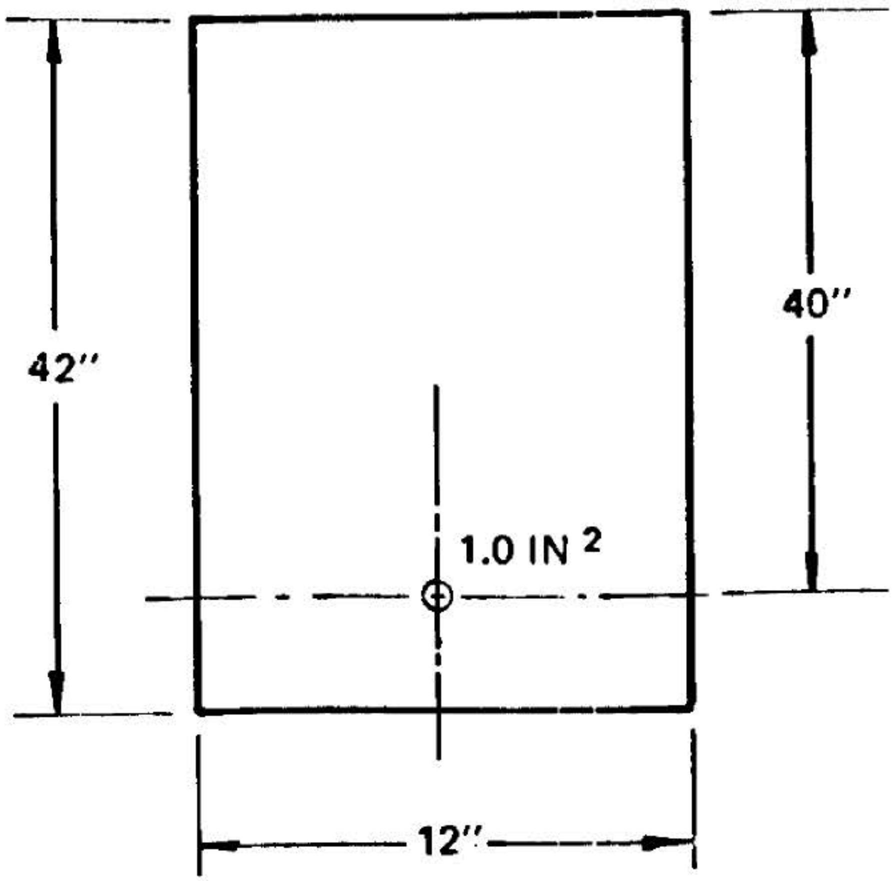
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

FINITE ELEMENT MODEL

FIGURE 3.8A-7, Rev. 47

Auto-Cad Figure Fsar 3_8A_7.dwg



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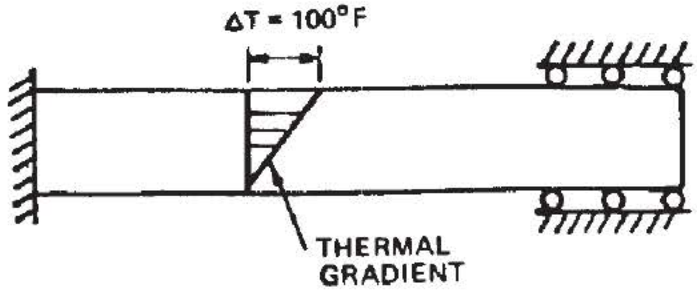
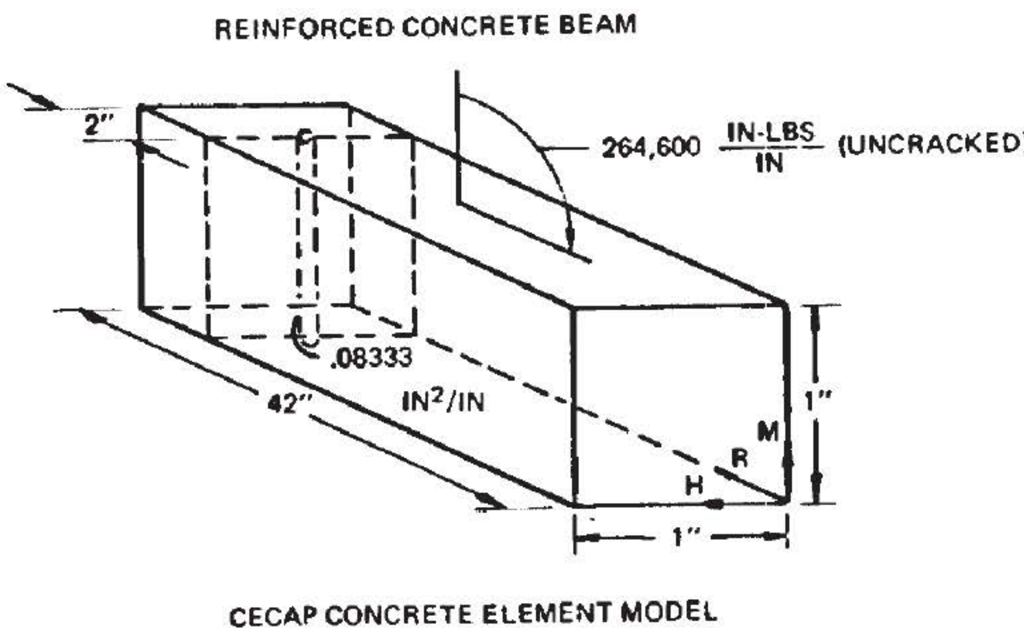
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

REINFORCED CONCRETE BEAM

FIGURE 3.8A-9, Rev. 47

Auto-Cad Figure Fsar 3_8A_9.dwg





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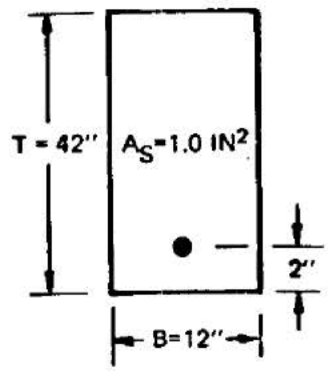
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

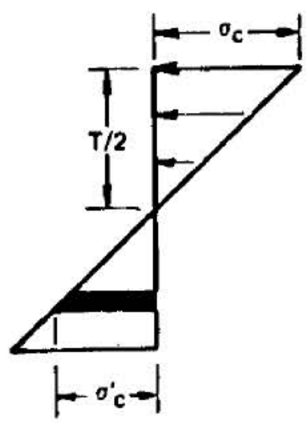
REINFORCED CONCRETE BEAM
AND
CECAP CONCRETE ELEMENT MODEL

FIGURE 3.8A-10, Rev. 47

Auto-Cad Figure Fsar 3_8A_10.dwg



BEAM CROSS - SECTION



INITIAL THERMAL STRESS DISTRIBUTION

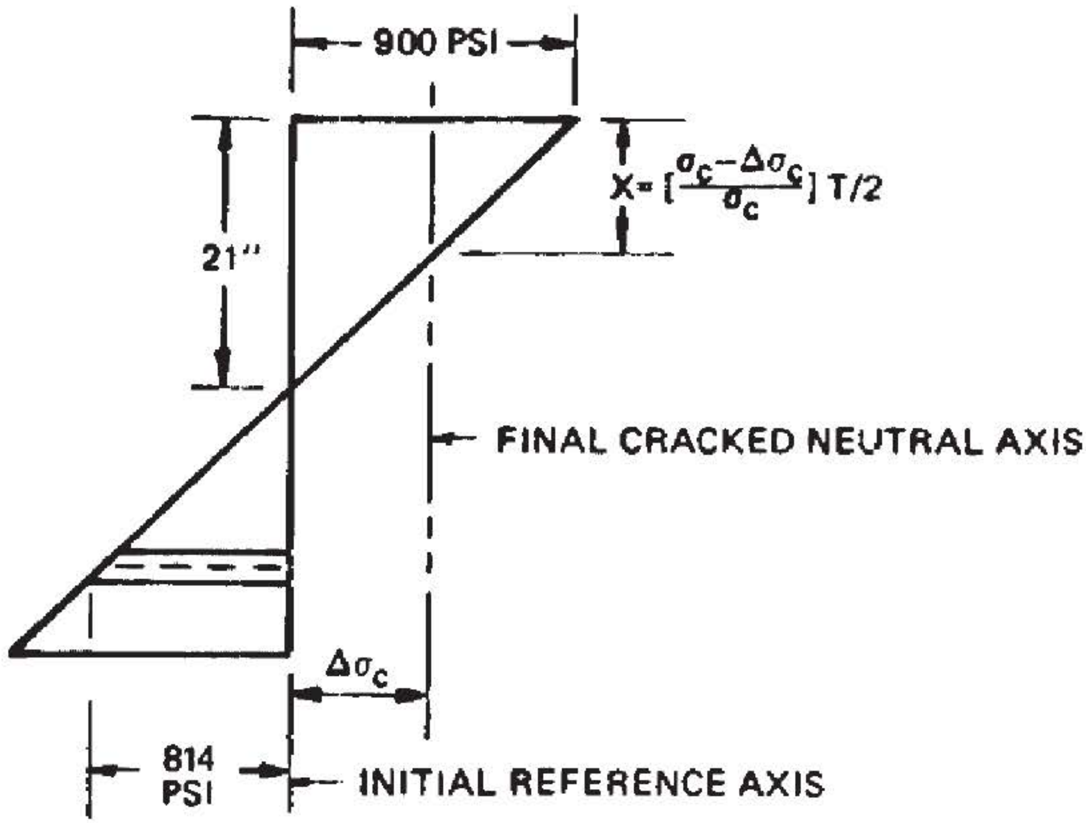
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

BEAM CROSS-SECTION
AND
INITIAL STRESS DISTRIBUTION

FIGURE 3.8A-11, Rev. 47

Auto-Cad Figure Fsar 3_8A_11.dwg



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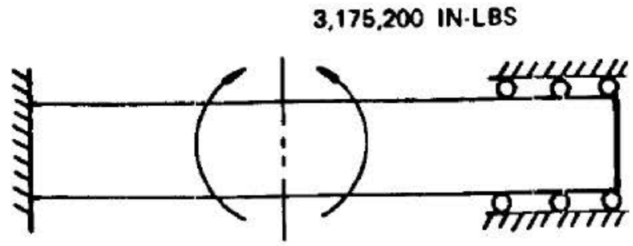
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

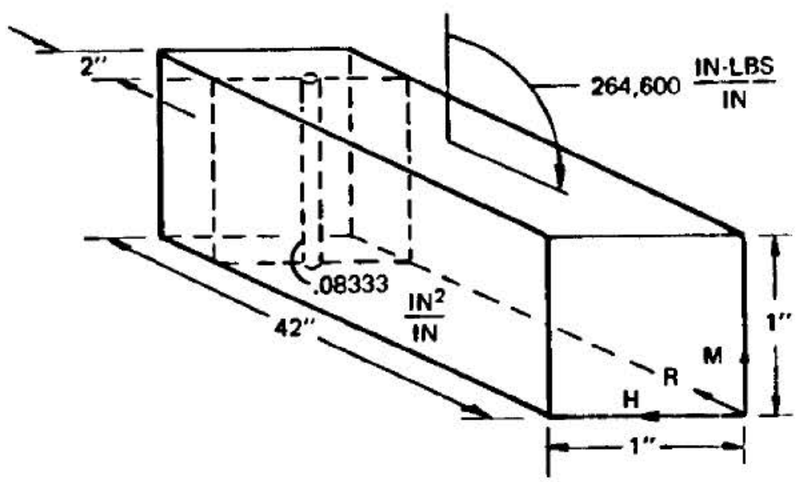
FINAL THERMAL STRESS DISTRIBUTION

FIGURE 3.8A-12, Rev. 47

Auto-Cad Figure Fsar 3_8A_12.dwg



REINFORCED CONCRETE BEAM



CECAP CONCRETE ELEMENT MODEL

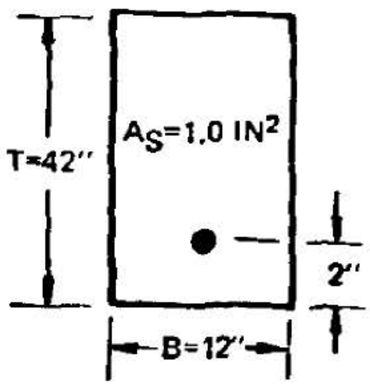
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

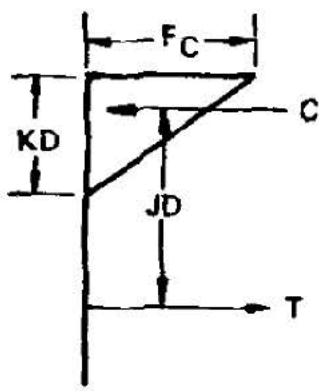
REINFORCED CONCRETE BEAM
AND
CECAP CONCRETE ELEMENT MODEL

FIGURE 3.8A-13, Rev. 47

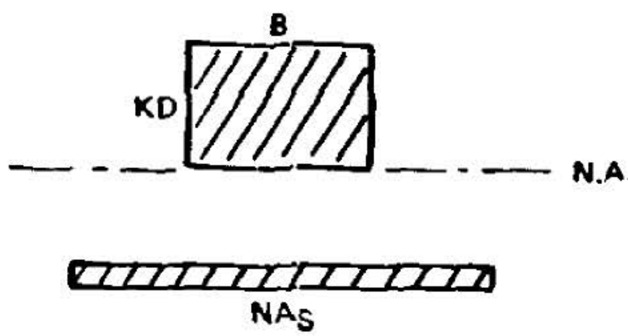
Auto-Cad Figure Fsar 3_8A_13.dwg



BEAM CROSS - SECTION



STRESS BLOCK



TRANSFORMED SECTION

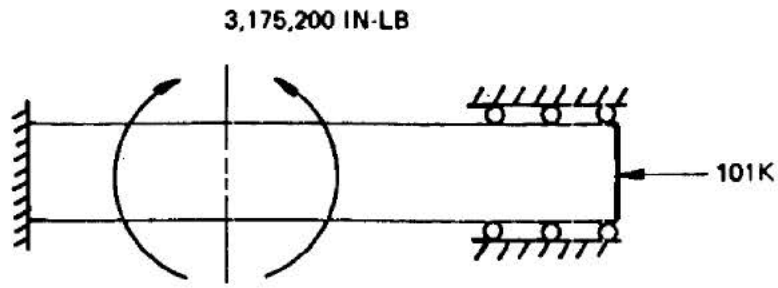
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

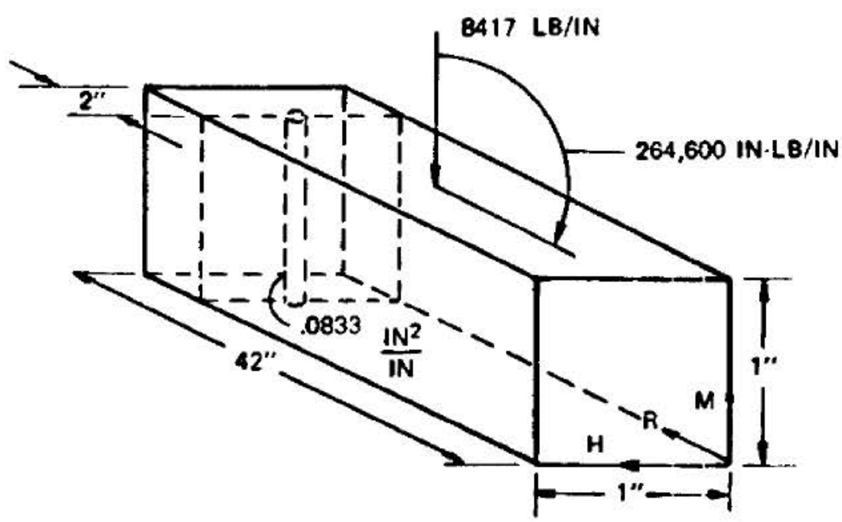
REINFORCED CONCRETE BEAM
AND
CECAP CONCRETE ELEMENT MODEL

FIGURE 3.8A-14, Rev. 47

Auto-Cad Figure Fsar 3_8A_14.dwg



REINFORCED CONCRETE BEAM



CECAP CONCRETE ELEMENT MODEL

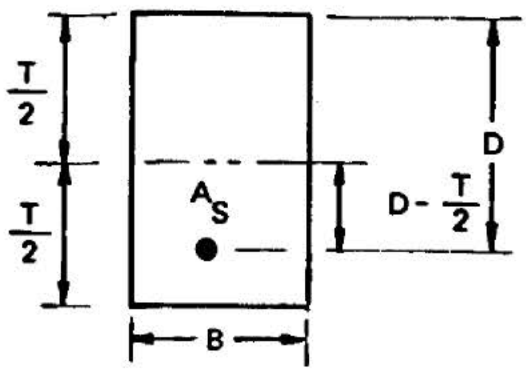
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

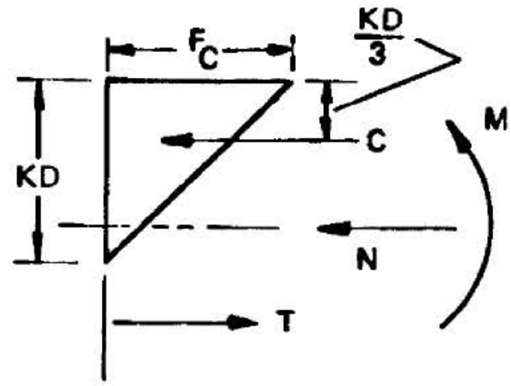
REINFORCED CONCRETE BEAM
AND CECAP MODEL

FIGURE 3.8A-15, Rev. 47

Auto-Cad Figure Fsar 3_8A-15.dwg



BEAM CROSS-SECTION



STRESS BLOCK

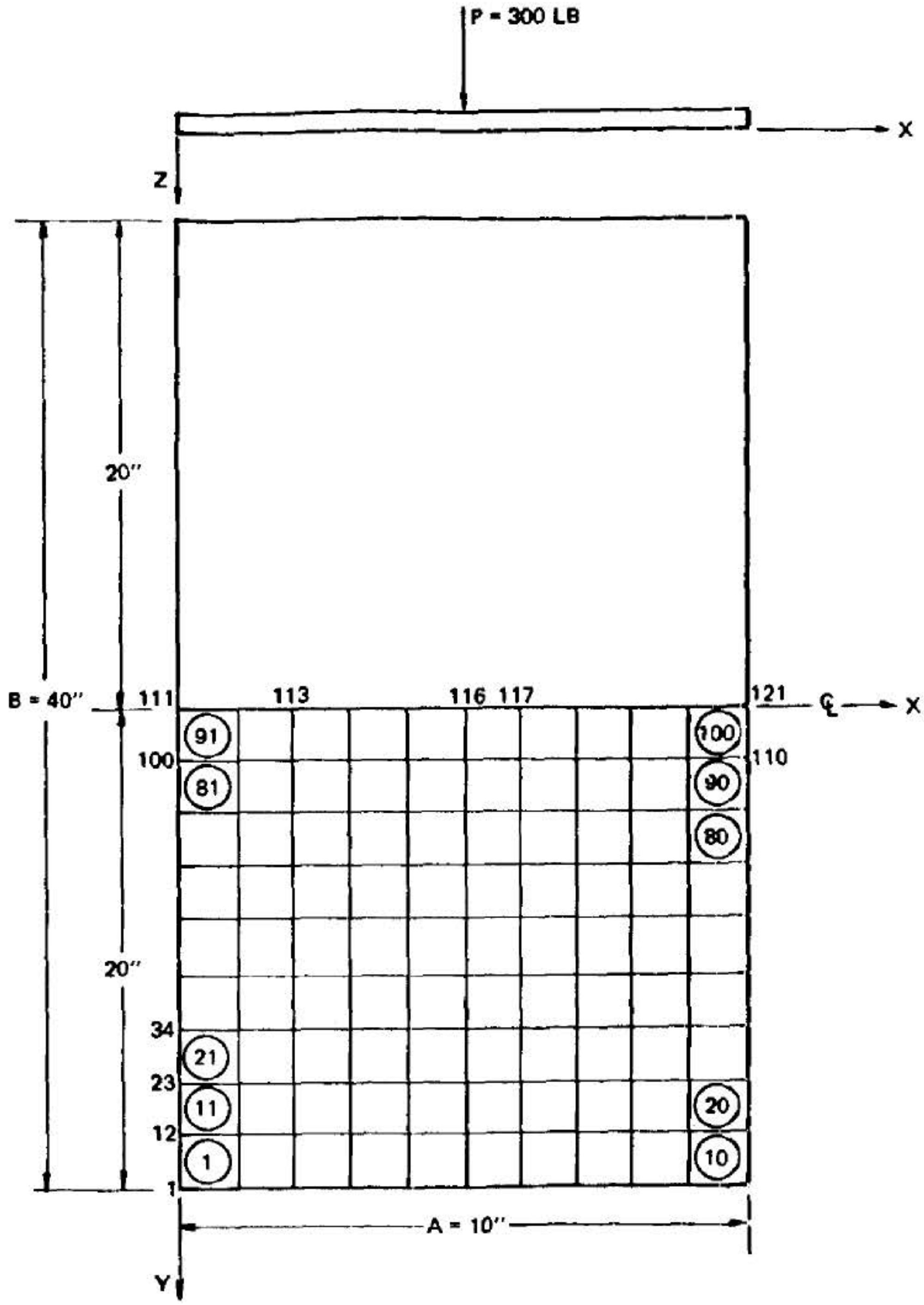
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

BEAM CROSS-SECTION
AND
STRESS BLOCK

FIGURE 3.8A-16, Rev. 47

Auto-Cad Figure Fsar 3_8A_16.dwg



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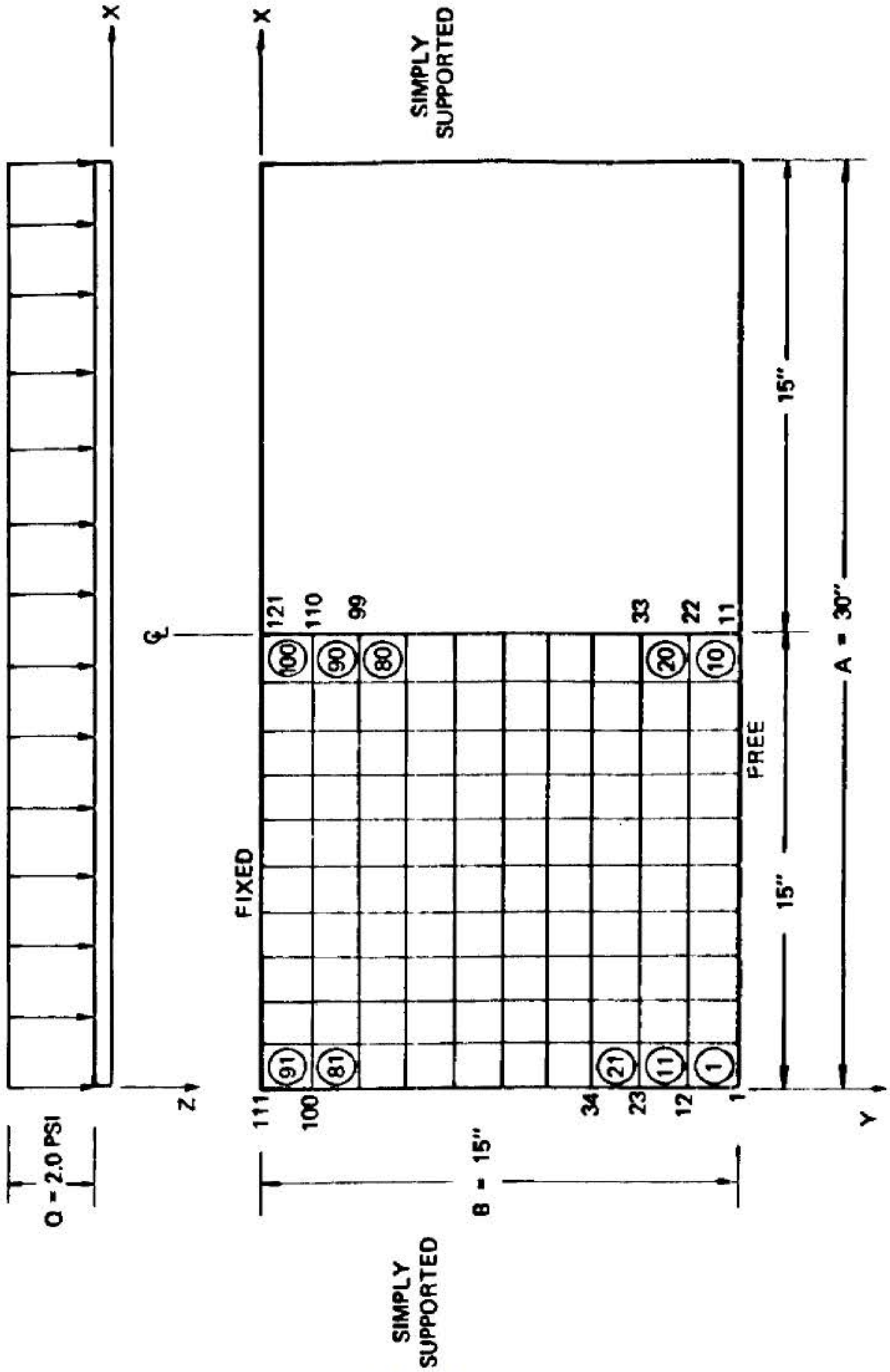
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

PLATE GEOMETRY, LOADING AND
FINITE ELEMENT MESH FOR
RECTANGULAR PLATE WITH A
CONCENTRATED LOAD AT THE CENTER

FIGURE 3.8A-17, Rev. 47

Auto-Cad Figure Fsar 3_8A-17.dwg



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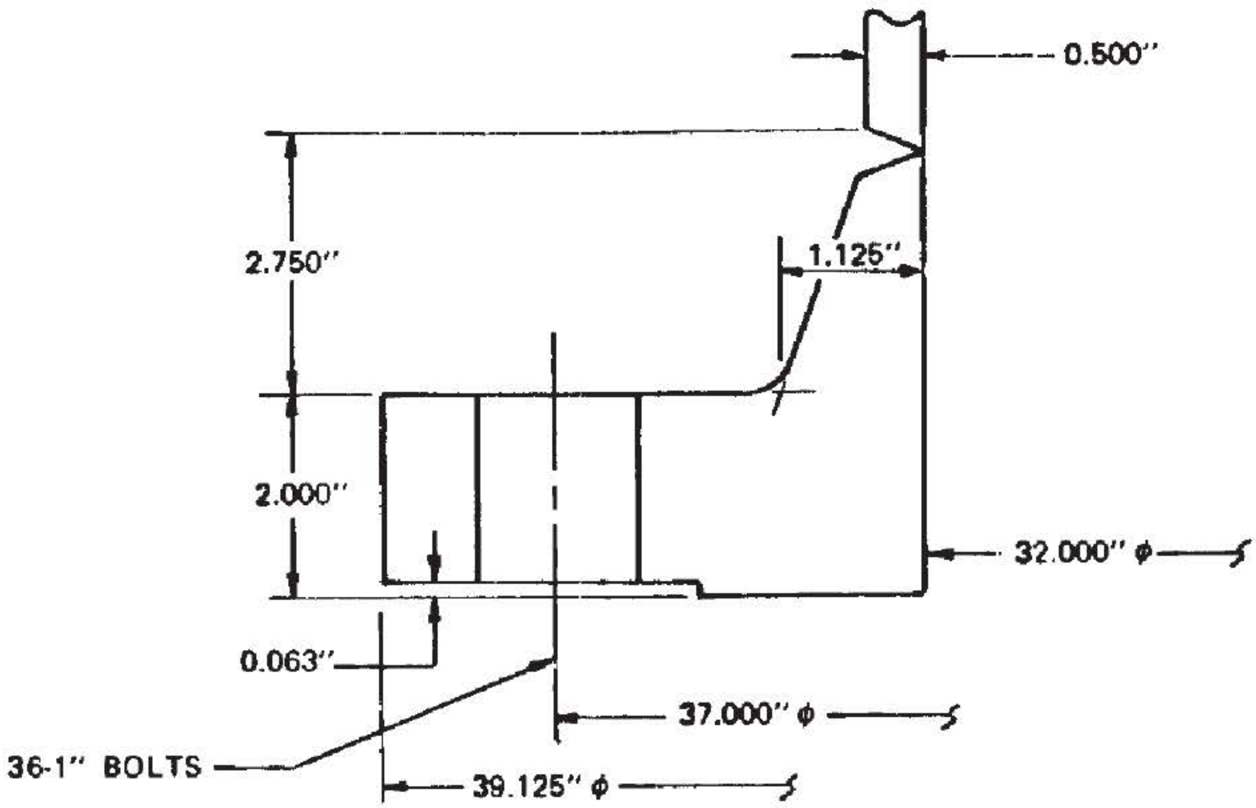
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

PLATE GEOMETRY, LOADING AND
FINITE ELEMENT MESH FOR
RECTANGULAR PLATE WITH
VARIOUS EDGE CONDITIONS

FIGURE 3.8A-18, Rev. 47

Auto-Cad Figure Fsar 3_8A-18.dwg



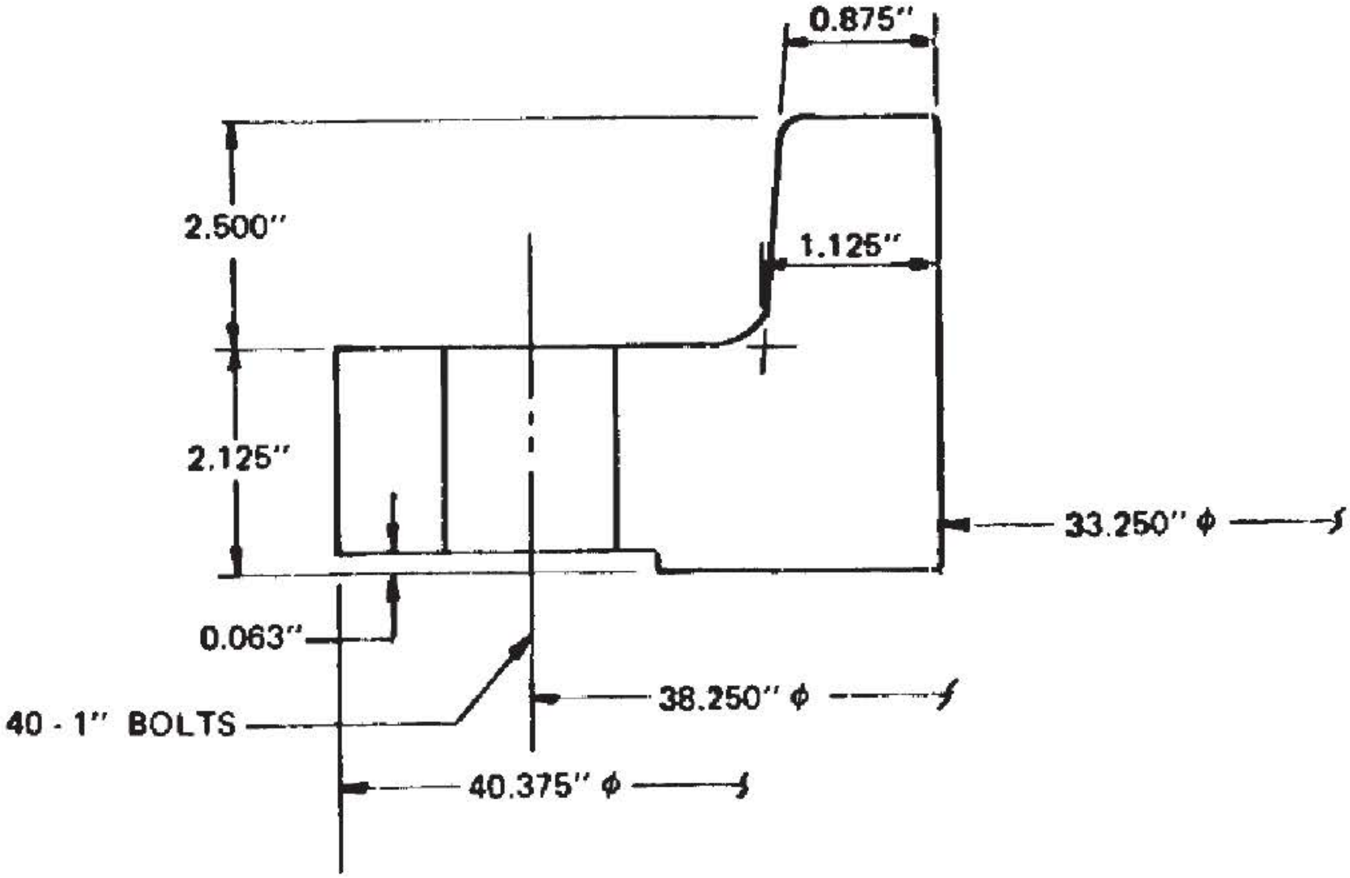
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

WELD NECK FLANGE DETAIL

FIGURE 3.8A-19, Rev. 47

Auto-Cad Figure Fsar 3_8A-19.dwg



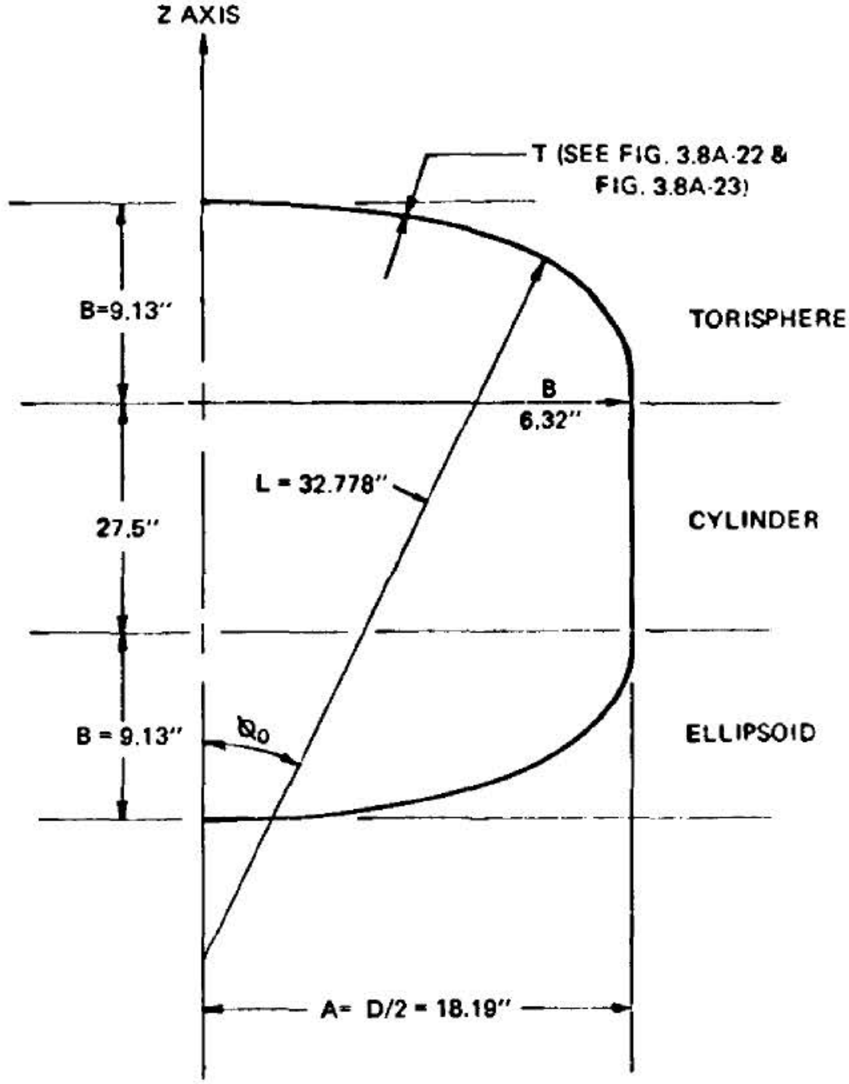
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

SLIP-ON FLANGE DETAIL

FIGURE 3.8A-20, Rev. 47

Auto-Cad Figure Fsar 3_8A_20.dwg



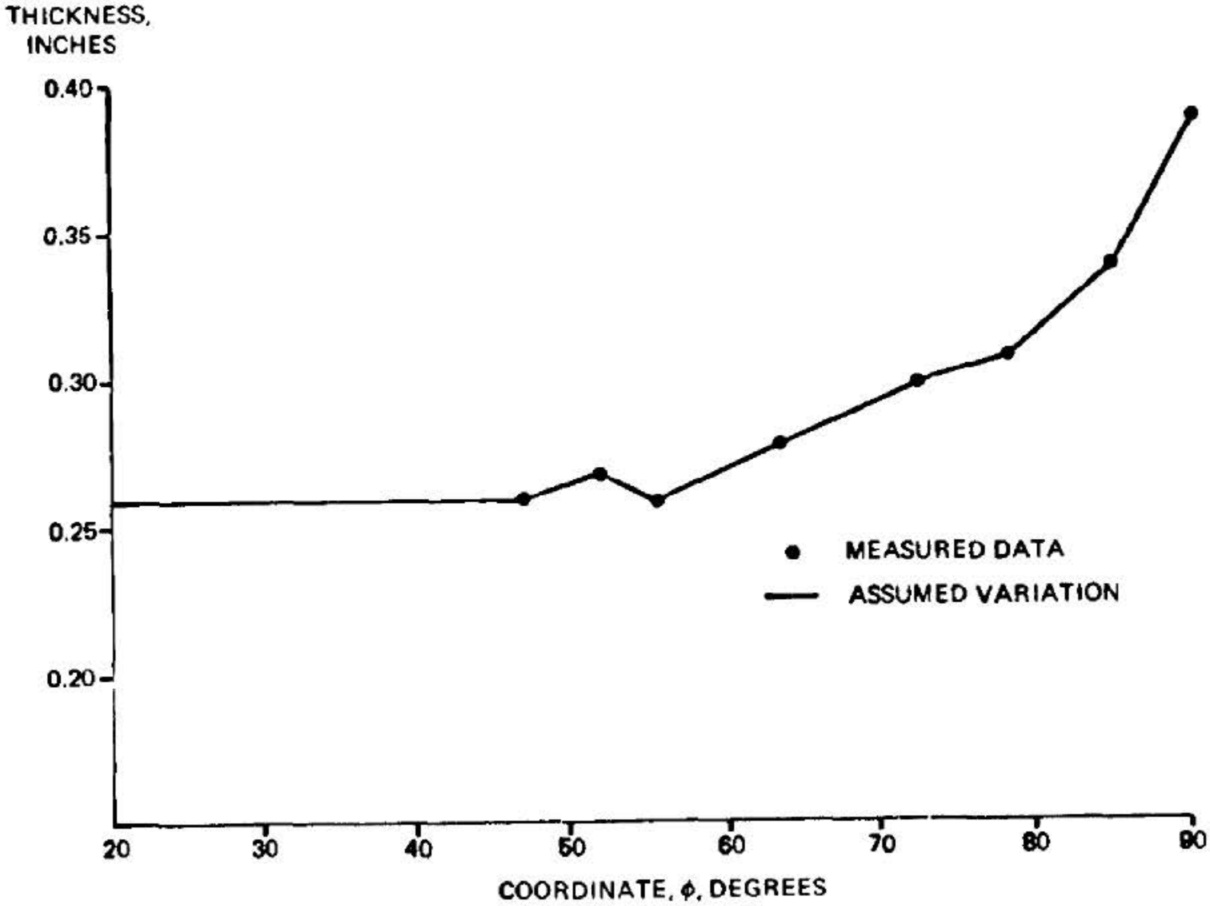
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

GEOMETRY OF TORISPHERICAL
AND ELLIPSOIDAL HEADS
(FIT TO CORRESPOND TO
GEOMETRY OF ELLIPSOIDAL HEAD
IN REFERENCE 3.8A-9)

FIGURE 3.8A-21, Rev. 47

Auto-Cad Figure Fsar 3_8A_21.dwg



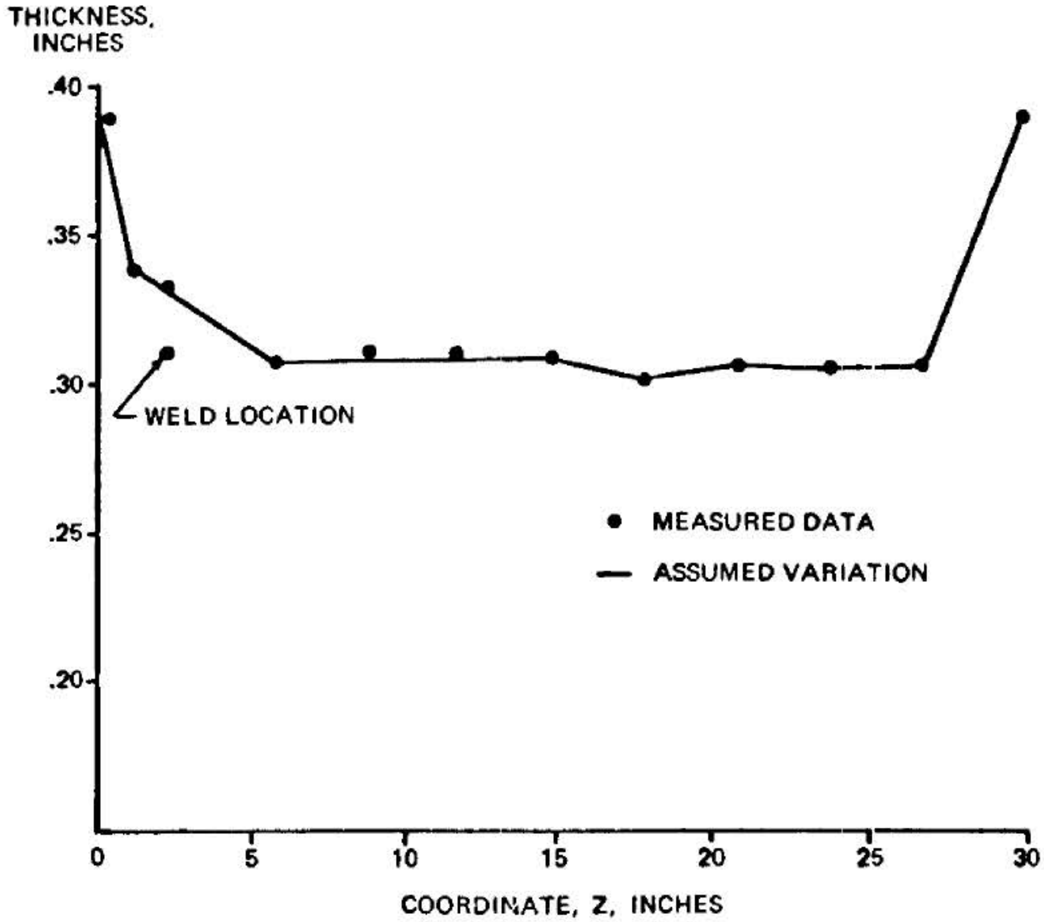
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

MEASURED THICKNESS VARIATION
IN EXPERIMENTAL HEAD NO. 1
(FROM REF. 3.8A-10 PAGE 18)

FIGURE 3.8A-22, Rev. 47

Auto-Cad Figure Fsar 3_8A_22.dwg



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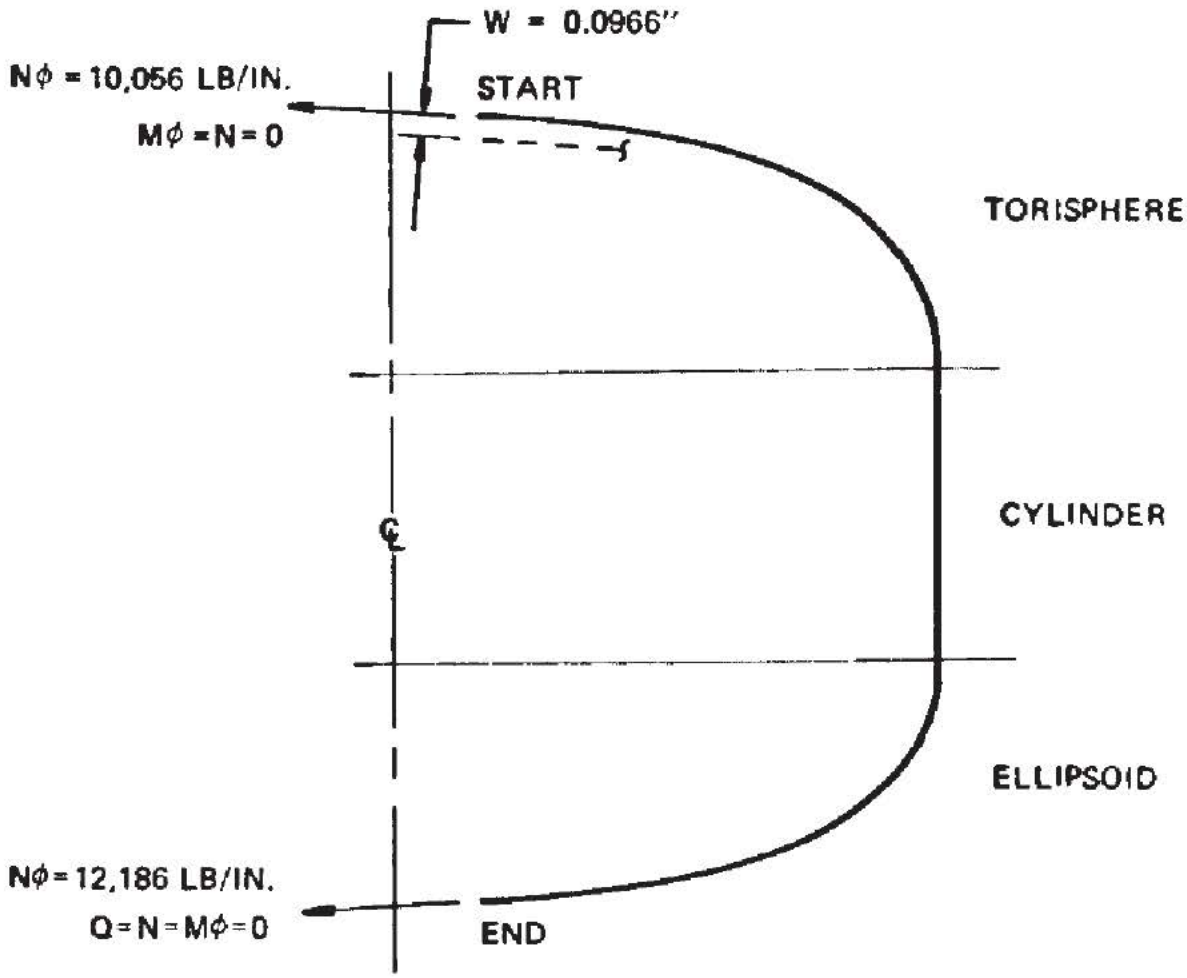
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

THICKNESS VARIATION
IN CYLINDER NO. 1
(FROM REF. 3.8A-11 FIG. 4)

FIGURE 3.8A-23, Rev. 47

Auto-Cad Figure Fsar 3_8A_23.dwg



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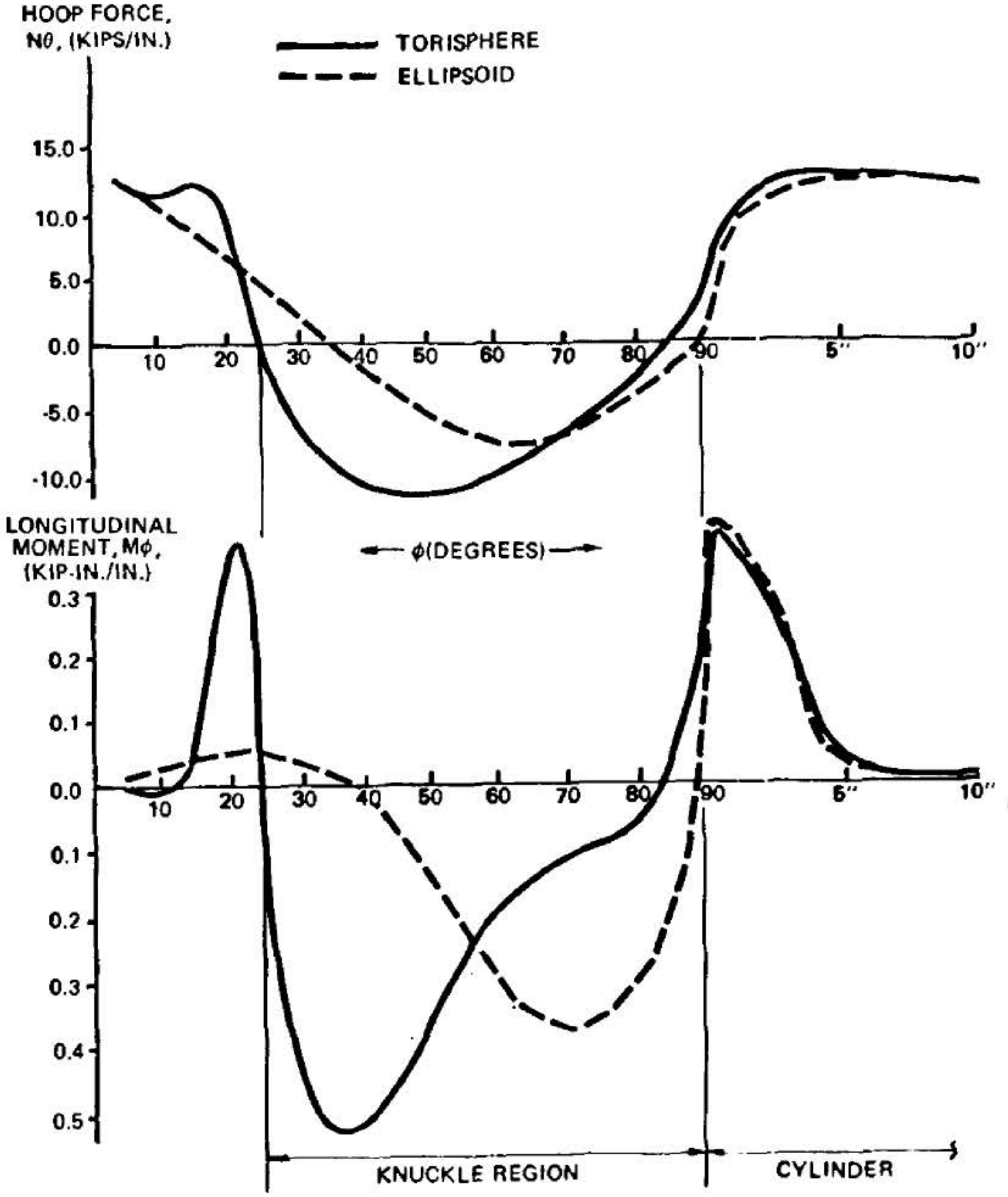
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

ANALYTICAL MODEL
WITH
BOUNDARY CONDITIONS

FIGURE 3.8A-24, Rev. 47

Auto-Cad Figure Fsar 3_8A_24.dwg



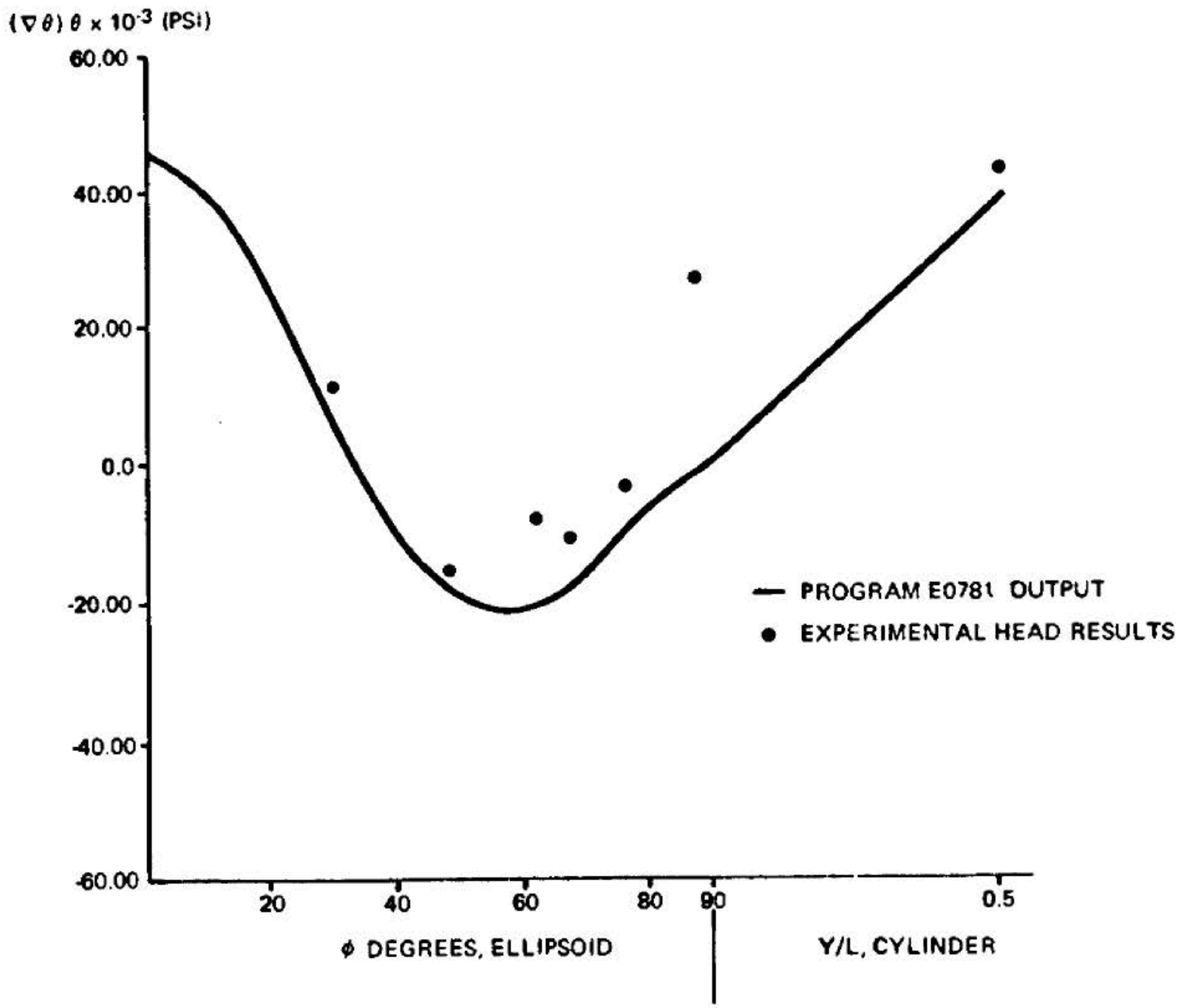
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

PLOT OF HOOP FORCE AND
LONGITUDINAL MOVEMENT
FROM E0781 OUTPUT

FIGURE 3.8A-25, Rev. 47

Auto-Cad Figure Fsar 3_8A_25.dwg



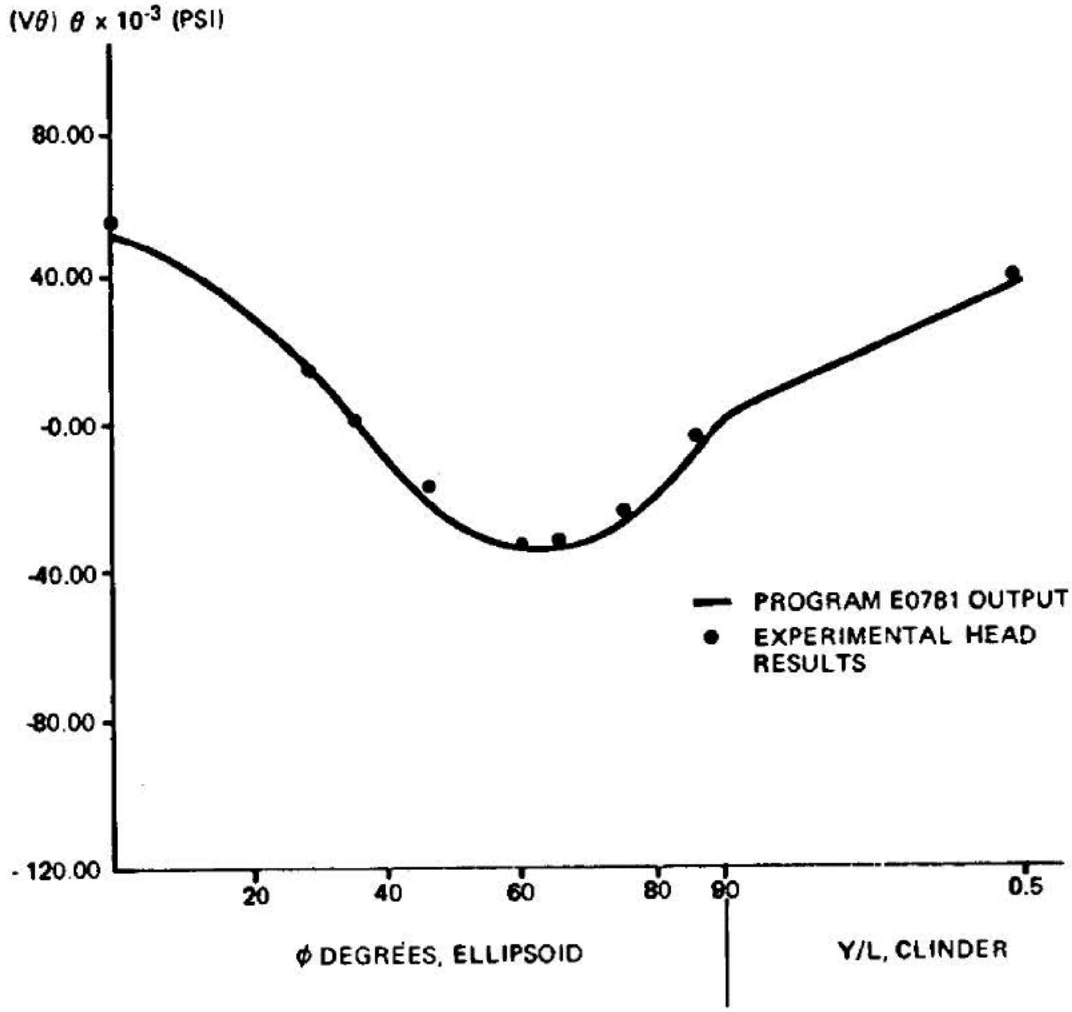
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

PLOT OF STRESS IN THE Θ (HOOP)
DIRECTION ON THE INSIDE
SURFACE (Θ = 0°)
(REF. 3.8A-12 PAGE G-14)

FIGURE 3.8A-26, Rev. 47

Auto-Cad Figure Fsar 3_8A_26.dwg



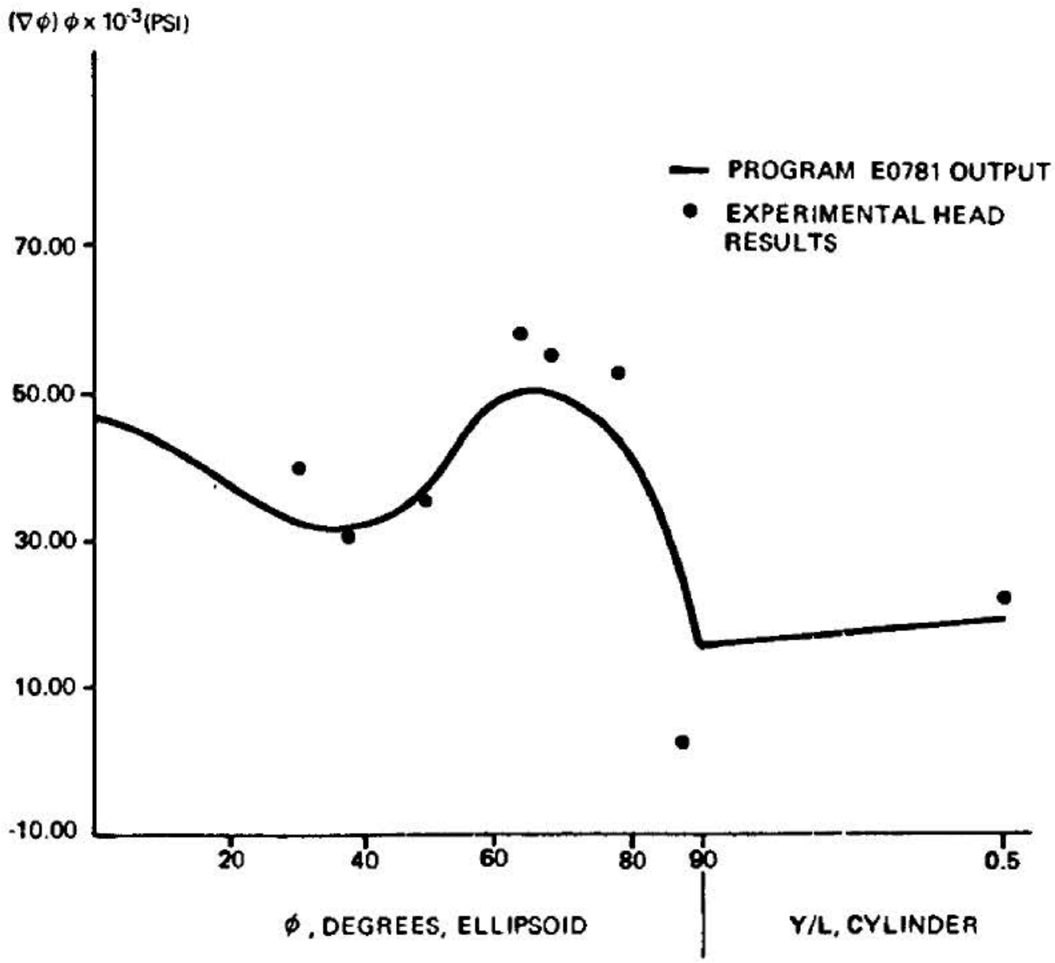
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

PLOT OF STRESS IN THE Θ (HOOP)
DIRECTION ON THE OUTSIDE
SURFACE (Θ = 0°)
(REF. 3.8A-12 PAGE G-11)

FIGURE 3.8A-27, Rev. 47

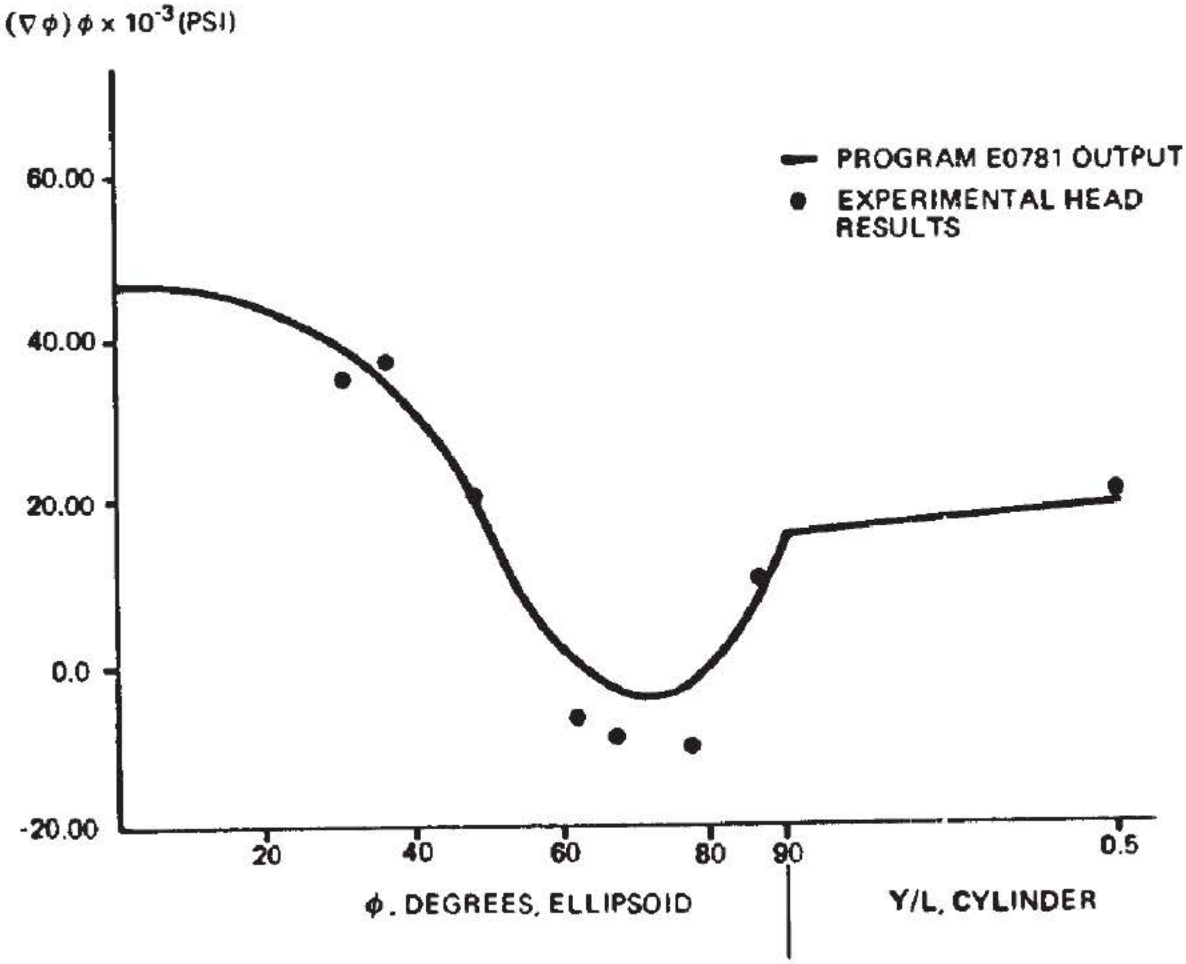
Auto-Cad Figure Fsar 3_8A_27.dwg



FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
PLOT OF STRESS IN THE ϕ (LONGITUDINAL)DIRECTION ON THE INSIDE SURFACE ($\Theta = 0°$) (REF. 3.8A-12 PAGE G-20) FIGURE 3.8A-28, Rev. 47

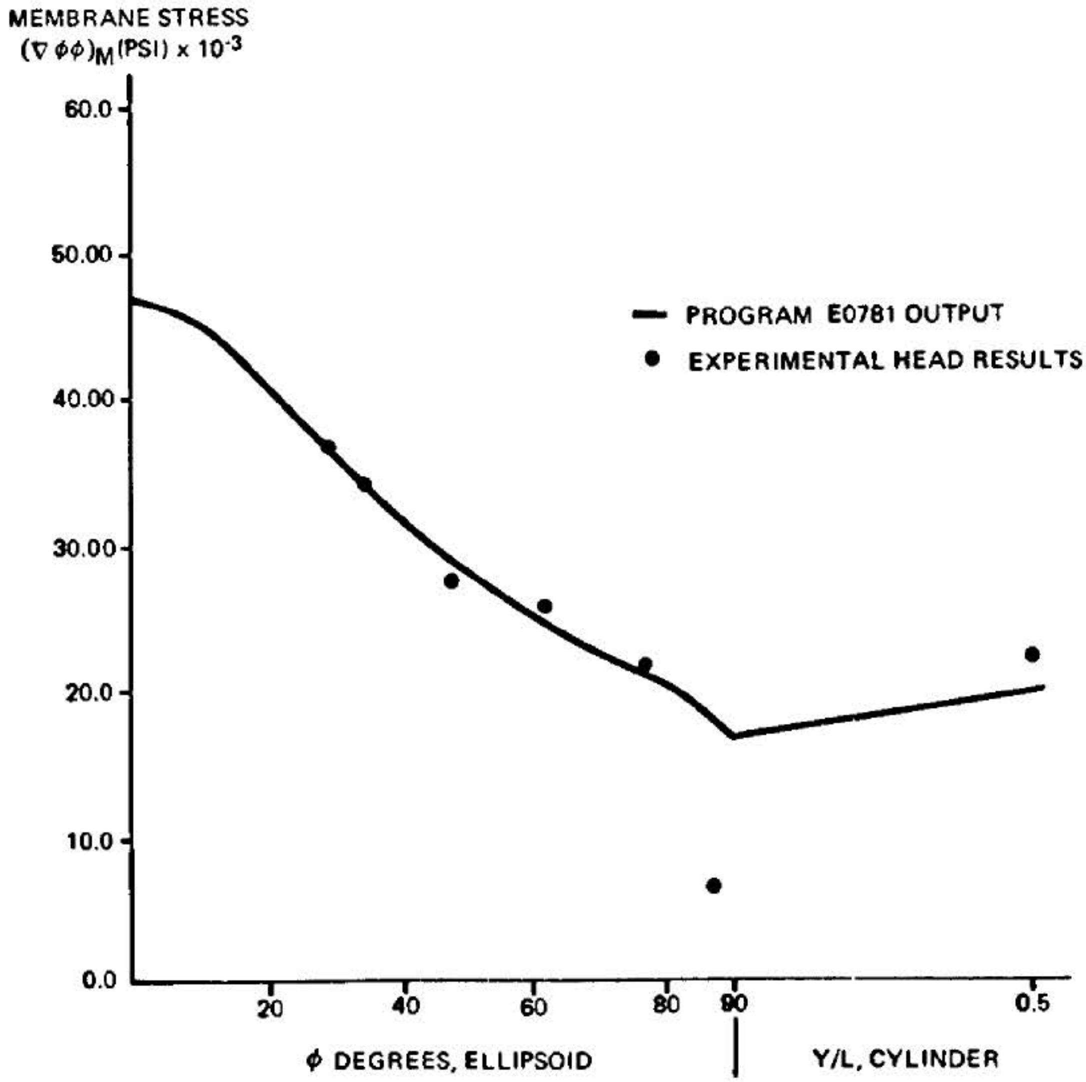
Auto-Cad Figure Fsar 3_8A_28.dwg



FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
PLOT OF STRESS IN THE ϕ (LONGITUDINAL) DIRECTION ON THE OUTSIDE SURFACE (ϕ = 0°) (REF. 3.8A-12 PAGE G-17) FIGURE 3.8A-29, Rev. 47

Auto-Cad Figure Fsar 3_8A_29.dwg



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SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
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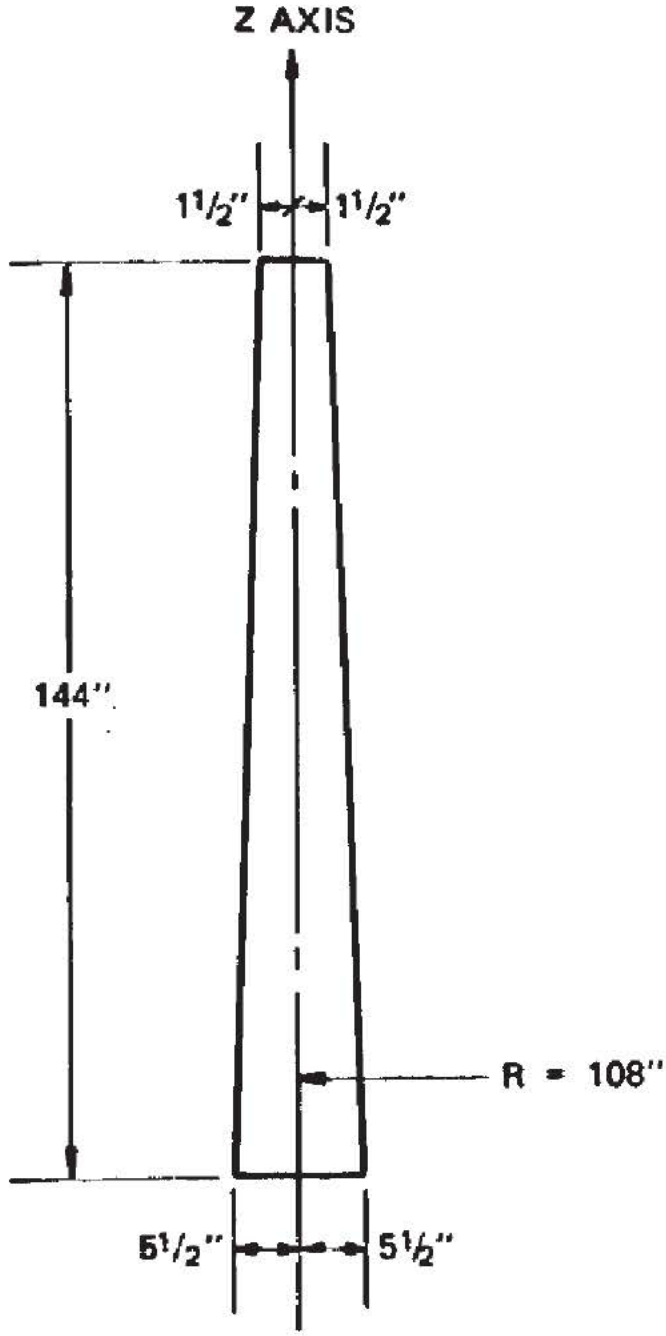
PLOT OF
MEMBRANE STRESS (Θ = 0°)
(REF. 3.8A-12 PAGE G-23)

FIGURE 3.8A-30, Rev. 47

Auto-Cad Figure Fsar 3_8A_30.dwg



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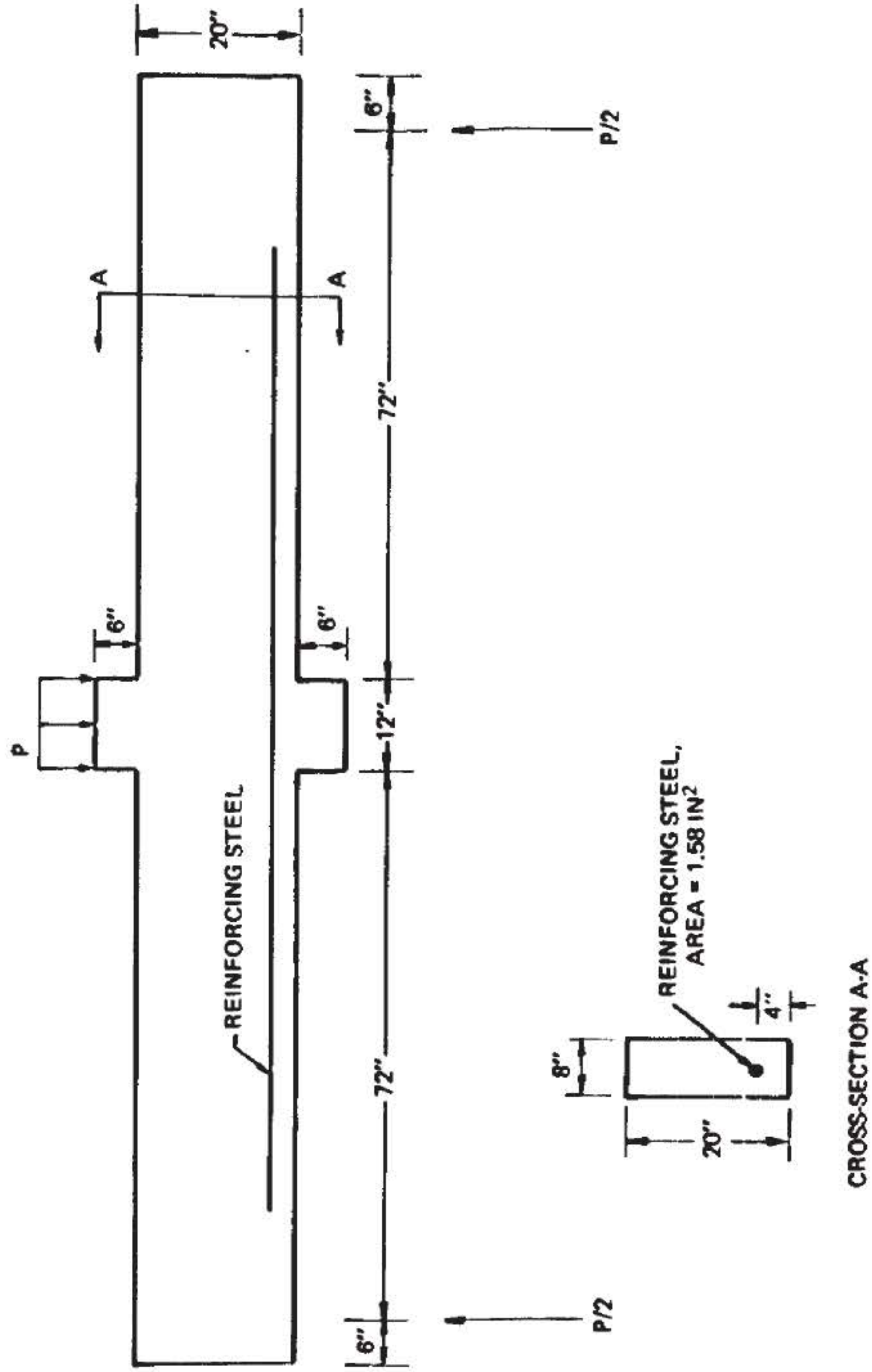
DIMENSIONS OF CYLINDRICAL WATER TANK

FIGURE 3.8A-31, Rev. 47

Auto-Cad Figure Fsar 3_8A_31.dwg



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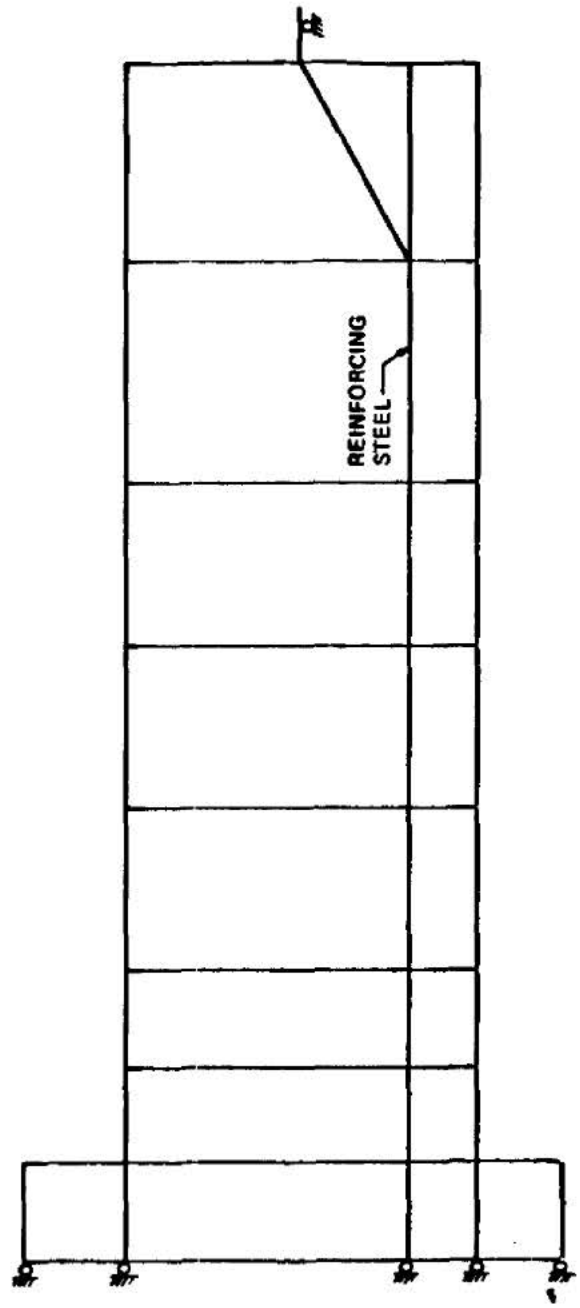
EXPERIMENTAL BEAM DIMENSIONS

FIGURE 3.8A-32, Rev. 47

Auto-Cad Figure Fsar 3_8A_32.dwg



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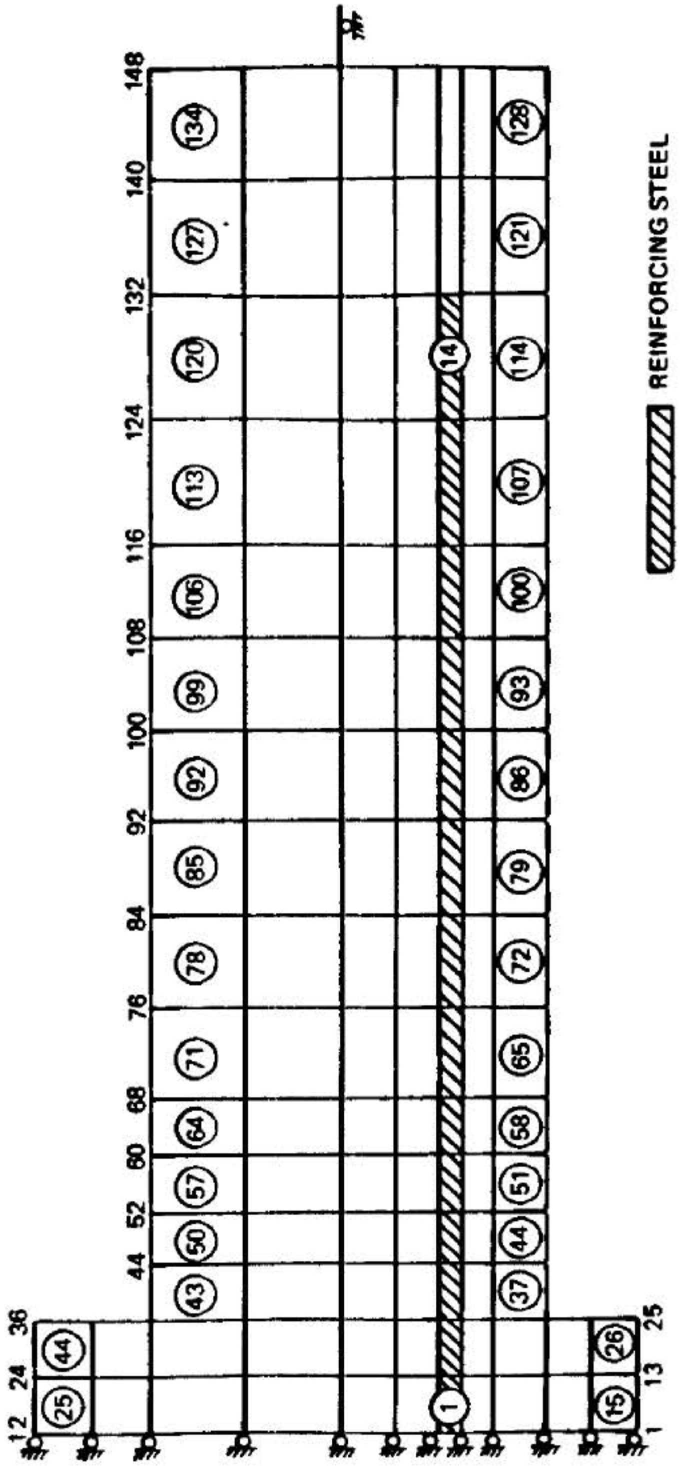
REFERENCE 3.8A-14 MESH

FIGURE 3.8A-33, Rev. 47

Auto-Cad Figure Fsar 3_8A_33.dwg



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FINEL FINITE ELEMENT MESH

FIGURE 3.8A-34, Rev. 47

Auto-Cad Figure Fsar 3_8A_34.dwg

REGIONS OF CRACKING
SEE FIG. 3.8A-35-1 & 3.8A-35-2

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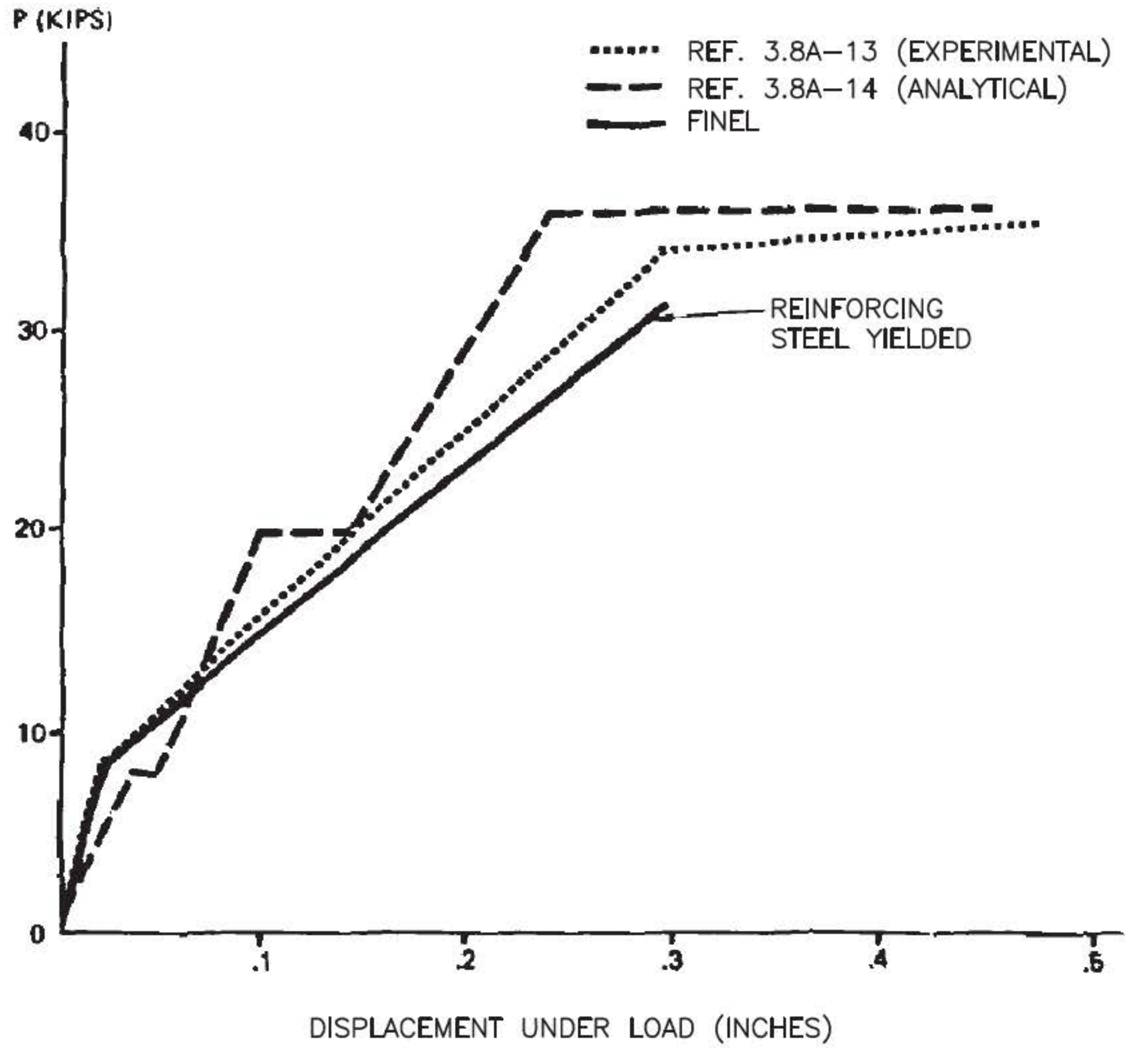
REGIONS OF CRACKING
SEE FIG. 3.8A-35-1 & 3.8A-35-2

FIGURE 3.8A-35, Rev. 36

Auto-Cad Figure Fsar 3_8A_35.dwg



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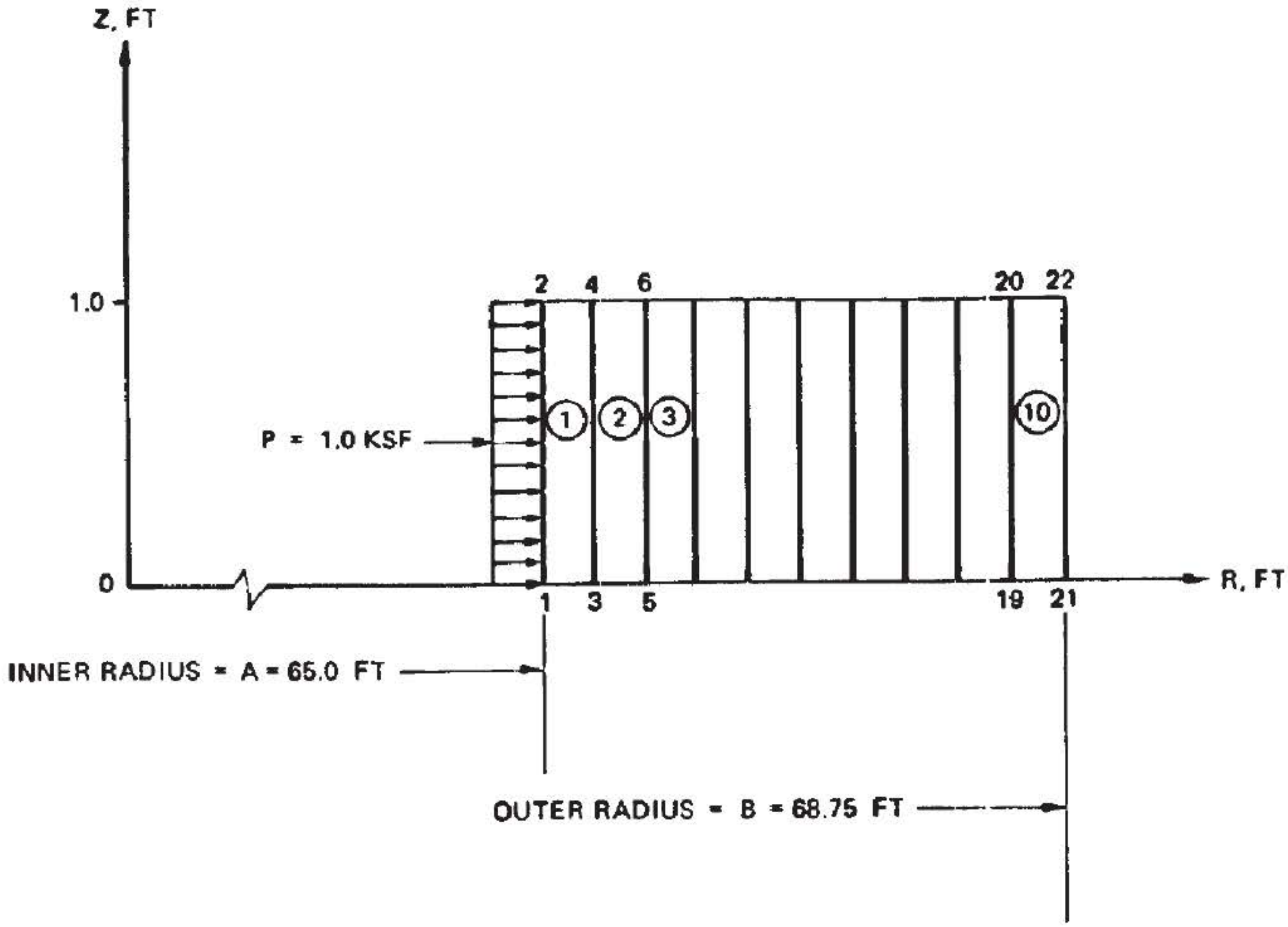
LOAD DISPLACEMENT CURVES
FROM FINEL VERIFICATION USING
A SIMPLY BEAM

FIGURE 3.8A-36, Rev. 47

Auto-Cad Figure Fsar 3_8A_36.dwg



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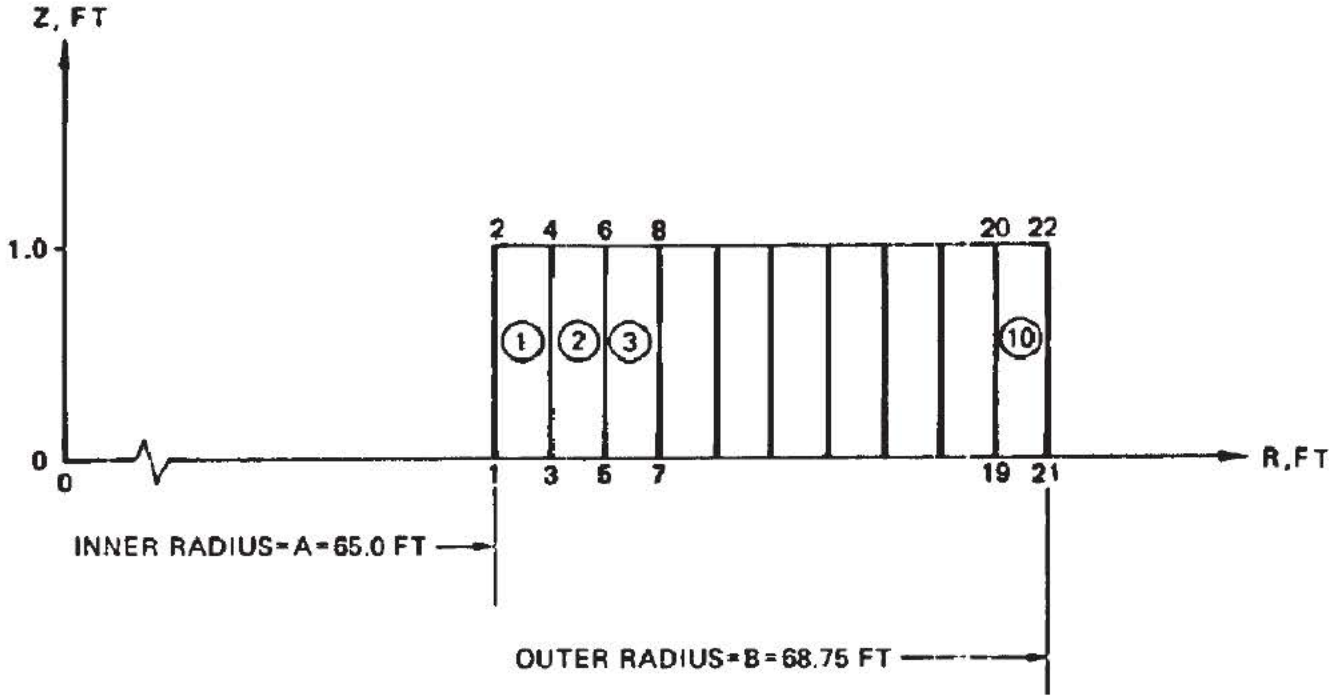
FINITE ELEMENT MODEL

FIGURE 3.8A-37, Rev. 47

Auto-Cad Figure Fsar 3_8A_37.dwg



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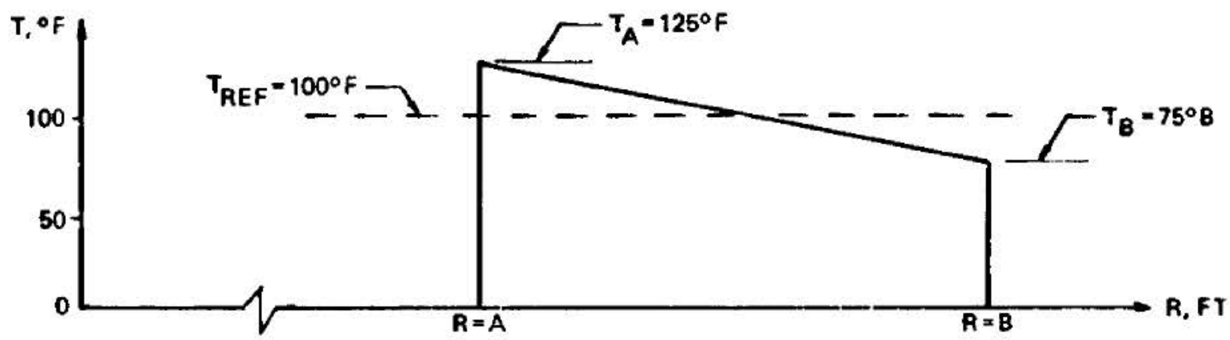
FINITE ELEMENT MODEL

FIGURE 3.8A-38, Rev. 47

Auto-Cad Figure Fsar 3_8A_38.dwg



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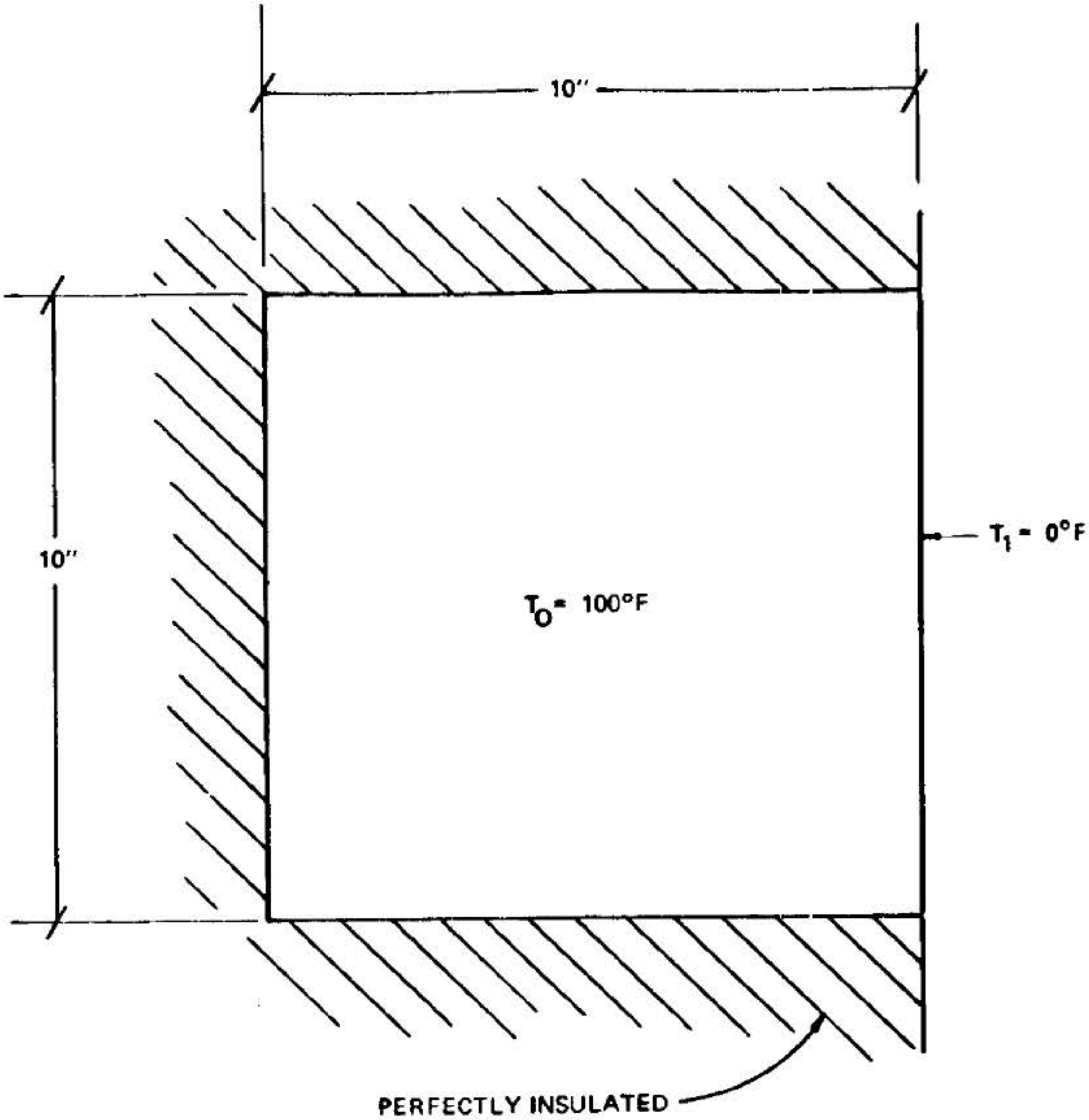
TEMPERATURE DISTRIBUTION

FIGURE 3.8A-39, Rev. 47

Auto-Cad Figure Fsar 3_8A_39.dwg



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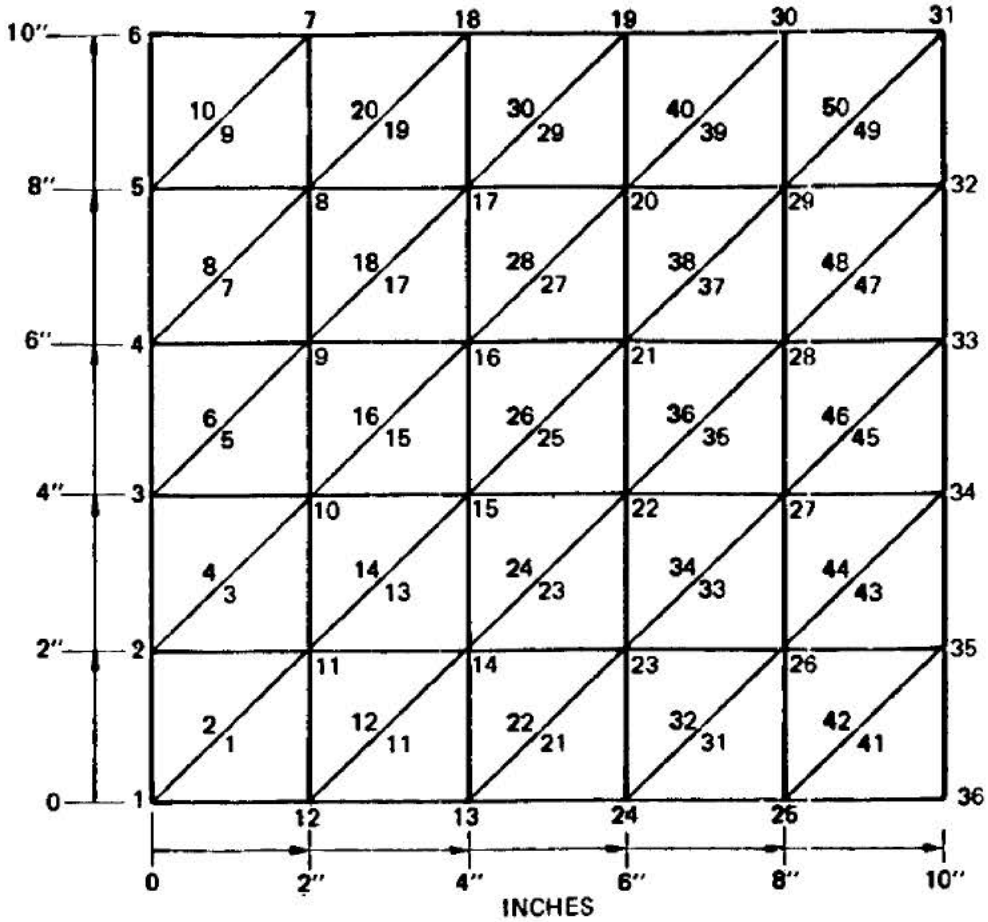
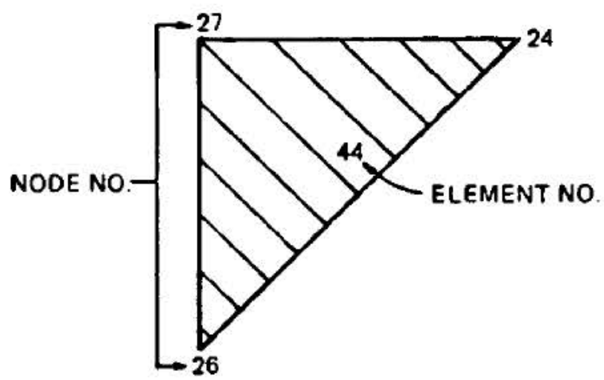
SCHEMATIC OF TEST PROBLEM

FIGURE 3.8A-40, Rev. 47

Auto-Cad Figure Fsar 3_8A_40.dwg





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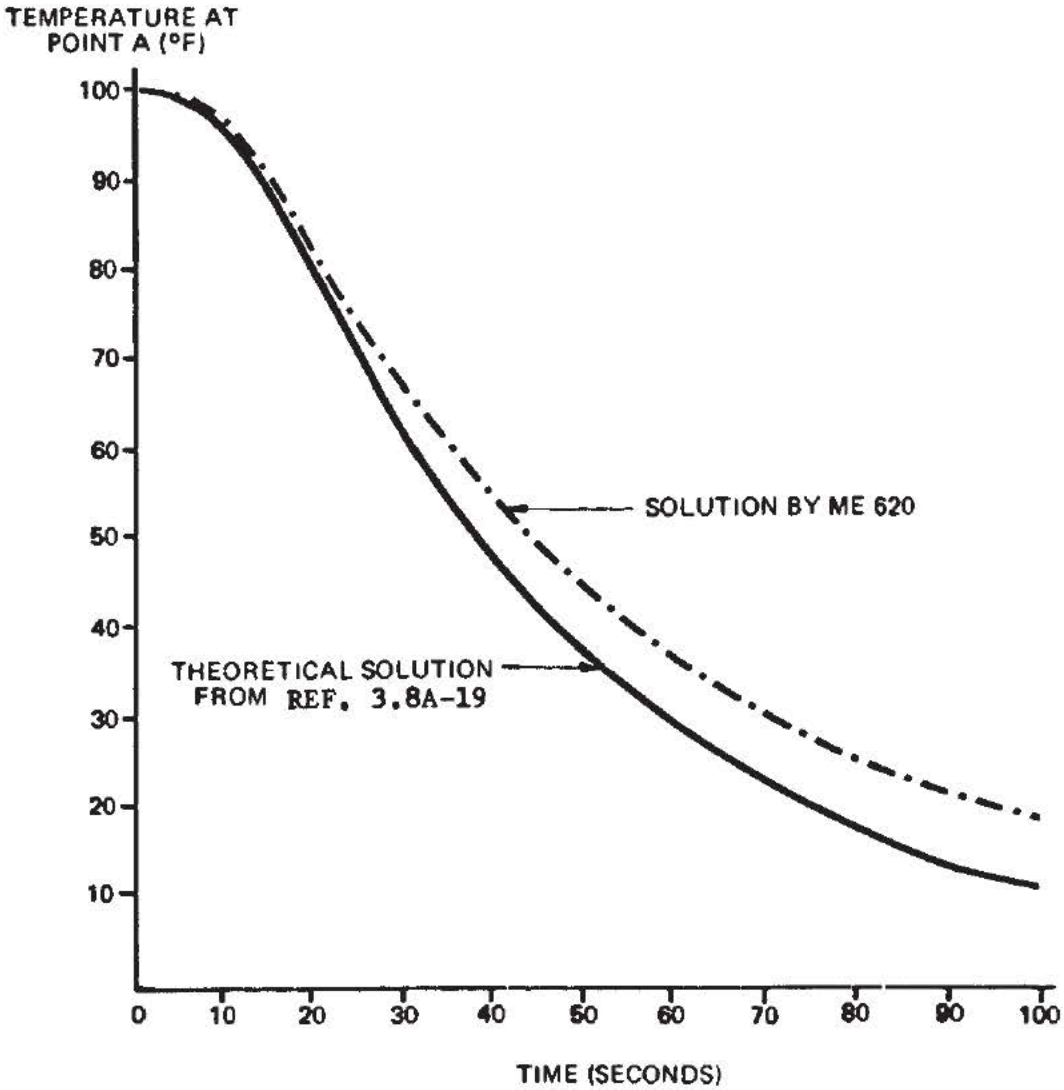
FINITE ELEMENT LAYOUT

FIGURE 3.8A-41, Rev. 47

Auto-Cad Figure Fsar 3_8A_41.dwg



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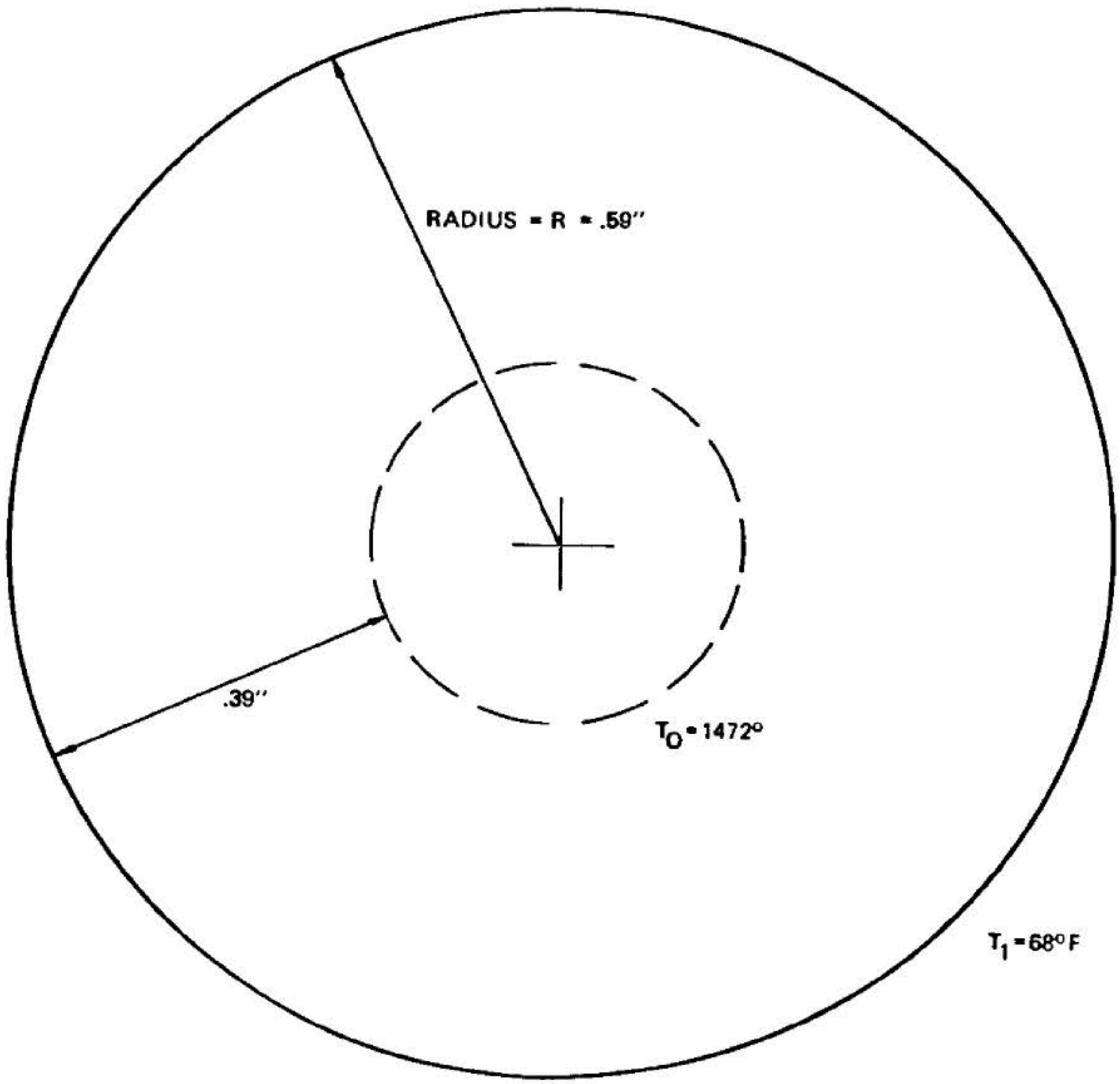
COMPARISON OF RESULTS

FIGURE 3.8A-42, Rev. 47

Auto-Cad Figure Fsar 3_8A_42.dwg



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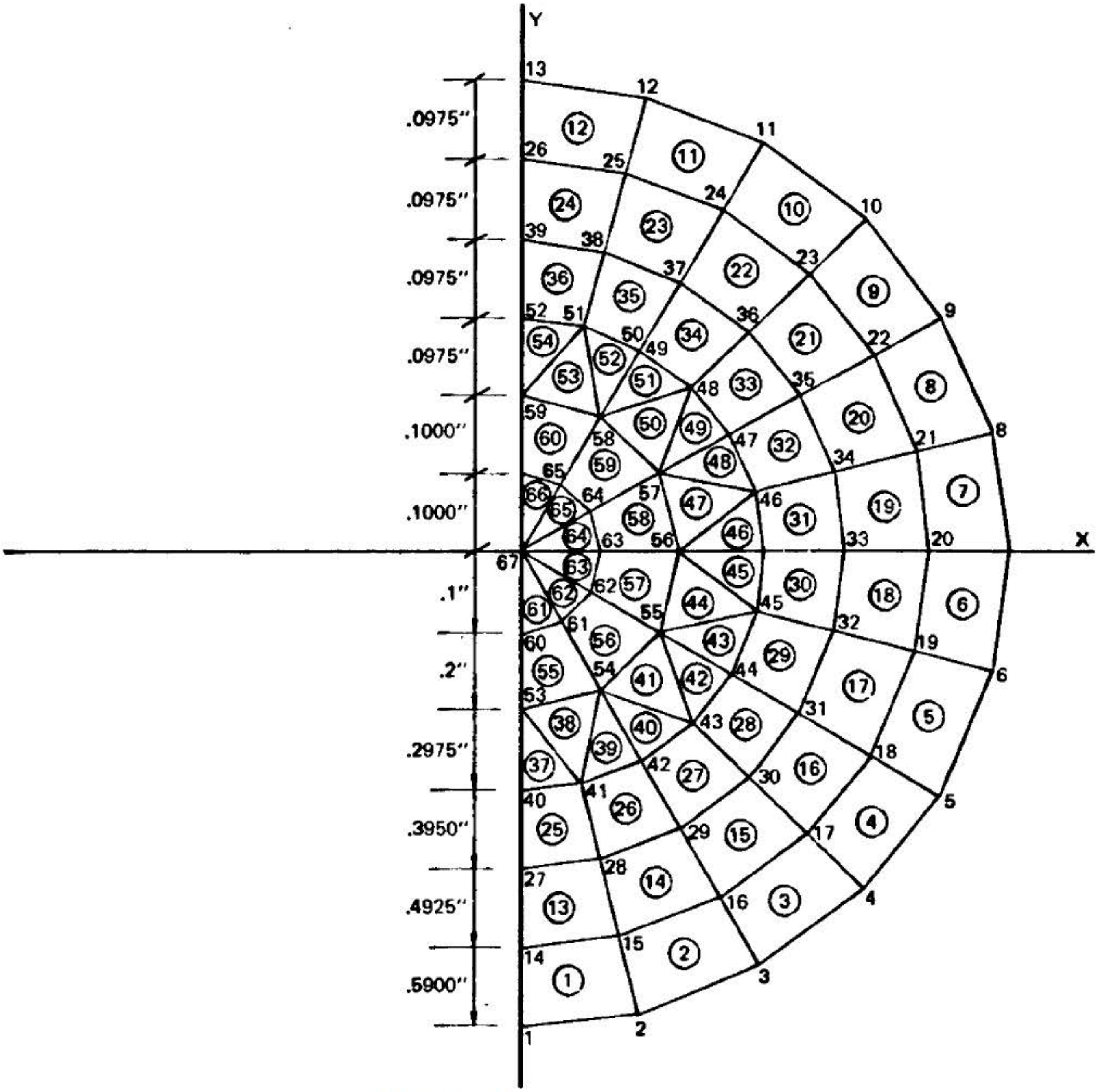
STEEL SPHERE

FIGURE 3.8A-43, Rev. 47

Auto-Cad Figure Fsar 3_8A_43.dwg



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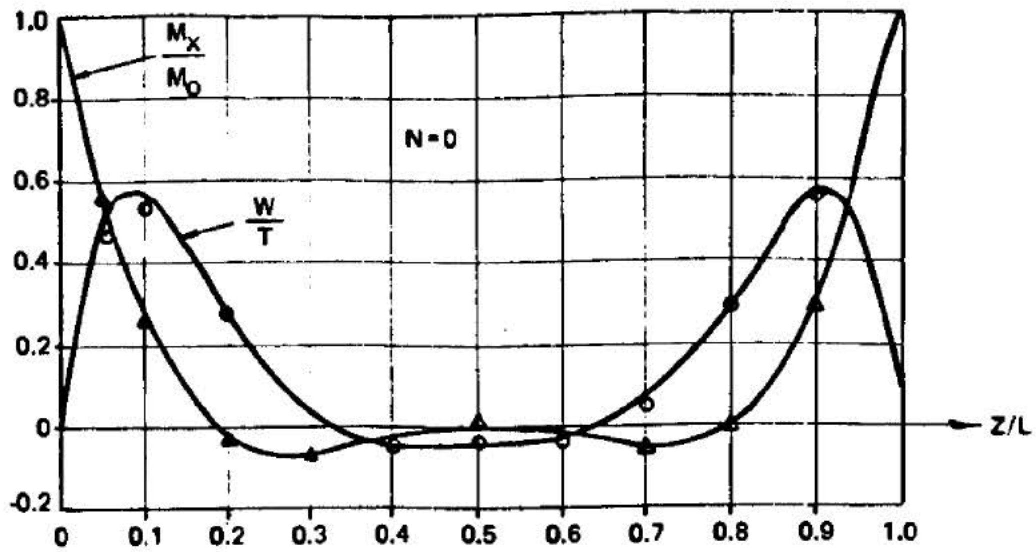
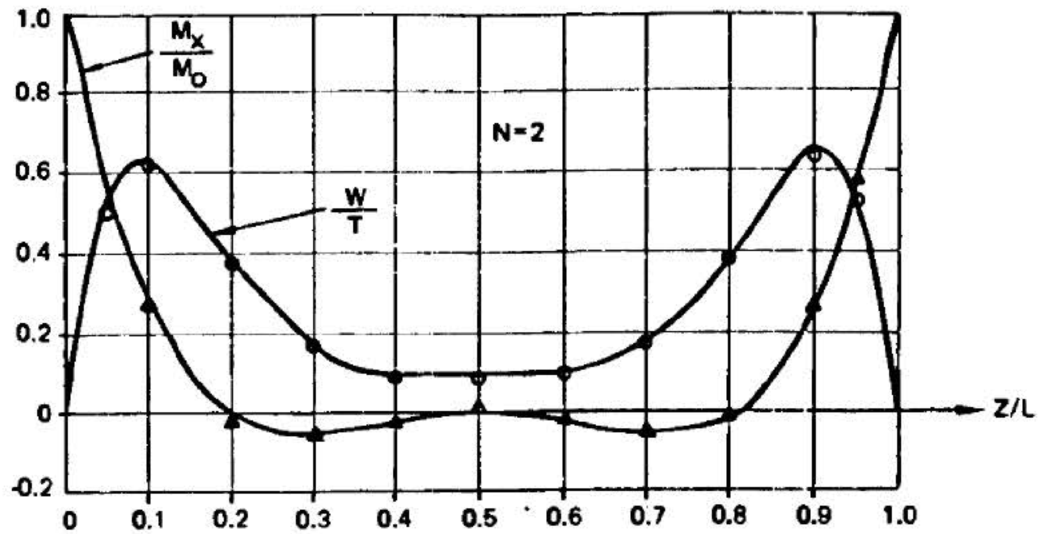
FINITE ELEMENT MODEL

FIGURE 3.8A-44, Rev. 47

Auto-Cad Figure Fsar 3_8A_44.dwg





o ASHSD (RADIAL DISPLACEMENT W)

Δ ASHSD (LONGITUDINAL MOMENT M_X)

— REFERENCE 3.8A-3

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SUSQUEHANNA STEAM ELECTRIC STATION
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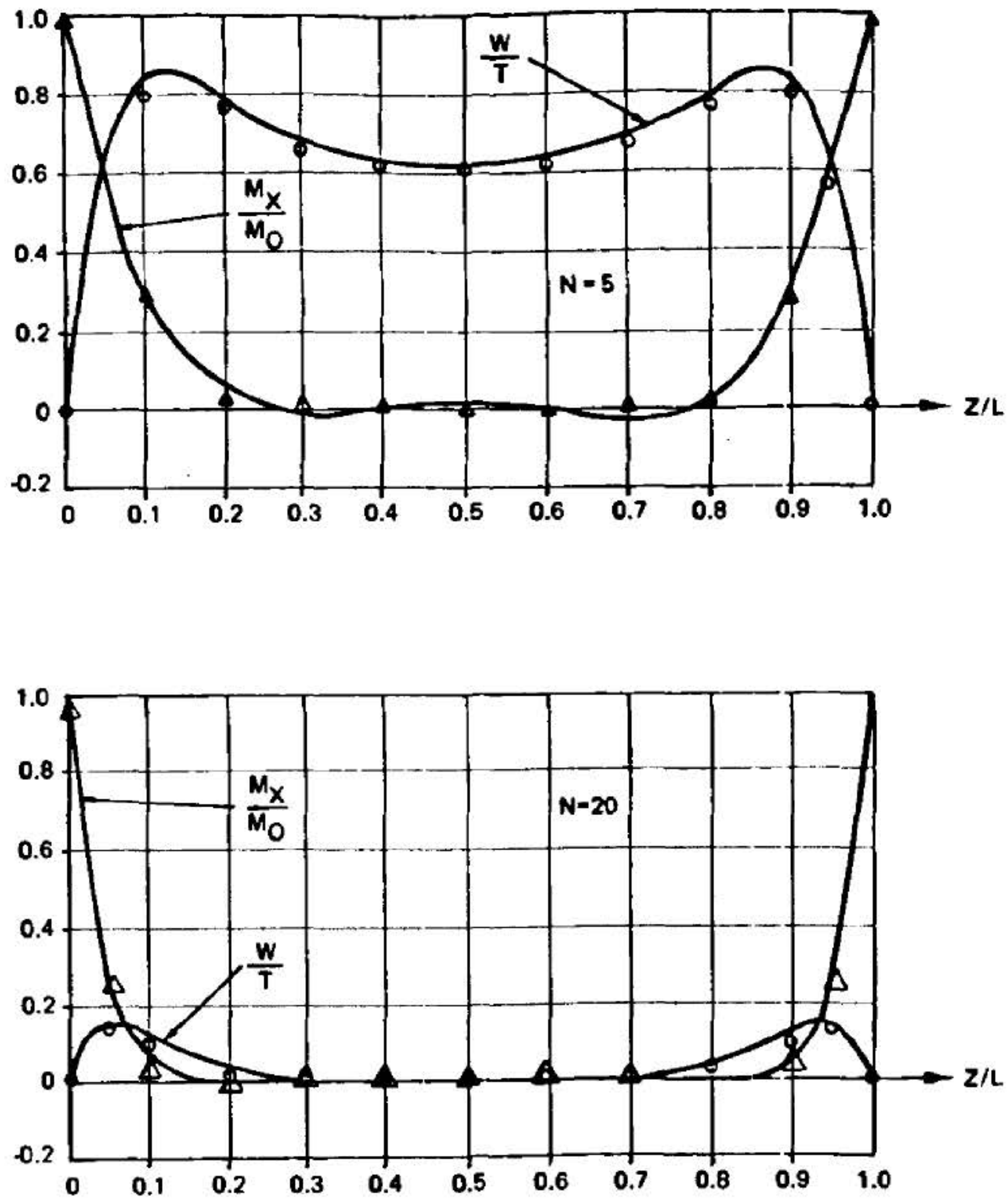
COMPARISON OF RESULTS FOR
CYLINDRICAL SHELL SUBJECTED
TO AN ASYMMETRIC BENDING

FIGURE 3.8A-8-1, Rev. 47

Auto-Cad Figure Fsar 3_8A_8_1.dwg



o ASHSD (RADIAL DISPLACEMENT W)

Δ ASHSD (LONGITUDINAL MOMENT M_X)

— REFERENCE 3.8A-3

FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
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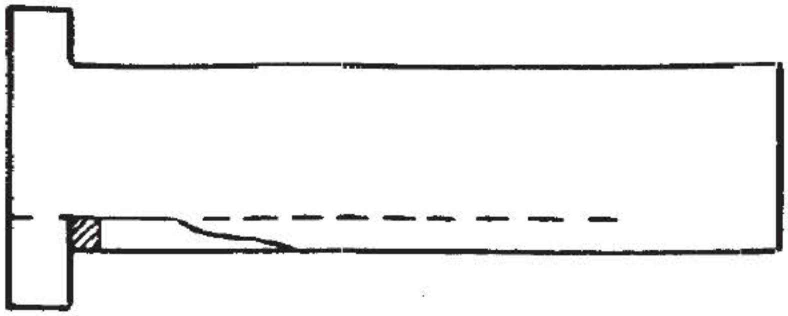
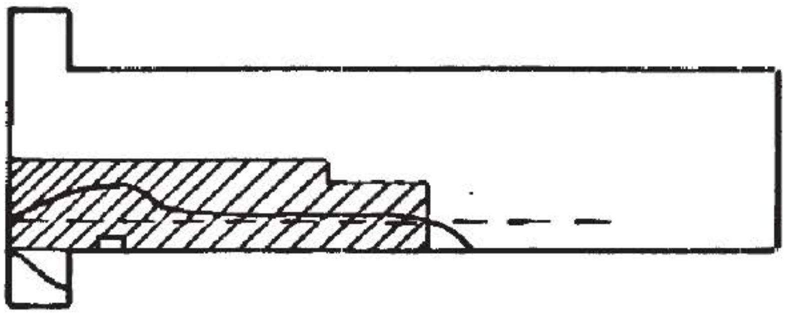
COMPARISON (CONT'D) OF RESULTS FOR
CYLINDRICAL SHELL SUBJECTED
TO AN ASYMMETRIC BENDING

FIGURE 3.8A-8-2, Rev. 47

Auto-Cad Figure Fsar 3_8A_8_2.dwg



FINEL - 8.7 kips KIPS REF. 3.8A-14 - 8.0 KIPS
(NO CRACKING IN FINEL BELOW 8.7 KIPS)



FINEL AND REF. 3.8A-14 - 20.0 KIPS

——— BOUNDARY OF CRACKED REGION, REF. 3.8A-14

 CRACKED REGION, FINEL

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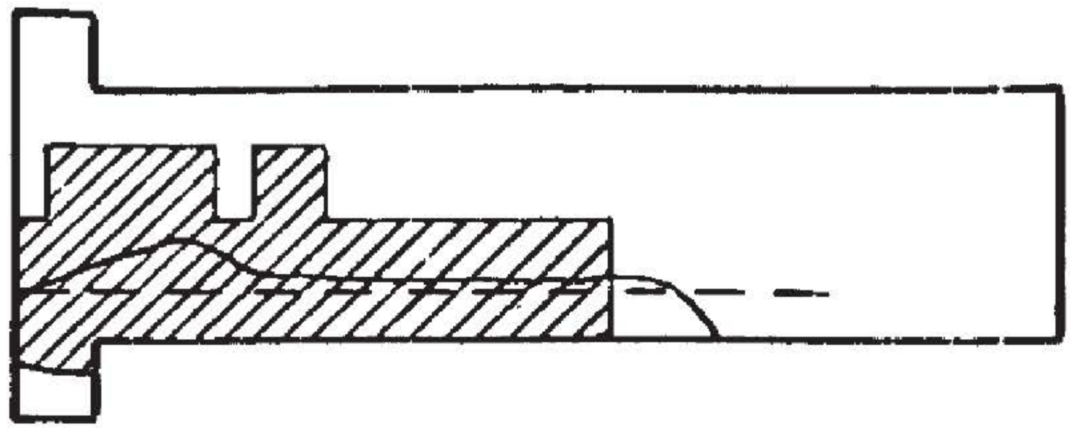
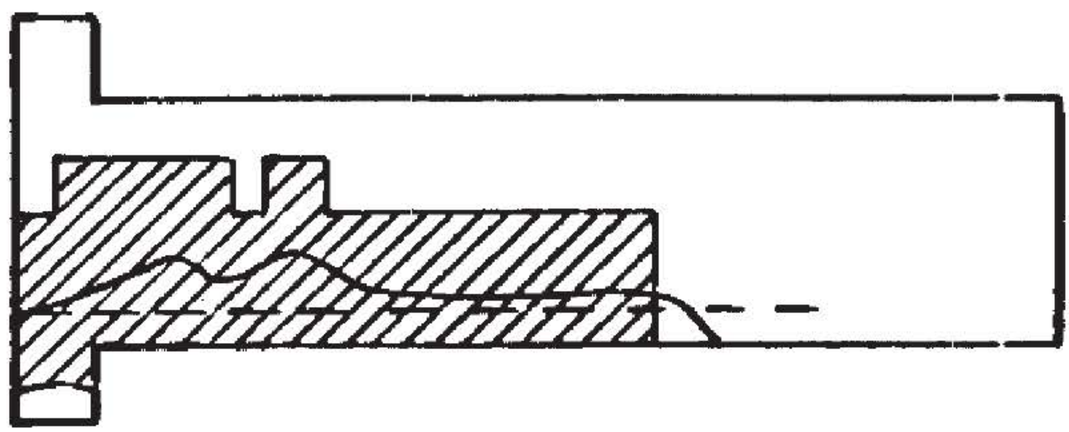
REGIONS OF CRACKING

FIGURE 3.8A-35-1, Rev. 47

Auto-Cad Figure Fsar 3_8A_35_1.dwg



FINEL AND REF. 3.8A-14 - 28.0 KIPS



FINEL AND REF. 3.8A-14 - 31.2 KIPS

FSAR REV.65

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REGIONS OF CRACKING (CONT.)

FIGURE 3.8A-35-2, Rev. 47

Auto-Cad Figure Fsar 3_8A_35_2.dwg

APPENDIX 3.8B

CONCRETE, CONCRETE MATERIALS, QUALITY CONTROL, AND SPECIAL CONSTRUCTION TECHNIQUES

Materials, workmanship, and quality control are based on the codes, standards, recommendations and specifications listed in Tables 3.8B-1, 3.8B-2, and 3.8B-3. Documents in Table 3.8B-1 are modified as required to suit the particular conditions associated with nuclear power plant design and construction while maintaining structural adequacy, for all structures except the diesel generator 'E' building. Extent of application and principal exceptions are indicated herein, and as follows:

ACI 301-72

a) Provisions of ACI 301-72, Chapter 12, Curing and Protection, shall be modified as follows:

i) Paragraph 12.2.1 shall be revised to read as follows:

"For concrete surfaces not in contact with forms, one of the following procedures shall be applied immediately after completion of placement and finishing except that the curing process may be interrupted as necessary not to exceed 8 hours providing requirements for weather protection are maintained. Such curing process may not be interrupted more than twice with a minimum of 8 hours elapsing between interruptions. If the curing is interrupted for up to 8 hours, the curing time shall be extended to provide a total of 7 days curing."

ii) Paragraph 12.2.3 shall be revised to read as follows:

"Curing in accordance with Section 12.2-1 and 12.2.2 shall be contained for at least 7 days in the case of all concrete except high-early-strength concrete for which the period shall be at least 3 days. Alternatively, if tests are made of cylinders kept adjacent to the structure and cured by the same methods, moisture retention measures may be terminated prior to 7 days when test results indicate that the average compressive strength, has reached 70 percent of the specified strength, f'c. Required period of initial curing need not be greater than the lesser of the two periods. If one of the curing procedures of Section 12.2.1.1 through 12.2.1.4 is used initially, it may be replaced by one of the other procedures of Section 12.2.1 any time after the concrete is one day old provided the concrete is not permitted to become surface dry during transition. Curing during periods of cold weather shall be in accordance with Section 12.3.1."

iii) Paragraph 12.3.1 shall be deleted and replaced with the following:

"Initial curing and protection measures for the concrete during periods of cold weather shall be in accordance with the recommendations of ACI 306-66 (1972)."

b) Provisions of ACI 301-72, Chapter 14, Massive Concrete, shall be modified as follows:

i) Paragraph 14.4.1 shall be deleted and replaced with the following:

"The slump of the concrete as placed shall be 3" or less except that a tolerance of up to 2" above this indicated maximum shall be allowed for batches provided the average for all batches or the most recent 10 batches tested, whichever is fewer, does not exceed 3". Concrete of lower than usual slump may be used provided it is properly placed and consolidated."

ii) Paragraph 14.4.3 Delete the first sentence of the paragraph and substitute the following:

"Concrete shall be placed in layers approximately 24" thick."

iii) Paragraph 14.5.1 shall be deleted and replaced with the following:

"The minimum curing period shall be in accordance with Section 12.2.3."

iv) Paragraph 14.5.4. The requirement for controlled cooling at the conclusion of the specified heating shall be accomplished by leaving the cold weather protection in place at least 24 hours after heating is discontinued. In extremely cold weather, the field engineer shall require that additional measures be taken to prevent rapid cooling of the concrete by this method.

ACI 318-71

a) Provision of ACI 318-71, Chapter 5, "Mixing and Placing Concrete" shall be modified as follows:

i) Paragraph 5.5 shall be revised by the addition of the following new paragraph 5.5.3:

5.5.3 The curing requirements as described in Sections 5.5.1 and 5.5.2 above may be interrupted as necessary not to exceed 8 hours providing requirements for weather protection are maintained. Such curing process may not be interrupted more than twice with a minimum of 8 hours elapsing between interruptions. If the curing is interrupted for up to 8 hours, the curing time shall be extended to provide a total of 7 days curing.

b) Provisions of ACI 318-71, Chapter 6, Formwork, Embedded Pipes, and Construction Joints, shall be modified as follows:

i) Paragraphs 6.3.2.4, 6.3.2.5, 6.3.2.6 and 6.3.2.7 shall be deleted and replaced with the following:

6.3.2.4 "All piping and fitting shall be tested in accordance with the requirements of the code governing that piping system (e.g., ASME Boiler and Pressure Vessel Code, ANSI B 31.1, state or local plumbing codes, etc.) or in accordance with applicable design or technical specifications, or design drawings.

Whenever the piping system is not governed by such applicable codes, code cases or design documents, then such systems shall be tested for leaks prior to concreting. The testing pressure above atmospheric pressure shall be 50 percent in excess of the pressure to which the piping and fittings may be subjected, but the minimum testing pressure shall not be less than 150 psig.

The pressure test shall be held for 4 hours with no drop in pressure except that which may be caused by air temperature."

6.3.2.5 "Drain pipes and other piping systems not governed by applicable codes and designed for pressures of not more than 1 psig need not be tested as required above."

6.3.2.6 "Piping systems which are not governed by applicable codes, code cases or design documents and which carry liquid, gas or vapor which is explosive or injurious to health, shall be retested in accordance with Section 6.3.2.4 subsequent to the hardening of the concrete."

6.3.2.7 "Piping systems may be energized with water not exceeding 50 psi nor 90°F if approved by the responsible Field Engineer".

Other piping systems, including systems governed by piping system codes or design documents exceeding 50 psi or 90°F or energized with other than water, may be energized 7 days after the concrete placement provided that the temperature does not exceed 150âF nor the pressure exceed 200 psig. Piping systems may be energized prior to and during the placement of concrete provided that: (a) the above temperature and pressure restrictions are applied, (b) the energized system is not shut down within 24 hours of concrete placement, and (c) if the pressure in the energized system drops, the lower pressure shall become the limiting pressure until the seven-day-post-placement time limit has elapsed. Piping systems which have been energized within 24 hours of concrete placement may be reenergized at any time more than 24 hours after concrete placement up to the limiting pressure.

3.8B.1 CONCRETE AND CONCRETE MATERIALS - QUALIFICATIONS

3.8B.1.1 Concrete Material Qualification

Cement

Cement is Type II, portland cement conforming to ASTM C150. Certified copies of material test reports showing chemical composition of the cement and verification that the cement being furnished complies with requirements are furnished by the manufacturer for each batch or lot.

Normal Weight Aggregate

Fine and coarse aggregates conform to ASTM C33. Aggregate source acceptability is based on the following test requirements:

Method of Test	Designation
Unit Weight of Aggregate	ASTM C29
Organic Impurities in Sands	ASTM C40
Effect of Organic Impurities in Fine Aggregate on Strength of Mortar	ASTM C87
Soundness of Aggregates	ASTM C88
Materials Finer Than No. 200 Sieve	ASTM C117
Lightweight Pieces in Aggregate	ASTM C123
Specific Gravity & Absorption of Fine Aggregate	ASTM C128
L. A. Abrasion	ASTM C131
Sieve or Screen Analysis of Fine & Coarse Aggregates	ASTM C136
Clay Lumps & Friable Particles	ASTM C142
Scratch Hardness of Coarse Aggregates	ASTM C235
Potential Reactivity of Aggregate	ASTM C289

Petrographic Examination	ASTM C295
Lightweight Aggregates	ASTM C330
Percentage of Particles of Less Than 1.95 Specific Gravity in Coarse Aggregate	AASHTO T150
Resistance of Concrete Specimens to Rapid Freezing and Thawing in Water	AASHTO T161
Flat and Elongated Particles	CRD C119

Coarse aggregate loss from the L.A. Abrasion Test (ASTM C131) using Grading A is limited to 40 percent by weight at 500 revolutions.

Coarse aggregate grading is for size numbers 4, 8, and 67 as defined in ASTM C33 and the quantity of flat and elongated particles is limited to 15 percent in any nominal size group.

When fine and coarse aggregates are tested per ASTM C117 to meet the requirements of ASTM C33, and when the results of any of the aggregate sizes exceed the stated limits for fines, the aggregate is accepted, provided the total amount of aggregate fines in a given mix is not greater than the total amount permitted for each aggregate size at ASTM C33 limits.

High Density Aggregates

The requirements for high density aggregates are the same as for normal density aggregates except as noted below.

Fine and coarse aggregate conforms to ASTM C637 except that grading is as follows:

Sieve Size U.S. Std. Sq. Mesh	Percentage Passing Fine Aggregate (Sand)	Coarse Aggregate 1-1/2 in.
2 in.		100
1-1/2 in.		95-100
3/4 in.		35-70
3/8 in.	100	10-30
No. 4	75-95	2-15
No. 8	55-85	0-10
No. 16	30-60	
No. 30	15-45	
No. 50	10-30	
No. 100	0-15	

The fineness modulus of the fine aggregate is not less than 3.2 nor more than 4.2. Both fine and coarse aggregate have a minimum bulk specific gravity of 4.0.

These aggregates are not tested per AASHTO T161 unless the structure is exposed to a design freeze-thaw environment and are also not tested per ASTM C330.

Certified test reports are prepared by an independent testing laboratory for each material shipment attesting to aggregate conformance to cleanliness requirements when tested per ASTM C117 and specific gravity requirements when tested per ASTM C127 and C128.

Pozzolan

Pozzolan, when used, conforms to ASTM C618 for Class F except that the maximum loss on ignition of 6 percent. Prior to shipment a minimum of one sample is taken and tested in accordance with ASTM C311 to demonstrate conformance with the above. Such documentation accompanies material shipment.

Mixing Water and Ice

Water and ice used in mixing concrete is free of injurious amounts of oil, acid, alkali, organic matter, or other deleterious substances as determined by AASHTO T26. Such water and ice does not contain impurities that would cause either a change in the setting time of portland cement of more than 25 percent or a reduction in compressive strength of mortar of more than 5 percent compared with results obtained with distilled water. The water and ice do not contain more than 250 ppm of chlorides as Cl, or more than 1000 ppm of sulphates as SO4. The pH range is between 4.5 and 8.5.

Admixtures

Air entraining admixtures, when used, conform to ASTM C260. Water reducing and retarding admixtures, when used, conform to ASTM C494 for types A and D. Types A and D are used in accordance with the manufacturer's recommendations. Certificates of conformance stating conformance to the applicable ASTM specification are furnished with each shipment. Use of calcium chloride is not permitted.

3.8B.1.2 Concrete Mix Design Concrete Properties

Concrete Properties

Concrete properties required for each type of mix design are verified by testing for the applicable properties indicated below:

Property	Test Designation
Compressive Strength	ASTM C39
Unit Weight	ASTM C138
Slump	ASTM C143
Air Content	ASTM C231

The following additional properties of selected mix designs have been determined to ascertain material compatibility with design assumptions:

Static Modulus of Elasticity	ASTM C469
Static Poizzon's Ratio	ASTM C469
Dynamic Modulus of Elasticity	ASTM C215
Dynamic Poizzon's Ratio	ASTM C215
Thermal Diffusivity	CRD C36
Thermal Coefficient of Expansion	CRD C39

Concrete Mix Proportions

Proportions of ingredients are determined and tests conducted in accordance with ACI 211.1, except as noted below, for combinations of materials established by trial mixes. These proportioning methods provide required concrete strength, durability, and unit weight while maintaining adequate workability and proper consistency to permit required consolidation without excessive segregation or bleeding.

The design strength (f'c) of mixes that contain pozzolan is measured at 90 days; for those that do not contain pozzolan, f'c is measured at 28 days. Three cylinders are tested for each mix design and age as follows:

Pozzolan Mix	Nonpozzolan Mix
3 days	3 days
7 days	7 days
28 days	28 days
90 days	

Concrete mixes for limited uses such as in radiation-sensitive facilities and high density concrete do not contain pozzolan. All other concrete mixes are based on use of approximately 15 to 20 percent pozzolan by weight as cement replacement. Further concrete mixes except limited application use, such as high density concrete, are based on 3 to 6 percent air entrainment for both 3/4 and 1-1/2 in. nominal maximum size coarse aggregate. These measures provide a concrete possessing both good freeze-thaw and sulphate resistance.

In lieu of establishing limits on water-cement ratio, the concrete is proportioned and mixed so as to be placed at specified slumps. The average slump at the point of placement is less than the"Working Limit," which is the maximum slump for estimating the quantity of mixing water to be used in the concrete. An "Inadvertency Margin" is the allowable deviation from the "Working Limit" for such occasional batches as may inadvertently exceed the "Working Limit." Jobsite tests have indicated that concrete with slumps at the "Inadvertency Margin" will produce acceptable quality concrete.

3.8B.1.3 Grout

Construction Grout

Construction grout for use at horizontal construction joints and similar applications is proportioned from the same materials as for concrete. Grout strength is determined in accordance with ASTM C109.

Starter Mix

Starter mixes are used in applications such as at the bottom of foundation slabs and in lieu of construction grout and are proportioned from the same materials as for concrete. These mixes are generally proportioned for a "Working Limit" slump 2 in. greater than the associated concrete mix. Trial mixes are prepared and tested for strength as described for general concrete mixes.

Nonshrink Grout

Nonshrink grout is prepared from proprietary materials such as Embeco LL-636 by Master Builders Company or Five Star Grout by US Grout Corporation. Such grouts are proportioned in accordance with the manufacturer's recommendations and are tested for expansion, compressive strength, and flow characteristics with maximum water content recommended by the manufacturer prior to use.

3.8B.2 CONCRETE AND CONCRETE MATERIALS - BATCHING, PLACING, CURING, AND PROTECTION

3.8B.2.1 Storage

Storage of aggregates, cement, pozzolan, and admixtures is in accordance with the recommendations of ACI 304.

3.8B.2.2 Batching, Mixing, and Delivering

Concrete for principal structures is provided as central mixed concrete from a batch plant located on the jobsite. Some limited amounts of concrete are obtained from an offsite batch plant. All such batch plant facilities are certified by the National Ready Mix Concrete Association (NRMCA) and measuring devices are calibrated at required intervals and more frequently when deemed appropriate.

Measuring of materials, batching, mixing, and delivering normal weight concrete conform to ASTM C94, Alternate No. 1 except as otherwise noted.

Regulatory Guide 1.69 has basically adopted ANSI N101.6. This ANSI standard is interpreted to be applicable only to high density concrete serving as radiation shields and is therefore not used on this project. As the concrete has a dual function of providing shielding and structural adequacy, the standard practices described herein are adopted for normal weight concrete. When higher density concrete is required for shielding purposes, the practices adopted are in general agreement with those outlined in the ACI Journal of August 1975 report by ACI Committee 304: "High Density Concrete Measuring, Mixing, Transporting, and Placing."

The delivery of materials from the batching equipment is within the following limits of accuracy:

	Over and Under Percent	
Material	Weight	Weight
	Less than or equal to 30 percent of scale capacity	Greater than 30 percent of scale capacity
Cement	Minus 0 Plus 4	1

Pozzolan	Minus 0 Plus 4	1
Water	1	1
Ice	1	1
Aggregate equal to or smaller than 1-1/2	3 (See note below)	2
Admixture when batched separately	3	1

Note: Or plus or minus 0.3 percent of scale capacity, whichever is less.

NRMCA Section 2.7 provides additional tolerances for batching recorders.

3.8B.2.3 Placing

Placing of normal weight concrete is in accordance with the recommendations of ACI 304. Placing of high density concrete is as described above.

3.8B.2.4 Consolidation

Consolidation of concrete is in accordance with the recommendations of ACI 309.

3.8B.2.5 Curing

Curing of concrete is in accordance with the recommendations of ACI 308.

3.8B.2.6 Hot and Cold Weather Concreting

Measures taken to mitigate the effects of hot and cold weather during each step of the concreting operation are in accordance with ACI 305 and 306 respectively.

3.8B.3 CONCRETE AND CONCRETE MATERIALS-CONSTRUCTION TESTING

An independent concrete and concrete materials testing laboratory has been established at the project site to monitor the quality of such work and materials and to promptly report any deviations from specified conditions. Such testing personnel are qualified to meet the requirements of NRC Regulatory Guide 1.58. Procedures and tests for accomplishing such work are reviewed and accepted by Bechtel prior to use. Qualifications and procedures in use by Bechtel quality control personnel and the extent of conformance to Regulatory Guide 1.94 are described in Section 3.13.

Production testing for concrete and concrete materials is as shown in Table 3.8B-1.

Materials that do not meet test requirements are not used in the construction.

If the measured concrete temperature, slump, unit weight, or air content falls outside the limits specified, a check is made. In the event of a second failure, the load of concrete represented is not in the construction.

Concrete cylinder tests results are reviewed for compliance with Chapter 17 of ACI 301 and are evaluated in accordance with ACI 214.

Materials or portions thereof that do not meet the above criteria but may inadvertently be used are handled as described in Appendix D and amendments to the FSAR.

3.8B.4 CONCRETE REINFORCEMENT MATERIALS - QUALIFICATIONS

Reinforcing steel for concrete structures conforms to ASTM A615, Grade 60, including Section S1 for bar sizes 14 and 18. Certified copies of material test reports indicating chemical composition, physical properties and dimensional compliance are furnished by the manufacturer for each heat.

When permitted by the design drawings, reinforcing steel is furnished by the supplier to special chemistry requirements to enhance reinforcing weld characteristics. The chemistry of such bars meets the following chemical analysis requirements expressed in maximum percentage by weight:

C	-	0.50%	P	-	0.05%
Mn	-	1.30%	S	-	0.05%

Weld splicing of reinforcing is not performed in the primary containment structures.

Each bundle of reinforcing steel is tagged to ensure unique heat traceability during production, while in transit and into storage. During storage and installation reinforcing steel is collectively traceable to the group of certified material test reports received.

Prior to installation at the jobsite all reinforcing steel is subjected to a testing program meeting the requirements of NRC Regulatory Guide 1.15. Any reinforcing steel which does not meet these requirements is not used in the construction.

Sleeves for reinforcing steel mechanical splices conform to ASTM A519 for Grades 1018 and 1026. Certified copies of material test reports indicating chemical composition and physical properties are furnished by the manufacturer for each sleeve lot.

3.8B.5 CONCRETE REINFORCEMENT MATERIALS - FABRICATION

3.8B.5.1 Bending Reinforcement

Hooks and bends are fabricated in accordance with ACI 318 Chapter 7.1. Bars partially embedded in concrete are bent subject to the following conditions.

Bending Partially Embedded Reinforcement

The minimum distance from existing concrete surface to the beginning of bend and the minimum inside diameter of bend is:

Bar Size	Min. Dist. from Surface to Beginning of Bend	Min. Inside Bend Diameter
No. 3 through No. 8	3 Bar Diameters	6 Bar Diameters
No. 9, No. 10, No. 11	4 Bar Diameters	8 Bar Diameters
No. 14, No. 18	5 Bar Diameters	10 Bar Diameters

Bars No. 3 to No. 5 inclusive may be bent cold once. Heating is required for subsequent straightening or bending.

Bars No. 6 and larger may be bent and straightened, provided that heating is used.

When heat is used, it is applied as uniformly as possible over a length of bar equal to 10 bar diameters, and is centered at the middle of the arc of the completed bend. The maximum bar temperature is between 1100 and 1200°F, and maintained at that level until bending (or straightening) is complete.

Temperature-measuring crayons or a contact pyrometer is used to determine the temperature. Heat is applied in such a way as to avoid damage to the concrete. Care is taken to prevent rapid quenching of heated bars.

Straightened bars are visually inspected to determine whether they are cracked, reduced in cross-section, or otherwise damaged. Any damaged portions are removed and replaced.

3.8B.5.2 Splicing Reinforcement

Lap Splices

In general, lapped splices are used for No. 11 and smaller bars. Such lap splices are in accordance with Sections 7.5, 7.6, and 7.7 of ACI 318.

Mechanical Splices

In general, mechanical (Cadweld) splices are used for all No. 14 and No. 18 splices, for splices across liner plates and in lieu of standard hooks when a plate anchorage is required or desirable. To obtain an effective level of quality control for this splicing process, a qualification, inspection, testing, and acceptance program in accordance with NRC Regulatory Guide 1.10 has been used. Welding of splice sleeves to liners, or other plates and shapes is in accordance with AWS D1.1.

Welded Splices

Whenever both lap and mechanical splices have been determined to be impractical, welded splices are used on a case-by-case approval basis. Such welding is performed by qualified welders using a procedure conforming to the basic recommendations of AWS D12.1.

3.8B.5.3 Placing Reinforcement

Reinforcement is securely tied with wire and held in position by spacers, chairs, and other supports to maintain placement accuracy within the tolerances established for reinforcement protection and the design requirements.

3.8B.5.4 Spacing Reinforcement

Spacing and reinforcement is in accordance with Sections 3.3.2, 7.4.1, and 7.4.5 of ACI 318.

3.8B.5.5 Surface Condition

Reinforcement surface condition at the time of concrete placement is in accordance with Section 7.2 of ACI 318.

3.8B.6 CONCRETE REINFORCEMENT MATERIALS - CONSTRUCTION TESTING

Inspection of reinforcement materials to ensure that bending, placing, splicing, spacing, and surface condition requirements are met is in accordance with the program described in Chapter 17 as is the extent of conformance to Regulatory Guide 1.94.

3.8B.7 FORMWORK AND CONSTRUCTION JOINTS

Formwork is designed and constructed in accordance with ACI 347. Such formwork maintains position and shape to keep deformations within limits established by the design requirements.

Prior to concrete placement, construction joints are cleaned to remove unsatisfactory concrete, laitance, coatings, debris, and other foreign material and to expose the aggregate. The joints are then saturated to produce a saturated surface dry condition. Horizontal construction joints then shall be covered with either approximately 1/4 in. of construction grout or a layer of starter mix which is approximately 4 to 6 in. deep.

Except as discussed below, concrete is placed in accordance with Regulatory Guide 1.55.

Regulatory positions 2 and 3 of the Regulatory Guide state the presumed functional responsibilities of the "Designer" and the "Constructor." Under the designer's role are listed the responsibilities for checking shop drawings and locations of construction joints. On this project, the former is fully delegated to the Bechtel field, although the design engineering office may check significant portions and may advise the field accordingly. The responsibility for construction joint location is partly delegated to the field in the sense that the field has to follow the guidelines set out in the design drawings and specifications prepared by the design engineering office. In interface areas, a delegation of the design engineering office's responsibility to the field office is within the definition of the terms "responsibility" and delegated responsibility" as discussed in Paragraph 1.3 of the proposed ANSI N45.2.5. Delegation of the responsibilities for checking the reinforcing drawings to the field engineering group is justified by the following:

a) The Bechtel field engineering group is segregated from the field supervision group, although both are located at the jobsite and eventually report to the project construction manager.

b) The field engineering group is staffed, for the most part, by graduate engineers who have been trained in the use of the ACI code and understand the design implication of the proper location, splicing, and embedment of reinforcing steel.

c) The field inspection of the actual rebar as placed in the forms is conducted using the engineering drawings as the primary source document. This ensures a check on any errors which may have passed the critical review of the field engineer in checking the shop detail or erection drawings.

d) It is standard practice in the civil engineering profession that engineering requirement drawings for reinforcing be converted to shop detail and erection drawings in accordance with ACI standards applied by steel detailers at the reinforcing steel vendor's shop. Most contractors installing reinforcing steel rely upon their superintendent and foreman for correct interpretation of these detail drawings in erecting the reinforcing steel. While this is also true of Bechtel field operation, we do have the additional help and guidance of the field engineers both during the installation phase and finally at the inspection phase prior to final sign-off on the report card.

e) The field engineers have the added benefit of being able to plan and witness the actual installation and can, therefore, better foresee any difficulties in meeting the intended design requirements. Their assessment of the situation is further assisted by regular telephone communication with the design engineers who also periodically visit the jobsite.

The above procedure of delegation of the design engineering office's responsibility to the field personnel and periodic monitoring by the engineering office ensures correctness and conformance of the shop drawings to the design drawings and therefore meets the intent of Regulatory Guide 1.55.

TABLE 3.8B-1

Minimum Testing Frequencies for Concrete Materials and Concrete
(Except for the Diesel Generator 'E' Building)

Material	Requirement	Test	Frequency
Cement	Standard Physical and Chemical Properties	ASTM C150	The lesser of each 5000 cubic yards of production concrete of each 1200 tons of cement used
Pozzolan	Chemical and Physical Properties per ASTM C618	ASTM C311	Each shipment of pozzolan by manufacturer and upon occasion by the jobsite
Aggregate	Unit Weight of Aggregate	ASTM C29	Once for each 5000 cubic yards of production
	Organic Impurities	ASTM C40	Once daily for each 1000 cubic yards of production
	Soundness of Aggregates	ASTM C88	Once for each 5000 cubic yards of production
	Material Finer than No. 200 Sieve	ASTM C117	Once daily for each 1000 cubic yards of production
	Lightweight Pieces in Aggregates	ASTM C123	Once for each 5000 cubic yards of production
	Specific Gravity and Absorption	ASTM C127/C128	Once for each 5000 cubic yards of production
	L. A. Abrasion	ASTM C131	Once for each 5000 cubic yards of production

TABLE 3.8B-1 (Continued)

Material	Requirement	Test	Frequency
	Gradation	ASTM C136	
	Coarse Aggregate		Once daily for each 1000 cubic yards of production
	Fine Aggregate		Twice daily for each 1000 cubic yards of production
	Petrographic Examination	ASTM C295	Once for each 10,000 cubic yards of production
	Moisture	ASTM C566	
	Coarse Aggregate		Once daily for each 1000 cubic yards of production
	Fine Aggregate		Twice daily for each 1000 cubic yards of production
	Flat and Elongated Particles	CRD C119	Once daily for each 1000 cubic yards of production
Water and Ice	Quality of Water to be Used in Concrete (To meet the requirements herein)	AASHTO T26	Once each three months or each 5000 cubic yards of production
Admixtures			

TABLE 3.8B-2

Testing Requirements for Concrete Materials Used in the
Diesel Generator 'E' Building

<u>Frequency of Test</u>

Material	Test (Specification)	By Manufacturer/Supplier/ Contractor	By Laboratory	Remarks
Cement	Complete physical & chemical analysis (ASTM C-150)	Initial Qualification (M)	-	Sample from mill
		Each shipment (M)	-	Sample from mill
		-	Per ASTM C-183	Sample from batch plant
	Compressive strength of Mortar Cubes (ASTM C-109)	-	Material stored 4 months or more	Sample from storage
		-	Each mill run	Sample from mill
Aggregates**	The following tests covered in ASTM C-33 plus C 128, C 295 and CRDC119 as follows:			
	Sieve Analysis (ASTM C136)	Initial Qualification (s)	Daily*	
	Material Finer than No. 200 Sieve (C 117)	Initial Qualification (s)	Daily*	
	Moisture Content (C566)	Initial Qualification (s)	Daily*	
	Clay Lumps (ASTM C142)	Initial Qualification (s)	Monthly	
	Organic Impurities (ASTM C40)	Initial Qualification(s)	Weekly	
	L. A. Abrasion (ASTM C131)	Initial Qualification (s)	Each 4000 tons or every 6 months	
	Potential Reactivity (ASTM C289)	Initial Qualification (s)	Each 4000 tons or every 6 months	
	Soundness (ASTM C88)	Initial Qualification (s)	Each 4000 tons or every 6 months	

TABLE 3.8B-2 (Continued)

Frequency of Test

Material	Test (Specification)	By Manufacturer/Supplier/ Contractor	By Laboratory	Remarks
	Lightweight Pieces (ASTM C123)	Initial Qualification (s)	Monthly	
	Scratch Hardness of Coarse Aggregate (C851)	Initial Qualification (s)	Monthly	Replaces Soft Fragments (C-235)
	Specific Gravity & Absorption, C.A. (C127)	Initial Qualification (s)	Each 4000 tons	
	Specific Gravity & Absorption, F.A. (C128)	Initial Qualification (s)	Each 4000 tons	
	Mortar Making Properties (C87)	Initial Qualification (s)	Each 4000 tons	
	Flat & Elongated Particles Corp of E, (CRD C-119)	Initial Qualification (s)	Each 4000 tons or every 6 months	
	Fineness Modulus F. A. (C33)	Initial Qualification (s)	Daily*	
	Petrographic Examination of Aggregates for Concrete (C295)	Initial Qualification (s)	Each 4000 tons	
Admixtures Water Reducer (Types A & D)	Composition and uniformity (ASTM C494)	Initial Qualification (M) Each lot shipped (M)		
Air Entraining Agent	Composition and uniformity (ASTM C260)	Initial Qualification (M) Each lot shipped (M)		
Water	Chloride Content (ASTM D512)	Initial Qualification (c)	Every 6 months	

TABLE 3.8B-2 (Continued)

Frequency of Test

Material	Test (Specification)	By Manufacturer/Supplier/ Contractor	By Laboratory	Remarks
	Compare following properties for mixing water vs. distilled water: Soundness (ASTM C151) Time of Set (ASTM C191) Compressive strength of Mortar Cubes (ASTM C109)	Initial Qualification (c)	Every 6 months	

* The daily tests on the aggregate shall be performed only on those actually being batched that day.

** 1. Additional tests shall be performed for each change in source of supply and for each change in Supplier's Quarry location.

2. Materials which fail to meet requirements of the tests shall not be used and shall be removed to a spoil area.

3. A tolerance of +5% on quantity of aggregate is acceptable for the aggregate tests to be performed at a frequency of "each 4000 tons of aggregate."

TABLE 3.8B-3
Testing Requirements for Concrete Used in the
Diesel Generator 'E' Building

Item	Test (Specification) or Activity	By	Frequency	Remarks
Design Mixes	Design and qualify test mixes. Establish mix properties of:			
	Cement Flyash Water Coarse Aggregate Fine Aggregate Air Entraining Admixtures Water Reducing Admixtures	Contractor	Initial qualification of each proposed mix	Concrete materials to meet qualification tests of Table 1
	Determine for each mix:			
	Compressive strength (ASTM C39) Static Modulus of Elasticity (ASTM C469) Poisson's Ratio (ASTM C469)	Contractor	Initial qualification of each proposed mix.	
Production Concrete	Compressive strength (ASTM C39) (Laboratory cured)	Laboratory	1 set of strength specimens for each 100 c.y. or fraction of each mix.	Following ACI 301, 16.3.4 except use 4 specimens. Test 2 at 7 and 2 at 28 days.
	• Compressive Strength of Grout/Mortar (ASTM C109) (Laboratory Cured)	Laboratory	1 set of 6-2 inch cubes for each 100 c.y. or fraction or fraction of each mix	Three cubes shall be tested at 7 days and three at 28 days.
	Sampling Method (ASTM C172)	Laboratory		
	Compressive strength (ASTM C31) (Field cured)	Laboratory	1 set of strength specimens for each structure or major part as directed by the Engineer	

TABLE 3.8B-3 (Continued)

Item	Test (Specification) or Activity	By	Frequency	Remarks
	Slump (ASTM C143)	Laboratory	Each strength test, first batch each day and every 50 c.y.	Measured at point of deposit as defined in Section 5.2.1 of this Specification
	Air Content (ASTM C231)	Laboratory	With each set of compression cylinders and every 50 cub. yd.	
Production Concrete	Unit Weight of Fresh Concrete (ASTM C138)	Laboratory	Each strength test	
	Temperature	Laboratory	Each strength test, First Batch each day and every 50 cub. yds.	
	Batch Ticket Information. Include the following: Date Time batched Location Operator Truck No. Mix Number Quantity batched Pour location "As Batched" quantities Maximum size of aggregate Amount of water withheld at the Batch Plant Amount of water subsequently added prior to placement	Laboratory	Each Batch Produced	Batch Tickets shall be forwarded to the Constructor's Q.C. Inspector with each truckload of concrete delivered to the site.
	Concrete Test Report In addition to the batch ticket information the following shall be reported:	Laboratory	Each strength specimen set	

TABLE 3.8B-3 (Continued)

Item	Test (Specification) or Activity	By	Frequency	Remarks
	Sampler			
	Times sampled and tested			
	Air temperature			
	Concrete temperature			
	Measured properties of fresh concrete			
	Cylinder numbers			
	Compressive strengths			
	Capping material			
	Type of break			
	Tested by			

* Note: Strength testing is not required for the Grout/Mortar used for buttering at horizontal construction joints per Section 4.11.4 of this specification

APPENDIX 3.8C
CONCRETE UNIT MASONRY, MASONRY MATERIALS AND QUALITY CONTROL

Materials, workmanship and quality control are based on the applicable codes, standards, recommendations and specifications listed in Table 3.8-1. These documents are modified as required to suit the particular conditions associated with nuclear power plant design and construction while maintaining structural adequacy.

3.8C.1 CONCRETE UNIT MASONRY AND MASONRY MATERIALS - QUALIFICATIONS

Concrete Unit Masonry

Concrete unit masonry conforms to either ASTM C90, Type 1, Grade N for hollow masonry units or ASTM C145, Type I, Grade S for solid masonry units.

Masonry Mortar

Masonry mortar conforms to ASTM C270, Type M, and is of the following ingredients:

Portland cement conforming to ASTM C150, Type I or II.

Hydrated lime conforming to ASTM C207, Type S.

Aggregate conforming to ASTM C144.

Masonry Grout

Masonry grout conforms to ASTM C476.

Concrete Infill

Concrete infill conforms to the program and requirements described in Appendix 3.8B.

Reinforcing Steel

Reinforcing steel conforms to the program and requirements described in Appendix 3.8B.

Horizontal Joint Reinforcement

Horizontal joint reinforcement is made of wire conforming to ASTM A82. Certificates of compliance stating conformance to ASTM A82 are furnished for the joint reinforcement.

3.8C.2 CONCRETE UNIT MASONRY AND MASONRY MATERIALS - CONSTRUCTION AND ERECTION

Construction and erection of concrete unit masonry and masonry materials is in conformance with the requirements of the Uniform Building Code.

3.8C.3 CONCRETE UNIT MASONRY AND MASONRY MATERIALS - CONSTRUCTION TESTING

An independent testing laboratory has been established at the project site to monitor the quality of concrete unit masonry and masonry materials and to promptly report any deviations from specified conditions. Procedures and tests for accomplishing such work are reviewed and accepted by Bechtel prior to use.

Production testing for concrete unit masonry and masonry materials is as follows:

Concrete Unit Masonry

Tests of concrete unit masonry are performed at a frequency of six units randomly selected from each lot of 5000 units or fraction thereof delivered to the jobsite. Such units are tested in accordance with ASTM C140 to demonstrate compliance with ASTM C90 for hollow masonry units and with ASTM C145 for solid masonry units. Such tests are performed and acceptability determined, prior to use of that lot of masonry units.

Masonry Mortar

Tests of masonry mortar are performed prior to use initially and then for each 5000 concrete masonry units placed. Such tests are performed in accordance with and meet the acceptance standards of ASTM C270.

Masonry Grout

Tests of masonry grout are performed at a frequency of once for each 100 cubic yards of each class of masonry grout produced. Each test consists of 6 two inch cubes made, cured and tested in accordance with ASTM C109. Three cubes are tested at 7 days and three at 28 days.

Concrete Infill

Concrete infill is tested at the same frequency and by the methods described for Appendix 3.8B.

Materials that do not meet test requirements are not used for construction.

Materials or portions thereof that do not meet the above criteria but may inadvertently be used are handled as described in Appendix D and amendments to the PSAR.