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 $(\Delta k/k)/(\Delta P/P)$ at the beginning of life and about -0.3$(\Delta k/k)/(\Delta P/P)$ at 10,000 MWd/T. These values are well within the range required for adequate damping of power and spatial xenon disturbances.

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

For further discussion, see the following sections:

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	

* 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 *
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

* 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 *
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

* 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 *
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

* 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 *
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

* 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 *
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

* 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 *
FUEL STORAGE AND HANDLING	9.1								
Storage Equipment	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	

* 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 *
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	

* 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 *
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.

* 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 *
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			

* 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 *
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	

* 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 *
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

* 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 *
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

* 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 *
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	

* 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 *
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			

* 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 *
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			

* 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 *
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	

* 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 *
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									

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

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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 (or ASME AG-1-1997)[71]	NA	N	
		P	R	NA	Other	MIL-F-51068C (or ASME AG-1-1997)[71]	NA	N	
Adsorber units		P	R	NA	Other	RDT M-16-1T	NA	N	
Coils, Cooling (RBCW) & Heating		R	R	NA	3	AACC CS-8 ARI	I	Y	
Damper-Isolation and Ductwork Connected to RB Recirculation System		P	R	NA	Other	AMCR,SMACNA	NA	N	
Ductwork - Other		P	R	NA	Other	AISI,AWS	NA	N	
Dampers - Other		P	R	NA	Other	SMACNA/AISI	NA	N	
Piping & Valves						AMCA			
Also See Plant Chilled Water System						B31.1			
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 system (PCAMS)		P	C,R	B,D	2	IEEE-344 III-2	I	Y	10, 41
Piping valves forming Containment Penetration Boundary		P	C,R	B	2	IEEE-344 III-2	I	Y	69

* 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 *
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			

* 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 *
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	

* 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 *
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	

* 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 *
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	

* 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 *
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		GE	C,R,T	NA	2	IEEE-279	I	Y	
All portions that must operate to control and safety shut down the reactor to a hot shutdown condition (Electronic modules)									
Cable with safety function		P	C,R,T	NA	2	IEEE-279/383	NA	Y	15
Alternate Rod Injection		P	R	NA	2	10CFR50.62	NA	Y	47
All portions that must operate to control and safely shut down the reactor to a hot shutdown condition (Electronic modules)									
Cable with safety function									

* 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 *
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	

* 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 *
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

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

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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

* 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 *
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	9.3.1								
Compressed Air and Instrument Gas Systems									
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

* 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 *
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	

* 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 *
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									
Isolation, Chilled Water to Primary Containment		P	R	B	2	III-2	I	Y	
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

* 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 *
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	

* 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 *
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	

* 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 *
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	

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

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.

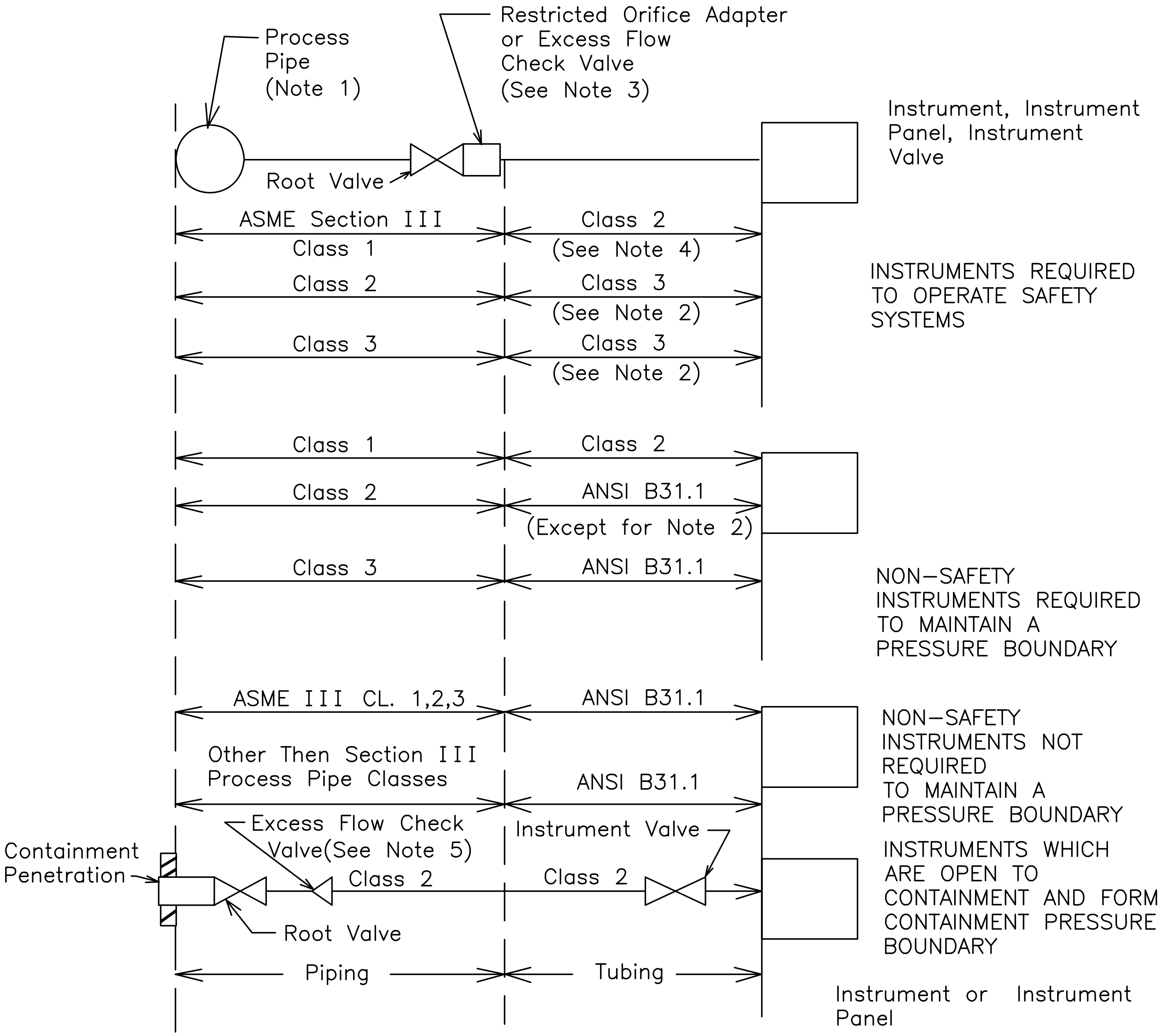
N/A - The equipment shall be constructed in accordance with the quality assurance requirements consistent with accepted practice for steam power plants.

(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 UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
CODE CLASSIFICATION OF PIPING AND VALVES FIGURE 3.2-1



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

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

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:

$q \quad = \quad 0.002558\ V^2$ where,

$q \quad = \quad$ Dynamic pressure in psf

$V \quad = \quad$ 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

$Cp \quad = \quad$ 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.

3.3.2.4 Safety-Related Equipment Not Protected By Reinforced Concrete

SECURITY-RELATED INFORMATION.
TEXT WITHHELD UNDER 10 CFR 2.390

SECURITY-RELATED INFORMATION.
TEXT WITHHELD UNDER 10 CFR 2.390

3.3.3 REFERENCES

3.3-1. H. C. S. Thom, "New Distributions of Extreme Winds in the United States," Journal of the Structural Division, ASCE (July 1968), p. 1787.

3.3-2. "Wind Forces on Structures", ASCE Paper No. 3269, Transactions, Volume 126, Part II (1961), p. 1124.

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

SECURITY-RELATED INFORMATION.
TEXT WITHHELD UNDER 10 CFR 2.390

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.

3.5.1.5 Site Proximity Missiles

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START – HISTORICAL INFORMATION

3.5.1.6 Aircraft Hazards

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3.5.1.6.1 Airport Operations

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3.5.1.6.2 Aircraft Crash Probability

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3.5.1.6.3 Critical Target Area for the Plant

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3.5.1.6.4 Striking Probabilities

The probability that an aircraft might strike the Susquehanna SES, resulting in a potential nuclear safety hazard, is the product of:

the annual traffic (number of aircraft) (3.5.1.6.1)

the crash probability (events per mi^2) (3.5.1.6.2)

the applicable target area (mi^2) (3.5.1.6.3)

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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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SECURITY-RELATED INFORMATION.
TEXT WITHHELD UNDER 10 CFR 2.390

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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3.5-5 "Design of Structures for Missile Impact," BC-TOP-9A, Rev. 2, Bechtel Power Corporation, San Francisco, California (September, 1974).

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3.5-13 Vasallo, F.A., Missile Impact Testing of Reinforced Concrete Panels, Prepared for Bechtel Corp., Calspan Corp., (January, 1975).

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3.5-16 GE Letter "Integral LP Rotor Differences," B.E. Nadler to M.J. Barberetta (October 11, 1985).

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 3.5-4

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.

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.

Table 3.5-5

HISTORICAL INFORMATION

BERWICK AIRPORT MOVEMENT SUMARY

Security-Related Information
Table Withheld Under 10 CFR 2.390

Table 3.5-6

HISTORICAL INFORMATION

PLANT TARGET AREAS

Security-Related Information
Table 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	17.05 ksi	28.3 ksi
d-e	19.54 ksi	30.7 ksi
Shearing Stress		
Zone	Calculated Stress	Stress Limit
a-b	7.60 ksi	11.34 ksi
c-f	10.83 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 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.

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

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

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

FIGURE 3.5-6, Rev. 55

THIS FIGURE HAS BEEN
REPLACED BY DWG.
A-21, Sh. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

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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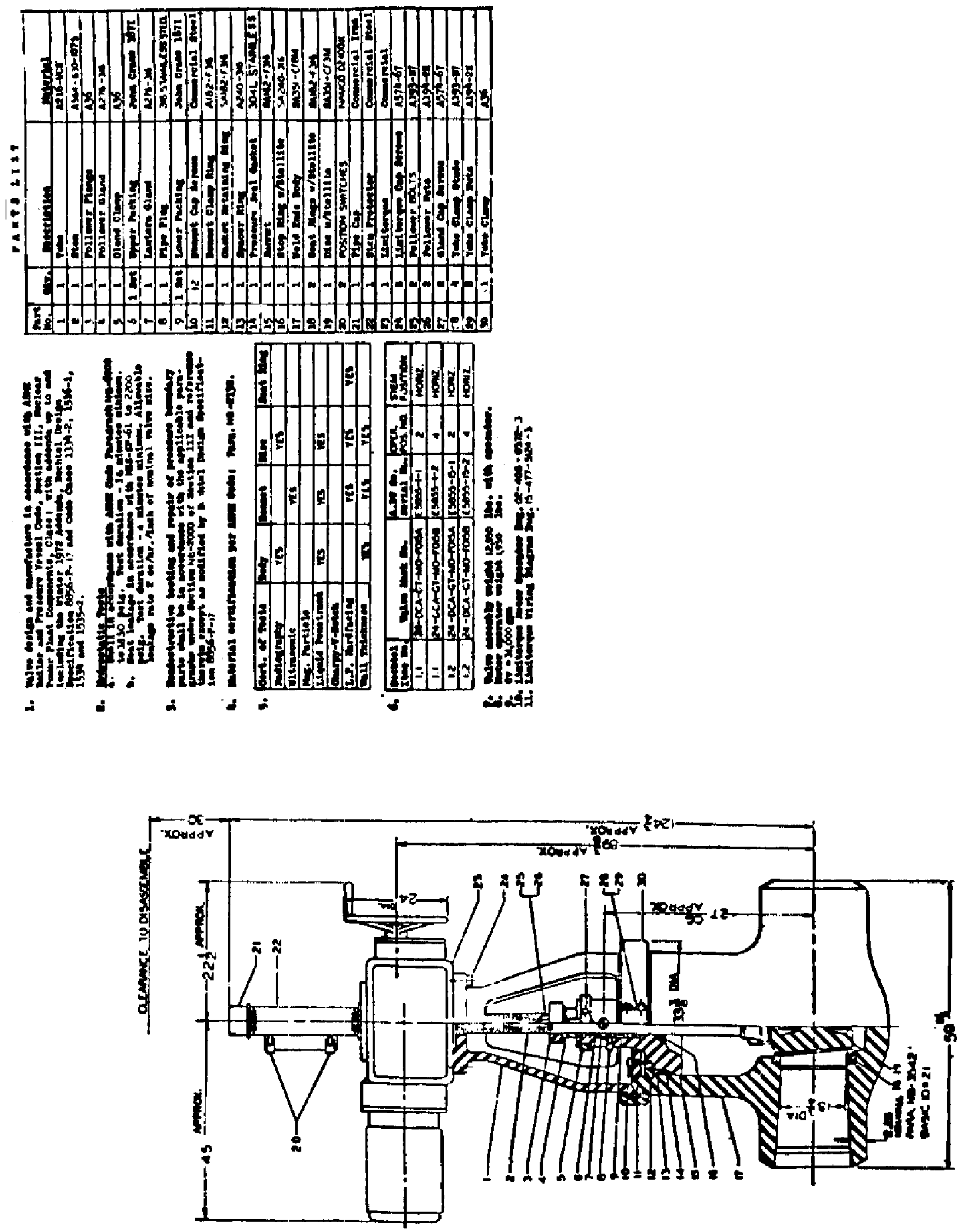
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

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

FIGURE 3.5-8, Rev. 48

AutoCAD Figure 3_5_8.doc



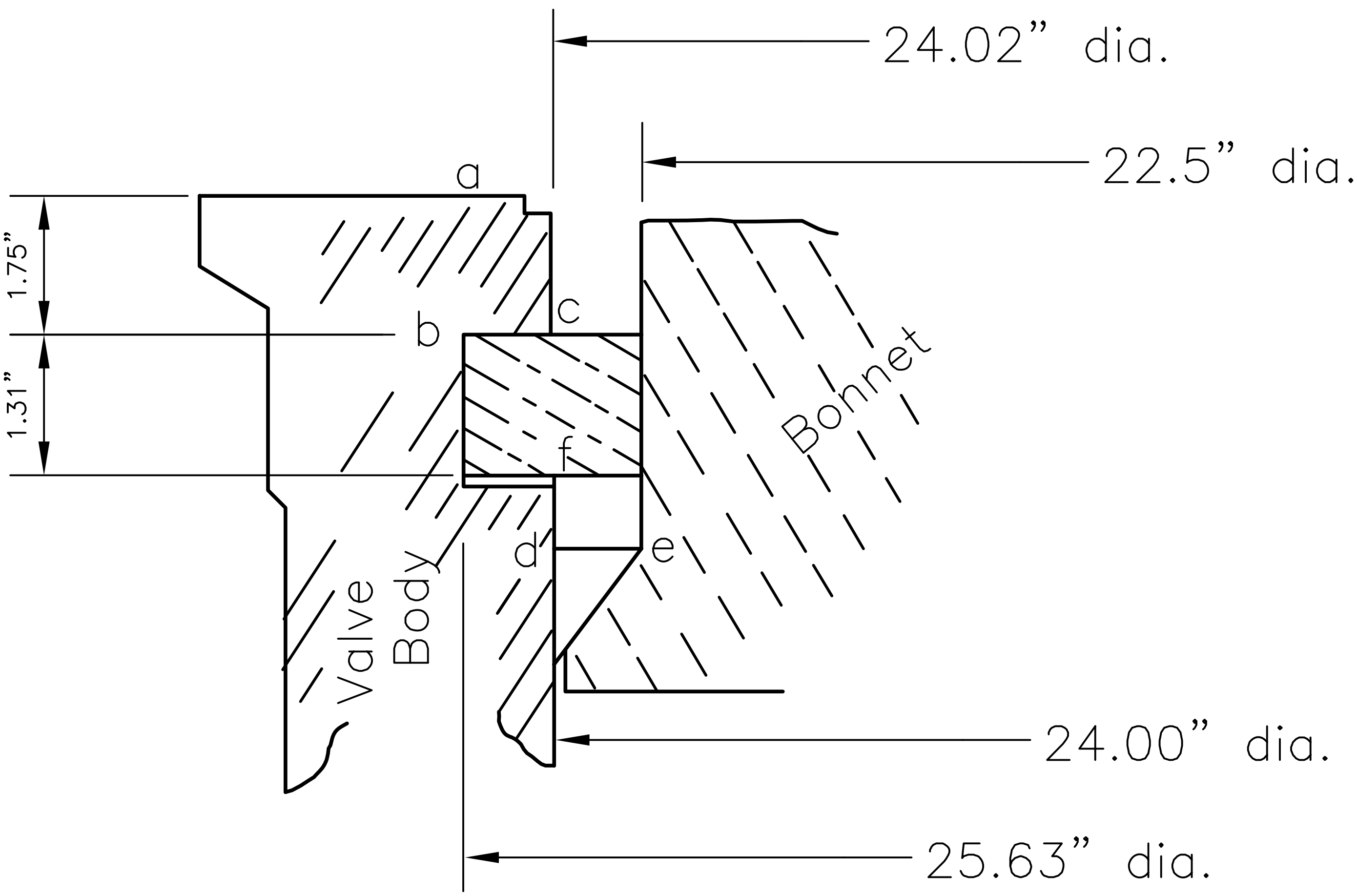
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

TYPICAL 900#
BONNET SEAL
TYPE VALVE

FIGURE 3.5-9, Rev. 47

Auto-Cad Figure Fsar 3_5_9.dwg



FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

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.

Fluid System Piping Other Than That Between Containment Isolation Valves

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:

$$\frac{T}{A_e} = P_E - P_{00} + \frac{G_E^2 \, \overline{V}_E}{g_c}$$

$$R = -\frac{T}{A_e} \, x \, A_{te}$$

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 3.6-1A
MODERATE ENERGY FLUID SYSTEM PIPING
(LOCATED IN SAFETY-RELATED STRUCTURES)

Security-Related Information
Table Withheld Under 10 CFR 2.390

Table 3.6-2

SAFETY COMPONENTS IN CLOSE PROXIMITY TO

HIGH ENERGY FLUID SYSTEM PIPING

(REQUIRING JET IMPINGEMENT PROTECTION) – PRIMARY CONTAINMENT

Security-Related Information
Table Withheld Under 10 CFR 2.390

Table 3.6-3
SAFETY COMPONENTS IN CLOSE PROXIMITY TO
HIGH ENERGY FLUID SYSTEM PIPING
(REQUIRING JET IMPINGEMENT PROTECTION) – REACTOR BUILDING

Security-Related Information
Table 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 PDA	No. of Bars NSC	Load (kips) PDA	Load (kips) NSC	Restraint Deflection (in.) PDA	Restraint Deflection (in.) NSC	% of Design Restraint Deflection PDA	% of Design Restraint Deflection NSC	Pipe Deflection (in.) PDS	Pipe Deflection (in.) 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 3.6-6a

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 3.6-7

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 3.6-7a

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 3.6-8

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 3.6-8a

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 3.6-9

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 3.6-9a

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 3.6-10

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 3.6-10a

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 3.6-11

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 3.6-11a

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 3.6-12a.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

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 3.6-12b.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

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 3.6-12c.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.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

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.

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

Security-Related Information
Figure Withheld Under 10 CFR 2.390

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
MAIN STEAM LINE 'B'
FIGURE 3.6-1B

Security-Related Information
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

Security-Related Information
Figure Withheld Under 10 CFR 2.390

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
MAIN STEAM LINE 'D'
FIGURE 3.6-1D

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

Security-Related Information
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

Security-Related Information
Figure Withheld Under 10 CFR 2.390

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
RCIC STEAM SUPPLY
FIGURE 3.6-4

Security-Related Information
Figure Withheld Under 10 CFR 2.390

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
CORE SPRAY
FIGURE 3.6-5

Security-Related Information
Figure Withheld Under 10 CFR 2.390

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
RHR SUPPLY
FIGURE 3.6-6

Security-Related Information
Figure Withheld Under 10 CFR 2.390

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
RHR RETURN LOOP 'A' FIGURE 3.6-7

Security-Related Information
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

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

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

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

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

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

Security-Related Information
Figure Withheld Under 10 CFR 2.390

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
REACTOR VESSEL HEAD VENT
FIGURE 3.6-8B

Security-Related Information
Figure Withheld Under 10 CFR 2.390

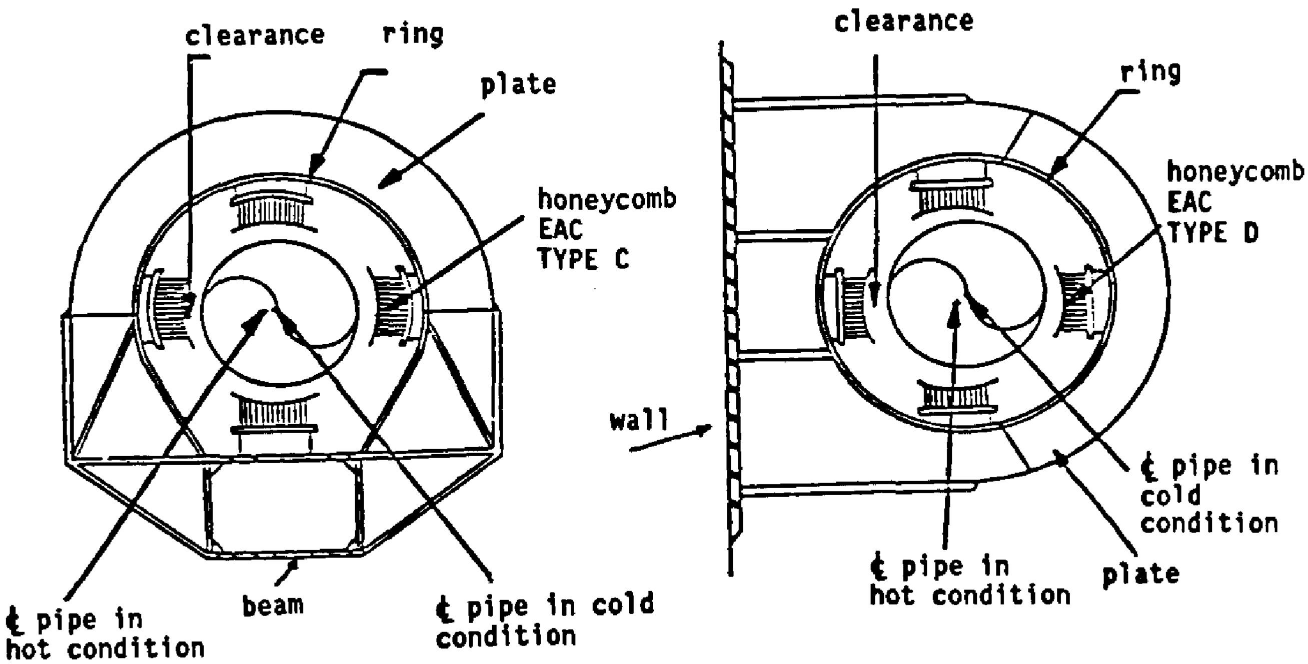
SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
HEAD SPRAY
FIGURE 3.6-8C

Security-Related Information
Figure Withheld Under 10 CFR 2.390

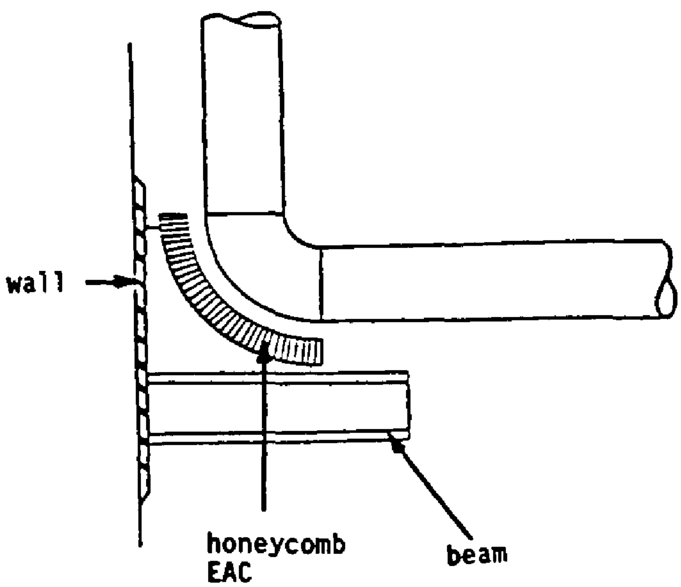
SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
STANDBY LIQUID CONTROL
FIGURE 3.6-8D

Security-Related Information
Figure Withheld Under 10 CFR 2.390

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
MSIV DRAINS
FIGURE 3.6-8E



a) Pipe whip restraints with honeycomb (EAC) and clearance all around the pipe



b) Bumpers

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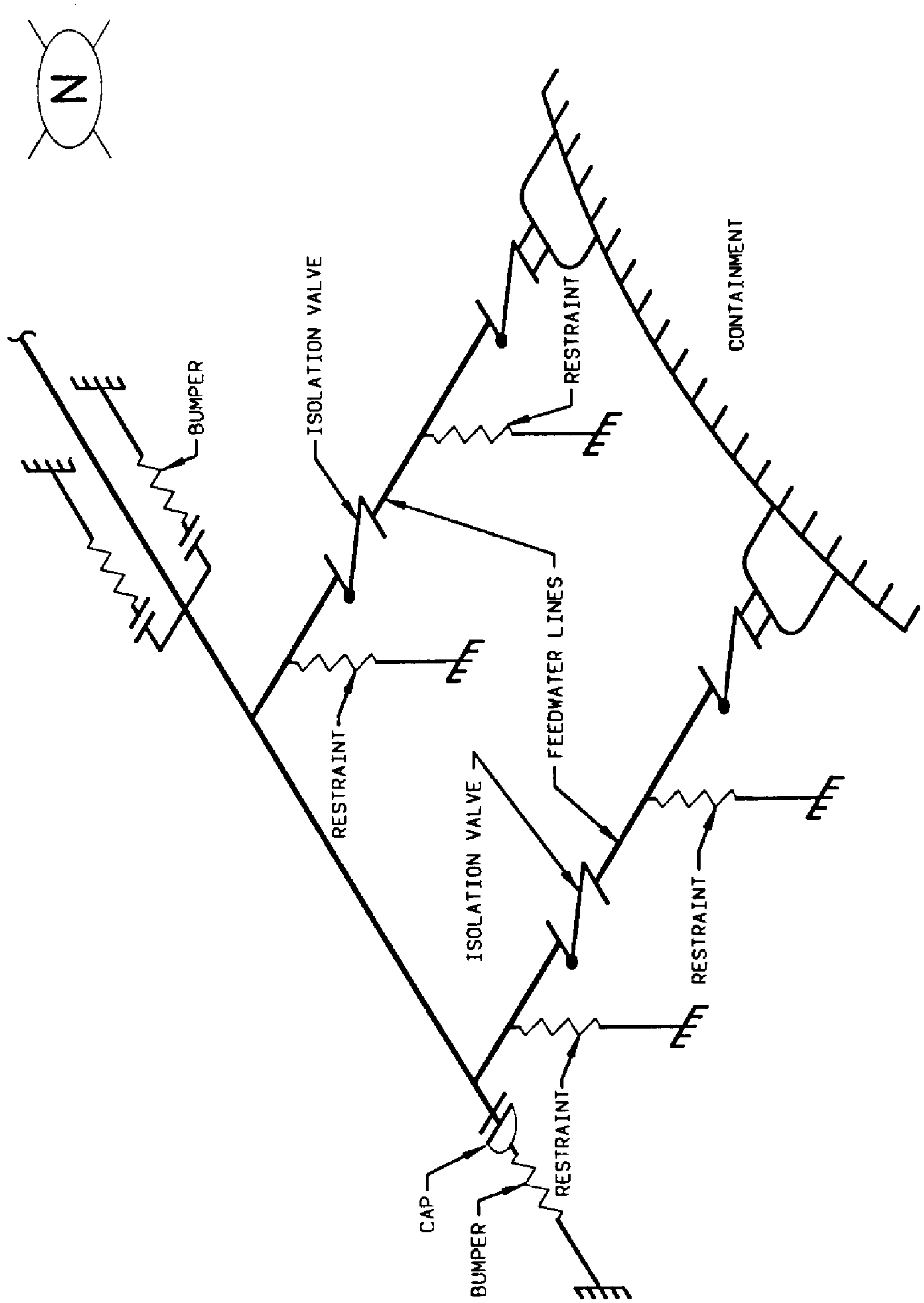
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

TYPICAL PIPE
WHIP RESTRAINTS

FIGURE 3.6-10, Rev. 47

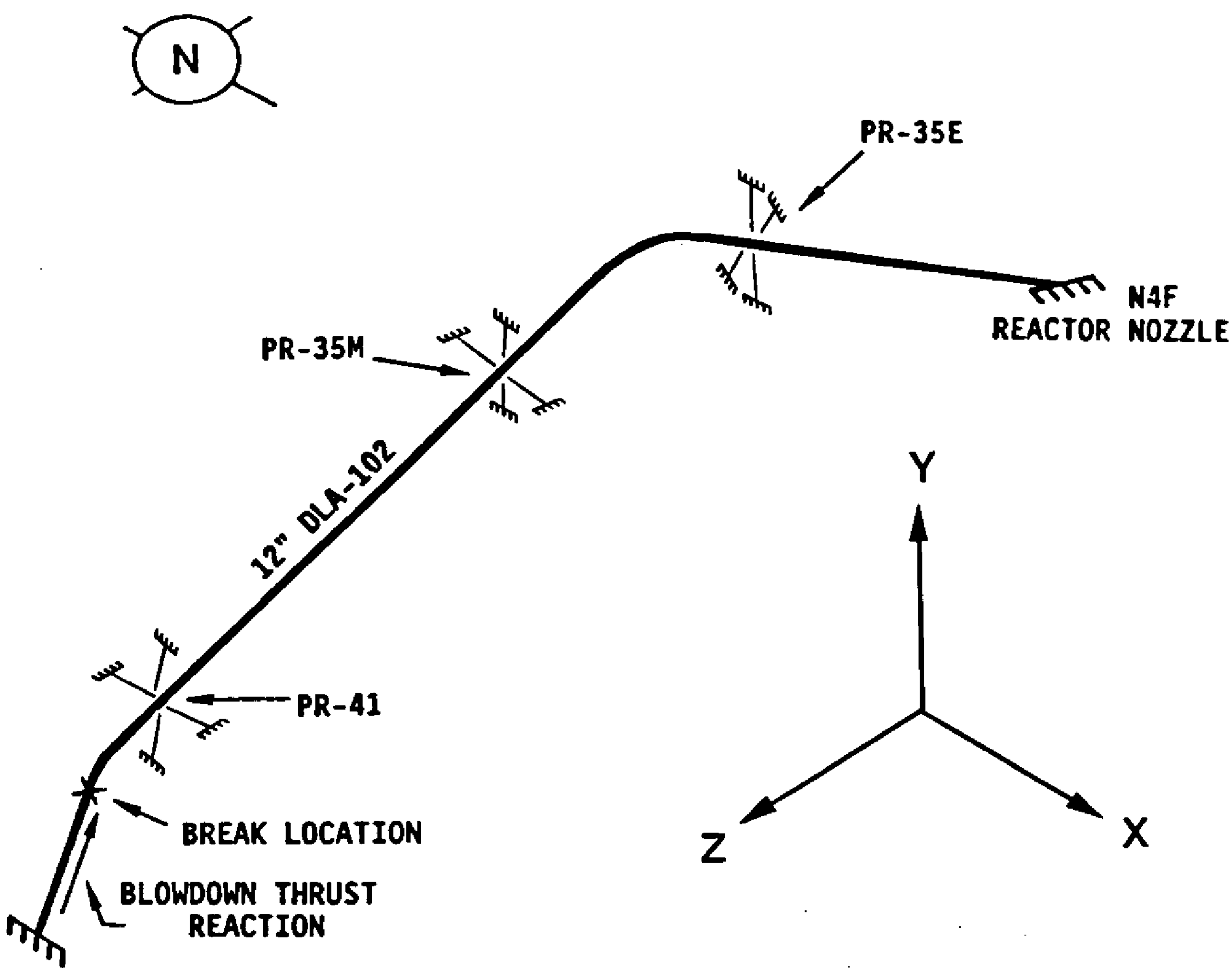
Auto-Cad Figure Fsar 3_6_10.dwg



FSAR REV.65

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



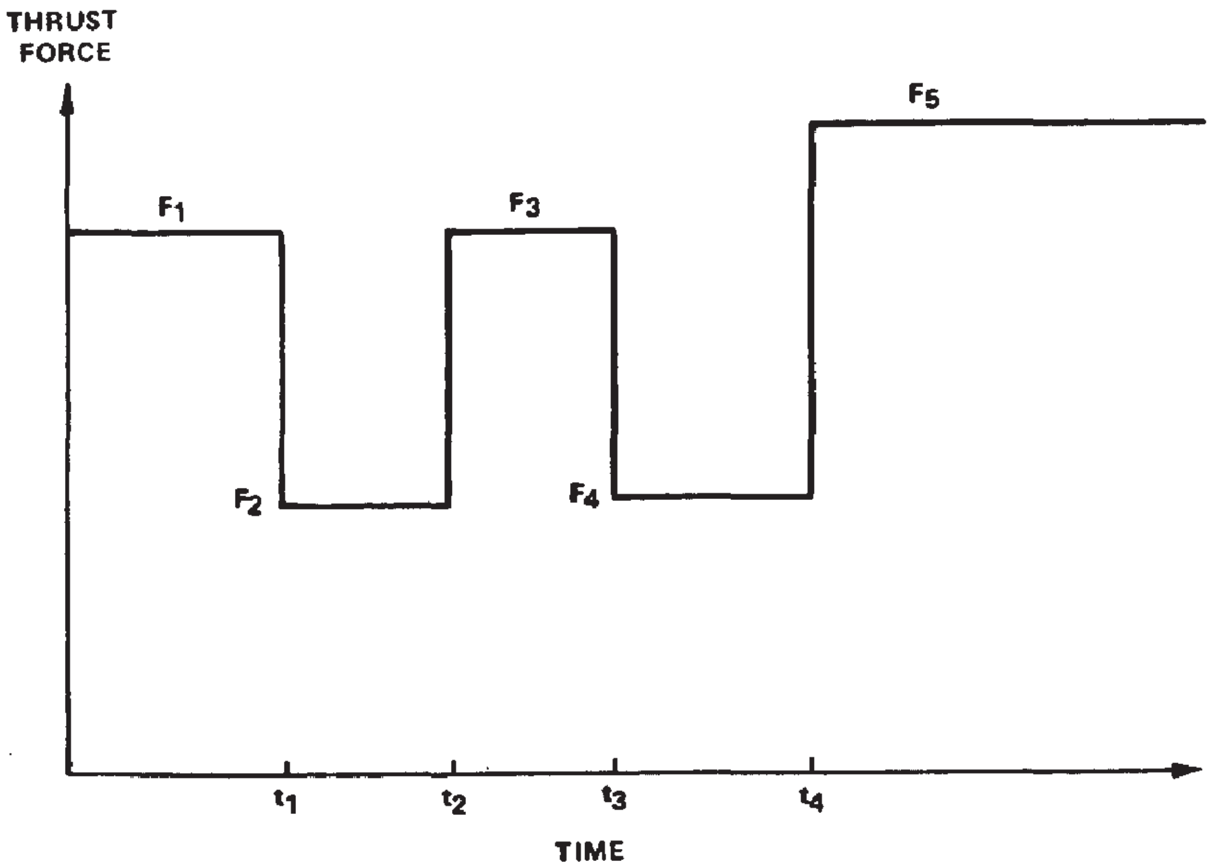
FSAR REV.65

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



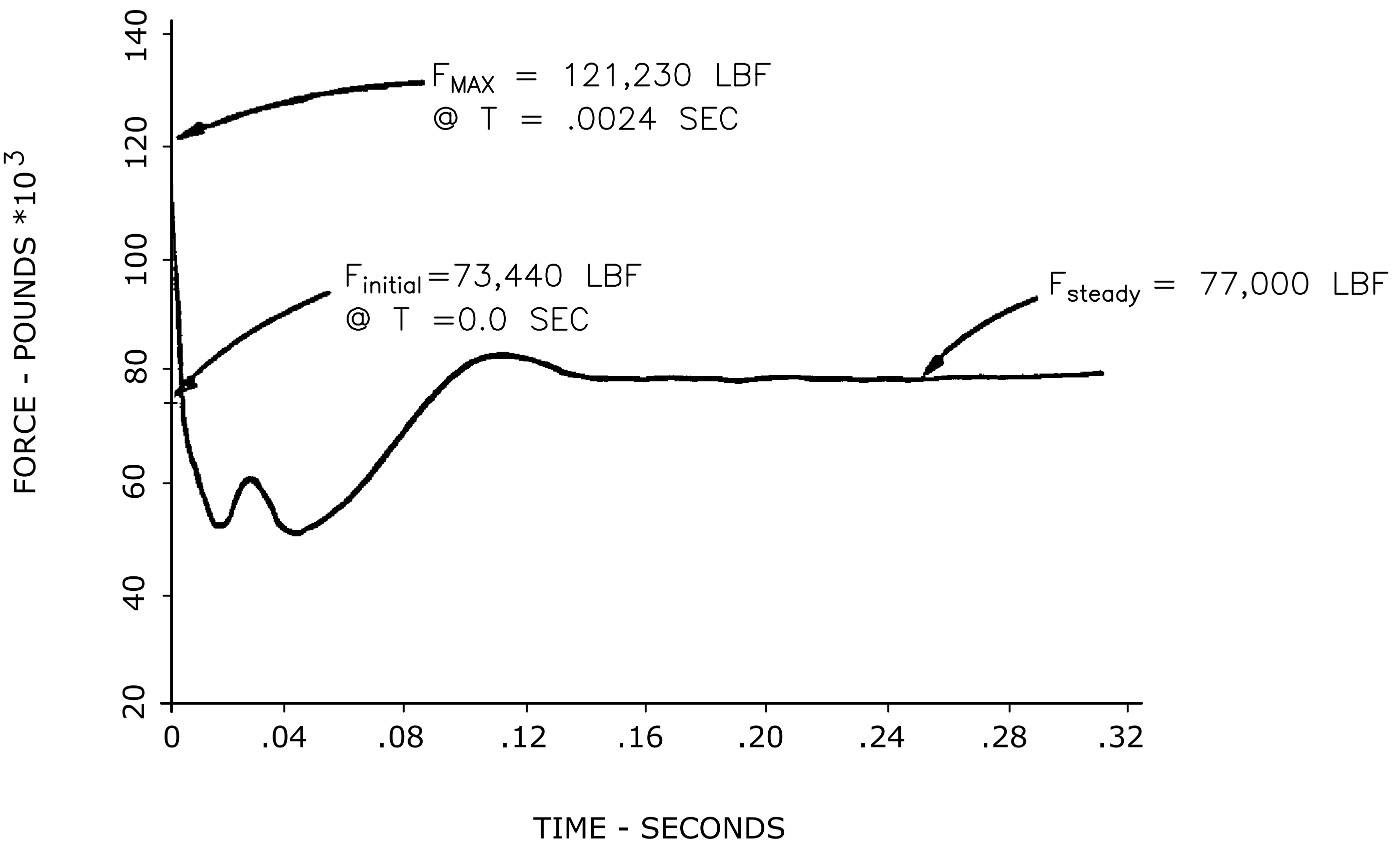
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

FORCING FUNCTIONS MODEL
ASSOCIATED WITH PIPE WHIP
DYNAMIC ANALYSIS

FIGURE 3.6-12, Rev. 47

Auto-Cad Figure Fsar 3_6_12.dwg



HPCI INSIDE CONTATINMENT
PIPE BREAK FORCING FUNCITON
REACTOR SIDE

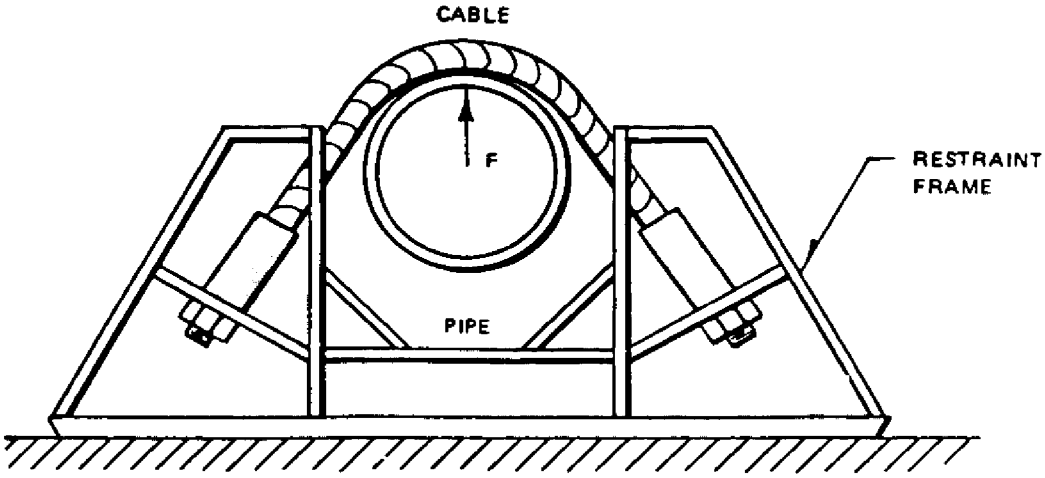
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

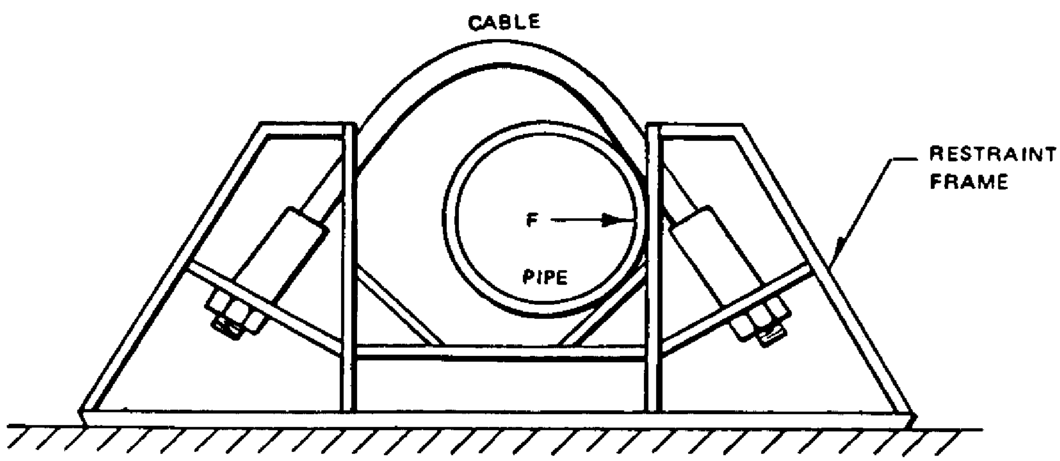
FORCING FUNCTIONS MODEL
ASSOCIATED WITH PIPE WHIP
DYNAMIC ANALYSIS

FIGURE 3.6-12A, Rev. 47

Auto-Cad Figure Fsar 3_6_12A.dwg



(a) LOAD APPLIED PERPENDICULAR TO RESTRAINT BASE AGAINST CABLES



(b) LOAD APPLIED PARALLEL TO FRAME BASE AGAINST ONE SIDE OF RESTRAINT FRAME

FSAR REV.65

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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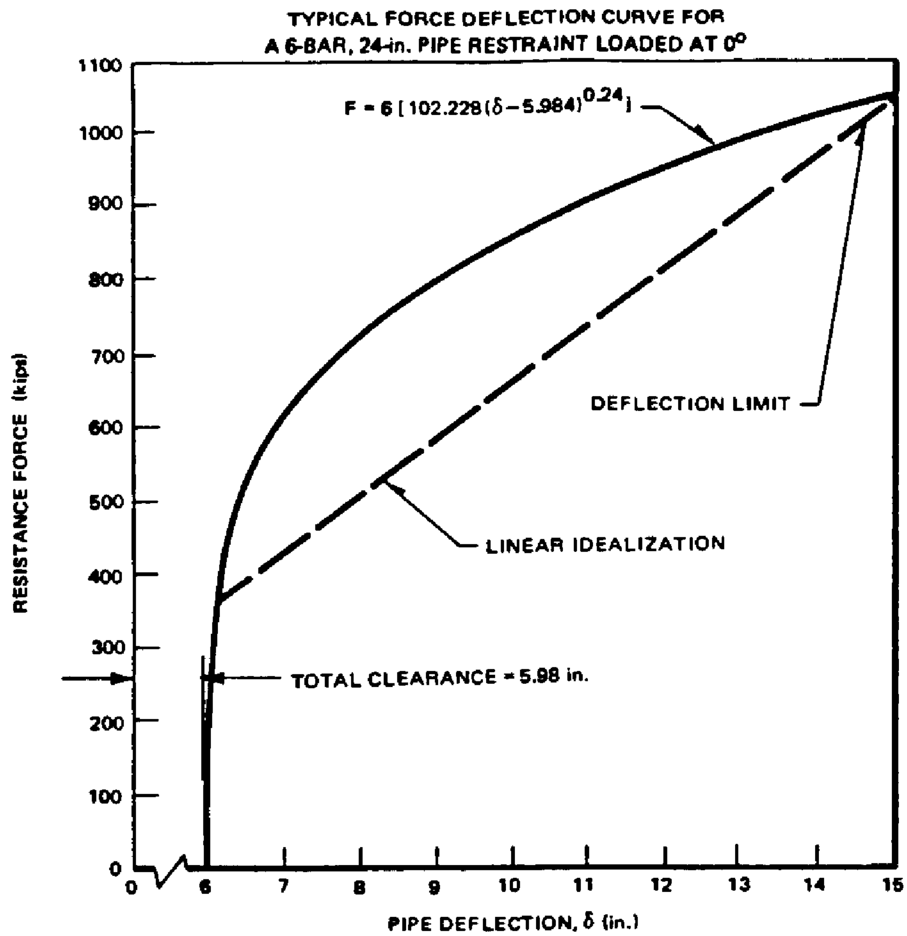
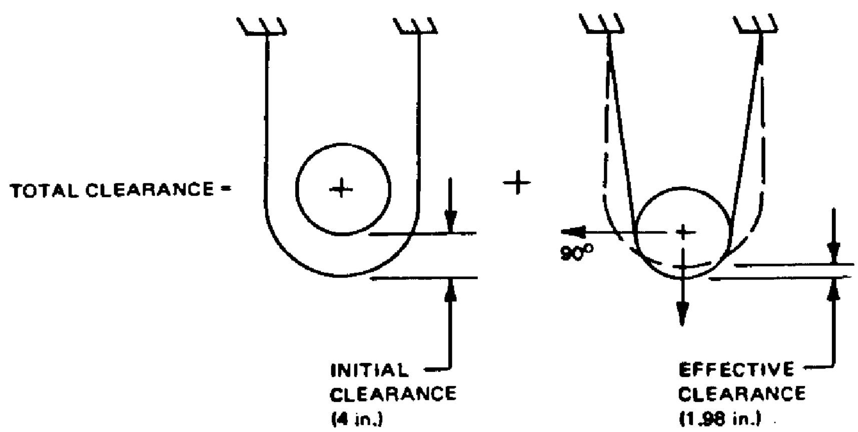
SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
RECIRCULATION SYSTEM POSTULATED BREAK LOCATIONS AND RESTRAINT LOCATIONS (LOOP A AND B SAME, UNLESS OTHERWISE SPECIFIED) FIGURE 3.6-14





FSAR REV.65

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

Security-Related Information

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

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

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

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

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}

Results (BDIDs Closed):

Peak Pressure:	1.95 psig
Peak Temperature:	295.6°F

Results (BDIDs Open, HPCI Steam Supply Break):

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

Results (BDIDs Open, RCIC Steam Supply Break, 4”-DBB-109):

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

ANALYSIS FOR HPCI PUMP ROOM (UNIT 1)

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:

$$\left(\frac{1}{g_c\,A(x)}\right)\frac{\partial}{\partial t}\left[A(x)\ G(x,t)\right] = -\left(\frac{1}{g_c\,A(x)}\right)\frac{\partial}{\partial x}\left[\frac{A(x)\,G^2(x,t)}{\rho(x,t)}\right] - \frac{\partial \mathrm{p(x,t)}}{\partial \mathrm{x}} - \frac{1}{A(x)}\frac{\partial F(x,t)}{\partial x} \qquad (1)$$

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:

$$\frac{1}{g_c}\int_{x_1}^{x_2}\frac{1}{\mathrm{A(x)}}\frac{\partial}{\partial t}\left[A(x)\,G(x,t)\right]dx = \frac{1}{\mathrm{g_c}}\frac{\partial}{\partial t}W(t)\int_{x_1}^{x_2}\frac{\mathrm{dx}}{\mathrm{A(x)}} = \frac{1}{\mathrm{g_c}}\frac{\mathrm{dW(t)}}{\mathrm{dt}}\sum_{\mathrm{i}}\left(\mathrm{L_i}/\mathrm{A_i}\right) \qquad (1a)$$

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:

$$-\frac{1}{g_c}\int_{x_1}^{x_2}\frac{1}{A(x)}\frac{\partial}{\partial x}\left[\frac{A(x)\,G^2(x,t)}{\rho(x,t)}\right]dx = -\frac{W^2(t)}{g_c\rho}\int_{x_1}^{x_2}\left(\frac{1}{A(x)}\right)\frac{d}{dx}\left(\frac{1}{A(x)}\right)dx = -\frac{W^2(t)}{2g_c\rho}\left[\frac{1}{A_2^2}-\frac{1}{A_1^2}\right] \quad (1b)$$

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 1b/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:

$$\frac{1}{g_c} \frac{dW(t)}{dt} \sum_i \left(\frac{L_i}{A_i} \right) = \mathrm{P}_1^* - \mathrm{P}_2^* - \frac{\mathrm{KiW}^2(t)}{2g_c \rho A_T^2} \qquad (1g)$$

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:

$$\frac{Ki}{2\rho g_c A_T^2} W^2(t) + \frac{\sum_i \left(L_i \big/ A_i \right)}{g_c \Delta t} W(t) - \left\{ \frac{\sum_i \left(L_i \big/ A_i \right)}{g_c \Delta t} W(t - \Delta t) + P_1^* - P_2^* \right\} = 0 \qquad (3)$$

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 $O \leq T \leq L_1/C$ 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. $At = L_1 / C \leq T \leq \dfrac{2*(1.1*L_1)}{0.9*C}\sec$ (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 $\dfrac{fl_1}{D}$.

3. $At\,\dfrac{2*(1.1*L_1)}{0.9*C} \leq T \leq 1.0\sec$ (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 $\dfrac{fl_1}{D}$.

(B) Calculation of mass and energy release rates from the reverse direction.

1. At $O \leq T \leq L_4/C$ 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. $At\, L_4 / C \leq T \leq \dfrac{2*(L_3 + L_4)}{C}\sec$ (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 $f\,\dfrac{(L_3 + L_4)}{D}$

3. $At\,\dfrac{2*(L_3 + L_4)}{C} \leq T \leq 1.00\sec$ (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

	$\frac{G_m(i-1)}{G_{comp}}$	$G_{mi}(\rho)$	$G_{mi}(\rho_{au})$	$G_{mi}(\rho_c)$	$\frac{G_{mi}(\rho_{au})}{G_{mi}(\rho)}$
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

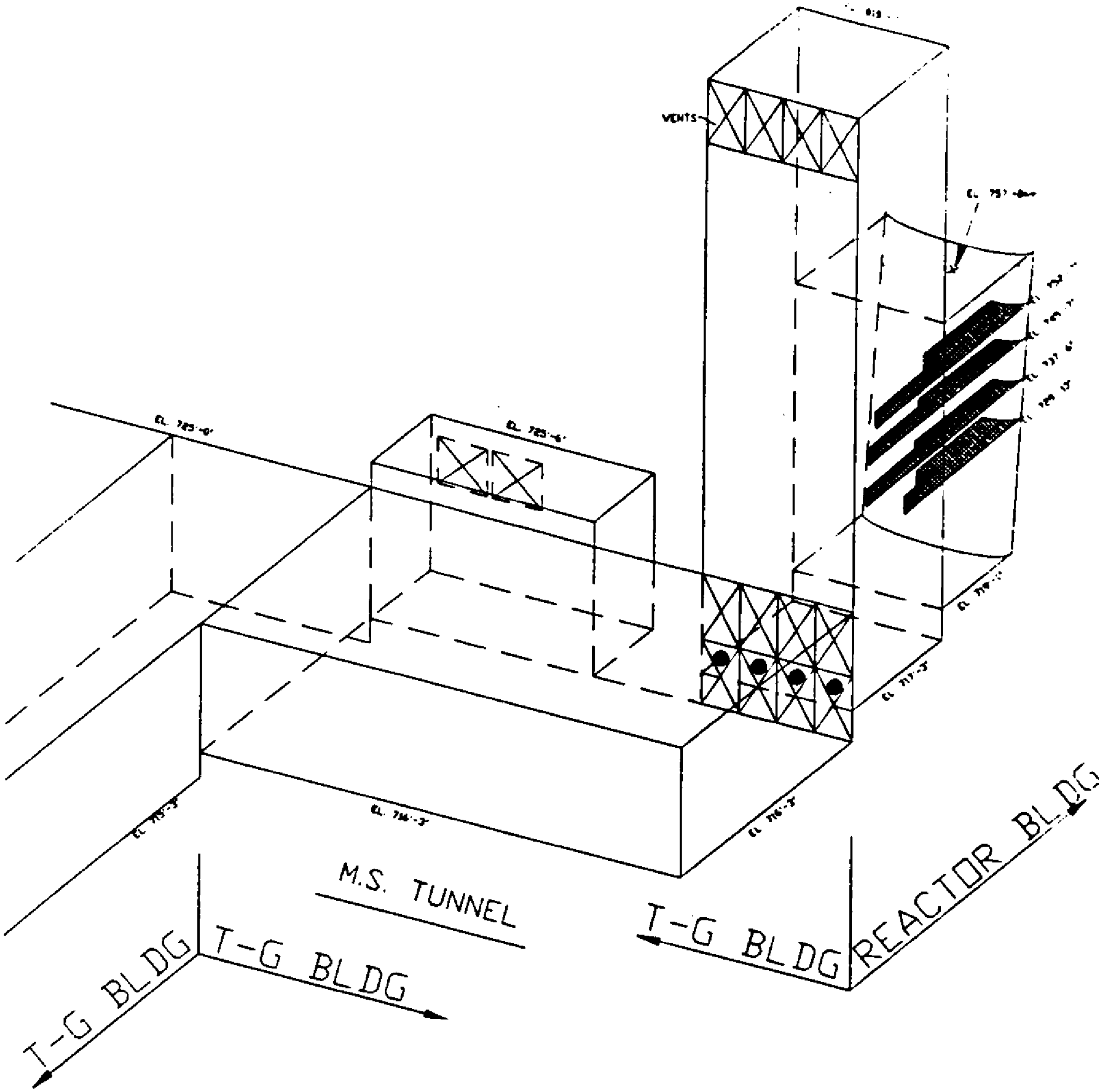
TABLE 3.6A-6

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

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
CASE A MSLB IN REACTOR BUILDING FIGURE 3.6A-1



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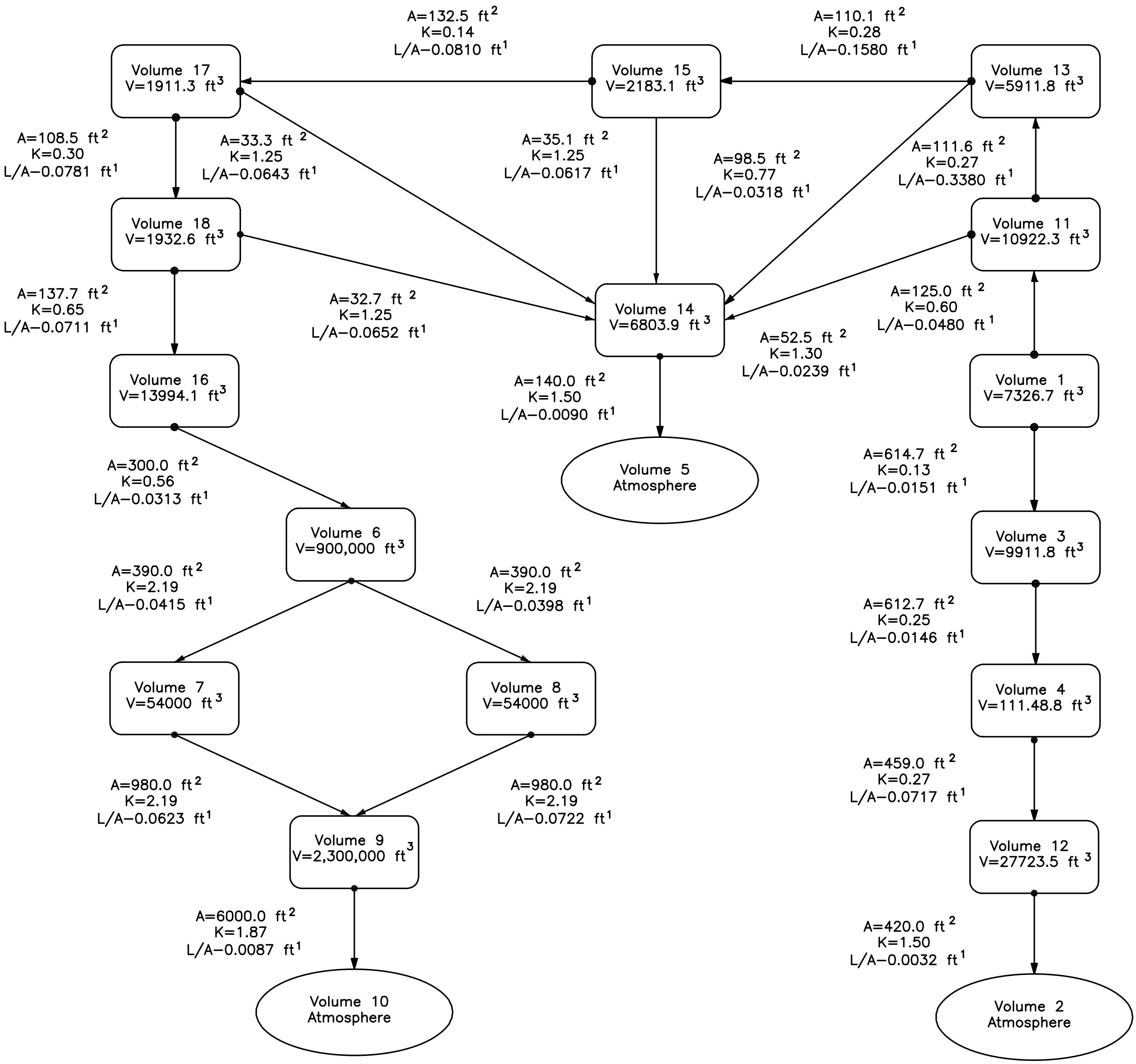
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

PANEL AND PLATFORM
LOCATIONS

FIGURE 3.6A-2, Rev. 47

Auto-Cad Figure Fsar 3_6A_2.dwg



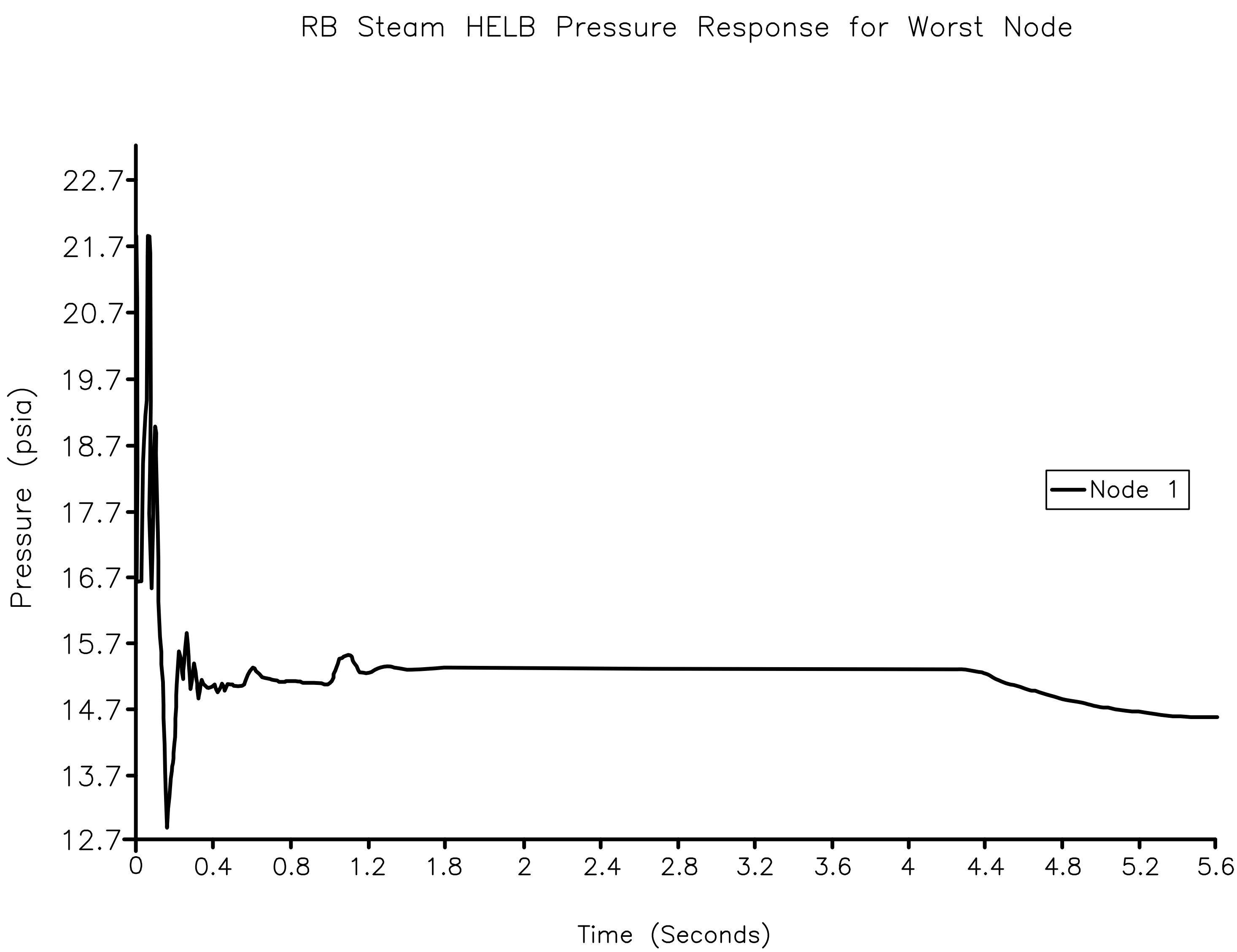
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

CASE A VOLUME FLOWS

FIGURE 3.6A-3, Rev. 48

Auto-Cad Figure Fsar 3_6A_3.dwg



FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

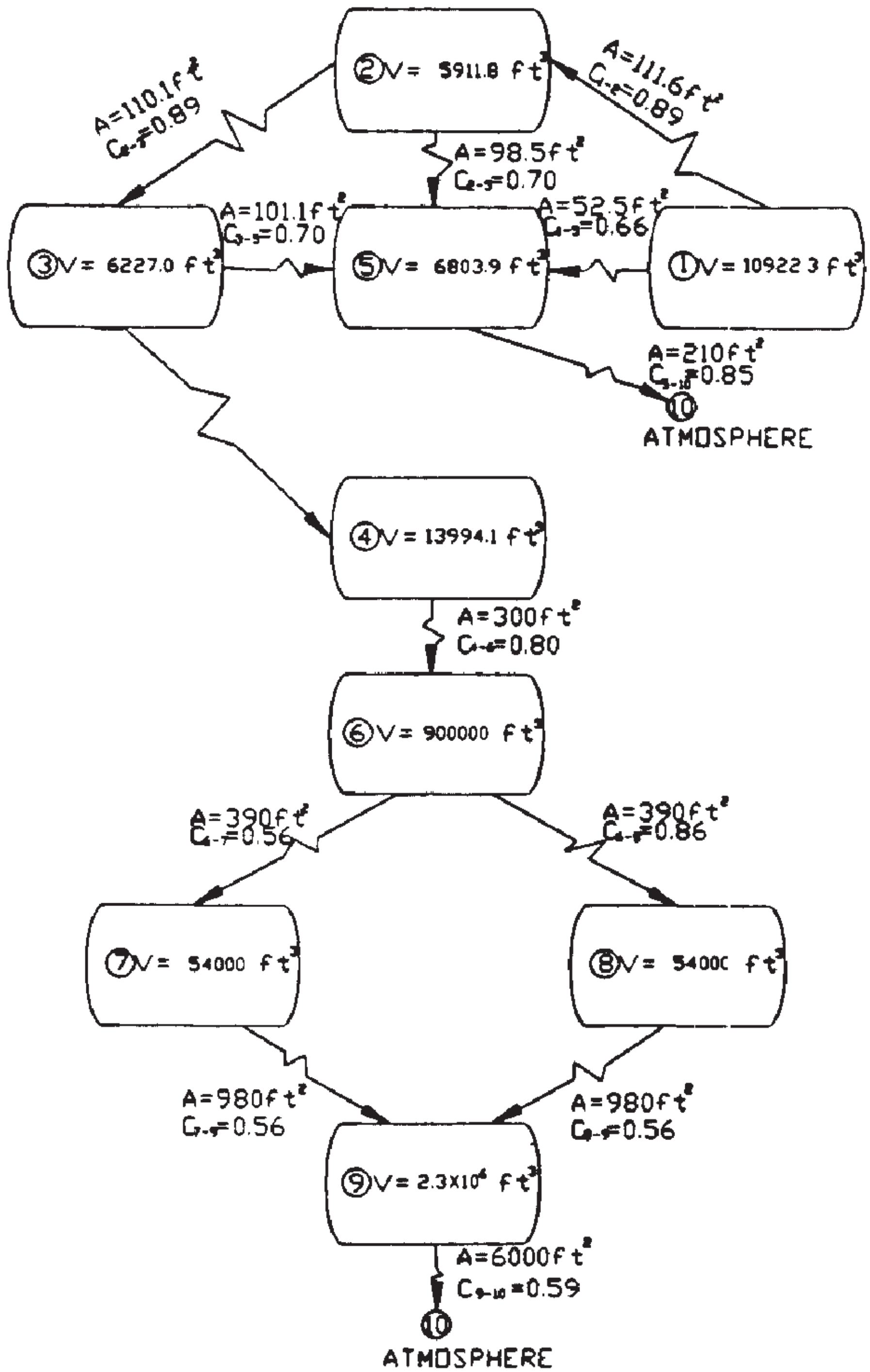
CASE A PRESSURE TRANSIENT

FIGURE 3.6A-4, Rev. 48

Auto-Cad Figure Fsar 3_6A_4.dwg

Security-Related Information
Figure Withheld Under 10 CFR 2.390

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
CASE B MSLB IN TURBINE BUILDING
FIGURE 3.6A-5



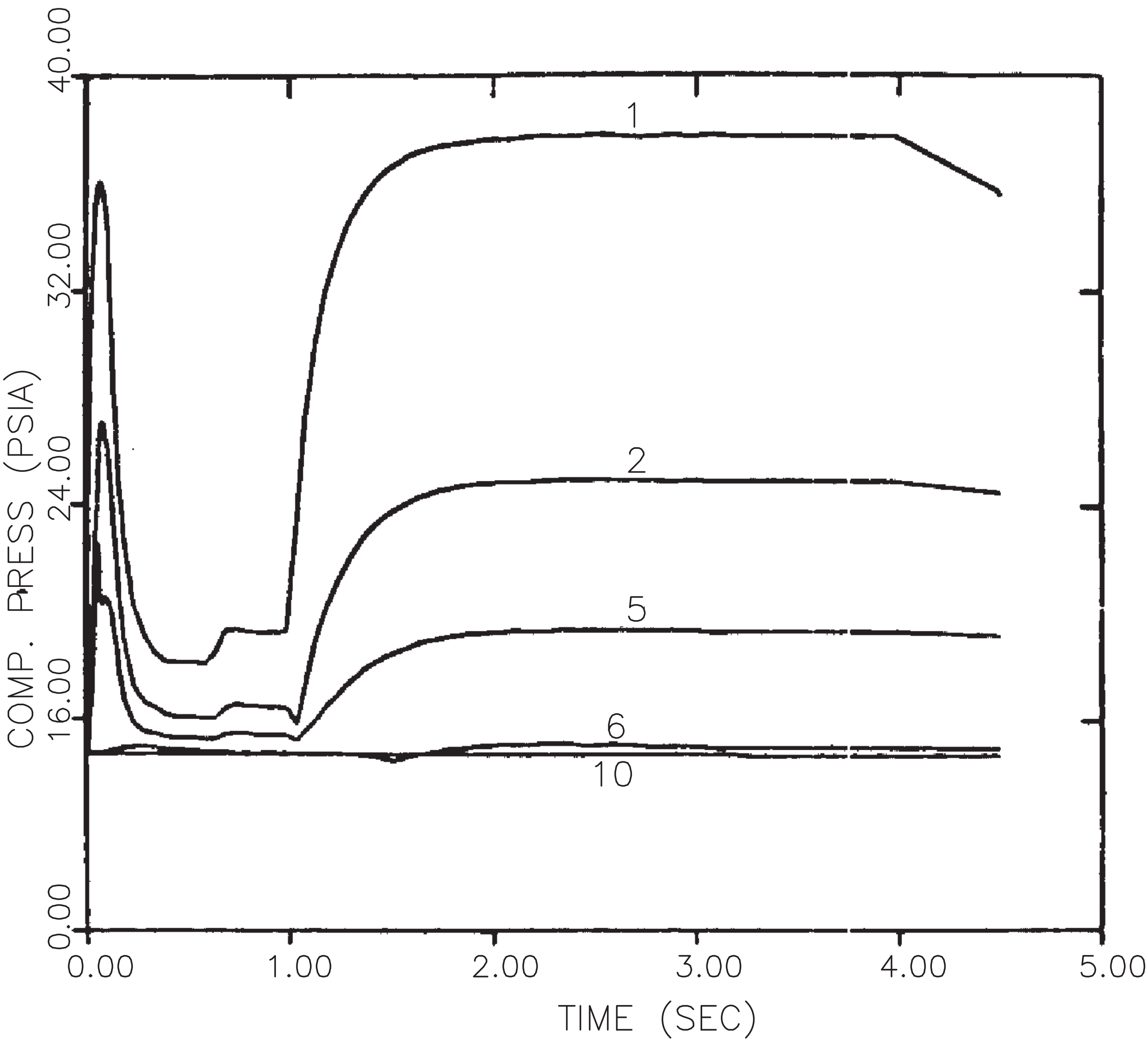
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

CASE B VOLUME FLOWS

FIGURE 3.6A-6, Rev. 47

Auto-Cad Figure Fsar 3_6A_6.dwg



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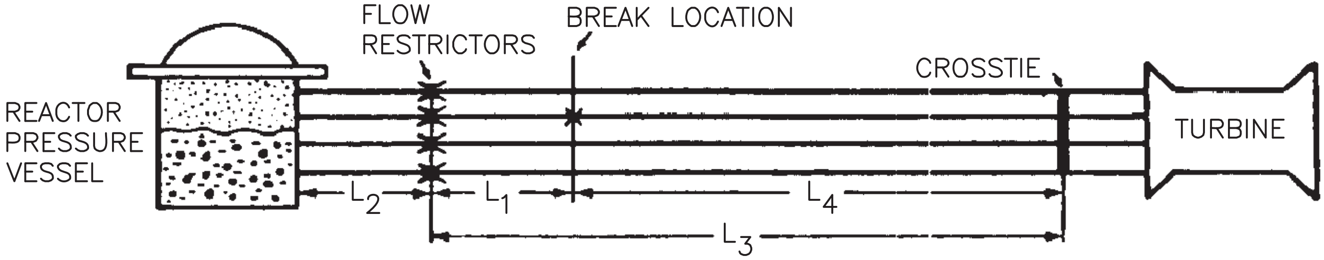
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

CASE B PRESSURE TRANSIENT

FIGURE 3.6A-7, Rev. 47

Auto-Cad Figure Fsar 3_6A_7.dwg



FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

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

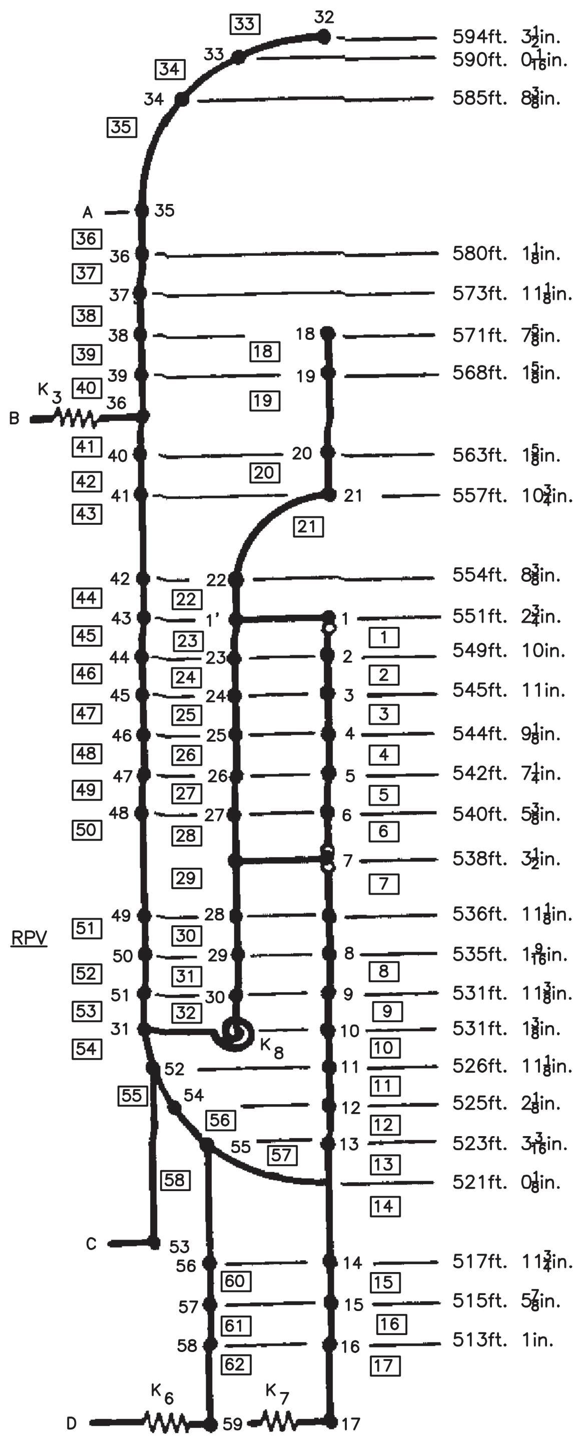
Other values may be used if they are indicated to be reliable by experiment or study.

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-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



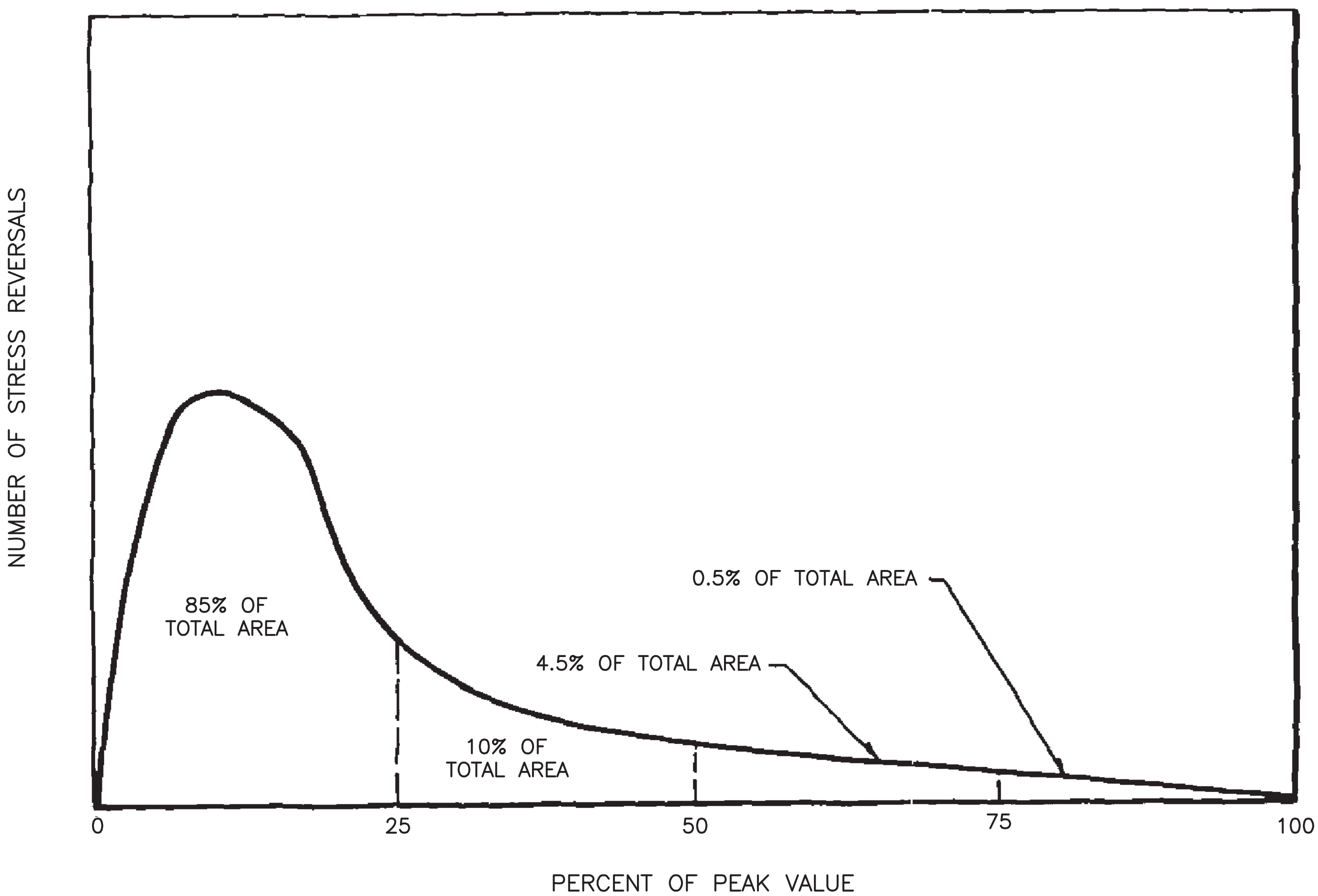
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

REACTOR PRESSURE VESSEL
AND INTERNAL SEISMIC MODEL

FIGURE 3.7A-1, Rev. 47

Auto-Cad Figure Fsar 3_7A_1.dwg



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DENSITY OF STRESS REVERSALS

FIGURE 3.7A-2, Rev. 47

Auto-Cad Figure Fsar 3_7A_2.dwg

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 ± 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:

$I_e = (M_{cr}/M_a)^3\, I_g + (1 - (M_{cr}/M_a)^3)\, I_{cr}$

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 = $\frac{fr\ I_g}{Y_t}$

fr = modulus of rupture for masonry = 50 psi

modulus of rupture for concrete = $6\sqrt{f'_c}$ 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

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* 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.

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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

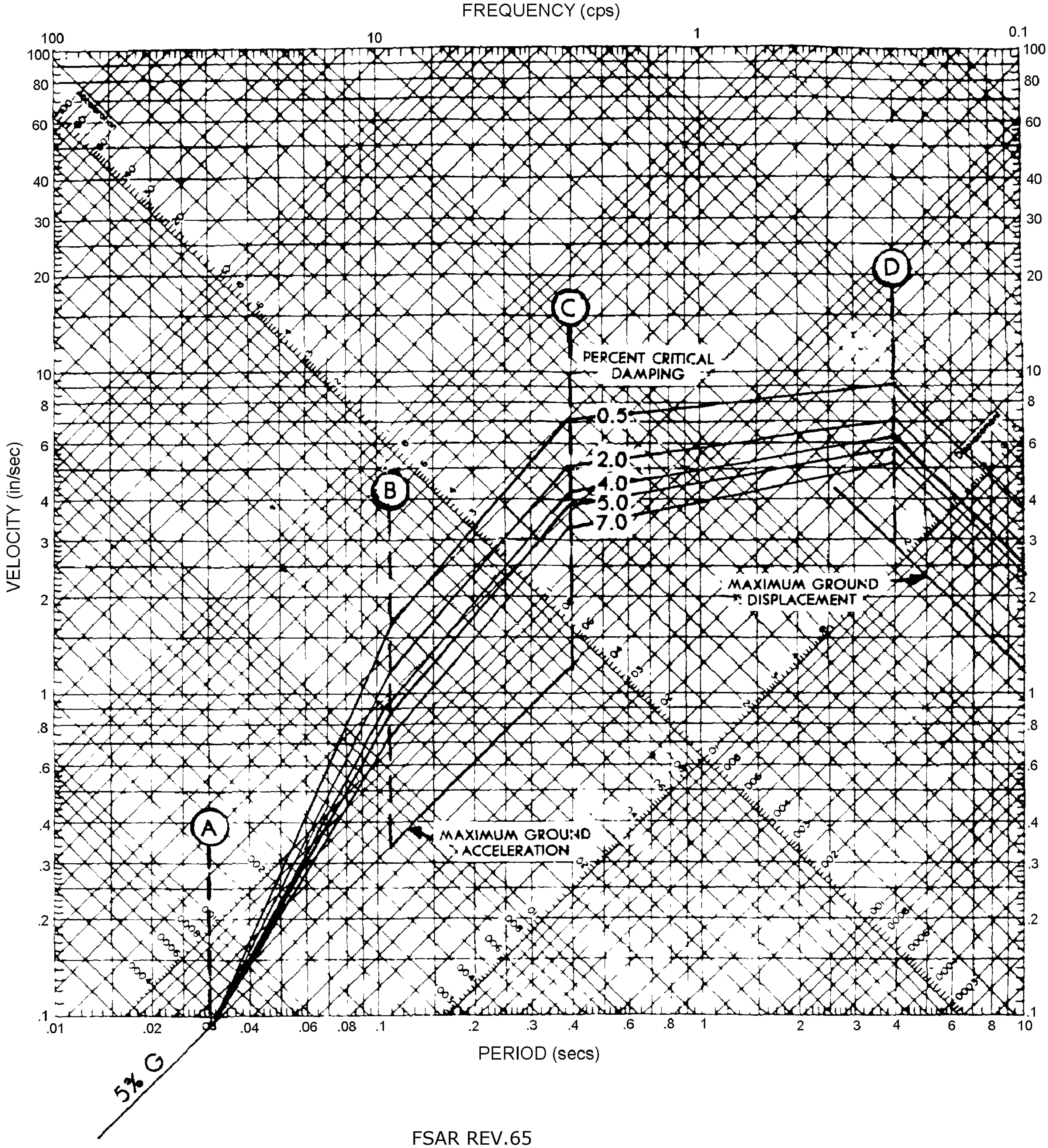
(SEE FIGURE 3.7B-126)

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



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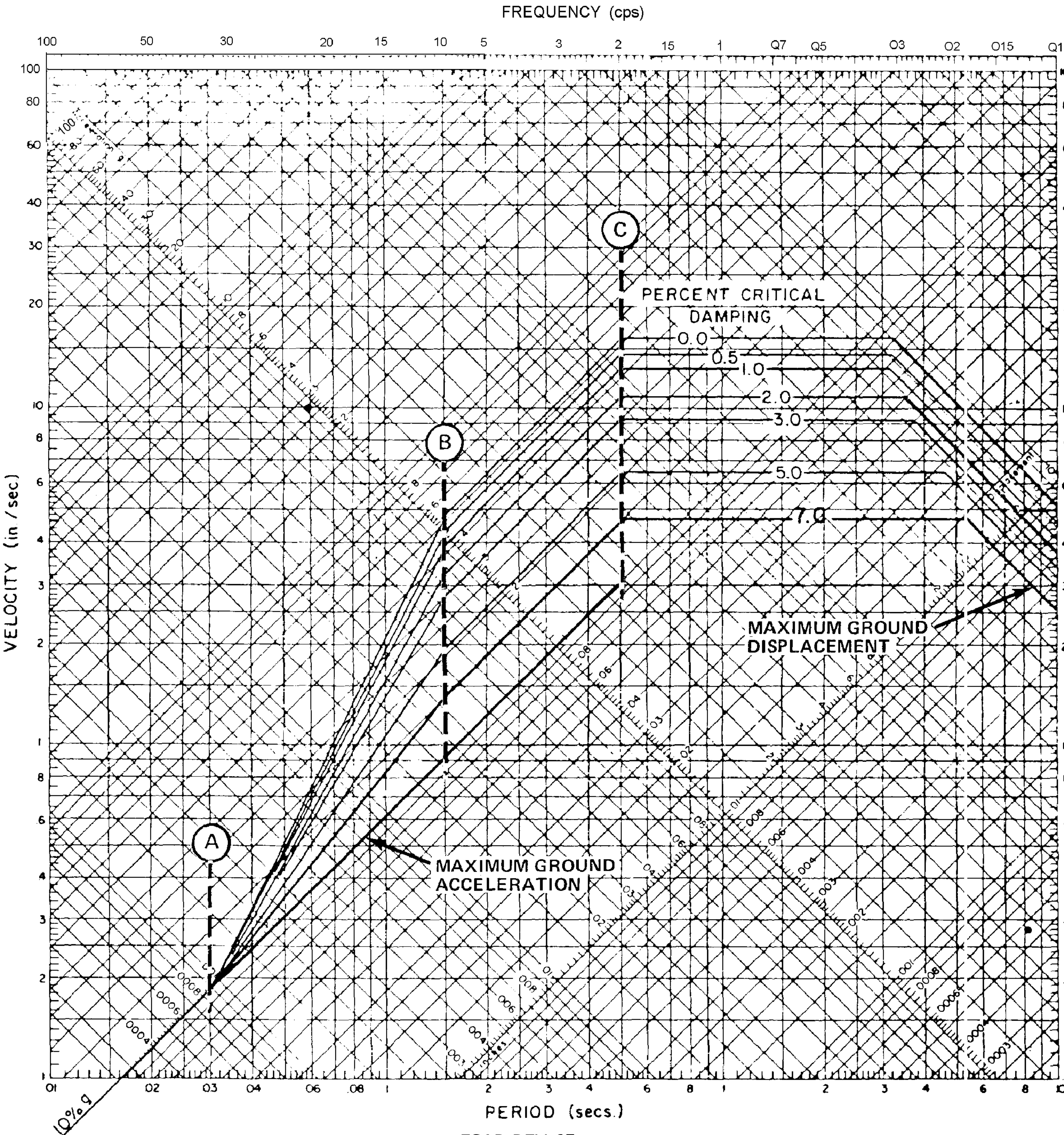
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

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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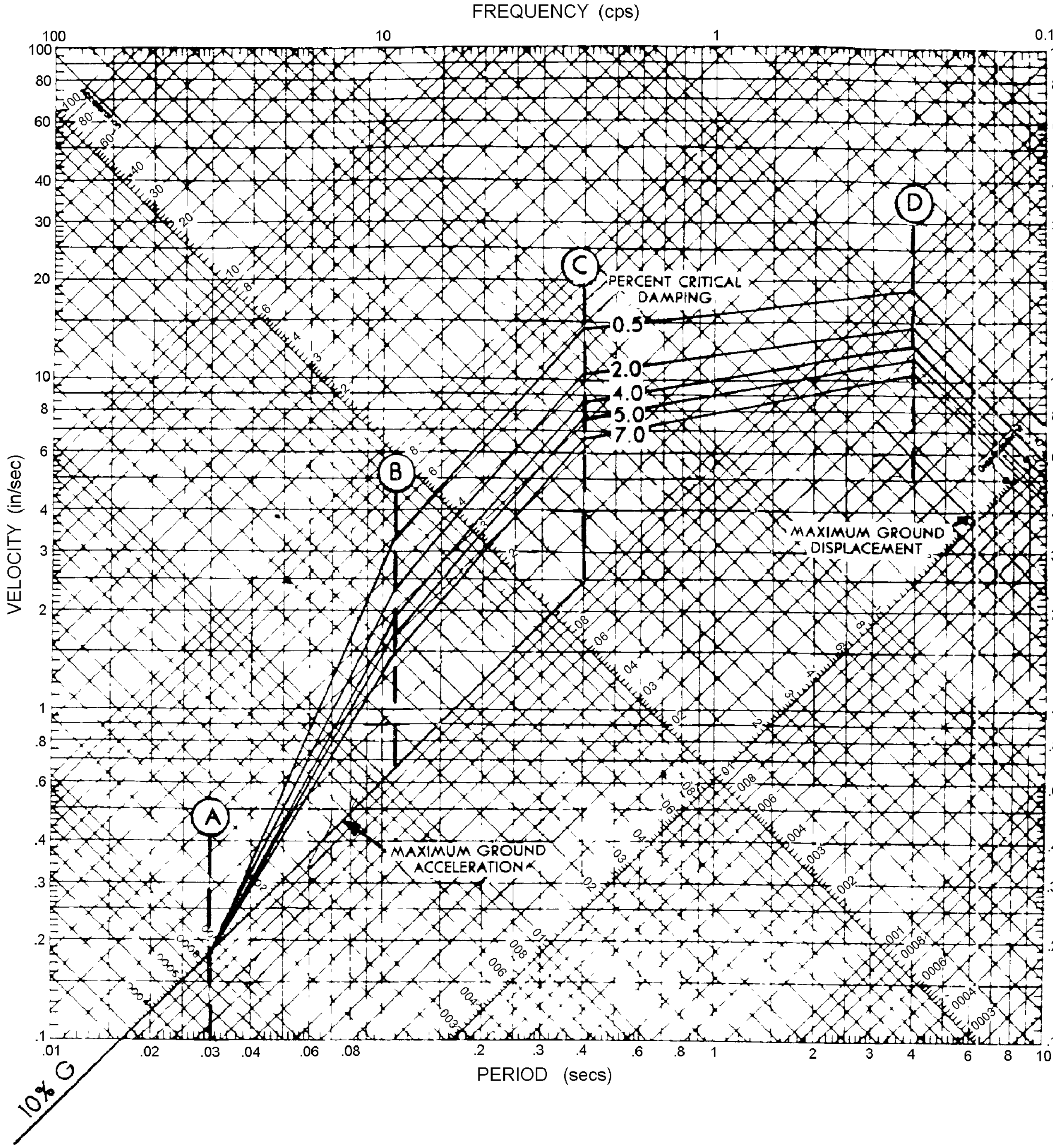
DESIGN RESPONSE SPECTRA
SAFE SHUTDOWN EARTHQUAKE
HORIZONTAL COMPONENT

FIGURE 3.7B-3, Rev. 55

Auto-Cad Figure Fsar 3_7B_3.dwg



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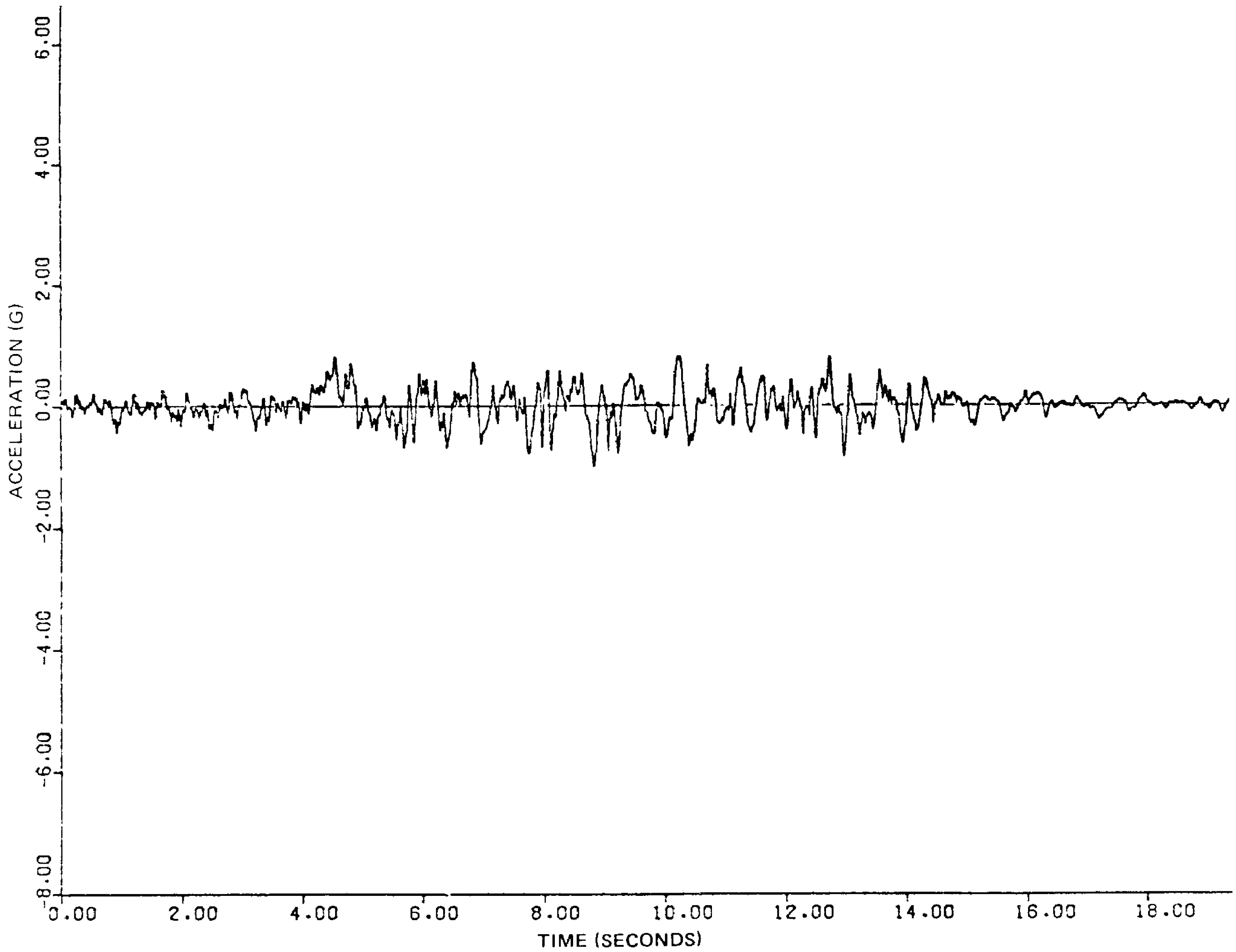
DIESEL GENERATOR 'E' BUILDING'S
DESIGN RESPONSE SPECTRA
SAFE SHUTDOWN EARTHQUAKE
HORIZONTAL COMPONENT

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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UNITS 1 & 2
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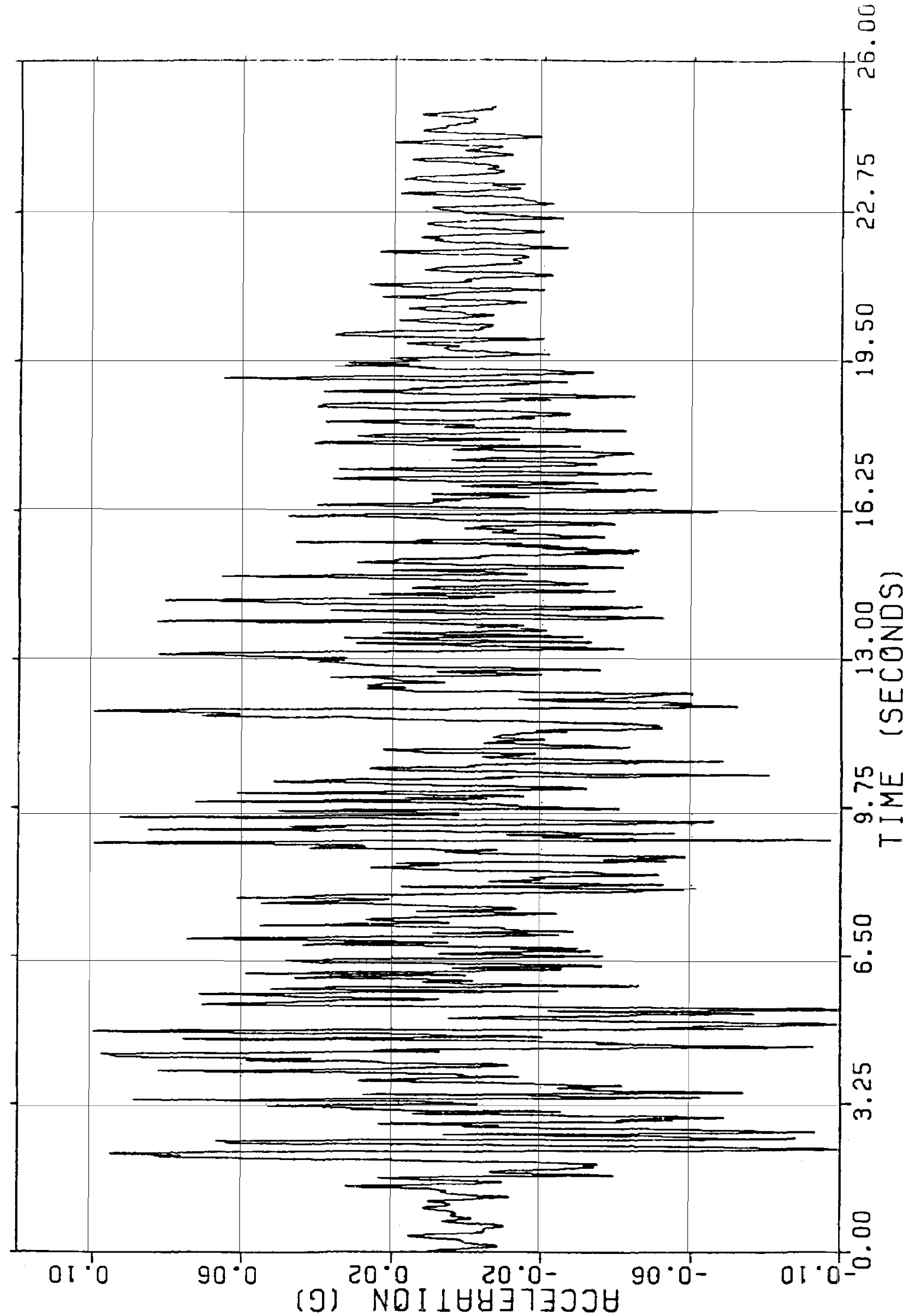
SYNTHETIC TIME HISTORY*
NORMALIZED TO 1G

FIGURE 3.7B-5, Rev. 55

Auto-Cad Figure Fsar 3_7B_5.dwg



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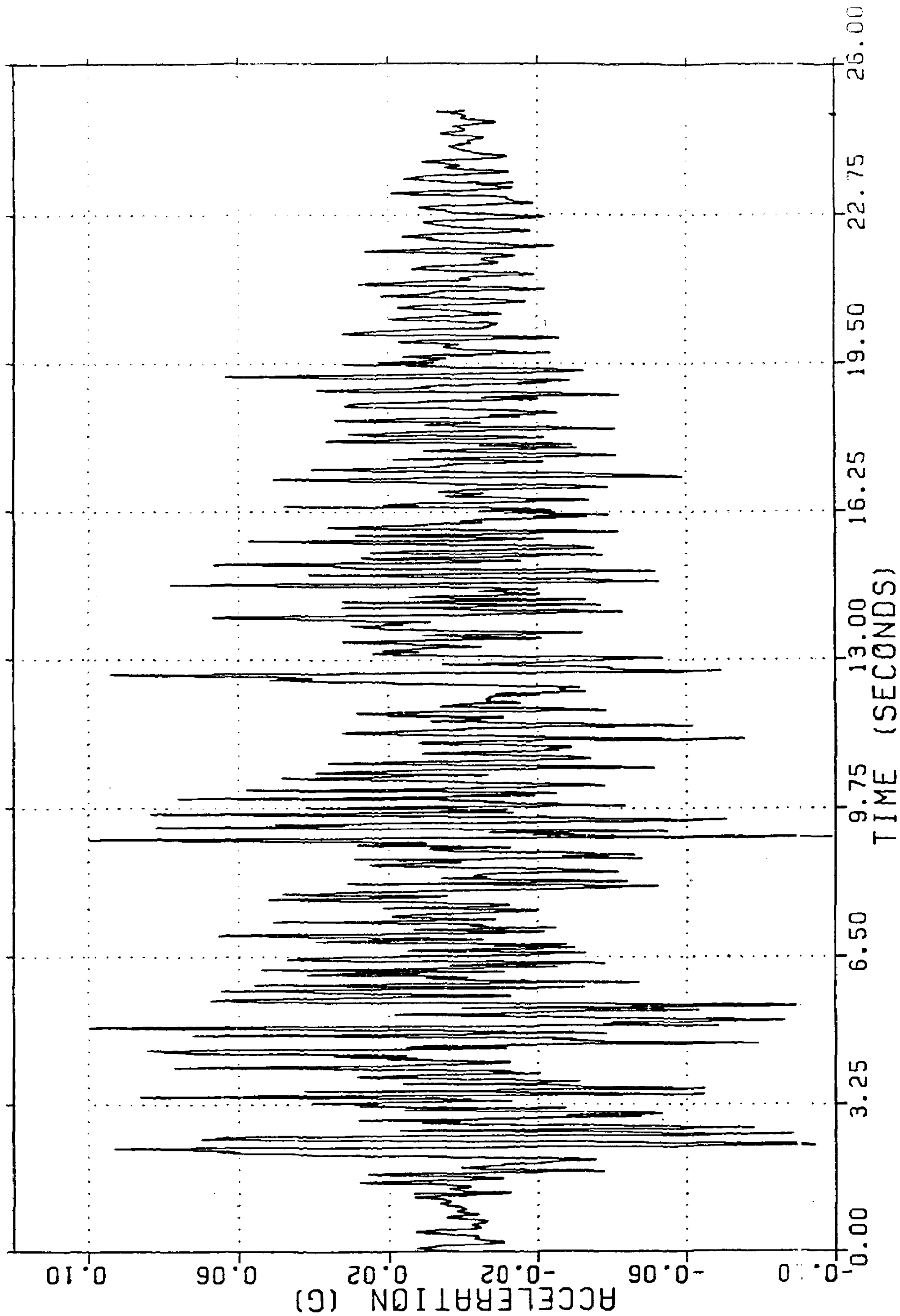
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

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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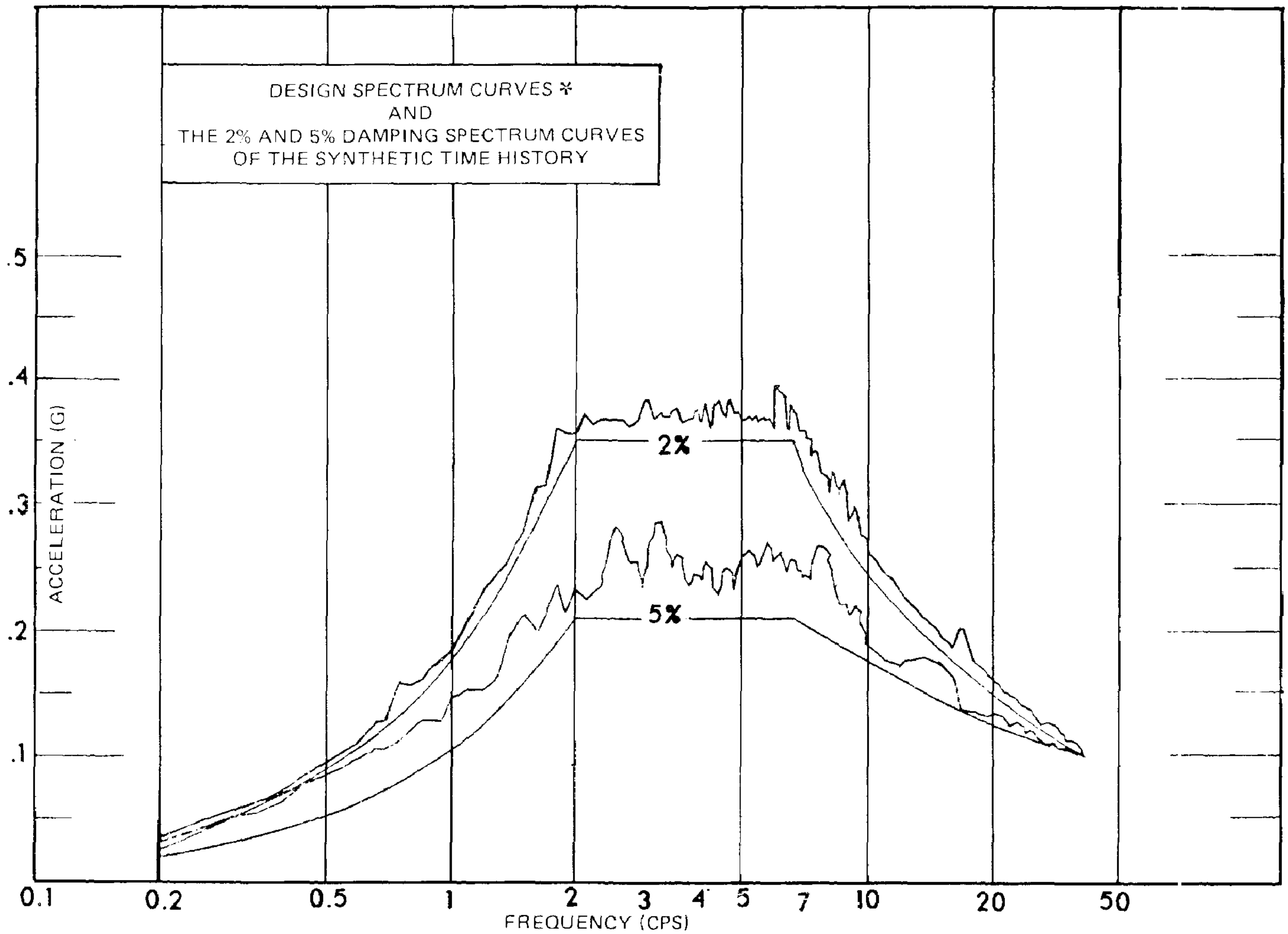
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

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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SUSQUEHANNA STEAM ELECTRIC STATION
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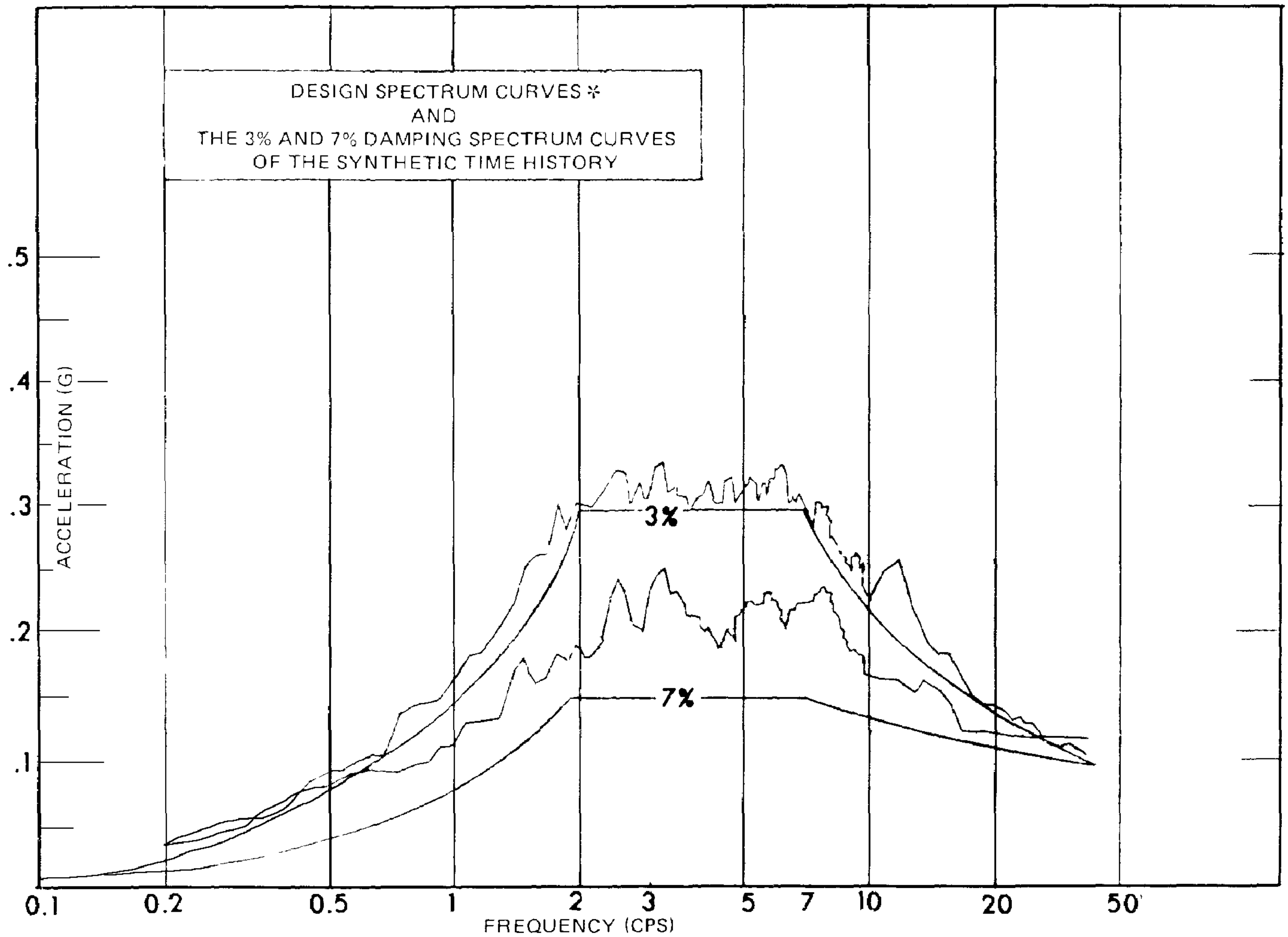
FINAL SAFETY ANALYSIS REPORT

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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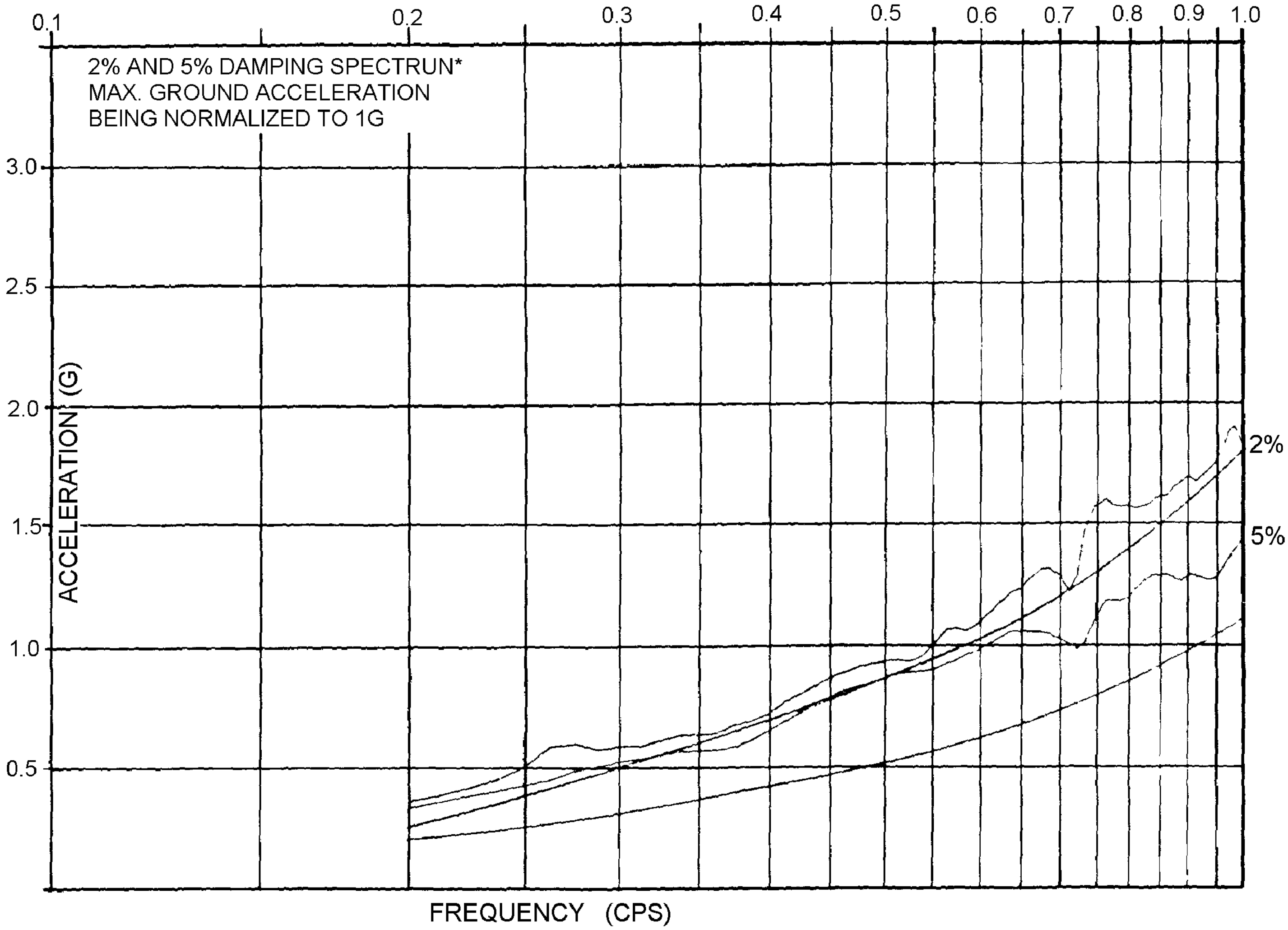
COMPARISON OF TIME HISTORY
RESPONSE SPECTRA AND
DESIGN RESPONSE SPECTRA
3% AND 7% DAMPING (0.2-30 CPS)

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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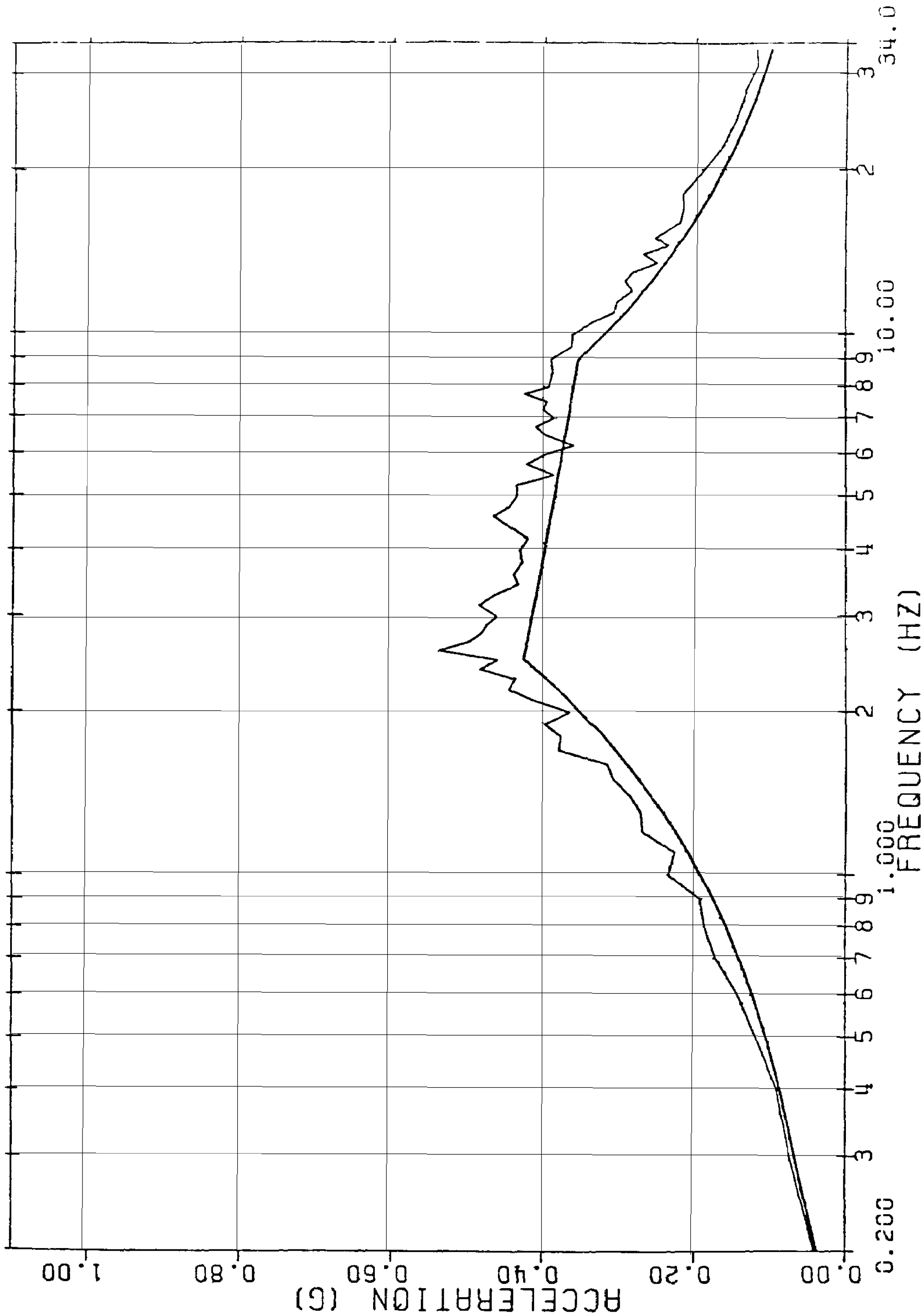
COMPARISON OF TIME HISTORY
RESPONSE SPECTA AND
DESIGN RESPONSE SPECTRA-
2% AND 5% DAMPING (0.2-1.0 CPS)

FIGURE 3.7B-10, Rev. 55

Auto-Cad Figure Fsar 3_7B_10.dwg



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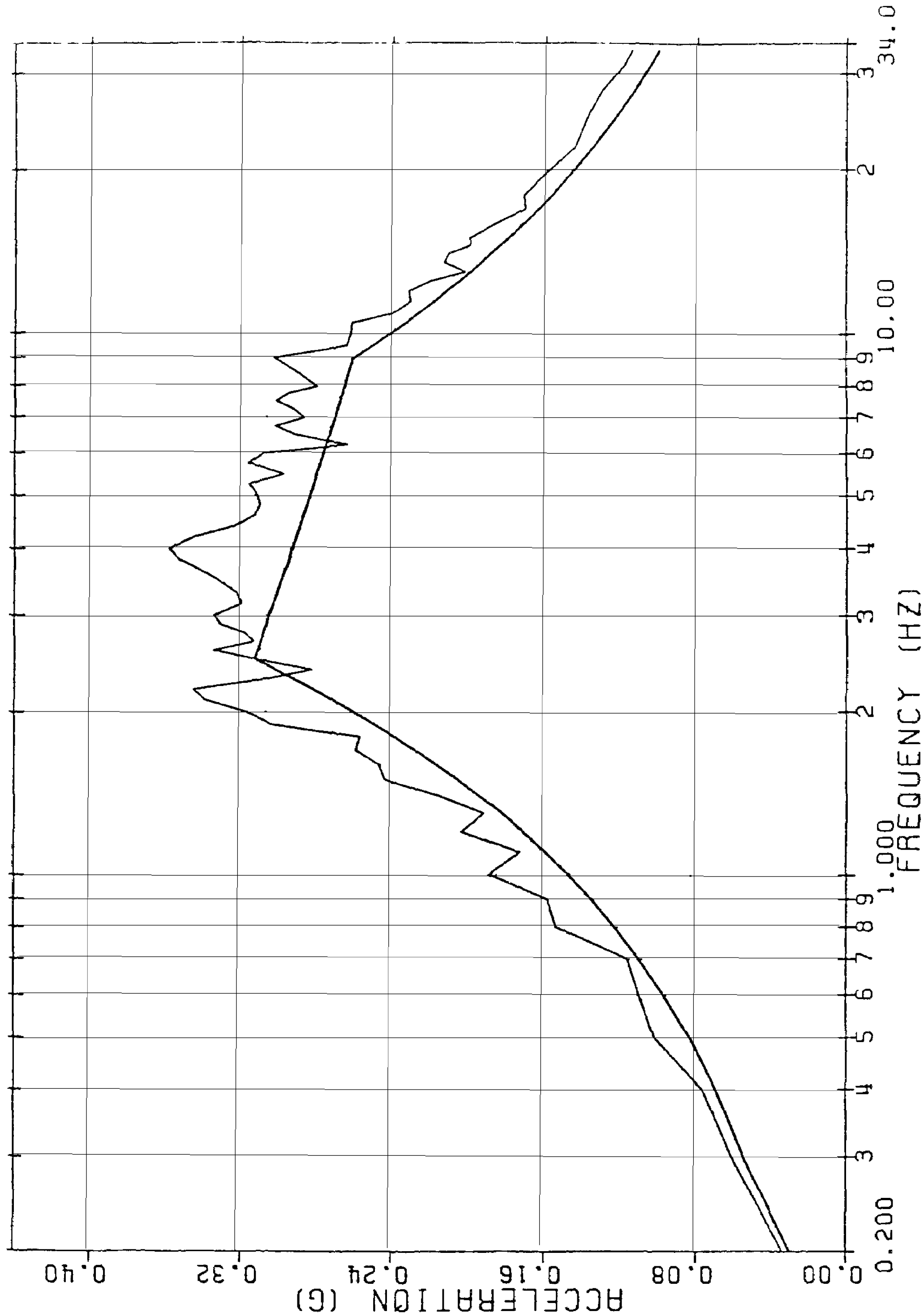
DIESEL GENERATOR 'E' BUILDING
COMPARISON OF HORIZONTAL TIME
HISTORY RESPONSE SPECTRUM AND
HORIZONTAL DESIGN RESPONSE
SPECTRUM 2% DAMPING

FIGURE 3.7B-11, Rev. 55

Auto-Cad Figure Fsar 3_7B_11.dwg



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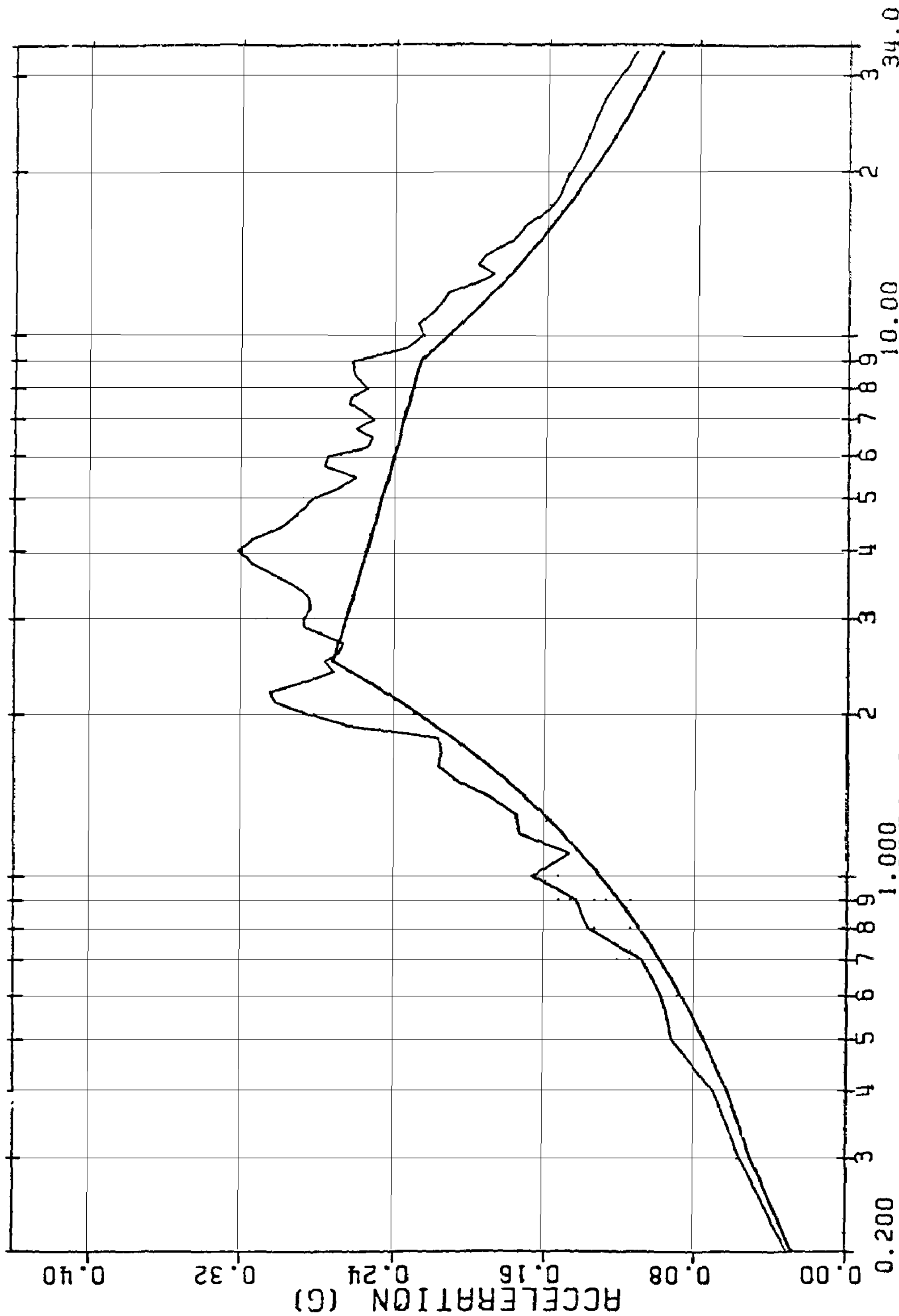
DIESEL GENERATOR 'E' BUILDING
COMPARISON OF HORIZONTAL TIME
HISTORY RESPONSE SPECTRUM AND
HORIZONTAL DESIGN RESPONSE
SPECTRUM 5% DAMPING

FIGURE 3.7B-12, Rev. 55

Auto-Cad Figure Fsar 3_7B_12.dwg



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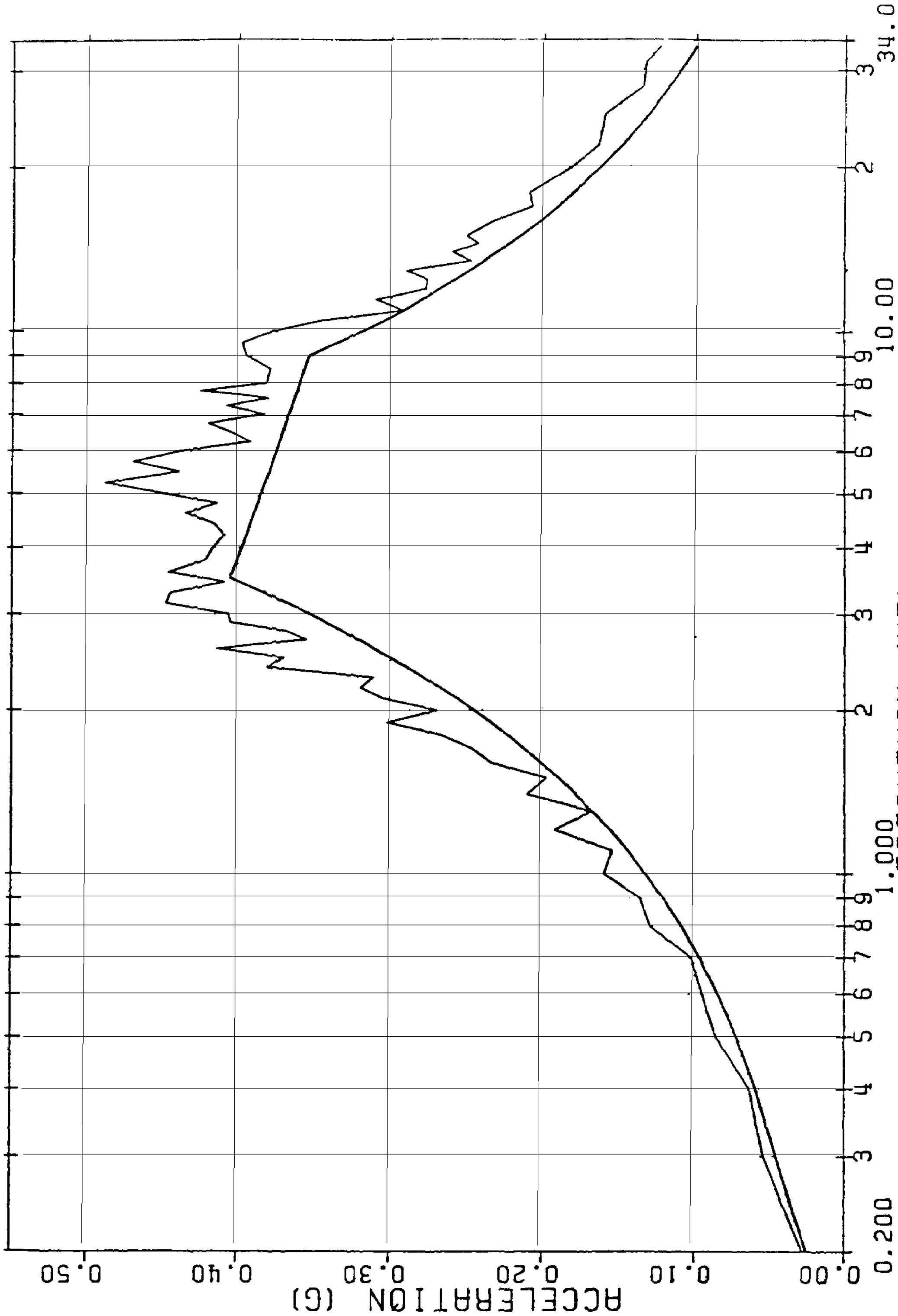
DIESEL GENERATOR 'E' BUILDING
COMPARISON OF HORIZONTAL TIME
HISTORY RESPONSE SPECTRUM AND
HORIZONTAL DESIGN RESPONSE
SPECTRUM 7% DAMPING

FIGURE 3.7B-13, Rev. 55

Auto-Cad Figure Fsar 3_7B_13.dwg



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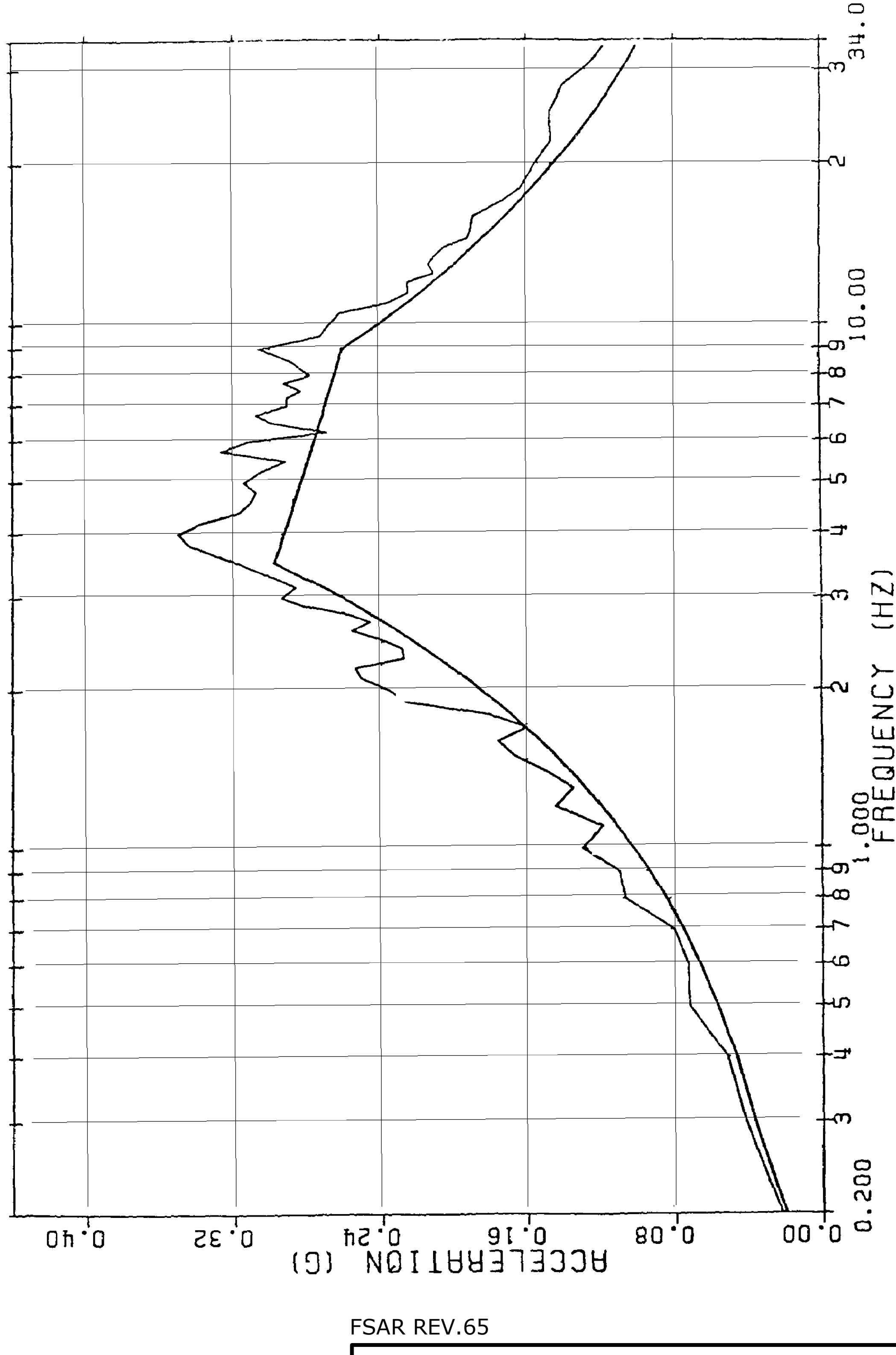
DIESEL GENERATOR 'E' BUILDING
COMPARISON OF VERTICAL TIME
HISTORY RESPONSE SPECTRUM AND
VERTICAL DESIGN RESPONSE
SPECTRUM 2% DAMPING

FIGURE 3.7B-14, Rev. 55

Auto-Cad Figure Fsar 3_7B_14.dwg



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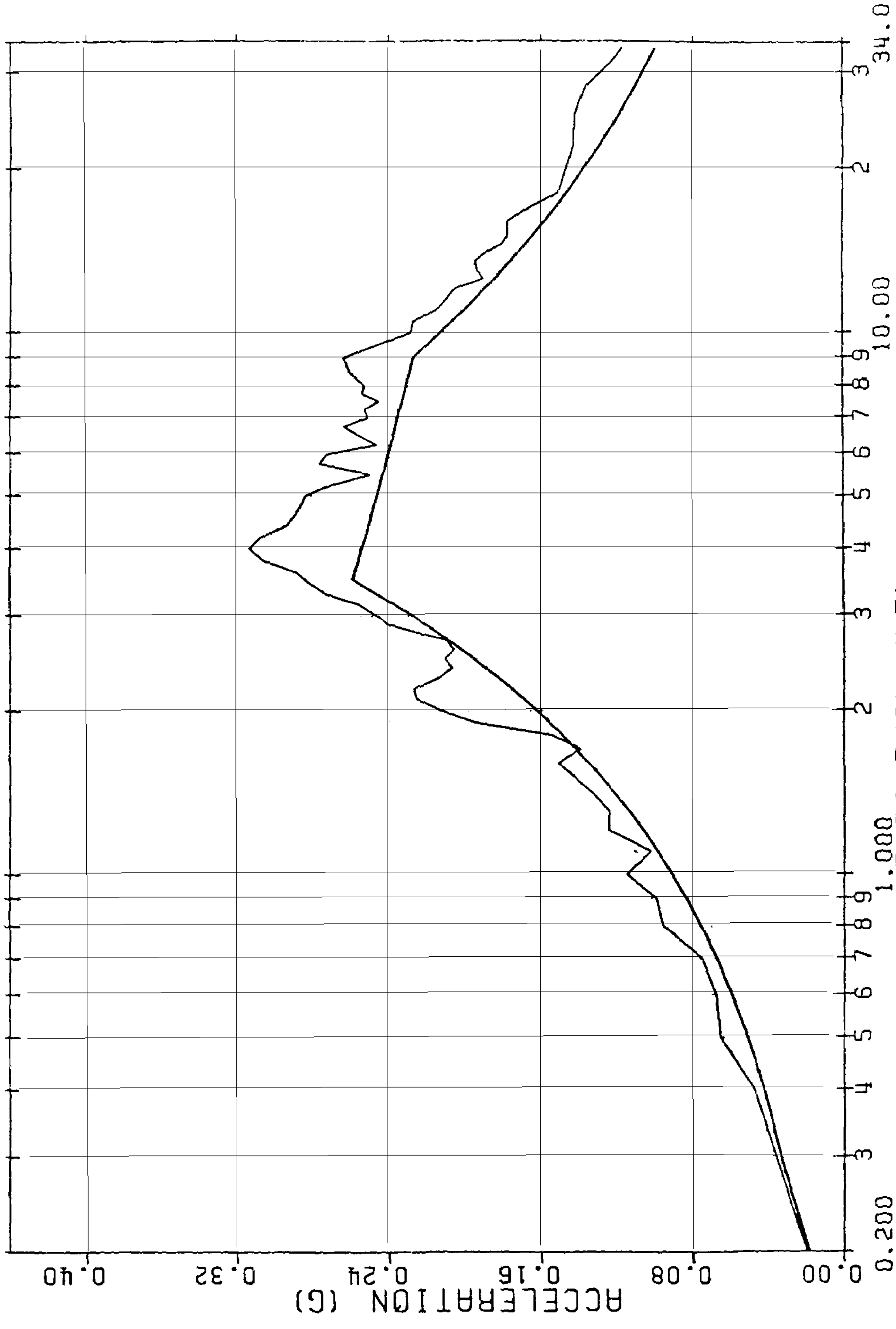
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

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



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UNITS 1 & 2
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DIESEL GENERATOR 'E' BUILDING
COMPARISON OF VERTICAL TIME
HISTORY RESPONSE SPECTRUM AND
VERTICAL DESIGN RESPONSE
SPECTRUM 7% DAMPING

FIGURE 3.7B-16, Rev. 55

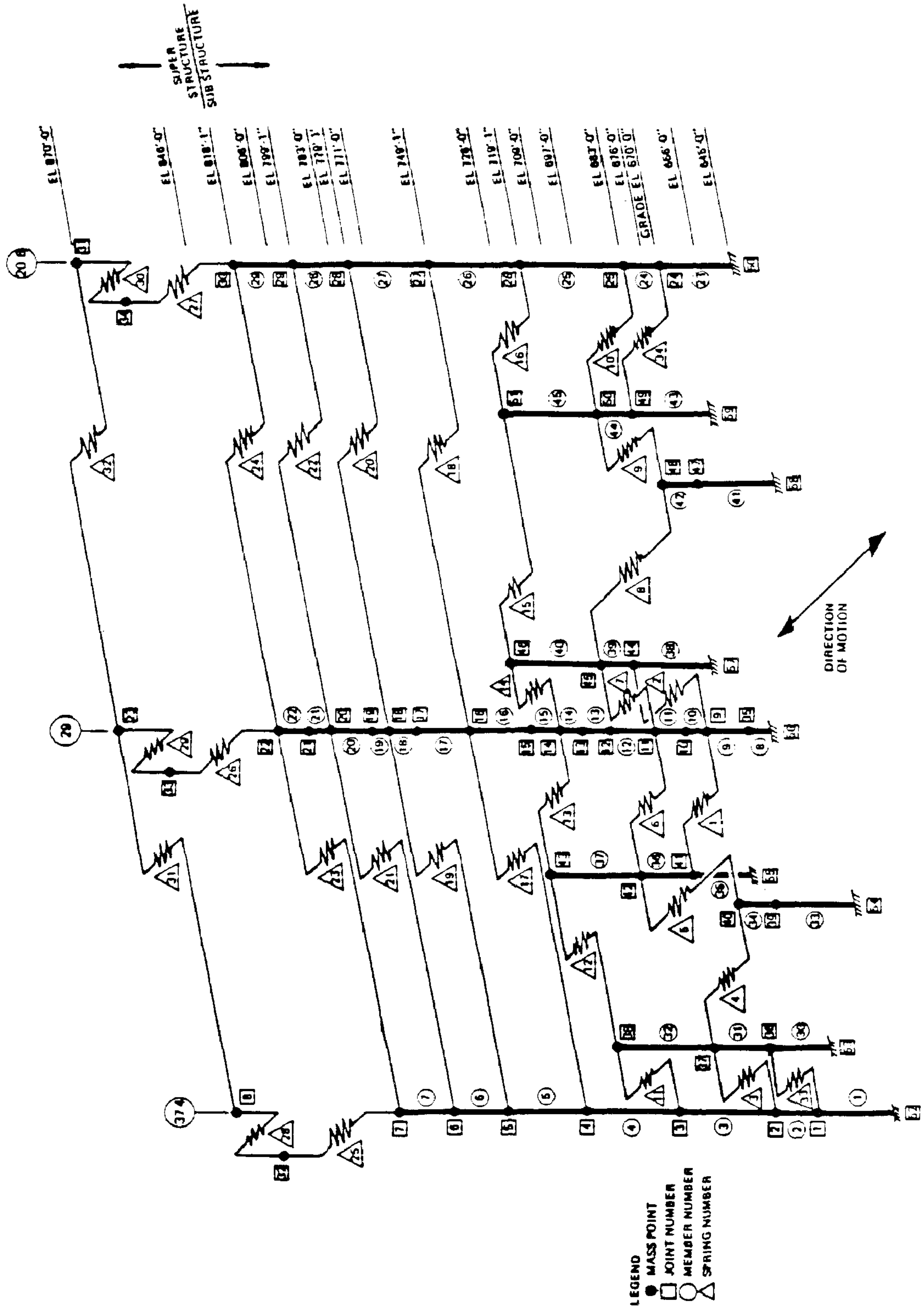
Auto-Cad Figure Fsar 3_7B_16.dwg

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

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



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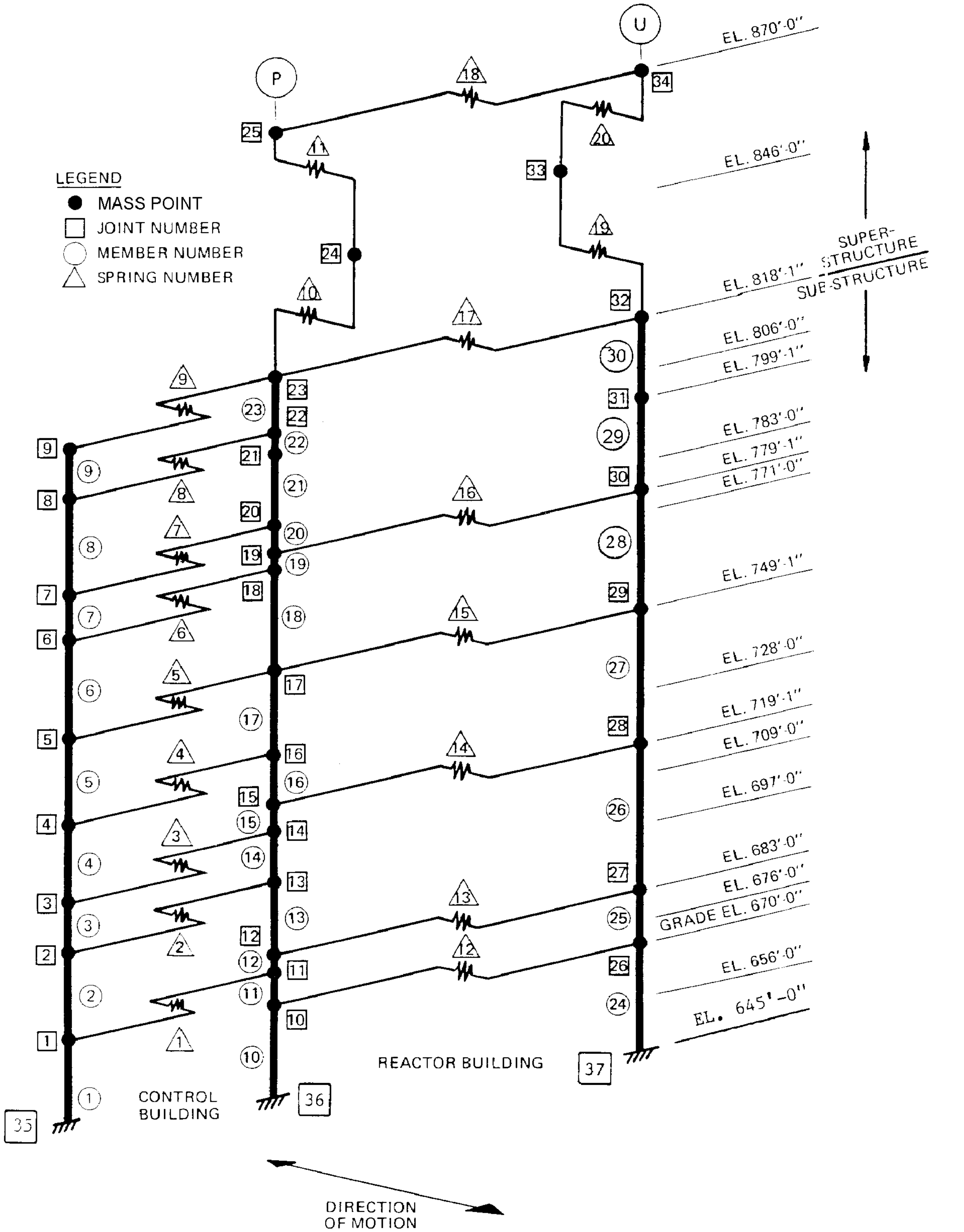
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

E-W SEISMIC MODEL OF
REACTOR AND CONTROL BUILDING

FIGURE 3.7B-19, Rev. 55

Auto-Cad Figure Fsar 3_7B_19.dwg



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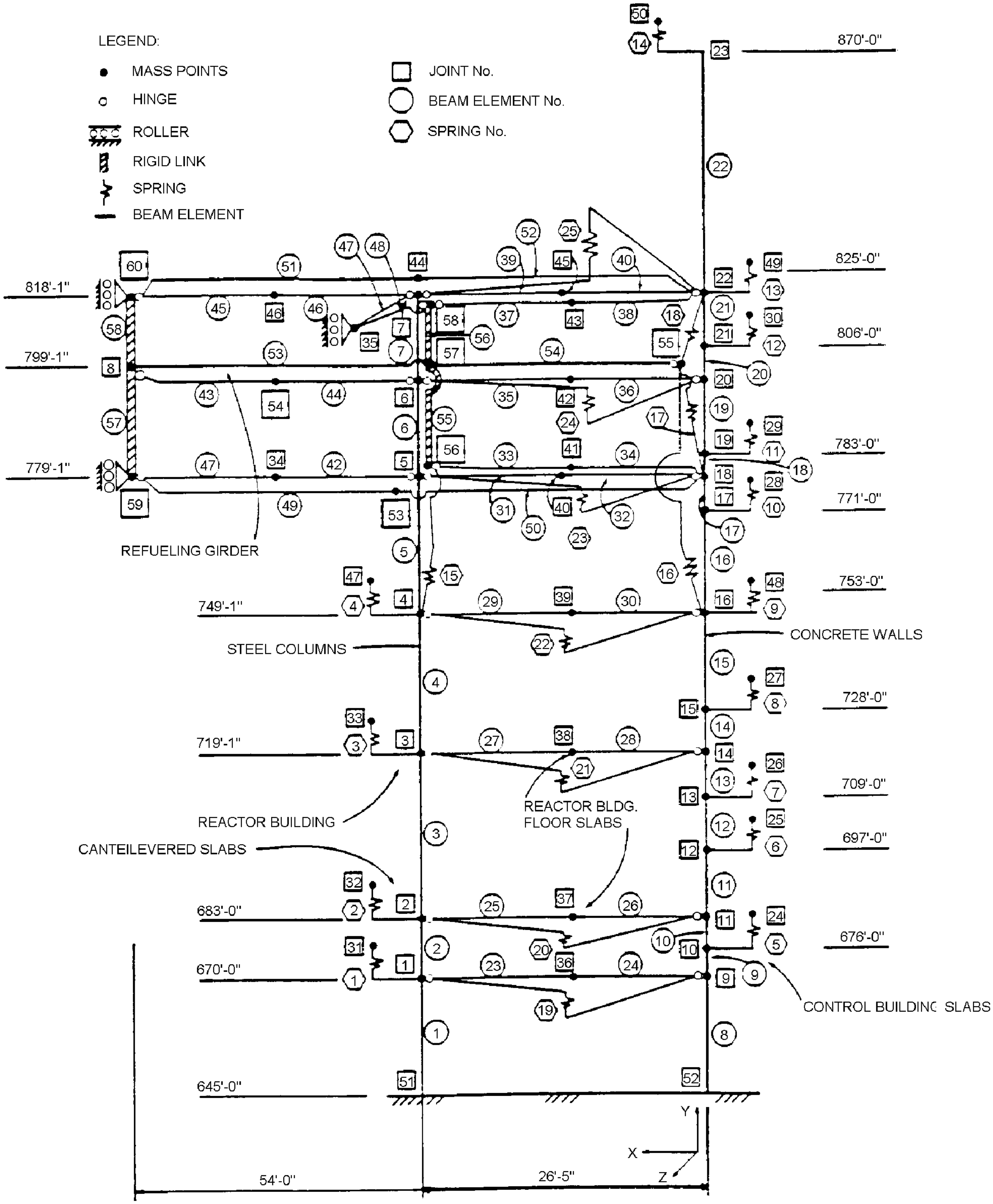
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

N-S SEISMIC MODEL OF
REACTOR AND CONTROL BUILDING

FIGURE 3.7B-20, Rev. 55

Auto-Cad Figure Fsar 3_7B_20.dwg



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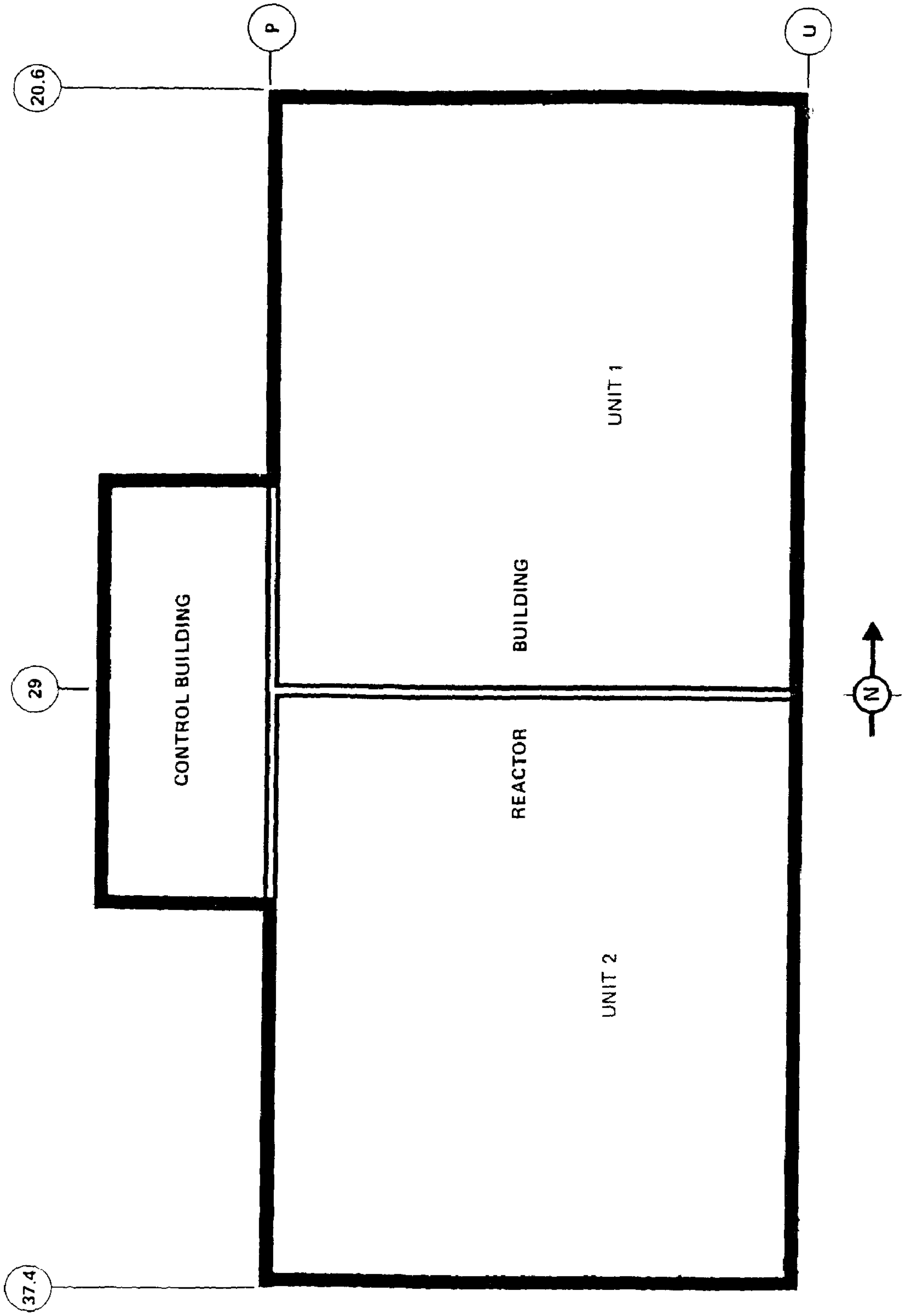
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

VERTICAL SEISMIC MODEL OF
REACTOR AND CONTROL BUILDING

FIGURE 3.7B-21, Rev. 55

Auto-Cad Figure Fsar 3_7B_21.dwg



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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

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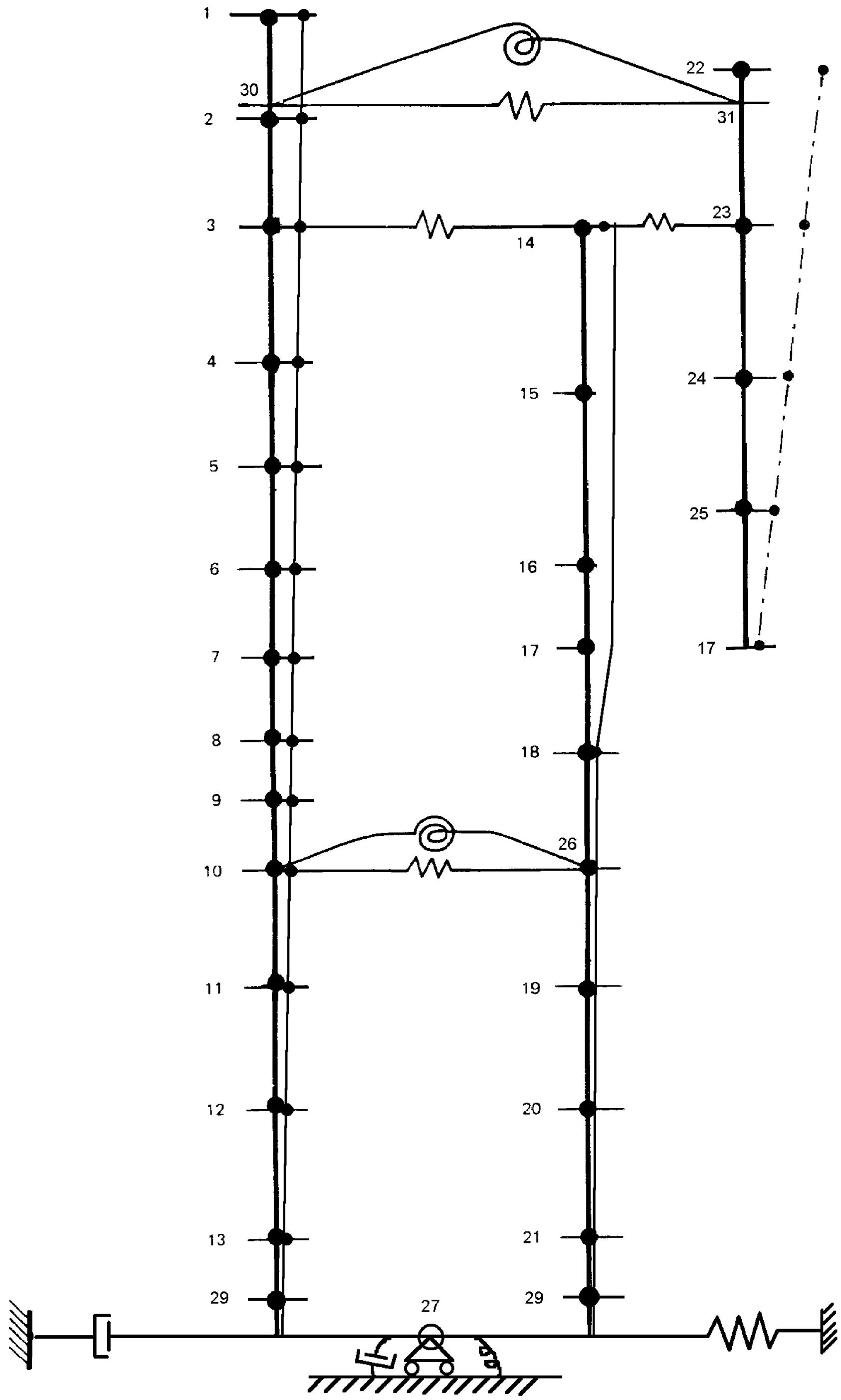
Figure Withheld Under 10 CFR 2.390

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



FREQUENCY = 4.99 CPS

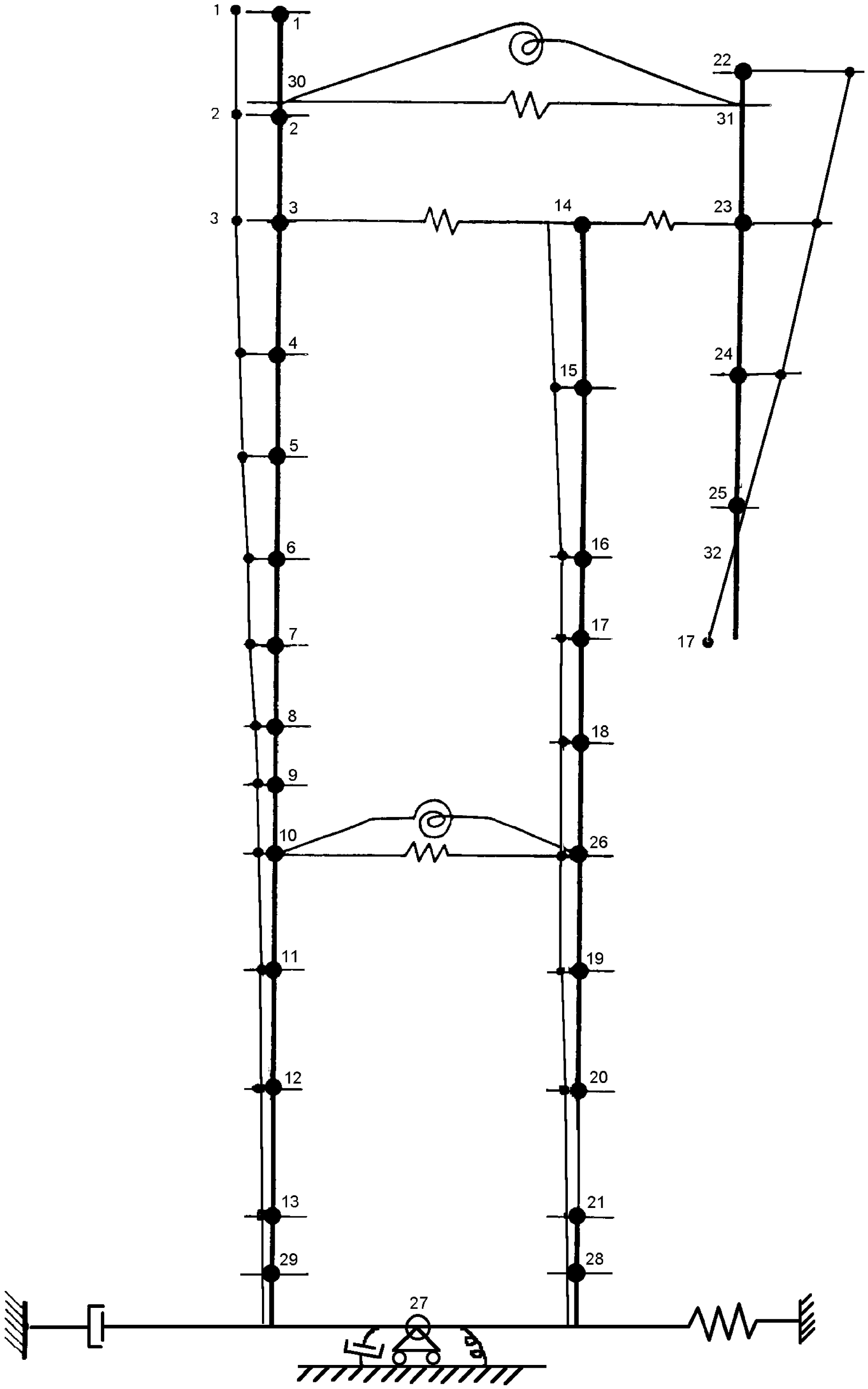
FSAR REV.65

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

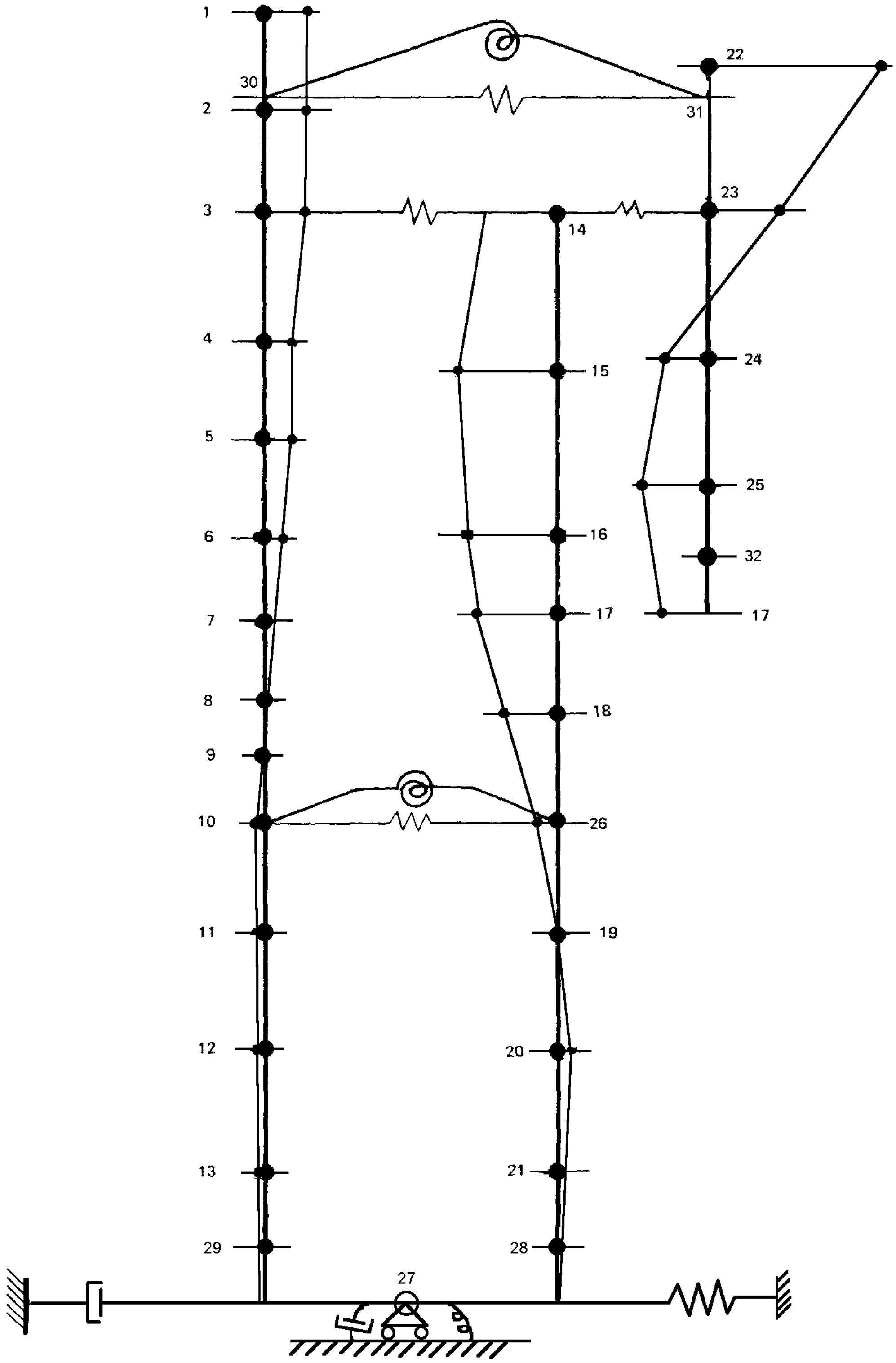
FSAR REV.65

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

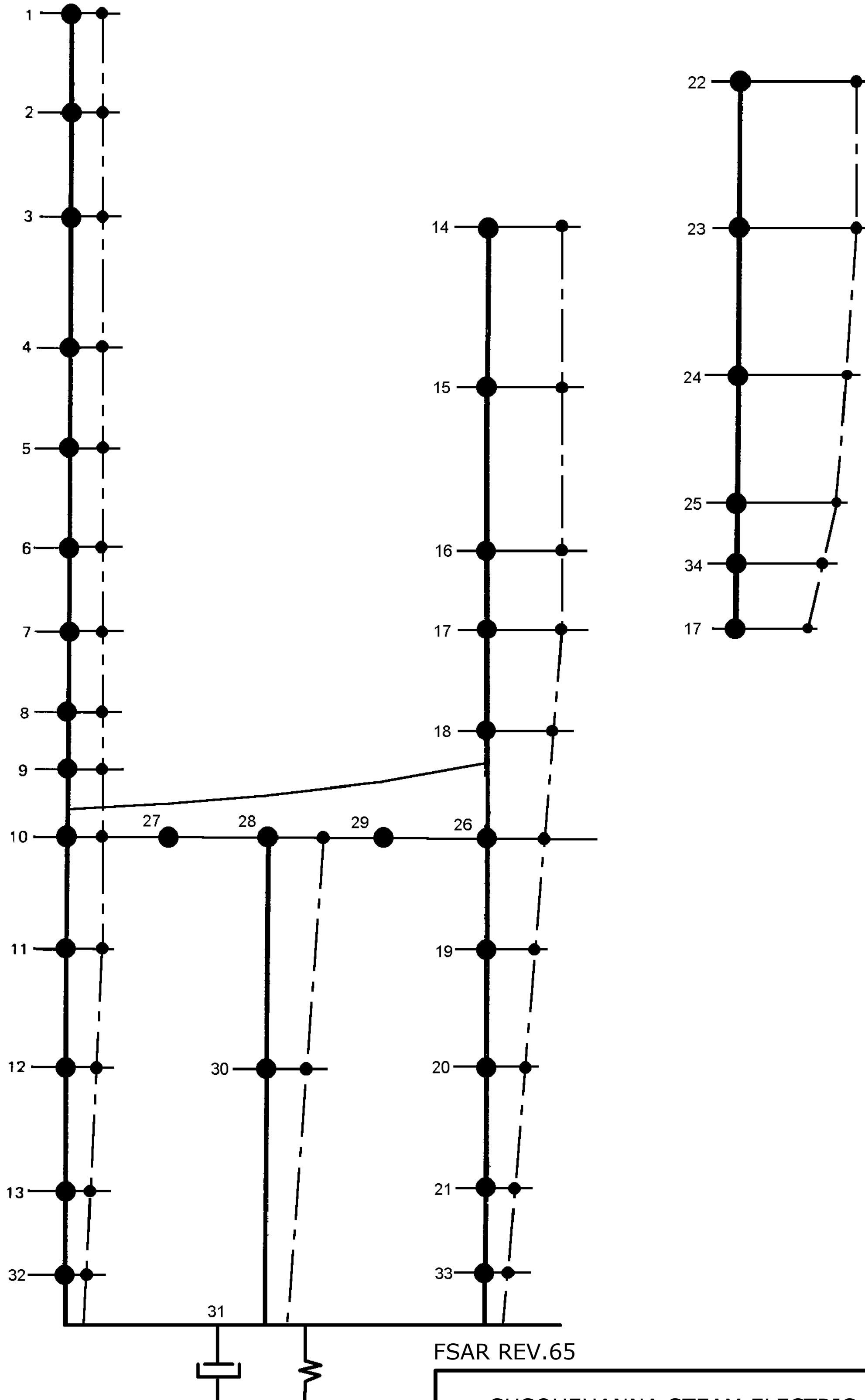
FSAR REV.65

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

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.

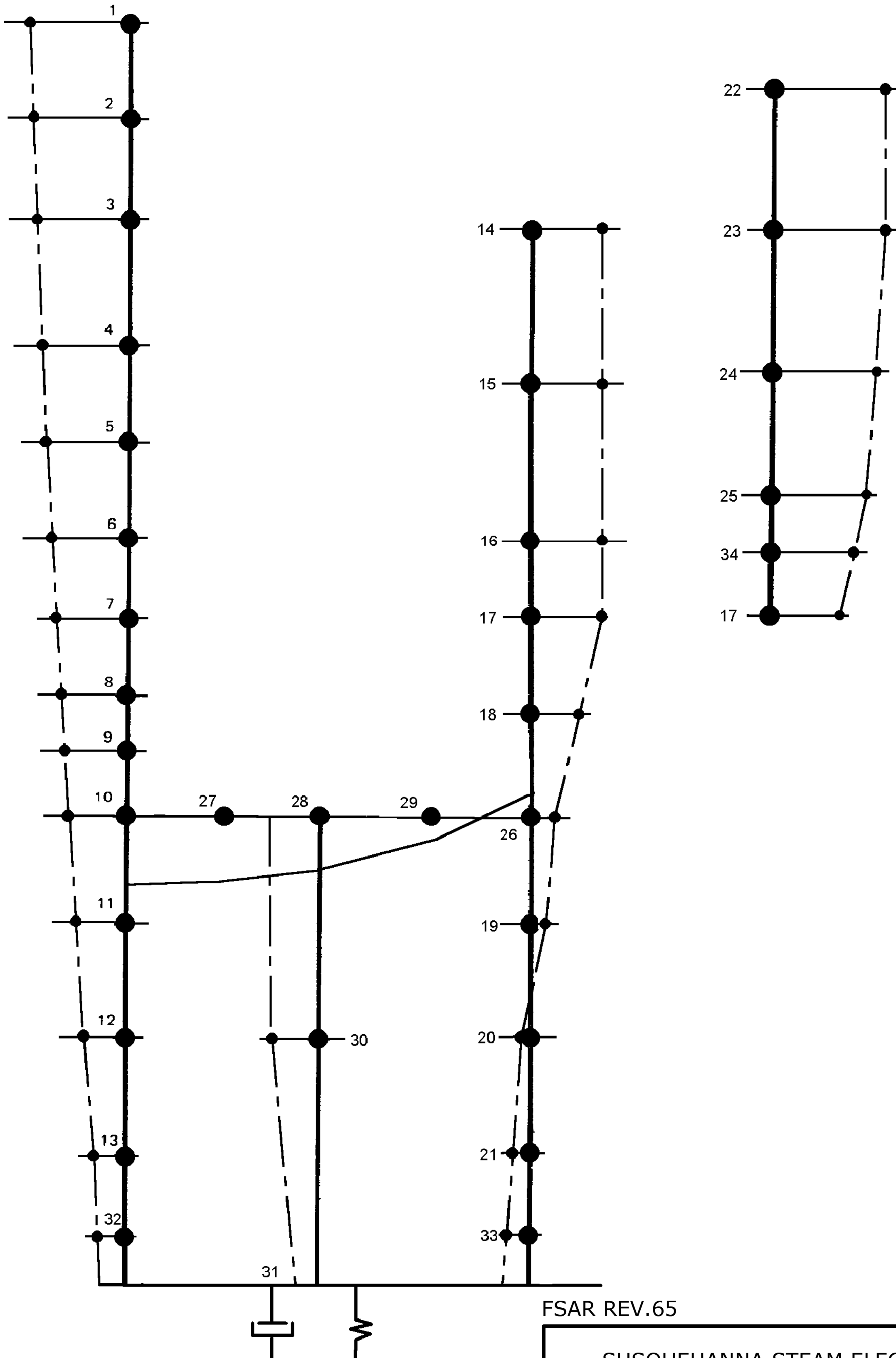
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

CONTAINMENT
VERTICAL MODE SHAPES
MODE 1

FIGURE 3.7B-27, Rev. 55

Auto-Cad Figure Fsar 3_7B_27.dwg



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
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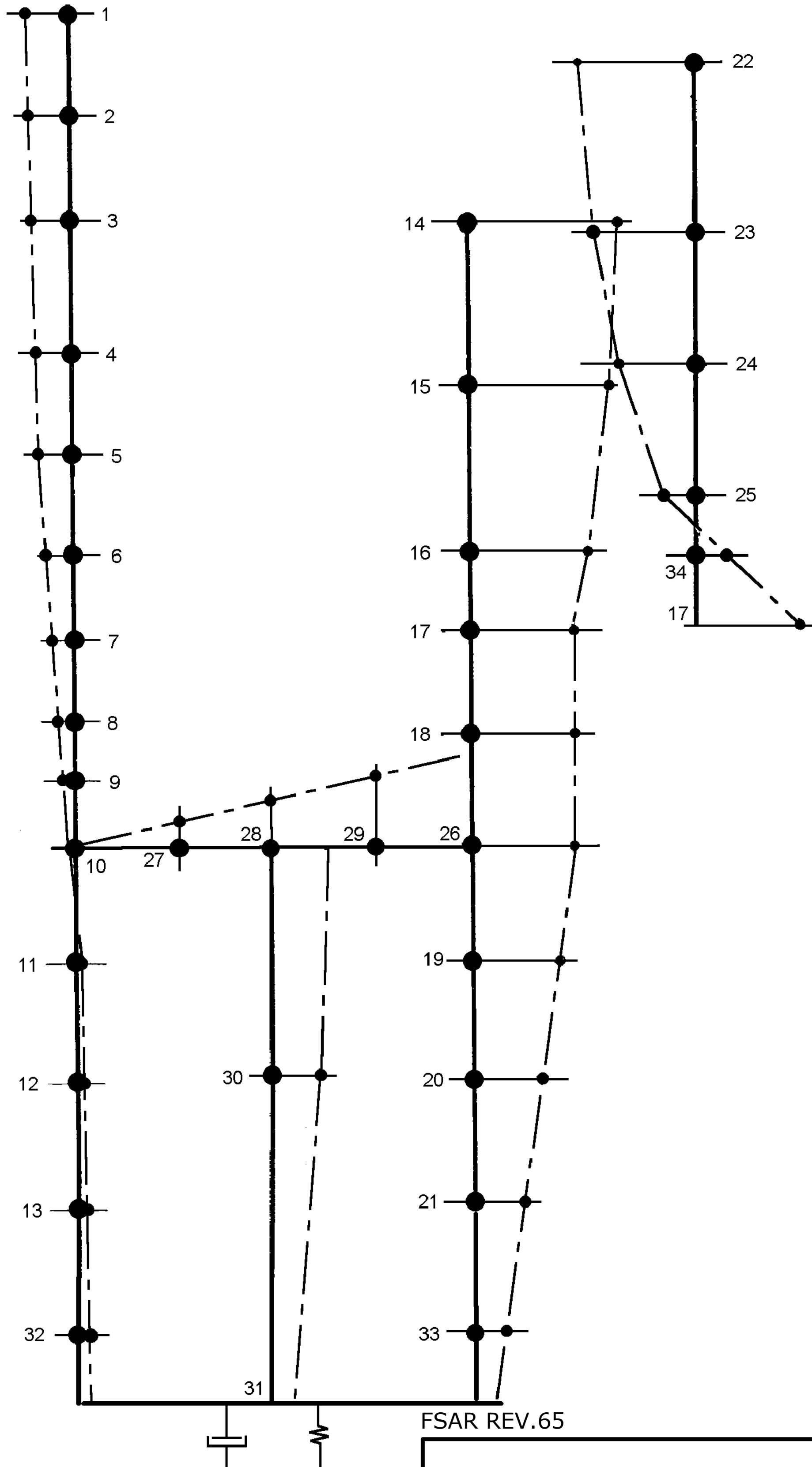
CONTAINMENT
VERTICAL MODE SHAPES
MODE 2

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.

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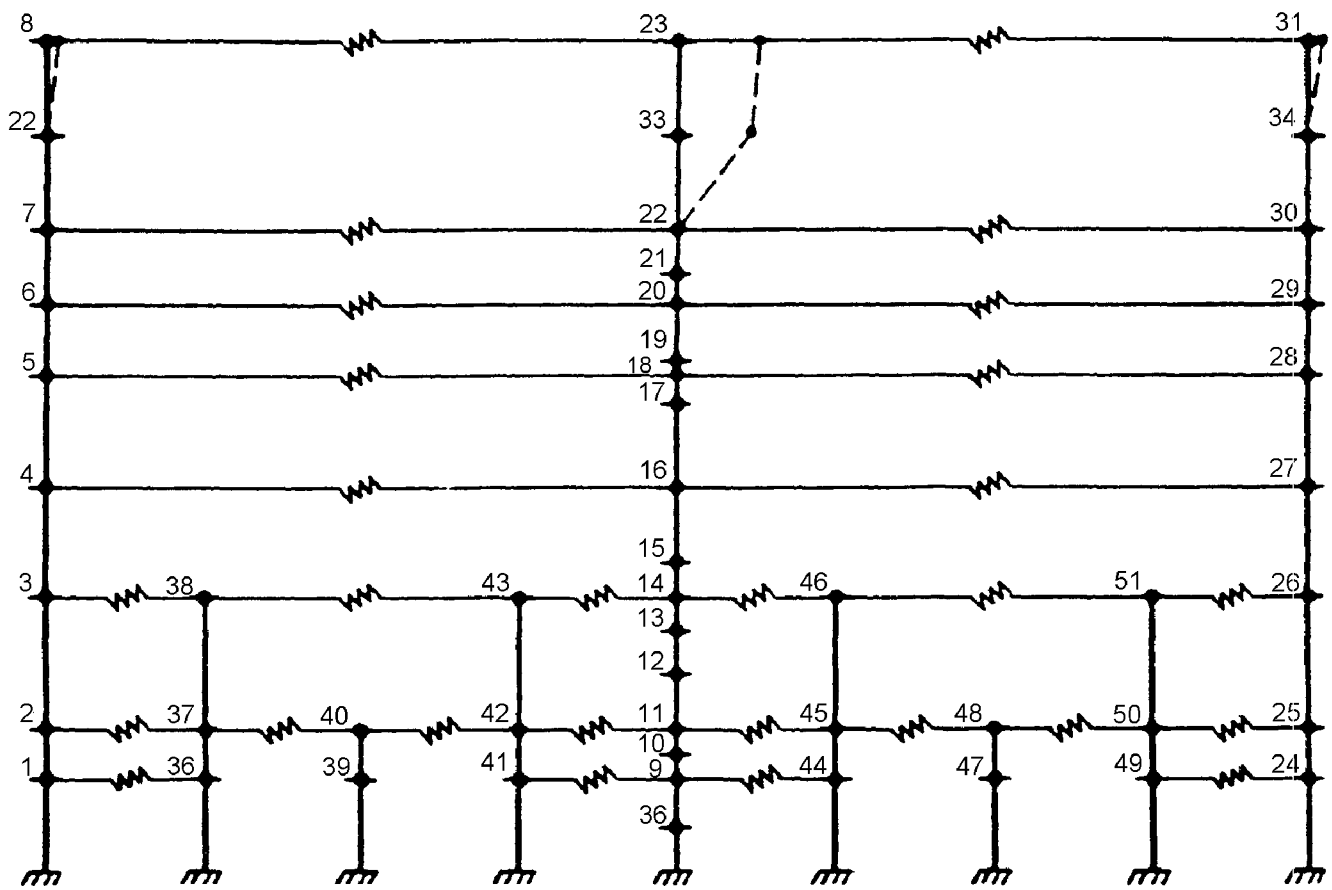
CONTAINMENT
VERTICAL MODE SHAPES
MODE 3

FIGURE 3.7B-29, Rev. 55

Auto-Cad Figure Fsar 3_7B_29.dwg



NOTE: CRANES ARE LOCATED
AT MASS POINTS 32 AND 33

FREQUENCY = 2.23 CPS

FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
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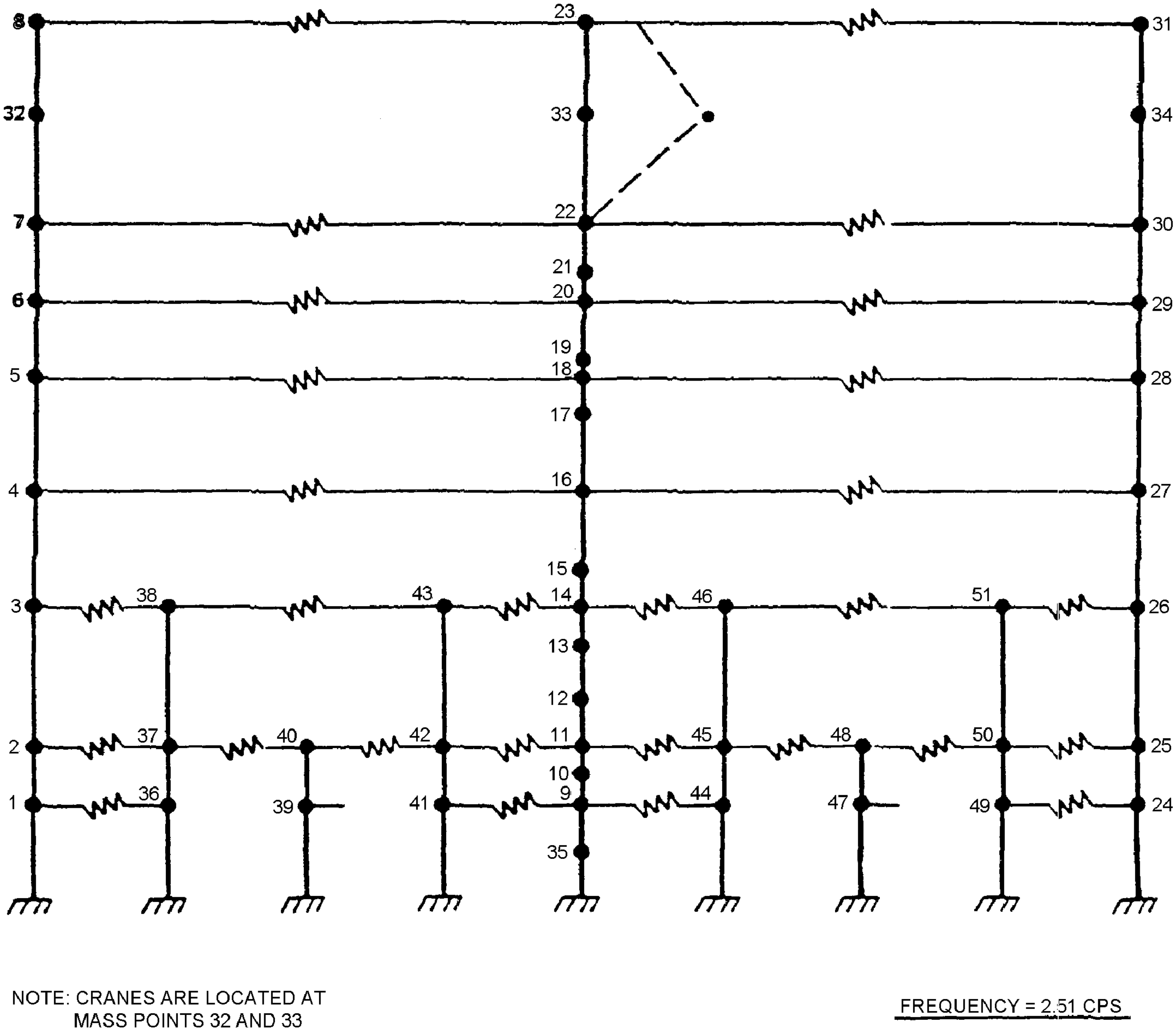
REACTOR AND CONTROL BUILDING
E-W MODE SHAPES - MODE 1
(CRANES AT POINTS 32 AND 33)

FIGURE 3.7B-30, Rev. 55

Auto-Cad Figure Fsar 3_7B_30.dwg



FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
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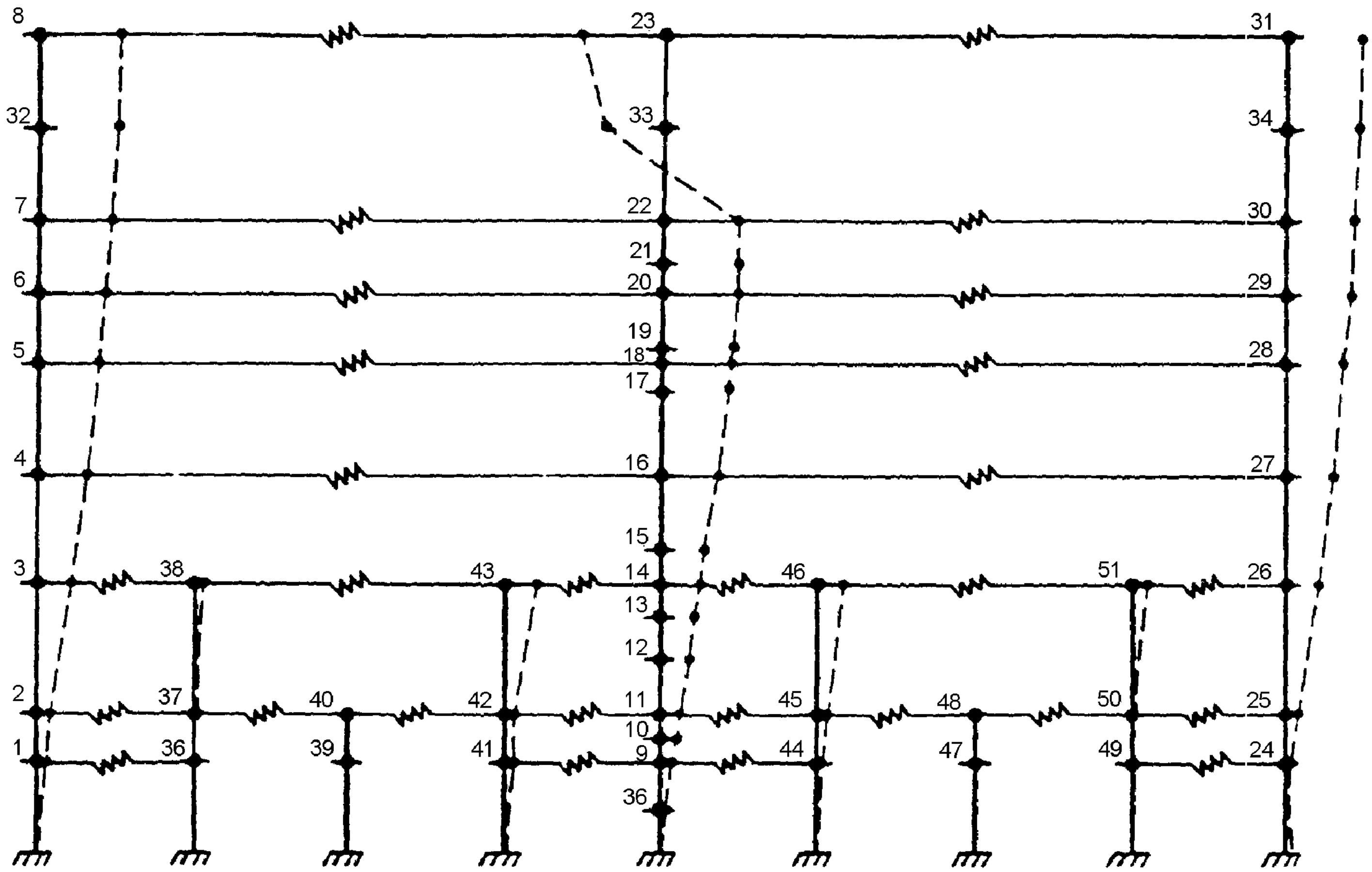
FINAL SAFETY ANALYSIS REPORT

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
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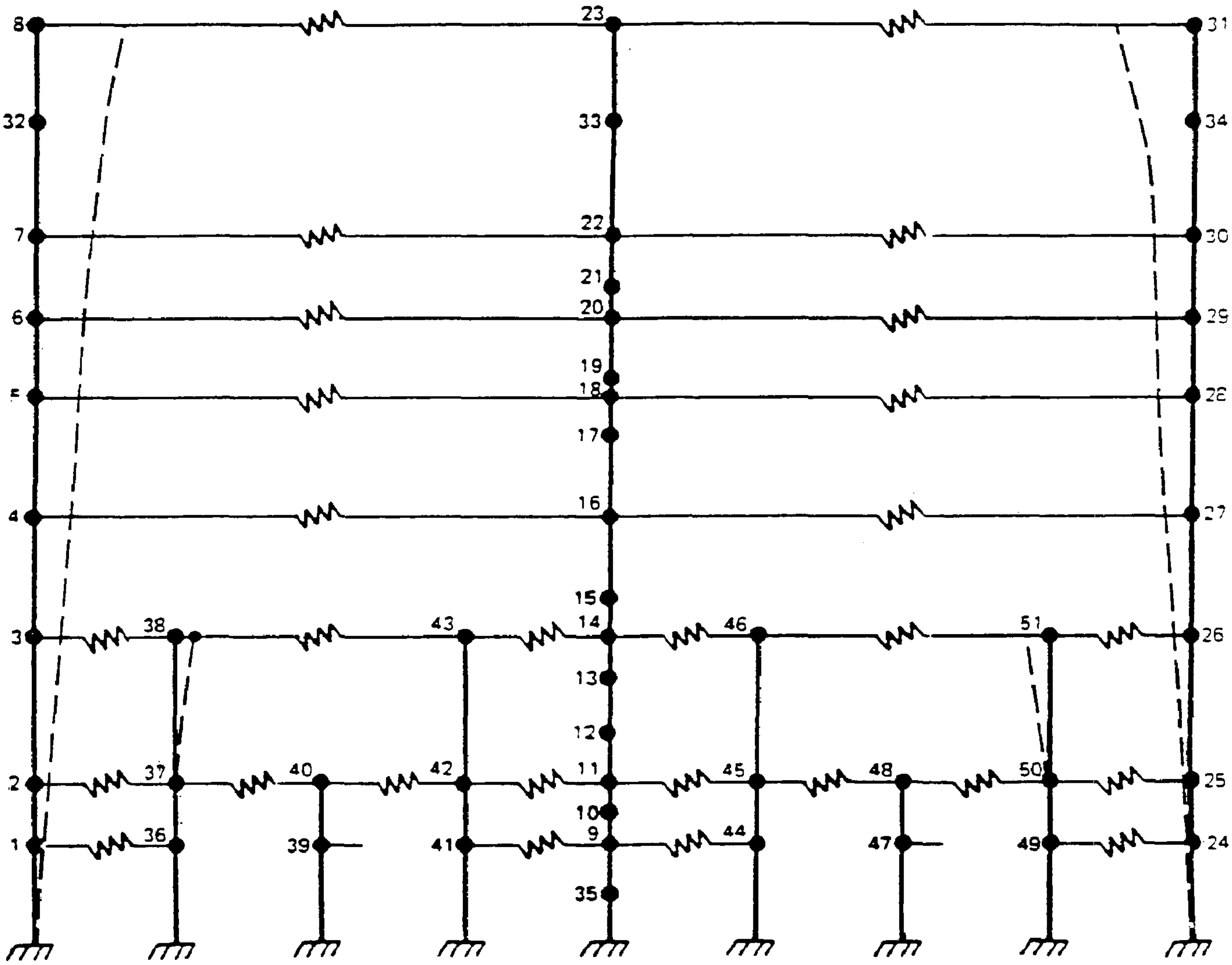
REACTOR AND CONTROL BUILDING
E-W MODE SHAPES - MODE 3
(CRANES AT POINTS 32 AND 33)

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
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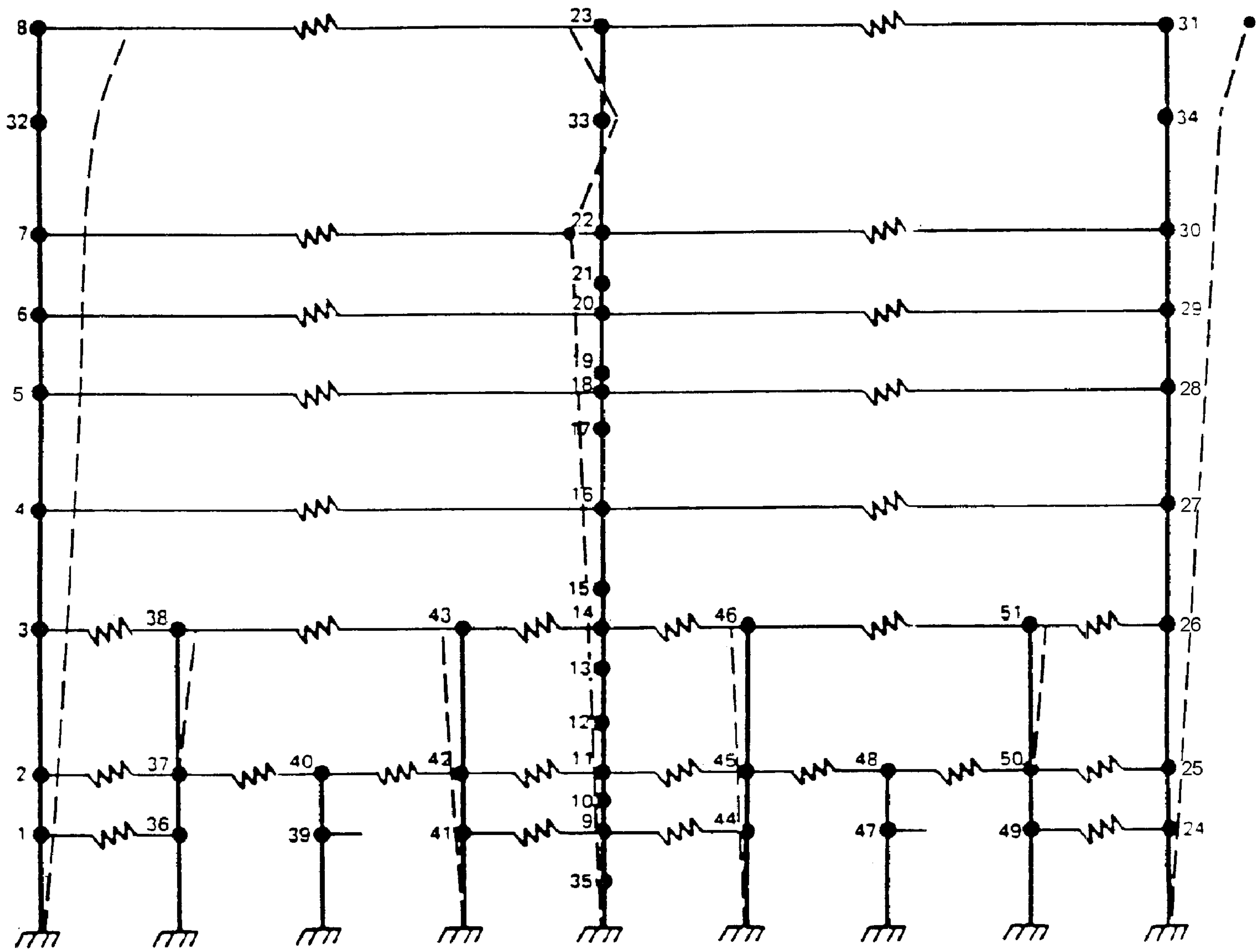
REACTOR AND CONTROL BUILDING
E-W MODE SHAPES
MODE 4

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

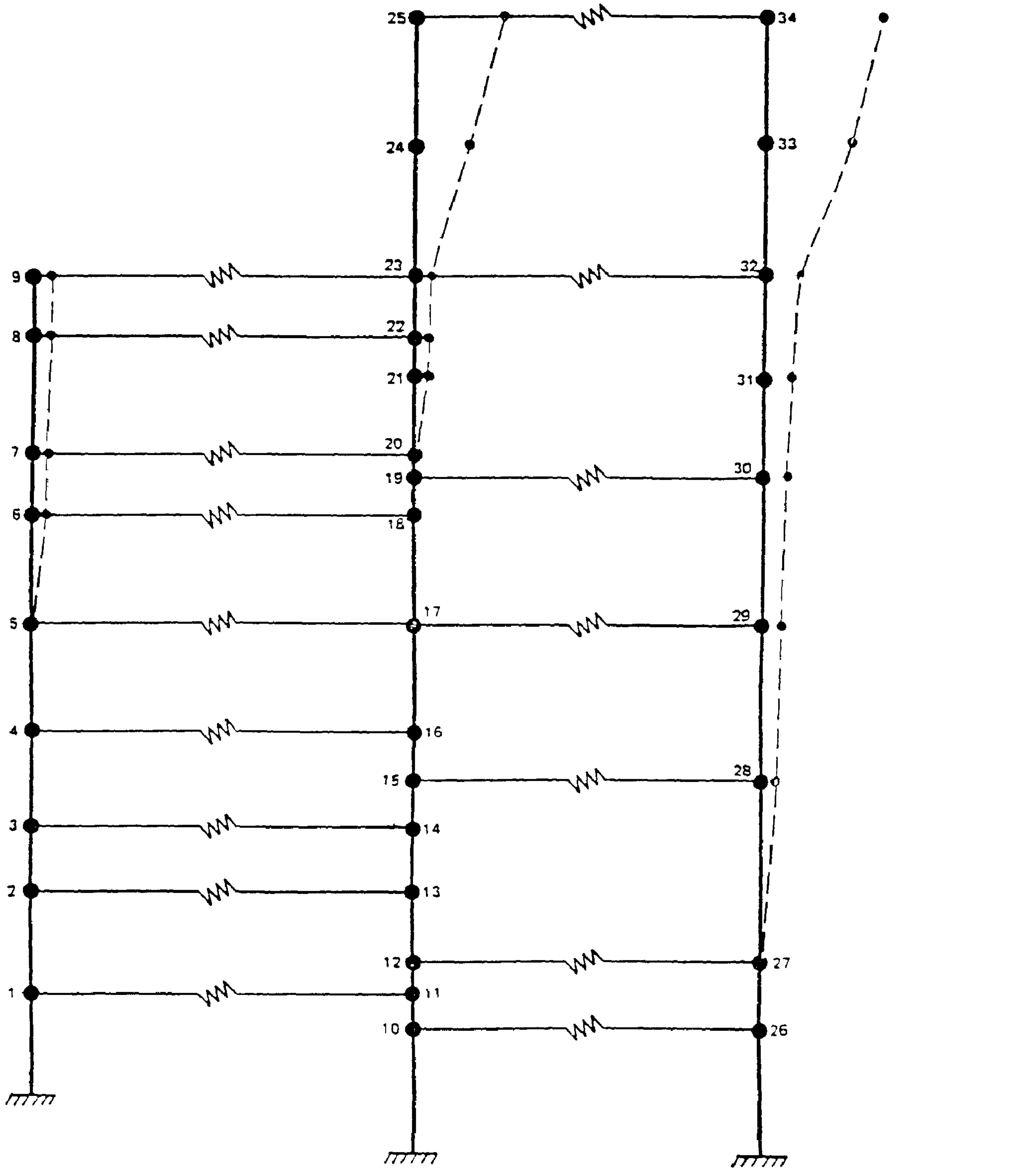
FSAR REV.65

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
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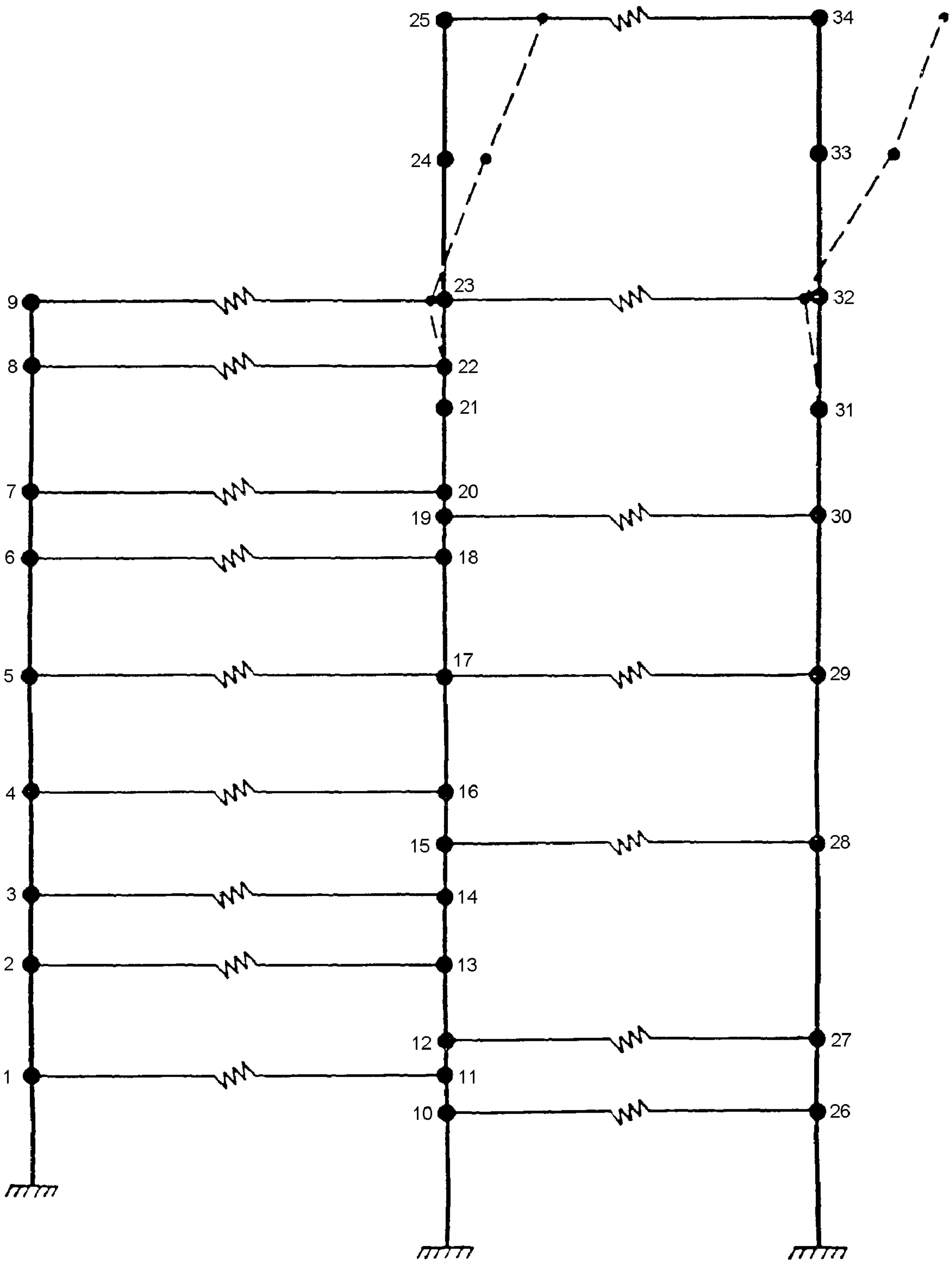
FINAL SAFETY ANALYSIS REPORT

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

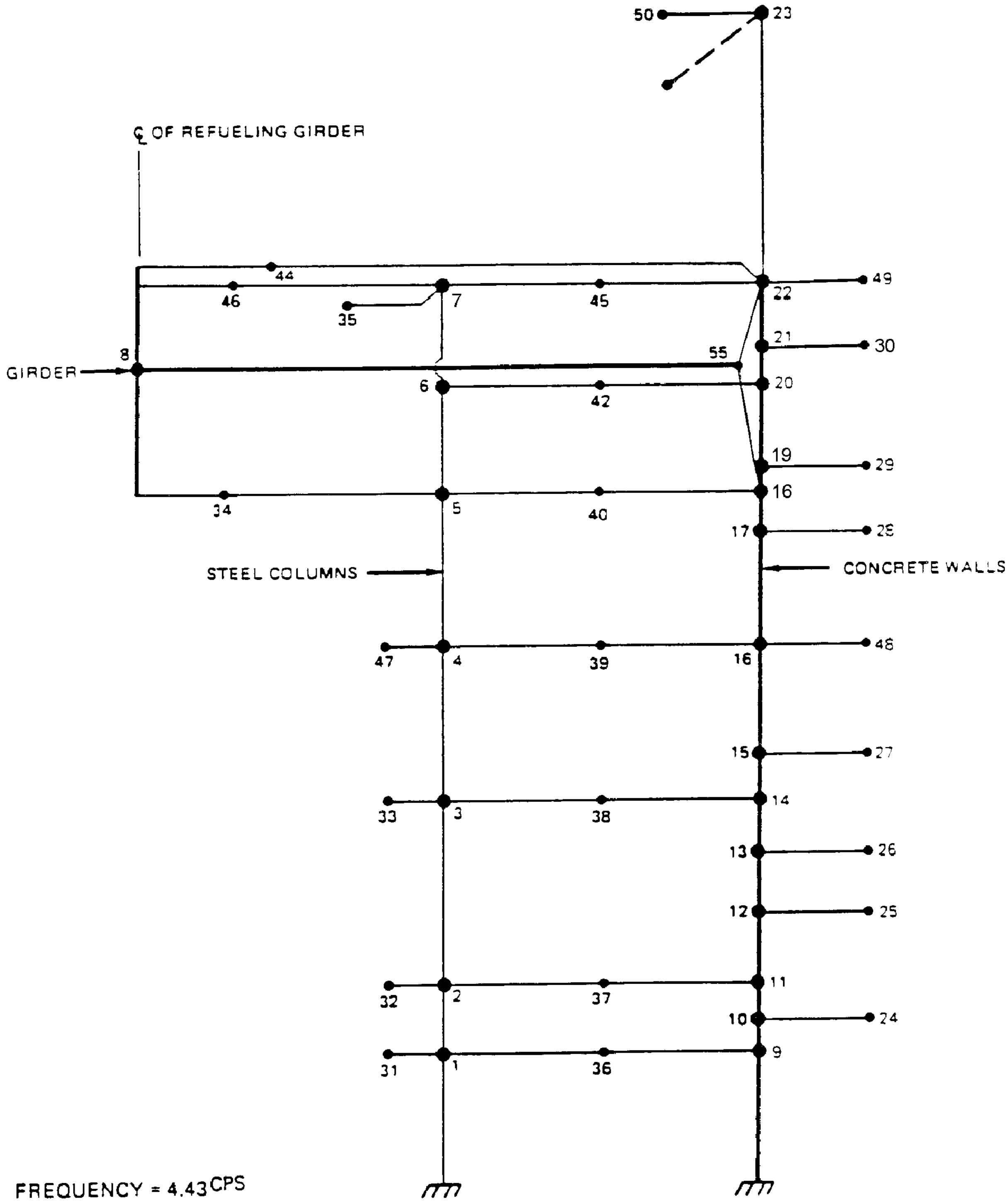
FSAR REV.65

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

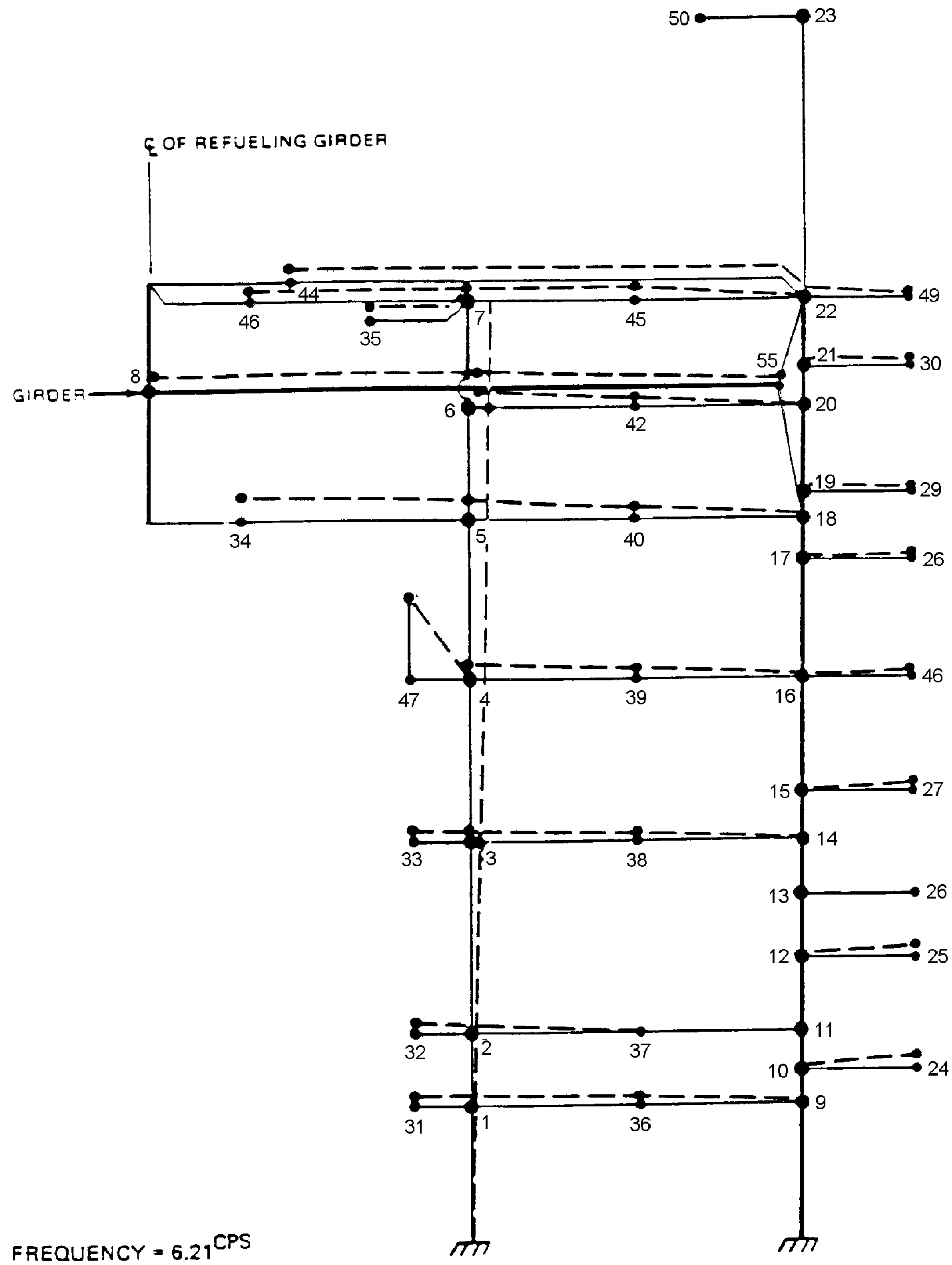
Auto-Cad Figure Fsar 3_7B_36.dwg



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



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
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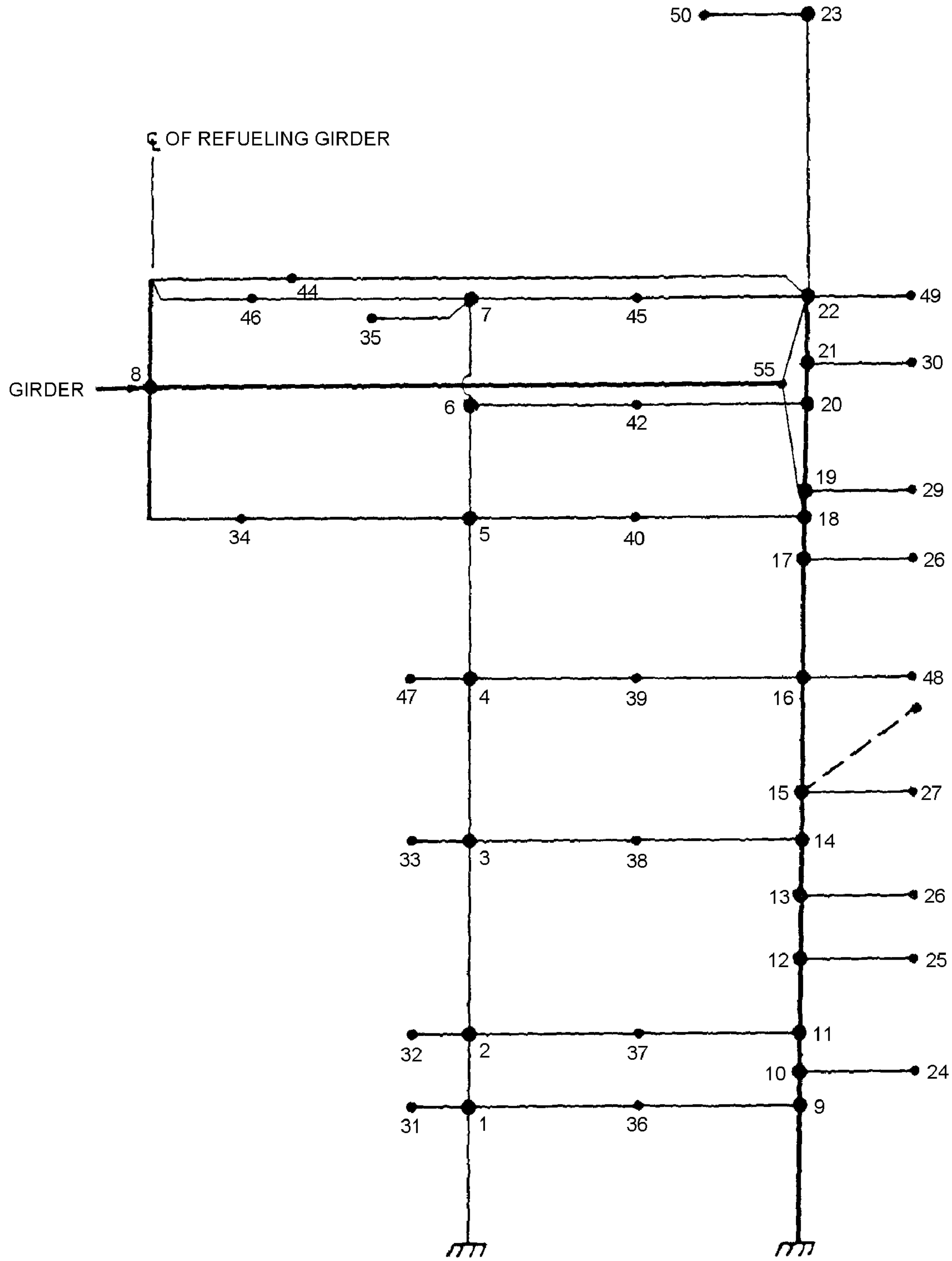
REACTOR AND CONTROL BUILDING
VERTICAL MODE SHAPES
MODE 2

FIGURE 3.7B-38, Rev. 55

Auto-Cad Figure Fsar 3_7B_38.dwg



FREQUENCY = 6.80 CPS

FSAR REV.65

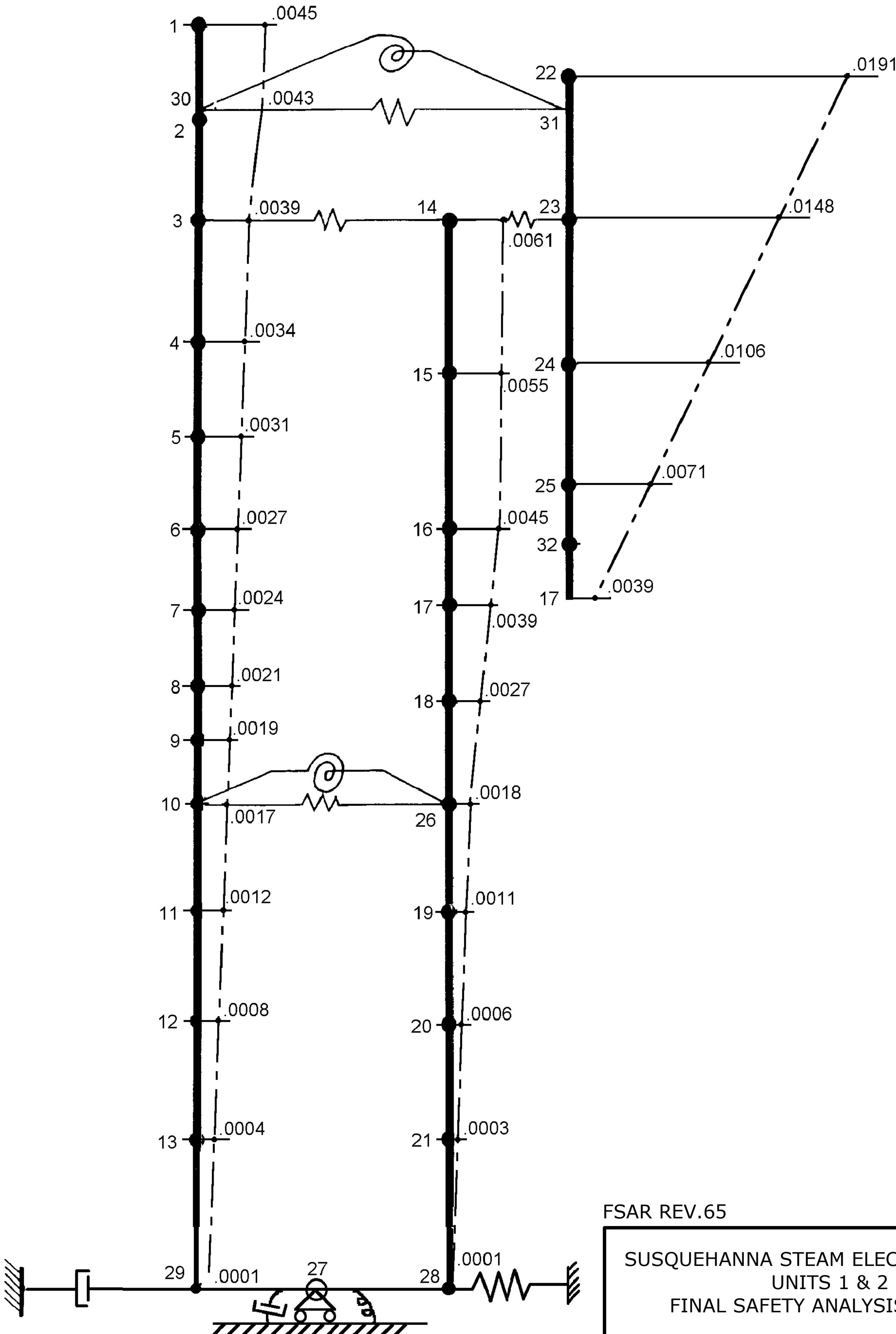
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

REACTOR AND CONTROL BUILDING
VERTICAL MODE SHAPES
MODE 3

FIGURE 3.7B-39, Rev. 55

Auto-Cad Figure Fsar 3_7B_39.dwg

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.



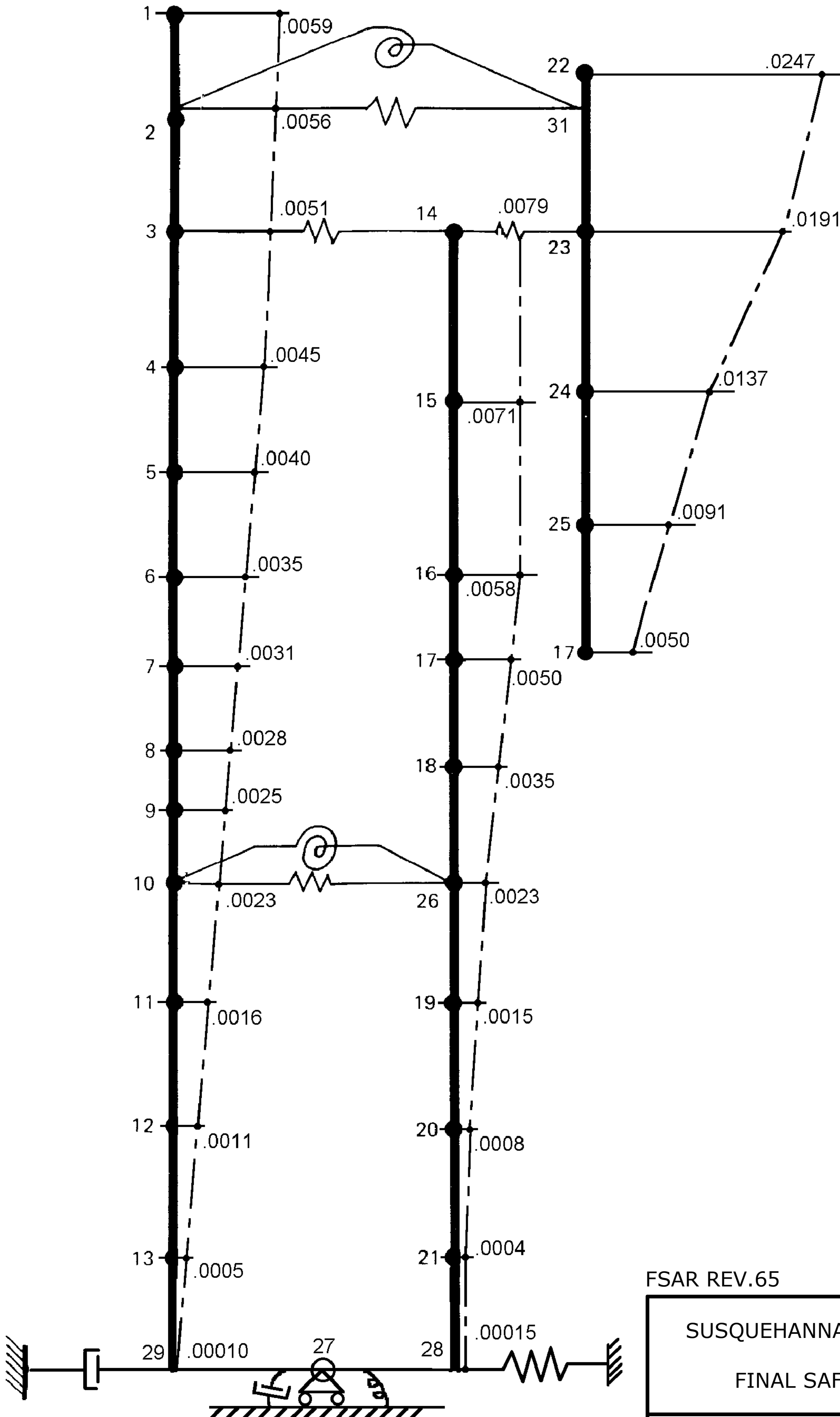
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

CONTAINMENT
HORIZONTAL REPLACEMENTS
OBE

FIGURE 3.7B-40, Rev. 55

Auto-Cad Figure Fsar 3_7B_40.dwg



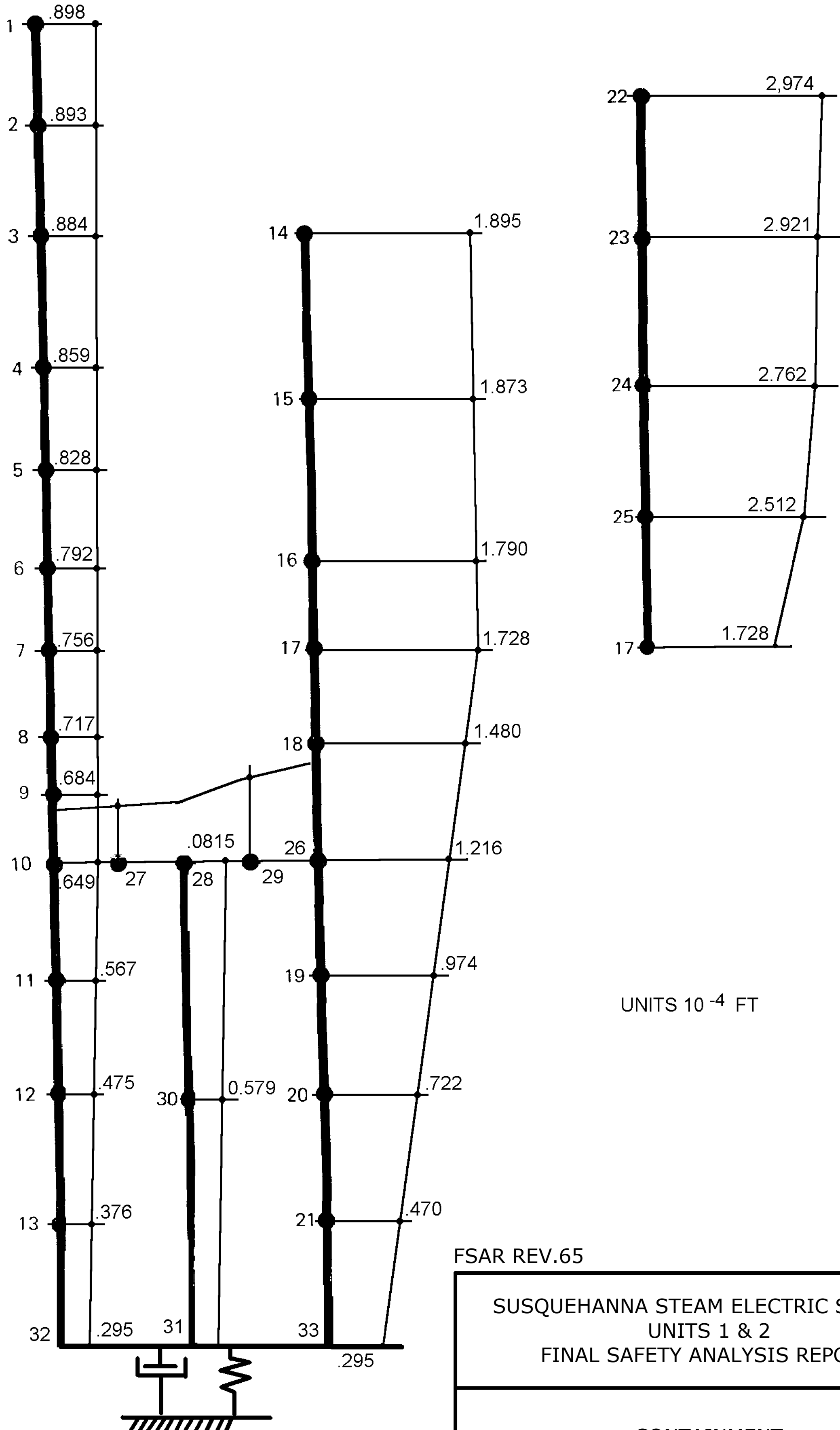
FSAR REV.65

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

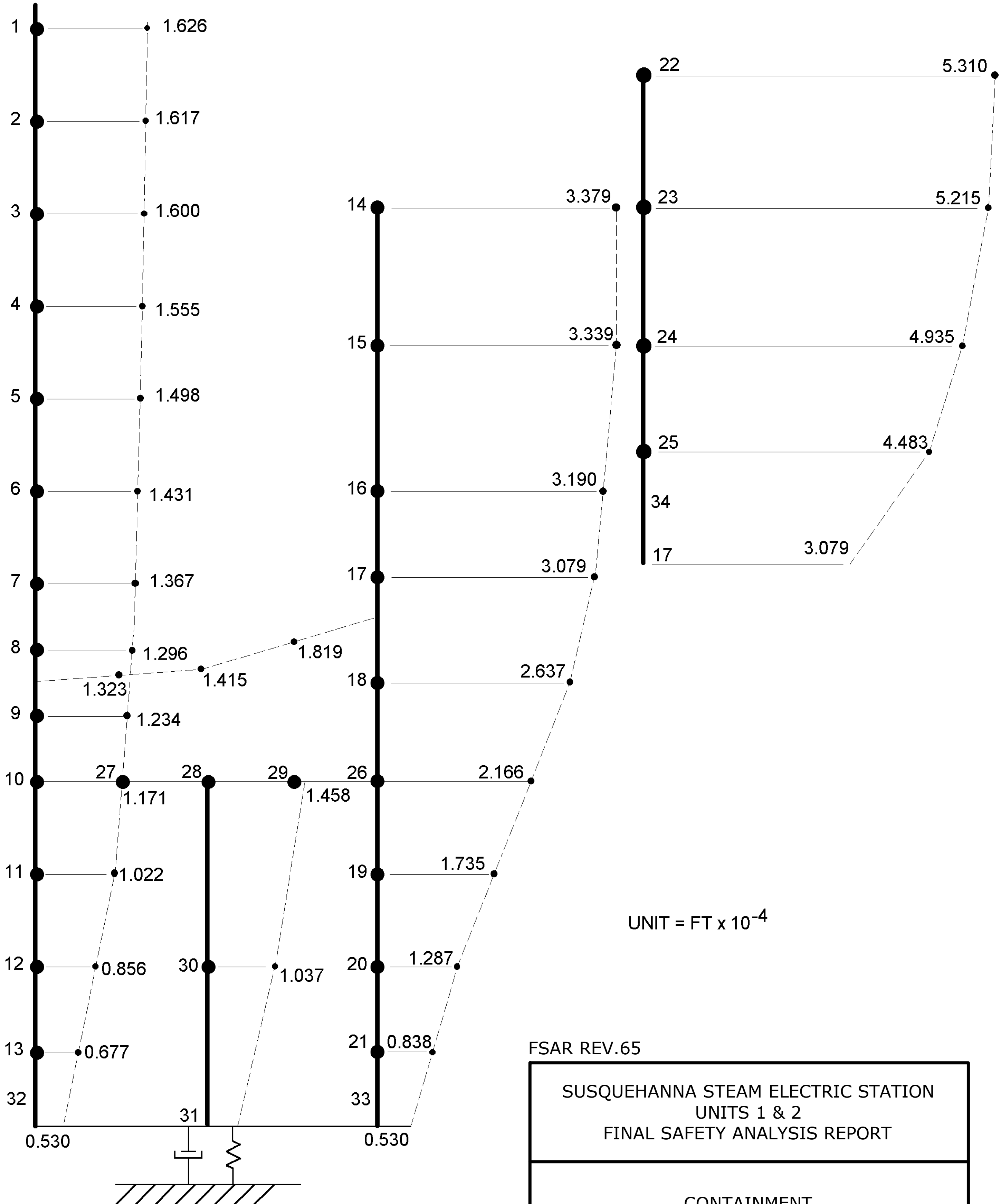


NOTE:
FOR CLARITY OF ILLUSTRATION, THE DISPLACEMENT VALUES ASSOCIATED WITH THE NODES ORIENTATED ALONG THE VERTICAL MEMBERS ARE DISPLAYED GRAPHICALLY IN THE HORIZONTAL DIRECTION. THE DISPLACEMENTS ARE IN THE VERTICAL DIRECTION.

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
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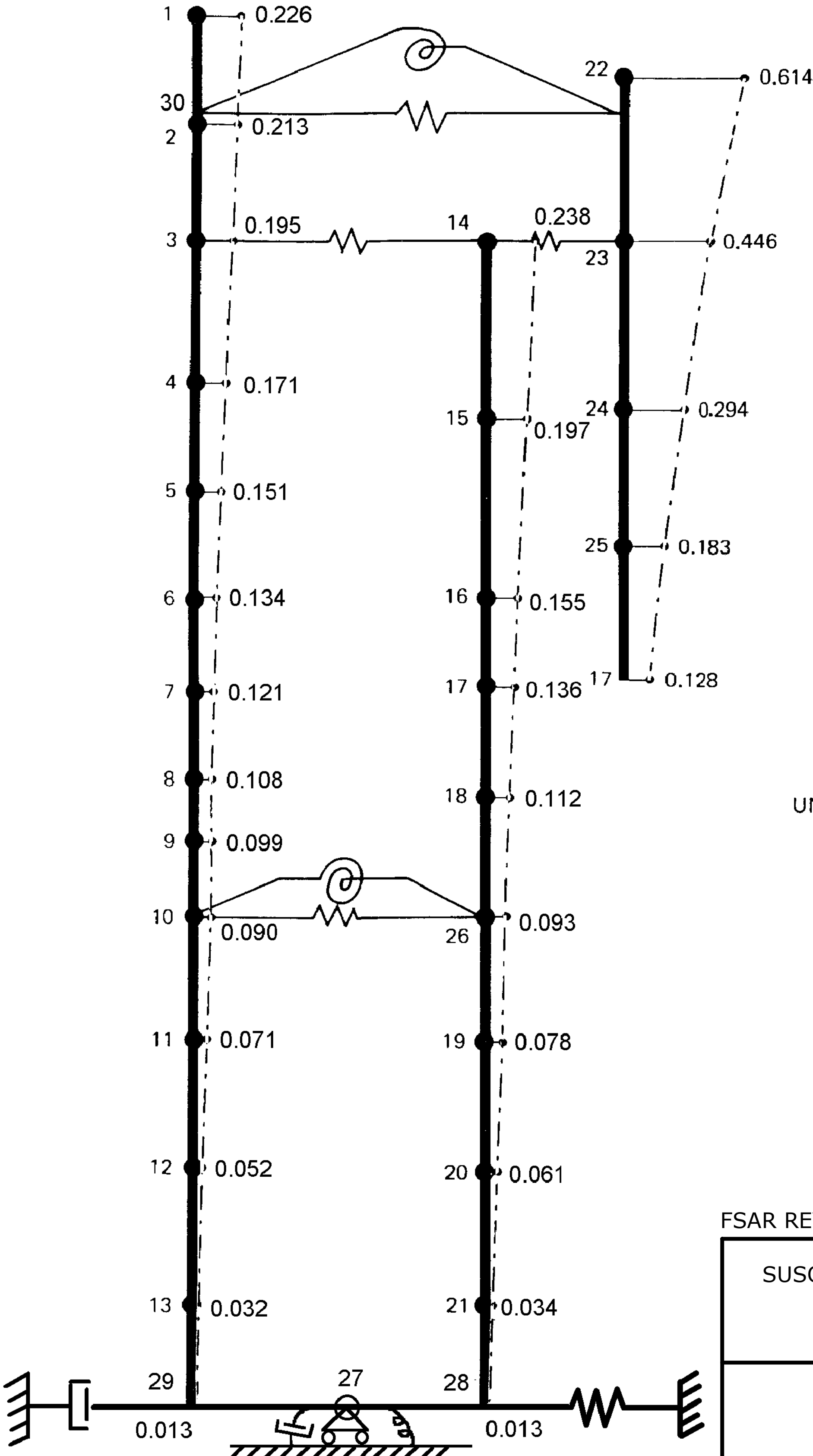
CONTAINMENT
VERTICAL DISPALCEMENTS
SSE

FIGURE 3.7B-43, Rev. 55

Auto-Cad Figure Fsar 3_7B_43.dwg



UNITS: G's

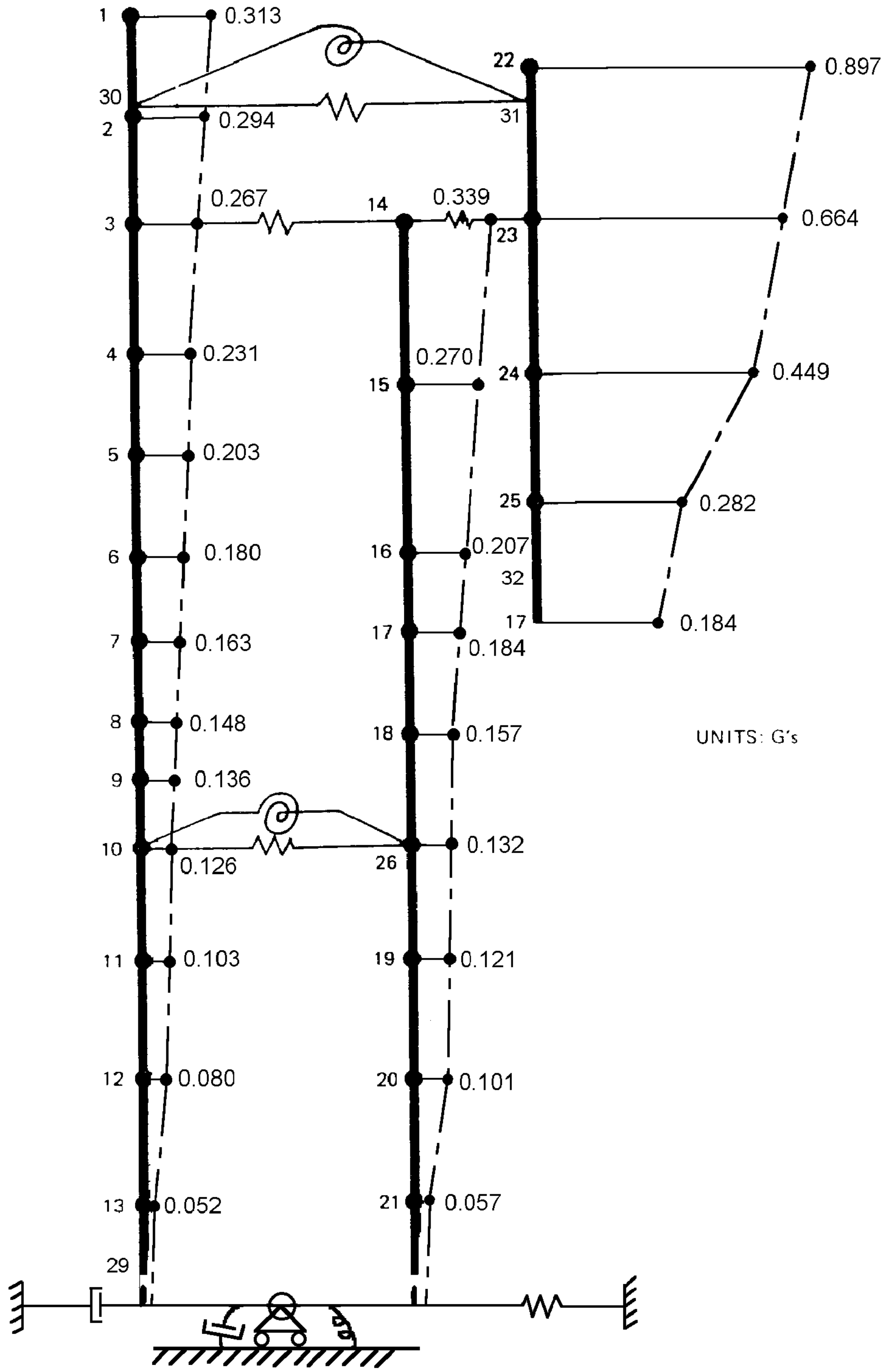
FSAR REV.65

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



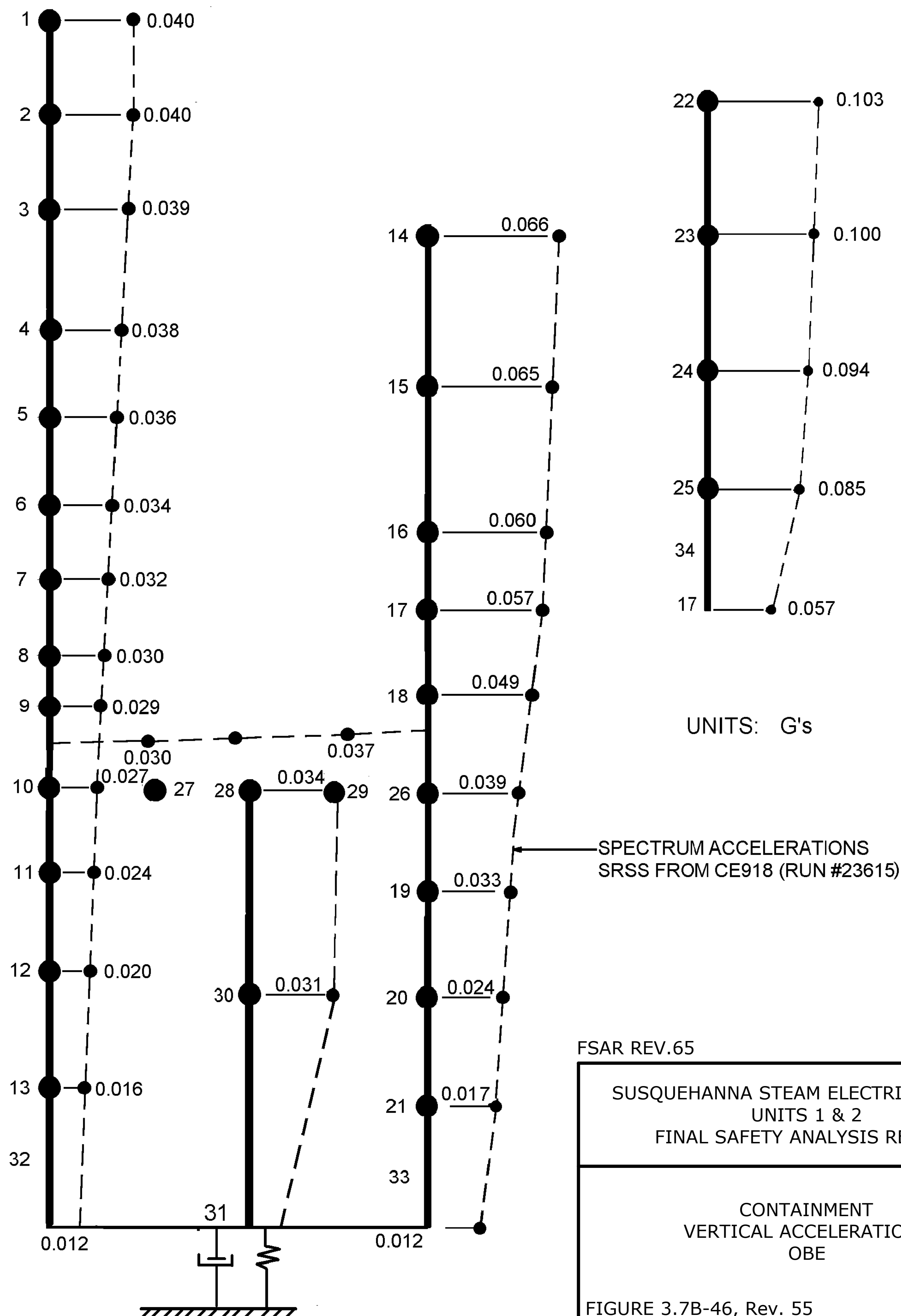
FSAR REV.65

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

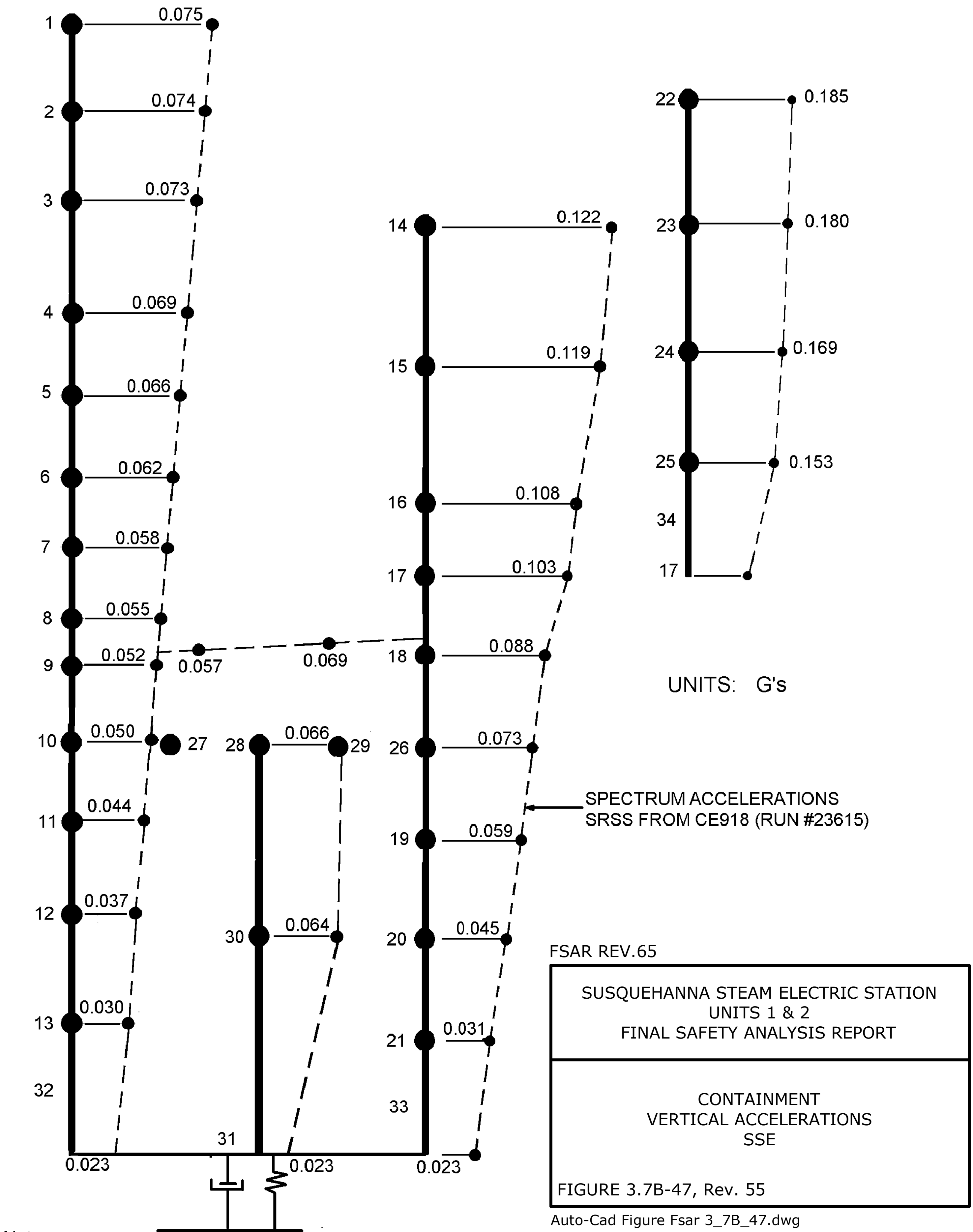
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

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

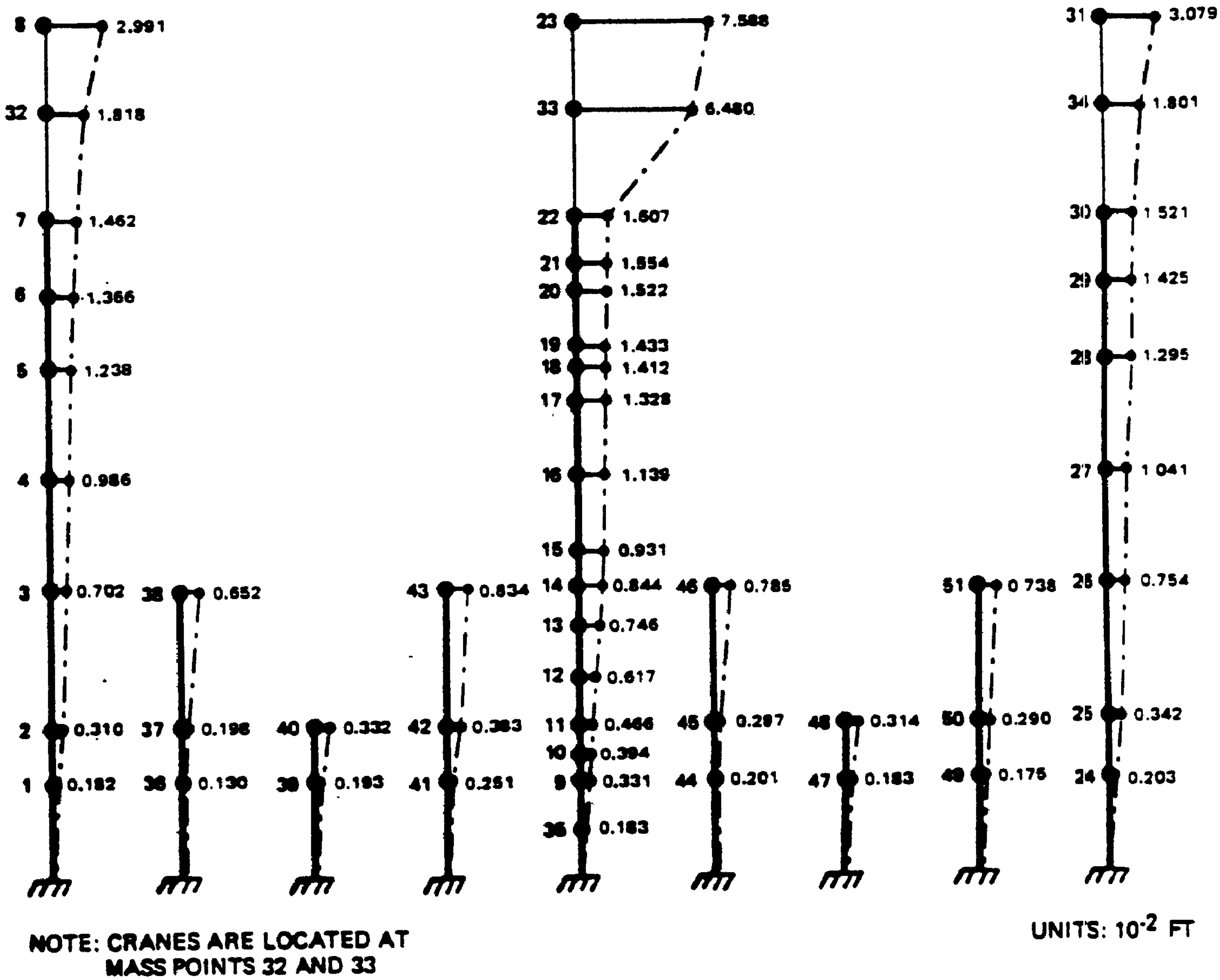
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

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.



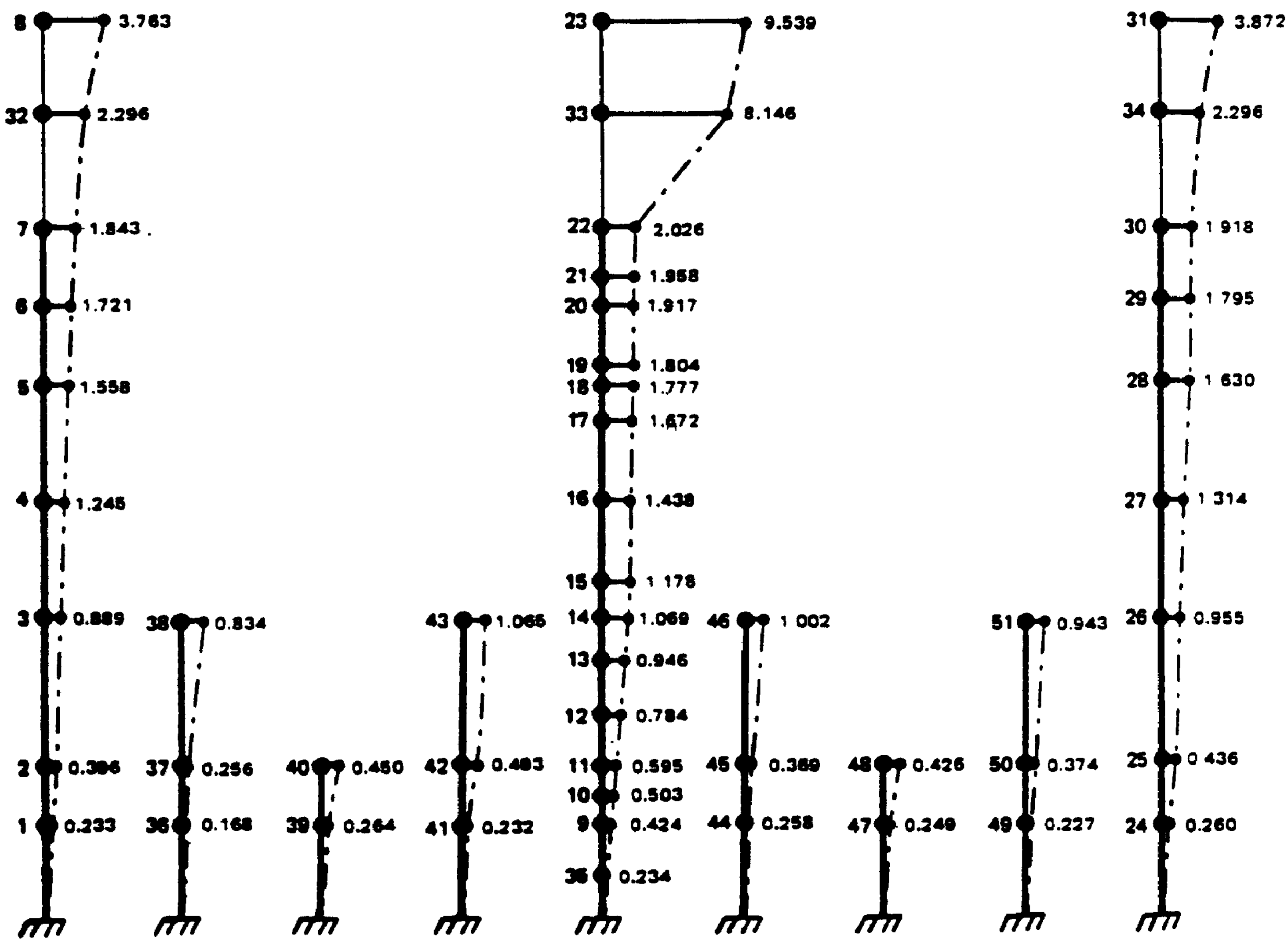
FSAR REV.65

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

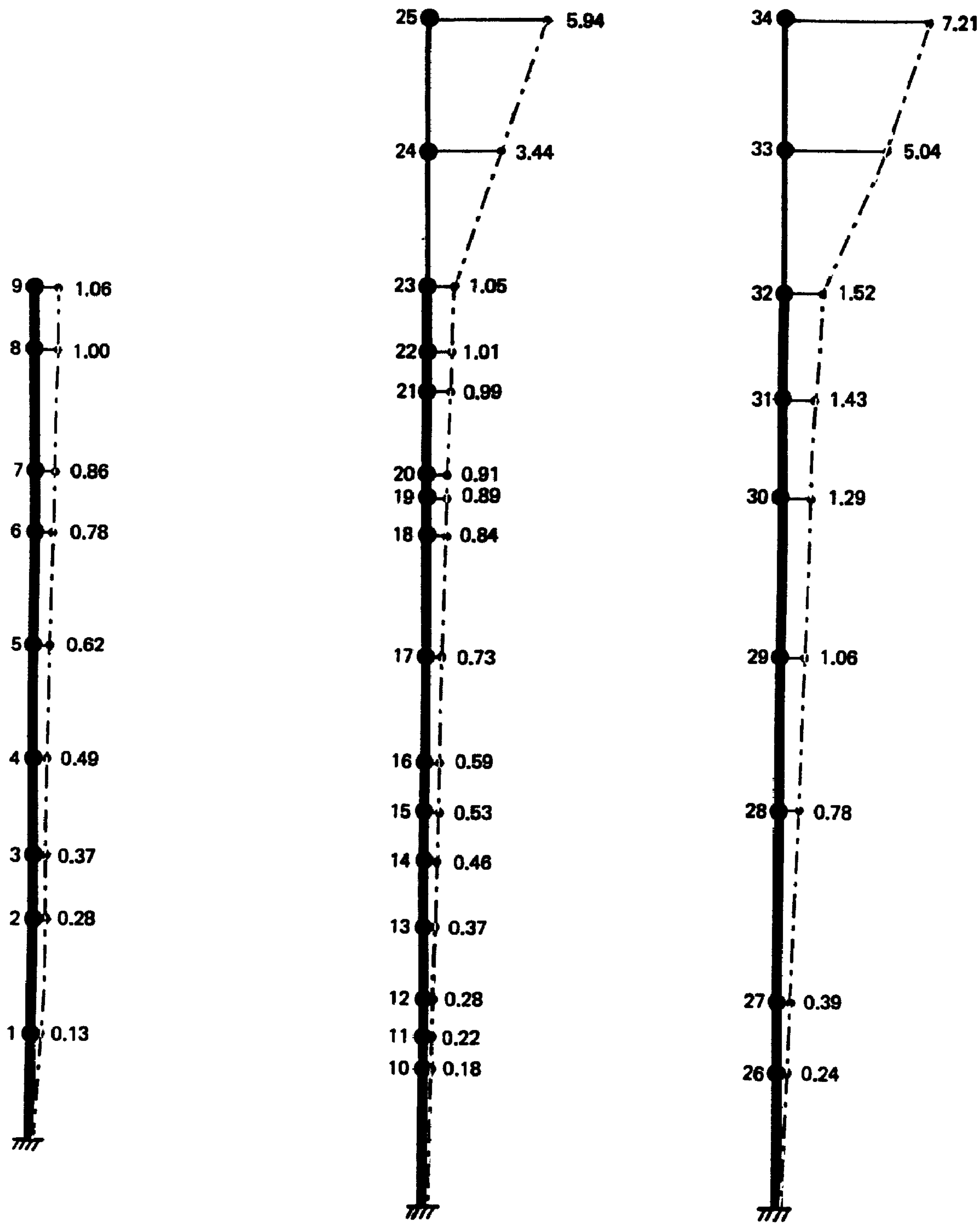
FSAR REV.65

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



UNITS: 10^{-2} FT

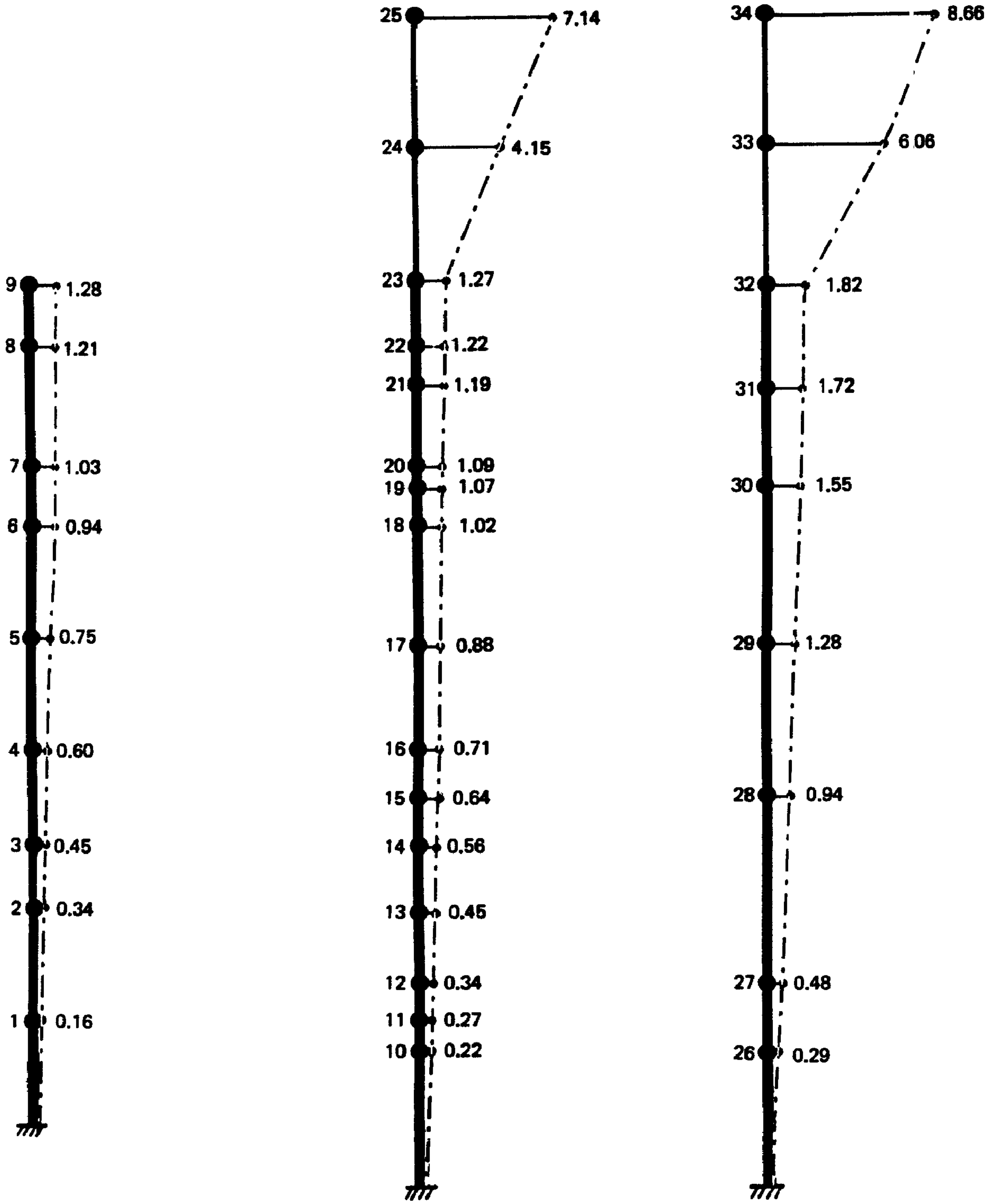
FSAR REV.65

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

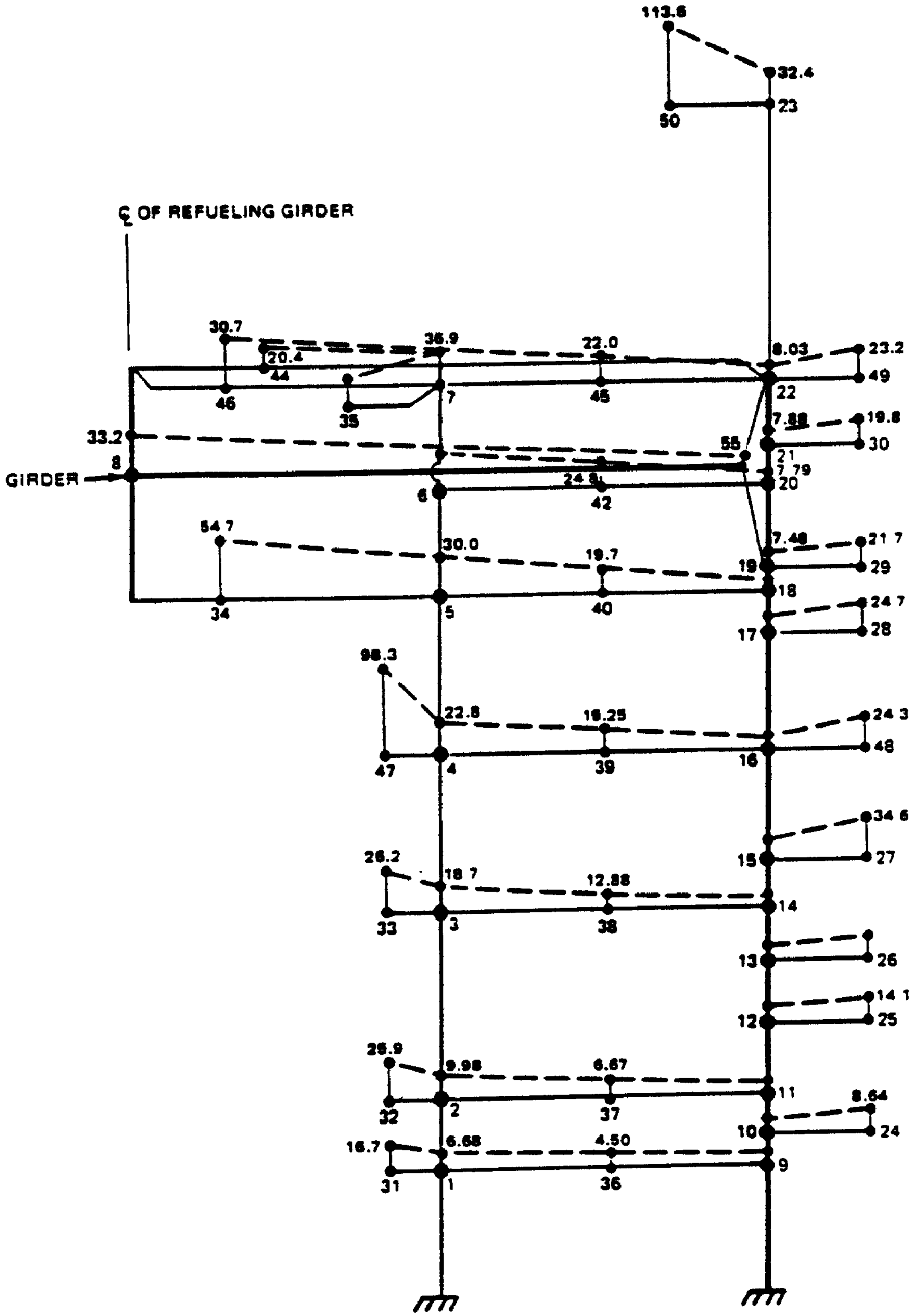
UNITS: 10^{-2} FT

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



UNITS = 10^{-4} FT

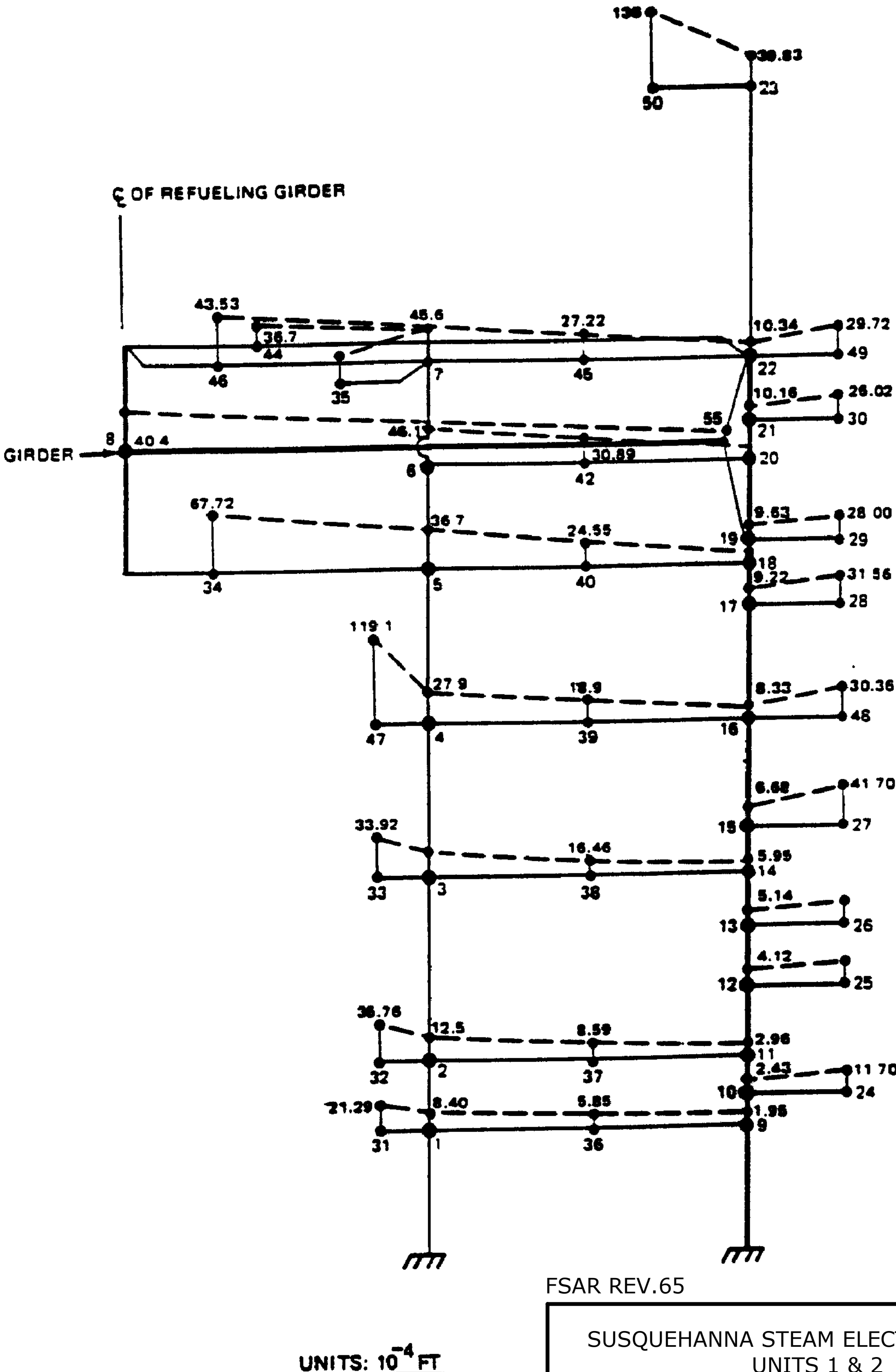
FSAR REV.65

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



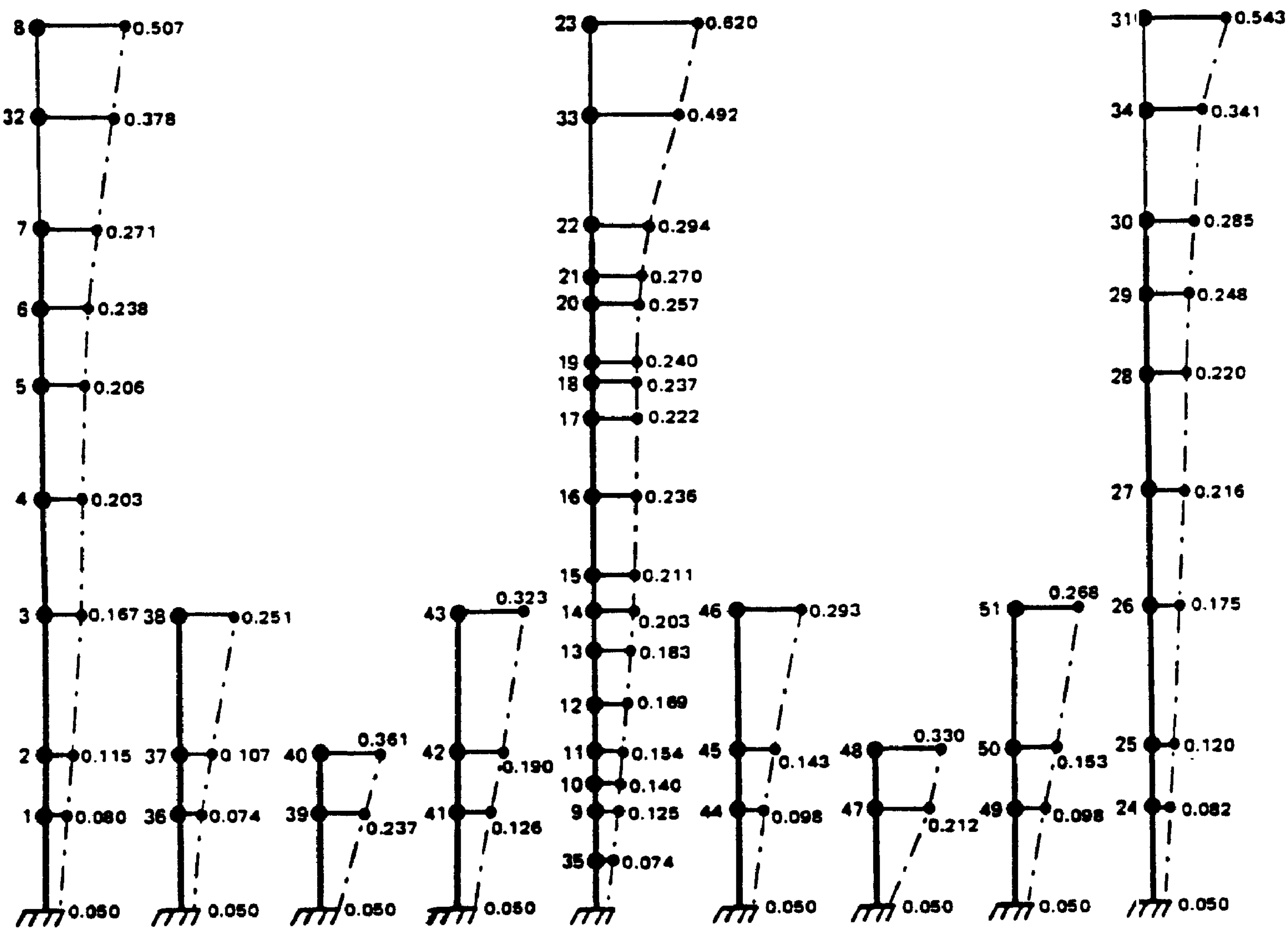
FSAR REV.65

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

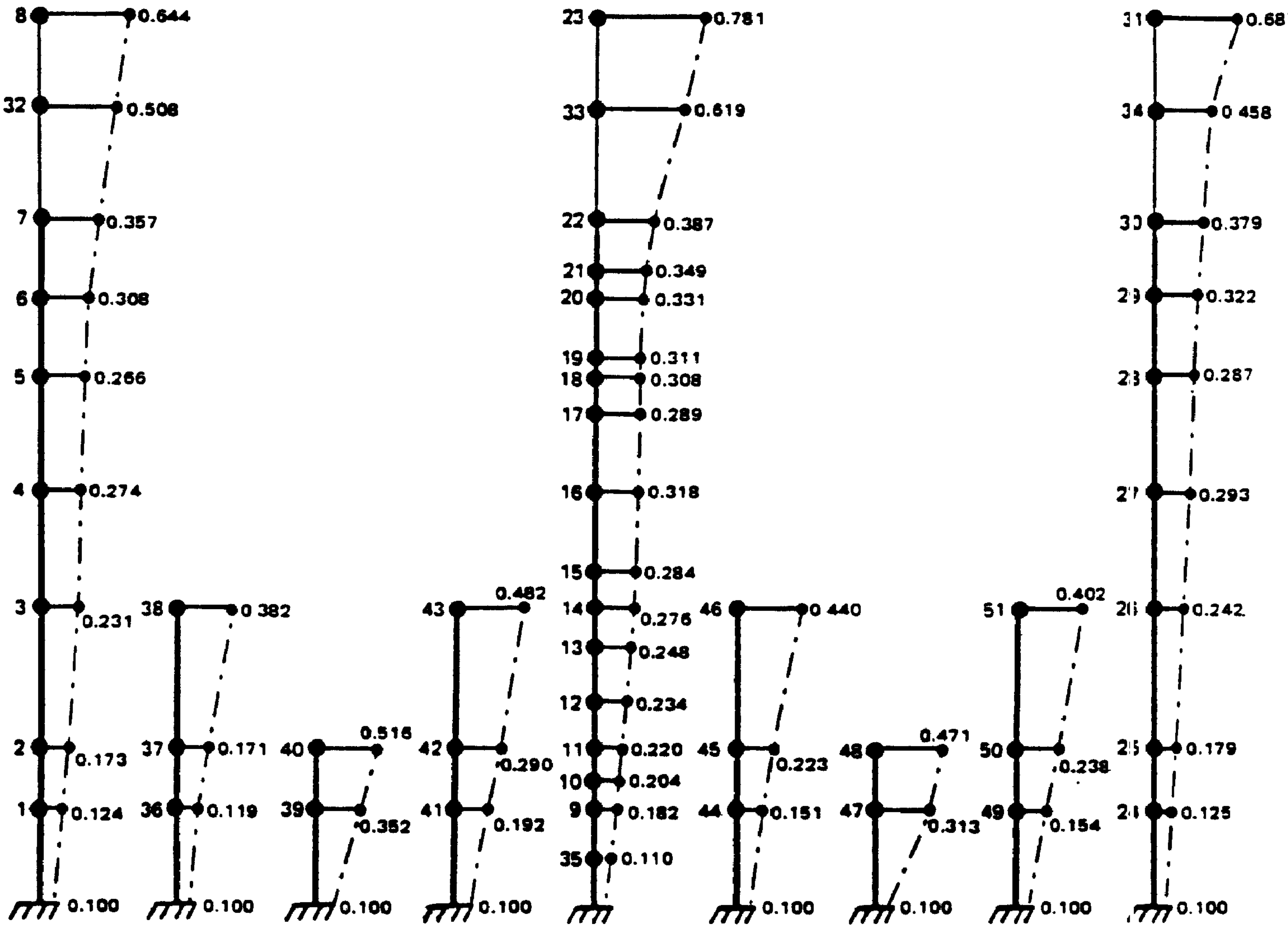
FSAR REV.65

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
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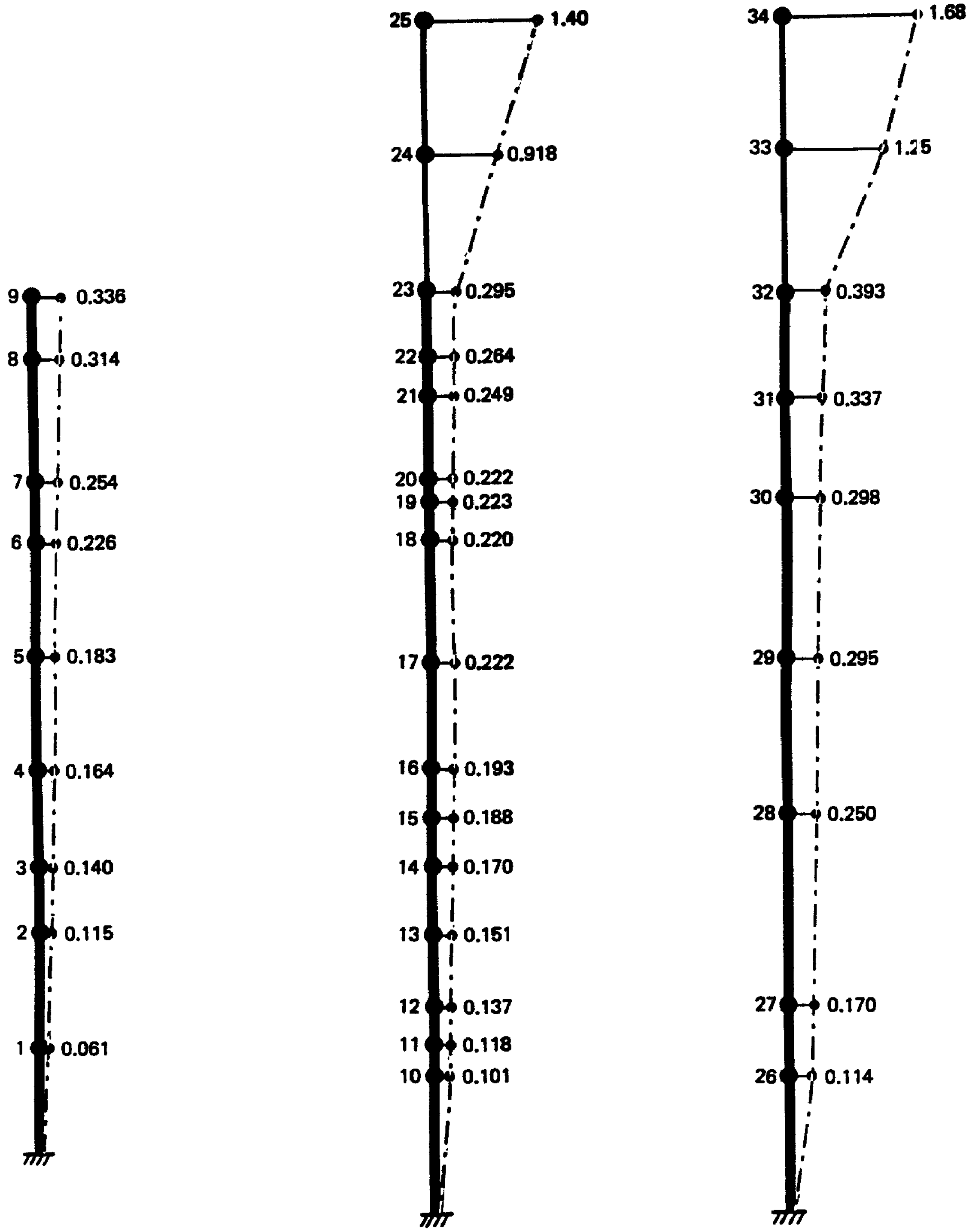
REACTOR AND CONTROL BUILDING
E-W ACCELERATIONS
SSE

FIGURE 3.7B-55, Rev. 55

Auto-Cad Figure Fsar 3_7B_55.dwg



UNITS: G's

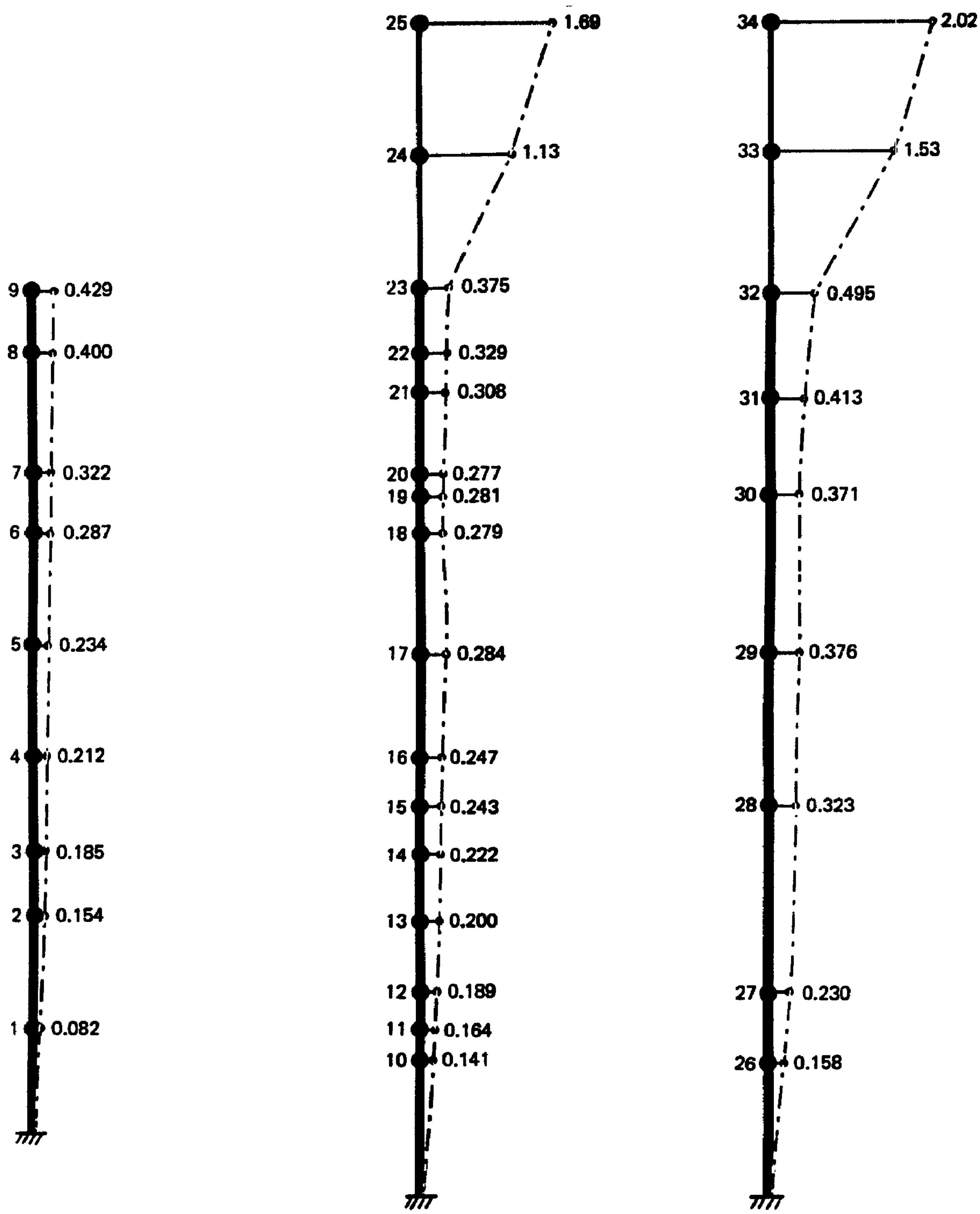
FSAR REV.65

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

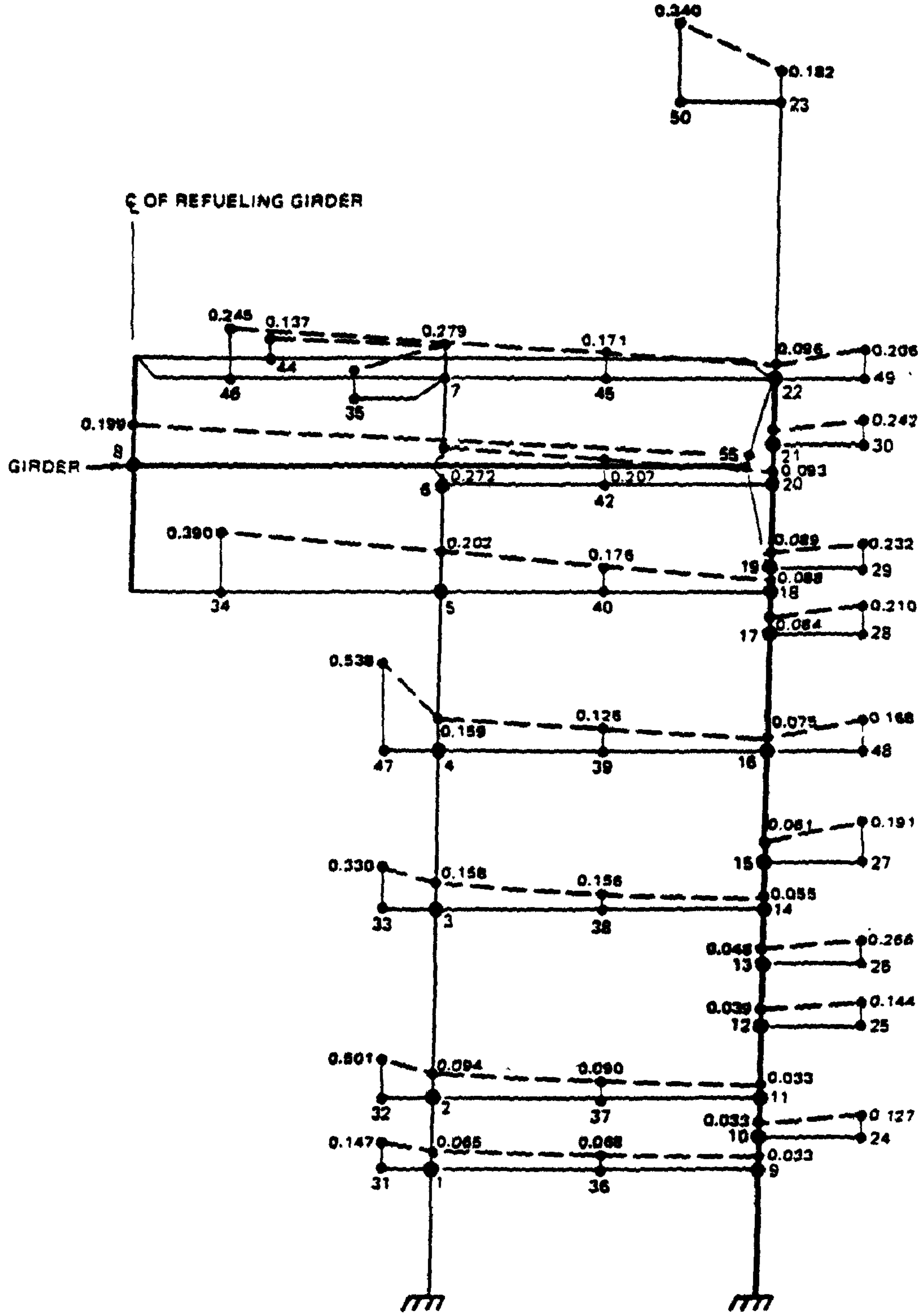
FSAR REV.65

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



UNITS = G's

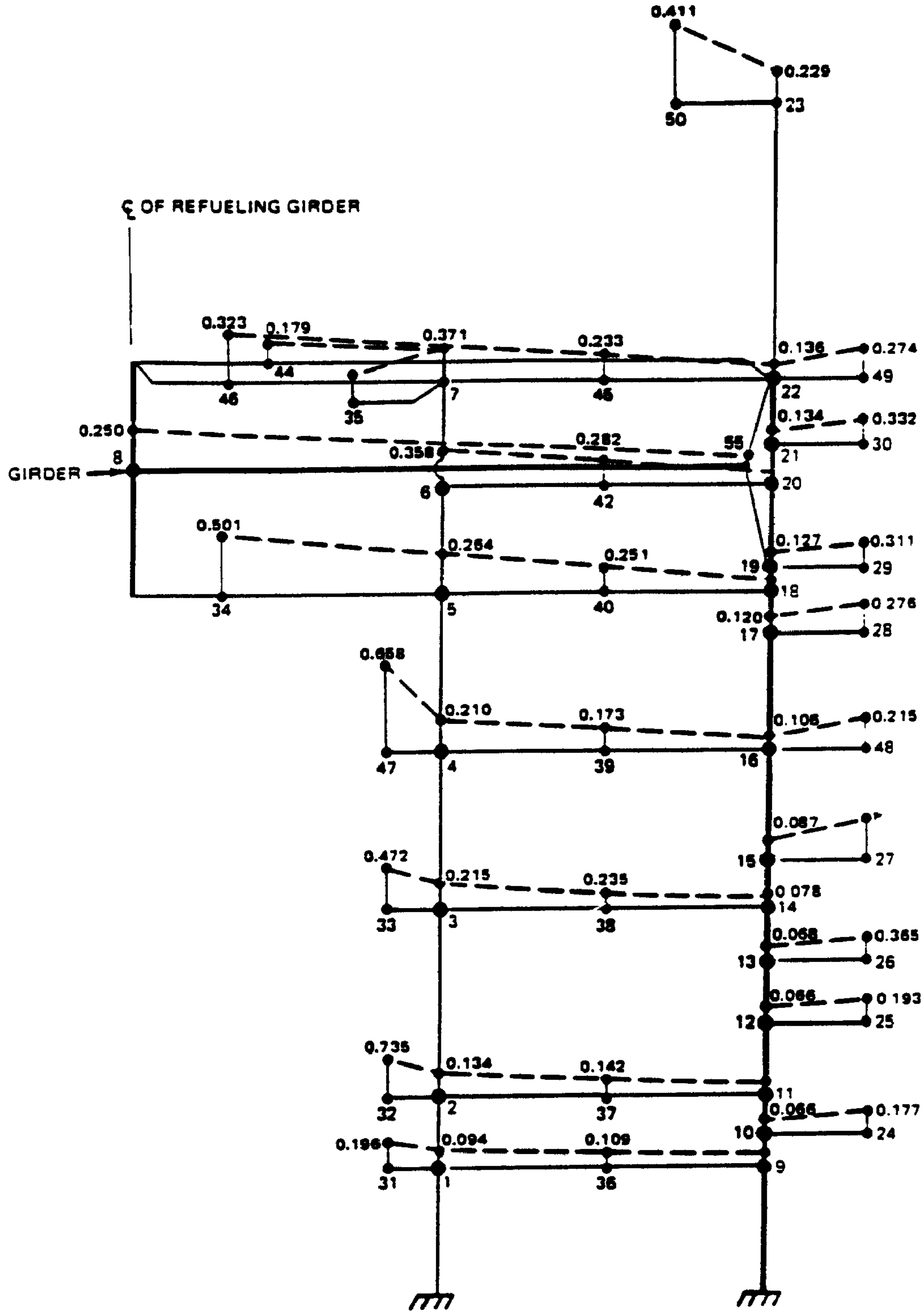
FSAR REV.65

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



FSAR REV.65

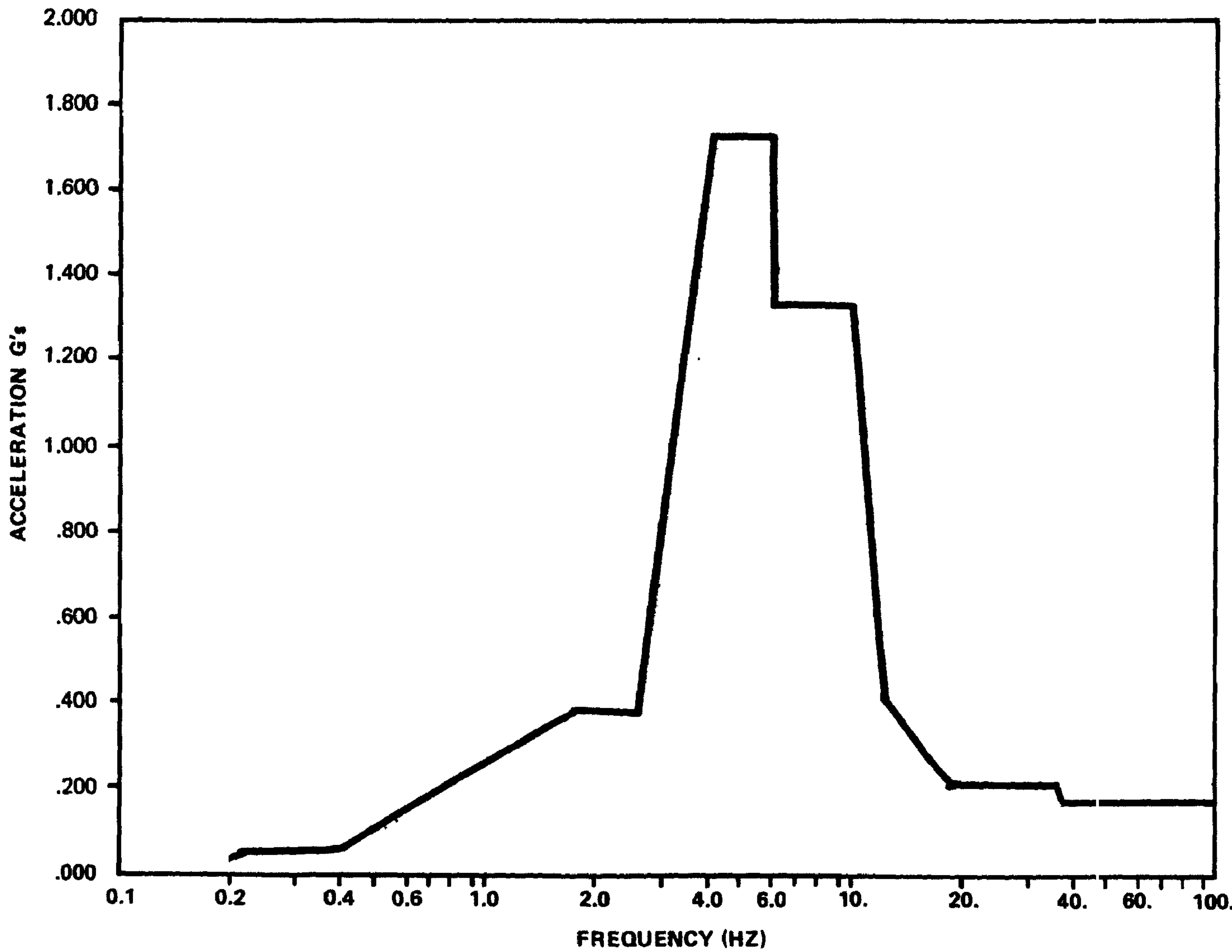
UNITS: G's

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

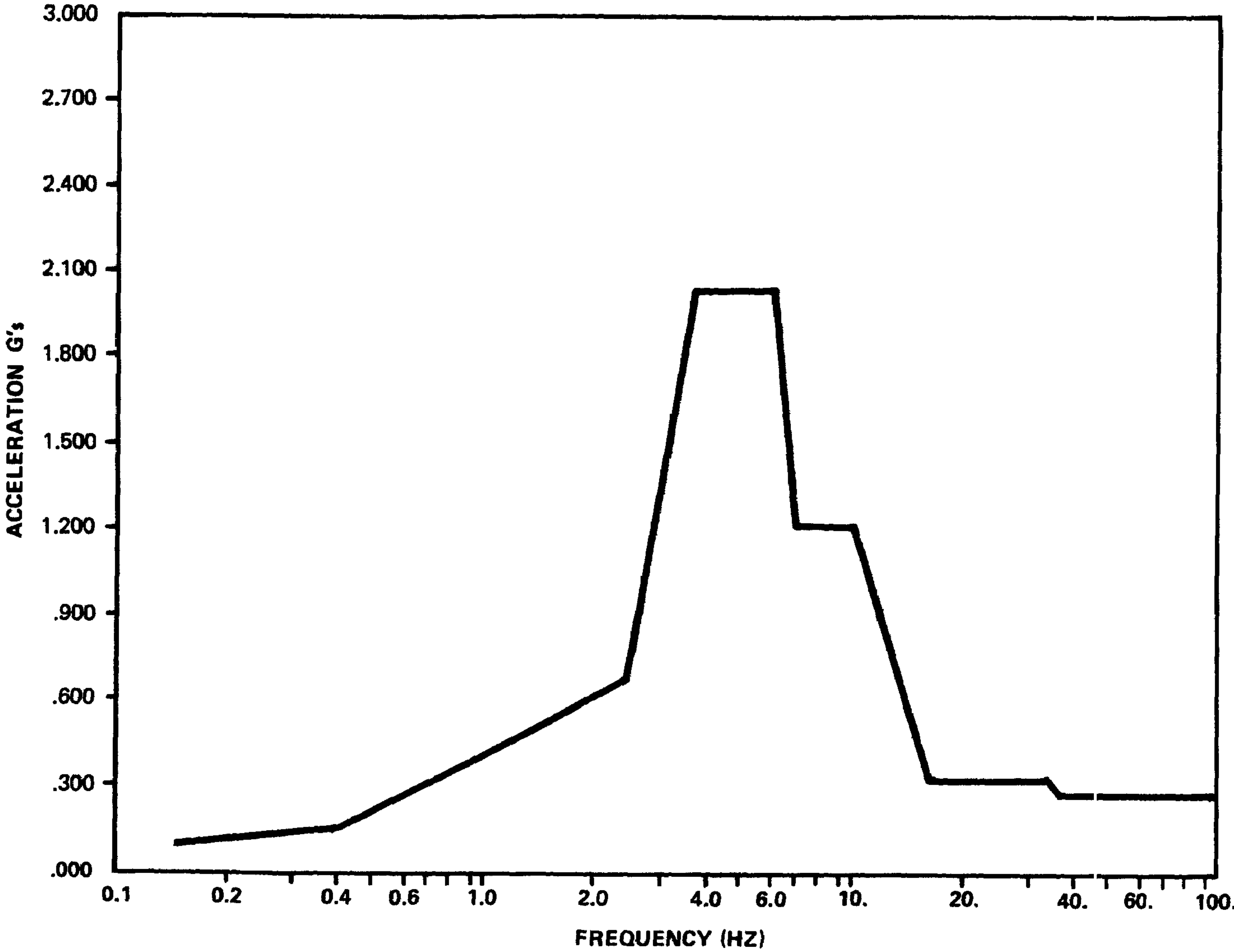
FSAR REV.65

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

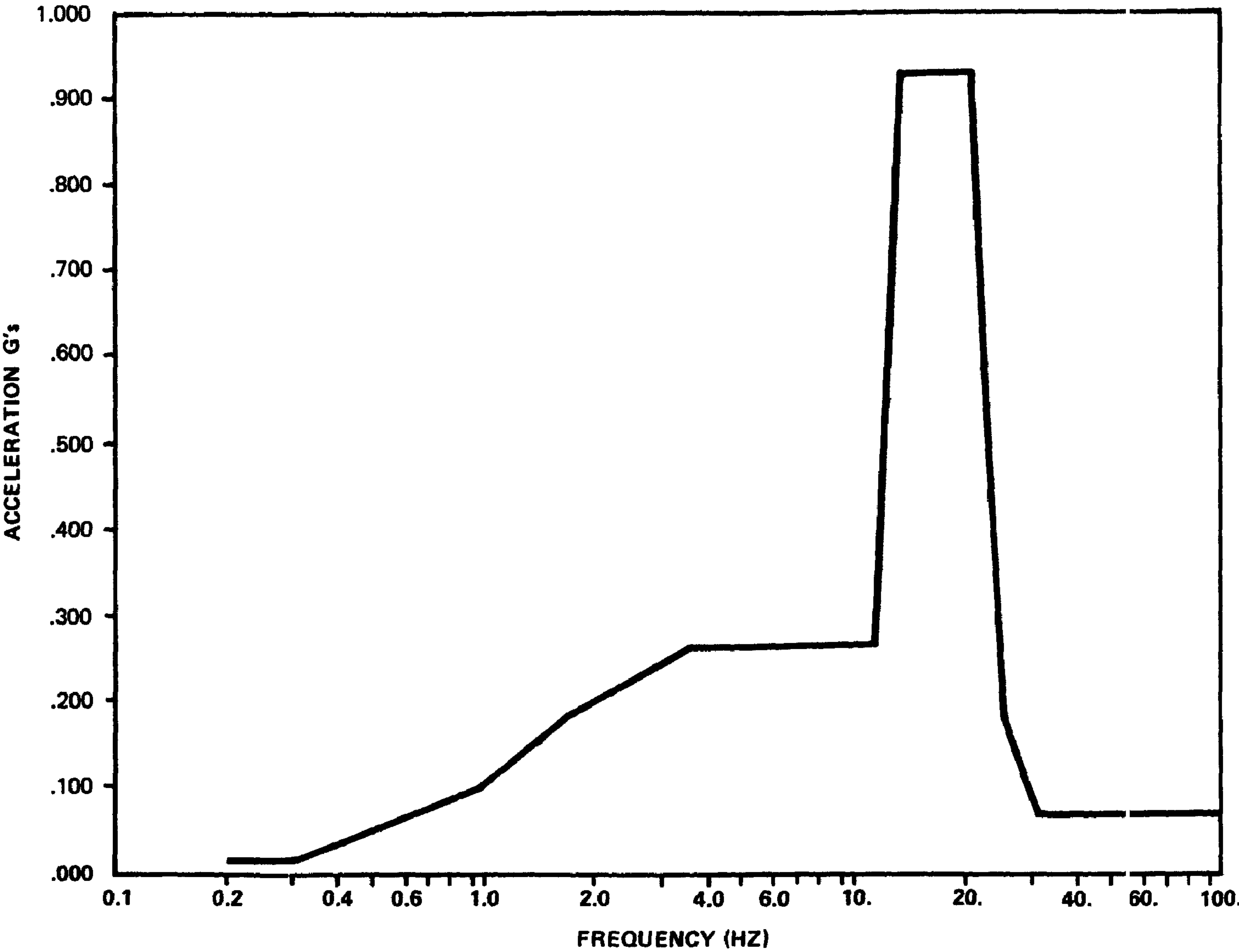
FSAR REV.65

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

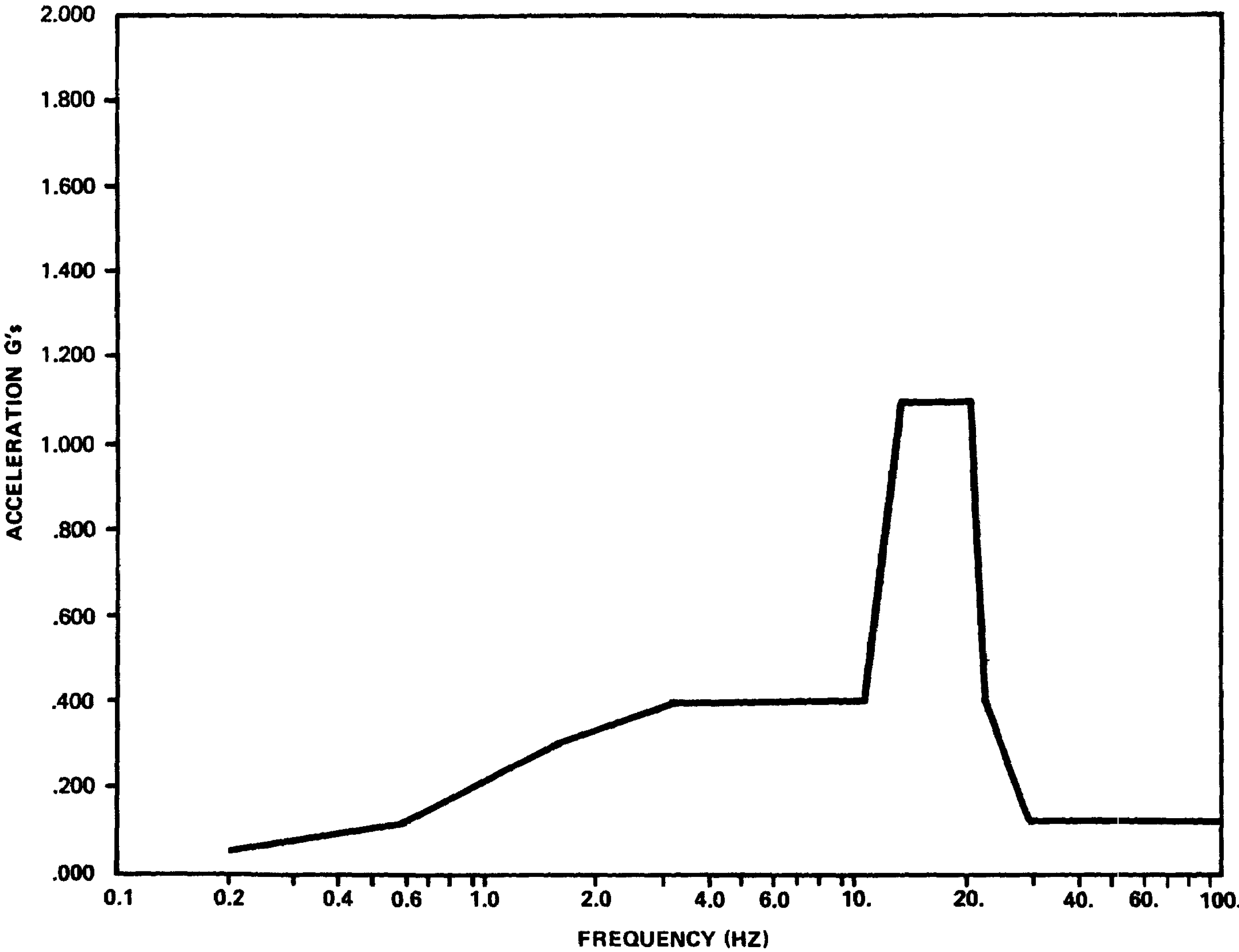
FSAR REV.65

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

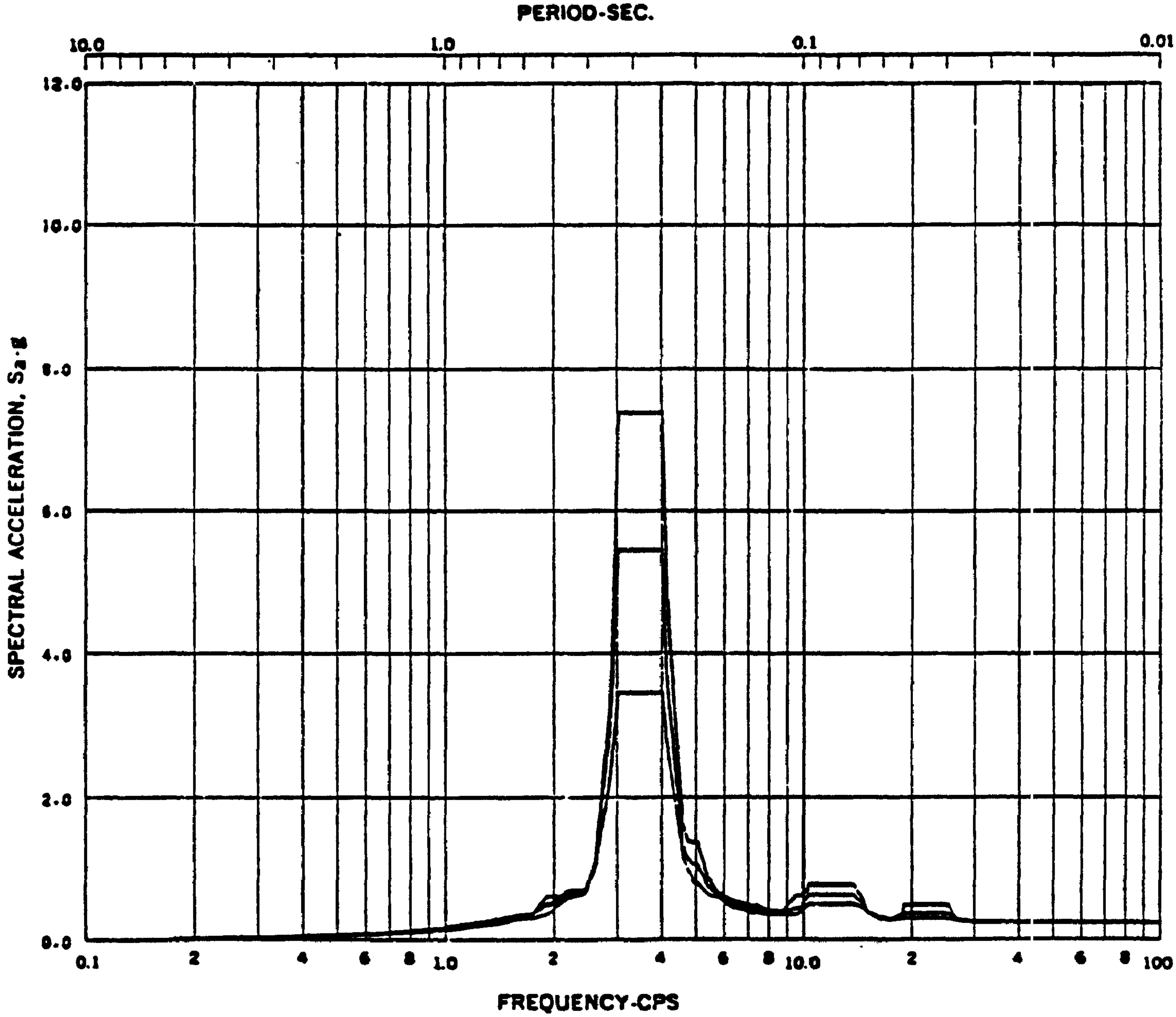
FSAR REV.65

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

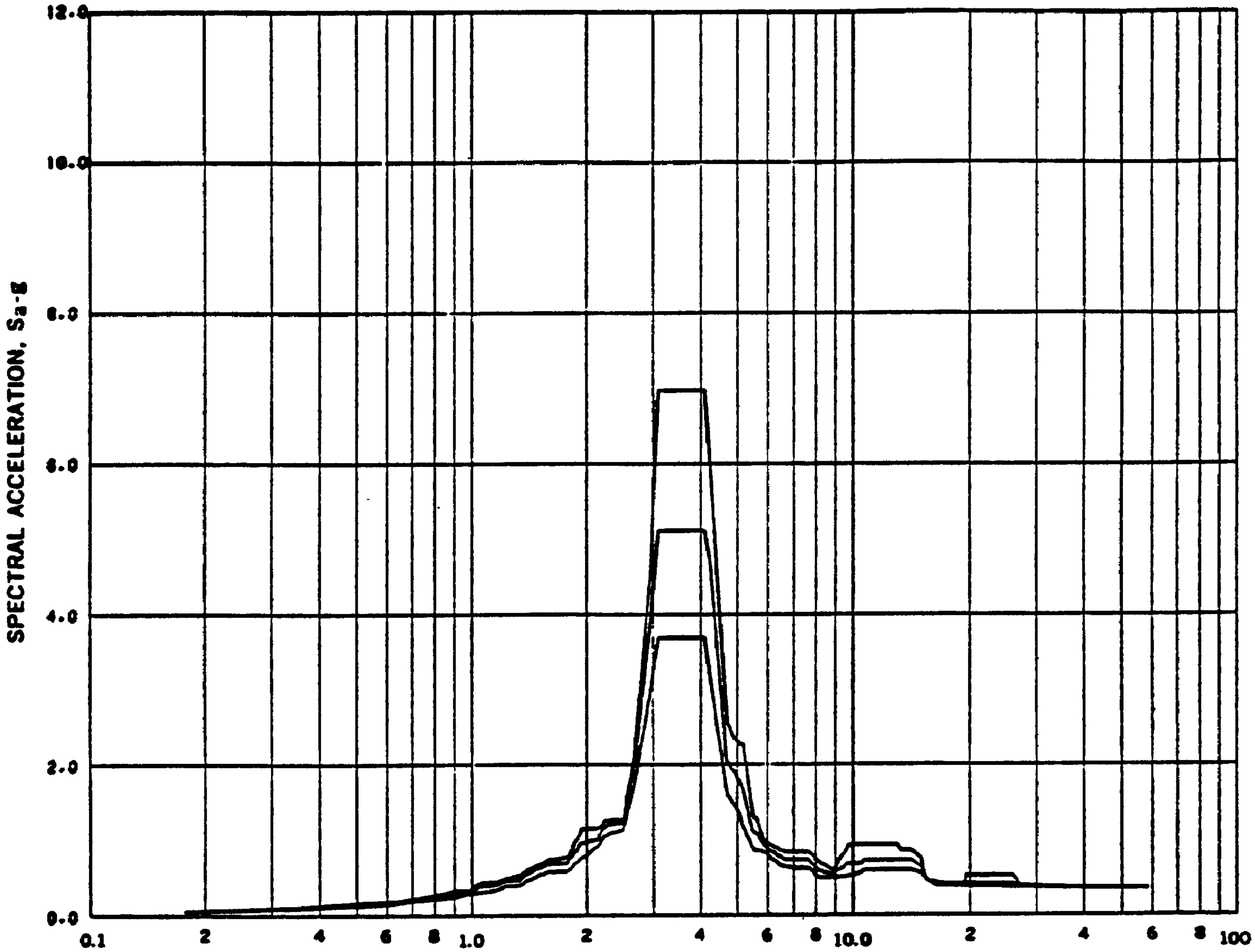
FSAR REV.65

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

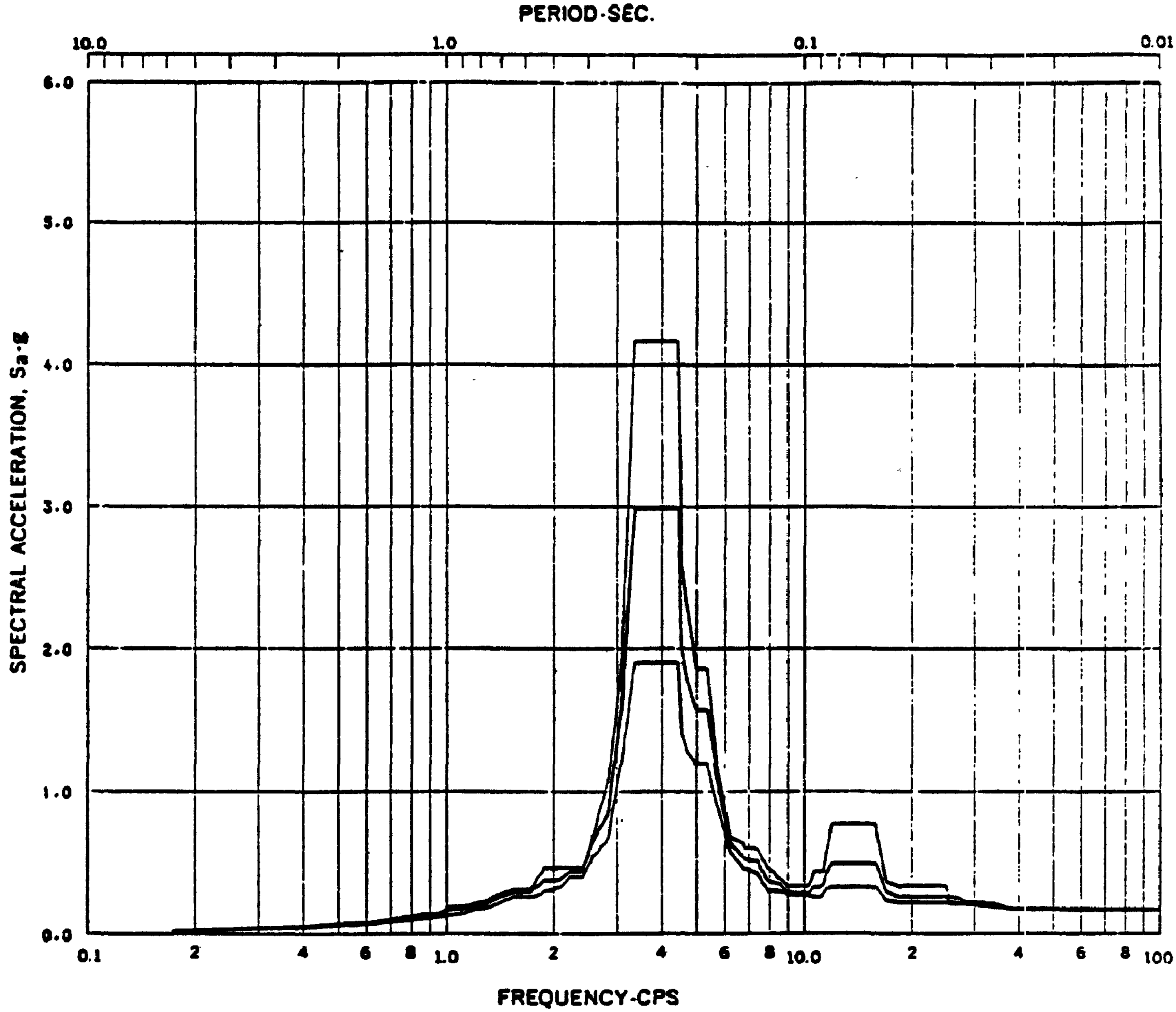
NODE 22, DIRECTION EW, ELEV 818'-1

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

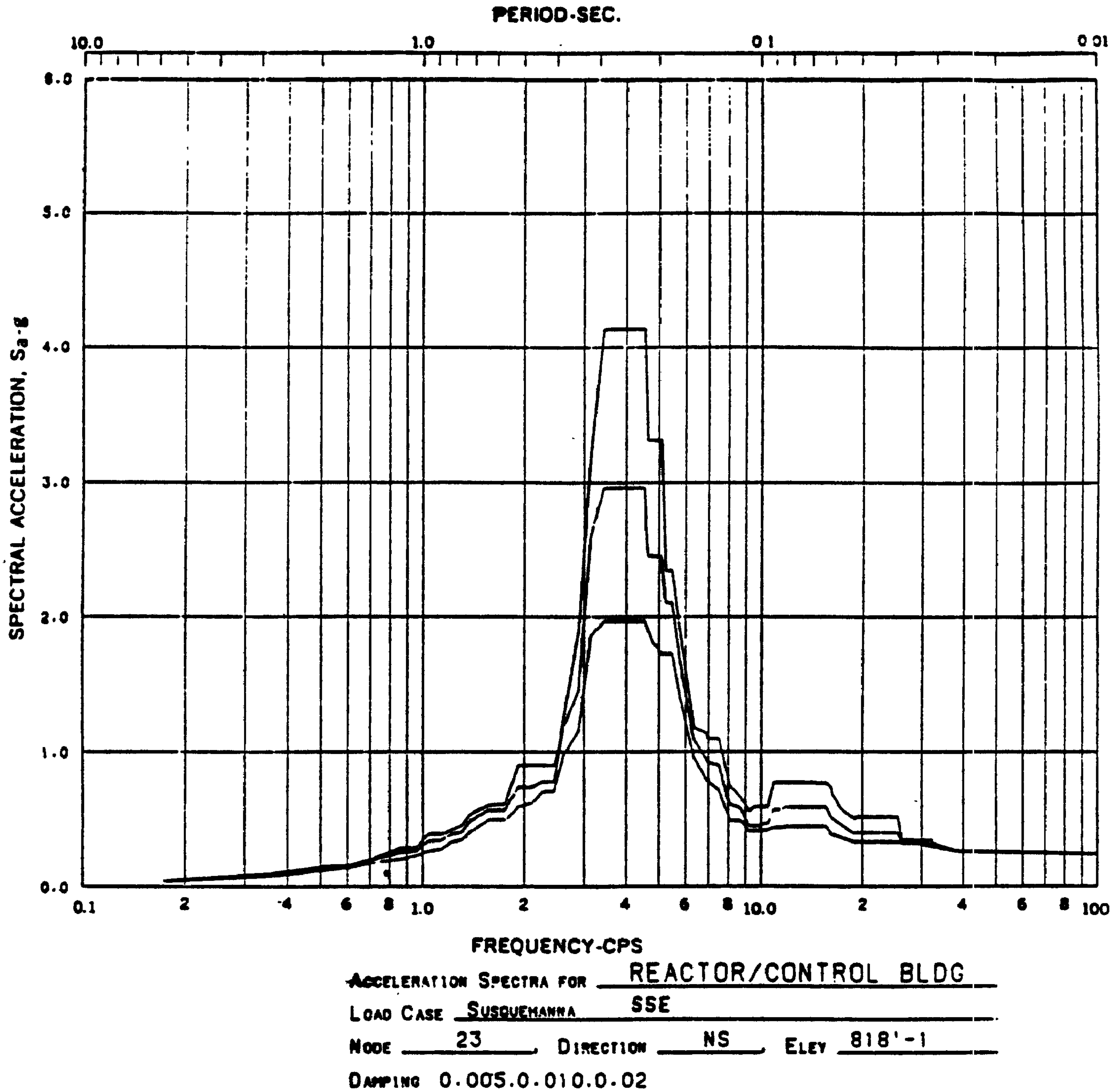
FSAR REV.65

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



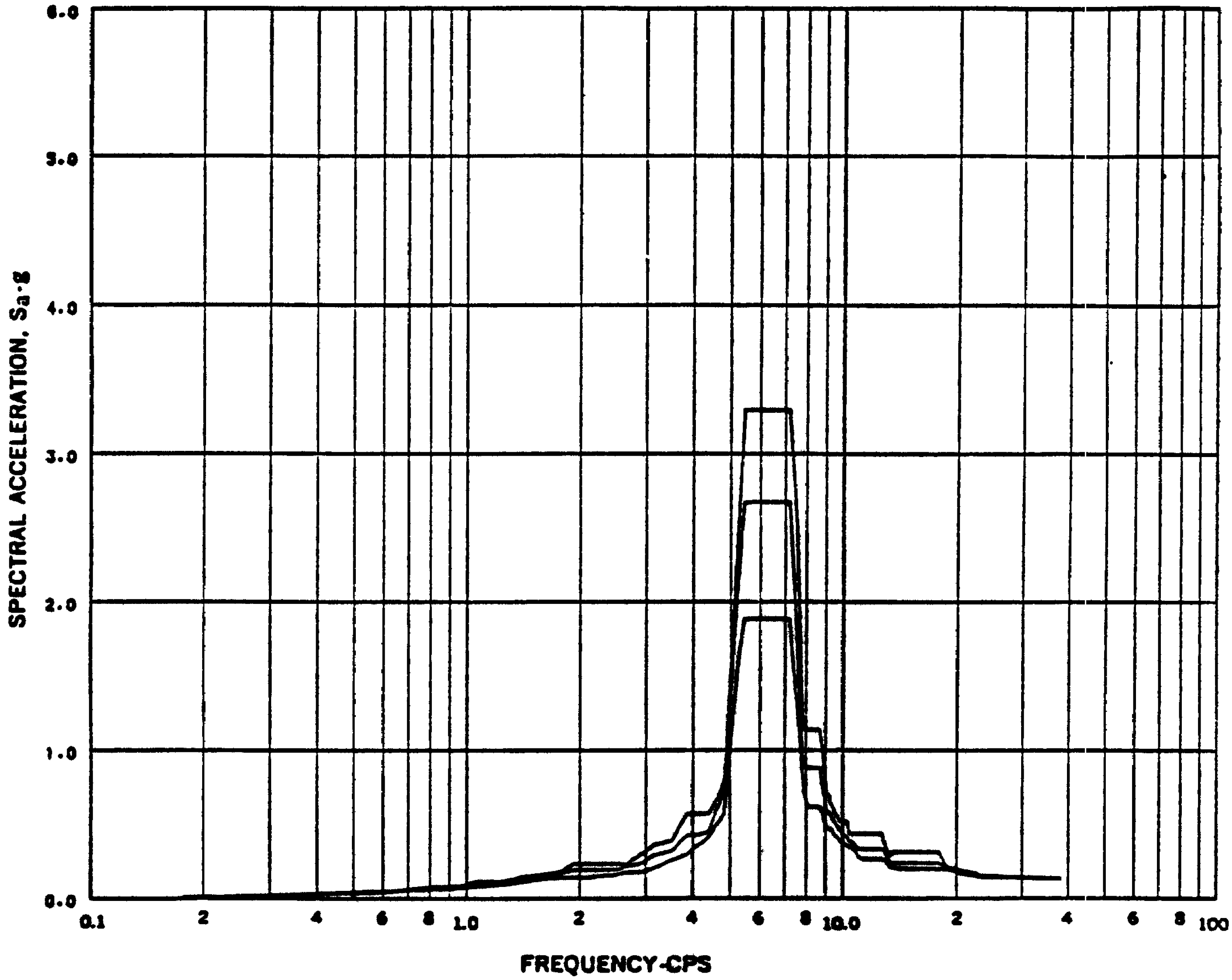
FSAR REV.65

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



ACCELERATION SPECTRA FOR REFUELING GIRDER

LOAD CASE SUSQUEHANNA OBE

NODE 8 DIRECTION VERT ELEV 799'-1

DAMPING 0.005.0.010.0.02

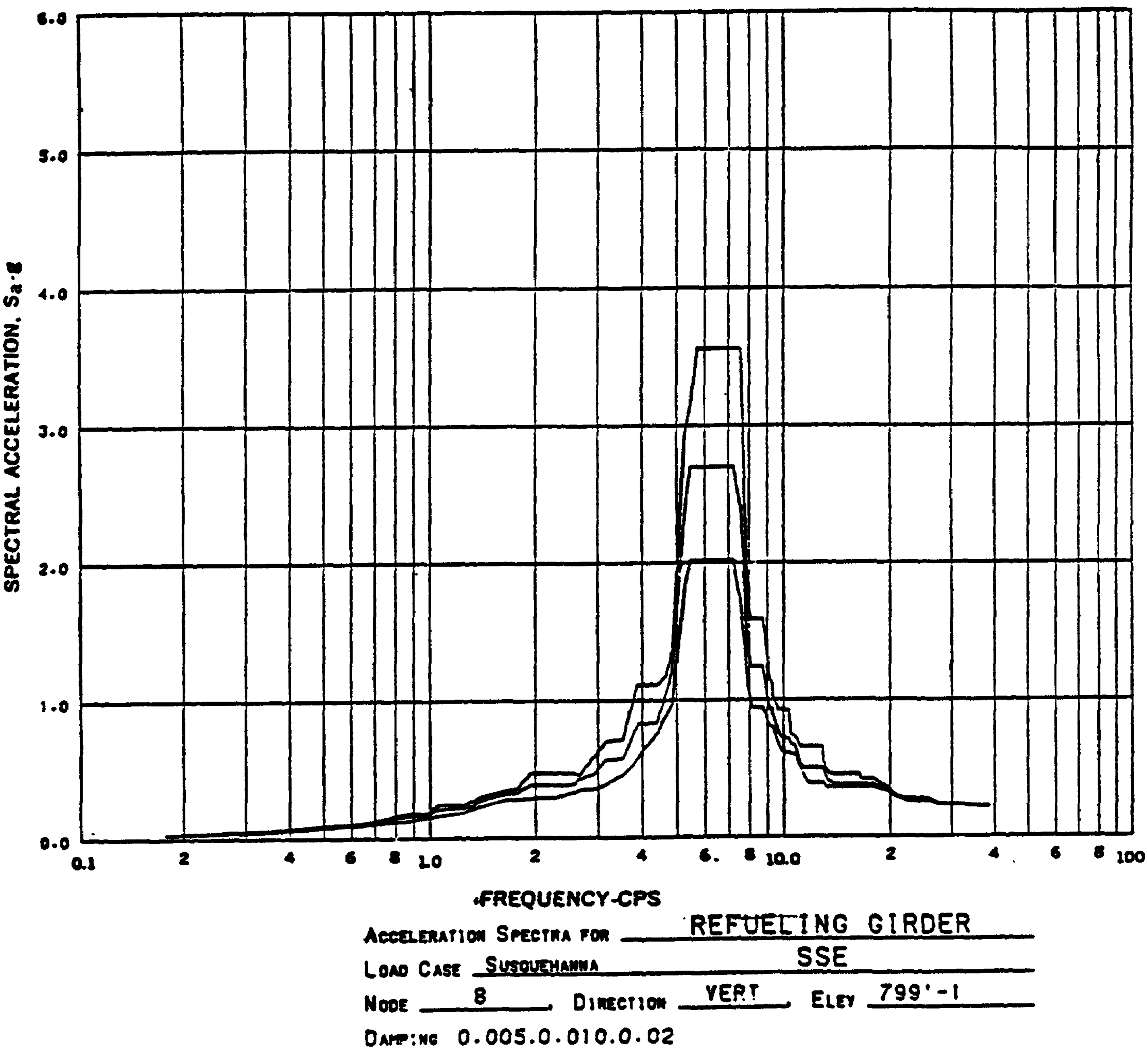
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

RESPONSE SPECTRUM
AT REFUELING AREA
VERTICAL OBE

FIGURE 3.7B-68, Rev. 55

Auto-Cad Figure Fsar 3_7B_68.dwg



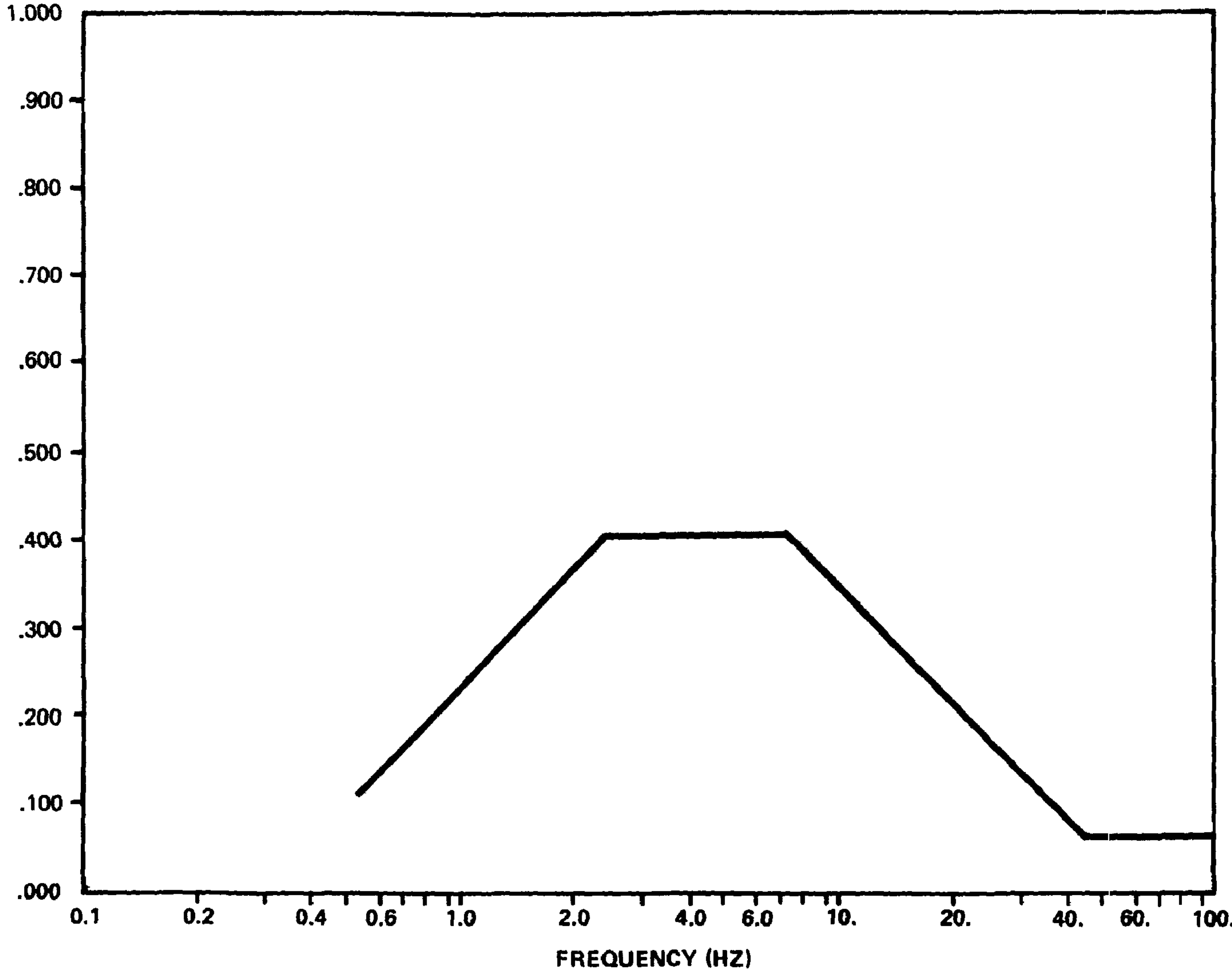
FSAR REV.65

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



LOCATION: TOP OF PEDESTAL FOR DIESELS 'A-D'
DIRECTION: E-W
EARTHQUAKE: OBE
DAMPING: 0.005

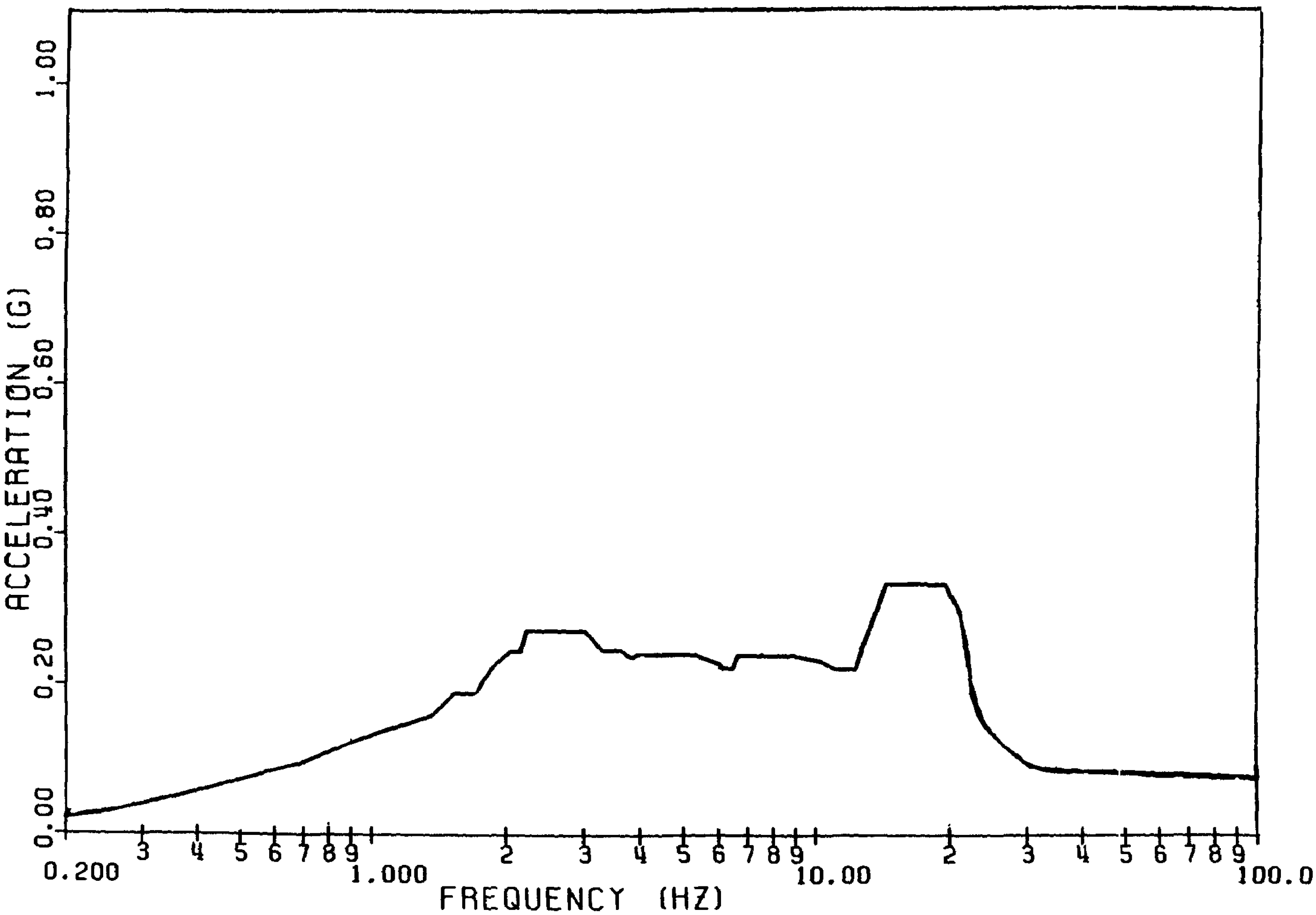
FSAR REV.65

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

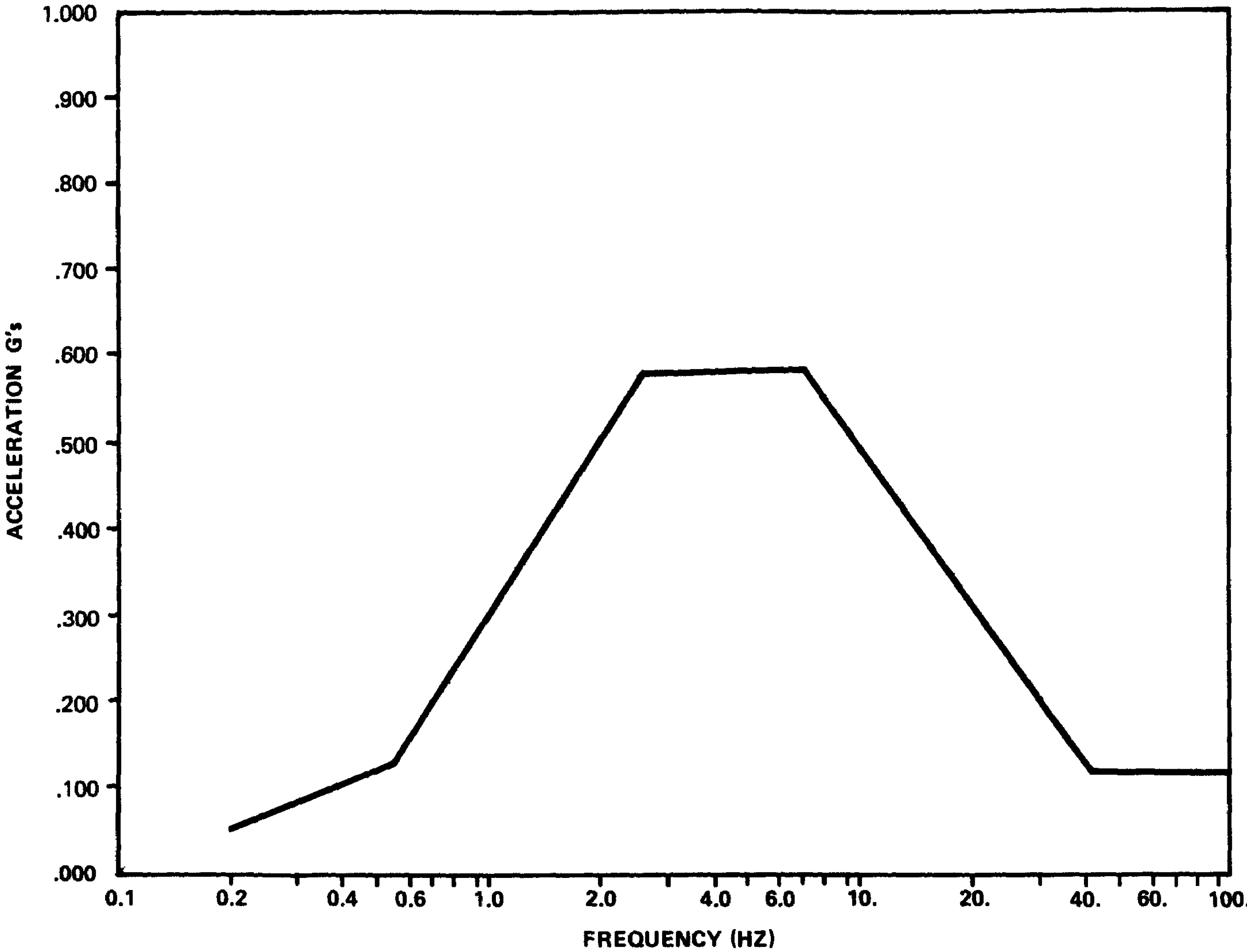
FSAR REV.65

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

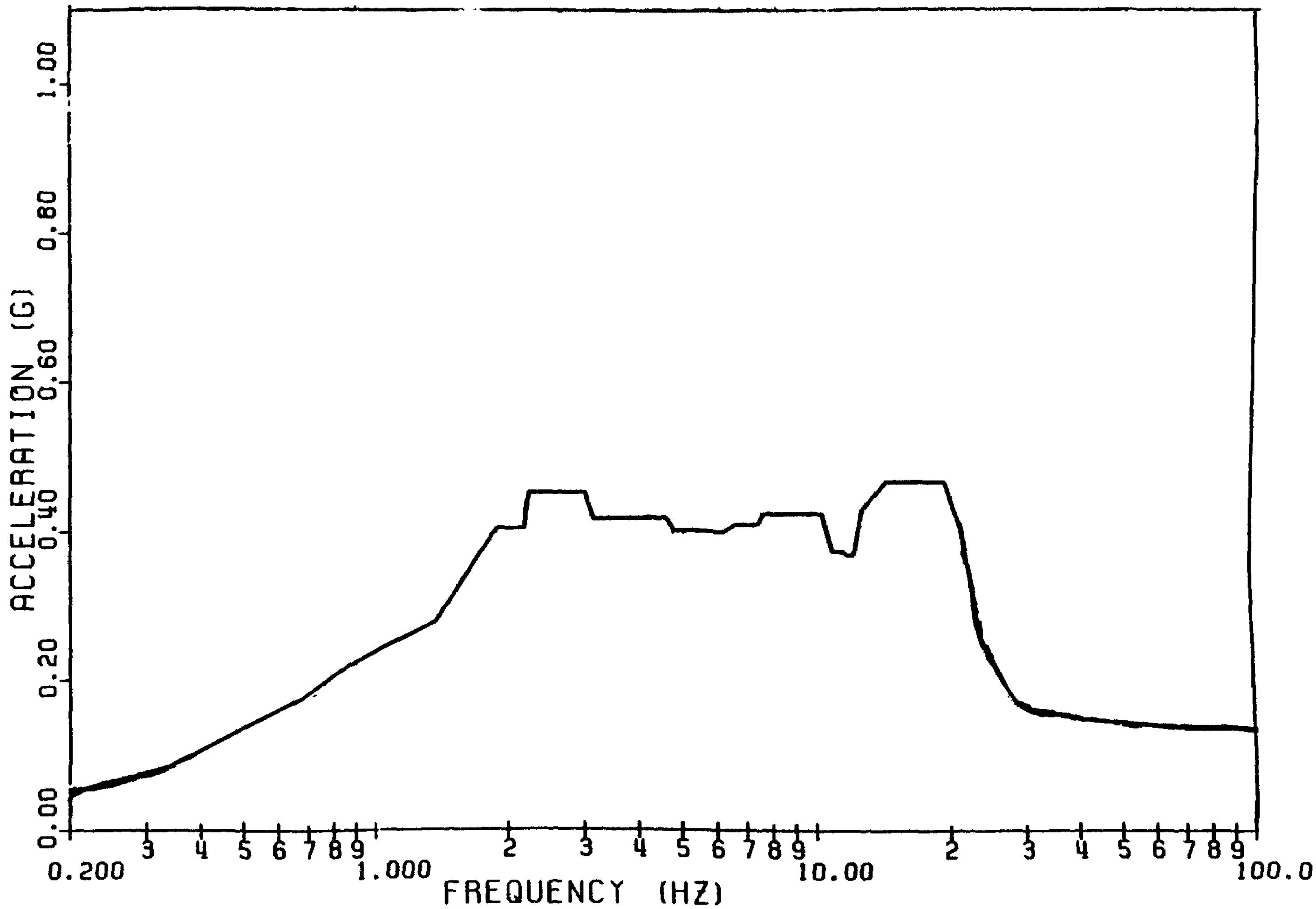
FSAR REV.65

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

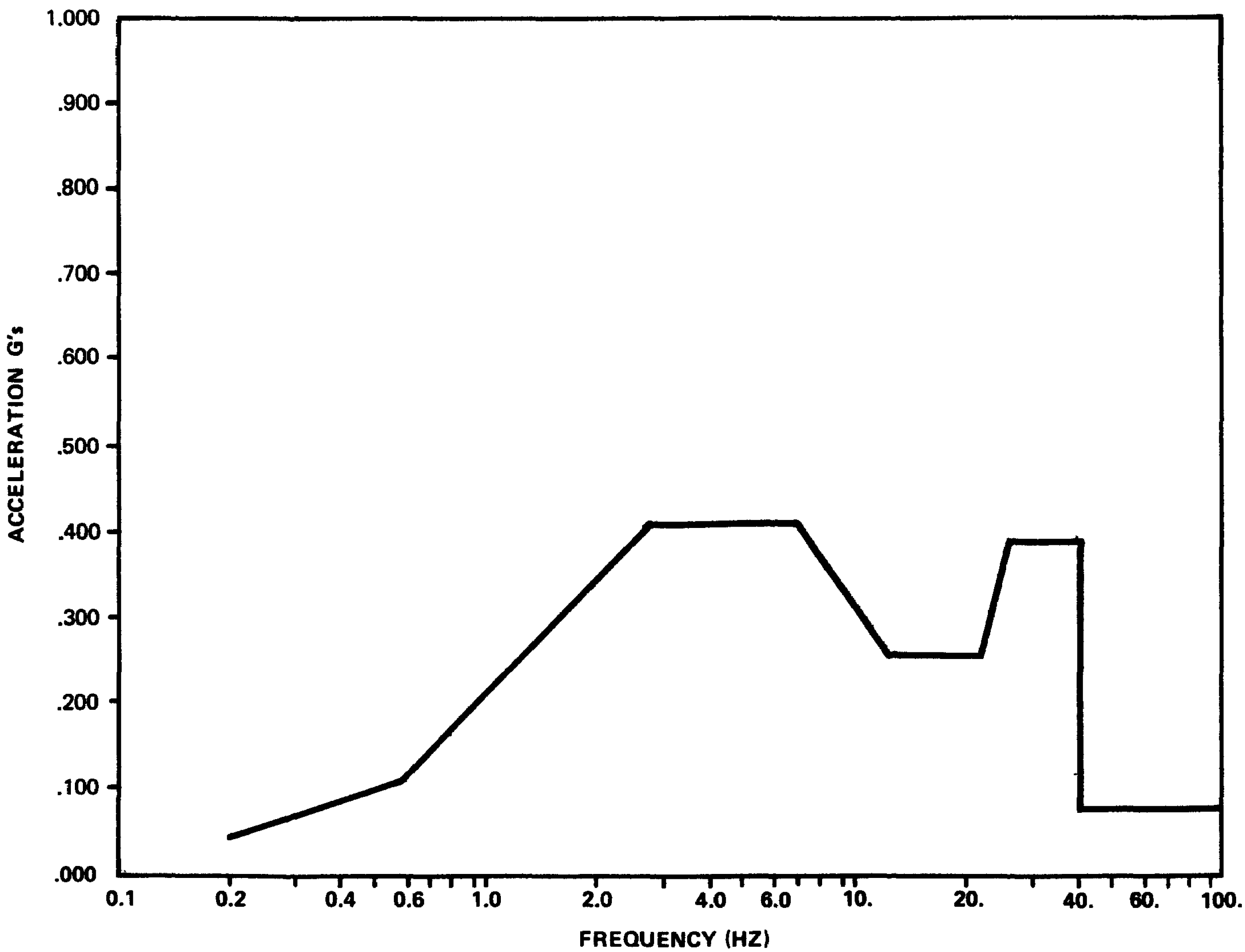
FSAR REV.65

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

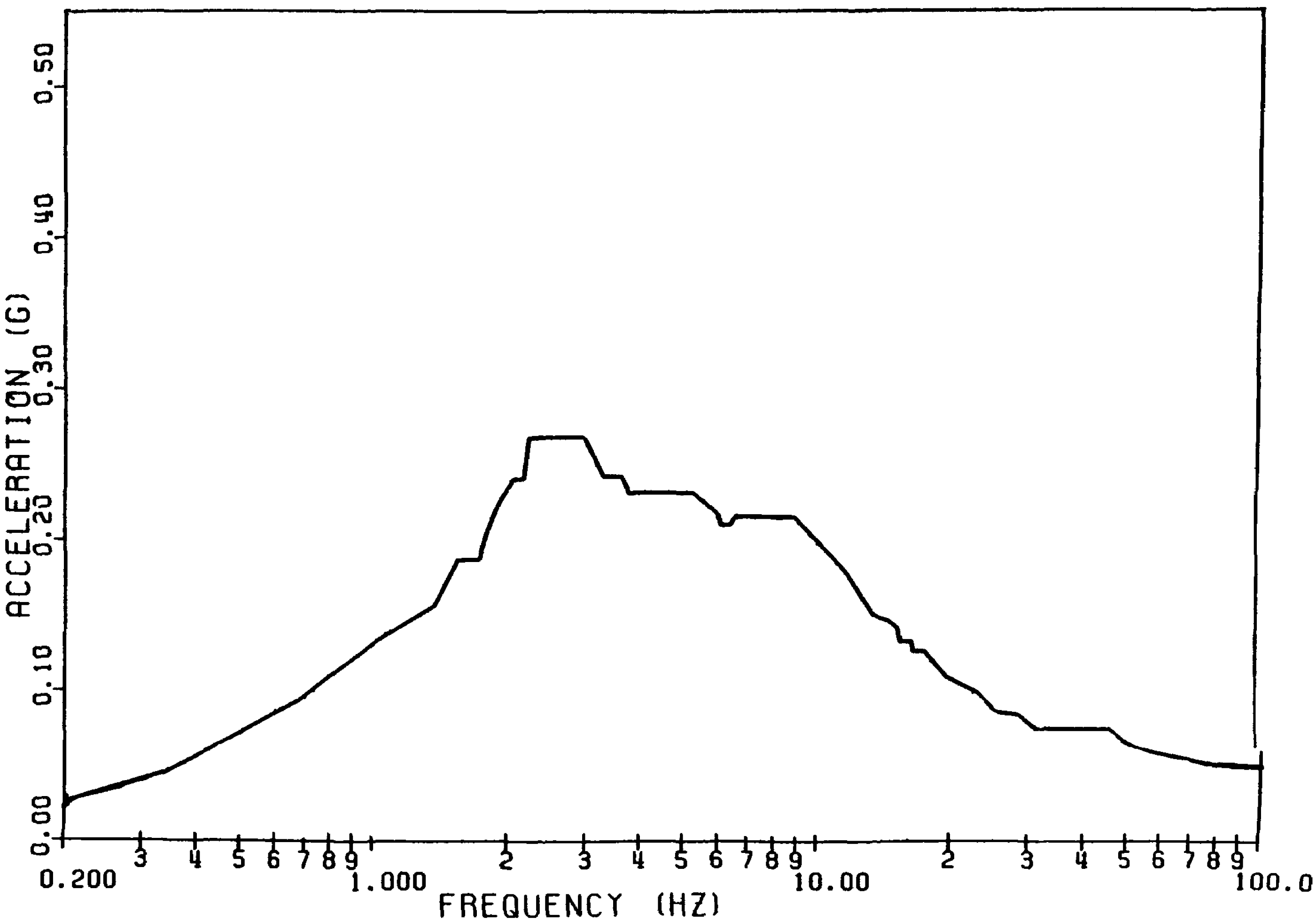
FSAR REV.65

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

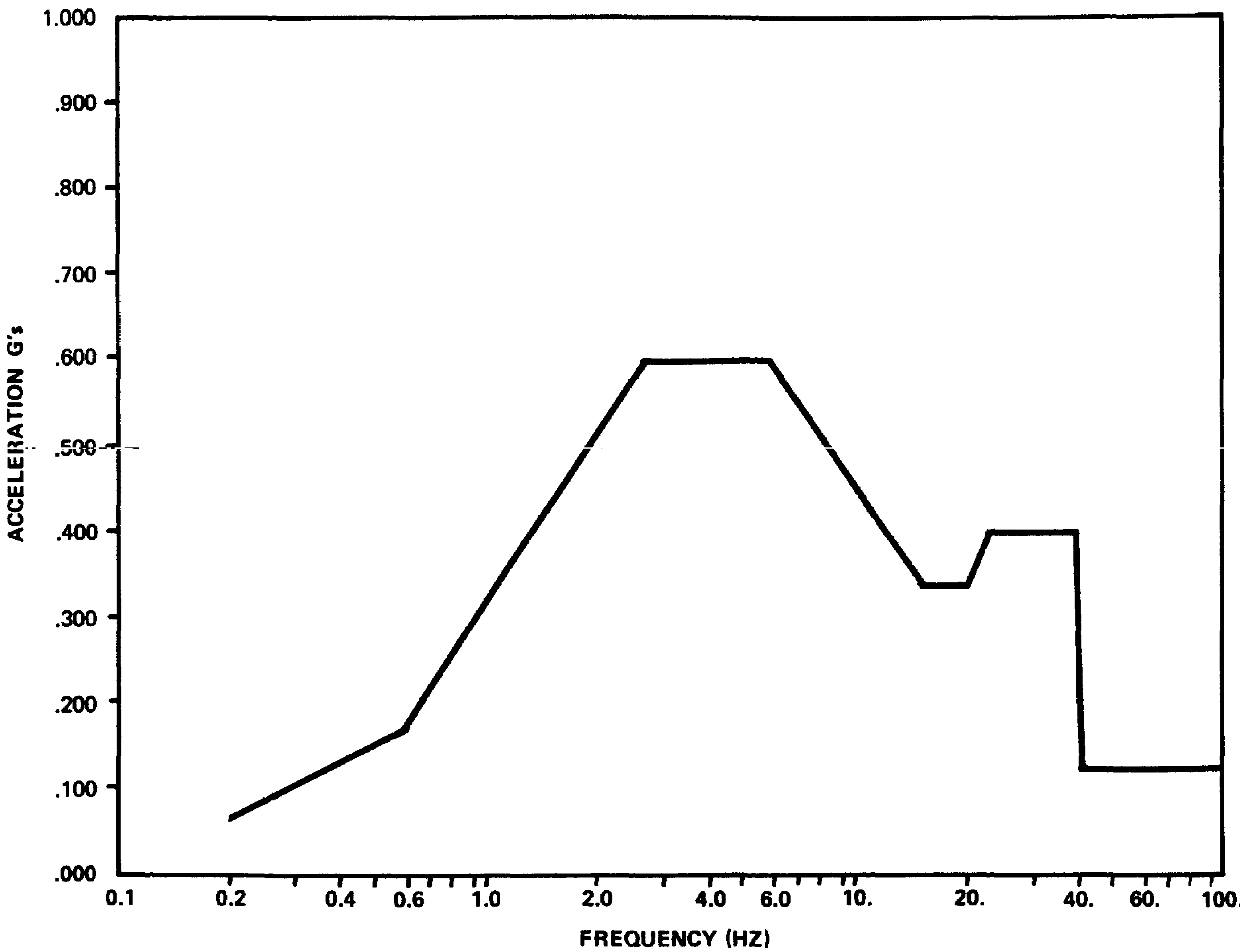
FSAR REV.65

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

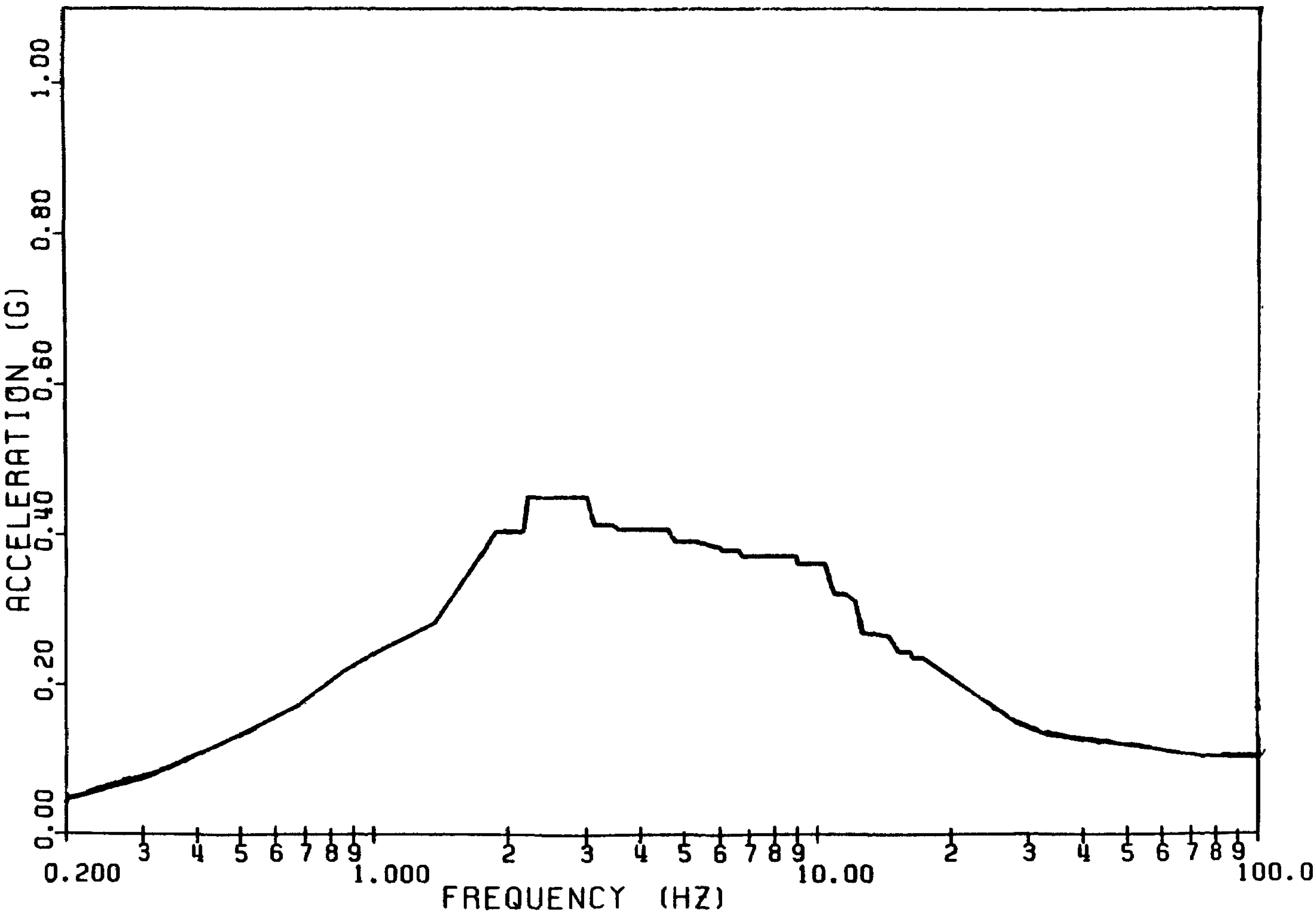
FSAR REV.65

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

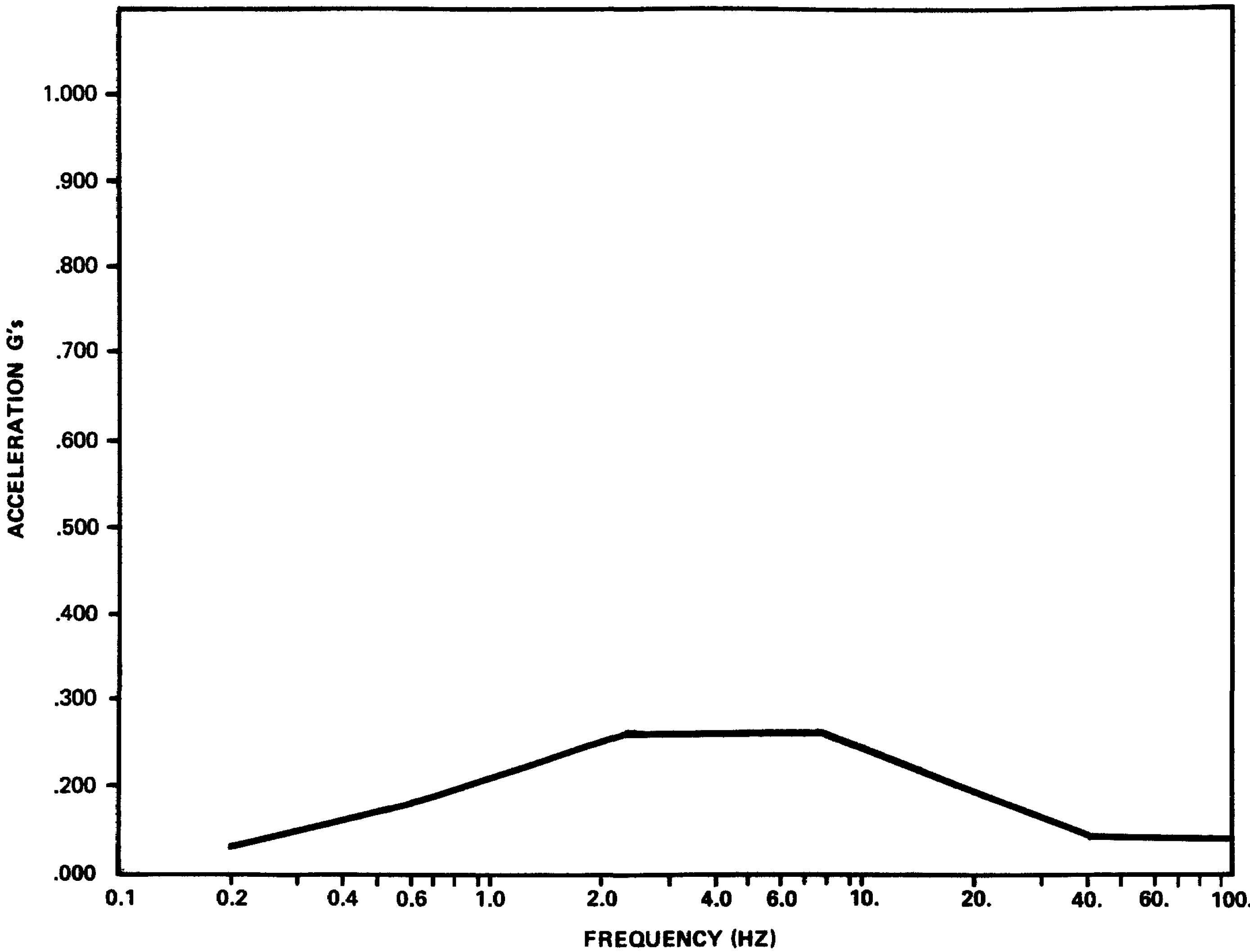
FSAR REV.65

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

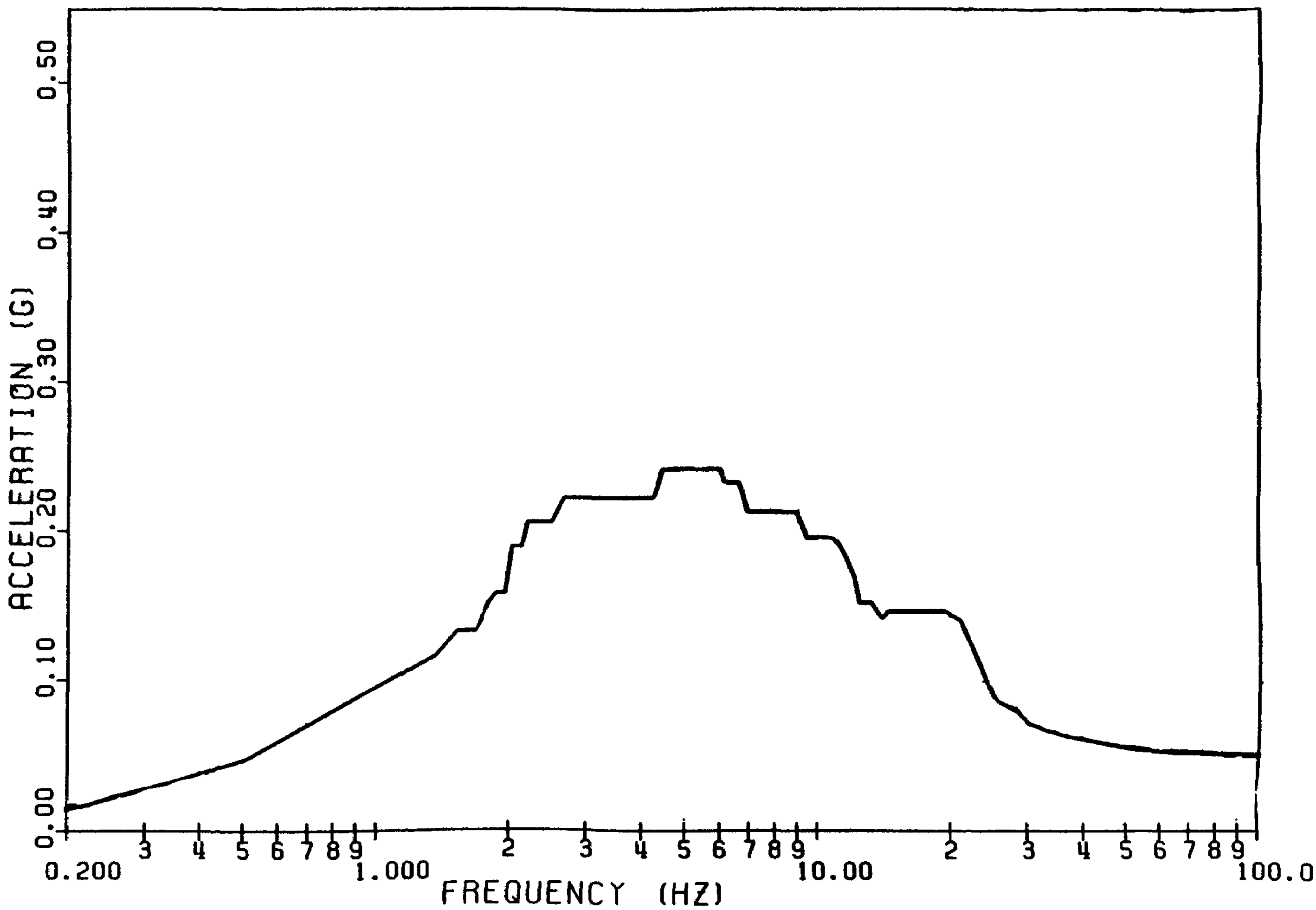
FSAR REV.65

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

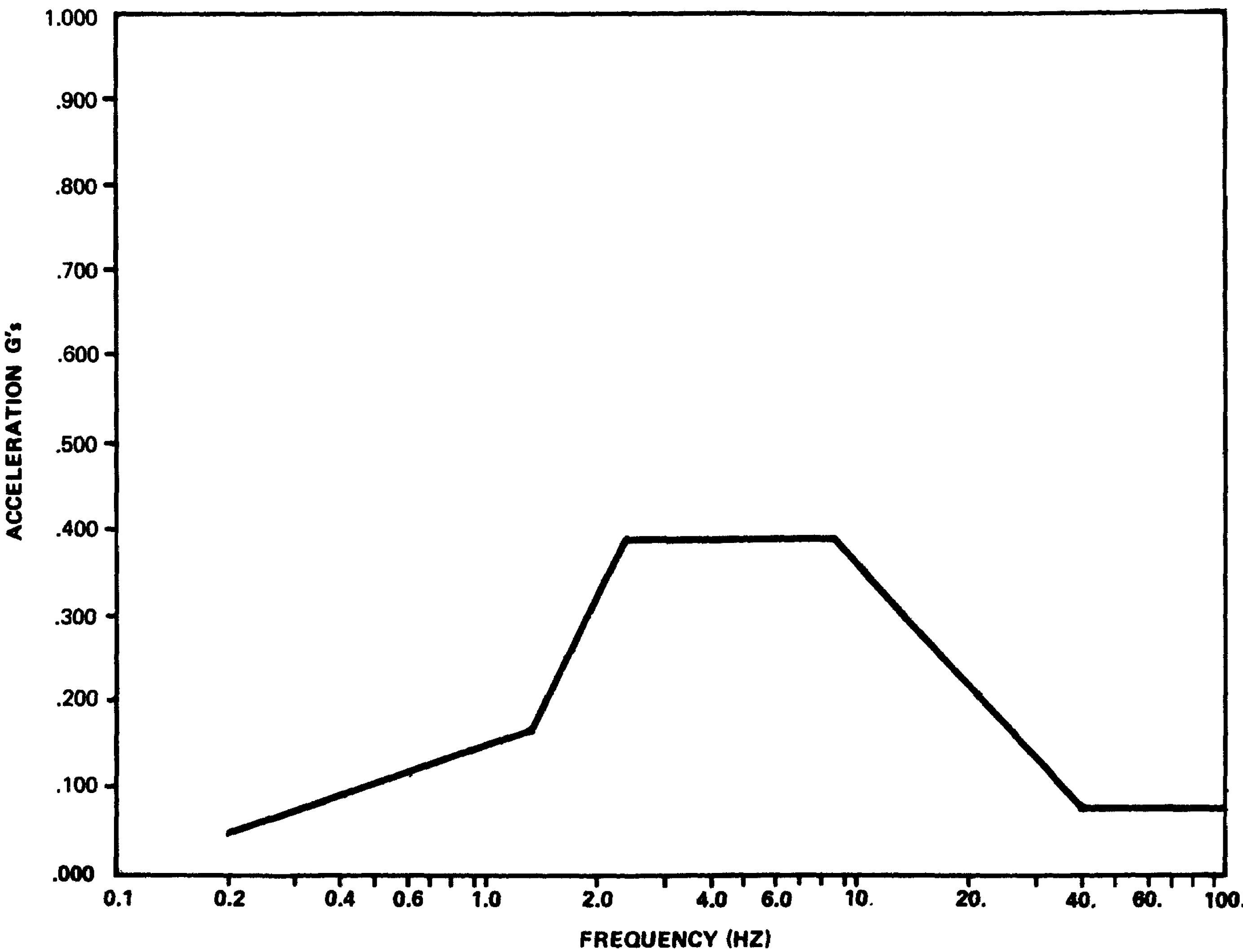
FSAR REV.65

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

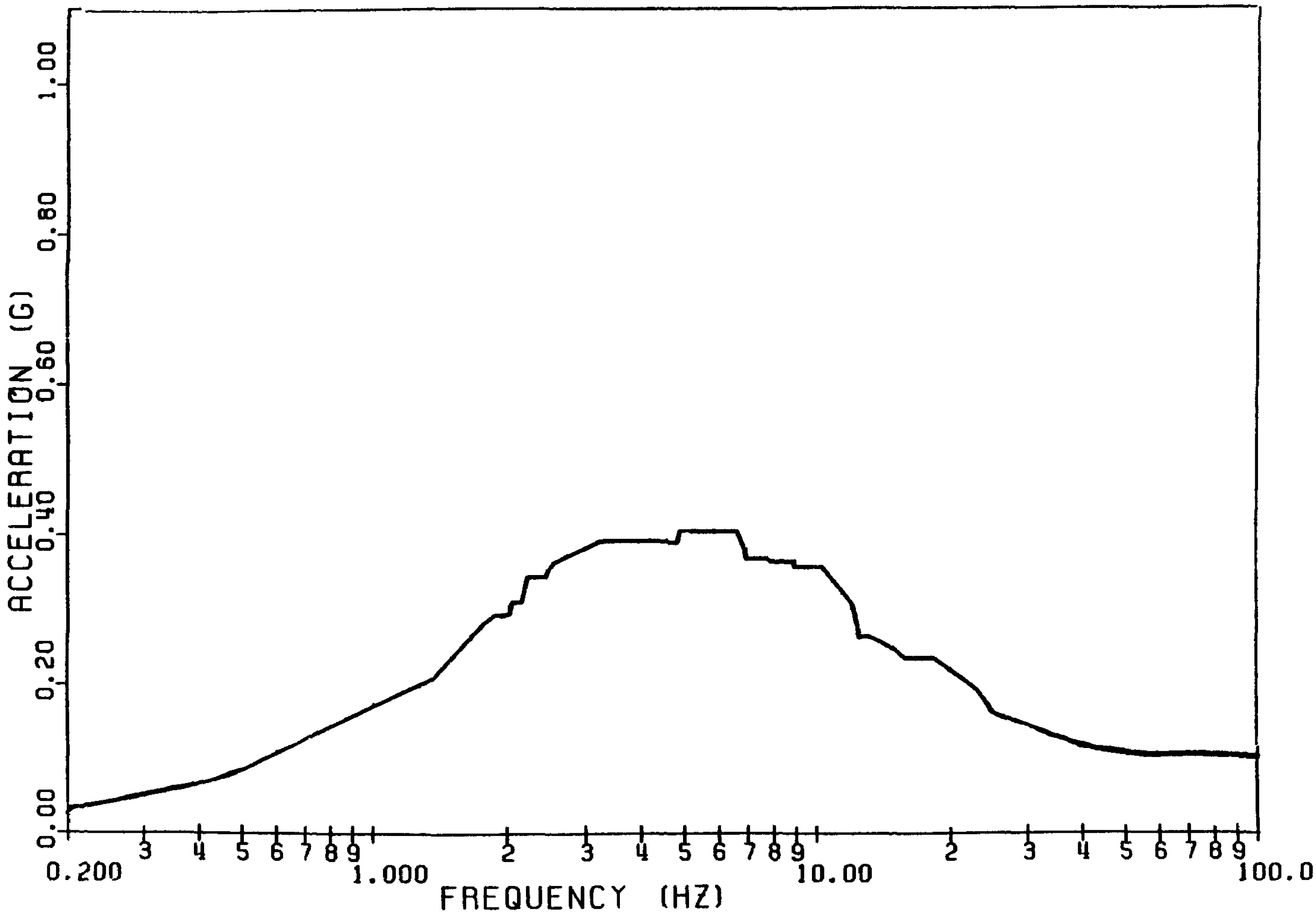
FSAR REV.65

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

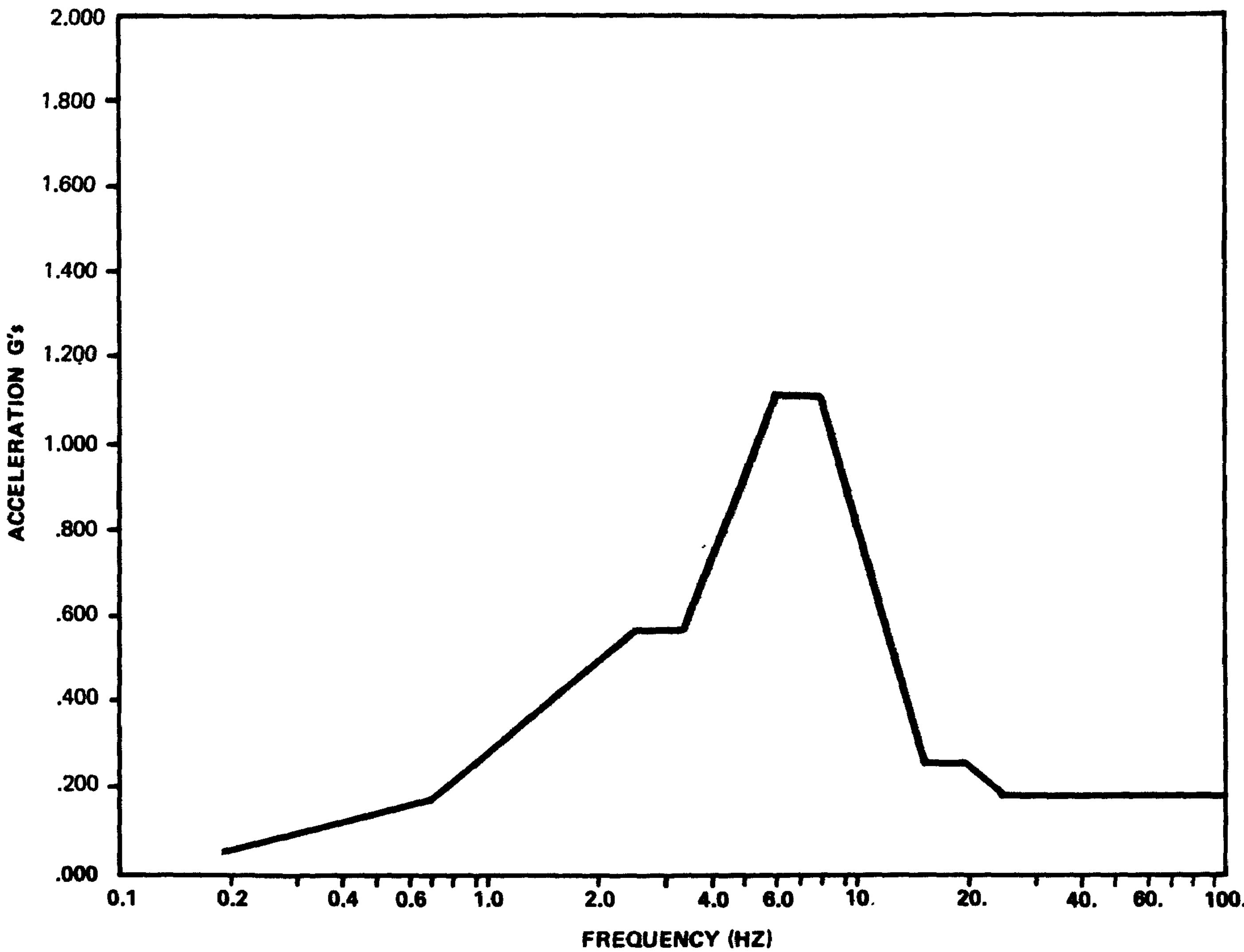
FSAR REV.65

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

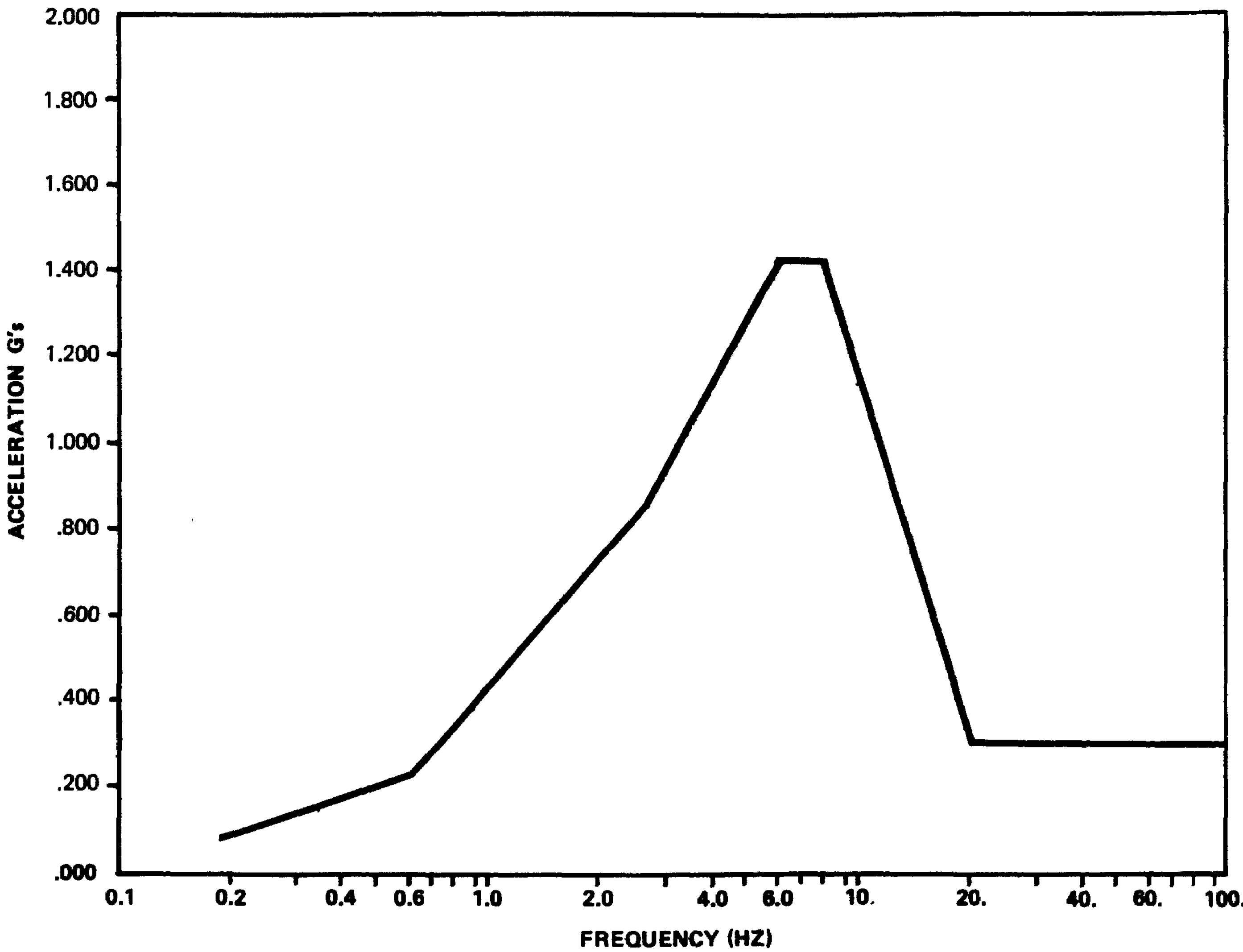
FSAR REV.65

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

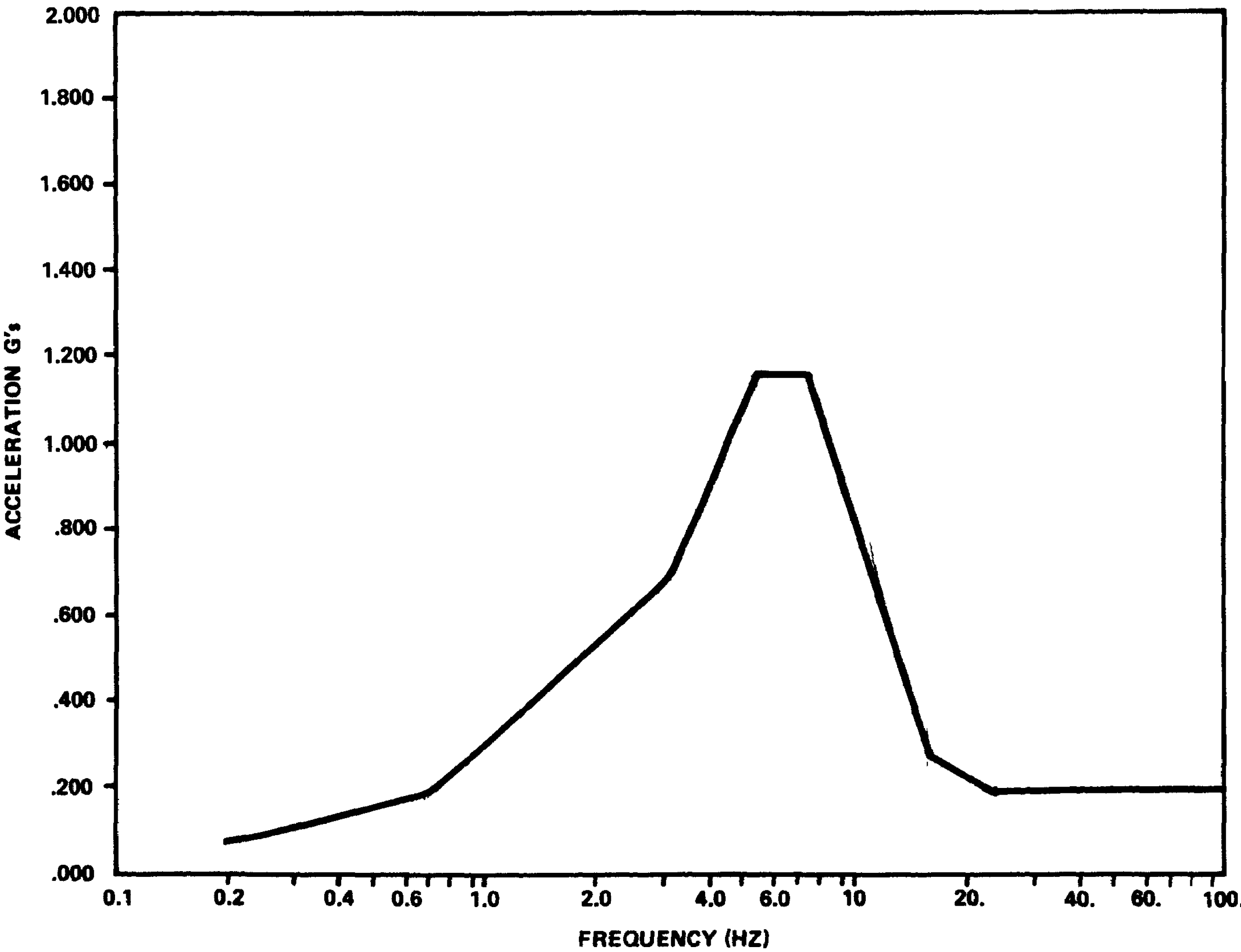
FSAR REV.65

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

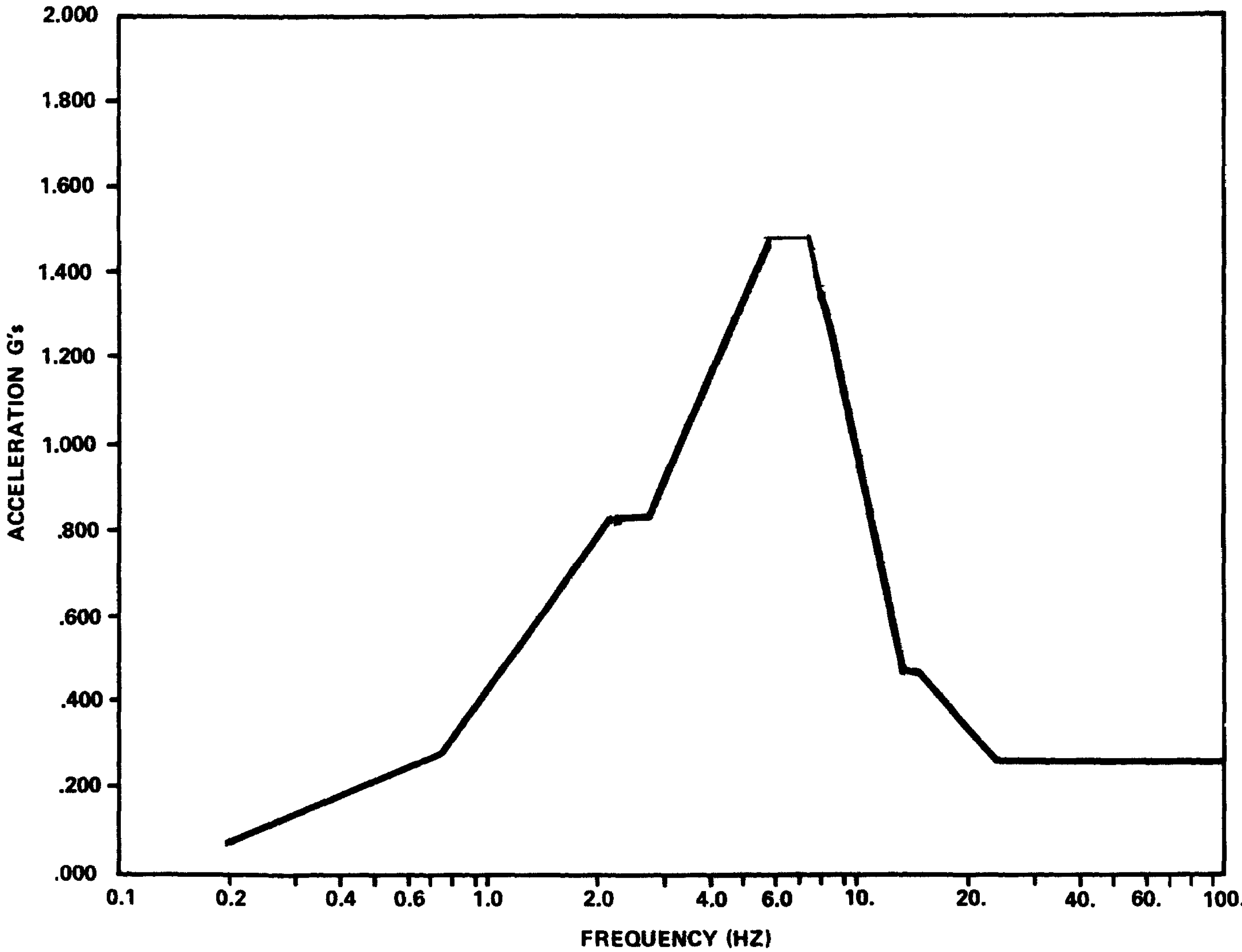
FSAR REV.65

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

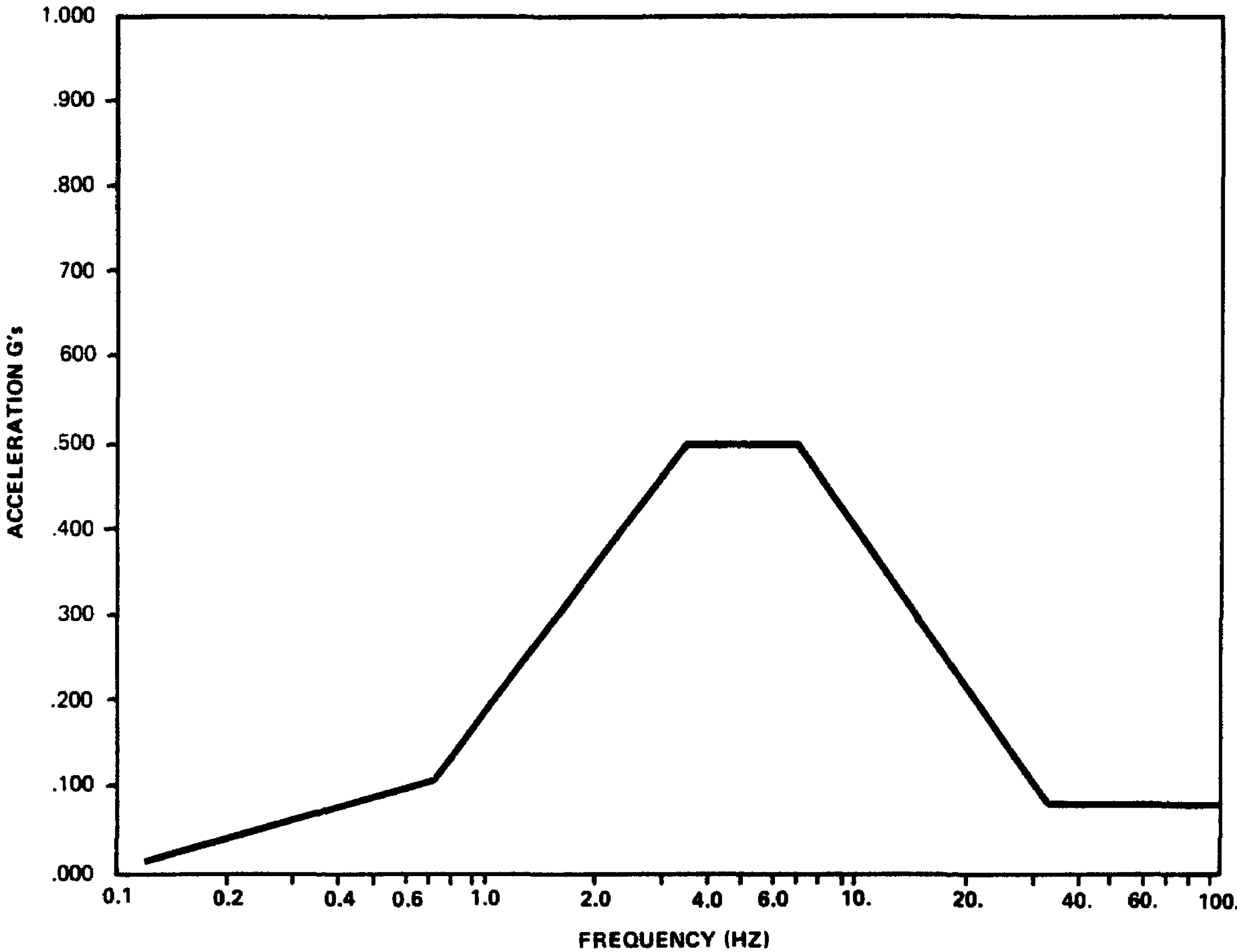
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

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

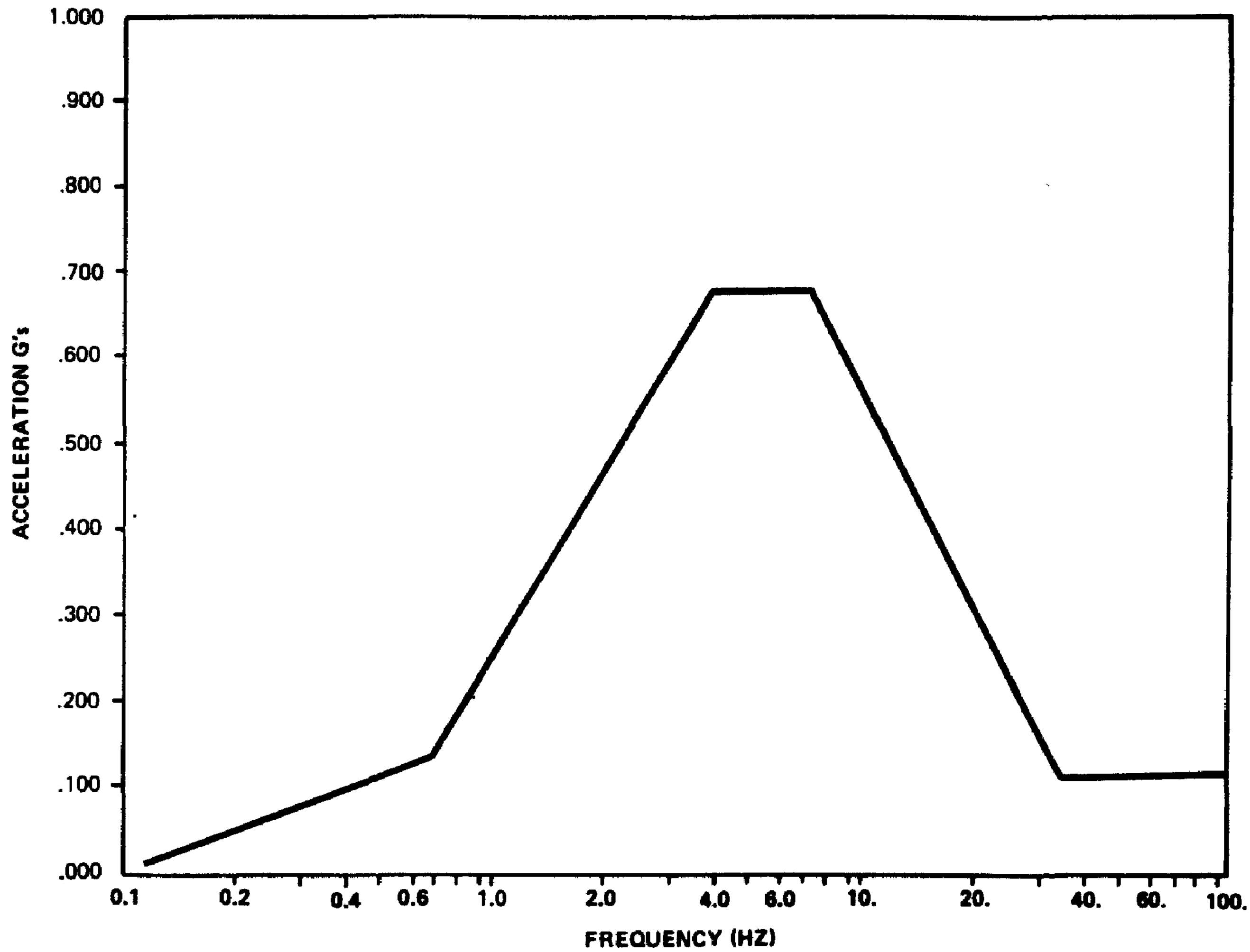


LOCATION: ESSW PUMPHOUSE (OPERATING FLOOR)
DIRECTION: VERTICAL
EARTHQUAKE: OBE
DAMPING. 0.005

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

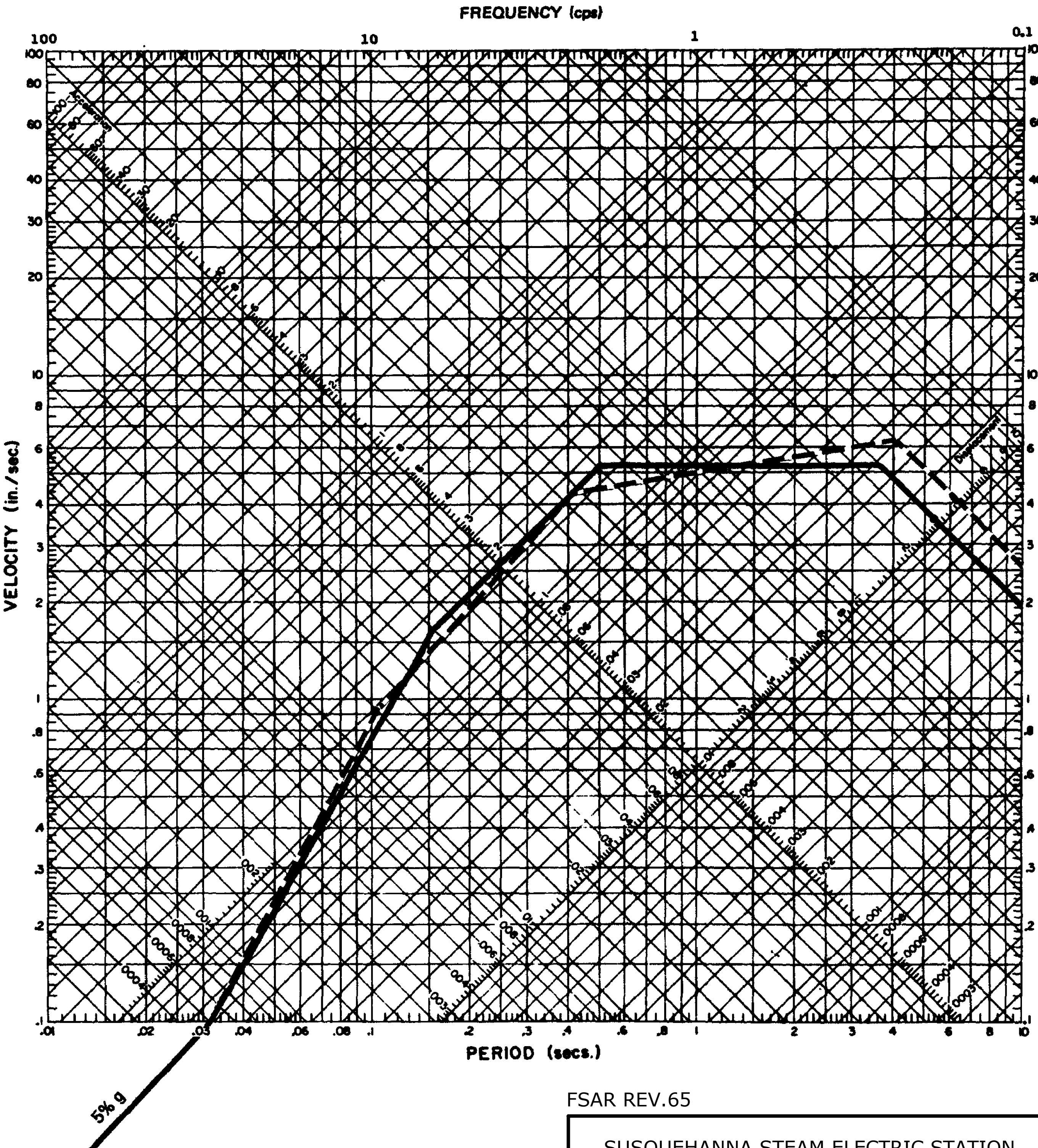
FSAR REV.65

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



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

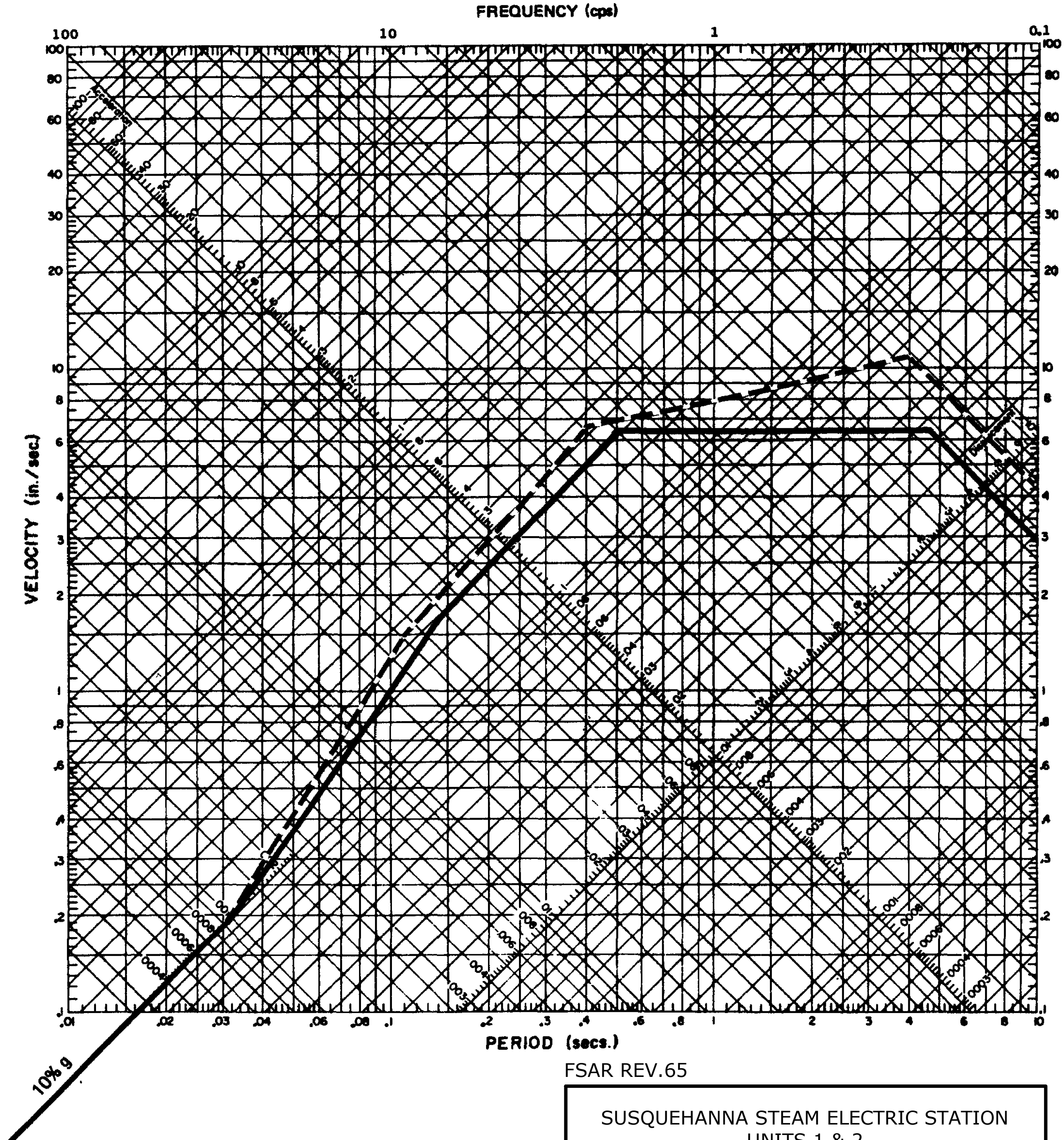
FSAR REV.65

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



FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

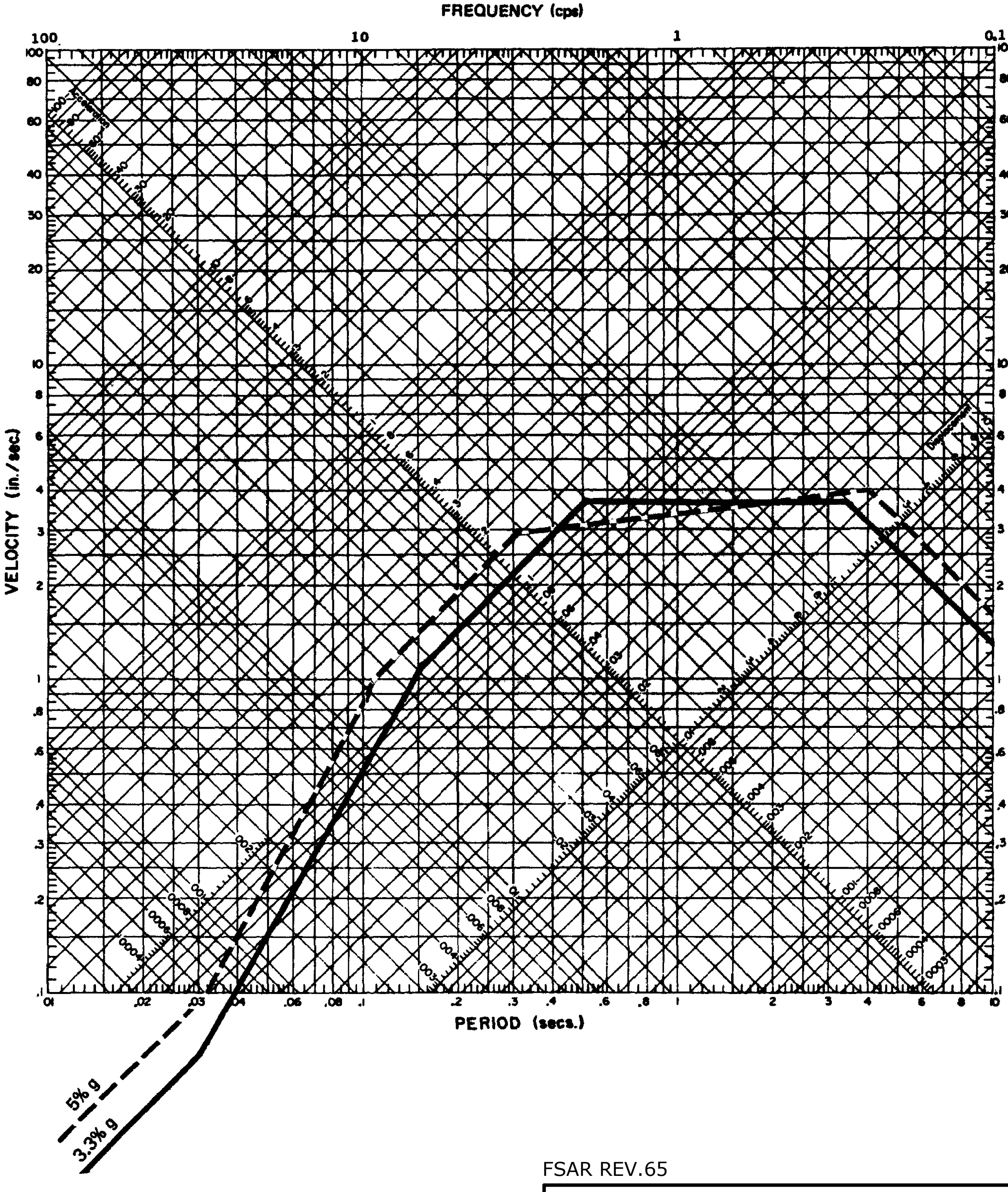
DESIGN SPECTRUM (5% DAMPING) FOR ALL ROCK FOUNDED SEISMIC CATEGORY I STRUCTURES EXCEPT THE DIESEL GENERATOR 'E' BUILDING

R.G. 1.60 SPECTRUM (7% DAMPING) USED FOR THE DESIGN OF THE DIESEL GENERATOR 'E' BUILDING

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 (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

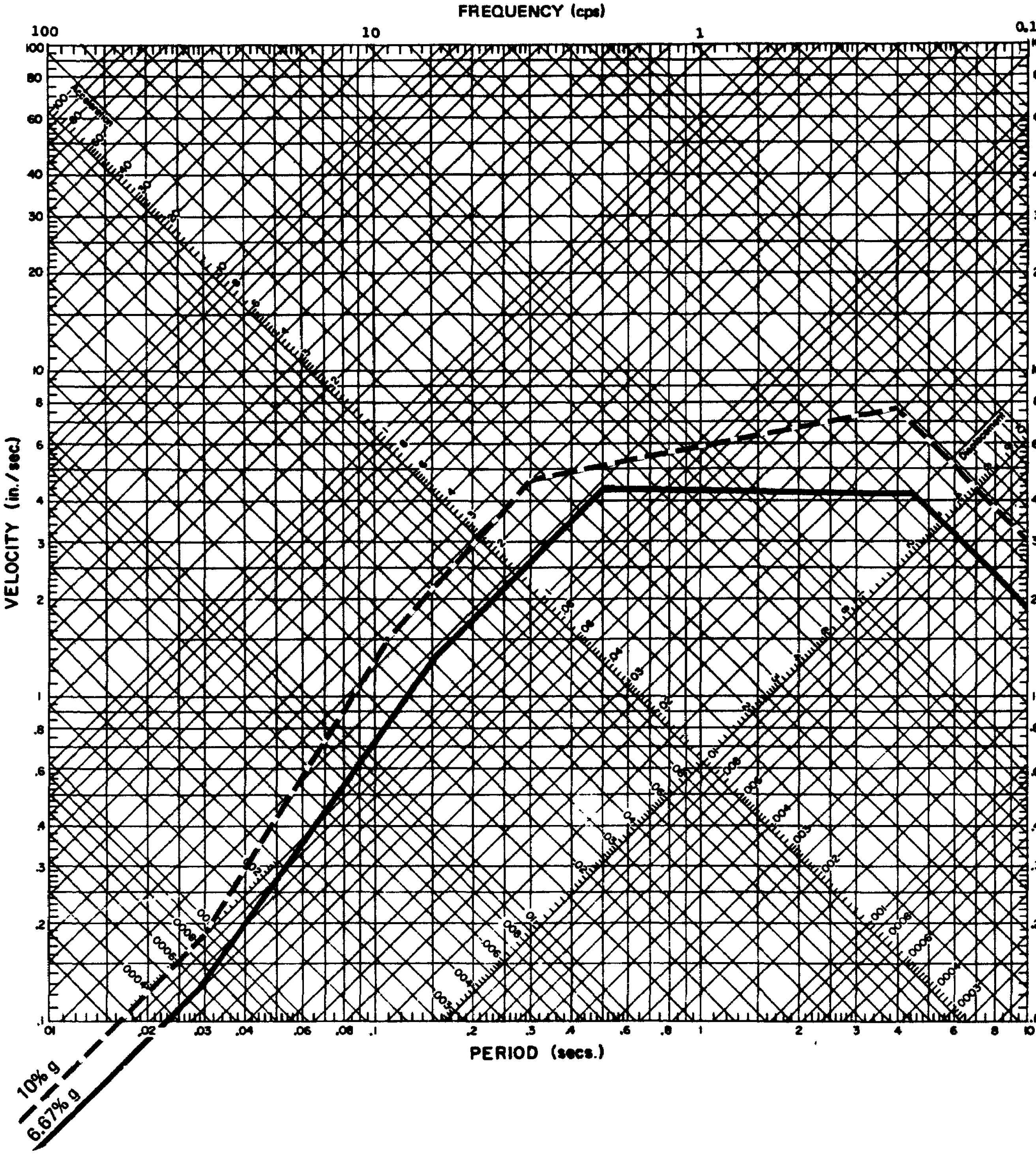
FSAR REV.65

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

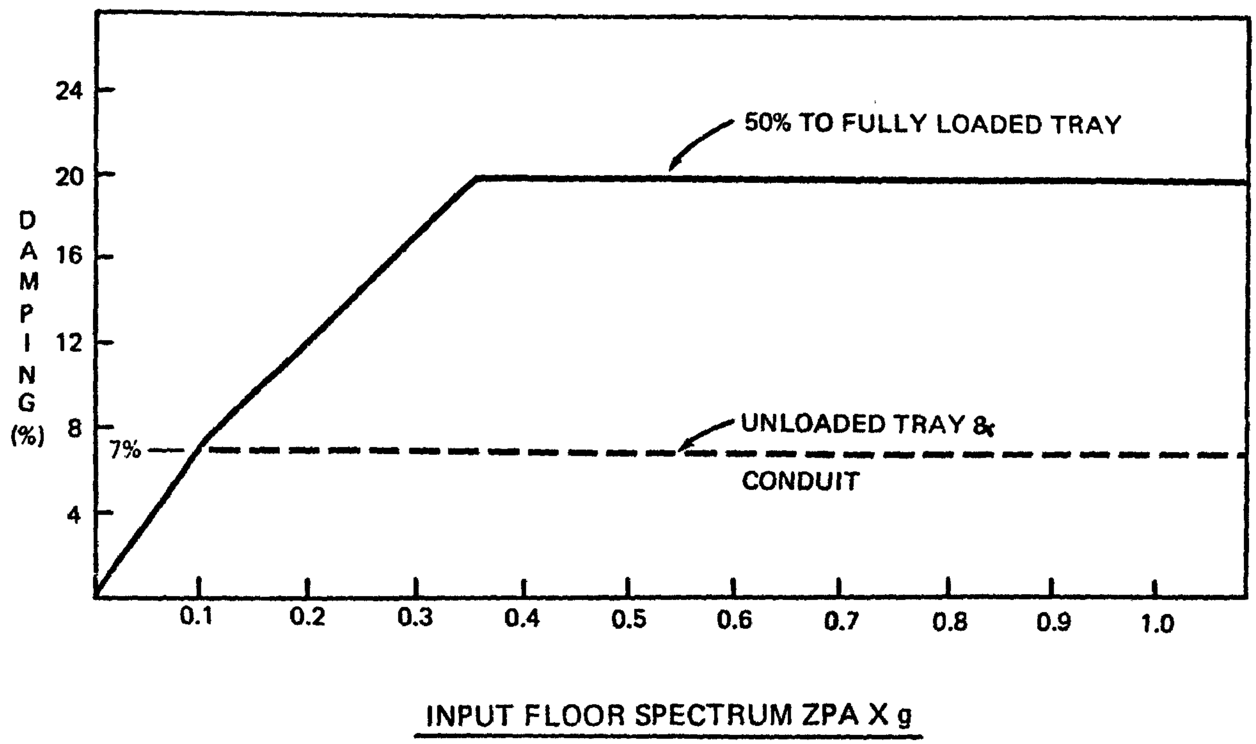
FSAR REV.65

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

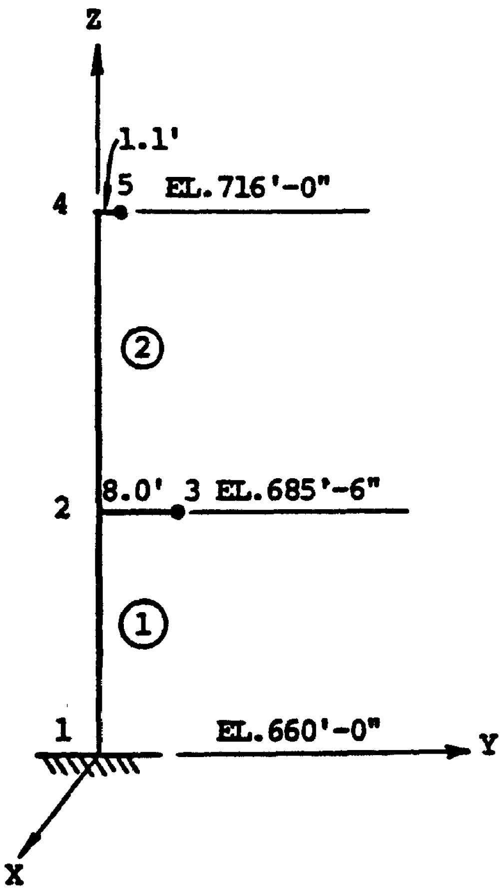
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
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



MASSES AT NODES 3 AND 5

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

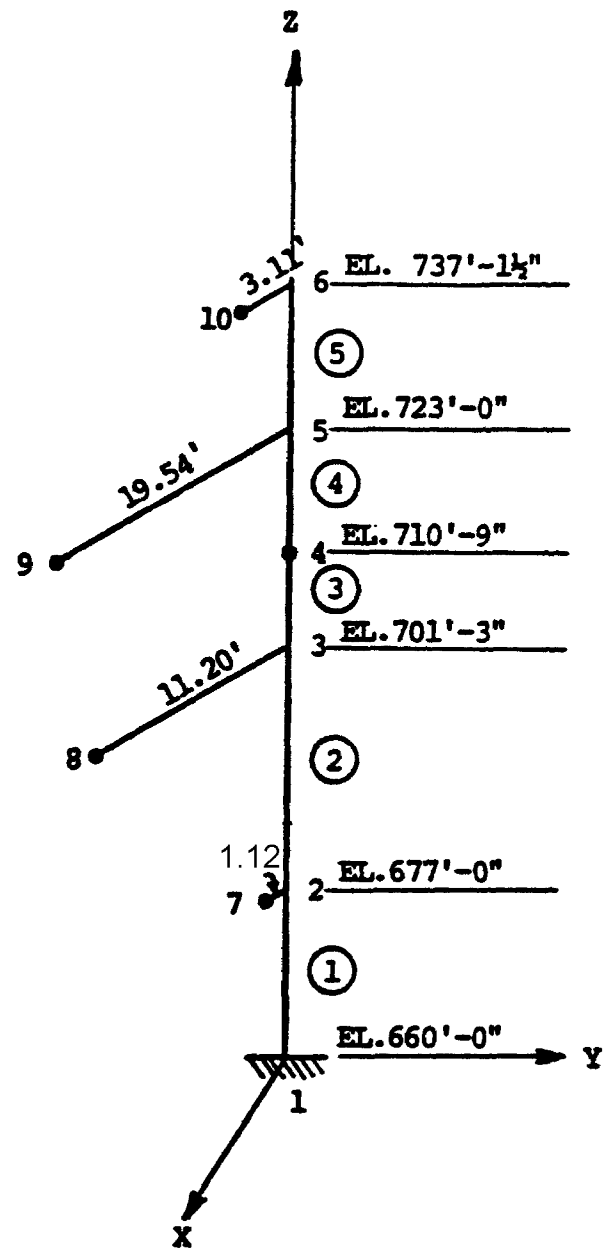
FSAR REV.65

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

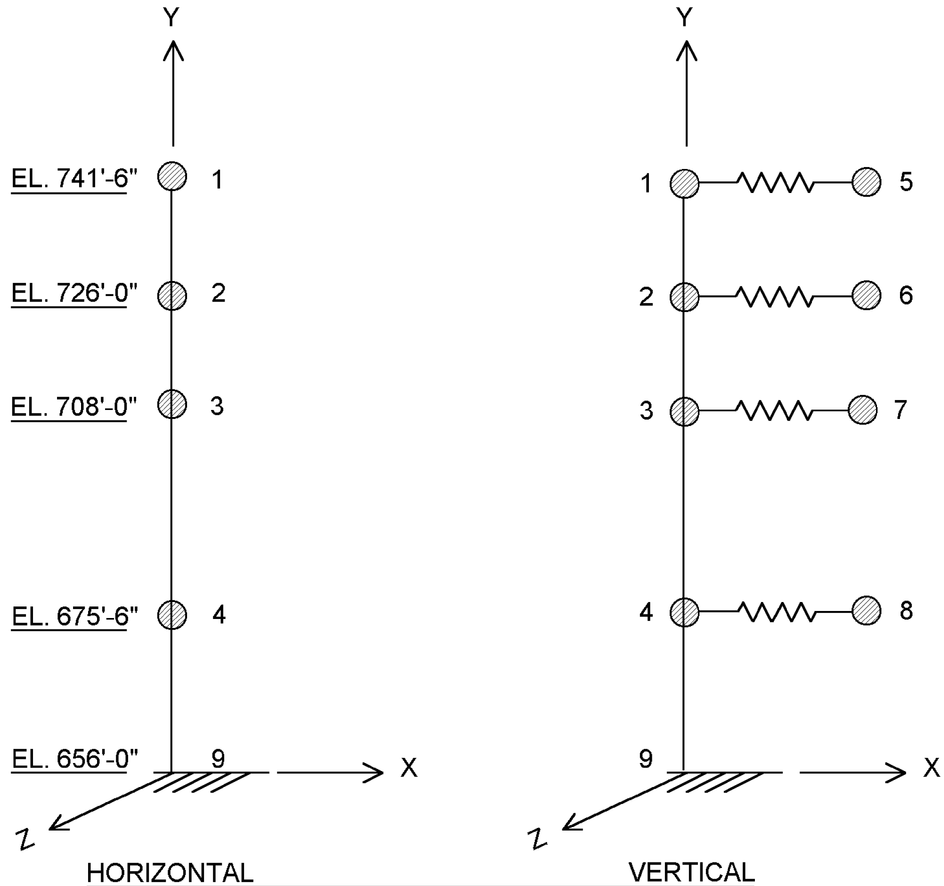
FSAR REV.65

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

FSAR REV.65

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

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				
	USD	1.0	0.67	fy*
Ratio	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 ±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 ±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.

<u>Containment</u>: The containment foundation is described in Subsection 3.8.1.

<u>Reactor Building and Control Building</u>

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.

<u>Diesel Generator Buildings:</u>

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.

<u>SSW Pumphouse</u>: 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.

<u>Turbine Building</u>: 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.

<u>Radwaste Building</u>: 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.

<u>3.8.5.2 Applicable Codes, Standards, and Specifications</u>

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.

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.

TABLE 3.8-1 (Continued)

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	Designation	Title Number	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.

TABLE 3.8-1 (Continued)

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*
4G**	ANSI N45.2.2	Packaging, Shipping, Receiving, Storage and Handling of Items for Nuclear Power Plants	1978
5G**	ANSI N45.2.6	Qualifications of Inspection, Examination and Testing Personnel for the Construction Phase of Nuclear Power Plants	1978
6G**	ANSI N45.2.9	Requirements for Collection, Storage, and Maintenance of Quality Assurance Records for Nuclear Power Plants	1974
7G**	ANSI N45.2.10	Quality Assurance Terms and Definitions	1973
8G**	ANSI N45.2.11	Quality Assurance Requirements for the Design of Nuclear Power Plants	1974
9G**	ANSI N45.2.12	Requirements for Auditing of Quality Assurance Programs for Nuclear Power Plants	1977
10G**	ANSI N45.2.13	Quality Assurance Requirements for Control of Procurement of Items and Services for Nuclear Power Plants	1976
11G**	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.

TABLE 3.8-1 (Continued)

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 3.8-2a
Load Combinations for Reactor Pedestal

Security-Related Information
Table Withheld Under 10 CFR 2.390

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 3.8-3a

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

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	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 3.8-4

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:	$D + L + T_o + E + H_o$ $D + L + T_o + W + H_o$	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:	$D+L+R+T_o + E'+P+H_o$ $D + L + R + (T_o + T_a) + P + E' + H_a$	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 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:

$$U = 1.0D+1.0L+1.0E'+1.0T_o +1.0R + 1.25 H_o$$

$$U = 1.0D+1.0L+1.0E'+1.0(T_o+T_a)+1.0R + 1.0 H_a$$

TABLE 3.8-9 (Continued)

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 3.8-9a

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. $1.6S = D+L+W_{ms}$

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.

THIS FIGURE HAS BEEN
REPLACED BY DWG.
C-331, Sh. 1

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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

FSAR REV. 65

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

AutoCAD Figure 3_8_3.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
C-1932, Sh. 4

Figure 3.8-4 replaced by dwg.
C-1932, Sh. 4

THIS FIGURE HAS BEEN
REPLACED BY DWG.
C-1932, Sh. 5

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SUSQUEHANNA STEAM ELECTRIC STATION
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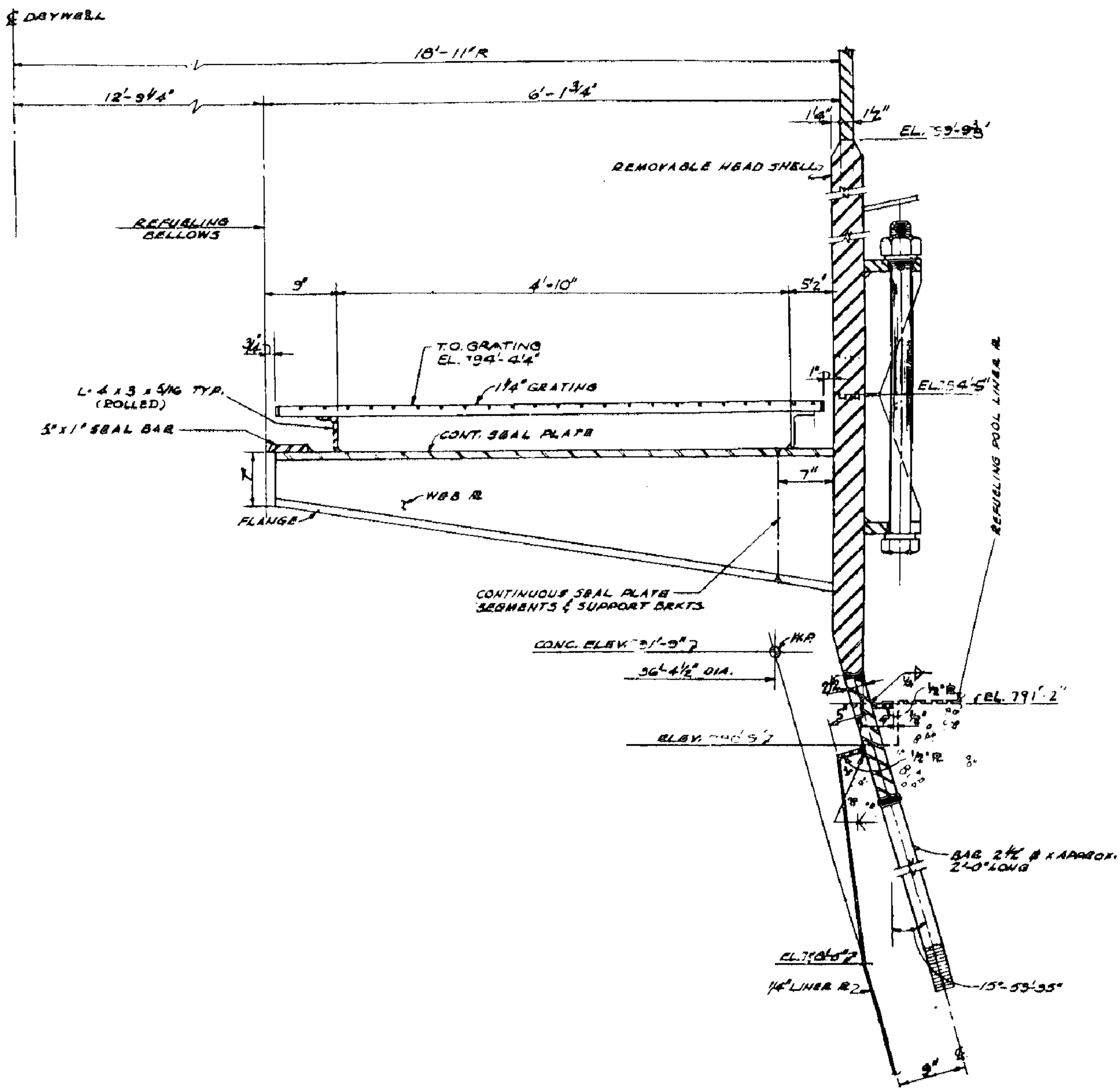
FINAL SAFETY ANALYSIS REPORT

Figure 3.8-5 replaced by dwg.
C-1932, Sh. 5

FIGURE 3.8-5, Rev. 55

AutoCAD Figure 3_8_5.doc



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
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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-332, Sh. 1

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SUSQUEHANNA STEAM ELECTRIC STATION
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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

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SUSQUEHANNA STEAM ELECTRIC STATION
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FINAL SAFETY ANALYSIS REPORT

Figure 3.8-11-2 replaced by dwg.
C-333, Sh. 1

FIGURE 3.8-11-2, Rev. 49

Figure Fsar 3_8_11_2.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
C-281, Sh. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
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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
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Figure 3.8-13 replaced by dwg.
C-281, Sh. 1

FIGURE 3.8-13, Rev. 55

Figure Fsar 3_8_13.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
C-370, Sh. 1

FSAR REV. 65

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-334, Sh. 1

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SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
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

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UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

Figure 3.8-15-3 replaced by dwg.
C-336, Sh. 1

FIGURE 3.8-15-3, Rev. 49

THIS FIGURE HAS BEEN
REPLACED BY DWG.
C-337, Sh. 1

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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

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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

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SUSQUEHANNA STEAM ELECTRIC STATION
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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

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FINAL SAFETY ANALYSIS REPORT

Figure 3.8-16-2 replaced by dwg.
C-352, Sh. 1

FIGURE 3.8-16-2, Rev. 49

THIS FIGURE HAS BEEN
REPLACED BY DWG.
C-353, Sh. 1

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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

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SUSQUEHANNA STEAM ELECTRIC STATION
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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

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SUSQUEHANNA STEAM ELECTRIC STATION
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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

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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

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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

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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

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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-360, Sh. 1

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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

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SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
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Figure 3.8-16-11 replaced by dwg.
C-393, Sh. 1

FIGURE 3.8-16-11, Rev. 49

Figure Fsar 3_8_16_11.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
C-394, Sh. 1

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SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Figure 3.8-16-12 replaced by dwg. C-394, Sh. 1 FIGURE 3.8-16-12, Rev. 49

Figure Fsar 3_8_16_12.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
C-395, Sh. 1

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SUSQUEHANNA STEAM ELECTRIC STATION
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Figure 3.8-16-13 replaced by dwg.
C-395, Sh. 1

FIGURE 3.8-16-13, Rev. 49

THIS FIGURE HAS BEEN
REPLACED BY DWG.
C-396, Sh. 1

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SUSQUEHANNA STEAM ELECTRIC STATION
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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

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SUSQUEHANNA STEAM ELECTRIC STATION
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FINAL SAFETY ANALYSIS REPORT

Figure 3.8-16-15 replaced by dwg.
C-397, Sh. 1

FIGURE 3.8-16-15, Rev. 49

THIS FIGURE HAS BEEN
REPLACED BY DWG.
C-398, Sh. 1

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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

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SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 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 FINAL SAFETY ANALYSIS REPORT
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

THIS FIGURE HAS BEEN
REPLACED BY DWG.
C-282, Sh. 1

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SUSQUEHANNA STEAM ELECTRIC STATION
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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

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
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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-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
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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

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SUSQUEHANNA STEAM ELECTRIC STATION
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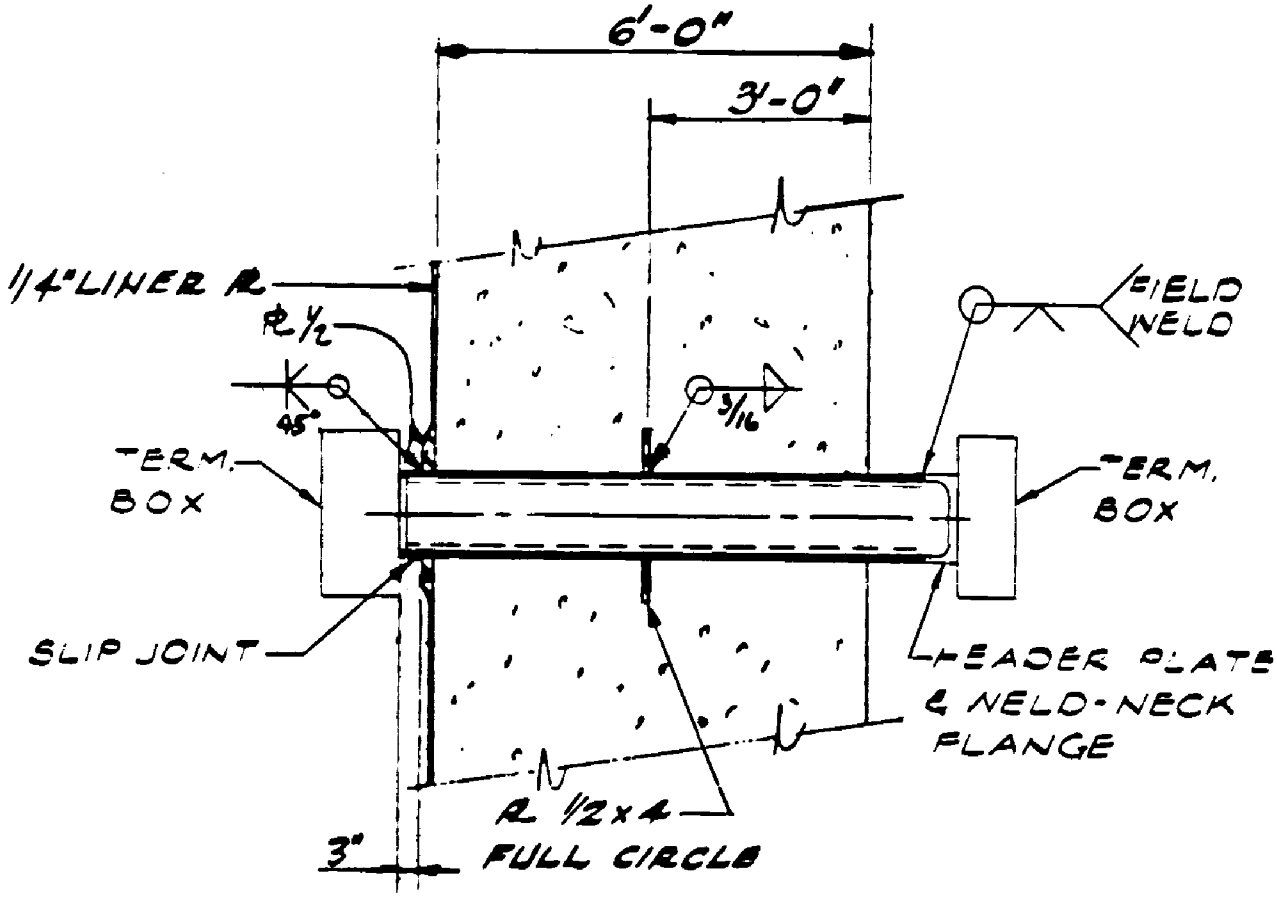
Figure 3.8-19-3 replaced by dwg.
C-283, Sh. 1

FIGURE 3.8-19-3, Rev. 55

AutoCAD Figure 3_8_19_3.doc



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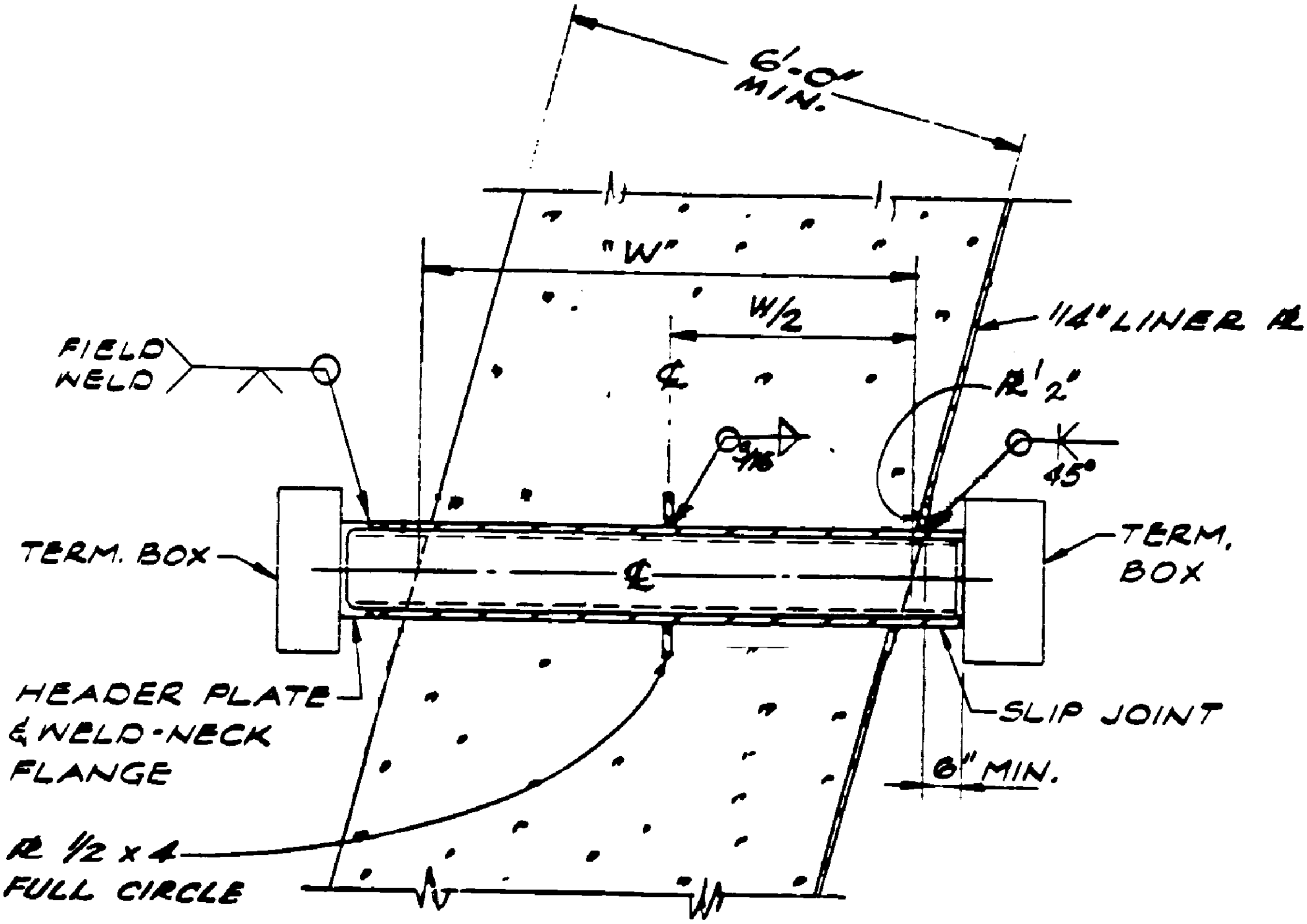
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

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-286, Sh. 1

FSAR REV. 65

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

FSAR REV. 65

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

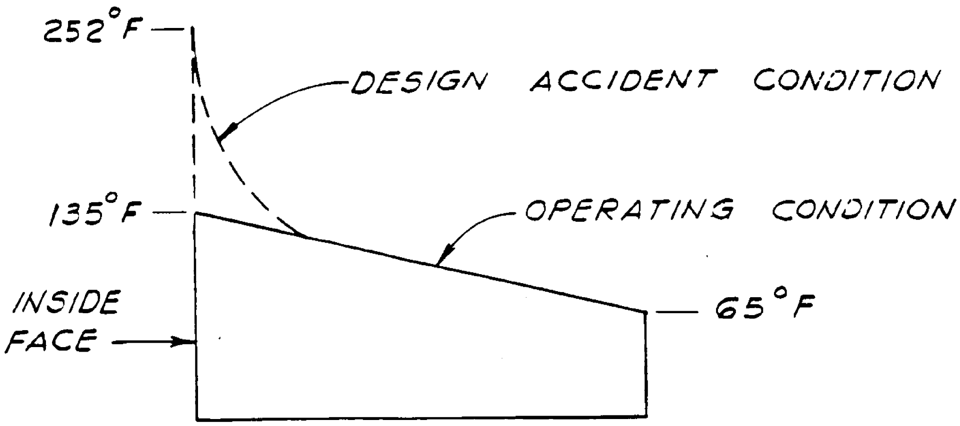
FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

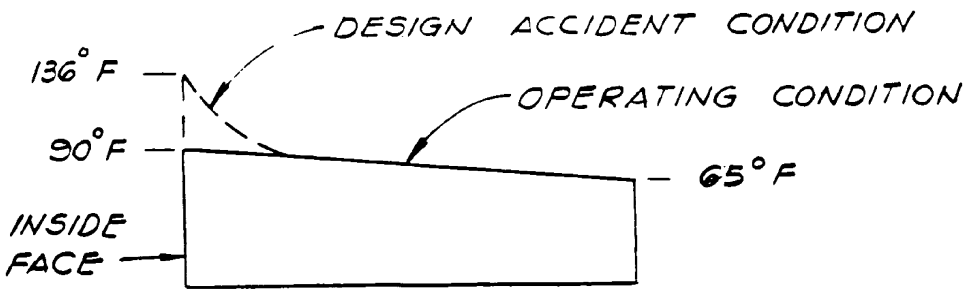
Figure 3.8-23 replaced by dwg.
C-278, Sh. 1

FIGURE 3.8-23, Rev. 55

AutoCAD Figure 3_8_23.doc



DRYWELL WALL



SUPPRESSION CHAMBER WALL

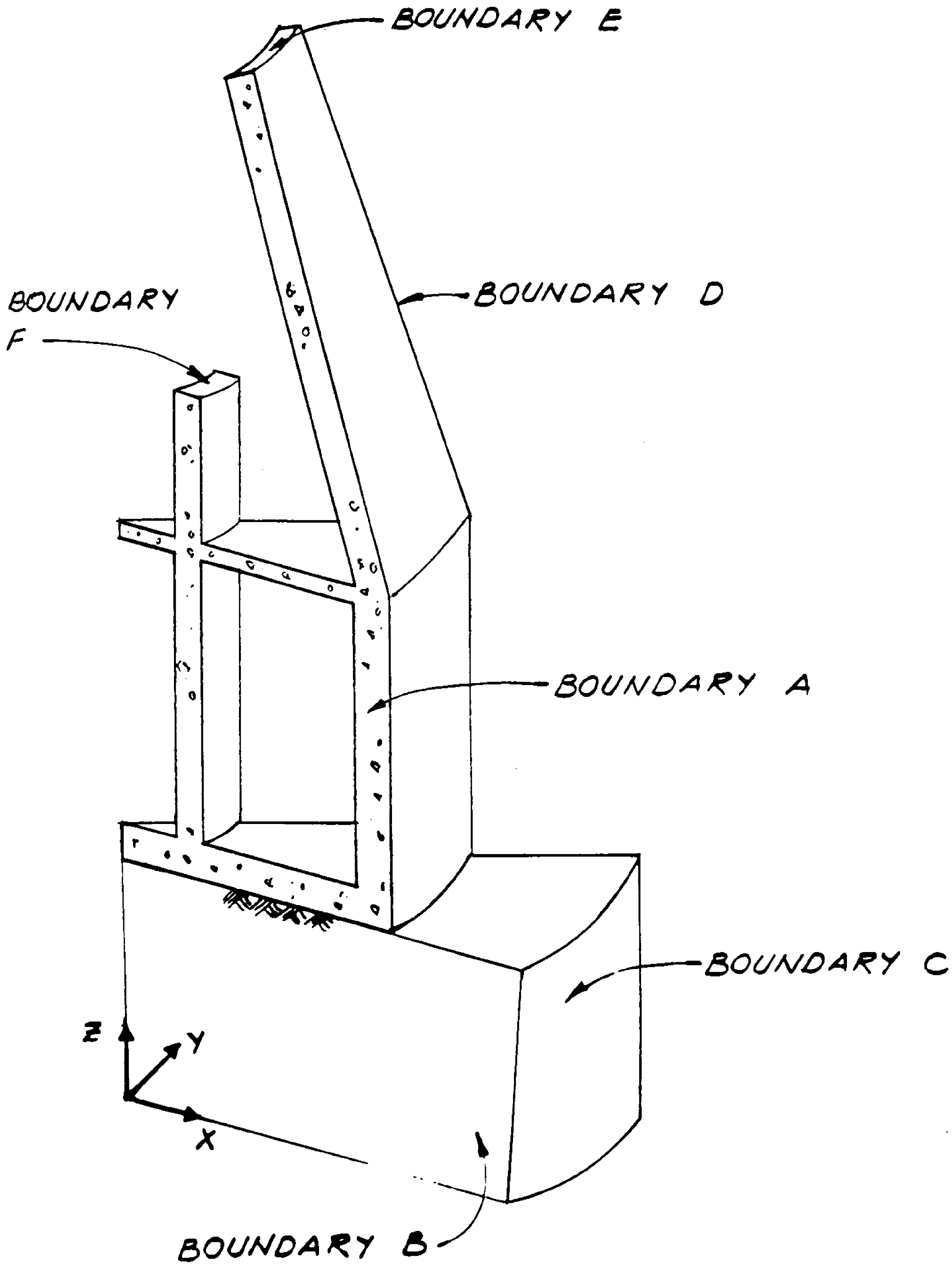
FSAR REV.65

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



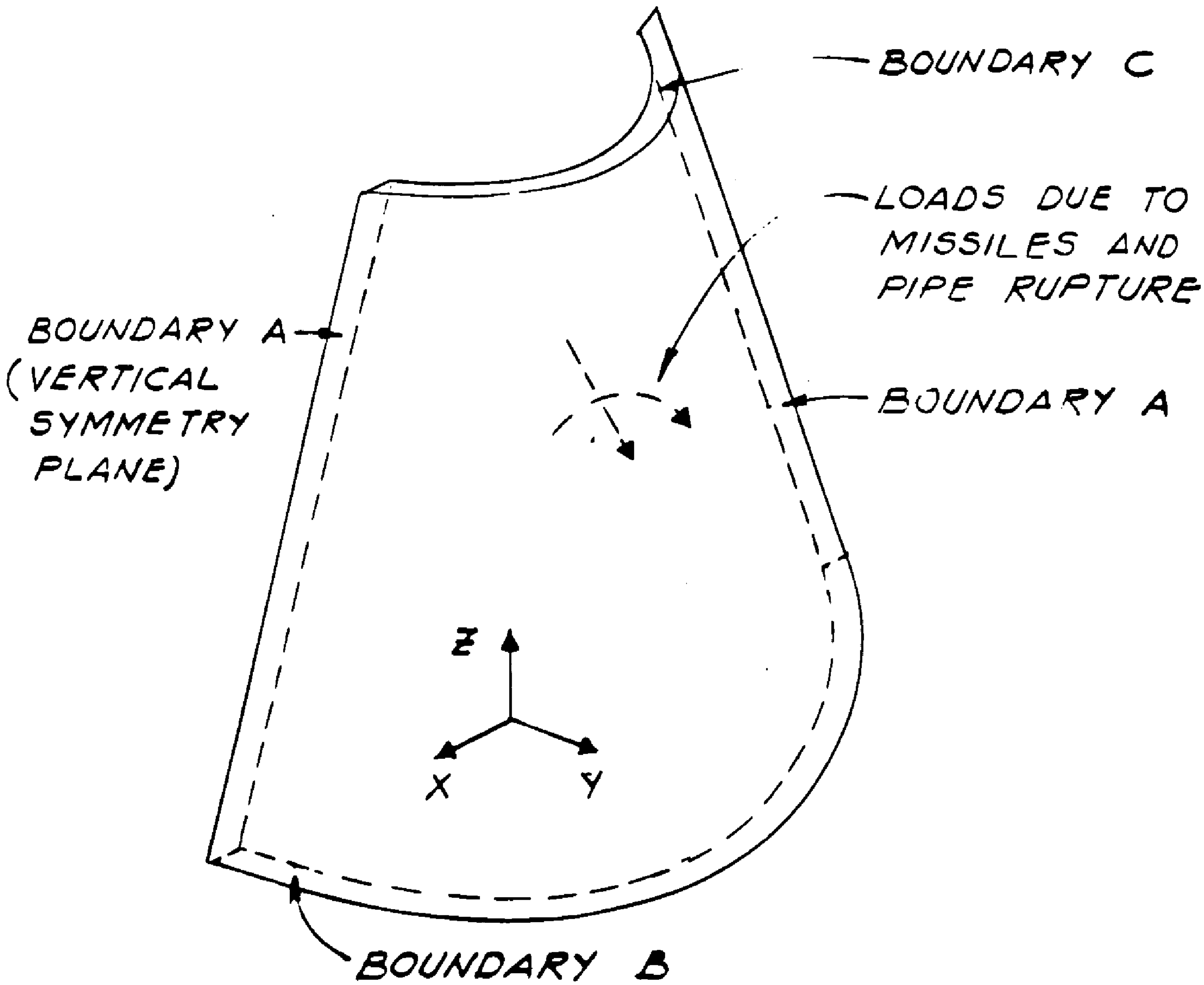
FSAR REV.65

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



FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
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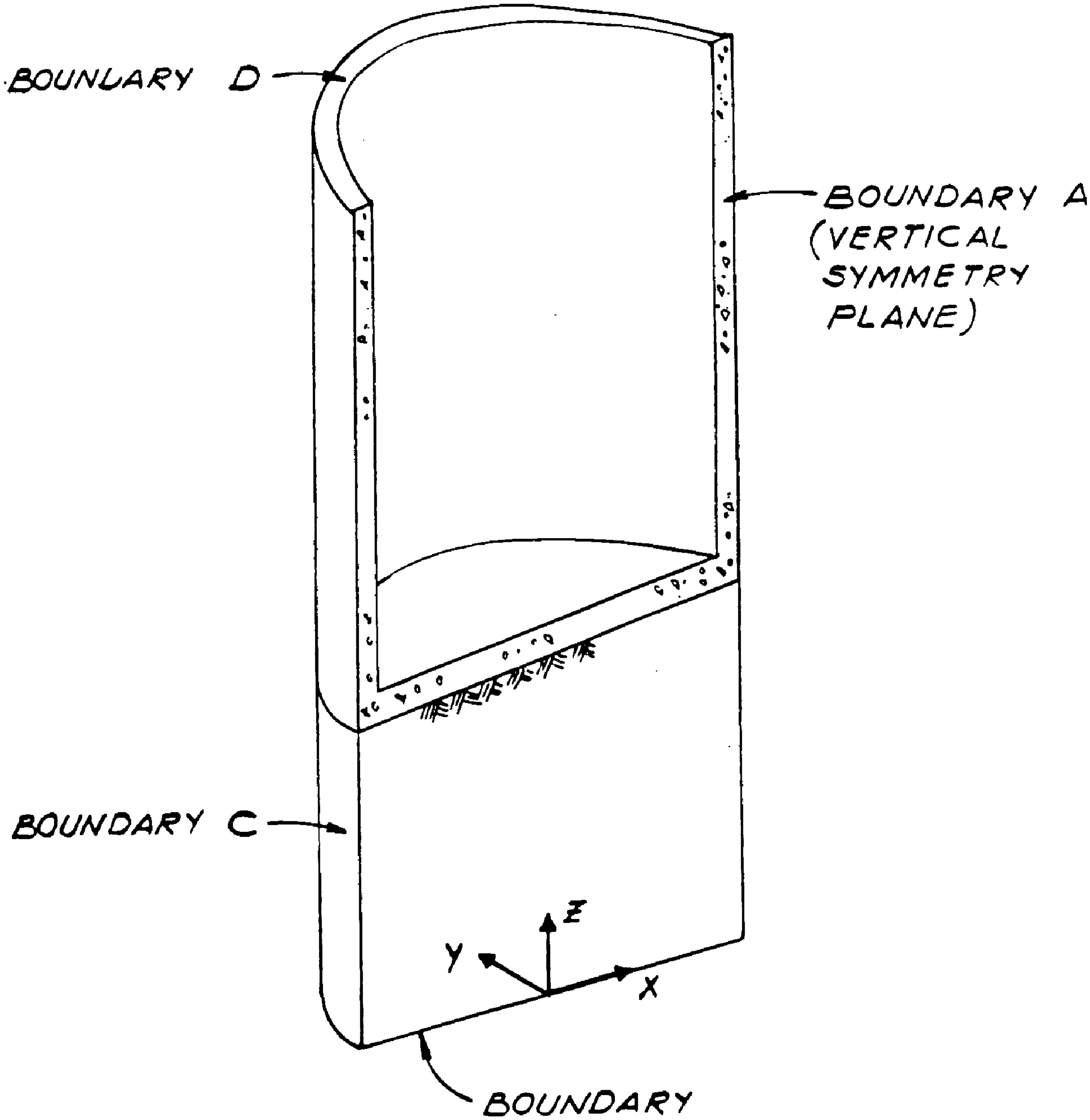
FINAL SAFETY ANALYSIS REPORT

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



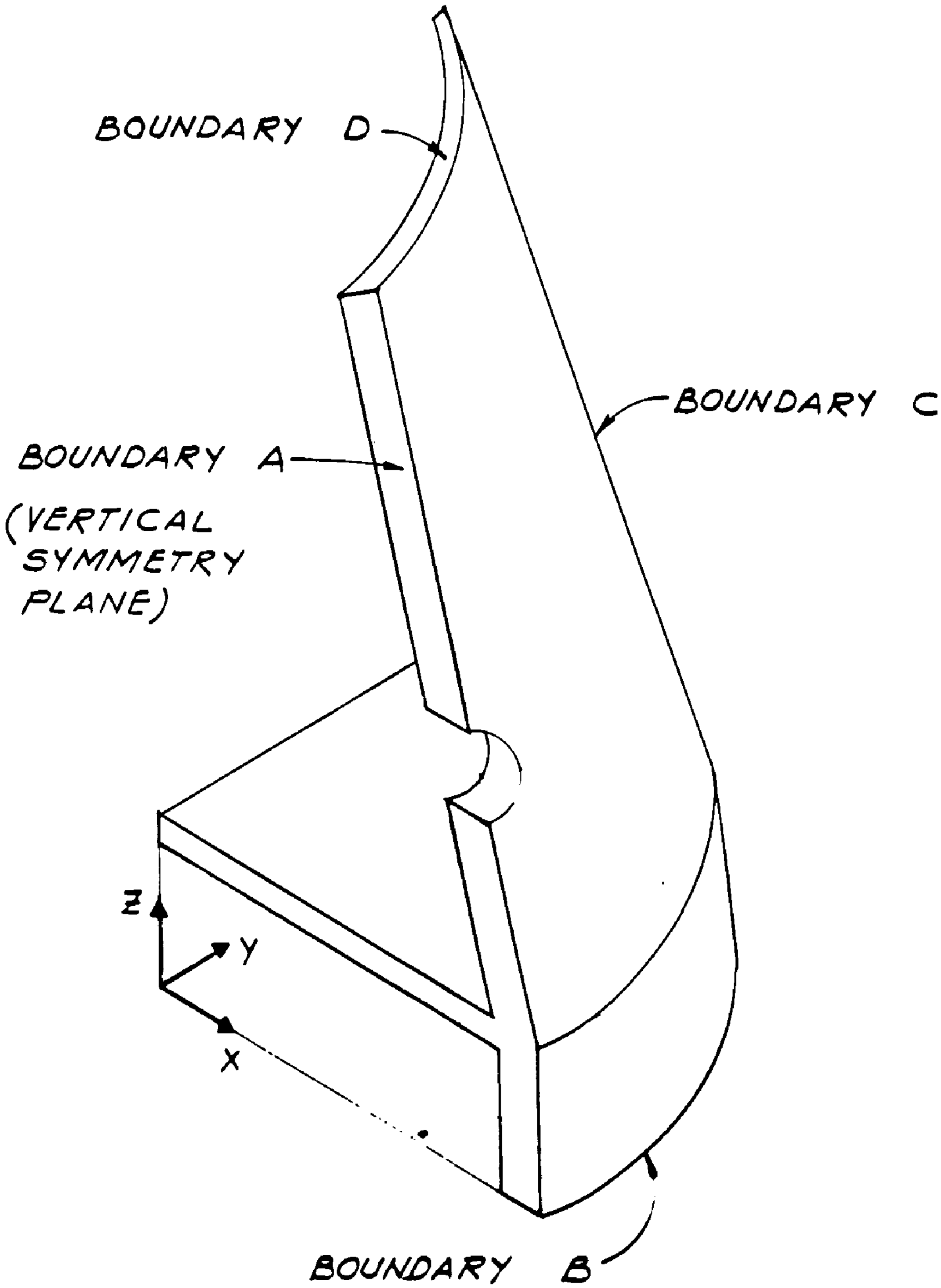
FSAR REV.65

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



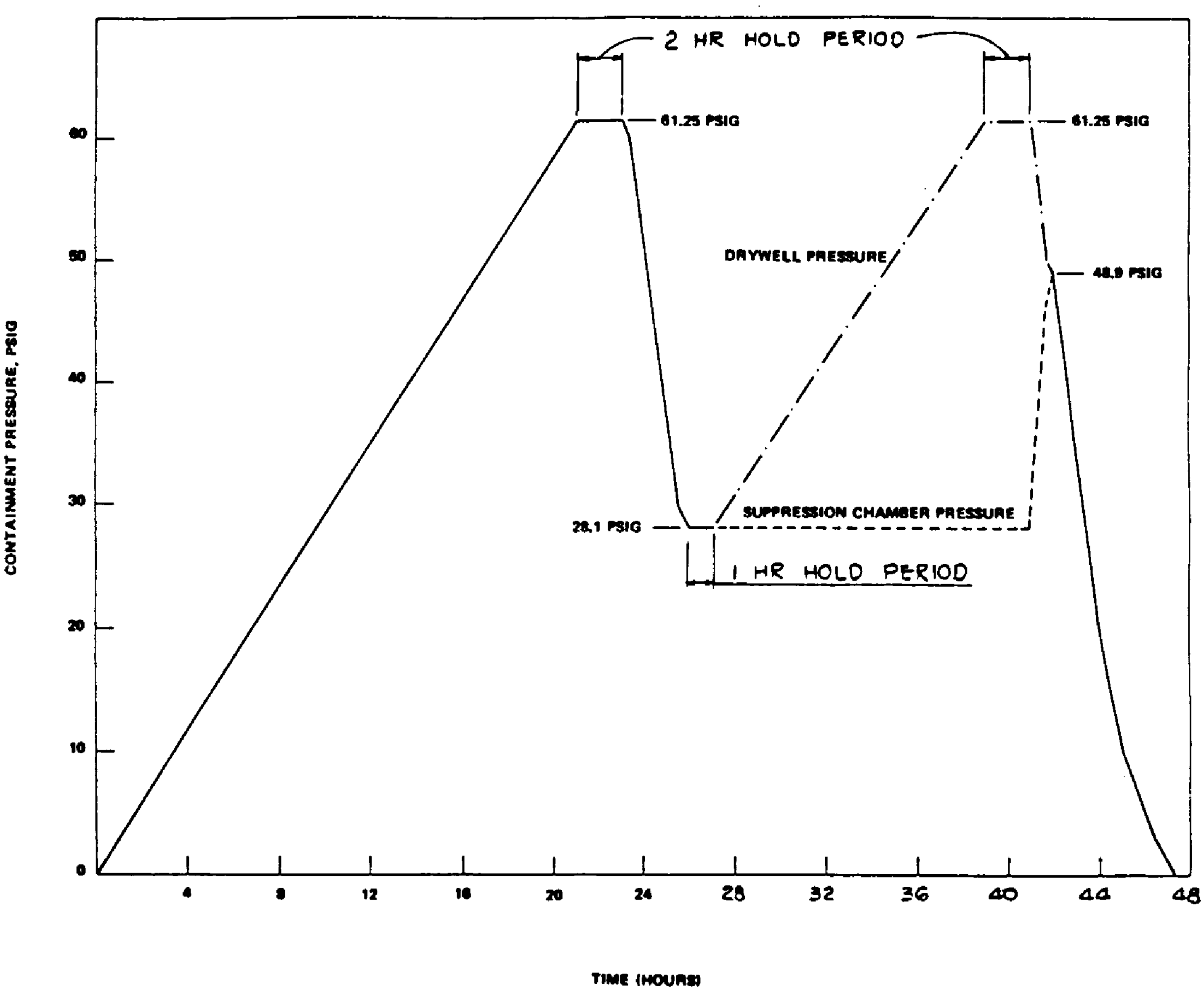
FSAR REV.65

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



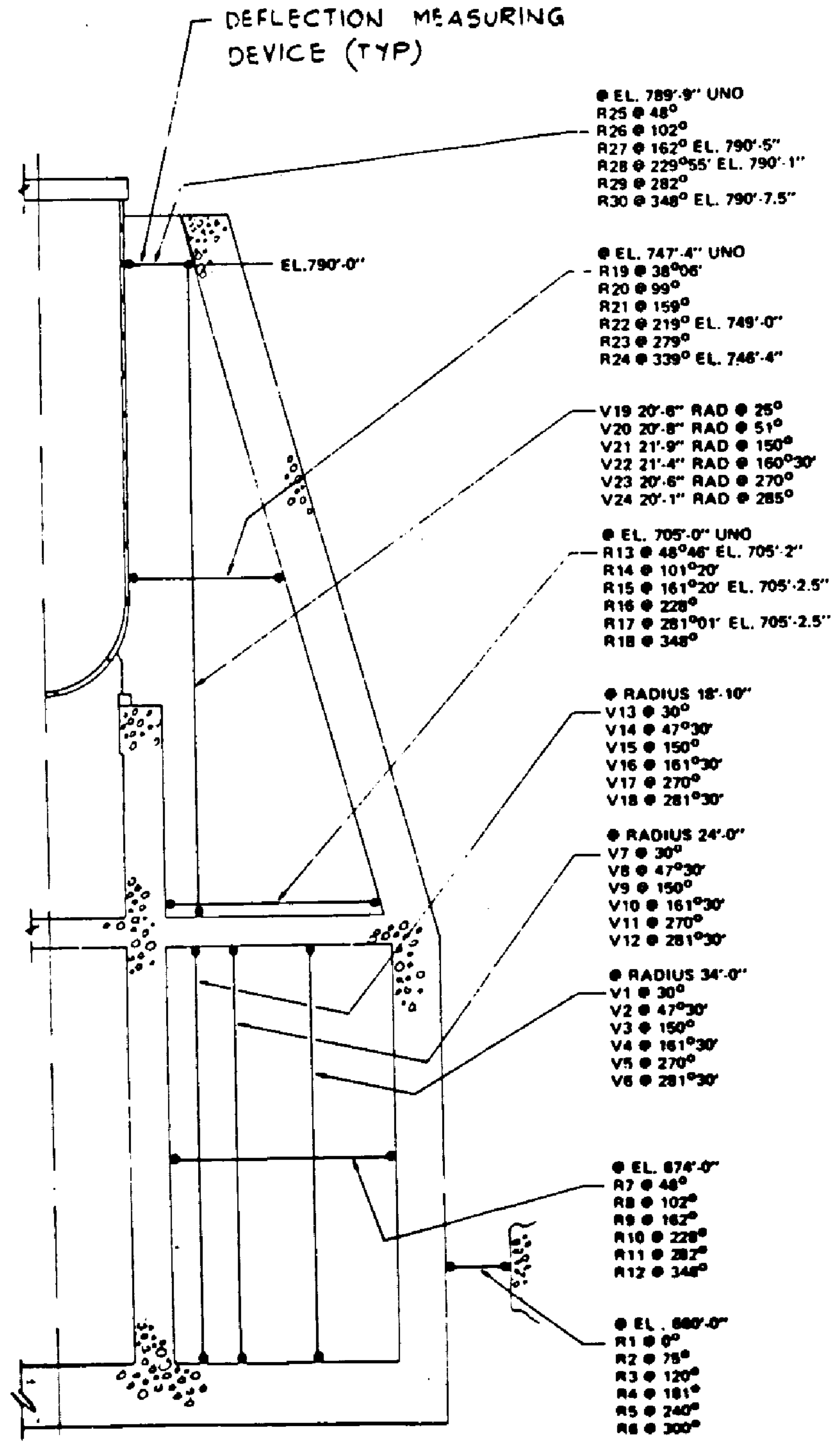
FSAR REV.65

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



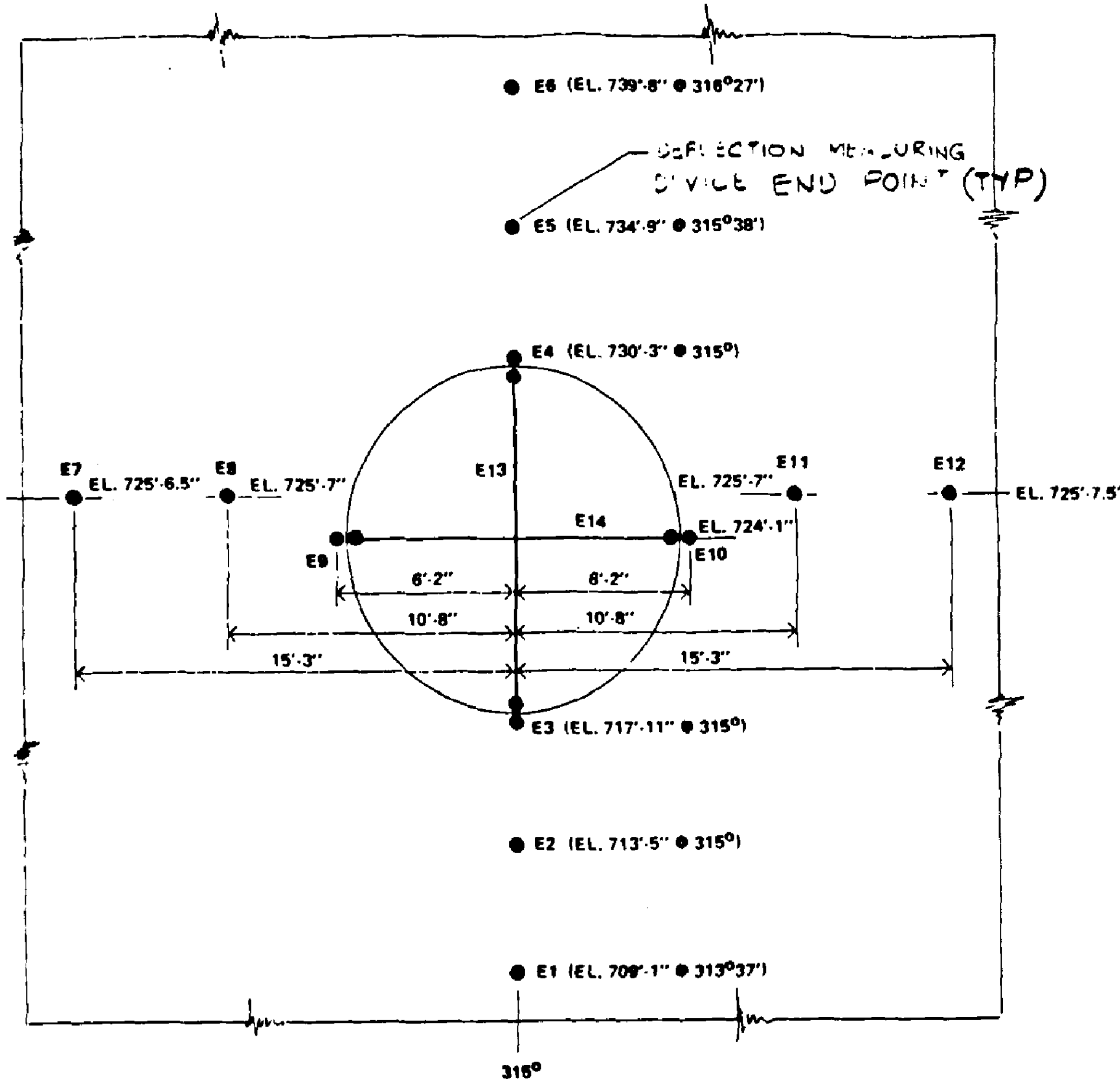
FSAR REV.65

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)

FSAR REV.65

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

FSAR REV. 65

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

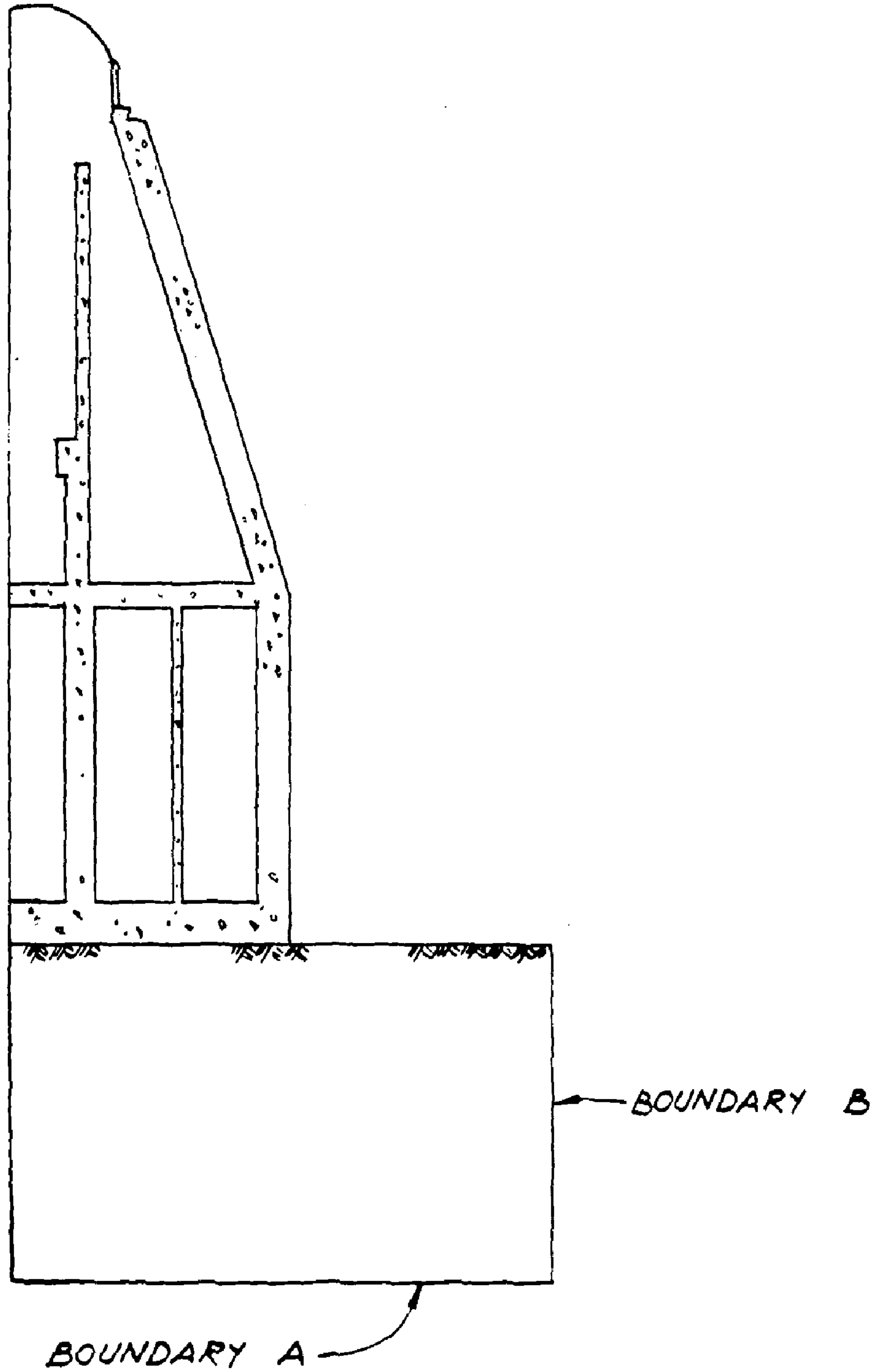
AutoCAD Figure 3_8_32.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
C-387, Sh. 1

FSAR REV. 65

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



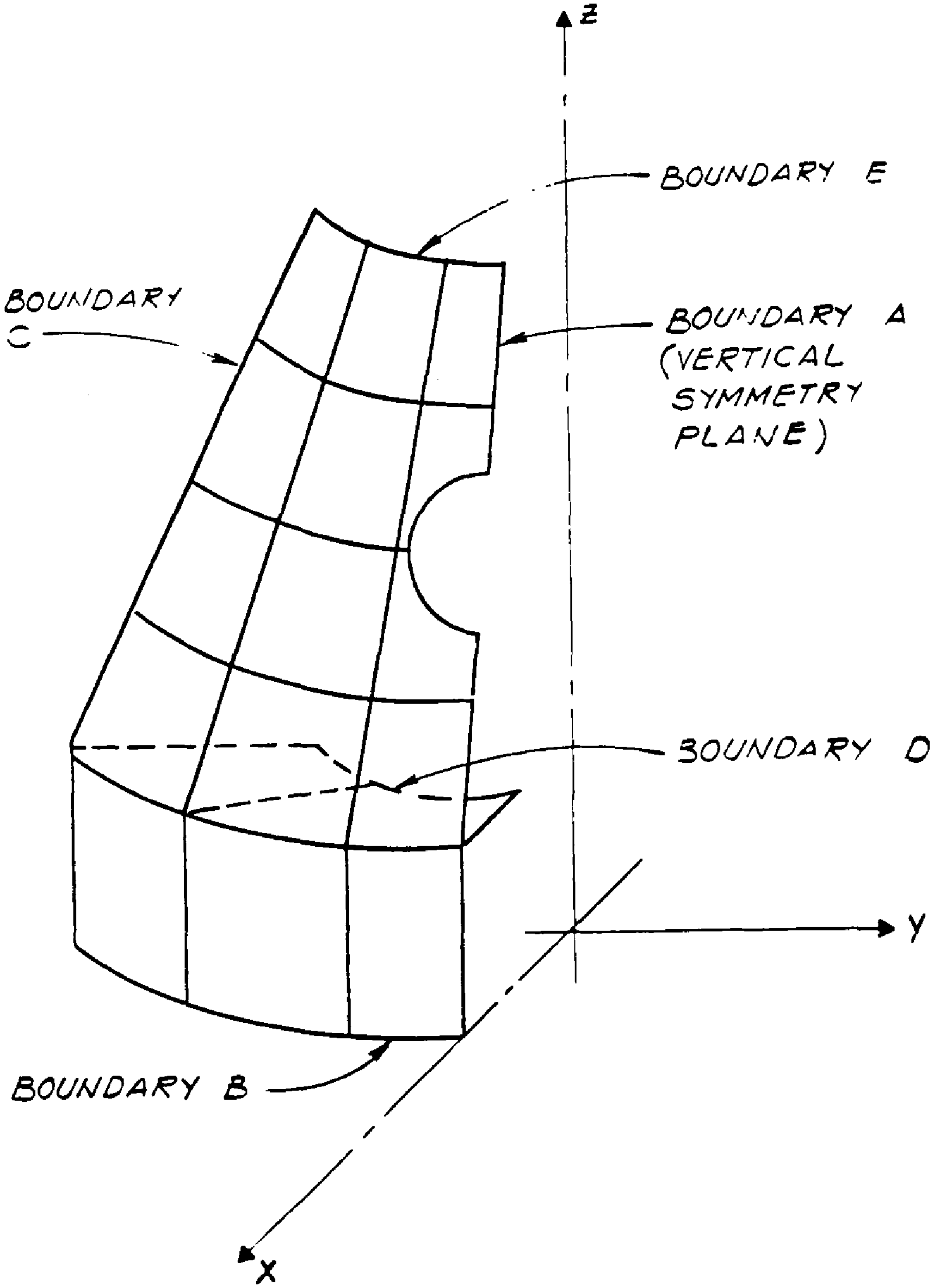
FSAR REV.65

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



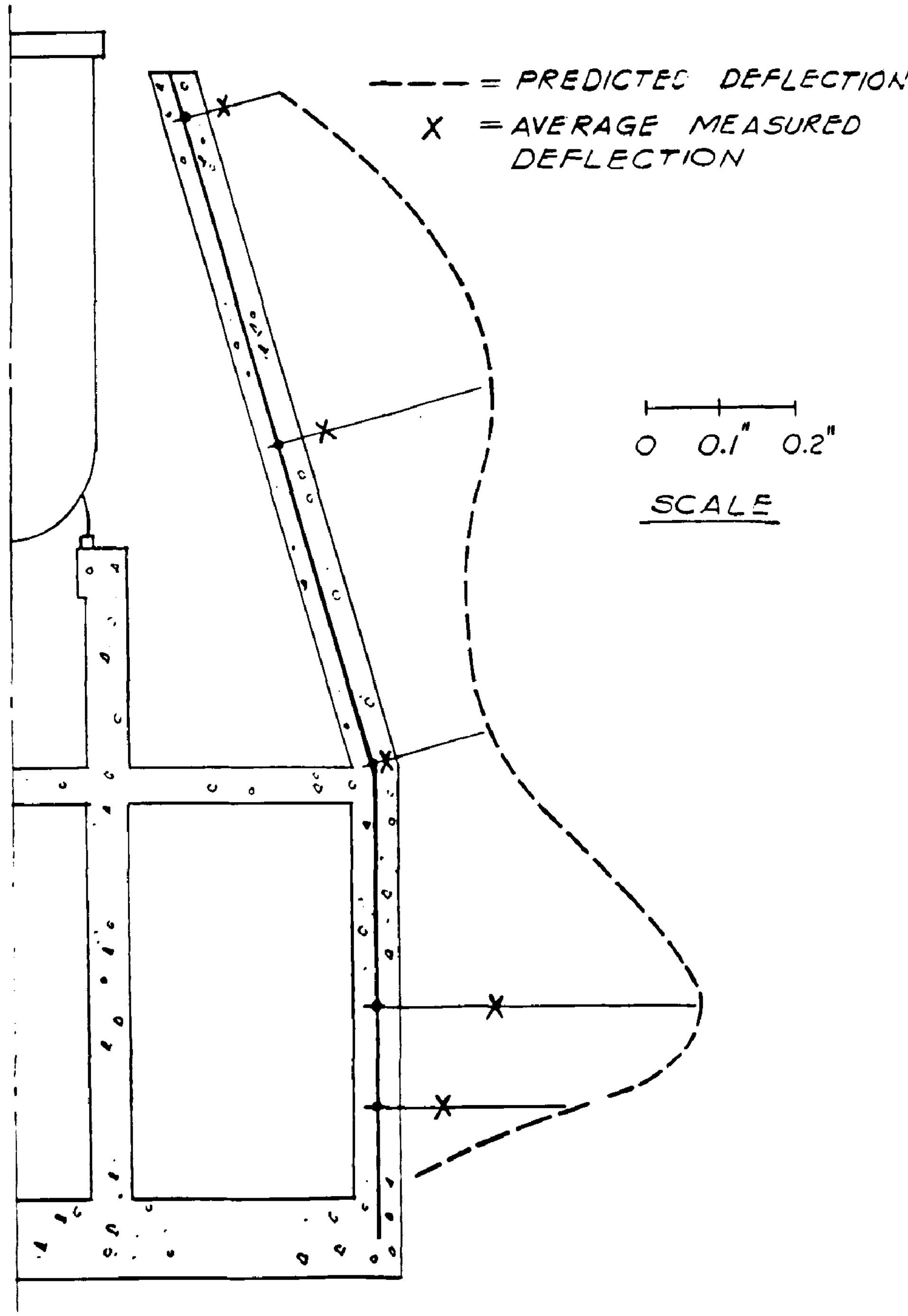
FSAR REV.65

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



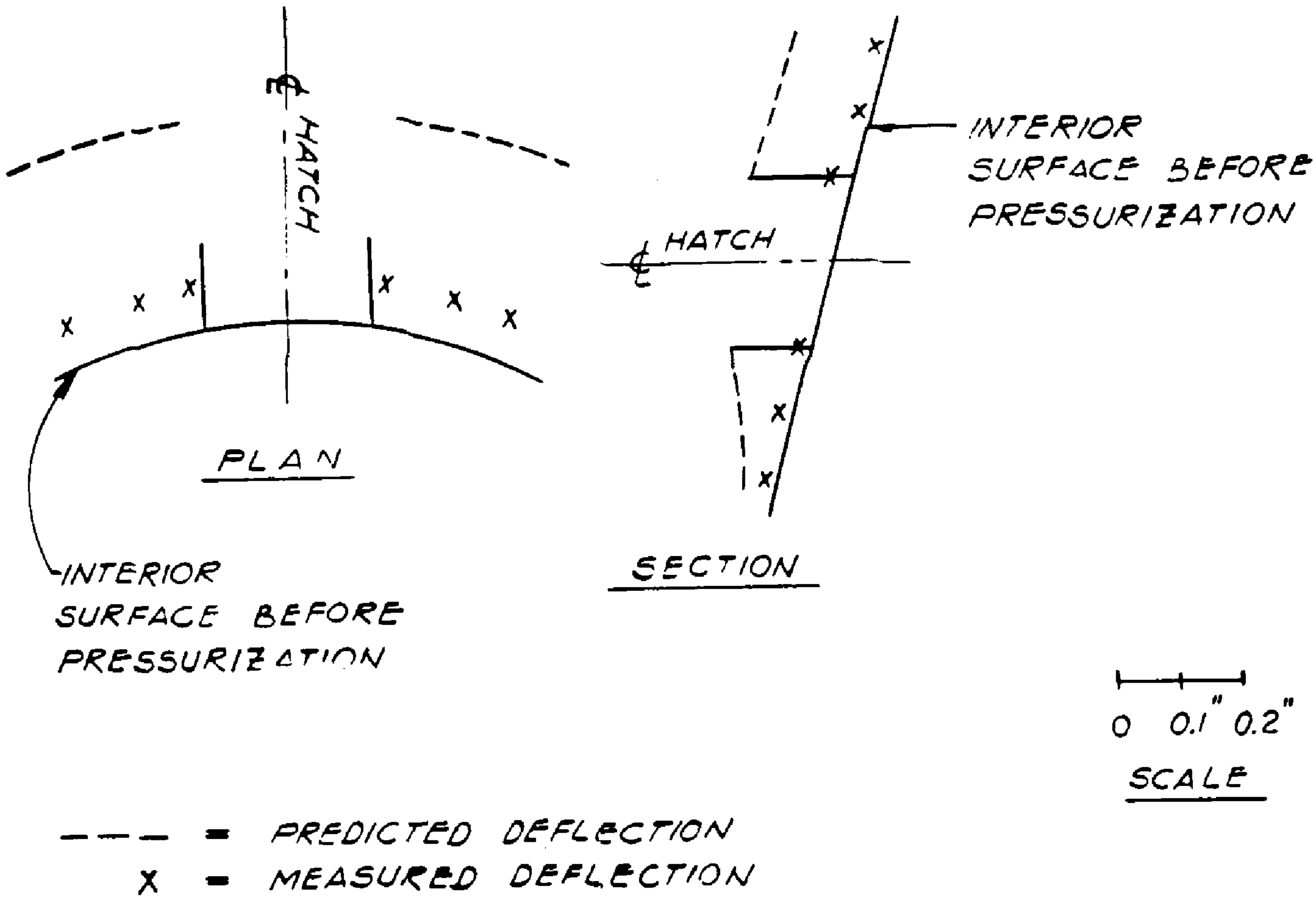
FSAR REV.65

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



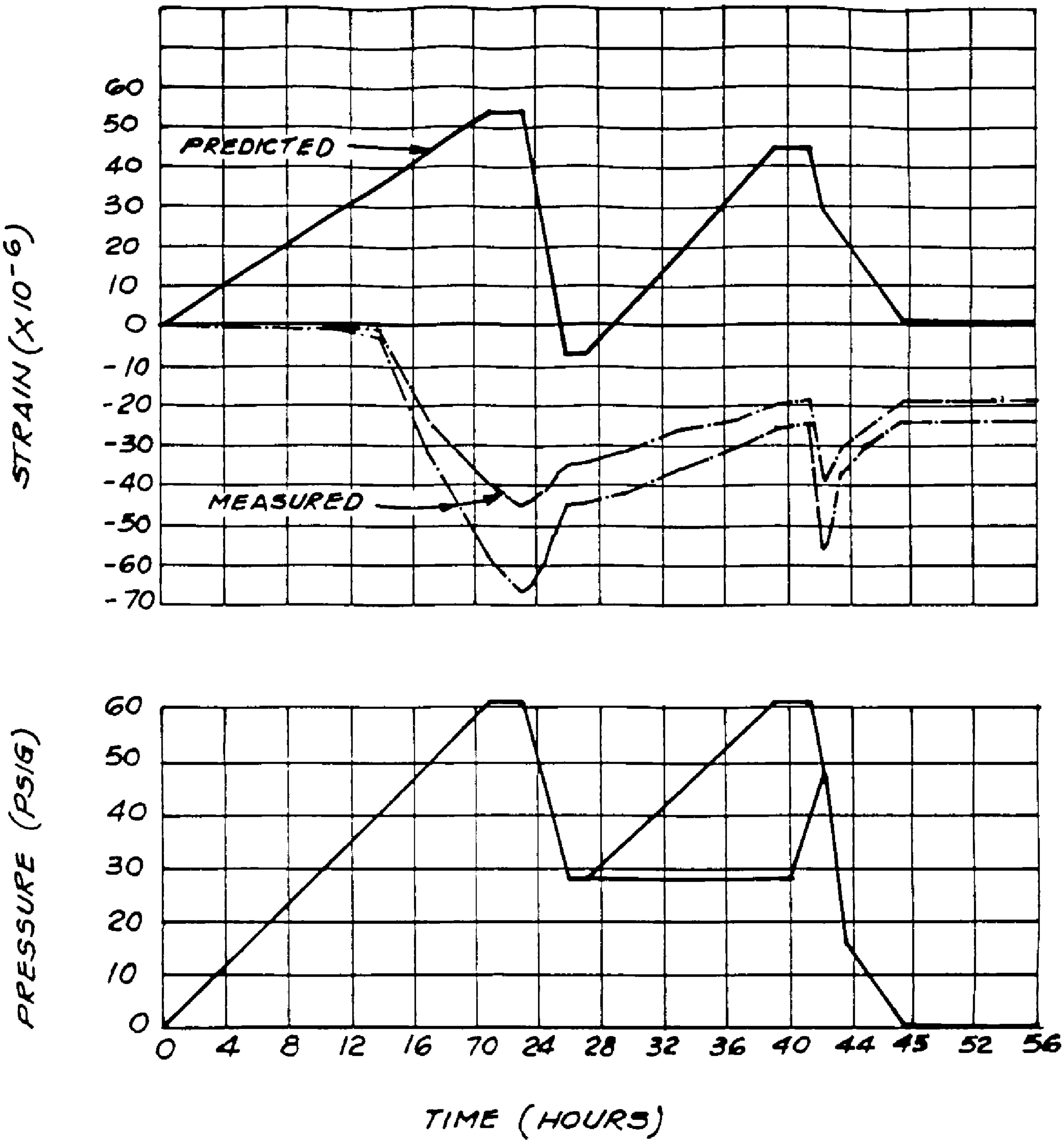
FSAR REV.65

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



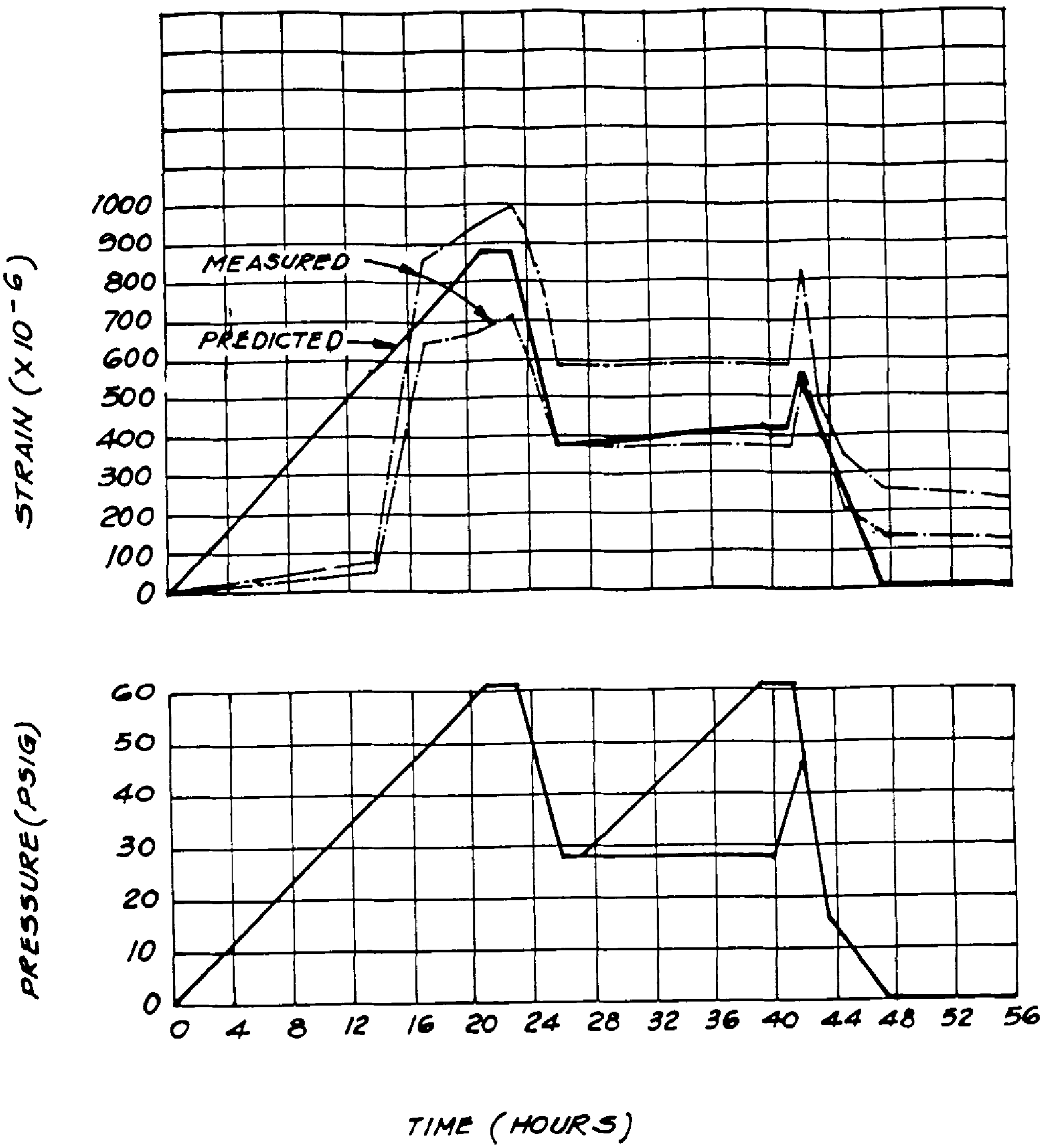
FSAR REV.65

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



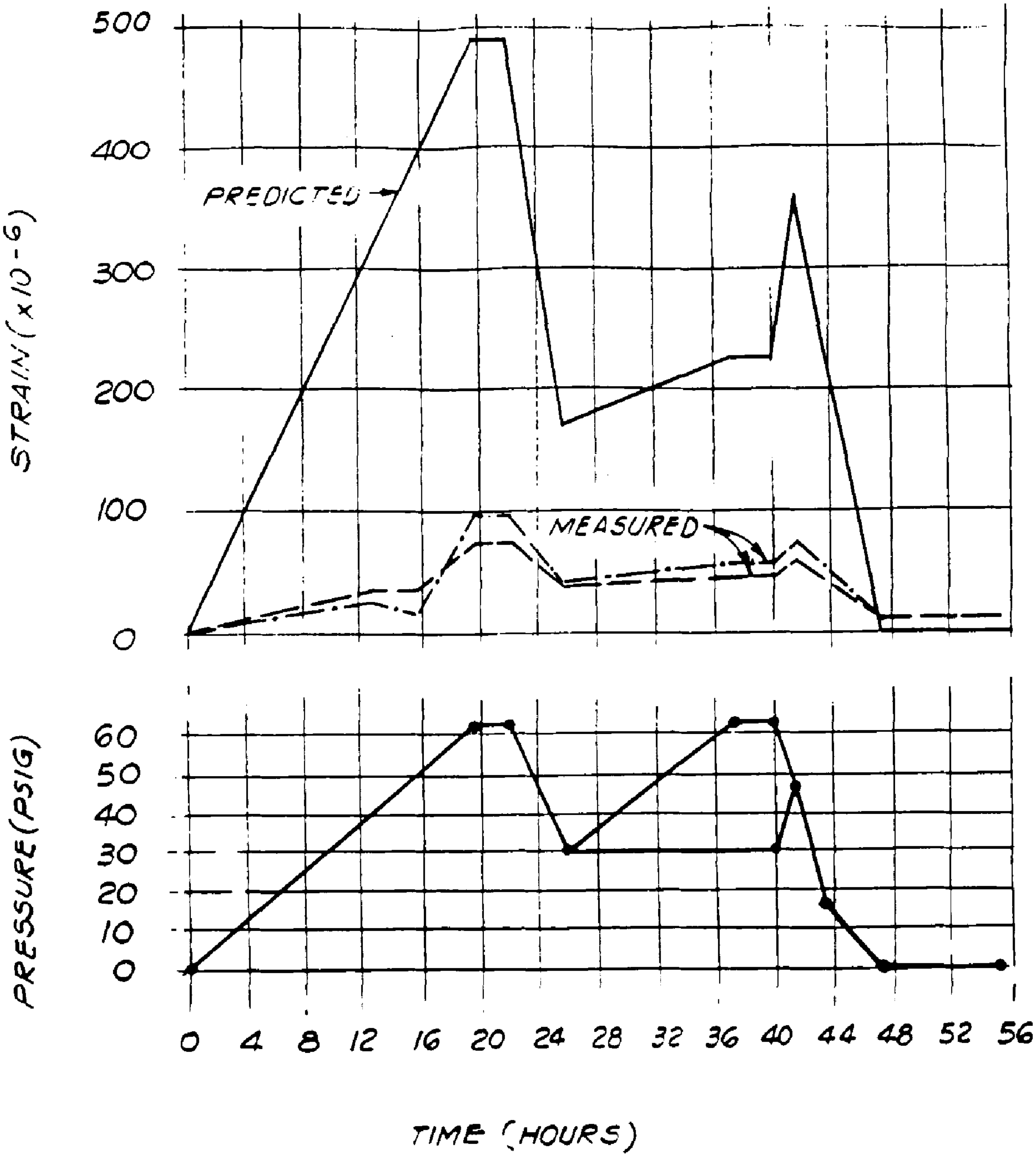
FSAR REV.65

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



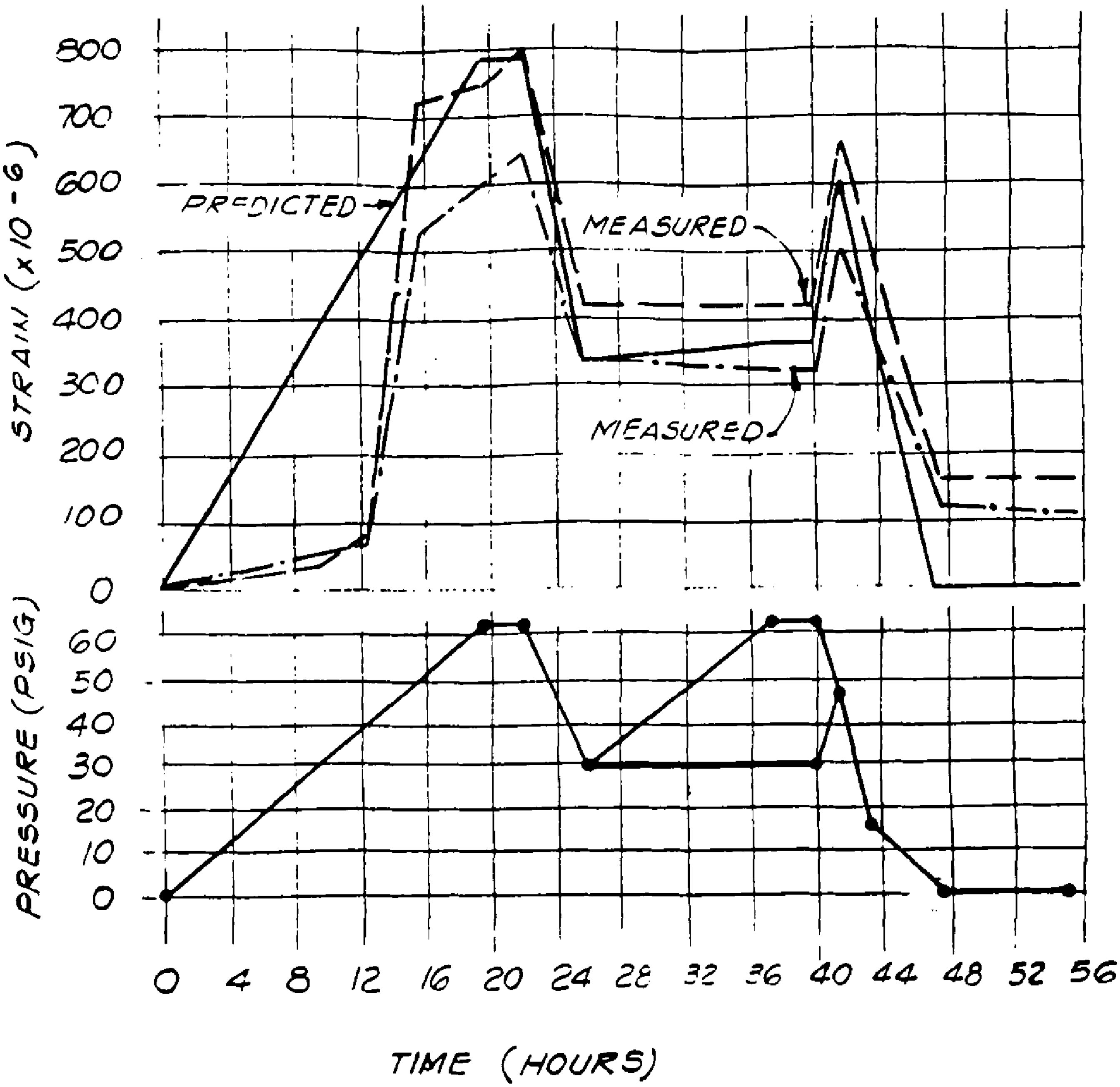
FSAR REV.65

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



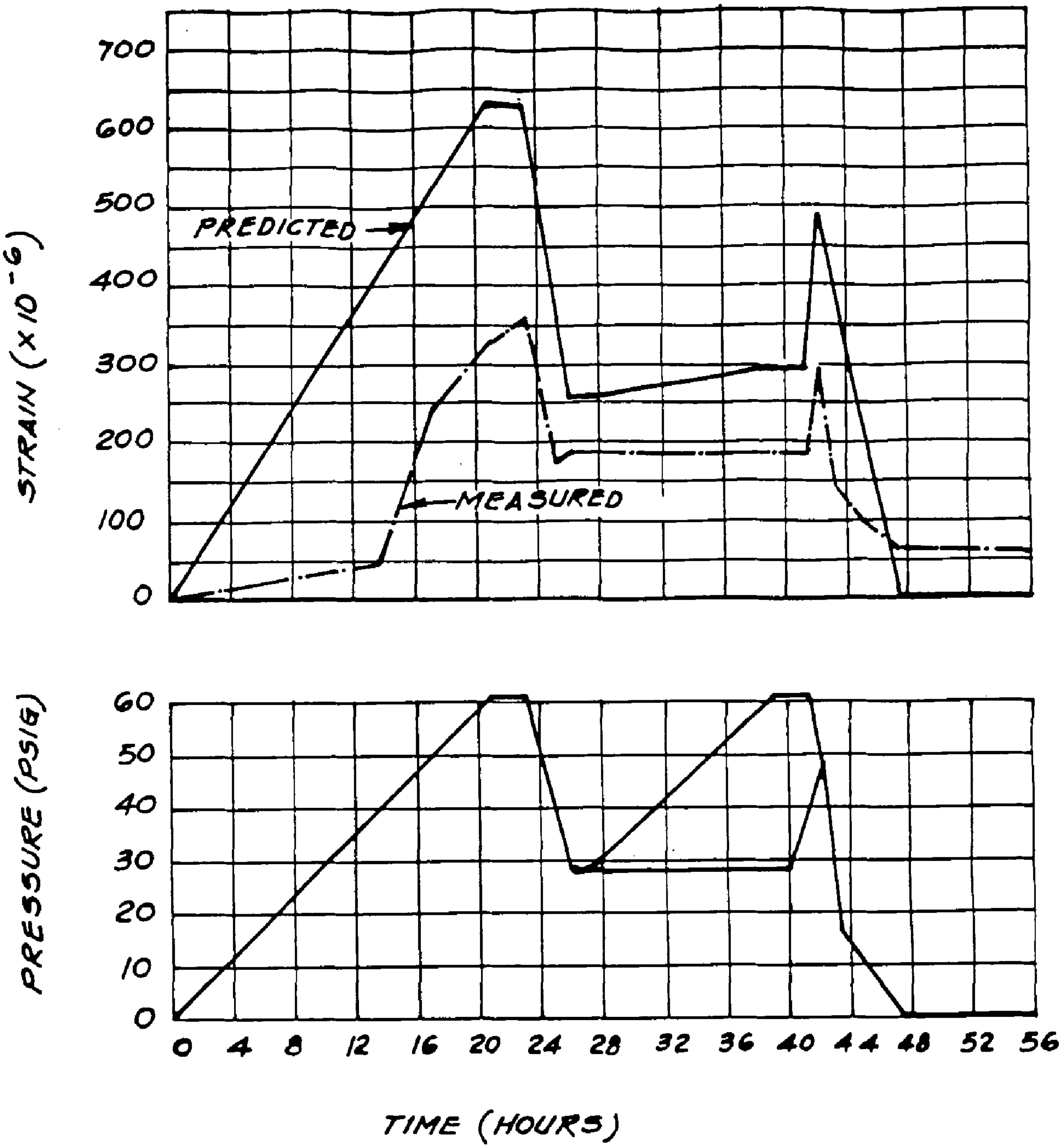
FSAR REV.65

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



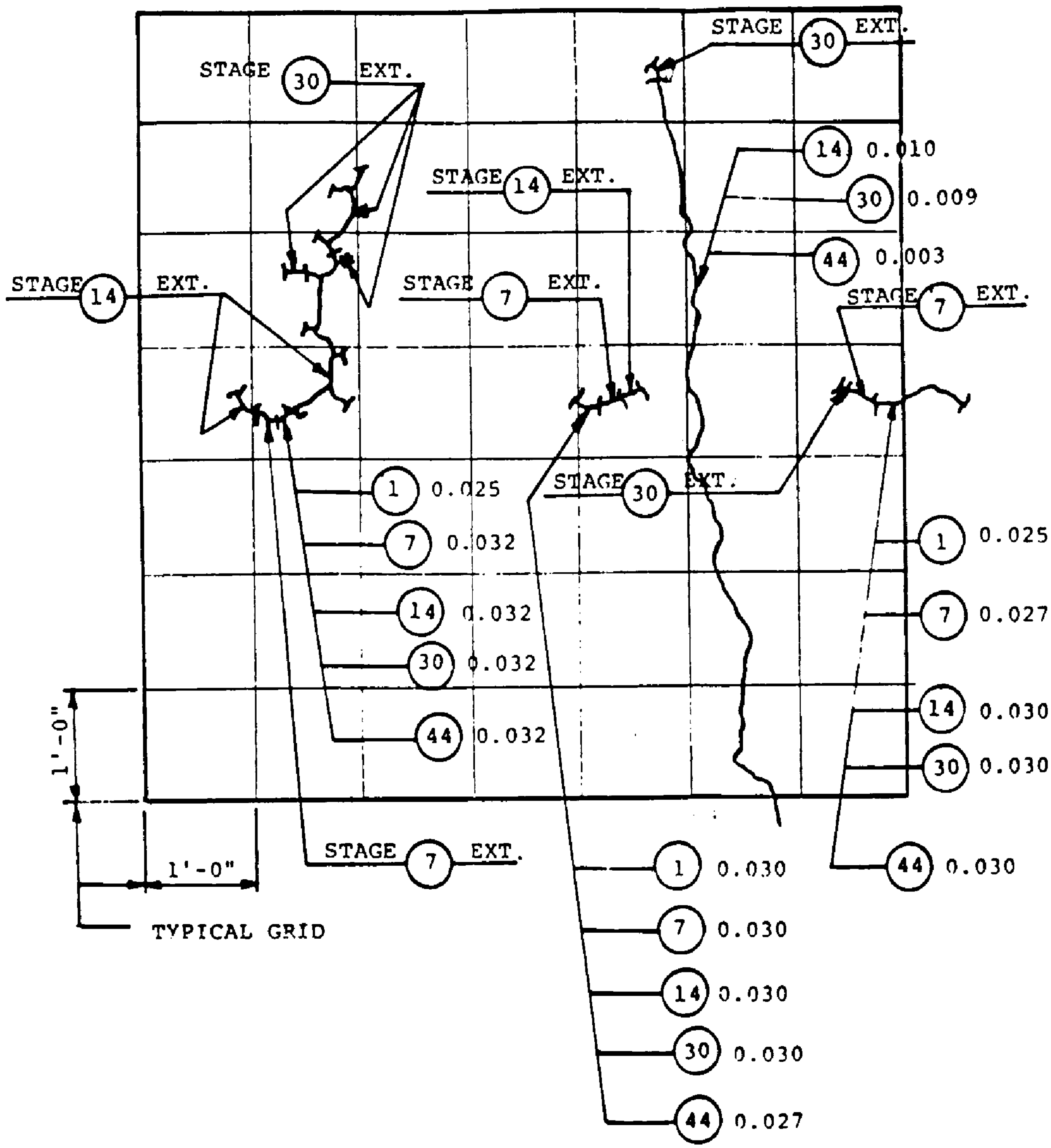
FSAR REV.65

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



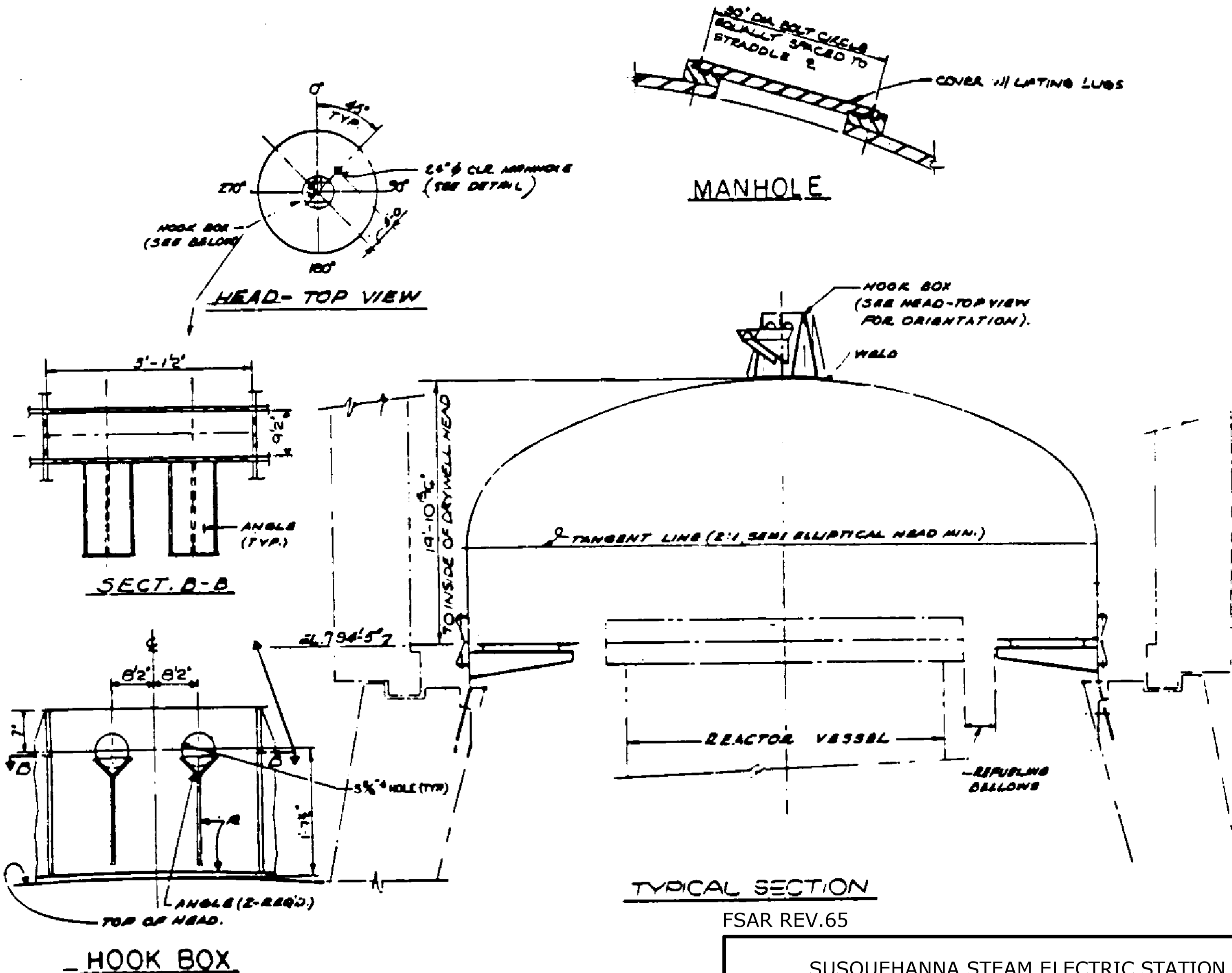
FSAR REV.65

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



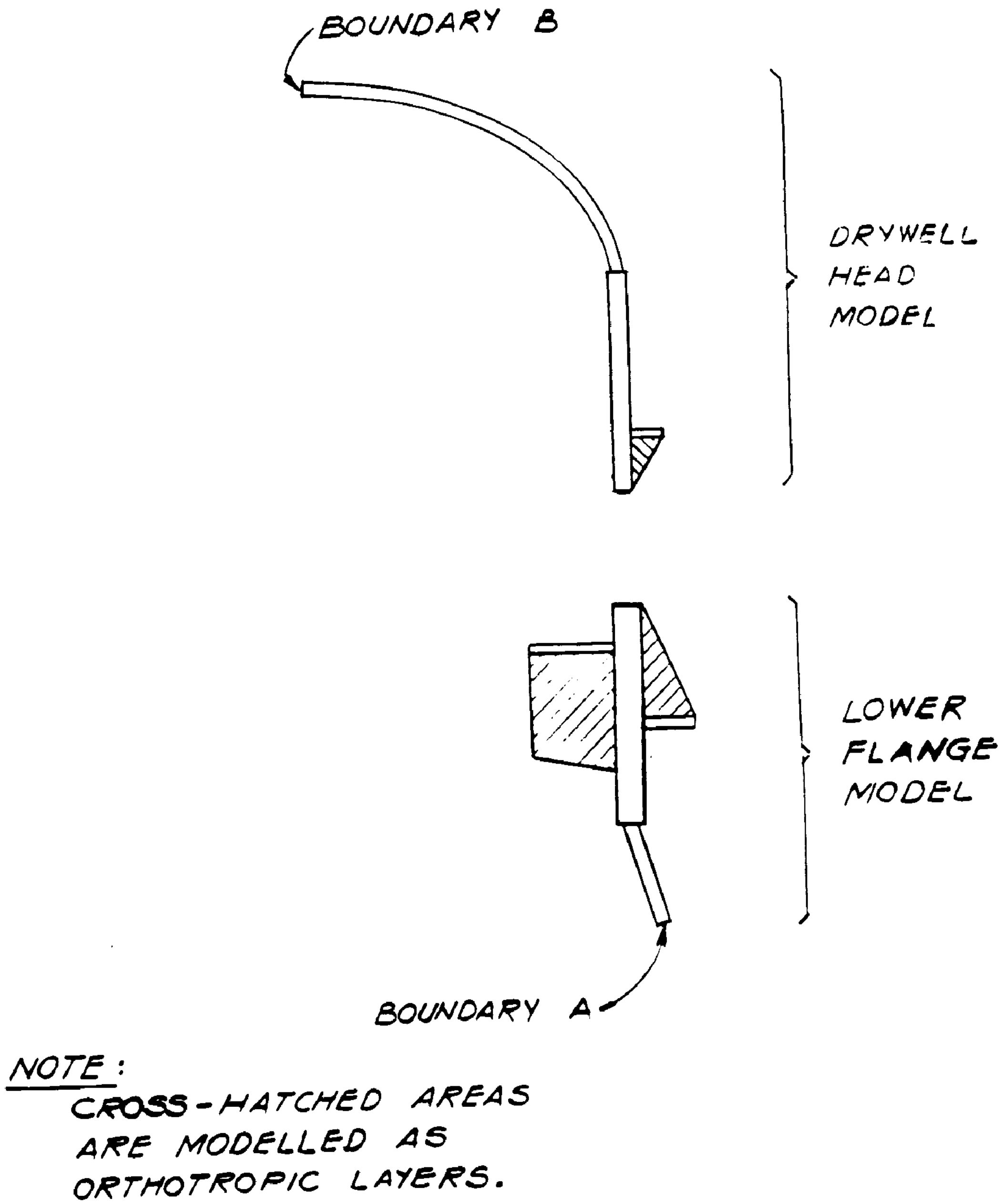
FSAR REV.65

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



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-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

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
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-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

AutoCAD Figure 3_8_50.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
C-344, Sh. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

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
REPLACED BY DWG.
C-377, Sh. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

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-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

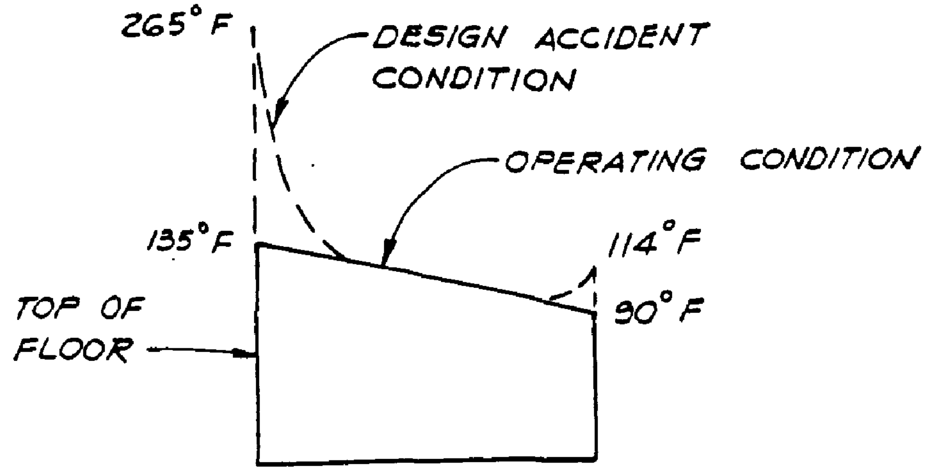
FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

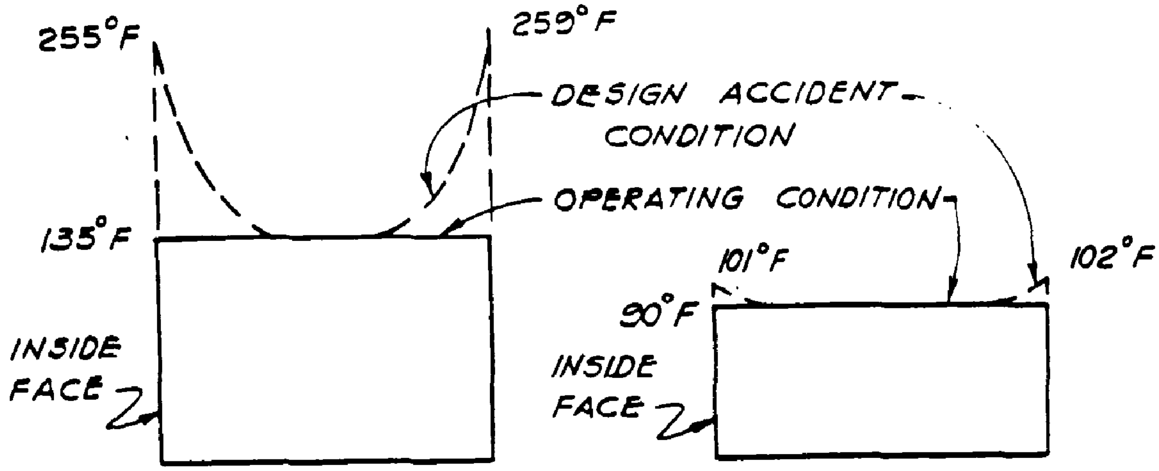
Figure 3.8-57 replaced by dwg.
C-380, Sh. 1

FIGURE 3.8-57, Rev. 48

AutoCAD Figure 3_8_57.doc



DRYWELL FLOOR



ABOVE DRYWELL FLOOR

BELOW DRYWELL FLOOR

REACTOR PEDESTAL

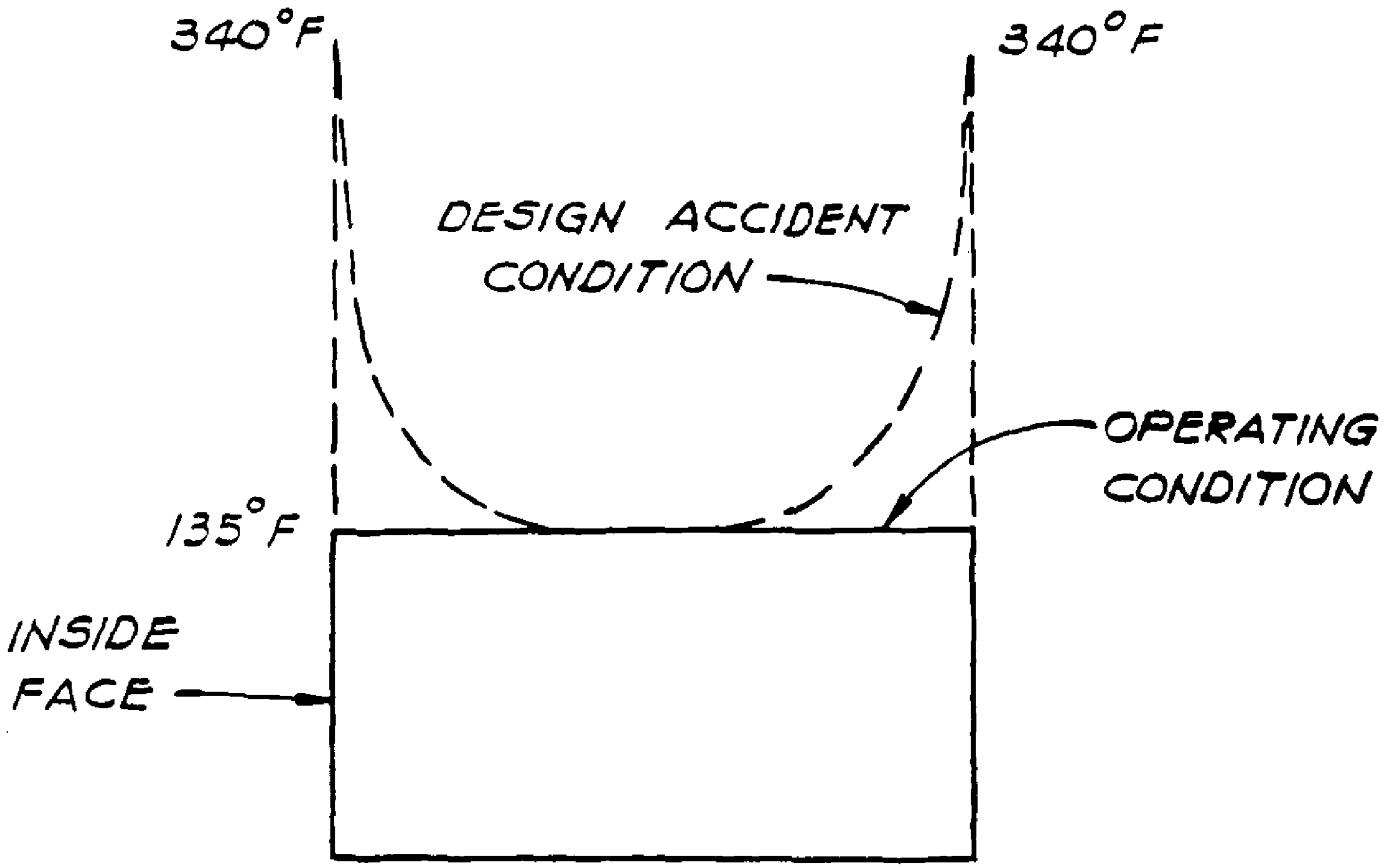
FSAR REV.65

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



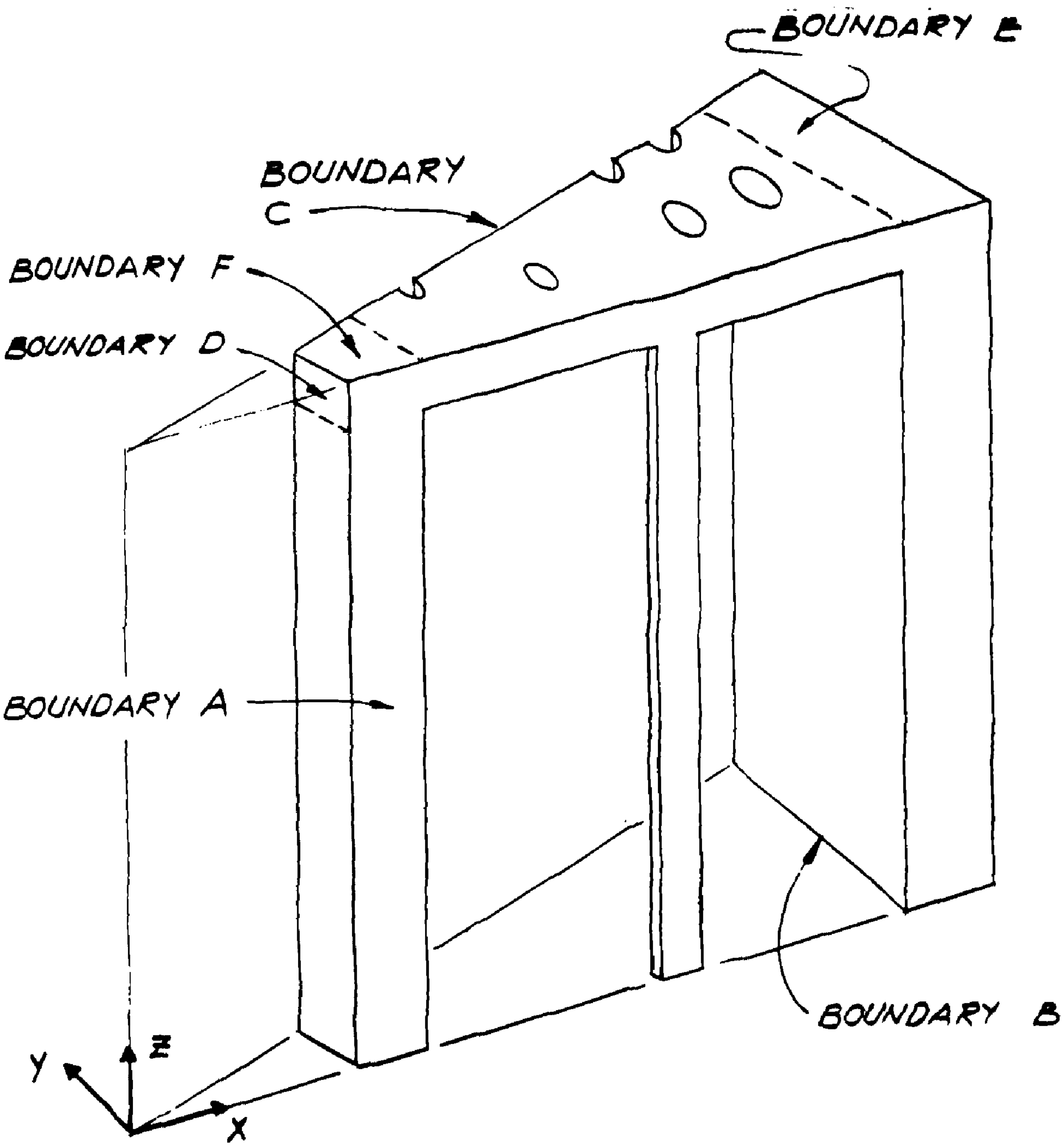
FSAR REV.65

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



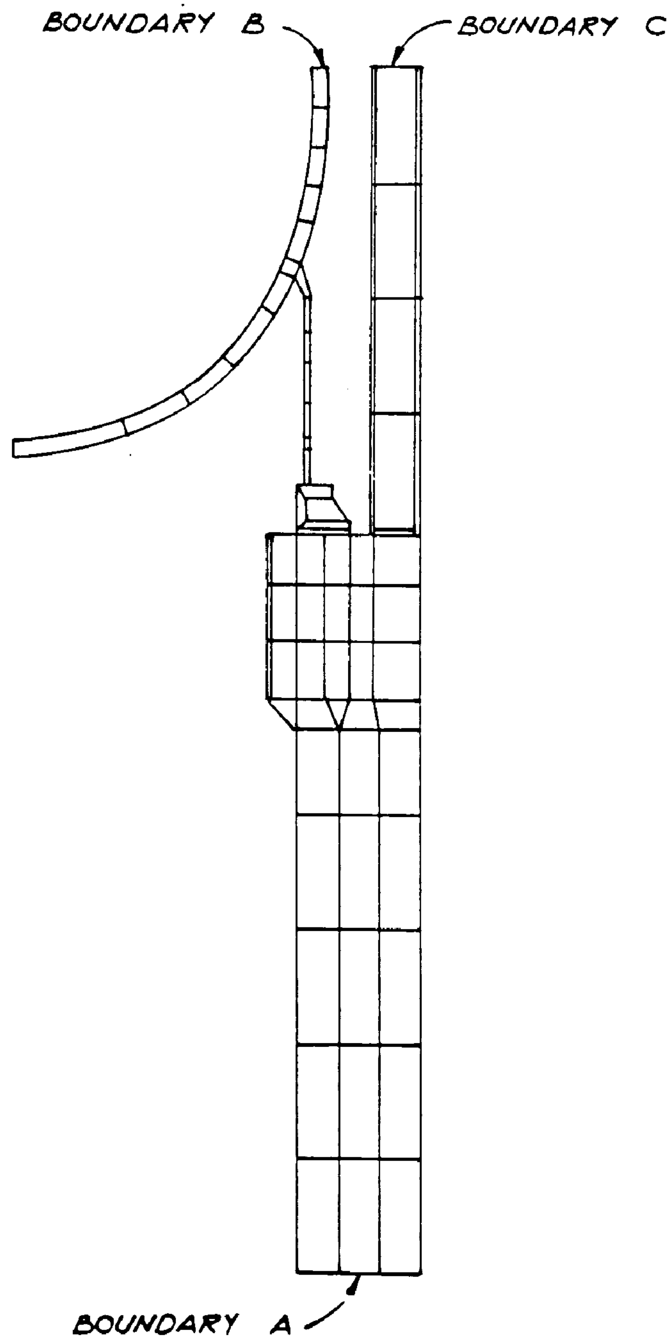
FSAR REV.65

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



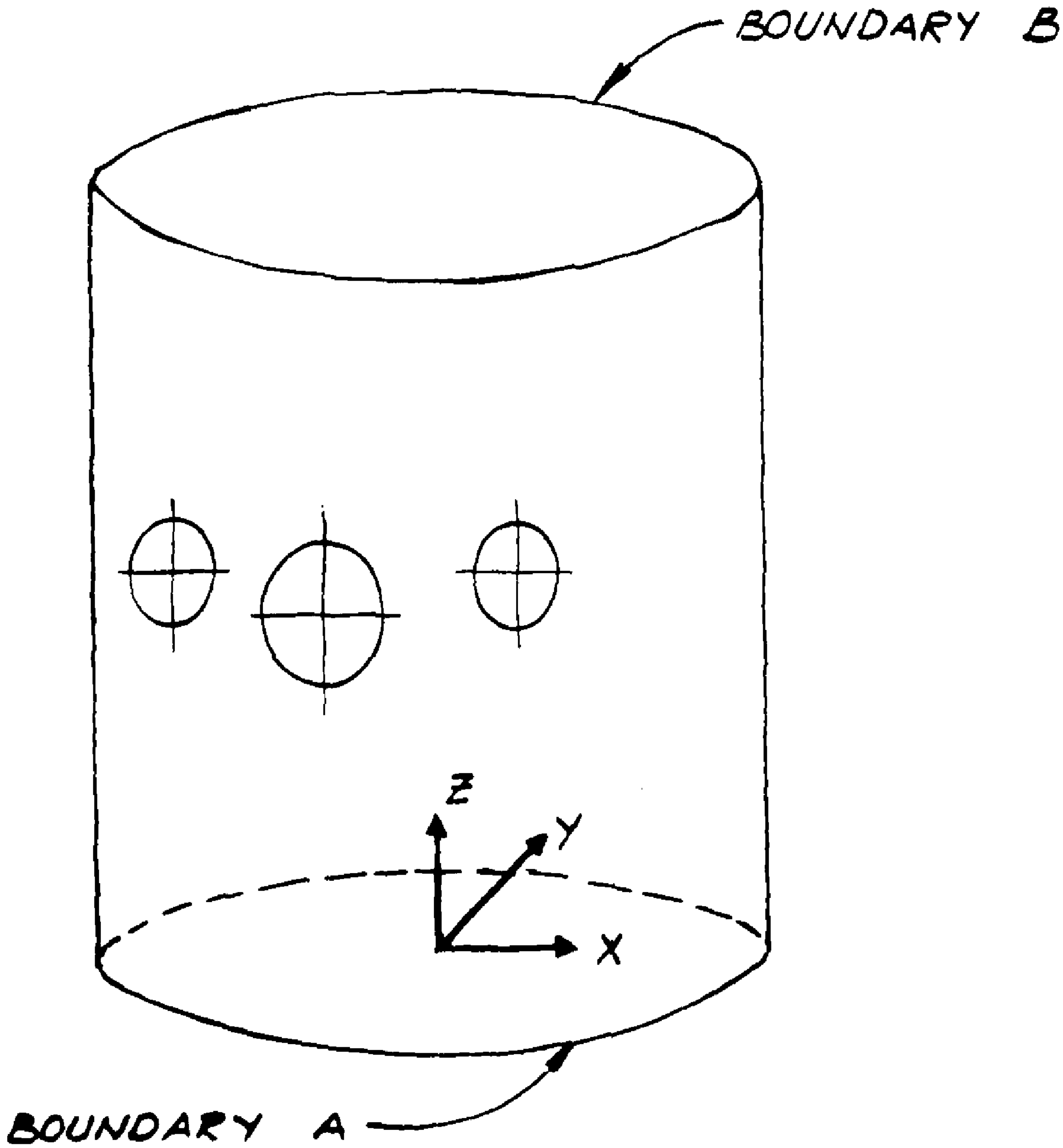
FSAR REV.65

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



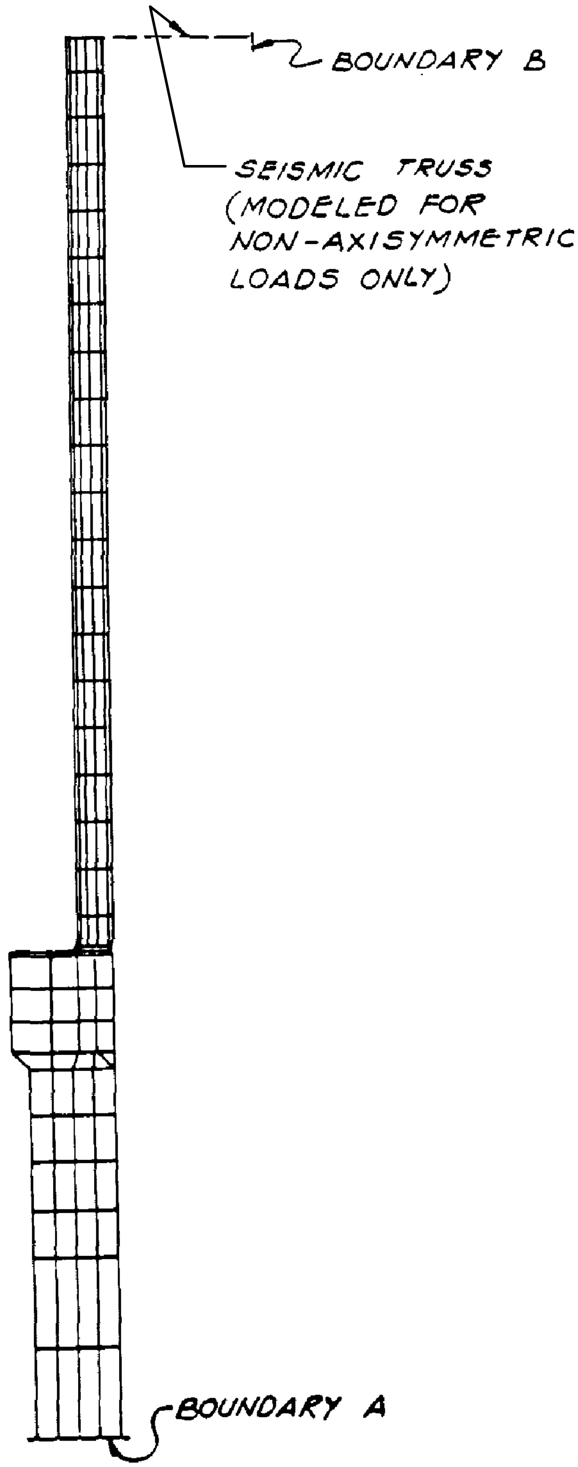
FSAR REV.65

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



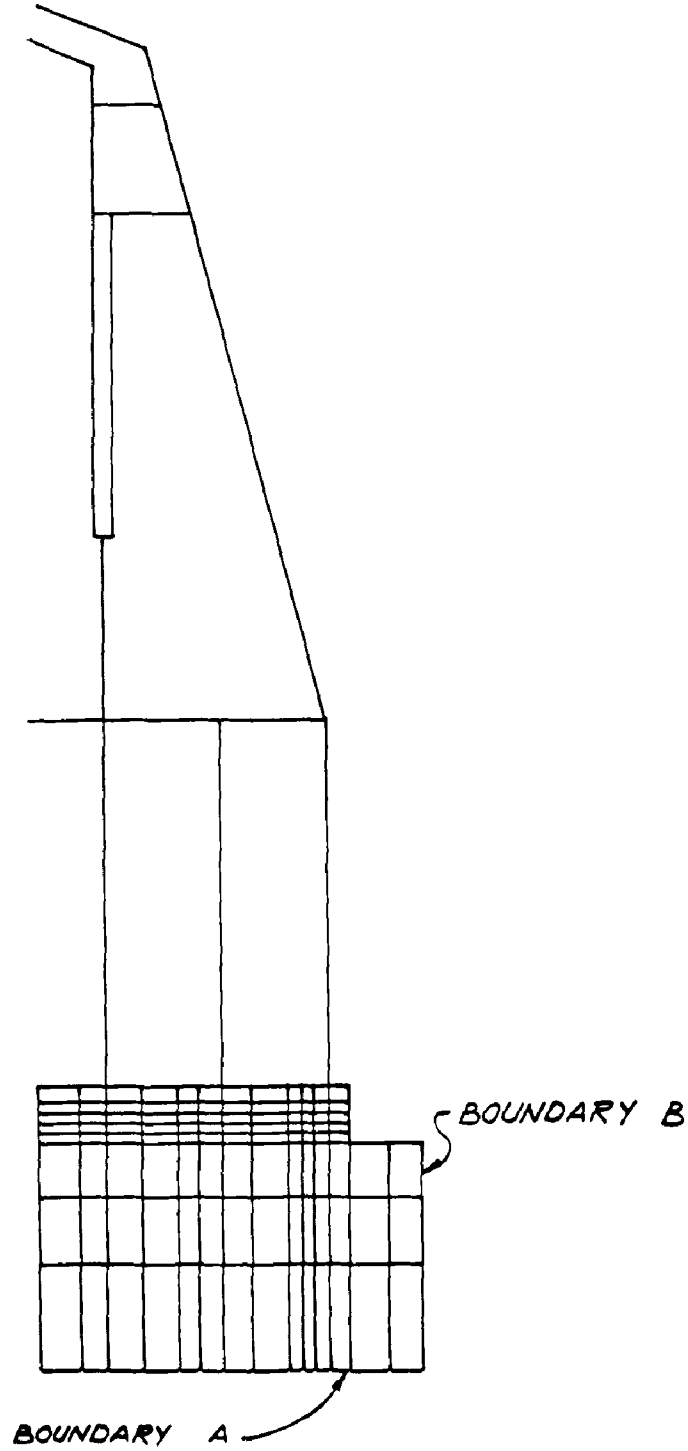
FSAR REV.65

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



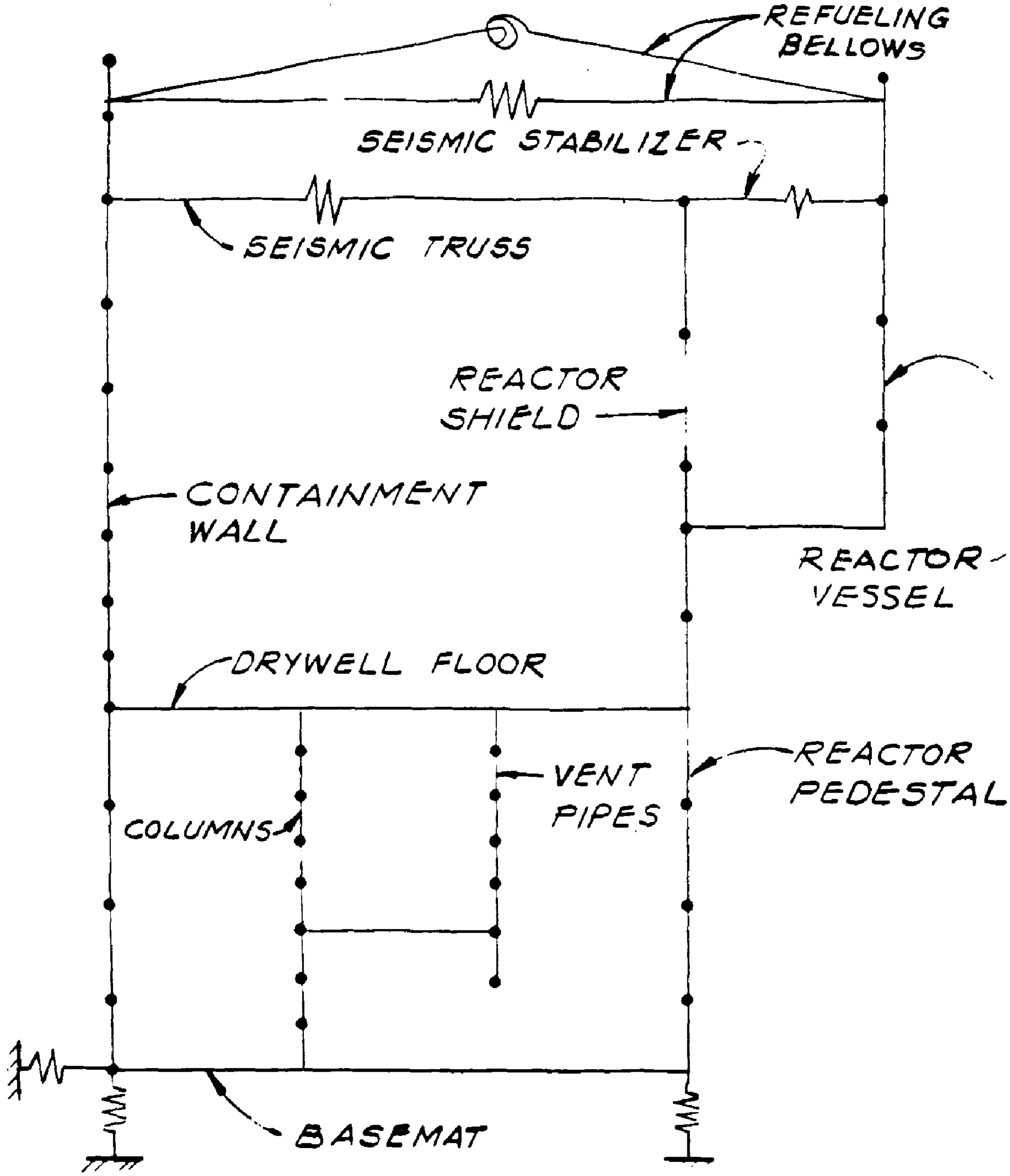
FSAR REV.65

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



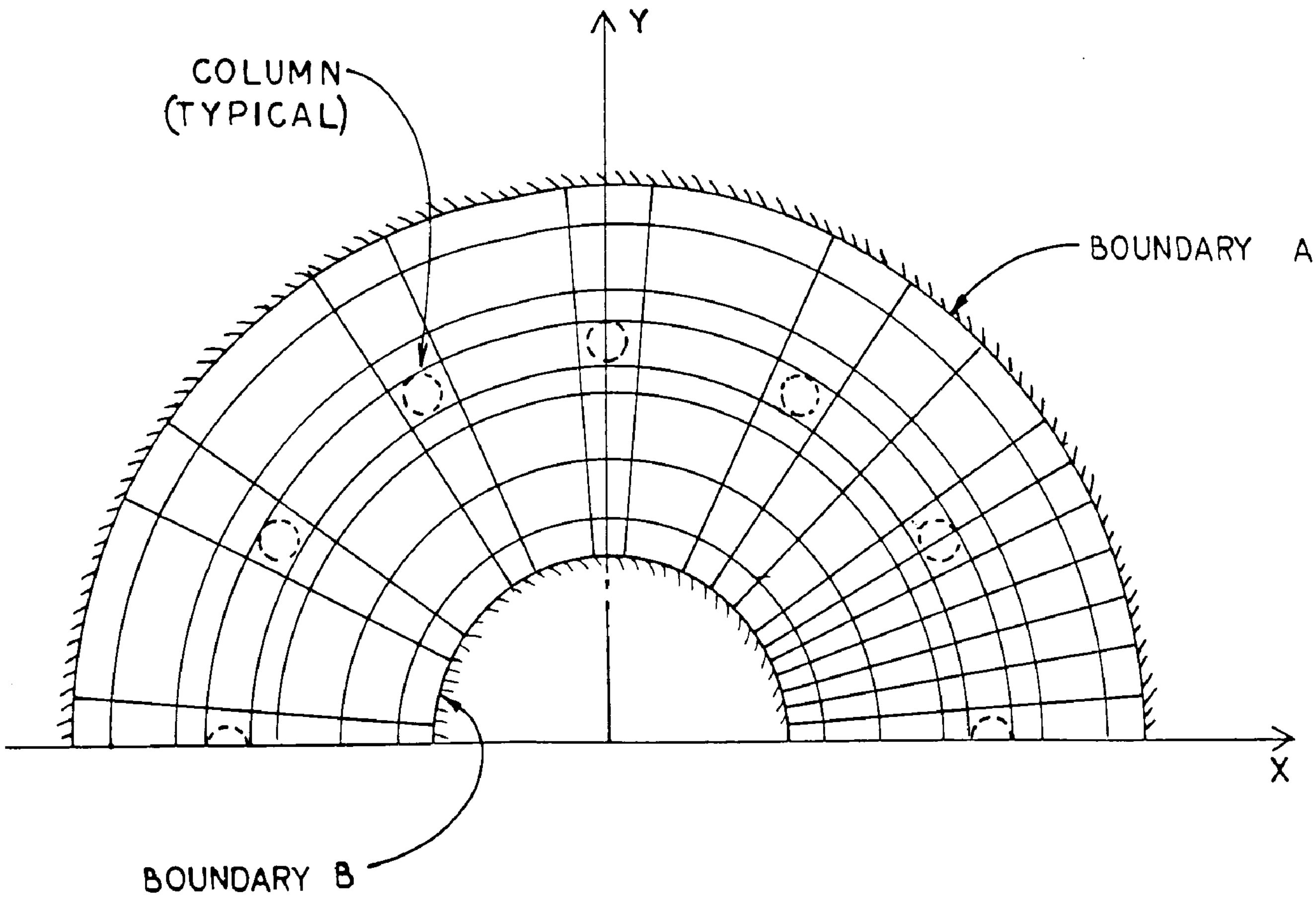
FSAR REV.65

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



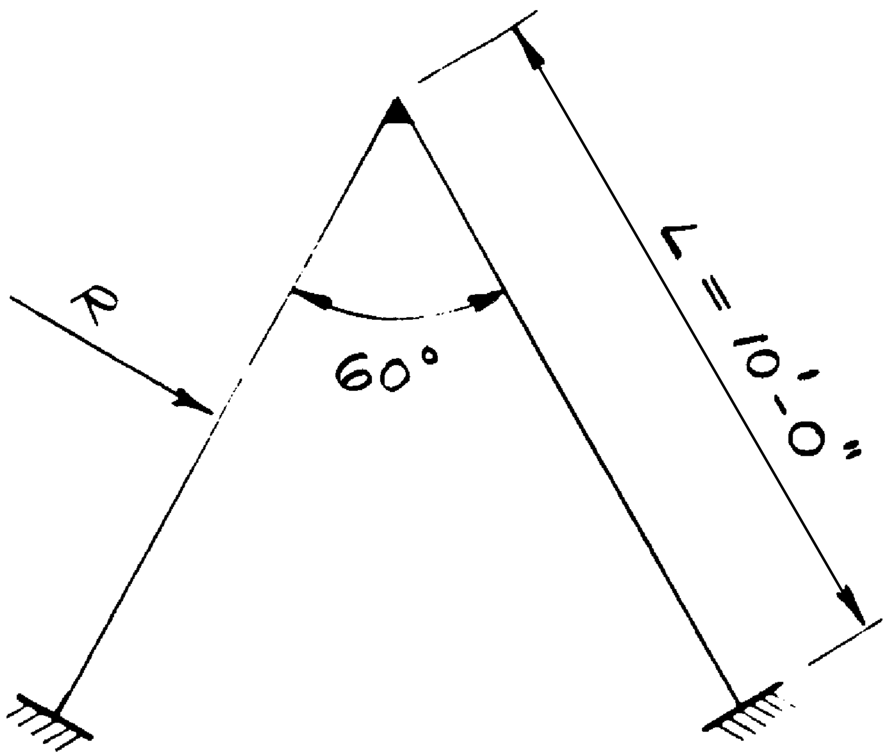
FSAR REV.65

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

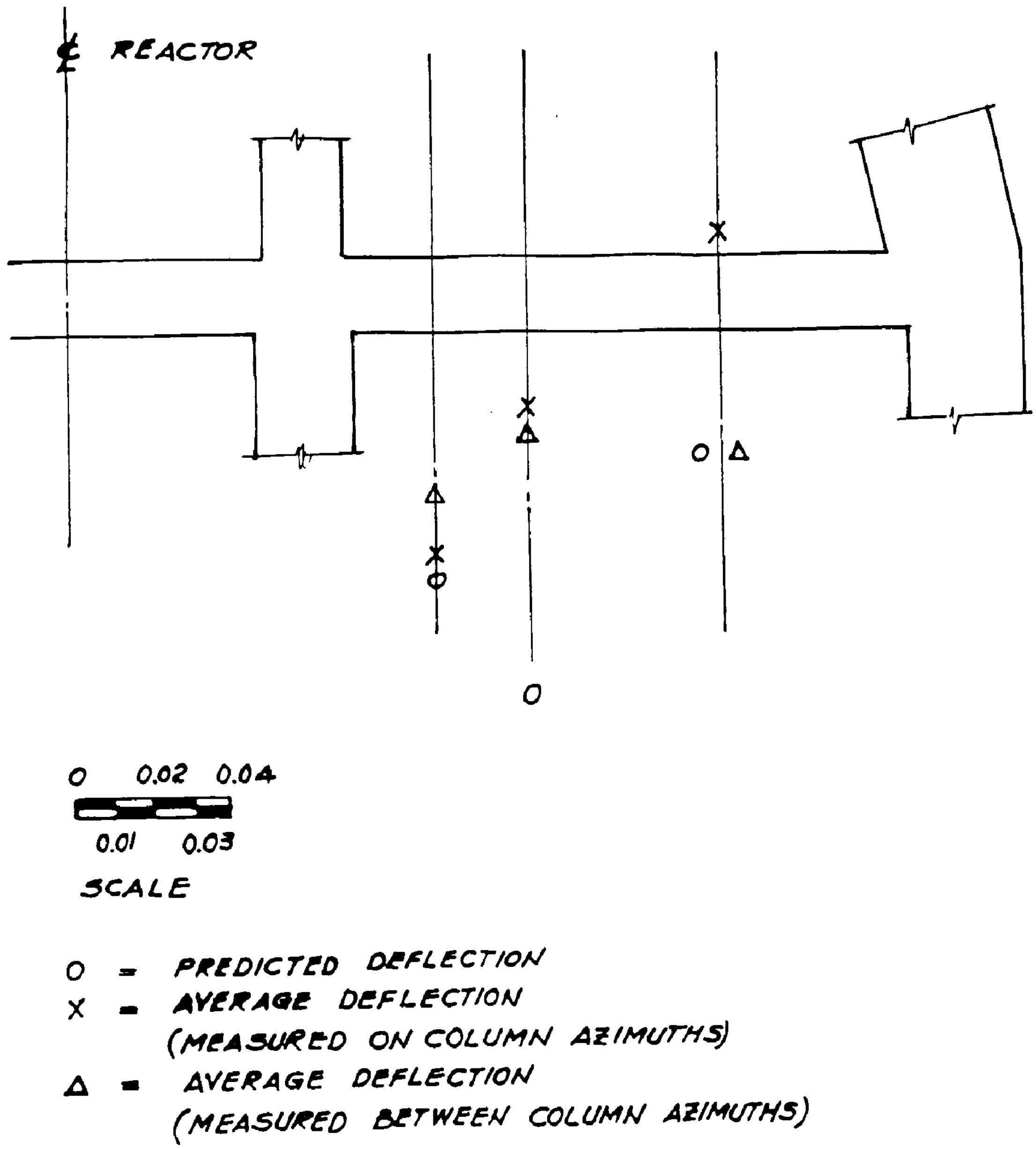
FSAR REV.65

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

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

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

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THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-204, Sh. 1

FSAR REV. 65

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

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
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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

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
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

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

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

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

FSAR REV. 65

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

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THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-260, Sh. 1

FSAR REV. 65

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

FSAR REV. 65

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

FSAR REV. 65

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

FSAR REV. 65

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

FSAR REV. 65

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
UNITS 1 & 2
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

FSAR REV. 65

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

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-272, Sh. 1

FSAR REV. 65

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

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
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

FSAR REV. 65

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

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
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

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 = ν = 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 = ν = 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:

$$M = \Delta T^{\alpha c} E_c b t^2/12 = (100)(6x10^{-6})(3x10^6)(12)(42)^2/12$$
$$= 3,175,000 \text{ in-lbs}$$

$$\sigma c = \Delta T^{\alpha c} E_c/2 = (100)(6x10^{-6})(3x10^6)/2 = 900 \text{ psi (compression)}$$

$$\sigma c = \frac{(t/2-2)}{t/2}\sigma_c = \frac{(21-2)}{21} 900 = 814 \text{ psi (tension)}$$

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:

$$M_T = \frac{f_S A_S \left(d-\frac{X}{3}\right)}{12} = \frac{13970(1)\ (40-2.47)}{12} = 43{,}690\,\frac{inch\text{-}lb}{inch}$$

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;

$$kd = 7.37\ in.$$

The resultant forces are:

$$C = T = \frac{M}{jd} = \frac{3{,}175{,}000}{\left(40 - \frac{7.37}{3}\right)}$$

$$C = T = 84{,}570 \text{ lb.}$$

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) $$f_c = \frac{f_s kd}{n(d - kd)} \quad for \ \frac{M}{N} \geq t/6$$

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

$$@\ center\ \omega = .01695\ \frac{Pa2}{D} = .01695\ \frac{300\ (100)\ 12\ (1-(\ .3)^2)}{(2.9 \times 10^7)\ \ (.5^3)}$$

$$\omega = .00153\ in.\ @\ Node\ 116$$

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

$$@x = 2, y = 0 \quad M\chi = \frac{-P(1+\nu)}{8\pi} \ln \left[\frac{1 - \sin \frac{\pi \chi}{a}}{1 + \sin \frac{\pi \chi}{a}}\right]$$

$$M\chi = \frac{-300(1.3)}{8\pi} \ln \left[\frac{1 - \sin \frac{\pi}{5}}{1 + \sin \frac{\pi}{5}}\right] = (-15.52)\ (-1.348)$$

$M\chi = 20.92$ lb - in. @ Node 113

My : (for $b >> a$)

$$@\chi = 6, y + 0 \quad My = \frac{-P(1 + \nu)}{8\pi} \ln \left[\frac{1 - \sin \frac{\pi \chi}{a}}{1 + \sin \frac{\pi \chi}{a}}\right]$$

$$= \frac{-300(1.3)}{8\pi} \ln \left[\frac{1 - \sin \frac{3\pi}{5}]}{1 + \sin \frac{3\pi}{5}}\right]$$

$$= (-15.52)\ (-3.685)$$

$$My = 57.198 \text{ lb-in @ Node } 117$$

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

$$@\chi = 15, y = 15 \omega = .0582\ \left(\frac{gb^4}{D}\right) = .0582 \left[\frac{2(15)^4(12)(1-(.3)^2)}{(2.9 \times 10^7)\ (.2)^3}\right]$$

$\omega = .277$ 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

$M\chi$:

$@\chi = 15, y = 15$ $\quad M\chi = .0293\ ga^2 = .0293\ (2)(30)^2$
$M\chi = 52.74$ *in-lbs* @ *Node* 11

My:

$@\chi = 15, y = 0$ $\quad My = .319\ gb^2 = .319\ (2)(15)^2$
$My = 143.55$ *in-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 EO781 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 EO781'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:

$$N\phi = (680/2)\ (32.778) = 11,144.5 \text{ lb/in.}$$

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 ($^{\mu}\phi$ = 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

$$p = \frac{(12\ ft)\ (62.5\ lb/ft^3)}{144\ in^2/ft^2} = 5.2083 psi$$

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
Uϕ - displacement component in ϕ direction
Bϕ - rotation of reference surface in ϕ direction
Q - transverse shear in ϕ direction
Nϕ - membrane force in ϕ direction
Mϕ - 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: Axially Constrained Hollow Cylinder with a Distributed Pressure Loading

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	ν	= 0.25
Young's Modulus	E	= 4.32×10^5 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: Axially Constrained Hollow Cylinder with a Linear Temperature Gradient

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 :

$$\delta_\theta = \frac{\alpha E}{1-\nu} \frac{1}{r^2} \left[\left(\frac{r^2 + a^2}{b^2 - a^2} \right) \int_a^b Trdr + \int_a^r Tr'\,dr' - TR^2 \right]$$

axial stress, δz :

$$\delta_z = \frac{\alpha E}{1-\nu} \frac{1}{r^2} \left[\left(\frac{r^2 + a^2}{b^2 - a^2} \right) \int_a^b Trdr - T \right]$$

radial stress, δ_r :

$$\delta_r = \frac{\alpha E}{1-\nu} \frac{1}{r^2} \left[\left(\frac{r^2 - a^2}{b^2 - a^2} \right) \int_b^a Trdr - \int_a^r Tr'dr' \right]$$

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 M_x @ Node 121	 52.74 143.55	 50.92 142.28

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

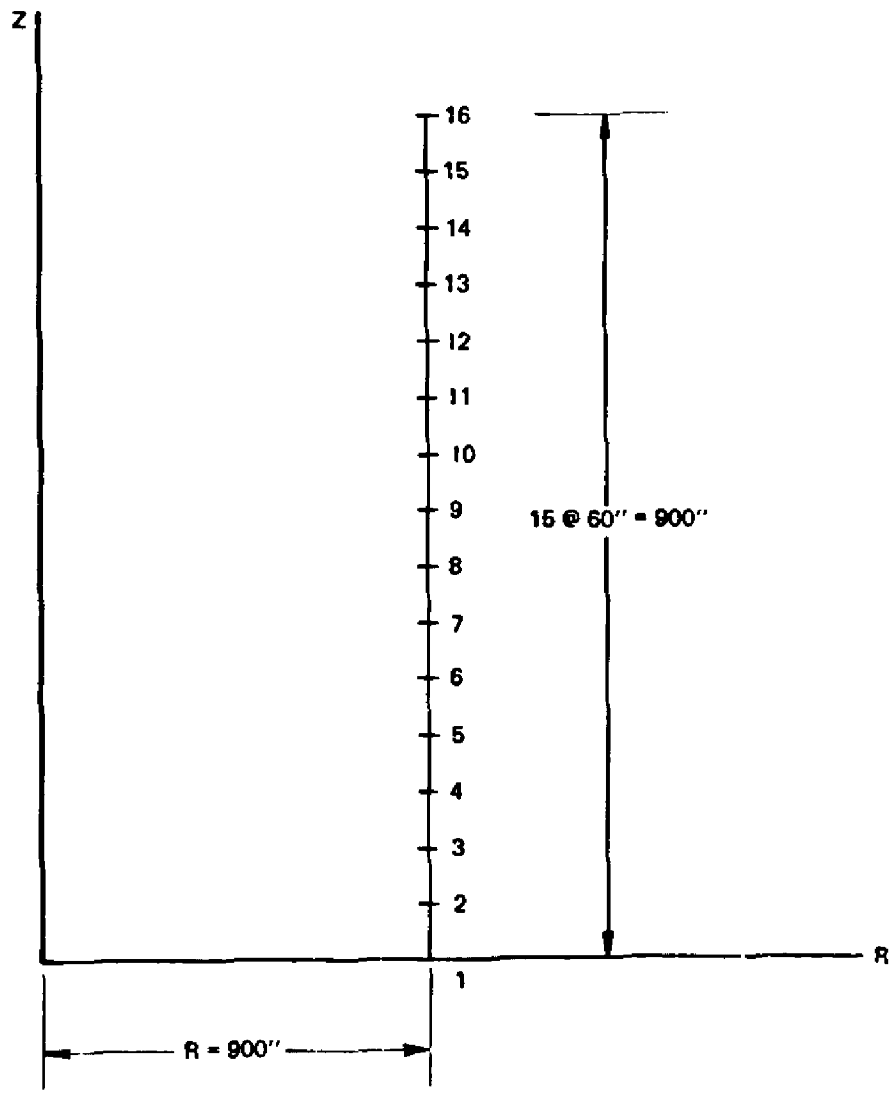
Comparison of Stress Results

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	-0.9
9	68.19	60.26	60.27	60.28	60.28	-0.59	-0.59
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



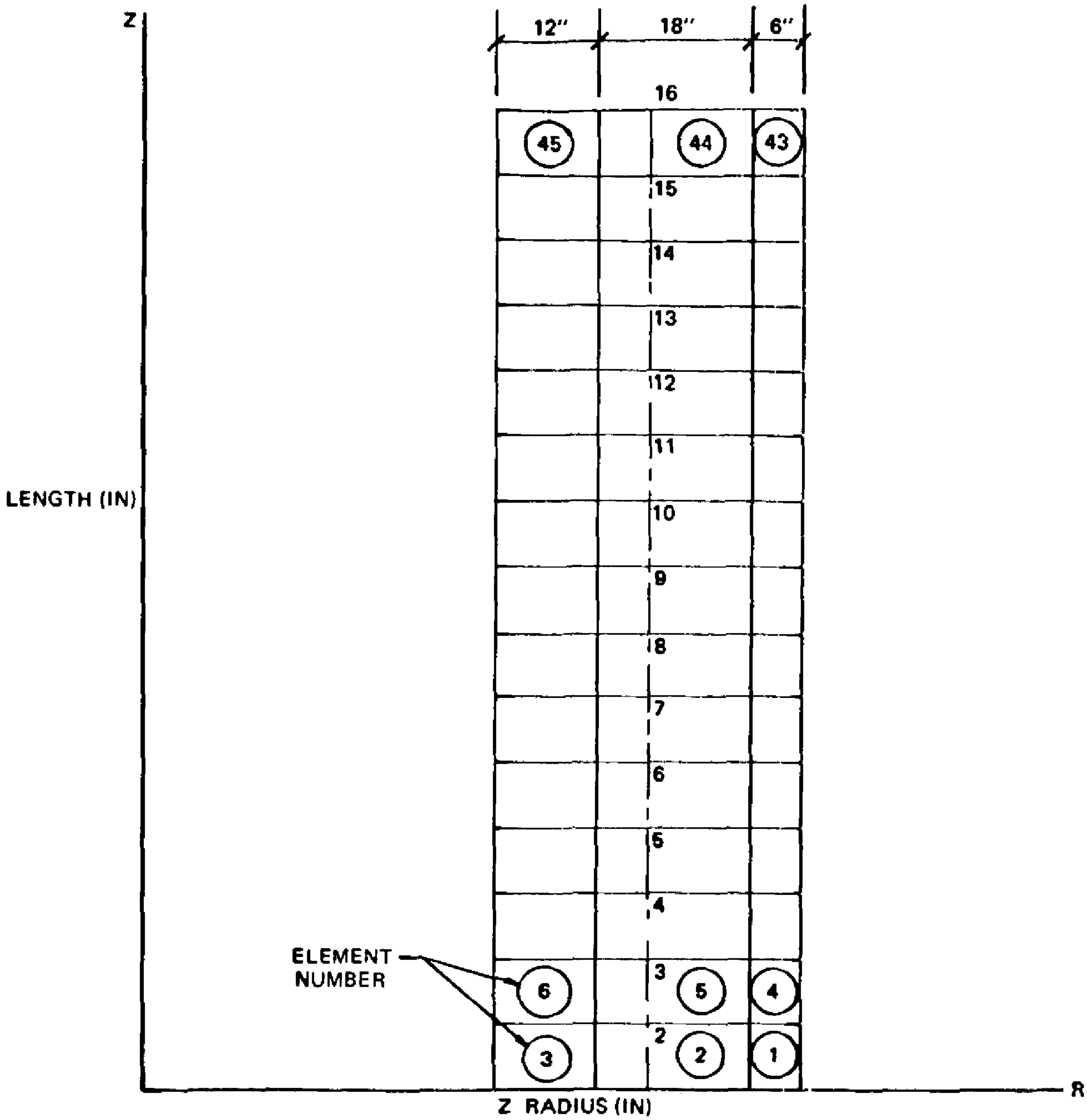
FSAR REV.65

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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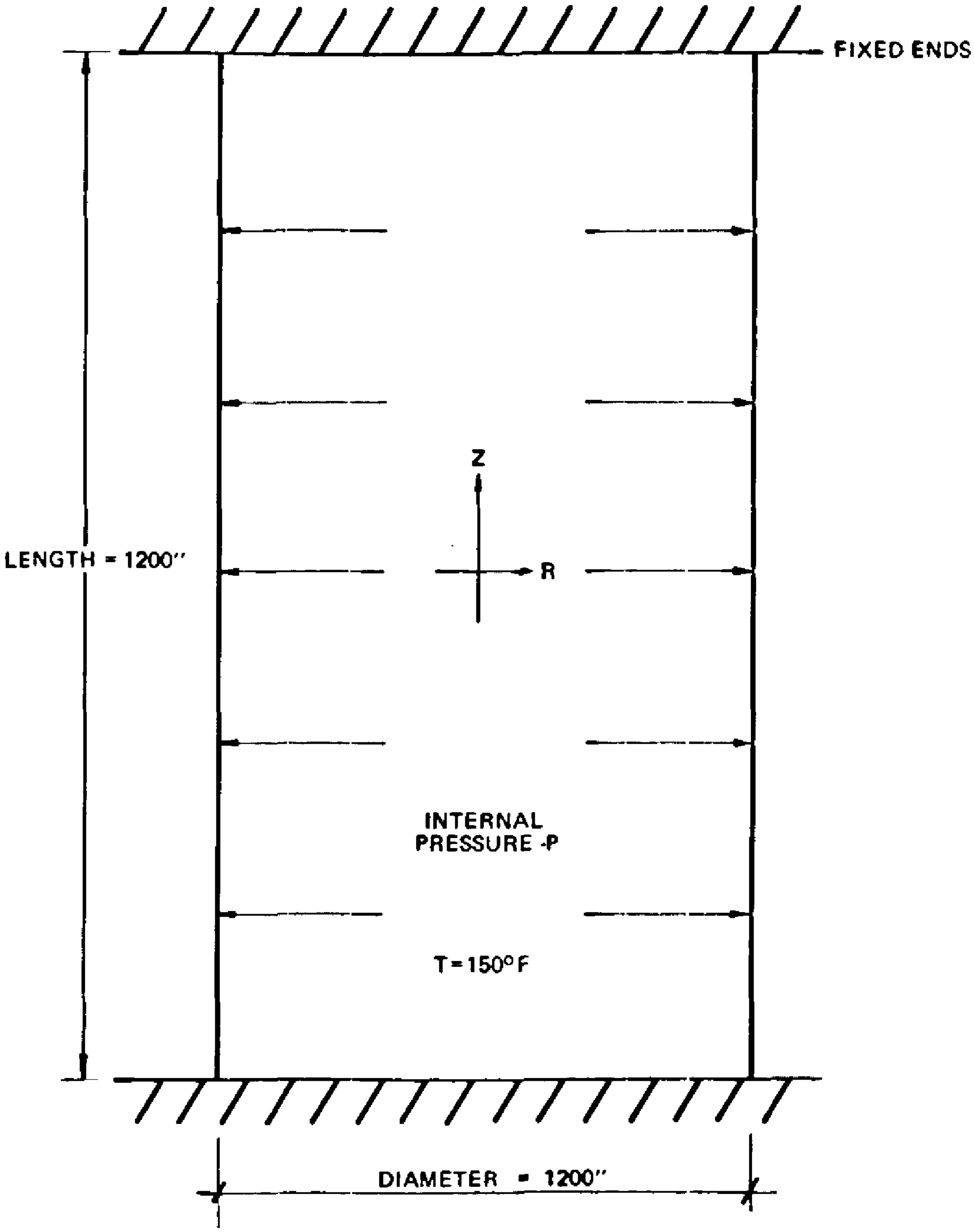
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

LAYERED CYLINDER

FIGURE 3.8A-2, Rev. 47

Auto-Cad Figure Fsar 3_8A_2.dwg



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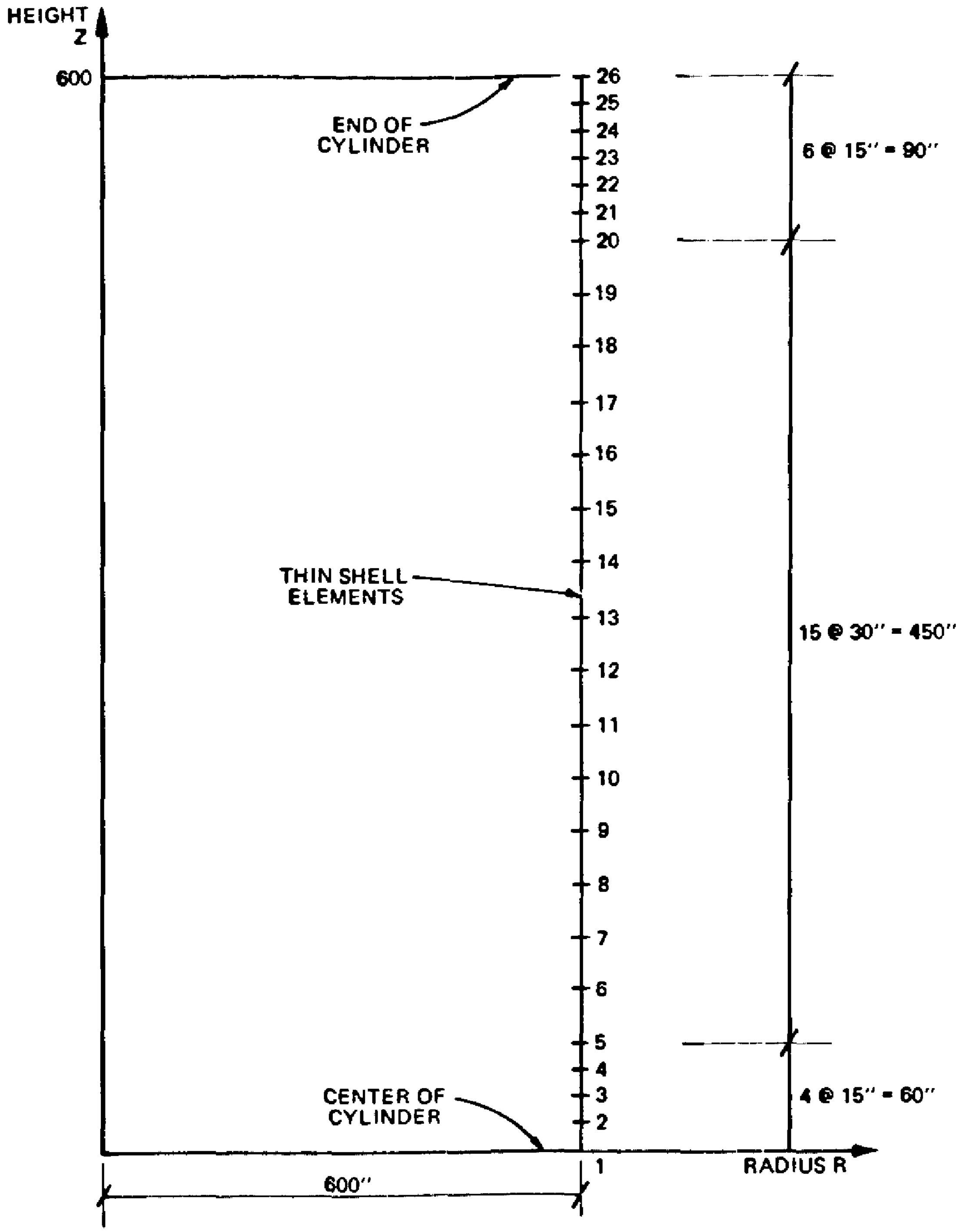
GENERAL LAYOUT OF CYLINDER

FIGURE 3.8A-3, Rev. 47

Auto-Cad Figure Fsar 3_8A_3.dwg



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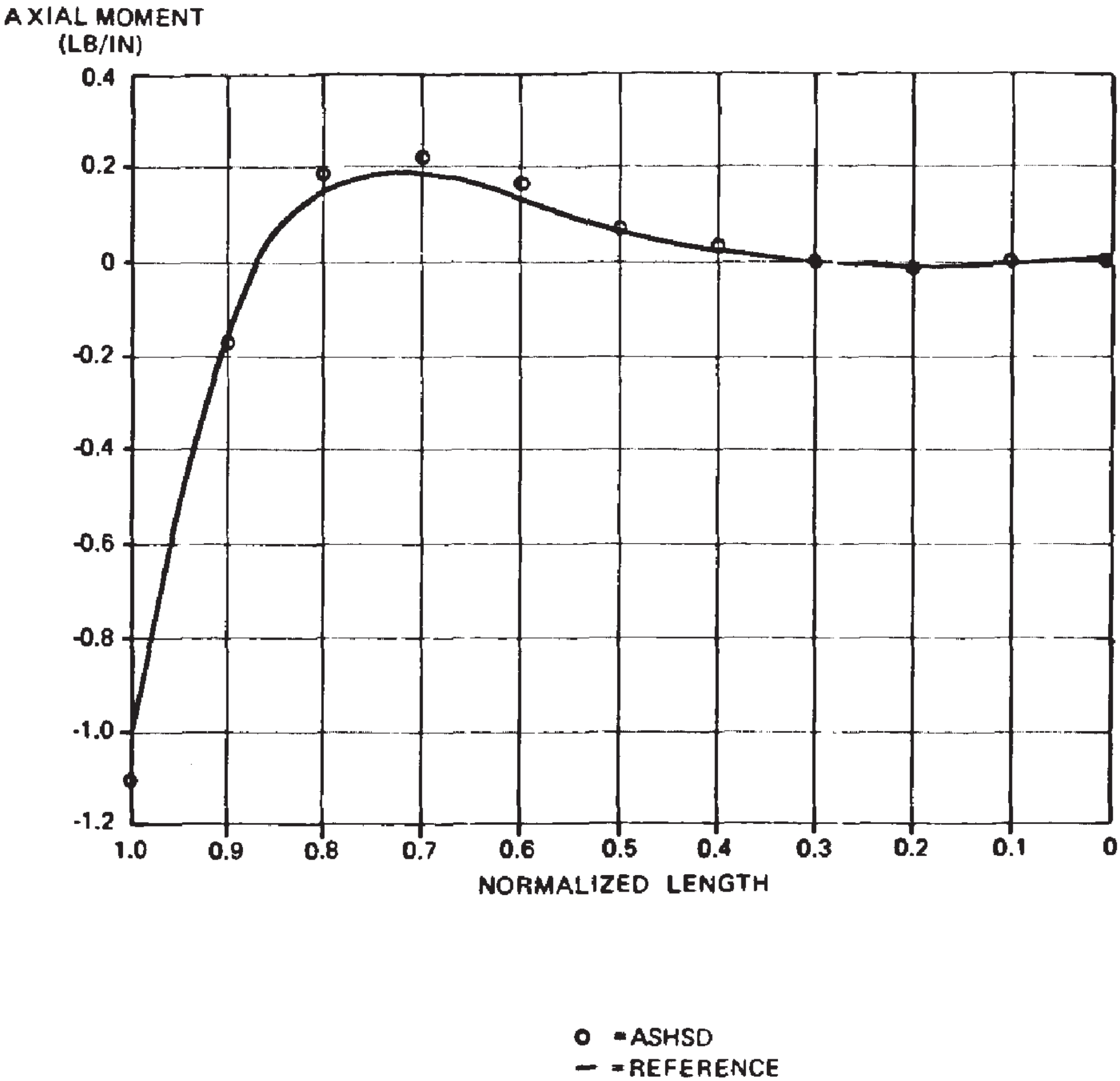
FINAL SAFETY ANALYSIS REPORT

FINITE ELEMENT MODEL

FIGURE 3.8A-4, Rev. 47

Auto-Cad Figure Fsar 3_8A_4.dwg



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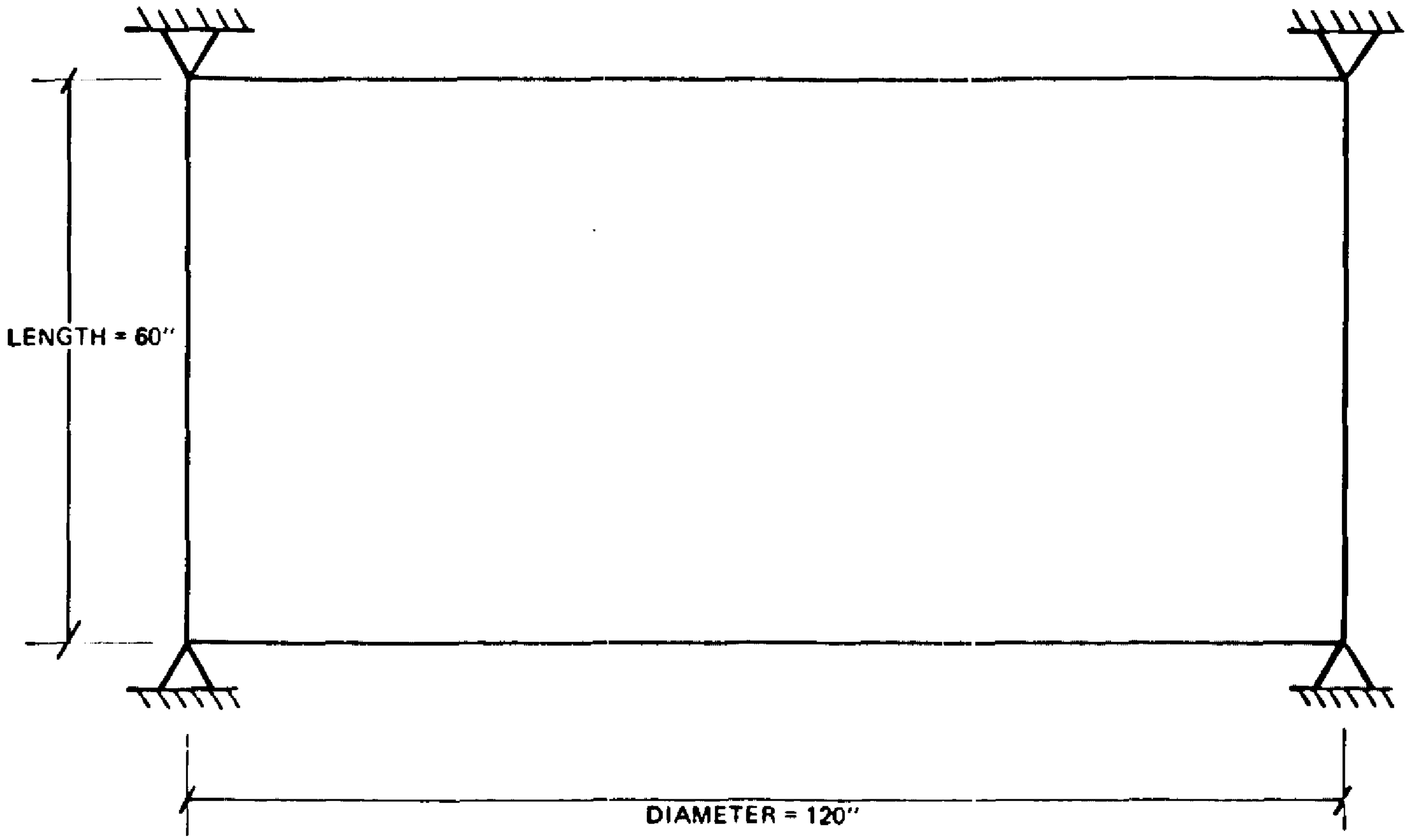
FINAL SAFETY ANALYSIS REPORT

AXIAL MOMENT

FIGURE 3.8A-5, Rev. 47

Auto-Cad Figure Fsar 3_8A_5.dwg



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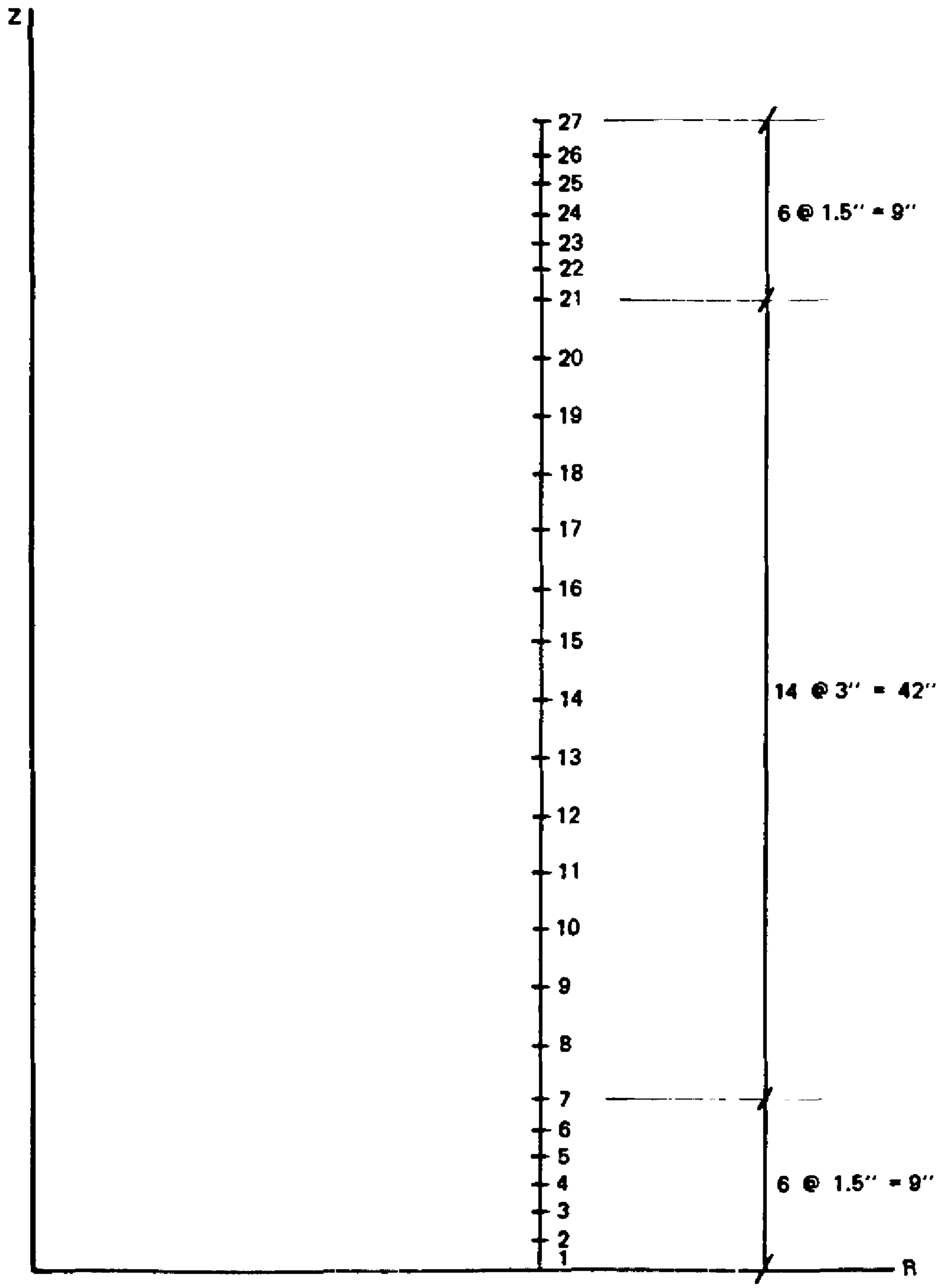
CYLINDER WITH HINGE ENDS

FIGURE 3.8A-6, Rev. 47

Auto-Cad Figure Fsar 3_8A_6.dwg



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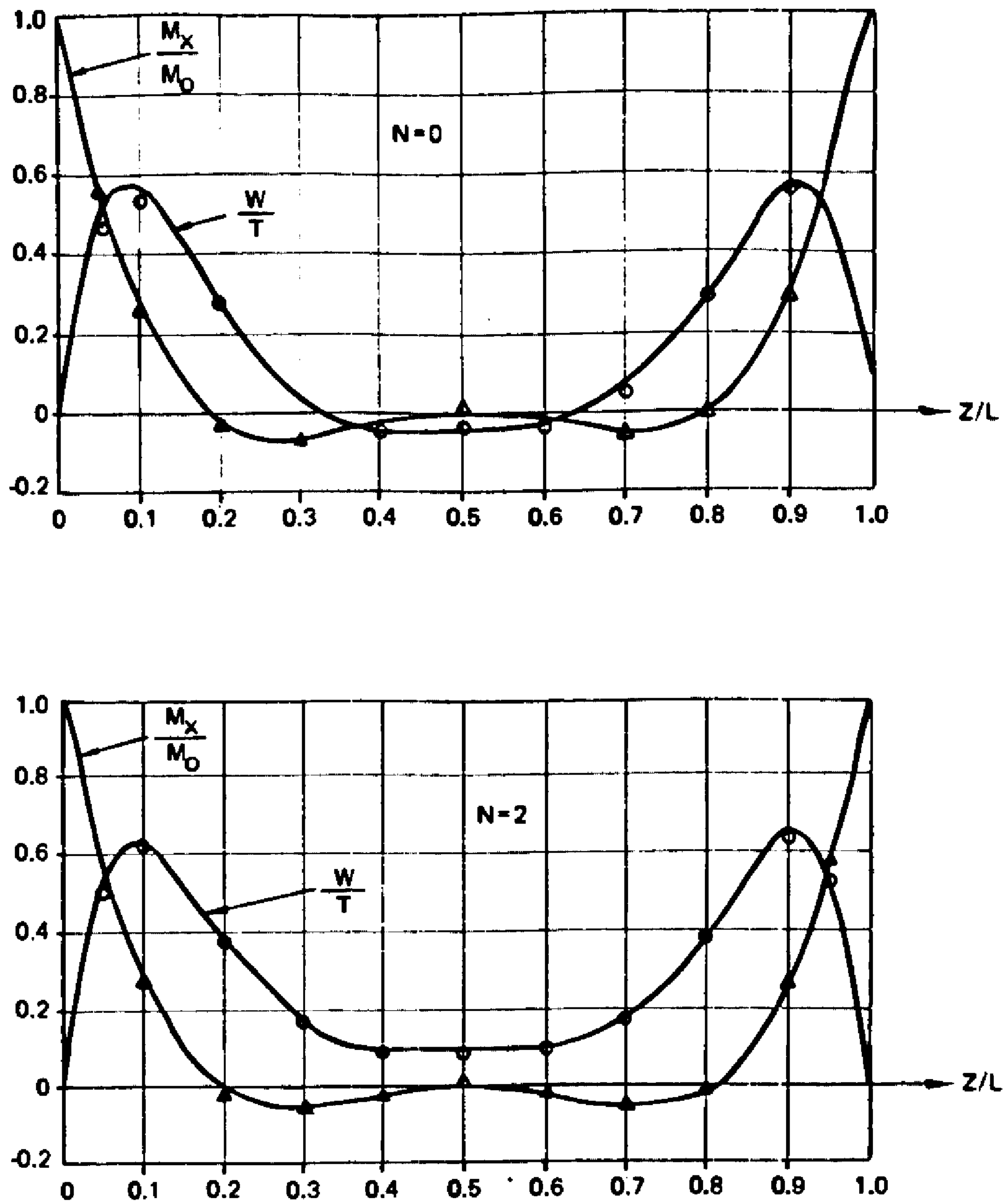
FINITE ELEMENT MODEL

FIGURE 3.8A-7, Rev. 47

Auto-Cad Figure Fsar 3_8A_7.dwg



o ASHSD (RADIAL DISPLACEMENT W)

Δ ASHSD (LONGITUDINAL MOMENT M_x)

— REFERENCE 3.8A-3

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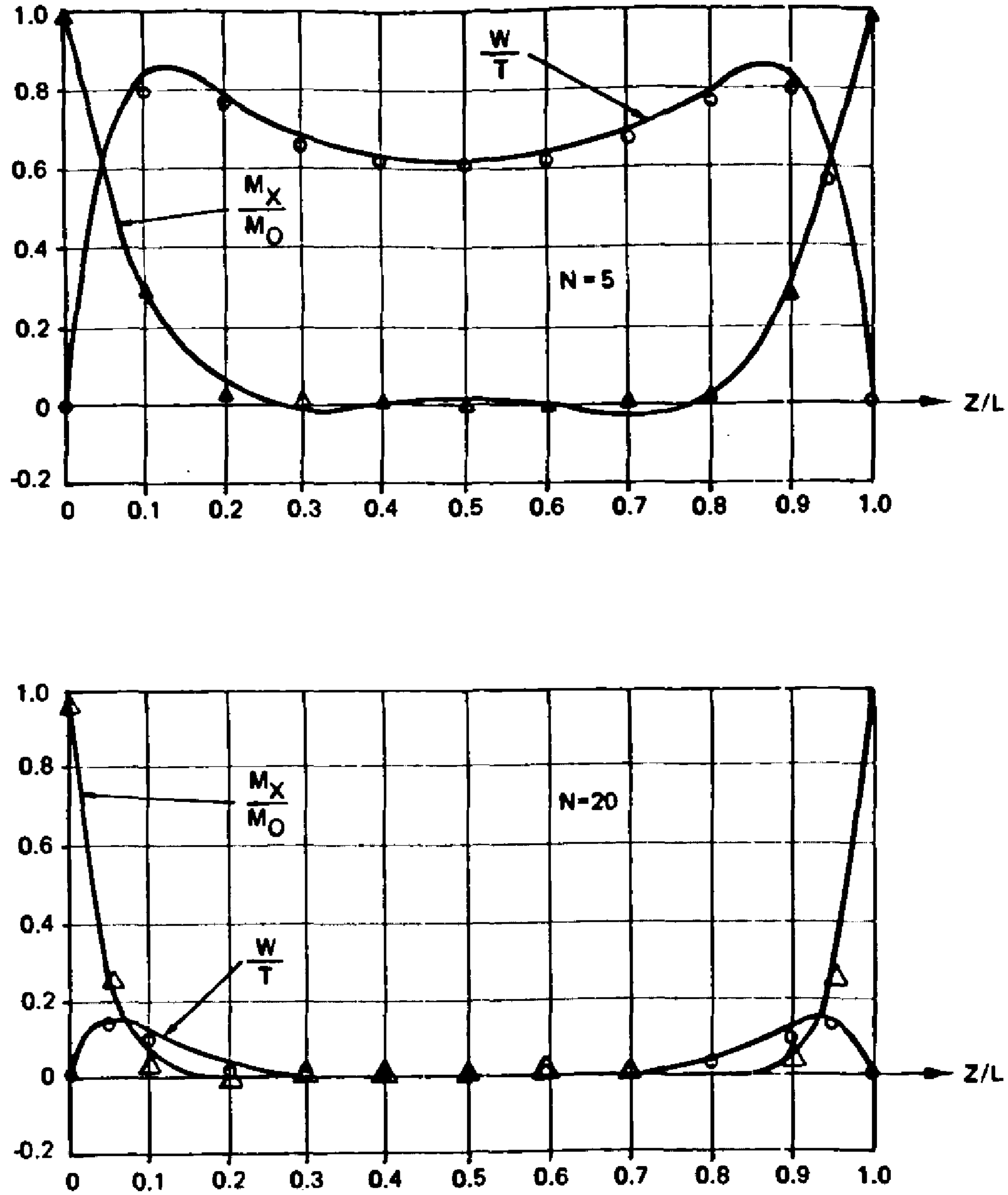
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

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



○ ASHSD (RADIAL DISPLACEMENT W)

Δ ASHSD (LONGITUDINAL MOMENT M_X)

— REFERENCE 3.8A-3

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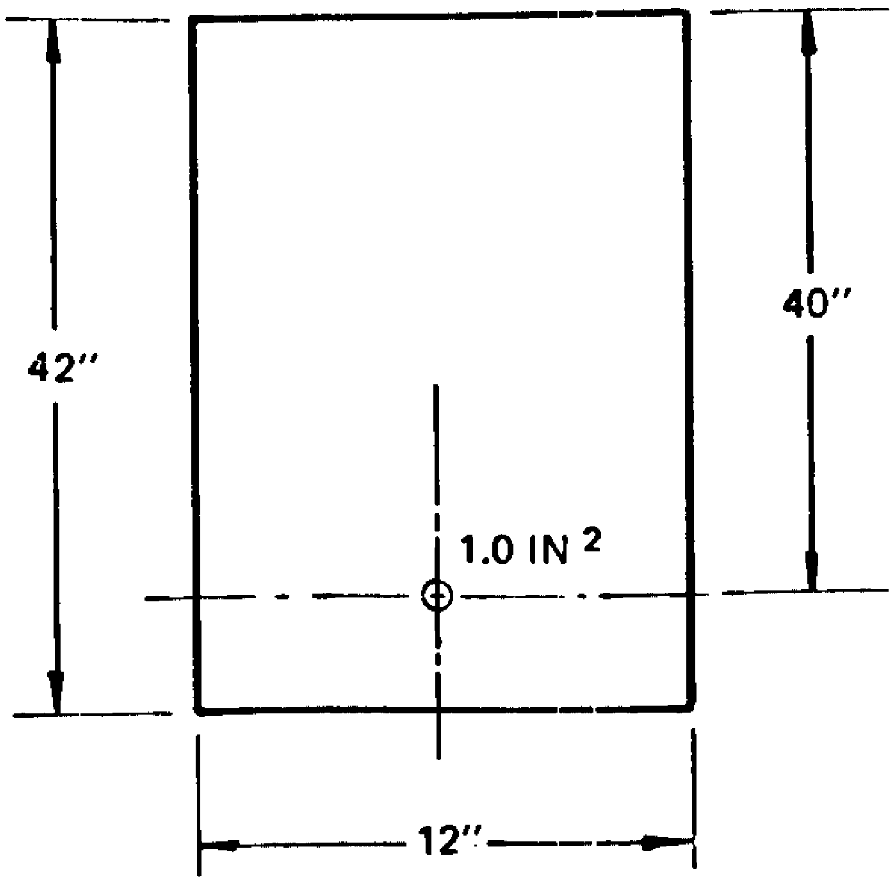
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

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



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SUSQUEHANNA STEAM ELECTRIC STATION
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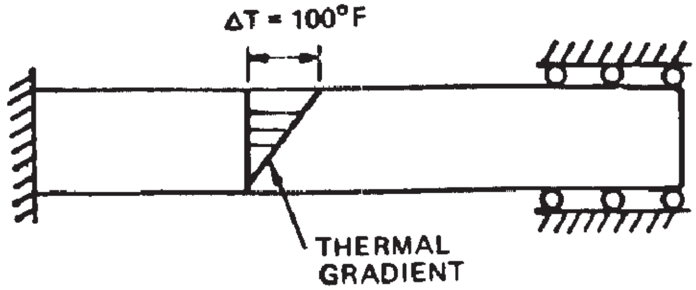
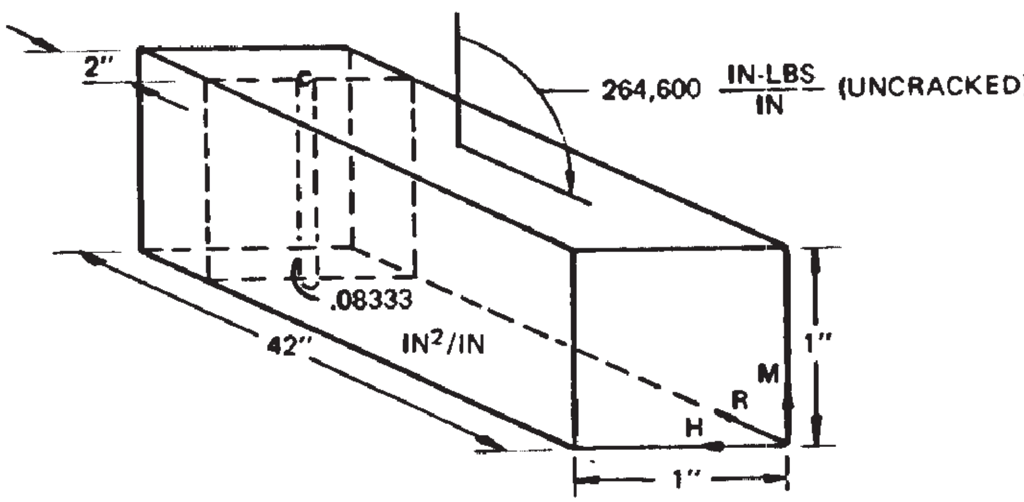
REINFORCED CONCRETE BEAM

FIGURE 3.8A-9, Rev. 47

Auto-Cad Figure Fsar 3_8A_9.dwg



REINFORCED CONCRETE BEAM



CECAP CONCRETE ELEMENT MODEL

FSAR REV.65

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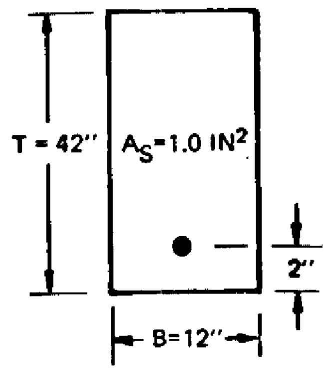
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

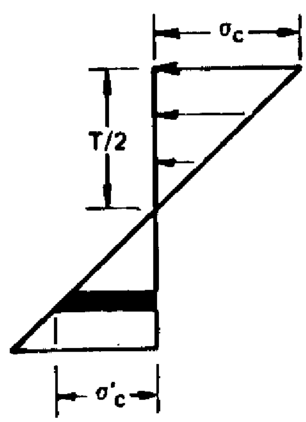
REINFORCED CONCRETE BEAM
AND
CECAP CONCRETE ELEMENT MODEL

FIGURE 3.8A-10, Rev. 47

Auto-Cad Figure Fsar 3_8A_10.dwg



BEAM CROSS - SECTION



INITIAL THERMAL STRESS DISTRIBUTION

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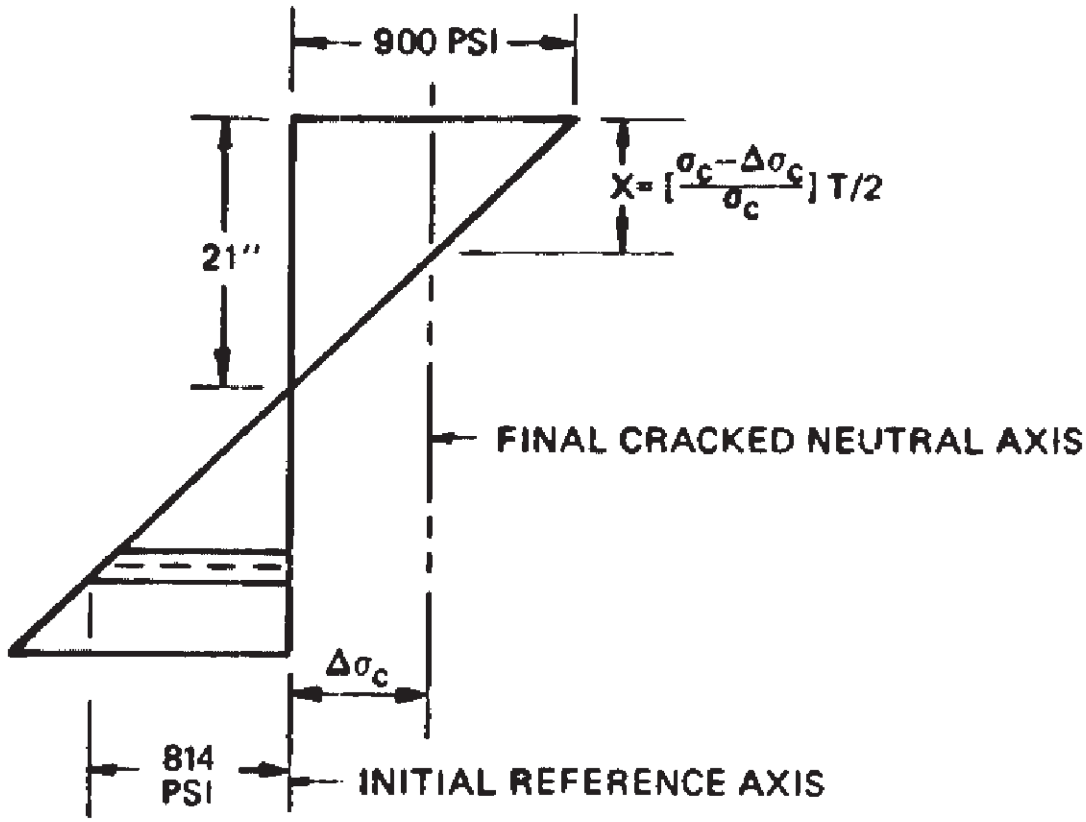
FINAL SAFETY ANALYSIS REPORT

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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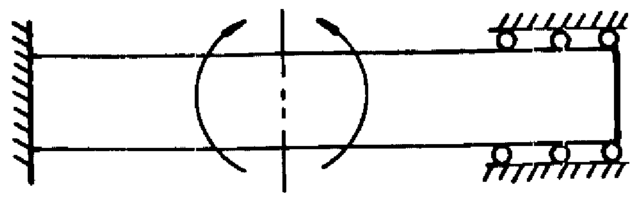
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

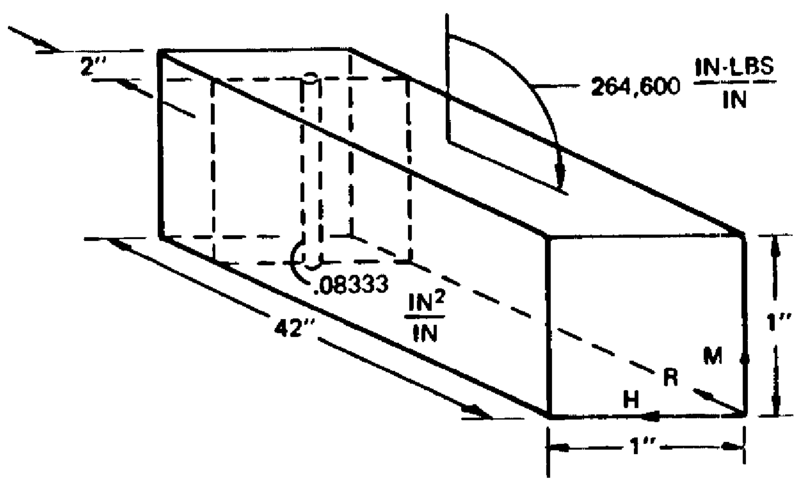
FINAL THERMAL STRESS
DISTRIBUTION

FIGURE 3.8A-12, Rev. 47

Auto-Cad Figure Fsar 3_8A_12.dwg



REINFORCED CONCRETE BEAM



CECAP CONCRETE ELEMENT MODEL

FSAR REV.65

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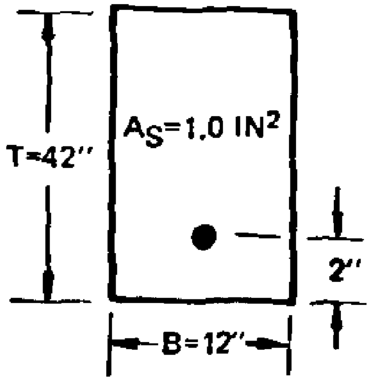
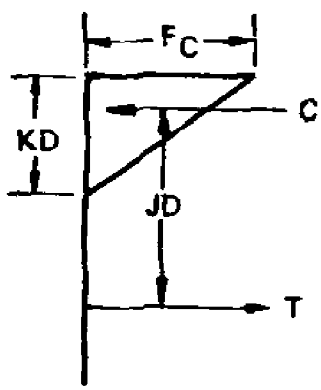
REINFORCED CONCRETE BEAM
AND
CECAP CONCRETE ELEMENT MODEL

FIGURE 3.8A-13, Rev. 47

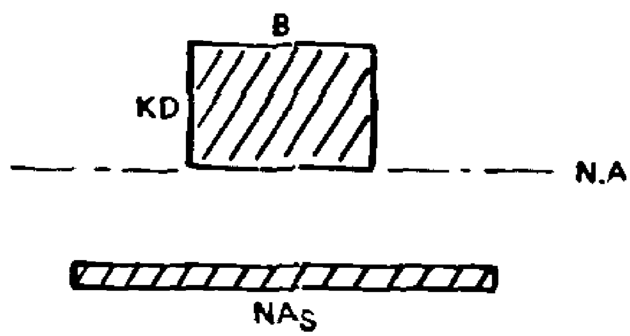
Auto-Cad Figure Fsar 3_8A_13.dwg



BEAM CROSS - SECTION



STRESS BLOCK



TRANSFORMED SECTION

FSAR REV.65

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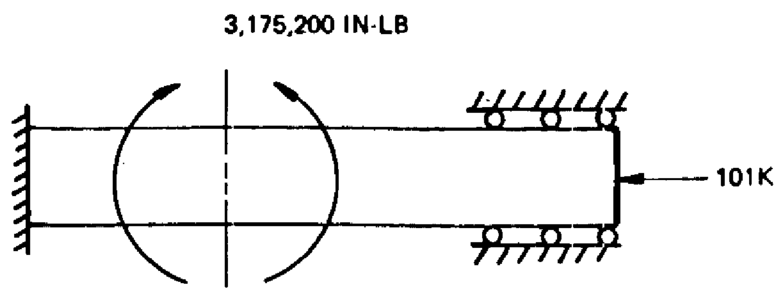
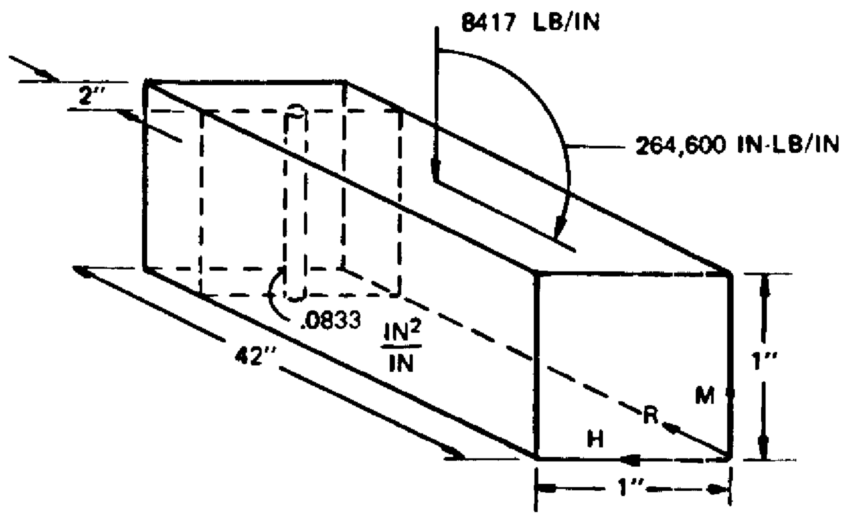
REINFORCED CONCRETE BEAM
AND
CECAP CONCRETE ELEMENT MODEL

FIGURE 3.8A-14, Rev. 47

Auto-Cad Figure Fsar 3_8A_14.dwg



REINFORCED CONCRETE BEAM



CECAP CONCRETE ELEMENT MODEL

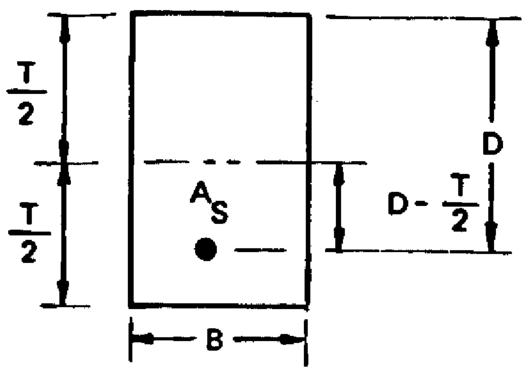
FSAR REV.65

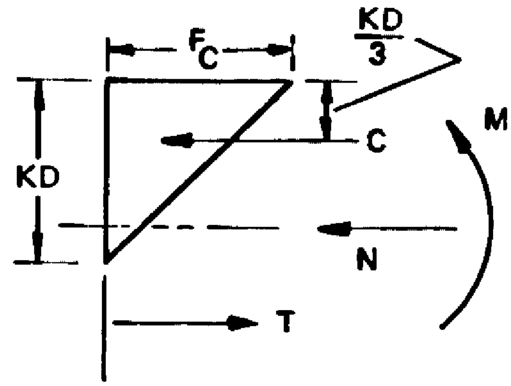
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

REINFORCED CONCRETE BEAM
AND CECAP MODEL

FIGURE 3.8A-15, Rev. 47

Auto-Cad Figure Fsar 3_8A-15.dwg





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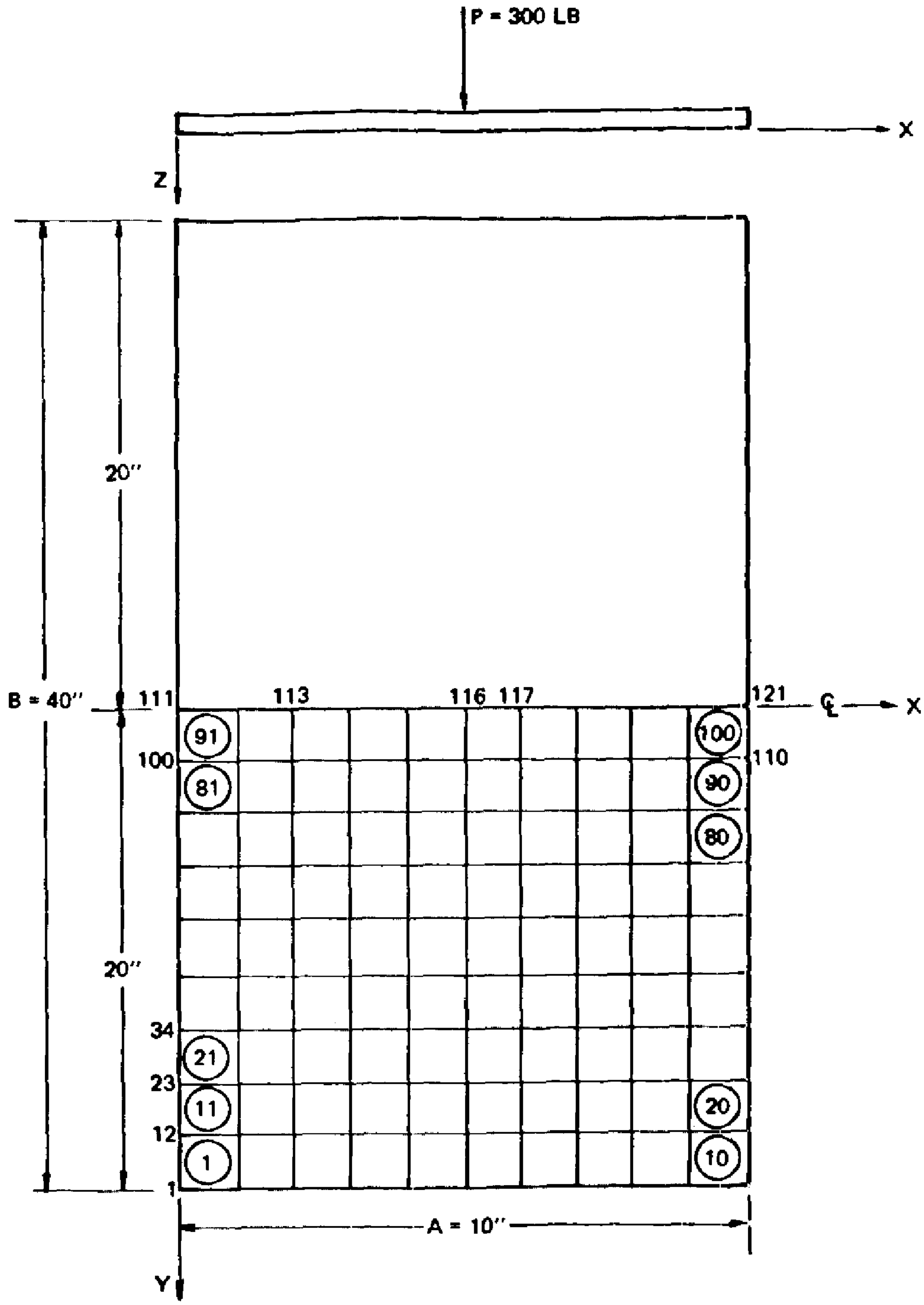
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

BEAM CROSS-SECTION
AND
STRESS BLOCK

FIGURE 3.8A-16, Rev. 47

Auto-Cad Figure Fsar 3_8A_16.dwg



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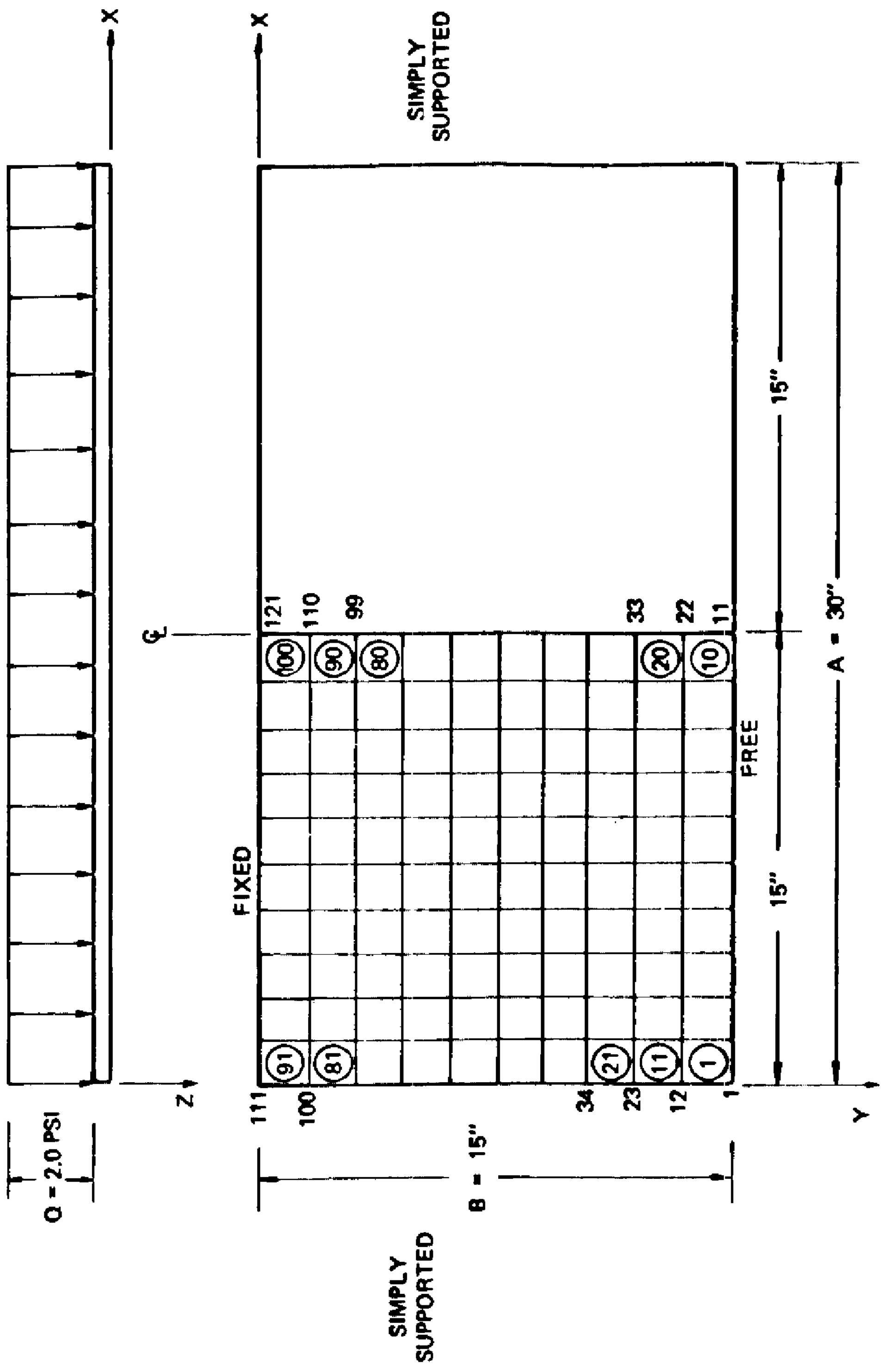
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

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



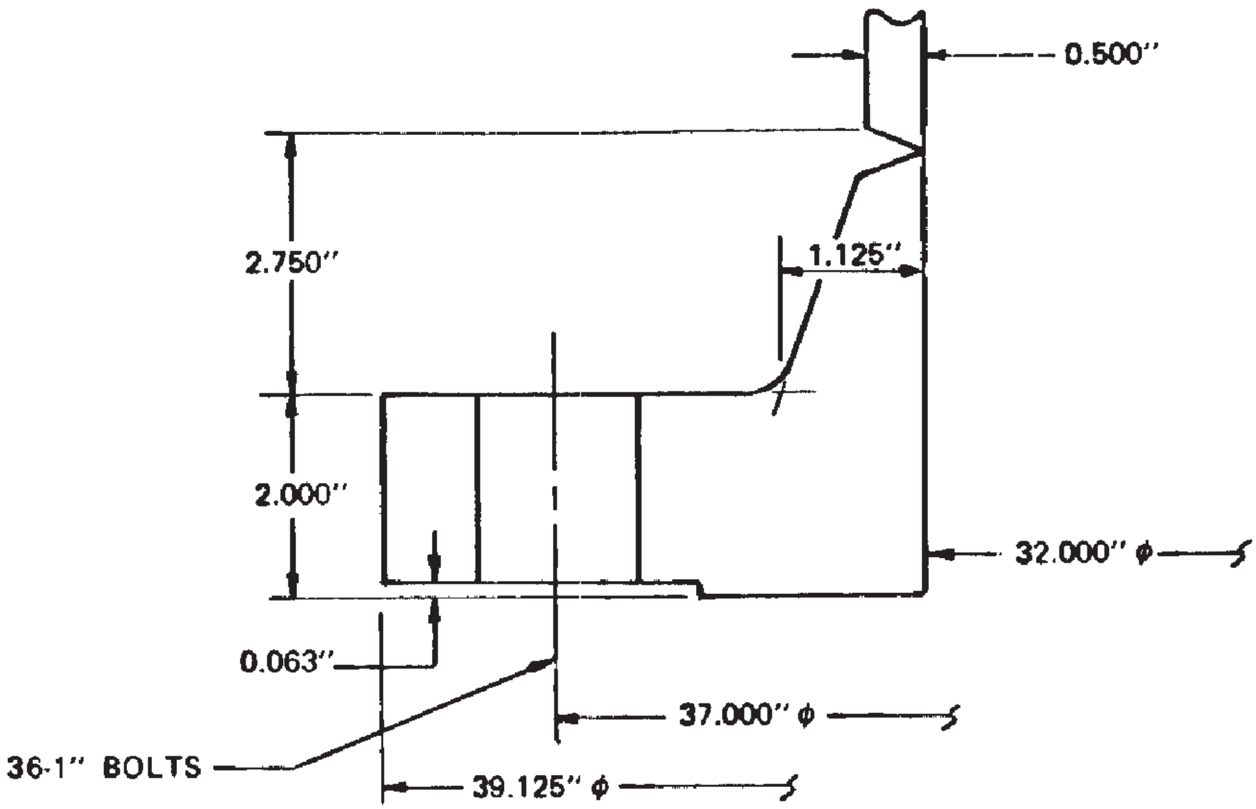
FSAR REV.65

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



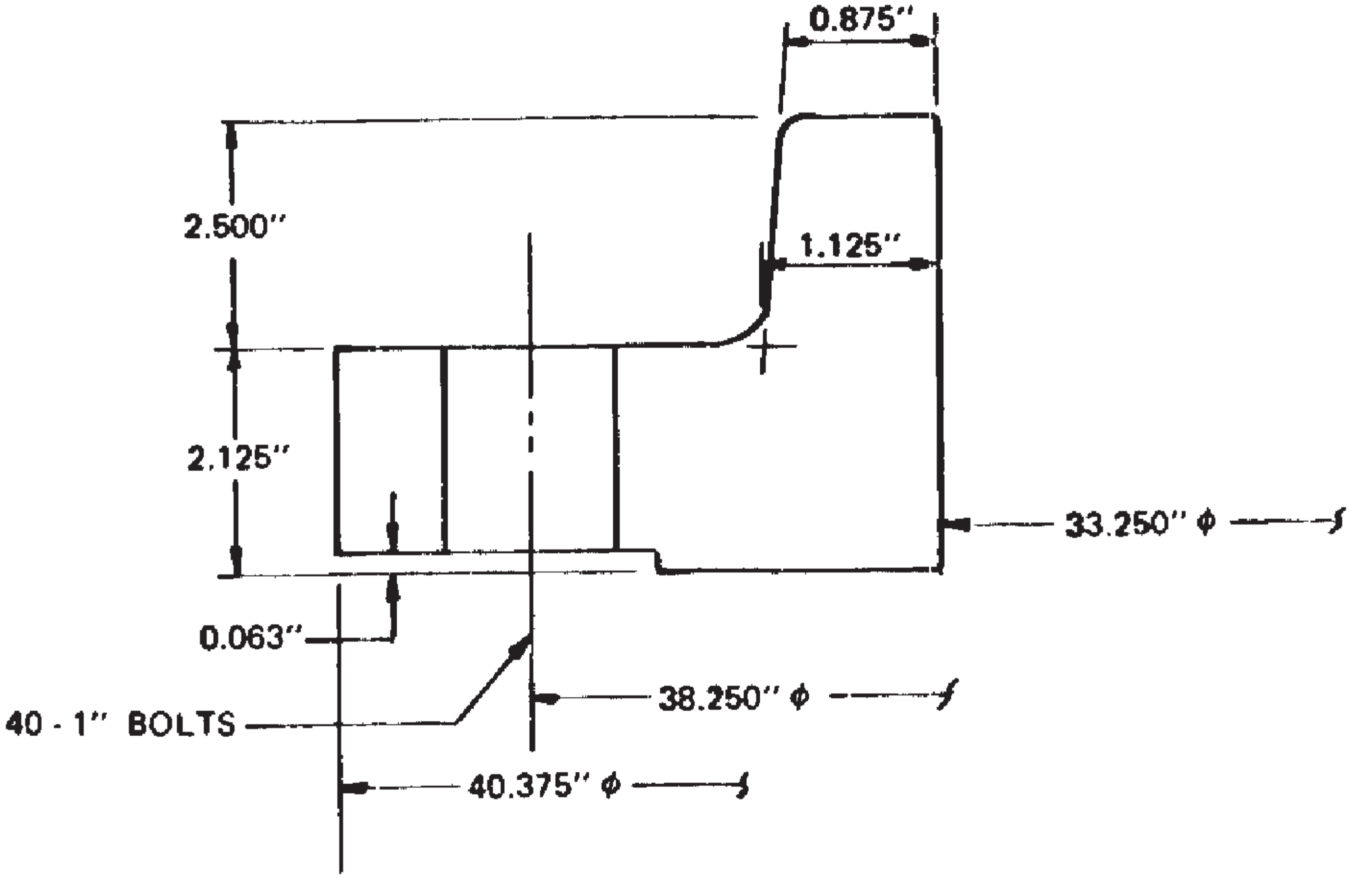
FSAR REV.65

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



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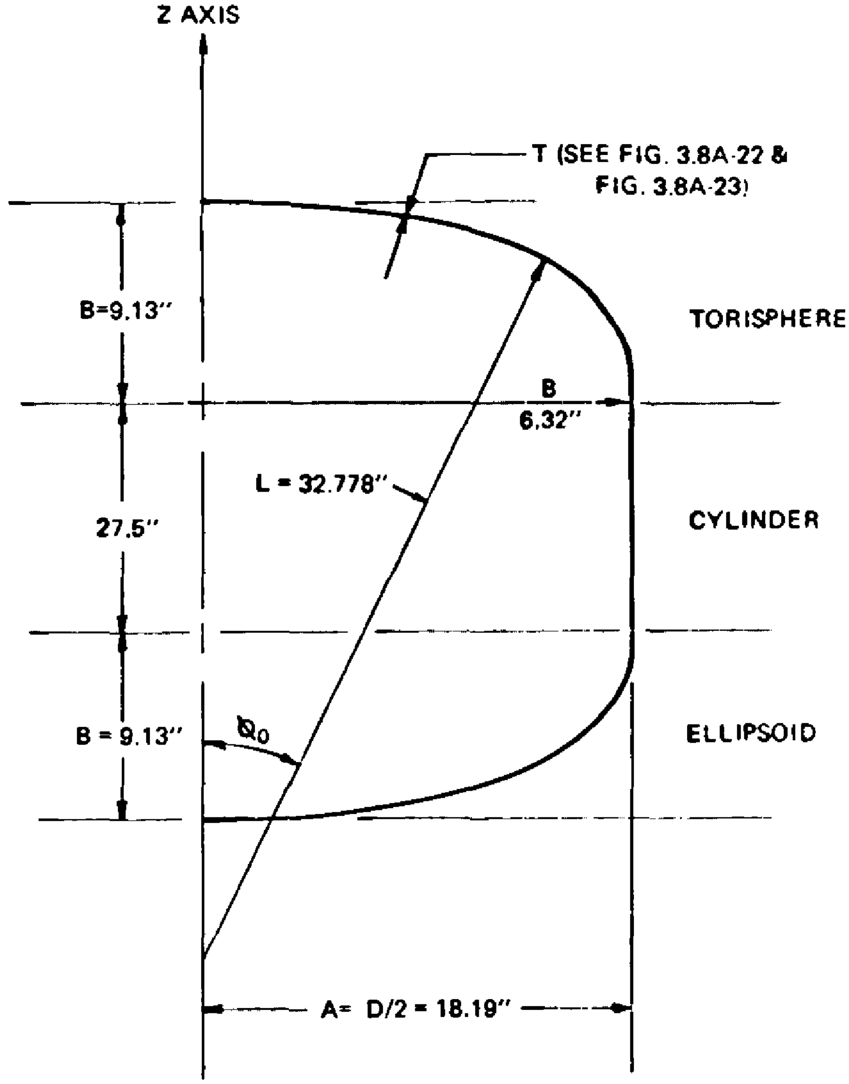
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

SLIP-ON FLANGE DETAIL

FIGURE 3.8A-20, Rev. 47

Auto-Cad Figure Fsar 3_8A_20.dwg



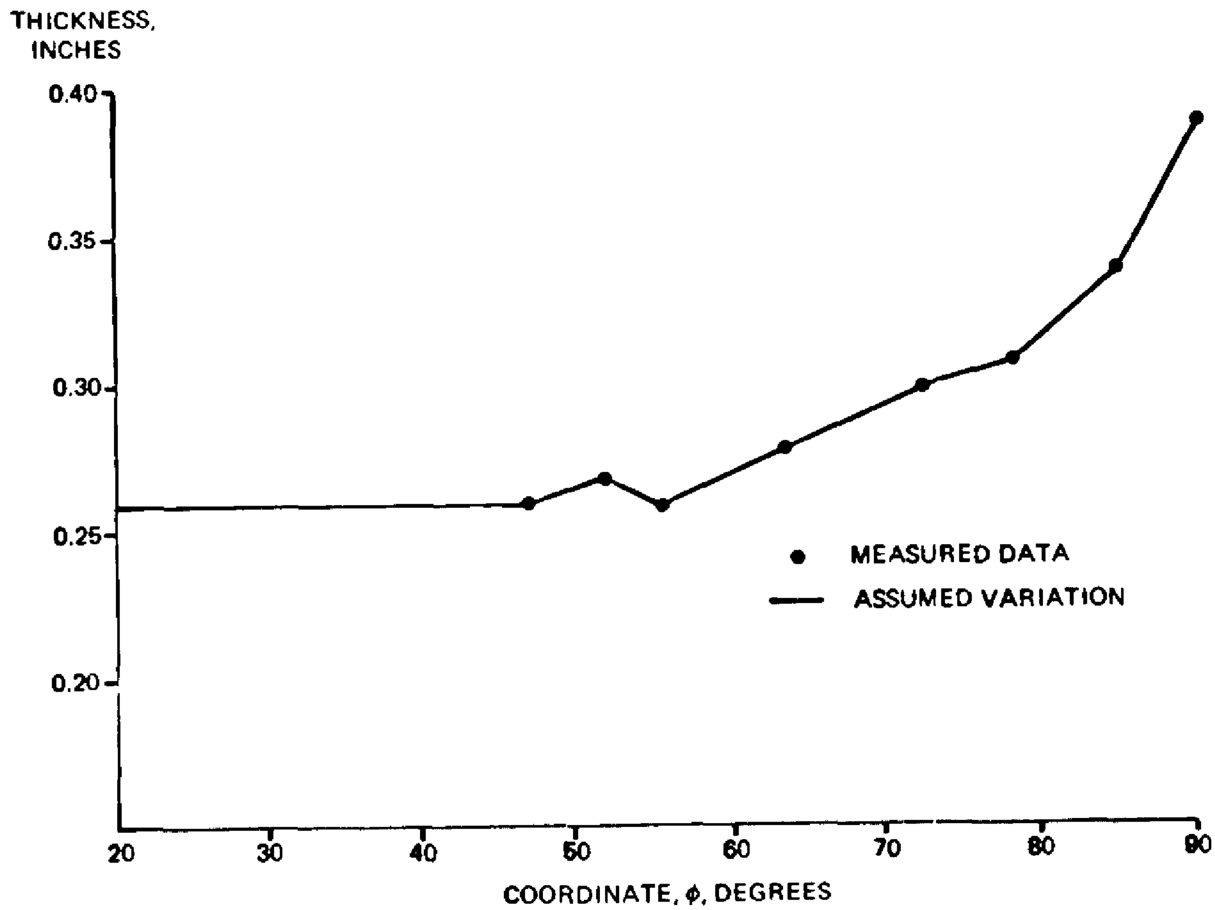
FSAR REV.65

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



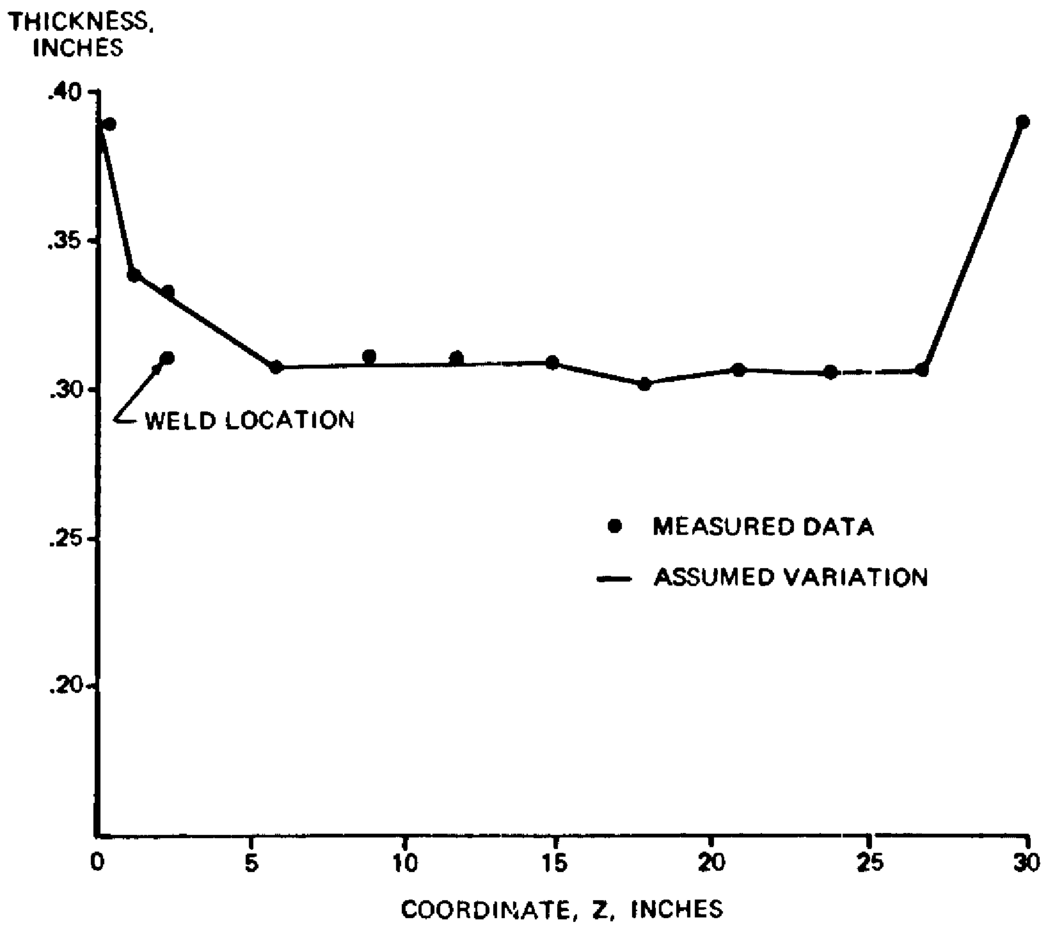
FSAR REV.65

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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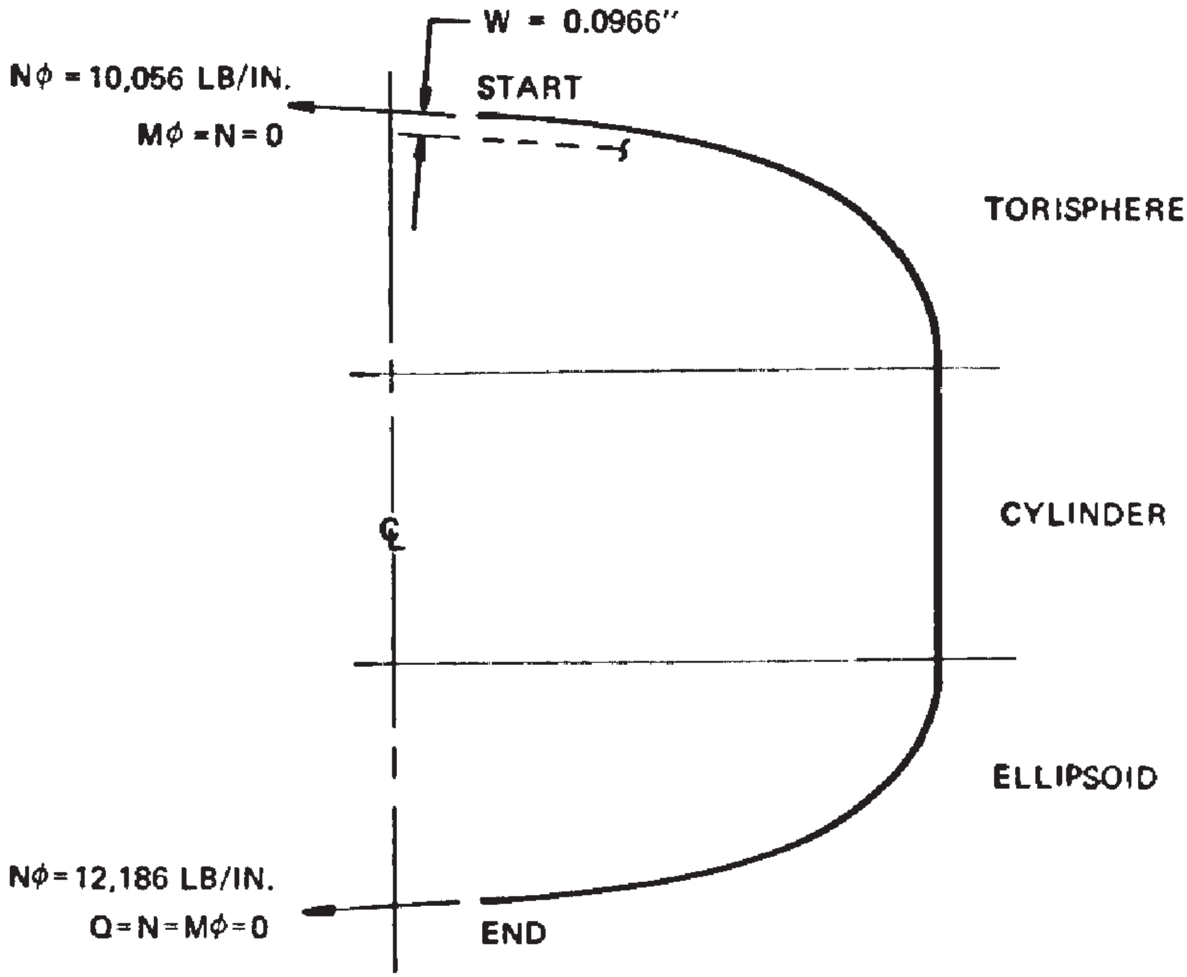
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

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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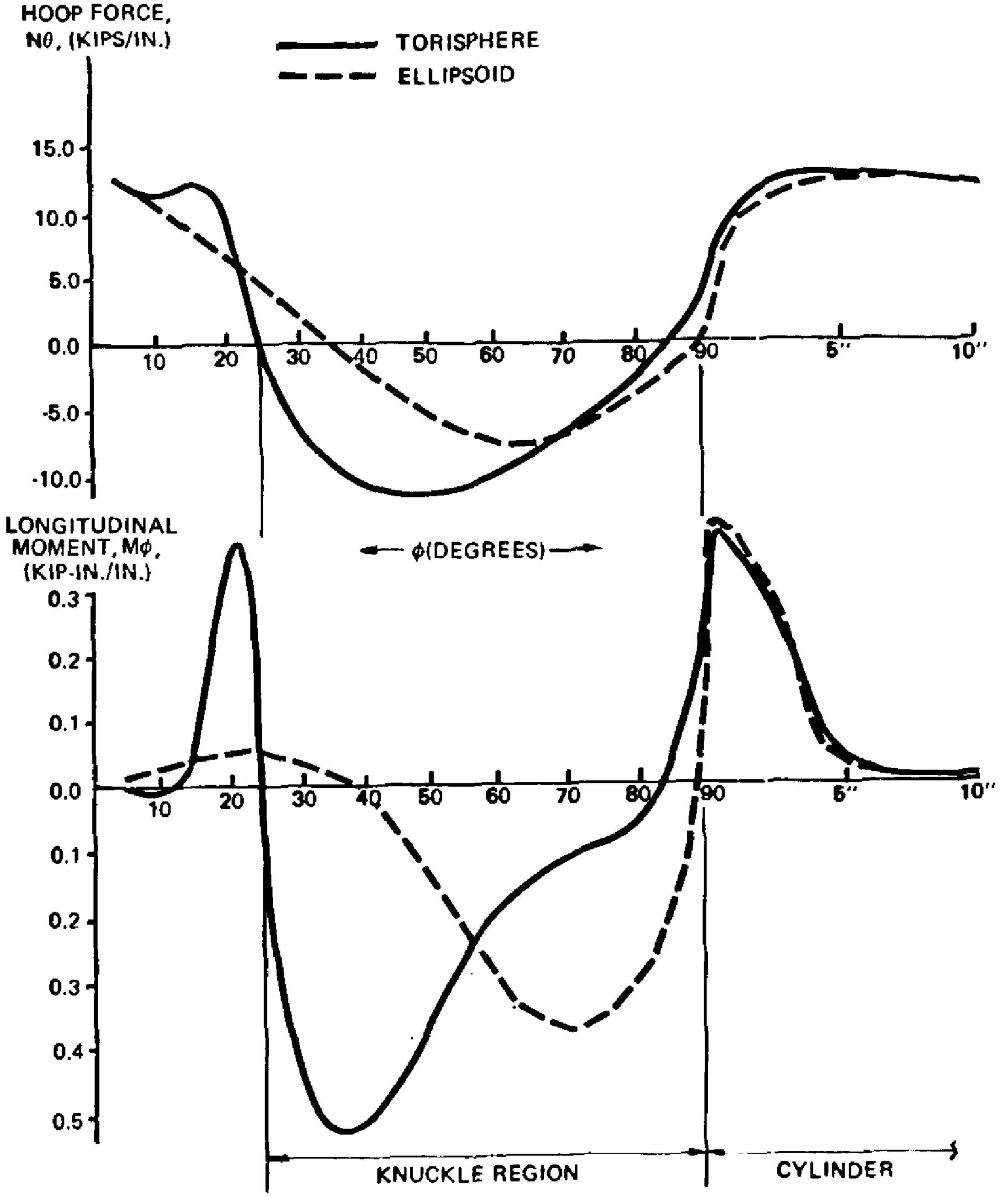
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

ANALYTICAL MODEL
WITH
BOUNDARY CONDITIONS

FIGURE 3.8A-24, Rev. 47

Auto-Cad Figure Fsar 3_8A_24.dwg



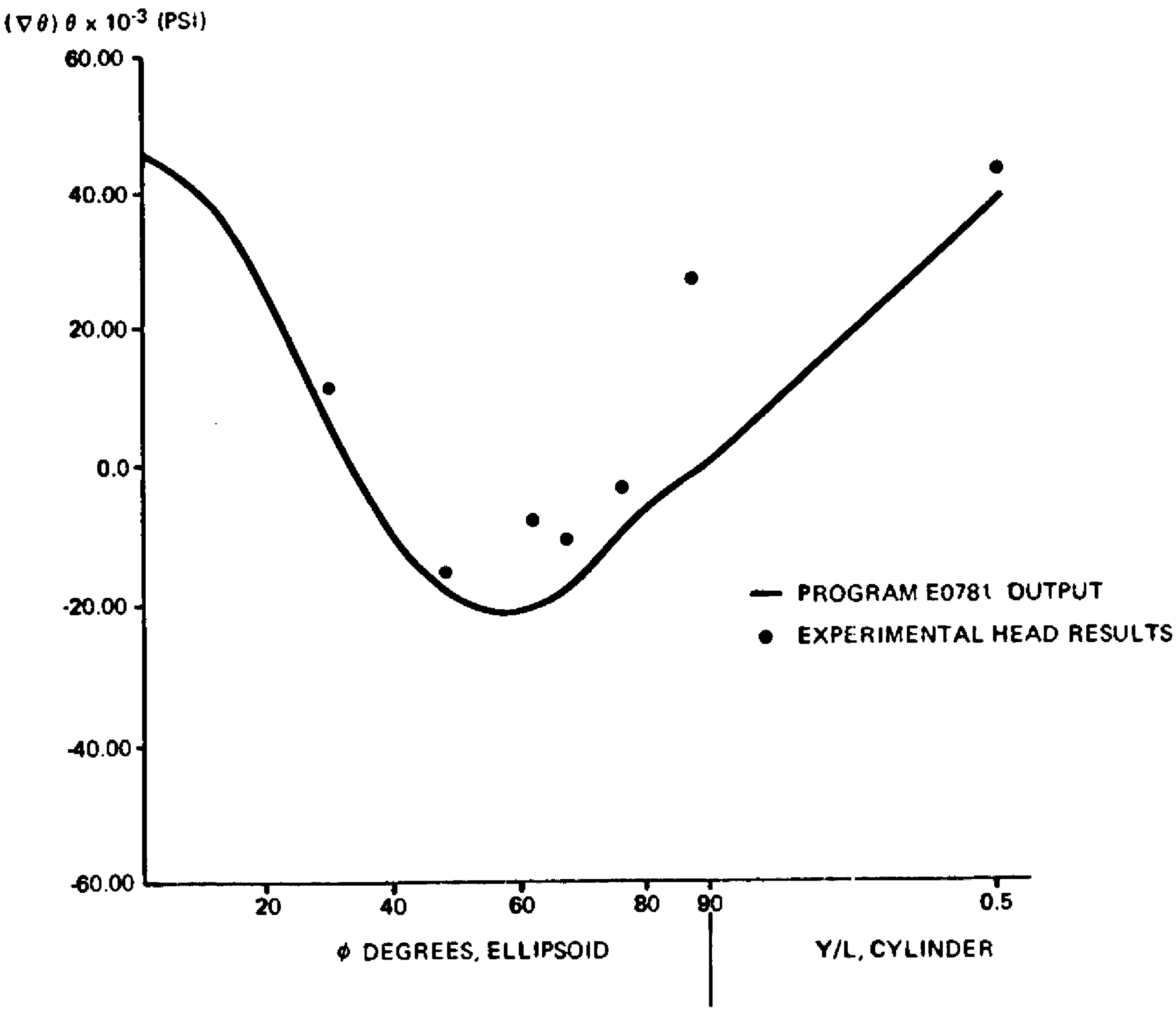
FSAR REV.65

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



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UNITS 1 & 2
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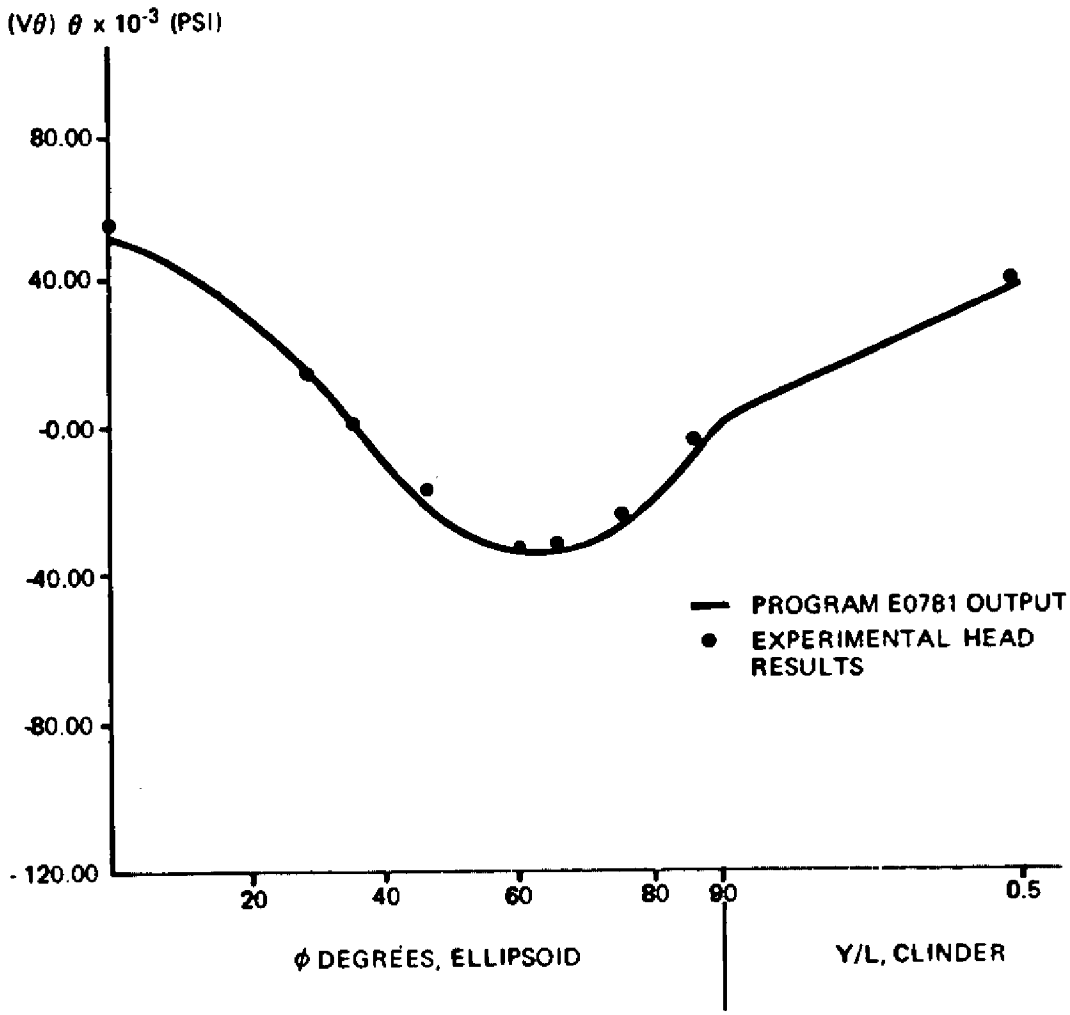
PLOT OF STRESS IN THE Θ (HOOP)
DIRECTION ON THE INSIDE
SURFACE (Θ = 0°)
(REF. 3.8A-12 PAGE G-14)

FIGURE 3.8A-26, Rev. 47

Auto-Cad Figure Fsar 3_8A_26.dwg



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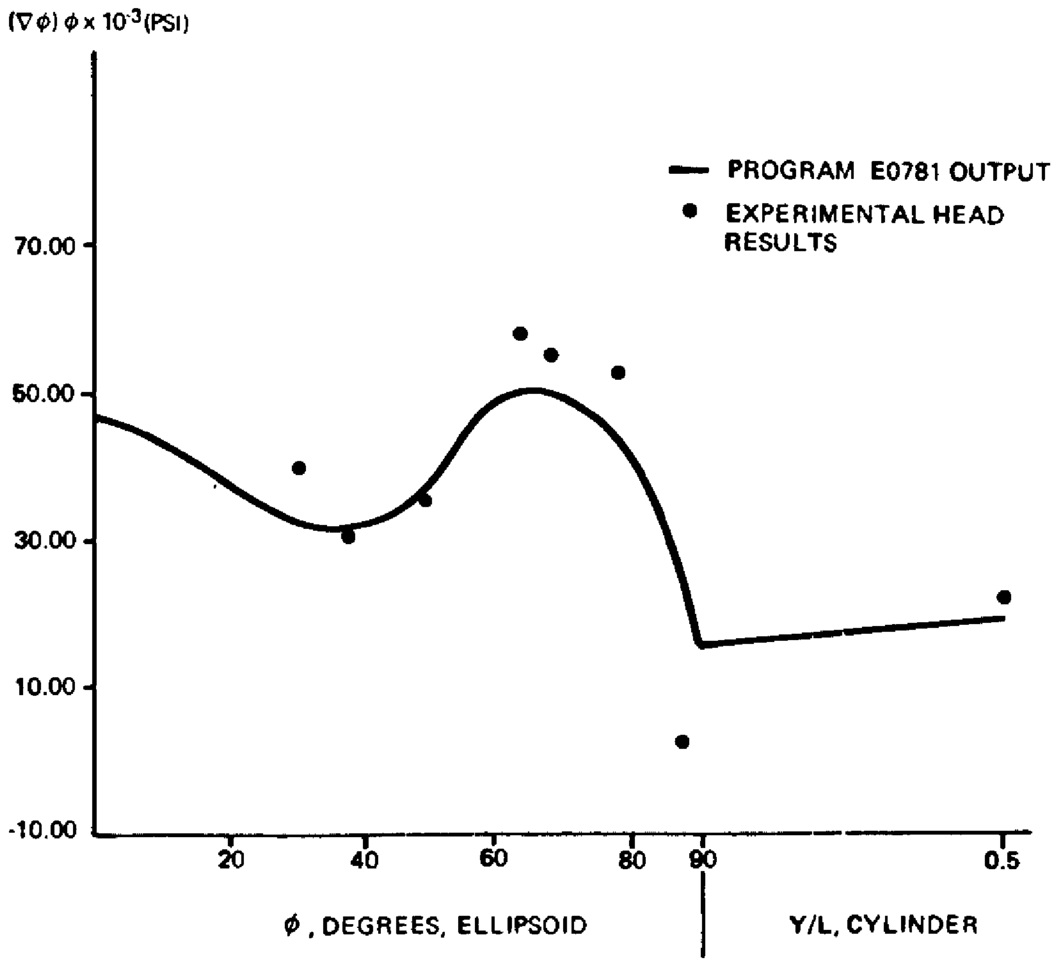
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

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



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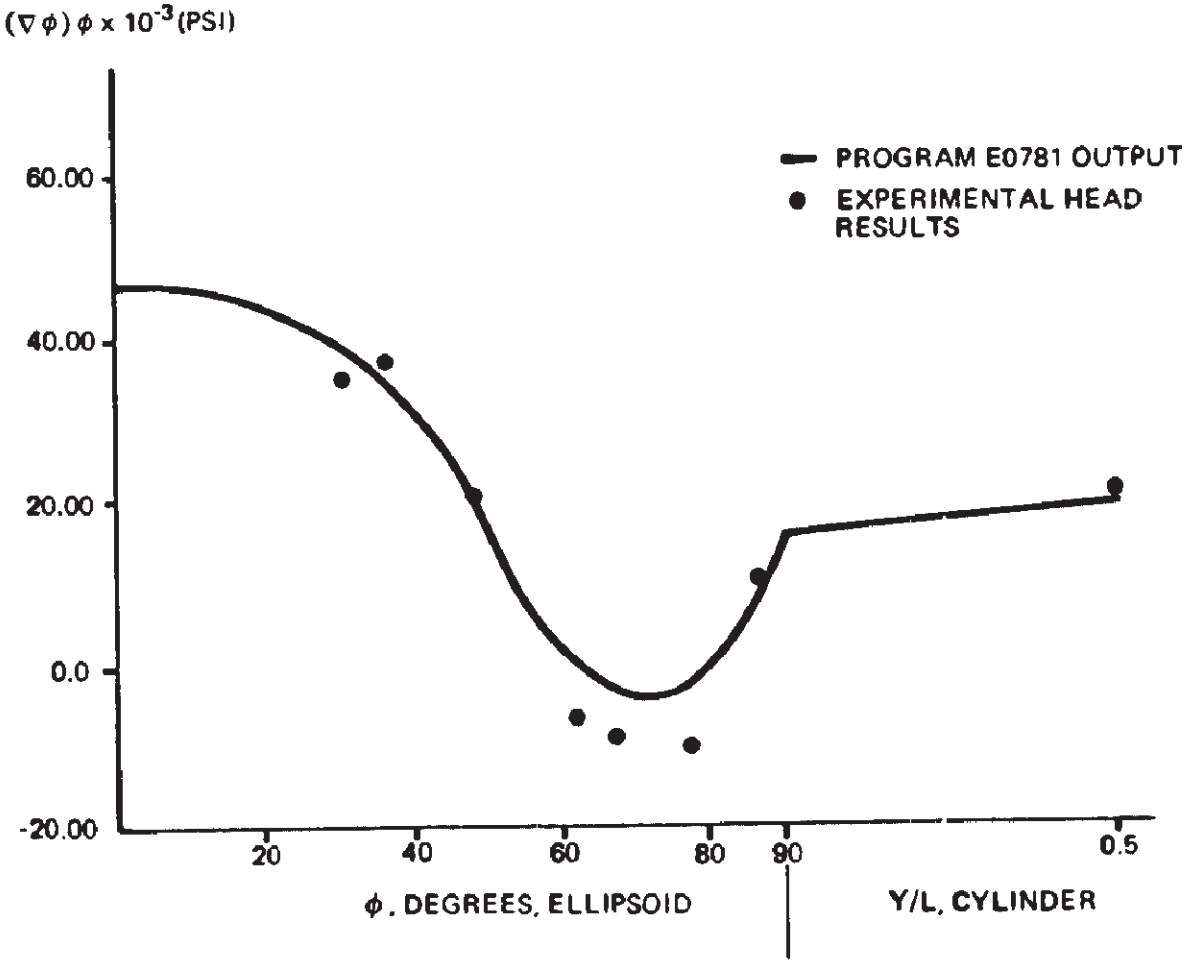
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

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



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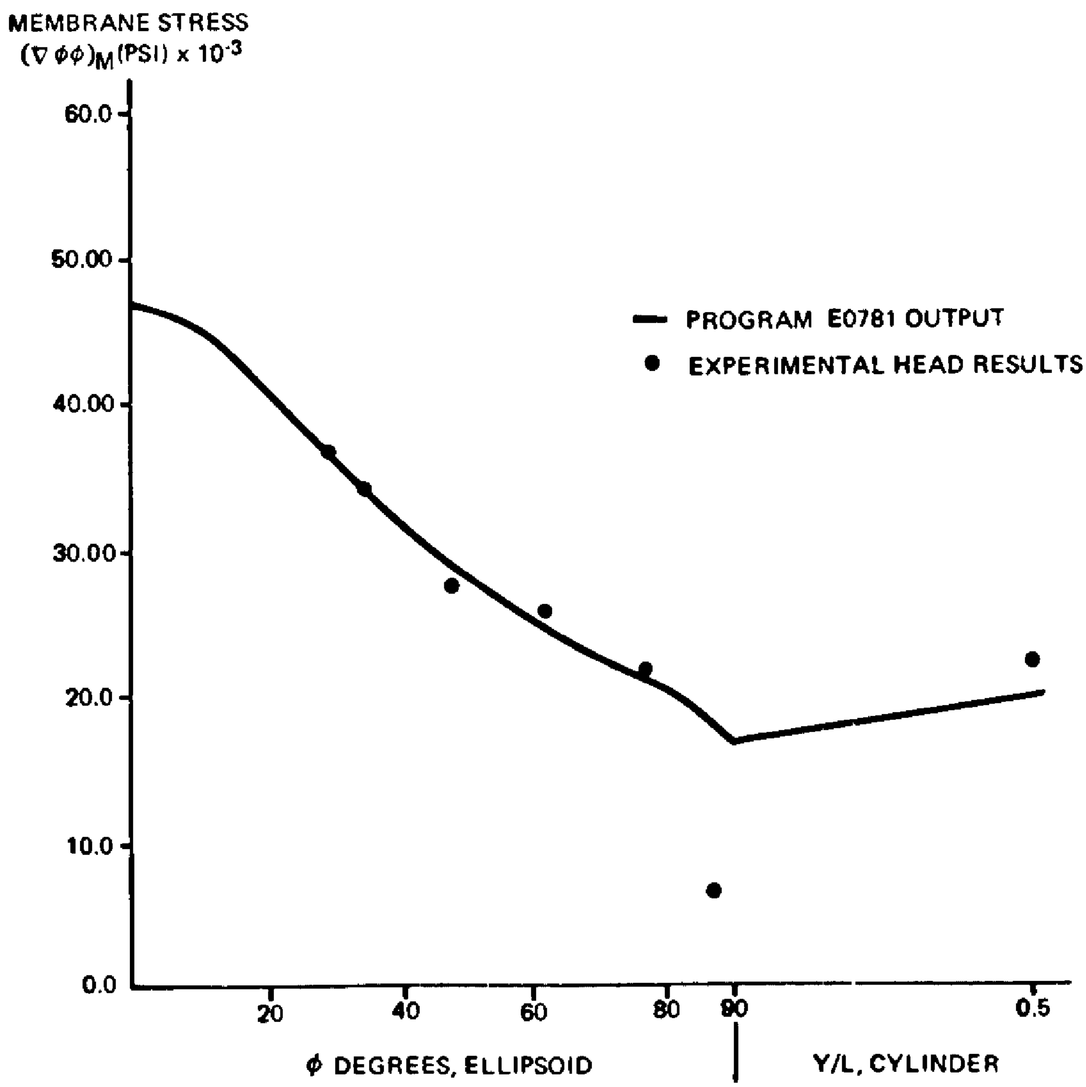
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

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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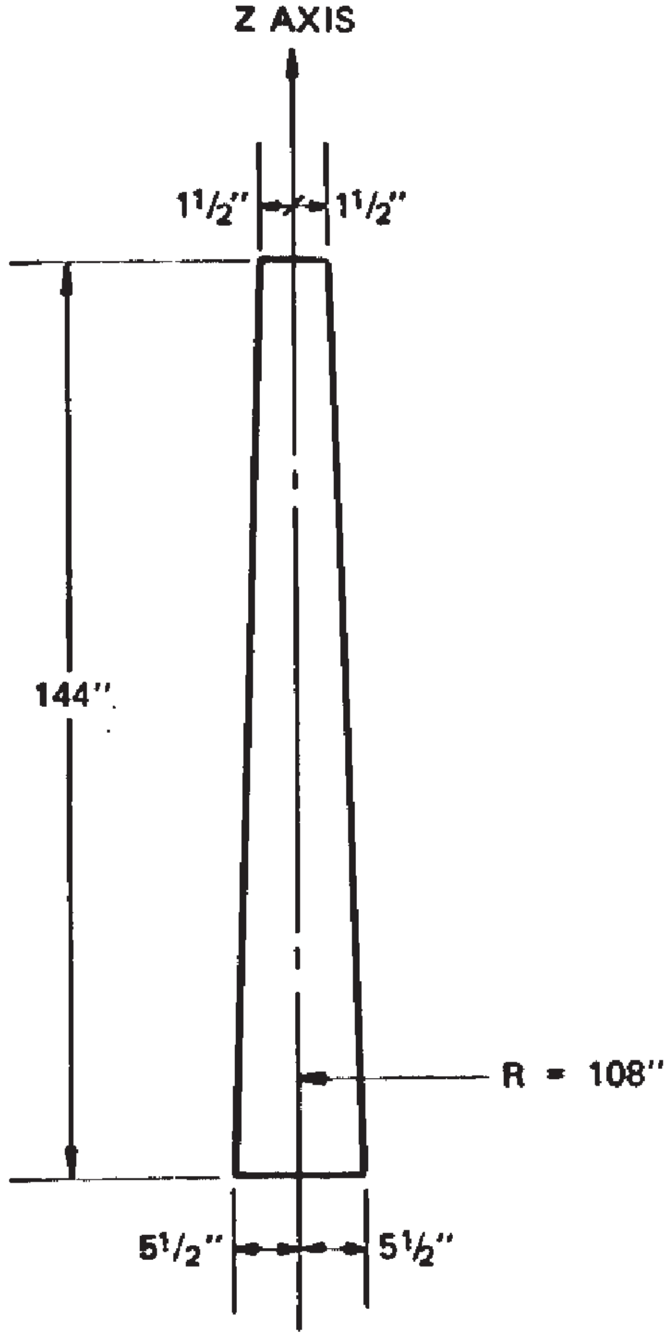
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

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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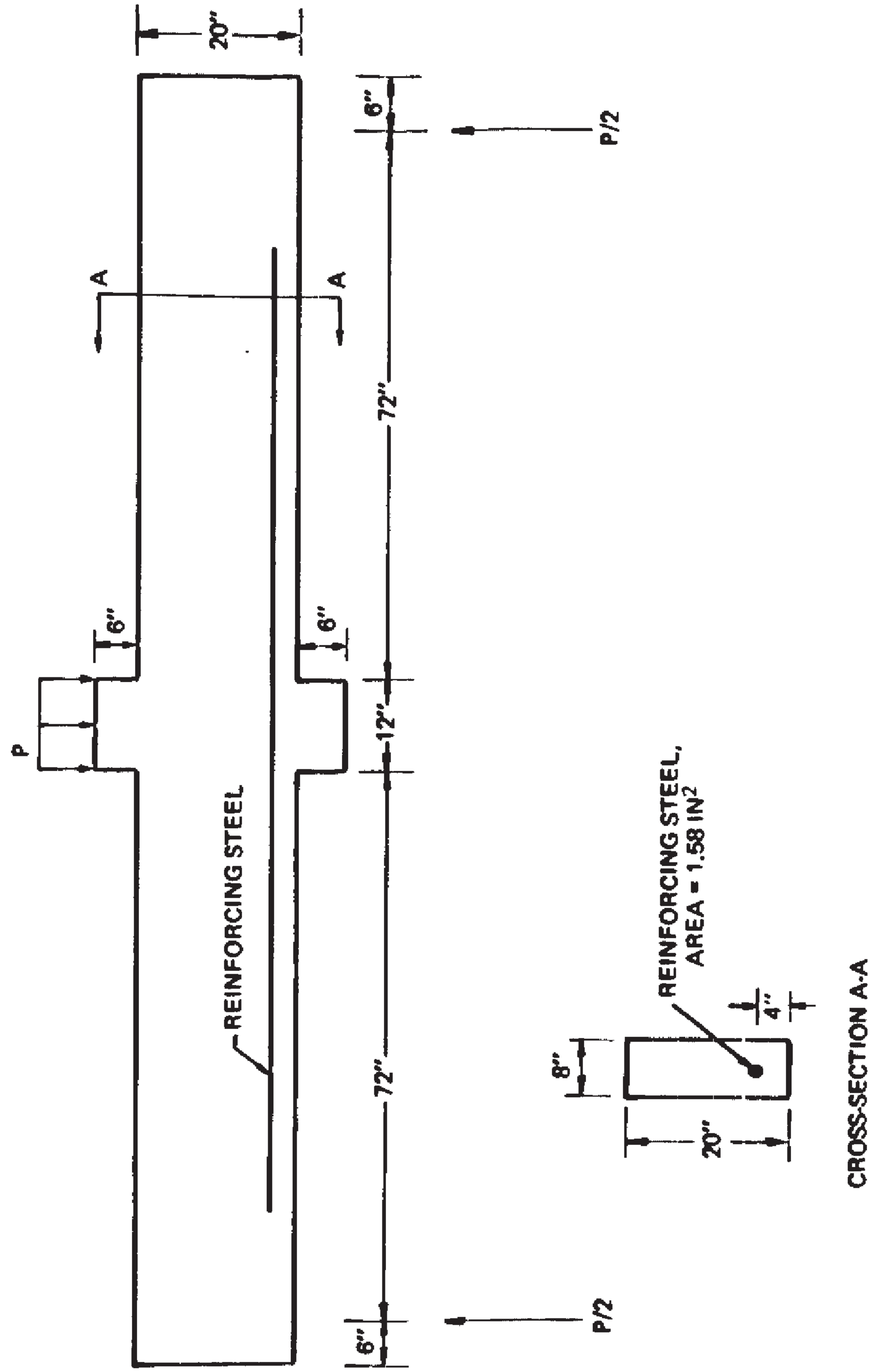
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

DIMENSIONS OF CYLINDRICAL WATER TANK

FIGURE 3.8A-31, Rev. 47

Auto-Cad Figure Fsar 3_8A_31.dwg



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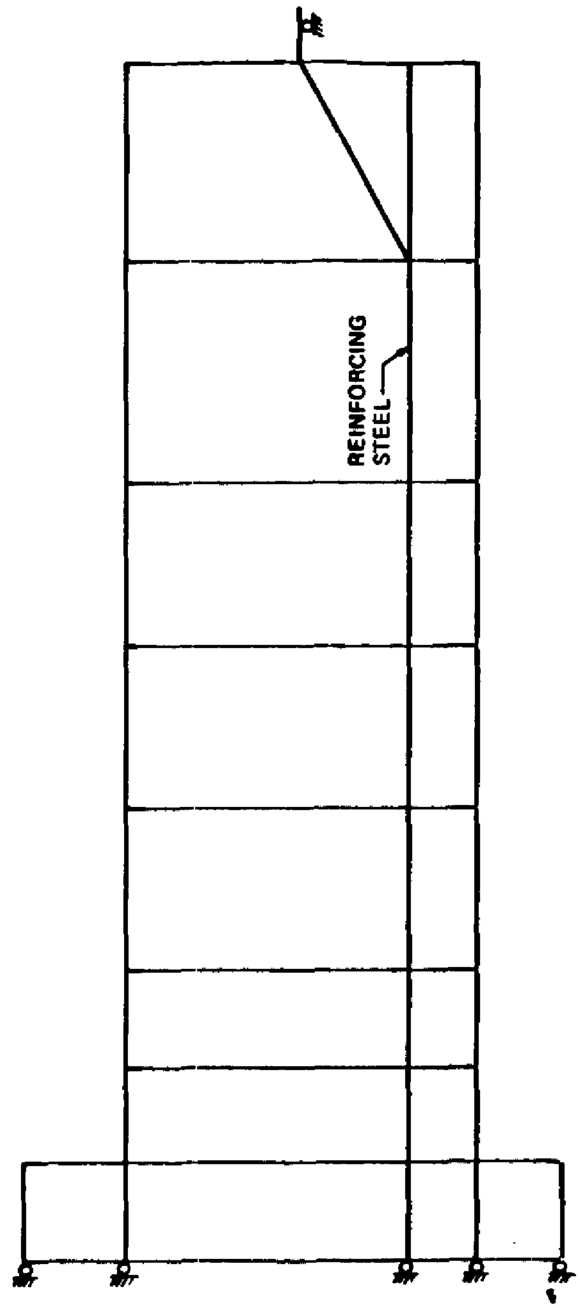
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

EXPERIMENTAL BEAM DIMENSIONS

FIGURE 3.8A-32, Rev. 47

Auto-Cad Figure Fsar 3_8A_32.dwg



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SUSQUEHANNA STEAM ELECTRIC STATION
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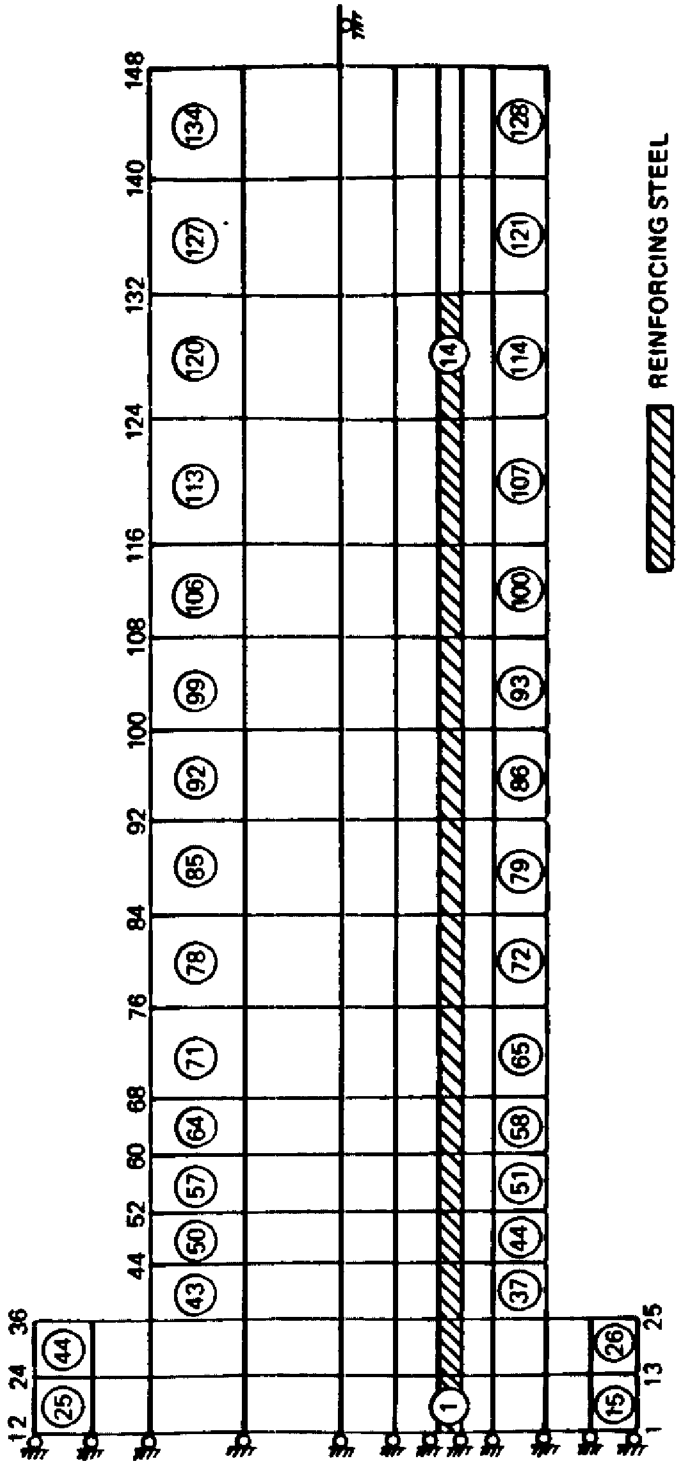
REFERENCE 3.8A-14 MESH

FIGURE 3.8A-33, Rev. 47

Auto-Cad Figure Fsar 3_8A_33.dwg



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SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

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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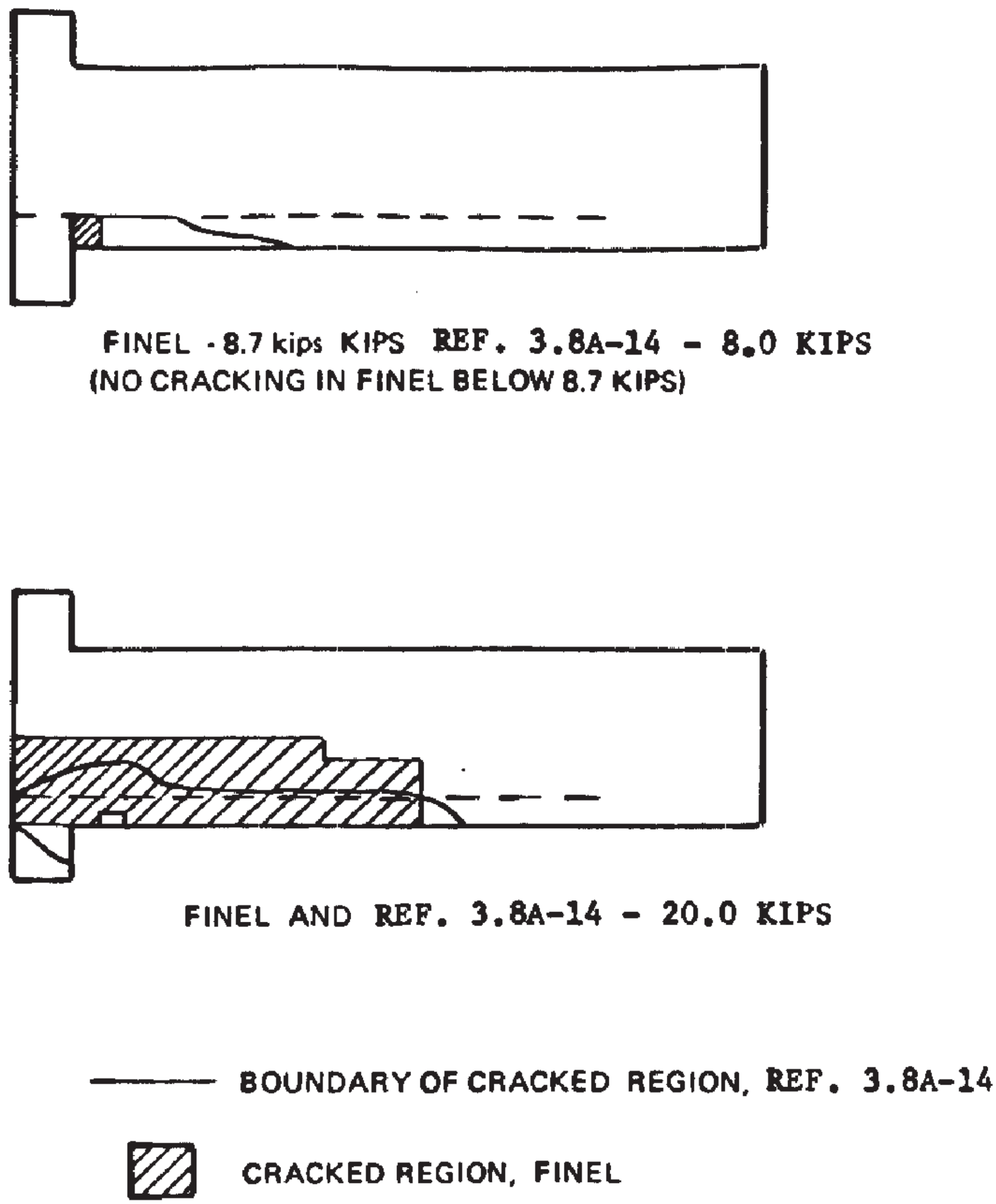
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

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



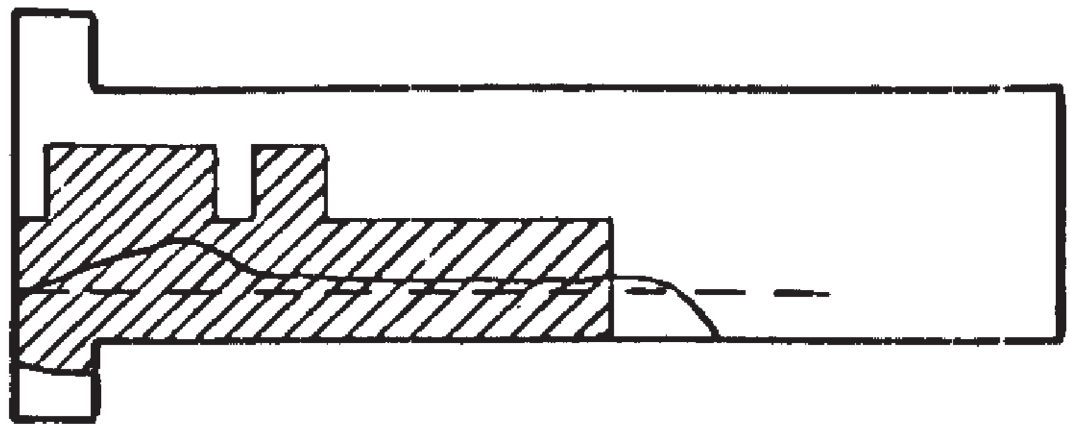
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

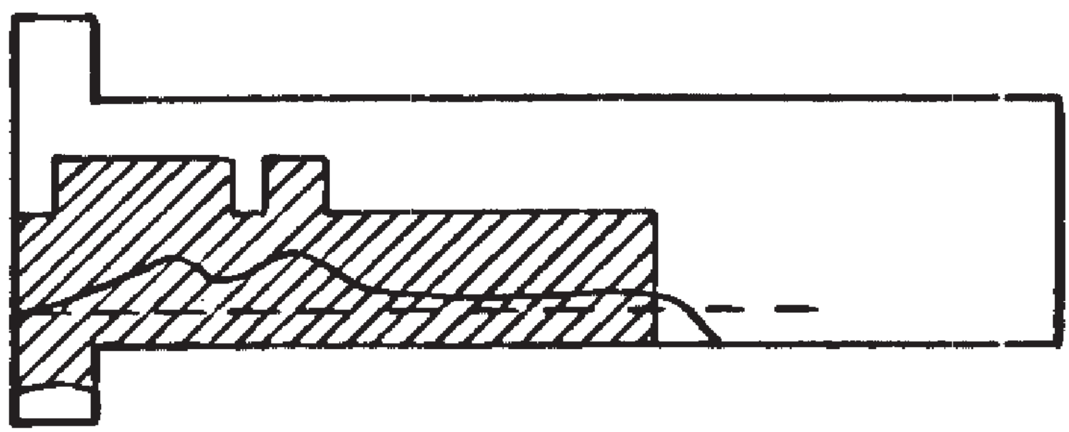
REGIONS OF CRACKING

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

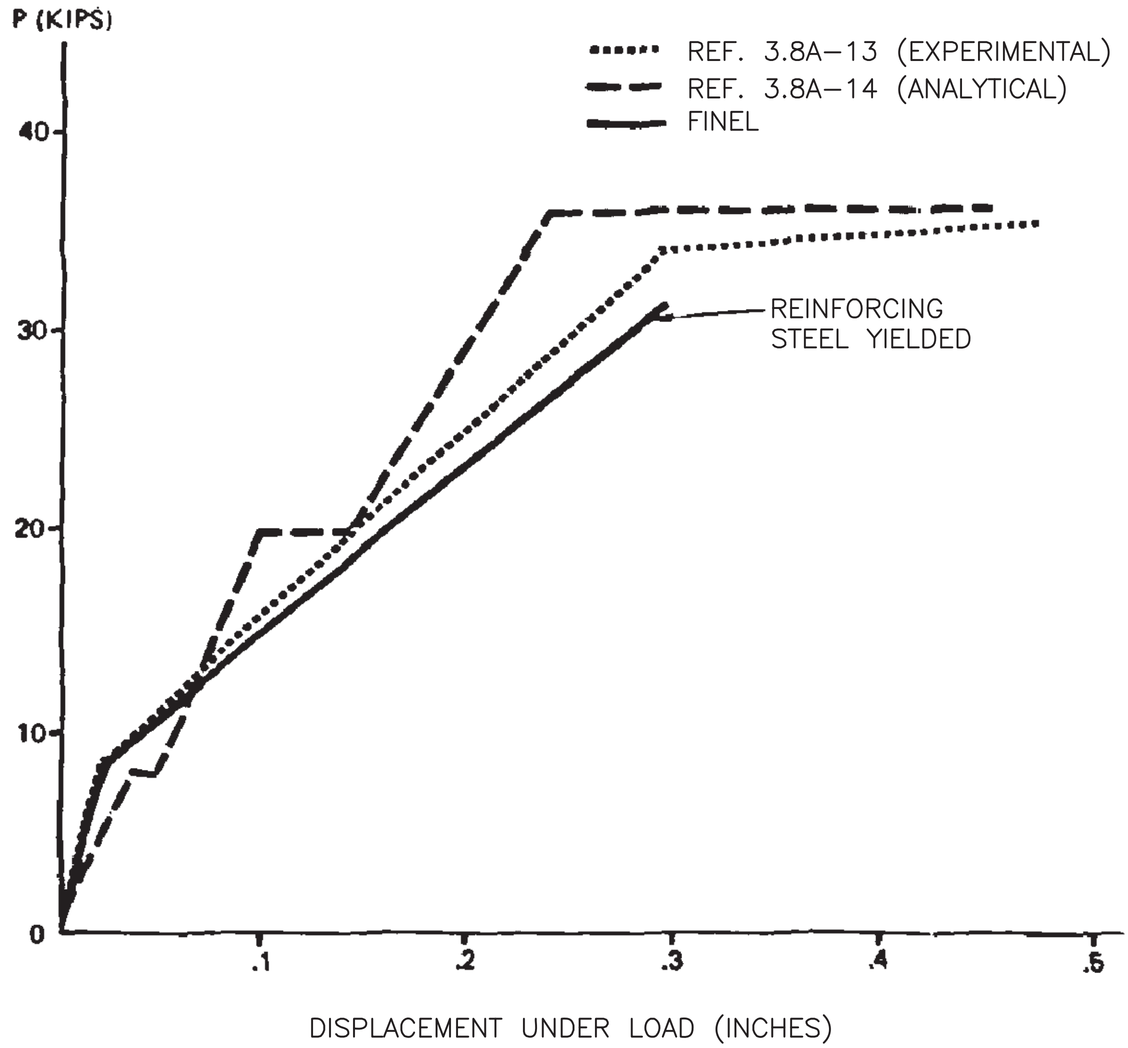
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

REGIONS OF CRACKING (CONT.)

FIGURE 3.8A-35-2, Rev. 47

Auto-Cad Figure Fsar 3_8A_35_2.dwg



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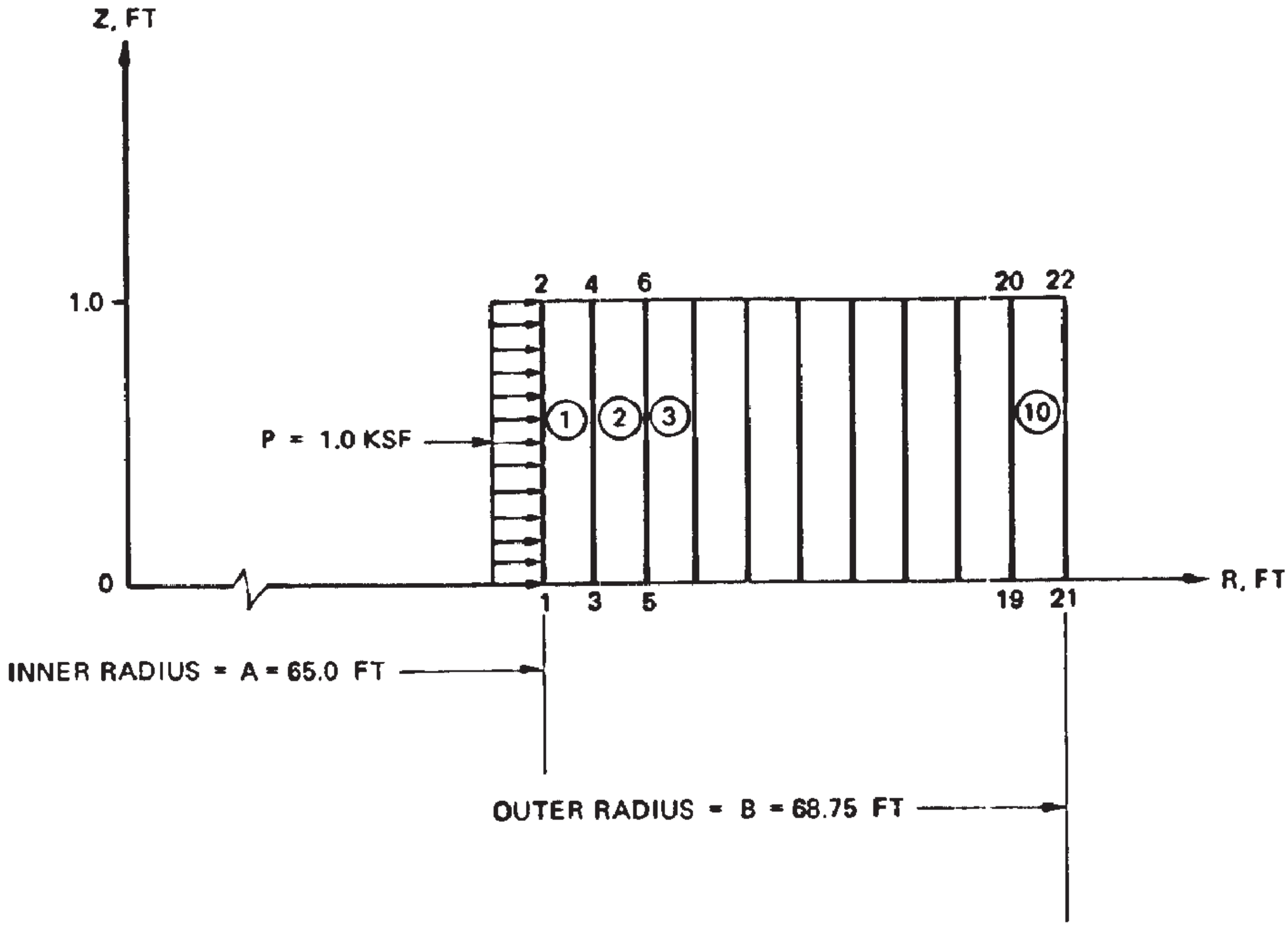
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

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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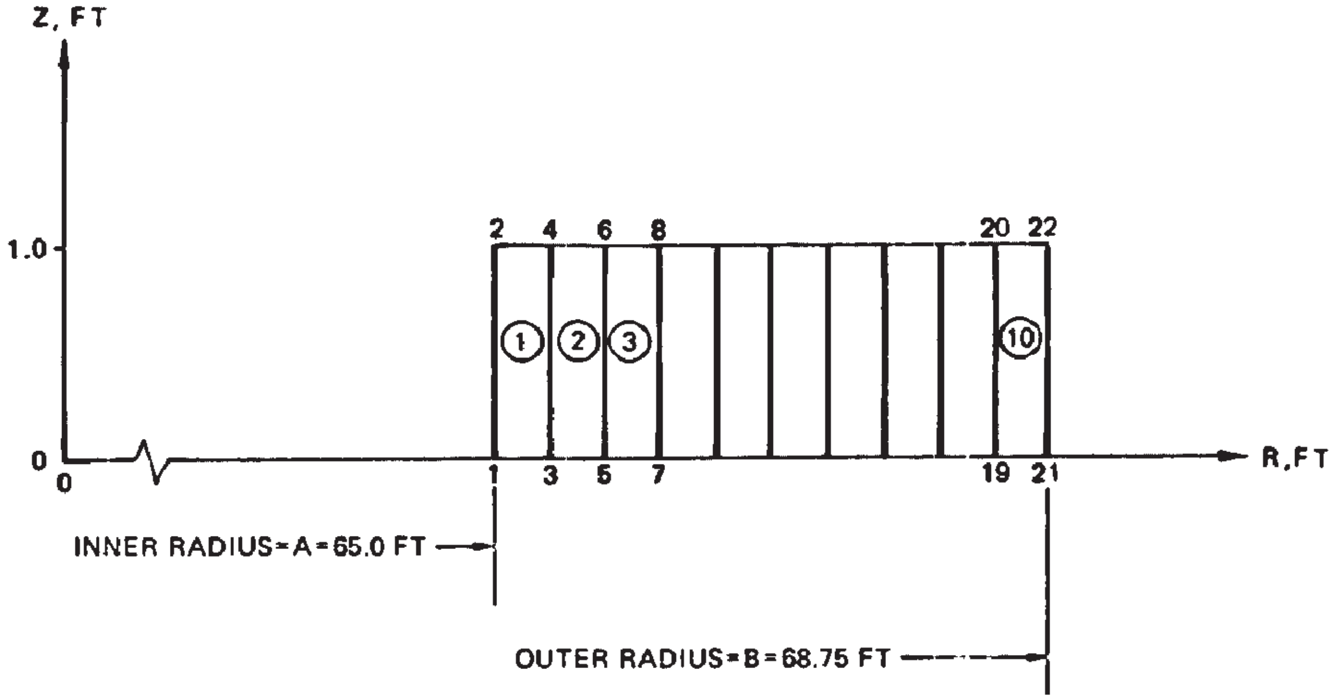
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

FINITE ELEMENT MODEL

FIGURE 3.8A-37, Rev. 47

Auto-Cad Figure Fsar 3_8A_37.dwg



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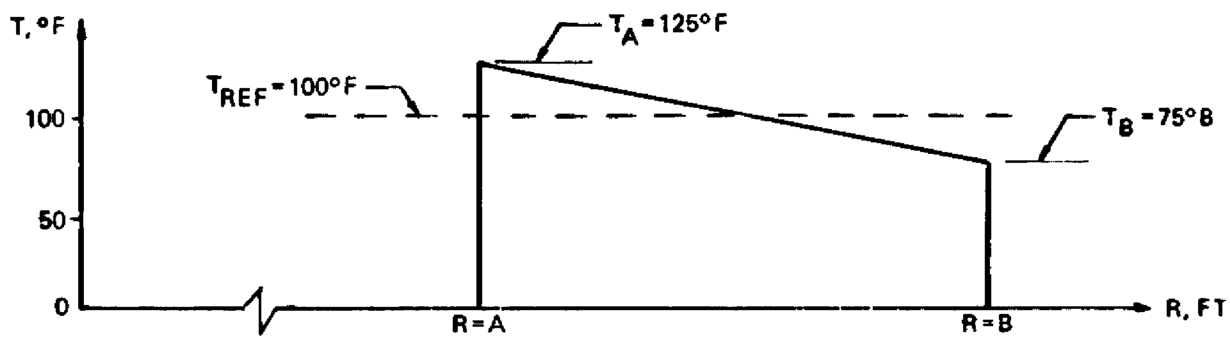
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

FINITE ELEMENT MODEL

FIGURE 3.8A-38, Rev. 47

Auto-Cad Figure Fsar 3_8A_38.dwg



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SUSQUEHANNA STEAM ELECTRIC STATION
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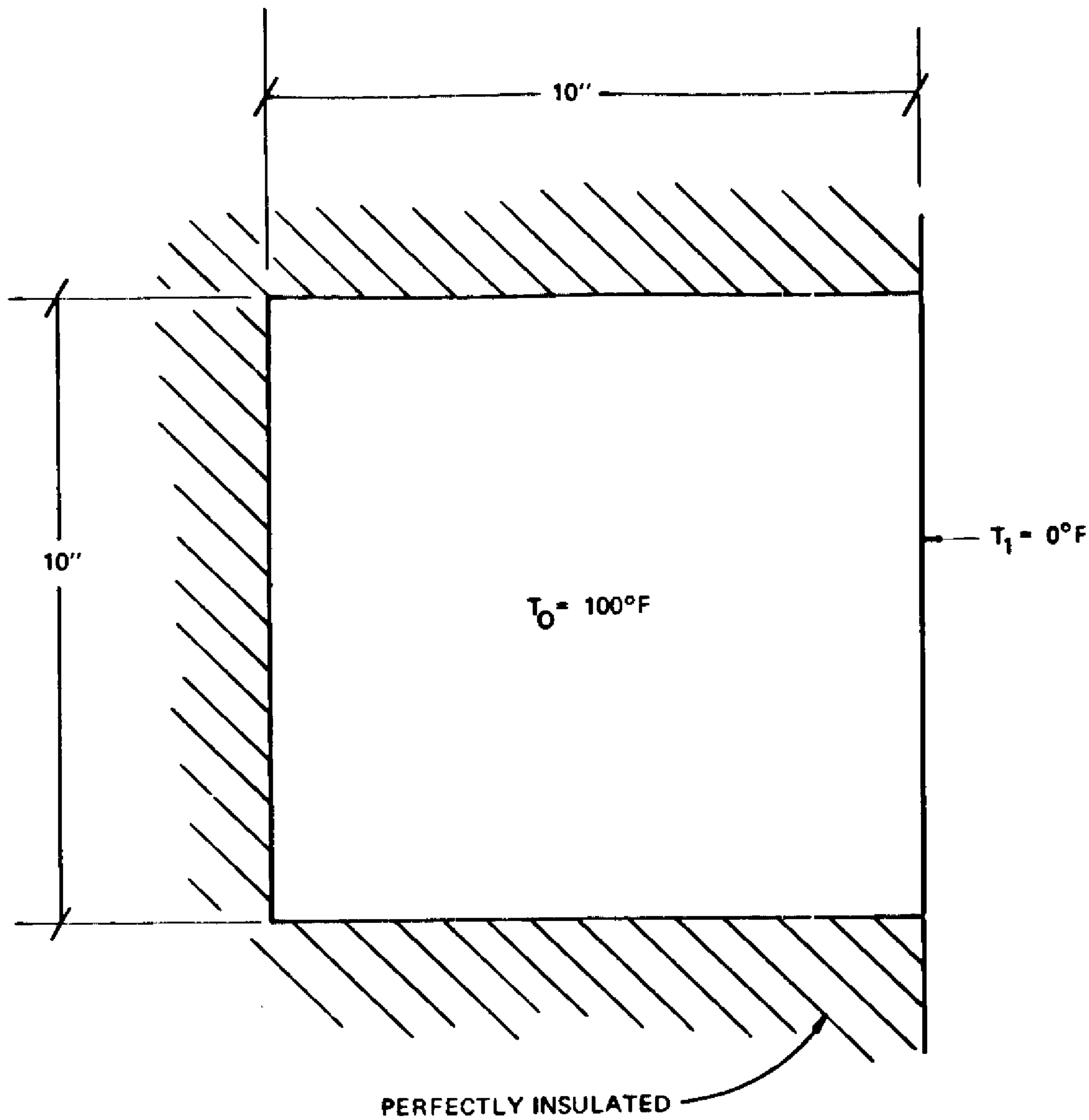
FINAL SAFETY ANALYSIS REPORT

TEMPERATURE DISTRIBUTION

FIGURE 3.8A-39, Rev. 47

Auto-Cad Figure Fsar 3_8A_39.dwg



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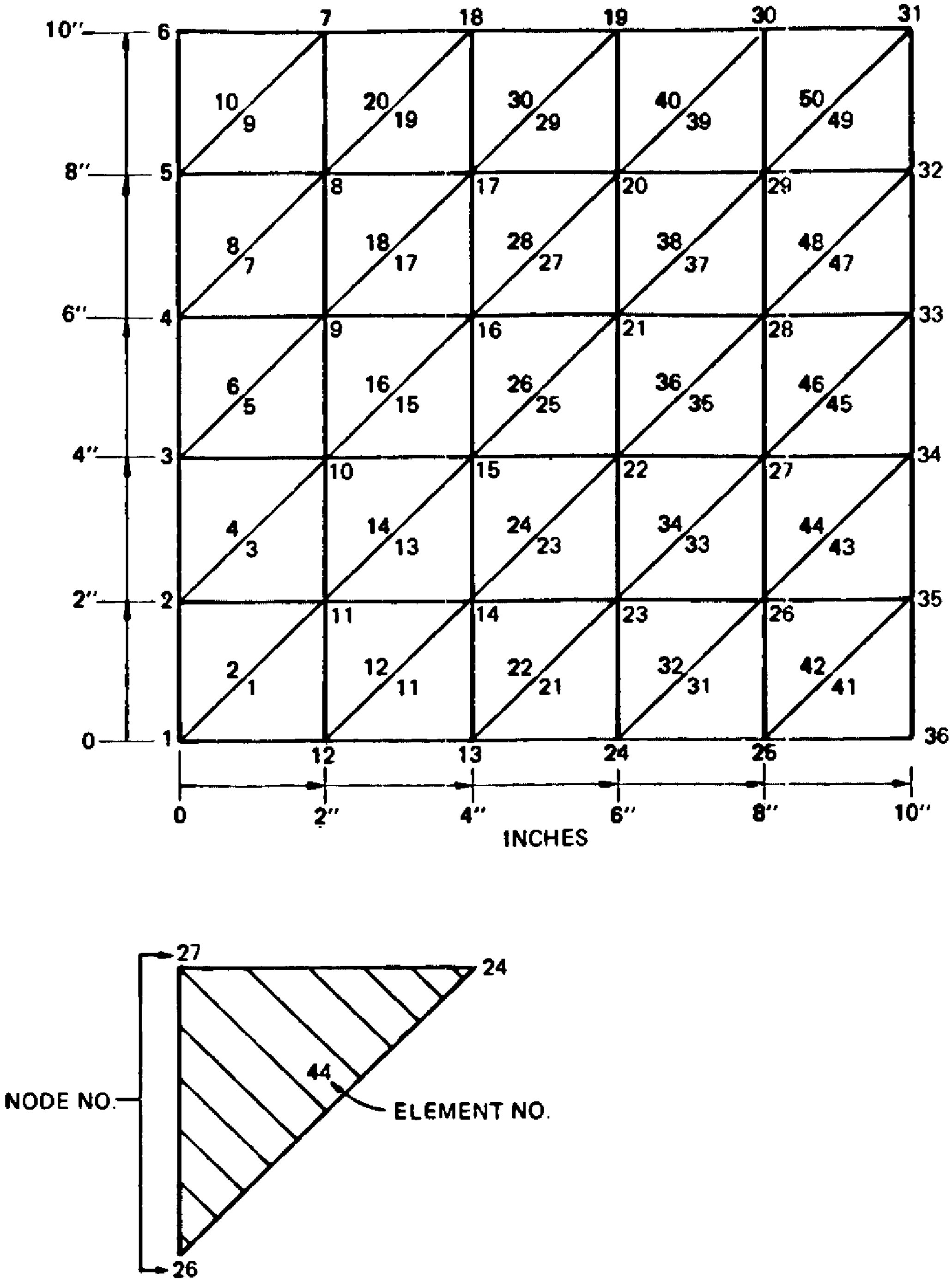
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

SCHEMATIC OF TEST PROBLEM

FIGURE 3.8A-40, Rev. 47

Auto-Cad Figure Fsar 3_8A_40.dwg



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SUSQUEHANNA STEAM ELECTRIC STATION
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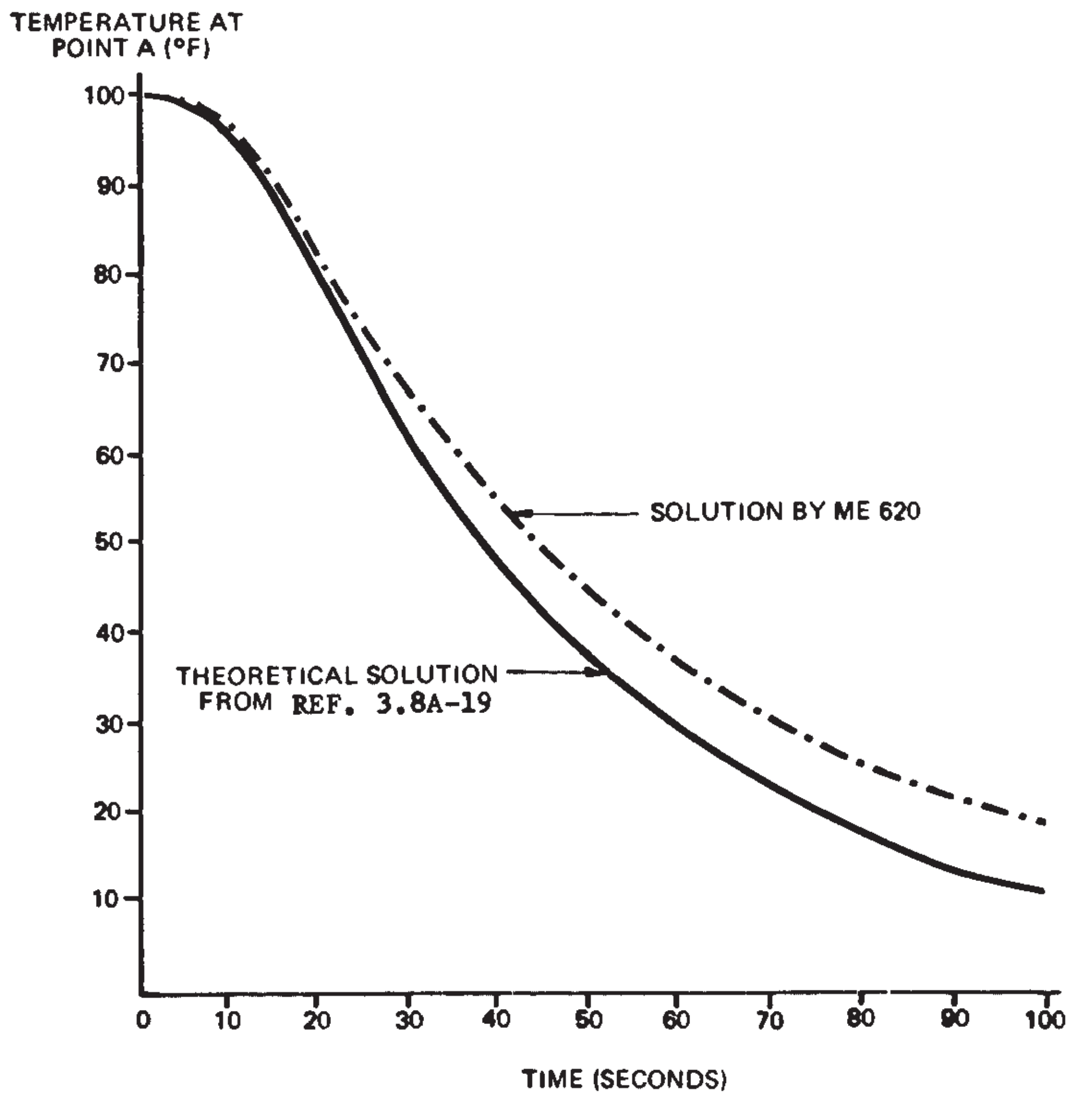
FINAL SAFETY ANALYSIS REPORT

FINITE ELEMENT LAYOUT

FIGURE 3.8A-41, Rev. 47

Auto-Cad Figure Fsar 3_8A_41.dwg



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SUSQUEHANNA STEAM ELECTRIC STATION
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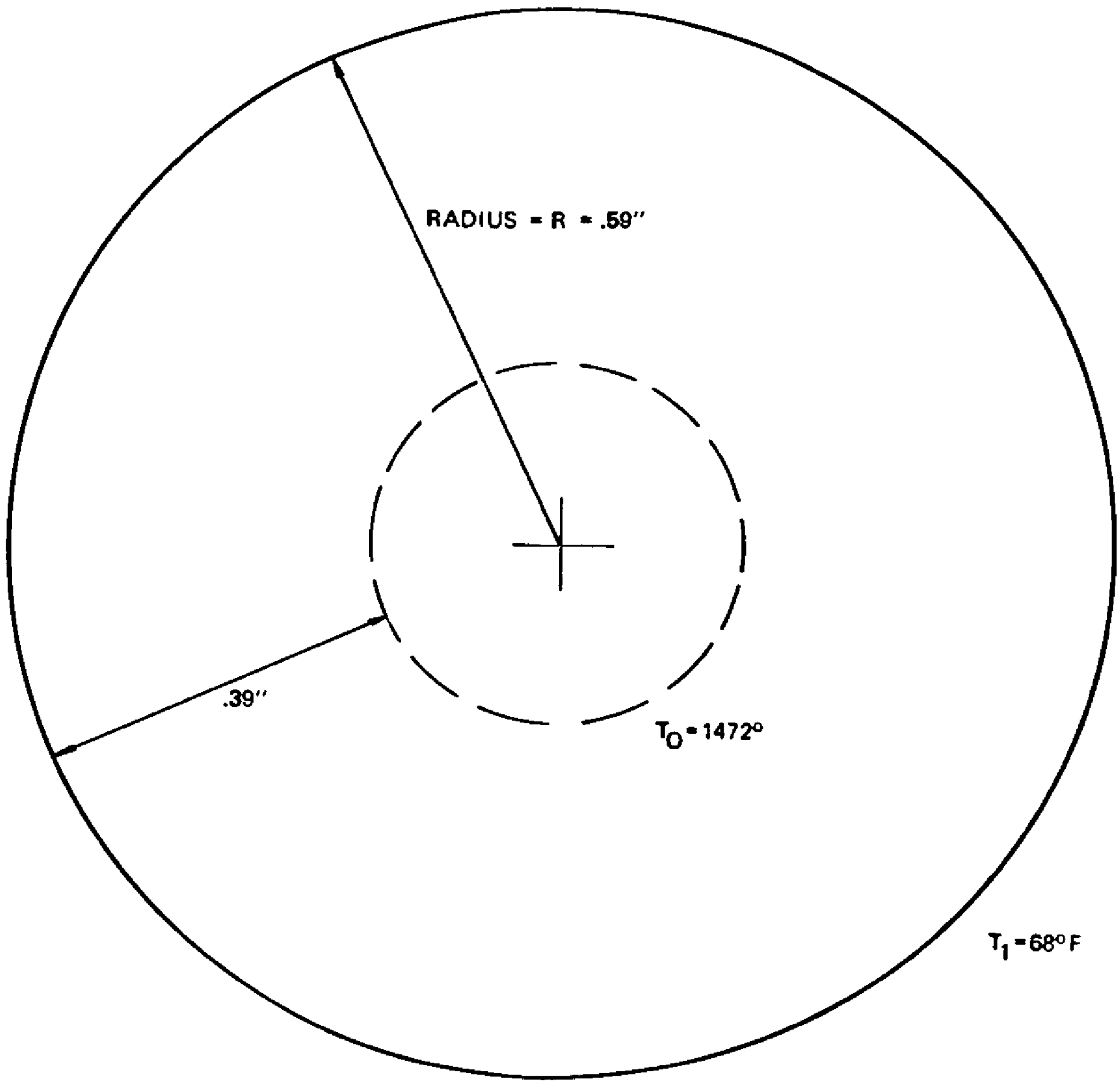
FINAL SAFETY ANALYSIS REPORT

COMPARISON OF RESULTS

FIGURE 3.8A-42, Rev. 47

Auto-Cad Figure Fsar 3_8A_42.dwg



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SUSQUEHANNA STEAM ELECTRIC STATION
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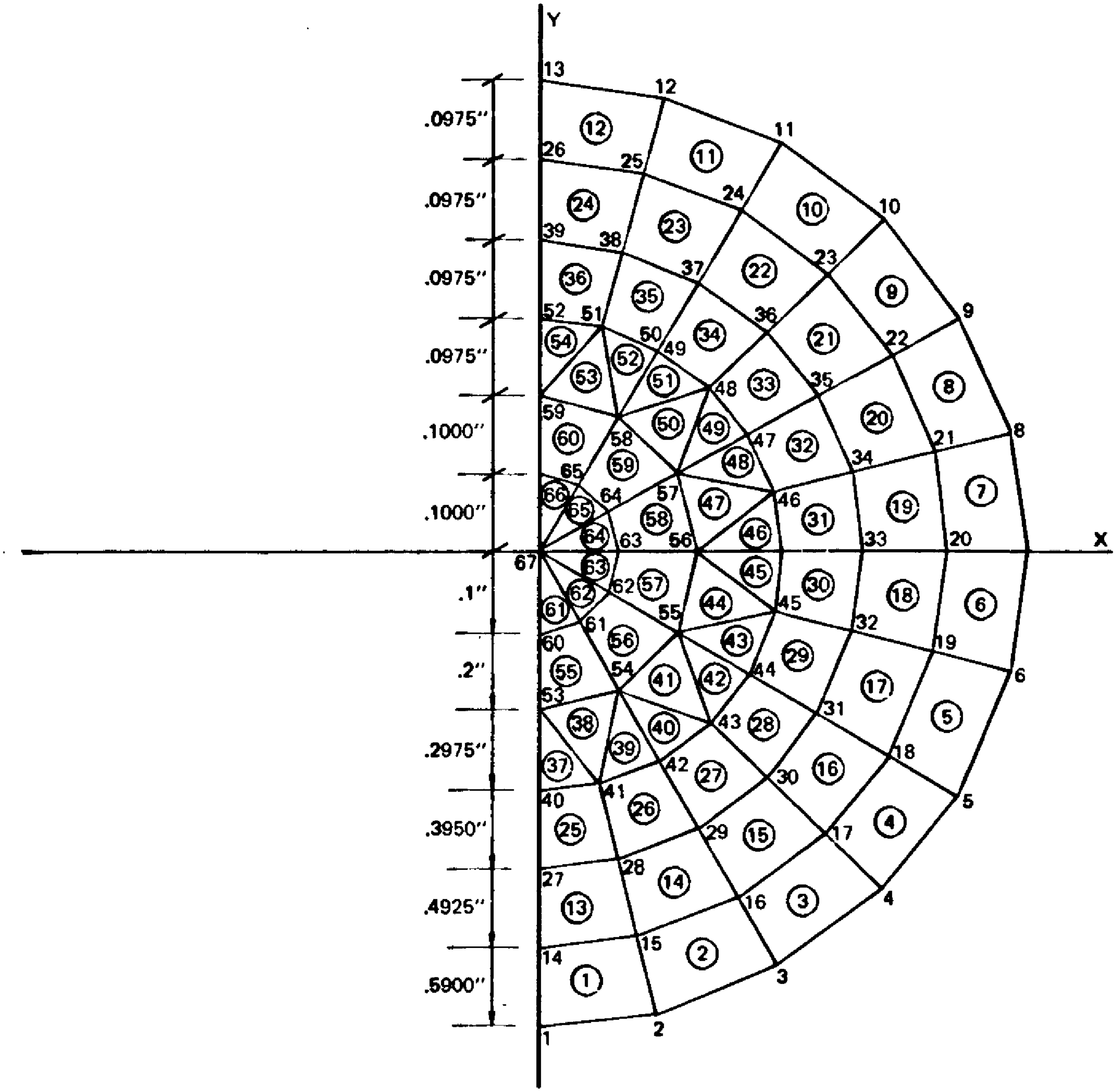
STEEL SPHERE

FIGURE 3.8A-43, Rev. 47

Auto-Cad Figure Fsar 3_8A_43.dwg



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SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

FINITE ELEMENT MODEL

FIGURE 3.8A-44, Rev. 47

Auto-Cad Figure Fsar 3_8A_44.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 150°F 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	Percentage Passing	
U.S. Std.	Fine Aggregate	Coarse Aggregate
Sq. Mesh	(Sand)	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 C1, 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 Proprietary Concrete

The above described concrete (3.8B.1.1 & 3.8B.1.2) is from original construction. EC 2198239 authorizes an alternate concrete made from proprietary materials from companies such as BASF and Five Star. Such concrete is proportioned in accordance with the manufacturer's recommendations and tested for compressive strength, bonding, and freeze/thaw resistance with water and aggregate content recommended by the manufacturer prior to use.

The dry concrete is a proprietary mix. Aggregates conform to ASTM C33. Mixing (potable) water will comply with ANSI/NSF 61.

Certified copies of material test reports showing chemical composition of the concrete and verification that the concrete being furnished complies with requirements are furnished by the manufacturer for each batch or lot OR the concrete will be dedicated. 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 OR the aggregate will be dedicated as part of the concrete. For admixtures, certificates of conformance stating conformance to the applicable ASTM specification are furnished with each shipment OR the admixtures will be dedicated as part of the concrete.

Concrete properties required for each mix are verified by testing for the applicable properties indicated below:

Property	Test Designation
Compressive Strength	ASTM C39
Bonding Strength	ASTM C882 (bonding agent not required)
Air Content	ASTM C231

The design strength (f 'c) of mixes is measured at 28 days. Three cylinders are tested for each mix design at 3, 7, and 28 days. The concrete will have established limits on aggregate-concrete mix ratio. The concrete mix will either have established limits on water-cement ratio OR the concrete will be 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.

Minimum testing requirements for concrete materials and concrete included in Table 3.8B-1 do not apply to the alternate proprietary concrete.

3.8B.1.4 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

Frequency of Test

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)

Material	Test (Specification)	Frequency of Test: By Manufacturer/Supplier/ Contractor	Frequency of Test: 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)

Material	Test (Specification)	Frequency of Test: By Manufacturer/Supplier/ Contractor	Frequency of Test: 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.

3.9 MECHANICAL SYSTEMS AND COMPONENTS

3.9.1 SPECIAL TOPICS FOR MECHANICAL COMPONENTS

3.9.1.1 Design Transients

This section shows the transients which are used in the design of the ASME Boiler and Pressure Vessel Code (ASME Code) Class 1 core support, reactor internals, and control rod drive (CRD) components. The number of cycles or events for each transient are included. The design transients shown in this section are included in the design specifications for the components. Transients or combinations of transients are classified with respect to the component operating condition categories identified as "normal," "upset," "emergency," "faulted," or "testing" in the ASME Code if applicable. (The first four conditions correspond to Service Levels A, B, C, and D, respectively for those components constructed to the 1976 or later Edition of the ASME Code.)

3.9.1.1.1 Control Rod Drive Transients

The normal and test service load cycles used for design purposes for the 40 year life of the Control Rod Drive (CRD) are as follows:

Transient		Category	Cycles
1.	Reactor startup/shutdown	Normal/upset	120
2.	Vessel pressure tests	Normal/upset	130
3.	Vessel overpressure tests	Normal/upset	10
4.	Scram test plus startup scrams	Normal/upset	300
5.	Operational scrams	Normal/upset	300
6.	Jog cycles	Normal/upset	30,000
7.	Shim/drive cycles	Normal/upset	1,000

In addition to the above cycles, the following have been considered in the design of the CRD.

Transient		Category	Cycles
8.	Operating Basis Earthquake* (OBE)	Normal/upset	10
9.	Safe Shutdown Earthquake** (SSE)	Faulted	1
10.	Scram with inoperative buffer	Normal/upset	10
11.	Scram with stuck control blade	Normal/upset	1

All ASME Code Class 1 components of the CRD have been analyzed according to the ASME Code.

The capability of the CRD's to withstand the emergency and faulted conditions is verified by test rather than analysis.

3.9.1.1.2 CRD Housing and Incore Housing Transients

The number of transients, their cycles, and classification as considered in the design and fatigue analysis of the CRD housing and incore housing are as follows:

Transient		Category	Cycles
1.	Normal startup & shutdown	Normal/upset	120
2.	Vessel pressure tests	Normal/upset	130
3.	Vessel overpressure tests	Normal/upset	10
4.	Interruption of feedwater flow	Normal/upset	80
5.	Scram	Normal/upset	200
6.	OBE	Normal/upset	10
7.	SSE	Faulted	1
8.	Stuck rod scram	Normal/upset	1

* The frequency of occurrence of this transient would indicate emergency category. However, for conservatism, this OBE condition was analyzed as an upset condition. Ten peak OBE cycles for each occurrence are postulated.

** SSES is a faulted condition; however, in the stress analysis report, it was treated as emergency with lower stress limits.

Transient		Category	Cycles
9.	Scram no buffer	Normal/upset	10

3.9.1.1.3 Hydraulic Control Unit Transients

The normal and test service load cycles used for the design and fatigue analysis for the 40 year life and the Hydraulic Control Unit (HCU) are as follows:

Transient		Category	Cycles
1.	Normal startup and shutdown	Normal/upset	120
2.	Vessel pressure tests	Normal/upset	130
3.	Vessel overpressure tests	Normal/upset	10
4.	Scram tests (cold)	Normal/upset	300
5.	Operational scrams (hot)	Normal/upset	300
6.	Jog cycles	Normal/upset	30,000
7.	Drive cycles	Normal/upset	1,000
8.	Scram with stuck scram discharge valve	Normal/upset	1
9	OBE	Normal/upset	10
10.	SSE	Faulted	1

3.9.1.1.4 Core Support and Reactor Internals Transients

The events and number of cycles used for the design and fatigue analysis for the 40-year life of the core support and internals are shown in Table 3.9-1.

3.9.1.1.5 Main Steam System Transients

The transients considered in the stress analysis of the main steam piping are included in Table 3.9-4.

3.9.1.1.6 Recirculation System Transients

The transients considered in the stress analysis of the recirculation piping are included in Table 3.9-4.

3.9.1.1.7 Reactor Assembly Transients

The reactor assembly includes the reactor pressure vessel, support skirt, shroud support, and shroud plate. The cycles listed in Table 3.9-1 were specified in the reactor assembly design and fatigue analysis.

Reactor design cycle or transient limits are as follows:

Transient	Design Cycle
Heatup and Cooldown (Envelopes items 3, 10, 13, 16b, 16d from Table 3.9-1)	70°F to 551°F to 70°F
Reactor Trip cycle (Item 9 from Table 3.9-1)	100% to 0% Rated Thermal Power
Hydrostatic Pressure and Leak Tests (Item 2 from Table 3.9-1)	Pressurized ≥930 psig and ≤1250 psig

3.9.1.1.8 Main Steam Isolation Valve Transients

The main steam isolation valves are designed for the following service conditions and thermal cycles:

Transient			Category	Cycles
1.	Pre-op @100°F/hr		Normal/upset	150
2.	Startup (heating 100°F/hr)		Normal/upset	120
3.	Shutdown			
	a.	cooling cycles @100°F/hr 540°F to 375°F	Normal/upset	120
	b.	cooling cycles @270°F/hr 375°F to 330°F	Normal/upset	120
	c.	cooling cycles @100°F/hr 330°F to 100°F	Normal/upset	120
4.	Scram cooling cycles @100°F/hr		Normal/upset	180
5.	Emergency and faulted transients			
	a.	546°F to 281°F in 15 sec	Emergency/faulted	1
	b.	546°F to 375°F in 3.3 min	Emergency/faulted	1
		375°F to 281°F @300°F/hr	Emergency/faulted	1
	c1.	546°F to 375°F in 10 min	Emergency/faulted	8

Transient		Category	Cycles
c2.	375°F to 281°F @100°F/hr	Emergency/faulted	8
d1.	546°F to 583°F in 2 sec	Emergency/faulted	1
d2.	583°F to 538°F in 30 sec	Emergency/faulted	1
d3.	538°F to 400°F @100°F/hr	Emergency/faulted	1
d4.	400°F to 546°F @100°F/hr	Emergency/faulted	1
e1.	561°F to 500°F in 7 min	Emergency/faulted	10
e2.	500°F to 400°F @100°F/hr	Emergency/faulted	10
e3.	400°F to 546°F @100°F/hr	Emergency/faulted	10

3.9.1.1.9 Safety/Relief Valves Transients

The transients used in the analysis of the safety/relief valves are as follows:

Transient		Category	Cycles
1.	Pre-op and in-service testing (100°F/hr).	Normal/upset	150
2.	Startup (100°F/hr) and pressure increase (0 psig to 1000 psig).	Normal/upset	120
3.	Shutdown (100°F/hr, pressure decrease to 0 psig).	Normal/upset	120
4.	Scram.	Normal/upset	180
5.	System pressure and temperature decay from 1000 psig and 546°F to 35 psig and 281°F within 15 sec.	Emergency/faulted	1
6.	System temperature change from 546°F to 375°F within 3.3 mins and from 375°F to 281°F at a rate of 300°F/hr. Pressure change from 1000 psig to 35 psig	Emergency/faulted	1
7.	System temperature change from 546°F to 375°F within 10 min. and from 375°F to 281°F at a rate of 100°F/hr. Pressure change from 1000 psig to 35 psig.	Emergency/faulted	8

Transient		Category	Cycles
8.	System temperature change from 546°F to 583°F within 2 sec, from 583°F to 538°F within 30 sec, and from 538°F to 400°F and return to 546°F at a rate of 100°F/hr. Pressure change from 1000 psig thence to 1350 psig , thence to 240 psig and return to 1000 psig.	Emergency/faulted	1
9.	System temperature changes, greater than 30°F, from 561°F to 500°F within 7 min. and from 500°F to 400°F and return to normal operating temperature at 546°F at a rate of 100°F/hr. Pressure change from 1000 psig to 1180 psig to 240 psig and return to normal operating of 1000 psig.	Emergency/faulted	10

Paragraph NB3552 of ASME III code excludes various transients and provides means for combining those which are not excluded. Review and approval of the equipment supplier's certified calculation provides assurance of proper accounting of the specified transients.

3.9.1.1.10 Recirculation Flow Control Valve Transients

Not applicable since Susquehanna SES has no flow control valve.

3.9.1.1.11 Recirculation Pump Transients

The following transients are listed in the design specification as a requirement for design considerations. However, a submitted certified analysis considering thermal stresses was not required. The vendor was required to submit a certification of compliance. The submitted certified design calculations only considered pressure transient. Nozzle piping loads were considered in accordance with the following paragraph:

> "The pump case was designed to withstand secondary stresses due to piping reactions in accordance with Paragraph 452.4b of the ASME Standard Code for Pumps and Valves for Nuclear Power (1968 Draft)."

Transient		Category	Cycles
1.	Heatup and cooldown at 100°F/hr	Normal/upset	300
2.	±29°F temperature changes	Normal/upset	600
3.	±50°F temperature changes	Normal/upset	200
4.	RPV pressure transients to 110% design pressure	Normal/upset	1

5.	SRV blowdowns	Emergency	8
6.	Improper pump startup, 100°F to 552°F in 15 sec	Emergency	1
7.	Cooling transient 552°F to 281°F in 15 sec	Faulted	1
8.	Hydrotest to 1300 psig	Testing	130
9.	Hydrotest to 1670 psig	Testing	3

3.9.1.1.12 Recirculation Gate Valve Transients

The following transients are considered in the design of the recirculation gate valves.

Transient		Cycles
1.	50°F - 575°F - 50°F at 100°F/hr	300
2.	±29°F between limits of 50°F and 575°F, instantaneous	600
3.	+50°F between limits of 50°F and 546°F, instantaneous	200
4.	546°F to 375°F, instantaneous	30
5.	546°F to 281°F, instantaneous	2
6.	130°F to 546°F, instantaneous	1
7.	110% design pressure at 575°F	1
8.	1300 psi at 100°F installed hydrostatic test	130
9.	1670 psi at 100°F installed hydrostatic test	3

3.9.1.2 Computer Programs Used in Analysis

The following sections discuss computer programs used in the analysis of specific NSSS components. (Computer programs were not used in the analysis of all components, thus, not all components are listed.) The NSSS programs can be divided into two categories.

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.

GE Programs

The verification of the following GE programs has been performed in accordance with the requirements of 10CFR50, Appendix B. Evidence of the verification of input, output, and methodology is documented in GE Design Record Files.

(a)	MASS	(i)	TSFOR	(q)	DYSEA
(b)	SNAP (MULTISHELL)	(j)	EZPYP	(r)	SPECA
(c)	HEATER	(k)	PDA	(s)	SEISM
(d)	SAP4	(l)	SAP4G		
(e)	ANSI7	(m)	FTFLG01		
(f)	LUGSTR	(n)	ANSYS		
(g)	PISYS	(o)	POSUM		
(h)	RVFOR	(p)	BILRD		

Vendor Programs

The verification of the following CB&I programs is assured by contractual requirements between GE and the vendor. Per the requirements, the quality assurance procedure of these proprietary programs used in the design of N-stamped equipment is in full compliance with 10CFR50, Appendix B.

(a)	711 GENOZZ	(i)	928 TGRV
(b)	948 NAPALM	(j)	962 E0962A
(c)	1027	(k)	984
(d)	846	(l)	992 GASP
(e)	781 KALNINS	(m)	1037 DUNHAM'S
(f)	979 ASFAST	(n)	1335
(g)	766 TEMAPR	(o)	1606 & 1647 HAP
(h)	767 PRINCESS	(p)	1634 N

A list of computer programs used in the BOP system components is provided in Table 3.9-5. This list consists of computer programs that are developed and/or owned by Bechtel Power Corporation, and computer programs that are recognized and widely used in industry.

The Bechtel developed and/or owned computer programs are documented, verified, and maintained by Bechtel and meet the requirements of 10CFR50, Appendix B. A brief description of each of these programs is provided in Appendix 3.9A.

3.9.1.2.1 Reactor Vessel and Internals

The computer programs used in the preparation of the reactor vessel stress report are identified and their use summarized in the following paragraphs.

3.9.1.2.1.1 Reactor Vessel

3.9.1.2.1.1.1 CB&I Program 711 "GENOZZ"

The GENOZZ computer program is used to proportion barrel and double taper type nozzle configurations to comply with the specifications of the ASME Code, Section III and contract documents. The program will either design such a configuration or analyze the configuration input into it. If the input configuration will not comply with the specifications, the program will modify the design and redesign it to yield an acceptable result.

3.9.1.2.1.1.2 CB&I Program 948 - "NAPALM"

The basis for the program NAPALM, Nozzle Analysis Program-All Loads Mechanical, is to analyze nozzles for mechanical loads and find the maximum stress intensity and location. The program analyzes at specified locations from the point of application of the mechanical loads. At each location the maximum stress intensity is calculated for both the inside and outside surfaces of the nozzle. The program gives the maximum stress intensity for both the inside and outside surfaces of the nozzle as well as its angular location around the circumference of the nozzle from the 0° reference location. The principle stresses are also printed. The stresses resulting from each component of loading (bending, axial, shear, and torsion) are printed, as well as the loadings which caused these stresses.

3.9.1.2.1.1.3 CB&I Program 1027

This program is a computerized version of the analysis method contained in the "Welding Research Council Bulletin F107," December, 1965.

Part of this program provides for the determination of the shell stress intensities (S) at each of four cardinal points at both the upper and lower shell plate surfaces (ordinarily considered outside and inside surfaces) around the perimeter of a loaded attachment on a cylindrical or spherical vessel. With the determination of each S, there is also determined the components of that S (2 normal stresses, and one shear stress). This program provides the same information as the manual computation and the input data is essentially the input of the geometry of the vessel and attachment.

3.9.1.2.1.1.4 CB&I Program 846

This program computes the required thickness of a hemispherical head with a large number of circular parallel penetrations by means of the area replacement method in accordance with the ASME Code, Section III.

In cases where the penetration has a counterbore, the thickness is determined so that the counterbore does not penetrate the outside surface of the head.

3.9.1.2.1.1.5 CB&I Program 781 - "KALNINS"

This program is a thin elastic shell program for shells of revolution. This program was developed by Dr. A. Kalnins of Lehigh University. Extensive revisions and improvements have been made by Dr. J. Endicott to yield the CB&I version of this program.

The basic method of analysis was published by Professor Kalnins in the Journal of Applied Mechanics, Volume 31, September, 1964, pages 467 through 476. The KALNINS thin shell program (Program 781) is used to establish the shell influence coefficient and to perform detail stress analysis of the vessel. The stresses and the deformations of the vessel can be computed for any combination of the following axi-symmetric loading:

a) Preload condition

b) Internal pressure

c) Thermal load

3.9.1.2.1.1.6 CB&I Program 979 - "ASFAST"

ASFAST Program (Program 979) performs the stress analysis of axi-symmetric, bolted closure flanges between head and cylindrical shell.

3.9.1.2.1.1.7 CB&I Program 766 - "TEMAPR"

This program will reduce any arbitrary temperature gradient through the wall thickness to an equivalent linear gradient. The resulting equivalent gradient will have the same average temperature and the same temperature-moment as the given temperature distribution. Input consists of plate thickness and actual temperature distribution. The output contains the average temperature and total gradient through the wall thickness. The program is written in FORTRAN IV language.

3.9.1.2.1.1.8 CB&I Program 767 - "PRINCESS"

The PRINCESS computer program calculates the maximum alternating stress amplitudes from a series of stress values by the method in Section III of the ASME Pressure Vessel Code.

3.9.1.2.1.1.9 CB&I Program 928 - "TGRV"

The TGRV program is used to calculate temperature distributions in structures or vessels. Although it is primarily a program for solving the heat conduction equations, some provisions have been made for including radiation and convection effects at the surfaces of the vessel. The TGRV program is a greatly modified version of the TIGER heat transfer program written about 1958 at Knolls Atomic Power Laboratory, by A. P. Bray. There have been many versions of TIGER in existence including TIGER II, TIGER II B, TIGER IV, and TIGER V, in addition to TGRV. This manual has been written for use with CB&I's version of TGRV.

This program utilizes an electrical network analogy to obtain the temperature distribution of any given system as a function of time. The finite difference representation of the three-dimensional equations of heat transfer are repeatedly solved for small time increments and continually summed. Linear mathematics are used to solve the mesh network for every time interval. Included in the analysis are the three basic forms of heat transfer, conduction, radiation and convection, as well as internal heat generation.

Given any odd-shaped structure, which can be represented by a three dimensional field, its geometry and physical properties, boundary conditions, and internal heat generation rates, TGRV will calculate and give as output the steady state or transient temperature distributions in the structure as a function of time.

3.9.1.2.1.1.10 CB&I Program 962 - "EO962A"

Program E0962A is one of a group of programs (E0953A, E1606A, E0962A, E0992N, E1037N, and E0984N) which are used together to determine the temperature distribution and stresses in pressure vessel components by the finite element method.

Program E0962A is primarily a plotting program. Using the nodal temperatures calculated by program E1606A or Program E0928A, and the node and element cards for the finite element model, it calculates and plots lines of constant temperature (isotherms). These isotherm plots are used as part of the stress report to present the results of the thermal analysis. They are also very useful in determining at which points in time the thermal stresses should be determined.

In addition to its plotting capability the program can also determine the temperatures of some of the nodal points by interpolation. This feature of the program is intended primarily for use with the compatible TGRV and finite element models that are generated by program E0953A.

3.9.1.2.1.1.11 CB&I Program 984

Program 984 is used to calculate the stress intensity of the stress differences, on a component level, between two different stress conditions. The calculation of the stress intensity of stress component differences (the range of stress intensity) is required by Section III of the ASME Code.

3.9.1.2.1.1.12 CB&I Program 992 - GASP

The GASP computer program, originated by Prof. E. L. Wilson of the University of California at Berkeley, uses the finite element method to determine the stresses and displacements of plane or axi-symmetric structures of arbitrary geometry and is written in FORTRAN IV. For a detailed account, see the following reference document.

> Wilson, E. L.; "A Digital Computer Program for the Finite Element Analysis of Solids with Non-Linear Material Properties," Aerojet General Corporation, Sacramento, California. Technical Memorandum No. 23, July 1965.

As mentioned above, the program determines the stresses and displacements of plane or axi-symmetric structures using the finite element method. The structures may have arbitrary geometry and have linear or non-linear material properties. The loadings may be thermal, mechanical, accelerational, or a combination of these.

The structure to be analyzed is broken up into a finite number of discrete elements or "finite-elements" which are interconnected at finite number of "nodal-points" or "nodes." The actual loads on the structure are simulated by statically equivalent loads acting at the appropriate nodes. The basic input to the program consists of the geometry of the stress-model and the boundary conditions. The program then gives the stress components at the center of each element and the displacements at the nodes, consistent with the prescribed boundary conditions.

3.9.1.2.1.1.13 CB&I Program 1037 - "DUNHAM'S"

DUNHAM'S program is a finite ring element stress analysis program. It will determine the stresses and displacements of axi-symmetric structures of arbitrary geometry subjected to either axi-symmetric loads or non-axi-symmetric loads represented by a Fourier series.

This program is similar to the GASP program (CB&I 992). The major differences are that DUNHAM'S can handle non-axi-symmetric loads (which requires that each node have three degrees of freedom) and the material properties for DUNHAM'S must be constant. As in GASP, the loadings may be thermal, mechanical, and accelerational.

3.9.1.2.1.1.14 CB&I Program 1335

To obtain stresses in the shroud support, the baffle plate must be made a continuous circular plate. The program makes this modification and allows the baffle plate to be included in CB&I program 781 as two isotropic parts and an orthotropic portion at the middle (where the diffuser holes are located).

3.9.1.2.1.1.15 CB&I Programs 1606 and 1657 - "HAP"

The HAP program is an axi-symmetric nonlinear heat analysis program. It is a finite element program and is used to determine nodal temperatures in a two-dimensional or axi-symmetric body subjected to transient disturbances. Programs 1606 and 1657 are identical except that 1606 has a larger storage area allocated and can thus be used to solve larger problems. The model for program 1606 is compatible with CB&I stress programs 992 and 1037.

3.9.1.2.1.1.16 CB&I Program 1634N

This program is used to analyze thin cylindrical shells subjected to local loading beyond the range where Bijloard's curves are directly applicable, i.e., R/t >300.

This program computes stresses and displacements in thin walled elastic cylindrical shells subjected to mechanical loading such as radial loads, longitudinal and circumferential moments.

3.9.1.2.1.2 Reactor Internals

The following programs are used in the analysis of core support structures and other safety related reactor internals: MASS, SNAP (MULTISHELL), and HEATER. These programs are described in detail in Section 4.1.

3.9.1.2.2 Piping

The computer programs used in the analysis of NSSS piping systems are identified and summarized below:

3.9.1.2.2.1 Structural Analysis Program SAP 4

The Structural Analysis Program SAP 4, for the static and dynamic analysis of linear structural system is the result of several years research and development experience. The program has proven to be a very flexible and efficient analysis tool. The first version of the SAP Program was published in September, 1970. An improved static analysis program, namely SOLID SAP, or SAP 2, was presented in 1971. Work was then started on a new static and dynamic analysis program. The program SAP 3 was released towards the end of 1972. SAP 4 has the additional analysis capability of out-of-core direct integration for the time history analysis.

The structural systems to be analyzed may be composed of combinations of a number of different structural elements. The program presently contains the following element types:

a) three-dimensional truss element

b) three-dimensional beam element

c) plane stress and plane strain element,

d) two-dimensional axisymetric solid,

e) three-dimensional solid,

f) thick shell element,

g) thin plate or thin shell element,

h) boundary element,

i) pipe element (tangent and bend).

These structural elements can be used in a static or dynamic analysis. The capacity of the program depends mainly on the total number of nodal points in the system, the number of eigenvalues needed in the dynamic analysis and the computer used. There is practically no restriction on the number of elements used, the number of load cases or the order and bandwidth of the stiffness matrix. Each nodal point in the system can have from zero to six displacement degrees of freedom. The element stiffness and mass matrices are assembled in condensed form; therefore, the program is equally efficient in the analysis of one-, two-, or three-dimensional systems.

The formation of the structure matrices is carried out in the same way in a static or dynamic analysis. The static analysis is continued by solving the equations of equilibrium followed by the computation of element stresses. In a dynamic analysis the choice is between:

a) frequency calculations only,

b) frequency calculations followed by response history analysis,

c) frequency calculations followed by response spectrum analysis,

d) response history analysis by direct integration.

To obtain the frequencies and vibration mode shapes, solution routines are used which calculate the required eigenvalues and eigenvectors directly without a transformation of the structure stiffness matrix and mass matrix. This way the program operation and necessary input data for a dynamic analysis is a simple addition to what is needed for a static analysis.

3.9.1.2.2.2 Component Analysis/ANSI 7

Application. The ANSI 7 Computer Program determines stress and accumulative usage factors for thermal weight, seismic relief valve lift and turbine stop valve closure (as applicable) conditions of loadings derived from the Structural System Analysis in accordance with NB-3600 of ASME Code Section III.

Program Organization. For Class 1 ASME Code stress analysis, the program generates and prints hoop, bending, thermal discontinuity, linear temperature gradient and nonlinear temperature gradient components of stress for each equation of subarticle NB-3600 of Section III. Load combination results from possible load sets for Class 1 stress equations. The total stress (sum of component stresses) and the stress ratio (total stress divided by appropriate stress intensity limit) is printed for each Class 1 equation. The total stress (sum of the component stresses) and the stress ratio (total stress divided by the appropriate stress intensity limit) is printed for each one of the equations 9, 10, 12, and 13 of NB-3600. The alternating stresses and usage factor are calculated per NB-3653.6.

3.9.1.2.2.3 Integral Attachment/LUGSTR

The computer program "LUGSTR" was prepared to evaluate the stress in the pipe wall that are produced by loads applied to the integral attachments. The program was prepared based on the Welding Research Council Bulletin 198 including the evaluation due to stress range and fatigue analysis.

3.9.1.2.2.4 Piping Analysis Program/PISYS

PISYS is a computer code specialized for piping load calculations. It utilizes selected stiffness matrices representing standard piping components, which are assembled to form a finite element model of a piping system. The technique relies on dividing the pipe model into several discrete substructures, called pipe elements, which are connected to each other via nodes called pipe joints. It is through these joints that the model interacts with the environment, and loading of the structure becomes possible. PISYS is based on the linear classical elasticity in which the resultant deformation and stresses are proportional to the loading, and the superposition of loading is valid.

PISYS has a full range of static and dynamic analysis options which include distributed weight, thermal expansion, differential support motion modal extraction, response spectra, and time history analysis by modal or direct integration. The PISYS program has been benchmarked against five Nuclear Regulatory Commission piping models for the option-of-response-spectrum analysis and the results are documented in a report to the Commission, "PISYS Analysis of NRC Problem," NEDO-24210, August, 1979.

3.9.1.2.2.5 Relief Valve Discharge Pipe Forces/RVFO R

The relief valve discharge pipe connects the relief valve to the suppression pool. When the valve is opened, the transient fluid flow causes time dependent forces to develop in the pipe wall. This computer program computes the transient fluid mechanics and the resultant pipe forces using the method of characteristics.

3.9.1.2.2.6 Turbine Stop Valve Closure/TSFOR

The TSFOR program computes the time history forcing function in the main steam piping due to turbine stop valve closure. The program utilizes the method of characteristics to compute fluid momentum and pressure loads at each change in pipe section or direction.

3.9.1.2.2.7 Piping Analysis Program/EZPYP

EZPYP links the ANSI-7 and SAP program together. The EZPYP program can be used to run several SAP cases by making user specified changes to a biasic SAP pipe model. By controlling files and SAP runs the EZPYP program gives the analyst the capability to perform a complete piping analysis in one computer run.

3.9.1.2.2.8 Pipe Whip Analysis/PDA

The pipe whip analysis was performed using the PDA computer program. PDA is a computer program used to determine the response of a pipe subjected to the thrust force occurring after a pipe break. The program treats the situation in terms of generic pipe break configuration, which involves a straight, uniform pipe fixed at one end and subjected to a time-dependent thrust-force at the other end. A typical restraint used to reduce the resulting deformation is also included at a location between the two ends. Nonlinear and time-independent stress-strain relations are used for the pipe and the restraint. Similar to the popular plastic-hinge concept, bending of the pipe is assumed to occur only at the fixed end and at the location supported by the restraint.

Shear deformation is also neglected. The pipe bending moment-deflection (or rotation) relation used for these locations is obtained from a static nonlinear cantilever beam analysis. Using the moment-rotation relation, nonlinear equations of motion of the pipe are formulated using an energy consideration and the equations are numerically integrated in small time steps to yield time-history information of the deformed pipe.

3.9.1.2.3 Recirculation and ECCS Pumps and Motors

3.9.1.2.3.1 Recirculation Pumps

No computer programs were used in the design of the recirculation pumps.

3.9.1.2.3.2 ECCS Pumps and Motors

3.9.1.2.3.2.1 Structural Analysis Program/SAP4G

SAP4G is used to analyze the structural and functional integrity of the ECCS pump/motor systems. This is a general structural analysis program for static and dynamic analysis of linear elastic complex structures. The finite element displacement method is used to solve the displacements and stresses of each element of the structure. The structure can be composed of unlimited number of three-dimensional truss, beam, plate, shell, solid, plate strain-plane stress and spring elements that are axisymmetric. The program can treat thermal and various forms of mechanical loading. The dynamic analysis includes mode superposition, time history, and response spectrum analysis. Seismic loading and time-dependent pressure can be treated. The program is versatile and efficient in analyzing large and complex structural systems. The output contains displacements of each nodal point as well as stresses at the surface of each element.

3.9.1.2.3.2.2 Effects of Flange Joint Connections/FTFLG01

The flange joints connecting the pump bowl castings are analyzed using FTFLG01. This program uses the local forces and moments determined by SAP4G to perform flat flange calculations in accordance with the rules set forth in Appendix II and Section III of the ASME Code.

3.9.1.2.3.2.3 Structural Analysis of Discharge Head/ANSYS

ANSYS is used to analyze the pump discharge head flange and bolting taking into account of the prying action developed by the flag face contact surface. The program is described in detail in 3.12.

3.9.1.2.3.2.4 Beam Element Data Processing/POSUM

POSUM is a computer code designed to process SAP generated beam element data for pump or heat exchanger models. The purpose is to determine the load combination that would produce the maximum stress in a selected beam element. It is intended for use on RHR heat exchangers with four nozzles or ECCS pumps with two nozzles.

3.9.1.2.4 RHR Heat Exchangers

3.9.1.2.4.1 Structural Analysis Programs/SAP4G

SAP4G is used to analyze the structural and functional integrity of the RHR heat exchangers. The description of this program is provided in Subsection 3.9.1.2.3.2.1.

3.9.1.2.4.2 Shell Attachment Parameters and Coefficients/BILRD

BILRD is used to calculate the shell attachment parameters and coefficients utilized in the stress analysis of the support to shell junction. The method, per Welding Research Council Bulletin No. 107, is implemented to calculate local membrane stress due to the support reaction loads on the heat exchanger shell.

3.9.1.2.4.3 Beam Element Data Processing/POSUM

POSUM is used to process SAP generated beam element data. The description of this program is provided in Subsection 3.9.1.2.3.2.4.

3.9.1.2.5 Dynamic Loads Analysis

3.9.1.2.5.1 Dynamic Analysis Program/DYSEA

DYSEA simulates a beam model in the annulus pressurization dynamic analysis. A detailed description of DYSEA is provided in Section 4.1. DYSEA employs a preprocessor program names GZAPL. GZAPL converts pressure time histories into time varying loads and forcing functions for DYSEA. The overall resultant forces and moments time histories at specified points of resolution can also be obtained from GZAPL.

3.9.1.2.5.2 Acceleration Response Spectrum Program/SPECA

SPECA generates acceleration response spectrum for an arbitrary input time history of piece-wise linear accelerations, i.e., to compute maximum acceleration responses for a series of single-degree-of-freedom systems subjected to the same input. It can accept acceleration time histories from a random file. It also has the capability of generating the broadened/enveloped spectra when the special points are generated equally spaced on a logarithmic scale axis of period/frequency. This program is also used in seismic and SRV transient analyses.

3.9.1.2.5.3 Fuel Support Loads Program/SEISM

SEISM02 computes the vertical fuel support loads using the component element methods in dynamics. The methodology is based on the publication "The Component Element Method in Dynamics," by S. Levy and J. P. D. Wilkinson, McGraw Hill Co., New York, 1976.

3.9.1.3 Experimental Stress Analysis

When experimental stress analysis is used in lieu of analytical methods for Seismic Category I ASME Code items, the requirements for experimental testing enumerated in the ASME Code applicable for the specific components under test are applied. When testing is required for Seismic Category I non-ASME Code items, account is taken of the effects of differences in size, dimensional tolerances, and ultimate strength (or other governing material properties) between the actual and tested parts to assure that the loads obtained from tests are realistic or conservative representation of the capability of the actual structure.

The following subsections in this section list the components upon which experimental stress analysis was used.

3.9.1.3.1 Experimental Stress Analysis of NSSS Piping Components

The following components have been tested to verify their design adequacy:

a) Snubbers

b) Pipe whip restraints

Descriptions of the snubber and whip restraint tests are contained in Subsections 3.9.3.4 and 3.6, respectively.

3.9.1.3.2 Seismic Category I Items Other Than NSSS

Experimental testing is performed in the qualification and acceptance of snubbers, compensating struts, and honeycomb material used in energy absorbing components for pipe break.

3.9.1.4 Considerations for the Evaluation of Faulted Conditions

All Seismic Category I equipment in the NSSS is evaluated for the faulted loading conditions. However, emergency stress limits rather than faulted stress limits were used in many cases. In essentially all cases, calculated stresses are within allowable limits. The following paragraphs in subsection 3.9.1.4 show examples of the treatment of faulted conditions for the major components on a component by component basis. Additional discussion of faulted analysis can be found in Subsections 3.9.3 and 3.9.5, and Table 3.9-2.

Subsection 3.9.2.2 and Section 3.7 discuss the treatment of dynamic loads resulting from the postulated seismic and hydrodynamic events. Section 3.9.2.5 discusses the dynamic analysis of loads affected on reactor internals equipment resulting from blowdown. Deformations under faulted conditions have been evaluated in critical areas and no cases are identified where design limits, such as clearance limits, are violated.

3.9.1.4.1 Control Rod Drive System Components

3.9.1.4.1.1 Control Rod Drives

The ASME Code components of the CRD have been analyzed for abnormal conditions shown in Subsection 3.9.1.1.1.

The load criteria, calculated and allowable stresses for various operating conditions are summarized in Table 3.9-2(u).

The design adequacy of non ASME code components of the CRD has been verified by analysis and extensive testing programs on both component parts, specially instrumented prototype drives and production drives. The testing has included postulated abnormal events as well as the service life cycle listed in Subsection 3.9.1.1.1.

3.9.1.4.1.2 Hydraulic Control Unit

The Hydraulic Control Unit (HCU) was analyzed for the faulted condition.

The analysis of the HCU under faulted condition loads establishes the structural integrity of the system.

Section 3.9.2.2a.2.4 discusses the dynamic qualification of the HCU.

3.9.1.4.2 Standard Reactor Internal Components

3.9.1.4.2.1 CR Guide Tube

The maximum calculated stress on the CR Guide Tube occurs in the base during the faulted condition. The loading criteria and calculated and allowable stresses are summarized in Table 3.9-2aa.

3.9.1.4.2.2 Incore Housing

The maximum calculated stress on the Incore Housing occurs at the outer surface of the vessel penetration during the faulted condition. The loading criteria and calculated and allowable stresses are shown in Table 3.9-2ab.

3.9.1.4.2.3 Jet Pump

The maximum stress in the jet pump occurs in the faulted condition due to impulse loading of the diffuser during a pipe rupture and blowdown. Table 3.9-2w summarizes loading criteria, and calculated and allowable stresses.

3.9.1.4.2.4 Orificed Fuel Support

A series of vertical and horizontal load tests were performed on the orificed fuel support (OFS) in order to verify the design. Results from these tests indicate that the component and seismic loading of the OFS are well below the stress limit allowables with a safety margin of 1.26 for the normal and upset conditions, and 1.5 for the faulted condition. The allowable stress limits were arrived at by applying a .65 quality factor to the ASME Code allowables of 1.5 S for the upset condition, and 1.5 x .7 Su for the faulted condition.

3.9.1.4.2.5 Control Rod Drive Housing

The CRD Housing is analyzed for the faulted condition considering SSE and hydrodynamic loads. Table 3.9-2v shows that the calculated stresses are within the allowable stresses.

3.9.1.4.3 Reactor Pressure Vessel Assembly

The reactor pressure vessel, support skirt, and the shroud support were evaluated using elastic analysis methods for the faulted conditions. For the support skirt and shroud support an elastic analysis was performed and buckling was evaluated for the compressive load. Table 3.9-2a lists the loading criteria and calculated and allowable stresses for the various loading combinations.

3.9.1.4.4 Core Support Structure

The evaluations for faulted conditions for the core support structure are discussed in Subsection 3.9.5. The loading criteria and calculated and allowable stresses are summarized in Table 3.9-2b.

3.9.1.4.5 Main Steam Isolation, Recirculation Gate, and Safety/Relief Valves

Tables 3.9-2g, 3.9-2h, and 3.9-2j provide a summary of the analyses of the safety/relief, main steam isolation, and recirculation gate valves, respectively.

Standard design rules, as defined in the ASME Code, Section III, are utilized in the analysis of pressure boundary components of Seismic Category I valves. Conventional elastic stress analysis is used to evaluate components not defined in the ASME code. The code allowable stresses are applied to determine acceptability of structure under applicable loading conditions including faulted condition.

3.9.1.4.6 Main Steam and Recirculation Piping

For Main Steam and Recirculation System piping, elastic analysis methods are used for evaluating faulted loading conditions. The equivalent allowable stresses using elastic techniques are obtained from ASME Code Section III, Appendix F, "Rules for Evaluation of Faulted Conditions," and these are above elastic limits. Additional information on the main steam and recirculation piping and pipe-mounted equipment is in Table 3.9-2d and 3.9-2e.

3.9.1.4.7 Nuclear Steam Supply System Pumps, Heat Exchanger, and Turbines

The recirculation, ECCS, RCIC, and SLC pumps, RHR heat exchangers and RCIC turbine have been analyzed for the faulted loading conditions identified in Subsection 3.9.1.1. In all cases, stresses were within the elastic limits. The analytical methods, stress limits, and allowable stresses are discussed in Subsections 3.9.2.2 and 3.9.3.1.

3.9.1.4.8 Control Rod Drive Housing Supports

Design adequacy of the CRD Housing Supports is shown by comparing the total static and dynamic loads to the original design loads. The comparison shows that the hydrodynamic loads and other dynamic loads combined by SRSS are less than the original design loads.

3.9.1.4.9 Fuel Storage Racks

The stress criteria, loadings, calculated stresses, and stress limits for the faulted conditions for the new fuel storage racks are shown in Table 3.9-2s. No inelastic stress analyses were used on these components.

Similar information for the spent fuel storage racks was provided by the rack vendor.

3.9.1.4.10 Fuel Assembly (Including Channels)

Seismic/ LOCA loading evaluations for channeled ATRIUM™-10 fuel and channeled Lead Use Assemblies are located in Section 4.2. (Fuel System)

3.9.1.4.11 Refueling Equipment

Refueling and servicing equipment that is important to safety is classified as essential equipment per the requirements of 10 CFR 50, Appendix A. This equipment and other equipment whose failure would degrade an essential component is defined in Section 9.1 and is classified as Seismic Category I. These components are subjected to an elastic dynamic finite

element analysis to generate loadings. This analysis utilizes appropriate seismic floor response spectra and combines loads at frequencies up to 33 Hz for seismic and up to 60 Hz for the hydrodynamic loads in three directions. Imposed stresses are generated and combined for normal, upset, and faulted conditions. Stresses are compared, depending on the specific safety class of the equipment, to Industrial Codes, ASME, ANSI or Industrial Standards, AISC, allowables. The calculated stresses and allowable limits for the faulted loads for the fuel preparation machine are provided in Table 3.9-2s. The refueling platform has also been examined; it can withstand the faulted loads due to seismic hydrodynamic events.

3.9.1.4.12 Seismic Category I Items Other than NSSS

For statically applied loads, the stress allowables of Appendix F of the ASME Code, Section III, Winter 1972 were used for code components. For non-code components, allowables were based on tests or accepted standards consistent with those in Appendix F of the code.

Dynamic loads for components loaded in the elastic range were calculated using dynamic load factors, time history analysis, or any other method that assumes elastic behavior of the component.

The limits of the elastic range are defined in Paragraph 1323 of Appendix F for the code components. The local yielding due to stress concentration is assumed not to affect the validity of the assumptions of elastic behavior. The stress allowables of Appendix F for elastically analyzed components were used for code components. For non-code components, allowables were based on tests or accepted material standards consistent with those in Appendix F for elastically analyzed components.

The methods used in evaluating the pipe break effects are discussed in Section 3.6.

3.9.2 DYNAMIC TESTING AND ANALYSIS

3.9.2.1a Preoperational Vibration and Dynamic Effects Testing on NSSS Piping

The test program is divided into three phases: preoperational vibration, startup vibration, and operational transients.

3.9.2.1a.1 Preoperational Vibration Testing

The purpose of the preoperational vibration test phase is to verify that operating vibrations in the recirculation piping are acceptable. This phase of the test uses visual observation.

3.9.2.1a.2 Small Attached Piping

There is no small attached piping in the NSSS scope of supply.

3.9.2.1a.3 Startup Vibration

The purpose of this phase of the program is to verify that the main steam and recirculation piping vibration are within acceptable limits. Because of limited access due to high radiation levels, no visual observation is made during this phase of the test. Remote measurements were made during the following steady state conditions:

a) Main steam flow at 25% of rated;

b) Main steam flow at 50% of rated;

c) Main steam flow at 75% of rated;

d) Main steam flow at 100% of rated.

3.9.2.1a.4 Operating Transient Loads

The purpose of the operating transient test phase is to verify that pipe stresses are within Code Limits. The amplitude of displacements and number of cycles per transient of the main steam and recirculation piping were measured and the displacements compared with acceptance criteria. The deflections are correlated with the calculated deflections to assure that the stresses remain within Code limits. Remote vibration and deflection measurements were taken during the following transients:

a) Recirculation pump starts;

b) Recirculation pump trip at 100% of rated flow;

c) Turbine stop valve closure at 100% power;

d) Manual discharge of representative S/R valves at 1,000 psig and at planned transient tests that result in S/R valve discharge.

3.9.2.1a.5 Test Evaluation and Acceptance Criteria

The piping response to test conditions shall be considered acceptable if the organization responsible for the stress report reviews the test results and determines that the tests verify that the piping responded in a manner consistent with the predictions of the stress report and/or that the tests verify that piping stresses are within Code limits. To insure test data integrity and test safety, criteria have been established to facilitate assessment of the test while it is in progress. These criteria, designated Level 1 and 2, are described in the following paragraphs.

3.9.2.1a.5.1 Level 1 Criterion

Level 1 establishes the maximum limits for the level of pipe motion which, if exceeded, makes a test hold or termination mandatory.

If the Level 1 limit is exceeded, the plant will be placed in a satisfactory hold condition, and the responsible piping design engineer will be advised. Following resolution, applicable tests must be repeated to verify that the requirements of the Level 1 limits are satisfied.

3.9.2.1a.5.2 Level 2 Criterion

Level 2 specifies the level of pipe motion which, if exceeded, requires that the responsible piping design engineer be advised. If the Level 2 limit is not satisfied, plant operating and startup testing plans would not necessarily be altered. Investigations of the measurements, criteria, and calculations used to generate the pipe motion limits would be initiated. An acceptable resolution must be reached by all appropriate and involved parties, including the responsible piping design engineer. Depending upon the nature of such resolution, the applicable tests may or may not have to be repeated.

3.9.2.1a.6 Corrective Actions

During the course of the tests, the remote measurements shall be regularly checked to determine compliance with the Level 1 criterion. If trends indicate that the Level 1 criterion may be violated, the measurements shall be monitored at more frequent intervals. The test will be held or terminated as soon as the criterion is violated. As soon as possible after the test hold or termination, the following corrective actions will be taken:

a) Installation Inspection. A walkdown of the piping and suspension will be made to identify any obstruction or improperly operating suspension components. If vibration exceeds criteria, the source of the excitation must be identified to determine if it is related to equipment failure. Action will be taken to correct any discrepancies before repeating the test.

b) Instrumentation Inspection. The instrumentation installation and calibration will be checked and any discrepancies corrected. Additional instrumentation will be added, if necessary.

c) Repeat Test. If actions (a) and (b) identify discrepancies that could account for failure to meet the Level 1 criterion, the test will be repeated.

d) Resolution of Findings. If the Level 1 criterion is violated on the repeat test or no relevant discrepancies are identified in (a) and (b), the organization responsible for the stress report shall review the test results and criteria to determine if the test can be safely continued.

If the test measurements indicate failure to meet the Level 2 criterion, the following corrective actions will be taken after completion of the test:

a) Installation Inspection. A walkdown of the piping and suspension shall be made to identify any obstruction or improperly operating suspension components. If vibration exceeds limits, the source of the vibration must be identified. Action, such as suspension adjustment, will be taken to correct any discrepancies.

b) Instrumentation Inspection. The instrumentation installation and calibration will be checked and any discrepancies corrected.

c) Repeat Test. If (a) and (b) above identify a malfunction or discrepancy that could account for failure to comply with Level 2 criterion and appropriate corrective action has been taken, the test may be repeated.

d) Documentation of Discrepancies. If the test is not repeated, the discrepancies found under actions (a) and (b) above shall be documented in the test evaluation report and correlated with the test condition. The test will not be considered complete until the test results are reconciled with the acceptance criteria.

3.9.2.1a.7 Measurement Locations

Remote shock and vibration measurements were made in the three orthogonal directions near the first downstream S/R valve on each steam line; and in the three orthogonal directions on the piping between the recirculation pump discharge and the first downstream valve. During preoperational testing prior to fuel load, visual inspection of the piping was made, and any visible vibration measured with a handheld instrument.

For each of the selected remote measurement locations, Levels 1 and 2 deflection and acceleration limits were prescribed in the startup test specification. Level 2 limits were based on the results of the stress report adjusted for operating mode and instrument accuracy; Level 1 limits were based on maximum allowable Code stress limits.

3.9.2.1b Preoperational Thermal Vibration and Dynamic Effects of Testing of Piping other than NSSS

The dynamic effects on all safety-related ASME Class 1, 2, and 3 piping systems, including their supports and restraints, are considered as required by NB-3622.3, NC-3622, and ND-3622 of Section III of ASME B&PV Code. The structural and functional integrity of the piping system is ensured under a postulated seismic event by dynamic analysis only. Piping systems having significant anticipated transients loads, e.g., main stop valve closure or relief valve blow for example, are analyzed for the time-dependent forces. In addition, piping steady state vibration and dynamic transient tests were performed as summarized below, to ensure that

a) Excessive steady state vibration is not present in the piping that would result in piping stresses and restraint loads above the allowables.

b) The piping is adequately restrained to withstand the dynamic transient loads.

Cognizant design personnel familiar with the systems to be tested developed the test plans, and evaluate the test results. Also the cognizant design personnel witnessed the test. The data acquired from the tests was compared with the expected results to determine the acceptability of the total system response.

A list of all piping systems in the BOP is provided in Table 3.9-20. ASME Section III Class 1, 2 and 3 piping systems, high energy piping systems, moderate energy piping systems, seismic Category I and seismic Category II systems are identified in the Table. The Table also identifies the tests to be performed for each system.

Piping thermal expansion tests are performed for the safety-related piping systems with normal operating temperature exceeding 300°F. Safety-related piping systems with normal operating temperature less than 300°F do not have enough significant thermal expansion to warrant thermal expansion tests. Engineering review of all seismic Category I piping systems, including their supports, restraints or snubbers, is performed after completion of construction and prior to fuel load to ensure that no restraint of normal thermal movement occurs due to interferences and obstructions, and that the support and restraints are in accordance with the design intent.

For the systems receiving thermal expansion tests, the pipe movements are monitored to ensure that no restraint of normal thermal movement occurs at locations other than at the designed restraint locations.

The thermal expansion test program verifies that the free thermal expansion of piping systems takes place at the snubbers by monitoring the thermal movement. Performance of the snubbers designed for transient loads such as due to Main Stop Valve Closure or Main Steam Relief Valve Discharge are verified by measuring the load in the snubber during the dynamic transient tests. The snubbers are qualified by dynamic testing for cyclic loading as described in Subsection 3.9.3.4.1.

The acceptance criterion for thermal expansion tests and dynamic transient tests is that the measured pipe displacements or restraint loads shall be below the calculated or design values.

The acceptance criterion for the steady-state vibration tests is:

Either

The maximum measured amplitude of the piping vibration shall not induce a stress in the pipe more than half the endurance limit of the material for B31.3 piping. The maximum stress in the pipe due to steady state vibration for class 1, 2 & 3 piping is limited to one-half of the endurance limit (allowable stress corresponding to 10^6 cycles in Appendix I of ASME Section III), the steady-state vibration-induced stress will not contribute to the reduction of fatigue life of piping.

Or

Acceptance criterion are divided into two categories, i.e., Level 1 and Level 2. If the Level 1 criterion is violated, the test must be placed on hold. If the Level 2 criterion is violated, the test can continue, but the measurements must be evaluated to verify that continued test operation will not result in exceeding piping fatigue requirements.

For steady state vibration the piping peak stress zero to peak due to vibration only (neglecting pressure) will not exceed 10,000 psi for the Level 1 criterion and 5,000 psi for the Level 2 criterion. These limits are below the piping material fatigue endurance limits as defined for 10^6 cycles in Appendix I of the ASME Code, Section III.

The Table 3.9-20 provides cross reference between the FSAR Section 3.9 and the appropriate test description in FSAR Chapter 14.

<u>3.9.2.1b.1 Piping Dynamic Transient Tests</u>

During the preoperational and/or startup testing, dynamic transient tests will be performed on the following piping for the indicated modes of operation.

a) Main steam piping outside the containment for main steam turbine stop valve closure at 30 percent ±10%, 75 percent ±10%, and 100 percent +0 -10% power.

b) Main steam bypass piping to the anchor near the bypass valves for the turbine stop valve closure.

c) Selected main steam safety/relief valve discharge piping for the main steam relief valve opening. The selected SRV discharge piping brackets all the SRV discharge piping.

d) HPCI turbine steam supply piping for HPCI turbine trip.

e) Feedwater system reactor feed pump trip/coastdown under operating conditions; Pump A only, B only, and C only and at normal pump flowrate.

From past experience, the dynamic transients in other piping systems are not significant.

Dynamic transient analysis of the subject lines has been performed to determine the response of the piping system and the restraint loads. During the test the displacement of the pipe, loads in the snubbers and restraints and pressure at representative locations will be measured.

Acceptance criteria for this test are that the measured loads in the snubbers and restraints shall be below the design values of the snubbers and restraints. In the case (e) the acceptance criteria is that the measured response shall be less than the acceptable response determined by analysis.

3.9.2.1b.2 Piping Steady State Vibration Testing

The piping system associated with the following components' operation will be observed for steady state vibration during preoperational test programs or power ascension:

a) RHR pump

b) HPCI pump and turbine

c) RCIC pump and turbine

d) Core spray pump

e) Main Steam

f) Feedwater

g) Reactor Water Cleanup

From experience on other nuclear power plants, the steady state vibration in other piping systems is not critical. However, abnormal vibrations of other systems during system walkdown on initial startup or power escalation will be noted and instrumented if necessary to determine the acceptability of such vibration.

Steady-state vibration in the subject piping systems is primarily induced by the flow in the pipe and the equipment motion. In general, the specific causes of the steady-state vibration is not known beforehand; therefore, design engineers with stress analysis experience and familiarity with the subject piping system will visually observe the lines or monitor inaccessible lines with suitable instrumentation during all significant modes of system operation and classify each line as acceptable if the vibration is not significant, or questionable if vibration is significant. The lines with questionable steady-state vibration will be monitored by suitable instrumentation to determine the system response.

The type of the instrumentation, if necessary, will be determined by the design engineer so that the maximum amplitude and frequency response of the piping system can be determined. The instrumentation will not screen out the significant frequencies.

For lines with questionable steady-state vibration, the acceptance criterion is as discussed in Subsection 3.9.2.1b.

When required, additional restraints will be provided to reduce the steady-state vibration and to keep the stresses below the acceptance criterion levels.

3.9.2.2a Seismic and Hydrodynamic Qualification of Safety-Related NSSS Mechanical Equipment

This subsection describes the criteria for dynamic qualification of mechanical safety-related equipment and also describes the qualification testing and/or analysis applicable to this plant for all the major components on a component-by-component basis. In some cases, a module or assembly consisting of mechanical and electrical equipment is qualified as a unit, for example, motor powered pumps. These modules are generally discussed in this paragraph rather than providing discussion of the separate electrical parts in Sections 3.10 and 3.11. Dynamic qualification testing for pumps and valves is also discussed in Subsection 3.9.3.2. Electrical supporting equipment such as control consoles, cabinets, and panels which are part of the NSSS are discussed in Subsection 3.10.

All safety related NSSS mechanical equipment located in the Containment and the Reactor and Control Buildings are qualified for the combined seismic and hydrodynamic vibratory loadings. Procedures for the assessment and requalification of safety-related NSSS mechanical equipment for the additional hydrodynamic loads are described in Section 7.1.6 of the Design Assessment Report (DAR.)

3.9.2.2a.1 Tests and Analysis Criteria and Methods

The ability of equipment to perform its safety-related function during and after the application of dynamic loads is demonstrated by tests and/or analysis. Selection of testing, or analysis, or a combination of the two is determined by the type, size, shape, and complexity of the equipment being considered. When practical, equipment operability is demonstrated by testing. Otherwise, operability is demonstrated by mathematical analysis.

Equipment which is large, simple, and/or consumes large amounts of power is usually qualified by analysis or test to show that the loads, stresses and deflections are less than the allowable maximum. Analysis and/or testing is also used to show there are no natural frequencies below 33 Hz for seismic loads and 60 Hz for hydrodynamic loads. If a natural frequency is discovered, dynamic tests may be conducted and in conjunction with mathematical analysis used to verify operability and structural integrity at the required dynamic input conditions. When the equipment is qualified by dynamic test, either the response spectrum or time history of the attachment point is used in determining input motion.

Natural frequency may be determined by running a continuous sweep frequency search using a sinusoidal steady-state input of low magnitude. Dynamic loading conditions are simulated by testing using random vibration input or single frequency input (within equipment capability) at frequencies through 35 Hz. Whichever method is used, the input amplitude during testing envelopes the actual input amplitude expected during dynamic loading conditions.

The equipment being dynamically tested is mounted on a fixture which simulates the intended service mounting and causes no dynamic coupling to the equipment.

Equipment having an extended structure, such as a valve operator, is analyzed by applying static equivalent dynamic loads at the center of gravity of the extended structure. In cases where the equipment structural complexity makes mathematical analysis impractical, a test is used to determine spring constant and operational capability at maximum equivalent dynamic loading conditions. Pipe-mounted equipment is analyzed in the piping system dynamic analysis.

3.9.2.2a.1.1 Test Input Motion

When random vibration input is used, the actual input motion envelopes the appropriate floor input motion at the individual modes. However, single frequency input, such as sine beats, can be used provided one of the following conditions are met:

a) The characteristics of the required input motion are dominated by one frequency.

b) The anticipated response of the equipment is adequately represented by one mode.

c) The input has sufficient intensity and duration to excite all modes to the required magnitude, such that the testing response spectra will envelope the corresponding response spectra of the individual modes.

3.9.2.2a.1.2 Application of Input Motion

When dynamic tests are performed, the input motion is applied to one vertical and one horizontal axis simultaneously. However, if the equipment response along the vertical direction is not sensitive to the vibratory motion along the horizontal direction, and vice versa, then the input motion is applied to one direction at a time. In the case of single frequency input, the time phasing of the inputs in the vertical and horizontal directions are such that a purely rectilinear resultant input is avoided.

3.9.2.2a.1.3 Fixture Design

The fixture design will simulate the actual service mounting and cause no dynamic coupling to the equipment.

3.9.2.2a.1.4 Prototype Testing

Equipment testing has been conducted on prototypes of the equipment installed in this plant.

3.9.2.2a.2 Seismic and Hydrodynamic Qualification of Specific Mechanical Components

The following sections discuss the testing or analytical qualification of NSSS equipment. Seismic and hydrodynamic qualification is also described in Subsections 3.9.1.4, 3.9.3.1, and 3.9.3.2.

3.9.2.2a.2.1 Jet Pumps

A dynamic analysis of the jet pumps was performed. The stresses resulting from the analysis are below the design allowables.

3.9.2.2a.2.2 CRD and CRD Housing

The dynamic qualification of the CRD housing (with enclosed CRD) was done analytically. The results of this analysis established the structural integrity of these components. Preliminary dynamic tests have been conducted to verify the operability of the Control Rod Drive during a dynamic event. A test was performed in which the CRD was shown to function satisfactorily, while a static bow in the fuel channels was used to simulate dynamic loading.

3.9.2.2a.2.3 Core Support (Orificed Fuel Support and CR Guide Tube)

A detailed analysis imposing dynamic effects due to seismic and hydrodynamic events has shown that the maximum stresses developed during these events are much lower than the maximum allowed for the component material.

3.9.2.2a.2.4 Hydraulic Control Unit (HCU)

The seismic and hydrodynamic load adequacy of the HCU for the faulted condition is demonstrated by test and analysis. With the HCU's mounted on a seismic support structure, the dynamic loading results from application of 3.0g vertical at the natural frequency of 7 to 30 Hz, and 1.0g horizontal at 2 to 6 Hz, and 5.0g horizontal at 10 Hz. At these frequencies, the maximum HCU capability demonstrated for dynamic loading is 20g vertical at 7 to 30 Hz, and greater than 4g horizontal at 2 to 6 Hz, and 8g horizontal at 10 Hz.

3.9.2.2a.2.5 Fuel Channels

Fuel channel loading is discussed in Section 3.9.1.4.10.

3.9.2.2a.2.6 Recirculation Pump and Motor Assembly

Calculations are made to determine that the recirculation pump and motor assembly are designed to withstand the specific static to seismic and hydrodynamic forces. The flooded assembly was analyzed as a free body supported by constant support hangers from the brackets on the motor mounting member with mechanical snubbers attached to brackets located on the pump case and the top of the motor frame.

Primary stresses due to horizontal and vertical dynamic forces were considered to act simultaneously and are conservatively added directly. Horizontal and vertical dynamic forces were applied at mass centers and equilibrium reactions determined for motor and pump brackets.

3.9.2.2a.2.7 ECCS Pump and Motor Assembly

Pump/motor assemblies were analyzed with static loading equivalent to seismic acceleration under faulted conditions since the natural frequencies are above 33 Hz. The maximum specified vertical and horizontal accelerations were applied simultaneously in the worst case

combination and the results of the analysis indicate that the pump is capable of sustaining the loadings without overstressing the pump components.

A motor of similar design has been dynamically qualified by a combination of static analysis and dynamic testing. The motor has been seismically qualified via dynamic testing in accordance with IEEE 344, 1975. The qualification test program included demonstration of startup and shutdown capabilities, as well as no load operability during seismic and hydrodynamic loading conditions.

3.9.2.2a.2.8 RCIC Pump Assembly

The barrel type RCIC pump is mounted on a large cross-section pedestal.

The RCIC pump assembly is qualified by analysis using static loading equivalent to seismic and hydrodynamic loading with the design operating loads and temperature. The results of this analysis confirm that the calculated stresses are substantially less than the allowable stresses.

3.9.2.2a.2.9 RCIC Turbine Assembly

The RCIC turbine is qualified by analysis using static loading equivalent to dynamic loading. The turbine assembly and its components were considered to be supported as designed. Horizontal/vertical accelerations were applied to the mass centers of gravity. The magnitude of the acceleration coefficients was 3.0g horizontal and 1.0g vertical. The results of the analysis indicate that the turbine assembly is capable of sustaining the above loadings without overstressing any components.

The turbine assembly is qualified by dynamic testing, in accordance with IEEE 344-1975. The qualification test program demonstrated start-up, steady-state operability, and shutdown capabilities.

3.9.2.2a.2.10 Standby Liquid Control Pump and Motor Assembly

The SLC positive displacement pump and motor are mounted on a common base plate which is qualified by static analysis using static loading equivalent to the dynamic loading with the design operating loads and temperature.

The results of this analysis confirm that the calculated stresses are substantially less than the allowable stresses.

3.9.2.2a.2.11 RHR Heat Exchangers

A three-dimensional finite element model of the RHR heat exchanger and its support was developed and analyzed using the response spectrum method to verify that the heat exchanger can withstand seismic and hydrodynamic loads. The same model was statically analyzed to evaluate the effects of the external piping loads and dead weight to ensure that the nozzle load criteria and stress limits are met. Critical location stresses were evaluated and found to be lower than the corresponding allowable values.

3.9.2.2a.2.12 Safety Relief Valves (SRV)

Three SRVs of the Susquehanna design were subjected to the following qualification test programs in order to demonstrate compliance with the performance requirements under the specified conditions.

1. Life Cycle Tests - These tests consisted of subjecting each of the prequalification production units to approximately 300 safety and relief actuations in order to verify acceptability of the design to meet the requirements for (a) set pressure, (b) opening and closing response time, (c) blowdown, (d) seat tightness, (e) achievement of rated-capacity flow lift (ASME) during each actuation, (f) proper reclosure after each actuation without a tendency to stick open, chatter, or resulting in disc oscillation, and (g) opening without any inlet pressure applied which requirement (h) simulates an emergency operability condition.

   Conditions such as operating temperature, pressure ramp rate, dynamic and static back-pressures, pneumatic operating pressure and solenoid voltage were varied to assure valve operability under normal and transient operating conditions. Upon completion of the tests, test units were disassembled and inspected. This test program established the qualified service life of the safety relief valve.

2. Seismic Tests - The test units were subjected to seismic tests to simulate the normal, upset, emergency, and faulted conditions.

   Post-OBE and post-SSE reference frame tests were performed to determine the operability effects due to repeated combinations of seismic simulations, nozzle loadings, temperature and pressure. These reference frame tests consisted of set pressure determination during safety actuation, response time determination during relief actuation, valve leakage, and an emergency operability test. These reference frame tests were performed with induced nozzle loads applied.

   In order to evaluate the design capability of the test unit, the OBE and SSE tests were repeated using a higher input level. The test conditions during these tests are shown in Table 5.2-3.

   After the seismic tests, the electro-pneumatic actuator assembly was removed from the test unit and subjected to post seismic reference frame tests, a negative pressure test, post-negative pressure reference frame tests, a postulated Loss of Coolant Accident (LOCA) environmental test, and a post-LOCA reference frame test and inspection.

3.9.2.2a.2.13 Standby Liquid Control Tank

The standby liquid control tank is a cylindrical tank 9 feet in diameter and 12 feet high bolted to the concrete floor. The Standby Liquid Control Tank is qualified by analysis for:

a) Stresses in the tank bearing plate

b) Belt stresses

c) Sloshing loads imposed at the natural frequency of sloshing, which is 0.58 Hz

d) Minimum wall thickness

e) Buckling

The results of the analysis confirm that the calculated stresses at the investigated locations are below the allowable stresses.

3.9.2.2a.2.14 Main Steam Isolation Valves

The main steam isolation valves were analyzed; representative models were statically tested to demonstrate operability at the specified faulted condition. Static testing consisted of loading the valve actuator mechanically to equivalent specified dynamic loading while valve closure was performed. Operation of the valve under simulated faulted conditions was demonstrated by this test.

3.9.2.2a.2.15 Main Steam Safety/Relief Valves

Due to the complexity of the SRVs and the performance requirements, the total assembly of the safety/relief valve (including electrical, pneumatic devices) was dynamically tested at accelerations equal to or greater than the combined specified SSE and hydrodynamic loading. Satisfactory operation of the valves was demonstrated during and after the test.

3.9.2.2a.2.16 HPCI Turbine

The HPCI turbine was qualified by static analysis equivalent seismic acceleration. The turbine assembly and its components were considered to be supported as designed, and loading equivalent to horizontal/vertical accelerations was applied to the center of mass. The results of the analysis indicate that the turbine assembly is capable of sustaining the loadings without overstressing any components. The turbine electronic governor assembly has been seismically qualified via dynamic testing, in accordance with IEEE 344-1975. The qualification test program demonstrated startup, steady state operability, and shutdown capabilities.

3.9.2.2a.2.17 HPCI Pump

The HPCI pump is a split body type comprising a booster pump and a main pump mounted on a common base plate. The pump assembly behaves as a rigid body; therefore, qualification by analysis was performed. Results are obtained by using acceleration forces acting simultaneously in two directions, one vertical and one horizontal and calculated using the square root of the sum of the squares method. The pump mass, support system, and accessory piping are shown by analysis to have a natural frequency less than 33 Hz.

3.9.2.2b Seismic Qualification Testing of Safety Related Non-NSSS Mechanical Equipment

All Non-NSSS Seismic Category I equipment has been designed to withstand simultaneously the horizontal and vertical accelerations caused by the OBE and the SSE, in conjunction with other applicable loads. All equipment classified as active have demonstrated through qualification that they will perform their design function before, during and after a design basis accident.

The criteria for the seismic qualification of non-NSSS mechanical and electrical equipment, with the exception of valves, valve operators other than relief valves and the equipment found in the

Diesel Generator 'E' Building, is contained in project specification No. 8856-G-10 for a seismic environment complemented by No. 8856-G-22 for a combined seismic and hydrodynamic environment. For the Diesel Generator 'E' facility, the criteria for seismic qualification of mechanical and electrical equipment is contained in project specification No. C-1041 and Cooper Energy Services Standard No. SD-140. The standard IEEE-344, "Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations", is referenced in the G-10 and C-1041 Specifications and is being used as a supplement to the G-10, G-22, and C-1041 Specifications in the individual equipment procurement documentation package. Specifications G-10, G-22, and C-1041 and Standard IEEE-344 address the requirements of the demonstration of the seismic adequacy of equipment by analysis and/or tests. NRC Regulatory Guide 1.100 Revision 1, August 1977 accepts the use of standard IEEE-344 with a few modifications. Table 3.9-18 shows the comparison of the specification G-10 with IEEE-344-1975.

Non-NSSS motor-and air-operated valves are addressed in Subsection 3.9.3.2b.2. Control valves are addressed in Section 3.10b.

The assessment and requalification of safety-related non-NSSS mechanical equipment for the additional hydrodynamic loads are described in Section 7.1.7 of the Design Assessment Report (DAR).

3.9.2.2b.1 Safety-Related and Safety-Impacted Mechanical Equipment Other than for the NSSS

3.9.2.2b.1.1 Dynamic Analysis Without Testing

Structural analysis without testing was used if structural integrity alone could ensure the intended design function. Equipment that falls into this category includes:

Safety-Related

a) Diesel oil storage tanks

b) Containment instrument gas accumulators

c) Suppression pool suction strainers

d) Nuclear safety/relief valves

e) Vacuum breakers

Safety-Impacted

f) Supports for air handling units

g) Diesel building supports for cranes

h) Reactor building supports for cranes

i) Fuel pool skimmer surge tanks

Rotational analysis without testing was used to qualify heavy rotating machinery where it had to be verified that deformations from seismic loading would not bind the rotating element so that the component could not perform its intended design function. Components that fall into this category include:

a) Diesel generators

b) Diesel oil transfer pumps

c) RHR service water pumps

d) Emergency service water pumps

e) Control room centrifugal water chiller pumps

Refer to Tables 3.9-16 and 3.9-17 for listings of dynamically qualified equipment.

3.9.2.2b.1.2 Dynamic Testing

The equipment subjected to dynamic testing are the hydrogen recombiners (NSD-E-JFW 1003 March 4, 1975) and rupture discs (Black Sivalls Bryson, January 3, 1977). The rupture discs are installed in the exhaust of the HPCI and RCIC turbines.

3.9.2.2b.2 Criteria

For dynamic analysis without testing the equipment listed under Subsection 3.9.2.2b.1.1, and for dynamically testing the rupture discs under Subsection 3.9.2.2b.1.2, the criteria are as follows.

Response Spectrum Curves

The appropriate response spectrum curves for the equipment in question were issued with the material requisition or the equipment specification, for OBE, SSE, LOCA and SRV (LOCA & SRV only when applicable). Response spectrum curves are based upon the seismic analysis of the supporting structure and represent the maximum seismic response, as a function of oscillator frequency, of an array of single degree of freedom damped oscillators at a particular location within the structure. Response spectrum curves, plotted in terms of acceleration versus frequency, correspond to various locations within the buildings and are identified with respect to the points noted on the mathematical model for each direction of vibration to be considered. This may include the vertical as well as both the north-south and the east-west horizontal directions. In addition, each response spectrum curve corresponds to a particular damping ratio, i.e., the ratio of damping of the single degree of freedom oscillators to critical damping. See Section 3.7 for the appropriate response spectrum curves.

Load Combinations and Allowable Stress Limits

Seismic Category I equipment has been designed to withstand the more severe of the following load combinations:

a) OBE Conditions

Gravity loads and operating loads (or Design Basis Accident loads, if applicable) including associated temperature and pressures combined by absolute sums with the dynamic seismic loading of the OBE. Allowable stresses in the structural steel portions may be increased to 125 percent of the allowable working stress limits as set forth in ASME Boiler and Pressure Vessel Code Section III, or other applicable industrial codes. The customary increase in normal allowable working stress due to an earthquake shall be used if, according to the appropriate code, it is less than 25 percent. Resulting deflections, misalignment or binding of parts, or effects on electrical performance (microphones, contact bounce, etc.) do not prevent operation of the equipment during or after the seismic disturbance.

b) SSE Conditions

Gravity loads and operating loads (or Design Basis Accident loads, if applicable), including associated temperatures and pressures combined by absolute sums with the dynamic seismic loading of the SSE. Allowable stresses in the structural portions may be increased to 150 percent of allowable working stress limits in accordance with the appropriate codes listed in (a); however, the stresses may not exceed 0.9 Fy in bending, 0.85 Fy for axial tension, and 0.5 Fy in shear, where Fy equals the material minimum yield stress at the design temperature. For equipment designed by the maximum shear stress theory, the difference between the maximum and minimum principal stresses will not exceed 0.9 Fy. The resulting deflections, misalignment, or binding of parts, or effects on electrical performance (microphones, contact bounce, etc.) will not prevent operation of the equipment during or after the seismic disturbance.

Prevention of Overturning and Sliding

Stationary equipment is designed to prevent overturning or sliding by using anchor bolts or other suitable mechanical anchoring devices. The effect of friction on the ability to resist sliding is neglected. The effect of upward vertical seismic loads on reducing overturning resistance is considered. Anchoring devices are designed in accordance with the requirements of Items a) and b) and the AISC Manual of Steel Construction. The proposed anchoring system is shown on the Seller's drawings so that the Buyer can provide the proper foundation.

Dynamic Testing

Seismic adequacy was established for the rupture discs by dynamically testing them to meet the criteria defined under a and b above. Actual testing of equipment was done with base connections simulating the actual installation in accordance with one of the following methods:

a) The equipment was subjected to an input excitation such that the measured response was equal to or greater than the specified design response.

b) The equipment was subjected to an input excitation whose response spectrum equaled or exceeded the specified response spectrum for that location.

Criteria for the Diesel Fuel Oil Storage Tanks

These tanks are buried below grade under a cover of 16 ft (8 ft for diesel 'E' fuel tank) of earth. Equivalent fluid pressure of soil is 110 lb/ft^3 (100 lb/ft^3 for diesel 'E' fuel tank).

Tanks and tank supports are designed to withstand an H-20 loading, according to AASHTO, applied above 16 ft (8 ft for diesel 'E' fuel tank) of saturated overburden. The H-20 loading acts simultaneously with normal fluid pressure. Tank walls and ends will not deflect more than 3 percent maximum under the most unfavorable loading conditions.

The diesel fuel oil storage tanks are designed, fabricated, tested, and stamped in accordance with the ASME Code, Section III, Class 3. The tanks, including vents and openings, are designed as underground atmospheric tanks in accordance with OSHA Section 1910.106.

Tanks and their supports are designated Seismic Category I, and are designed to resist the increased earth pressure from the OBE and the SSE. For the OBE, the lateral earth pressure is 90 psf (180 psf for diesel 'E' fuel tank), for the SSE, 180 psf (330 psf for diesel 'E' fuel tank). When combined with other normal operating conditions, the stresses are limited to 125 percent of the ASME Code, Section III allowable stresses for the OBE condition, and are limited to 90 percent of the material's yield stress for the SSE condition.

Tanks are designed to withstand external pressure resulting from being buried in ground having a water table surface at ground level when the tanks are empty. Hydraulic uplift forces on buried tanks are resisted by the weight of the empty tank and the foundation mat plus 16 ft (8 ft for diesel 'E' fuel tank) of saturated overburden.

3.9.2.3 Dynamic Response of Reactor Internals under Operational Flow Transients and Steady State Conditions

The major reactor internal components within the vessel are subjected to extensive testing. In addition, dynamic system analyses are conducted to describe and evaluate the flow-induced vibration phenomena resulting from normal reactor operation and from anticipated operational transients.

In general, the vibration forcing functions for operational flow transients and steady state conditions are not predetermined by detailed analysis. Special analysis of the response signals measured from reactor internals of many similar designs are performed to obtain the parameters which determine the amplitude and modal contributions in the vibration responses. These studies are useful for extrapolating the results from tests to components of similar design. This vibration prediction method is appropriate where standard hydrodynamic theory cannot be applied due to complexity of the structure and flow conditions. Elements of the vibration prediction method are outlined as follows:

1) Dynamic analysis of major components and subassemblies is performed to identify natural vibration modes and frequencies. The analysis models used for Seismic Category 1 structures are similar to those outlined in subsection 3.7.2.

2) Data from previous plant vibration measurements is assembled and examined to identify predominant vibration response modes of major components. In general, response modes are similar but response amplitudes vary among BWRs of differing size and design.

3) Parameters are identified which are expected to influence vibration response amplitudes among the several reference plants. These include hydraulic parameters such as velocity and steam flow rates, and structural parameters such as natural frequency and significant dimensions.

4) Correlation functions of the variable parameters are developed which, multiplied by response amplitudes, tend to minimize the statistical variability between plants. A correlation function is obtained for each major component and response mode.

5) Predicted vibration amplitudes for components of the prototype plant are obtained from these correlation functions, based on applicable values of the parameters for the prototype plant. The predicted amplitude for each dominant response mode is stated in terms of a range, taking into account the degree of statistical variability in each of the correlations. The predicted mode and frequency are obtained from the dynamic analyses of paragraph 1 above.

The dynamic modal analysis also forms the basis for interpretation of the prototype plant preoperational and initial startup test results (Subsection 3.9.2.4). Modal stresses are calculated and relationships are obtained between sensor response amplitudes and peak component

3.9.2.4 Confirmatory Flow-Induced Vibration Testing of Reactor Internals

Reactor internals were tested in accordance with provisions of Regulatory Guide 1.20, Revision 2, for Non-prototype Category I plants. The test procedure required operation of the recirculation system at rated flow with internals important to safety installed. Inspection for evidence of vibration, wear, or loose parts followed. Blade guides, incore instruments, neutron sources, dryer and fuel were not installed. Control rods were either not installed or fully withdrawn and prevented from inserting. The test duration was sufficient to subject critical components to at least 10^6 cycles of vibration during two-loop and single-loop operation of the recirculation system. At the completion of the flow test, the vessel head and shroud head were removed, the vessel was drained and major components will be inspected on a selected basis. The inspection covered all components which were examined on the prototype design, including the shroud, shroud head, core support structures, jet pumps, peripheral control rod drive guide tubes and peripheral in-core guide tubes. Access will be provided to the reactor lower plenum.

Reactor internals for Susquehanna are substantially the same as the internals design configurations that have been tested in prototype BWR/4 plants. Results of the prototype tests are presented in a Licensing Topical Report (Ref. 3.9-7). This report also contains additional information on the confirmatory inspection program.

A labyrinth seal, consisting of five circumferential grooves on each jet pump mixer at the slip joint interface with the jet pump diffuser collar, reduces leakage at the slip joint. Tests performed by General Electric Company (Reference 3.9-10) demonstrated that the labyrinth seals reduce leakage through the slip joints. However, SSES no longer credits the labyrinth seals with this function. Jet pumps are equipped with either Slip Joint Diffuser Rings or Slip Joint Clamps. Slip Joint Diffuser Rings reduce leakage and leakage induced vibration, and are

designed to clamp to the Diffuser Collar Ears holding the Ring in place. Slip Joint Clamps reduce Jet Pump vibration.

3.9.2.5 Dynamic System Analysis of the Reactor Internals Under Faulted Conditions

In order to assure that no significant dynamic amplification of load occurs as a result of the oscillatory nature of the blowdown forces, a comparison was made of the periods of the applied forces and the natural periods of the core support structures being acted upon by the applied forces. These periods were determined from a 12-node vertical dynamic model of the RPV and internals. Besides the real masses of the RPV and core support structures, account was made for the water inside the RPV.

The time-varying pressures are applied to the dynamic model of the reactor internals described above. Except for the dynamic model and the nature and locations of the forcing functions, the dynamic analysis method is identical to that described for seismic analysis and is detailed in Subsection 3.7.2.1.

Dynamic loads are combined by SRSS. The results are then combined with other static and steady state loads on an ABS basis to confirm the adequacy of design loads. The results of the dynamic analysis are summarized in Tables 3.9-2, 3.9-2b, 3.9-2w, and 3.9-2aa.

3.9.2.6 Correlations of Reactor Internals Vibration Test Results with the Analytical Results

Prior to initiation of the instrumented vibration test program for the prototype plant, extensive dynamic analyses of the reactor and internals are performed. The results of these analyses are used to generate the allowable vibration levels during the vibration test. The vibration data obtained during the test are analyzed in detail. The results of the data analysis, vibration amplitudes, natural frequencies and mode shapes, are then compared to those obtained from the theoretical analysis.

Such comparisons provide the analysts with added insight into the dynamic behavior of the reactor internals. The additional knowledge gained is utilized in the generation of the dynamic models for seismic and LOCA analyses for this plant. The models used for this plant are the same as those used for the vibration analysis of the prototype plant.

The vibration test data are supplemented by data from forced oscillation tests of reactor internal components to provide the analysis with additional information concerning the dynamic behavior of the reactor internals.

3.9.3 ASME CODE CLASS 1, 2, AND 3 COMPONENTS, COMPONENT SUPPORTS, AND CORE SUPPORT STRUCTURES

3.9.3.1 Loading Combinations, Design Transients, and Stress Limits

This section delineates the criteria for selection and definition of design limits and loading combinations associated with normal operation, postulated accidents, and specified seismic and hydrodynamic events for the design of safety-related ASME code components (except containment components, which are discussed in Section 3.8.)

This section also lists the major ASME Class 1, 2, and 3 equipment and associated pressure retaining parts on a component by component basis and identifies the applicable loadings, calculation methods, calculated stresses, and allowable stresses. Design transients for ASME Class 2 equipment are not addressed in this section. They are covered in Subsection 3.9.1.1. Seismic and hydrodynamic related loads are discussed in Subsections 3.9.2.2a, 3.9.2.2b and Section 3.7.

Table 3.9-2 is the major part of this section; it presents the loading combination, analytical methods (by reference or example) and also the calculated stress or other design values for the most critical areas of the ASME Code Class 1, 2 and 3 components, supports, and core support structures. These design values are also compared to applicable code allowables.

3.9.3.1.1 Plant Conditions

All events that the plant might credibly experience during a reactor year are evaluated to establish a design basis for plant equipment. These events are divided into four plant conditions. The plant conditions described in the following paragraphs are based on event probability (i.e., frequency of occurrence) and correlated design conditions defined in the ASME Boiler and Pressure Vessel Code, Section III.

3.9.3.1.1.1 Normal Condition

Normal conditions are any conditions in the course of System startup, operation in the design power range, normal hot standby (with main condenser available), and System shutdown other than Upset, Emergency, Faulted, or Testing.

3.9.3.1.1.2 Upset Condition

Any deviations from Normal Conditions anticipated to occur often enough that design should include a capability to withstand the conditions without operational impairment. The Upset Conditions include those transients which result from any single operator error or control malfunction, transients caused by a fault in a system component requiring its isolation from the system, and transients due to loss of load or power, or an operating basis earthquake. Hot standby with the main condenser isolated is an Upset Condition.

3.9.3.1.1.3 Emergency Condition

Those deviations from Normal Conditions which require shutdown for correction of the conditions or repair of damage in the RCPB. The conditions have a low probability of occurrence but are included to provide assurance that no gross loss of structural integrity will result as a concomitant effect of any damage developed in the system. Emergency condition events include, but are not limited to, transients caused by one of the following: a multiple valve blowdown of the reactor vessel; loss of reactor coolant from a small break or crack which does not depressurize the reactor system nor result in leakage beyond normal makeup system capacity, but which requires the safety functions of isolation of containment, and reactor shutdown; improper assembly of the core during refueling; and seizure of one recirculation pump.

3.9.3.1.1.4 Faulted Condition

Those combinations of conditions associated with extremely low probability, postulated events whose consequences are such that the integrity and operability of the system may be impaired to the extent that considerations of public health and safety are involved. Faulted conditions encompass events that are postulated because their consequences would include the potential for the release of significant amounts of radioactive material. These postulated events are the most drastic that must be designed against and thus represent limiting design bases. Faulted condition events include, but are not limited to, one of the following: a control rod drop accident, a fuel-handling accident, a main steam line break, a recirculation loop break, the combination of any pipe break plus the seismic motion associated with SSE and hydrodynamic loading plus a loss of offsite power, or the safe shutdown earthquake.

3.9.3.1.1.5 Correlation of Plant Conditions with Event Probability

The probability of an event occurring per reactor year associated with the plant conditions is listed below. This correlation can be used to identify the appropriate plant condition for any hypothesized event or sequence of events.

Plant Conditions	Event Encountered Probability per Reactor Year
Normal (planned)	1.0
Upset (moderate probability)	1.0 > p > 10-2
Emergency (low probability)	10-2 > p > 10-4
Faulted (extremely low probability)	10-4 > p > 10-6

3.9.3.1.1.6 Safety Class Functional Criteria

For any normal or upset design condition event, Safety Class 1, 2, and 3 equipment shall be capable of accomplishing its safety functions as required by the event and shall incur no permanent changes that could impair its ability to accomplish its safety functions as required by any subsequent design condition event.

For any emergency or faulted design condition event, Safety Class 1, 2, and 3 equipment shall be capable of accomplishing its safety functions as required by the event but repairs could be required to ensure its ability to accomplish its safety functions as required by any subsequent design condition event.

Functional capability of safety-related essential piping components will be assured using the criteria given in Enclosure 110-1 to NRC questions and the Rodabaugh criteria.

3.9.3.1.1.7 Compliance with Regulatory Guide 1.48

Regulatory Guide 1.48 was issued after the design of this plant was established. Compliance with this Regulatory Guide is addressed in Section 3.13.

3.9.3.1.2 Reactor Vessel Assembly

The reactor vessel assembly consists of the reactor pressure vessel support skirt, shroud support and shroud plate.

The reactor pressure vessel, vessel support skirt, and shroud support are constructed in accordance with Section III of the ASME Code. The shroud support consists of the shroud support plate and the shroud support cylinder and its legs. The reactor pressure vessel is an ASME Code Class I component. Complete stress reports on these components have been prepared in accordance with ASME requirements. Table 3.9-2a provides a summary of the stress criteria, load combinations, calculated and allowable stresses. The stress analysis performed for the reactor vessel assembly, including the faulted condition, were completed using elastic methods. The stress Load combinations and stress analyses for the core support structures and other reactor internals are discussed in Subsection 3.9.5.

3.9.3.1.3 Main Steam Piping

The main steam piping discussed in this paragraph includes that piping extending from the reactor pressure vessel to the outboard main steam isolation valve. This piping is designed in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Subsection NB-3600. The load combinations and stress criteria for the main steam piping and pipe-mounted equipment are shown in Table 3.9-2d.

The rules contained in Appendix F of ASME Code Section III will be used in evaluating faulted loading conditions independently of all other design and operating conditions. Stresses calculated on an elastic basis will be evaluated in accordance with F-1360.

3.9.3.1.4 Recirculation Loop Piping

This piping is designed in accordance with the ASME for the recirculation piping and pipe-mounted equipment Code Section III, Subsection NB-3600. The load combinations and allowables are shown in Table 3.9-2e. The rules contained in Appendix F of ASME Code Section III are used in evaluating faulted loading conditions, independently of all other design and operating conditions. Stresses calculated on an elastic basis are evaluated in accordance with F-1360.

3.9.3.1.5 Recirculation System Valves

The recirculation system valves are designed in accordance with the ASME Code, Section III, Class I, Subsection NB-3500. The discharge gate valves are required to close for LPCI flow injection. Loading combinations and other stress analysis information are presented in Table 3.9-2(j).

3.9.3.1.6 Recirculation Pump

In the design of the recirculation pumps, the ASME Code, Section VIII, Division 1, 1971 Edition with latest addenda was used as a guide in calculations made for determining the thickness of pressure-retaining parts, and in sizing the pressure-retaining bolting.

The pump vendor made calculations for the design of the pressure-containing components to include the determination of minimum wall thickness, allowable stress and pressures. The design calculations are shown in Table 3.9-2i.

Load, shear, and moment diagrams were constructed to scale, using live loads, dead loads, and calculated snubber reactions. Combined bending, tension and shear stresses were determined for each major component of the assembly, including the pump driver mount, motor flange bolting, and pump case.

Replacement pump cover gaskets have been upgraded by the pump Original Equipment Manufacturer (OEM) to eliminate the use of asbestos and to improve reliability. The replacement pump cover gaskets require a higher bolt preload than the original gaskets. The OEM prepared a Gasket Upgrade Design Report that concludes the pump subcomponents are acceptable and meet the ASME Code requirements.

The maximum combined tensile stress in the cover bolting was calculated using tensile stress from design pressure.

Combined primary stresses did not exceed 150 percent of the code allowable stress shown in Section VIII of the ASME Code, 1971 Edition.

These methods and calculations demonstrate that the pump will maintain pressure integrity at all times.

3.9.3.1.7 Standby Liquid Control (SLC) Tank

The SLC tank is designed in accordance with the ASME Code, Section III. A summary of the design calculations and stress criteria used are shown in Table 3.9-2m.

3.9.3.1.8 Residual Heat Removal Heat Exchangers

The RHR heat exchanger is designed in accordance with the ASME B&PV Code, Section III. The loading combinations considered and other stress analysis are presented in Table 3.9-2o.

3.9.3.1.9 RCIC Turbine

Although not under the jurisdiction of the ASME Code, the RCIC turbine has been designed and fabricated following the basic guidelines for an ASME Code Section III, Class 2 component.

The RCIC Turbine is surveillance tested according to current Technical Specifications.

Design conditions for the RCIC turbine include:

a) Auto Quick start per Technical Specification surveillance requirements.

b) Turbine Inlet - 1250 psig at saturated temperature

c) Turbine Exhaust - 165 psig at saturated temperature

Table 3.9-9 contains a summary of the RCIC turbine components calculated and allowable loads.

3.9.3.1.10 RCIC Pump

The RCIC pump has been designed and fabricated to the requirements for an ASME Code Section III Class 2 component.

The RCIC pump is surveillance tested in conjunction with the RCIC turbine. Surveillance testing is performed according to current Technical Specification surveillance requirements. Design conditions for the RCIC pump include:

a)	Maximum NPSHR -		21.3 feet
b)	Total head, maximum	High speed Low speed	3060 feet 525 feet
c)	Constant flow rate:		625 gpm
d)	Normal ambient operating temperature -		60°F to 100°F

e) Normal/Upset conditions which control the pump design include:

Design pressure -	1500 psig
Design temperature -	40°F - 140°F
Seismic loads -	2/3 of SSE

Table 3.9-2r contains a summary of the design calculation for the RCIC pump components.

3.9.3.1.11 ECCS Pumps

The RHR, CS and HPCI pumps are designed in accordance with the ASME Code, Section III. The stress criteria and calculated and allowable stresses are summarized in Table 3.9-2n.

Design condition for RHR and core spray pumps are as follows:

	RHR	CORE SPRAY
Design pressure		
Suction	220 psig	125 psig
Discharge	500 psig	500 psig
Design Temperature	40-360°F	40-212°F

3.9.3.1.12 Standby Liquid Control Pump

The standby liquid control pump has been designed and fabricated to the 1968 P&V Code for Class 2 component.

The SLC pumps and motors are functionally tested by pumping demineralized water through a closed test loop. The SLC pumps are capable of injecting the net contents of the storage tank into the reactor in less than an hour. The pumps are capable of injecting flow into the reactor against pressure up to the second lowest spring set pressure (1195) of the reactor safety relief valves.
Design conditions for the SLC pump include:

a) Flow rate — 43 gpm

b) Maximum operating discharge pressure — 1250 psig

c) Ambient conditions:

Temperature	70°F - 120°F
Relative Humidity	20% - 95%

d) Normal/upset conditions which control the pump design include:

Design pressure	1500 psig
Design temperature	150°F
Seismic Loads	2/3 of SSE

Stress limits for the pressure boundary are the ASME Code allowable stress (1.0S) for general membrane

e) Faulted or emergency conditions include:

Design pressure	1500 psig
Design temperature	150°F
Safe shutdown earthquake	Horizontal 1.5g Vertical 0.14g

A summary of the design calculations for the SLC pump components is contained in Table 3.9-2l.

3.9.3.1.13 Main Steam Isolation Valves and Safety/Relief Valves

The main steam isolation and safety relief valves are designed in accordance with the requirements of the ASME Code Section III, Subsection NB-3500 for Class 1 components.

Load combination, analytical methods, calculated stresses, and allowable limits are shown in Table 3.9-2g and 3.9-2(h), respectively.

3.9.3.1.14 Safety Relief Valve Piping

See Subsection 3.9.3.1.19.

3.9.3.1.15 High Pressure Coolant Injection (HPCI) Turbine

Although not under the jurisdiction of the ASME Code, the HPCI turbine has been designed and fabricated to the basic guidelines for an ASME Code Section III as a Class 2 component. Surveillance testing is performed according to current Technical Specification surveillance requirements.

Design conditions for the HPCI turbine include:

a) Auto-Startup -- 30 cycles per year with reactor pressure at 1150 psig peak and saturated temperature, turbine exhaust pressure at 50 psig peak and saturated temperature.

b) Turbine Inlet - 1250 psig at saturated temperature.

c) Turbine Exhaust - 200 psig at saturated temperature

d) Upset conditions, which control the turbine design include:

   Design pressure

   Design temperature

   Operating basis earthquake

   Inlet and exhaust piping nozzle loads

   Stress limits for pressure boundary are ASME Code allowable stress (1.0S) for general membrane and 1.5S for bending (local membrane).

e) Faulted, or emergency conditions include:

   Design pressure

   Design temperature

   Safe shutdown earthquake

   Inlet and exhaust piping nozzle loads

   Stress limits for pressure boundary are 120% of ASME Code Section III allowable stress (1.2S) for general membrane and 1.8S for bending (local membrane).

f) Nozzle loading definition includes:

Upset	-	Inlet F	=	(20,000 - M)/2.5, but < 5,000 lbs.
	-	Exhaust F	=	(20,000 - M)/0.8, but < 11,500 lbs.

Faulted (or - Emergency)	Inlet F	=	(30,000 - M)/2.5, but < 7,500 lbs.
	Exhaust F	=	(30,000 - M)/0.8, but < 17,250 lbs.

Where F (lbs) and M (ft-lb) are the resultant force and moment on the respective nozzle.

A summary of the design calculations for the HPCI turbine components are shown in Table 3.9-2(ae).

3.9.3.1.16 High Pressure Coolant Injection (HPCI) Pump

The HPCI pump has been designed and fabricated to the requirements for an ASME Code Section III Class 2 component.

The HPCI pump is surveillance tested in conjunction with the HPCI turbine. The HPCI pump is surveillance tested according to the current Technical Specification surveillance requirements. The HPCI pump takes condensate from the above-ground storage tank and at design flow discharges condensate back to the above-ground storage tank via a closed test loop.

Design conditions for HPCI pump include:

a) Total head , maximum- High speed: 3060 feet

Low speed: 525 feet

b) Constant flow rate - 5000 gpm

c) Normal ambient operating temperature - 60°F to 100°F

d) Normal plus Upset conditions which control the pump design include:

Design pressure	-	1500 psig
Design temperature	-	40°F - 140°F
Seismic Loads	-	2/3 of SSE
Suction nozzle loads	-	F = 1940 lb M = 2460 ft-lbs
Discharge nozzle loads	-	F = 3715 lbs M = 4330 ft-lbs

Stress limits for pressure boundary are ASME Code allowable stress (1.0S) for general membrane and 1.5S for bending (local membrane).

e) Faulted, or Emergency conditions include:

Design pressure	-	1500 psig
Design temperature	-	40°F - 140°F
Safe shutdown earthquake	-	Horizontal - 1.50g Vertical - 0.14g

Stress limits for pressure boundary are 120% of ASME Code allowable stress (1.2S) for general membrane and 1.8S for bending (local membrane).

f) Nozzle loading

Pump nozzles are subject to loading from the connecting pipe. The method of analysis shows the maximum resultant moment is due to pipe reaction. The maximum resultant force shall not exceed the allowable. Allowable nozzle forces and moments are expressed as:

Normal plus Upset

Suction	- F = 33,000-0.79M
Discharge	- F = 32,000-1.54M

Emergency:

Suction	- F = 43,000-0.74M
Discharge	- F = 47,000-1.23M

The calculated stress values are compared to allowable stresses for critical components in Table 3.9-2t.

3.9.3.1.17 Reactor Water Cleanup (RWCU) System

The RWCU pump and heat exchangers are not part of a safety system and are not designed to Seismic Category I requirements.

However, the requirements for ASME Code Section III, Class 3 components are used as guidelines in evaluating the RWCU system components.

The design loading combinations and limits for the pump include the following:

a) Normal plus upset loads: This includes the simultaneous effect of normal operating loads, design pressure, temperature, nozzle loads from connected piping, dead weight loads, seismic loads, plus torsional loads due to rotating parts.

b) Seismic loading: This equipment and supports are designed to withstand the static seismic forces applied at the mass center, assuming that the pump is flooded.

c) Stresses in the supports and the anchor bolts due to seismic loads are combined with the stresses due to other live and dead loads and operating loads. The allowable stress for this combination of loads is based on the allowable stress as set forth in the ASME Code Section III.

d) Equipment operates between 70°F and 532.3°F. Transient analysis is not required for Class III components in this temperature range.

Tables 3.9-2(p) and 3.9-2(c) show the calculated stress values and allowable stress limits for the pump and heat exchangers, respectively.

3.9.3.1.18 This Section Has Been Intentionally Deleted

3.9.3.1.19 ASME Code Constructed Items Not Furnished with the NSSS

The design loading combinations categorized with respect to plant operating conditions identified as normal, upset, emergency, and faulted for ASME code constructed items are presented in Table 3.9-6.

The method of combining the peak loads on components and supports resulting from different dynamic events was addressed by the Mark II Owners Subgroup on SRSS. The generic resolution has been reviewed and applies to Susquehanna SES.

The design criteria and stress limits associated with each of the plant operating conditions for each type of ASME code constructed item are presented in Tables 3.9-7, 3.9-8, 3.9-9, 3.9-10, 3.9-11 and 3.9-12.

The component operating condition will be the same as the plant operating condition, except for active pumps or valves for which, the emergency or faulted plant condition is considered a normal operating condition.

3.9.3.2a NSSS Pump and Valve Operability Assurance

The active NSSS pumps and valves are listed in Table 3.9-3.

Active mechanical equipment classified as Seismic Category I are designed to perform their function during the life of the plant under postulated plant conditions. Equipment with faulted condition functional requirements include "active" (active equipment must perform a mechanical motion during the course of accomplishing a safety function) pumps and valves in fluid systems such as the emergency core cooling system. Operability is assured by satisfying the requirements of the following programs. Safety-related valves are qualified by prototype testing and analysis satisfying stress and deformation criteria at all critical locations and safety-related active pumps by analysis with suitable stress limits and nozzle loads. The content of these programs is detailed below.

3.9.3.2a.1 ECCS Pumps

All active pumps are qualified for operability by first being subjected to rigid tests before installation in the plant. The in-shop tests include (1) hydrostatic tests of pressure-retaining parts to 125% of the design pressure times the ratio of material allowable stress at room temperature to the allowable stress value at the design temperature, (2) seal leakage tests, (3)

performance tests, while the pump is operated with flow, to determine total developed head, minimum and maximum head, Net Positive Suction Head (NPSH) requirements and other pump/motor parameters. Also monitored during these operating tests are bearing temperatures (except water cooled bearings) and vibration levels. Both are shown to be below specified limits. After the pump is installed in the plant, it undergoes the cold hydro tests, functional tests, and the required periodic in-service inspection and operation. These tests demonstrate reliability of the pump for the design life of the plant.

In addition to these tests, the safety-related active pumps are analyzed for operability during a faulted condition by assuring that (1) the pump will not be damaged during the seismic and hydrodynamic event, and (2) the pump will continue operating despite the faulted loads.

3.9.3.2a.1.1 Analysis of Loading, Stress, and Acceleration Conditions

In order to avoid damage during the faulted plant condition, the stresses caused by the combination of normal operating loads, SSE, and dynamic system loads are limited to the material elastic limit, as indicated in Table 3.9-2. A three-dimensional finite element model of the pump/motor and its supports is developed using the response spectrum method of dynamic analysis. The same model is analyzed for static nozzle loads, pump thrust loads, and deadweight. Critical location stresses are evaluated and compared with the allowable criteria. The average membrane stress σ_m) for the faulted conditions loads are maintained at 1.2S, or approximately 0.75 σ_y(σ_y - yield stress), and the maximum stress in local fibers (σ_m + bending stress o) is limited to 1.8S, or approximately 1.1 σ_y. The maximum dynamic nozzle loads are considered in an analysis of the pump supports to assure that a system misalignment cannot occur.

Performing these analyses with the conservative loads stated and with the restrictive stress limits of Table 3.9-2 as allowables, will assure that critical parts of the pump will not be damaged during the faulted condition; therefore, the reliability of the pump for post-faulted condition operation will not be impaired by the seismic and hydrodynamic events.

A dynamic analysis is made to determine the seismic load from the applicable floor response spectra. Analysis is made to check that faulted condition nozzle loads and dynamic accelerations will not impair the operability of the pumps during or following the faulted condition. Components of the pump, when having a natural frequency above 33 Hz, are considered essentially rigid. This frequency is considered sufficiently high to avoid problems with amplification between the component and structure for all seismic areas.

If the natural frequency is found to be below 33 Hz, an analysis is performed to determine the amplified input accelerations necessary to perform the static analysis. The adjusted accelerations will be determined using the same conservatisms contained in the horizontal and vertical accelerations used for "rigid" structures. The static analysis is performed using the adjusted accelerations; the stress limits stated in Table-3.9-2 must still be satisfied.

3.9.3.2a.1.2 Pump Operation During and Following Faulted Condition Loading

Active pump/motor rotor combinations are designed to rotate at a constant speed under all design conditions. Motors are designed to withstand short periods of severe overload. The high rotary inertia in the operating pump rotor, and the nature of the random, short duration loading characteristics of the seismic and hydrodynamic event, will prevent the rotor from becoming seized. In actuality, the dynamic loadings will cause only a slight increase, if any, in

the torque (i.e., motor current) necessary to drive the pump at the constant design speed. Therefore the pump will not shutdown during the faulted event and will operate at the design speed despite the faulted loads.

The functional ability of the active pumps after a faulted condition is assured since only normal operating loads and steady state nozzle loads exist. For the active pumps, the faulted condition is more severe than the normal condition only due to seismic and hydrodynamic loads on the equipment itself and the increase in nozzle loads due to the SSE on the connecting pipe. The SSE event is infrequent and of relatively short duration compared to the design life of the equipment. Since it is demonstrated that the pumps would not be damaged during the faulted condition, the post-faulted condition operating loads will be no worse then the normal plant operating limits. This is assured by requiring that the imposed nozzle loads (steady-state loads) for normal conditions and post-faulted conditions be limited by the magnitudes of the normal condition nozzle loads. The post-faulted condition ability of the pumps to function under these applied loads is proven during the normal operating plant conditions for active pumps.

3.9.3.2a.2 SLC Pump and Motor Assembly and RCIC Pump Assembly

These equipment assemblies are small, compact, rigid assemblies, with natural frequencies well above 33 Hz. With this fact verified, each equipment assembly is qualified by static analysis only. This static qualification verifies operability under seismic and hydrodynamic conditions, and assures structural loading stresses within Code limitations.

3.9.3.2a.3 RCIC Turbine

Analysis and testing done on the RCIC turbine is covered by Subsections 3.9.2.2, 3.9.3.1, and Table 3.9-2q.

3.9.3.2a.4 ECCS Motors

Qualification of the Class 1E motors used for the ECCS motors is in compliance with IEEE Standard 323-1971. The qualification of motors of all sizes is based on completion of a type test, followed up with review and comparison of design and material details and seismic analysis of production units, ranging from 500 to 3500 Bhp. The motor is used in the type test. All manufacturing, inspection, and routine tests performed by motor manufacturers on production units are performed on the test motor.

The type test has been performed on a 1250 HP vertical motor in accordance with IEEE Standard 323-1971. Normal operation during the design life is first simulated, then the motor is subjected to a number of seismic events. Then the abnormal environmental condition possible during and after a loss of coolant accident (LOCA) is simulated. The test plan for the type test was as follows:

a) Thermal aging of the motor electrical insulation system (which is a part of the stator only) was based on IEEE Standard 275-1966. The amount of aging equaled the total estimated operating days at maximum insulation surface temperature.

b) Radiation aging of the motor electrical insulation equals the maximum estimated integrated dose of gamma radiation during normal and abnormal conditions.

c) The normal operating inducted vibration effect on the insulation system has been simulated by 1.5g horizontal vibration acceleration of 60 Hz current frequency for one hour duration.

d) Motor bearings are selected and their operating life is established based on bearing manufacturer's test and operating data using the loads calculated to act on the bearing.

e) The dynamic load deflection analysis on the rotor shaft, performed to ensure adequate rotation clearance, has been verified by static loading and deflection of the rotor for the type-test motor.

f) Dynamic load aging and testing has been performed on a biaxial test table in accordance with IEEE 344-1975. During this type test the shake table was activated simulating the maximum design limit of the safe shutdown earthquake and hydrodynamic loads with motor starts and operation combination as may possibly occur during a plant life.

g) An environmental test simulating a LOCA condition with 100 days duration time has been performed with the test motor fully loaded to simulate pump operation. The test consisted of startup and six hours operation at 212°F ambient temperature and 100% steam environment. Another startup and operation of the test motor after one hour stand-still in the same environment was followed by sufficient operation at high humidity and temperature. The operation was based on the temperature-life characteristic curve from IEEE 275-1966 for the insulation type used on the ECCS motors.

3.9.3.2a.5 NSSS Valves

The Class 1 Active Valves are the Main Steam Isolation Valves, Safety/Relief Valves, Recirculation Discharge and Bypass Gate Valves, Standby Liquid Control Valves and Control Rod Drive Scram Discharge Volume Vent and Drain Valves. Each of these valves is dynamically qualified for operability in a manner unique to its design. Therefore, each method of qualification is detailed individually below.

3.9.3.2a.5.1 Main Steam Isolation Valve

The MSIV's are evaluated for operability during dynamic acceleration by both analysis and test. This analysis for MSIV operability is completed in two separate ways. First the valve body is designed in accordance with the ASME Code Section III Class 1 which limits deformation to be within the elastic limit of the material by limiting pressure and pipe reaction input loads (including seismic and hydrodynamic loads). This assures only small deformation in the operating area of the valve body, hence, there is no interference with valve operability.

A static deflection test was conducted on a MSIV of similar design to assure operability under maximum deformation from seismic loading. A maximum static load equivalent to 8 g's applied perpendicular to the actuator axis centerline resulted in no significant change in valve closure rate and no change in measured seat leakage following termination of the load.

To assure that design limits are not exceeded for both piping input loads and actuator dynamic loads, the MSIV is mathematically modeled in the Main Steam Line System Analysis. The valve input loads, amplified accelerations, and resonance frequencies are determined based on site excitation input to the system for MSIV as a part of the overall steamline analysis. Pipe anchors

and restraints are applied as required to limit pipe system resonance frequencies and amplified accelerations to within acceptable limits for the MSIV's. Additional details on the analysis of these valves is shown on Table 3.9-2(h).

The main steam isolation valve operability during LOCA conditions was demonstrated as defined in the report APED-5750 (March 1969). The test specimen was a 20" valve of a design representative of the MSIV's.

3.9.3.2a.5.2 Main Steam Safety/Relief Valves

The safety/relief valves are qualified for operability during seismic and hydrodynamic events by both design and test.

The valve is designed for the largest moments that can occur in service. These are 400,000 in-lbs and 300,000 in-lbs at the inlet and outlet, respectively. These moments are resultants due to dead weight plus dynamic seismic effects of 3 g's horizontal and 1 g vertical of both valve and connecting pipe, thermal expansion of the connecting pipe, and reaction forces from valve discharge. A production S/R valve demonstrated operability while being dynamically (shake table) tested at loads greater than equipment design limit loads.

A mathematical model of this valve is included in the main steam line system analysis. This analysis assures the design limits are not exceeded.

The safety relief valves are generically qualified by testing for seismic and hydrodynamic loads. The natural frequencies are determined to be greater than 33 Hz for seismic and 60 Hz for hydrodynamic loads.

Additional details on the analysis of these valves are shown in Table 3.9-2g.

3.9.3.2a.5.3 Recirculation (Discharge and Bypass) Gate Valves

Recirculation discharge and bypass gate valves are evaluated for operability during seismic and hydrodynamic events by both analysis and test.

Motor operators were generically qualified to IEEE 382-1980 which requires a dynamic test to verify the absence of any natural frequencies below 33 Hz and then a demonstration of operability during dynamic testing. The operators have been qualified to acceleration levels of 10 g from 2 Hz to 100 Hz.

The valve are designed in accordance with the ASME Code, Section III Class 1 design rules. The discharge valves are designed to seismic accelerations of 9.8 g’s horizontal and 2.188 g’s vertical including gravity. Both valves’ extended structures were analyzed to show that they could withstand both compressive stresses and bending stresses imposed by the seismic accelerations. The valve fundamental frequencies were determined by frequency analysis to be less than the seismic cut-off frequency of 33 Hz. This required dynamic analysis considering multinode response. However, since the valves are pipe-mounted and the required response spectra at the valve location were not available, it was necessary to perform a dynamic analysis on the entire piping system. A simple lumped-mass model of the valve and its actuator was developed based on the valve fundamental frequency, and was used to represent the valve dynamic characteristics in the piping analysis.

Dynamic piping analysis indicated that the loads imposed on these valves are less than the design allowable loads. Additional details on analysis of the discharge valves are shown in Table 3.9-2(j).

3.9.3.2a.5.4 Explosive Valves

The SLC explosive valve has been qualified to IEEE 344-1975. The qualification test included a demonstration of the absence of natural frequencies below 35 Hz and the ability to remain operable under a horizontal seismic coefficient of 6.5g and a vertical seismic coefficient of 4.5g at 33 Hz.

3.9.3.2a.5.5 CRD Scram Discharge Volume Vent and Drain Valves

The CRD Scram Discharge Vent and Drain Valves are evaluated for operability during dynamic acceleration by test. The testing consisted of a combination of vibration aging testing. SRV cycling induced fatigue load testing and seismic testing at acceleration levels based upon both upset and faulted required response spectra (RRS). The valve successfully passed all qualification testing.

3.9.3.2a.6 HPCI Turbine

The HPCI turbine is dynamically qualified by static analysis. The turbine assembly and its components were considered to be supported as designed, and horizontal/vertical accelerations were applied to the mass's center of gravity. The magnitude of the acceleration coefficients was 1.50 horizontal and 0.48 vertical. The results of the analysis indicate that the turbine assembly is capable of sustaining the above loadings without overstressing any component.

The turbine was dynamically qualified via dynamic testing by the 1st quarter 1983 in accordance with IEEE 344-1975. The qualification test program demonstrated start-up, steady state operability, and shutdown capabilities.

3.9.3.2b Non-NSSS Pump and Valve Operability Assurance

The pumps under this category are:

a) Diesel oil transfer pumps

b) RHR service water pumps

c) Emergency service water pumps

d) Control structure chiller - cooling water pumps.

All the above pumps are Class 3.

3.9.3.2b.1 Pumps

The pumps listed above are subjected to testing both in the manufacturer's shop and following their installation to verify that they meet the criteria required by the respective specifications.

During manufacture, nondestructive test procedures including liquid penetrant examination, radiographic examination, magnetic particle inspection, and ultrasonic inspection are applied to the pumps. All these procedures are performed in accordance with ASME Code, Section III.

After the pumps have been assembled they are hydrostatically tested and performance tested in the manufacturer's shop in accordance with the Hydraulic Institute's standards.

After the pumps are installed they will undergo functional tests. Provisions will be made for in-service inspection and operational testing.

All these tests demonstrate that the pumps are reliable and will function as specified.

3.9.3.2b.1.1 Analysis of Loading, Stress, and Acceleration Conditions

In addition to the tests and procedures referred to above, the pumps are seismically analyzed to ensure that they will be capable of operating both during and after the OBE and DBE.

In performing these analyses, conservative seismic accelerations and stress criteria were used; this ensures that critical parts of the pump are not damaged during a seismic event and that the pump can still operate following such an event.

3.9.3.2b.1.2 Pump Operation During and Following SSE Loading

Each pump/motor combination is designed to rotate at a constant speed under all conditions unless the rotor becomes completely seized, i.e., with no rotation. Motors are designed to withstand short periods of severe overload and, typically, the rotor can be seized five full seconds before a circuit breaker shuts down the pumps. However, the high rotary inertia in the operating pump rotor, and the nature of the random, short duration loading characteristics of the seismic event, will prevent the rotor from becoming seized. In actuality, the seismic loadings will cause only a slight increase in the torque (i.e., motor current) necessary to drive the pump at the constant design speed. Therefore, the pump will not shut down during the event and will operate at the design speed despite the seismic loads.

From previous discussions, it is evident that the pump/motor units will withstand seismic loadings and, therefore, will perform their intended functions. These proposed requirements take into account the complex characteristics of the pump and are sufficient to demonstrate and ensure the seismic operability of these pumps. Post-seismic condition operating loads will be no worse than the normal plant operating limits.

3.9.3.2b.2 Valves

Active ASME Class 1, 2, and 3 valves are identified in the Plant’s ISI program manual.

Safety related active valves are subjected to a series of tests prior to service and during the plant life. Prior to installation, the following tests are performed: shell hydrostatic test in accordance with ASME Code Section III requirements, backseat and main seat leakage tests, disc hydrostatic test, functional tests which verify that the valve will open and close within the specified time limits, and operability qualification of motor operators for the environmental conditions over the installed life (i.e., aging, radiation, accident environment simulation, etc). in accordance with IEEE 382-1972. After installation, cold hydrostatic construction tests, functional tests in accordance with the requirements of Chapter 14, and periodic in-service

operation in accordance with the requirements of Chapter 16, are performed to verify and ensure the functional ability of the valve.

The valves are designed using either stress analyses or the pressure-containing minimum wall thickness requirements. On all active valves with extended topworks, an analysis is also performed for static equivalent SSE loads applied at the center of gravity of the extended structure. The maximum stress limits allowed in the analyses demonstrate structural integrity and are equal to the limits recommended by the ASME Code for the particular ASME class of valve analyzed as listed in Tables 3.9-7 and 3.9-12. Loading combinations are listed in Tables 3.9-6 and 3.9-14. In addition to these tests and analyses, a representative valve of each design type is tested for verification of operability during a simulated seismic event by demonstrating operational capabilities within the specified limits.

A. Selection of Representative Valve

The valves requiring operability qualification are divided into different groups: by valve manufacturer, valve type, size, pressure class, material type (carbon steel, stainless steel, and alloy steel) and actuator type (AC electric, DC electric, air, hydraulic, etc.) Valve sizes that cover the range of sizes in service are qualified as shown in Table 3.9-15 by tests, and the results are used to qualify all valves within the intermediate range of sizes. A tabulation is made of the weight of the valve actuator, the actuator thrust margin (a ratio of the maximum thrust available from the actuator divided by the design thrust required for the valve), and the yoke configuration (as related to stiffness) for each valve assembly. For a range of qualified valve sizes, as defined by the qualification table, the valve assembly with the heaviest actuator, lowest thrust margin, and least stiff yoke is picked as the test unit. In those cases where a test unit is not readily apparent, more than one unit is tested to provide a conservative test position. This procedure is repeated within each group until all listed units are represented by a test unit, and for each group until all the necessary valves are represented by a test unit.

In addition to the tests, the stress calculations for each valve assembly are reviewed. A tabulation is made for all qualified valve assemblies comparing the yoke stress for all valve classes, the yoke-flange to body and the yoke-flange to actuator-bolting stresses, as applicable, for all classes of valves, and the body stress for Class 1 valves. This is done to provide further analytical justification for the qualification of non-tested valves by tested valves.

B. Qualification Testing Procedures

The valve is mounted in a manner that will conservatively represent typical valve installations. The valve unit includes the actuator and all appurtenances usually attached to the valve in service. The operability of the valve during a SSE is demonstrated by satisfying the following criteria:

a) All the active valves with topworks are designed to have a first natural frequency greater than 33 Hz. This is shown by suitable test or analysis.

b) The extended topworks of the valve are subjected to a statically applied equivalent seismic load. Load is applied at the center of gravity of the topworks in the direction of the weakest axis of the yoke. The design pressure of the valve or the design pressure of the system is simultaneously applied to the valve during the static load tests.

c) The valve is then operated at minimum specified actuation supply voltage or pressure with equivalent seismic static load applied. The valve must perform its safety-related function within the specified operating time limits.

d) Motor operators and other electrical appurtenances necessary for operation are qualified as operable during the SSE in accordance IEEE-344-1975 prior to their installation on the valve.

For valves with and without the topworks supported, the statically applied load envelopes the specified G-force times the weight of the topworks. **This load is generally greater than would result from 3.0 g horizontal and 3.0 g vertical**. For valves with the topworks supported, additional loading from thermal and/or anchor movements may be imposed upon the valve through the support(s). If the loads due to thermal and/or anchor movements are greater than the statically applied load (which envelopes inertial forces), **stress and critical deflection analyses are** performed on the valve (considering the maximum applied loading) **as an acceptable qualification alternative.**

An exception to the above described seismic qualification approach is the RHR throttle valves (HV151F017A/B and HV251F017A/B) which were not tested with a static seismic load but instead were qualified by a combination of static seismic analysis and static deflection operability analysis. The static deflection operability analysis verified that adequate internal clearances exist to insure the binding does not occur within the valve during and after a design basis event.

The piping designer limits the valve accelerations and support loads to allowable values as determined by the qualification test and analysis.

The valve is leak tested following the test described above to show that the valve has not been damaged. The leak rates must not exceed the original allowable leakage rate specified for the valve.

The above testing program applies only to valves with overhanging structures, i.e., the motor operator or air actuator assembly. Because of their simple characteristics, check and other compact valves are not affected by seismic acceleration. Check valves have no extended structures to distort the valve and cause a malfunction. Check valve discs are designed to allow sufficient clearance around the disc to prevent distortions from nozzle or other imposed loads. Accordingly, check valves are qualified by a combination of the following tests and analysis:

a) The air-operated check valves are analyzed to ensure that the air cylinder cannot impair the ability of the valve to operate as a simple check valve during seismic loading. No functional test simulating seismic loading is performed. Air operators on check valves do not perform a safety function.

b) In-shop hydrostatic test

c) In-shop seat leakage test

d) Periodic valve exercise and inspection to ensure the functional ability of the valve in accordance with the requirements of Chapter 16.

Using the methods described, all the safety-related active valves in the systems are qualified for operability during a seismic event. These methods conservatively simulate the seismic event and ensure that the active valves will perform their safety-related functions when necessary.

3.9.3.3a Design and Installation of NSSS Supplied Pressure Relief Devices

3.9.3.3a.1 Main Steam Safety/Relief Valves

Safety/relief valve lift results in a transient that produces momentary unbalanced forces acting on the discharge piping system for the period from opening of the safety/relief valve until a steady discharge flow from the reactor pressure vessel to the suppression pool is established. This period includes clearing of the water slug from the end of the discharge piping submerged in the suppression pool. Pressure waves traveling through the discharge piping following the relatively rapid opening of the safety/relief valve cause the safety/relief valve discharge piping to vibrate. This in turn produces forces that act on the main steam piping.

The analysis of the relief valve discharge transient consists of a stepwise time history solution of the fluid flow equation to generate a time-history of the fluid properties at numerous locations along the pipe. Simultaneously, reaction loads on the pipe are determined at each location corresponding to the position of an elbow. These loads are composed of pressure-times-area, momentum change, and fluid friction terms. Figure 3.9-2 shows a set of fluid property and pipe section load transients typical of those produced by relief valve discharge.

The method of analysis applied to determine piping system response to relief valve operation is time history integration. The forces are applied at locations on the piping system where fluid flow changes direction, thus causing momentary reactions. The resulting loads on the safety/relief valve, the main steam line, and the discharge piping are combined with loads due to other effects as specified in Subsection 3.9.3.1. The Code stress limits corresponding to load combinations classification as normal, upset, emergency and faulted, are applied to the steam and discharge pipe.

3.9.3.3b Design and Installation Details for Mounting of Pressure Relief Devices in ASME Code Class 1 and 2 Systems

The design of pressure relieving devices can be grouped into two categories: open discharge and closed discharge.

a) Open Discharge

There are no open discharge pressure relieving devices mounted on ASME Code Class 1 and 2 systems.

b) Closed Discharge

A closed discharge system is characterized by piping between the valve and a tank, or some other terminal end. Under steady-state conditions, there are no net unbalanced

forces. The initial transient response and resulting stresses are determined by using either a time-history computer solution or a conservative equivalent static solution. In calculating initial transient forces, pressure and momentum terms are included. Water slug effects are also considered.

Time history dynamic analysis is performed for the discharge piping and its supports. The effect of the loading on the header is also considered. The design load combinations for a given transient are shown in Table 3.9-2, and the design criteria and stress limits for the relief valve are shown in Table 3.9-2g.

3.9.3.4 Component Supports

3.9.3.4.1 Recirculation Piping Supports

The NSSS-designed recirculation piping supports are designed in accordance with Subsection NF of ASME Code Section III. (Non-NSSS designed pipe supports on recirculation piping are in accordance with Subsection 3.9.3.4.6.) Supports are either designed by load rating per paragraph NF-3260, or to the stress limits for linear supports per paragraph NF-3231. In general, the load combinations for the various operating conditions correspond to those used to design the supported pipe. Design transient cyclic data are not applicable to piping supports as no fatigue evaluation is necessary to meet the Code requirements.

The design criteria and dynamic testing requirements for component supports are as follows:

Component Supports

All components supports are designed, fabricated, and assembled so that they cannot become disengaged by the movement of the supported pipe or equipment after they have been installed.

Hangers

The design load on hangers is the load caused by dead weight. The hangers are calibrated to ensure that they support the design load at both their hot and cold load settings. Hangers provide a specified down travel and up travel in excess of the specified thermal movement.

Snubbers

The design load on snubbers includes those loads caused by dynamic forces (OBE and SSE) and hydrodynamic loads, system anchor movements, and reaction forces caused by relief valve discharge, turbine stop valve closure, etc.

The snubbers are designed and load rated in accordance with NF-3000 to be capable of carrying the design load for all operating conditions. Faulted condition design uses the criteria outlined in Appendix F of the ASME code. They are designed to be able to carry the load under normal, upset, emergency, and faulted loading conditions.

The snubbers are also tested dynamically to ensure that they can perform as required in the following manner:

a) The snubber will be subjected to either force or displacement that varies approximately as the sine wave.

b) The frequency (Hz) of the input motion or force will be verified at small increments within the specified range.

c) The resulting relative displacements and corresponding loads across the working components, including end attachments, will be recorded.

d) The test will be conducted with the snubber at various temperatures representative of operating conditions.

e) The rated load in both tension and compression will be equal to or higher than the peak load.

f) The duration of the test at each frequency will be specified.

Dynamic Testing

The criterion used to demonstrate the operability of the snubber under dynamic loading conditions is that the total travel of the unit, including lost motion and deflection during dynamic load cycling, shall not exceed ±.060 inches (.120 inch total).

The dynamic testing on a prototype snubber consists of the following:

(a) The snubber is subjected to load cycling at 100%, 75%, 50% of the rated load and this load varies approximately as the sine wave function.

(b) The frequency of the input force is varied from 3 to 33 Hz in 3 Hz steps.

(c) The duration of the test (load cycles can be determined by this time interval) at each frequency is 10 seconds as a minimum.

Struts

The design load on struts includes those loads caused by dead weight, thermal expansion, primary dynamic forces, i.e., (OBE) and SSE, system anchor displacements, and reaction forces caused by relief valve discharge, turbine stop valve closure, etc.

Struts are designed in accordance with NF-3000 to be capable of carrying the design load for all operating conditions.

3.9.3.4.2 Reactor Pressure Vessel Support Skirt

The permissible compressive load on the reactor vessel support skirt cylinder (modeled as plate and shell type component support) was limited by the GE design specification to 90 percent of the load which produces yield stress, divided by the safety factor for the condition being evaluated. The effects of fabrication and operational eccentricity was included. The safety factor for faulted conditions was 1.125.

An analysis of reactor pressure vessel support skirt buckling for faulted conditions shows that the support skirt has the capability to meet ASME Code Section III, Paragraph F-1370(c) faulted condition limits of 0.67 times the critical buckling strength of the support at temperature. The

faulted condition analyzed included the compressive loads due to the design basis maximum earthquake, the overturning moments and shears due to the jet reactor load resulting from a severed pipe, and the compressive effects on the support skirt due to the thermal and pressure expansion of the reactor vessel. The expected maximum earthquake loads for the reactor vessel support skirts are less than 60% of the maximum design basis loads used in the buckling analysis described; therefore, the expected faulted loads are well below the critical buckling limits of Paragraph F-1370(c) for the vessel support skirt. The expected earthquake loads were determined using the seismic dynamic analysis methods described in Section 3.7.

The loading condition, stress criterion, calculated and allowable stresses are summarized in Table 3.9-2a.

3.9.3.4.3 NSSS Floor-Mounted Equipment (Pumps, Heat Exchangers and RCIC Turbine)

The RHR pump, core spray pump, RHR heat exchangers, RCIC pump, SLC pump, and RCIC turbine are all analyzed to verify the adequacy of their support structures under various plant operating conditions. In all cases, the calculated loads in the critical support areas are within the ASME Code allowables.

3.9.3.4.4 Supports for ASME Code Class 1, 2 and 3 Active Components

ASME Code Class 1, 2 and 3 active components are either pumps or valves. Since valves are supported by piping and not tied to building structures, pipe design criteria govern.

Seismic Category I active pumps supports are qualified for seismic and hydrodynamic loads by testing when the pump supports along with the pumps are fulfilling the following conditions:

1) Simulate actual mounting conditions.

2) Simulate all static and dynamic loadings on the pump.

3) Monitor pump operability during testing.

4) The normal operation of the pump during and after the test indicates that the supports are adequate. Any deflection or deformation of the pump supports which precludes the operability of the pump, is not accepted; and,

5) Supports are inspected for structural integrity after the test. Any cracking or permanent deformation is not accepted.

Seismic and hydrodynamic qualification of component supports by analysis is generally accomplished as follows:

1) Stresses at all support elements and parts such as pumps holddown, and baseplate holddown bolts, pump support pads, pump pedestal, and foundation are checked to be within the allowable limits as specified in ASME Subsection NF.

2) For Normal and Upset plant conditions, the deflections and deformations of the supports are assured to be within the elastic limits and not to exceed the values permitted by the designer based on his design verification tests to ensure the operability of the pumps.

3) For Emergency and Faulted plant conditions, the deformations must not exceed the values permitted by the designer to ensure the operability of the pumps. Elastic/plastic analysis will be performed if the deflections are above the elastic limits.

3.9.3.4.5 HPCI Turbine

This section has been intentionally deleted.

3.9.3.4.6 Non-NSSS Supports

The design loading combinations for supports for ASME Code Class 1, 2, and 3 components, categorized with respect to plant operating conditions identified as normal, upset, emergency, and faulted, are given in Table 3.9-14. This table also provides the stress limits for each plant operating condition.

The loads imposed on the ASME Class 1, 2 and 3 active valves and pumps are limited to values meeting both the manufacturer's and code allowables to insure operability of the active components by the design of the supports. The supports are designed to remain elastic under the maximum loads. The minor local deformations associated with the elastic deformation of the support will not impair operability of the active components.

3.9.3.4.6.1 Snubbers

Snubbers will be used in seismic Category I systems. Snubbers will be purchased with load ratings appropriate for the design conditions and load combinations.

3.9.3.4.6.1.1 Snubber Design Specification

The specification for the purchase of shock suppressors (snubbers) covers the following related to snubber design, supplier's performance qualification tests and load tests. Mechanical snubbers specified for Susquehanna SES are addressed below. Design specification information pertaining to hydraulic snubbers is also provided below.

Design Criteria

Mechanical Snubbers

a) The frictional resistance of purchased suppressors shall not exceed 2% of the rated load. However, for suppressors tested during in-service inspection, the frictional resistance may exceed 2% of the rated load provided that the effects of this increase on the Seismic Category I systems on which the suppressors are used have been evaluated and shown to be acceptable.

b) All purchased snubbers shall be designed such that they limit the acceleration of the pipe to a maximum of 0.02g when subjected to any load up to rated load. An evaluation of installed snubbers and the Seismic Category I systems on which they are used indicates that the permissible acceleration can be increased to 0.04g maximum, with some exceptions that are limited to 0.02g as defined in the snubber program.

c) The total pipe movement a long the axis of the suppressor shall not exceed ±.060 inches due to any applied dynamic cycle load from 3 to 33 cps up to the rated load at the unit.

d) The suppressor shall be designed for an exposure to a temperature of 40°F prior to initial startup and 200°F during continuous operations and to a relative humidity of 55 percent normally and 90 percent during shutdown. The radiation exposure shall be 100 Roentgen/hour.

Performance Test: Two types of tests are required.

Production Test: This type of test is required to be performed on each unit.

a) Check unit to confirm acceleration level is less than specified maximum.

b) Check unit to confirm that it operates freely over the total stroke.

c) Measure and record the force required to initiate motion over the stroke in tension and compression.

d) Measure and record lost motion of the snubber mechanism.

Qualification Tests: These types of tests are to be performed on randomly selected production models. These tests are used to demonstrate the required load performance (load rating) and specified displacement when subjected to dynamic load cycling. Also included in these tests are low temperature, high temperature, humidity, salt/sand and dust spray test, life test and faulted load test.

Hydraulic Snubbers

The following environmental conditions apply to each hydraulic snubber:

	Normal Operation	Emergency	Faulted
Temperature range	32°F to 176°F 0°C to 80°C	32°F to 302°F 0°C to 150°C	32°F to 302°F 0°C to 150°C
Humidity	Up to 100%	Up to 100%	Up to 100%
Pressure	1 bar	1 bar	5 bar
Radiation (*total accum)	10^7 rad	10^7 rad	10^7 rad
Max. no. cycles	20000	40	1

The following performance testing is required for each hydraulic snubber:

The frictional resistance, including breakaway, in compression and tension shall not exceed 1.5% of normal load for a normal load > 4500 lbs.

The standard lock-up velocity in compression and tension shall be between 4.7 ipm to 14.2 ipm. The standard bleed velocity shall be between 0.47 imp to 4.7 ipm at normal load.

The piston rod travel during a dynamic functional test under rated load from 2 to 35 cps shall be kept at $< \pm 0.16"$.

3.9.3.4.6.1.2 Snubber Analysis Model

A piping system is idealized as a mathematical model consisting of lumped masses connected by massless elastic members. The elastic members are given the properties of the piping system being analyzed. The lumped masses are carefully located to adequately represent the dynamic and elastic properties of the piping system. A lumped mass is located at the beginning and end of every elbow, valve, at the extended valve operator, and at the intersection of every tee. On straight runs, lumped masses are located at spacings no greater than the span length corresponding to 33 cps. A mass point is located at every extended mass to account for torsional effects on the piping system. In addition, the increased stiffness and mass of valves is considered in the modeling of a piping system.

The three-dimensional stiffness matrix of the mathematical model is determined by the direct stiffness method. Axial, shear, flexural and torsional deformations of each member are included. For curved members and branch connectors a decreased stiffness is used in accordance with ASME Section III. The mass matrix is also calculated.

Snubbers are considered to be rigid members in the dynamic model. Differences in tension and compression spring rates will not effect design calculations; similarly entrapped air and temperature do not effect mechanical snubbers. Hydraulic snubbers manufactured by LISEGA have a pressurized reservoir that precludes air entrapment and environmental temperature range limits are provided in the design specification.

The load conditions and combinations are being addressed as a generic issue and are included in the SSES plant Design Assessment Report (DAR). The transients analyzed will include:

1. Seismic

2. Hydrodynamic

    a) LOCA induced
    b) SRV induced

3. Flow disruption transients (e.g., fast valve closure)

4. Normal Operating Loads.

Snubber locations and sizes are chosen to maintain the stresses due to the above listed loads to below the ASME code allowable stresses. Since all of the loads described above will be maintained below the code allowable stresses for the plant condition indicated in the DAR, the snubber with the appropriate load rating will be used.

3.9.4 CONTROL ROD DRIVE SYSTEM

This plant is equipped with a hydraulic control rod drive system. The discussion in this section includes the Control Rod Drive Mechanism (CRDM), the Hydraulic Control Unit (HCU), the Condensate Supply System and the Scram discharge volume and extends to the coupling interface with the control rods.

3.9.4.1 Descriptive Information on CRDS

Descriptive information on the control rod drives as well as the entire control and drive system is contained in Section 4.6.

3.9.4.2 Applicable CRDS Design Specifications

The Control Rod Drive System (CRDS) is designed to meet the functional design criteria as outlined in Section 4.6 and consists of the following:

a) Locking piston control rod drive;

b) Hydraulic control unit;

c) Hydraulic power supply (pumps),

d) Interconnecting piping,

e) Flow and pressure and isolation valves,

f) Instrumentation and electrical controls.

Those CRD components forming part of the primary pressure boundary are designed according to ASME Code Section III.

The quality group classification of the CRD hydraulic system is outlined in Table 3.2-1, and the components are designed according to the codes and standards governing the individual quality groups.

Pertinent aspects of the design and qualification of the CRD components are discussed in the following locations: transients in Subsection 3.9.1.1, faulted conditions in Subsection 3.9.1.4, and dynamic testing in Subsection 3.9.2.2.

3.9.4.3 Design Loads, Stress Limits, and Allowable Deformation

The ASME Code components of the CRDs and CRD Housings have been evaluated analytically and the design load combinations and stress limits for the CRD housing are listed in Table 3.9-2v. For the non-code components, experimental testing was used to assure the CRD performance under all possible conditions as described in Subsection 3.9.4.4.

Deformations are not a limiting factor in the analysis of the CRDs components based upon the results of numerous tests on the drive.

3.9.4.3.1 Control Rod Drive Housing Supports

The Control Rod Drive (CRD) housing support system functions are described in Section 4.6.

The American Institute of Steel Construction (AISC) Manual of Steel Construction, "Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings," was used in designing the CRD housing support system. However, to provide a structure that absorbs as much energy as practical without yielding, the allowable tension and bending stresses used were 90% of yield and the allowable shear stress used was 60% of yield. These design stresses are 1.5 times the AISC allowable stresses (60% and 40% of yield, respectively).

The CRD housing supports are designed as Seismic Category I equipment.

3.9.4.4 CRD Performance Assurance Program

The CRD test program consists of the following tests:

a) Development tests

b) Factory Quality Control Tests

c) 5 year Maintenance Life tests
d) 1.5X Design Life tests

e) Operational tests

f) Acceptance tests

g) Surveillance tests

All of the above tests except c) and d) are discussed in Subsections 4.6.3 through 4.6.3.1.1.5. Tests c) and d) are discussed below:

c) "5 Year Maintenance Life" Tests

   Four Control Rod Drives are picked at random from the production stock each year and subjected to various tests under simulated reactor conditions and 1/8 of the cycles specified in Subsection 3.9.1.1. Upon completion of the test program, the control rod drive parts are checked to the drawings and all parts must meet or surpass the minimum specified requirements.

d) 1.5X Design Life Tests

   When a significant design change is made to the components of the drive, the drive is subjected to a series of tests equivalent to 1.5 times the life test cycles specified in Subsection 3.9.1.1.

Two CRDs were tested in 1976. Upon completion of the test program, the CRDs were disassembled and the parts checked to the drawing for wear and/or damage. All parts met or surpassed the minimum specified requirements.

3.9.5 REACTOR PRESSURE VESSEL INTERNALS

This subsection identifies and discusses the structural and functional integrity of the major reactor pressure vessel internals.

3.9.5.1 Design Arrangements

The core support structures and reactor vessel internals (exclusive of fuel, control rods, CRDs, and incore nuclear instrumentation) are identified below:

Core Support Structures

Shroud

Shroud support

Core plate and holddown bolts

Top guide (including bolts and keepers)

Fuel supports

Control rod guide tubes

Control rod drive housing

Reactor Internals

Jet Pump assemblies and instrumentation

*Feedwater spargers

Vessel head spray nozzle

Differential pressure and liquid control lines

In-core flux monitor tubes

*Initial startup neutron sources

*Surveillance sample holders

Core spray lines and spargers

*In-Core instrument housings

*Steam dryer

*Shroud head and steam separator assembly

*Guide rods

CRD thermal sleeves

* Non-safety class components

A general assembly drawing of the important reactor components is shown in Figure 3.9-3.

The floodable inner volume of the reactor pressure vessel can be seen in Figure 3.9-4. It is the volume inside the core shroud up to the level of the jet pump suction inlet.

The design arrangement of the reactor internals, such as the jet pumps, steam separators and guide tube, is such that one end is unrestricted and thus free to expand.

3.9.5.1.1 Core Support Structures

The core support structures consist of those items listed in Subsection 3.9.5.1. These structures form partitions within the reactor vessel, to sustain pressure differentials across the partitions, direct the flow of the coolant water, and laterally locate and support the fuel assemblies. Figure 3.9-4 shows the reactor vessel internal flow paths.

3.9.5.1.1.1 Shroud

The shroud support, shroud, and top guide make up a stainless steel cylindrical assembly that provides a partition to separate the upward flow of coolant through the core from the downward recirculation flow. This partition separates the core region from the Downcomer annulus, thus providing a floodable region following a recirculation line break. The volume enclosed by this assembly is characterized by three regions. The upper portion surrounds the core discharge plenum, which is bounded by the shroud head on top and the top guide's grid plate below. The central portion and the shroud surrounds the active fuel and forms the longest section of the assembly. This section is bounded at the bottom by the core support. The lower portion, surrounding part of the lower plenum, is welded to the reactor pressure vessel shroud support.

3.9.5.1.1.2 Shroud Head and Steam Separator Assembly

The shroud head and steam separator assembly is bolted to the top of the top guide to form the top of the core discharge plenum. This plenum provides a mixing chamber for the steam-water mixture before it enters the steam separators. Individual stainless steel axial flow steam separators are attached to the top of standpipes that are welded into the shroud head. The steam separators have no moving parts. In each separator, the steam-water mixture rising through the standpipe passes vanes that impart a spin to establish a vortex separating the water from the steam. The separated water flows from the lower portion of the steam separator into the Downcomer annulus.

3.9.5.1.2 Core Plate

The core plate consists of a circular stainless steel plate with bored holes stiffened with a rim and beam structure. The plate provides lateral support and guidance for the control rod guide tubes, in-core flux monitor guide tubes, peripheral fuel supports, and startup neutron sources. The last two items are also supported vertically by the core support plate.

The entire assembly is bolted to a support ledge on the lower portions of the shroud.

3.9.5.1.3 Top Guide

The top guide is formed by a series of stainless steel beams joined at right angles to form square openings and fastened to a peripheral rim. Each opening provides lateral support and guidance for four fuel assemblies or in the case of peripheral fuel, one or two fuel assemblies. Sockets are provided in the bottom of the beam intersections to anchor the in-core flux monitors and startup neutron sources. The rim of the top guide rests on a ledge between the upper and central portions of the shroud. The top guide has alignment pins that engage and bear against slots in the shroud which are used to correctly position the assembly before it is secured. Lateral restraint is provided by wedge blocks between the top guide and the shroud wall.

3.9.5.1.4 Fuel Support

The fuel supports, shown in Figure 3.9-5 are of two basic types; namely, peripheral supports and four-lobed orificed fuel supports. The peripheral fuel support is located at the outer edge of the active core and is not adjacent to control rods. Each peripheral fuel support will support one fuel assembly and contains a single orifice assembly designed to assure proper coolant flow to the peripheral fuel assembly. Each four-lobed orificed fuel support will support four fuel assemblies and is provided with four orifice plates to assure proper coolant flow distribution to each rod-controlled fuel assembly. The four-lobed orificed fuel supports rest in the top of the control rod guide tubes which are supported laterally by the core support. The control rods pass through slots in the center of the four-lobed orificed fuel support. A control rod and the four adjacent fuel assemblies represent a core cell. (see Subsection 4.4.2).

3.9.5.1.5 Control Rod Guide Tubes

The control rod guide tubes, located inside the vessel, extend from the top of the control rod drive housings up through holes in the core support plate. Each tube is designed as the guide for a control rod and as the vertical support for a four-lobed orificed fuel support piece and the four fuel assemblies surrounding the control rod. The bottom of the guide tube is supported by the control rod drive housing, which in turn transmits the weight of the guide tube, fuel support, and fuel assemblies to the reactor vessel bottom head. A thermal sleeve is inserted into the control rod drive housing from below and is rotated to lock the control rod guide tube in place. A key is inserted into a locking slot in the bottom of the control rod drive housing to hold the thermal sleeve in position.

3.9.5.1.6 Jet Pump Assemblies

The jet pump assemblies are located in two semi-circular groups in the downcomer annulus between the core shroud and the reactor vessel wall. The design and performance of the jet pump is covered in detail in References 3.9-1 and 3.9-2. Each stainless steel jet pump consists of driving nozzles, suction inlet, throat or mixing section, and diffuser (see Figure 3.9-6). The driving nozzle, suction inlet, and throat are joined together as a removable unit, and the diffuser is permanently installed. High pressure water from the recirculation pumps is supplied to each pair of jet pumps through a riser pipe welded to the recirculation inlet nozzle thermal sleeve. A riser brace consists of cantilever beams welded to a riser pipe and to pads on the reactor vessel wall.

The nozzle entry section is connected to the riser by a metal-to-metal, spherical-to-conical seal joint. Firm contact is maintained by a hold- down clamp. The throat section is supported laterally by a bracket attached to the riser. There is a slip-fit joint between the throat and

diffuser. The diffuser is a gradual conical section changing to a straight cylindrical section at the lower end.

3.9.5.1.7 Steam Dryers

The steam dryers remove moisture from the wet steam leaving the steam separators. The extracted moisture flows down the dryer vanes to the collecting troughs, then flows through tubes into the downcomer annulus. A skirt extends from the bottom of the dryer vane housing to the steam separator standpipe, below the water level. This skirt forms a seal between the wet steam plenum and the dry steam flowing from the top of the dryers to the steam outlet nozzles.

The steam dryer and shroud head are positioned in the vessel during installation with the aid of vertical guide rods. The dryer assembly rests on steam dryer support brackets attached to the reactor vessel wall. Upward movement of the dryer assembly, which would occur only under accident conditions, is restricted by steam dryer hold-down brackets attached to the reactor vessel top head.

3.9.5.1.8 Feedwater Spargers

The feedwater spargers are stainless steel headers located in the mixing plenum above the downcomer annulus. A separate sparger is fitted to each feedwater nozzle and is shaped to conform to the curve of the vessel wall. Sparger end brackets are pinned to vessel brackets to support the spargers. Feedwater flow enters the center of the spargers and is discharged radially inward to mix the cooler feedwater with the downcomer flow from the steam separators and steam dryer before it contacts the vessel wall. The feedwater also serves to condense the steam in the region above the downcomer annulus and to subcool the water flowing to the jet pumps and recirculation pumps.

3.9.5.1.9 Core Spray Lines

The core spray lines are the means for directing flow to the core spray nozzles which distribute coolant during accident conditions.

Two core spray lines enter the reactor vessel through the two core spray nozzles. (see Section 5.4). The lines divide immediately inside the reactor vessel. The two halves are routed to opposite sides of the reactor vessel and are supported by clamps attached to the vessel wall. The lines are then routed downward into the downcomer annulus and pass through the upper shroud immediately below the flange. The flow divides again as it enters the center of the semicircular sparger, which is routed halfway around the inside of the upper shroud. The two spargers are supported by brackets designed to accommodate thermal expansion. The line routing and supports are designed to accommodate differential movement between the shroud and vessel. The other core spray line is identical except that it enters the opposite side of the vessel and the spargers are at a slightly different elevation inside the shroud. The correct spray distribution pattern is provided by a combination of distribution nozzles pointed radially inward and downward from the spargers (see Section 6.3).

3.9.5.1.10 Vessel Head Spray Nozzle

When reactor coolant is returned to the reactor vessel part of the flow can be diverted to a spray nozzle in the reactor head. This spray maintains saturated conditions in the reactor vessel head

volume by condensing steam being generated by the hot reactor vessel walls and internals. The spray also decreases thermal stratification in the reactor vessel coolant. This ensures that the water level in the reactor vessel can rise. The higher water level provides conduction cooling to more of the mass of metal of the reactor vessel and, therefore, helps maintain the cooldown rate.

The vessel head spray nozzle is mounted to a short length of pipe and a flange, which is bolted to a mating flange on the reactor vessel head nozzle (see Subsection 5.4.7).

3.9.5.1.11 Differential Pressure and Liquid Control Line

The differential pressure and liquid control line serves a dual function within the reactor vessel - to provide a path for the injection of the liquid control solution into the coolant stream and to sense the differential pressure across the core support plate (described in Section 5.4). This line enters the reactor vessel at a point below the core shroud as two concentric pipes. In the lower plenum, the two pipes separate. The inner pipe terminates near the lower shroud with a perforated length below the core support plate. It is used to sense the pressure below the core support plate during normal operation and to inject liquid control solution if required. This location facilitates good mixing and dispersion. The inner pipe also reduces thermal shock to the vessel nozzle should the standby liquid control system be actuated. The outer pipe terminates immediately above the core support plate and senses the pressure in the region outside the fuel assemblies.

3.9.5.1.12 In-Core Flux Monitor Guide Tubes

In-core flux monitor guide tubes provide a means of positioning fixed detectors in the core as well as provide a path for calibration monitors (TIP System).

The in-core flux monitor guide tubes extend from the top of the in-core flux monitor housing (see Section 5.4) in the lower plenum to the top of the core support plate. The power range detectors for the power range monitoring units and the dry tubes for the source range monitoring and intermediate range monitoring (SRM/IRM) detectors are inserted through the guide tubes. A lattice work of clamps, tie bars, and spacers give lateral support and rigidity to the guide tubes. The bolts and clamps are welded, after assembly, to prevent loosening during reactor operation.

3.9.5.1.13 Surveillance Sample Holders

The surveillance sample holders are welded baskets containing impact and tensile specimen capsules (see Section 5.4). The baskets hang from the brackets that are attached to the inside wall of the reactor vessel and extend to mid-height of the active core. The radial positions are chosen to expose the specimens to the same environment and maximum neutron fluxes experienced by the reactor vessel itself while avoiding jet pump removal interference or damage.

3.9.5.2 Design Loading Conditions

3.9.5.2.1 Events to be Evaluated

Examination of the spectrum of conditions for which the safety design basis must be satisfied reveals four significant faulted events:

a) Recirculation Line Break: a break in a recirculation line between the reactor vessel and the recirculation pump suction.

b) Steam line break accident: a break in one main steam line between the reactor vessel and the flow restrictor. The accident results in significant pressure differentials across some of the structures within the reactor.

c) Earthquake: subjects the core support structures and reactor internals to significant forces as a result of ground motion.

d) Safety relief valve discharge in combination with an SSE.

Analysis of other conditions existing during normal operation, abnormal operational transients, and accidents shows that the loads affecting the core support structures and other engineered safety feature reactor internals are less severe than these three postulated events.

The faulted conditions for the reactor pressure vessel internals are discussed in Subsection 3.9.1.4. Loading combination and analysis for the reactor pressure vessel internals are discussed in Subsection 3.9.3.l, Tables 3.9-1 and 3.9-2.

3.9.5.2.2 Pressure Differential During Rapid Depressurization

A digital computer code is used to analyze the transient conditions within the reactor vessel following the recirculation line break accident and the steam line break accident. The analytical model of the vessel consists of nine nodes, which are connected to the necessary adjoining nodes by flow paths having the required resistance and inertial characteristics. The program solves the energy and mass conservation equations for each node to give the depressurization rates and pressure in the various regions of the reactor. Figure 3.9-7 shows the nine reactor nodes. The computer code used is the General Electric Short-Term Thermal-Hydraulic Model described in Reference 3.9-3. This model has been approved for use in ECCS conformance evaluation under 10CFR50, Appendix K. In order to adequately describe the blowdown pressure effect on the individual assembly components, three features are included in the model that are not applicable to the ECCS analysis and are, therefore, not described in Reference 3.9-3. These additional features are discussed below:

a) The liquid level in the steam separator region and in the annulus between the dryer skirt and the pressure vessel is tracked to more accurately determine the flow and mixture quality in the steam dryer and in the steamline.

b) The flow path between the bypass region and the shroud head is more accurately modeled since the fuel assembly pressure differential is influenced by flashing in the guide tubes and bypass region for a steamline break. In the ECCS analysis, the momentum equation is solved in this flow path, but its irreversible loss coefficient is conservatively set at an arbitrary low value.

c) The enthalpies in the guide tubes and the bypass are calculated separately, since the fuel assembly ΔP is influenced by flashing in these regions. In the ECCS analysis, these regions are lumped.

3.9.5.2.3 Recirculation Line and Steam Line Break

3.9.5.2.3.1 Accident Definition

Both a recirculation line break (the largest liquid break) and an inside steam line break (the largest steam break) are considered in determining the design basis accident for the engineered safety feature reactor internals. The recirculation line break is the same as the design basis loss-of-coolant accident described in Section 6.3. A sudden, complete circumferential break is assumed to occur in one recirculation loop. The pressure differentials on the reactor internals and core support structures are in all cases lower than for the main steam line break.

The analysis of the steam line break assumes a sudden, complete circumferential break of one main steam line between the reactor vessel and the main steam line restrictor. A steam line break upstream of the flow restrictors produces a larger blowdown area and thus a faster depressurization rate than a break downstream of the restrictors. The larger blowdown area results in greater pressure differentials across the reactor internal structures.

The steam line break accident produces significantly higher pressure differentials across the reactor internal structures than does the recirculation line break. This results from the higher reactor depressurization rate associated with the steam line break. Therefore, the steam line break is the design basis accident for internal pressure differentials.

3.9.5.2.3.2 Effects of Initial Reactor Power and Core Flow

The maximum internal pressure loads can be considered to be composed of two parts: steady-state and transient pressure differentials. For a given plant the core flow and power are the two major factors which influence the reactor internal pressure differentials. The core flow essentially affects only the steady-state part. For a fixed power, the greater the core flow, the larger will be the steady-state pressure differentials. The core power affects both the steady-state and the transient parts. As the power is decreased, there is less voiding in the core and consequently the steady-state core pressure differential is less. However, less voiding in the core also means that less steam is generated in the reactor pressure vessel and thus the depressurization rate and the transient part of the maximum pressure load is increased. As a result, the total loads on some components are higher at low power.

To ensure that the calculated pressure differences bound those which could be expected if a steam line break should occur, an analysis is conducted at a low power-high recirculation flow condition in addition to the standard safety analysis condition at high power, rated recirculation flow. The power chosen for analysis is the minimum value permitted by the recirculation system controls at rated recirculation drive flow (that is, the drive flow necessary to achieve rated core flow at rated power.)

This condition maximizes those loads which are inversely proportional to power. It must be noted that this condition, while possible, is unlikely; first, because the reactor will generally operate at or near full power; second, because high core flow is neither required nor desirable at such a reduced power condition.

3.9.5.2.4 Seismic and Hydrodynamic Loads

The seismic and hydrodynamic loads acting on the structures within the reactor vessel are based on a dynamic analysis as described in Section 3.7. Seismic analysis is performed by coupling the lumped mass model of the reactor vessel and internals, as described in Section 3.7, with the building model to determine the acceleration force and moment time histories in the reactor vessel and internals. This is accomplished by using the modal superpostion method. Acceleration response spectra are also generated for subsystem analyses of selected components.

3.9.5.3 Design Loading Categories

Loading combinations for the core support structures are shown in Table 3.9-2. The basis for determining faulted loads on the reactor internals is shown for dynamic loads in Section 3.7 and for pipe rupture loads in Subsection 3.9.5.2.3 and 3.9.5.4.3. Table 3.9-2b shows allowable and calculated stress values for highly stressed core support structures and selected reactor internal components. Table 3.9-2aa provides this same type of information for the CRD guide tubes.

Stress intensity and other design limits are discussed in Subsection 3.9.5.4.4. The core support structures which are fabricated as part of the reactor pressure vessel assembly are discussed in Subsection 3.9.1.3 in conjunction with the reactor pressure vessel.

The design requirements for equipment classified as "other" e.g., steam dryers and shroud heads, were specified by the designer with appropriate consideration of the intended service of the equipment and expected plant and environmental conditions under which it will operate. Where possible, design requirements are based on applicable industry codes and standards. If these are not available, the designer relies on accepted industry or engineering practices.

3.9.5.4 Design Bases

3.9.5.4.1 Safety Design Bases

The reactor core support structures and internals shall meet the following safety design bases:

1) They shall be arranged to provide a floodable volume in which the core can be adequately cooled in the event of a breach in the nuclear system process barrier external to the reactor vessel.

2) Deformation shall be limited to assure that the control rods and core standby cooling systems can perform their safety functions.

3) Mechanical design of applicable structures shall assure that safety design bases (1) and (2), above, are satisfied so that the safe shutdown of the plant and removal of decay heat are not impaired.

3.9.5.4.2 Power Generation Design Bases

The reactor core support structures and internals shall be designed to the following power generation design bases:

1) They shall provide the proper coolant distribution during all anticipated normal operating conditions to full power operation of the core without fuel damage.

2) They shall be arranged to facilitate refueling operations.

3) They shall be designed to facilitate inspection.

3.9.5.4.3 Response of Internals Due to Inside Steam Break Accident

It is concluded that the maximum pressure loads acting on the reactor internal components result from an inside steam line break, and on some components the loads are greatest with operation at the minimum power associated with the maximum core flow. This has been substantiated by the analytical comparison of liquid versus steam breaks and by the investigation of the effects of core power and core flow.

It has also been pointed out that, although possible, it is not probable that the reactor would be operating at the rather abnormal condition of minimum power and maximum core flow. More realistically, the reactor would be at or near a full power condition and thus the maximum pressure loads acting on the internal components would be less.

3.9.5.4.4 Stress, Deformation, and Fatigue Limits for Reactor Internals (Except Core Support Structure)

These limits are summarized in Table 3.9-2(b).

Design Condition	SF
Normal	2.25
Upset	2.25
Emergency	1.5
Faulted	1.125

Components inside the reactor pressure vessel such as control rods which must move during accident condition have been examined to determine if adequate clearances exist during emergency and faulted conditions. No mechanical clearance problems have been identified. No plastic deformation occurs in the reactor internal components during emergency or faulted conditions as shown in Subsections 3.9.4 and 3.9.3.1, and Table 3.9-2. This is used in demonstrating that no mechanical interferences exist. No fatigue analysis is required under the faulted conditions due to the low encounter frequency of faulted events and the low number of cycles. The forcing functions applicable to the reactor internals are discussed in Subsection 3.9.2.5.

3.9.5.4.5 Stress, Deformation, and Fatigue Limits for Core Support Structures

These limits are summarized in Tables 3.9-2a, 3.9-2b, and 3.9-2v.

3.9.6 IN-SERVICE TESTING OF PUMPS AND VALVES

The construction permits for the Susquehanna Steam Electric Station were issued in November 1973. Relating this date to the requirements of 10CFR50.55a(g), the preservice examination

program, including provisions for design and access to enable operational readiness testing of pumps and valves, complied, as a minimum, with the 1971 Edition of the ASME B&PV Code Section XI including Addenda through Summer 1972.

This ASME Code Edition does not require preservice and in-service testing of pumps and valves to ensure operational readiness. The requirements for in-service testing of pumps and valves were added as Subsections IWV and IWP to ASME B&PV Code Section XI, Summer 1973 Addenda, effective December 30, 1973. By then, design and procurement for SSES was under way; however, the preservice testing program for pumps and valves for assessing operational readiness was conducted, to the extent practical, so that it complied with requirements of the 1974 Edition, ASME B&PV Code through Winter 1975 Addenda.

The first 120 months' in-service tests will assess operational readiness of pumps and valves. These tests complied, to the extent practical within design limitations, with the requirements of 10CFR50.55a.

During successive 120-month periods, in-service tests of pumps and valves for assessing operational readiness will comply, to the extent practical within design limitations, with the requirements of 10CFR50.55a.

3.9.7 REFERENCES

3.9-1 "Design and Performance of G.E. BWR Jet Pumps," General Electric Company, Atomic Power Equipment Department, APED-5460, July 1968.

3.9-2 Moen, H.H., "Testing of Improved Jet Pumps for the BWR/6 Nuclear System," General Electric Company, Atomic Power Equipment Department, NEDO-10602, June 1972.

3.9-3 General Electric Company, "Analytical Model for Loss-of-Coolant Analysis in Accordance with 10 CFR 50, Appendix K," Proprietary Document, General Electric Company, NEDE-20566.

3.9-4 Not Used

3.9-5 Not Used

3.9-6 Seismic Analysis of Piping Systems, BP-TOP-1, Bechtel Power Corporation, San Francisco, California, Rev. 2, January, 1975.

3.9-7 "Assessment of Reactor Internals Vibration in BWR/4 and BWR/5 Plants," NEDE-24057-P (Class III) and NEDO-24057 (Class I), November, 1977.

3.9-8 "Functional Capability Criteria for Essential Mark II Piping," NEDO-21985, 78 NED174 (Class I), September, 1978.

3.9-9 "Power Uprate Engineering Report for Susquehanna Steam Electric Station, Units 1 and 2," NEDC-32161P, As Revised by PP&L Calculation EC-PUPC-1001, Revision 0, March, 1994.

3.9-10 “BWR Jet Pump Assembly Maintenance Issues,” General Electric Company, San Jose, CA, June 2002

TABLE 3.9-1

TRANSIENTS AND THE NUMBER OF ASSOCIATED CYCLES CONSIDERED IN THE DESIGN AND FATIGUE ANALYSIS OF THE RPV ASSEMBLY AND INTERNAL TRANSIENTS

	No. of Cycles
Normal, Upset, and Testing Conditions	
1. Bolt Up*	123
2. Design Hydrostatic Test	130
3. Startup (100°F/hr Heatup Rate)**	117
4. Daily Reduction to 75% Power*	10,000
5. Weekly Reduction to 50% Power*	2,000
6. Control Rod Pattern Change*	400
7. Loss of Feedwater Heaters (80 Cycles Total):	80
8. 50% Safe Shutdown Earthquake Event at Rated Operating Conditions	10****
9. Scram:	
a. Turbine Generator Trip, Feedwater On, Isolation Valves Stay Open	40
b. Other Scrams	140
10. Reduction to 0% Power, Hot Standby with main condenser available, Shutdown (100°F/hr Cooldown Rate)**	111
11. Unbolt	123
12. Pre-op Blowdown	10
13. Natural Circulation Startup	3
14. Loss of AC Power, Natural Circulation Restart	5
Emergency Conditions	
15. Scram:	
a. Reactor Overpressure with Delayed Scram, Feedwater Stays On, Isolation Valves Stay Open	1***

TABLE 3.9-1 (Continued)

Transients	No. of Cycles
Emergency Conditions (Continued)	
16. Scram	
a. --	--
b. Automatic Blowdown	1***
c. Loss of Feedwater Pumps, Isolation Valves Closed	5
d. Single Safety or Relief Valve Blowdown	8
17. Improper Start of Cold Recirculation Loop	1***
18. Sudden Start of Pump in Cold Recirculation Loop	1***
19. Improper Startup with Reactor Drain Shut Off	1***
Faulted Condition	
20. Pipe Rupture and Blowdown	1***
21. Safe Shutdown Earthquake at Rated Operating Conditions	1***

---

* Applies to reactor pressure vessel only.

** Bulk average vessel coolant temperature change in any 1-hour period.

*** The annual encounter probability for the one cycle events is < 10^{-2} for emergency and 10 for faulted events.

**** Includes 10 maximum load cycles per event. Not required to be considered in fatigue analysis due to low encounter frequency (<10^{-2}) and low number of cycles.

TABLE 3.9-2

INTRODUCTION

This table lists the major mechanical safety related mechanical components in the plant. Various parts of the table are referenced in Section 3.9. The format in the various parts of the table is consistent, since variations exist on analytical methods and depth of detail necessary to demonstrate the safety aspects of various components.

TABLE 3.9-2 INDEX

TABLE 3.9-2 INDEX -- (Continued)

3.9-2p	REACTOR WATER CLEANUP (RWCU) PUMP
3.9-2q	RCIC TURBINE
3.9-2r	RCIC PUMP
3.9-2s	REACTOR REFUELING & SERVICING EQUIPMENT (i) Fuel Storage Racks (ii) Fuel Preparation Machine
3.9-2t	HIGH PRESSURE COOLANT INJECTION (HPCI) PUMP
3.9-2u	CONTROL ROD DRIVE (INDEX TUBE)
3.9-2v	CONTROL ROD DRIVE HOUSING
3.9-2w	JET PUMPS
3.9-2x	NOT USED
3.9-2y	NOT USED
3.9-2z	NOT USED
3.9-2aa	CONTROL ROD GUIDE TUBE
3.9-2ab	INCORE HOUSING
3.9-2ac	NOT USED
3.9-2ad	NOT USED
3.9-2ae	HPCI TURBINE DESIGN CALCULATIONS

TABLE 3.9-2

LOAD COMBINATION AND ACCEPTANCE CRITERIA
FOR ASME CODE CLASS 1, 2, AND 3
NSSS PIPING AND EQUIPMENT

Load Combination	Design Basis	Evaluation Basis	Service Level
N + $SRV_{(ALL)}$	Upset	Upset	(B)
N + OBE	Upset	Upset	(B)
N + OBE + $SRV_{(ALL)}$	Emergency	Upset	(B)
N + SSE + $SRV_{(ALL)}$	Faulted	Faulted*	(D)
N + SBA + SRV	Emergency	Emergency*	(C)
N + IBA + SRV	Faulted	Faulted*	(D)
N + SBA + $SRV_{(ADS)}$	Emergency	Emergency*	(C)
N + SBA + OBE + $SRV_{(ADS)}$	Faulted	Faulted*	(D)
N + IBA + OBE + $SRV_{(ADS)}$	Faulted	Faulted*	(D)
N + SBA/IBA + SSE + $SRV_{(ADS)}$	Faulted	Faulted*	(D)
N + LOCA** + SSE	Faulted	Faulted*	(D)

NOTE: All dynamic loads are combined by SRSS

LOAD DEFINITION LEGEND

Normal(N) - Normal and/or abnormal loads depending on acceptance criteria.

OBE - Operational basis earthquake loads.

SSE - Safe Shutdown earthquake loads.

SRV_{ALL} - The loads induced by the actuation of all safety/relief valves which activate within milliseconds of each other (e.g., turbine trip operational transient).

SRV_{ADS} - The loads induced by the actuation of safety/relief valves associated with Automatic Depressurzation System which actuate within milliseconds of each other during the postulated small or intermediate size pipe rupture.

SRV - Safety/relief-valve-discharge-induced loads from two adjacent valves (one valve actuated when adjacent valve is cycling).

LOAD COMBINATION TABLE 3.9-2 (Cont'd)

LOCA	- The loss of coolant accident associated with the postulated pipe rupture of large pipes (e.g., main steam, feedwater, recirculation piping).
$LOCA_1$	- Pool swell drag/fallout loads on piping and components located between the main vent discharge outlet and the suppression pool water upper surface.
$LOCA_2$	- Pool swell impact loads on piping and components located above the suppression pool water upper surface.
$LOCA_3$	- Oscillating pressure induced loads on submerged piping and components during condensation oscillations.
$LOCA_4$	- Building motion induced loads from chugging.
$LOCA_5$	- Building motion induced loads from main vent air clearing.
$LOCA_6$	- Vertical and horizontal loads on main vent piping.
$LOCA_7$	- Annulus pressurization loads.
SBA	- The abnormal transients associated with a Small Break Accident.
IBA	- The abnormal transients associated with an Intermediate Break Accident.

* All ASME Code Class 1, 2, and 3 piping systems that are required to function for safety shutdown under the postulated events shall meet the requirements of NRC's "Interim Technical Position-Function Capability of passive components" - by MEB.

** The most severe load combination among LOCA.

TABLE 3.9-2a

REACTOR PRESSURE VESSEL AND SHROUD SUPPORT ASSEMBLY

ASME B&PV CODE SEC. III SUBSECTION NB PRIMARY STRESS LIMIT CRITERIA	LOADING	PRIMARY STRESS TYPE	ALLOWABLE STRESS -(psi)	MAXIMUM CALCULATED STRESS (psi)
(i) VESSEL SUPPORT SKIRT				
MATERIAL: SA-516 GR-70				
A. NORMAL & UPSET CONDITION:				
$P_m \leq Sm$ Sm = 19,150 @ 575°F $P_m + P_b \leq 1.5$ Sm 1.5 Sm = 28,725 @ 575°F	1. Deadweight 2. Pressure 3. OBE 4. SRV	PRIMARY MEMBRANE PRIMARY MEMBRANE PLUS BENDING	19,150 28,725	14,566 U1 < 14,566 U2 20,330 U1 < 20,330 U2
B. EMERGENCY CONDITION:				
$P_m \leq Sy$ Sy = 29,425 @ 546°F $P_m + P_b \leq 1.5$ Sy 1.5 Sy = 44,138 @ 546 °F	1. Deadweight 2. Pressure 3. DBA 4. SRV	PRIMARY MEMBRANE PRIMARY MEMBRANE PLUS BENDING	29,425 44,150	20,565 U1 < 20,565 U2 29,377 U1 < 29,377 U2
C. FAULTED CONDITION:				
$P_m \leq 2.4$ Sm 2.4 Sm = 45,960 @ 546°F $P_m + P_b \leq 3.6$ Sm 3.6 Sm = 68,940 @ 546°F	1. Deadweight 2. Pressure 3. Annulus Pressurization 4. SSE 5. LOCA	PRIMARY MEMBRANE PRIMARY MEMBRANE PLUS BENDING	45,960 68,940	36,410 60,570
D. MAXIMUM CUMULATIVE FATIGUE USAGE FACTOR: 0.913 U1 and 0.888 U2 At Skirt Base Junction These CUFs account for the original 40-year plant design. During the period of extended operation, actual fatigue design margins are periodically evaluated in accordance with the Plant Transient and Fatigue Monitoring Program.				

TABLE 3.9-2a (Continued)

REACTOR PRESSURE VESSEL AND SHROUD SUPPORT ASSEMBLY

ASME B&PV CODE SEC. III SUBSECTION NB PRIMARY STRESS LIMIT CRITERIA	LOADING	PRIMARY STRESS TYPE	ALLOWABLE STRESS -(psi)	MAXIMUM CALCULATED STRESS (psi)
(ii) SHROUD SUPPORT				
MATERIAL: INCONEL SB-168				
A. NORMAL & UPSET CONDITION:				
$P_m \leq Sm$ Sm = 23,300 @ 575°F $P_m + P_b \leq 1.5$ Sm 1.5 Sm = 34,950 @ 575°F	1. Deadweight 2. Pressure 3. OBE 4. SRV	PRIMARY MEMBRANE PRIMARY MEMBRANE PLUS BENDING	23,300 35,000	7,600 11,600
B. EMERGENCY CONDITION:				
$P_m \leq Sy$ Sy = 28,125 @ 575°F $P_m + P_b \leq 1.5$ Sy 1.5 Sy = 42,188 @ 575°F	1. Deadweight 2. Pressure 3. SBA 4. SRV	PRIMARY MEMBRANE PRIMARY MEMBRANE PLUS BENDING	35,000 54,400	7,600 11,600
C. FAULTED CONDITION:				
$P_m \leq Sy$[(1)] Sy = 28,125 @ 575°F $P_m + P_b \leq 1.5$ Sy 1.5 Sy = 42,188 @ 575°F	1. Deadweight 2. Pressure 3. Annulus Pressurization 4. SSE	PRIMARY MEMBRANE PRIMARY MEMBRANE PLUS BENDING	35,000 52,400	15,300 23,400
D. MAXIMUM CUMULATIVE FATIGUE USAGE FACTOR: 0.41 At Vessel - Support Plate Junction. This CUF accounts for the original 40-year plant design. During the period of extended operation, actual fatigue design margins are periodically evaluated in accordance with the Plant Transient and Fatigue Monitoring Program.				

(1) Emergency allowable values used.

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TABLE 3.9-2a (Continued)

REACTOR PRESSURE VESSEL AND SHROUD SUPPORT ASSEMBLY

ASME B&PV CODE SEC. III SUBSECTION NB PRIMARY STRESS LIMIT CRITERIA	LOADING	PRIMARY STRESS TYPE	ALLOWABLE STRESS (psi)	MAXIMUM CALCULATED STRESS (psi)
(iii) RPV FEEDWATER NOZZLE				
MATERIAL: SA 508 CL. I.				
A. NORMAL & UPSET CONDITION:				
$P_m \leq Sm$ Sm = 17,700 @ 525°F $P_m + P_b \leq 1.5$ Sm 1.5 Sm = 26,550 @ 525°F	1. Deadweight 2. Pressure 3. OBE 4. SRV	PRIMARY MEMBRANE PRIMARY MEMBRANE PLUS BENDING	17,700 26,550	14,200 19,700
B. EMERGENCY CONDITION:				
$P_m \leq Sy$ Sy = 25,900 @ 594°F $P_m + P_b \leq 1.5$ Sy 1.5 Sy = 38,850 @ 594°F	1. Deadweight 2. Pressure 3. SBA 4. SRV	PRIMARY MEMBRANE PRIMARY MEMBRANE PLUS BENDING	26,500 39,800	15,600 20,800
C. FAULTED CONDITION:				
$P_m \leq 3.0$ Sm 3.0 Sm = 53,100 @ 575°F $P_m + P_b \leq 1.5$ Sy 1.5 Sy = 38,850 @ 594°F	1. Deadweight 2. Pressure 3. LOCA 4. SSE	PRIMARY MEMBRANE PRIMARY MEMBRANE PLUS BENDING	53,100 39,800	28,300(2) 20,800
D. MAXIMUM CUMULATIVE FATIGUE USAGE FACTOR: 0.82 At Safe End. This CUF accounts for the original 40-year plant design. During the period of extended operation, actual fatigue design margins are periodically evaluated in accordance with the Plant Transient and Fatigue Monitoring Program.				

(2) Without Thermal Bending.

TABLE 3.9-2a (Continued)

REACTOR PRESSURE VESSEL AND SHROUD SUPPORT ASSEMBLY

ASME B&PV CODE SEC. III SUBSECTION NB PRIMARY STRESS LIMIT CRITERIA	LOADING	PRIMARY STRESS TYPE	ALLOWABLE STRESS (psi)	MAXIMUM CALCULATED STRESS (psi)
(iv) IV CRD PENETRATION - CRD HOUSING				
MATERIAL: TYPE 304SS				
A. NORMAL & UPSET CONDITION:				
$P_m \leq Sm$ Sm = 16,600 @ 575°F $P_m + P_b \leq 1.5$ Sm 1.5 Sm = 24,990 @ 575°F	1. Deadweight 2. Pressure 3. OBE 4. SRV	PRIMARY MEMBRANE PRIMARY MEMBRANE PLUS BENDING	16,600 24,900	8,506 8,480
B. EMERGENCY CONDITION:				
$P_m \leq 1.2$ Sm 1.2 Sm = 20,000 @ 575°F $P_m + P_b \leq 1.8$ Sm 1.8 Sm = 30,000 @ 575°F	1. Deadweight 2. Pressure 3. SRV 4. SBA	PRIMARY MEMBRANE PRIMARY MEMBRANE PLUS BENDING	19,920 29,880	12,470 12,710
C. FAULTED CONDITION:				
$P_m \leq 2.4$ Sm 2.4 Sm = 39,984 @ 575°F $P_m + P_b \leq 3.6$ Sm 3.6 Sm = 59,975 @ 575°F	1. Deadweight 2. Pressure 3. LOCA 4. SCRAM 5. SSE	PRIMARY MEMBRANE PRIMARY MEMBRANE PLUS BENDING	39,840 59,760	12,470 12,710
D. MAXIMUM CUMULATIVE FATIGUE USAGE FACTOR: 0.208 at Lower CRD Housing. This CUF accounts for the original 40-year plant design. During the period of extended operation, actual fatigue design margins are periodically evaluated in accordance with the Plant Transient and Fatigue Monitoring Program.				

TABLE 3.9-2a (Continued)

REACTOR PRESSURE VESSEL AND SHROUD SUPPORT ASSEMBLY

ASME B&PV CODE SEC. III SUBSECTION NB PRIMARY STRESS LIMIT CRITERIA	LOADING	PRIMARY STRESS TYPE	ALLOWABLE STRESS -(psi)	MAXIMUM CALCULATED STRESS (psi)
(v) CRD PENETRATION - STUB TUBE				
MATERIAL: SB-167 INCONEL				
A. NORMAL & UPSET CONDITION:				
$P_m \leq Sm$ Sm = 20,000 @ 575°F $P_m + P_b \leq 1.5$ Sm 1.5 Sm = 30,000 @ 575°F	1. Deadweight 2. Pressure 3. OBE 4. SRV	PRIMARY MEMBRANE PRIMARY MEMBRANE PLUS BENDING	20,000 30,000	5,005 28,200
B. EMERGENCY CONDITION:				
$P_m \leq Sy$ Sy = 24,100 @ 575°F $P_m + P_b \leq 1.5$ Sy 1.5 Sy = 36,150 @ 575°F	1. Deadweight 2. Pressure 3. SBA 4. SRV	PRIMARY MEMBRANE PRIMARY MEMBRANE PLUS BENDING	24,100 36,150	6,755 30,260
C. FAULTED CONDITION:				
$P_m \leq 2.4$ Sm 2.4 Sm = 48,000 @ 575°F $P_m + P_b \leq 3.6$ Sm 3.6 Sm = 72,000 @ 575°F	1. Deadweight 2. Pressure 3. LOCA 4. SSE	PRIMARY MEMBRANE PRIMARY MEMBRANE PLUS BENDING	48,000 72,000	6,755 30,260
D. MAXIMUM CUMULATIVE FATIGUE USAGE FACTOR: 0.36 At Bottom of Weld. This CUF accounts for the original 40-year plant design. During the period of extended operation, actual fatigue design margins are periodically evaluated in accordance with the Plant Transient and Fatigue Monitoring Program.				

TABLE 3.9-2aa

CONTROL ROD GUIDE TUBE FLANGE

CRITERIA	LOADING	PRIMARY STRESS TYPE	ALLOWABLE STRESS (psi)	CALCULATED STRESS (psi)
CONTROL ROD GUIDE TUBE				
Primary Stress Limit – The allowable primary membrane stress plus bending stress is based on the ASME Boiler and Pressure Vessel Code, Section III for Type 304				
S.S. S_m = 16,000 psi @ 575°F				
For normal and upset condition: $S_{limit} = 1.5\ S_m$ $P_m + P_b$	1. Dead weight 2. Pressure drop across guide tube 3. OBE 4. SRV 5. Scram	Primary membrane and bending (The maximum bending stress occurs at the guide tube base.)	24,000	11,563
For emergency condition: $S_{limit} = 2.25\ S_m$ $P_m + P_b$	1. Dead weight 2. Pressure 3. OBE 4. SRV 5. Chugging	Primary membrane and bending	36,000	26,103
For faulted condition: $S_{limit} = 2.4\ S_m$ $P_m + P_b$	1. Dead weight 2. Pressure 3. Chugging 4. SRV 5. SSE	Primary membrane and bending	38,400	27,010

TABLE 3.9-2ab

INCORE HOUSING

CRITERIA	LOADING	PRIMARY STRESS TYPE	ALLOWABLE STRESS (psi)	CALCULATED STRESS (psi)
Primary Stress Limit – The allowable primary membrane stress is based on ASME Boiler and Pressure Vessel Code, *Section III* for Class I vessels, for Type 304 stainless steel				
P_m = 16,660 psi @ 575°F				
For normal and upset conditions: $S_{limit} = P_m$	1. Design pressure 2. OBE 3. SRV	Maximum membrane stress occurs at the outer surface of the vessel penetration	16,660	6,000[(1)] 15,548[(2)]
For faulted condition: $S_{limit} = 2.4\ P_m$	1. Design pressure 2. SSE 3. LOCA 4. Annulus pressurization	Maximum membrane stress occurs at the outer surface of the vessel penetration	39,840	6,450[(1)] 27,830[(2)]

NOTE: The emergency condition loads are less than the normal/upset loads.

(1) Unit No. 1
(2) Unit No. 2

TABLE 3.9-2 (ac)

REACTOR VESSEL SUPPORT EQUIPMENT

CRD HOUSING SUPPORT

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TABLE 3.9-2 (ae)

HPCI TURBINE DESIGN CALCULATIONS

TURBINE PART	CALCULATED	ALLOWABLE
Pressure Boundary Castings		
Stop Valve	8,975 psi	14,000 psi
Turbine Inlet (high pres.)	6,550 psi	14,000 psi
Turbine Wheel Case (low pres.)	6,000 psi	14,000 psi
Pressure Boundary Bolting		
Stop Valve	17,700 psi	20,000 psi
Turbine	18,290 psi	20,000 psi
Non-Pressure Boundary Components		
Turbine Shaft	5,000 psi	60,500 psi
Thrust Bearing	4,400 lbf	5,600 lbf
Journal Bearing	2,680 lbf	19,500 lbf
Stop Valve Yoke	13,500 psi	33,000 psi
Pedestal Dowel Pins	29,800 psi	61,100 psi
Pedestal Bolts	11,400 lbf	28,300 lbf

TABLE 3.9 - 2 (af), page 1 of 2

HIGH DENSITY SPENT FUEL RACKS

TYPES OF ANALYSIS PERFORMED

DYNAMIC ANALYSIS:

A dynamic modal analysis using the seismic, SRV, and LOCA response spectra was performed on a simplified model consisting of 6 racks (1 quadrant). The resulting loads on the corner module were extracted and a more detailed analysis performed.

STATIC ANALYSIS:

A detailed finite element (1364 elements) model of the corner module was developed and a static analysis performed using the loading results of the dynamic analysis. The section descriptions, allowable stresses and stress ratios for the detailed model are given on page 2 of this table.

FUEL RATTLING ANALYSIS:

A time history analysis was performed to determine local impact loads due to fuel rattling. A comparison of the support loads from the fuel rattling analysis with those of the response spectrum analysis showed that the fuel rattling results are less than or equal to the response spectrum results. Analysis of the poison can was completed using the local impact loads.

MODEL IMPACT ANALYSIS:

An equivalent static load was determined for the following drop conditions:

1) 18" fuel drop on corner of top casting

2) 18" fuel drop on middle of top casting

3) fuel drop full length through the cavity impacting bottom casting at the middle.

For the first 2 cases the equivalent static loads calculated were combined with dead load and applied to the detailed model. For the 3rd case, the ultimate load of the bundle shearing out of the fuel seat was determined and combined with dead load. This combined load was then applied to the detailed model.

TABLE 3.9-2 (af), page 2 of 2
HIGH DENSITY SPENT FUEL RACK
SUMMARY OF RESULTS FOR THE DETAILED MODEL ELEMENTS

SECT. NO.	SECTION DESCRIPTION	NORMAL ALLOWABLE STRESSES			NORMAL OPERATING CONDITION				DESIGN ACCIDENT AND EXTREME ENVIRONMENTAL CONDITIONS			
		F_a	F_{by}	F_{bx}	$\frac{f_a}{F_a}$	$\frac{f_{by}}{F_{by}}$	$\frac{f_{bx}}{F_{bx}}$	MAX STRESS RATIO (1)	$\frac{f_a}{F_a}$	$\frac{f_{by}}{F_{by}}$	$\frac{f_{bx}}{F_{by}}$	MAX STRESS RATIO(1)
1	Top Grid Outer Section	9941	15760	15760	.026	.009	.747	.78	.018	.006	.715	.74
2	Top Grid Inner Section	9420	15760	15760	.057	.055	.813	.93	.040	.039	.766	.85
3	Bottom Grid Outer Sect.	8830	15760	12120	.062	.248	.108	.42	.062	.248	.108	.42
4	Bottom Grid Inner Sect.	8550	15760	12120	.005	.831	.013	.85	.005	.831	.013	.85
5	Bottom Grid Outer Section Near Leg	9650	15760	12120	.047	.249	.269	.57	.047	.249	.269	.57
6	Bottom Grid Inner Section Near Leg	9530	15760	12120	.046	.508	.248	.80	.046	.508	.248	.80
7	Bottom Grid Foot	10250	15760	12120	.132	0	.001	.13	.160	0	.003	.16
8	Bottom Grid Foot	11020	14180	14180	.161	0	.003	.16	.195	0	.006	.20
9	1/2" Plate	3320	F_v = 1390		-	-	-	$.99^{(2)}$	-	-	-	$.76^{(2)}$
10	7/8" Plate	17370	F_v = 10970		-	-	-	$.92^{(2)}$	-	-	-	$.92^{(2)}$

(1) Stress Ratio $= \frac{f_a}{F_a} + \frac{f_{by}}{F_{by}} + \frac{f_{bx}}{F_{bx}}$

(2) Plate Stress Ratio $= \frac{f_y}{F_a} + \frac{f_x}{F_a}$

NOTE
Allowable stresses are factored up per Table 9.1-7a of the SSES-FSAR.

TABLE 3.9-2b

REACTOR INTERNALS AND ASSOCIATED EQUIPMENT

ASME B&PV CODE SEC. III SUBSECTION NG PRIMARY STRESS LIMIT CRITERIA	LOADING	PRIMARY STRESS TYPE	ALLOWABLE STRESS (psi)	MAXIMUM CALCULATED STRESS (psi)
(i) TOP GUIDE - BEAM WITH HIGHEST STRESS				
MATERIAL: SA-240 TYPE 304SS				
A. NORMAL & UPSET CONDITION:				
$P_m \leq Sm$ Sm = 16,900 @ 550°F $P_m + P_b \leq 1.5$ Sm 1.5 Sm = 25,350 @ 550°F	1. Normal 2. Pressure 3. OBE 4. SRV	PRIMARY MEMBRANE PRIMARY MEMBRANE PLUS BENDING	16,950 25,430	2,770 11,500
B. EMERGENCY CONDITION:				
$P_m \leq 1.5$ Sm 1.5 Sm = 25,350 @ 550°F $P_m + P_b \leq 2.25$ Sm 2.25 Sm = 38,025 @ 550°F	1. Normal 2. Pressure 3. Chugging 4. SRV	PRIMARY MEMBRANE PRIMARY MEMBRANE PLUS BENDING	25,430 38,140	730 9,600
C. FAULTED CONDITION:				
$P_m \leq 2.4$ Sm 2.4 Sm = 40,560 @ 550°F $P_m + P_b \leq 3$ Sm 3 Sm = 50,7000 @ 550°F	1. Normal 2. Pressure 3. SRV 4. SSES 5. LOCA	PRIMARY MEMBRANE PRIMARY MEMBRANE PLUS BENDING	40,680 61,020	3,500 11,800
D. MAXIMUM CUMULATIVE USAGE FACTOR: 0.22 At Beam-Slot Location This CUF was evaluated for the period of extended operation and found to be acceptable.				

TABLE 3.9-2b (Continued)

REACTOR INTERNALS AND ASSOCIATED EQUIPMENT

ASME B&PV CODE SEC. III SUBSECTION NG PRIMARY STRESS LIMIT CRITERIA	LOADING	PRIMARY STRESS TYPE	ALLOWABLE STRESS (psi)	MAXIMUM CALCULATED STRESS (psi)
(ii) CORE PLATE (LIGAMENT IN TOP PLATE)				
MATERIAL: TYPE 304SS				
A. NORMAL & UPSET CONDITION:				
$P_m \leq Sm$ Sm = 16,900 @ 550°F $P_m + P_b \leq 1.5$ Sm 1.5 Sm = 25,350 @ 550°F	1. Normal 2. Pressure 3. OBE 4. SRV	PRIMARY MEMBRANE PRIMARY MEMBRANE PLUS BENDING	16,900 25,470	783 10,140
B. EMERGENCY CONDITION:				
$P_m \leq 1.5$ Sm 1.5 Sm = 25,350 @ 550°F $P_m + P_b \leq 2.25$ Sm 2.25 Sm = 38,025 @ 550°F	1. Normal 2. Pressure 3. Chugging 4. SRV	PRIMARY MEMBRANE PRIMARY MEMBRANE PLUS BENDING	25,350 38,200	647 [1] 12,460
C. FAULTED CONDITION:				
$P_m \leq 2.4$ Sm 2.4 Sm = 40,560 @ 550°F $P_L + P_b \leq 3$ Sm 3 Sm = 50,700 @ 550°F	1. Normal 2. Pressure 3. SRV 4. SSE 5. LOCA	PRIMARY MEMBRANE PRIMARY MEMBRANE PLUS BENDING	40,560 61,020	9,570 [1] 15,710
D. MAXIMUM CUMULATIVE USAGE FACTOR: 0.005 At Beam-Rim Junction. This CUF was evaluated for the period of extended operation and found to be acceptable.				
[1] Value is not given in New Loads Report. However, Primary and Bending Stresses are, and these are bounding.				

TABLE 3.9-2b (Continued)

REACTOR INTERNALS AND ASSOCIATED EQUIPMENT

ASME B&PV CODE SEC. III SUBSECTION NG PRIMARY STRESS LIMIT CRITERIA	LOADING	PRIMARY STRESS TYPE	ALLOWABLE STRESS (psi)	MAXIMUM CALCULATED STRESS (psi)
(ii) VESSEL HEAD SPRAY NOZZLE				
MATERIAL: CARBON STEEL SA508,CL1				
A. NORMAL & UPSET CONDITION:				
$P_m \leq$ * Sm = 16,670 @ 575°f $P_L + P_b \leq 3.5$ Sm 1.5 Sm = 50,000 @ 575°F	1. Normal 2. Pressure 3. OBE 4. SRV	PRIMARY MEMBRANE PRIMARY MEMBRANE PLUS BENDING	50,000	28,700
B. EMERGENCY CONDITION:				
$P_L + P_b \leq 1.8$ Sm 1.8 Sm - 30,000 @ 575°F	1. Normal 2. Pressure 3. Chugging 4. SRV	PRIMARY MEMBRANE PLUS BENDING	30,000	28,700
C. FAULTED CONDITION:				
$P_L + P_b \leq 3$ Sm 3 Sm = 50,000 @ 575°F $P_L + P_b \leq 3$ Sm 3 Sm = 50,7000 @ 550°F	1. Normal 2. Pressure 3. SRV 4. SSE 5. LOCA	PRIMARY MEMBRANE PLUS BENDING	50,000	28,700
D. MAXIMUM CUMULATIVE USAGE FACTOR: Satisfied per N-415-1				

* Not required per NB-3222.

TABLE 3.9-2(c)

REACTOR WATER CLEANUP HEAT EXCHANGERS

Part	Required Thickness (In.)	Allowable Stresses (psi)	Actual Thickness (in.)
REGENERATIVE CU HX			
Shell	.858	15,000	1.156
Shell head	.704	17,500	1.0
Channel shell	.917	15,900	1.0*
Tubesheet	2.87	15,900	2.875*
Tubes	.084	14,025	.095
Piping	.240	15,000	.337
Channel cover	3.53	17,500	3.75*
NON-REGENERATIVE CU HX			
Shell	.168	15,000	.375
Shell head	.144	17,500	.375
Channel shell	.917	15,900	1.0*
Channel cover	3.53	17,500	3.75*
Tubesheet	2.87	15,900	2.875*
Tubes	.056	11,900	0.65
Channel piping	.240	15,000	.337
Shell piping	.073	15,000	.322

Values within 10%

Table 3.9-2d

ASME CODE CLASS 1 MAIN STEAM PIPING AND PIPE MOUNTED EQUIPMENT – HIGHEST STRESS SUMMARY – UNIT 1

ACCEPTANCE CRITERIA	LIMITING STRESS TYPE	CALCULATED STRESS OR USAGE FACTOR	ALLOWABLE LIMITS	RATIO ACTUAL/ ALLOWABLE	LOADING	IDENTIFICATION OF LOCATIONS OF HIGHEST STRESS POINTS – NODG POINT NUMBERS
ASME B&PV Code Section III, NB-3600						
Design Condition: Eq.9 $\leq$ 1.5 S_m	Primary	26,473 psi	26,550 psi	0.997	1. Pressure 2. Weight 3. OBE	M.S. Line ‘A’ Sweetpolet 543
Service Levels A&B (Normal & Upset) Conditions: Eq.12 $\leq$ 3.0 S_m	Secondary	27,403 psi	53,100 psi	0.516	1. Thermal	M.S. Line ‘B’ Sockolet 266
Service Levels A&B (Normal & Upset) Condition: Eq.13 $\leq$ 3.0 S_m	Primary Plus Secondary (Except Thermal Expansion)	52,566 psi	54,252 psi	0.969	1. Pressure 2. Weight 3. OBE 4. Operating transient 5. SRV	M.S. Line ‘D’ Sweepolet 850
Service Levels A&B (Normal & Upset) Conditions: Cumulative Usage Factor	N/A	0.8497*	1.0	0.8497	1. Pressure 2. Thermal 3. OBE 4. Operating Transient 5. SRV	M.S. Line ‘D’ TTJ @ Sweepolet 990
Service Level B (Upset) Condition: Eq.9 $\leq$ 1.8 S_m & 1.5S_y	Primary	27,770 psi	31,860 psi	0.872	1. Pressure 2. Weight 3. OBE 4. Operating transient 5. SRV	M.S. Line ‘D’ Sweepolet 850
Service Level C (Emergency) Condition: Eq.9 $<$ 2.25 S_m & 1.8 S_y	Primary	28,432 psi	39,820 psi	0.714	1. Pressure 2. Weight 3. OBE 4. Operating transient 5. SRV 6. SBA	M.S. Line ‘D’ Sweepolet 880
Service Level D (Faulted) Condition: Eq. 9 $<$ 3.0 S_m	Primary	57,585 psi	60,000 psi	0.960	1. Pressure 2. Weight 3. SSE 4. LOCA 5. Operating Transient 6. IBA 7. SRV	M.S. Line ‘C’ Sweepolet 620

Table 3.9-2d (Continued)

ASME CODE CLASS 1 MAIN STEAM PIPING AND PIPE MOUNTED EQUIPMENT – UNIT 1

COMPONENT LOAD TYPE	HIGHEST CALCULATED LOAD	ALLOWABLE LOAD	RATIO CALCULATED/ ALLOWABLE	LOADING	IDENTIFICATION OF EQUIPMENT WITH HIGHEST LOADS
Snubber/ Service Level 'B' Loads	27,693	50,000	0.554	1. Pressure 2. Weight 3. OBE 4. Operating transient	M.S. Line 'A' (S035) H35
Snubber/ Service Level 'C' Loads	35,798	159,600	0.224	1. Pressure 2. Weight 3. SBA 4. Operating transient	M.S. Line 'D' (S038) H38
Snubber/ Service Level 'D' Loads	58,063	160,000	0.363	1. Pressure 2. Weight 3. IBA 4. SSE 5. Operating transient	M.S. Line 'D' (S038) H38
SRV/Horizontal Acceleration	3.981 g	5.2g	0.766	1. Pressure 2. Weight 3. LOCA 4. SSE	Line C SRR Inlet
SRV/Vertical Acceleration	2.650 g	4.4 g	0.602	1. Pressure 2. Weight 3. LOCA 4. SRV	Line B SRP Inlet
MSIV Bonnet/ Axial Force	7,682 lb	41,221 lb	0.186	1. Pressure 2. Weight 3. LOCA 4. SSE	Line A MSIV Bonnet
MSIV Bonnet/ Bending Moment	1,242,707 in-lbs	1,589,000 in-lb	0.782	1. Pressure 2. Weight 3. LOCA 4. SSE	Line B MSIV Bonnet

* This CUF accounts for the original 40-year plant design. During the period of extended operation, actual fatigue design margins are periodically evaluated in accordance with the Plant Transient and Fatigue Monitoring Program.

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Table 3.9-2d.1
ASME CODE CLASS 1 MAIN STEAM PIPING AND PIPE MOUNTED EQUIPMENT –

HIGHEST STRESS SUMMARY – UNIT 2

ACCEPTANCE CRITERIA	LIMITING STRESS TYPE	CALCULATED STRESS OR USAGE FACTOR	ALLOWABLE LIMITS	RATIO ACTUAL/ ALLOWABLE	LOADING	IDENTIFICATION OF LOCATIONS OF HIGHEST STRESS POINTS – NODG POINT NUMBERS
ASME B&PV Code Section III, NB-3600						
Design Condition: Eq. 9 ≤ 1.5 S_m	Primary	26,480 psi	26,550 psi	0.997	1. Pressure 2. Weight 3. OBE 4. SRV	M.S. Line 'A' Sweepolet 990
Service Levels A&B (Normal & Upset) Condition: Eq. 12 ≤ 3.0 S_m	Secondary	32,295 psi	54,252 psi	0.595	1. Thermal	M.S. Line 'C' Sweepolet 641
Service Levels A&B (Normal & Upset) Condition: Eq.13 ≤ 3.0 S_m	Primary Plus Secondary (Except Thermal Expansion)	54,046 psi	54,252 psi	0.996	1. Pressure 2. Weight 3. OBE 4. SRV	M.S. Line 'A' Sweepolet 790
Service Levels A&B (Normal & Upset) Conditions: Cumulative Usage Factor	N/A	0.683*	1.0	0.683		M.S. Line 'B' TTJ @ Sweepolet 932
Service Level B (Upset) Condition: Lower{Eq.9 ≤ 1.8 S_m Of {EQ 9 ≤ 1.5S_y	Primary	30,010 psi	31,860 psi	0.942	1. Pressure 2. Weight 3. OBE 4. SRV	M.S. Line 'D' Sweepolet 90
Service Level C (Emergency) Condition: Lower{Eq.9<2.25 S_m Of {Eq 9 < 1.8 S_y	Primary	30,110 psi	39,820 psi	0.756	1. Pressure 2. Weight 3. SBA 4. OBE 5. SRV	M.S. Line 'D' Sweepolet 75
Service Level D (Faulted) Condition: Eq. 9 <3.0 S_m	Primary	58,915psi	60,000 psi	0.982	1. Pressure 2. Weight 3. SSE 4. LOCA 5. IBA 6. SRV	M.S. Line 'A' Sweepolet 790

Table 3.9-2d.1 (Continued)

ASME CODE CLASS 1 MAIN STEAM PIPING AND PIPE MOUNTED EQUIPMENT – UNIT 2

COMPONENT/LOAD TYPE	HIGHEST CALCULATED LOAD	ALLOWABLE LOAD	RATIO CALCULATED/ALLOWABLE	LOADING	IDENTIFICATION OF EQUIPMENT WITH HIGHEST LOADS
Snubber/ Service Level ‘B’ Loads	44,848	50,000	0.897	1. Pressure 2. Weight 3. OBE 4. Operating transient	M.S. Line ‘A’ 120; H42
Snubber/ Service Level ‘C’ Loads	42,043	70,350	0.598	1. Pressure 2. Weight 3. SBA 4. Operating transient	M.S. Line ‘A’ 120; H42
Snubber/ Service Level ‘D’ Loads	74,120	91,000	0.815	1. Pressure 2. Weight 3. IBA 4. SSE 5. Operating transient	M.S. Line ‘A’ 120; H42
SRV/Horizontal Acceleration	4.063	5.2 g	0.781	1. Pressure 2. Weight 3. LOCA 4. SSE	Line ‘C’ SRV ‘R’
SRV/Vertical Acceleration	2.499 g	4.4 g	0.568	1. Pressure 2. Weight 3. LOCA 4. SRV	Line ‘D’ SRV ‘K’
MSIV Bonnet/ Axial Force	7,340	41,221 lb	0.178	1. Pressure 2. Weight 3. LOCA 4. SSE	Line ‘A’ MSIV Bonnet
MSIV Bonnet/ Bending Moment	1,369,899 in-lbs	1,589,000 in-lb	0.862	1. Pressure 2. Weight 3. LOCA 4. SSE	Line ‘B’ MSIV Bonnet

* This CUF accounts for the original 40-year plant design. During the period of extended operation, actual fatigue design margins are periodically evaluated in accordance with the Plant Transient and Fatigue Monitoring Program.

Table 3.9-2e

ASME CODE CLASS 1 RECIRCULATION PIPING AND PIPE MOUNTED EQUIPMENT – HIGHEST STRESS SUMMARY – UNIT 1

ACCEPTANCE CRITERIA	LIMITING STRESS TYPE	CALCULATED STRESS OR USAGE FACTOR	ALLOWABLE LIMITS	RATIO ACTUAL/ ALLOWABLE	LOADING	IDENTIFICATION OF LOCATIONS OF HIGHEST STRESS POINTS - NODG POINT NUMBERS
ASME B&PV Code Section III, NB-3600						
Design Condition: Eq. 9 $\leq$ 1.5 S_m	Primary	24,097 psi	25,013 psi	0.963	1. Pressure 2. Weigh 3. OBE	Loop 'B' Elbow-784
Service Levels A & B (Normal & Upset) Conditions: Eq. 12 $\leq$ 3.0 S_m	Secondary	36,019 psi	50,025 psi	0.720	1. Thermal	Loop 'B' Weldolet 508
Service Levels A & B (Normal & Upset) Condition: Eq. 13 $\leq$ 3.0 S_m	Primary Plus Secondary (Except Thermal Expansion)	49,544 psi	50,025 psi	0.990	1. Pressure 2. Weight 3. OBE 4. Operating Transient 5. SRV	Loop 'B' TTJ@Valve End 782
Service Levels A & B (Normal and Upset) Conditions: Cumulative Usage Factor	N.A.	0.5555**	1.0	0.555	1. Pressure 2. Thermal 3. OBE 4 .Operating Transient 5. SRV	Loop 'B' Half Coupling 727
Service Level B (Upset) Condition: Eq. 9 $\leq$ 1.8 S_m & 1.5 S_y	Primary	25,298 psi	27,750 psi	0.912	1. Pressure 2. Weight 3. OBE 4. Operating Transient 5. SRV	Loop 'B' Elbow 784

Table 3.9-2e

ASME CODE CLASS 1 RECIRCULATION PIPING AND PIPE MOUNTED EQUIPMENT – HIGHEST STRESS SUMMARY – UNIT 1

ACCEPTANCE CRITERIA	LIMITING STRESS TYPE	CALCULATED STRESS OR USAGE FACTOR	ALLOWABLE LIMITS	RATIO ACTUAL/ ALLOWABLE	LOADING	IDENTIFICATION OF LOCATIONS OF HIGHEST STRESS POINTS - NODG POINT NUMBERS
Service Level C (Emergency) Condition: Eq. 9 < 2.25 S_m & 1.8 S_y	Primary	28,801 psi	33,300 psi	0.865	1. Pressure 2. Weight 3. OBE 4. Operating Transient 5. SRV 6. SBA	Loop 'B' Weldolet 508
Service Level D (Faulted) Condition: Eq. 9 < 3.0 S_m	Primary	41,019 psi	50,025 psi	0.820	1. Pressure 2. Weight 3 .LOCA 4. Operating Transient 5. SSE 6. IBA 7. SRV	Loop 'B' Elbow 784

Table 3.9-2e (Continued)

ASME CODE CLASS 1 RECIRCULATION AND PIPE MOUNTED EQUIPMENT – UNIT 1

COMPONENT/LOAD TYPE	HIGHEST CALCULATED LOAD	ALLOWABLE LOAD	RATIO CALCULATED/ ALLOWABLE	LOADING	IDENTIFICATION OF EQUIPMENT WITH HIGHEST LOADS
Snubber/ Service Level 'B' Loads	56,085	120,000	0.467	1. Pressure 2. Weight 3. OBE 4. Operating transient	Loop 'A' (SAI) H46
STRUT Service Level 'C' Loads	68,094	87,840	0775	1. Pressure 2. Weight 3. SBA 4. Operating transient	Loop 'A' (PO1) H50
STRUT Service Level 'D' Loads	94,493	121,896	0.775	1. Pressure 2. Weight 3. LOCA 4. Operating transient	Loop 'A' (PO1) H50
Discharge Gate Valve/ Horizontal Acceleration	7.296	9.8	0.744	1. Pressure 2. Weight 3. LOCA 4. Operating transient	Loop 'B' Operator
Discharge Gate Vale/ Vertical Acceleration	2.2	2.2	1.0	1. Pressure 2. Weight 3. LOCA 4. Operating transient	Loop 'A' Operator
By-Pass Valve/ Horizontal (East/West) Acceleration	4.336	4.336	1.0	1. Pressure 2. Weight 3. LOCA 4. Operating transient	Loop 'A' Operator
By-Pass Valve/ Horizontal (North South) Acceleration	2.605	2.605	1.0	1. Pressure 2. Weight 3. LOCA 4. Operating transient	Loop 'A' Operator
By-Pass Valve/ Vertical Acceleration	2.514	2.62	0.960	1. Pressure 2. Weight 3. LOCA 4. Operating transient	Loop 'A' Operator
Suction Gate Valve/ Horizontal & Vertical Acceleration*	5.24	9.87	0.531	1. Pressure 2. Weight 3. LOCA 4. Operating transient	Loop 'A' Operator
Recirc. Pump/ Horizontal Acceleration	2.338	2.7	0.866	1. Pressure 2. Weight 3. LOCA 4. SSE	Loop 'A' Pump Motor C.G.
Recirc. Pump/ Vertical Acceleration	1.004	2.1	0.478	1. Pressure 2. Weight 3. LOCA 4. SSE 5. SRN	Loop 'B' Pump Motor C.G.

* Allowable acceleration applies to operator only; valve body is qualified based upon allowable moment and allowable axial force.

** This CUF accounts for the original 40-year plant design. During the period of extended operation, actual fatigue design margins are periodically evaluated in accordance with the Plant Transient and Fatigue Monitoring Program.

Table Rev. 51

Table 3.9-2e.1
ASME CODE CLASS 1 RECIRCULATION PIPING AND PIPE MOUNTED EQUIPMENT –
HIGHEST STRESS SUMMARY – UNIT 2

ACCEPTANCE CRITERIA	LIMITING STRESS TYPE	CALCULATED STRESS OR USAGE FACTOR	ALLOWABLE LIMITS	RATIO ACTUAL/ ALLOWABLE	LOADING	IDENTIFICATION OF LOCATIONS OF HIGHEST STRESS POINTS – NODG POINT NUMBERS
ASME B&PV Code Section III, NB-3600						
Design Condition: Eq.9 ≤ 1.5 S_m	Primary	24,712 psi	25,013 psi	0.988	1. Pressure 2. Weight 3. OBE 4. SRV	Loop 'B' Elbow 765
Service Levels A & B (Normal & Upset) Condition: Eq.12 ≤ 3.0 S_m	Secondary	35,891 psi	50,025 psi	0.717	1. Thermal	Loop 'B' Weldolet 305
Service Levels A & B (Normal & Upset) Condition: Eq.13 ≤ 3.0 S_m	Primary Plus Secondary (Except Thermal Expansion)	49,585 psi	50,025 psi	0.991	1. Pressure 2. Weight 3. OBE 4. SRV	Loop 'B' Half Coupling 300
Service Levels A & B (Normal & Upset) Conditions: Cumulative Usage Factor	N/A	0.7938**	1.0	0.7938		Loop 'B' Half Coupling 380
Service Level B (Upset) Condition: Lower{Eq. 9 ≤ 1.8 S_m Of {EQ 9 ≤ 1.5 S_y	Primary	26,160 psi	27,750 psi	0.943	1. Pressure 2. Weight 3. OBE 4. SRV	Loop 'A' Elbow 575
Service Level C (Emergency) Condition: Lower{Eq. 9 < 2.25 S_m Of {Eq. 9 < 1.8 S_y	Primary	26,650 psi	33,300 psi	0.800	1. Pressure 2. Weight 3. SRV 4. SBA 5. OBE	Loop 'B' Weldolet 275
Service Level D (Faulted) Condition: Eq. 9 < 3.0 S_m	Primary	41,120 psi	50,025 psi	0.822	1. Pressure 2. Weight 3. LOCA 4. SSE 5. SRV 6. IBA	Loop 'A' 4 inch Weldolet 315

Table 3.9-2e.1
ASME CODE CLASS 1 RECIRCULATION AND PIPE MOUNTED EQUIPMENT – UNIT 2

COMPONENT/ LOAD TYPE	HIGHEST CALCULATED LOAD	ALLOWABLE LOAD	RATIO CALCULATED/ ALLOWABLE	LOADING	IDENTIFICATION OF EQUIPMENT WITH HIGHEST LOADS
Strut/ Service Level 'B' Loads	57,612	62,800	0.92	1. Pressure 2. Weight 3. OBE 4. Operating transient	Loop 'A' PO1; H50
Strut/ Service Level 'C' Loads	79,132	86,250	0.92	1. Pressure 2. Weight 3. SBA 4. Operating transient	Loop 'A' PO1; H50
Strut/ Service Level 'D' Loads	96,643	105,340	0.92	1. Pressure 2. Weight 3. LOCA 4. Operating transient	Loop 'A' PO1; H50
Discharge Gate Valve/Horizontal Acceleration	7.678	9.8	0.783	1. Pressure 2. Weight 3. LOCA 4. Operating transient	Loop A' Operator
Discharge Gate Valve/ Vertical Acceleration	2.118	2.188	0.968	1. Pressure 2. Weight 3. LOCA 4. Operating transient	Loop 'B' Operator
By-Pass Valve/ Horizontal Acceleration (East-West)	5.438	5.468	0.995	1. Pressure 2. Weight 3. LOCA 4. Operating transient	Loop 'B' Operator
By-Pass Valve/ Horizontal (North-South)	2.539	2.539	1.000	1. Pressure 2. Weight 3. LOCA 4. Operating transient	Loop A'
Vertical Acceleration	2.376	2.376	1.000	5. Pressure 6. Weight 7. LOCA 8. Operating transient	Loop 'A' Body
Suction Valve/ Horizontal & Vertical Acceleration*	7.714	9.87	0.782	1. Pressure 2. Weight 3. LOCA 4. Operating transient	Loop 'A' Operator
Recirc. Pump/Horizontal Acceleration	2.386	2.7	0.884	1. Pressure 2. Weight 3. LOCA 4. SSE	Loop 'A' Pump Motor C.G.
Vertical Acceleration	1.388	2.1	0.661	1. Pressure 2. Weight 3. LOCA 4. SSE 5. SRN	Loop 'B' Pump Motor C.G.

* Allowable acceleration applies to operator only; valve body is qualified based upon allowable moment and allowable axial force.

** This CUF accounts for the original 40-year plant design. During the period of extended operation, actual fatigue design margins are periodically evaluated in accordance with the Plant Transient and Fatigue Monitoring Program.

TABLE 3.9-2g

SAFETY/RELIEF VALVES (MAIN STEAM)

CRITERIA	METHOD OF ANALYSIS	ALLOWABLE STRESS OR MINIMUM THICKNESS REQUIRED	CALCULATED STRESS OR ACTUAL THICKNESS
	Unless otherwise specified, all references are to ASME B and PVC Section III (July 1971).		
1. Body Inlet and Outlet Flange Stresses	Based on ANSI B31.7-1969, PARA. 1-704.5.1 & ASME BPVC Criteria		
	$S_H = \frac{fM_o}{Lg^2B} + \frac{P_bB}{4g_1} < 1.5\ S_m$		
	$S_R = \frac{(4/3te + 1)M_o}{IT^2J} < 1.5\ S_m$		
	$S_T = \frac{YM_o}{t^2J} - Z\ S_R < 1.5\ S_m$		
	Where		
	S_H = Longitudinal "Hub" Wall Stress, PSI S_R = Residual "Flange" (Body Base, Inlet) Stress, PSI S_T = Tangential "Flange" Stress, PSI		
	For Inlet		
	As S_H < 27,300 PSI (1.5 S_m @ 575°F for A-105GII)	27,300 PSI	See Note 1
	S_R<27,300 PSI (1.5 S_m)	27,300 PSI	See Note 1
	S_T< 27,300 PSI (1.5 S_m) Criteria Satisfied	27,300 PSI	See Note 1
	For Outlet		

TABLE 3.9-2g

SAFETY/RELIEF VALVES (MAIN STEAM)

CRITERIA	METHOD OF ANALYSIS	ALLOWABLE STRESS OR MINIMUM THICKNESS REQUIRED	CALCULATED STRESS OR ACTUAL THICKNESS
	As S_H < 27,300 PSI (1.5 S_m)	29,100 PSI	See Note 1
	S_R <27,300 PSI (1.5 S_m)	29,100 PSI	See Note 1
	S_T < 27,300 PSI (1.5 S_m)	29,100 PSI	See Note 1
2. Inlet and Outlet Stud Area Requirements	Based on USAS B31.7-1969, PARA. 1-704.5.1 and ASME BPVC Total Cross-Sectional Area of Bolt Shall Equal or Exceed the Greater of		
	$AM_1 = \frac{Wm_1}{S_b}$ or $AM_2 = \frac{WM_2}{S_a}$		
	Where:		
	AM_1 = The total required bolt (stud) area for operating condition, in^2.		
	AM_2 = Total required bolt (stud) area for gasket seating, in^2.		
	For Inlet		
	AM_1 = 9,060 in^2 > AM_2 = 0.807 in^2	9.060 in^2	See Note 1
	As AM_1 = 9,060 in^2		
	(Am, actual stud area Criterion Satisfied)		

TABLE 3.9-2g

SAFETY/RELIEF VALVES (MAIN STEAM)

CRITERIA	METHOD OF ANALYSIS	ALLOWABLE STRESS OR MINIMUM THICKNESS REQUIRED	CALCULATED STRESS OR ACTUAL THICKNESS
	For Outlet		
	$Am_1 = 4.76\ in^2 > Am_2 = 1.55\ in^2$		
	As $Am_1 = 4.76\ in^2$	$4.76\ in^2$	See Note 1
	(Am, actual stud area Criterion Satisfied)		
3. Nozzle Wall Thickness	Based on PARA. NB-3530, NB-3540 and NB-3550 for non-standard primary pressure rating.		
	1. Valve Wall Thickness Criterion		
	$t_{min.} < t_A$		
	Where $t_{min.}$ = Minimum Calculated Required Thickness With Corrosion Allowable, in.		
	t_A = Actual Nozzle Wall Thickness, in.		
	i) For thinnest section near valve seat		
	As $t_{min.}$ = 0.467 in. (t_a, actual)	0.467 in.	See Note 1.
	ii) For section about mid-section of nozzle		
	As $t_{min.}$ = 0.468 in.. (t_a, actual)	0.468 in.	See Note 1.

TABLE 3.9-2g

SAFETY/RELIEF VALVES (MAIN STEAM)

CRITERIA	METHOD OF ANALYSIS	ALLOWABLE STRESS OR MINIMUM THICKNESS REQUIRED	CALCULATED STRESS OR ACTUAL THICKNESS
	2. Cyclic Rating (NB-3550) i) The thermal cyclic index criterion $I_t = \Sigma \frac{N_{ri}}{N_i}$ As $I_t = 0.032 < 1$ Criterion Satisifed ii) The fatigue requirement Criterion $Na \leq 2000$ cycles $S_{P1} = \frac{2}{3} Q_P + \frac{P_{eb}}{2} + Q_{t2} + 1.3Q_{t1}$ $S_{P2} = 0.4\, Q_P + \frac{K}{2} (P_{eb} + Q_{t2})$ Where: S_{P1} = The fatigue stress intensity at the inside surface of the crotch, PSI S_{P2} = The fatigue stress intensity at the outside surface of the crotch, PSI S_{P1} = 9,130 PSI > S_{P2} = 545 PSI The permissible number of startup/shutdown cycles. NA > 10^6 > 2,000 cycles Criterion Satisfied		

TABLE 3.9-2g

SAFETY/RELIEF VALVES (MAIN STEAM)

CRITERIA	METHOD OF ANALYSIS	ALLOWABLE STRESS OR MINIMUM THICKNESS REQUIRED	CALCULATED STRESS OR ACTUAL THICKNESS
4. Bonnet	Based on Nuclear Power Piping Code USAS B31.7-1969, PARA. 1-704.5		
	1. Body to Bonnet Stud Stress Criterion Total Cross-Sectional Area of Bolt Shall Equal or Exceed the Greater of		
	$Am_1 = \frac{Wm_1}{S_b}$ or $Am_2 = \frac{Wm_2}{S_a}$		
	Am_1 = 5.19 in.^{2} > AM_2 = 0.849 in.^{2}		
	As AM_1 = 5.19 in.^{2}	5.19 in.^{2}	See Note 1
	(Am, actual stud area criterion satisfied)		
	2. Bonnet Flange Strength (B31.7-1969, PARA. 1-704.5.1) Criterion		
	$S_H < 1.5\ S_m$		
	$S_R < 1.5\ S_m$		
	$S_T < 1.5\ S_m$		
	As S_H < 29,100 psi (15 S_m @ 500°F A105 GI	29,100 PSI	See Note 1
	S_R < 29,100 psi	29,100 PSI	See Note 1
	S_T < 29,100 psi (1.5 S_m)	29,100 PSI	See Note 1

TABLE 3.9-2g

SAFETY/RELIEF VALVES (MAIN STEAM)

CRITERIA	METHOD OF ANALYSIS	ALLOWABLE STRESS OR MINIMUM THICKNESS REQUIRED	CALCULATED STRESS OR ACTUAL THICKNESS
5. DISC Insert	Bending Stress of Disc Insert per Roark's formulas for stress and strain, 4th Edition pg. 222-223 superposition of case no. 21 and case no. 22 for flat plates. Alternatively, finite element analysis may be used to calculate stress. $S_t = S_{r_1} + S_{r_2} = \frac{3W_1}{4t^2_{avg}} \frac{4a^4(m+1)\ln(a/b) - a^4(m+3) + b^4(m-1) + 4}{a^2(m+1) + b^2(m-1)}$ $+ \frac{3W_2}{2\pi t^2} \frac{2a^2(m+1)\ln(a/b) + a^2(m-1) - b^2(m-1)}{a^2(m+1) + b^2(m-1)}$ Where: W_1 = Pressure, psig W_2 = Seat Load, lbs. a = ½ Disc. Insert Outside Dia., 2.782 in. b = Hub Radius 0.592 in. t = Average Thickness, 1.022 in. S_{r1} = Stress Due to Pressure, psi S_{r2} = Stress Due to Seat Load, psi m = Reciprocal of poisson ratio, 3.333 1. The normal operating condition of 80% of design pressure. S_t < 45,225 psi (S_m @ 575°F for ASME SA-637 TYPE 718) Criterion Satisfied Where W_1 = 1,000 psi W_2 = 5,422 lbs.	45,225 PSI	See Note 1

TABLE 3.9-2g

SAFETY/RELIEF VALVES (MAIN STEAM)

CRITERIA	METHOD OF ANALYSIS	ALLOWABLE STRESS OR MINIMUM THICKNESS REQUIRED	CALCULATED STRESS OR ACTUAL THICKNESS
	2. The maximum possible full flow pressure at 10% above design pressure (the stress is entirely due to pressure).		
	$S_t < 45{,}225$ psi (S_m) Criterion Satisfied	45,225 PSI	See Note 1
	Where $W_1 = 1{,}375$ psi $W_2 = 0$		
	3. The Set Spring Load, Zero Inlet Pressure, Load Temperature (The Stress is Entirely Due to Seat Load, Zero Inlet Pressure).		
	$S_t < 50{,}000$ psi (S_m @ Room Temperature of ASME SA-637 TYPE 718)	50,000 PSI	See Note 1
	Where $W_1 = 0$ $W_2 = 27{,}110$ psi		

Note 1: The calculated stress was determined by calculation to be less than the allowable stress. The actual thickness is greater than the minimum thickness required as determined by calculation. The actual thickness satisfies minimum thickness requirements for the period of extended operation.

TABLE 3.9-2(h) (page 1 of 9)

MAIN STEAM ISOLATION VALVE

Criteria	Method of Analysis	Allowable Stress or Minimum Thickness (in.)	Calculated Stress or Actual Thickness (in.)
Design of Pressure Retaining parts	All references are made to ASME Boiler & Pressure Vessel Code, Section III, Nuclear Power Plant Components, 1971 Ed. plus addendum July 1971. Reference the same code for explanation of the symbols used.		
Body Minimum Wall Thickness	Reference paragraph NB 3543, Nonstandard Pressure-Rated Valve, Table NB 3542-1,		
	For design condition of 1250 psig and 575°F. The primary service rating = 495 based on a core diameter of 23.00 in. t_m = 1.539 in. (including a corrosion allowance of 0.12 in).	1.58	2.12
Body Shape Rule	Reference paragraph NB 3544, Body Shape Rules		
Radius of Crotch	Reference paragraph NB 3544.1(a), Radius of Crotch		
	criterion $r_2 \geq 0.3\ t_m$ as r_2 = 1.00 in., t_m = 1.539 + 1.00 $\geq$ 0.3 x 1.937 = 0.581 criterion satisfied.		
Corner Radii on Internal Surfaces	Reference paragraph NB 3544.1(b), Corner Radii on Internal Surfaces		
	criterion $r_4 < r_2$ as r_4 = 1.00 in., r_2 = 1.125 in. + 1.00 < 1.125 criterion satisfied.		
Out of Roundness	Reference paragraph 3544.5, Out of Roundness, Figure NB 3545.1-2		
	The body is not out of round in excess of 5%, the requirements of this article are satisfied.		
Longitudinal Curvature	Reference paragraph NB 3544.6 Longitudinal Curvature		
	criterion $\frac{1}{r_{Long.}} + \frac{1}{r_{Lat.}} \geq \frac{4}{3d_m}$ as $r_{Long.}$ = 27 in.,		
	$r_{Lat.}$ = 11.5 in., d_m = 23.00 in., + .124 $\geq$ .058 criterion satisfied.		
Flat Wall Limitation	Reference paragraph NB 3544.7, Flat Wall Limitation		
	Since no flat sections were built into the valve body design, the requirements of this article are satisfied.		

TABLE 3.9-2(h) (page 2 of 9)

MAIN STEAM ISOLATION VALVE

Criteria	Method of Analysis	Allowable Stress or Minimum Thickness (in.)	Calculated Stress or Actual Thickness (in.)
Minimum Wall at Weld End	Reference paragraph 3544.8, Minimum Wall at Weld End Actual thickness at > 1.937 in. (i.e., 1.937 in.) measured along. The run direction is 1.492 in. Actual run distance is $\geq$ 9.5 in.		
Primary Crotch Stress Due to Internal Pressure	Reference paragraph NB 3545.1 criterion $P_m = \left(\frac{A_f}{A_m} + 0.5\right) P_s < S_m$ where A = 503 in.^{2}, A_m = 57 in.^{2}, P_s = 1375 psig, P_m = 12,821 psi, S_m = 19,400 psi, since $S_m > P_m$ criterion satisfied		
Valve Body Secondary Stress	Reference paragraph NB 3545.2		
Primary Plus Secondary Stress Due to Internal Pressure	Reference paragraphs NB 3545.2(a)(1), NB 3545.2(a)(2) $Q_p = C_p \left(\frac{r_i}{t_e} + 0.5\right) P_s\ Ca$ where C_p = 3, r_i = 11.625 in., P_s = 1375 psi, t_e = 2.75 for wye-type valve C_a = 1.33 + Q_p = 25.965 psi		
Secondary Stress Due to Pipe Reaction	Reference paragraph NB 3545.2(b), Figures NB 3545.2-3, NB 3545.2-5, NB 3545.2-6		
Direct or Axial Load Effect	$P_{ed} = \frac{F_d S}{G_d}$ where S = 30,750, F_d = 30 in.^{2}, G_d = 183 in.^{2} P_{ed} = 5041 psi		
Bending Load Effect	$P_{eb} = C_b \frac{F_b S}{G_b}$ where S = 30,750, F_b = 340 in.^{3},	29,100	5,041
	i.d. = 23.25 in., r_i = 11.625 t_e = 2.75 r = 13.91 in. as $\frac{t_e}{r}$ = 0.197 0.19 + C_b = 1 $C_b = \frac{I}{r_i + t_e}$ where I = 15,028 in.^{4}, r_i = 11.625 in., t_e = 2.75 in. + G_b = 1048.2 in.^{3}		
	+ P_{eb} = = 9,974 psi	29,100	9,974

TABLE 3.9-2(h) (page 3 of 9)

MAIN STEAM ISOLATION VALVE

Criteria	Method of Analysis	Allowable Stress or Minimum Thickness (in.)	Calculated Stress or Actual Thickness (in.)
Torsion Load Effect	Reference paragraphs 3545.2(b)(1), 3545.2(b)(6)(c)		
	$P_{et} = 2 \frac{F_b S}{G_t}$ where $F_b = 340$ in.^{3}, $S = 30,750$ psi		
	$G_t = 2161$ in.^{3}		
	$\rightarrow P_{et} = 9,676$ psi	29,100	9,676
	For special requirement of $S = 2 S_m = 41,000$ psi		
	$P_{ed} = 6,722 \leq 29,100$		
	$P_{eb} = 13,299 \leq 29,100$	29,100	13,299
	$P_{et} = 12,896 \leq 29,100$		
	$S_h = 33,733 \leq 58,200$ psi	58,200	33,733
Thermal Secondary Stress at Crotch Region	Reference paragraph NB 3545.2(c), Figures NB 3545.2(c)-2, NB 3545.2(c)(2), NB 3545.2(C)-3, NB 3545.2(c)-3, NB 3545.2(c)-4		
	$Q_T = Q_{T_1} + Q_{T_2}$		
	where $Te_1 = 3$ in., $Q_{T_1} = 1,100$		
	$Q_{T_2} = C_6 C_2 \Delta T_2$ where $C_2 = 0.53$ $C_6 = 210$ and $\Delta T_2 = 4.7$		
	$Q_{T_2} = 523$ psi, $Q_T = 1,623$ psi		
	criterion $S_N = Q_P + P_{ed} + 2Q_{T_2} \leq 3 S_m$		
	where $Q_P = 25,965$ $P_{ed} = 5,041$ $Q_{T_2} = 523$		
	as $32,052 \leq 58,200$ criterion satisfied	58,200	32,052
Normal Duty Valve Fatigue Requirements	Reference paragraphs 3545.3, NB 3545.3(a), NB 3545.3a,		
	Figure 1-9-1		
	criterion $N_a \geq 2,000$ cycles		

TABLE 3.9-2(h) (page 4 of 9)

MAIN STEAM ISOLATION VALVE

Criteria	Method of Analysis	Allowable Stress or Minimum Thickness (in.)	Calculated Stress or Actual Thickness (in.)
Normal Duty Valve Fatigue Requirements (cont'd.)	$S_{P_1} = \frac{2}{3} Q' \; p + \frac{P_{eb}}{2} + Q_{T_3} + 1.3 \, Q_{T_1} \; S_{P_2} = 0.4 \, Q' \; p +$		
	$\frac{K}{2} (P_{eb} + 2Q_{T_3})$		
	where $Q' \; P = 25{,}965$ $P_{eb} - 9{,}974$ $K=2$, $Q_{T_1} = 1{,}100$ $Q_{T_3} = 622$ psi		
	$\rightarrow S_{P_1} = 24{,}349$ $S_{P_2} = 21{,}604$ S_a equal to the larger of S_{P_1} and S_{P_2} $\rightarrow$ $S_a = 24{,}349$ psi		
	$\rightarrow N_a = 50{,}000 \geq 2{,}000$ criterion satisfied		
Cyclic Loading Requirements at Valve crotch	Reference paragraph 3550 For the largest temperature change range criterion $Q_p + P_{ed} + C_6 C_2 C_4 \, \Delta T_f \max \leq 3 \, S_m$		
	Where $Q' \; p = 25{,}965$ psi, $P_{ed} = 5{,}041$ $C_6 = 210$ at ΔT_f max of 171 F,		
	$C_2 = 0.53$ $C_4 = 0.03$, S_m, 19,400		
	$\rightarrow 31{,}577 \leq 58{,}200$ criterion satisfied	58,200	31,577
	Thermal Transients Not Excluded by Code criterion $\frac{\Sigma N_{ri}}{\Sigma N_i} < 1$		
	Calculate the fatigue usage factor (I_t) as follows:		
	S_n Max $= Q' \; p + P_{eb} + C_6 C_3 C_4 \, \Delta T_f \max \rightarrow S_n \max = 36{,}618$ psi	58,200	36,618
	Since $S_n \max < 3 \, S_m$ (= 58,200) the following equation is used:		
	$S_i = \frac{4}{3} Q' \; p + P_{eb} + C_6 \, (C_3 C_4 + C_5) \, \Delta T_{fi}$		
	for ΔT_{fi} + 45°F, $N_{ri} = 120$ $S_i = 54{,}719$ psi, $N_i = 25{,}000$,		
	$N_{ri}/N_i = .0048$		
	$\Delta T_{fi} = 121°F$ $N_{ri} = 8$, $S_i = 83{,}069$ psi, $N_i = 6{,}000$		
	$N_{ri}/N_i = .0013$		
	$\Delta T_{fi} = 61°F$, $N_{ri} = 10$, $S_i = 58{,}319$ psi, $N_i = 20{,}000$		
	$N_{ri}/N_i = .0005$		

TABLE 3.9-2(h) (page 5 of 9)

MAIN STEAM ISOLATION VALVE

Criteria	Method of Analysis	Allowable Stress or Minimum Thickness (in.)	Calculated Stress or Actual Thickness (in.)
	as $I_t = \Sigma \frac{N_{ri}}{N_i} = 0.0066 < 1$ criterion satisfied	1.00	.0066
Disk Design Calculation	Reference paragraph NB 3546.3, Table I-1.1, Roark, 3rd Ed., Pages 198, 200, 201		
	Disk design conditions, $P_R = 1250$ psi at 500°F, $S_m = 17,800$ psi at 500°F		
	Case No. 13 $S = \frac{3W}{4mt^2(a^2-b^2)} (a^4 (3m+1) + b^4 (m-1) -$		
	$4_m a^2b^2 - 4(m+1) a^2b^2(\ln(a/b))$		
	where W = 1250 psi, $m - \frac{10}{3}$, t = 5,875 in., a = 10.75 in., b = 1.750 inc.,		
	$s_t = 9,488$ psi		
	Case No. 14 $S = \frac{3W}{2\pi mt^2} \frac{(2a^2(m+1)}{a^2-b^2} \ln \frac{(a)}{(b)} + (m-1))$		
	where W = 58,123 lb_f, t = 5.875 in., $m = \frac{10}{3}$,		
	a = 10.75 in., b = 1.75 in., $s_t = 4,460$ psi		
	$S_t = S_{t_{Case\ no.\ 13}} + S_{t_{Case\ No.\ 14}} = 13,949 \leq 17,800$ psi	17,800	13.949
	Case No. 21 $S_r = \frac{3W}{4t^2} \frac{[4a^4(m+1) \ln \frac{(a)}{(b)} a^4 (m+3) + b^4(m-1) + 4a^2b^2]}{a^2 (m+1) + b^2 (m-1)}$		
	where W = 1250 m = 10/3, t = 3.125 in., a = 10.75 in., b = 7.25 in.		
	$\rightarrow S_r = 5,761$ psi		
	Case No. 22 $S_r = \frac{3W}{2\pi t^2} \frac{[2a^2 (m+1) \ln \frac{(a)}{(b)} + a^2 (m-1) - b^2 (m-1)]}{a^2 (m+1) + b^2 (m-1)}$		
	where W = 252,755 m = 10/3, t = 3.125 a = 10.75, b = 7.25		
	$S_r = 10,738$ psi		

TABLE 3.9-2(h) (page 6 of 9)

MAIN STEAM ISOLATION VALVE

Criteria	Method of Analysis	Allowable Stress or Minimum Thickness (in.)	Calculated Stress or Actual Thickness (in.)
Disk Design Calculation (cont'd.)	Total stress = $Sr_{22} + Sr_{22}$ = 16,499 psi, allowable stress 17,800 psi S_{shear} at inner edge disk[2]	17,800	16,499
	$S_{shear} = \frac{F}{A}$ where F = 264,536 lb, A = 71.18 in.^{2} S_{shear} = 3,716 psi		
	Allowable shear stress = 0.6 x allowable stress = 0.6 x 17,800 = 10,680 psi	10,680	3,716
Tensile Stress at thread Relief	$S_A = \frac{F}{A_t}$ where F= 46,342 lb, A_t = 2.624 in.^{2}, S = 17,661 lb/in^2		
	S_n = 17,661 psi, S_m = 30,600 psi	30,600	17,661
Bonnet Design Calculations including Seismic accelerations for SSE	Reference paragraph NB 3546.1		
Minimum Thickness	$P_{fd} = P + P_{eg}$ $P_{eg} = \frac{16M}{\pi G^3} + \frac{4F}{\pi G^2}$		
	where M = 335,253 in.-lb, F = 46,342 lb, G = 24.75 in.		
	→ P_{eg} = 209 psi, P_{fd} = 1,459 psi		
	$t - d\sqrt{\frac{CP}{S} + \frac{1.78\ W\ hg}{S_d^{\ 3}}}$		
	where C = 0.3, P = 1,459 psi, S = 17,800 psi, hg = 2.625 in., w = 910,144 lb, d = 24.75 in.		
	→ t = 4.97 in., t - 4.97 + 0.120 = 5.09 in. (Corrosion allowance is 0.120 in.)	5.09	5.344

TABLE 3.9-2(h) (page 7 of 9)

MAIN STEAM ISOLATION VALVE

Criteria	Method of Analysis	Allowable Stress or Minimum Thickness (in.)	Calculated Stress or Actual Thickness (in.)
Reinforcement	Reference paragraph NB 3643.3		
	To account for the opening for stem in the bonnet		
	Required Reinforcement $A_r = 7.207\ in^2$	8.6844	7.207
	Available Reinforcement $A = 8.6844\ in^2$		
Bonnet Studs Design Calculation	Reference paragraph 3232.1 and Article E-1000		
	Bolt used 20 pieces of 2 - 8 UNC Bolts		
	Total bolt area = 53.04 in.^{2}		
Normal Operation	1. Pressure stress at Operating Condition		
	$S_1 = \frac{W_{m1}}{A_b} = 17,159\ lb/in.^2$ where $W_{m1} = 910,144$ lb,		
	$A_b = 53.04\ in.^2$ $A_{m1} = \frac{W_{m1}}{S_{Bo}} = \frac{910,144}{27,700} = 32.85\ in^2$	53.04	32.85
	2. Gasket load at ambient condition with no internal pressure		
	$S_2 = \frac{W_{m2}}{A_b} = 2,019\ lb/in.^2$ where $W_{m2} = 107,065\ lb_f$		
	$A_b = 53.04\ in.^2$ $A_{m2} = \frac{W_{m2}}{S_{Ba}} = \frac{107,065}{35,000} = 3.06\ in^2$	53.04	3.06
	Maximum tensile stress = 17,159 lb/in.^{2}		
	Thermal stress is assumed negligible because the coefficients of thermal expansion of bonnet place and stud are the same.		
Body Flange Design Calculations	Reference paragraph NB 3647.1		
	Total flange moment under operating conditions		
	$M_o = M_D + M_G + M_T$		
	$M_D = H_D h_D$, $H_D = 0.785\ B^2 P$, $h_D = R + 0.591$		
	where B = 21.75 in., P = 1,459 psi → $H_D = 542,080\ lb_f$, $h_D = 2.813$ in.,		
	$M_D = 1,524,871$ in.-lb		
	$M_G = H_G h_g$, $H_G = W - H$, $h_G = \frac{C-G}{2}$		
	where W is the higher of W_{m1} and W_{m2}		
	$W_{m1} = 0.785\ G^2 P + (2b \times 3.14\ G\ m\ P)$		

TABLE 3.9-2(h) (page 8 of 9)

MAIN STEAM ISOLATION VALVE

Criteria	Method of Analysis	Allowable Stress or Minimum Thickness (in.)	Calculated Stress or Actual Thickness (in.)
Body Flance Design Calculations (cont'd.)	where G = 24.75 in., b = 0.306 in., m = 3, y = 4,500		
	$\rightarrow W_{m1}$ = 910,144 lb, W_{m2} = 107,065 lb		
	$\rightarrow H_G$ = 208,210 lb, h_G = 2.625 in. $\rightarrow M_G$ = 546,551 in.-lb		
	$M_T = H_T h_T$, $H_T = H - H_D$, $h_T = \frac{R + g_{1} + h_G}{2}$		
	where H = 701,934 H_D = 542,080 R = 1.5 in., g_1 = 2.625 in., h_G = 2.625 in.		
	$\rightarrow H_T$ = 159,854 lb, h_T = 3.375 in., M_T = 539,507 in.-lb_f		
	M_o = 2,610,929 in.-lb. where M_D = 1,524,871 in.-lb,		
	M_G = 546,551 in.-lb, M_T = 539,507 in.-lb		
	Total flance moment under gasket seating condition		
	$M_o = W(\frac{C-G}{2})$, $W = (\frac{A_m + A_b}{2})\ s_a$		
	where C = 30 in., A_b = 53.04 in.^{2}, G = 24.75 in.,		
	A_m = 32.86 in.^{2}, s_a = 35,000 psi at 100°F		
	$\rightarrow$ W = 1,503,198 lb $\rightarrow M_o$ = 3,014,460 lb-in.		
Longitudinal Hub Stress	Reference Paragraph NB 3647.1(c)		
	$S_h = \frac{fM_o}{L g_1^2 B} + \frac{PB}{4 g_o}$ = 21,794 lb/in.^{2} < 1.5 S_{fo} = 26,700 lb/in.^{2}	26,700	21,794
Radial Stress	Reference UA-51(1), Equation (7) of Section VIII of ASME B&PV Code, 1971 Edition		
	$S_R = \frac{(1.33\ t_e + 1)\ M_o}{L t^2 B}$ = 12,303 psi < 1.5 S_{fo} = 26,700	26,700	12,303
Tangential Stress	$S_T = (\frac{YM_o}{t^2 B}) - ZS_R$ = 7,126 psi < 1.5 S_{fo} = 26,700	26,700	7,126
	where Y = 4.5, t = 4.125 in., Z = 2.4, B = 21.75 in.		

TABLE 3.9-2(h) (page 9 of 9)

MAIN STEAM ISOLATION VALVE

Criteria	Method of Analysis	Allowable Stress or Minimum Thickness (in.)	Calculated Stress or Actual Thickness (in.)
Flange Stress Criteria	$\frac{S_H + S_R}{2} < S_m$, $\frac{S_H + S_R}{2} < S_m$		
	$\frac{S_H + S_R}{2} = 17{,}049$ psi	17,800	17,049
	$\frac{S_H + S_T}{2} = 14{,}460$ psi		
Stem Calculation			
Back Seated Stress	$S = \frac{F}{A}$		
	where F = 30,697 lb net upward force		
	A = 2.535 in.^{2}, the smallest cross sectional area on the stem		
	S = 12,110 psi < 26,880 psi		
Valve Close Stem Stress	$S = \frac{F}{A}$		
	where F = 46,959 lb net down force		
	A = 3.141 in.^{2}, the smallest cross sectional area on the stem		
	S = 14,950 psi < 26,880 psi		
Stem Thread Strength	Reference Federal Thread Standard		
	Stem Thread Mating With Disk		
	Thread 2.00 in. - 8 UN - 2 Thread		
	A_{S1} = 3.49 in.^{2}/inch engagement.		
	A_{S1} = 6.54 in.^{2}		
	$\tau = \frac{F}{A_{S1}}$		
	where F = 46,959 lb_f, A_{S1} = 6.54 in.^{2} → τ_{Sd} = 7,177 psi		

TABLE 3.9-2i

RECIRCULATION PUMP

CRITERIA	ANALYTICAL RESULTS	ALLOW. STRESS OR ACTUAL THICKNESS
1. Casing Minimum Wall Thickness $$t = \frac{PR}{SE - 0.6P} + C$$ Loads: Normal and Upset Condition — where: Design pressure & temperature — t = min. req'd thickness, in.; P = design pressure, psig Primary membrane stress limit: — R = max. internal radius, in.; S = allowable working stress, psi Allowable working stress per ASME Sect. III, Class C — E = joint efficiency; C = corrosion allowance, in.	t = 2.69 inches	$S_{all.}$ = 15,075 psi $t_{act.}$ = 3.00 inches
2. Casing Cover Minimum Thickness $$S_s = \frac{F}{A}$$ Loads: Normal and Upset Condition — F = force; A = area at shear point Design pressure & temperature Primary bending & shear stress limit: $S_b = \frac{K^{2qa}}{h^2}$ 1.5 S_m per ASME Code for Pumps and Valves for Nuclear Power Class I	S_s = 3,380 psi S_b = 5,950 psi	$S_{all.}$ = 8,750 psi $t_{act.}$ = 3.5 inches $S_{all.}$ = 1.5 x 15,075 $t_{act.}$ = 7 inches

TABLE 3.9-2i

RECIRCULATION PUMP

CRITERIA	ANALYTICAL RESULTS	ALLOW. STRESS OR ACTUAL THICKNESS
Allowable working stress per q = pressure load a = radius of O.D. b = radius of I.D. h = plate thickness		
3. Cover and Seal Flange Bolt Areas Bolting loads, areas and stresses shall be calculated in accordance with “Rules for Bolted Flange Connections” – ASME Sect. VIII, Para. UA-49 Loads: Normal and Upset Condition Design pressure & temperature Design gasket load Bolting Stress Limit: Allowable working stress per ASME Sect. III, Class C	Cover Flange Bolts $S_{act.}$ = 19,050 psi A_m = 90.2 sq. in. A_m = 79 sq. in. (For Upgraded Gasket) Seal Flange Bolts $S_{act.}$ = 18,000 psi A_m = 9.85 sq. in.	$S_{all.}$ = 20,000 psi $A_{all.}$ = 101 sq. in. $A_{all.}$ = 95 sq. in. (For Upgraded Gasket) $S_{all.}$ = 20,000 psi $A_{all.}$ = 11.1 sq. in.
4. Cover Clamp Flange Thickness Flange thickness and stress shall be calculated in accordance with “Rules for Bolted Flange Connections” – ASME Sect. VIII, Para. UA-51 Loads: Normal and upset Condition Design pressure & temperature Design gasket load Design bolting load	Flange Thickness Stress t = 8.9 inches	$t_{act.}$ = 9.25 inches $S_{all.}$ = 26,250 psi

TABLE 3.9-2i

RECIRCULATION PUMP

CRITERIA	ANALYTICAL RESULTS	ALLOW. STRESS OR ACTUAL THICKNESS
<u>Tangential Flange Stress Limit</u> Allowable working stress per ASME Sect. III, Class C		
5. Seal Cover Loads: $S_b = \frac{KP}{t^2}$ <u>Normal and Upset Condition</u> Design pressure & temperature P = Bolt load due to pressure t = Thickness, in. K = Constant of shape factor	S_b = 2,870 psi t = 1.10	$S_{all.}$ = 15,075 psi t_{act} = 2 9/16
6. Seal Chamber Minimum Wall Thickens $t = \frac{PR}{SE - 0.6P} + C$ Loads: <u>Normal and Upset Condition</u> Design pressure & temperature Piping reactions during normal operation <u>Combined Stress Limit</u>: 1.5 S_m per ASME Section III Code for Pumps and Valves for Nuclear Power Class I where: t = min. req'd thickness, in. P = design pressure, psig R = max. internal radius, in. S = allowable working stress, psi E = joint efficiency C = corrosion allowance, in.	t= 0.741 inches	$S_{all.}$ = 1.5 x 17,075 psi $t_{act.}$ = 1.375 inches

TABLE 3.9-2i

RECIRCULATION PUMP

CRITERIA	ANALYTICAL RESULTS	ALLOW. STRESS OR ACTUAL THICKNESS
7. Mounting Bracket Combined Stress Loads: Flooded weight DBE horizontal seismic force = 2.50g DBE vertical seismic force = 1.61g Bracket vertical loads shall be determined by summing the equipment and fluid weights and vertical seismic forces. Bracket horizontal loads shall be determined by applying the specified seismic force at mass center of pump-motor assembly (flooded). Combined Stress Limit: 1.5 S_m per ASME Code Section III for Pumps and Valves for Nuclear Power Class I Horizontal and vertical loads shall be applied simultaneously to determine tensile, shear and bending stresses in the brackets. Tensile, shear and bending stresses shall be combined to determine max. combined stresses.	Lug #1 S_C = 2,270 psi Lug #2 S_C = 24,429 psi Lug #3 S_C = 14,178 psi Lug #4 S_C = 4,591 psi	1.5 S_m = 25,013 psi
8. Stresses Due to Seismic Loads Loads: Operation pressure and temperature DBE horizontal seismic force = 2.50g DBE vertical seismic force = 1.61g The flooded pump-motor assembly shall be analyzed as a free body supported by constant support hangers from the pump brackets. Horizontal and vertical seismic forces shall be applied at mass center of assembly and equilibrium reactions shall be determined for the motor and pump brackets. Loads, shear, and moment Combined Stress Limit: Yield stress	Motor Bolt Tensile Stress: $S_{act.}$ = 22,471 psi Pump Cover Bolt Tensile Stress: $S_{act.}$ = 19,417 psi Motor Support Barrel Combined Stress: $S_{act.}$ = 3,307 psi	 $S_{all.}$ = 30,800 psi $S_{all.}$ = 32,000 psi $S_{all.}$ = 22,400 psi

TABLE 3.9-2i

RECIRCULATION PUMP

CRITERIA	ANALYTICAL RESULTS	ALLOW. STRESS OR ACTUAL THICKNESS
diagrams shall be constructed using live loads, dead loads, and calculated snubber reactions. Combined bending, tension and shear stresses shall be determined for each major component of the assembly including motor support barrel, bolting and pump casing. The maximum combined tensile stress in the cover bolting shall be calculated using tensile stresses determined from loading diagram plus tensile stress from operating pressure.		

TABLE 3.9-2 (j)

REACTOR RECIRCULATION SYSTEM GATE VALVES, *DISCHARGE*

STRUCTURAL & MECHANICAL LOADING CRITERIA

0. 0.1 0.2	Component Loads Design	Design Procedure	Required Design Value	Actual Design Value
1.	**Body and Bonnet**			
1.1	Loads: Design pressure, design temp.	Vendor's Design calculation	1525 psi 575°F	1525 psi 575°F
1.2	Pressure Rating. lb.	Used NB-3543, Table NB-3531-4 & NB-3531-5 of ASME SECTION III	P_r = 826 lb.	P_r = 826 lb.
1.3	Minimum wall thickness, inches	Used ASME SECTION III PARA NB-3543, Table NB-3542-1	$t_m \geq$ 2.42 inches	t_m = 2.4224 inches
1.4	Primary membrane stress, psi	Used ASME SECTION III PARA NB-3545.1	$P_m \leq S_m$ (500°F) = 19600 psi	P_m = 9954 psi
1.5	Secondary stress due to pipe reaction	Used ASME SECTION III PARA NB-3545.2 (b) (S = 30,000 psi)	$P_{ed} \leq 1.5\ S_m$ = 29400 psi $P_{eb} \leq 1.5\ S_m$ = 29400 psi $P_{et} \leq 1.5\ S_m$ = 29400 psi	P_{ed} = 4271 psi P_{eb} = 7581 psi P_{et} = 7170 psi
1.6	Primary plus secondary stress due to internal pressure	Used ASME SECTION III PARA NB-3545.2 (a)	58,000 psi (See 1.8 below)	Q_p = 24218 psi
1.7	Thermal secondary stress	Used ASME SECTION III PARA NB-3545-2 (c)	58,000 psi (See 1.8 below)	Q_{T1} = 5800 psi Q_{T2} = 1757 psi Q_{T3} = 2044 psi
1.8	Sum of primary plus secondary stress	Used ASME SECTION III PARA NB-3545.2	$S_n = Q_p + P_{ed} + 2Q_{T2}$ $S_n \leq 3S_m$ (500°F) = 58800 psi	S_n = 31674 psi
1.9	Fatigue requirements	Used ASME SECTION III PARA NB-3545.3	$N_a \geq$ 2000 cycles	$N_a > 10^5$cycles
1.10	Cyclic rating	Used ASME SECTION III PARA NB-3550	$I_t \leq 1$	I_t = .00335
2.0	**Body to Bonnet Bolting**			
2.1	Loads: Design pressure & temp., gasket loads, stem operational load, seismic load (design basis earthquake)	Used ASME SECTION III PARA NB-3546.1 NB-3647.1 and Section VIII		
2.2	Bolt Area	Used ASME SECTION III PARA NB-3546.1 NB-3647.1 and Section VIII	$A_b \geq$ 42.79 sq. in. $S_b \leq$ 27975 psi	A_b = 55.86 sq. in. S_b = 21288 psi
2.3	Body Bonnet Flange Stresses			
2.3.1	Operating condition	Used ASME SECTION III PARA NB-3546.1 NB-3647.1 and Section VIII	$S_H \leq 1.5\ S_m$ (575°F) = 28837 psi $S_R \leq 1.5\ S_m$ (575°F) = 28837 psi $S_T \leq 1.5\ S_m$ (575°F) = 28837 psi	S_H = 24206 psi S_R = 7307 psi S_T = 7812 psi
2.3.2	Gasket seating condition	Used ASME SECTION III PARA NB-3546.1 NB-3647.1 and Section VIII	$S_H \leq 1.5\ S_m$ (100°F) = 30000 psi $S_R \leq 1.5\ S_m$ (100°F) = 30000 psi $S_T \leq 1.5\ S_m$ (100°F) = 30000 psi	S_H = 29837 psi S_R = 11050 psi S_T = 11815 psi

NOTES

(1) SECTION III = ASME BOILER AND PRESSURE VESSEL CODE, SECTION III. 1971, "NUCLEAR POWER PLANT COMPONENTS"

(2) VALVE DIFFERENTIAL PRESSURE 200 PSI

TABLE 3.9-2 (j)

REACTOR RECIRCULATION SYSTEM GATE VALVES, *DISCHARGE*

STRUCTURAL & MECHANICAL LOADING CRITERIA

0. 0.1 0.2	Component Loads Design	Design Procedure	Required Design Value	Actual Design Value
3.0	**Stress in Stem**			
3.1	Load Operator thrust and torque			
3.2	Buckling on Stem	Calculate slenderness ratio. If greater than 30, calculate allowable load from Rankine's formula using safety factor of 9.	Maximum allowable load = 82,635 lbs.	Slenderness ratio = 61.1 Actual load on Stem = 34,342 lbs. Therefore no buckling.
3.3	Stem Thrust Stress	Calculate stress due to operator thrust in critical cross-section	$S_T \leq S_m$ = 44100 psi	S_T = 6968 psi
3.4	Stem Torque Stress	Calculate shear stress due to operator torque in critical cross-section	$S_c \leq .6S_m$ = 26460 psi	S_c = 4285 psi
4.	**Disc Analysis**			
4.1	Loads: Maximum differential pressure			
4.2	Maximum Stress	Calculate maximum stress according to chapter of R. J. Roark "Formulas for Stress and Strain"	$S_{max} \leq 1.5S_m$ (575°F) = 28500 psi	S_{max} = 22191 psi
5.	**Yoke and Yoke Connections**			
5.1	Loads: Stem operational load	Calculate stresses in the yoke and yoke connections to acceptable structural analysis methods		
5.2	Tensile stress in yoke legs bolts		$S_{max} \leq S_m$ = 23.300 psi	Max. stress = 8869 psi
5.3	Bending stress of yoke legs		$S_b \leq 1.5\ S_m$ = 34,950 psi	S_b = 6359 psi

TABLE 3.9-2 (j)

REACTOR RECIRCULATION SYSTEM GATE VALVES, *SUCTION*

STRUCTURAL & MECHANICAL LOADING CRITERIA

0. 0.1 0.2	Component Loads Design	Design Procedure	Required Design Value	Actual Design Value
1.	**Body and Bonnet**			
1.1	Loads: Design pressure, design temp.	Vendor's design calculation	1275 psi 575°F	1275 psi 575°F
1.2	Pressure Rating , lb.	Used NB-3543, Table NB-3531-4 & NB-3531-5 of ASME SECTION III	P_r = 695 lb.	P_r = 695 lb.
1.3	Minimum wall thickness inches	Used ASME SECTION III PARA NB-3543, Table NB-3542-1	$t_m \geq 2.03$ inches	$t_m = 2.0254$ in.
1.4	Primary membrane stress, psi	Used ASME SECTION III PARA NB-3545.1	$P_m \leq S_m$ (500°F) = 19600 psi	P_m = 13419 psi
1.5	Secondary stress due to pipe reaction	Used ASME SECTION III[1] PARA NB-3545.2 (b) (S = 30,000 psi)	$P_{ed} \leq 1.5\ S_m$ = 29,400 psi $P_{eb} \leq 1.5\ S_m$ = 29,400 psi $P_{et} \leq 1.5\ S_m$ = 29,400 psi	P_{ed} = 4391 psi P_{eb} = 10740 psi P_{et} = 9220 psi
1.6	Primary plus secondary stress due to internal pressure	Used ASME SECTION III PARA NB-3545.2 (a)	58,800 psi (See 1.8 below)	Q_p = 23527 psi
1.7	Thermal secondary stress	Used ASME SECTION III PARA NB-3545.2 (c)	58,800 psi (See 1.8 below)	Q_{T1} = 3400 psi Q_{T2} = 1127 psi Q_{T3} = 1332 psi
1.8	Sum of primary plus secondary stress	Used ASME SECTION III PARA NB-3545.2	$S_n = Q_p + P_{ed} + 2Q_{T2}$ $S_n \leq 3S_m$ (500°F) = 58800 psi	S_n = 30332 psi
1.9	Fatigue requirements	Used ASME SECTION III PARA NB-3545.3	$N_a \geq 2000$ cycles	$N_a > 10^5$ cycles
1.10	Cyclic rating	USED ASME SECTION III PARA NB-3550	$I_t \leq 1.0$	I_t = .002617
2.0	**Body to Bonnet Bolting**			
2.1	Loads: Design pressure & temp., gasket loads, stem operational load, seismic load (design basis earthquake)	Used ASME SECTION III PARA NB-3546.1 NB-3647.1 and Section VIII		
2.2	Bolt Area	Used ASME SECTION III PARA NB-3546.1 NB-3647.1 and Section VIII	$A_b \geq 35.70$ sq. in. $S_b \geq 27975$ psi	A_b = 55.86 sq. in. S_b = 17881 psi
2.3	Body Bonnet Flange Stresses	Used ASME SECTION III PARA, NB-3546.1 NB-3647.1 and Section VIII		
2.3.1	Operating condition	Used ASME SECTION III PARA NB-3546.1 NB-3647.1 and Section VIII	$S_H \leq 1.5\ S_m$ (575°F) = 28837 psi $S_R \leq 1.5\ S_m$ (575°F) = 28837 psi $S_T \leq 1.5\ S_m$ (575°F) = 28837 psi	S_H = 21000 psi S_R = 5677 psi S_T = 8049 psi
2.3.2	Gasket seating condition	Used ASME SECTION III PARA NB-3546.1 NB-3647.1 and Section VIII	$S_H \leq 1.5\ S_m$ (100°F) = 30000 psi $S_R \leq 1.5\ S_m$ (100°F) = 30000 psi $S_T \leq 1.5\ S_m$ (100°F) = 30000 psi	S_H = 27342 psi S_R = 9351 psi S_T = 13257psi

NOTES

(1) SECTION III = ASME BOILER AND PRESSURE VESSEL CODE, SECTION III, 1971, "NUCLEAR POWER PLANT COMPONENT"

(2) VALVE DIFFERENTIAL PRESSURE 50 PSI

TABLE 3.9-2 (j)

REACTOR RECIRCULATION SYSTEM GATE VALVES, *SUCTION*

STRUCTURAL & MECHANICAL LOADING CRITERIA

0. 0.1 0.2	Component Loads Design	Design Procedure	Required Design Value	Actual Design Value
3.0	**Stress in Stem**			
3.1	Load Operator thrust and torque			
3.2	Buckling on Stem	Calculate slenderness ratio. If greater than 30, calculate allowable load from Rankine's formula using safety factor of 9.	Maximum allowable load = 82,635 lbs.	Slenderness ratio = 61.1 Actual load on Stem = 17047 lbs. Therefore no buckling.
3.3	Stem Thrust Stress	Calculate stress due to operator thrust in critical cross-section	$S_T \leq S_m$ = 44100 psi	S_T = 3473 psi
3.4	Stem Torque Stress	Calculate shear stress due to operator torque in critical cross-section	$S_c \leq .6\ S_m$ = 26460 psi	S_c = 2140 psi
4.	**Disc Analysis**			
4.1	Loads: Maximum differential pressure			
4.2	Maximum Stress	Calculate maximum stress according to chapter of R. J. Roark "Formulas for Stress and Strain:	$S_{max} \leq 1.5\ S_m$ (575°F) = 28500 psi	S_{max} = 18393 psi

TABLE 3.9-2L

STANDBY LIQUID CONTROL PUMP

CRITERIA/LOADING		COMPONENT	LIMITING STRESS TYPE	ALLOWABLE STRESS (PSI)	CALCULATED STRESS (PSI)
The standby liquid control pump has been designed and fabricated to the 1968 P&V Code for Class 2 component.					
Pressure boundary parts:					
1) Fluid cylinder – SA182-F304	S_Y = 30,000 psi				
2) Discharge valve stop stuffing box and cylinder head extension SA 479-304	S_Y = 30,000 psi				
3) Discharge valve cover, cylinder head and stuffing box flange plate, SA 285 GR. C	S_Y = 30,000 psi				
4) Stuffing box gland, ASTM A461, GR. 630	S_Y = 90,000 psi				
5) Studs, SA 193B7	S_Y = 105,000 psi				
6) Dowel pins[(2)] alignment, SAE 4140	S_Y = 117,000 psi				
7) Studs, cylinder tie, SA 193-B7	S_A = 105,000 psi				
8) Pump holddown bolts, SAE GR. 1	T_A = 15,000 psi Q_A = 12,000 psi				
9) Power frame, foot area, cast iron	S_A = 15,000 psi				
10) Motor holddown bolts, SAE GR. 1	T_A = 15,000 psi				
	Q_A = 12,000 psi				
11) Motor frame foot area, cast iron	S_A = 15,000 psi				

TABLE 3.9-2L (Continued)

STANDBY LIQUID CONTROL PUMP

CRITERIA/LOADING	COMPONENT	LIMITING STRESS TYPE	ALLOWABLE STRESS (PSI)	CALCULATED STRESS (PSI)
Normal & Upset Condition Loads:				
1. Design pressure	1. Fluid Cylinder	General Membrane	17,800	See Note (3)
2. Design temperature	2. Discharge valve stop	General Membrane	17,800	
3. Operating basis earthquake	3. Cylinder head extension	General Membrane	17,800	
4. Nozzle loads(1)	4. Discharge valve cover	General Membrane	17,800	
5. Dead weight	5. Cylinder head	General Membrane	17,800	
6. Thermal expansion	6. Stuffing box flange plate	General Membrane	17,800	
7. SRV discharge	7. Stuffing box gland	General Membrane	35,000	
	8. Cylinder head studs	Tensile	25,000	
	9. Stuffing box studs	Tensile	25,000	
Emergency or Faulted Condition:				
1. Design pressure	1. Fluid cylinder	General Membrane	21,360	4,450
2. Design temperature	2. Discharge valve stop	General Membrane	21,360	13,600
3. Weight of structure	3. Cylinder head extension	General Membrane	21,360	13,600
4. Thermal expansion	4. Discharge valve cover	General Membrane	21,360	8,150
5. Safe shutdown earthquake	5. Cylinder head	General Membrane	21,360	8,150
6. SRV	6. Stuffing box flange plate	General Membrane	21,360	10,390
7. LOCA	7. Stuffing box gland	General Membrane	42,000	11,420
8. Nozzle Loads(1)	8. Cylinder head studs	Tensile	25,000	18,820
	9. Dowel pins(2)	Shear only(2)	23,400	19,400
	10. Studs, cylinder tie	Tensile(2)	25,000	8,685
	11. Pump holddown bolts	Shear	12,000	9,415
	12. Pump holddown bolts	Tensile	15,000	11,675
	13. Power frame-foot area	Shear	15,000	1,850
	14. Power frame-foot area	Tensile	15,000	11,390
	15. Motor holddown bolts	Shear	12,000	3,020
	16. Motor holddown bolts	Tensile	15,000	5,290
	17. Motor frame-foot area	Shear	15,000	2,070
	18. Motor frame-foot area	Tensile	15,000	4,125

TABLE 3.9-2L (Continued)

STANDBY LIQUID CONTROL PUMP

CRITERIA/LOADING	COMPONENT	LIMITING STRESS TYPE	ALLOWABLE ACCELERATION (G)	CALCULATED ACCELERATION (G)
Faulted Condition				
Dynamic Loads	SLC Pump Assembly	Equivalent static acceleration	1.75g (vertical) 1.75g (horizontal)	0.45g 0.73g
1. SSE				
2. SRV				
3. LOCA				

TABLE 3.9-2L (Continued)

STANDBY LIQUID CONTROL PUMP

CRITERIA/LOADING	COMPONENT	LIMITING STRESS	ALLOWABLE LOADS	CALCULATED LOADS
Nozzle Load Definition:				
Units: Forces – lbs.				
Moments – ft. - lbs.				
Allowable combination of forces and moments are as follows:				
Where:				
F_1 = The largest absolute value of the three actual external orthogonal forces (F_x, F_y, F_z) that may be imposed by the interface pipe, and,				
M = The largest absolute value of the three actual internal orthogonal moments (M_x, M_y, M_z) permitted from the pipe when they are combined simultaneously for a specific condition.				
			SUCTION:	
Normal and Upset Condition Loads:		F_o = Allowable value of F_1 when all	F_o = 770 lb.	207
		moments are zero.		
1. Design pressure		M_o = Allowable value of M_1 when	M_o = 490 ft.-lb.	388
2. Design temperature		all forces are zero.		
3. Dead weight			DISCHARGE:	
4. Thermal expansion				
5. Operating Basis Earthquake			F_o = 370 lb.	173
			M_o = 110 ft.-lb.	95

TABLE 3.9-2L (Continued)

STANDBY LIQUID CONTROL PUMP

CRITERIA/LOADING	COMPONENT	LIMITING STRESS	ALLOWABLE LOADS	CALCULATED LOADS
Emergency of Faulted Condition Loads:				
1. Design pressure				
2. Design temperature			SUCTION:	
3. Dead weight				
4. Thermal expansion			F_o = 920 lb.	207
5. Safe Shutdown Earthquake			M_o = 590 ft.-lb.	388
6. SRV				
7. LOCA			DISCHARGE:	
8. Nozzle Loads				
			F_o = 440 lb.	173
			M_o = 130 ft.-lb.	95

NOTES:

(1) Nozzle loads produce shear loads only.

(2) Dowel pins take all shear.

(3) Calculated stresses for emergency or faulted condition are less than the allowable stresses for the normal and upset condition loads, therefore, the normal and upset condition is not evaluated.

(4) Operability: The sum of the plunges and rod assembly, pounds mass times 1.75, acceleration is much less than the thrust loads encountered during normal operating conditions. Therefore, the loads during the faulted condition have no significant effect on pump operability.

Table 3.9-2m

Standby Liquid Control Tank

	Criteria	Method of Analysis	Allowables		Actuals			
1.	Shell Thickness							
	Loads: Normal and Upset Design Pressure and Temperature	Brownell and Young "Process Equipment Design" $t = \frac{PR}{SE - 0.6\,P}$	0.01542 in.		0.1875 in.			
	Stress Limit	ASME Section III	30,000 psi		1,602 psi			
2.	Nozzle Loads							
	Loads: Normal and Upset Design Pressure and Temperature	The maximum moments due to pipe reaction and maximum forces shall not exceed the allowable limits.	F_o(lb.)	M_o (ft.-lb.)	F_o (lb.)	M_o	(ft.	- lb.)
	Overflow Nozzle		440	330	298[1]	167[2]	218[1]	203[2]
	Discharge Nozzle		440	330	298[1]	167[2]	218[1]	203[2]
	Loads: Faulted Dead Weight, Thermal Expansion and SSE Earthquake	The maximum moments due to pipe reaction and maximum forces shall not exceed the allowable limits.	F_o(lb.)	M_o (ft.-lb.)	F_o (lb.)	M_o	(ft.	- lb.)
	Overflow Nozzle		528	360	313[1]	176[2]	231[1]	219[2]
	Discharge Nozzle		528	360	313[1]	176[2]	231[1]	219[2]
3.	Anchor Bolts	ASME Section III	18,750 psi		9.617 psi			
4.	Dynamic Loads	Equivalent Static	(Horizonta) 1.5g (Vertical) 1.5g		0.41g 0.53g			
	1. SSE							
	2. SRV							
	3. LOCA							

(1) Unit 1

(2) Unit 2

TABLE 3.9-2n

(i) RHR Pumps and (ii) Core Spray Pumps

Location	Loading Condition	Criteria	Calculated Stress (psi)	Allowable Stress (psi)
(i) RHR PUMPS				
Suction Barrel Shell	FAULTED CONDITION Design Pressure Static Loads Dynamic Loads	ASME Boiler & Pressure Vessel Code, Section III	17,672	21,000
Stuffing Box Pipe	Design Pressure Static Loads Dynamic Loads	ASME Boiler & Pressure Vessel Code, Section III	8,451	18,750
Nozzle Shell Intersection	FAULTED CONDITION Design Pressure Static Loads Dynamic Loads	ASME Boiler & Pressure Vessel Code, Section III	15,193 (Suction) 20,392 (Discharge)	34,650 34,650
Discharge Elbow or Suction Pipe (Maximum)	FAULTED CONDITION Design Pressure Static Loads	ASME Boiler & Pressure Vessel Code, Section III	10,926 (Disch. Elbow)	21,000
Motor Stand	FAULTED CONDITION Dynamic Loads Static Loads	Bolting Loads & Stresses per ASME B&PV, Section III, Subsection NF	5,717	21,000
Motor Bolting	FAULTED CONDITION Dynamic Loads Static Loads	Bolting Loads & Stresses per ASME B&PV, Section III, Subsection NF	5,977	30,000
(ii) CORE SPRAY PUMPS				
Suction Barrel Shell	FAULTED CONDITION Design Pressure Static Loads Dynamic Loads	ASME Boiler & Pressure Vessel Code, Section III	13,499	21,000
Stuffing Box Pipe	Design Pressure Static Loads Dynamic Loads	ASME Boiler & Pressure Vessel Code, Section III	6,766	18,750
Nozzle Shell Intersection	FAULTED CONDITION Design Pressure Static Loads Dynamic Loads	ASME Boiler & Pressure Vessel Code, Section III	33,417 (Suction) 31,562 (Discharge)	34,650 34,650
Discharge Elbow or Suction Pipe (Maximum)	FAULTED CONDITION Design Pressure Static Loads	ASME Boiler & Pressure Vessel Code, Section III	11,061 (Disch. Elbow)	21,000
Motor Stand	FAULTED CONDITION Static Loads Dynamic Loads	Bolting Loads & Stresses per ASME B&PV, Section III, Subsection NF	4,505	21,000
Motor Bolting	FAULTED CONDITION Static Loads Dynamic Loads	Bolting Loads & Stresses per ASME B&PV, Section III, Subsection NF	1,960	30,000

TABLE 3.9-2o

RHR HEAT EXCHANGER

Criteria	Method of Analysis	Allowable Min. Thickness Required	Actual Minimum Thickness
1. Closure Bolting Loads: Normal and Upset Design pressure and temperature Design gasket load	Bolting loads and stresses calculated per "Rules for Bolted Flange Connections" ASME Section III 1971, Winter 1972 Addenda		
Bolting Stress Limit			
Allowable working stress per ASME Section III	a. Shell-to-tube sheet bolts	1-3/8" dia.	1-3/8" dia.
	b. Channel cover bolts	1-3/8" dia.	1-3/8" dia.
2. Wall Thickness Loads: Normal and Upset Design pressure and temperature	Shell side ASME Section III Class 2 and TEMA Class C Tube side ASME Section III Class 3 and TEMA Class C		
Stress Limit			
ASME Section III	a. Shell	0.905 in.	1.0 in.*
	b. Shell cover	0.895 in.	0.895 in. min.
	c. Channel	0.932 in.	1.0 in.*
	d. Tubes 16 BWG	0.049 in.	0.065 in.*
	e. Channel cover	8.98 in.	9.00 in.*
	f. Tube sheet	7.42 in.	7.50 in.*

*Stresses within 10% of allowable.

TABLE 3.9-2o

RHR HEAT EXCHANGER

Criteria	Method of Analysis	Allowable Nozzle Forces and Moments Force in lbs., Moment in in.-lbs.	Calculated Nozzle Forces and Moments
3. Nozzle Loads Design pressure and temperature	The maximum moments due to pipe reaction and the maximum forces shall not exceed the allowable limits.	See below.	$\frac{Fi}{Fo} + \frac{Mi}{Mo}$ = 0.86 (Inlet) $\frac{Fi}{Fo} + \frac{Mi}{Mo}$ = 0.53 (Exhaust)
Dead weight, thermal expansion design basis earthquake	Primary stress less than 1.5 ASME Section III allowable	Maximum allowable piping loads for emergency conditions (including DBE) shall not exceed the following relationship for each nozzle.	
Allowable limits (design basis)		$\frac{Fi}{Fo} + \frac{Mi}{Mo} \leq 1$	

N1	N2	N3	N4	
Fx = 15,500	15,500	21,500	21,500	where Fi (lbs.) is the maximum of the three (3) orthogonal forces
Fy = 15,500	15,500	21,500	21,500	(Fx Fy Fz)
Fz = 15,500	15,500	21,500	21,500	
Mx = 60,000	60,000	100,000	100,000	and Mi (ft.-lbs.) is the maximum of the three (3) orthogonal moments
My = 60,000	60,000	100,000	100,000	
Mz = 60,000	60,000	100,000	100,000	(Mx My Mz)

TABLE 3.9-2o

RHR HEAT EXCHANGER

Criteria	Method of Analysis	Allowable Min. Thickness Required (psi)	Actual (psi)
4. Support Brackets & Attachment Welds	SAP-4G Finite Element Computer Code Stress Analysis		
Loads: Faulted			
Design pressure and temperature, dead weight, nozzle loads, safe shutdown earthquake			
Stress Limits			
Allowables as per ASME Section III, Subsection NF (Upset Condition)	a. Lower Bracket Welds - Bending Stress - Shear Stress	 14,438 21,000	 5,990 3,304
	b. Upper Bracket Welds		
	- Bending Stress - Shear Stress	14,438 21,000	9,887 3,197
5. Anchor Bolts	Bolt loads calculated using SAP-4G finite Element Computer Code		
Loads: Faulted			
Design pressure and temperature, dead weight, nozzle loads, safe shutdown earthquake			
Bolting Stress Limits			
Allowable stresses as per ASME III, Appendix XVII	Lower Support Bolting		
Tension - 2461.1 Shear - 2461.2	Tension Shear	29,000 11,990	13,115 8,647

TABLE 3.9-2o

RHR HEAT EXCHANGER

Criteria	Method of Analysis	Allowable Min. Thickness Required (psi)	Actual (psi)
6. Shell Adjacent to Support Brackets	Finite Element and Localized Shell Stress Analysis		
Loads: Faulted			
Design pressure and temperature, dead weight, nozzle loads, safe shutdown earthquake			
Shell Stress Limit			
Allowables as per ASME Section III, Subsection NC	a. Maximum Principal Stress adjacent to upper support	28,875	22,432
	b. Maximum Principal Stress adjacent to lower support	28,875	23,463
7. Shell	Finite Element Dynamic Stress Analysis		
Loads: Faulted			
Design pressure and temperature, dead weight, nozzle loads, safe shutdown earthquake			
Stress Limits			
Allowable as per ASME Section III, Subsection NC (Upset Condition)	- Principal Stress	19,250	18,560

TABLE 3.9-2(p)

RWCU PUMP

The following is a summary of design calculations by Hayward Tyler Fluid Dynamics Ltd. on the RWCU Pump:

ASME Code Calculation	Required Thickness (in)	Allowable Stress (psi)	Actual Thickness (in)
Pump Part			
Body	0.8833	17,500	2.081
Drain Nozzle	0.0421	17,500	0.181
Suction Nozzle	0.2513	17,500	0.422
Suction End	3.688	17,500	3.713
Suction Bend	0.2518	15,000	0.43
Discharge Branch	0.1693	17,500	0.8615
Pump/motor Flange Studs	1.3393 (Note 1)	25,000	1.375 (Note 1)
Motor Part			
Motor Case	3.09	17,500	3.21
End Plate	1.5943	17,500	2.07
Gasket Seating	2.21	17,500	3.00
Thermal Neck	0.1871	17,500	1.3457
Motor Case Shell Body			
Pump End	0.54	17,500	0.995
Cover End	0.6058	17,500	2.8025
Motor Cover (Fill, Drain & Purge Nozzle)	0.0421	17,500	0.1986
Flange	2.585	17,500	2.826
Motor Case/Motor Cover Studs	1.3393 (Note 1)	25,000	1.375 (Note 1)

Note 1: Engagement Length

TABLE 3.9-2(p) (Continued)

ASME Code Calculation	Required Thickness (in)	Allowable Stress (psi)	Actual Thickness (in)
Heat Exchanger Part			
Heat Exchanger Shell Body	0.2404	17,500	0.331
Nozzles	0.4775	17,500	0.884
Vent Nozzle	0.0401	17,500	0.199
HP Nozzles	0.0884	17,500	0.47725
Heat Exchanger End Cover Body	0.0216	17,500	0.5575
Heat Exchanger End Cover End Plate	0.2076	17,500	1.12
Heat Exchanger LP Tubes		3,461 (Note 2)	1,450 (Note 2)

END COVER - FLANGE

ASME Code Calculation	Calculated Stress (psi)	Allowable Stress (psi)
Operating Condition		
SH	2,354	(1.5S) 26,250
SR	1,307	(1.0S) 17,500
ST	962	(1.0S) 17,500
(SH + SR)/2	1,830	(1.0S) 17,500
(SH + ST)/2	1,658	(1.0S) 17,500
Gasket Seating Condition		
SH	11,916	(1.5S) 26,250
SR	6,617	(1.0S) 17,500
ST	4,865	(1.0S) 17,500
(SH + SR)/2	9,267	(1.0S) 17,500
(SH + ST)/2	8,391	(1.0S) 17,500

Note 2: Pressure, (psi)

TABLE 3.9-2q

RCIC TURBINE

Criteria

The highest stressed components of the RCIC Turbine assembly are identified.
Allowable stresses for pressure retaining components are based on ASME B&PV Code, Section III.

Normal Condition:

Pressure Boundary Castings: SA216-WCB @ 500°F
 S = 17,500 psi
 S_A (General Membrane) = 0.8 x S
 S_A (Bending) = 1.5 x 0.8 x S
Pressure Boundary Bolting: SA193-B7 @ 500°F
 S_A = 1.0 x S S = 25,000 psi
Alignment Taper Pins: AISI 4037, Rc 28-35
 T_a = 61,100psi S_y = 106,000 psi

Faulted Condition:

Pressure Boundary Castings: SA216-WCB @ 500°F
 S = 17,500 psi
 S_A = (General Membrane) = 1.2 x 0.8 x S
 S_A = (Bending) = 1.8 x 0.8 x S
Pressure Boundary Bolting: SA193-B7 @ 500°F
 S_A = 1.0 x S S = 25,000 psi
Alignment Taper Pins: AISI 4037, Rc 28-35
 T_a = 61,100psi S_y = 106,000 psi

LOADING	COMPONENT	LIMITING STRESS TYPE	ALLOWABLE STRESS (PSI)	CALCULATED STRESS (PSI)
Normal Condition Loads:	Castings: 1) Stop valve	General Membrane	14,000	
1. Design Pressure	2) Governor valve	General Membrane	14,000	
2. Design Temperature	3) Turbine inlet	Local Bending	21,000	(1)
3. Deadweight	4) Turbine case	Local Bending	21,000	
4. Inlet Nozzle Loads	Pressure boundary bolts	Tensile	25,000	
5. Exhaust Nozzle Loads	Alignment taper pins	Shear	61,100	
Faulted Condition:	Castings: 1) Stop valve	General Membrane	16,800	9,800
1. Design Pressure	2) Governor valve	General Membrane	16,800	13,200
2. Design Temperature	3) Turbine inlet	Local Bending	25,200	15,300
3. Deadweight	4) Turbine case	Local Bending	25,200	18,000
4. Inlet Nozzle Loads	Pressure boundary bolts	Tensile	25,000	20,100
5. Exhaust Nozzle Loads	Alignment taper pins	Shear	61,100	46,880
6. Controlling combination of SSE, SRV, and LOCA				

TABLE 3.9-2q

RCIC TURBINE

Nozzle Load Definition:	Allowable Nozzle Load Criteria:
Turbine vendor has defined allowable nozzle loads for the turbine assembly. The above calculated stresses assume these allowable nozzle loads have been satisfied.	
Normal Loads: 1. Design Pressure 2. Design Temperature/Thermal Expansion 3. Deadweight	Inlet: $F = \frac{(2620 - M)}{3}$ Exhaust: $F = \frac{(6000 - M)}{3}$ Where: F = Resultant force (lbs.) M = Resultant moment (ft.-lbs.)
Faulted Loads:(3) 1. Design Pressure 2. Design Temperature/Thermal Expansion 3. Deadweight 4. Controlling combination of SSE, SRV, and LOCA	Inlet: $F = \frac{(7000 - M)}{4.7}$ Exhaust: $F = \frac{(8500 - M)}{0.34}$ Not to exceed 7000 lbs. Where: F = Resultant force (lbs.) M = Resultant moment (ft.-lbs.)

NOTES:

(1) Calculated stresses for the faulted condition are lower than the allowable stresses for the normal condition. Therefore the normal, upset, and emergency conditions are not evaluated.

(2) Operability: Analysis indicates that shaft deflection with faulted loads is 0.008 inch; which is fully acceptable. And, maximum bearing loads under faulted conditions are acceptable. Furthermore, the turbine assembly has been seismically qualified via dynamic testing. This qualification included demonstration of start-up and shutdown capabilities, as well as no load operability during seismic loading conditions.

TABLE 3.9-2q

RCIC TURBINE

(3) The nozzle loads for the Upset loading condition, as determined by the piping analysis, shall satisfy the allowable criteria for the Faulted loading condition.

TABLE 3.9-2r

RCIC PUMP

Criteria

The critical components of the RCIC pump assembly are identified.
Allowable and calculated values are based on ASME Section III or VIII, as applicable.

COMPONENT	ALLOWABLE	CALCULATED (1)
Pump Hold Down Bolts: $\frac{ft^2}{Ftb^2} + \frac{fv^2}{Fvb^2}$	1	0.502
Anchor Bolt: Shear Stress Tensile Stress	10,000 psi 21,600 psi	7,422 psi 17,090 psi
Pump Outer Case	17,500 psi	7,052 psi
Pump Outer Case at Discharge Nozzle	26,250 psi	7,855 psi
Discharge Nozzle	17,500 psi	3,600 psi
Suction Elbow	18,000 psi	7,900 psi
Pump Shaft (2)	32,000 psi	5,975 psi
Impeller Key	9,000 psi	4,810 psi
Mounting Feet	17,500 psi	6,208 psi
Mounting Feet Weld Stress	9,625 psi	6,208 psi
Pump Pedestal Weld Stress	9,625 psi	1,868 psi
Base Plate & Plate Stiffener	21,600 psi	13,818 psi
Outboard Bearing – Axial (2)	17,200 lb.	1,323 lb.
Inboard Bearing – Axial (2)	7,670 lb.	376 lb.
Seal Circulation Piping - ½" - ¾"	15,000 psi 15,000 psi	10,000 psi 4,630 psi
Bypass Piping	15,000 psi	8,592 psi
Shaft: Relative Radial Displacement Between Shaft & Sleeve (2) Relative Radial Displacement Between Shaft & Mech. Seal (2) Angular Misalignment at Coupling (2)	.0055 in. .0055 in. .017 Rad.	.00383 in. .0008 in. .0022 Rad.
Impeller: Relative Radial Displacement Between Impeller & Casing (2)	.0075 in.	.00084 in.
Suction End and Discharge End Bolting	25,000 psi	20,740 psi

TABLE 3.9-2r

RCIC PUMP

Nozzle Load Definition:

Allowable nozzle loads for the pump have been defined. The above calculated stresses assume these allowable nozzle loads have been satisfied.

Units: Forces = lbs.
Moments = ft-lbs.

The allowable combinations of forces and moments are as follows:

$$\frac{F_i}{F_o} + \frac{M_i}{M_o} \leq 1$$

Where:

F_i = Largest absolute value of the three actual orthogonal forces (F_x, F_y, F_z) imposed by the interface pipe.
M_i = Largest absolute value of the three actual orthogonal moments (M_x, M_y, M_z) imposed by the interface pipe.
F_o = Allowable value of F_i when all moments are zero.
M_o = Allowable value of M_i, when all forces are zero.

Nozzle	Allowable Load (3)
Suction	F_o = 2906 M_o = 3688
Discharge	F_o = 4450 M_o = 5200

NOTES:

(1) Calculated values are due to the highest faulted condition loads and are less than 1.2 times the upset allowables. Therefore, the normal plus upset and emergency conditions are not evaluated.

TABLE 3.9-2r

RCIC PUMP

(2) Operability is addressed by the evaluation of the relative displacements between rotating and stationary components, shaft stress, and bearing loads under faulted condition loads. All criteria are met. Therefore, operability is ensured.

(3) Nozzle load allowables are applicable to all loading conditions.

TABLE 3.9-2s

REACTOR REFUELING & SERVICING EQUIPMENT
(i) NEW FUEL STORAGE RACKS

CRITERIA	LOADING	LOCATION	ALLOWABLE STRESS (.7 ULT)	CALCULATED STRESS
1. NEW FUEL STORAGE RACKS Stress due to normal upset or emergency loading shall not cause a failure so as to result in a critical array.	FAULTED CONDITION "A" 1. Dead Loads 2. Full Fuel Load in rack 3. S.S.E. 4. Thermal (not applicable)	1. Beam (Axial) 2. Beam (Trans.) 3. Combined	1. 26,000#/in^2 2. 26,000#/in^2 3. 26,000#/in^2	1. 18,905#/in^2 2. 7,005#/in^2 3. 25,910#/in^2

2. SOURCE OF ALLOWABLE STRESS (.7 ULT)

a. ASTM B308 Alloy 6061-T6
b. ASME Code – Boilers and Pressure Vessels, Sect. III, NA
c. Product Safety Standards for BWR-6-Mark III, Sect. VI, A. (3)
d. ASME – Pressure Vessels and Piping: Design and Analysis, Volume One, Page 69.
e. ASTM code for Boilers and Pressure Vessels was selected on the premise that data used from this source would necessarily be on the conversative side as applied to the fuel storage rack calculations.

3. S. S. E. loads derived by dynamic analysis. Total stress refers to combined earthquake and thermal load at highest expected pool temperature. Earthquake stresses obtained by square root of the sum of the squares method for a response due to tri-axial excitation. Stress given is the highest in the total structural array. The calculated stresses are conservative since they are based on the original fuel assembly type which has the greatest mass of all fuel assemblies in use or in storage at SSES.

TABLE 3.9-2s

REACTOR REFUELING & SERVICING EQUIPMENT
(i) NEW FUEL STORAGE RACKS

CRITERIA	LOADING	LOCATION	ALLOWABLE STRESS (.7 ULT)	CALCULATED STRESS
4. <u>NEW FUEL STORAGE RACKS</u> Stresses due to normal upset or emergency loading shall not cause a failure so as to result in a critical array.	<u>FAULTED CONDITION "B"</u> (See Below, Par. 5)	(Location – See Par. 6, Below)	Not Applicable	Not Applicable
5. <u>FAULTED CONDITION "B"</u>	Condition "B" is an emergency condition in which the stress limit is equal to the yield strength at 0.2% offset. The racks were tested to determine their capability to safely withstand the accidental, uncontrolled, drop of a fuel bundle from its fully retracted position into the weakest portion of the rack.			
6. <u>METHOD OF TESTING</u>:	Four (4) rack castings were subjected to impact loads ranging from 1908 ft. lbs. To 4070 ft. lbs., which were generated by dropping simulated fuel bundles weighing 660 lbs. from heights varying from 3.0' to 6.17'. Racks were aligned in pairs and simulated bundles were dropped on both racks at the flange area. Both center impact and end impact tests were conducted. (Two (2) of the racks were X-ray examined prior to testing. Strain gauges were mounted on racks to ascertain max. strain and accelerometers were mounted on bundles to determine "G" loads.)			
7 <u>TEST RESULTS</u>:	A total of nineteen (19) tests were performed with drop height increased at each test. First failure occurred due to a central impact on rack No. 3 from a max. height of 6.17', (Test #13). Racks #1 and #2 both failed from a center impact caused by a load dropped from a height of 5.33', (Test #19). Accelerometer readings are not available due to the inability to adequately affix the accelerometer to the simulated fuel bundle.			

TABLE 3.9-2s

REACTOR REFUELING & SERVICING EQUIPMENT
(ii) FUEL PREP MACHINE

ACCEPTABLE CRITERIA	LIMITING LOAD COMBINATION	PRIMARY LOADING	ALLOWABLE STRESS (psi)	CALCULATED STRESS (psi)	OPERABILITY ASSURANCE DEMONSTRATED BY
AISC Code; ASME, Sect. III			Fu = 75,000 Fy = 30,000 S_m200 = 17,800		
Chain 302 Stainless Side Plates 17-4 PH or 17-7 PH Rollers					
Normal Condition	Static	Axial Load	17,800	12,300	Analysis
Upset Condition	N+OBE+SRV	Axial Load	24,000	18,900	Analysis
Emergency Condition	Analyzed As Upset Cond.	Axial Load	--	N/A	--
Faulted Condition	N+SSE+SRV+LOCA	Axial Load	52,500	32,200	Analysis

Note: The calculated stresses are conservative since they are based on the original fuel assembly type which has the greatest mass of all fuel assemblies in use or in storage at SSES.

TABLE 3.9-2t

HIGH PRESSURE COOLANT INJECTION PUMP

Location	Loading Condition	Criteria	Calculated Stress (psi)	Allowable Stress (psi)	Material
Pressure Boundary Parts					
Closure Bolting (Main)	Emergency/Faulted 1. Pressure 2. Design Temperature 3. Seismic 4. SRV 5. LOCA 6. Nozzle Loads	Allowables Based on Normal/Upset Condition Per ASME B&PV Code Section for Pressure Boundary Parts @ 140°F	15,498	25,000	A-193 GR.B7 Tensile Stress
Closure Bolting (Booster)			15,878	25,000	
Casing Wall (Main)			12,050	14,000	A-216 GR. WCB General Membrane
Casing Wall (Booster)			3,650	14,000	
Non Pressure Boundary Parts					
Pump Bolts (Booster) (Tensile) Booster Pump Mounting Foot	Emergency/Faulted 1. Pressure 2. Design Temperature 3. Seismic 4. Nozzle Loads 5. SRV 6. LOCA	Allowables Based on ASME B&PV Code Section IV @ 140°F 0.5 Su for Bolts 0.4 Sy for Pins	15,870	21,000	A=307 GR.B7 Su = 60,000 psi
Pump Bolts (Main) (Tensile) Main Pump Mounting Foot			19,918	21,000	
Dowel Pins (Booster) (Shear) Booster Pump Taper Pin			20,978	33,600	A-193 GR.B7 Sy =105,000 psi
Dowel Pins (Main) (Shear) Main Pump Taper Pin			22,517	33,600	

NOTE: Eight (8) anchor bolts, each carries the stresses for both units mounted on a common base plate.

TABLE 3.9-2u

CONTROL ROD DRIVE (Index Tube)

Criteria	Loading	Primary Stress Type	Allowable Stress (psi)	Calculated Stress (psi) (ABS)*
Allowable Primary Membrane Stress plus Bending Stress is based on ASME Boiller and Pressure Vessel Code, Section III for Type 316 Stainless Steel @ 250°F S_m = 20000 psi				
For Normal and Upset Condition: S_{allow} = S_y for General Membrane = 1.49 for General Membrane and Bending	For Normal & Upset Condition: 1. Normal Loads [(1)] 2. Scram 3. OBE 4. SRV	General Membrane General Membrane and Bending	28,500 42,500	18,700 32,700
For Emergency Condition: (2)	For Emergency Condition : 1. Normal Loads [(1)] 2. Chugging 3. SRV 4. Scram	General Membrane and Bending	(2)	(2)
For Faulted Condition: S_{allow} = 0.80 Su for General Membrane = 2.16 S_y for General Membrane and Bending	For Faulted Condition : 1. Normal Loads [(1)] 2. SSE 3. Chugging 4. SRV	General Membrane General Membrane and Bending	56,500 61,560	<29,400 <32,700 [(3)] 29,400 [(4)]

NOTES:

(1) Normal loads include pressure, temperature, weight and mechanical loads.

(2) Less severe than the upset condition.

(3) Unit 1

(4) Unit 2

* The points of highest stress using the absolute sum (ABS) methodology are given. The ABS approach results in higher values than the square root sum of the squares (SRSS) methodology for corresponding load combination.

TABLE 3.9-2v

CONTROL ROD DRIVE HOUSING

Criteria	Loading	Primary Stress Type	Allowable Stress (psi)	Calculated Stress (psi)
Primary Stress Limit – The allowable primary membrane stress is based on the ASME Boiler and Pressure Vessel Code, Section III, for Class I Vessels, for Type 304 Stainless Steel.	Normal and Upset Condition Loads: 1. Design Pressure 2. Stuck Rod Scram Loads 3. Operational Basis Earthquake, with Housing Lateral Support Installed.	Maximum membrane stress intensity occurs at the tube to tube weld near the center of the housing for the normal, upset and emergency conditions.	16,600	15,710
For Normal and Upset Condition: S_m = 16,660 psi @ 575°F				
For Faulted Conditions: * S_{limit} = 2.4 S_m = 2.4 x 16,600 = 39,984 psi Note: Analyzed to Emergency Condition Limits.	Faulted Condition Loads: 1. Design Pressure 2. Stuck Rod Scram Loads 3. Design Basis Earthquake, with Housing Lateral Support Installed 4. Jet Reaction 5. Annulus Pressurization.		39,800	16,700

* Analyzed to Normal upset Condition Limits.

Table 3.9-2w

JET PUMPS

CRITERIA	LOADING CONDITIONS	STRESS TYPE	ALLOWABLE STRESS (psi)	CALCULATED STRESS (psi)
Primary Membrane Plus Bending Stress Based on ASME B&PV Code Section III				
MATERIAL: TYPE 304SS				
A. For Service Levels A & B – NORMAL & UPSET CONDITION:				
S_m = 16,900 psi @ 550^{o}F $P_m + P_b$ $S_{limit} = 1.5\ S_m$	1. Deadweight 2. Pressure 3. SRV 4. OBE	PRIMARY MEMBRANE PLUS BENDING	25,350	12,700
B. For Service Level C – EMERGENCY CONDITION:				
$P_m + P_b$ $S_{limit} = 2.25\ S_m$	1. Deadweight 2. Pressure 3. SRV 4. SBA	PRIMARY MEMBRANE PLUS BENDING	38,025	16,010
C. For Service Level D – FAULTED CONDITION:				
$P_m + P_b$ $S_{limit} = 3.0\ S_m$	1. Deadweight 2. Pressure 3. Chugging 4. SRV 5. SSE	PRIMARY MEMBRANE PLUS BENDING	60,840[(1)] 50,800[(2)]	38,416
D. MAXIMUM CUMULATIVE FATIGUE USAGE FACTOR ACCEPTANCE CRITERIA: <1.0: UNIT 1 PROJECTED CUMULATIVE USAGE FACTOR: = 0.94 UNIT 2 PROJECTED CUMULATIVE USAGE FACTOR: = 0.67 These CUFs account for the original 40-year plant design. These components are presently inspected on a regular basis. These inspections continue during the period of extended operation to ensure components continue to perform their intended functions.				
(1) Unit 1 (2) Unit 2				

Table Rev. 37

TABLE 3.9-3

NSSS SEISMIC CATEGORY I ACTIVE PUMPS AND VALVES

COMPONENT NAME	IDENTIFICATION AS SHOWN ON APPLICABLE FIGURES
Main Steam Isolation Valves	B 21 F 022 B 21 F 028
Safety Relief Valves	B 21 F 013
Control Rod Drive Globe Valves	C 12 F 010/180 (Unit 1) C 12 F 011/181 (Unit 1) C 12 F 010A/B (Unit 2) C 12 F 011A/B (Unit 2)
Recirculation System (Discharge and Bypass) Gate Valves	B 31 F 031 B 31 F 032
Standby Liquid Control Pump	C 41 C 001
Standby Liquid Control Valve	C 41 F 004
RHR Pump	E 11 C 002
Core Spray Pump and Motor	E 21 C 001
RCIC Pump	E 51 C 001
RCIC Turbine	E 51 C 002
HPCI Pump	E 41 C 001
HPCI Turbine	E 41 C 002

TABLE 3.9-4 (Page 1 of 21)

APPLICABLE THERMAL TRANSIENTS

(PRE STARTUP LEAK TEST)
130 CYCLES
CONDITION - TEST

Pipeline	Initial Temp. °F	Final Temp. °F	Time	Temp. Rate °F/Hour	Temp. °F
Main Steam Line	70	100	30 min	60	30
Head Spray (RHR)	70	100	30 min	60	30
Recirculation Suction	70	100	30 min	60	30
Recirculation Discharge	70	100	30 min	60	30
Bottom Drain	70	100	30 min	60	30
Standby Liquid Control	70	100	30 min	60	30
	100	50	Step	10 min Duration	50
	50	100	Step		50
Core Spray	70	100	30 min	60	30
Feedwater	70	100	30 min	60	30
CRDHS Return	70	100	30 min	60	30

Remarks: After temperature is raised to 100°F, reactor pressure is increased to 1250 psig and then decreased to 0 psig.

TABLE 3.9-4 (Page 2 of 21)

APPLICABLE THERMAL TRANSIENTS

(STARTUP)
120 CYCLES
CONDITION - NORMAL

Pipeline	Initial Temp. °F	Final Temp. °F	Time	Temp. Rate °F/Hour	Temp. °F
Main Steam Line	100	551	-	100	451
Head Spray (RHR)	100	551	-	100	451
Recirculation Suction	100	551	-	100	451
	551	543	Step	-	8
	543	527	Step	-	16
Recirculation Discharge	543	527	Step	-	16
Bottom Drain	543	527	Step	-	16
Standby Liquid Control	543	527	Step	-	16
Core Spray only occurs 10 times	100	406	-	100	306
	406	50	Step	-	356
	50	406	Step	-	356
	406	551	-	100	145
Feedwater	100	551	-	100	451
	551	90	Step	-	461
	90	420	30 min	660	330
CRDHS Return	100	50	Step	-	50

Remarks: Reactor pressure increases from 0 to 1038 psig at rate of temperature increase.

TABLE 3.9-4 (Page 3 of 21)

APPLICABLE THERMAL TRANSIENTS

(DAILY POWER REDUCTION AND ROD PATTERN CHANGE)
10,400 CYCLES
CONDITION - NORMAL

Pipeline	Initial Temp. °F	Final Temp. °F	Time	Temp. Rate °F/Hour	Temp. °F
Main Steam Line	551	551			
Head Spray (RHR)	551	551			
Recirculation Suction	527	527			
Recirculation Discharge	527	527			
Bottom Drain	527	527			
Standby Liquid Control	527	527			
Core Spray	527	527			
Feedwater	420	354	15 min	264	66
	354	420	15 min	264	66
CRDHS Return	50	50			
Clean Return	435	435			
Remarks: Reactor pressure remains at 1038 psig.					

TABLE 3.9-4 (Page 4 of 21)

APPLICABLE THERMAL TRANSIENTS

(WEEKLY POWER REDUCTION)
2000 CYCLES
CONDITION - NORMAL

Pipeline	Initial Temp. °F	Final Temp. °F	Time	Temp. Rate °F/Hour	Temp. °F
Main Steam Line	551	551			
Head Spray (RHR)	551	551			
Recirculation Suction	527	527			
Recirculation Discharge	527	527			
Bottom Drain	527	527			
Standby Liquid Control	527	527			
Core Spray	527	527			
Feedwater	420	324	30 min	192	96
	324	420	30 min	192	96
CRDHS Return	50	50			
Cleanup Return	435	435			
Remarks: Reactor pressure remains at 1038 psig.					

TABLE 3.9-4 (Page 5 of 21)

APPLICABLE THERMAL TRANSIENTS

(TURBINE TRIP 100 PERCENT BYPASS)
10 CYCLES
CONDITION - UPSET

Pipeline	Initial Temp. °F	Final Temp. °F	Time	Temp. Rate °F/Hour	Temp. °F
Main Steam Line	551	551			
Head Spray (RHR)	551	551			
Recirculation Suction	527	495	1.5 min	1280	32
	495	527	4 min	480	32
Recirculation Discharge	495	527	4 min	480	32
Bottom Drain	495	527	4 min	480	32
Standby Liquid Control	495	527	4 min	480	32
Core Spray	495	527	4 min	480	32
Feedwater	420	100	1.5 min	12,800	320
	100	420	4 min	4800	320
CRDHS Return	50	50			

Remarks: Reactor pressure remains at 1038 psig.

TABLE 3.9-4 (Page 6 of 21)

APPLICABLE THERMAL TRANSIENTS

(PARTIAL FEEDWATER HEATER BYPASS)
70 CYCLES
CONDITION - UPSET

Pipeline	Initial Temp. °F	Final Temp. °F	Time	Temp. Rate °F/Hour	Temp. °F
Main Steam Line	551	551			
Head Spray (RHR)	551	551			
Recirculation Suction	527	517	2 min	300	10
	517	527	4 min	150	10
Recirculation Discharge	517	527	4 min	150	10
Bottom Drain	517	527	4 min	150	10
Standby Liquid Control	517	527	4 min	150	10
Core Spray	517	527	4 min	150	10
Feedwater	420	265	1.5 min	6200	155
	265	420	3 min	3100	155
CRDHS Return	50	50			

Remarks: Reactor pressure remains at 1038 psig.

TABLE 3.9-4 (Page 7 of 21)

APPLICABLE THERMAL TRANSIENTS

(SCRAM - T/G TRIP FEEDWATER ON - MSIV OPEN)
40 CYCLES
CONDITION - UPSET

Pipeline	Initial Temp. °F	Final Temp. °F	Time	Temp. Rate °F/Hour	Temp. °F
Main Steam Line	551	565	10 sec	5040	14
	565	543	15 sec	5280	22
	543	400	-	100	143
	400	551	-	100	151
Head Spray (RHR)	400	551	-	100	151
Recirculation Suction	527	400	-	100	127
	400	551	-	100	151
	551	543	Step	-	8
	543	527	Step	-	16
Recirculation Discharge	543	527	Step	-	16
Bottom Drain	543	527	Step	-	16
Standby Liquid Control	543	527	Step	-	16
Core Spray	543	527	Step	-	16
Feedwater	420	275	1 min	8700	145
	275	100	15 min	700	175
	100	250	Step	-	150
	250	420	30 min	340	170
CRDHS Return	50	50			

Remarks: Reactor pressure increases to 1163 psig all relief valves open. Pressure decreases to 240 psig and then increases to 1038 psig.

TABLE 3.9-4 (Page 8 of 21)

APPLICABLE THERMAL TRANSIENTS

(ALL OTHER SCRAMS)
140 CYCLES
CONDITION - UPSET

Pipeline	Initial Temp. °F	Final Temp. °F	Time	Temp. Rate °F/Hour	Temp. °F
Main Steam Line	551	543	15 sec	1920	8
	543	400	-	100	143
	400	551	-	100	151
Head Spray (RHR)	400	551	-	100	151
Recirculation Suction	527	400	-	100	127
	400	551	-	100	151
	551	543	Step	-	8
	543	527	Step	-	16
Recirculation Discharge	543	527	Step	-	16
Bottom Drain	543	527	Step	-	16
Standby Liquid Control	543	527	Step	-	16
Core Spray	543	527	Step	-	16
Feedwater	420	275	1 min	8700	145
	275	100	15 min	700	175
	100	250	Step	-	150
	250	420	30 min	340	170
CRDHS Return	50	50			

Remarks: Reactor pressure decreases to 240 psig and then increases to 1038 psig.

TABLE 3.9-4 (Page 9 of 21)

APPLICABLE THERMAL TRANSIENTS

(RATED POWER)
SEE BELOW
CONDITION - NORMAL

Pipeline	Initial Temp. °F	Final Temp. °F	Time	Temp. Rate °F/Hour	Temp. °F
Main Steam Line	551	551			
Head Spray (RHR)	551	551			
Recirculation Suction	527	527			
Recirculation Discharge	527	527			
Bottom Drain *240 Times	527	150	1 hr	377	377
	150	527	Step	-	377
Standby Liquid Control *10 Times	527	60	Step	-	467
	60	527	60 min	467	467
Core Spray	527	527			
Feedwater	420	420			
CRDHS Return	50	50			
Cleanup Return	435	435			
Remarks: Reactor pressure remains at 1038 psig.					

TABLE 3.9-4 (Page 10 of 21)

APPLICABLE THERMAL TRANSIENTS

(SHUT DOWN)
111 CYCLES
CONDITION - NORMAL

Pipeline	Initial Temp. °F	Final Temp. °F	Time	Temp. Rate °F/Hour	Temp. °F
Main Steam Line	551	375	-	100	176
	375	330	10 min	270	45
	330	100	-	100	230
Head Spray (RHR) and RHR Return	551	375	-	100	176
	375	50	Step	15 sec Duration	325
	50	300	Step		250
	300	100	-	100	200
Recirculation Suction	527	551	Step	-	24
	551	375	-	100	176
	375	330	10 min	270	45
	330	100	-	100	230
Bottom Drain	330	100	-	100	230
Standby Liquid Control	330	100	-	100	230
Core Spray	330	100	-	100	230
Recirculation Discharge	527	551	Step	-	24
	551	375	-	100	176
	375	300	Step	-	75
	300	260	10 min	240	40
	260	100	-	100	160
Feedwater *5 step changes to 100°F and back during cooldown	420	265	30 min	310	155
	265	420	Step	-	155
	420	551	-	100	131
	551	100	-	100	451
CRDHS Return	50	50			

Remarks: Reactor pressure decreases from 1038 psig to 0 psig.

TABLE 3.9-4 (Page 11 of 21)

APPLICABLE THERMAL TRANSIENTS

(LOSS OF FEEDWATER PUMPS - MSIV CLOSED)
10 CYCLES
CONDITION - EMERGENCY

Pipeline	Initial Temp. °F	Final Temp. °F	Time	Temp. Rate °F/Hour	Temp. °F
Main Steam Line	551	571	3 sec	24,000	20
	571	565	10 sec	2,160	6
	565	536	3 min	580	29
	536	565	73 min	24	29
	565	505	7 min	514	60
	505	400	-	100	105
	400	551	-	100	151
Head Spray (RHR)	400	551	-	100	151
Recirculation Suction	527	300	30 min	454	227
	300	551	-	100	251
	551	543	Step	-	8
	543	527	Step	-	16
Recirculation Discharge	543	527	Step	-	16
Bottom Drain	527	300	3.7 min	3,681	227
	300	505	23 min	535	205
	505	300	3 min	4,100	205
	300	505	73 min	169	205
	505	300	7 min	1,757	205
	300	551	-	100	251
	551	543	Step	-	8
	543	527	Step	-	16
Standby Liquid Control	543	527	Step	-	16
Core Spray	543	527	Step	-	16

TABLE 3.9-4 (Page 12 of 21)

APPLICABLE THERMAL TRANSIENTS

(LOSS OF FEEDWATER PUMPS - MSIV CLOSED) **(continued)**

Pipeline	Initial Temp. °F	Final Temp. °F	Time	Temp. Rate °F/Hour	Temp. °F
Feedwater	420	551	Step	-	131
	551	40	Step	-	511
	40	551	23 min	1,333	511
	551	40	Step	-	511
	40	551	51 min	601	511
	551	40	Step	-	511
	40	300	5 min	3120	260
	300	551	-	100	251
	551	100	Step	-	451
	100	250	Step	-	150
	250	420	30 min	340	170
CRDHS Return	50	545	10 min	2,970	495

Remarks: Reactor pressure increases to 1218 psig. All relief valves open. Pressure decreases to 1163 psig and relief valves close. RCIC initiates and pressure decreases to 913 psig. RCIC trips off on high level and pressure increases to 1163 psig and one relief valve opens and then closes as pressure decreases at rate of 100°F/hr.

TABLE 3.9-4 (Page 13 of 21)

APPLICABLE THERMAL TRANSIENTS

(REACTOR OVERPRESSURE DELAYED SCRAM)
1 CYCLE
CONDITION - EMERGENCY

Pipeline	Initial Temp. °F	Final Temp. °F	Time	Temp. Rate °F/Hour	Temp. °F
Main Steam Line	551	583	2 sec	57,600	32
	583	543	30 sec	4,800	40
	543	400	-	100	143
Head Spray (RHR)	543	400	-	100	143
Recirculation Suction	527	562	11 sec	11,455	35
	562	400	-	100	162
Recirculation Discharge	562	400	-	100	162
Bottom Drain	562	400	-	100	162
Standby Liquid Control	562	400	-	100	162
Core Spray	562	400	-	100	162
Feedwater	420	276	1 min	8640	144
	276	100	15 min	704	176
	100	250	Step	-	150
	250	420	30 min	340	170
CRDHS Return	100	50	Step	-	50

Remarks: Reactor pressure increases to 1350 psig. All relief valves and safety valves open. Pressure decreases to 240 psig.

TABLE 3.9-4 (Page 14 of 21)

APPLICABLE THERMAL TRANSIENTS

(SINGLE SAFETY/RELIEF VALVE BLOWDOWN)
8 CYCLES
CONDITION - EMERGENCY

Pipeline	Initial Temp. °F	Final Temp. °F	Time	Temp. Rate °F/Hour	Temp. °F
Main Steam Line	551	375	10 min	1,056	176
	375	100	-	100	275
Head Spray (RHR)	375	100	-	100	275
Recirculation Suction	527	375	10 min	912	152
	375	100	-	100	275
Recirculation Discharge	375	100	-	100	275
Bottom Drain	275	100	-	100	275
Standby Liquid Control	375	100	-	100	275
Core Spray	375	100	-	100	275
Feedwater	420	276	1 min	8640	144
	276	100	15 min	704	176
CRDHS Return	50	50			

Remarks: Reactor pressure decreases to 0 psig with one relief valve or safety valve open.

TABLE 3.9-4 (Page 15 of 21)

APPLICABLE THERMAL TRANSIENTS

(AUTOMATIC DEPRESSURIZATION)
1 CYCLE
CONDITION - EMERGENCY

Pipeline	Initial Temp. °F	Final Temp. °F	Time	Temp. Rate °F/Hour	Temp. °F
Main Steam Line	551	375	3.3 min	3,200	176
	375	281	-	300	94
Head Spray (RHR)	375	281	-	300	94
Recirculation Suction	527	375	3.3 min	2,764	152
	375	281	-	300	94
Recirculation Discharge	375	281	-	300	94
Bottom Drain	375	281	-	300	94
Standby Liquid Control	375	281	-	300	94
Core Spray	375	281	-	300	94
Feedwater	420	276	1 min	8640	144
	276	100	15 min	704	176
CRDHS Return	50	50			

Remarks: Reactor pressure decreases to 35 psig with auto-blowdown relief valves open.

TABLE 3.9-4 (Page 16 of 21)

APPLICABLE THERMAL TRANSIENTS

(IMPROPER START OF COLD RECIRCULATION LOOP)
1 CYCLE
CONDITION - EMERGENCY

Pipeline	Initial Temp. °F	Final Temp. °F	Time	Temp. Rate °F/Hour	Temp. °F
Main Steam Line	551	551			
Head Spray (RHR)	551	551			
Recirculation Suction	527	130	Step	26 sec Duration	397
	130	527	Step		397
Recirculation Discharge	527	130	Step	34 sec Duration	397
	130	527	Step		397
Bottom Drain	527	527			
Standby Liquid Control	527	527			
Core Spray	527	268	Step	34 sec Duration	259
	268	527	Step		259
Feedwater	420	420			
CRDHS Return	50	50			
Remarks: Reactor pressure remains at 1038 psig.					

TABLE 3.9-4 (Page 17 of 21)

APPLICABLE THERMAL TRANSIENTS

(SUDDEN PUMP START IN COLD LOOP)
1 CYCLE
CONDITION - EMERGENCY

Pipeline	Initial Temp. °F	Final Temp. °F	Time	Temp. Rate °F/Hour	Temp. °F
Main Steam Line	551	551			
Head Spray (RHR)	551	551			
Recirculation Suction	527	527			
Recirculation Discharge	527	130	Step	34 second Duration	397
	130	527	Step		397
Bottom Drain	527	350	Step		177
	350	527	Step		177
Standby Liquid Control	350	527	Step		177
Core Spray	527	527			
Feedwater	420	420			
CRDHS Return	50	50			
Remarks: Reactor pressure remains at 1038 psig.					

TABLE 3.9-4 (Page 18 of 21)

APPLICABLE THERMAL TRANSIENTS

(IMPROPER START WITH RECIRCULATION PUMPS OFF)
1 CYCLE
CONDITION - EMERGENCY

Pipeline	Initial Temp. °F	Final Temp. °F	Time	Temp. Rate °F/Hour	Temp. °F
Main Steam Line	100	551	-	100	451
Head Spray (RHR)	100	551	-	100	451
Recirculation Suction	100	551	-	100	451
Recirculation Discharge	100	551	-	100	451
	551	130	Step	34 sec Duration	421
	130	551	Step		421
Bottom Drain	100	551	5 min	5,412	451
Standby Liquid Control	100	551	5 min	5,412	451
Core Spray	100	551	-	100	451
Feedwater	90	551	-	100	461
	551	90	Step	-	461
	90	420	30 min	660	330
CRDHS Returns	50	50			

Remarks: Reactor pressure increases to 1038 psig as temperature increases.

TABLE 3.9-4 (Page 19 of 21)

APPLICABLE THERMAL TRANSIENTS

(PIPE RUPTURE AND BLOWDOWN)
1 CYCLE
CONDITION - FAULTED

Pipeline	Initial Temp. °F	Final Temp. °F	Time	Temp. Rate °F/Hour	Temp. °F
Main Steam Line	551	281	15 sec	64,800	270
Head Spray (RHR)	551	281	15 sec	64,800	270
Recirculation Suction	527	281	15 sec	59,040	246
Recirculation Discharge	527	281	15 sec	59,040	246
	281	223	35 sec	5,966	58
	223	50	Step	90 sec Duration	173
	50	130	Step		80
Bottom Drain	527	281	15 sec	59,040	246
	281	273	35 sec	822	8
	273	50	Step	90 sec Duration	223
	50	130	Step		80
Standby Liquid Control	50	130	Step		80
Core Spray	527	406	10 sec	43,560	121
	406	50	Step	90 sec Duration	356
	50	130	Step		80
Feedwater	420	281	15 sec	33,400	139
CRDHS Return	50	50			
Remarks: Reactor pressure decreases from 1038 psig to 35 psig in 15 seconds.					

TABLE 3.9-4 (Page 20 of 21)

APPLICABLE THERMAL TRANSIENTS

(BECHTEL CRITERIA)

1/2 SSE Cycles (Operating Basis Earthquake) Condition-Upset	
Expected number of equivalent 1/2 SSE in life of pipe system	5
Average duration of strong motion vibration 1/2 SSE	15 sec
Average number of maximum seismic load cycles of pipe system for each 1/2 SSE	10
Total lifetime number of maximum seismic load cycles of piping system	50
SSE Cycles (Design Basis Earthquake) Condition-Faulted	
Expected number of equivalent SSE in life of pipe system	1
Average duration of strong motion vibration SSE	15 sec

TABLE 3.9-4 (Page 21 of 21)

APPLICABLE THERMAL TRANSIENTS

(GENERAL ELECTRIC CRITERIA)
FOR NSSS PIPING

1. 1/2 SSE Cycles Condition - Upset

Expected number of equivalent 1/2 SSE in life of pipe system	1
Average duration of strong motion vibration 1/2 SSE	30 sec
Average number of maximum seismic load cycles of pipe system for each 1/2 SSE	10
Total lifetime number of maximum seismic load cycles of piping system	10

2. SSE Cycles Condition-Faulted

Expected number of equivalent SSE in life of pipe system	1
Average duration of strong motion vibration SSE	30 sec
Average number of maximum seismic load cycles of pipe system for each SSE	1
Total lifetime number of maximum seismic load cycles of piping system	1

3. Turbine Stop Valve* Closure (TSVC) Condition-Upset 120 cycles

4. Relief Valve Lift* (RVL) Condition-Upset 20,100 cycles

* not applicable to recirculation piping due to negligible effect.

Table 3.9-5
LIST OF COMPUTER PROGRAMS USED IN BOP MECHANICAL
SYSTEMS AND COMPONENTS

Security-Related Information
Table Withheld Under 10 CFR 2.390

TABLE 3.9-6

DESIGN LOADING COMBINATIONS FOR ASME CODE CLASS 1, 2, AND 3 COMPONENTS (NON-NSSS)

Condition	Design Loading Combinations(1)
Design	PD
Normal	PD + DW
Upset	(a) PO + DW + $(OBE^2 + SRV_x^2)1/2$ (b) PO + DW + $(RVC^2 + OBE^2)1/2$ (c) PO + DW + FV (d) PO + DW + OBE + RVO
Emergency	(a) PO + DW + $(OBE^2 + SRV^2_{ADS} + SBA^2)1/2$ (b) PO + DW + $(FV^2 + OBE^2)1/2$
Faulted	(a) PO + DW + $(OBE^2 + SRV^2_{ADS} + IBA^2)1/2$ (b) PO + DW + $(SSE^2 + SRV^2_{ADS} + IBA^2)1/2$ (c) PO + DW + $(SSE^2 + DBA^2)1/2$

NOTE: (1) As required by the appropriate subsection, i.e. NB, NC, or ND, of ASME Section III Division I. Other loads, such as loads from thermal transient, thermal gradients, and the anchor point displacement portion of the OBE or SRV may require consideration in addition to those primary stress-producing loads listed.

Legend:

- PD - Design Pressure
- PO - Operating Pressure
- DW - Dead weight
- OBE - Operating basis earthquake (inertia portion)
- SSE - Safe shutdown earthquake (inertia portion)
- SRV_X - Loads due to Safety Relief Valve Blow - Axisymmetric or Asymmetric
- SRV_{ADS} - Loads due to Automatic Depressurization SRV Blow-Axisymmetric
- SBA - Small Break Accident
- IBA - Intermediate Break Accident
- DBA - Design Basis Accident
- FV - Transient response of the piping system associated with fast valve closure. Transients associated with valve closure time less than 5 seconds are considered.
- RVC - Transient response of the piping system associated with relief valve opening in a closed system.
- RVO - Sustained load or response of the piping system associated with relief valve opening in an open system or last segment of the closed system with steady state load.

SBA, IBA, and DBA include all event induced loads, as applicable, such as chugging, condensation oscillation, pool swell, drag loads, annulus pressurization, etc.

For the NSSS load combination, see Table 3.9-2.

TABLE 3.9-7

DESIGN CRITERIA FOR ASME CODE CLASS 1 VALVES
(NON-NSSS)

Condition	Stress Limits
Design	NB-3521[1]
Normal and Upset	NB-3200 or NB-3500[1] (Standard Design Rules)
Emergency[2]	NB-3526[3]
Faulted[2]	NB-3527[3]

(1) As specified by ASME Code Section III, 1971 thru Winter 1972 Addenda.
(2) Where valve function must be ensured (active valve) during the emergency or faulted conditions, the specified emergency or faulted condition for the plant shall be considered the normal condition for the valve.
(3) As specified by ASME Code Section III, 1971, Winter 1973 Addenda.

TABLE 3.9-8

DESIGN CRITERIA FOR ASME CODE CLASS 2 AND 3 VESSELS DESIGNED TO NC-3300 AND ND-3300

Condition	Stress Limits[1][2]
Design and Normal	The vessel shall conform to the requirements of NC-3300 and ND-3300.
Upset, Emergency, and Faulted	The vessel shall conform to the requirements of ASME Code Case 1607-1.

(1) As specified by ASME Code Section III, 1971 thru 1972 Winter Addenda.
(2) For the Diesel Generator E Facility, the design criteria for class 3 vessels is governed by ASME code, Section III, 1980 edition through summer 1981 addenda.

TABLE 3.9-9

DESIGN CRITERIA FOR ASME CODE CLASS 2 VESSELS
DESIGNED TO ALTERNATE RULES OF NC-3200

Page 1 of 2

Condition	Stress Limits(1)(2)
Design and Normal	The vessel shall conform to the requirements of NC-3200.
Upset(3)	$P_e \leq 3\ S_m$ $P_m \leq 1.1\ S_m$ $(P_m \text{ or } P_L) + P_b \leq 1.65\ S_m$
Emergency	$P_m \leq$ greater of $1.2\ S_m$ or $1.0\ S_y$ $(P_m \text{ or } P_L) + P_b \leq$ greater of $1.8\ S_m$ or $1.5\ S_y$
Faulted (4)	$P_m \leq 2.0\ S_m$ $(P_m \text{ or } P_L) + P_b \leq 2.4\ S_m$

(1) Definition of symbols:

P_m	=	General primary membrane stress intensity. This stress intensity is derived from the average value across the solid section under consideration. Excludes discontinuities and concentrations. Produced only by pressure and other mechanical loads.
P_L	=	Local primary membrane stress intensity. Same as P_m except that discontinuities are considered.
P_b	=	Primary bending stress intensity. Component of primary stress intensity proportional to distance from centroid of solid section. Excludes discontinuities and concentrations. Produced only by pressure and other mechanical loads.
P_e	=	Secondary stress intensity range. Developed by constraint of adjacent parts or by self-constraint of a structure. Considers discontinuities but not concentrations. Produced by mechanical loads and by thermal expansion.
S_m	=	Design stress intensity value, Appendix I, Table I-1.0.
S_y	=	Yield strength value, Appendix I, Table I-2.0.

(2) These limits do not take into account either local or general buckling that might occur in thin-wall vessels. Such buckling shall be considered for upset conditions, but need not be considered for emergency or faulted conditions unless required by the design specification.

TABLE 3.9-9

DESIGN CRITERIA FOR ASME CODE CLASS 2 VESSELS DESIGNED TO ALTERNATE RULES OF NC-3200

Condition	Stress Limits(1)(2)

(3)	Fatigue analysis requirements of NC-3219 and Appendix XIV shall also be considered.
(4)	As an alternative to satisfying these limits, the faulted condition stress limits of Appendix F may be applied provided that a complete analysis in accordance with NC-3211.1(c) is performed.

TABLE 3.9-10

DESIGN CRITERIA FOR ASME CODE CLASS 2 AND 3 PIPING

Condition	Stress Limits[1]
Design, Normal, Upset, and Emergency [2]	The piping shall conform to the requirements of Section III, paragraphs NC-3600 and ND-3600.
Faulted[2]	The piping shall conform to the requirements of ASME Code Case 1606.

(1) As specified by ASME Code Section III, 1971 through 1972 Winter Addenda. For diesel generator 'E' facility piping on the auxiliary and air start skids the applicable code is ASME Section III, 1980 through Summer 1981 Addenda. See Table 3.2-3 for later code editions used for snubber elimination or other piping modifications.

(2) Functional capability of essential piping will be assured per Rodabaugh Criteria for emergency and faulted conditions only, Ref. 3.9-8.

TABLE 3.9-11

DESIGN CRITERIA FOR ASME CODE CLASS 2 AND 3 PUMPS

Condition	Stress Limits(2)(3)
Design and Normal	The pump shall conform to the requirements of Section III, Paragraphs NC-3400 and ND-3400.
Upset, Emergency(1) and Faulted(1).	The pump shall conform to the requirements of ASME Code Case 1636-1.

(1) Where pump function must be ensured (active pumps) during the emergency or faulted condition, the pumps nozzle loads due to the specified emergency or faulted plant conditions shall be considered in satisfying the normal condition stress limits for the pump.

(2) As specified by ASME Code Section III, 1971 and Winter 1972 Addenda.

(3) For pumps mounted on the diesel generator 'E' auxiliary skid, the design criteria is governed by ASME Code, Section III, 1980 edition through Summer 1981 Addenda.

TABLE 3.9-12

DESIGN CRITERIA FOR ASME CODE CLASS 2 AND 3 VALVES

Condition	Stress Limits[2][3]
Design and Normal	The valve shall conform to the requirements of Section III, Paragraphs NC-3500 and ND-3500.
Upset, Emergency[1] and Faulted[1]	The valve shall conform to the requirements of ASME Code Case 1635-1.

(1) Where valve function must be ensured (active valve) during the emergency or faulted condition, the specified emergency or faulted conditions for the plant shall be considered as the normal condition for the valve.

(2) As specified by ASME Code Section III, 1971 and Winter 1972 Addenda.

(3) For valves supplied on the diesel generator 'E' auxiliary and air start skids, the design criteria is governed by the ASME Code, Section III, 1980 edition through Summer 1981 Addenda.

TABLE 3.9-14

DESIGN LOADING COMBINATIONS FOR SUPPORTS FOR ASME CODE CLASS 1, 2, AND 3 COMPONENTS[3] (NON-NSSS)

Condition	Design Loading Combinations[1]	Allowable Stress*[2]
Hydrostatic Test	(a) HTDW	0.8 Sy*
Normal and Upset	(a) $DW + TH + (OBE^2 + SRV^2)1/2$ (b) $DW + TH + (RVC^2 + OBE^2)1/2$ (c) $DW + TH + FV$ (d) $DW + TH + OBE + RVO$	S_h
Emergency	(a) $DW + TH + (OBE^2 + SRV^2_{ADS} + SBA^2)1/2$ (b) $DW + TH + (OBE^2 + FV^2)1/2$	1.8 S_h
Faulted	(a) $DW + TH + (SSE^2 + SRV^2_{ADS} + IBA^2)1/2$ (b) $DW + TH + (OBE^2 + SRV^2_{ADS} + IBA^2)1/2$ (c) $DW + TH + (SSE^2 + DBA^2)1/2$	0.9 Sy

* Snubbers, compensating struts and struts comply with all the requirements of ASME Section III, Subsection NF. (They are not commercially available to the requirements of ANSI B31.1.)

NOTES:
(1) Loads due to OBE, SSE, SRV_x, SRV_{ADS}, SBA, IBA and DBA include both inertia portion and anchor motion portion when response spectra method is used. The loads from inertia portion and anchor motion are combined by the method of Square Root of the Sum of Squares (SRSS).
(2) The allowable stress shall be limited to two-thirds of the critical buckling stress.
(3) Supports on the diesel generator E and air start skids comply with all the requirements of ASME, Section III, Subsection JF, 1980 edition through Summer 1981.

Legend:
- HTDW - Piping dead weight due to hydrostatic test
- TH - Reaction at the support due to thermal expansion of the pipe
- Sy - Yield stress
- S_h - Allowable stress per ANSI B31.1

See Table 3.9-6 for additional nomenclature.
See Table 3.9-2 for NSSS support.

TABLE 3.9-15

VALVE QUALIFICATION TEST RANGE

Size of Qualified Valve	Qualification Extends To These Valve Sizes																			
	.5	1	1.5	2	3	4	6	8	10	12	14	16	18	20	22	24	26	28	30	36
.5	X	X																		
1	X	X	X																	
1.5		X	X	X																
2			X	X	X															
3				X	X	X														
4					X	X	X													
6						X	X	X												
8							X	X	X	X										
10								X	X	X	X									
12								X	X	X	X	X								
14									X	X	X	X	X	X						
16										X	X	X	X	X	X					
18											X	X	X	X	X	X				
20											X	X	X	X	X	X	X	X		
22												X	X	X	X	X	X	X	X	
24													X	X	X	X	X	X	X	
26														X	X	X	X	X	X	X
28														X	X	X	X	X	X	X
30															X	X	X	X	X	X
36																	X	X	X	X

Table 3.9-16

HISTORICAL INFORMATION

LISTING OF DYNAMICALLY QUALIFIED EQUIPMENT

Security-Related Information
Table Withheld Under 10 CFR 2.390

Table 3.9-17

HISTORICAL INFORMATION

DIESEL GENERATOR ‘A-D’ SEISMIC TEST OR ANALYSIS SUBMITTAL CHART

Security-Related Information
Table Withheld Under 10 CFR 2.390

Table 3.9-18

SUMMARY COMPARISON – PROJECT SPECIFICATOIN – 10, "GENERAL PROJECT REQUIREMENTS FOR A SEISMIC DESIGN AND ANALYSIS OF CLASS 1 EQUIPMENT AND SUPPORTS"*

SPEC. G-10	IEEE-344-1975	REMARKS
1. Analysis		
Equipment is classified as (1) structurally simple and (2) structurally complex. Structurally simple equipment is one which can be adequately represented as a single degree of freedom system or equipment whose fundamental frequency is greater than 33 cps. Otherwise the equipment is structurally complex.	There are two methods of analysis; one approach is based upon equivalent static analysis and the other on dynamic analysis. For the static coefficient analysis, no determination of natural frequency is made but the response of equipment is assumed to be the peak of the RRS at a conservative value of damping multiplied by a static coefficient of 1.5 to take into account the effects of both multifrequency excitation and multifrequency response. A lower value of static coefficient can be used, if it can be shown to yield conservative results.	Reg. Guide 1.100 Rev. 1 of August 1977 indicates that the static coefficient of IEEE-344-1975 of 1.5 is acceptable for verifying structural integrity of frame type structures such as columns, beams that can be represented by a simple model. For equipment having configurations other than frame type structure justification should be provided for the use of static coefficient.
For equipment which is structurally simple due to single degree of freedom system the seismic load consists of a static load corresponding to the equipment weight times the acceleration selected from the response spectrum curve for the natural frequency of the equipment. If the equipment frequency is not known the acceleration shall correspond to the maximum value of the response spectrum.	For the dynamic analysis the equipment shall be modeled to best represent its mass distribution and stiffness characteristics, and this model is used to determine if the equipment is rigid or flexible. If there is no response in the frequency range below the high-frequency asymptote (ZPA) of the RRS, it is considered rigid. Then the seismic forces on each component of the equipment are obtained by concentrating its main at its center of the gravity and multiplying the values of main and the appropriate maximum floor acceleration. If flexible, the model can be analyzed using response spectrum model analysis technique or time history analysis. The response of interest is determined by combining each model response considering all significant modes by SRSS. The absolute sum of similar effects should be considered for closely spaced modes which are those with frequencies differing by 10% or less. In the analysis the effects of each of the two major horizontal directions and the vertical direction should be considered.	
For equipment which is structurally simple due to fundamental frequency greater than 33 cps the seismic load shall consist of a static load corresponding to the acceleration at 33 cps selected from appropriate response spectrum curve with an increase of 50%.		

* The siesmic design of the deisel generator 'E' facility confirms to project specification C-1041 or Cooper Energy Services Standard No. DS-140 and IEEE Standard 344-75 in lieu of project specification G-10.

Table 3.9-18

SUMMARY COMPARISON – PROJECT SPECIFICATOIN – 10, "GENERAL PROJECT REQUIREMENTS FOR A SEISMIC DESIGN AND ANALYSIS OF CLASS 1 EQUIPMENT AND SUPPORTS"*

SPEC. G-10	IEEE-344-1975	REMARKS
For equipment which is structurally complex for the analysis purposes, the equipment shall be idealized by a lumped mass model. Frequencies and mode shapes are determined for the vertical, and two orthogonal horizontal directions. Spectral acceleration per mode shall be obtained from appropriate response spectrum curves, the value chosen to be the largest value on the curve when the frequency is varied by ±0%. The results of the individual modes shall be combined by the square root of sum of the squares method. For closely spaced modes which have frequencies that do not differ by more than 10% the responses of all these modes are combined by sum of the absolute values before combining with other modes by SRSS method.		
2. Damping		
The damping values specified are 1/2 % for OBE and 1% for SSE in the response curves attached with the material requisition and equipment specifications. If there is evidence (such as test results) of different damping values these can be used.	The allowable damping values for the equipment are 2% for the OBE and 3% for SSE. Damping values higher than these can be used if justified by documented test data. If equipment damping is not known a value of 5% is recommended.	Reg. Guide 1.61 for equipment also allows damping values of 2% for OBE and 4% for SSE and higher values are permitted in a dynamic seismic analysis for documented test data are provided.
3. Testing		
Testing for both OBE and SSE loads must be done unless it can be shown for a particular item that the SSE is a more severe condition than the OBE.	Seismic qualifications tests designed to show adequacy of performance during and following a SSE must be preceded by one or more OBE test. The number of tests shall be justified for each site or shall produce the equivalent effect of 5 OBE's.	
4. Testing		
Perform frequency sweep at line amplitude acceleration input varying frequency (at sweep rate of 1/6 of forcing frequency per minute) and determine all resonant frequencies below 33 cps and first resonant frequency above 33 cps if below 65 cps.	Exploratory test may be run in the form of low level continuous sinusoidal sweep (such as 0.2g) at a rate no greater than 2 octaves per minute over the frequency range equal to or greater than that to which the equipment is to be qualified.	

Table 3.9-18

SUMMARY COMPARISON – PROJECT SPECIFICATOIN – 10, "GENERAL PROJECT REQUIREMENTS FOR A SEISMIC DESIGN AND ANALYSIS OF CLASS 1 EQUIPMENT AND SUPPORTS"*

SPEC. G-10	IEEE-344-1975	REMARKS
5. Testing		
Input acceleration, a, for each of horizontal and vertical directions at each reasonant (frequency is determined by, a = 1.5 x 1-(ff/fe) – Sa where. SA is spectral response acceleration for equipment frequency fe. and ff is the forcing frequency of the shaking device (ff 0.8 fe). The factor 1.5 is used to account for the possible excitation of other modes.	The maximum acceleration of shake table should be at least equal to ZPA on RRS. For equipment with more than one predominant frequency the shake motion should provide TRS acceleration at the test frequency of 1.5 times that of RRS or less if justified. The choice of the preceding factor (with largest values of 1.5) is applicable to broadband RRS. As a consequence the TRS need not envelope RRS provided proper justification is given.	Reg Guide 1.100 indicates that the use of factor 1.5 in IEEE-344, and the concept that TRS need not envelope the RRS as a consequence of using 1.5 should not, in the absence of justification, be considered acceptable.
6. Testing		
Duration of excitation of the input acceleration shall not be less than 30 seconds.	The duration of each test shall be least equal the strong motion portion of the original time history used to obtain RRS for the SSE.	The duration of design earthquake for the Susquehanna project is 20 seconds.

TABLE 3.9-20

BOP PIPING SYSTEMS POWER ASCENSION TESTING

Piping System	Code(s) SC/HE/ME (1)	Temp. > 200°F	Thermal Expansion Test (2)	Dynamic Transient Test (3)	Steady State Vibration Test (4)	Remarks
Main Steam	ASME III - 2 B31.1 SC II HE	Yes	Yes	Yes	Yes	Main Stop Valve Closure and SRV Opening Transients
Extraction Steam	B31.1 SC II HE	Yes	N/A(5)	N/A	N/A	
Condensate & Refueling Water Storage	B31.1 SC II ME	No	N/A	N/A	N/A	
Feedwater	ASME III - 1,2 B31.1 HE, SC I SC II	Yes	Yes	Yes	Yes	Power Ascension Test for Safety Related Piping Portion Only
Air Removal and Seal Steam	B31.1 SC II HE	Yes	N/A	N/A	N/A	
Service Water	ASME III - 2,3 B31.1 SC I & SC II ME	No	N/A	N/A	N/A	A Portion of the System has Temperature >200°F But is Less than 300°F
Raw Water Treatment	B31.1 SC II ME	No	N/A	N/A	N/A	
Lube Oil & Diesel Oil Storage & Transfer	ASME III - 3 B31.1 SC I & II ME	No	N/A	N/A	N/A	
Auxiliary Steam	B31.1 SC II, HE	Yes	N/A	N/A	N/A	

TABLE 3.9-20

BOP PIPING SYSTEMS POWER ASCENSION TESTING

Piping System	Code(s) SC/HE/ME (1)	Temp. > 200°F	Thermal Expansion Test (2)	Dynamic Transient Test (3)	Steady State Vibration Test (4)	Remarks
Fire Protection	SC II, ME	No	N/A	N/A	N/A	
Process Sampling	B31.1. SC II, ME	No	N/A	N/A	N/A	
Chlorination	B31.1, SC II ME	No	N/A	N/A	N/A	
Compressed Air	B31.1, SC II ME	No	N/A	N/A	N/A	
Instrument Gas	ASME III - 2,3 B31.1, SC I SC II, ME	No	N/A	N/A	N/A	
Feed Pump Turbine Steam	B31.1, SC II HE	Yes	N/A	N/A	N/A	
Makeup Water	B31.1, SC II ME	No	N/A	N/A	N/A	
Valve Steam Leakoff	B31.1 SC II HE	Yes	N/A	N/A	N/A	
Acid Injection	B31.1, SC II ME	No	N/A	N/A	N/A	
Hydrogen Storage	B31.1, SC II, ME	No	N/A	N/A	N/A	
Diesel Engine Auxiliaries	ASME III - 3 B31.1, SC I & II, ME	Yes	Yes(6)	N/A	N/A	Emergency Diesel Exhaust Has T >300°F and Thermal Expansion Test Performed
				(See Remarks)		
MSIV Leakage Control	ASME III - 1,2 B31.1, SC I SC II, HE	Yes	N/A	N/A	N/A	
Reactor Recirc Motor/Generator	B31.1, SC II ME	No	N/A	N/A	N/A	

Table Rev. 48

TABLE 3.9-20

BOP PIPING SYSTEMS POWER ASCENSION TESTING

Piping System	Code(s) SC/HE/ME (1)	Temp. > 200°F	Thermal Expansion Test (2)	Dynamic Transient Test (3)	Steady State Vibration Test (4)	Remarks
High Pressure Coolant Injection	ASME III - 1,2 B31.1, SC I SC II, HE, ME	Yes	Yes	Yes	Yes	HPCI Turbine Stop Valve Closure Transients for Steam Supply, Steady State Vibration for Steam Supply and Turbine Exhaust HPCI Pump Suction and discharge lines under steady state vibration.
Reactor Core Isolation Cooling	ASME III - 1,2 B31.1, SC I SC II, HE, ME	Yes	Yes	N/A	Yes	Steady State Vibration for RCIC Steam Supply and Turbine Exhaust RCIC pump suction and discharge piping under steady state vibration test
Reactor Water Cleanup	ASME III - 1,2 B31.1, SC I SC II, HE, ME	Yes	Yes	N/A	Yes(6)	Steady State Vibration for RWCU Line Inside Containment
Residual Heat Removal (Includes Head Spray)	ASME III - 1,2,3 B31.1, SC I & II, HE, ME	Yes	Yes	N/A	Yes(6)	Majority of the System has Normal Operating Temperature less than 300°F. Thermal Expansion Tests are done for SCI Systems WAA T >300°F. Steady State Vibration for Inside Containment Piping and RHR Pump Discharge.
		(See Remarks)				
Cleanup Filter Demineralizer	ASME III - 2 B31.1, SC II ME	No	N/A	N/A	N/A	
Control Rod Drive	ASME III - 2 B31.1, SC I, SC II, ME	No	N/A	N/A	Yes(6)	CRD insert/withdrawal pipe.
Standby Liquid Control	ASME III - 1,2 B31.1, SC I & II, HE, ME	No	N/A	N/A	N/A	Only a small portion of the Line near RPV has Temperature >200°F.
		(See Remarks)				
Core Spray	ASME III - 1,2 B31.1, SC I & II HE, ME	Yes	Yes(7)	N/A	Yes(6)	Steady State Vibration For Core Spray Pump Discharge.

Table Rev. 48

TABLE 3.9-20

BOP PIPING SYSTEMS POWER ASCENSION TESTING

Piping System	Code(s) SC/HE/ME (1)	Temp. > 200°F	Thermal Expansion Test (2)	Dynamic Transient Test (3)	Steady State Vibration Test (4)	Remarks
Fuel Pool Cooling, Cleanup & Demineralizer	ASME III - 3 B31.1, SC I SC II, ME	No	N/A	N/A	N/A	
Containment Atmospheric Control	ASME III - 2 B31.1, SC I & II ME	No	N/A	N/A	N/A	
Solid Radwaste and Radwaste Solidification	B31.1 SC II ME	No	N/A	N/A	N/A	
Off-Gas Recombiner	ASME III - 3 B31.1, SC I SC II, HE	Yes	N/A	N/A	N/A	
Ambient Temperature Charcoal Off-Gas Treatment	B31.1, SC II ME	No	N/A	N/A	N/A	
Chilled Water	ASME III - 3 B31.1, SC I SC II, ME	No	N/A	N/A	N/A	

TABLE 3.9-20

BOP PIPING SYSTEMS POWER ASCENSION TESTING

Piping System	Code(s) SC/HE/ME (1)	Temp. > 200°F	Thermal Expansion Test (2)	Dynamic Transient Test (3)	Steady State Vibration Test (4)	Remarks

NOTES: (1) Code(s): ASME III Boiler and Pressure Vessel Code, -1, -2, or -3 Denotes Nuclear Class 1, 2 or 3 Piping.
SC I or II Denote Seismic Category I or II
HE: Denotes High Energy Piping System i.e. Pressure ≥275 PSI or Temperature ≥200° During Normal Plant Operation
ME: Denotes Moderate Energy Piping System

(2) Thermal Expansion Tests for the indicated systems corresponds to test description ST-38, Chapter 14.

(3) Dynamic Transient Tests for the indicated systems corresponds to test description ST-39, Chapter 14.

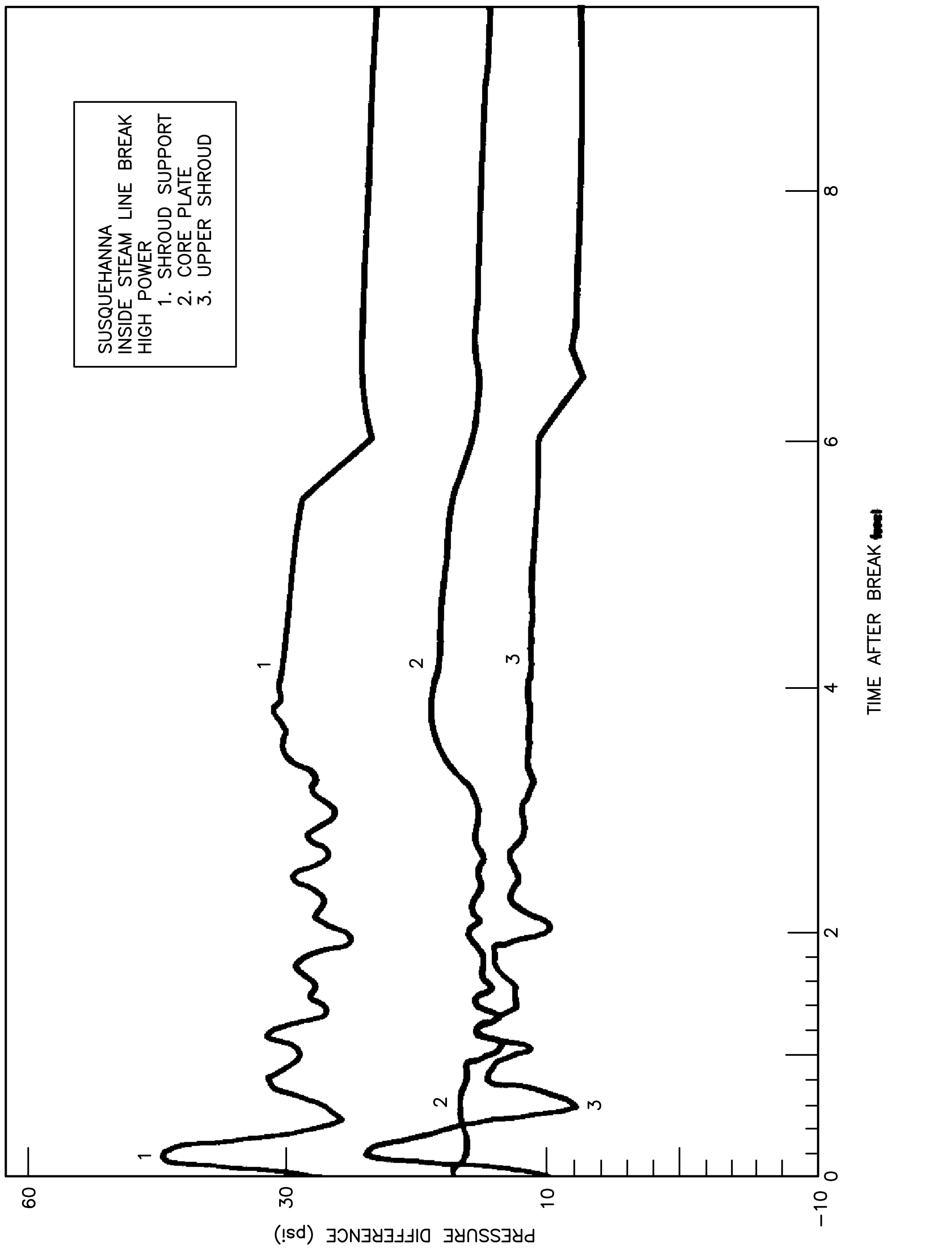
(4) Steady State Vibration Tests for the indicated systems corresponds to test description ST-40, Chapter 14.

(5) N/A - Denotes Not Applicable and it means test is not performed for the reasons given below:

A) For Thermal Expansion Tests: Either the system is not safety-related or the normal operating temperature is less than 300°F.
B) For Dynamic Transient Test: Either the system is not safety-related or the system does not experience any significant transients.
C) For Steady-State Vibration Tests: Either the system is not safety-related or no significant vibration is expected.

(6) Test may be done during Peroperational Test Program.

(7) For the effect of RPV expansion only. No flow in the Core Spray line.



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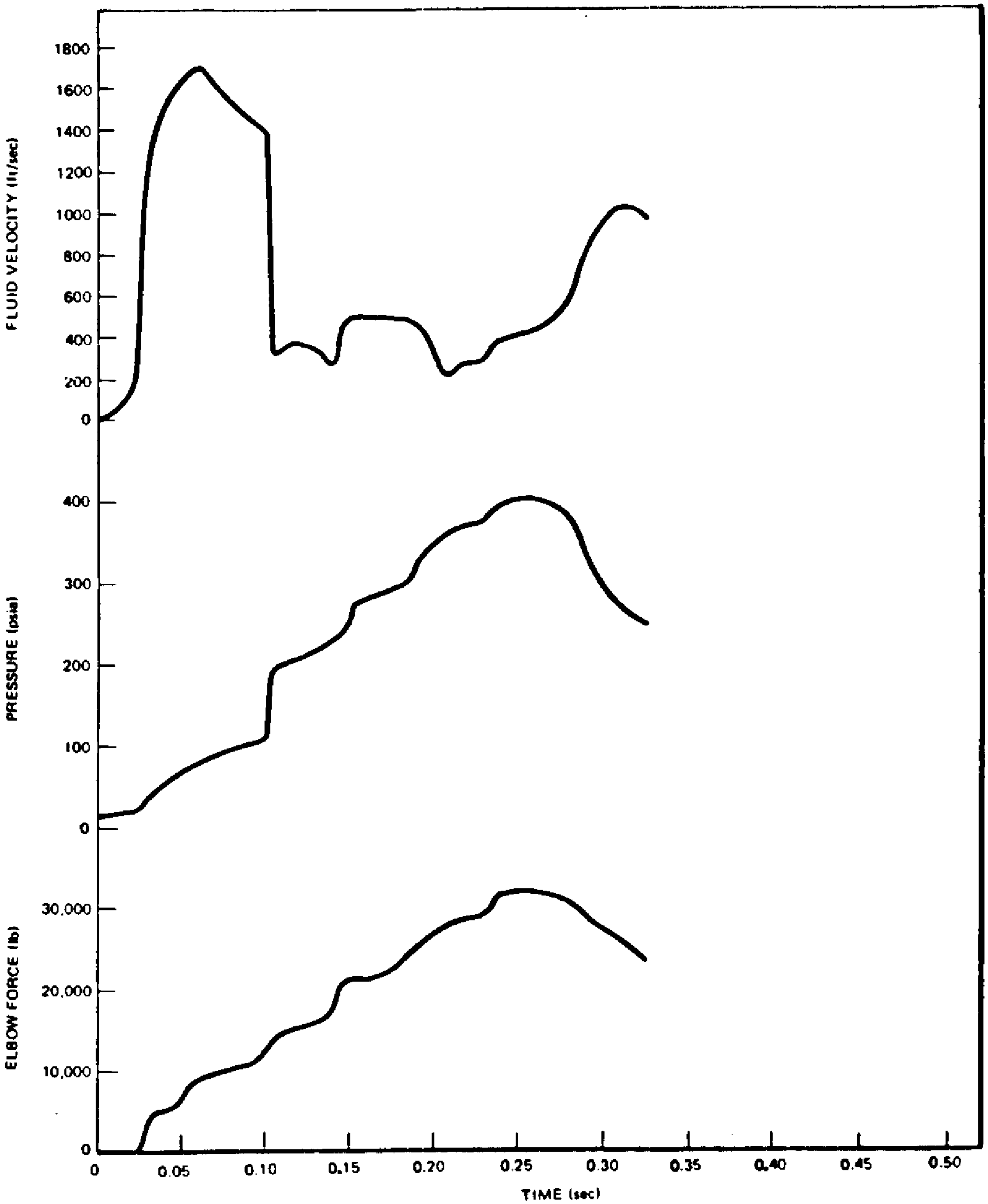
TRANSIENT PRESSURE DIFFERENTIALS
FOLLOWING A STEAMLINE BREAK AT
105% RATED STEAM FLOW
100% RECIRCULATION FLOW

FIGURE 3.9-1, Rev. 48

Auto-Cad Figure Fsar 3_9_1.dwg



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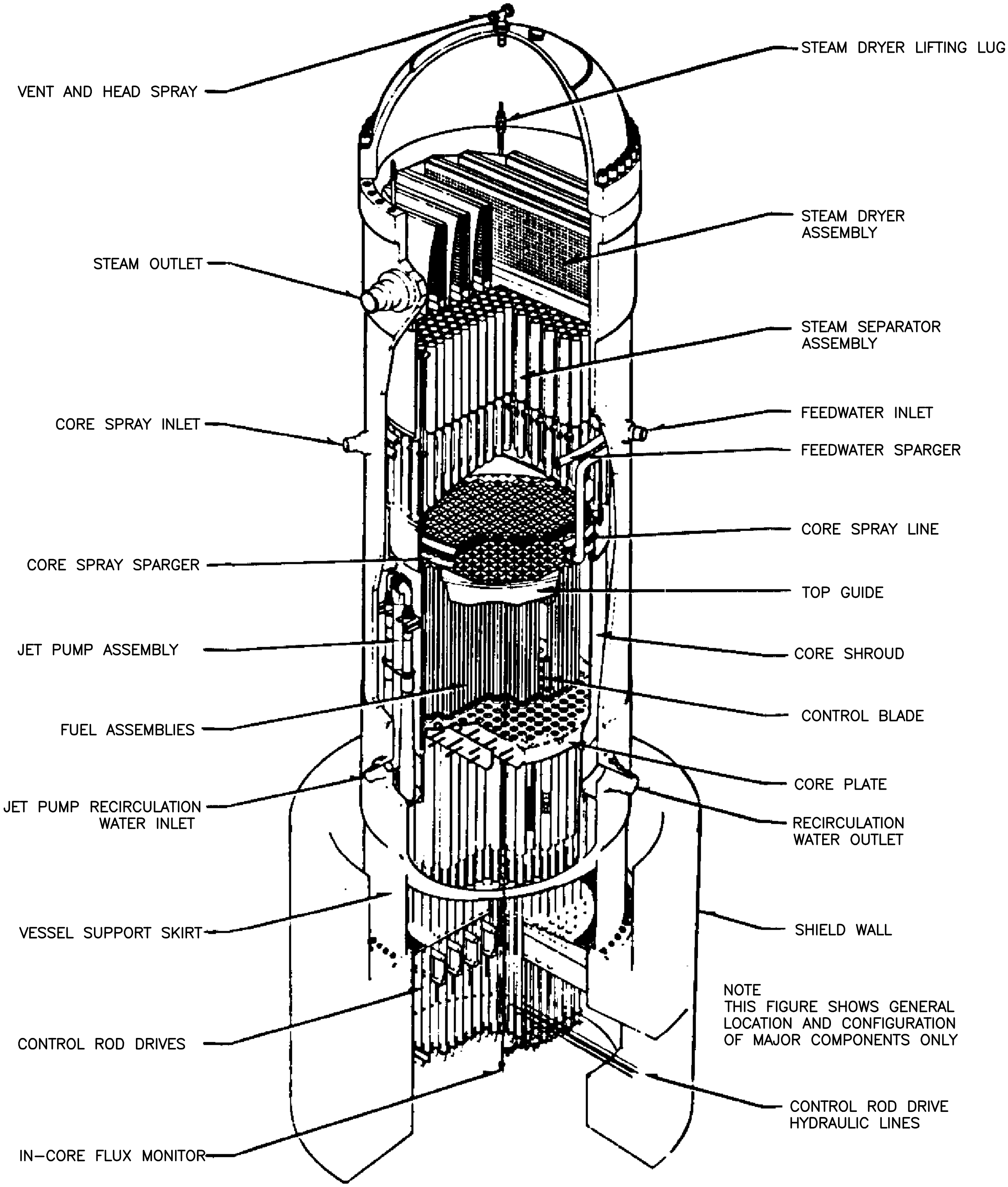
TYPICAL RELIEF VALVE TRANSIENT

FIGURE 3.9-2, Rev. 48

Auto-Cad Figure Fsar 3_9_2.dwg



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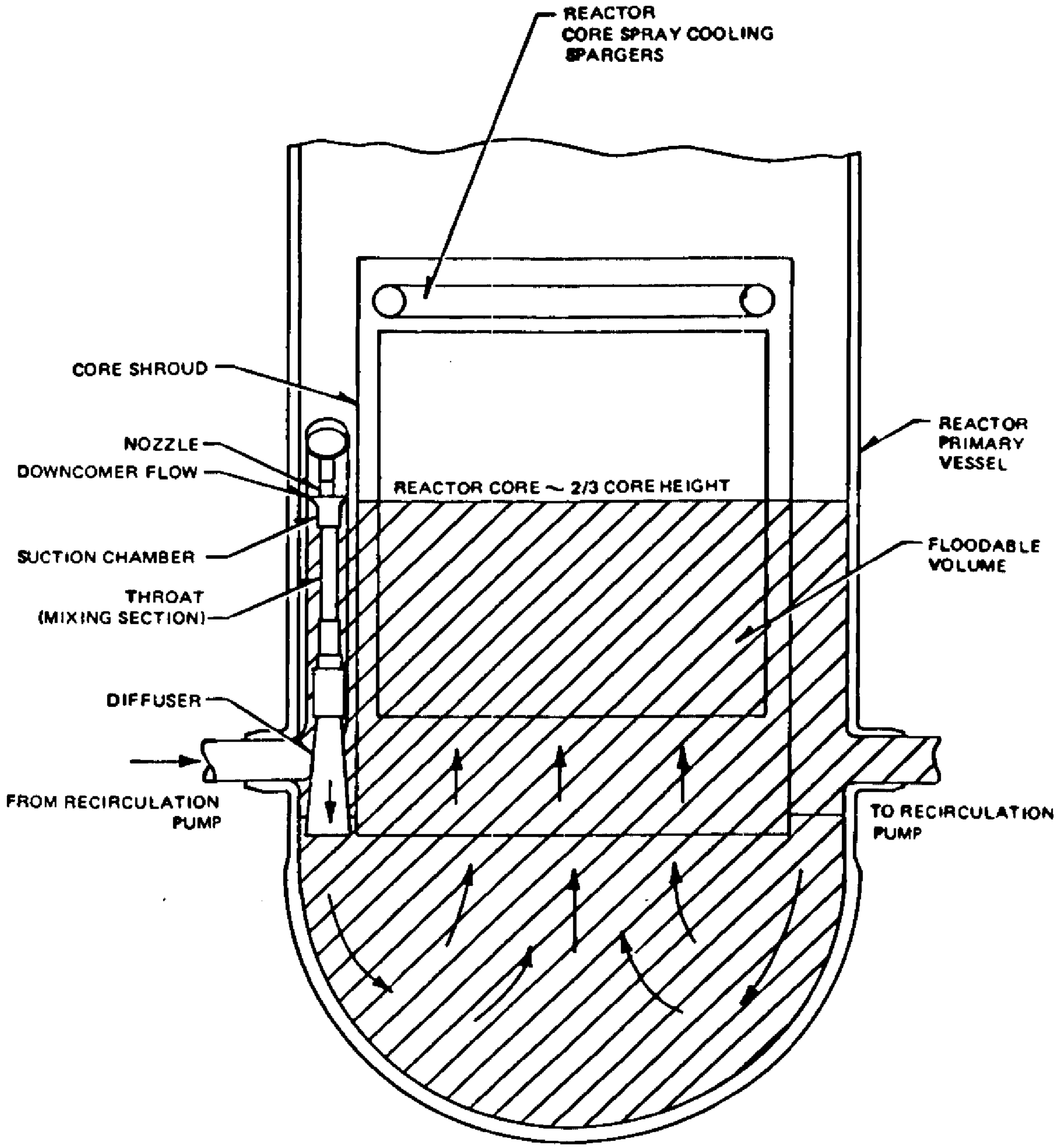
SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
REACTOR VESSEL CUTAWAY FIGURE 3.9-3, Rev. 48

Auto-Cad Figure Fsar 3_9_3.dwg



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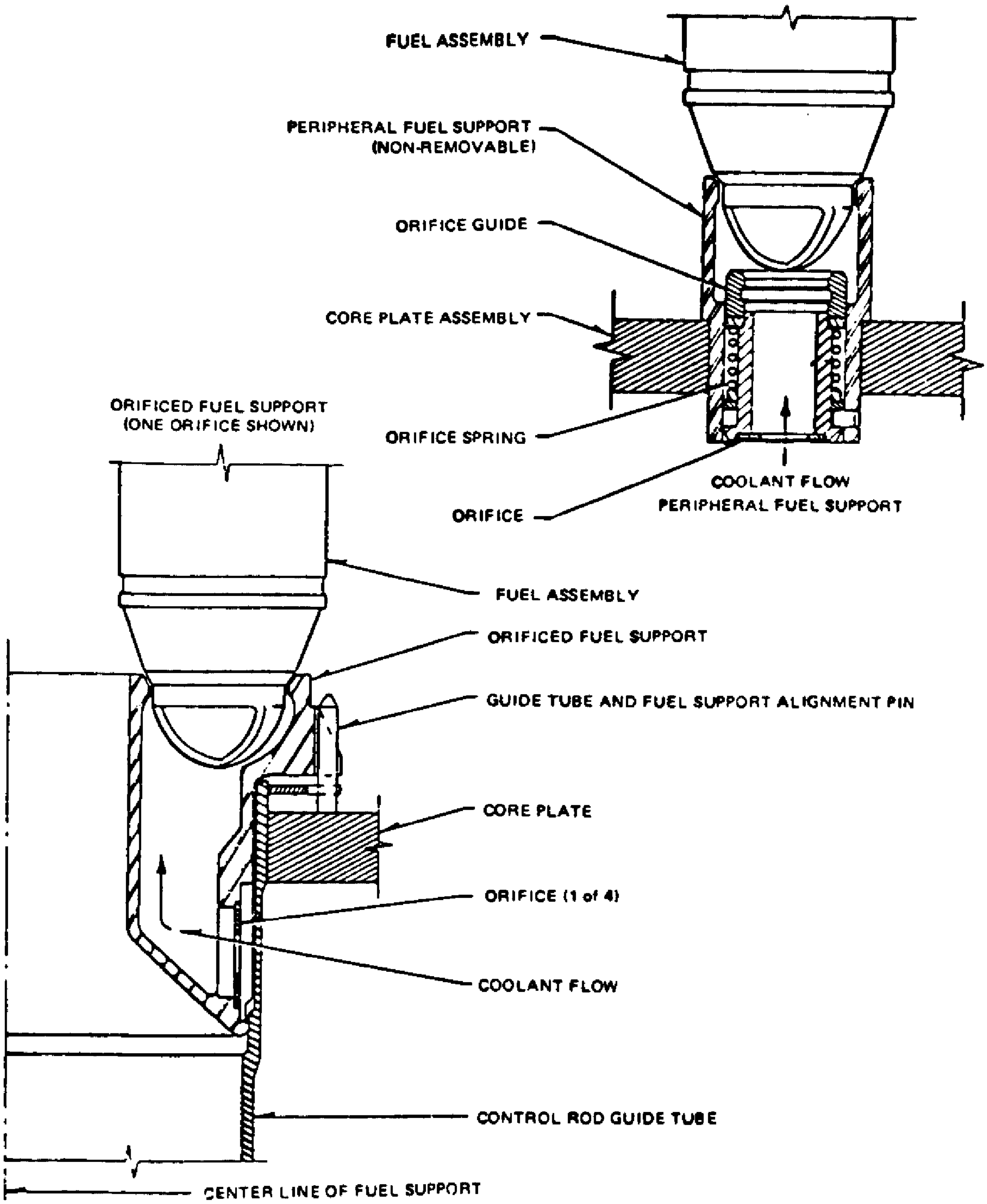
REACTOR INTERNALS FLOW PATHS

FIGURE 3.9-4, Rev. 48

Auto-Cad Figure Fsar 3_9_4.dwg



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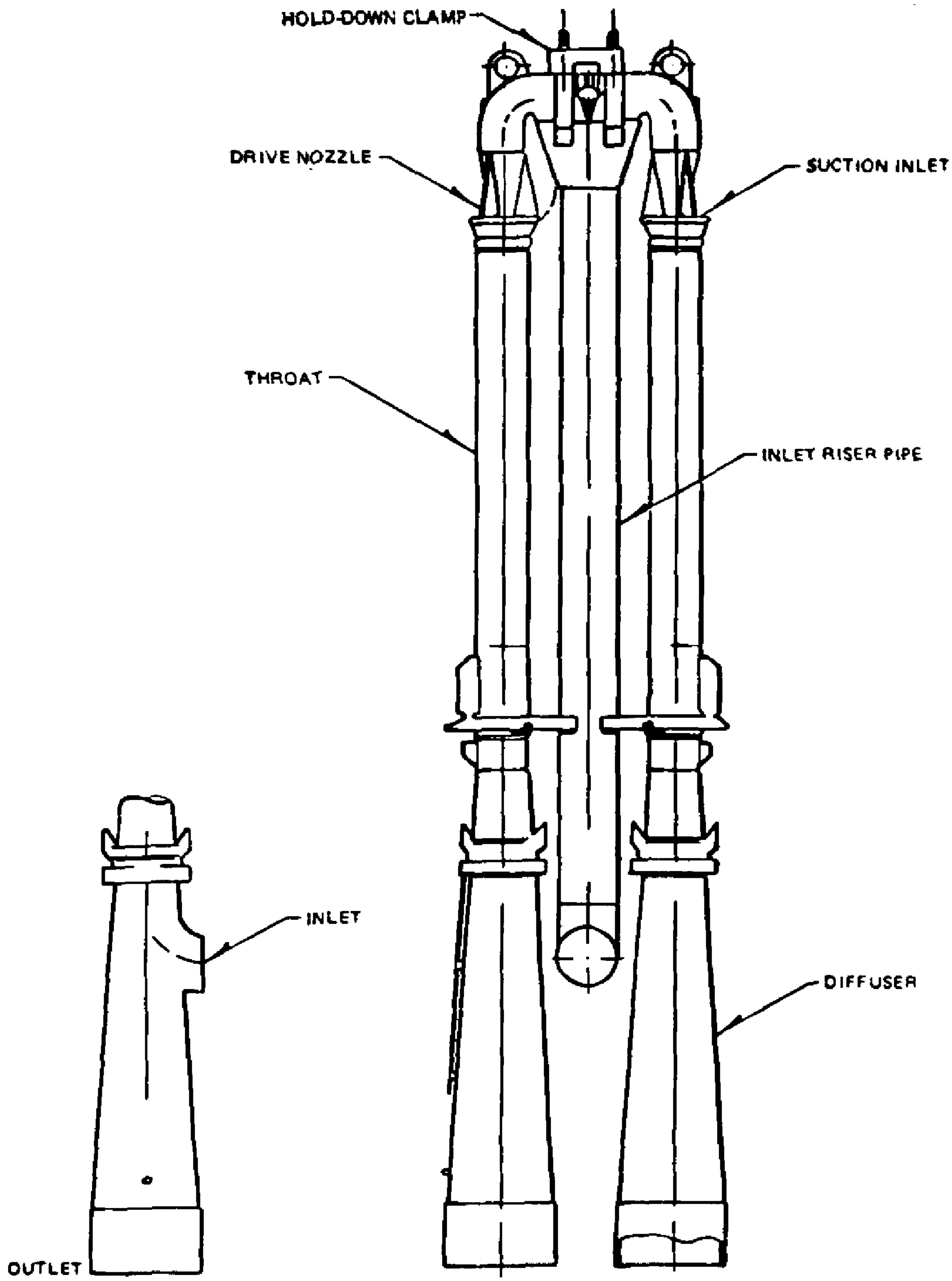
FUEL SUPPORT PIECES

FIGURE 3.9-5, Rev. 48

Auto-Cad Figure Fsar 3_9_5.dwg



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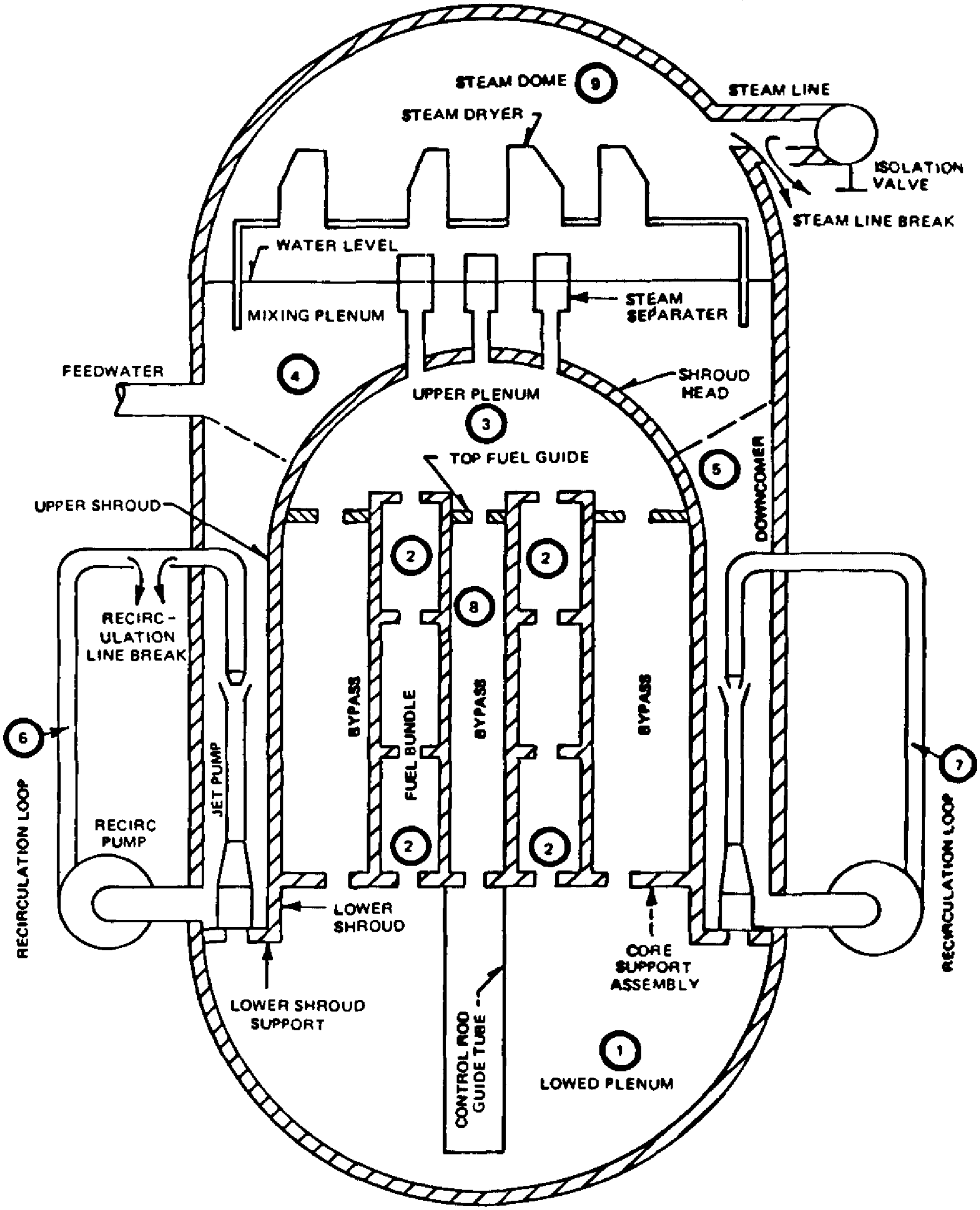
JET PUMP

FIGURE 3.9-6, Rev. 48

Auto-Cad Figure Fsar 3_9_6.dwg



FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
PRESSURE NODES USED FOR DEPRESSURIZATION ANALYSIS FIGURE 3.9-7, Rev. 48

Auto-Cad Figure Fsar 3_9_7.dwg

APPENDIX 3.9A

COMPUTER PROGRAMS

START HISTORICAL

Introduction

The computer programs discussed in this section are those programs used by Bechtel 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.

3.9A.1 ME101

Program Description

ME101 is a finite element computer program which performs linear elastic analysis of piping systems using standard beam theory techniques. The input data format is specifically designed for pipe stress engineering, and the English system of units is used. A thorough checking of the input has been coordinated in the program. In addition, modifications aimed at achieving an improved model are performed automatically.

The output may be used directly for piping design and for conformation to code and other regulatory requirements. Two piping codes, ASME BPV code 1974 and B31.1 Summer 1973 addenda are incorporated in the program to the extent of computing flexibility factors, stress intensification factors and stresses.

ME101 may be used for static and seismic load analysis of piping systems and also performs effective weight calculations.

Static analysis considers one or more of the following: thermal expansion, dead weight, uniformly distributed loads, and externally applied forces, moments, displacements and rotations, or individual force loads.

Seismic analysis is based on standard normal mode techniques and uses response spectrum data. Two methods of eigen value solution are available. Determinant Search or Subspace Iteration considers all data points as mass points. Kinematic Reduction and Householder QR considers masses only at specified data points in designated directions. Differential seismic anchor movement analysis and static seismic analysis are also provided.

ME101 generates isometric plots of the piping configuration with optional node numbering. The plots are obtained by either ZETA or CALCOMP 1036 plotter. The program uses out-of-core solution techniques for both static and dynamic analysis, and

has no practical limitations to the number of equations or band width. However; very large systems may become prohibitive due to cost of computation. The maximum number of mode shapes allowable is currently 125.

Program Version and Computer

The current UNIVAC version (C3) of ME101 is being used by Bechtel Power Corporation.

Extent of Application

ME101 is a piping program developed by Bechtel Power Corporation (BPC). Its development began in July 1975 and is being continuously supported by BPC. It has been used by various projects in the BPC.

Test Problems

The ASME Benchmark Problem No. 1 demonstrates the solution for natural frequencies of a three-dimensional structure as described in Reference 3.9A-1.

The following table lists the natural frequencies from ME101 and Reference 3.9A-1:

Natural Frequency Comparisions, CPS		
Mode No.	*Reference 3.9A-1*	*ME 101*
1	*110*	*112*
2	*117*	*116*
3	*134*	*138*

Additional test problems can be found in Reference 3.9A-2.

3.9A.2 ME632

Program Description

ME632 performs stress analysis of 3-dimensional piping systems. The effects of thermal expansion, uniform load of the pipe, pipe contents and insulation, concentrated loads, movements of the piping system supports, and other external loads, such as wind and snow, may be considered. The input data format is specifically designed for pipe stress engineering, and the English system of units is used. A thorough checking of the input has been coordinated in the program.

The output may be used directly for piping design and for conformation to code and other regulatory requirements. Piping codes, ASME BPV code, B31.1 code and B31.3

code have been incorporated in the program to the extent of computing flexibility factors, stress intensification factors and stresses.

A response spectrum analysis may be performed to analyze the effect of earthquake forces on the piping system, and transient effects of water hammer, steam hammer, or other impulsive type dynamic loading are also handled by the program. Also, a plot of piping geometry and/or response spectrum curves may be obtained to verify the accuracy of the model.

Program Version and Computer

The current UNIVAC version (B9) of ME632 is being used by Bechtel Power Corporation.

Extent of Application

ME632 is a piping program developed by Bechtel Power Corporation (BPC). Its development began in 1970 and is being continuously supported by BPC. It has been used by various projects in the BPC.

Test Problems

The ASME Benchmark Problem No. 1 demonstrates the solution for natural frequencies of a three-dimensional structure as described in Reference 3.9A-1.

The following table lists the natural frequencies for ME632 and Reference 3.9A-1:

Natural Frequency Comparisions, CPS		
Mode No.	Reference 3.9A-1	ME632
1	110	111
2	117	116
3	134	137

Additional test problems can be found in Reference 3.9A-3.

3.9A.3 ME912

Program Description

Finite-difference representation of the heat diffusion equation is used for the pipes or component wall section in contact with fluid of specified temperature and flow rate time histories. The program is quasi-two-dimensional, so that reduction of severity of a given transient with distance from inlet is accounted for.

Thermal properties of water, liquid sodium, stainless and carbon steel are built in the program. Film transfer coefficients for water or liquid sodium are computed by the program for each time step and pipe section. For other fluids such as steam, the program is used on a one-dimensional basis with user supplied film coefficients. Sequential computations are done for pipe lengths of different diameters or wall thicknesses. Fluid outlet temperature data from one pipe length are stored for use as inlet to the next length downstream. Average temperature differences T_a - T_b are thus calculated for structural discontinuity.

Program Version and Computer

The ME912 program has been used by Bechtel Power Corporation in Gaithersburg, and San Francisco offices on various BPC projects. A Univac 1110 computer is used to run the ME912 program.

Extent of Application

The ME912 program was developed from References 3.9A-4, 3.9A-5 and 3.9A-6 by the Stress Group of Gaithersburg and San Francisco offices of BAC. The ME912 program has been extensively used since 1975 for nuclear Class I component design on FFTF project.

Test Problem

For local gradients, the program has been compared with analytical flat plate data of Ref. 3.9A-5 and numerical results by in-house program ME643, Ref. 3.9A-7. The results were acceptable. For axial variations of fluid and wall temperatures, the program agrees closely with the analytical solution of Ref. 3.9A-6. Table 3.9A-2 shows the comparison of ME912 with ME 643 and analytical results.

The ME643 program was developed from References 3.9A-11 and 3.9A-12 by the Stress Group of Los Angeles Power Division of BPC. The results of ME 643 transient temperature responses on both inside and outside surfaces are compared with Chart 36 of Reference 3.9A-13 and plotted Figure 3.9A-1.

3.9A.4 ME 913

Program Description

ME913 can determine stress intensity levels for Class 1 nuclear power piping components for Equations 9 through 14 of subarticle NB-3650, ANALYSIS OF PIPING COMPONENTS of Section III, ASME Boiler and Pressure Vessel Code. Before attempting to exercise this program, the user should be familiar with the requirements and procedures set forth in subarticle NB-3650.

Prior to using this program, the user should have the following information external to the program.

1. *Piping configuration*

2. *Piping and piping component properties*

3. *Moment reactions due to*

    a. *Thermal expansion loads*
    b. *Weight loads and*
    c. *Earthquake loads*

4. *The thermal response of the piping system due to the specified transients:*

    ΔT_1, ΔT_2, *and the* $(T_a - T_b)$ *values for the key points during system life.*

Program Version and Computer

The current ME913 version is being used by Bechtel Power Corporation in its Gaithersburg, Los Angeles, Ann Arbor and San Francisco offices. A Univac 1100 computer is used to run the ME913 program.

Extent of Application

ME913 is the revised and expanded version of the LOTEMP program which was originally developed by the pipe stress group of the San Francisco Power Division of BPC and made available for use through the CDC 6600 computer. The LOTEMP program has been extensively used by the Bechtel Fast Flux Test Facility (FFTF) Systems Analysis Group since 1972 in the preliminary design of FFTF Class I piping. The ME913 program has been used to analyze nuclear Class I piping for Bechtel nuclear power plant projects.

Test Problems

The Grand Gulf Project feedwater line was selected as a test problem. Hand calculations of a selected component in the piping system were performed in accordance with the sample problem (Reference 3.9A-8). Their results were compared with the computer output for code equations 9 through 14 in ME 913 (Reference 3.9A-10).

Table 3.9A-1 shows the comparison between the ASME sample problem (Reference 3.9A-8) and ME913 results (Butt Welding Tee, Location 10).

3.9A.5 References

3.9A-1 *Pressure Vessel and Piping 1972 Computer Programs Verification, the American Society of Mechanical Engineers.*

3.9A-2 *Verification Report on ME101, Linear Elastic Analysis of Piping Systems, Revision 1, February, 1977, Bechtel Power Corporation.*

3.9A-3 *Verification Report on ME632, Seismic Analysis of Piping Systems, October 1977, Bechtel Power Corporation.*

3.9A-4 *Tung, T.K. and Chern,, C.Y., "DELTAT, a Quasi-Two-Dimensional Program for Pipe Thermal Transients," ASME PVP-36, June 1979.*

3.9A-5 *McNeil, D. R. and Brock, J. E., "Charts for Transient Temperatures in Pipes," Heating/Piping/Air Conditioning, Nov. 1979, pp. 107-119.*

3.9A-6 *Carslaw, H.S. and Jaeger, J.C., "Conduction of Heat in Solids," Oxford University Press, 1959, pp. 392-394.*

3.9A-7 *ME643 Thermal and Stress Analyses Program Stress Group, Los Angeles Power Division, October 1977, Bechtel Power Corporation.*

3.9A-8 *"Sample Analysis of a Class I Piping System," prepared by the Working Group on Piping (SGD, ScIII) of the ASME Boiler and Pressure Vessel Code, December 1971.*

3.9A-9 *ASME Boiler and Pressure Vessel Code, Section III, Nuclear Power Plant Components, 1974 Edition.*

3.9A-10 *ME9I3 Verification by Bechtel Power Corporation, February 1975.*

3.9A-11 *Wilson, E. and Nickell, S.R., "Application for the Finite Element Method to Heat Conduction Analysis," Nuclear Engineering and Design 4, 1966.*

3.9A-12 *Wilson, E., "Structural Analysis of Axisymmetric Solids", AIAA Journal, Vol. 3, No. 1 12, December 1965.*

3.9A-13 *Schneider, P. J., "Temperature Response Charts," John Wiley and Sons, Inc., 1963.*

END HISTORICAL

TABLE 3.9A-1

COMPARISON BETWEEN SAMPLE PROBLEM AND COMPUTER PROGRAM ME 913 RESULTS

	ME 913	Sample Problem (Ref. 3.9A-8)
Eq. 9	20,810 psi	20,825 psi
Eq. 10	65,567 psi	65,596 psi
Eq. 11	128,950 psi	128,920 psi
Eq. 12	39,536 psi	39,564 psi
Eq. 13	23,152 psi	23,155 psi
Total Usage Factor	0.3439	0.3699

NOTE:

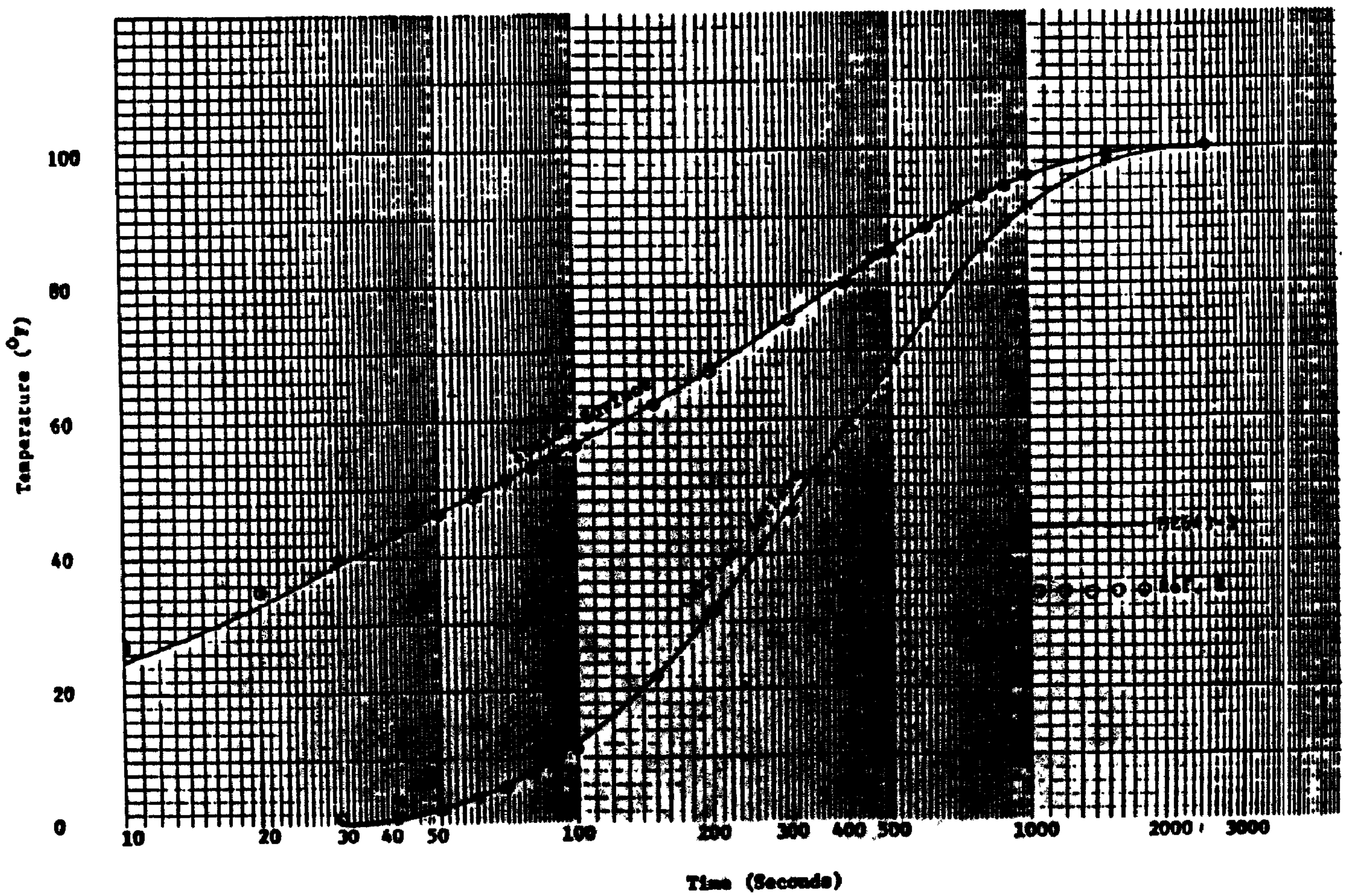
Comparison made for Butt Welding Tee, Location 10.

TABLE 3.9A-2

COMPARISON OF ME 912 WITH ME 643 AND ANALYTICAL RESULTS

Case	Gradients of Program	Pipe		Ta - Tb
		ΔT_1	ΔT_2	
450° to 553°F Step	Me643	79.0	38.0	24.0
3" Sch. 160, Stainless	ME912	79.7	40.6	24.3
Thicknesses 1.50:1	B/M**	82.0	41.0	--
408° to 100°F Step 12" Sch. 80 Carbon Steel Thicknesses 1.69:1	ME643 ME912 B/M**	136.2 134.4 139.0	40.1 41.9 43.0	83.0 81.6 --

** Ref. 3.9A-5



HISTORICAL

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TRANSIENT TEMPERATURE RESPONSE

FIGURE 3.9A-1, Rev. 55

Auto-Cad Figure Fsar 3_9A_1.dwg

3.10 SEISMIC QUALIFICATION* OF SEISMIC CATEGORY I INSTRUMENTATION AND ELECTRICAL EQUIPMENT

The seismic qualification* of Seismic Category I instrumentation and electrical equipment is described in the following subsections:

- 3.10a - NSSS Instrumentation and Electrical Equipment
- 3.10b - Non-NSSS Instrumentation
- 3.10c - Non-NSSS Electrical Equipment

In addition to seismic qualification, all Seismic Category I instrumentation and electrical equipment located in the Containment and the Reactor and Control Buildings are qualified for the combined seismic and hydrodynamic vibratory loadings. Procedures for the assessment and requalification of Seismic Category I instrumentation and electrical equipment for the additional hydrodynamic loads are described in Sections 7.1.6 and 7.1.7 of the Design Assessment Report (DAR).

* The term "Seismic Qualification" in this section is synonymous with "Seismic and Hydrodynamic Qualification."

3.10a SEISMIC QUALIFICATION OF SEISMIC CATEGORY I NSSS INSTRUMENTATION AND ELECTRICAL EQUIPMENT

3.10a.1 SEISMIC QUALIFICATION CRITERIA

3.10a.1.1 Seismic Category I Equipment Identification

Seismic Category I instrumentation and electrical equipment, as well as other equipment, can be found in Table 3.2-1. Pumps and valves which are qualified as seismically "active" are listed in Table 3.9-3.

All NSSS Seismic Category I instrumentation and electrical equipment will be designed to resist and withstand the effects of the postulated earthquakes. Seismic Category I instrumentation and electrical equipment is designed to withstand the effects of the Safe Shutdown Earthquake (SSE) defined in Subsection 3.7a, and to withstand the effects of hydrodynamic loads without functional impairment.

From the basic input ground motion data, a series of response curves at various building elevations are developed after the building layout is completed. This information is included in the purchase specifications for Seismic Category I equipment. Suppliers of equipment such as batteries and racks, instrument racks, control consoles, etc., are required to submit test data, operating experience and/or calculations to substantiate that their components, systems, etc., will not suffer loss of function during or after seismic and hydrodynamic loadings. The magnitude and frequency of the dynamic loadings which each component will experience is determined by its specific location within the plant.

The Class 1E instrumentation and electrical equipment (excluding motors and valve-mounted equipment) supplied by GE requiring seismic qualification are identified in Table 3.10a-1.

3.10a.1.2 Dynamic Design Criteria

3.10a.1.2.1 NSSS Equipment

The seismic criteria used in the design and subsequent qualification of all Class 1E instrumentation and electrical equipment supplied by GE was as follows: The Class 1E equipment shall be capable of performing all safety-related functions during (1) normal plant operation, during (2) anticipated transients, during (3) design basis accidents, and during (4) post-accident operation, while being subjected to, and after the cessation of the accelerations resulting from the OBE and SSE at the point of attachment of the equipment to the building or supporting structure.

The criteria for each of the devices used in the Class 1E systems depend on the use in a given system; for example, a relay in one system may have as its safety function to deenergize and open its contacts within a certain time, while in another system it must energize and close its contacts. Since GE supplies devices for many applications, the approach taken was to test the device in all modes in which it might be used. In this way, the capability of protective action initiation and the proper operation of safety-failure circuits is ensured.

3.10a.2 METHODS AND PROCEDURES FOR QUALIFYING ELECTRICAL EQUIPMENT AND INSTRUMENTATION (EXCLUDING MOTORS AND VALVE MOUNTED EQUIPMENT)

3.10a.2.1 Methods of Showing NSSS Equipment Compliance with IEEE 344-1971

(a) Scope - Compliance not applicable.

(b) Definition - Compliance not applicable.

(c) Procedures - GE supplied Class 1E equipment meets the requirement that the qualification should demonstrate the capability to perform the required function during and after the seismic and hydrodynamic load event. Both analysis and testing were used but most equipment was tested. Analysis was primarily used to determine the adequacy of mechanical strength (mounting bolts, etc.) after operating capability was established by testing.

1. Analysis - GE supplied Class 1E equipment performing primarily a mechanical safety function (pressure boundary devices, etc.) was analyzed since the passive nature of its critical safety role usually made testing impractical. Analytical methods sanctioned by IEEE 344-1971 were used in such cases (see Table 3.10a-1 for indication of which items were qualified by analysis).

2. Testing - GE supplied Class 1E equipment having primarily an active electrical safety function was tested in compliance with IEEE 344-1971, Section 3.2.

(d) Documentation - Available documentation verifies that the seismic qualification of GE supplied Class 1E equipment is in accordance with the requirements of IEEE 344-1971, Section 4.

3.10a.2.2 Testing Procedures for Qualifying Electrical Equipment and Instrumentation (Excluding Motors and Valve-Mounted Equipment)

(The following procedures are not applicable for the Diesel Generator 'E' facility where the seismic qualification conforms to project specification C-1041 or SD-140 and IEEE Standard 344-75.) In addition, replacement equipment may be seismically qualified to a version of the IEEE Standard 344 that is more recent than the 1971 version. Non-GE supplied replacement equipment is qualified to the provisions of FSAR Section 3.10b.

The test procedure required that the devices be mounted on the table of the vibration machine in a manner similar to which it was to be installed. The device was tested in the operating states that it is to be used in performing its Class 1E functions. These states were monitored before, during, and after the test to ensure proper function and absence of spurious function. In the case of a relay, both energized and deenergized states and normally open and normally closed contact configurations were tested if the relay is used in those configurations in its Class 1E functions.

The dynamic excitation was a single frequency continuous test in which the applied vibration was a sinusoidal table motion at a fixed peak acceleration and a discrete frequency at any given time. Each frequency and acceleration combination was maintained for about 30 seconds except when a resonance search was made (see IEEE 344-1971, paragraph 3.2).

The vibratory excitation was applied in three orthogonal axes individually with the axes chosen as those coincident with the most probable mounting configuration.

The first step was to search for resonances in each device. This was done since resonances cause amplification of the input vibration and are the most likely cause of malfunction. The resonance search was usually run at low acceleration levels (0.2G) to avoid destroying the test sample in case a severe resonance was encountered. The resonance search was performed in accordance with IEEE 344 in no less than 7 minutes; if the device was large enough, the vibrations were monitored by accelerometers placed at critical locations. Resonances were determined by comparing the acceleration level with that at the table of the vibration machine. Usually, the devices were either too small for an accelerometer, had their critical parts in an inaccessible location, or had critical parts that would be adversely affected by the mounting of an accelerometer. In these cases, the resonances were detected by visual (strobe light), audible observation, or performance.

Following the frequency scan and resonance determination, the devices were tested to determine their malfunction limit. This test was a necessary adjunct to the assembly test as will be shown later. The malfunction limit test was run at each resonant frequency as determined by the frequency scan. In this test, the acceleration level was gradually increased until either the device malfunctioned or the limit of the device (usually the case) was considered to be rigid (all parts move in unison) and the malfunction limit was therefore independent of frequency.

To achieve maximum acceleration from the vibration machine, rigid devices were malfunction tested at the upper test frequency since that allowed the maximum acceleration to be obtained from deflection- limited machines. The summary of the tests on the devices used in Class 1E applications given in Table 3.10a-1 includes the qualification limit for each device tested.

The above procedures were required of purchased devices as well as those manufactured by GE. Vendor test results were reviewed and if unacceptable, the tests were repeated either by GE or the vendor. If the vendor tests were adequate, the device was considered qualified to the limits of the test.

3.10a.2.3 Qualification of Valve-Mounted Equipment

The piping analyses establishes the response spectra, power spectral density function or time history characteristics, and develops a horizontal and a vertical acceleration for the pipe-mounted equipment. Class 1E motor-operated valves actuators were qualified per IEEE 382-1972, with the exception of DG-E motor-operated valve actuators which were qualified to IEEE 382-1980.

The safety/relief valve, including the electrical components mounted on the valve, are subjected to a dynamic test. This testing is described in Subsections 3.9.2.2a.2.15 and 3.9.3.2a.5.2.

3.10a.2.4 Qualification of NSSS Motors

Seismic qualification of the ECCS motors is discussed in Subsection 3.9.2.2a.2.7, in conjunction with the ECCS pump and motor assembly. Seismic qualification of the Standby Liquid Control (SLC) pump motor is discussed in Subsection 3.9.2.2a.2.10 in conjunction with the SLC pump motor assembly.

3.10a.3 METHODS AND PROCEDURE OF ANALYSIS OR TESTING OF SUPPORTS OF ELECTRICAL EQUIPMENT AND INSTRUMENTATION

3.10a.3.1 Dynamic Analysis Testing Procedures and Restraint Measures

3.10a.3.1.1 NSSS Equipment (Other Than Motors and Valve Mounted Equipment)

The Class 1E equipment supplied by GE is used in many systems on many different plants under widely varying seismic requirements. The dynamic qualification was performed in accordance with IEEE-344.

Some GE supplied Class 1E devices were qualified by analysis only (as noted in Table 3.10a-2). One of the analysis methods is shown in Subsection 3.10a.5. Analysis was used for passive mechanical devices and was sometimes used in combination with testing for larger assemblies containing Class 1E devices. For instance, a test might have been run to determine if there were natural frequencies in the equipment within the critical seismic frequency range (see IEEE 344-1971, paragraph 3.2). If the equipment was determined to be free of natural frequencies, then it was assumed to be rigid and a static analysis was performed (see IEEE 344-1971, paragraph 3.2). If it had natural frequencies in the critical frequency range, then calculations of transmissibility and responses to varying input accelerations were determined to see if Class 1E devices mounted in the assembly would operate without malfunctioning. In general, the testing of Class 1E equipment was accomplished using the following procedure.

Assemblies (i.e., control panels) containing devices which have had dynamic load malfunction-limits established were tested by mounting the assembly on the table of a vibration machine in the manner in which it was to be mounted in use. It was vibration tested by running a low level resonance search. As with the devices, the assemblies were tested in the three major orthogonal axes. The resonance search was run in the same manner as described for devices. If resonances were present, the transmissibility between the input and the location of each Class 1E device was determined by measuring the accelerations at each device location and calculating the magnification between it and the input. Once known, the transmissibilities could be used analytically to determine the response at any Class 1E device location for any given input. (It was assumed that the transmissibilities were linear as a function of acceleration even though they actually decrease as acceleration is increased-therefore a conservative assumption.).

Since control panels and racks constitute the majority of Class 1E electric assemblies supplied by GE, seismic qualification testing of these will be discussed in more detail. There are basically four generic panel types. One or more of each type was tested using the above procedures.

Figures 3.10a-1, 3.10a-2, 3.10a-3 and 3.10a-4 illustrate the four basic panel types referenced above and show typical accelerometer locations. The status of the dynamic tests on the Class 1E panels supplied by GE is summarized in Table 3.10a-2.

The full acceleration level tests described above disclosed that most of the panel types had more than adequate mechanical strength. A given panel design acceptability was shown to be only a function of its amplification factor and the malfunction levels of the devices mounted in it. Subsequent panels were, therefore, tested at lower acceleration levels and the transmissibilities measured to the various devices as described above. By dividing the devices' malfunction levels by the panel transmissibility between the device and the panel input, the panel seismic qualification level could be determined. Several high level tests have been run on selected generic panel designs to ensure the conservatism in using the transmissibility analysis described.

3.10a.4 OPERATING LICENSE REVIEW

3.10a.4.1 NSSS Control and Electrical Equipment (Other Than Motors and Valve Mounted Equipment)

The dynamic test results for safety-related panels and control equipment within the NSSS scope are maintained in a permanent file by GE and can be readily audited in all cases. The equipment used in Class 1E applications passed the prescribed tests. Where equipment failed to pass the tests, it was rejected. In some cases, equipment which failed one test was modified to meet the performance requirements and retested. If the retested equipment passed the latter test, it could be used in a Class 1E application.

Table 3.10a-1 lists the NSSS control devices by item number and vendor. Also, a summary of the test conditions for the devices used in Class 1E applications is given in Table 3.10a-2.

The acceleration level shown in the right hand columns of Table 3.10a-1 is the acceleration at which either the device malfunctioned or the limit of the vibration machine was reached.

3.10a.4.2 NSSS Motors

Seismic qualification test results for the ECCS motors are discussed in Subsection 3.9.2.2a.2.7 in conjunction with the ECCS pump and motor assembly. Seismic qualification test results for the Standby Liquid Control (SLC) motor are discussed in Subsection 3.9.2.2a.2.10 in conjunction with the SLC pump motor assembly.

3.10a.4.3 Valve-Mounted Equipment

The safety relief valves (including the electrical components mounted on the valve) are subjected to dynamic tests. The results of these tests are discussed in Subsections 3.9.2.2a.2.15 and 3.9.3.2a.5.2.

3.10a.5 Dynamic Analysis By Response Spectrum Method

The system stiffness and mass matrices are generated using standard techniques. A dynamic analysis is performed using the following equations of motion and procedure to uncouple these equations

The equations of motion in matrix form are as follows:

$$M(\ddot{X} + \ddot{Y}) + C\dot{X} + KX = 0 \qquad \text{(Eq. 3.10a-1)}$$

where

M = mass matrix, nxn (this includes the hydrodynamic mass)
X = column vector of displacement relative to ground* (nxl)
C = damping matrix (nxn)
K = stiffness matrix (nxn)
$\ddot{Y}$ = column vector of ground accelerations (nxl)
. = first derivative with respect to time
.. = second derivative with respect to time

It should be noted that for equipment containing fluid, a hydrodynamic mass coupling exists between real structural masses. This hydrodynamic mass appears as diagonal and off-diagonal terms in the mass matrix. The overall system stiffness matrix K is determined by either the matrix force method or the matrix displacement method. The resulting stiffness matrix is similar.

Removing the driving-point acceleration vector to the right side of Eq. 3.10a-1, the equation reduces to the classical form:

$$M\ddot{X} + C\dot{X} + KX = M\ddot{Y} \qquad \text{(Eq. 3.10a-2)}$$

In order to decouple Eq. 3.10a-2, we set:

$$X = \phi q \qquad \text{(Eq. 3.10a-3)}$$

Eq. 3.10a-2 then becomes

$$M\phi q + C\phi q + K\phi q = -M\ddot{Y} \qquad \text{(Eq. 3.10a-4)}$$

Pre-multiplying by ϕT, the transpose of ϕ, and performing the coordinate transformation described in Eq. 3.10a-4 such that is defined by the following orthogonality conditions:

$$\phi\, TM\, \phi = I \qquad \text{(Eq. 3.10a-5)}$$

$$\phi\, TK\, \phi = w^2 \qquad \text{(Eq. 3.10a-6)}$$

where I is an identifying matrix (Nxn) and w^2 is a diagonal matrix of the eigenvalues. Then Eq. 3.10a-4 becomes

$$\phi^T M\phi\ddot{q} + \phi^T C\phi\dot{q} + \phi^T K\phi q = \phi^T M\ddot{Y} \quad \text{(Eq. 3.10a-7)}$$

$$\ddot{q} + \phi^T O\phi\dot{q} = w^2\ q = \phi^T M\ddot{Y} \quad \text{(Eq. 3.10a-8)}$$

The above procedure for decoupling the equation of motion by using the modal matrix of the undamped system assumes that damping in the system is small. It will further be assumed that the damping matrix C is such that $\phi^T C\phi$ is a diagonal matrix. The elements of this diagonal-matrix are the modal damping values.

With the above assumptions, Eq. 3.10a-8 may be written in the following uncoupled form:

$$\ddot{q}_i + \ + 2\beta_i w_i \dot{q}_i + w_i^2 q_i = S\ U \quad \text{(Eq. 3.10a-9)}$$

i = 1, 2, --- n

where

$$x_i = \begin{matrix} \chi^1 i \\ \chi^2 i \\ \cdot \\ \cdot \\ \cdot \\ \cdot \\ \chi^n i \end{matrix} \qquad \phi_i = \begin{matrix} \phi 1_i \\ \phi 2_i \\ \cdot \\ \cdot \\ \cdot \\ \cdot \\ \phi n_i \end{matrix}$$

The maximum physical displacement for each mass is then taken to the square root of the sums of the squares of each of the maximum displacement responses for each mode, i.e.,

where:

$$X\,max. = \left[\sum_{j=1}^{n} x_{ij}^2\right]^{1/2}$$

(X) maximum. is the column vector of maximum displacements. Similarly, the maximum load response for the i mode is found from

$$L_{ji} = \beta_j X_i$$

$$L_{ji} = \begin{matrix} L_{1i} \\ L_{2i} \\ \cdot \\ \cdot \\ \cdot \\ L_{mi} \end{matrix}$$

where

β_j is the stress matrix for element j, j=1, ...m

m = total number of elements.

where

β_i = damping ratio for the i^{th} mode expressed as percent of critical damping

w_i = i^{th} natural angular frequency of the system

S_i = modal participation factor the i^{th} mode = ϕ_i^t MD

U_g = ground or floor acceleration time history

ϕ_i^t = transpose of the i^{th} mode shape

D = earthquake direction vector

The response is calculated using the response spectra specified for the location of the input to the analytical model. The analytical procedure is described briefly in the following paragraphs.

The system of one degree-of-freedom equations represented by Eqs. 3.10a-8 or 3.10a-9 can be solved by the response spectrum method. With this method, the maximum modal response for each natural frequency of interest is found from the applicable response spectra. Response spectrum curves are essentially plots of the maximum responses of single degrees-of-freedom systems described by Eq. 3.10a-9 with S = 1.0 as a function of their natural frequencies.

Having found the maximum modal displacements q, i = l...m, the maximum physical displacement for the i[th] mode is given by:

$$X_i = \phi_i S_i q_i$$

The maximum load response is taken to be the square root of the sums of the squares of each of the maximum responses for each mode, i.e.,

$$L_j \max. = \left[\sum_{i=1}^{n} L_{ji}^2 \right]^{1/2} \quad : j = 1, 2, \ldots m$$

where (L) max is the column vector of maximum loads

The accelerations for each mode are determined by multiplying the displacements vector for that mode (X_i) by the natural frequency of (w_{2_i}) that mode.

$$A_i = X_i w_{2_i}$$

The maximum accelerations are then determined by

$$A \max. = \left[\sum_{i=1}^{n} A_i^2 \right]^{1/2}$$

TABLE - 3.10a-1

ESSENTIAL ELECTRICAL COMPONENTS AND INSTRUMENTS

-------------------- DESCRIPTION --------------------					-------------------- SEISMIC AND ENVIRONMENTAL QUALIFICATIONS --------------------					
ITEM NO.	NAME	VENDOR	QUANTITY	ENVIRONMENT(1)	OTHERS OF SAME TYPE IN SAME ENVIRONMENT			SEISMIC QUALIFICATION(4)		
								X	Y	Z
SYSTEM TITLE - REACTOR										
B11-D193	Power Range Detector	GE	43	In Vessel						
SYSTEM TITLE – NUCLEAR BOILER										
B21-N002	Pressure Switch		1	Area II						
B21-N004	Temp Element	PYCO	16	Area II.7				Note 2		
B21-N006	Diff Press Switch	BARTON	18	Area II	N007	N008	N009,	5	10	10
					N021	A,B,C,D				
B21-N010	Temp Element	CALIF. ALLOY	13	Area II	N014,	N016,	N017	5	5	5
B21-N015	Press Switch	BARKSDALE	4	Area III.3				5	10	10
B21-N020	Press Switch	BARKSDALE	34	Area II.1			N023,	5	15	15
					N039	N044,	N022			
B21-N024	Level Ind Switch	BARTON	10	Area II.1	N031,	N042		15	15	15
B21-N025	Level Ind Switch	YARWAY	4	Area II.1				1.5	1.5	1.5
B21-N026	Level Ind Trans Switch	BARTON	6	Area II.1	N037			5	5	5
B21-N027	Level Trans	ROSEMOUNT	25	Area II.1	N033,	N034		Note 2		
B21-N043	Press Trans	ROSEMOUNT	1	Area II.1				3	3	3
B21-N055	Press Trans	ROSEMOUNT	2	Area II.1				3	3	3
B21-N056 A&C	Vacuum Switch (Unit 1)	STATIC-O-RING	2	Area III.5						
B21-N056 B&D	Vacuum Switch (Unit 1)	BARKSDALE	2	Area III.5						
B21-N056B	Vacuum Switch (Unit 2)	STATIC-O-RING	1	Area III.5						
B21-N056 A,C, & D	Vacuum Switch (Unit 2)	BARKSDALE	3	Area III.5						

TABLE - 3.10a-1

ESSENTIAL ELECTRICAL COMPONENTS AND INSTRUMENTS

ITEM NO.	NAME	VENDOR	QUANTITY	ENVIRONMENT(1)	OTHERS OF SAME TYPE IN SAME ENVIRONMENT			SEISMIC QUALIFICATION(4) X	Y	Z
B21-N064	Temp Element	CALIF. ALLOY	1	Area II				2	2	2
B21-N600	Temp Switch	RILEY	8	Area V	N603			Note 2		
B21-R004	Press Indicator	ROBERTSHAW	2	Area II.1				4	4	4
B21-R005	Diff Press Ind	BARTON	1	Area II.1				Note 2		
B31-N014	Flow Trans	ROSEMOUNT	8	Area II.1	N024			2	2	2
B31-N015	Diff Press Trans	ROSEMOUNT	1	Area II.1				Note 2		
B31-N016	Diff Press Switch	BARTON	13	Area II.1	N018A,	N019 thru		5	10	10
					N022					
B31-N018B	Press Switch	STATIC-O-RING	1	Area II.1				15	15	15
B31-N023	Temp Element	ROSEMOUNT	2	Area 1.4				Note 2		
B31-N035	Temp Element		2					Note 2		
SYSTEM TITLE – CRD HYDRAULIC CONTROL										
C12-N013	Level Switch	MAGNETROL	5	Area II.1				4.1	5	9.5
SYSTEM TITLE – FEEDWATER CONTROL										
C32-N003	Transmitter (Diff Press)	ROSEMOUNT	6	Area II.1	N004			Note 2		
C32-N005	Transmitter (Pressure)	ROSEMOUNT	2	Area II.1	N008			Note 2		
C32-N017	Diff Press Trans	STATHOM	2	Area II.1				Note 2		
SYSTEM TITLE – STAND BY LIQUID										
C41-N003	Temp Switch	CALIF. ALLOY	1	Area II.8				Note 2		
C41-N004	Press Trans	ROSEMOUNT	1	Area II.8				Note 2		
C41-N006	Temp Element		1	Area II.8				Note 2		
C41-R003	Press Indicator	ROBERTSHAW	1	Area II.8				Note 2		
SYSTEM TITLE – NEUTRON MONITORING										

TABLE - 3.10a-1

ESSENTIAL ELECTRICAL COMPONENTS AND INSTRUMENTS

-------------------- DESCRIPTION -------------------- -------------------- SEISMIC AND ENVIRONMENTAL QUALIFICATIONS --------------------

ITEM NO.	NAME	VENDOR	QUANTITY	ENVIRONMENT[1]	OTHERS OF SAME TYPE IN SAME ENVIRONMENT			SEISMIC QUALIFICATION[4]		
								X	Y	Z
C51-J004	Valve, Guide Tube	GE	5	Area II.10						
C51-J008	Guidetubes	GE	1	Area II.10				Note 2		
C51-K002	Volt Preamplifier	GE	8	Area V				4	5	5
C51-K601	Intermediate Range Mon	GE	8	Area V				6	6	6
C51-K605	Pwr Rnge Instr	GE	1	Area V						
C51-N002	Detector	GE	8	Area I.3						
SYSTEM TITLE – REACTOR PROTECTION										
C72-N002	Prim Cont Press Switch	STATIC-O-RING	4	Area II.1				15	15	15
C72-N003	Turbine 1st Stage Pr SW	BARKSDALE	4	Area III.3				15	15	15
C72-N005	Turbine EMC Press SW	BARKSDALE	4	Area III.3						
C72-N006	Turb Stop Vlv POS SW	ACME CLEVELAND	4	Area III.3						
C72-N008	Turbine Bypass Vlv POS SW	ACME CLEVELAND	4	Area III.3						
C72-S003(A-H)	Elec. Prot. Assy.	GE	8	Not Required				Later		
SYSTEM TITLE – PROCESS RADIATION MONITORING										
D12-K603	Rad Mon & Ind (Mn St Ln)	GE	4	Area V						
D12-K609	Ind & Trip Unit	GE	12	Area V	K615,	K616	K617	3	3	3
					K618					
D12-N006	Detector (Mn St Ln)	GE	4	Area II.7						
D12-N015	Detector	GE	8	Area II.9	N016	N017,	N018	15	15	15
SYSTEM TITLE – RESIDUAL HEAT REMOVAL										

TABLE - 3.10a-1

ESSENTIAL ELECTRICAL COMPONENTS AND INSTRUMENTS

-------------------- DESCRIPTION --------------------					-------------------- SEISMIC AND ENVIRONMENTAL QUALIFICATIONS --------------------					
ITEM NO.	NAME	VENDOR	QUANTITY	ENVIRONMENT(1)	OTHERS OF SAME TYPE IN SAME ENVIRONMENT			SEISMIC QUALIFICATION(4)		
								X	Y	Z
E11-N001	Cond Element	BALSBAUGH	2	Area II.5				Note 2		
E11-N007	Diff Press Trans	ROSEMOUNT	5	Area II.5	N013,	N015		Note 2		
E11-N008	Diff Press Trans	BARTON	2	Area II.5				Note 2		
E11-N009	Temp Element	CALIF ALLOY	12	Area II.5	N029,	N030		2	2	2
E11-N010	Press Switch	STATIC-O-RING	8	Area II.5	N011			15	15	15
E11-N016	Press Switch	STATIC-O-RING	9	Area II.5	N018,	N020		15	15	15
E11-N019	Diff Press Switch	BARTON	2	Area II.5				5	10	10
E11-N021	Diff Press Ind Switch	BARTON	2	Area II.5				15	15	15
E11-N022	Press Switch	BARKSDALE	2	Area II.5				Note 3		
E11-N023	Level Switch	MAGNETROL	3	Area II.5		N024		Note 3		
E11-N026	Press Trans	ROSEMOUNT	3	Area II.5		N028		Note 3		
E11-N033	Flow Switch	FISHER & PORTER	2	Area II.5				Note 3		
E11-N600	Temp Switch		8	Area V	N601			4.5	4.5	4
E11-R002	Press Indicator	ROBERTSHAW OR CONTROL SPECIALTIES	8	Area II.5	R003			Note 3		
SYSTEM TITLE – CORE SPRAY										
E21-N001	Press Trans	ROSEMOUNT	2	Area II.5				Note 2		
E21-N003	Diff Press	ROSEMOUNT	2	Area II.5				Note 2		
E21-N004	Diff Press	BARTON	2	Area II.5				5	10	10
E21-N006	Flow Switch		2	Area II.5						
E21-N007	Press Switch		2	Area II.5				Note 2		
E21-N008	Press Switch		4	Area II.5						

TABLE - 3.10a-1

ESSENTIAL ELECTRICAL COMPONENTS AND INSTRUMENTS

DESCRIPTION					SEISMIC AND ENVIRONMENTAL QUALIFICATIONS					
ITEM NO.	NAME	VENDOR	QUANTITY	ENVIRONMENT[1]	OTHERS OF SAME TYPE IN SAME ENVIRONMENT			SEISMIC QUALIFICATION[4]		
								X	Y	Z
E21-R001	Pressure Indicator	ROBERTSHAW OR CONTROL SPECIALTIES	2	Area II.5				Note 2		
SYSTEM TITLE – MSIV LEAKAGE CONTROL										
E32-K601	Power Supply	GE	2	Area V				2.5	2.5	2.5
E32-N006	Flow Element	S & K INSTRUMENTS	4	Area II						
E32-N050	Press Trans	ROSEMOUNT	8	Area II	N055,	N058,	N060	3	3	3
					N061					
E32-N051	Press Trans	ROSEMOUNT	5	Area II	N056			3	3	3
E32-N053	Flow Trans	S & K INSTRUMENTS	4	Area II						
E32-N054	Diff Press Trans	ROSEMOUNT	2	Area II	N059			3	3	3
E32-N600	Timer	EAGLE SIGNAL	13	Area V	N601,	N602,	N604	2.5	2.5	2.5
E32-N650	Alarm	BAILEY METER	19	Area V	N651,	N653,	N654,	9	9.5	13
					N655,	N656,	N658,			
					N659,	N660,	N661			
E32-R601	MV/I	BAILEY METER	4	Area V				Note 2		
E32-R651	Meter	GE	18	Area V	R653 thru	R656,		Note 2		
					R658 thru	R661				
SYSTEM TITLE – HIGH PRESSURE COOLANT INJECTION										
E41-K600	Power Supply	GE	1	Area V				2.5	2.5	2.5
E41-K601	SO Root Converter	BAILEY METER	1	Area V				9	9	13

TABLE - 3.10a-1

ESSENTIAL ELECTRICAL COMPONENTS AND INSTRUMENTS

---- DESCRIPTION ----				---- SEISMIC AND ENVIRONMENTAL QUALIFICATIONS ----						
ITEM NO.	NAME	VENDOR	QUANTITY	ENVIRONMENT(1)	OTHERS OF SAME TYPE IN SAME ENVIRONMENT			SEISMIC QUALIFICATION(4)		
								X	Y	Z
E41-K603	Inverter	TOPAZ	1	Area V				5	10	8.5
E41-N001	Press Switch	BARKSDALE	6	Area II	N027	N031		29	29	29
E41-N002	Level Switch	MAGNETROL	5	Area II.4	N003,	N015,	N018	1.2	6	9.5
E41-N604	Diff Press Switch	BARTON	2	Area II	N005			5	10	10
E41-N005	Diff Press Switch	BARTON	1	Area II				Note 3		
E41-N008	Diff Press Trans	ROSEMOUNT	1	Area II				3	3	3
E41-N009	Press Trans	ROSEMOUNT	4	Area II	N013,	N016,	N019	Note 2		
E41-N010	Press Switch	STATIC-O-RING	7	Area II	N012,	N017		15	15	15
E41-N014	Level Switch	MAGNETROL	1	Area II				Note 2		
E41-N024	Temp Element	CALIF. ALLOY	16	Area II	N025,	N028,	N029,	2	2	2
					N030					
E41-N600	Temp Switch	GE	6	Area V	N601,	N602				
E41-R601	Press Indicator	ROBERTSHAW	4	Area II	R003,	R004,	R005	Note 2		
E41-R002	Temp Indicator	MOELLER	1	Area II				Note 2		
E41-R600	Controller	BAILEY METER	1	Area V				9	9	8
SYSTEM TITLE – REACTOR CORE ISOLATION COOLING										
E51-K603	Inverter(DC to AC)	TOPAZ	1	Area V				5	10	8.5
E51-N602	Timer		4	Area V	N603					
E51-N003	Diff Press Switch	BARTON	1	Area II.4				15	15	15
E51-N003	Diff Press Transmitter	ROSEMOUNT	1	Area II.4				3	3	3
E51-N004	Press Transmitter	ROSEMOUNT	4	Area II.4	N005,	N007,	N008	Note 2		
E51-N006	Press Switch	STATIC-O-RING	5	Area II.4	N019			15	15	15
E51-N009	Press Switch	BARKSDALE	8	Area II.4	N012,	N020,	N030	29	29	29

TABLE - 3.10a-1

ESSENTIAL ELECTRICAL COMPONENTS AND INSTRUMENTS

ITEM NO.	NAME	VENDOR	QUANTITY	ENVIRONMENT(1)	OTHERS OF SAME TYPE IN SAME ENVIRONMENT			SEISMIC QUALIFICATION(4) X	Y	Z
E51-N010	Level Switch	MAGNETROL	1	Area II.4				Note 2		
E51-N011	Temp Element		20	Area II	N021,	N022,	N023			
					N025,	N026,	N027			
E51-N017	Diff Press Switch	BARTON	2	Area II	N018			5	10	10
E51-N600	Temp Switch	GE	14	Area V	N601 thru	N604				8
E51-R001	Press Indicator	ROBERTSHAW	4	Area II.4				Note 2		
E51-R005	Temp Indicator	MOELLER	1	Area II.4				5	5	5
E51-R600	Flow Indicator Cont	BAILEY METER	1	Area V				9	9	8
SYSTEM TITLE – REACTOR WATER CLEANUP										
G33-K600	Power Supply	GE	1	Area V				2.5	2.3	2.5
G33-K602	SQ Root Conv	GE	3	Area V	K603,	K605				
G33-K604	Summer	BAILEY METER	1	Area V				4	9	13
G33-N011	Flow Element	VICKERY SIMS	2	Area II.2.b	N035,	N040		Note 2		
G33-N012	Diff Press Trans	ROSEMOUNT	3	Area II	N036,	N041		3	3	3
G33-N016	Temp Element	CALIF. ALLOY	18	Area II	N022,	N023		2	2	2
G33-N044	Diff Press Switch	BARTON	2	Area II				15	15	15
G33-N600	Temp Switch	GE	12	Area V	N602					
G33-N603	Alarm	BAILEY METER	12	Area V				9	9.5	13

NOTES:

TABLE - 3.10a-1

ESSENTIAL ELECTRICAL COMPONENTS AND INSTRUMENTS

-------------------- DESCRIPTION --------------------				-------------------- SEISMIC AND ENVIRONMENTAL QUALIFICATIONS --------------------				
ITEM NO.	NAME	VENDOR	QUANTITY	ENVIRONMENT(1)	OTHERS OF SAME TYPE IN SAME ENVIRONMENT	SEISMIC QUALIFICATION(4)		
						X	Y	Z

1. Refer to Tables 3.11-1, 3.11-2 and 3.11-3.
2. Classified as Pressure Integrity Instrument; Seismic qualification not required.
3. Hydrostatic test only required for qualification.
4. This table is based on the original seismic qualification effort performed by GE. Replacement equipment may be seismically qualified to a version of the IEEE Standard 344 that is more recent than the 1971 version. Non-GE supplied replacement equipment is qualified to the revisions of FSAR Section 3.10b.

TABLE 3.10a-2

SEISMIC QUALIFICATION TEST SUMMARY
CLASS 1E CONTROL PANELS AND LOCAL PANELS & RACKS

PANEL	DESCRIPTION	TYPE	CLASS 1E EQUIPMENT DESCRIPTION	COMMENTS
H12-P601	Reactor Core Cooling System	Benchboard	SBM & CR 2940 switches GEMAC instruments	Too long for test table – not tested [1] qualified by analysis
H12-P680	Unit Operating Bd. Reactor Water Cleanup & Recirculation	Benchboard	SBM & CR 2940 switches BEMAC instruments	Seismic test on similar type panel[2]
H12-P680	Reactor Control	Benchboard	Mode switch, range switches	Seismic test completed[3]
H12-P606	Radiation Monitor	2 Bay instrument rack	Startup neutron monitoring electronics	Seismic test completed
H12-P609	Reactor Protection System Division 1 & 2 Logic	Vertical board	HFA & HMA Relays, CR 105 contactor	Identical to U13-P611 panel tested[4]
H12-P611	Reactor Protection System Division 3 & 4 Logic	Vertical board	HFA & HMA Relays, CR 105 contactor	Seismic test completed
H12-P612	FW & Recirc Instruments	2 Bay instrument rack	GEMAC Instruments	Seismic test completed
H12-P613	NSSS Process Instruments	2 Bay instrument rack	GEMAC Instruments	Seismic test completed
H12-P618	Division 2 RHR/RCIC Relay	Vertical board	HFA & HMA Relays	Seismic test on similar type panel
H12-P621	Reactor Core Isolation Cooling Relays	Vertical board	HFA & HMA Relays	Seismic test on similar type panel
H12-P622	Inboard Isolation Valve Relays	Vertical board	HFA & HMA Relays	Seismic test on similar type panel
H12-P623	Outboard Isolation Valve Relays VB	Vertical board	HFA & HMA Relays	Seismic test on similar type panel
H12-P628	ADS Channel A Relay VB	Vertical board	HFA & HMA Relays	Seismic test on similar type panel
H22-P001	CS System Loc. Pnl. A	Local rack	Pressure Switch	Seismic test completed
H12-P631	ADS Channel B Relay VB	Vertical board	HFA & HMA Relays	Seismic test on similar type panel

TABLE 3.10a-2

SEISMIC QUALIFICATION TEST SUMMARY
CLASS 1E CONTROL PANELS AND LOCAL PANELS & RACKS

PANEL	DESCRIPTION	TYPE	CLASS 1E EQUIPMENT DESCRIPTION	COMMENTS
			Mon., Timers	
H12-P633	Radiation Monitor Instrument Panel B	2 Bay instrument rack	Startup Neutron Monitoring Electronics	Identical to H13-P606; Panel tested
H22-P002	Reactor Water Cleanup	Local rack	Pressure transmitters	Seismic test completed
H22-P004	Reactor Vessel Level & Pressure – A	Local rack	Pressure switches, level indicator/transmitter	Seismic test on similar type panel
H22-P005	Reactor Vessel Level & Pressure – B	Local rack	Pressure switches, level indicator/transmitter	Seismic test on similar type panel
H22-P006	Recirc A/Main Steam Flow A	Local rack	Pressure transmitter	Seismic test on similar type panel
H22-P009	Jet Pump Division 1	Local rack	Pressure transmitter	Seismic test completed
H22-P010	Jet Pump Division 2	Local rack	Pressure transmitter	Identical to H22-P009 panel tested
H22-P015	Main Steam Flow A	Local rack	Pressure switch	Identical to H22-P025 panel tested
H22-P017	RCIC Division 1	Local rack	Pressure transmitter/switches	Seismic test on similar type panel
H22-P018	RHR Panel A	Local rack	Pressure switches	Seismic test on similar type panel
H22-P021	RHR Panel B	Local rack	Pressure transmitter/switches	Seismic test on similar type panel
H22-P025	Main Steam Flow D	Local rack	Pressure switches	Seismic test completed
H22-P030	SRM & IRM Preamp A-D	NEMA 12 – Enclosures	SRM-IRM Preamplifiers	Seismic test completed
H22-P031	SRM & IRM Preamp A-D	NEMA 12 – Enclosures	SRM-IRM Preamplifiers	Identical to H22-P030 enclosure tested
H22-P032	SRM & IRM Preamp A-D	NEMA 12 – Enclosures	SRM-IRM Preamplifiers	Identical to H22-P039 enclosure tested

TABLE 3.10a-2

SEISMIC QUALIFICATION TEST SUMMARY
CLASS 1E CONTROL PANELS AND LOCAL PANELS & RACKS

PANEL	DESCRIPTION	TYPE	CLASS 1E EQUIPMENT DESCRIPTION	COMMENTS
H22-P033	SRM & IRM Preamp A-D	NEMA 12 – Enclosures	SRM-IRM Preamplifiers	Identical to H22-P030 enclosure tested
H12-P700	Termination Cabinet	4 Bay Cabinet	Cables	
H12-P701	Termination Cabinet	4 Bay Cabinet	Cables	
H12-P702	Termination Cabinet	4 Bay Cabinet	Cables	
H12-P703	Termination Cabinet	4 Bay Cabinet	Cables	
H12-P704	Termination Cabinet	4 Bay Cabinet	Cables	
H12-P705	Termination Cabinet	4 Bay Cabinet	Cables	
H12-P706	Termination Cabinet	4 Bay Cabinet	Cables	
H12-P732	Termination Cabinet	4 Bay Cabinet	Cables	

FOOTNOTES:

Seismic tests on essential C&I panels fall into the following categories:

1. Panels not tested — Due to size limitations, qualification completed by analysis.
2. Tests on similar panels — When panel size and configuration are very similar to but not necessarily identical, test results for a similar panel are used.
3. Seismic test completed — Tests run on essentially identical panels but possibly build for a different plant.
4. Tests on identical panels — When two panels are exact duplicates of one another, tests are run on only one panel (e.g., H12-P609 and H12-P611 are identical – only H12-P611 was tested).

TABLE 3.10a-3

SUMMARY OF SAMPLE SEISMIC STATIC ANALYSIS
FOR THREE TYPICAL CABINETS

Panel Description	Center of Gravity (in)	Number of Studs	Panel Forces (lb)/Ft of Panel FR to BK (F-B)	Panel Forces Side to Side (S-S)	Panel Forces Up	Panel Forces Down	Axial Load/Stud Tension (lb)	Axial Load/Stud Comp (lb)	Shear Load/ Stud (lb)	Combined Stress Shear (psi)	Combined Stress Normal (psi)	Margin of Safety* Yield	Stud Tensile Load (lb)**
NSSS Cabinet H-12-P608 Power Range Monitor	45	40	736	736	1656	2576	1748	174.8	349.6	6633	12,793	Tensile 0.95 Shear 1.07	1815
PGCC Computer Cabinet	40.5	4	561	561	1262	1963	1797	88	380	6878	13,206	Tensile 0.89 Shear 1.0	1874
Electro-Hydraulic Cabinet H-12-P863	43	24	637	637	1432	2228	2111	321	398	7953	15,398	Tensile 0.63 Shear 0.73	2185

* - A value for the margin of safety which is greater than zero (> 0) represents an adequate installation.

** - A value for the stud tensile load which is less than 4800 lbs. (< 4800) represents an adequate installation.

TABLE 3.10a-4

SEISMIC DESIGN VERIFICATION DATA SHEET

Cabinet Name: Area Radiation Monitor, H12-P605

Applied Horizontal Acceleration	1.6 G
Applied Vertical Acceleration	4.6 G
Tension Stress (Maximum-Yield)	25,000 PSI
Shear Stress (Maximum-Yield)	13,750 PSI
Weight of Cabinet (Approx.)	720 LBS
Number of Mounting Bolts	4
Height of Center of Gravity	45 Inches
Combined Stress (Tensile)	10,592 PSI
Combined Stress (Shear)	5,490 PSI
Margin of Safety (Tensile)	1.36
Margin of Safety (Shear)	1.50

Cabinet Name: TIP Control, H12-P607

Applied Horizontal Acceleration	1.5G
Applied Vertical Acceleration	2.6 G
Allowable Shear Stress in Weld	21,000 PSI
Weight of Cabinet (Approx.)	755 LBS
Number of Plug Welds Used for Mounting	8
Height of Center of Gravity	50 Inches
Total Normal Force per Plug Weld	858.8 LBS
Total Shear Force per Plug Weld	141.6 LBS
Stress in Weld	1,243 PSI
Margin of Safety (Shear)	15.9

Cabinet Name: Division A Radiation Monitor, H12-P606

Applied Horizontal Acceleration	1.6 G
Applied Vertical Acceleration	4.6 G
Tension Stress (Maximum-Yield)	25,000 PSI
Shear Stress (Maximum-Yield)	13,750 PSI
Weight of Cabinet (Approx.)	1,440 LBS
Number of Mounting Bolts	8
Height of Center of Gravity	45 Inches
Combined Stress (Tensile)	10,539 PSI
Combined Stress (Shear)	5,465 PSI
Margin of Safety (Tensile)	1.37
Margin of Safety (Shear)	1.52

Cabinet Name: Power Range Monitor, H12-P608

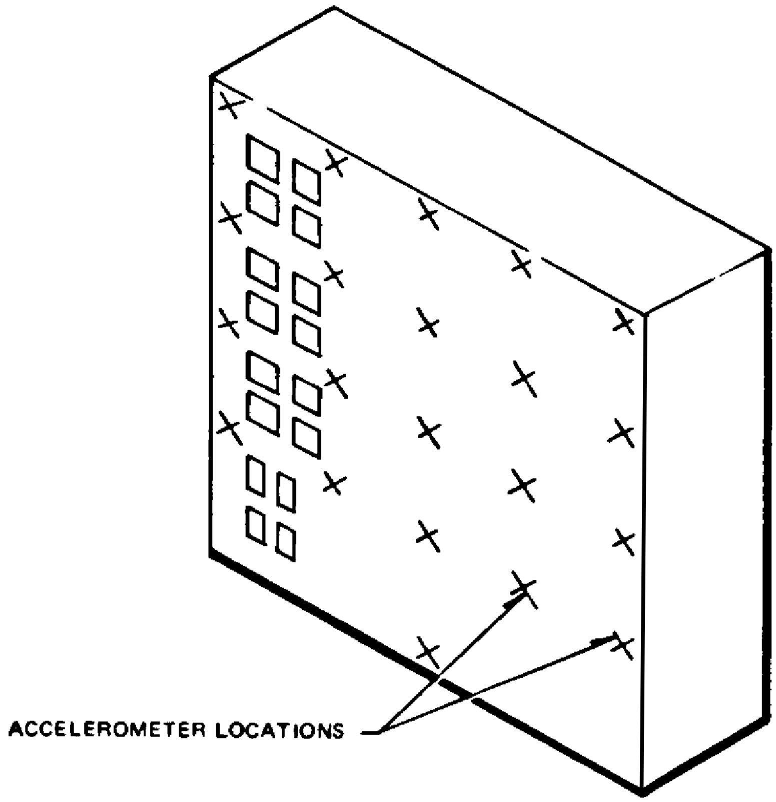
Applied Horizontal Acceleration	1.6 G
Applied Vertical Acceleration	4.6 G
Tension Stress (Maximum-Yield)	25,000 PSI
Shear Stress (Maximum-Yield)	13,750 PSI
Weight of Cabinet (Approx.)	5,750 LBS
Number of Mounting Bolts	40
Height of Center of Gravity	45 Inches
Combined Stress (Tensile)	12,793 PSI
Combined Stress (Shear)	6,633 PSI
Margin of Safety (Tensile)	0.95
Margin of Safety (Shear)	1.07

Cabinet Name: Rod Position Information System, H12-P615

Applied Horizontal Acceleration	1.6 G
Applied Vertical Acceleration	4.6 G
Tension Stress (Maximum-Yield)	25,000 PSI
Shear Stress (Maximum-Yield)	13,750 PSI
Weight of Cabinet (Approx.)	1,425 LBS
Number of Mounting Bolts	12
Height of Center of Gravity	45 Inches
Combined Stress (Tensile)	6,953 PSI
Combined Stress (Shear)	3,605 PSI
Margin of Safety (Tensile)	260
Margin of Safety (Shear)	2.81

IV. CONCLUSION

Review of the Margin of Safety for each standard cabinet indicates that the mounting bolts of each cabinet are capable of withstanding a seismic disturbance.



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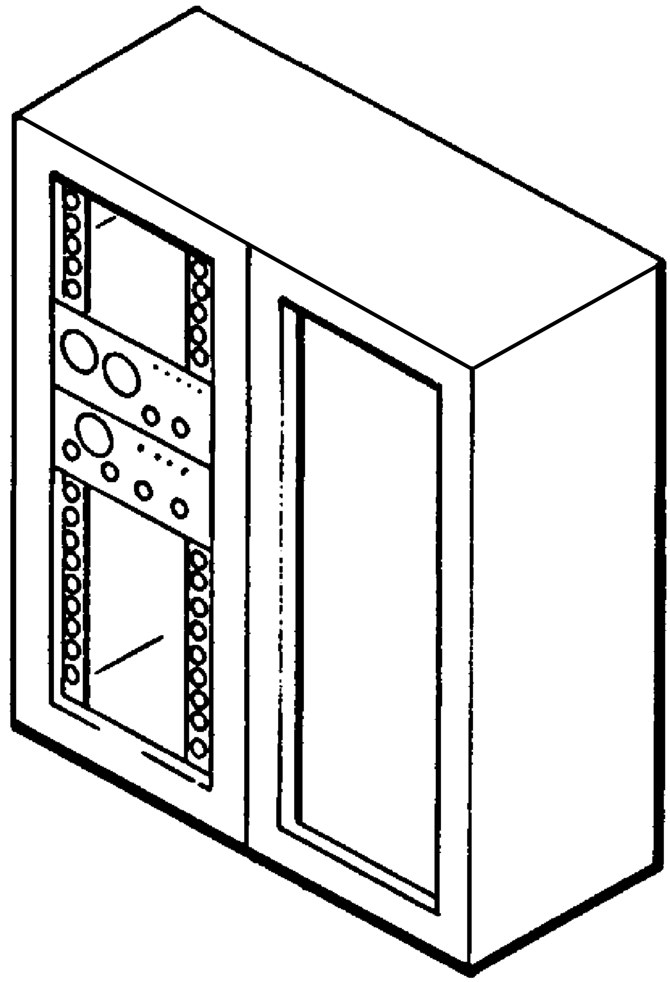
TYPICAL VERTICAL BOARD
(BENCHBOARD WOULD BE THE
SAME WITH A BENCH SECTION
PROTRUDING ABOUT
HALF-WAY DOWN)

FIGURE 3.10A-1, Rev. 47

Auto-Cad Figure Fsar 3_10A_1.dwg



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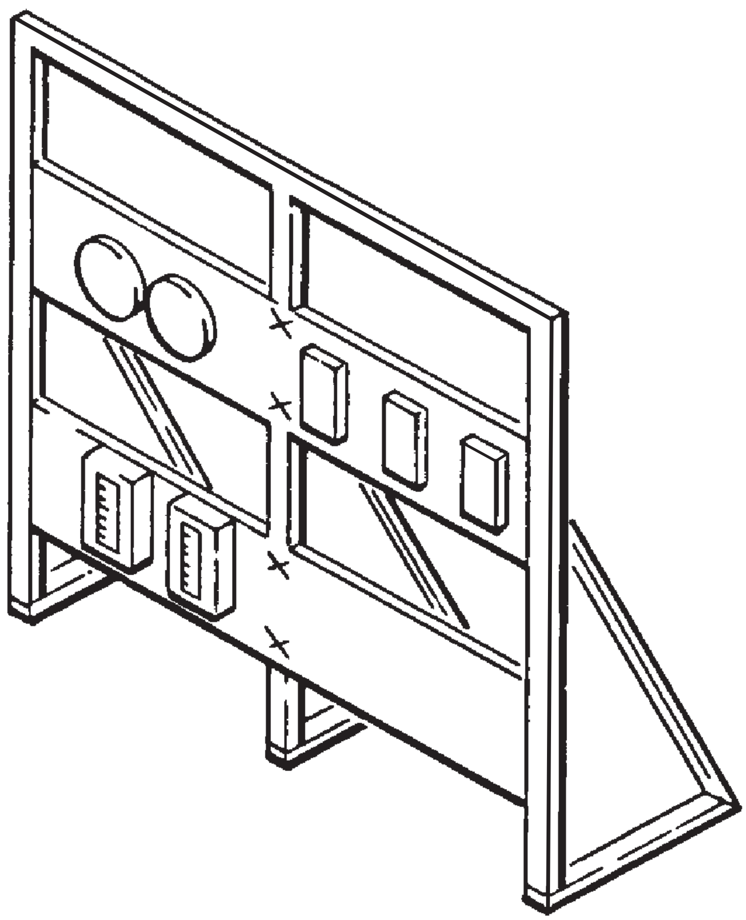
INSTRUMENT RACK

FIGURE 3.10A-2, Rev. 47

Auto-Cad Figure Fsar 3_10A_2.dwg



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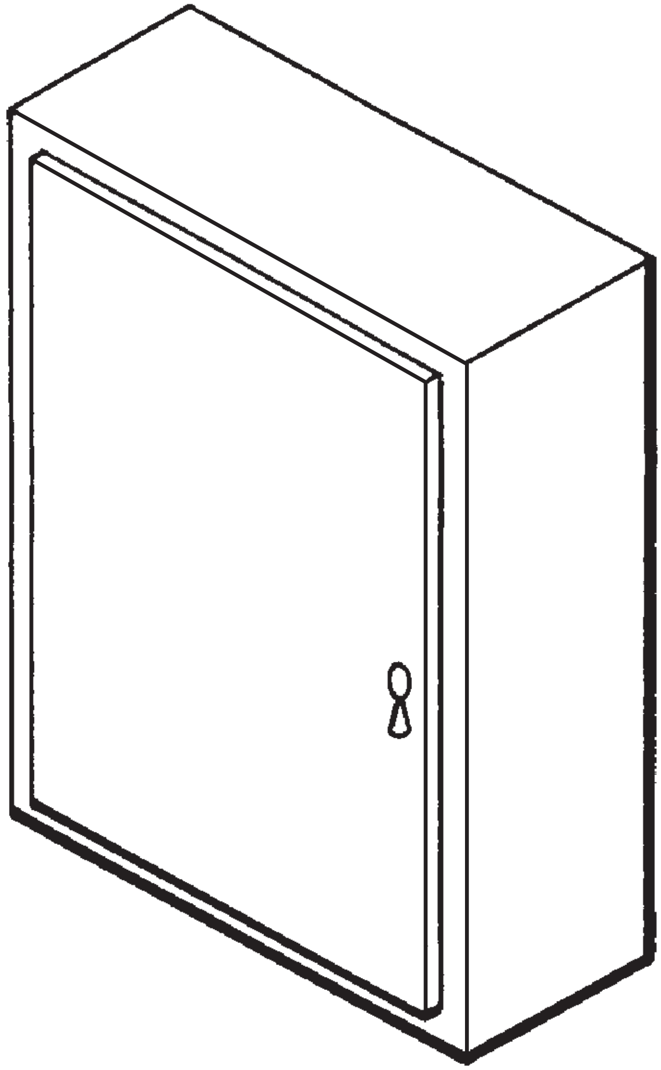
TYPICAL LOCAL RACK
(PIPING AND OTHER EXTERNAL
CONNECTIONS NOT SHOWN)

FIGURE 3.10A-3, Rev. 47

Auto-Cad Figure Fsar 3_10A_3.dwg



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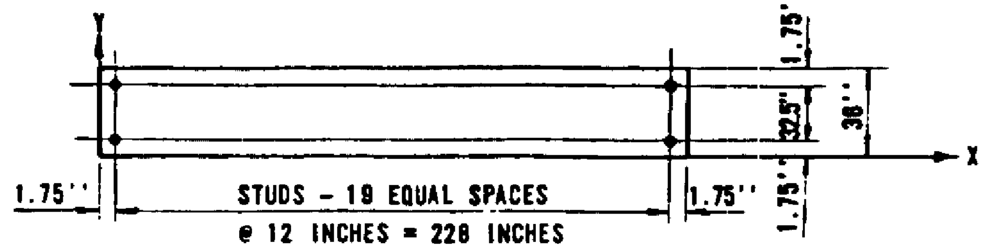
NEMA TYPE-12 ENCLOSURE
(INSTRUMENTS MOUNTED INSIDE
ON INTERNAL MEMBRANE
MOUNTED ON STANDOFFS
ATTACHED TO BACK)

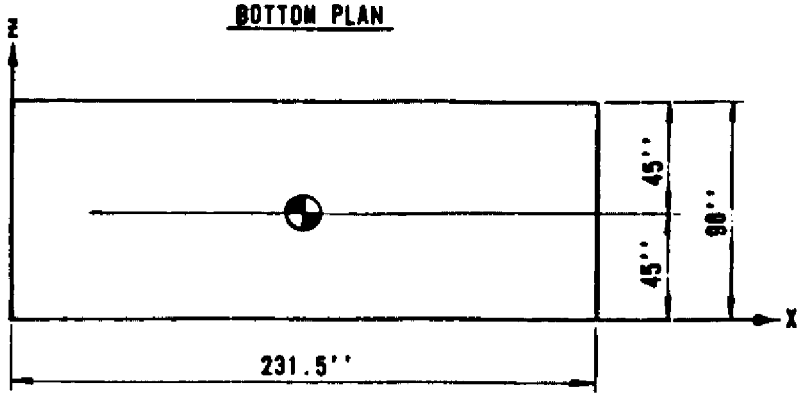
FIGURE 3.10A-4, Rev. 47

Auto-Cad Figure Fsar 3_10A_4.dwg

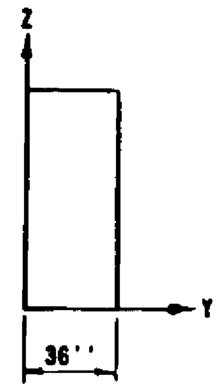
STEP (1): H12-P608, POWER RANGE MONITOR CABINET



BOTTOM PLAN



FRONT ELEVATION



SIDE ELEVATION

LEGEND OF TERMS

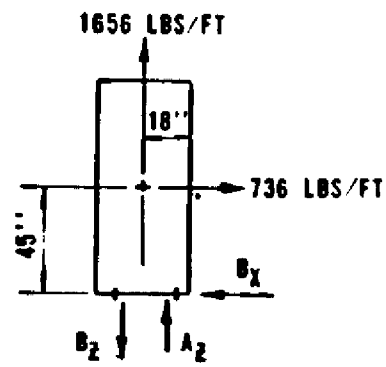
B_Z, A_Z = TENSION/COMPRESSION LOAD IN MOUNTING BOLT

B_X = SHEAR LOAD IN MOUNTING BOLT

B_X' = MAXIMUM COMBINED SHEAR LOAD AT A POINT IN BOLT DUE TO OVERTURNING AND UPLIFT

B_Y' = MAXIMUM COMBINED TENSION LOAD AT A POINT IN BOLT DUE TO OVERTURNING AND UPLIFT

S_t = MAXIMUM COMBINED TENSILE STRESS AT A POINT IN BOLT



FREE BODY DIAGRAM

STEP (2):

ABSOLUTE COMBINED LOADS (G's @ 3% DAMPING)

HORIZONTAL (3 TO 80HZ) 1.5G TO 1.6G [Gx, Gy]
VERTICAL (13 TO 18HZ) 4.6G [Gz]

PANEL FORCES PER FOOT OF PANEL

H12-P608

FRONT TO BACK	Fx = 1.6G (460) = 736LB	
SIDE TO SIDE	Fy = 1.6G (460) = 736LB	
UP	+Fz = (4.6-1G) 460 = 1656LB	
DOWN	-Fz = (4.6+1G) 460 = 2576LB	

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CABINET INSTALLATION FOR
SEISMIC AND HYDRODYNAMIC
LOADS - SAMPLE CALCULATION
(CABINET H12-P608)

FIGURE 3.10A-5-1, Rev. 47

Auto-Cad Figure Fsar 3_10A_5_1.dwg

STEPS (3) & (5): CASE 1

FRONT TO BACK

$$\overset{+}{\curvearrowleft}\Sigma M_A \; \big[(3.75\ FT)(736\ LBS/FT)(19\ FT) + (1.5\ FT)(1656\ LBS/FT)(19\ FT) - (2.65\ FT)(B_Z)\big] \div 20\ STUDS = 0$$

$$B_Z = \frac{34{,}960\ LBS}{20\ STUDS}$$

$$B_Z = LBS\ PER\ STUD = \frac{34{,}960\ LBS}{20\ STUDS} = 1{,}748\ LBS/STUD\ TENSION$$

$$\Sigma F \updownarrow = 0 \quad \big[(1656\ LBS/FT)(19\ FT) + A_Z - 34{,}960\ LBS\big] \div 20 = 0$$

$$A_Z = \frac{3496\ LBS}{20\ STUDS}$$

$$A_Z = LBS\ PER\ STUD = \frac{3496\ LBS}{20\ STUDS} = 174.8\ LBS/STUD\ COMPRESSION$$

STEPS (4) & (5): SHEAR LOAD

$$B_X = \frac{(736\ LBS/FT)(19\ FT)}{40\ STUDS} = 349.6\ LBS/STUD$$

STEP (6): COMBINED STRESS

SHEAR

$$B'_X = \sqrt{B_X^2 + \left(\frac{B_Z}{2}\right)^2} = \sqrt{(349.6)^2 + \left(\frac{1748}{2}\right)^2}$$

$$B'_X = 941.33\ LBS/STUD$$

FORMULAS FOR COMBINED TENSION AND SHEAR AT A POINT MAY BE FOUND IN STRENGTH OF MATERIALS, 2ND ED. BY SINGER, HARPER & ROW PUBLISHERS, 1962.

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CABINET INSTALLATION FOR
SEISMIC AND HYDRODYNAMIC
LOADS - SAMPLE CALCULATION
(CABINET H12-P608)

FIGURE 3.10A-5-2, Rev. 47

Auto-Cad Figure Fsar 3_10A_5_2.dwg

NORMAL (TENSILE)

$$B'_Y = \frac{B_Z}{2} + \sqrt{B_X{}^2 + \left(\frac{B_Z}{2}\right)^2} = \frac{1748}{2} + \sqrt{349.6^2 + \left(\frac{1748}{2}\right)^2}$$

$$B'_Y = 874 + 941.33 = 1815.33 \text{ LBS/STUD}$$

$$S_t = \frac{B'_Y}{A_t} = \frac{1815.33}{.1419} = 12{,}793 \text{ PSI}$$

STEP (7): **MARGIN OF SAFETY**

TENSILE

$$\text{M.S. YIELD STRENGTH} = \frac{25{,}000}{12{,}793} - 1 = +.95$$

SHEAR

USE SHEAR YIELD = .55 TENSILE YIELD

$$\text{M.S. YIELD STRENGTH} = \frac{.55(25{,}000)}{\left(\frac{941.33}{.1419}\right)} - 1$$

$$= +1.07$$

ALL MARGINS OF SAFETY ARE POSITIVE, THERFORE, MOUNTING OF CABINET IS ADEQUATE TO RESTRAIN DESIGN LOADS.

STEP (8): STUD PRE-LOAD VS. STUD TENSILE LOAD

PRE-LOAD = 4800 LBS > B'_Y = 1815 LBS, THEREFORE, MOUNTING OF CABINET IS ADEQUATE TO RESTRAIN DESIGN LOADS.

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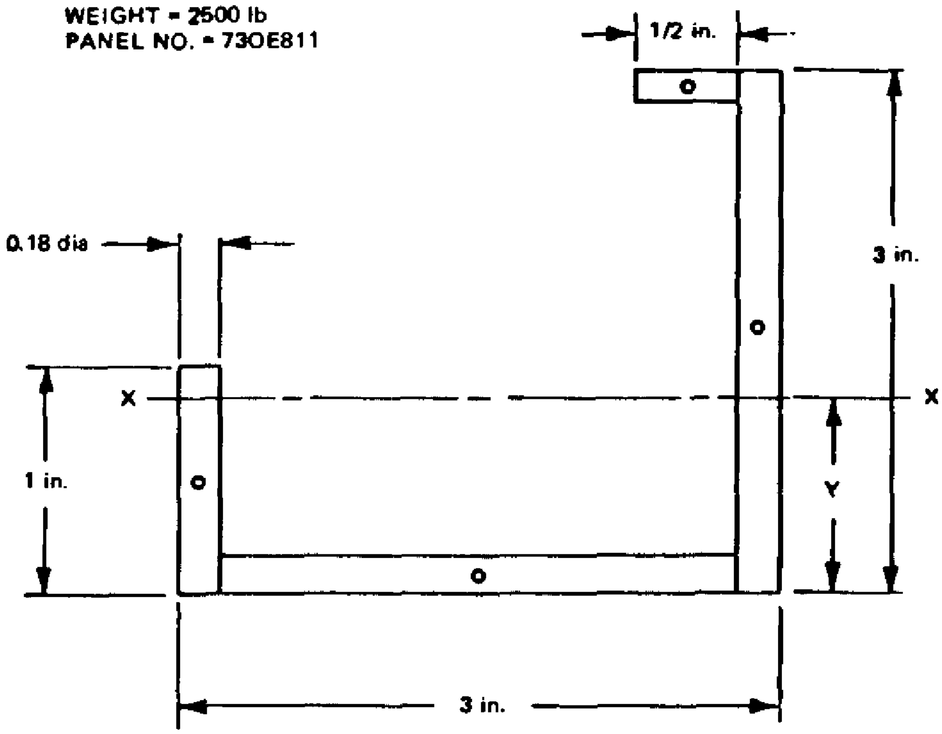
CABINET INSTALLATION FOR SEISMIC AND HYDRODYNAMIC LOADS - SAMPLE CALCULATION (CABINET H12-P608) FIGURE 3.10A-5-3, Rev. 47

Auto-Cad Figure Fsar 3_10A_5_3.dwg



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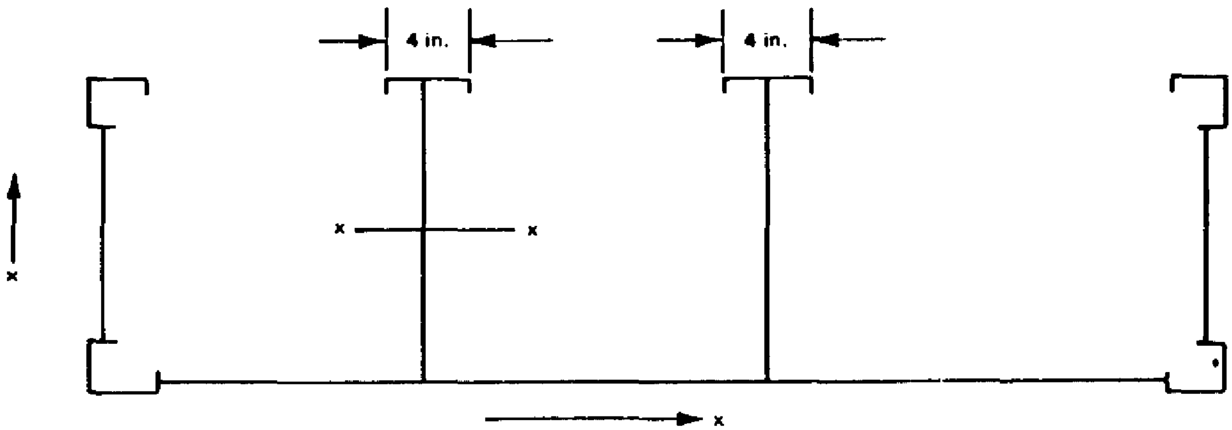
CORNER POST

FIGURE 3.10A-6, Rev. 47

Auto-Cad Figure Fsar 3_10A_6.dwg

SECOND APPROXIMATION

For a second approximation, consider two 0.18 in. x 20 in. barriers in addition to the corner posts. The plan view of the panel is shown in Figure 3.10C-2.



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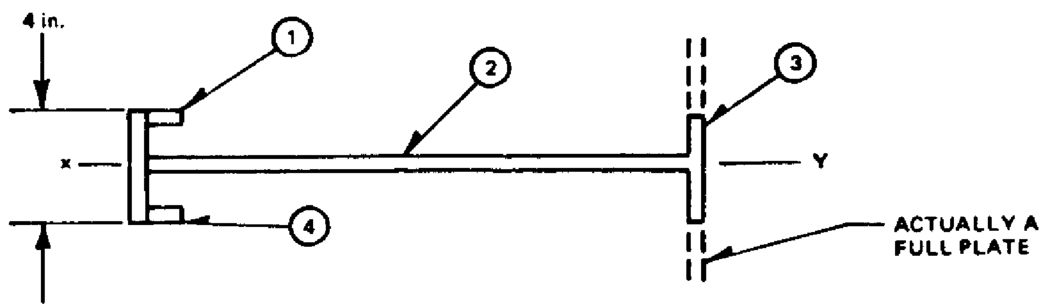
PLAN VIEW OF PANEL

FIGURE 3.10A-7, Rev. 47

Auto-Cad Figure Fsar 3_10A_7.dwg

In the X direction just one barrier will raise the frequency to 30HZ. Use 4 inches of the back panel for each of the two barriers (see Figure 3.10C-3) and the natural frequency in the Y direction becomes 4 HZ.



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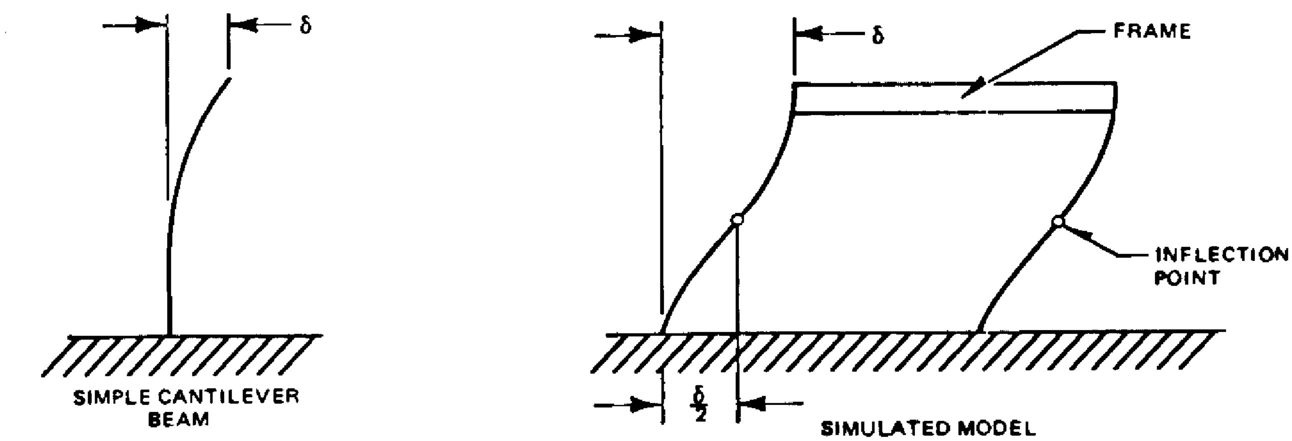
BARRIER WITH TWO END PLATES

FIGURE 3.10A-8, Rev. 47

Auto-Cad Figure Fsar 3_10A_8.dwg



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PANEL DEFLECTIONS

FIGURE 3.10A-9, Rev. 47

Auto-Cad Figure Fsar 3_10A_9.dwg

3.10b SEISMIC QUALIFICATION OF NON-NSSS SUPPLIED SEISMIC CATEGORY I INSTRUMENTATION

3.10b.1 SEISMIC QUALIFICATION CRITERIA

3.10b.1.1 Seismic Category I Equipment Identification

Seismic Category I instrumentation devices and panels were designed to withstand the dynamic effects of the Operating Basis Earthquake (OBE) and the Safe Shutdown Earthquake (SSE) established for this power plant. In addition, when required due to equipment location, instrumentation devices and panels are designed to withstand vibration due to Safety Relief Valve (SRV) and LOCA conditions in combination with OBE and SSE loads.

Instrumentation devices are mounted in instrumentation panels, on equipment racks, on piping, and mounted on building walls or wall structural elements. All devices, panels, and racks that are classified Seismic Category I are mounted on similarly classified supporting members and located within Seismic Category I structures.

Seismic qualification requirements were imposed on equipment through the purchasing documents that were used to procure that equipment. Accordingly, seismic qualification was achieved through analysis and/or testing of items identified on each purchase order for each type of device.

Qualification of instrumentation devices and of assemblies through testing was not mandatory. Equipment suppliers were permitted to qualify their equipment by any of the methods allowable in IEEE 344. However, when practical, testing was the preferred method. Analysis methods were not used when equipment was required to perform an active function under seismic conditions.

Instrument racks, instrument tube tray and their supports, and instrument impulse sensing lines were qualified by analysis.

3.10b.1.2 Seismic and Hydrodynamic Design Criteria

The seismic and hydrodynamic (i.e., SRV and LOCA) design and test criteria for qualification of safety related instrumentation for balance-of-plant systems are described below.

3.10b.1.2.1 Functional Criterion

Every instrumentation device shall be capable of performing its safety related function during plant operating conditions of startup, constant power operation, and normal or emergency shutdown without impairment of its safety related function while undergoing seismic and hydrodynamic excitation. The safety related function of instrumentation devices can be either passive or active. Where one type of device is used in both types of applications, the device is qualified for the worst-case application.

3.10b.1.2.2 Qualification Levels

From the plant OBE, SSE, SRV, and LOCA conditions a family of acceleration required response spectra (RRS) were generated for each building elevation for north-south, east-west and vertical directions. The spectra for each elevation where instrumentation is located were examined to establish the worst-case response spectra.

Pipe-mounted devices are procured for certain generic acceleration values (such as 3g or 6g) applied in the vertical and the weakest lateral axis simultaneously. These values are checked against the piping analysis to ensure that the piping response does not exceed the qualification level. Where equipment was not capable of meeting this generic value, the actual "g" value for that equipment was used for qualification.

For devices mounted in panels, the RRS used was derived from the panel analysis or from the panel shaker table test data.

3.10b.1.2.3 Instrumentation Supports

Instrumentation devices, assemblies, and control panels shall be seismically qualified using the supports that will be used during in-plant installation. These items of equipment are required to maintain their functional capability while undergoing earthquake excitation at the equipment supports.

3.10b.1.3 Device Qualification Test Criteria

Devices that were qualified by test were tested in accordance with IEEE Standard 344-1975. In general, test requirements and acceptance criteria are summarized as follows:

a) Devices under test are mounted in a manner that simulates intended use.

b) Devices are tested while in their normal operating condition (e.g., energized) to determine that vibratory conditions do not produce a malfunction or failure. Seismic Category I devices shall not malfunction during or after a safe shutdown earthquake.

c) Devices are tested in all three axes. Simultaneous excitation in all three axes is preferred; however, tests may be run one axis at a time and then be repeated for the other two axes as an acceptable alternative.

d) Where appropriate a frequency sweep (varying the frequency of excitation with time) is conducted at a low "g" value, e.g., 0.2g as noted in IEEE Standard 344. This test was performed to identify resonant frequencies in the range of interest.

e) Devices that are floor- or panel-mounted are subjected to five OBEs and one SSE in each axis tested. Each OBE and SSE consists of random input motion that envelopes the RRS for that device.

f) Devices that are pipe-mounted are subjected to sine-beat tests over the frequency range of 1 to 100 Hz where required. Each sine-beat test is performed at certain generic peak acceleration values (such as 3g or 6g) or to the peak acceleration for the specific mounting location. If used, the generic acceleration values are checked against the piping analysis to insure that the piping response does not exceed the qualification level.

g) The criteria for malfunction or failure include as many of the following characteristics as are applicable to the safety related function of the device during and after testing:

   1) Loss of output signal; e.g., open or short circuit
   2) Output variations greater than ±10 percent of full range
   3) Spurious or unwanted output; e.g., relay contact bounce
   4) Major calibration shift; e.g., greater than ±10 percent of range
   5) Structural failure; e.g., broken or loosened parts.

3.10b.2 SEISMIC CATEGORY I EQUIPMENT QUALIFICATION

Detailed information about seismic qualification of Non-NSSS Supplied Seismic Category I Instrumentation is maintained in a central file within PP&L. A synopsis of this information was by SQRT forms previously submitted to the NRC.

3.10b.3 Methods and Procedures of Analysis or Testing of Supports of Instrumentation

Instrumentation equipment was qualified by test. The instrument support design was considered during the qualification process.

3.10b.4 Operating License Review

Results of tests and analyses were provided in individual SQRT Forms.

3.10c SEISMIC QUALIFICATION OF NON-NSSS SEISMIC CATEGORY I ELECTRICAL EQUIPMENT

3.10c.1 SEISMIC QUALIFICATION CRITERIA

Seismic qualification of Seismic Category I electrical equipment and supports was demonstrated by the suppliers test laboratories, or consultants by analysis and/or by tests.

Seismic qualification of electrical equipment and supports was performed by analysis when the equipment could be modeled and the structural and functional integrity was adequately represented.

The analysis were performed by an equivalent static analysis or by a dynamic analysis. See Subsection 3.7b.3.5 for details of equivalent static load method of analysis.

The dynamic analysis was performed by the response spectrum method. See Subsection 3.7.3.1 for details of dynamic analysis. When analysis was not sufficient to determine seismic integrity, then tests or a combination of tests and analysis was performed to qualify the electrical equipment and supports.

3.1Oc.1.1 Equipment Location

Electrical equipment and supports are located within the several buildings on the Susquehanna Steam Electric Stations Units 1 and 2.

3.10c.1.2 Response Spectrum Curves for the Electrical Equipment and Supports

Response spectrum curves are based upon the seismic analysis of the supporting structure and represent the maximum seismic response, as a function of oscillator frequency, of an array of single degree of freedom damped oscillators at a particular location within the structure (See Section 3.7).

3.10c.1.3 Seismic Category I Electrical Equipment Loads

Seismic Category I electrical equipment will withstand simultaneously the horizontal and vertical accelerations caused by the OBE and the design SSE as defined herein, in conjunction with applicable electrical, mechanical, and thermal loads. The functions of electrical equipment or components, which are necessary for the functional requirements of the equipment, shall not be impaired when the equipment is subjected to the OBE or the SSE in conjunction with applicable electrical, mechanical, and thermal loads.

3.10c.1.4 Safe Shutdown Earthquake (SSE) Conditions

SSE is defined as an earthquake that produces the maximum vibratory ground motion for which certain structures, systems, and components are designed to remain functional. These structures, systems, and components are necessary to ensure the following:

a) Integrity of reactor coolant pressure boundary

b) Capability to shut down the reactor and maintain it in safe shutdown condition

c) Capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures to the radioactive material released to the environment.

The load combinations include gravity loads and operating loads. Allowable stresses in the structural portions may be increased to 150 percent of allowable working stress limits. The resulting deflections, misalignment or binding of parts, or effects on electrical performance (microphonics, contact bounce, etc) do not prevent the operation of the equipment during or after the seismic disturbance.

3.10c.1.5 Operating Basis Earthquake (OBE) Conditions

The load combinations include gravity loads and operation loads.

Allowable stresses in the structural steel portions may be increased to 125 percent of the allowable working stress limits as set forth in the appropriate design standards, that is, AISC Manual of Steel Construction, ANSI and other applicable industrial codes. The customary increase in normal allowable working stress due to earthquake is used if, according to the appropriate code, it is less than 25 percent. The resulting deflections, misalignment or binding of parts, or effects on electrical performance (microphonics, contact bounce, etc); does not prevent continuous normal operation of the equipment during and after the seismic disturbance.

For the Diesel Generator 'E' facility, the above 25% increase in allowable working stress limit is not allowed.

3.10c.1.6 Prevention of Overturning and Sliding

Stationary electrical equipment is designed to prevent overturning or sliding by the use of anchor bolts, welding, or other suitable mechanical anchorage devices.

3.10c.2 METHODS AND PROCEDURES FOR QUALIFYING ELECTRICAL EQUIPMENT

3.10c.2.1 Seismic Analysis Method

For the purpose of analysis, the equipment has been idealized as a mathematical model consisting of lumped masses connected by massless elastic structural members. For dynamic analysis, the frequencies and mode shapes have been determined for vibration in the vertical and two orthogonal horizontal directions, termed global directions. The effects of coupling between vibrations in all three global directions have been considered. The spectral acceleration per mode has been obtained from the appropriate response spectrum curve, which has been provided for the appropriate damping value. For determining the spectral acceleration from the response spectrum curves, the value chosen is the largest value on the curve when the

frequency in question varies by ±10 percent. Seismic response in terms of inertia forces, shears, moments, stresses, and deflections are determined for response to seismic excitation in each of the global directions for each mode. (See Subsection 3.7b.3.7)

For the consideration of stress or deflection at any point, the total seismic load consists of the most severe seismic load in one of the horizontal global directions combined by the sum of the absolute values method with the vertical seismic load. (See Subsection 3.7b.3.6)

For the Diesel Generator 'E' facility, responses at a point are obtained by taking the square root sum of the squares of corresponding responses due to three orthogonal components of earthquake acting simultaneously.

3.10c.2.2 Seismic Qualification for Electrical Equipment Operability

The seismic qualification of Category I electrical equipment, equipment supports, and material except in the Diesel Generator 'E' Building meets as a minimum the requirements of IEEE 344-1971 and project specification G-10, "General Project Requirements for A Seismic Design and Analysis of Class I Equipment and Equipment Supports" and complemented by Project Specification G-22, "Design Assessment and Qualification of Seismic Category I Equipment & Equipment Supports for Seismic & Hydrodynamic Loads." Project Specification G-10 is summarized in comparison to IEEE-344-1975 and Regulatory Guide 1.100 in Table 3.9-18.

Electrical equipment is qualified for functional operability during and after an earthquake of magnitude up to and including the SSE according to at least one of the following input excitation tests:

a) Single frequency sinusoidal motion or sine beat motions were continuously inputted during the test at specified frequencies to cover the frequency range up to 33 Hz.

b) Random waveform, multifrequency tests.

For the Diesel Generator 'E' facility, the seismic qualification of Category I electrical equipment, equipment supports and material meets as a minimum the requirements of IEEE 344-1975 and project specification C-1041, "General Specification for Seismic Criteria for Design and Qualification of Seismic Category I Equipment and Equipment Supports Located in the Diesel Generator 'E' Building".

3.10c.2.3 Seismic Test Report Analysis and Methods

The analysis and test reports furnished by the supplier demonstrate the ability of electrical equipment to perform its required function during and after the time it is subjected to the forces resulting from one SSE and a required number of OBE.

Four categories of reports are provided by the supplier of electrical equipment and material applicable to Seismic Category I qualification:

a) Electrical equipment qualified by testing method

b) Electrical equipment support and material qualified by analysis and calculation method

c) Electrical equipment qualified by supplier's certification of Seismic Category I requirements.

d) Combination of analysis and testing.

3.10c.2.3.1 Electrical Equipment Qualified by Testing and Combination of Testing and Analysis Method

Qualification of the electrical equipment listed below is based on testing performed by the suppliers or test laboratories. (Qualification may be based on tested equipment which is similar in design and assembly).

a) Indoor secondary unit substation and indoor power transformers (see Table 3.10c-1)
b) 480 V ac motor control centers (see Table 3.10c-2)
c) Soother monitors and fuse boxes (see Table 3.10c-3)
d) DC distribution panels (see Table 3.10c-4)
e) Battery racks (see Table 3.10c-5)
f) Electrical cable penetrations (see Table 3.10c-6)
g) Battery charger racks and cabinets (see Table 3.10c-8)
h) Panels and termination cabinets (see Table 3.10c-10)
i) Battery chargers (see Table 3.10c-11)
j) 4.16 kV switchgear (see Table 3.10c-12)
k) DC control and load centers (see Table 3. 10c-13)
l) Instrument ac transformers (see Table 3.10c-14)
m) Automatic transfer switches (see Table 3.10c-15)
n) Load isolation motor generator sets (see Table 3.10c-16)
o) Inverters and 120V AC instrument panels (see Table 3-10c-18)
p) Battery Shunt Box (See Table 3.10c-3)

3.10c.2.3.2 Electrical Equipment and Supports Qualified by Analysis Method

Cable trays were qualified by analysis method based on similarity in design and assembly, and representing the type of equipment shown in Table 3.10c-7.

3.10c.2.3.3 Electrical Equipment Qualified by Suppliers' Certification

Large induction motors (see Table 3.10c-9) were certified by the suppliers the motors had been previously qualified by tests equivalent to those described in Subsections 3.10c.1, or were analyzed and calculated.

3.10c.2.3.4 Minimum Operating Voltage of Voltage Relays

All non-NSSS and non-ACR voltage relays which must be energized or must remain energized to perform safety functions during a seismic event are tabulated in Table 3.10c.17.

3.10c.3 Methods and Procedures of Analysis or Testing of Supports of Electrical Equipment

Electrical equipment supports were qualified or tested with their associated equipment. See Subsection 3.10c.2 for a description of the applicable method or procedure.

3.10c.4 Operating License Review

A summary of tests and analyses is identified in Tables 3.10c-1 to 3.10c-16.

TABLE 3.10c-1 SECONDARY UNIT SUBSTATIONS AND POWER TRANSFORMERS

ITEM NO.	EQUIPMENT IDENTIFICATION DESCRIPTION	EQUIPMENT NO.	LOCATION BLDG.	ELEV.	UNIT NO.	SUPPLIER	TESTING FACILITIES	QUALIFICATION CRITERIA	QUALIFICATION "E1" SIGNED BY:
8856-E-117-57&58	Single Ended Secondary Unit Substation Consisting of: a. Terminal Chamber, b. 750 kVA Transformer, c. L.W. Switchgear	1B-210 1B-220 1B-230 1B-240 2B-210 2B-220 2B-230 2B-240	Reactor	749 749 719 719 749 749 719 719	1 1 1 1 2 2 2 2	I.T.E. Imperial Corporation	Wyle Laboratories Norco, California	Project Spec G-10* & IEEE-344-1975	Report # 26340-2 26340-3 26340-4 By: G. Shipway
Spec E-1023	1000 kVA Transformer 4.16KV-480V	0X565	D. Gen. 'E' Bldg.	675	Comm.	B.B.C.	Wyle Laboratories Huntsville, Alabama	Spec C-1041 & IEEE 344-1975	Report # 37-55778-STA By: C. E. Kunkel
Spec E-1023	5kV Switch	0S569	D. Gen. 'E' Bldg.	675	Comm.	B.B.C.	Wyle Laboratories Huntsville, Alabama	Spec C-1041 & IEEE 344-1975	Report # 37-55778-STA By: C. E. Kunkel

* NOTE: Specification G-10 is complemented by Specification G-22. For G-10 Specification Summary, See Table 3.9-31.

TABLE 3.10c-2 MOTOR CONTROL CENTERS (Page 1 of 2)

ITEM NO.	EQUIPMENT IDENTIFICATION DESCRIPTION	EQUIPMENT NO.	LOCATION BLDG.	ELEV.	UNIT NO.	SUPPLIER	TESTING FACILITIES	QUALIFICATION CRITERIA	QUALIFICATION "E1" SIGNED BY:
8856-E-118	Motor Control Center	0B-136	Control	783	Cmmn	Cutler-Hammer	Wyle Laboratories Huntsville, Alabama	Project Spec G-10* & IEEE-344-1975	Report #42966-1
		0B-146	Control	783	Cmmn				
		0B-516	D.Gen. 'A-D'	677	Cmmn				By: J. Foreman
		0B-517		677					Wyle
		0B-526		677	Cmmn				Report #45590-1
		0B-527		677	Cmmn				#45590-2
		0B-536		677	Cmmn				
		0B-546		677	Cmmn				By: Vincent F. Kearns III
		1B-216	Reactor	683	1				
		1B-217		749	1				C. H. Eaton
		1B-219		670	1				Report #DA57-3251 By:
		1B-226		683	1				Vincent F. Kearns III
		1B-227		749	1				
		1B-229		719	1				
		1B-236		719	1				
		1B-237		670	1				
		1B-246		719	1				
		1B-247		670	1				
		2B-216	Reactor	683	2				
		2B-217		749	2				
		2B-226		683	2				
		2B-227		749	2				
		2B-236		719	2				
		2B-237		670	2				
		2B-246		719	2				
		2B-247		670	2				

* Note: Specification G is complemented by Specification G-22

TABLE 3.10c-2 MOTOR CONTROL CENTERS (Page 2 of 2)

ITEM NO.	EQUIPMENT IDENTIFICATION DESCRIPTION	EQUIPMENT NO.	LOCATION BLDG.	ELEV.	UNIT NO.	SUPPLIER	TESTING FACILITIES	QUALIFICATION CRITERIA	QUALIFICATION "E1" SIGNED BY:
		1Y-216	Reactor	683	1				
		1Y-218		719	1				
		1Y-226		683	1				
		1Y-236		719	1				
		1Y-246		719	1				
		2Y-216		683	2				
		2Y-218		719	2				
		2Y-226		683	2				
		2Y-236		719	2				
		2Y-246		719	2				
Spec E-1024	Motor Control Center	OB-565	D. Gen. 'E'	675	Comm. Telemecanique		Wyle Laboratories Huntsville, Alabama	Spec C-1041 & IEEE 344-1975	Telemecanique Report No. SC-655 By: Paul W. Higgins

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TABLE 3.10c-3
BATTERY MONITORS AND FUSE BOXES

ITEM NO.	EQUIPMENT IDENTIFICATION: DESCRIPTION	EQUIPMENT IDENTIFICATION: EQUIPMENT NO.	LOCATION: BLDG	LOCATION: ELEV.	UNIT NO.	SUPPLIER	TESTING FACILITIES	QUALIFICATION CRITERIA	QUALIFICATION "E1" SIGNED BY:
8856-E-0119AC	Battery Monitor 24V	1D-675	Control	771	1	Power Conversion Products, Inc.	Wyle. Laboratories, Huntsville, Alabama	Project Spec G-10* & IEEE 344-1975	Vincent F. Kearns Test Report #45463-1 Rev. A
		1D-676			1				
		1D-685			1				
		1D-686			1				
		2D-675			2				
		2D-676			2				
		2D-685			2				
		2D-686			2				
	Battery Monitor 125V	1D-691			1				
		1D-692			1				
		1D-693			1				
		1D-694			1				
		2D-691			2				
		2D-692			2				
		2D-693			2				
		2D-694			2				
	Battery Monitor 250V	1D-695			1				
		1D-696			1				
		2D-695			2				
		2D-696			2				
	125V Fuse Box 2-1000A	1D-611			1				
		1D-621			1				
		1D-631			1				
		1D-641			1				
		2D-611			2				

*NOTE: Specification G-10 is complemented by Specification G-22.

TABLE 3.10c-3
BATTERY MONITORS AND FUSE BOXES

ITEM NO.	EQUIPMENT IDENTIFICATION		LOCATION		UNIT NO.	SUPPLIER	TESTING FACILITIES	QUALIFICATION CRITERIA	QUALIFICATION "E1" SIGNED BY:
	DESCRIPTION	EQUIPMENT NO.	BLDG	ELEV.					
		2D-621			2				
		2D-631			2				
		2D-641			2				
	250V Fuse Box 2-1600A	1D-651			1				
		1D-661			1				
		2D-651			2				
		2D-661			2				
	Fuse Box, 24V, 2-100A	1D-671			1				
		1D-681			1				
		2D-671			2				
		2D-681			2				
Spec E-1025	Battery Monitor 125V	0D-601	D. Gen. 'E'	656	Comm.	Vitro Corp.	Wyle Laboratories Huntsville, Alabama	Spec C-1041 & IEEE 344-1975	C&D Power Systems Test Report No. QR2-13201-1 By: Paul Wagner
	Battery Float Current Shunt Box 125V	0D595A			0				
		1D610A, 2D610A			1 2				
		1D620A, 2D620A			1 2				
		1D630A, 2D630A			1 2				
		1D640A, 2D640A			1 2				
	Battery Float Current Shunt Box 250V	1D650A, 2D650A			1 2				
		1D660A, 2D660A			1 2				

TABLE 3.10c-4 DC DISTRIBUTION PANELS

ITEM NO.	EQUIPMENT IDENTIFICATION DESCRIPTION	EQUIPMENT NO.	LOCATION BLDG.	ELEV.	UNIT NO.	SUPPLIER	TESTING FACILITIES	QUALIFICATION CRITERIA	QUALIFICATION "E1" SIGNED BY:
8856-E-120	DC Distribution	1D-614	Control	771'	1	I.T.E.	Wyle	Project Spec	Report #26340-5
	Panels	1D-615			1	Imperial	Laboratories	G-10* & IEEE-	By: G. Shipway
	125V 225A	1D-624			1	Corporation	Novco,	344-1975	
	Main Bus	1D-625			1		California		Report #26340-3
		1D-634			1				26340-6
		1D-635			1				By: G. Shipway
		1D-644			1				
		1D-645			1				
	24V 100A	1D-672			1		*NOTE: Specification G-10 is complemented by		
	Main Bus	1D-682			1		Specification G-22.		
	125V 225A	2D-614			2				
	Main Bus	2D-615			2				
		2D-624			2				
		2D-625			2				
		2D-634			2				
		2D-635			2				
		2D-644			2				
		2D-645			2				
	24V 100A	2D-672			2				
	Main Bus	2D-682			2				
Spec E-1027	DC Switchboard 125V	OD-597	D. Gen 'E'	656	Comm	Square 'D' Company	(Qualified by Test) Farwell & Hendricks, Inc. Milford, Ohio	Spec C-1041 & IEEE 344-1975	Square 'D' Report No. 8998-10.09-L74 8998-10.09-L84 By: R. A. Diersing
	DC Distribution Panel 125V	OD-599	D Gen. 'E'	660	Comm	Square 'D' Company	(Qualified by Test) Farwell & Hendricks, Inc. Milford, Ohio	Spec C-1041 & IEEE 344-1975	Square 'D' Report No. 8998-10-09-L74 By: R. A. Diersing

TABLE 3.10c-5
BATTERY RACKS

ITEM NO.	EQUIPMENT IDENTIFICATION		LOCATION		UNIT NO.	SUPPLIER	TESTING FACILITIES	QUALIFICATION CRITERIA	QUALIFICATION "E1" SIGNED BY:
	DESCRIPTION	EQUIPMENT NO.	BLDG	ELEV.					
8856-E-119B	Stationary Batteries 24V 75AH	1D-670	Control	771	1	C&D Batteries Co.	Structural Dynamic Research Corporation, Milford, Ohio for Corporate Consulting Development Co.	Project Spec G-10* & IEEE-344-1975	Report #A379-81-01 Stephen A. Lehrman Dr. John Roland Yow
		1D-680			1				
		2D-670			2				
		2D-680			2				
	Stationary Batteries 125V 720AH	1D-610			1				
		1D-620			1				
		1D-630			1				
		1D-640			1				
		2D-610			2				
		2D-620			2				
		2D-630			2				
		2D-640			2				
	Stationary Batteries 250V 1800AH	1D-650			1				
		1D-660			1				
		2D-650			2				
		2D-650			2				
Spec E-1025	Stationary Batteries 125V 825AH	OD-595	D Gen. 'E'	656	Comm	C&D Power Systems	Wyle Laboratories Huntsville, Alabama	Spec C-1041 & IEEE 344-1975	C&D Power Systems Report No. QR2-54035-1 By: Paul Wagner

NOTE: Specification G-10 is complemented by Specification G-22.

TABLE 3.10c-6 ELECTRICAL CABLE PENETRATION (Page 1)

ITEM NO.	EQUIPMENT IDENTIFICATION DESCRIPTION	EQUIPMENT NO.	LOCATION BLDG.	ELEV.	UNIT NO.	SUPPLIER	TESTING FACILITIES	QUALIFICATION CRITERIA	QUALIFICATION "E1" SIGNED BY:
8856-E-135	Electrical Cable Penetrations:					Westinghouse	Action Labs	Project Spec G-10* & IEEE-344-1975	Report #PEN-TR-16404-81N By: A. Lebourdais
	Neutron Monitor	1W-100A	Reactor	707'	1				
		1W-100B		707'	1				
		1W-100C		707'	1				
		1W-100D		707'	1				
		2W-100A		707'	2				
		2W-100B		707'	2				
		2W-100C		707'	2				
		2W-100D		707'	2				
	Medium Voltage	1W-101A		735'-9"	1				
		1W-101B		733'	1				
		1W-101C		733'	1				
		1W-101D		700'	1				
		1W-101E		730'-2"	1				
		1W-101F		727'-0"	1				
		2W-101A		735'-9"	2				
		2W-101B		733'	2				
		2W-101C		733'	2				
		2W-101D		700'	2				
		2W-101E		730'-2"	2				
		2W-101F		727'-0"	2				
	Low Level Signal	1W-102A		729'-1"	1				
		1W-102B		729'-1"	1				
		2W-102A		729'-1"	2				
		2W-102B		729'-1"	2				
		1W-103A		707'	1				
		1W-103B		712'	1				
		2W-103A		707'	2				
		2W-103B		712'	2				
	Control Rod Drive	1W-104A		707'	1				
		1W-104B		712'	1				
		1W-104C		712'	1				
		1W-104D		712'	1				
		2W-104A		707'	2				
		2W-104B		712'	2				
		2W-104C		712'	2				
		2W-104D		712'	2				

* NOTE: Specification G-10 is complemented by Specification G-22.

TABLE 3.10c-6 ELECTRICAL CABLE PENETRATION (Page 2)

ITEM NO.	EQUIPMENT IDENTIFICATION DESCRIPTION	EQUIPMENT NO.	LOCATION BLDG.	ELEV.	UNIT NO.	SUPPLIER	TESTING FACILITIES	QUALIFICATION CRITERIA	QUALIFICATION REPORT SIGNED BY:
8856-E-135	Electrical Cable Penetrations					Westinghouse	Action Labs	Project Spec G-10* & IEEE-344-1975	Report #15404-81" By: A. Labouritis
	Power	1W-105A	Reactor	729'-1"	1				
		1W-105B		729'-1"	1				
		1W-105C		729'-1"	1				
		1W-105D		741'	1		Seismic Test for Modular Penetration 5" Dia.		
		2W-105A		729'-1"	2				
		2W-105B		729'-1"	2				
		2W-105C		729'-1"	2				
		2W-105D		741'	2				
	Low Voltage	1W-106A		729'-1"	1				
		1W-106B		729'-1"	1				
		1W-106C		729'-1"	1				
		1W-106D		741'	1				
		1W-107		741'	1				
		1W-108		729'-1"	1				
		2W-106A		729'-1"	2				
		2W-106B		729'-1"	2				
		2W-106C		729'-1"	2				
		2W-106D		741'	2				
		2W-107		741'	2				
		2W-108		729'-1"	2				
	Suppression Pool Low Voltage, Control and Power	1W-300		688'-6"	1				
		1W-301		688'-1"	1				
		2W-300		688'-6"	2				
		2W-301		688'-1"	2				

* NOTE: Specification G-10 is complemented by Specification G-22.

TABLE 3.10c-7 CABLE TRAYS "SAFEGUARD" (Page 1 of 2)

ITEM NO.	EQUIPMENT IDENTIFICATION DESCRIPTION	EQUIPMENT NO.	LOCATION BLDG.	ELEV.	UNIT NO.	SUPPLIER	TESTING FACILITIES	QUALIFICATION CRITERIA	QUALIFICATION "E1" SIGNED BY:
8856-E-132	Cable Trays: 3"D x 24"W	S9N1-24-144	Control Reactor	670' to 770'	1&2	Husky Product Inc.	Husky Products, Inc. 7405 Industrial Rd Florence, Kentucky	Project Spec G-10 & IEEE-344-1975	1-29-76 a. Test No. 977-978 Load Test- (Trays) By: T. O'Hara B. Heinz b. (Hold Down Test 4/12/76 Test No. 1127-L,H,V, 5/14/76 1151& 1152 7/21/76 1188 8/10/76 1196-H,V c. Electric Test 12/12/72 Harper-Morrez B. Schuster d. Seismic Calculation 8/11/76 By: B. Schuster
	3"D x 18"W	S9N1-18-144							
	3"D x 12"W	S9N1-12-144							
	5"D x 24"W	S9N1-24-144							
	5"D x 18"W	S9N1-18-144							
	5"D x 12"W	S9N1-12-144							

TABLE 3.10c-7 CABLE TRAYS "SAFEGUARD" (Page 2 of 2)

ITEM NO.	EQUIPMENT IDENTIFICATION DESCRIPTION	EQUIPMENT IDENTIFICATION EQUIPMENT NO.	LOCATION BLDG.	LOCATION ELEV.	UNIT NO.	SUPPLIER	TESTING FACILITIES	QUALIFICATION CRITERIA	QUALIFICATION "E1" SIGNED BY:
Spec E-1032	Cable Trays: 3"D x 12"W		D. Gen 'A-D' and 'E'			T. J. Cope	(By Analysis)	Spec C-1041 & IEEE 344-1975	PP&L Calc. No. JI-CMHR-102 By: T. A. Gorman
	3"D x 18"W								
	3"D x 24"W								
	5"D x 12"W								
	5"D x 18"W								
	5"D x 24"W								
	5"D x 36"W								

TABLE 3.10c-8
BATTERY CHARGER RACKS AND CABINETS

ITEM NO.	EQUIPMENT ID		LOCATION		UNIT NO.	SUPPLIER	TESTING FACILITIES	QUALIFICATION CRITERIA	QUALIFICATION "E1" SIGNED BY:
	DESCRIPTION	EQUIPMENT NO.	BLDG	ELEV.					
8856-E-119-AC	Battery Chargers 125V 100A	1D-613	Control	771'	1	Power Conversion Products Inc. 42 East Street, Crystal Lake, Illinois 66014	Wyle Laboratories Huntsville, Alabama	Project Spec G-10* & IEEE-344-1975	Test Report #45463-1 Rev. A Vincent F. Kearns
		1D-623			1				
		1D-633			1				
		1D-643			1				
		2D-613			2				
		2D-623			2				
		2D-633			2				
		2D-643			2				
		0D-673			Cmmn				
	Battery Chargers 250V 300A	1D-653A			1				
		1D-653B			1				
		1D-663			1				
		2D-653A			2				
		2D-653B			2				
		2D-663			2				
		0D-683			Cmmn				
	Battery Chargers 24V 25A	1D-673			1				
		1D-674			1				
		1D-683			1				
		1D-684			1				
		2D-673			2				
		2D-674			2				
		2D-683			2				
		2D-684			2				
		0D-685			Cmmn				
Spec E-1025	Battery Charger 125 V 200A	0D-596	D. Gen. 'E'	656	Comm	C&D Power Systems	Wyle Laboratories Huntsville, Alabama	Spec. C-1041 & IEEE 344-1975	C&D Test Report QR2-52666-1 By: Paul Wagner

*NOTE: Specification G-10 is complemented by Specification G-22.

TABLE 3.10c-9 LARGE INDUCTION MOTORS 4000V

ITEM NO.	EQUIPMENT IDENTIFICATION DESCRIPTION	EQUIPMENT NO.	LOCATION BLDG.	LOCATION ELEV.	UNIT NO.	SUPPLIER	TESTING FACILITIES	QUALIFICATION CRITERIA	QUALIFICATION "RIM" SIGNED BY:
8856-E-112	Large Induction						The motors are	Project Spec.	C. K. McDonald
	Motors 4000V 3ø						qualifieid by	G-10* IEEE	Report #
	450 HP 1800 RPM	0P-504-A		685	Comn	General	seismic analysis	344-1975	ME-573
	Emergency	0P-504-B			Comn	Electric	by McDonald		ME-574
	Service Water	0P-504-C			Comn		Engineering		
	Pump	0P-504-D			Comn		Analysis, Inc.,		
	600 HP 4000V	1P-506-A		685	1		Birmingham,		
	1200 RPM	1P-506-B			1		Alabama		
	RHR Service	2P-506-A			2				
	Water Pump	2P-506-B			2				

* NOTE: Specification G-10 is complemented by Specification G-22.

TABLE 3.10c-10
PANELS AND TERMINATION CABINETS

ITEM NO.	EQUIPMENT IDENTIFICATION		LOCATION		UNIT NO.	SUPPLIER	TESTING FACILITIES	QUALIFICATION CRITERIA	QUALIFICATION "E1" SIGNED BY:
	DESCRIPTION	EQUIPMENT NO.	BLDG	ELEV.					
	Transfer Panels	0C512E-A	D. Gen. 'E'	656'-6"	1	York Electro-Panel	(By Analysis)	Project Spec C-1041* & IEEE 344-1975	N&S Reports 1290-1 and 1290-2 By: M. Randall
		0C512E-B			2				
		0C512E-C			1				
		0C512E-D			2				
	Termination Cabinets	0TC512-A/C			1				
		0TC512-B/D			2				
	Transfer Panels	OC512-A	D. Gen. 'A-D'	710'-9"	1				
		0C512-B			2				
		0C512-C			1				
		0C512-D			2				
	Synchronizing Panel	0C619	D. Gen. 'E'	675'-6"	Comm	Golden Gate Switchboard Co.	Wyle Labs Norco Calif.	Project Spec. C-1041# IEEE 344-1975	Wyle Labs Report No. 53444 By: C. C. Lee

* NOTE: Spec C-1041 is complemented by Spec. E-1026.
# NOTE: Spec C-1041 is complemented by Spec. E-1022.

TABLE 3.10c-11
BATTERY CHARGERS

EQUIPMENT IDENTIFICATION									
ITEM NO.	DESCRIPTION	EQUIPMENT NO.	BLDG	ELEV.	UNIT NO.	SUPPLIER	TESTING FACILITIES	QUALIFICATION CRITERIA	QUALIFICATION "E1" SIGNED BY:
8856-E-119-AC	Battery Chargers 125V 100A	1D-613	Control	771	1	Power Conversion Products, Inc.	Wyle Laboratories, Huntsville, Alabama	Project Spec G-10* & IEEE-344-1975	Test Report #45463-1 Rev. A Vincent F. Kearns
		1D-623			1				
		1D-633			1				
		1D-643			1				
		2D-613			2				
		2D-623			2				
		2D-633			2				
		2D-643			2				
		2D-673			Comm				
	Battery Chargers 250V 300A	1D-653A			1				
		1D-653B			1				
		1D-663			1				
		2D-653A			1				
		2D-653B			2				
		2D-663			2				
		2D-683			2				
		2D-684			2				
		0D-685			Comm				
	Battery Chargers 24V 25A	1D-673			1				
		1D-674			1				
		1D-683			1				
		1D-684			1				
		2D-673			2				
		2D-674			2				
		2D-683			2				
		2D-684			2				
		0D-685			Comm				
Spec E-1025	Battery Charger 125V 200A	0D596	D. Gen. 'E'	656	Comm	C&D Power Systems	Wyle Laboratories Huntsville, Alabama	Spec. C-1041 & IEEE 344-1975	C&D Test Report QR2-52666-1 QR2-52666-1 By: Paul Wagner

*NOTE: Specification G-10 is complemented by Specification G-22.

TABLE 3.10c-12 4.16 KV SWITCHGEAR

ITEM NO.	EQUIPMENT IDENTIFICATION DESCRIPTION	EQUIPMENT IDENTIFICATION EQUIPMENT NO.	LOCATION BLDG.	LOCATION ELEV.	UNIT NO.	SUPPLIER	TESTING FACILITIES	QUALIFICATION CRITERIA	QUALIFICATION "E1" SIGNED BY:
8856-E-109-	4.16 kV	1A-201	Reactor	749	1	Westinghouse	Wyle	Project Spec	Report #'s
34	Switchgear	1A-202		749	1		Laboratory,	G-10* & IEEE-	57577-1
		1A-203		719	1		Huntsville,	344-1975	57588
		1A-204		719	1		Alabama		
		2A-201		749	2		and		58642
		2A-202		749	2		Wyle		58664
		2A-203		719	2		Laboratory		
		2A-204		719	2		Novco, CA		G. Shipway
Spec E-1022	4.16 kV	OA510	D. Gen. 'E'	657	Comm	B.B.C.	Wyle	Spec C-1041 &	C&D
	Switchgear						Laboratories	IEEE 344-1975	Report No.
							Huntsville,		37-55736-SSA
							Alabama		
									By: C. E. Kunkel
		OA510A	D. Gen. 'A-D'	710	Comm	B.B.C.			
		OA510B							
		OA510C							
		OA510D							

* NOTE: Specification G-10 is complemented by Specification G-22.

TABLE 3.10c-13 DC CONTROL AND LOAD CENTERS

ITEM NO.	EQUIPMENT IDENTIFICATION DESCRIPTION	EQUIPMENT NO.	LOCATION BLDG.	ELEV.	UNIT NO.	SUPPLIER	TESTING FACILITIES	QUALIFICATION CRITERIA	QUALIFICATION "E1" SIGNED BY:
8856-E-121-22-1	DC Control Centers 250V	1D-254	Reactor	670	1	General Electric Co.	Wyle Laboratory, Novco, CA	Project Spec G-10* & IEEE-344-1971	Report# 26340-8 By: G. Shipway
		1D-264	Reactor	683	1				
		1D-274	Reactor	683	1				
		2D-254	Reactor	670	2				
		2D-264	Reactor	683	1				
		2D-274	Turbine	729	2				
8856-E-121-22-3	DC Load Centers 250V	1D-652	Control	771	1			Project Spec G-22 & IEEE-344-1975	Report #'s 26340-2 26340-3 26340-7 By: G. Shipway
		1D-662	Control	771	1				
		2D-652	Control	771	2				
		2D-662	Control	771	2				
	DC Load Centers 125V	1D-612	Control	771	1				
		1D-622	Control	771	1				
		1D-632	Control	771	1				
		1D-642	Control	771	1				
		2D-612	Control	771	2				
		2D-622	Control	771	2				
		2D-632	Control	771	2				
		2D-642	Control	771	2				
Spec E-1024	DC Motor Control Center 125V	OD598	D Gen 'E'	657	Comm	Telemecanique	Wyle Laboratories Huntsville, Alabama	Spec. C-1041 & IEEE 344-1975	Telemecanique Report No. SC-655 By: Paul Wiggins

* NOTE: Specification G-10 is complemented by Specification G-22.

TABLE 3.10c-14 INSTRUMENT AC TRANSFORMERS

ITEM NO.	EQUIPMENT IDENTIFICATION DESCRIPTION	EQUIPMENT IDENTIFICATION EQUIPMENT NO.	LOCATION BLDG.	LOCATION ELEV.	UNIT NO.	SUPPLIER	TESTING FACILITIES	QUALIFICATION CRITERIA	QUALIFICATION "E1" SIGNED BY:
8856-E-136	Instrument AC Transformers 37.5 kVA, 3Ø 4 Wire, 480-208y/120V	1X-216	Reactor	683	1	Federal Pac. Electric Co.	Wyle Laboratory, Huntsville, Alabama	Proj Spec G-10* & IEEE 344-1975	Report # 45455-1 By: Vincent F. Kearns III
		1X-226		683	1				
		1X-236		719	1				
		1X-246		719	1				
		2X-216		683	2				
		2X-226		683	2				
		2X-236		719	2				
		2X-246		670	2				
	25 kVa, 1Ø 480-120/240 V	1X-201A	Reactor	761	1				
		1X-201B		761	1				
		2X-201A		761	2				
		2X-201B		761	2				
	15 kVa, 1Ø 480-120/240V	0X-507	D. Gen. 'A-D'	710	Comm				
		0X-508		710	Comm				
		0X-509		710	Comm		Qualified by Analysis		
		0X-510		710	Comm				
		0X-512	ESSW Pump-house	685	Comm				
		0X-513		685	Comm				
Spec E-1024	30 kVa, 3Ø 480-480/277	OLX-5B	D. Gen. 'E'	675	Comm	Telemecanique	Wyle Laboratories Huntsville, Alabama	Spec C-1041 & IEEE 344-1975	Telemecanique Report No. SC-665 By: J. D. Owens

* NOTE: Specification G-10 is complemented by Specification G-22.

TABLE 3.10c-15 AUTOMATIC TRANSFER SWITCHES

ITEM NO.	EQUIPMENT IDENTIFICATION DESCRIPTION	EQUIPMENT NO.	LOCATION BLDG.	ELEV.	UNIT NO.	SUPPLIER	TESTING FACILITIES	QUALIFICATION CRITERIA	QUALIFICATION "EI" SIGNED BY:
8856-E-152	Automatic Transfer Switch	0ATS-516	D Gen. 'A-D'	677'	Comm		Russelectric, Inc. Alabama for C.C & D Company Ltd.	Wyle Laboratory, 344-1975	ProjectSpec G-10* & IEEE- By: James W. Foreman Report # 44434-1
		0ATS-526	Gen.	677'	Comm				
		0ATS-536		677'	Comm				
		0ATS-546		677'	Comm				
		1ATS-219	Reactor	670'	1				
		1ATS-229		719'	1				
		2ATS-219	Reactor	670'	2				
		2ATS-229		719'	2				
		0ATS-556	D Gen. 'E'	656'-6"	Comm	Gould/ Telemecanique (TE)	Wyle Laboratory Huntsville, Alabama	Project Spec. C-1041# & IEEE 344-1975	TE Report SC-657, Rev. 1 By: P. Higgins

* NOTE: Specification G-10 is complemented by Specification G-22.
# NOTE: Spec. C 1041 is complemented by Spec. E-1024.

TABLE 3.10c-16

LOAD ISOLATION MOTOR-GENERATOR SETS

ITEM NO.	EQUIPMENT IDENTIFICATION		LOCATION		UNIT NO.	SUPPLIER	TESTING FACILITIES	QUALIFICATION CRITERIA	QUALIFICATION "E" SIGNED BY:
	DESCRIPTION	EQUIPMENT NO.	BLDG.	ELEV.					
8856-E-151	Load Isolation Motor Generator Sets: Motor 150 HP. 3ϕ, 480 V	1S-246 1G-202	Reactor	670' 670'	1 1	Engine Power Co./ Kato Engineering	Wyle Labs and R. W. Siegfried Associates	Project Spec G-10* & IEEE-344-1975	Wyle Report # 58393 58411 By: W. M. West Report #4058 R. W. Siegfried Report #4058
	Generator 100 kv, 3ϕ 480 V	1S-247 1G-203		719' 719'	1 1				
		2S-246 2G-202		670' 670'	2 2				
		2S-247 2G-203		719' 719'	2 2				
	Control Panels for Motor-Generator Sets	1C-246 1C-247		670' 719'	1 1				
		2C-246 2C-247		670' 719'	2 2				

NOTE: Specification G-10 is complemented by Specification G-22.

TABLE 3.10c-17

NON-NSSS AND NON-ACR RELAYS REQUIRED TO BE ENERGIZED
(Units 1 & 2 Devices Are Identical)

Device No.	Relay Function	Location	Manufacturer Type	Operating (Volt) Normal	Minimum	Seismically Type Tested	Remarks
27	Supervise 480 V Auto Transfer Switches	0ATS219, 0ATS229 0ATS516, 0ATS526 0ATS536, 0ATS546	Russel-Electric UV-100/42 480/500	480 ac	432 ac	432 Vac	
27A	Initiates 4 kV Bus Auto Transfer	1A201, 1A202 1A203, 1A204	ABB/West. – SVF31	119	107	24 Vac	
27AI	Permissive to Close 4 kV Incom Breakers	1A201, 1A202 1A203, 1A204	ITE-270	119	107	110 Vac	
43	Transfer Relay for 480 V Auto Transfer Switches 0ATS536, 0ATS546	1ATS219, 1ATS229 0ATS516, 0ATS526	Ward-Leon ARD Bul-130	480 ac	432 ac	432 Vac	
44	Initiation of 4 kV ESF Loads	1A201, 1A202 1A203, 1A204	ABB/West – SSV-T	120 ac	90 ac	90 ac	
59N	Trip ±24 vdc Battery Charger on Overvoltage	1D672, 1D682	GE – NSV	24 dc	28 dc	30 dc	
51V	480 V Swing Bus M-G Set Protection	1C246, 1C247	ABB/West – Cov-9	120 ac	108 ac	80 ac	
62	Time Delay Relay	Various inplant Locations	Agastat 7000 Series	125 dc	105 dc	120 dc	
62	Time Delay Relay	Various inplant Locations	Agastat 7000 Series	120 ac	108 ac	120 ac	
62	Time Delay Relay (480 V Auto Transfer Switches	1AT219, 1AT229 0ATS516, 0ATS526 0ATS536, 0ATS546	Ind. Timer CSF-30M	120 ac	108 ac	120 Vac	
X	Auxiliary Control Relays	1ATS219, 1ATS229 0ATS516, 0ATS526 0ATS536, 0ATS536	Ward – Leon ARD 130-6429	125 dc	105 dc	125 Vdc	
X	Auxiliary Control Relays	1C246, 1C247	GE-HFA	125 dc	105 dc	125 Vdc	

TABLE 3.10c-17

NON-NSSS AND NON-ACR RELAYS REQUIRED TO BE ENERGIZED
(Units 1 & 2 Devices Are Identical)

Device No.	Relay Function	Location	Manufacturer Type	Operating (Volt) Normal	Operating (Volt) Minimum	Operating (Volt) Seismically Type Tested	Remarks
X	Auxiliary Control Relays	1C661A, 1C661B 0C877A, 0C877B 0C876A, 0C876B	GE-HFA	120 ac	108 ac	125 Vdc	
X	Auxiliary Control Relays	1C661A, 1C661B	GE-HMA	120 ac	108 ac	96 ac	
X	Auxiliary Control Relays	1C661A, 1C661B	GE-HMA	125 dc	105 dc	125 Vdc	
X	Auxiliary Control Relays	1C661A, 1C661B 0C578, 0C681 1C681, 0C877A 0C877B, 0C883A 0C883B, 0C876A 0C876B	Agastat-GPI	120 ac	108 dc	96 Vac	
X	Auxiliary Control Relays	0C519A, 0C519B 0C519C, 0C519D 0C519E, 0C521A 0C521B, 0C521C 0C521D, 0C521E	Agastat-GPD	125 dc	105 dc	125 Vdc	
X	Auxiliary Control Relays (prevent cycling of 4 kV Bkr)	1A201, 1A202 1A203, 1A204	ABB/West – AR	125 dc	105 dc	125 Vdc	
X	Auxiliary Control Relays	1A201, 1A202 1A203, 1A204	ABB/West – MG6	125 dc	105 dc	125 Vdc	
X	Isolation Relays	1A201, 1A202 1A203, 1A204	P.B. MDR – 5062/ 5151	125 dc	105 dc	105 dc	(1)
X	Isolation Relays	1C661A, 1C661B 0C877A, 0C877B 0C876A, 0C876B 0C529A, 0C529B	P.B. MDR – 4094/ 4094-1/4165	120 ac	92 ac	90 ac	(1)

Remarks

(1) Test made by Arkansas Unit 1.

TABLE 3.10C-18 - INVERTERS AND 120 VAC INSTRUMENT PANELS

Item No.	Equipment Identification		Location		Unit No.	Supplier	Testing Facilities	Qualification Criteria	Qualification "E1" Signed By:
	Description	Equipment No.	Bldg.	Elev.					
	Inverters 2KVA, 120VAC	1D115 1D125 2D115 2D125	Control	754' 714' 754' 714'	1 1 2 2	General Electric	Wyle Laboratories Novco, California	Project spec G10* and IEEE 344-1975	ADO-01294-1 and -9 signed by D.A. Kneerea
	Instrument Distr. Panels 120VAC, 100A Main Bus	1Y115 1Y125 2Y115 2Y125	Control	754' 714' 754' 714'	1 1 2 2	Eaton Corp.	Wyle Laboratories Novco, California	Project Spec G10* and IEEE 344-1975	45590-1 signed by V.F. Kearns, III

*NOTE: Specification G-10 is complemented by Specification G-22.
For G-10 specification summary, see Table 3.9-31.

3.11 ENVIRONMENTAL DESIGN OF MECHANICAL AND ELECTRICAL EQUIPMENT

Environmental design criteria for the facilities conform to 10CFR50, Appendix A, General Design Criteria 1, 4, and 23. Compatibility of mechanical and electrical equipment with environmental conditions is provided to fulfill the following design criteria:

a) For normal operation, systems and components required to mitigate the consequences of a design basis accident (DBA) or required for a safe shutdown, are designed to remain functional after exposure to the following environmental conditions:

   1) Design temperatures, pressures, and relative humidity values maintained at the equipment location during normal operating by the heating, ventilating, and cooling systems described in Section 9.4.

   2) Maximum expected integrated radiation exposures, for 40 years at the equipment location during normal operation. For service beyond 40 years, the SSES EQ Program is used to manage the aging of equipment to ensure the components continue to perform their intended function.

The environmental conditions expected during normal operation are given in Table 3.11-1 and Dwgs. C-1815, Sh. 1, C-1815, Sh. 2, C-1815, Sh. 3, C-1815, Sh. 4, C-1815, Sh. 5, C-1815, Sh. 6, C-1815, Sh. 7, C-1815, Sh. 8, C-1815, Sh. 9, C-1815, Sh. 10, C-1815, Sh. 11, and C-1815, Sh. 12. The environmental conditions identified in Table 3.11-1 are for the turbine building and for all elevations in the control building except elevation 806'. The environmental conditions in Dwgs. C-1815, Sh. 1, C-1815, Sh. 2, C-1815, Sh. 3, C-1815, Sh. 4, C-1815, Sh. 5, C-1815, Sh. 6, C-1815, Sh. 7, C-1815, Sh. 8, C-1815, Sh. 10, C-1815, Sh. 11, and C-1815, Sh. 12 are for primary containment, the reactor building and elevation 806' of the control structure.

b) In addition to the normal operation environmental requirements listed in a) above, the systems and components required to mitigate the consequences of a DBA, or to effect a safe shutdown of the reactor are designed to remain functional after exposure to the applicable accident environmental conditions. The applicable environmental conditions are those anticipated to follow the DBA that the systems or components are intended to mitigate and are listed below:

   1) Components Inside Containment

      The temperature, pressure, and humidity inside containment after a design basis Loss of Coolant Accident (LOCA) conditions are indicated in Dwgs. C-1815, Sh. 1, C-1815, Sh. 2, C-1815, Sh. 3, C-1815, Sh. 4, C-1815, Sh. 5, C-1815, Sh. 6, C-1815, Sh. 7, C-1815, Sh. 8, C-1815, Sh. 9, C-1815, Sh. 10, C-1815, Sh. 11, and C-1815, Sh. 12.

Containment zones are shown on Dwgs. C-1815, Sh. 1, C-1815, Sh. 2, C-1815, Sh. 3, C-1815, Sh. 4, C-1815, Sh. 5, C-1815, Sh. 6, C-1815, Sh. 7, C-1815, Sh. 8, C-1815, Sh. 9, C-1815, Sh. 10, C-1815, Sh. 11, and C-1815, Sh. 12.

The post-LOCA radiation environment is calculated assuming that 100 percent of the core noble gas inventory, 50 percent of the core halogen inventory, and 1 percent of the core solid fission product inventory are released. The calculational method is in accordance with NUREG 0588, Rev. 1, Section 1.4 and appendix D. The total calculated post-accident dose is the integrated dose from the time of the LOCA to 180 days.

2) Components Outside Containment

The expected temperature, pressure and humidity conditions are specified in Table 3.11-1 and Dwgs. C-1815, Sh. 1, C-1815, Sh. 2, C-1815, Sh. 3, C-1815, Sh. 4, C-1815, Sh. 5, C-1815, Sh. 6, C-1815, Sh. 7, C-1815, Sh. 8, C-1815, Sh. 9, C-1815, Sh. 10, C-1815, Sh. 11, and C-1815, Sh. 12.

In computing the expected integrated accident doses for equipment in contact with or in proximity to Emergency Core Cooling System (ECCS) - water, it is assumed that 50 percent of the core halogen inventory and 1 percent of the core solid fission product inventory are diluted by the Reactor Coolant System water plus the suppression pool water after a design basis LOCA. For equipment located remotely from ECCS water, the appropriate accidental release is assumed.

3.11.1 EQUIPMENT IDENTIFICATION AND ENVIRONMENTAL CONDITIONS

Class 1E safety-related equipment is installed in accordance with mechanical and electrical separation requirements and designed and qualified in accordance with the provisions of IEEE 323-1971 and 1974, with appropriate margins, to function properly in the environments listed.

An Environmental Qualification Master Equipment List (EQMEL) is maintained for the SSES equipment which requires environmental qualification through the current Procedure.

1) required to detect a steam or water line accident condition;
2) required to perform a steamline isolation function;
3) required to perform a water line isolation function and could be subjected to the steam environment such as electrical cable or valve operator;
4) required for safety system operation and is located so a steamline break in some other system exposes the safety system equipment to the local accident environment; and,
5) required to track the post-accident environment condition such as pressure, temperature and radiation monitors.

Electrical switchgear and motor control centers required for safety system operation are located outside of the drywell accident environment to ensure operation.

Harsh environments may arise in primary and secondary containments as a result of a Loss of Coolant Accident (LOCA) inside primary containment. In addition, harsh environments may arise in localized areas outside primary containment as a result of a High Energy Line Break (HELB). The environmental conditions (both normal and maximum) to which Class 1E equipment is exposed are given in Table 3.11-1 and Dwgs.
C-1815, Sh. 1, C-1815, Sh. 2, C-1815, Sh. 3, C-1815, Sh. 4,
C-1815, Sh. 5, C-1815, Sh. 6, C-1815, Sh. 7, C-1815, Sh. 8,
C-1815, Sh. 9, C-1815, Sh. 10, C-1815, Sh. 11, and C-1815, Sh. 12.

The nonseismic vibration of safety-related equipment conforms to requirements of the following standards:

Equipment	Standard
Diesel Fuel Oil Transfer Pumps	Hydraulics Institute Standards
RHR Service Water Pumps	Hydraulics Institute Standards
Emergency Service Water Pumps	Hydraulics Institute Standards
Control Structure Chilled Water Pumps	Hydraulics Institute Standards
All other Safety-Related Pumps	API 610, Section IV or better
HVAC Fans	ASHRAE Systems Handbook 1973 Edition, Chapter 35, Page 24
Diesel Generator Engines	DEMA Standard Practices for low and medium speed stationary diesel and gas engines
Electric Motors	NEMA MGI

The absence of any significant nonseismic vibration caused by pipe vibration interaction with above equipment is verified in Subsection 3.9.2.1.

A containment spray system may be utilized following the LOCA; therefore, exposed safety-related systems located in the containment are designed to withstand the effects of the containment spray.

3.11.2 QUAIIFICATION TEST AND ANAIYSIS

The qualification tests conform to the requirements of Appendix B, Section XI of 10CFR50.

For the Class 1E equipment, qualification tests and analysis performed on electrical equipment, including motors, are maintained in an auditable manner as discussed in Section 3.11.3. Class 1E equipment installed at SSES is subject at a minimum to the requirements of NUREG-0588, Category II as detailed in Section 3.11.2a.1. Certain new or replacement Class 1E equipment installed at SSES is subject at a minimum to the requirements of IEEE Standard 323-1974 as detailed in Section 3.11.2a.2.

3.11.2.1 CLASS 1E EQUIPMENT QUALIFIED TO IEEE STANDARD 323-1971

The original Class 1E equipment at SSES have been qualified at a minimum to NUREG-0588, Category II and IEEE Standard 323-1971. This is because SSES received its construction permit before July 1, 1974. Nuclear power plants which received their construction permits after

July 1, 1974 are required to qualify the Class 1E equipment to NUREG-0588, Category I and IEEE Standard 323-1974. New or replacement equipment purchased after May 23, 1980 is qualified to the requirements of 10CFR50.49 (NUREG-0588, Category I; IEEE Standard 323-1974) except in cases where sound reasons to the contrary have been established per the guidelines of Regulatory Guide 1.89.

3.11.2.2 CLASS 1E EQUIPMENT QUALIFIED TO IEEE STANDARD 323-1974

A large number of pieces of Class 1E equipment at SSES are qualified to NUREG-0588, Category I and to IEEE Standard 323-1974. Qualification to IEEE Standard 323-1974 requires type testing of a prototype to demonstrate that the equipment will perform its safety function in the combined temperature, pressure, humidity, chemical and radiation environment.

Equipment qualified to IEEE 323-1974 is qualified to the test sequence and margins specified in IEEE 323-1974 unless justification for using another sequence or other margins is provided in the SSES EQ documentation.

3.11.2.3 Class 1E Component Environment Design and Qualification for Normal Operation

Class 1E equipment is designed for 40 years of continuous operation in the most severe temperature, pressure, humidity, and radiation environments that exist at the equipment location during normal operation, assuming that proper routine preventive maintenance is performed, such as periodic replacement of seals, packing, and consumable materials. For service beyond 40 years, the SSES EQ Program is used to manage the aging of equipment to ensure the components continue to perform their intended function.

Table 3.11-1 and Dwgs. C-1815, Sh. 1, C-1815, Sh. 2, C-1815, Sh. 3, C-1815, Sh. 4, C-1815, Sh. 5, C-1815, Sh. 6, C-1815, Sh. 7, C-1815, Sh. 8, C-1815, Sh. 9, C-1815, Sh. 10, C-1815, Sh. 11, and C-1815, Sh. 12 provide the maximum normal and maximum DBE values for the environmental parameters of temperature, pressure, and humidity for each area in which Class 1E safety-related equipment is located, as well as the exposures to radiation.

For most equipment, special qualification tests to verify operability at normal operating temperature, pressure, and humidity conditions are not required. Verification for this equipment is based on proven operating capability in similar environments in industrial and previous nuclear power plant applications. The pre-operational and post-operational test programs for safety-related components further ensure that safety-related components will be available when required. Since the normal and accident integrated radiation doses have cumulative effects, the integrated radiation dose during normal operation is discussed in Subsection 3.11.5.3.

3.11.2.4 Class 1E Component Environmental Design and Qualification for Operation After a Design Basis Accident

Class 1E safety-related equipment is designed to remain functional in the most severe combination of temperature, pressure and humidity conditions that exist at the equipment location after a design basis LOCA. This equipment is also designed for the maximum calculated integrated radiation exposure of the LOCA or of the accident, as discussed in Subsection 3.11.5. The temperature, pressure, and humidity environment inside the primary containment after a LOCA is presented and discussed in detail in Subsection 6.2.1. The integrated post accident radiation dose for plant locations in which the equipment is located is given in Table 3.11-1 and Dwgs. C-1815, Sh. 1, C-1815, Sh. 2, C-1815, Sh. 3, C-1815, Sh. 4, C-1815, Sh. 5, C-1815, Sh. 6, C-1815, Sh. 7, C-1815, Sh. 8, C-1815, Sh. 9, C-1815, Sh. 10, C-1815, Sh. 11, and C-1815, Sh. 12. In addition, possible steam and feedwater line breaks outside the containment are analytically checked to ensure that no additional qualifications need be applied to components that could be affected by these breaks.

The requirements of the general design criteria, 1, 4, and 23 of Appendix A to 10CFR50, are met as discussed in Section 3.1.

The recommendations contained in the regulatory guides listed below (listings a) through g)) have been utilized as described in Section 3.13. Additional discussion is included in listings f) through j).

a) Regulatory Guide 1.30, Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electrical Equipment.

b) Regulatory Guide 1.40, Qualification Test of Continuous Duty Motors Installed Inside the Containment of Water Cooled Nuclear Power Plants.

   Continuous duty motors used inside the containment are type tested under simulated LOCA conditions. IEEE Standard 334-1971 is used. Insofar as practicable, auxiliary equipment that is part of the installed motor assembly is likewise qualified in accordance with IEEE Standard 334 under simulated design bases event conditions.

c) Regulatory Guide 1.63, Electrical Penetrations Assemblies in Containment Structures for Water Cooled Nuclear Plants.

   Electrical containment penetrations are tested in accordance with IEEE Standard 317-1972. Refer to Section 8.1 for discussion on this guide.

d) Regulatory Guide 1.73, Qualification Test of Electric Valve Operator, Installed Inside the Containment of Nuclear Power Plants.

   Motor operated valves used inside the containment are type tested as a minimum in accordance with IEEE 382-1972 (ANSI N41.6).

e) Regulatory Guide 1.89, Qualification of Class 1E Equipment for Nuclear Power Plants.

The qualification methods and documentation requirements of IEEE 323-1971 are discussed in Section 3.11.

f) Regulatory Guide 1.97, Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident, Rev. 2, December 1980.

g) Regulatory Guide 1.131, Qualification Tests of Electric Cables, Filed Splices, and Connections for Light-Water-Cooled Nuclear Power Plants, August 1977.

h) Type tests to ensure acceptability for use in the containment post accident environment are performed for each type of cable in accordance with IEEE Standard 383-1974.

i) Pressure boundary components inside the containment are designed for the temperature, pressure, and humidity environment in accordance with the applicable codes to which the component is constructed. Qualification testing is not considered necessary for such components.

j) A total (normal plus accident) integrated dose of less than 10^4 rad will not affect the strength or properties of materials used; hence, further qualification analyses and tests for components which will be exposed to less than 10^4 rad are not necessary. However, certain electronic equipment such as metal oxide semi-conductive devices are sensitive to radiation levels of less than 10^4. Therefore, radiation qualification is evaluated on a case by case basis even when the postulated accident dose is less than 10^4. For higher integrated doses, components are qualified either by qualification testing or by evaluation of materials used. Reliable accumulated data on radiation effects, such as contained in Reference 2, is used to analyze the dose effects of particular materials.

k) The sources used in calculating radiation levels following LOCA are consistent with those set forth in NUREG-0588, Rev. 1. All the active non-NSSS safety-related equipment located inside the primary containment is designed to withstand the maximum integrated doses during the life of the plant. Suitability of materials used is verified by test for all electrical penetration assembly materials, for an integrated dose rate of at least 5 x 10^7 rad, in accordance with requirements of IEEE 317-1972.

l) The materials used in the fabrication of the reactor coolant system pressure boundary and other mechanical and structural components are selected to minimize corrosion and hydrogen generation.

3.11.3 QUALIFICATION TEST RESULTS

Environmental qualification documentation for Class 1E electrical equipment is contained in EQ binders maintained by Susquehanna Nuclear Records in File R34. In some cases, additional details of test results for GE safety-related equipment are maintained in a permanent file by GE and can be readily audited. In all cases, the equipment used in Class 1E applications passed the prescribed tests. Dwgs. C-1815, Sh. 1, C-1815, Sh. 2, C-1815, Sh. 3, C-1815, Sh. 4, C-1815, Sh. 5, C-1815, Sh. 6, C-1815, Sh. 7, C-1815, Sh. 8, C-1815, Sh. 9, C-1815, Sh. 10, C-1815, Sh. 11, and C-1815, Sh. 12, identify the temperature pressure, humidity, and radiation environments to which the Class 1E equipment has been qualified.

3.11.4 LOSS OF VENTILATION

The maximum temperatures considered in the sizing of air conditioning systems serving safety-related systems are determined by additive analysis of the following factors:

a) Maximum outdoor design temperature for the geographical area of the plant (both wet bulb and dry bulb readings)

b) Maximum internal piping thermal loads, if applicable, for the room, using maximum normal operating temperatures for the pipe contents and maximum footage of active pipe for each mode of operation.

c) Maximum internal electrical load assuming full lighting for the room, and using, if applicable, the maximum control and equipment resistance losses for each mode of operation.

d) Maximum heat transfer from miscellaneous equipment surfaces, if applicable (e.g., outer surface of the diesel generator).

e) Maximum heat transfer from the surface of open pools and tanks, if applicable, using the maximum operating temperature of the contents.

f) Maximum heat transfer from the room envelope including walls, floor, and ceiling, or roof (this value may be negative).

Seismic Category I air conditioning and air cooling systems, described in Section 9.4 are powered from the Class 1E electrical power supplies and are provided for the locations listed in Dwgs. C-1815, Sh. 1, C-1815, Sh. 2, C-1815, Sh. 3, C-1815, Sh. 4, C-1815, Sh. 5, C-1815, Sh. 6, C-1815, Sh. 7, C-1815, Sh. 8, C-1815, Sh. 9, C-1815, Sh. 10, C-1815, Sh. 11, and C-1815, Sh. 12.

These Category I systems are designed so that the single failure of an active mechanical component, or an active or passive electrical component, after a DBA, cannot impair the ability of the systems served by the air conditioning equipment to fulfill their safety functions. Should a train in a Seismic Category I air conditioning system become inoperative during normal operation, sufficient equipment is still available to mitigate the consequences of a DBA.

Two redundant Seismic Category I emergency air conditioning trains are provided for the control room.

Power cable is rated for a conductor temperature of 194°F (90°C). Class 1E cables are qualified for the plant specific worst case temperatures expected during normal operation and post accident conditions by testing and analysis. The allowable current carrying capacity of the cable is based on not exceeding this insulation design temperature while the surrounding air is at an ambient temperature of 150°F (65.5°C) inside the containment and for the rest of the plant area a temperature of 122°F (50°C) or 104°F (40°C) depending on location. The cable ampacity is determined as discussed in Section 8.3.3.1.

Instrumentation cable is rated for a conductor temperature of 194°F (90°C). Class 1E cables are environmentally qualified for the plant specific worst case temperatures expected during normal operation and post accident conditions by testing and analysis. Operating currents of these cables are low and will not cause this temperature to be exceeded at maximum design ambient temperature. Instruments required to operate following a DBA are not located in pipe tunnels, but are mounted outside the tunnels.

3.11.5 ESTIMATED CHEMICAL, PHYSICAL, AND RADIATION ENVIRONMENT

3.11.5.1 Suppression Pool, Residual Heat Removal System, and Emergency Core Cooling System Water Quality

The water in these systems shall not be chemically inhibited. The maximum limits for the suppression pool have been established to be compatible with the primary coolant limits for the shutdown condition and are listed in Table 3.11-7 for comparison.

Observations made of suppression pool water quality over a period of several years in suppression pool with and without coatings, indicate that the feed and bleed to radwaste that occurs during normal system testing and level adjustments maintain the water quality well within the above limits.

During reactor shutdown cooling, the RHR system water mixes with reactor water. Therefore, to insure reactor water quality, as much as practicable of the shutdown cooling piping and equipment shall be flushed with either reactor water or water of the quality specified above for maximum limit.

3.11.5.2 Physical Environment

Engineered safety feature (ESF) systems are designed to perform their safety-related functions in the temperature, pressure, and humidity conditions described in Subsection 3.11.2, and Sections 6.2 and 6.3.

The containment atmosphere is maintained below 4 percent by volume hydrogen consistent with the recommendations of Regulatory Guide 1.7 as discussed in Subsection 6.2.5.

3.11.5.3 Radiation Environment

ESF systems and components are designed to perform their safety-related functions after the normal operational exposure plus an accident exposure. The normal operational exposure is based on the design source terms presented in Chapter 11 and Subsection 12.2.1 and the equipment and shielding configurations presented in Section 12.3.

Post-accident ESF system and component radiation exposures are dependent on equipment location. In the containment and control room area, exposures are due to a hypothesized LOCA. Source terms and other accident parameters are consistent with the recommendations of NUREG 0588, Rev. 1.

In the reactor building, post-accident exposures to ECCS systems recirculating depressurized reactor fluids are based on source terms assuming an inventory of 50 per cent of the core halogens and one percent of the core solid fission products. Post Accident exposures to the High-Pressure Coolant Injection and Reactor Core Isolating Cooling Systems are based on steam source terms of control rod drop accident as described in Section 15.4.9. The minimum exposure within the Reactor Building is based on airborne sources originating from one per cent per day drywell leakage of post-accident airborne activity.

Normal, accident, and design (normal plus accident) radiation exposures for plant areas, based on the above assumptions, are presented in Table 3.11-1 and Dwgs. C-1815, Sh. 1, C-1815, Sh. 2, C-1815, Sh. 3, C-1815, Sh. 4, C-1815, Sh. 5, C-1815, Sh. 6, C-1815, Sh. 7, C-1815, Sh. 8, C-1815, Sh. 9, C-1815, Sh. 10, C-1815, Sh. 11, and C-1815, Sh. 12.

Organic materials that exist within the containment are identified in Subsection 6.1.2. The design radiation exposures identified in Table 3.11-1 and Dwgs. C-1815, Sh. 1, C-1815, Sh. 2, C-1815, Sh. 3, C-1815, Sh. 4, C-1815, Sh. 5, C-1815, Sh. 6, C-1815, Sh. 7, C-1815, Sh. 8, C-1815, Sh. 9, C-1815, Sh. 10, C-1815, Sh. 11, and C-1815, Sh. 12 are based on gamma radiation exposure only except as noted. The attenuation of beta radiation by small amounts of shielding, such as conduits and jackets for cable and casings for equipment, is evaluated. Where such shielding is not completely effective, the equipment is qualified for radiation exposures which include the appropriate portion of the postulated beta TID.

3.11.6 REFERENCES

3.11-1. J.J. DiNunno, R.E. Baker, F.D. Anderson, and R.L. Waterfield, "Calculation of Distance Factors for Power and Test Reactor Sites", TID-14844, Division of Licensing and Regulation, AEC, Washington, D.C. (1962).

3.11-2. J.F. Kircher and R.E. Bowman, "Effects of Radiation on Materials and Components", Van Nostrand Reinhold, New York, 1964.

Table Rev. 55

TABLE 3.11-1

NORMAL AND MAXIMUM PLANT ENVIRONMENTAL CONDITIONS

ABEA	KEY (3)	PRESSURE	NORMAL OPERATING CONDITIONS					MAXIMUM CONDITIONS WITH NUREG-0588 SOURCE TERM			
			TEMP °F MAX/MIN	RELATIVE HUMIDITY MAX/MIN %	DOSE RATE (R/HR) (4)	INTEGRATED DOSE (RAD)	PRESSURE (2)	TEMP °F (2)	RELATIVE HUMIDITY % (2)	LOCA DOSE RATE (RAD/HR) (6) (7)	TOTAL INTEGR. DOSE (1) (RAD) (6) (7)
Control Room	CS1	+.125" wg	80 / 70	55 / 45	.0005	$1.8X10^2$	+.125" wg	80	55	<1.0	$1.8X10^2$
Cable Spreading Rooms & HVAC Equipment Room, Relay Rooms Elect. Equip. Rooms	CS2	+.125" wg	80 / 60	60 / 10	.0005	$1.8X10^2$	+.125" wg	90	60	<1.0	$1.8X10^2$
Battery Room	CS5	+.125" wg	80 / 60	60 / 10	.0005	$1.8X10^2$	+.125" wg	80	60	<1.0	$1.8X10^2$
Computer Room	CS3	+.125" wg	85 / 65	60 / 40	.0005	$1.8X10^2$	+.125" wg	85	60	<1.0	$1.8X10^2$
Diesel Generator 'A-D' Rooms (5)	G	Atmos	104 / 72	90 / 5	.0005	$1.8X10^2$	Atmos	120	50	<1.0	$1.8X10^2$
ESW Pumphouse	SW	Atmos	104 / 40	100 / 5	.0005	$1.8X10^2$	Atmos	104	100	<1.0	$1.8X10^2$
UPS Rooms	CS3	+.125" wg	104 / 65	60 / 40	.0005	$1.8X10^2$	+.125" wg	104	60	<1.0	$1.8X10^2$
Turbine Building Operating Floor	T2a	Atmos	104	90 / 10	.0025	$8.8X10^2$	-.125" wg	104	90	≤1.0	$8.8X10^2$
Diesel Generator E Building (5)	G	Atmos	104 / 72	90 / 5	.0005	$1.8X10^2$	Atmos	120	50	≤1.0	$1.8X10^2$
Turbine Building General Areas (Shielded)	T1	-.125" wg	104	90 / 10	.0025	$8.8X10^2$	-.125" wg	104	100	≤1.0	$8.8X10^2$
HP Turbine	T2b	-.125" wg	-	-	.5	$1.8X10^4$	-.125" wg	-	-	≤1.0	$1.8X10^3$

Table Rev. 55

TABLE 3.11-1

NORMAL AND MAXIMUM PLANT ENVIRONMENTAL CONDITIONS

			NORMAL OPERATING CONDITIONS					MAXIMUM CONDITIONS WITH NUREG-0588 SOURCE TERM			
ABEA	KEY (3)	PRESSURE	TEMP °F MAX/MIN	RELATIVE HUMIDITY MAX/MIN %	DOSE RATE (R/HR) (4)	INTEGRATED DOSE (RAD)	PRESSURE (2)	TEMP °F (2)	RELATIVE HUMIDITY % (2)	LOCA DOSE RATE (RAD/HR) (6) (7)	TOTAL INTEGR. DOSE (1) (RAD) (6) (7)
LP Turbine	T2c	-.125" wg	-	-	.1	$3.5X10^4$	-.125" wg	-	-	≤1.0	$3.5X10^4$
Feedwater Heaters Condensers	T3	-.125" wg	120	90 / 10	5	$1.8X10^6$	-.125" wg	120	100	≤1.0	$1.8X10^6$
Steam Jet Air Ejectors	T4	-.125" wg	120	90 / 10	15	$5.3X10^6$	-.125" wg	120	100	≤1.0	$5.3X10^6$
Condensate Treatment	T5	-.125" wg	120	90 / 10	10	$3.5X10^6$	-.125" wg	120	100	≤1.0	$3.5X10^6$

(1) Includes integrated accident and normal dose TID for 180 days. After 180 days, the TID is essentially saturated and will not increase significantly.
(2) Pressure, temperature, and humidity maximum are not simultaneous. Above normal pressure, temperature and humidity are considered to persist for 100 days. After 100 days, the thermal environment will be equal to or less than the "maximum" given for normal operation.
(3) Key letter and number identifies a particular group of environmental parameters.
(4) If not otherwise noted, dose is gamma.
(5) For DG rooms: Normal operation means DG in Standby, maximum condition means DG operating.
(6) Maximum Condition Dose rates and TIDs are maximum contact doses in each room, and specific equipment may be subject to a reduced dose based upon the appropriate attenuation factors.
(7) For Beta Sensitive equipment only, the post-accident airborne Beta doses shown in this note must be corrected with appropriate attenuation factors and then added to the tabulated gamma doses to determine total TID.

Area	Max Beta Dose Rate (R/HR)	Beta TID (RAD)
Control Building	2.0	$1.0X10^2$
Turbine Building	20.0	$1.0X10^3$

TABLE 3.11-7

WATER QUALITY

PARAMETER	REACTOR WATER LIMITS SHUTDOWN CONDITION	PRESSURE SUPPRESSION POOL WATER QUALITY EXPECTED	SUPPRESSION POOL WATER MAXIMUM LIMIT
Conductivity	≤ 10 μs/cm at 25°C	≤ 5 μs/cm at 25°C	≤ 10 μs/cm at 25°C
Chlorides (as C1)	≤ 0.5 ppm	≤ 0.2 ppm	≤ 0.5 ppm
pH	5.3 to 8.6 at 25°C	6.0 to 7.5 at 25°C	5.3 to 8.6 at 25°C

THIS FIGURE HAS BEEN
REPLACED BY DWG.
C-1815, Sh. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

Figure 3.11-1 replaced by dwg.
C-1815, Sh. 1

FIGURE 3.11-1, Rev. 56

AutoCAD Figure 3_11_1.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
C-1815, Sh. 2

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Figure 3.11-2 replaced by dwg. C-1815, Sh. 2 FIGURE 3.11-2, Rev. 51

AutoCAD Figure 3_11_2.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
C-1815, Sh. 3

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Figure 3.11-3 replaced by dwg. C-1815, Sh. 3 FIGURE 3.11-3, Rev. 51

AutoCAD Figure 3_11_3.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
C-1815, Sh. 4

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

Figure 3.11-4 replaced by dwg.
C-1815, Sh. 4

FIGURE 3.11-4, Rev. 56

AutoCAD Figure 3_11_4.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
C-1815, Sh. 5

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Figure 3.11-5 replaced by dwg. C-1815, Sh. 5 FIGURE 3.11-5, Rev. 56

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THIS FIGURE HAS BEEN
REPLACED BY DWG.
C-1815, Sh. 6

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

Figure 3.11-6 replaced by dwg.
C-1815, Sh. 6

FIGURE 3.11-6, Rev. 56

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THIS FIGURE HAS BEEN
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C-1815, Sh. 7

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

Figure 3.11-7 replaced by dwg.
C-1815, Sh. 7

FIGURE 3.11-7, Rev. 56

AutoCAD Figure 3_11_7.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
C-1815, Sh. 8

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

Figure 3.11-8 replaced by dwg.
C-1815, Sh. 8

FIGURE 3.11-8, Rev. 56

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THIS FIGURE HAS BEEN
REPLACED BY DWG.
C-1815, Sh. 9

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Figure 3.11-9 replaced by dwg. C-1815, Sh. 9 FIGURE 3.11-9, Rev. 56

AutoCAD Figure 3_11_9.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
C-1815, Sh. 10

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Figure 3.11-10 replaced by dwg. C-1815, Sh. 10 FIGURE 3.11-10, Rev. 58

AutoCAD Figure 3_11_10.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
C-1815, Sh. 11

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

Figure 3.11-11 replaced by dwg.
C-1815, Sh. 11

FIGURE 3.11-11, Rev. 56

AutoCAD Figure 3_11_11.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
C-1815, Sh. 12

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

Figure 3.11-12 replaced by dwg.
C-1815, Sh. 12

FIGURE 3.11-12, Rev. 56

AutoCAD Figure 3_11_12.doc

Appendix 3.11A has been deleted.

3.12 SEPARATION CRITERIA FOR SAFETY RELATED MECHANICAL AND ELECTRICAL POWER EQUIPMENT

3.12.1 INTRODUCTION

This section describes the various separation criteria utilized in the design of mechanical and electrical safety related systems and auxiliary support systems; outside of the NSSS scope delineated in Section 7.1

3.12.2 MECHANICAL SYSTEMS

The mechanical safety related auxiliary support and safety related systems to which the separation criteria apply are identified in Table 3.12-1 and Table 3.12-2. Mechanical descriptions of the systems covered by this section are given in Chapters 6 and 9.

3.12.2.1 Criteria

3.12.2.1.1 General Criteria

Redundant systems are separated from each other so that single failure of a component or channel will not interfere with the proper operation of its redundant/diverse counterpart.

The affected mechanical systems and equipment are separated so that systems important to safety are protected from the following hazards:

a) The pipe break dynamic effects outlined in Section 3.6.

b) Environmental effects as a result of pipe breaks and as outlined in Section 3.11

c) Flooding effects as a result of pipe breaks and as outlined in Section 3.4.

d) Missiles as defined in Section 3.5.

e) Fires capable of damaging redundant mechanical safety equipment.

The need for adequacy of separation to protect the safety equipment from the above hazards are determined in conjunction with the criteria specified in Sections: 3.4 (flood protection), 3.5 (missile protection), 3.6 (pipe rupture), 3.11 (environmental design), and the Fire Protection Review Report.

3.12.2.1.2 Svstem Separation Criteria

Piping from a redundant safety system is run independently of its counterparts, unless it can be shown that no single credible event, e.g., LOCA, is capable of causing piping failure that could prevent reactor shutdown. Supports and restraints of redundant mechanical components and piping are not shared, unless such sharing does not significantly impair their ability to perform their safety function.

Penetrations to the primary containment are separated or other adequate provisions are made so that the initial break of one piping branch of a system does not render its redundant counterpart(s) inoperable.

3.12.2.1.3 Phvsical Separation Criteria

Mechanical equipment and piping are separated from each other so that single failure of a device or component will not interfere with the proper operation of its redundant counterpart.

3.12.2.2 Separation Techniques

The methods used to protect redundant Auxiliary Support Systems from the above hazards (Subsection 3.12.2.1.1) fall into four categories of separation techniques: plant arrangement, barriers, spatial separation, and alternatives.

a) Plant Arrangement

A basic design consideration of plant layout is that redundant divisions of a safety system should not share common equipment areas. However, equipment common to a particular safety system division can share a common area if that equipment does not constitute a hazard within itself to another safety system of the same division.

Failure of any non-safety related structure, system, or component shall not result in failure of any safety related structures, system, or component.

To accomplish Auxiliary Support Systems separations through plant arrangement, redundant division of a safety system may be placed in different compartments or even on different elevations. Non-safety equipment, components, or piping should not be run above safety equipment unless they are adequately restrained or it can be demonstrated that failure will not impair function of the safety equipment.

b) Barriers

Barriers are most often used in restricted areas where a particular hazard (e.g., small turbine missiles) is more easily identified or where other techniques are inappropriate (e.g., separation between control boards). Separation by barriers is an extension of separation by the use of compartments in plant arrangement. Separation was also accomplished through the use of suitably designed equipment that in itself acts as a barrier. In many cases, the barrier may enclose the hazard (e.g., a compartment around a high speed turbine driven pump) in lieu of effecting a direct separation between redundant systems.

c) Spatial Separation

Spatial separation is another method of separating redundant safety systems and protecting them from the hazards described in Subsection 3.12.2.1.1.

For example, in areas where a barrier would be impractical, piping has been rerouted so that jet impingement resulting from a break would be dissipated by the distance traveled. In this example, partial barriers or restraints could also be used, as well as by hardening design (e.g., heavier housing construction) of system components within the hazard area. When it can be shown that a hazard would have only a certain sphere of effectiveness (e.g., for pipe whip, a rotation about a plastic hinge at the next restraint), spatial separation was considered adequate.

d) Alternatives

When one of the above techniques is impractical, a suitable alternative was used, some of which were additional restraints, hardening design, or temporary system isolation under accident conditions. When the redundant safety component cannot be held safe from common hazards by the alternatives outlined above, more resistant components were selected. An example would be the use of high pressure piping in a low pressure safety system to ensure its ability to withstand the effect of a break in adjacent high pressure lines.

3.12.3 ELECTRICAL SYSTEMS AND EQUIPMENT SEPARATION CRITERIA

Electrical and actuation systems are described in detail in Chapters 8 and 7, respectively.

3.12.3.1 Affected Svstems

The electrical portions of the systems identified in Tables 3.12-1 and 3.12-2 are designed to the criteria of Subsection 3.12.3.2. Equipment covered by the requirements in this Subsection include: instrument channels, trip systems, and trip actuators.

3.12.3.2 General Criteria

The resulting installations satisfy the criteria of IEEE 279-1971, 10CFR50 Appendix A, General Design Criteria 3, 17, and 21, as further clarified and limited below. The affected electrical systems and equipment are separated such that systems important to safety are protected from the following hazards:

a) Fires in cable raceways due to an electrical fault that could cause failure of insulation on other cables.

b) Mechanical damage of electrical equipment in a single location.

c) Single Design Base Event (DBE) should not disable essential automatic or manual protective function, i.e., reactor scram, primary containment isolation, core cooling, etc.

Identification

Identification and division/channels conform to the following:

a) Panels and racks, not part of the PGCC, are labeled with distinctive marker plates. The marker plates include identification of the proper division/channel, as listed in Table 3.12-1.

b) Junction and/or pull boxes, not part of the PGCC, have identification similar to and compatible with the panels and racks considered above.

c) Cables external to cabinets and/or panels, not part of the PGCC, are marked to distinguish them in color from other cables and to identify their separation division/channel as applicable.

d) Raceways, not part of the PGCC, identified as described in Subsection 3.12.3.4.2.1.b.

e) For PGCC panels and racks refer to Section 7.1.2a.3.

f) For cables external to panels and racks but within the PGCC, refer to Section 7.1.2a.3.

3.12.3.3 Svstem Separation Criteria

See Section 7.1.2a.3.

3.12.3.4 Electrical Phvsical Separation Criteria

3.12.3.4.1 General Separation Criteria

Methods of Separation

The separation of circuits and equipment is achieved by separate safety class structures, distance, or barriers, or any combination thereof.

Compatibilitv with Mechanical Svstems

Class 1E circuits are routed and/or protected such that failure of related mechanical equipment of one redundant system cannot disable Class 1E circuits or equipment essential to the operation of the other redundant system(s).

Racewav Sharing of Class 1E and non-Class 1E Circuits

See Subsection 8.1.6.1.

3.12.3.4.2 Specific Separation Criteria

3.12.3.4.2.1 Cables and Racewavs

a) General

The minimum separation distances specified in paragraphs d) and e) are based on open ventilated trays. Where these distances are used to provide adequate physical separation:

1) Cable splices in raceways are prohibited
2) Cables and raceways involved are flame retardant
3) The design basis is that the cable trays will not be filled above the side rails
4) Hazards will be limited to failures or faults internal to the electric equipment or cables.

In areas where the raceway separation criteria cannot be met due to physical limitation, each circuit in the raceway is to be analyzed to assure that the Class 1E function is not degraded to an unacceptable level. The specific analysis of each case is documented by a controlled document.

b) Identification of Non-PGCC Cables and Racewavs

Exposed Class 1E raceways are identified in a distinct and permanent manner at intervals not to exceed 15 ft. In addition, these raceways are also identified where

they pass through walls and/or floors, and enclosed areas. Class 1E raceways are identified prior to the installation of their cables.

Cables installed in these raceways are identified with the separation group color at intervals not exceeding 5 ft. to facilitate initial verification that the installation conforms to the separation criteria. These cable identifications are applied prior to or during their installation.

Class 1E cables are identified by a permanent marker at each end in accordance with the design drawings or cable schedule.

Color coding is used to meet the above requirements and to distinguish between redundant Class 1E cables and non-Class 1E cables.

c) Identification of PGCC Cables and Racewavs

Refer to Subsection 7.1.2a.3.2.

d) Cable Spreading Areas/Control Structure Complex

The control structure complex consists of two elevations of relay rooms, two cable spreading areas, and the main control room. Below the main control room is the lower cable spreading room, which facilitates cable convergence from the computer room and the lower relay rooms (which are below the lower cable spreading room) to the general plant areas, and to the cable entrance areas at the bottom of the control room panels. The lower relay rooms consist mainly of control and instrument panels of non-Class 1E systems and one division (i.e., Division II) of redundant systems as listed in Subsection 3.12.3.1. The main control room panels are mounted on a raised floor assembly with cable trays and wireways that enter the bottom of the main control room panels. Above the main control room are the upper relay rooms and the upper cable spreading area. The upper cable spreading area facilitates cable convergence from the upper relay room to the general plant areas, to the top of the main control room panels and to the control room raised floor. The upper relay room consists mainly of control and instrument panels of non-Class 1E systems and the other division (i.e., Division I) of the redundant systems listed in Subsection 3.12.3.1. The relay room panels and cabinets are integrated with a module type floor assembly with lateral and longitudinal ducts that act as raceways and barriers. The cabling interface between the PGCC and the spreading area is made at termination cabinets on the periphery of the relay room floor assemblies.

The relay rooms and spreading room areas do not contain high energy equipment (such as switchgear, transformers) or potential sources of missiles or pipe whip and are not used for storing flammable materials.

Circuits in the relay room and main control room are limited to control functions, instrument functions, and those power supply circuits and facilities serving the main control room and instrument systems.

Where for operational reasons redundant channel/division Class 1E cables are not separated by different safety class structures (e.g., two relay rooms and spreading areas), the minimum separation distance between the redundant Class 1E cable trays is 1 ft. horizontally and 3 ft. vertically. Where 1 ft. horizontal separation is not possible, one of the two following requirements is met: a fire barrier is placed between the redundant cable trays 1 ft. above the trays or to the ceiling; or cables of each channel/division are installed in rigid steel conduit or totally enclosed raceway up to a point where the 1 ft. spacing requirement is met. Where cables of redundant channel/divisions must be stacked one above the other with less than 3 ft. vertical spacing, one of the following requirement is met: a) a fire barrier is placed between the trays and extended to 6 in. of each side of the tray system or to the wall, or b) a solid steel tray cover is installed on the lower cable tray and the upper tray has a solid bottom up to a point where 3 ft. vertical separation is met; or c) the cables of each redundant channel/division are installed in rigid steel conduit or totally enclosed raceway to a point where the 3 ft. vertical separation exists. The minimum separation distance between these rigid steel conduit and totally enclosed raceway is 1 inch, except as noted in Section 8.1.6.1 (Regulatory Guide 1.75 (1/75), Part 7).

Separation requirements between Class 1E trays and non-Class 1E trays are the same as separation of redundant channel/division, except that the minimum separation distance between Class 1E tray and non-Class 1E conduit or totally enclosed raceway is 1 inch.

Free air temporary cables can be installed with no separation distance from totally enclosed Class 1E raceways. Temporary cables are non-Class 1E and have a specified removal date or removal event. Tests have demonstrated the acceptability of a single solid metal cable cover as a barrier when the worst case electrical fault occurs to a cable resting on the metal cable tray cover. The cables inside the cable tray maintained their functional capability during the testing.

Permanent free air telephone (PABX) cables can be installed with a separation distance of one-foot horizontal and three-foot vertical from Class 1E ventilated cable tray. Physical separation between the permanent free air telephone (PABX) cables and Class 1E enclosed raceway shall be 6 inches. Tests have demonstrated the acceptability of a single solid metal conduit as a barrier when the worst case electrical fault occurs to a cable resting on the barrier. The cables inside the conduit maintained their functional capability during the test[jcb1]

In confined spaces, where fire barriers cannot be installed, lesser separation distance between open trays than those specified above shall be allowed in the following areas that are protected by ionization type detectors with total flooding or

manual spurt CO_2 suppression systems. (Refer to Fire Protection Review Report, Table 6.1-1, "SSES Fire Areas" for room numbers and fire zones.) Examples of confined spaces are:

1) Cable chases/soffits

2) Raised floor section

e) General Plant Areas

In plant areas from which potential hazards such as missiles, external fires, and pipe whip are excluded, the minimum separation distance between redundant Class 1E cable trays is 3 ft. between trays separated horizontally if no physical barrier exists between trays. If a horizontal separation of less than 3 ft. exists, alternate methods as stated in paragraph d) above are required. Vertical stacking of trays is avoided wherever possible; however, where cable trays of redundant channel/divisions are stacked, a minimum vertical separation distance of 5 ft. is required, or alternate methods as stated in paragraph d) above are required. Where a cross-over of one tray over another carrying redundant channel/division is made, and minimum vertical separation distance cannot be maintained, one of the following requirements is met; a) a solid cover is installed on the lower tray to extend 1 ft. 0 in. minimum either side of the upper tray, b) fire barriers are installed minimum 1 in. from the upper tray and extend 1 ft. 0 in. minimum beyond the crossing tray.

Separation requirements between Class 1E and non-Class 1E trays are the same as separation of redundant channel/division, except that the minimum separation distance between Class 1E tray and non-Class 1E conduit or totally enclosed raceway is 1 inch.

Free air temporary cables can be installed with no separation distance from totally enclosed Class 1E raceways. Temporary cables are non-Class 1E and have a specified removal date or removal event. Tests have demonstrated the acceptability of a single solid metal cable cover as a barrier when the worst case electrical fault occurs to a cable resting on the metal cable tray cover. The cables inside the cable tray maintained their functional capability during the testing.

Permanent free air telephone (PABX) cables can be installed with a separation distance of one-foot horizontal and three-foot vertical from Class 1E ventilated cable tray. Physical separation between the permanent free air telephone (PABX) cables and Class 1E enclosed raceway shall be 6 inches. Tests have demonstrated the acceptability of one-foot horizontal and three-foot vertical distances to prevent migration of electrical faults from the low energy free air telephone cables to the Class 1E cables. Tests also have demonstrated the acceptability of a single solid metal conduit as a barrier when the worst case

electrical fault occurs to a cable resting on the barrier. The cables inside the conduit maintained their functional capability during the test[jcb2].

f) Power Generation Control Complex - (PGCC)

Refer to Subsection 7.1.2a.3.3.6.

g) The Lighting Fixture Cords

The non-Class 1E lighting fixture cord connects a lighting fixture to a single phase, 277V power supply. The cord is a #14 AWG SO insulated cable (with grounding conductor) installed in free air carrying a maximum load current of 1.0 ampere. The minimum separation between the free air lighting fixture cord and a Class 1E open tray is 6 inches. If the above minimum separation cannot be satisfied, tray covers or conduits will be provided for the Class 1E raceway in the vicinity of the free air lighting fixture cord. The lighting fixture cord does not have sufficient combustible material or energy which could cause failure of the nearby Class 1E cables.

Non-Class 1E lighting conduits, containing a single circuit rated less than 300 VAC and 10 amp or 20 amp for receptacles, are treated as control conduits for separation purposes. The separation criteria is as stated in Section 8.1.6.1 (Regulatory Guide 1.75(1/75), Part 7). The lighting fixture cords, which are exposed in free air containing #12 AWG wires and about 3 to 5 feet in length, shall be maintained at 6" minimum separation from a Class 1E raceway.

h) An exception to the above subsections d) and e) is the 450 MHz radio antenna cable network.

The 450 MHz radio antenna is the plant security communication radio system (non-Class 1E). This system utilizes an antenna cable network installed exposed (not enclosed in raceway) on the cable raceway supports throughout the plant. The jacketing material of the antenna cable is flame retardant. The cable has been tested and passed IEEE 383 and ASTM Proc. D2633 part 30.

Separation between this antenna cable and other class 1E raceways is not required because:

1) The antenna cable is a low energy circuit. A short circuit of the antenna cable would not produce enough energy to cause degradation of any other circuits.

2) The antenna cable is not routed with any other cables.

3) The antenna cable jacket is made of flame retardant material.

4) The antenna cable does not terminate in close proximity or routed through any equipment with voltage level higher than 120V AC.

5) The maximum radio frequency (rf) power output level of the antenna cable is 37.5 watts.

6) Where redundant safe shutdown raceways are separated by less than 20 feet, fire barriers have been provided to protect one division per Fire Protection Review Report Section 4.11.

3.12.3.4.2.2 Standbv Power Supply

a) Emergencv Diesel Generators

Redundant Class 1E diesel generator units are located in separate safety class structures and have independent air and fuel supplies.

b) Auxiliaries and Local Controls

The auxiliaries and local controls for diesel generators are in the same safety class structure as the unit they serve, except for the Diesel Generator A, B, C and D fuel oil transfer pumps that are located in separate safety class structures at the fuel oil storage tanks (see Subsection 9.5.4).

3.12.3.4.2.3 DC Svstem

a) Batteries

Redundant Class 1E batteries are placed in separate safety class structures. The structures are served by redundant ventilation equipment.

b) Battery Chargers

Battery chargers and their respective switchgears are placed in separate safety class structures from their respective redundant Class 1E batteries.

3.12.3.4.2.4 Distribution Svstem

a) Switchgear

Redundant Class 1E distribution switchgear groups are placed in separate safety class structures.

b) Motor Control Centers

Redundant Class 1E motor control centers are physically separated in accordance with the requirements of Subsection 3.12.3.4.1.

c) Distribution Panels

Redundant Class 1E distribution panels are physically separated in accordance with the requirements of Subsection 3.12.3.4.1.

3.12.3.4.2.5 Primary Containment Electrical Penetrations

Redundant Class 1E primary containment electrical penetrations are physically separated in accordance with the requirements of Subsection 3.12.3.4.1. The minimum physical separation for redundant penetrations meets the requirements for cables and raceways given in Subsections 3.12.3.4.2.1through 3.12.3.4.2.6.

3.12.3.4.2.6 Main Control Room and Relay Room Panels

a) For NSSS panels see Subsection 7.1.2a.3.1.1.

b) All non-NSSS panels containing safety-related equipment and circuits are provided as follows:

1) Generally, panels are divisionalized (i.e., are devoted to one (1) division only) and are physically separated from the redundant division's panels.

2) In cases where redundant channel/division Class 1E circuits, or RPS and other Class 1E and non-Class 1E circuits are located in the same enclosure, physical separation is achieved by minimum of 6" spatial separation, steel barriers, metallic enclosure, or metallic flexible conduit.

   Where the above separation methods are not feasible, one of the separation group circuits are to be covered with a qualified non-metallic barrier material. A description of the material and analysis to regulatory requirements is provided in Subsection 8.1.6.t.14(Conformance to Reg. Guide 1.75).

3) All requirements for connection of control circuits between separated divisions are accomplished with MDR relays to provide positive isolation of the circuits.

4) All the annunciator and computer digital inputs are classified as non-Class 1E circuits. These circuits are not separated from the Class 1E circuits within the Class 1E panels in which the non-Class 1E input is

derived. The interface devices used in the Class 1E circuits to develop the annunciator and computer digital inputs are listed in Table 3.12-3. An analysis for each circuit in which these devices are used has shown that a failure mode which prevents the Class 1E circuits from meeting their minimum performance requirements does not exist. This is based upon the application/function of the interface devices in each individual circuit.

TABLE 3.12-1

ESP DIVISION SEPARATION

Division I	Division II
Core Spray Loop A	Core Spray Loop B
Automatic Depressurization System A	Automatic Depressurization System B
Residual Heat Removal Loop A	Residual Heat Removal Loop B
High Pressure Coolant Injection System (Inboard Valve)	High Pressure Coolant Injection System except the inboard steam line isolation valve
Reactor Core Isolation Cooling System except the inboard steam line isolation valve	Reactor Core Isolation Cooling System (Inboard Valves)
Nuclear Steam Supply Shutoff System (Inboard Valves)	Nuclear System Supply Shutoff System (Outboard Valves)
Recirculation Pump Trip Loop A	Recirculation Pump Trip Loop B
Emergency Service Water Loop A	Emergency Service Water Loop B
RHR Service Water Loop A	RHR Service Water Loop B
Containment Instrument Gas Loop A	Containment Instrument Gas Loop B
Containment Atmospheric Control System A	Containment Atmospheric Control System B
Standby Gas Treatment System Train A	Standby Gas Treatment System Train B
Reactor Building HVAC Isolation and Recirculation System A	Reactor Building HVAC Isolation and Recirculation System B

TABLE 3.12-1

ESP DIVISION SEPARATION

Division I	Division II
Drywell HVAC System A	Drywell HVAC System B
Control Structure HVAC System Train A	Control Structure HVAC System Train B
Control Structure Chilled Water System Loop A	Control Structure Chilled Water System Loop B
Battery Room Ventilation System A	Battery Room Ventilation System B
HVAC Coolers for Div I	HVAC Coolers for Div II
Standby liquid Control System Pumps A[1] and B[1] and Explosive Valves A[1] and B[1]	
Class 1E 250V DC Supply System I	Class 1E 250V DC Supply System II
480V Swing Bus and Associated Motor-Generator Set Div I	480V Swing Bus and Associated Motor-Generator Set Div II
Class 1E 480V AC MCCs	Class 1E 480V AC MCCs
Class 1E 120V AC Distribution Panels	Class 1E 120V AC Distribution Panels
Class 1E 125V DC Distribution Panels	Class 1E 125V DC Distribution Panels

[1]The redundant standby liquid control pumps and explosive valves are powered from different electrical buses.

TABlE 3.12-2

CHANNEl SEPARATION

Channel A	Channel B	Channel C	Channel D
Standby Diesel Generator & Auxiliaries A (Common to Units 1 and 2)	Standby Diesel Generator & Auxiliaries B (Common to Units 1 and 2)	Standby Diesel Generator & Auxiliaries C (Common to Units 1 and 2)	Standby Diesel Generator & Auxiliaries D (Common to Units 1 and 2)
Standby Diesel A Ventilation System (Common to Units 1 and 2)	Standby Diesel B Ventilation System (Common to Units 1 and 2)	Standby Diesel C Ventilation System (Common to Units 1 and 2)	Standby Diesel D Ventilation System (Common to Units 1 and 2)
Class 1E 4160 V Switchgear	Class 1E 4160 V Switchgear	Class 1E 4160 V Switchgear	Class 1E 4160 V Switchgear
Class 1E 480 V Load Center	Class 1E 480 V Load Center	Class 1E 480 V Load Center	Class 1E 480 V Load Center
Class 1E 480 V MCC (Common to Units 1 and 2)	Class 1E 480 V MCC (Common to Units 1 and 2)	Class 1E 480 V MCC (Common to Units 1 and 2)	Class 1E 480 V MCC (Common to Units 1 and 2)
Class 1E 125 V Distribution Panel	Class 1E 125 V Distribution Panel	Class 1E 125 V Distribution Panel	Class 1E 125 V Distribution Panel

Note:

Additionally, a fifth diesel generator is provided which can be manually realigned as a replacement for any one of the other four diesel generators. This fifth diesel generator has its own ventilation and electrical support systems.

When this fifth diesel generator is substituted for any one of the other four diesel generators, the fifth diesel generator and it's auxiliaries assimilate the separation channel of the diesel generator which was substituted.

TABLE 3.12-3

Main Control Room and Relay Panel
Annunciator and Computer Interface Device

ANNUNCIATOR INTERFACE DEVICES	
Agastat Type EGP	GE Type HFA
Riley 86 T/C Monitor	P&B Type KH-4690
Westronics Recorder	Agastat Type E7000
GE Type CR2940 SW	GE Type 2820
Bailey 745 Alarm	P&B Type MDR
GE Type CR105	Agastat Type TR
C-H Type 10250T PB	GE Type HMA

COMPUTER INTERFACE DEVICES	
GE Type CR 105	Agastat Type E7024
GE Type HFA	P&B Type KH-4690
GE Type HMA	GE IRM Switch 216X494G19
Agastat Type EGP	

3.13 COMPLIANCE WITH NRC REGULATORY GUIDES

This section discusses the compliance of the plant design with the guidelines presented in the NRC Regulatory Guides. Where applicable, reference is made to the Final Safety Analysis Report (FSAR) section(s) in which the appropriate design feature is described.

Since the application for the construction permit for this station was docketed in March 1971; therefore, it should be noted that the implementation paragraphs of many of the Regulatory Guides render the provisions contained therein inapplicable to Susquehanna SES by virtue of their effective dates. Nonetheless, the Applicant has evaluated the design and construction against versions of the Regulatory Guides which were current when the application for an Operating License was tendered and has complied with the listed revisions to the extent practicable.

Where compliance to the regulatory guide has been qualified by an interpretation of the regulatory guide, these variances are discussed in either this section or in an appropriate section referenced in a particular response.

The use of an Alternative Source Term (AST) requires changes to source term assumptions and dose acceptance criteria. Regulatory Guides have not been reviewed in detail to determine if exceptions are required for the below listed items. This note captures that the following criteria may be applicable exception(s) to a specific Regulatory Guide.

a. New AST analyses performed in accordance with the guidance in Regulatory Guide 1.183 for the following accidents: Loss of Coolant Accident, Main Steam Line Break, the Refueling Accident and the Control Rod Drop Accident.

b. Dose acceptance criteria is based on the Total Effective Dose Equivalent (TEDE) versus thyroid, whole body and beta dose.

c. Changed from 10CFR100.11 to 10CFR50.67 for dose acceptance criteria.

d. Changed from 10CFR50, Appendix A, General Design Criteria 19 to 10CFR50.67 for control room personnel dose acceptance criteria.

e. No longer committed to Regulatory Guides 1.3, 1.5 and 1.25.

3.13.1 DIVISION 1, REGULATORY GUIDES - POWER REACTORS

Regulatory Guide 1.1 - NET POSITIVE SUCTION HEAD FOR EMERGENCY CORE COOLING AND CONTAINMENT HEAT REMOVAL SYSTEM PUMPS (November 2, 1970)

As discussed in Subsections 6.2.2 and 6.3.2, Susquehanna SES has been designed to comply with this regulatory guide.

Regulatory Guide 1.2 - THERMAL SHOCK TO REACTOR PRESSURE VESSELS (November 2, 1970)

With respect to the regulatory positions of this regulatory guide, Susquehanna SES is in compliance as follows:

The reactor pressure vessel utilized for Susquehanna SES employs no significant core or vessel design changes from previously approved BWR pressure vessels such as Browns Ferry.

NOTE: Although this regulatory guide has been withdrawn, any prior or existing commitments based on its use are not altered.

An investigation of the structural integrity of boiling water reactor pressure vessels during a design basis accident (DBA) has been conducted (refer to NEDO-10029). It has been determined, based upon methods of fracture mechanics, that no failure of the vessel by brittle fracture as a result of a DBA will occur.

The investigation included:

a. a comprehensive thermal analysis considering the effect of blowdown and the low-pressure coolant injection (LPCI) system reflooding;
b. a stress analysis considering the effects of pressure, temperature, seismic load, jet load, dead weight, and residual stresses;

c. the radiation effect on material toughness (NDTT shift and critical stress intensity); and

d. methods for calculating crack tip stress intensity associated with a non-uniform stress field following the design basis accident.

This analysis incorporated very conservative assumptions in all areas (particularly in the areas of heat transfer, stress analysis, effects of radiation on material toughness, and crack tip stress intensity). Therefore, the results reported in NEDO-10029 provide an upper bound limit on brittle fracture failure mode studies. Because of the upper bound approach, it is concluded that catastrophic failure of the pressure vessel due to the DBA is shown to be impossible from a fracture mechanics point of view. In the case studied, even if an acute flaw does form on the vessel inner wall, it will not propagate as the result of the DBA.

For further discussion of fracture toughness of the Reactor Pressure Vessel refer to Subsection 5.2.3.3.1.

Regulatory Guide 1.3 - ASSUMPTIONS USED FOR EVALUATING THE POTENTIAL RADIOLOGICAL CONSEQUENCES OF A LOSS-OF-COOLANT ACCIDENT FOR BOILING WATER REACTORS (Revision 2, June 1974)

Not Applicable.

Regulatory Guide 1.4 - ASSUMPTIONS USED FOR EVALUATING THE POTENTIAL RADIOLOGICAL CONSEQUENCES OF A LOSS-OF-COOLANT ACCIDENT FOR PRESSURIZED WATER REACTORS (Revision 2 June, 1974)

Not Applicable.

Regulatory Guide 1.5 - ASSUMPTIONS USED FOR EVALUATING THE POTENTIAL RADIOLOGICAL CONSEQUENCES OF A STEAM LINE BREAK ACCIDENT FOR BOILING WATER REACTORS (March 10, 1971)

Not Applicable.

Regulatory Guide 1.6 - INDEPENDENCE BETWEEN REDUNDANT - STANDBY (ONSITE) POWER SOURCES AND BETWEEN THEIR DISTRIBUTION SYSTEMS (March 10, 1971)

As discussed in Subsections 7.1.2.6, 8.1-6.1.a, and 8.3.1.4, independence between redundant standby (onsite) power sources and between their distribution systems is provided with the exception of Position D.4.c. Swing buses Supply power to the LPCI injection valves and the recirculation piping isolation and bypass valves. Motor generator sets are used to protect redundant power sources from faults that might develop on the swing bus, thus ensuring the requisite degree of independence between the redundant power sources.

Regulatory Guide 1.7 - CONTROL OF COMBUSTIBLE GAS CONCENTRATIONS IN CONTAINMENT FOLLOWING A LOSS-OF-COOLANT ACCIDENT (Revision 3, March 2007)

The design guidance and assumptions of this regulatory guide are followed as discussed in Subsection 6.2.5.

Regulatory Guide 1.8 - PERSONNEL SELECTION AND TRAINING (Revision 2, April 1987)

Commitment to this regulatory guide is described in FSAR Table 17.2-1 and Technical Specification 5.3.1. Additional information relating to personnel selection and training can be found in FSAR Sections 12.5, 13.1 and 13.2.

Regulatory Guide 1.9 - SELECTION OF DIESEL GENERATOR SET CAPACITY FOR STANDBY POWER SUPPLIES (March 10, 1971, For Diesel Generators 'A-D' and December 1979 for Diesel Generator 'E')

The standby power system is discussed fully in Subsections 8.1.6.1.b and 8.3.1, AC Power Systems. Standby diesel generator power supplies comply with Regulatory Guide 1.9.

Except as indicated below,

(1) Reference: Position C.4 Power quality is in accordance with IEEE 387-1972, Section 5.1.2(4) for Diesel Generators A-D and IEEE 387-1977, Section 5.1.2(5) for Diesel Generator E. At no time during the loading sequence will the frequency or voltage drop to a level which will degrade the performance of any of the loads below their minimum requirements.

(2) Reference: Position C.5. The suitability of each Susquehanna SES Diesel Generator is confirmed by factory qualification testing and preoperational test. Discussion of the factory test results is in Section 8.3.

(3) Reference: Position C.11 (December 1979 Revision only; there is no commensurate requirement in the March 1971 Revision). Exception is taken to the statement that IEEE 387-1977, Section 6.6, "Periodic Testing," should be supplemented by RG 1.108. Regulatory Position C.2.a of RG 1.108 requires that testing of diesel generators occurs at least once every 18 months. The 18 month requirement is interpreted to mean once per refueling cycle. Periodic testing is performed in accordance with the Technical Specifications, and the test interval may be replaced with performance-based, risk-informed test intervals. This statement in Regulatory Position C.11 of RG 1.9, and by extension Regulatory Position C.2.a of RG 1.108, clarifies the statement in Section 6.6.2 of IEEE 387- 1977 that operational tests be performed "at acceptable intervals." By taking exception to Regulatory Position C.11 of RG 1.9, exception is also being taken to Regulatory Position C.2.a of RG 1.108 that the frequency of diesel generator testing should be at least once every 18 months. Despite the exceptions to Regulatory Position C.11 of RG 1.9 and C.2.a of RG 1.108, exception is not being taken to the statement in Section 6.6.2 of IEEE 387-1977 that the testing be performed at acceptable intervals. The performance based, risk-informed test intervals are determined in accordance with Technical Specification 5.5.15 as approved by the NRC and are acceptable.

Regulatory Guide 1.10 - MECHANICAL (CADWELD) SPLICES IN REINFORCING BARS OF CATEGORY I CONCRETE STRUCTURES (Revision 1, January 2, 1973)

The testing and inspection program for all mechanical (Cadweld) splices in reinforcing bars of Category I structures are in compliance with this regulatory guide.

Regulatory Guide 1.11 - INSTRUMENT LINES PENETRATING PRIMARY REACTOR CONTAINMENT (March 10, 1971)

The design of the instrument lines penetrating the primary reactor containments of the Susquehanna SES complies with the provisions of this Regulatory Guide. Instrument lines which directly communicate with containment atmosphere, or do not communicate with reactor coolant pressure boundary are treated as extensions of the containment.

Regulatory Guide 1.12 - INSTRUMENTATION FOR EARTHQUAKES (Revision 1 April, 1974)

As described in Subsection 3.7b.4, seismic instrumentation is provided. The instrumentation meets the regulatory position set forth in this guide except for Section C.1.a. since no triaxial peak accelerographs are provided.

Regulatory Guide 1.13 - SPENT FUEL STORAGE FACILITY DESIGN BASIS (Revision 1, December 1975)

The fuel storage facility design basis is described in Section 9.1 and Appendix 9A. Regulatory positions are complied with subject to the following exceptions and clarifications:

(1) Reference: Position C.2. The fuel pool is designed to prevent significant loss of watertight integrity caused by tornadic winds and missiles generated by these winds. The reactor building above the refueling floor consists of steel framing with metal siding.

(2) Reference: Positions C.3 and C.5.a. Interlocks are provided to prevent the 125 ton hook of the reactor building crane from passing over or near stored fuel. These interlocks preclude any load suspended from this crane from tipping over on the stored fuel in the event of a crane failure. The 5 ton auxiliary hook suspended from the same crane trolley is prevented from passing over stored fuel when fuel handling is not in progress by administrative controls. There are no planned transfers of loads heavier than a new fuel element over the stored fuel.

(3) Reference: Position C.8. A Seismic Category I makeup water supply from each emergency service water loop is permanently connected to each spent fuel pool by two independent Seismic Category I piping routes. The make-up is provided for filling the Spent Fuel Pool to the proper level to support operation of the RHR Fuel Pool Cooling mode, and to provide for make-up from evaporative losses during cooling by RHR. The make-up rate is sized based on boiling so as to be conservative. The normal makeup system to the fuel pool is not Seismic Category I.

Regulatory Guide 1.14 - REACTOR COOLANT PUMP FLYWHEEL INTEGRITY (Revision 1, August 1975)

Not Applicable.

Regulatory Guide 1.15 - TESTING OF REINFORCING BARS FOR CATEGORY I CONCRETE STRUCTURES (Revision 1, December 28, 1972)

Testing of reinforcing bars for Category I concrete structures is in compliance with this regulatory guide.

Regulatory Guide 1.16 - REPORTING OF OPERATING INFORMATION-APPENDIX A TECHNICAL SPECIFICATIONS (Revision 4, August 1975)

In lieu of the positions stated in this Regulatory Guide, the reporting of operating information for the Susquehanna SES complies with Technical Specifications, 10CFR50.73 and G.L. 97-02.

Regulatory Guide 1.17 - PROTECTION OF NUCLEAR POWER PLANTS AGAINST INDUSTRIAL SABOTAGE (June 1973)

In lieu of the positions stated in this regulatory guide, the protection of Susquehanna SES against industrial sabotage complies with 10CFR73.

Regulatory Guide 1.18 - STRUCTURAL ACCEPTANCE TEST FOR CONCRETE PRIMARY REACTOR CONTAINMENTS (Revision 1, December 28, 1972)

The compliance with this regulatory guide is achieved subject to certain test modifications as discussed in Subsection 3.8.1.7.1.1.

Regulatory Guide 1.19 - NONDESTRUCTIVE EXAMINATION OF PRIMARY CONTAINMENT LINER WELDS (Revision 1, August 11, 1972)

Nondestructive examination of the primary containment liner welds is conducted as discussed in Subsection 3.8.1.

Regulatory Guide 1.20 - COMPREHENSIVE VIBRATION ASSESSMENT PROGRAM FOR REACTOR INTERNALS DURING PREOPERATIONAL AND INITIAL STARTUP TESTING (Revision 2, May 1976)

The vibration assessment program for reactor internals as discussed in Subsections 1.5.1, 3.9.2.4 and NEDE 24057P complies with this regulatory guide.

Regulatory Guide 1.21 - MEASURING, EVALUATING AND REPORTING RADIOACTIVITY IN SOLID WASTES AND RELEASES OF RADIOACTIVE MATERIALS IN LIQUID AND GASEOUS EFFLUENTS FROM LIGHT-WATER-COOLED NUCLEAR POWER PLANTS (Revision 1, June 1974)

Operation of the radwaste systems will be conducted in accordance with this regulatory guide as permitted by the design. Operation and design of the systems are discussed in Section 11.5.

Regulatory Guide 1.22 - PERIODIC TESTING OF PROTECTION SYSTEM ACTUATION FUNCTION (February 17, 1972)

As discussed in Subsections 7.1.2.6.4, 7.2.2.1.2.1.2, 7.2.4.1.1.2.2.1, 7.3.2a.1.2.1.3, 7.3.2a.2.2.1.2, 7.3.2a.5.2.1, 7.4.2.1.2.1.3, 7.4.2.2.2.1.3, 7.6.2a.3.2.3.1, 7.6.2a.5.4.2, 8.1.6.1d and 8.3.1.3.1.5 periodic testing of protection system actuation functions complies with this regulatory guide.

Regulatory Guide 1.23 - METEOROLOGICAL MEASUREMENT PROGRAM FOR NUCLEAR POWER PLANTS Second Proposed Revision 1, (April, 1986)

The onsite meteorological system and program comply with this regulatory guide.

The commitment to accuracy criteria for delta temperature is Atomic Energy Commission Safety Guide 23, February 1972 (Regulatory Guide 1.23 Rev. 0).

Regulatory Guide 1.24 - ASSUMPTIONS USED FOR EVALUATION THE POTENTIAL RADIOLOGICAL CONSEQUENCES OF A PRESSURIZED WATER GAS STORAGE TANK FAILURE (March 23, 1972)

Not Applicable.

Regulatory Guide 1.25 - ASSUMPTIONS USED FOR EVALUATING THE POTENTIAL RADIOLOGICAL CONSEQUENCES OF A FUEL-HANDLING ACCIDENT IN THE FUEL HANDLING AND STORAGE FACILITY FOR BOILING AND PRESSURIZED WATER REACTORS (March 23, 1972)

Not Applicable.

Regulatory Guide 1.26 - QUALITY GROUP CLASSIFICATION AND STANDARDS (Revision 3, February 1976)

In general the requirements of Regulatory Guide 1.26 are met for the Susquehanna Plant. Some exceptions exist due to the purchase date of the NSSS equipment and design changes in a few systems. These exceptions have been documented in correspondence with NRC. Quality Group classifications are detailed in Tables 3.2-1, 3.2-2, and SSES-FSAR 3.2-3.

Regulatory Guide 1.27 - ULTIMATE HEAT SINK FOR NUCLEAR POWER PLANTS (Revision 2, January 1976)

Subject to the exception indicated below, the design of the ultimate heat sink satisfies the requirements of this regulatory guide.

(1) Reference: Position C.2. Position C.2 states that the ultimate heat sink features which are not required to be designed to withstand the Safe Shutdown Earthquake (SSE) should, nonetheless, be designed and constructed to withstand the Operating Basis Earthquake and waterflow based on severe historical events in the region. The requirements of this regulatory guide without the necessity for any makeup operations following the occurrence of an SSE. Therefore, the potential makeup water sources, such as the cooling tower basins and the makeup system from the Susquehanna River, are not required to perform any safety-related function following the occurrence of any seismic event, and design criteria specified in Position C.2 have not been employed in their design.

Compliance is discussed in Subsection 9.2.7. A description of the analysis performed to demonstrate the ability of the ultimate heat sink to meet the requirements of this Regulatory Guide is presented in Subsection 9.2.7.3.

Regulatory Guide 1.28 - QUALITY ASSURANCE PROGRAM REQUIREMENTS (DESIGN AND CONSTRUCTION) (Revision 1, March 1978)

The Quality Assurance Program for the construction of Susquehanna SES is described in the PSAR, Appendix D and amendments. Compliance of the Operational Quality Assurance Program with this guide is discussed in Section 17.2.

Regulatory Guide 1.29 - SEISMIC DESIGN CLASSIFICATION (Revision 2, February 1976) (Revision 3, September 1978 for the Diesel Generator 'E' Facility)

Subject to the clarifications and/or exceptions indicated below, Susquehanna SES complies with this regulatory guide.

(1) Reference: Position C.1.b. For the NSSS, application of this guide is limited to the reactor core and reactor internals which are engineered safety features.

(2) Reference: Position C.1.d and C.1.g. The normal spent fuel pool cooling system is non-seismic Category I. If a seismic event would occur, cooling of the spent fuel is achieved by use of the RHR Fuel Pool Cooling (RHRFPC) mode as described in Sections 5.4.7.1.1.6, 5.4.7.2.6c, 9.1.3.1, and 9.1.3.3. Either or both of two Seismic Category I ESW makeup water supplies to each pool can provide make-up in support of the RHRFPC mode. Additionally, ESW is capable of supplying make-up for the boiling Spent Fuel Pool (SFP) analysis as described in Appendix 9A.

(3) Reference: Position C.1.e. The Main Steam System (MSS) beyond the outer isolation valves up to and including the turbine stop valves and all branch lines 2 1/2 in. in diameter and larger, up to and including the first valve (including their restraints) are not classified

Seismic Category I; because portions of the pipe are routed in a non-Seismic Category I building (the Turbine Building). However, the turbine building has been designed to withstand an SSE as stated in Subsection 3.7b.2.8. Further description of the turbine building is given in Subsection 3.8.4.1; applicable load combinations are given in Table 3.8-10. The subject piping is designed in accordance with ASME Section III, Class 2 requirements for the OBE and SSE as described in Subsection 10.3.3.

(4) Reference: Position C.1.h. The component cooling water portions of the reactor recirculation pumps are not Seismic Class I since they do not involve a safety function.

(5) Reference: Paragraph C.2 of the Regulatory Guide. Items which would otherwise be classified non-Seismic Category I, "but whose failure could reduce the functioning" of items important to safety "to an unacceptable safety level" are to be "designed and constructed so that the SSE would not cause such failure." In addition, Paragraph C.4 of the guide requires that the "pertinent quality assurance requirement of Appendix B to 10 CFR Part 50 should be applied to the safety requirements" of such items. Both of these positions are considered to be adequately met by applying the following practices to such items:

(a) Design and design control for such items are carried out in the same manner as that for items directly important to safety. This includes the performance of appropriate design reviews.

(b) Field work is performed under the direction of experienced field construction superintendents and is inspected by the staff of field engineers stationed at the site. The field engineers are responsible for verifying that construction is performed in accordance with the design drawings and specifications and with applicable standard codes and specifications.

(6) Reference: Paragraph C.3 of the Regulatory Guide. Seismic Category I design requirements are required to be extended "to the first seismic restraint beyond the defined boundaries." Since seismic analysis of a piping system requires division of the systems into discrete segments terminated by fixed points, this means that the seismic design cannot be terminated at a seismic restraint, but is extended to the first point in the system which can be treated as an anchor to the plant structure. In addition, Paragraph C.4 of the Regulatory Guide takes the position that "the pertinent quality assurance requirement of Appendix B to 10CFR Part 50 should be applied to the safety requirements" of such items. Both these requirements are considered to be met adequately by applying the following practices to such items:

(a) Design and design control for such items are carried out in the same manner as that for items directly important to safety. This includes the performance of appropriate design reviews.

(b) Field audits are performed by representatives of the originating design group to assure that the final installation of such items is in accordance with documents that formed the basis for the seismic analysis of the items.

Regulatory Guide 1.30 - QUALITY ASSURANCE REQUIREMENTS FOR THE INSTALLATION, INSPECTION, AND TESTING OF INSTRUMENTATION AND ELECTRICAL EQUIPMENT (August 11, 1972)

The Susquehanna SES quality program for construction of safety related items was conducted in accordance with the program described in PSAR Appendix D and amendments. Compliance of the Operational Quality Program with this guide is described in Table 17.2-1.

Regulatory Guide 1.31 - CONTROL OF STAINLESS STEEL WELDING

Control of stainless steel welding (except NSSS scope of supply) complies with Interim Position on Regulatory Guide 1.31 (Branch Technical Position MTEB 5-1 dated November 24, 1975) except as discussed below.

(1) Reference: Paragraph B.1.b of Interim Position. Austenitic stainless steel welding filler materials used in the fabrication and installation of ASME Section III, Class 1, 2, and 3 components are controlled to deposit from 8 to 25 percent delta ferrite. Welding filler materials 309 and 309L are controlled to deposit from 5 to 15 percent delta ferrite and are used only for welding carbon or low alloy steel to austenitic stainless steel. The use of 309L welding filler material is further limited to the overlay deposit on the carbon or low alloy steel component nozzles or connecting pipe when postweld heat treatment is required.

These limits for delta ferrite in austenitic stainless steel welding materials comply with Interim Regulatory Guide 1.31 because the upper limit of 20 percent delta ferrite does not apply to welds that are not heat treated after welding (Paragraph 3b). Solution heat treatment, although not required after welding, is permitted to avoid sensitization.

The procedure for determining the amount of delta ferrite in each heat or lot of austenitic stainless steel welding material does not comply with the Interim Position of the Regulatory Guide. Determination of delta ferrite is in accordance with ASME Section III, Division 1, 1974 Edition,

Paragraph NB02433, except that an undiluted weld deposit is required for each heat of bare wire used with the Gas Metal-Arc process.

(2) Reference: Paragraph B.2 of the Interim Position. This paragraph is complied with for all tests and examinations required by ASME Section III, Division 1, 1974 Edition.

(3) Reference: Paragraph B.3.a of the Interim Position. This paragraph is not complied with. Magnetic measurement of production welds for delta ferrite is unnecessary when austenitic stainless steel welding materials are controlled to deposit 8 to 25 percent delta ferrite based on chemistry, except for 309 and 309L welding materials which are controlled to deposit 5 to 15 percent delta ferrite based on chemistry.

Three Bechtel projects are committed to measuring production welds for delta ferrite in order to collect data and demonstrate that the welding material controls described above are more than adequate for the purpose of avoiding microfissuring. Since this represents a

sufficient number of welds for the purpose of collecting data, measurement of production welds for delta ferrite on this project is not planned.

(4) Reference: Paragraph B.3.b of the Interim Position. This paragraph is complied with for welding material certification.

(5) Reference: Paragraphs B.4.a, b, and c of the Interim Position. These paragraphs are not complied with since measurement of production welds for delta ferrite is not performed.

The NSSS scope of supply control of welding is described in Subsection 5.2.3.4.2.

Regulatory Guide 1.31 - CONTROL OF FERRITE CONTENT IN (DIESEL GENERATOR E ONLY) STAINLESS STEEL WELD METAL (Revision 3, April 1978)

The design of the Diesel Generator "E" meets the intent of this Regulatory Guide. Methods of control of welding, in fabricating and joining safety-related austenitic stainless steel components and systems, were implemented which led to meeting the intent of 10CFR50, Appendix A, GDC 1.

Regulatory Guide 1.32 - CRITERIA FOR SAFETY RELATED ELECTRIC POWER SYSTEMS FOR NUCLEAR POWER PLANTS (Revision 1, March 1976) for Diesel Generators "A-D" and (Revision 2, February 1977 for the Diesel Generator 'E' Facility)

Diesel Generators "A-D" are designed in accordance with Regulatory Guide 1.32, Revision 1 (March 1976). Diesel Generator "E" and the transfer points in the Diesel Generator "A-D" rooms are designed to Regulatory Guide 1.32, Revision 2 (February 1977), as discussed in FSAR Sections 8.1.6.1 and 8.3.2.2. Positions C.1.e and C.1.f of this regulatory guide are addressed in the responses to Regulatory Guides 1.75 and 1.9 respectively.

Regulatory Guide 1.33 - QUALITY ASSURANCE PROGRAM REQUIREMENTS (OPERATION) (Revision 2, February 1978)

Compliance of the Operational Quality Assurance Program with this guide is described in Section 17.2.

Regulatory Guide 1.34 - CONTROL OF ELECTROSLAG WELD PROPERTIES (December 28, 1972)

The electroslag weld method was not used for the fabrication of any core support structures or any ASME B&PV Section III, Class 1 or 2 vessels and components. Therefore, this regulatory guide is not applicable to Susquehanna SES.

Regulatory Guide 1.35 - IN-SERVICE INSPECTION OF UNGROUTED TENDONS IN PRESTRESSED CONCRETE CONTAINMENT STRUCTURES (Revision 2, January 1976)

Not Applicable.

Regulatory Guide 1.36 - NONMETALLIC THERMAL INSULATION FOR AUSTENITIC STAINLESS STEEL (February 1973)

As discussed in Subsections 4.5.2.4 and 6.1.1.1, the use of nonmetallic thermal insulation for austenitic stainless steel complies with this regulatory guide.

Regulatory Guide 1.37 - QUALITY ASSURANCE REQUIREMENTS FOR CLEANING OF FLUID SYSTEMS AND ASSOCIATED COMPONENTS OF WATER COOLED NUCLEAR POWER PLANTS (March 16, 1973)

The Quality Assurance Program for the construction of Susquehanna SES is described in the PSAR, Appendix D and amendments.

Compliance of the operational Quality Assurance program with this guide is discussed in Section 17.2.

Regulatory Guide 1.38 - QUALITY ASSURANCE REQUIREMENTS FOR PACKAGING, SHIPPING, RECEIVING, STORAGE, AND HANDLING OF ITEMS FOR WATER-COOLED NUCLEAR POWER PLANTS (Revision 2, May 1977)

The Susquehanna SES construction quality program is being conducted in accordance with the program described in PSAR Appendix D and amendments. Compliance of the Operational Quality Assurance Program is described in Section 17.2.

Regulatory Guide 1.39 - HOUSEKEEPING REQUIREMENTS FOR WATER-COOLED NUCLEAR POWER PLANTS (Revision 2, September 1977)

Compliance of the Operational Quality Assurance Program is described in Section 17.2.

Regulatory Guide 1.40 - QUALIFICATION TESTS OF CONTINUOUS DUTY MOTORS INSTALLED INSIDE THE CONTAINMENT OF WATER COOLED NUCLEAR POWER PLANTS (March 16, 1973)

As described in Subsection 3.11.2.2, the present design of Susquehanna SES complies with the provisions of this regulatory guide.

Regulatory Guide 1.41 - PREOPERATIONAL TESTING OF REDUNDANT ONSITE ELECTRIC POWER SYSTEM TO VERIFY PROPER LOAD DESIGN ASSIGNMENTS (March 16, 1973)

The requirements of this regulatory guide were met. The testing procedures are outlined in Section 14.2.

Regulatory Guide 1.42 - INTERIM LICENSING POLICY ON AS LOW AS PRACTICABLE FOR GASEOUS RADIOIODINE RELEASES FROM LIGHTWATER-COOLED NUCLEAR POWER REACTORS

Withdrawn March 18, 1976.

Regulatory Guide 1.43 - CONTROL OF STAINLESS STEEL WELD CLADDING OF LOW-ALLOY STEEL COMPONENTS (May 1973)

This regulatory guide prescribes qualification and production cladding controls for ASME SA 508-2 material made to coarse grain practice. This material is not used for any of the safety class components within the NSSS. ASME SA 508-2 composition material employed on the reactor pressure vessel for Susquehanna SES is produced to fine grain practice.

Regulatory Guide 1.44 - CONTROL OF THE USE OF SENSITIZED STAINLESS STEEL (May 1973)

Subject to the following clarifications and exceptions, the use of unstabilized austenitic stainless steel for components that are part of (a) the reactor coolant pressure boundary, (b) systems required for reactor shutdown, (c) systems required for emergency core cooling, (d) reactor vessel internals required for emergency core cooling, and (e) reactor vessel internals which are relied upon to permit adequate core cooling during any mode of normal operation or postulated accident conditions complies with Regulatory Guide 1.44.

(1) Reference: Position C.1. Contamination of austenitic stainless steel (Type 300 series) by compounds that could cause stress corrosion cracking is avoided during all stages of fabrication and installation. Cleaning is limited to solutions that contain not more than 100 ppm of chlorides. Rinsing or flushing is with water containing less than 100 ppm of chlorides. Foreign substances in contact with austenitic stainless steel (die lubricants, penetrant materials, marking materials, masking tape, etc.) either are controlled to contain the following amounts of contaminates, or are removed immediately following the operation in which they are used.

1. The inorganic halogen content shall be less than 200 ppm by weight.

2. The halogen (inorganic and organic) content shall be less than 1 percent by weight measured in accordance with ASTM D808-63.

3. Sulfur content shall be less than 1 percent by weight as measured in accordance with ASTM D129-64.

4. Total low melting point metal (lead, bismuth, zinc, mercury, antimony, and tin) content shall be less than 200 ppm by weight and no individual metal content shall be greater than 50 ppm by weight.

Completed components are packaged so that they are protected from the weather, dirt, wind, water spray, and other deleterious environmental conditions that may be encountered during shipment and subsequent site storage.

In the field, austenitic stainless steel components are stored clean and dry to prevent contamination. System hydrostatic tests are performed with demineralized water containing less than 10 ppm of chlorides. The influent water quality during final flushing or preoperational testing of the completed system is at least equivalent to the quality of demineralized water as defined in ANSI N45.2.1-1973.

Leachable chlorides and fluorides in nonmetallic insulation materials, which come in contact with austenitic stainless steel, are held to the lowest practical level by the inclusion of the requirements of Regulatory Guide 1.36 in the insulation purchase specifications.

(2) Reference: Position C.2. All grades of austenitic stainless steels (Type 300 series) are required to be furnished in the solution heat treated condition before fabrication or assembly into components or systems. The solution heat treatment varies according to the applicable ASME or ASTM material specification.

(3) Reference: Position C.3. All austenitic stainless steels are furnished in the solution heat treated condition in accordance with the material specification. For material that has been solution heat treated by the material manufacturer, testing to determine susceptibility to intergranular corrosion is performed only when required by the material specification. During fabrication and installation austenitic stainless steels are not permitted to be exposed to temperatures in the range of 800° to 1500°F, except for welding and hot forming. Welding practices are controlled to avoid severe sensitization, as described in (6), and solution heat treatment in accordance with the material specification is required following hot forming in the temperature range 800° to 1500°F. Unless otherwise required by the material specification, the maximum time for cooling from the solution heat treat temperature to below 800°F is 3 minutes. Corrosion testing in accordance with ASTM A 262-70, Practice A or E, or ASTM A 393 may be required if the maximum length of time for cooling to below 800°F is exceeded or the solution heat treat condition is in doubt.

(4) Reference: Position C.4. Use of low carbon (0.03 percent maximum) unstabilized austenitic stainless steel is not required since the reactor coolant meets the conductivity and chloride limits of Table 2 of Regulatory Guide 1.56. However, it is used as described in Section 6(c) below.

(5) Reference: Position C.5. Heat treating austenitic stainless steel in the temperature range 800 to 1500°F is not permitted and solution heat treatment is required following hot forming. Since sensitization is avoided, testing to determine susceptibility to intergranular attack is not performed.

(6) Reference: Position C.6. Welding practices are controlled to avoid severe sensitization in the heat-affected zone of unstabilized austenitic stainless steel as described below. Unless otherwise stated, the position applies to both Bechtel and Bechtel suppliers and subcontractors.

a) Weld Heat Input

Bechtel controls weld heat input during field installation by using shielded metal-arc welding and gas tungsten-arc welding processes only, and by limiting the size of electrodes for each process to 5/32 in. and 1/8 in. diameter maximum, respectively. In addition to these two processes, Bechtel suppliers and subcontractors are permitted to use automatic submerged-arc welding and gas metal-arc welding. Hardsurfacing operations are not included.

b) Interpass Temperature

The interpass temperature is controlled so as not to exceed 350°F.

c) Carbon Content

Susceptibility to sensitization is reduced significantly by selecting materials with the lowest reported carbon content. Specifically Type 304 stainless steel with carbon content limited to .030 maximum or 304L Stainless steel with carbon content limited to .030 maximum was used as follows:

Pipe Description	Size	Material
Head Spray	6"	304SS
Core Spray Influent	12"	304SS
Recirculation System	4"	304SS
Standby Liquid Control	1 1/2"	304LSS
Reactor Water Cleanup (Effluent from Reactor)	4"	304LSS
Instrument Piping	1" & 2"	304LSS
Bottom Drain	4"	304LSS
Vent, Drain, and Test Connections	1"	304LSS

d) Solution Heat Treatment

All austenitic stainless steels are provided in the solution annealed condition. This is accomplished by following the manufacturer solution annealing by water quenching from the solution annealing temperature to below 800°F in 3 minutes. Solution heat treatment is not required after welding.

Severe sensitization is avoided by not permitting heat treatment in the temperature range 800 to 1500°F following welding. This requires a special technique when welding stainless steel safe ends (transition pieces) to carbon or low-alloy steel component nozzles or piping. Specifically, a 309L stainless steel overlay or an Inconel weld overlay is deposited on the component and the component is postweld heat treated. Following final postweld heat treatment of the component, the

stainless steel safe end is welded to the weld overlay using 308 or 308L austenitic stainless steel or Inconel type welding materials.

Intergranular corrosion testing is not performed on a routine basis. Performing intergranular corrosion tests for each welding procedure serves no useful purpose when welding practices and reactor coolant water chemistry are controlled as described above.

Regulatory Guide 1.45 - REACTOR COOLANT PRESSURE BOUNDARY LEAKAGE DETECTION SYSTEMS (May 1973)

The design of the leakage detection systems is described in Subsection 5.2.5.1.

Regulatory Guide 1.46 - PROTECTION AGAINST PIPE WHIP INSIDE CONTAINMENT (May 1973)

The criteria given in NRC Branch Technical Position MEB3-1, dated 11-24-75, is used in lieu of the criteria prescribed in Regulatory Guide 1.46 dated May 1973 for the non-NSSS scope of supply. Section 3.6 describes how the present design meets these protection requirements.

This regulatory guide is applicable to the main steam, HPCI, RCIC, RWCU, SLCS and recirculation pipelines within the NSSS scope of supply.

The design of the containment structure, component arrangement, Class 1 pipe runs, pipe restraints and compartmentalization was done in consonance with the acknowledgement of protection against dynamic effects associated with postulated rupture of piping. Analytically sized and positioned pipe restraints were engineered to preclude damage based on the pipe break evaluation.

Pipe whip requirements for fluid system piping within the primary containment that under normal operation, has service temperatures 200º F or pressures greater than 275 psig comply with ANS 58.2 - "Design Basis for Protection of Light Water Nuclear Power Plants Against the Effects of Postulated Pipe Rupture" and Regulatory Guide 1.46 except as delineated in the following criteria for no breaks in Class 1 piping:

1. If Equation 10 of NB-3653-1, ASME Code III results in S_n less than or equal to 2.4 S_m for ferritic or austenitic steels, no other requirement need be met. Stress range should be calculated between any two load sets (including zero load set) according to NB-3600 for upset and an OBE event transient.

2. If Equation 10 results in S_n between 2.4 S_m and 3,0 S_m for ferritic or austenitic steels, the cumulative usage factor, U, calculated on the basis of Equation 14 of NB-3653.6, must be less than 0.1.

3. If Equation 10 results in S_n greater than or equal to 3.0 S_m for ferritic or austenitic steels, then the stress value in Equations 12 and 13 of NB-3653.6 must not exceed 2.4 S_m and the cumulative usage factor, U, must be less than 0.1.

Regulatory Guide 1.47 - BYPASSED AND INOPERATIONAL STATUS INDICATION FOR NUCLEAR POWER PLANT SAFETY SYSTEMS (May 1973)

The design, as discussed in Subsections 7.1.2, 7.1.2.6.10, 7.2.2.1.2.1.5), 7.3.2a.1.2.1.7, 7.3.2a.2.2.1.5, 7.3.2a.5.2.5, 7.4.2.1.2.1.7, 7.4.2.2.2.1.71, 7.5.1b.7, 7.5.2a.5.4, 7.5.2b.5, and 8.1.6.1.m (Regulatory Guide 1.47), complies with the provisions set forth in this regulatory guide.

Regulatory Guide 1.48 - DESIGN LIMITS AND LOADING COMBINATIONS FOR SEISMIC CATEGORY I FLUID SYSTEM COMPONENTS (May 1973)

The design loading combinations for non-NSSS systems for Positions C.1 to C.12 are described in Table 3.9-6. Operability of active pumps and valves is assured as described in Subsection 3.9.3.2.

The design limits and loading combinations for seismic category I fluid system components for the Diesel Generator 'E' facility are in compliance with this regulatory guide.

GE practice is representative of industry practice and is in general agreement with the requirements of Regulatory Guide 1.48 with the following clarifications:

a. The probability of an OBE of the magnitude postulated for the Susquehanna SES is consistent with its classification as an Emergency Event. However, for design conservatism, loads due to the OBE vibratory motion have been included under upset conditions. Loads due to the OBE vibratory motion plus associated transients, such as turbine trip, have been considered in the equipment design under emergency conditions consistent with the probability of the OBE occurrence.
b. The use of increased stress levels for Class 2 components is consistent with industry practice as specified in ASME B&PV Code Section III.

For a comparison of NSSS compliance with Regulatory Guide 1.48 see Table 3.13-1. This comparison reflects a GE practice on BWR 4's and 5's and therefore, is applicable to the Susquehanna SES (see Subsections 3.9.2 and 3.9.3).

Regulatory Guide 1.49 - POWER LEVELS OF NUCLEAR POWER PLANTS (Revision 1, December 1973)

Regulatory Guide 1.49 states, in part,

> Anaylses and evaluation in support of the application should be made at an assumed core power level equal to 1.02 times the proposed licensed power level. . . for (a) normal operating conditions, (b) transient conditions anticipated during the life of the facility. . . and (c) accident conditions necessary to evaluate the adequacy of structures, systems and components provided for the prevention of accidents and the mitigation of the consequences of accidents.

Fuel dependent analyses include the effects of a 2 percent power uncertainty factor discussed in Regulatory Guide 1.49. Most of the analyses were performed at 100% power level and the impact of the two percent power uncertainty factor is accounted for either statistically or through the

inherent conservatism of the methodology. For three of the analyses, ASME over-pressurization (Section 5.2), loss of feedwater flow (Section 15.2.7), and LOCA-ECCS analyses (Section 6.3.3), the effects of the 2 percent power uncertainty factor are not directly included in methodology used for the analyses. Therefore, these analyses were performed at 102% of CPPU rated power to account for the 2 percent power uncertainty factor. Non-fuel dependent analysis power levels include a two percent power uncertainty factor unless a smaller value is specifically justified or the uncertainty is accounted for in the analysis methods. Additionally, Regulatory Guide 1.49 does not apply to some events that have been historically analyzed from nominal initial conditions, which include the Anticipated Transient Without Scram (section 15.8) and Station Blackout (section 15.9) events.

Thus, the SSES units continue to meet the intent of the Guide, which is to assure that all design calculations are performed at the highest possible power level that the plant can be operating.

Regulatory Guide 1.50 - CONTROL OF PREHEAT TEMPERATURE FOR WELDING OF LOW-ALLOY STEEL (May 1973)

The control of preheat temperature for welding of low-alloy steel is described in Subsection 5.2.3.3.2.1.

Regulatory Guide 1.51 - INSERVICE INSPECTION OF ASME CODE CLASS 2 AND 3 NUCLEAR POWER PLANT COMPONENTS

Withdrawn July 15, 1975.

Regulatory Guide 1.52 - DESIGN, TESTING, AND MAINTENANCE CRITERIA FOR ENGINEERED SAFETY FEATURE ATMOSPHERE CLEANUP SYSTEM AIR FILTRATION AND ADSORPTION UNITS OF LIGHT WATER COOLED NUCLEAR POWER PLANTS (Revision 1, July 1976 and Revision 2, March 1978)

The filter adsorber systems are designed to mitigate exposures resulting from a design basis accident. The Control Structure Emergency Outside Air Supply System (CSEOASS) and the Standby Gas Treatment System (SGTS) are the only systems that are subject to the requirements of this regulatory guide.

Subject to the clarifications and/or exceptions indicated below, the general intent of this regulatory guide has been met by the current design of the plant. Items (1) through (10) and (13) apply to Revision 1 and items (11) and (12) apply to Revision 2 of the Regulatory Guide 1.52.

(1) Reference: Position C.2.a. Moisture separators are used only where moisture impingement may be a problem. The SGTS is the only system with moisture separators. Heaters are used on both systems (SGTS and CSEOASS) to control humidity before filtration.

(2) Reference: Position C.2.d. Devices, such as pressure relief valves, are not used on either the CSEOASS or SGTS. Neither filtration system is subject to containment pressures (internal or external) or hazardous pressure surges.

(3) Reference: Position C.2.g. The pertinent pressure drop which is instrumented to signal, alarm, and record in the control room is the pressure drop across the first HEPA filter. The SGTS also alarms on high differential pressure across the entire filter system. The flow rate and low flow alarm also indicate proper functioning of the fan.

(4) Reference: Position C.2.i. Overall design considerations include reduction of radiation exposures during routine maintenance and testing insofar as effectually possible. It is envisioned, however, that workers will not handle filter units after a design basis accident and will thereby avoid exposures associated with immediate post-accident filter handling. Accordingly, no efforts were made toward a unitized atmosphere cleanup train design in the interest of accident exposure reduction.

(5) Reference: Position C.3.d. Since none of the HEPA filters separators are exposed to potential iodine removal spray, the units are not designed for contact with the spray. 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.

(6) Reference: Position C.3.e through C.3.h. In these sections and all others where reference is made to ORNL-NSIC-65, the reference is understood to be to ERDA 76-21 or ANSI N509 where appropriate.

(7) Reference: Position C.3.i. The adsorber beds are designed for 2.5 mg of iodine (both stable and radioactive) per gram of activated carbon averaged over the bed depth. This is consistent with the background information. Each replacement batch of impregnated, activated carbon shall meet the qualification and batch test results summarized in Table 5-1 of ANSI N509-1980, rather than those of Table 2 in Regulatory Guide 1.52, Revision 1, except that the 350 ft^3 batch size limit specified in Table 2 shall be retained.

(8) Reference: Position C.3.k. All systems are designed for low flow in order to control temperature rise. Oxidation effects are considered. A water spray system, provided only to minimize property loss in the event of fire, is designed for the control structure OA supply units, to extinguish a fire by flooding the adsorber units. The fire extinguishing system in the SGTS filter, sprays large quantities of water over the charcoal adsorber, until the charcoal temperature drops below its ignition temperature. The water is removed from the SGTS housing through automatic drain valves. The sprays or quenches are not provided for prevention of fire inception.

(9) Reference: Positions C.4.c and C.4.d. The spacing requirement is applicable to systems requiring operator access to remove filters and adsorber trays. Where unnecessary, the space is not provided, e.g., gasketless carbon absorbers which are filled and emptied externally.

(10) Reference: Position C.4.d. The length of pipe associated with manifolding would promote plate-out of the constituents of the sampled gas stream, thereby resulting in erroneous test results. The test probes are located in readily accessible locations; a minimum run of piping is used and manifolding is not employed.

(11) Reference: Positions C.5.a, C.5.c and C.5.d - In-place testing criteria identified in Paragraph C.5.a, C.5.c and C.5.d of Rev. 2 dated March, 1978 of the Regulatory Guide 1.52 are implemented, however, the testing frequency for C.5.c is 24 months rather than 18 months.

(12) Reference: Positions C.6.a and C.6.b - Laboratory Testing Criteria for activated carbon identified in Paragraph C.6.a and C.6.b of Rev. 2 dated March, 1978 of the Regulatory Guide 1.52 are implemented with the following exceptions:

a. Representative samples of used activated carbon will be tested at ≤30°C and ≥70% relative humidity and in accordance with ASTM D3803-89.

b. New activated carbon will meet the performance requirements and physical property specifications given in Table 5-1 of ANSI N509-1980.

Note: Table 6.5-2 provides details of how various positions (C1 to C4) of Revision 0 of Regulatory Guide 1.52 are met in the design of SGTS and CSEAOSS.

(13) Reference: Position C.4.e. The frequency and duration of operating the cleanup train with the heater in operation is in accordance with the Plant Technical Specifications and Surveillance Requirements.

Regulatory Guide 1.53 - APPLICATION OF THE SINGLE-FAILURE CRITERION TO NUCLEAR POWER PLANT PROTECTION SYSTEMS (June 1973)

The Susquehanna SES complies with this guide in the design of protection, safeguards actuation, and Class 1E electrical systems. Related considerations pertaining to cable separation and associated circuits are considered in the discussion of Regulatory Guide 1.75 and environmental considerations in the discussion of Regulatory Guide 1.89.

Regulatory Guide 1.54 - QUALITY ASSURANCE REQUIREMENTS FOR PROTECTIVE COATINGS APPLIED TO WATER COOLED NUCLEAR POWER PLANTS (June 1973)

For the non-NSSS scope of supply, a quality assurance program for coatings was in compliance with this regulatory guide.

For the NSSS scope of supply, the quality assurance records requirements in this regulatory guide were not imposed on painting material and paint application for Susquehanna SES since these coatings cover a relatively small exposed surface area.

Regulatory Guide 1.55 - CONCRETE PLACEMENT IN CATEGORY I STRUCTURES (June 1973)

Concrete placement in Seismic Category I structures is in accordance with the regulatory positions of this guide as described in Appendix 3.8B.

Concrete placement for the Diesel Generator 'El Building is in accordance with ACI 349, "Code Requirements for Nuclear Safety-Related Concrete Structures" and ANSI N45.2.5 "Supplementary Quality Assurance Requirements for Installation, Inspection and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants."

Regulatory Guide 1.56 - MAINTENANCE OF WATER PURITY IN BOILING WATER REACTORS (June 1973)

GE Report, NEDO-10899, "Chloride Control in BWR Coolants," establishes General Electric's position on water purity.

Regulatory Guide 1.57 - DESIGN LIMITS AND LOADING COMBINATIONS FOR METAL PRIMARY REACTOR CONTAINMENT SYSTEM COMPONENTS (June 1973)

The primary containments for Susquehanna SES are reinforced concrete structures. Nonetheless, the provisions of this regulatory guide are applicable to the following components of each containment:

1) Equipment hatch with personnel lock
2) Equipment hatch
3) Drywell head assembly
4) CRD removal hatch
5) Suppression chamber access hatches
6) Pipe and electrical penetrations

These items were designed in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Subsection NE, for Class MC components, the 1971 Edition with addenda through Summer 1972. Allowable stress limits used for the design are in conformance with Regulatory Guide 1.57, Paragraph C.1.d and the ASME Boiler and Pressure Vessel Code, Section III, Subsection NE-3131.2 as specified in the Winter 1973 Addenda. A detailed discussion of these design features is contained in Subsection 3.8.2.

Regulatory Guide 1.58 - QUALIFICATIONS OF NUCLEAR POWER PLANT INSPECTION, EXAMINATION, AND TESTING PERSONNEL (Revision 1, September 1980)

The Quality Assurance Program for the construction of Susquehanna SES is described in the PSAR, Appendix D and amendments. Compliance of the Operational Quality Assurance Program with this guide is discussed in Section 17.2.

Regulatory Guide 1.59 - DESIGN BASIS FLOODS FOR NUCLEAR POWER PLANTS (Revision 2 August 1977)

The design basis flood, discussed in Section 2.4, is determined in accordance with the regulatory positions of this guide.

Regulatory Guide 1.60 - DESIGN RESPONSE SPECTRA FOR SEISMIC DESIGN OF NUCLEAR POWER PLANTS (Revision 1, December 1973)

The design response spectra used in the analysis of Susquehanna SES, except the Diesel Generator 'E' facility, are different from those of the regulatory guide. A detailed discussion of the design response spectra is presented in Subsection 3.7b.1.

Regulatory Guide 1.61 - DAMPING VALUES FOR SEISMIC DESIGN OF NUCLEAR POWER PLANTS (October 1973)

The damping values used in the original seismic design of Susquehanna SES, except the Diesel Generator 'E' facility, are different from the regulatory guide. A detailed discussion of the damping values is presented in Subsection 3.7b.1. For snubber elimination or other piping modifications, damping values from this Regulatory Guide may be used.

Regulatory Guide 1.62 - MANUAL INITIATION OF PROTECTIVE ACTIONS (October 1973)

The provisions for manual initiation of protective actions are described in Subsections 7.2.2.1.2.1.7, 7.2.4.1.1.2.2.4, 7.3.2a.1.2.1.9, 7.3.2a.2.2.1.7, 7.3.2a.5.2.7, 7.4.2.1.2.1.9, 7.4.2.2.2.1.9, 7.6.1b.3.1 and 8.1.6.1.0.

Regulatory Guide 1.63 - ELECTRIC PENETRATION ASSEMBLIES IN CONTAINMENT STRUCTURES FOR WATER COOLED NUCLEAR POWER PLANTS (Revision 1, May 1977)

Since the construction permit for Susquehanna SES was issued in November 1973, the provisions of Revision 1 to this regulatory guide (which supplements IEEE 317-1976) were not specifically considered in the design of Susquehanna SES. The design of the electric penetration assemblies is therefore in compliance with Regulatory Guide 1.63 dated October 1973 (which supplements IEEE 317-1972). Specifically, Sections 4.2.3, 4.2.4, 5.1.6, 5.2.2, 6.2, 6.3.3, and 6.4 of IEEE 317-1976 have not been incorporated.

The penetration assemblies are type tested. There are no provisions for periodic testing under simulated fault conditions.

Electrical penetration circuits are summarized as follows:

1. 480 Volt Circuits

Loads powered from 480 volt motor control centers are supplied via electrical penetrations equipped with #4 awg, #10 awg, or 4/0 awg copper conductors. Typical single line diagrams for each penetration conductor size are shown on Figure 3.13-1.

Overcurrent protection for penetrations considers both connected load characteristics and penetration time-current capabilities in accordance with the following guidelines:

a. 480V motor, less than 1.5 hp (Figure 3.13-2A)

    1. Penetration conductor #10 awg

    2. Overcurrent protection redundant, adjustable, magnetic only circuit breakers with a maximum setpoint of 45 amperes and a minimum setpoint which exceeds 200% of the motor locked rotor current.

b. 480V motor, 10 hp or less (Figure 3.13-2B)

    1. Penetration conductor #10 awg

    2. Overcurrent protection redundant, fixed, thermal magnetic circuit breakers with a maximum thermal trip rating of no more than 40 amperes and a minimum thermal trip rating which exceeds 200% of the motor full load current.

c. 480V motor, 10.1 to 20 hp (Figure 3.13-3)

    1. Penetration conductor #4 awg

    2. Overcurrent protection redundant, fixed, thermal magnetic circuit breakers with a maximum thermal trip rating of no more than 100 amperes and a minimum thermal trip rating which exceeds 200% of the motor full load current.

d. 480V non-motor loads (presently only non-motor loads are hydrogen recombiners.)

    1. Penetration conductor 4/0 awg (Figure 3.13-8)

    2. Overcurrent protection-redundant, fixed thermal magnetic circuit breakers with a maximum thermal trip rating of no more than 150 amperes and a minimum thermal trip rating which exceeds 150% of the connected load full load current.

Credit is not taken for penetration protection provided by other overload detectors, such as, motor overloads shown in Figure 3.13-1. Penetration seal withstand curves (Figures 3.13-2, 3.13-3, and 3.13-8) apply for the condition when mechanical seal integrity is maintained, but electrical integrity may be comprised.

Adequacy of the subject molded case circuit breaker selections are demonstrated on Figures 3.13-2, 3.13-3 and 3.13-8 by showing time-current characteristic curves with the protective device total clearing time below, and to the left of the penetration seal withstand curve.

2. 120 Volt AC Control Circuits

There are two types of 120 volt AC control circuits to be considered: (1) circuit powered by a control transformer located in an MCC cubicle and (2) circuit powered by a 120 volt AC instrument distribution panel (see Figure 3.13-4). #14 AWG is the minimum size used for control circuits.

a. The motor control circuits are powered by control transformers located in the respective MCC cubicles. Control transformers are sized to meet the requirement of the control circuit (one for each starter). Typically a 120 VA control transformer is used with a NEMA Size 1 starter and a 200 VA is used with a NEMA Size 2 starter. The largest control transformer used in connection with a penetration is 350 VA. The maximum short circuit current that can be delivered by a 350 VA transformer in a control circuit is approximately 100 amps. A fuse, rated 3.2 amp or less, located in the respective MCC provides the circuit protection. Since sustained short circuit current will destroy the control transformer before the integrity of the penetration assembly is compromised, backup fuses are not utilized (refer to Figure 3.13-4).

b. Control circuits that emanate from the fuse control panel will have a 10 amp or lower rated fuse at the panel and as a back-up breaker either an identical fuse in series or a breaker in the 120 V instrument AC distribution panel. 20 amp or smaller breaker or fuse will be used for backup protection (refer Figure 3.13-4). All breakers shown in Figure 3.13-4, Case 2, are molded case and self-actuated (short circuit current trip with manual closing).

3. 125 VDC Control Circuits

Each 125 VDC control circuit is protected by a 20 amp or smaller fuse located in a control panel with back-up protection provided by a 20 amp breaker (ITE type E) in the dc distribution panel. The mechanical integrity of the penetration assembly is maintained under overload or faults conditions (refer to Fig. 3.13-5).

4. 120 Volt Lighting and Space Heater Circuits

Mechanical integrity of all 120 V lighting and space heater circuits is maintained under overload or fault condition. Each type of circuit is discussed below:

a. Each 120 V lighting circuit is provided with a 20 amp breaker with back-up protection of a 50 amp breaker as shown on Fig. 3.13-6.

b. Each 120 V motor (except reactor recirc pump motor) or motor operated valve (MOV) space heater is provided with a 20 amp or smaller fuse protection. Backup protection is provided by a 20 amp or smaller breaker located in a lighting panel (see 4(a) of above).

c. Each of the 120 V space heater circuits for the reactor recirc pump motors is provided with a 40 amp breaker as the primary protection. Backup protection is provided by a 50 amp breaker located in a 280/120 V lighting panel.

5. Medium Voltage Circuits (above 480 volt)

The only medium voltage equipment located inside the containment are two variable frequency reactor recirc. pump motors. These two recirc. pump motors are fed by two independent 13.8 kV M-G (motor-generator) sets with the generator output rated at 3,920 volts. The M-G sets are located in the turbine building. A calculated M-G set generator decrement curve together with the penetration cable thermal curves is shown on Figure 3.13-7. The maximum short circuit current that can be produced by the generator is 7000 amp asymmetrical. Figure 3.13-7 shows that a line-to-line short circuit (across lines which do not furnish input power to the voltage regulator) of about 6,000 amps can be sustained if the generator field breaker and the main feeder to the M-G set drive motor are not tripped. However, the existing protection schemes have redundant detection devices and redundant protection devices for the penetration circuits. In addition, redundant Class 1E overcurrent protection is provided (one overcurrent relay for each Recirculation Pump Trip (RPT) breaker). This protection scheme ensures that the fault current would be cleared before any damage. Therefore, as shown by Figure 3.13-7, the mechanical integrity of the penetration assembly is maintained under the most severe fault condition.

6. Instrumentation Circuits

The instrumentation circuits are low level signal circuits (ma or mv range) such as thermocouples, RTD, etc. These circuits are current limiting. In addition, instrument cables are not routed with any other types of circuits in the same raceway. Therefore, backup protection is not needed.

The requirement for periodic testing and inspecting of fuses, breakers, and containment circuit protection schemes are stated in SSES Technical Requirements Manual. Requirements for back-up penetration circuit protection are met.

Regulatory Guide 1.64 - QUALITY ASSURANCE REQUIREMENTS FOR THE DESIGN OF NUCLEAR POWER PLANTS (Revision 2, June 1976)

The quality program for the design for the Susquehanna SES is described in PSAR Appendix D and amendments. Compliance of the Operational Quality Assurance Program with this regulatory guide is described in Section 17.2.

Regulatory Guide 1.65 - MATERIALS AND INSPECTIONS FOR REACTOR VESSEL CLOSURE STUDS (October 1973)

Subsections 5.2.4 and 5.3.1.7 describe the materials and inspections for reactor vessel closure studs.

Operation will be conducted in accordance with this regulatory guide with the following exception:

All studs will not be removed from the vessel prior to flooding the reactor cavity. Other means will be employed to prevent corrosion.

Regulatory Guide 1.66 - NONDESTRUCTIVE EXAMINATION OF TUBULAR PRODUCTS (October 1973)

Withdrawn September 28, 1977.

Regulatory Guide 1.67 - INSTALLATION OF OVERPRESSURE PROTECTION DEVICES (October 1973)

This regulatory guide is not applicable to Susquehanna SES since the main steamline safety/relief valves discharge into closed systems.

Regulatory Guide 1.68 - INITIAL TEST PROGRAM FOR WATER-COOLED REACTOR POWER PLANTS (Revision 1, January 1977)

Subject to the clarifications and/or exceptions indicated in Subsection 14.2.7, the provisions of this regulatory guide were met by the test programs instituted during the startup of each unit.

The design of the A-D Diesel Generators meets the intent of this Regulatory Guide. The applicable Appendix A of Regulatory Guide 1.68 provides acceptable preoperational testing criteria for emergency/standby AC power supplies. These testing requirements were implemented and led to meeting the intent of 10CFR50, Appendix B.

Regulatory Guide 1.68 - (TASK SC 704-5) INITIAL TEST (DIESEL GENERATOR E ONLY) PROGRAMS FOR WATER-COOLED NUCLEAR POWER PLANTS (Revision 2, August 1978)

The design of the Diesel Generator "E" meets the intent of this Regulatory Guide. This revision added "Emergency loads supplied should be confirmed to be in agreement with design sizing assumptions used for power supplies" to the applicable Appendix A, Section g(3) of Regulatory Guide 1.68. Methods of control of welding, in fabricating and joining safety-related austenitic stainless steel components and systems, were implemented which led to meeting the intent of 10 CFR 50, Appendix B.

Regulatory Guide 1.68.1 - PREOPERATIONAL AND INITIAL STARTUP TESTING OF FEEDWATER AND CONDENSATE SYSTEMS FOR BOILING WATER REACTOR POWER PLANTS (Revision 1, January 1977)

The preoperational testing and initial startup testing of the feedwater and condensate systems associated with Susquehanna SES were conducted in accordance with the provisions of this regulatory guide.

Regulatory Guide 1.68.2 - INITIAL STARTUP TEST PROGRAM TO DEMONSTRATE REMOTE SHUTDOWN CAPABILITY FOR WATER-COOLED NUCLEAR POWER PLANTS (January 1977)

The portion of the initial startup test program used to demonstrate the remote shutdown capability of each unit was conducted in accordance with the provisions of this regulatory guide.

Regulatory Guide 1.69 - CONCRETE RADIATION SHIELDS FOR NUCLEAR POWER PLANTS (December 1973)

The design and placement of the concrete used for radiation shielding differs from the provisions of this Regulatory Guide. The practices employed for the Susquehanna SES are described in Appendix 3.8B.

Regulatory Guide 1.70 - STANDARD FORMAT AND CONTENT OF SAFETY ANALYSIS REPORTS FOR NUCLEAR POWER PLANTS-LWR EDITION (Revision 2, September 1975)

The format of this FSAR complies with this regulatory guide except that replacement pages will contain a change indicator (vertical bar in the margin of text and table pages) and a page change identification consisting of a revision number. The date of the change will not be shown on replacement pages. This is consistent with the requirements of 10 CFR 50.71(e)(5).

With respect to physical specifications, material submitted may be submitted on CD-ROM in accordance with the guidance of NRC Regulatory Issue Summary 2001-05, dated January 25, 2001.

Regulatory Guide 1.71 - WELDER QUALIFICATION FOR AREAS OF LIMITED ACCESSIBILITY (December 1973)

Exceptions are taken to this regulatory guide as specified below:

(1) Reference: Position C.I. Performance qualifications for field personnel who weld under conditions of limited access, as defined in Regulatory Position C.1, are maintained in accordance with the applicable requirements of ASME Sections III and IX.

For the welder qualifications for the reactor coolant pressure boundary, see Subsection 5.2.3.3.2.3.

Regulatory Guide 1.72 - SPRAY POND PLASTIC PIPING December, 1973

Plastic piping is not used in safety related applications.

Regulatory Guide 1.73 - QUALIFICATION TESTS OF ELECTRIC VALVE OPERATORS INSTALLED INSIDE THE CONTAINMENT OF NUCLEAR POWER PLANTS (January 1974)

The requirements of this regulatory guide are met as described in Section 3.11.

Regulatory Guide 1.74 - QUALITY ASSURANCE TERMS AND DEFINITIONS (February 1974)

The Susquehanna SES construction quality program is described in PSAR Appendix D and amendments. Compliance of the operational Quality Assurance Program with this guide is described in Section 17.2.

Regulatory Guide 1.75 - PHYSICAL INDEPENDENCE OF ELECTRIC SYSTEMS (Revision 1, January 1975)

A. This Regulatory Guide endorses IEEE 384-1974 subject to the additions and clarifications delineated in Section C of the guide. Although there is no requirement for Susquehanna SES to comply with Regulatory Guide 1.75 and IEEE 384, Susquehanna SES follows this separation criteria, subject to the clarifications and exceptions below for the NSSS scope of supply. The paragraphs below reference sections of the IEEE standard in all cases and the specific paragraphs of the regulatory position statement where applicable.

(1) Reference: Regulatory Guide 1.75 and Section 4.5 of IEEE 384-1974

Certain power cables are subject to the same requirements as a Class 1E circuit. This applies to derating, environmental qualification, flame retardance, splicing restrictions, and raceway fill.

(2) Reference: Sections 4.5 and 4.6 of IEEE 384-1974

See Section 8.1.6.1 (Regulatory Guide 1.75).

(3) Reference: Sections 5.1.3, 5.1.4, and 5.6.2 of IEEE 384-1974

All annunciator and computer input circuits are classified as non-Class 1E circuits. These non-Class 1E circuits are not separated from Class 1E control circuits within Class 1E panels in which the non-Class 1E circuit derives its input, i.e., circuit breaker auxiliary contact used for computer input, etc., nor are they separated in PGCC cable ducts. These non-Class 1E instrument circuits are considered to be low energy and the probability of these non-Class 1E circuits providing a mechanism of failure to the Class 1E circuits is extremely low.

(4) Reference: Position C.15 of Regulatory Guide 1.75 and Section 5.3.1 of IEEE 384-1974

See Section 8.1.6.1 (Regulatory Guide 1.75).

(5) Reference: <u>Sections 5.6.2 and 5.6.3 of IEEE 384-1974</u>

In general, the circuits for redundant Class 1E systems and the circuits for non-Class 1E systems are located in separate panelboards, boxes, racks, and enclosures. Panels, racks, and boxes that contain wiring and devices for Class 1E circuits are labeled distinctly to externally identify the separation system and grouping. Internal to the enclosures, devices such as relays, switches, and instruments, are uniquely identified. In addition, external cables are color coded and marked to be readily identifiable. These methods of identification are described in Subsection 3.12.3.2.

Where required, physical separation is achieved either by a minimum of 6" horizontal and vertical separation, steel barriers, metallic enclosures, or metallic flexible conduit.

Where the above separation methods are not feasible, one of the separation groups are to be covered with one of the following qualified non-flammable materials:

i. Haveg Industries, Siltemp sleeving type S or woven tape type WT65.

ii. Carborundum, Fiberfrax sleeving type HP144T or woven tape type 3L144T.

These materials have been qualified to be used as separation barriers (Wyle Lab. Test Report No. 56669 dated May, 1980). Applications of these materials are controlled and documented.

(6) Reference: <u>Section 5.5 of IEEE 384-1974</u>

See Section 8.1.6.1 (Regulatory Guide 1.75).

Additional information is found in Sections 7.1, 7.2, 7.3, 7.4 and 7.6

B. Compliance to this Regulatory Guide for the non-NSSS scope of supply is discussed in Section 8.1.6.1 (Regulatory Guide 1.75), and for D/G E in Subsection 8.1.6.1.r.

<u>Regulatory Guide 1.76</u> - <u>DESIGN BASIS TORNADO FOR NUCLEAR POWER PLANTS (April 1974)</u>

For all tornado-resistant structures except the Diesel Generator 'E' Building, the following parameters were used in lieu of those presented in Table I of this regulatory guide.

Tangential speed:	300 mph
Translational speed:	60 mph
Rate of pressure drop:	1 psi/sec (for 3 seconds)

A detailed discussion of these parameters is contained in Subsection 3.3.2.

The design basis tornado used for the Diesel Generator 'E' Building is in accordance with this regulatory guide.

Regulatory Guide 1.77 - ASSUMPTIONS USED FOR EVALUATING A CONTROL ROD EJECTION ACCIDENT FOR PRESSURIZED WATER REACTORS (May 1974)

Not Applicable.

Regulatory Guide 1.78 - ASSUMPTIONS FOR EVALUATING THE HABITABILITY OF A NUCLEAR POWER PLANT CONTROL ROOM DURING A POSTULATED HAZARDOUS CHEMICAL RELEASE (Revision 1 December 2001)

As described in Subsection 6.4.4.2, the steps taken to protect control room habitability conform to requirements of this regulatory guide.

Regulatory Guide 1.79 - PREOPERATIONAL TESTING OF EMERGENCY CORE COOLING SYSTEMS FOR PRESSURIZED WATER REACTORS (Revision 1, September 1975)

Not Applicable.

Regulatory Guide 1.80 - PREOPERATIONAL TESTING OF INSTRUMENT AIR SYSTEMS (June 1974)

The primary containment instrument gas system will be tested in accordance with the requirements of Regulatory Guide 1.80, Sections C1 through C.6. The portions of the instrument air system which supply safety-related equipment will also be tested in accordance with Section C.1 through C.6 of the Regulatory Guide 1.80 (June, 1974). Loss of air testing will be done in the various system preoperational tests. Systems/components which have separate accumulators will be tested with a loss of air/gas to ensure that the accumulators function in accordance with design. Testing described in Regulatory Guide 1.80, Sections C7 through C10 will not be done in the instrument air system or primary containment gas system tests.

Regulatory Guide 1.81 - SHARED EMERGENCY AND SHUTDOWN ELECTRIC SYSTEMS FOR MULTI-UNIT NUCLEAR POWER PLANTS (Revision 1, January 1975)

The design of the standby electric power systems which use shared diesels complies with this Regulatory Guide by invoking the provisions of Position C.2. See Section 8.1.6.1.

Regulatory Guide 1.82 - SUMPS FOR EMERGENCY CORE COOLING AND CONTAINMENT SPRAY SYSTEMS (June 1974)

Not Applicable.

Regulatory Guide 1.83 - IN-SERVICE INSPECTION OF PRESSURIZED WATER REACTOR STEAM GENERATOR TUBES (Revision 1, July 1975)

Not Applicable.

Regulatory Guide 1.84 - CODE CASE ACCEPTABILITY, ASME SECTION III DESIGN AND FABRICATION

The revision date to this Regulatory Guide is intentionally not stated. Regulatory Guide 1.84 is frequently revised to append new Code Cases. No previous compliance requirements to earlier revisions are negated by updates to this Regulatory Guide.

The Susquehanna SES for the non-NSSS scope of supply will use only those code cases listed in this regulatory guide. In accordance with Section D of the Regulatory Guide, Code Cases approved by earlier revisions of the Regulatory Guide may have been invoked. In addition, ASME Code Case N-316 and N-640 have been approved by the NRC for use at Susquehanna Steam Electric Station. Should it be deemed necessary or beneficial to apply other code cases, a specific request shall be made to the NRC.

ASME Code Case N-516-2 has been approved by the NRC for use during the second 10-year inspection interval at Susquehanna SES in a letter dated February 21, 2002 (Relief Request for Authorization to use Code Case N-516-2 as an Alternative to the ASME Code). The use of Code Case N-516-2 is subject to the following three conditions and limitations:

1. Performance qualifications shall be in accordance with Paragraph 3.2 in Code Case N-516-2, except that immediate retest following a failed mechanical bend test shall be in accordance with ASME, Section IX, QW-320.

2. Procedure qualification shall be in accordance with Paragraph 3.1 in Code Case N-516-2. The Alternative Procedure Qualification Requirements of Paragraph 5.0 shall not be used except as noted in Paragraph 4.(b)(4) for the additional requirements for qualification of filler metal.

3. When welding is to be performed on high neutron fluence Class 1 material, then a mockup, using material with similar fluence levels, should be welded to verify that adequate crack prevention measures were used.

The NSSS scope of supply procedure for meeting the regulatory requirements is to obtain NRC approval for code cases applicable to Class I components only. Approval of code cases for Class 2 and 3 equipment was not required at the time of the design of the Susquehanna SES, and is not currently required by 10CFR50.55a. Therefore, GE believes that this procedure in conjunction with 10CFR50APPB and other regulatory requirements provide adequate assurance of quality in the design and fabrication of safety-related equipment (see Subsection 5.2.1.2).

Regulatory Guide 1.85 - CODE CASE ACCEPTABILITY, ASME SECTION III MATERIALS

The revision date to this Regulatory Guide is intentionally not stated. Regulatory Guide 1.85 is frequently revised to append new code cases, no previous compliance requirement to earlier revisions are negated by updates to this Regulatory Guide. The Susquehanna SES for non-NSSS scope of supply will use only those code cases listed in this regulatory guide. In accordance with Section D of the Regulatory Guide, Code Cases approved by earlier revisions of the Regulatory Guide may have been invoked. In addition, ASME Code Case 1481-1 has been approved by the NRC for use at Susquehanna Steam Electric Station. Should it be deemed necessary or beneficial to apply other code cases a specific request shall be made to the NRC.

The NSSS scope of supply procedure for meeting the regulatory requirements is to obtain NRC approval for code cases applicable to Class 1 components only. Approval of code cases for Class 2 and 3 equipment was not required at the time of the design of the Susquehanna SES, and is not currently required by 10CFR50.55a. Therefore, GE believes that this procedure in conjunction with 10CFR50APPB and other regulatory requirements provide adequate assurance of quality in the materials of safety-related equipment (see Subsection 5.2.1.2).

Regulatory Guide 1.86 - TERMINATION OF OPERATING LICENSES FOR NUCLEAR REACTORS (June 1974)

The Susquehanna SES will comply with this regulatory guide.

Regulatory Guide 1.87 - CONSTRUCTION CRITERIA FOR CLASS 1 COMPONENTS IN ELEVATED TEMPERATURE REACTORS (Supplement to ASME Section III Code Cases 1592, 1593, 1594, 1595, 1596) (Revision 1, June 1975)

Not Applicable.

Regulatory Guide 1.88 - COLLECTION, STORAGE, AND MAINTENANCE OF NUCLEAR POWER PLANT QUALITY ASSURANCE RECORDS (Revision 2, October 1976)

The quality assurance program for the construction of the Susquehanna SES is described in the PSAR, Appendix D and amendments. Compliance of the Operational Quality Assurance Program with this guide is described in Section 17.2.

Regulatory Guide 1.89 - QUALIFICATION OF CLASS 1E EQUIPMENT FOR NUCLEAR POWER PLANTS (November 1974)

For the non-NSSS scope of supply, the degree of compliance with this regulatory guide and justification for any exceptions to this guide are provided in Section 3.11.

For the NSSS scope of supply, see Subsections 3.9.2.2a.2.7, 3.9.2.2a.2.9, 3.11.2, 7.2.2.1.2.1.11, and 7.3.2a.2.2.1.10.

Regulatory Guide 1.90 - IN-SERVICE INSPECTION OF PRESTRESSED CONCRETE CONTAINMENT STRUCTURES WITH GROUTED TENDONS (Revision 1, August 1977)

Not Applicable.

Regulatory Guide 1.91 - EVALUATION OF EXPLOSIONS POSTULATED TO OCCUR ON TRANSPORTATION ROUTES NEAR NUCLEAR POWER PLANT SITES (January 1975)

An examination of the historical data for the surface transportation of explosive material near the Susquehanna SES indicates that the commercial truck and rail transportation routes are farther from the station's vital structures than the distances delineated in Figure 2 of this regulatory guide for plants situated in Tornado Region 1, as defined in Regulatory Guide 1.76. Although the closest point of approach from the Susquehanna River to the vital structures of the Susquehanna SES is less than the distance specified in this regulatory guide for the largest probable quantity of explosive material transported by ship (i.e., 5,000 tons of TNT), the Susquehanna River is not commercially navigable for purposes of transporting large quantities of explosives. Therefore, this type of accident is not considered to be probable enough to evaluate.

Regulatory Guide 1.92 - COMBINING MODAL RESPONSES AND SPATIAL COMPONENTS IN SEISMIC RESPONSE ANALYSIS (Revision 1, February 1976)

Since the construction permit for the Susquehanna SES was issued in November 1973, the methods of combining modal responses and spatial components in seismic response analysis, as described in this regulatory guide, were not specifically considered in the original design, except for the Diesel Generator 'E' facility. The methods of design and analysis for structures, components, and piping systems that have been employed are described in Sections 3.7a, 3.7b, and 3.9.

Regulatory Guide 1.92 shall be invoked for the analysis of snubber elimination or other piping modifications whenever Code Case N-411 or Regulatory Guide 1.61 damping values are used.

Regulatory Guide 1.93 - AVAILABILITY OF ELECTRIC POWER SOURCES (December 1974)

Compliance with this guide is discussed in Subsection 8.1.6.1.u.

Regulatory Guide 1.94 - QUALITY ASSURANCE REQUIREMENTS FOR INSTALLATION, INSPECTION, AND TESTING OF STRUCTURAL CONCRETE AND STRUCTURAL STEEL DURING THE CONSTRUCTION PHASE OF NUCLEAR POWER PLANTS (Revision 1, April 1976)

The quality assurance program for the construction of Susquehanna SES is described in the PSAR, Appendix D and amendments. Compliance of the Operational Quality Assurance Program with this guide is described in Section 17.2.

Regulatory Guide 1.95 - PROTECTION OF NUCLEAR POWER PLANT CONTROL ROOM OPERATORS AGAINST AN ACCIDENTAL CHLORINE RELEASE (February 1975)

This regulatory guide was superceded by Revision 1 of Regulatory Guide 178.

Regulatory Guide 1.96 - DESIGN OF MAIN STEAM ISOLATION VALVE LEAKAGE CONTROL SYSTEMS FOR BOILING WATER REACTOR NUCLEAR POWER PLANTS (Revision 1, June 1976)

Subject to the clarification indicated below, the provisions of this regulatory guide are met by the current plant design.

(1) Reference: Appendix A, Paragraph 6. The design and inspection of this portion of the leakage control system is in accordance with the provisions of Section XI of the ASME Boiler and Pressure Vessel Code. The 100% volumetric inspection of this portion of the system is specifically exempted from the requirement for volumetric inspection by paragraph IWB-1220(a) of Section XI of the ASME Boiler and Pressure Vessel Code (Summer 1975 Addenda).

Note: MSIV-LCS information maintained here for historical purposes. The MSIV-LCS has been deleted. The function is now performed by the Isolated Condenser Treatment Method (Section 6.7), approved by the NRC as an alternative.

Regulatory Guide 1.97 - INSTRUMENTATION FOR LIGHT WATER COOLED NUCLEAR POWER PLANTS TO ASSESS PLANT CONDITIONS DURING AND FOLLOWING AN ACCIDENT (Revision 2, December 1980)

The accident monitoring instrumentation was designed prior to this Regulatory Guide being issued. The instrumentation for accident monitoring has not been evaluated against Revision 1 to the Regulatory Guide. With the exception of the item discussed below, our position on Revision 2 to the Regulatory Guide is provided in PLA-965, dated November 13, 1981. Compliance is described in more detail in the applicable sections of Chapter 7.

The instrumentation for accident monitoring is not specifically identified on the control panels. The control panel layouts and instrument identification at Susquehanna SES are based on good human factors engineering for the presentation of information to the control room operator. The

Susquehanna SES Detailed Control Room Design Review validated our choice of instrument identification scheme. That review indicated that the use of redundant labeling as would have been required to specifically identify accident monitoring equipment would serve to confuse and distract operator performance rather than enhance it.

Subject to the clarifications delineated in PLA-2222, the exception above and the exceptions discussed in FSAR section 7.1.2.6.21, the provisions of this regulatory guide are met.

Regulatory Guide 1.98 - ASSUMPTIONS USED FOR EVALUATING THE POTENTIAL RADIOLOGICAL CONSEQUENCES OF A RADIOACTIVE OFFGAS SYSTEM FAILURE IN A BOILING WATER REACTOR (March 1976)

Subject to the clarifications or exceptions indicated below, the assumptions of Regulatory Guide 1.98 are followed in the analyses of the offgas system failure in Subsection 15.7.1.

(1) Reference: Position C.4.a. Dose consequences are expressed in terms of REM TEDE. Dose conversion factors are given in Appendix 15B.

Regulatory Guide 1.99 - RADIATION EMBRITTLEMENT OF REACTOR VESSEL MATERIALS (Revision 2, May 1988)

The methods of Regulatory Guide 1.99, Revision 2 are followed in the analyses of reactor pressure vessel fracture toughness in Subsection 5.3.

Regulatory Guide 1.100 - SEISMIC QUALIFICATION OF ELECTRIC EQUIPMENT FOR NUCLEAR POWER PLANTS (March 1976)

The implementation paragraph of this regulatory guide states that the requirements of the position statements will only be applied to plants that received construction permits after November 16, 1976. The Construction Permit for Susquehanna SES was issued in November 1973 and therefore the guidelines of this regulatory guide were not utilized in the design of this nuclear power station. However, PP&L conducted a reassessment of the original equipment qualification using the criteria contained in Regulatory Guide 1.100, Rev. 1.

Seismic qualification of the safety related electric equipment (non-NSSS scope of supply) has been conducted in accordance with the IEEE Standard 344-1971. Section 3.10 describes the complete qualification methods and procedures that have been utilized.

The safety-related electric equipment (NSSS scope of supply) meets IEEE 323-1971 and IEEE 344-1971.

The Diesel Generator 'E' facility is in accordance with Regulatory Guide 1.100, Rev. 1 and IEEE Standard 344-1975.

Regulatory Guide 1.101 - EMERGENCY PLANNING FOR NUCLEAR POWER PLANTS (Revision 3, August 1992)

The Susquehanna Emergency Plan complies with the provisions of this regulatory guide.

Regulatory Guide 1.102 - FLOOD PROTECTION FOR NUCLEAR POWER PLANTS (Revision 1, September 1976)

The present design of the Susquehanna SES complies with the provisions of this regulatory guide.

Regulatory Guide 1.103 - POST-TENSIONED PRESTRESSING SYSTEMS FOR CONCRETE REACTOR VESSELS AND CONTAINMENTS (Revision 1, October 1976)

Not Applicable.

Regulatory Guide 1.104 - OVERHEAD CRANE HANDLING SYSTEMS FOR NUCLEAR POWER PLANTS (February 1976)

Subject to the clarifications and exceptions indicated below, the safety related overhead crane handling systems of this station comply with the provisions of this regulatory guide.

(1) Reference: Position C.1.b(2). The nil-ductility transition temperature for the structural steel associated with the cranes was not determined as suggested by this position. Position C.1.b(3) states that a cold proof test represents an acceptable alternative to the requirements of Position C.1.b(2). Accordingly, a cold proof test will be performed in accordance with the general procedures and testing frequency suggested in Position C.4.d except as modified in Item 7 below.

(2) Reference: Position C.3.f. The paragraph states that "The fleet angles between individual sheaves for rope should not exceed 1 1/2 degrees.," however, the fleet angles for the cranes that have been purchased for the Susquehanna SES are 3 degrees 7 minutes.

This position also states that "the pitch diameter of the lead sheave should be 30 times the rope diameter for the 180-degree reverse bend, 26 times the rope diameter for running sheaves and drum,...". The pitch diameter of the running sheaves is 24 times that of the wire rope diameter.

Consistent with established industry standards and in light of the limited number of loading cycles, the present design is acceptable.

(3) Reference: Position C.3.g. This position states that the head block, rope reeving system and load block should be subjected to a load test of 200 percent of the design rated load. No such test has been specified for these components. ANSI Standard B30.2 allows for a 125 percent load test of these components and the purchase orders for these cranes have so specified this type of test.

(4) Reference: Position C.3.j. This position suggests that the designer should provide means within the reeving system located on the head or on the load block combinations to absorb or control the kinetic energy of rotating machinery prior to the incident of two blocking or load hangup. As an alternative to the regulatory position, each crane is provided with dual upper limit switches to preclude the possibility of a "two block" occurrence and an overload switch combined with overcurrent and rate of current rise cutouts to prevent a load hangup.

(5) Reference: Position C.3.p. This paragraph states that provisions should be made for manual operation of the brakes.

This regulatory position has not been incorporated in that there has been no provision made for manual bridge and trolley holding brake operation.

The position also recommends that the trolley and bridge speeds be limited to a maximum speed of 30 fpm for the trolley and 40 fpm for the bridge. The maximum speed for the bridge is actually 50 fpm, but the potential effects of this difference in maximum speed is compensated for by the substantial runway length (approximately 320 ft.) and a stepless type bridge speed control. Administrative controls will be instituted to ensure that a maximum trolley speed of 30 fpm is used when the main hoist is loaded.

(6) Reference: Position C.4.b. This position states that the complete hoisting machinery should be allowed to "two block" during the hoisting test and should also be tested for ability to sustain a load hangup condition. The testing of these conditions was not specified in the purchasing documents for these cranes since the reeving system is not designed for two blocking or load hangup as indicated in Item 4 above, which describes the alternate design that has been incorporated in lieu of the regulatory position C.3.j.

(7) Reference: Position C.4.d. This position states that the cold proof test should be performed at or below the minimum operating temperature of the structural members essential to the structural integrity of the crane and should use a dummy load equal to 1.25 times the maximum working load. It further states that "If it is not feasible to achieve the minimum operating temperature during the test, the dummy load should be increased beyond the design rated load 1.5 percent per °F temperature difference." The minimum operating temperature for the cranes is 60°F and by appropriate scheduling of the cold proof test and adjustment of the HVAC systems, it will be attempted to maintain this temperature. However, in no case will the dummy load be increased above an amount of 125 percent of the rated load. Crane testing in excess of 125 percent of the rated load will adversely affect the safety of the cranes, since any such tests may propagate undetectable material defects and thus increase the probability of crane component failures. Furthermore, such testing would violate the ANSI Standard B30.2 and Title 29CFR Part/1910.179(k). This position also states that cold-proof "test frequency should be approximately 40 months or less ...." The cold proof test will be conducted only once because the manufacturer recommends against repeatedly overloading the crane. Thereafter, all accessible welds whose failure could cause a critical load drop will be nondestructively examined every 4 years or less. This exception is consistent with Section 2.4 of NUREG 0554.

Regulatory Guide 1.105 - INSTRUMENT SPANS AND SET POINTS (Revision 1, November 1976)

Subject to the clarifications and/or exceptions indicated below, the provisions of this regulatory guide are met by the current plant design.

(1) Reference: Position C.4. The guide requires that instrumentation not exhibit certain mechanical characteristics during the environmental qualification tests performed in accordance with Regulatory Guide 1.89. These tests have not been performed for the Susquehanna SES as discussed in the response to Regulatory Guide 1.89.

(2) Reference: Position C.5. The guide requires that a securing device or equivalent be installed on all instrument set point mechanisms. All set points are provided with multiturn, "screwdriver" adjustments that have "built-in" friction to obviate the effect of vibratory inputs.

Regulatory Guide 1.106 - THERMAL OVERLOAD PROTECTION FOR ELECTRIC MOTORS ON MOTOR-OPERATED VALVES (Revision 0, November 1975; Revision 1, March 1977)

The requirements of Revision 0, November 1975 of this regulatory guide are met for Susquehanna SES except for the 'E' diesel generator. The requirements of Revision 1, March 1977 of this regulatory guide are met for the 'E' diesel generator. Compliance is discussed in Subsection 8.1.6.1.

Regulatory Guide 1.107 - QUALIFICATIONS FOR CEMENT GROUTING FOR PRESTRESSING TENDONS IN CONTAINMENT STRUCTURES (Revision 1, February 1977

Not Applicable.

Regulatory Guide 1.108 - PERIODIC TESTING OF DIESEL GENERATORS USED AS ONSITE ELECTRIC POWER SYSTEMS AT NUCLEAR POWER PLANT (Revision 1, August 1977) (INCLUDING ERRATA, SEPTEMBER 1977)

Subject to the clarifications and exceptions indicated below, the design of the diesel generators is in compliance with the provisions of this regulatory guide (including errata).

(1) Reference: Position C.1.b(5). Test according to requirements in Sections 14.2 and SSES Technical Specifications.

(2) Reference: Position C.2.a. Exception is taken to the statement that testing of diesel generators occur at least once every 18 months. The 18 month requirement is interpreted to mean once per refueling cycle. Periodic testing is performed in accordance with the Technical Specifications, and the test interval may be replaced with performance-based, risk-informed test intervals. This statement in Regulatory Position C.2.a clarifies the statement in Section 6.6.2 of IEEE 387-1977 that operational tests be performed “at acceptable intervals.” Despite the exception to Regulatory Position C.2.a

of RG 1.108, exception is not being taken to the statement in Section 6.6.2 of IEEE 387-1977 that the testing be performed at acceptable intervals. The performance-based, risk-informed test intervals are determined in accordance with Technical Specification 5.5.15 as approved by the NRC and are acceptable.

Regulatory Guide 1.109 - CALCULATION OF ANNUAL DOSES TO MAN FROM ROUTINE RELEASES OF REACTOR EFFLUENTS FOR THE PURPOSE OF EVALUATING COMPLIANCE WITH 10CFR50, APPENDIX I (Revision 1,October, 1977)

The assumptions of Regulatory Guide 1.109 are followed in the analysis of annual doses to man from routine releases presented in Sections 11.2 and 11.3.

Regulatory Guide 1.110 - COSTS-BENEFIT ANALYSIS FOR RADWASTE SYSTEMS FOR LIGHT WATER COOLED NUCLEAR POWER REACTOR (March 1976)

The requirements of this regulatory guide are met.

Regulatory Guide 1.111 - METHODS FOR ESTIMATING ATMOSPHERIC TRANSPORT AND DISPERSION OF GASEOUS EFFLUENTS IN ROUTINE RELEASE FROM LIGHT WATER COOLED REACTORS (Revision 1, July 1977)

The assumptions of Regulatory Guide 1.111 are followed in the analyses of atmospheric dispersion factors presented in Section 2.3 and of deposition rates presented in Section 11.3.

Regulatory Guide 1.112 - CALCULATION OF RELEASES OF RADIOACTIVE MATERIALS IN GASEOUS AND LIQUID EFFLUENTS FROM LIGHT WATER COOLED POWER REACTORS (April 1976)

The requirements of Regulatory Guide 1.112 are met.

Regulatory Guide 1.113 - ESTIMATING AQUATIC DISPERSION OF EFFLUENTS FROM ACCIDENTAL AND ROUTINE REACTOR RELEASES FOR THE PURPOSE OF IMPLEMENTING APPENDIX I (Revision 1, April 1977)

The requirements of Regulatory Guide 1.113 are met.

Regulatory Guide 1.114 - GUIDANCE ON BEING OPERATOR AT THE CONTROLS OF A NUCLEAR POWER PLANT (Revision 1, November 1976)

PP&L will be in compliance with this regulatory guide.

Regulatory Guide 1.115 - PROTECTION AGAINST LOW-TRAJECTORY TURBINE MISSILES (Revision 1, July 1977)

Since the construction permit for the Susquehanna station was issued in November 1973, the methods of turbine missile protection, as described in this Regulatory Guide, were not specifically considered in the design. The design methods and demonstrative analyses employed for the tangential turbine orientation of Susquehanna SES are described in Section 3.5.

Regulatory Guide 1.116 - QUALITY ASSURANCE REQUIREMENTS FOR INSTALLATION, INSPECTION AND TESTING OF MECHANICAL EQUIPMENT AND SYSTEMS (Revision 0-R, May 1977)

The Susquehanna SES quality program for construction of safety related items was conducted in accordance with the program described in PSAR Appendix D and amendments. Compliance of the Operational Quality Assurance Program with this guide is described in Section 17.2.

Regulatory Guide 1.117 - TORNADO DESIGN CLASSIFICATION (June 1976)

All the structures, systems, and components listed in the appendix to this regulatory guide are protected against the effects of tornadoes except that the spent fuel pool is only designed to prevent significant loss of watertight integrity caused by tornadic winds and missiles generated by these winds.

Regulatory Guide 1.118 - PERIODIC TESTING OF ELECTRIC POWER AND PROTECTION SYSTEMS (June 1976) and (June 1978 for the 'E' Diesel Generator only)

For compliance with this regulatory guide refer to Section 8.1.6.1.

Regulatory Guide 1.119 - SURVEILLANCE PROGRAM FOR NEW FUEL ASSEMBLY DESIGNS (June 1976)

Withdrawn June 23,1977.

Regulatory Guide 1.120 - FIRE PROTECTION GUIDELINES FOR NUCLEAR POWER PLANTS (June 1976)

Regulatory Guide 1.120 was withdrawn on August 15, 2001. Susquehanna SES is not committed to this Regulatory Guide. Refer to the Fire Protection Review Report for the Susquehanna SES Fire Protection Program Commitments.

Regulatory Guide 1.121 - BASES FOR PLUGGING DEGRADED PWR STEAM GENERATOR TUBES (August 1976)

Not Applicable.

Regulatory Guide 1.122 - DEVELOPMENT OF FLOOR DESIGN RESPONSE SPECTRA FOR SEISMIC DESIGN OF FLOOR-SUPPORTED EQUIPMENT OR COMPONENTS (September 1976)

The methods used for developing the floor design response spectra for Susquehanna SES are in compliance with the positions of this regulatory guide except as follows:

1. The frequencies used for the calculation of the response spectra are different and are described in Subsection 3.7b.2.5.

2. The procedure for smoothing the spectra (broadening of peaks) is different and is discussed in Subsection 3.7b.2.9.

The above exceptions are Not Applicable for the Diesel Generator 'E' Building where the methods used for developing floor response spectra are in compliance with Regulatory Guide 1.122, Rev. 1 (February 1978).

Regulatory Guide 1.123 - QUALITY ASSURANCE REQUIREMENTS FOR CONTROL OF PROCUREMENT OF ITEMS AND SERVICES FOR NUCLEAR PLANTS (Revision 1, July 1977)

The Susquehanna SES quality assurance program for the construction phase is detailed in PSAR Appendix D and amendments. Compliance of the operational Quality Assurance Program with this regulatory guide is discussed in Section 17.2.

Regulatory Guide 1.124 - DESIGN LIMITS AND LOADING COMBINATIONS FOR CLASS 1 LINEAR-TYPE COMPONENT SUPPORTS (November 1976)

Since the construction permit for Susquehanna SES was issued in November 1973, this regulatory guide was not specifically considered in the design. The methods used to determine design loading combinations for Susquehanna SES are described in Section 3.9.

Regulatory Guide 1.125 - PHYSICAL MODELS FOR DESIGN AND OPERATION OF HYDRAULIC STRUCTURES AND SYSTEMS FOR NUCLEAR POWER PLANTS (March 1977)

No physical models were used during the design of Susquehanna SES.

Regulatory Guide 1.126 - AN ACCEPTABLE MODEL AND RELATED STATISTICAL METHODS FOR THE ANALYSIS OF FUEL DENSIFICATION (Revision 1, March 1978)

This position will be supplied later.

Regulatory Guide 1.127 - INSPECTION OF WATER-CONTROL STRUCTURES ASSOCIATED WITH NUCLEAR POWER PLANTS (April 1977)

Susquehanna SES will comply with this regulatory guide.

Regulatory Guide 1.128 - INSTALLATION DESIGN AND INSTALLATION OF LARGE LEAD STORAGE BATTERIES FOR NUCLEAR POWER PLANTS (April 1977 and October 1978 For the Diesel Generator 'E' Facility)

Installation design and installation of Class 1E batteries are in compliance with this regulatory guide except the design ambient temperature of the battery rooms is 80°F - 5°F and 104°F (Max.) to 65°F (Min.) for the Diesel Generator 'E' facility.

Regulatory Guide 1.129 - MAINTENANCE, TESTING AND REPLACEMENT OF LARGE LEAD STORAGE BATTERIES FOR NUCLEAR POWER PLANTS (April 1977 and February 1978 for the Diesel Generator 'E' Facility)

The Susquehanna SES complies with Regulatory Guide 1.129 dated April, 1977 and February, 1978 which invoke IEEE Std. 450-1975. As a result of the conversion from the Current Technical Specification to the Improved Technical Specification, IEEE Std. 450-1995 has been established as the applicable IEEE standard for the Class 1E station battery maintenance, testing and replacement. Compliance with Regulatory Guide 1.129 dated April, 1977 and February, 1978 remains unchanged because the intent of Regulatory Guide 1.129 is not changed as a result of the commitment to IEEE Std. 450-1995.

Regulatory Guide 1.130 - DESIGN LIMITS AND LOADING COMBINATIONS FOR CLASS I PLATE AND-SHELL-TYPE COMPONENT SUPPORTS (July 1977)

Since the construction permit for Susquehanna SES was issued in November 1973, this regulatory guide was not specifically considered in the design. The methods used to determine design loading combinations for Susquehanna SES are described in Subsection 5.4.14.

Regulatory Guide 1.131 - QUALIFICATION TESTS OF ELECTRIC CABLES, FIELD SPLICES, AND CONNECTIONS FOR LIGHT-WATER-COOLED NUCLEAR POWER PLANTS (August 1977)

The electric cables, field splices, and connections for the non-NSSS scope supply are qualified in accordance with IEEE 383-1974. Exceptions to the regulatory positions are as follows:

(1) Paragraph C.2 - The design basis event conditions meet the most severe postulated conditions for Susquehanna SES. Factors for margin given in Section 6.3.1.5 of IEEE 323-1974 were not used.

(2) Paragraph C.4 - Only one aging data point (121 C) has been applied to the cables used on Susquehanna SES.

(3) Paragraph C.6 - Flame tests were done in accordance with IEEE 383-1974. No tests were performed on aged specimen.

(4) Paragraph C.10 - Gas burner position is in accordance with IEEE 383-1974.

(5) Panel internal wires are not qualified to Regulatory Guide 1.131.

The electric cables, field splices, and connections for the NSSS scope, of supply have not been evaluated against this regulatory guide.

The electric cables, field splice, and connections for the Diesel Generator 'E' facility and ties to the transfer points in each of the Diesel Generator A-D bays are qualified in accordance with IEEE 383-1974. Exceptions, to the regulatory positions are as follows:

(1) Paragraph C.2 - The design basis event conditions meet the most severe postulated conditions for the Diesel Generator 'E' facility and Diesel Generator A/D bays. Factors for margin given in Section 6.3.1.5 of IEEE 323-1974, were not used.

(2) Paragraph C.6 - Flame tests were done in accordance with IEEE 383-1974. No tests were performed on aged specimen.

(3) Paragraph C.10 - Gas burner position is in accordance with IEEE 383-1974.

(4) Paragraph C. - The gas and air pressures of Section 2.5.4.4.3 were utilized.

(5) Panel internal wires are not qualified to Regulatory Guide 1.131.

Regulatory Guide 1.137 - FUEL-OIL SYSTEMS FOR STANDBY DIESEL GENERATORS (Revision 0, January 1978 for Diesels A-D, Revision 1, October 1979 for Diesel E)

The design of diesel generators "A-D" meet the intent of the fuel oil quality and testing requirements in Regulatory Guide 1.137, Revision 0. The design of the "E" diesel generator meets the intent of Regulatory Guide 1.137, Revision 1. The diesel generator fuel oil storage and transfer system is discussed/clarified in FSAR Section 9.5.4 and the Fuel Oil Monitoring Program.

Regulatory Guide 1.144 - AUDITING OF QA PROGRAMS (January 1979)

Compliance of the Operational Quality Assurance Program with this guide is described in Section 17.2.

Regulatory Guide 1.145 - ATMOSPHERIC DISPERSION MODELS FOR POTENTIAL ACCIDENT CONSEQUENCE ASSESSMENT AT NUCLEAR POWER PLANTS (NOVEMBER 1982)

Compliance with the Regulatory Guide and Methodology used for the Atmospheric Diffusion Models is described in Section 2.3.

Regulatory Guide 1.146 - QUALIFICATION OF QA PROGRAM AUDIT PERSONNEL (August 1980)

Compliance of the Operational Quality Assurance Program with this guide is described in Section 17.2.

Regulatory Guide 1.148 - FUNCTIONAL SPECIFICATION OF ACTIVE (DIESEL GENERATOR E ONLY) VALVE ASSEMBLIES IN SYSTEMS IMPORTANT TO SAFETY IN NUCLEAR POWER PLANTS (March 1981)

The design of the "E" Diesel Generator System meets the intent of this Regulatory Guide, which defines operating requirements for safety related valve assemblies. This regulatory guide endorses the use of ANSI Standard N278.1-1975.

Regulatory Guide 1.163 - PERFORMANCE-BASED CONTAINMENT LEAK-TEST PROGRAM (Revision 0, July 1995)

Leakage rate testing of the primary containment for compliance with 10CFR50, Appendix J, Option B is performed in accordance with the guidance provided in this regulatory guide. This regulatory guide also endorses the methodology for testing presented in NEI 94-01 and ANSI/ANS-56.8-1994.

Regulatory Guide 1.183 - ALTERNATIVE RADIOLOGICAL SOURCE TERMS FOR EVALUATING DESIGN BASIS ACCIDENTS AT NUCLEAR POWER REACTORS (JULY 2000)

Regulatory positions in this guide are complied with subject to the following exceptions and clarifications:

Reference: Section 6.0. An Alternative Source Term (AST) assessment was not performed for equipment qualification. TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," will continue to be used as the radiation dose basis for equipment qualification and radiation zone maps/shielding calculations.

Reference: Section 3.3, Appendix A. No credit is conservatively taken in the AST analyses for fission product reduction due to the initiation of the drywell sprays.

Reference: Section 3.4, Appendix A. No credit is taken in the AST analyses for reduction of airborne radioactivity in the containment by in-containment recirculation filter systems.

Reference: Section 3.6, Appendix A. No credit is taken in the AST analyses for reduction of airborne radioactivity in the containment by retention in ice condensers, or other engineering safety features not addressed above.

Reference: Section 3.8, Appendix A. The primary containment is not routinely purged during power operations.

Reference: Section 5.4, Appendix A. The temperature of the leakage does not exceed 212°F.

Reference: Section 7.0, Appendix A. Primary containment purging as a combustible gas or pressure control measure was analyzed and not deemed to be required. Containment purging capabilities are maintained for purposes of severe accident management and are not credited in any design basis analysis.

Reference: Sections 5.1 to 5.5, Appendix B. Fuel handling and an equipment handling accidents were evaluated within the Reactor Building outside containment.

Reference: Section 2.0, Appendix C. Postulated accident assumes fuel clad failure and fuel melt.

Reference: Section 3.5, Appendix C. The MSIVs and main steam drain lines do not automatically trip closed following a CRDA. The operators manually scram the reactor and close all MSIVs and drain line valves in the event of a main steam line radiation monitor (MSLRM) high-high radiation alarm in accordance with station procedures.

Regulatory Guide 1.194 - ATMOSPHERIC RELATIVE CONCENTRATIONS FOR CONTROL ROOM RADIOLOGICAL ASSESSMENTS AT NUCLEAR POWER PLANTS (JUNE 2003)

Regulatory positions in this guide are complied with subject to the following exceptions and clarifications:

Reference: Sections 3.2.1 – 3.2.4, 3.2.4.1 – 3.2.4.8. The diffusion models used are based on point-source formulations and ground level releases.

Reference: Sections 3.3.2 – 3.3.4. The Control Room ventilation system has only a single outside air intake.

Reference: Sections 4.1 – 4.4. Source-receptor geometry, type and locations.
All ground level releases were determined per the ARCON96 methodology utilizing standard time intervals; no χ/Q correction per wind speed averaging.

Reference: Section 6.0. Plume rise was not considered in the χ/Q determinations.

Reference: Section 7.0. No experimental data was utilized to calculate the χ/Qs.

Regulatory Guide 1.197 - DEMONSTRATING CONTROL ROOM ENVELOPE INTEGRITY AT NUCLEAR POWER REACTORS (May 2003)

The following regulatory portions are compiled with:

Requirements for (i) determining the unfiltered air inleakage past the CRE boundary into the CRE in accordance with the testing methods and at the Frequencies specified in Section C.1 and C.2 and

(ii) assessing CRE habitability at the Frequencies specified in Sections C.1 and C.2.

TABLE 3.13-1

COMPARISON WITH REGULATORY GUIDE 1.48

COMPONENT	PLANT CONDITION	NRC REGULATORY GUIDE 1.48: LOADING COMBINATION 1/	NRC REGULATORY GUIDE 1.48: DESIGN LIMIT		REGULATORY GUIDE PARAGRAPH	SSES: LOADING COMBINATION (F)	SSES: CODE ALLOWABLE STRESSES	SSES: ASME SECTION 111 REFERENCE	HOW SSES COMPARES WITH NRC REGULATORY GUIDE 1.48
Class 1 Vessels	Upset (U)	(NPC or UPC) + 0.5 SSE	NB-3223	2/	1.a	(NPC OR UPC), 0.5 SSE	3.0 Sm (INCLUDES SECONDARY STRESSES)	NB-3223	GE REFLECTS INDUSTRY POSITION
	Emergency (E)	EPC	NB-3224		1.b	EPC,.5 SSE + TRANSIENT	1.8 Sm	NB-3224	
	Faulted (F)	NPC + SSE +DSL	NB-3225		1.c	NPC + SSE + DSL	APP F - SECT 111	NB-3225	
Class 1 PIPING	U	(NPC or UPC) + 0.5 SSE	NB-3654	2/	1.a	(NPC or UPC), 0.5 SSE	3.0 Sm (INCLUDES SECONDARY STRESSES)	NB-3654	GE REFLECTS INDUSTRY POSITION
	E	EPC	NB-3655		1.b	EPC,.5 SSE + TRANSIENT	2.25 Sm	NB-3655	
	F	NPC + SSE + DSL	NB-3656		1.c	NPC + SSE + DSL	3.0 Sm	NB-3656	
Class 1 Pumps (Inactive)	U	(NPC or UPC) + 0.5 SSE	NB-3223 5/	1/	2.a	(NPC or UPC), 0.5 SSE	1.65 Sm	NB-3223	GE REFLECTS INDUSTRY POSITION
	E	EPC	NB-3224		2.b	EPC, .5 SSE + TRANSIENT	1.8 Sm	NB-3224	
	F	NPC + SSE + DSL	NB-3225		2.c	NPC + SSE + DSL	APP F - SECTION 111	NB-3225	
Class 1 Pumps (Active)	U	(NPC or UPC) + 0.5 SSE	NB-3222	5/	4.a	(NPC or UPC), 0.5 SSE	NOT APPLICABLE	NOT APPLICABLE	NOT APPLICABLE
	E	EPC	NB-3222	6/	4.a	EPC			
	F	NPC + SSE + DSL	NB-3222	7/ 8/	4.a	NPC + SSE + DSL			
Class 1 Valves (Inactive) By Analysis	U	(NPC or UPC) + 0.5 SSE	NB-3223 5/	4/	2.a	(NPC or UPC), 0.5 SSE	NOT APPLICABLE	NOT APPLICABLE	NOT APPLICABLE
	E	EPC	NB-3224 2/		2.b	EPC			
	F	NPC + SSE + DSL	NB-3225 2/		2.c	NPC + SSE + DSL			
Class 1 Valves (Inactive) Designed by Either Std or Alternative Design Rules	U	(NPC or UPC) + 0.5 SSE	1.1 Pr		3.a	(NPC or UPC), 0.5 SSE	1.1 Pr	NB-3525	GE REFLECTS INDUSTRY POSITION
	E	EPC	1.2 Pr		3.b	EPC, .5 SSE + TRANSIENT	1.2 Pr	NB-3526	
	F	NPC + SSE + DSL	1.5 Pr		3.c	NPC + SSE + DSL	1.5 Pr	NB-3527	
Class 1 Valves (Active) By Analysis	U	(NPC or UPC) 0.5 SSE	NB-3222	5/	4a.1	(NPC or UPC), 0.5 SSE	NOT APPLICABLE	NOT APPLICABLE	NOT APPLICABLE
	E	EPC	NB-3222	6/ 7/	4a.2	EPC			
	F	NPC + SSE + DSL	NB-3222	8/	4a.3	NPC + SSE + DSL			
Class 1 Valves (Active) Designed by Std or Alternative Design Rules	U	(NPC or UPC) 0.5 SSE	1.0 Pr	6/	5a.1	(NPC or UPC), 0.5 SSE	1.0 Pr (a)	NB-3525	GE REFLECTS INDUSTRY POSITION
	E	EPC	1.0 Pr		5a.2	EPC, .5 SSE + TRANSIENT	1.0 Pr	NB-3526	
	F	NPC • SSE + DSL	1.0 Pr		5a.3	NPC + SSE + DSL	1.0 Pr	NB-3527	

TABLE 3.13-1 (Continued)

COMPARISON WITH REGULATORY GUIDE 1.48

COMPONENT	PLANT CONDITION	NRC REGULATORY GUIDE 1.48: LOADING COMBINATION 1/	NRC REGULATORY GUIDE 1.48: DESIGN LIMIT	REGULATORY GUIDE PARAGRAPH	SSES: LOADING COMBINATION (F)	SSES: CODE ALLOWABLE STRESSES	SSES: ASME SECTION 111 REFERENCE	HOW SSES COMPARES WITH NRC REGULATORY GUIDE 1.48
Class 2 & 3 Vessels (Division 1) of Section VIII of the ASME Code	U	(NPC or UPC) + 0.5 SSE	1.1S) 9/	6a	(NPC or UPC), 0.5 SSE	σm 1.15) (c)	CODE CASE 1607	FAULTED CONDITION NRC MORE CONSERVATIVE GE REFLECTS INDUSTRY POSITION
	E	EPC	1.1S)	6b	EPC, .5 SSE + TRANSIENT		NC/NB3321 1 Ob)	
	F	NPC + SSE + DSL	1.5S)	6c	NPC + SSE + DSL	σm 2.08)		
Class 2 Vessels (Divison 2) of Section VIII of the ASME Code	U	(NPC or UPC) + 0.5 SSE	NB-3223) 2/	7a	(NPC or UPC), 0.5 SSE	NOT APPLICABLE	NOT APPLICABLE	NOT APPLICABLE
	E	EPC	NB-3224)	7b	EPC			
	F	NPC + SSE + DSL	NB-3225)	7c	NPC + SSE + DSL			
Class 2 & 3 Piping	U	(NPC or UPC) + 0.5 SSE	NC3611.1(b) (4)) (c) (b) (1))	8a	(NPC or UPC), 0.5 SSE	1.2 Sh	NC/ND3611 3(b)) NC/ND3611 3(c)) (4) (b)) (b)	NRC MORE CONSERVATIVE
	E	EPC	NC3611.1(b) (4))10/ (c) (b) (1))	8a	EPC,.5 SSE + TRANSIENT	1.8 Sh	CODE CASE 1606)	GE REFLECTS INDUSTRY POSITION
	F	NPC + SSE + DSL	NC3611.1(b) (4)) (c) (b) (2)	8b	NPC + SSE + DSL	2.4 Sh	NC/ND3611.3(d)) (SEE NOTE (b)))	
Class 2 & 3 Pumps (Inactive)	U	(NPC or UPC) + 0.5 SSE	$\sigma m < 1.1S > \frac{\sigma m + \sigma b}{1.5}$	9.a	(NPC or UPC), 0.5 SSE	NOT APPLICABLE	NOT APPLICABLE	NOT APPLICABLE
	E	EPC	$\sigma m < 1.1S > \frac{\sigma m + \sigma b}{1.5}$	9.a	EPC			
	F	NPC + SSE + DSL	$\sigma m < 1.2S > \frac{\sigma m + \sigma b}{1.5}$	9.b	NPC + SSE + DSL			
Class 2 & 3 Pumps (Active)	U	(NPC or UPC) + 0.5 SSE	$\sigma m < 1.0S > \frac{\sigma m + \sigma b}{1.5}$	10.a	(NPC or UPC), 0.5 SSE	σm 1.1S)	CODE CASE 1636	GE RELECTS INDUSTRY POSITION
	E	EPC	$\sigma m < 1.0S > \frac{\sigma m + \sigma b}{1.5}$ 11/	10.a	EPC,.5 SSE + TRANSIENT	) (e)) (c)	NC/ND 3423	
	F	NPC + SSE + DSL	$\sigma m < 1.0S > \frac{\sigma m + \sigma b}{1.5}$	10.a	NPC + SSE + DSL	σm 1.2S)	(See Note (b))	
Class 2 & 3 Valves (Inactive)	U	(NPC or UPC) + 0.5 SSE	.1 Pr	11.a	(NPC or UPC), 0.5 SSE	σm 1.1S (c)	CODE CASE 1636	EQUALLY CONSERVATIVE
	E	EPC	1.1 Pr	11.a	EPC,.5 SSE + TRANSIENT	} (c)	NC/ND 3521	
	F	NPC • SSE + DSL	1.2 Pr	11.b	NPC + SSE + DSL	σm 2.0S)	(See Note (b))	
Class 2 & 3 Valves (Active)	U	(NPC or UPC) + 0.5 SSE	1.0 Pr	12.a	(NPC or UPC), 0.5 SSE	σm 1.1S)	CODE CASE 1636	EQUALLY CONSERVATIVE
	E	EPC	1.0 Pr.) 11/	12.a	EPC,.5 SSE + TRANSIENT	} (a)	NC/ND 3621	(a)
	F	NPC + SSE + DSL	1.0 Pr)	12.a	NPC + SSE + DSL	σm 1.2S } (c)	(See Note (b))	

TABLE 3.13-1 (CONTINUED)

NOTES FOR COMPARISON TABLE

1. Numerical indicators in the regulatory guide portion of the table (1/, 2/, etc.) correspond to footnotes of the Regulatory Guide 1.48.

2. Alphabetical indicators in Susquehanna SES portion of table (or comparative column) correspond to the following:

   a. In addition to compliance with the design limits specified, assurance of operability under all design loading combinations shall be in accordance with Subsection 3.9.3.2.

   b. Referenced paragraphs of code currently in course of preparation.

   c. The design limit for local membrane stress intensity or primary membrane plus primary bending stress intensity is 150% of that allowed for general membrane (except as limited to 2.4S for inactive components under faulted condition). Refer to Subsection 3.9.5.2.

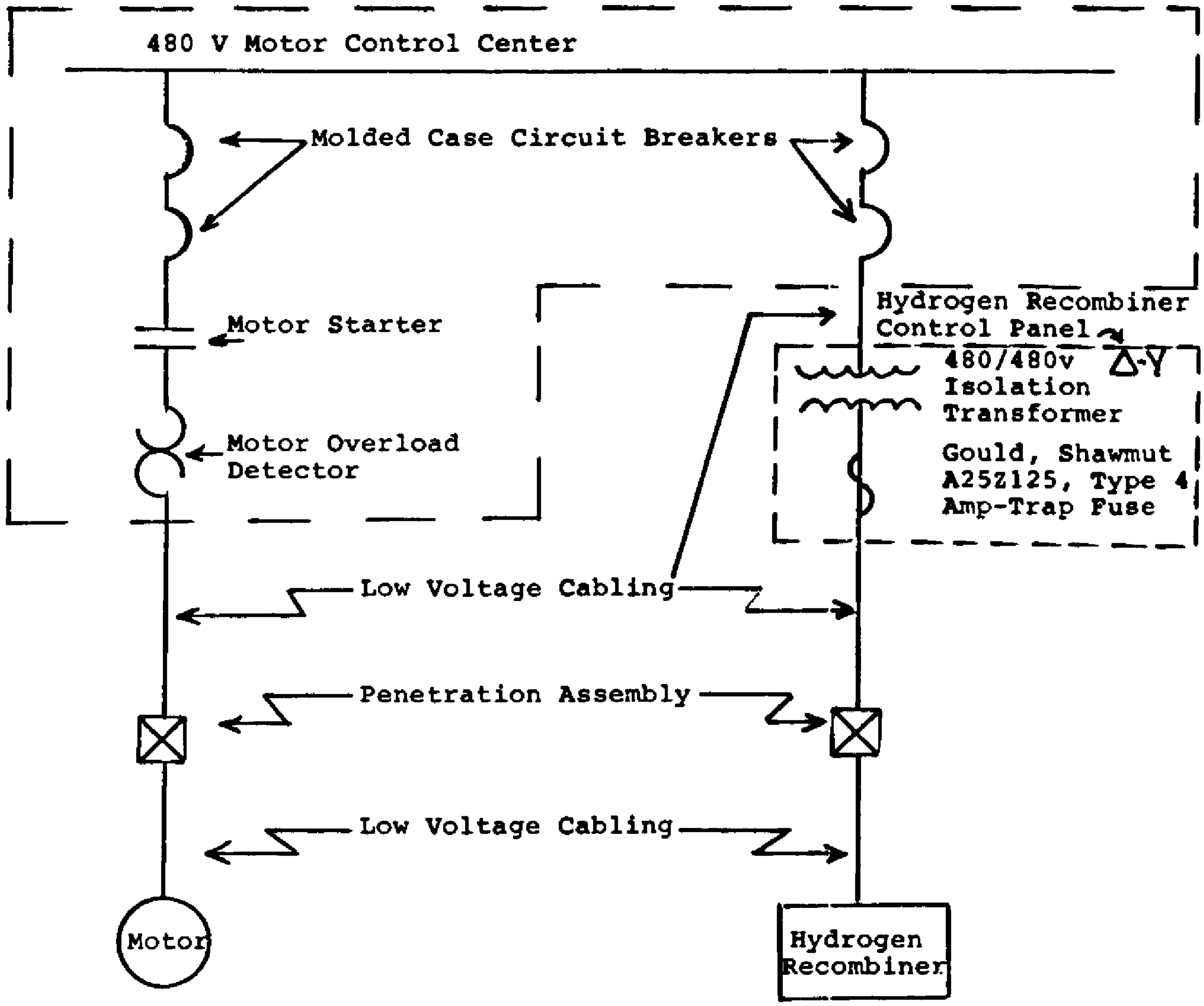
   d. Not used.

   e. Inactive limits may be used since operability will be demonstrated in accordance with Subsection 3.9.3.2.

   f. When selecting plant events for evaluation, the choice of the events to be included in each plant condition is selected based on the probability of occurrence of the particular load combination. The combination of loads are those identified in Table 3.9.2(a).

3. Acronyms

   UPC Upset plant conditions
   NPC Normal plant conditions
   EPC Emergency plant conditions
   DSL Dynamic System Loadings
   SSE Safe Shutdown Earthquake



FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

RELATIONSHIP OF LOADS,
PENETRATION ASSEMBLIES &
PROTECTIVE DEVICES FOR CURVES
PRESENTED IN FIGURES
3.13-2, 3.13-3 & 3.18-8

FIGURE 3.13-1, Rev. 47

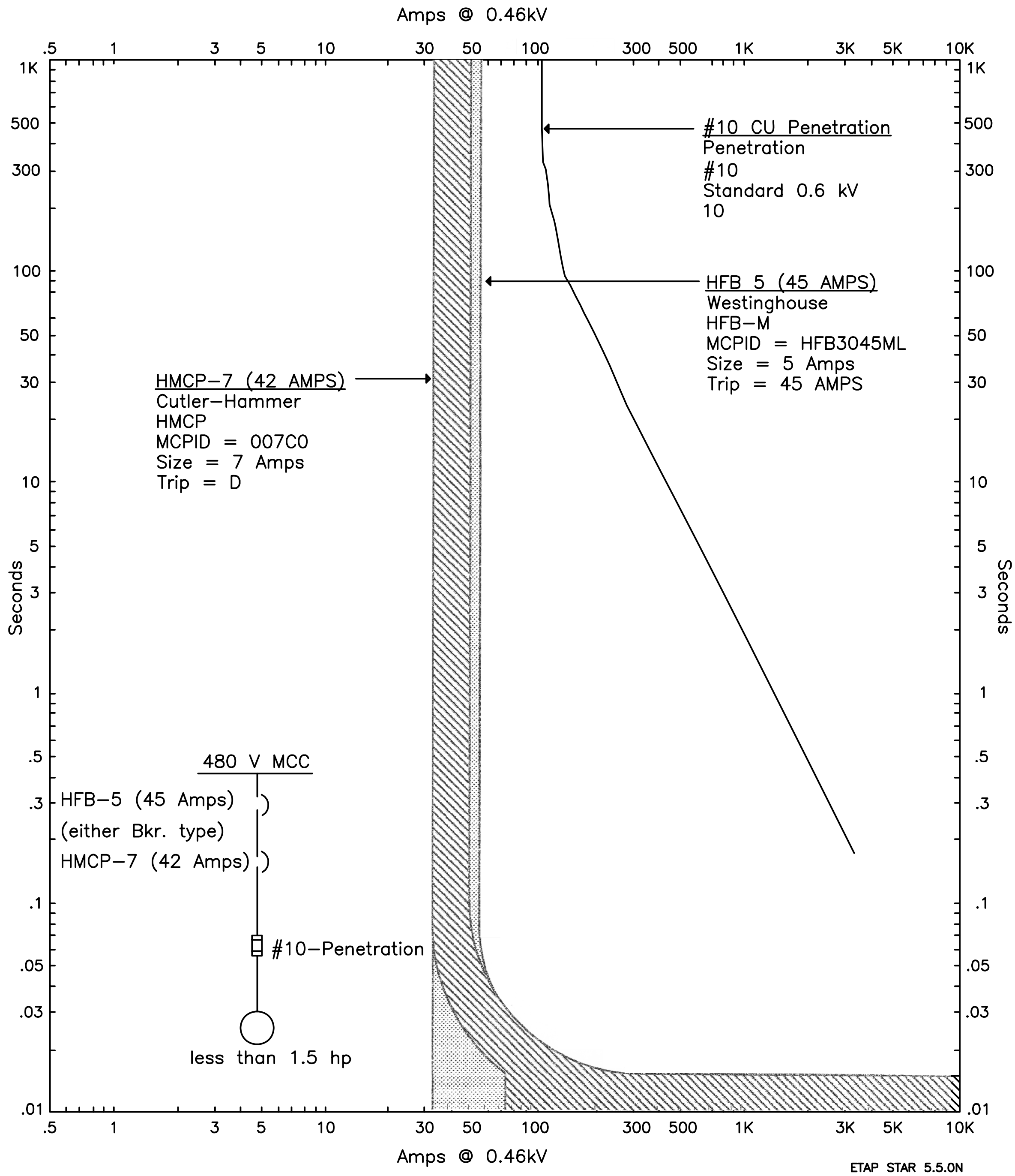
Auto-Cad Figure Fsar 3_13_1.dwg

THIS FIGURE HAS BEEN
REPLACED BY FIGURE 3.13-2A & FIGURE 3.13-2B

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Figure Replaced by FIGURE 3.13-2A & FIGURE 3.13-2B FIGURE 3.13-2, Rev. 48

AutoCAD; Figure Fsar 3_13_2.doc



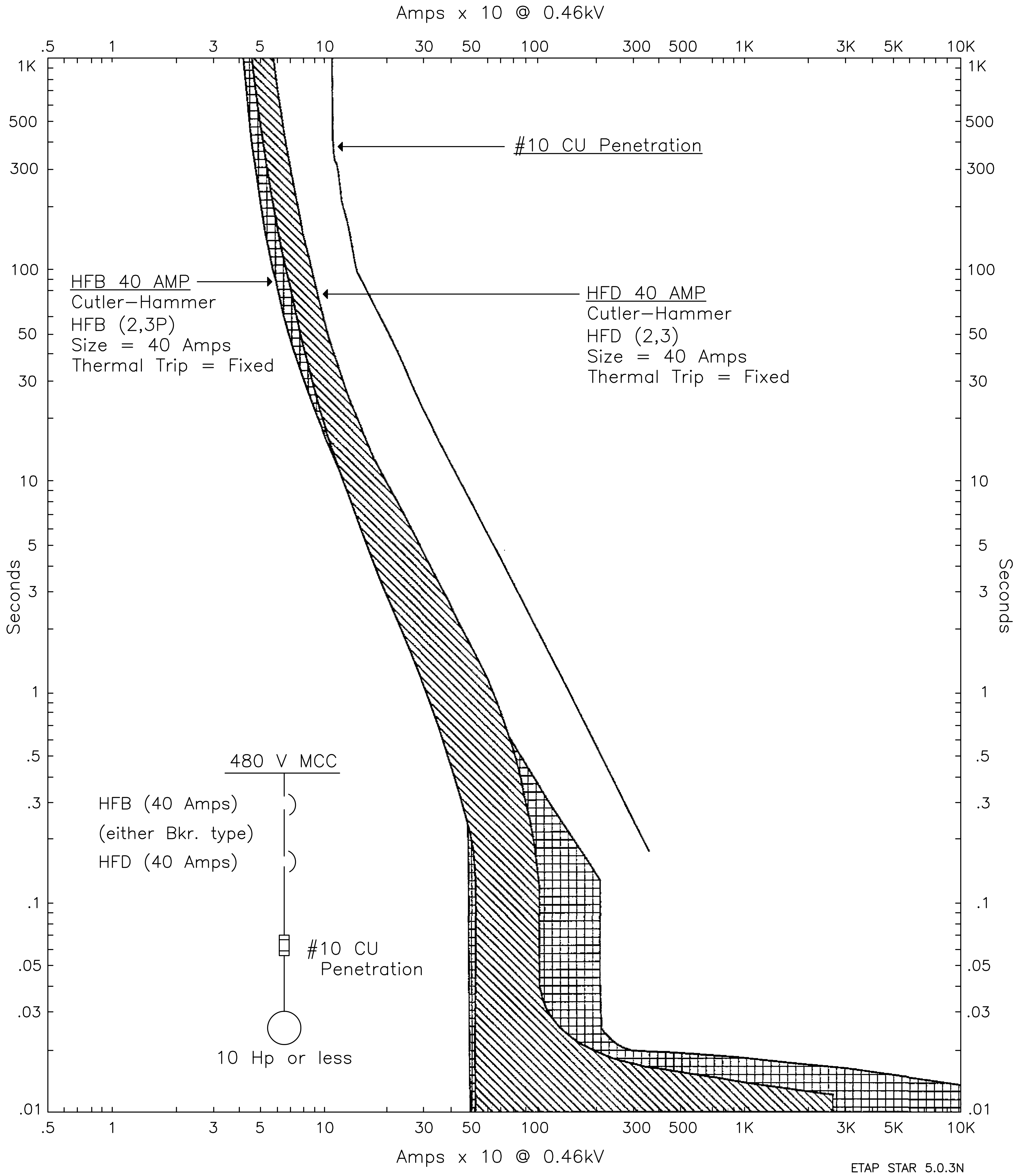
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

TIME-CURRENT
CHARACTERISTIC CURVES FOR
OVERCURRENT PROTECTION OF
#10 COPPER, CONTAINMENT PENETRATIONS

FIGURE 3.13-2A, Rev. 1

Auto-Cad Figure Fsar 3_13_2A.dwg



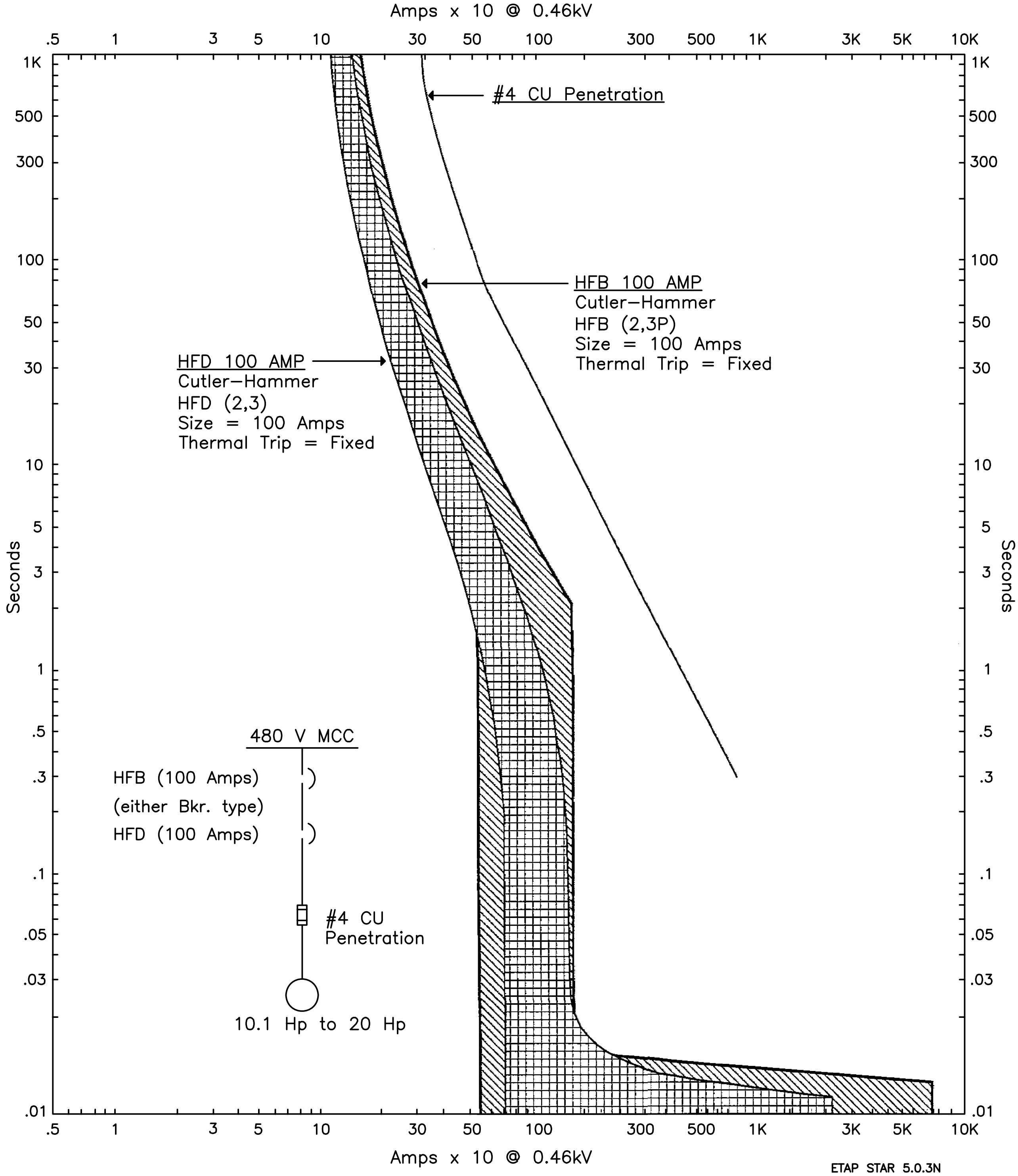
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

TIME-CURRENT
CHARACTERISTIC CURVES FOR
OVERCURRENT PROTECTION OF
#10 COPPER, CONTAINMENT PENETRATIONS

FIGURE 3.13-2B, Rev. 1

Auto-Cad Figure Fsar 3_13_2B.dwg



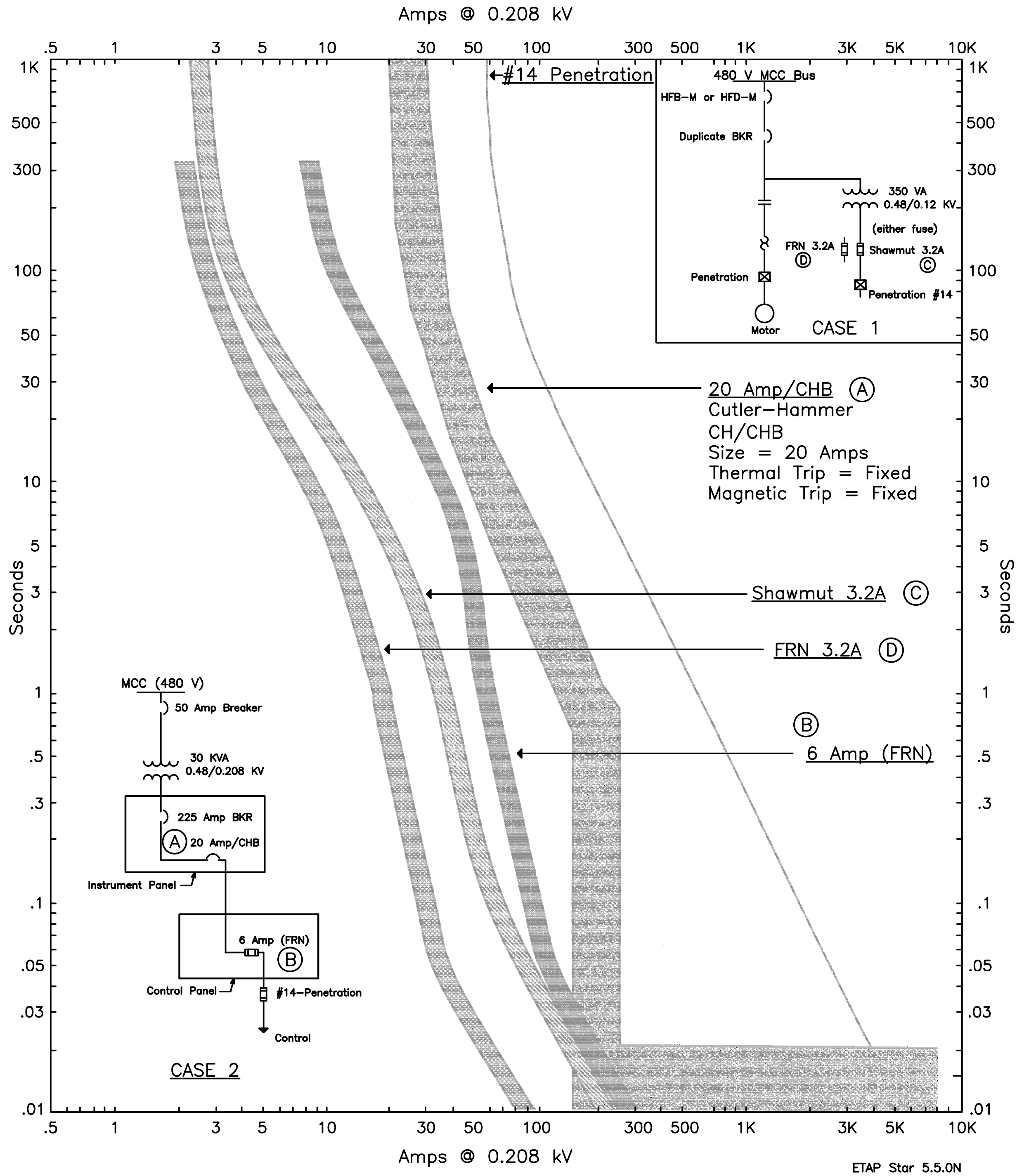
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

TIME-CURRENT
CHARACTERISTIC CURVES FOR
OVERCURRENT PROTECTION OF
#4 COPPER, CONTAINMENT PENETRATIONS

FIGURE 3.13-3, Rev. 48

Auto-Cad Figure Fsar 3_13_3.dwg



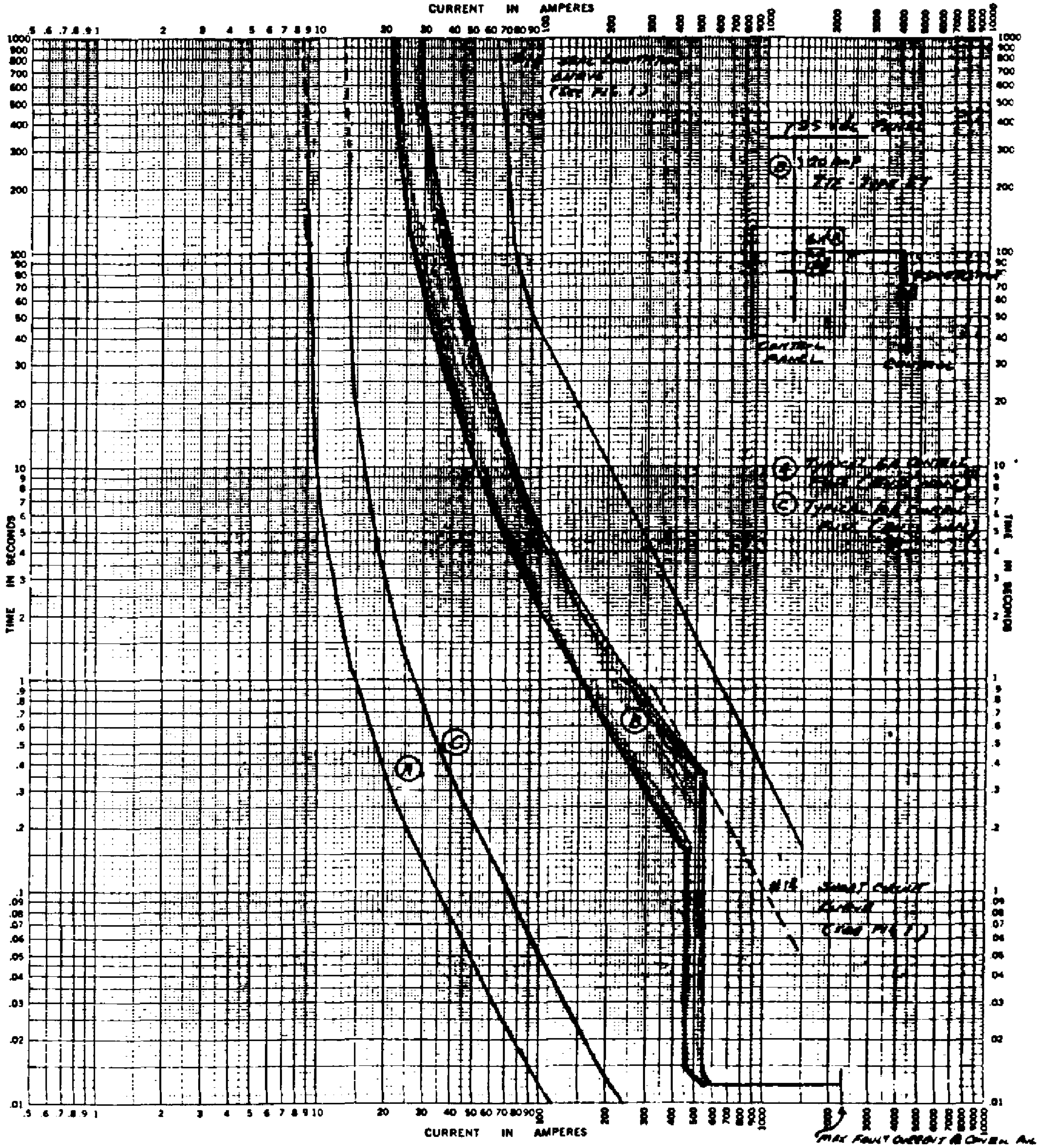
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

120V AC
CONTROL CIRCUITS

FIGURE 3.13-4, Rev. 48

Auto-Cad Figure Fsar 3_13_4.dwg



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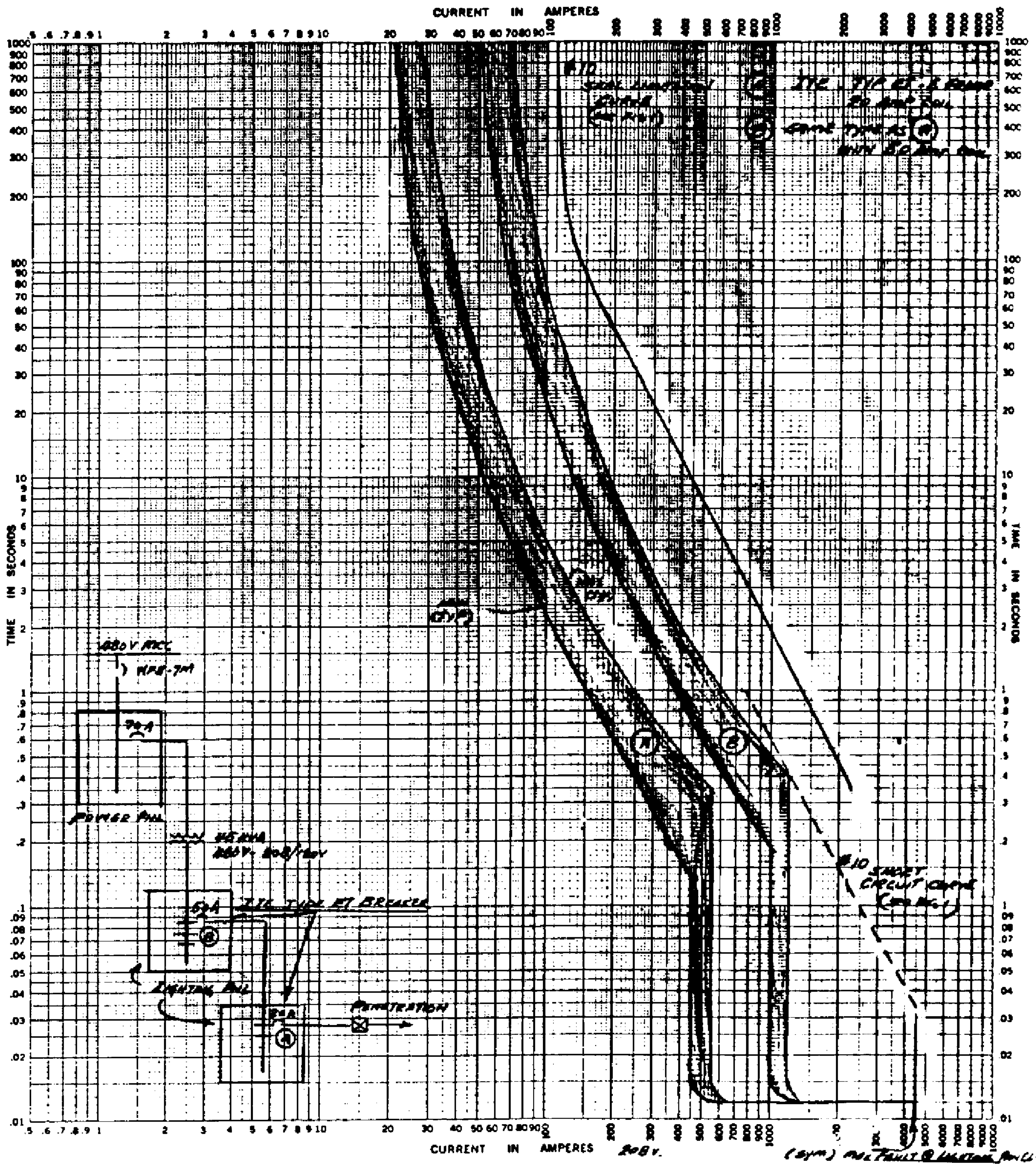
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

125 VOLT DC CONTROL CIRCUITS

FIGURE 3.13-5, Rev. 47

Auto-Cad Figure Fsar 3_13_5.dwg



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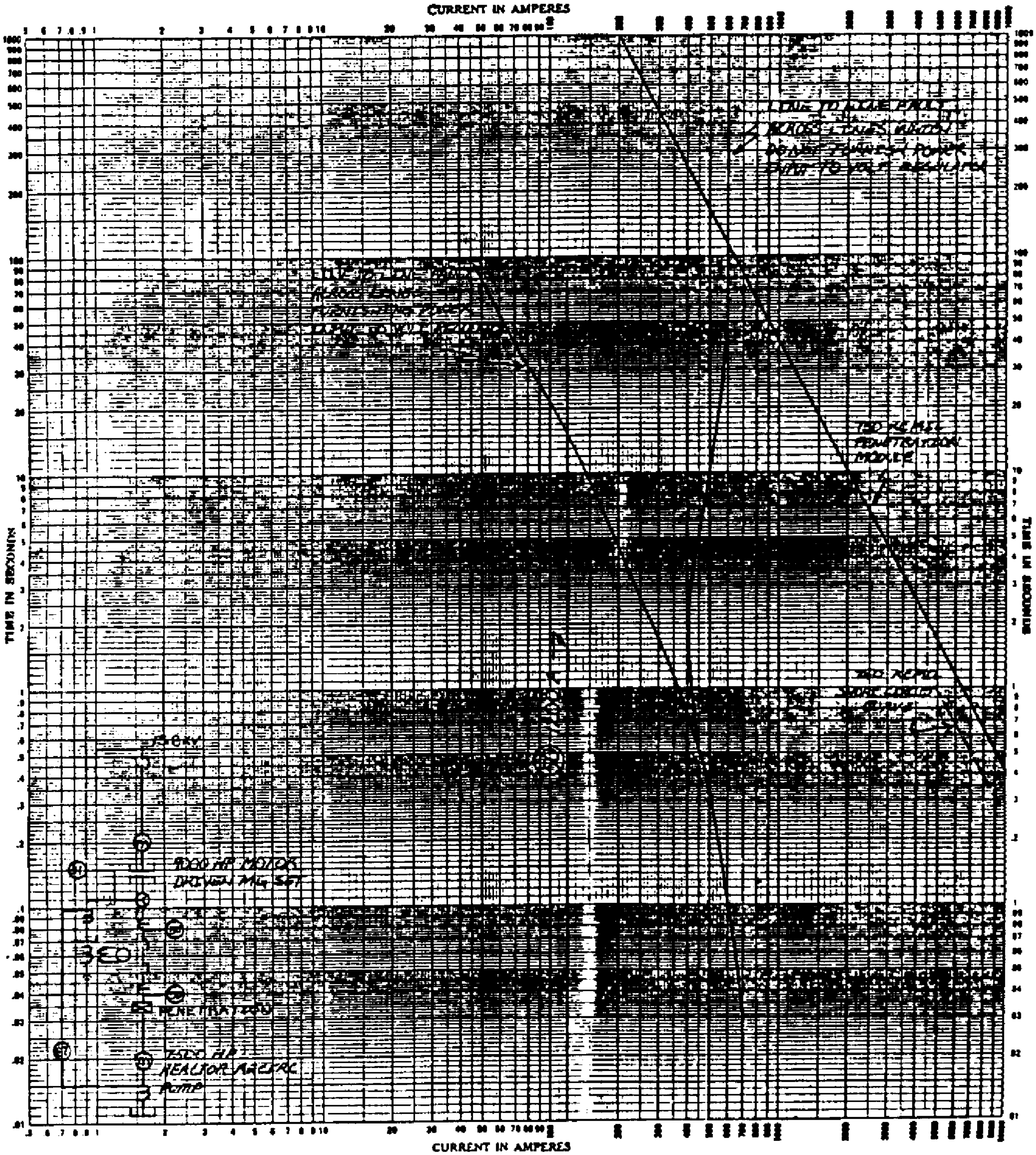
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

208/120 LIGHTING CIRCUITS

FIGURE 3.13-6, Rev. 47

Auto-Cad Figure Fsar 3_13_6.dwg



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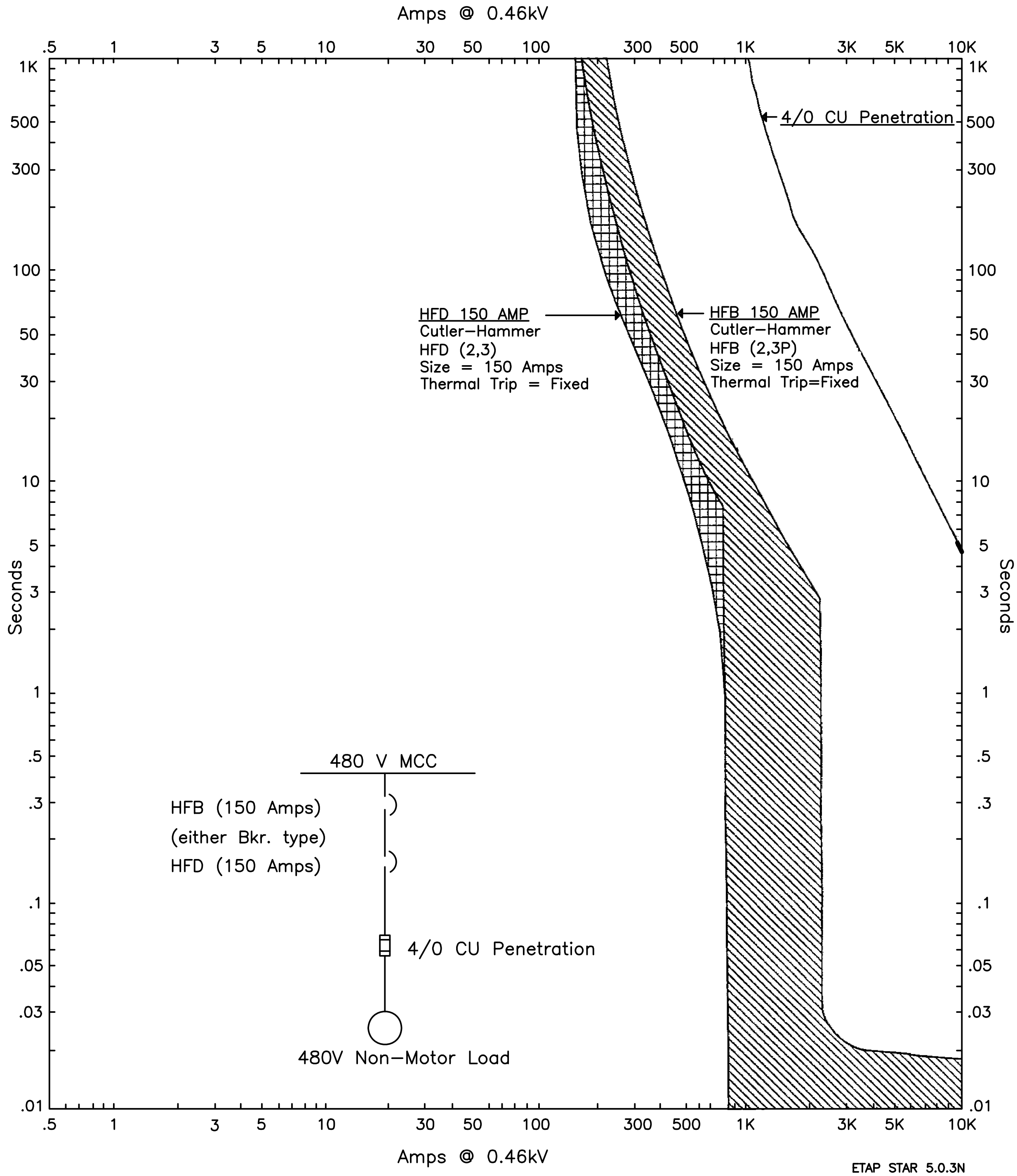
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

REACTOR RECIRCULATION PUMP-
MAXIMUM SHORT
CIRCUIT CAPABILITY

FIGURE 3.13-7, Rev. 47

Auto-Cad Figure Fsar 3_13_47.dwg



FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

TIME-CURRENT CHARACTERISTIC
CURVES FOR OVERCURRENT
PROTECTION OF 4/0 COPPER,
CONTAINMENT PENETRATIONS
USED WITH HYDROGEN RECOMBINERS

FIGURE 3.13-8, Rev. 48

Auto-Cad Figure Fsar 3_13_8.dwg

3.14 LICENSE RENEWAL PROGRAMS, TLAA, AND COMMITMENTS

3.14.1 INTRODUCTION

The License Renewal Rule, 10 CFR Part 54, governs the issuance of renewed operating licenses for nuclear power plants. The renewed SSES operating licenses were granted on November, 24, 2009 to be effective until July 17, 2042 for Unit 1 and March 23, 2044 for Unit 2.

This section contains the license renewal information to be included in the Final Safety Analysis Report as required by 10 CFR 54.21(d). The Safety Evaluation Report (SER) NUREG-1931, for renewed operating licenses for SSES Units 1 & 2 contains the results of the NRC review of technical information required by 10 CFR 54.21(a) and (c) as submitted in the SSES License Renewal Application (LRA) and supplemental RAI responses. The programs and activities that will be implemented to manage the effects of aging for the period of extended operation are listed throughout NUREG-1931. In addition, this section contains a listing of commitments associated with license renewal as identified in Appendix A of NUREG-1931.

3.14.2 AGING MANAGEMENT PROGRAMS (AMP)

The license renewal integrated plant assessment identified existing and new aging management programs necessary to provide reasonable assurance that components within the scope of license renewal will continue to perform their intended functions consistent with the current licensing basis (CLB) for the period of extended operation. This section describes the required aging management programs identified during the integrated plant assessment. Except for one-time inspections, these programs will be implemented during the period of extended operation. One-time inspections will be conducted within the 10-year period prior to beginning the period of extended operation.

Three elements of an effective aging management program that are common to all aging management programs are corrective actions, confirmation process, and administrative controls. These elements are included in the SSES Operational Quality Assurance (OQA) Program, which implements the requirements of 10 CFR 50, Appendix B. Prior to the period of extended operation, the elements of corrective action, confirmation process, and administrative controls in the SSES OQA Program will be applied to required aging management programs for both safety-related and nonsafety-related structures and components determined to require aging management during the period of extended operation.

3.14.2.1 Area-Based NSAS Inspection (NUREG 1931, Section 3.0.3.3.1)

The Area-Based NSAS Inspection detects and characterizes the conditions on the internal surfaces of nonsafety-related components exposed to non-radioactive equipment/area drainage water or potable water environments. The Area-Based NSAS Inspection also detects and characterizes specific conditions on the internal surface of copper alloys exposed to raw water from the spray pond/cooling tower. Components identified as non-safety affecting safety (NSAS) are those nonsafety-related components with the potential to prevent a safety-related system or component from performing its safety function. The conditions in these environments are not expected to result in sufficient degradation to cause spatial interaction with safety-related components or are expected to result in effects that progress very slowly. To ensure

that spatial interactions do not occur that impair or prevent a safety-related function, a focused characterization of conditions is performed to provide confirmation of a lack of degradation or to serve as the basis for recurring actions during the period of extended operation, if required.

The Area-Based NSAS Inspection is a new one-time inspection that will be implemented prior to the period of extended operation. The inspection activities will be conducted within the 10-year period prior to the period of extended operation.

3.14.2.2 Bolting Integrity Program (NUREG 1931, Section 3.0.3.2.5)

The Bolting Integrity Program is a combination of existing SSES activities that, in conjunction with other credited programs, addresses the management of aging for the bolting of subject mechanical components within the scope of license renewal. The Bolting Integrity Program relies on manufacturer/vendor information and industry recommendations for the proper selection, assembly and maintenance of bolting for pressure-retaining closures. The Bolting Integrity Program includes, through the Inservice Inspection (ISI) Program and System Walkdown Program, the periodic inspection of bolting for indication of degradation such as leakage, loss of material due corrosion, or cracking.

Prior to the period of extended operation, the Bolting Integrity Program will be enhanced to include a specific precaution against the use of sulfur (sulfide) containing compounds as a lubricant for bolted connections.

3.14.2.3 Buried Piping and Tanks Inspection Program (NUREG 1931, Section 3.0.3.2.13)

The Buried Piping and Tanks Inspection Program manages the effects of corrosion on the external surfaces of piping and tanks exposed to a buried environment. The Buried Piping and Tanks Inspection Program will be a combination of a prevention program (consisting of protective coatings and wrappings, where appropriate) and a condition monitoring program (consisting of visual inspections).

The Buried Piping and Tanks Inspection Program is a new aging management program that will be implemented prior to the period of extended operation.

3.14.2.4 Buried Piping Surveillance Program (NUREG 1931, Section 3.0.3.2.10)

The Buried Piping Surveillance Program manages the effects of corrosion on the external surfaces of piping with damaged coatings exposed to a buried environment. The Buried Piping Surveillance Program will be a combination of a prevention program (consisting of cathodic protection) and a condition monitoring program (consisting of periodic testing).

The Buried Piping Surveillance Program is a new aging management program that will be implemented prior to the period of extended operation.

3.14.2.5 BWR CRD Return Line Nozzle Program (NUREG 1931, Section 3.0.3.2.2)

The BWR CRD Return Line Nozzle Program is an existing program that manages cracking of the control rod drive return line (CRDRL) nozzle, cap, and connecting weld. The program was developed in response to industry events involving the CRD return line nozzle.

SSES modified the CRD System by cutting the CRDRL and capping the nozzle prior to initial plant startup. The CRDRL was not rerouted. SSES has completed all of the requirements specified in NUREG-0619 for the CRD System modifications performed at SSES, including the final liquid penetrant testing (PT) inspection and system performance testing. The SSES BWR CRD Return Line Nozzle Program monitors the effects of cracking on the intended function of the CRDRL nozzle by performing inservice inspections in conformance with the ASME Boiler and Pressure Vessel Code, Section XI, Subsection IWB, Table IWB 2500-1 (edition and addenda described in 3.14.2.23). Any cracks that are detected will be dispositioned in accordance with the requirements of ASME Section XI.

3.14.2.6 BWR Feedwater Nozzle Program (NUREG 1931, Section 3.0.3.1.4)

The BWR Feedwater Nozzle Program is an existing program that manages cracking of the feedwater nozzles.

The program includes (a) enhanced inservice inspection in accordance with the requirements of the ASME Boiler and Pressure Vessel Code, Section XI, Subsection IWB, Table IWB 2500-1 (edition and addenda described in 3.14.2.23) and the recommendations of report GE-NE-523-A71-0594, and (b) system modifications (completed on the spargers prior to initial startup) to mitigate cracking. The program specifies periodic ultrasonic inspection of critical regions of the feedwater nozzles.

3.14.2.7 BWR Penetrations Program (NUREG 1931, Section 3.0.3.2.3)

The BWR Penetrations Program is an existing program that manages cracking of selected reactor vessel penetrations. The BWR Penetrations Program is implemented via the Inservice Inspection (ISI) Program in compliance with ASME Section XI and the Boiling Water Reactor Vessel and Internals Project (BWRVIP) guidelines.

The program includes (a) inspection and flaw evaluation in conformance with the guidelines of NRC-approved BWRVIP reports and BWRVIP-27-A, BWRVIP-47-A, BWRVIP-49-A, and BWRVIP-74-A and (b) monitoring and control of reactor coolant water chemistry in accordance with the guidelines of BWRVIP-29 to ensure the long-term integrity and safe operation of reactor vessel internal components. The BWRVIP-27-A report addresses the standby liquid control system nozzle or housing, the BWRVIP-47-A report addresses the control rod drive and flux monitor penetrations in the lower plenum, the BWRVIP-49-A report provides guidelines for instrument penetrations, and the BWRVIP-74-A report addresses the reactor vessel flange leak off penetrations and the reactor vessel drain penetrations.

3.14.2.8 BWR Stress Corrosion Cracking (SCC) Program (NUREG 1931, Section 3.0.3.1.5)

The BWR Stress Corrosion Cracking (SCC) Program is an existing program that manages stress corrosion cracking for stainless steel piping, valves, flow instruments, and pump casings. The program to manage stress corrosion cracking in pressure boundary piping made of stainless steel is delineated in NUREG-0313, Revision 2, and NRC Generic Letter 88-01 and its Supplement 1.

The program includes (a) preventive measures to mitigate intergranular stress corrosion cracking (IGSCC), and (b) inspection and flaw evaluation to monitor IGSCC and its effects. The

NRC-approved report BWRVIP-75-A allows for modifications of inspection scope in the Generic Letter 88-01 program.

3.14.2.9 BWR Vessel ID Attachment Welds Program (NUREG 1931, Section 3.0.3.1.3)

The BWR Vessel ID Attachment Welds Program is an existing program that manages cracking of the welds for internal attachments to the reactor pressure vessel.

The program includes (a) inspection and flaw evaluation in accordance with the guidelines of the NRC-approved report BWRVIP-48-A, and (b) monitoring and control of reactor coolant water chemistry in accordance with the guidelines of BWRVIP-29 to ensure the long-term integrity and safe operation of the vessel inside diameter (ID) attachment welds. The BWR Vessel ID Attachment Welds Program is based on the inspection and flaw evaluation guidelines of the BWRVIP, and is implemented by the Inservice Inspection (ISI) Program in accordance with the ASME Code, Section XI, Table IWB 2500-1.

3.14.2.10 BWR Vessel Internals Program (NUREG 1931, Section 3.0.3.2.4)

The BWR Vessel Internals Program is an existing program that manages aging of the reactor vessel internals in accordance with the requirements of ASME Section XI and the BWRVIP documents. The purpose of the BWR Vessel Internals Program is to manage cracking, loss of material, and reduction of fracture toughness for various subcomponents of the reactor vessel internals.

The program includes (a) inspection and flaw evaluation in conformance with the guidelines of applicable and NRC-approved BWRVIP reports, and (b) monitoring and control of reactor coolant water chemistry in accordance with the guidelines of BWRVIP-29 to ensure the long-term integrity and safe operation of reactor vessel internal components.

3.14.2.11 BWR Water Chemistry Program (NUREG 1931, Section 3.0.3.1.1)

The BWR Water Chemistry Program is an existing program that mitigates damage due to loss of material and cracking of plant components that are within the scope of license renewal and contain treated water. The program manages the relevant conditions that could lead to the onset and propagation of a loss of material or cracking through proper monitoring and control consistent with pertinent EPRI water chemistry guidelines. The SSES BWR Water Chemistry Program currently implements BWRVIP-29 and is in the process of incorporating the recommendations of BWRVIP-130. The relevant conditions are specific parameters such as sulfates, halogens, dissolved oxygen, and conductivity that could lead to or are indicative of conditions for, corrosion, erosion, or stress corrosion cracking (SCC) of susceptible materials. The BWR Water Chemistry Program is a mitigation program.

The BWR Water Chemistry Program is supplemented by the Chemistry Program Effectiveness Inspection which provides verification of the effectiveness of the BWR Water Chemistry Program in mitigating the effects of aging.

3.14.2.12 Chemistry Program Effectiveness Inspection (NUREG 1931, Section 3.0.3.1.10)

The Chemistry Program Effectiveness Inspection detects and characterizes the condition of materials in representative low flow and stagnant areas of plant systems influenced by the BWR

Water Chemistry Program, the Closed Cooling Water Chemistry Program, and the Fuel Oil Chemistry Program. The inspection provides direct evidence as to whether, and to what extent, cracking or a loss of material has occurred. The Chemistry Program Effectiveness Inspection will provide confirmation of the effectiveness of the BWR Water, Closed Cooling Water, and Fuel Oil Chemistry Programs in managing the effects of aging.

The Chemistry Program Effectiveness Inspection is a new one-time inspection that will be implemented prior to the period of extended operation. The inspection activities will be conducted within the 10-year period prior to the period of extended operation.

3.14.2.13 Closed Cooling Water Chemistry Program (NUREG 1931, Section 3.0.3.2.7)

The Closed Cooling Water Chemistry Program is an existing program that mitigates damage due to loss of material and cracking of plant components that are within the scope of license renewal and that contain treated water in a closed cooling water system or component (e.g., a heat exchanger) served by a closed cooling water system. The program manages the relevant conditions that could lead to the onset and propagation of a loss of material or cracking through proper monitoring and control of corrosion inhibitor concentrations consistent with the pertinent EPRI water chemistry guideline. The Closed Cooling Water Chemistry Program is a mitigation program.

The Closed Cooling Water Chemistry Program is supplemented by the Chemistry Program Effectiveness Inspection which provides verification of the effectiveness of the Closed Cooling Water Chemistry Program in mitigating the effects of aging.

3.14.2.14 Condensate and Refueling Water Storage Tanks Inspection (NUREG 1931, Section 3.0.3.1.9)

The Condensate and Refueling Water Storage Tanks Inspection detects and characterizes the conditions on the bottom surfaces of the Condensate Storage Tanks and the Refueling Water Storage Tank. The inspection provides direct evidence through volumetric and/or visual examination as to whether, and to what extent, a loss of material due to crevice, general or pitting corrosion has occurred or is likely to occur in inaccessible areas (i.e., tank base/bottom) that could result in a loss of intended function.

The Condensate and Refueling Water Storage Tanks Inspection is a new one-time inspection that will be implemented prior to the period of extended operation. The inspection activities will be conducted within the 10-year period prior to the period of extended operation.

3.14.2.15 Containment Leakage Rate Test Program (NUREG 1931, Section 3.0.3.1.22)

The Containment Leakage Rate Test Program is an existing program that manages aging effects for the Primary Containment and systems penetrating the Primary Containment, which are the containment liner and Primary Containment penetrations including personnel airlock, equipment hatches, and control rod drive hatch. The Containment Leakage Rate Test Program provides assurance that leakage from the Primary Containment will not exceed maximum values for containment leakage.

3.14.2.16 Cooling Units Inspection (NUREG 1931, Section 3.0.3.1.11)

The Cooling Units Inspection detects and characterizes the condition of aluminum, carbon steel, copper alloy, and stainless steel cooling unit components that are exposed to a ventilation environment or to an uncontrolled raw water environment from cooling unit drain pans, and of certain heat exchanger components exposed to treated water or ventilation environments in the Control Structure Chilled Water, the Primary Containment Atmosphere Circulation, and the Control Structure and Reactor Building HVAC systems. The inspection provides direct evidence as to whether, and to what extent, a loss of material due to crevice, galvanic, general, or pitting corrosion, or reduction in heat transfer due to fouling of heat exchanger tubes and fins, has occurred or is likely to occur in these systems that could result in a loss of intended function.

The Cooling Units Inspection is a new one-time inspection that will be implemented prior to the period of extended operation. The inspection activities will be conducted within the 10-year period prior to the period of extended operation.

3.14.2.17 Crane Inspection Program (NUREG 1931, Section 3.0.3.1.8)

The Crane Inspection Program is an existing program that manages loss of material for cranes (including bridge, trolley, rails, and girders), monorails, and hoists within the scope of license renewal. The Crane Inspection Program is based on guidance contained in ANSI B30.2 for overhead and gantry cranes, ANSI B30.11 for monorail systems and underhung cranes, and ANSI B30.16 for overhead hoists. The inspections monitor structural members for signs of corrosion other than minor surface corrosion.

3.14.2.18 Fire Protection Program (NUREG 1931, Section 3.0.3.2.8)

The Fire Protection Program is an existing program that is described in the Fire Protection Review Report (FPRR) and which is credited with aging management of components with fire barrier functions in the scope of license renewal. Periodic visual inspections and functional tests are performed, as appropriate, of fire dampers, fire barrier walls, ceilings and floors, fire rated penetration seals (fire stops), fire wraps, fireproofing, and fire doors to ensure that functionality and operability are maintained. The Fire Protection Program is a condition monitoring program, comprised of tests and inspections generally in accordance with the applicable National Fire Protection Association (NFPA) recommendations.

3.14.2.19 Fire Water System Program (NUREG 1931, Section 3.0.3.2.9)

The Fire Water System Program (sub-program of the overall Fire Protection Program) is an existing program that is described in the Fire Protection Review Report (FPRR) and which is credited with aging management of the water suppression components in the scope of license renewal. Periodic inspection and testing of the water-based fire suppression systems provides reasonable assurance that the systems will remain capable of performing their intended function. Periodic inspection and testing activities include hydrant and hose station inspections, fire main flushing, flow tests, and sprinkler inspections. The Fire Water System Program is a condition monitoring program, comprised of tests and inspections generally in accordance with the applicable National Fire Protection Association (NFPA) recommendations.

Prior to the period of extended operation, the Fire Water System Program will be enhanced to incorporate:

a) Sprinkler head sampling/replacements, in accordance with NFPA 25;

b) Ultrasonic testing of representative above ground portions of water suppression piping that are exposed to water but which do not normally experience flow, are associated with a dry-pipe sprinkler system and may contain stagnant water, or is pre-action or deluge piping that is normally dry but may have been wetted and not completely dried;

c) At least one visual inspection (opportunistic or focused) of the internal surface of buried fire water piping within the 10 year period prior to the period of extended operation; and

d) At least one inspection per year of 'wet' fire protection piping for wall thickness and pipe blockage, if no opportunistic inspection is completed.

3.14.2.20 Flow-Accelerated Corrosion (FAC) Program (NUREG 1931, Section 3.0.3.1.7)

The Flow-Accelerated Corrosion (FAC) Program is an existing program that manages loss of material for carbon steel components located in systems that are susceptible to flow-accelerated corrosion, also called erosion/corrosion. The Flow-Accelerated Corrosion (FAC) Program is a condition monitoring program which ensures that the integrity of piping systems susceptible to flow-accelerated corrosion is maintained. The program was developed in response to NRC Bulletin 87-01 and NRC Generic Letter 89-08. The Flow-Accelerated Corrosion (FAC) Program follows the guidance and recommendations of EPRI NSAC-202L and combines the elements of predictive analysis, inspections (to baseline and monitor wall-thinning), industry experience, station information gathering and communication, and engineering judgment to monitor and predict flow-accelerated corrosion wear rates.

3.14.2.21 Fuel Oil Chemistry Program (NUREG 1931, Section 3.0.3.2.11)

The Fuel Oil Chemistry Program is an existing program that maintains fuel oil quality in order to mitigate damage due to loss of material and cracking of susceptible materials for plant components that are within the scope of license renewal and that contain fuel oil. The program manages the relevant conditions that could lead to the onset and propagation of loss of material or cracking through proper monitoring and control of fuel oil contamination consistent with pertinent plant technical specifications/requirements and American Society for Testing of Materials (ASTM) standards for fuel oil. The relevant conditions are specific contaminants such as water or microbiological organisms in the fuel oil that could lead to corrosion or stress corrosion cracking (SCC) of susceptible materials. Exposure to these contaminants is minimized by verifying the quality of new fuel oil before it enters the storage tanks and by periodic sampling to ensure that the tanks are free of water and particulates. The Fuel Oil Chemistry Program is a mitigation program.

The Fuel Oil Chemistry Program is supplemented by the Chemistry Program Effectiveness Inspection which provides verification of the effectiveness of the Fuel Oil Chemistry Program in mitigating the effects of aging.

Heat Exchanger Inspection (NUREG 1931, Section 3.0.3.1.12)

The Heat Exchanger Inspection detects and characterizes the condition of heat exchanger tubes in the Control Structure Chilled Water (CSCW), Fire Protection (FP) High Pressure Coolant Injection (HPCI), and Reactor Core Isolation Cooling (RCIC) systems.

The scope of the Heat Exchanger Inspection includes the CSCW chiller oil cooler and chiller evaporator, the FP diesel engine driven fire pump heat exchanger and lube oil cooler, and the HPCI and RCIC lube oil coolers. The inspection provides direct evidence as to whether, and to what extent, cracking due to SCC or reduction in heat transfer due to fouling has occurred or is likely to occur that could result in a loss of intended function.

The Heat Exchanger Inspection is a new one-time inspection that will be implemented prior to the period of extended operation. The inspection activities will be conducted within the 10-year period prior to the period of extended operation.

3.14.2.22 Inservice Inspection (ISI) Program (NUREG 1931, Section 3.0.3.2.1)

The Inservice Inspection (ISI) Program is an existing program that manages cracking and loss of material of multiple reactor coolant system pressure boundary components, including the reactor vessel, a limited number of internals components, and the reactor coolant system pressure boundary. The Inservice Inspection (ISI) Program was developed as required by 10 CFR 50.55a. The program is in accordance with the requirements detailed in the 1998 Edition through the 2000 Addenda, of ASME Boiler and Pressure Vessel Code, Section XI, Division 1, Subsections IWA, IWB, IWC, IWD, IWE, IWF, IWL, Mandatory Appendices, Inspection Program B of IWA-2432, and approved ASME Code Cases.

The inservice inspections conducted throughout the service life of SSES will comply, to the extent practical, with the requirements of the ASME Code Section XI Edition and Addenda incorporated by reference in 10 CFR 50.55a(b) 12 months prior to the start of the inspection interval, subject to prior approval via letter from the NRC. This is consistent with NRC statements of consideration (SOC) for 10 CFR 54 associated with the adoption of new editions and addenda of the ASME Code in 10 CFR 50.55a.

3.14.2.23 Inservice Inspection (ISI) Program – IWE (NUREG 1931, Section 3.0.3.1.19)

The Inservice Inspection (ISI) Program – IWE is an existing program that establishes responsibilities and requirements for conducting IWE inspections as required by 10 CFR 50.55a. The Inservice Inspection (ISI) Program – IWE includes visual examination of all accessible surface areas of the steel liner for the reinforced concrete Primary Containment and its integral attachments, containment seals and gaskets, and containment pressure-retaining bolting in accordance with the requirements of the ASME Boiler and Pressure Vessel Code, Section XI, Division 1 (edition and addenda described in 3.14.2.23) for Subsection IWE.

The inservice examinations conducted throughout the service life of SSES will comply, to the extent practical, with the requirements of the ASME Section XI Edition and Addenda incorporated by reference in 10 CFR 50.55a(b) 12 months prior to the start of the inspection interval, subject to prior approval via letter from the NRC. This is consistent with NRC statements of consideration (SOC), for 10 CFR 54, associated with the adoption of new editions and addenda of the ASME Code in 10 CFR 50.55a.

3.14.2.24 Inservice Inspection (ISI) Program – IWF (NUREG 1931, Section 3.0.3.1.21)

The Inservice Inspection (ISI) Program – IWF is an existing program that establishes responsibilities and requirements for conducting IWF Inspections for ASME Class 1, 2, and 3 component supports as required by 10 CFR 50.55a. The Inservice Inspection (ISI) Program – IWF visual examination for supports is based on sampling of the total support population. The sample size varies depending on the ASME Class. The largest sample size is specified for the most critical supports (ASME Class 1). The sample size decreases for the less critical supports (ASME Class 2 and 3). The primary inspection method employed is visual examination. Degradation that potentially compromises support function or load capacity is identified for evaluation. Supports requiring corrective actions are re-examined during the next inspection period in accordance with the requirements of ASME Section XI, Division 1 (edition and addenda described in 3.14.2.23) for Subsection IWF.

The inservice examinations conducted throughout the service life of SSES will comply, to the extent practical, with the requirements of ASME Section XI Edition and Addenda incorporated by reference in 10 CFR 50.55a(b) 12 months prior to the start of the inspection interval, subject to prior approval via letter from the NRC. This is consistent with NRC statements of consideration (SOC), for 10 CFR 54, associated with the adoption of new editions and addenda of the ASME Code in 10 CFR 50.55a.

3.14.2.25 Inservice Inspection (ISI) Program - IWL (NUREG 1931, Section 3.0.3.1.20)

The Inservice Inspection (ISI) Program – IWL is an existing program that establishes responsibilities and requirements for conducting IWL Inspections as required by 10 CFR 50.55a. The Inservice Inspection (ISI) Program – IWL includes visual examination of all accessible surface areas of the reinforced concrete Primary Containment in accordance with the requirements of ASME Section XI, Division 1 (edition and addenda described in 3.14.2.23) for Subsection IWL.

No applicable aging effects have been identified for the Primary Containment concrete. However, the Inservice Inspection (ISI) Program – IWL will be used to confirm the absence of significant aging effects for the extended period of operation.

The inservice examinations conducted throughout the service life of SSES will comply, to the extent practical, with the requirements of ASME Section XI Edition and Addenda incorporated by reference in 10 CFR 50.55a(b) 12 months prior to the start of the inspection interval, subject to prior approval via letter from the NRC. This is consistent with NRC statements of consideration, for 10 CFR 54, associated with the adoption of new editions and addenda of the ASME Code in 10 CFR 50.55a.

3.14.2.26 Leak Chase Channel Monitoring Activities (NUREG 1931, Section 3.0.3.3.2)

The Leak Chase Channel Monitoring Activities is an existing program that consists of observation and surveillance activities to detect leakage from the spent fuel pool and the fuel shipping cask storage pool liners due to aging and age-related degradation. The Leak Chase Channel Monitoring Activities is a condition monitoring program.

3.14.2.27 Lubricating Oil Analysis Program (NUREG 1931, Section 3.0.3.2.15)

The Lubricating Oil Analysis Program is an existing program that mitigates damage due to loss of material and reduction of heat transfer due to fouling for plant components that are within the scope of license renewal and exposed to lubricating oil. The program manages the relevant conditions that could lead to the onset and propagation of a loss of material, or reduction in heat transfer for heat exchanger tubes, through proper monitoring consistent with manufacturer's recommendations, the equipment's importance to safe plant operation, equipment accessibility and American Society for Testing of Materials (ASTM) standards for lubricating oil. The relevant conditions are specific parameters including particulate and water concentrations, viscosity, neutralization number, and flash point that could lead to, or are indicative of, conditions for age-related degradation of susceptible materials. The Lubricating Oil Analysis Program is a mitigation program.

Prior to the period of extended operation, the Lubricating Oil Analysis Program will be enhanced to include sampling of the lubricating oil from the Control Structure Chiller, Reactor Building Chiller, and Diesel Engine Driven Fire Pump when the oil is changed. The oil will be tested for water and for particle count.

The Lubricating Oil Analysis Program is supplemented by the Lubricating Oil Inspection which provides verification of the effectiveness of the Lubricating Oil Analysis Program in mitigating the effects of aging.

3.14.2.28 Lubricating Oil Inspection (NUREG 1931, Section 3.0.3.1.13)

The Lubricating Oil Inspection detects and characterizes the condition of materials in systems and components for which the Lubricating Oil Analysis Program is credited with aging management. The inspection provides direct evidence as to whether, and to what extent, a loss of material or a reduction in heat transfer due to fouling has occurred.

The Lubricating Oil Inspection is a new one-time inspection that will be implemented prior to the period of extended operation. The inspection activities will be conducted within the 10-year period prior to the period of extended operation.

3.14.2.30 Not Used (NUREG 1931, Section 3.0.3.1.14)

3.14.2.31 Masonry Wall Program (NUREG 1931, Section 3.0.3.2.16)

The Masonry Wall Program is an existing program that consists of inspection activities to detect cracking of masonry walls within the scope of license renewal. Masonry walls that perform a fire barrier intended function are also managed by the Fire Protection Program. The Masonry Wall Program is implemented as part of the Structures Monitoring Program. The Masonry Wall Program performs visual inspection of external surfaces of masonry walls.

Prior to the period of extended operation, the Masonry Wall Program will be enhanced to specify that for each masonry wall, the extent of observed masonry cracking and/or degradation of steel edge supports/bracing be evaluated to ensure that the current evaluation basis is still valid.

3.14.2.32 Metal-Enclosed Bus Inspection Program (NUREG 1931, Section 3.0.3.1.26)

The Metal-Enclosed Bus Inspection Program manages the aging of the metal-enclosed bus within the scope of license renewal. The program provides for inspection of the applicable metal-enclosed bus on a 10-year interval, in order to determine if age-related degradation is occurring.

The Metal-Enclosed Bus Inspection Program is a new aging management program that will be implemented prior to the period of extended operation.

3.14.2.33 Monitoring and Collection System Inspection (NUREG 1931, Section 3.0.3.1.15)

The Monitoring and Collection System Inspection detects and characterizes the conditions on the internal surfaces of subject components that are exposed to equipment/area drainage water and other potential contaminants/fluids. The inspection provides direct evidence as to whether, and to what extent, a loss of material has occurred or is likely to occur in the Liquid Waste Management System that could result in a loss of intended function.

The Monitoring and Collection System Inspection is a new one-time inspection that will be implemented prior to the period of extended operation. The inspection activities will be conducted within the 10-year period prior to the period of extended operation.

3.14.2.34 Non-EQ Cables and Connections Used in Low-Current Instrumentation Circuits Program (NUREG 1931, Section 3.0.3.1.24)

The Non-EQ Cables and Connections Used in Low-Current Instrumentation Circuits Program manages the age-related degradation associated with non-EQ, low-current instrumentation cables and connections within the scope of license renewal.

The program applies to in-scope, non-EQ electrical cables and connections used in neutron monitoring and radiation monitoring circuits with sensitive, low-current signals. The Non-EQ Cables and Connections Used in Low-Current Instrumentation Circuits Program will perform testing of the applicable cable systems to identify reduction in insulation resistance. The tests will be performed at least every ten years, with the frequency to be determined by engineering evaluation.

The Non-EQ Cables and Connections Used in Low-Current Instrumentation Circuits Program is a new aging management program that will be implemented prior to the period of extended operation.

3.14.2.35 Non-EQ Electrical Cables and Connections Visual Inspection Program (NUREG 1931, Section 3.0.3.1.23)

The Non-EQ Electrical Cables and Connections Visual Inspection Program manages the aging of non-EQ electrical cables and connections within the scope of license renewal. The program provides for visual inspection on a 10-year interval of accessible, non-EQ electrical cables and connections, in order to determine if age-related degradation is occurring, particularly in plant areas with high temperatures and/or high radiation levels.

The Non-EQ Electrical Cables and Connections Visual Inspection Program is a new aging management program that will be implemented prior to the period of extended operation.

3.14.2.36 Non-EQ Inaccessible Medium-Voltage Cables Program (NUREG 1931, Section 3.0.3.1.25)

The Non-EQ Inaccessible Medium-Voltage Cables Program manages the aging of non-EQ inaccessible medium-voltage electrical cables subject to wetting within the scope of license renewal. The program provides for the periodic testing of non-EQ inaccessible medium-voltage electrical cables, in order to determine if age-related degradation is occurring, and includes provision for the inspection of associated manholes to identify any collection of water. The cable testing frequency will be based on plant operating experience, but will be performed at least once every ten years. The electrical manhole inspection frequency will be based on plant operating experience, but will be performed at least once every two years.

The Non-EQ Inaccessible Medium-Voltage Cables Program is a new aging management program that will be implemented prior to the period of extended operation.

3.14.2.37 Non-EQ Electrical Cable Connections Program (NUREG 1931, Section 3.0.3.1.27)

The Non-EQ Electrical Cable Connections Program manages the aging for the metallic parts of non-EQ electrical cable connections within the scope of license renewal. The program addresses cable connections that are used to connect cable conductors to other cables or electrical devices. Aging management for the metallic parts of the non-EQ electrical cable connections that are subject to aging stressors will be provided by testing. A representative sample of non-EQ electrical cable connections will be selected for testing, considering the effects of their application (high, medium, and low voltage), circuit loading, and location with respect to electrical connection stressors. Thermography will be used to test a representative sample of cable connections to provide an indication of the integrity of the connections. The tests will be performed at least every ten years, with the frequency to be determined by engineering evaluation.

The Non-EQ Electrical Cable Connections Program is a new aging management program that will be implemented prior to the period of extended operation.

3.14.2.38 Piping Corrosion Program (NUREG 1931, Section 3.0.3.2.6)

The Piping Corrosion Program is an existing program that manages fouling due to particulates (e.g., corrosion products) and biological material (micro- and/or macro-organisms), and loss of material due to corrosion and erosion for components located in systems within the scope of the program that are exposed to a raw water environment. The program also manages the applicable aging effects for the internal environments of heat exchanger components within the scope of the program.

The Piping Corrosion Program is a combination of condition monitoring program (consisting of inspections, surveillances, and testing to detect the presence of, and to assess the extent of, damaged coatings, fouling and loss of material) and a mitigation program (consisting of chemical treatments and cleaning activities to minimize fouling and loss of material, and use of protective coatings in areas vulnerable to erosion).

Prior to the period of extended operation, the Piping Corrosion Program will be enhanced to include the Standby Gas Treatment System loop seals, and to also incorporate performance, documentation and trending of opportunistic visual inspections (during normal maintenance/repair activities).

3.14.2.39 Preventive Maintenance Activities – RCIC/HPCI Turbine Casings (NUREG 1931, Section 3.0.3.3.3)

The Preventive Maintenance Activities – RCIC/HPCI Turbine Casings is an existing program that manages loss of material on the internal surfaces of the Reactor Core Isolation Cooling (RCIC) and High Pressure Coolant Injection (HPCI) pump turbine casings, and on the internal surfaces of associated piping and piping components (rupture disks and valve bodies), that are constructed of carbon steel or cast iron.

The Preventive Maintenance Activities – RCIC/HPCI Turbine Casings is a condition monitoring program, consisting of inspections and surveillance activities to detect aging and age-related degradation.

Prior to the period of extended operation, the Preventive Maintenance Activities – RCIC / HPCI Turbine Casings will be enhanced to incorporate:

a) A specific step to perform a visual inspection of the RCIC turbine casing.

b) Performance of inspections by qualified personnel using VT-3 or equivalent inspection methods, and reporting and trending of inspection results.

c) Specific acceptance criteria for inspections.

3.14.2.40 Reactor Head Closure Studs Program (NUREG 1931, Section 3.0.3.1.2)

The Reactor Head Closure Studs Program is an existing program that manages cracking for the reactor head closure studs. The Reactor Head Closure Studs Program includes (a) inservice inspection in conformance with the requirements of ASME Code, Section XI, Subsection IWB (edition and addenda described in 3.14.2.23), Table IWB 2500-1, and (b) preventive measures in accordance with Regulatory Guide 1.65 to mitigate cracking. The Reactor Head Closure Studs Program is implemented by the design of the plant and the Inservice Inspection (ISI) Program.

3.14.2.41 Reactor Vessel Surveillance Program (NUREG 1931, Section 3.0.3.2.12)

The Reactor Vessel Surveillance Program is an existing program that manages reduction of fracture toughness for the low alloy steel reactor vessel shell and welds in the beltline region. The Reactor Vessel Surveillance Program is a condition monitoring program developed in response to 10 CFR 50 Appendix H.

The SSES Reactor Vessel Surveillance Program is part of the Integrated Surveillance Program (ISP) described in BWRVIP-78, BWRVIP-86-A and BWRVIP-116, and approved by the NRC staff. BWRVIP-116 extends the ISP to cover the period of extended operation. SSES will follow

the requirements of the BWRVIP ISP and will apply the ISP data to SSES Units 1 and 2. The NRC approved the use of the BWRVIP ISP in place of a unique plant program at SSES.

The SSES Reactor Vessel Surveillance Program will be enhanced, as necessary, to ensure that additional requirements that are specified in the NRC safety evaluation dated March 1, 2006, for BWRVIP-116 will be addressed before the period of extended operation. The program will include a requirement that, if a standby capsule is removed from either of the SSES Unit 1 or Unit 2 reactor vessels, without the intent to test it, the capsule will be stored in a manner which maintains it in a condition which would permit its future use, including during the period of extended operation, if necessary,

3.14.2.42 RG 1.127 Water-Control Structures Inspection (NUREG 1931, Section 3.0.3.2.18)

The RG 1.127 Water-Control Structures Inspection is an existing program that consists of inspection and surveillance activities to detect aging and age-related degradation. The RG 1.127 Water-Control Structures Inspection ensures the structural integrity and operational adequacy of the Spray Pond (including concrete liner, emergency spillway, and riser encasements), ESSW Pumphouse (including pump intake chambers, overflow weir and chamber, and structural components within the ESSW Pumphouse), and the earthen embankments along the Spray Pond.

Prior to the period of extended operation, the RG 1.127 Water-Control Structures Inspection will be enhanced to add the Spray Pond (including concrete liner, emergency spillway, riser encasements, and earthen embankments) to its scope for inspection. The program will be enhanced to include RG 1.127 inspection elements and degradation mechanisms for water-control structure inspection and to include acceptance criteria for water-control structures.

3.14.2.43 Selective Leaching Inspection (NUREG 1931, Section 3.0.3.1.17)

The Selective Leaching Inspection detects and characterizes the conditions on internal and external surfaces of subject components. The inspection provides direct evidence through a combination of visual examination and hardness testing of whether, and to what extent, a loss of material due to selective leaching has occurred or is likely to occur that could result in a loss of intended function.

The Selective Leaching Inspection is a new one-time inspection that will be implemented prior to the period of extended operation. The inspection activities will be conducted within the 10-year period prior to the period of extended operation.

3.14.2.44 Small Bore Class 1 Piping Inspection (NUREG 1931, Section 3.0.3.1.18)

The Small Bore Class 1 Piping Inspection is a one-time inspection to detect cracking resulting from thermal and mechanical loading or intergranular stress corrosion. The inspection will provide assurance that cracking of small bore Class 1 piping is not occurring or an evaluation of any detected crack indications will be performed to justify continued operation with no further monitoring, such that an aging management program (AMP) is not warranted. The inspection will also confirm the effectiveness of the BWR Water Chemistry Program mitigating cracking due to intergranular stress corrosion. The Small Bore Class 1 Piping Inspection is applicable to

small bore ASME Code Class 1 piping and systems less than 4 inches nominal pipe size (NPS 4), which includes pipes, fittings, and branch connections.

The Small Bore Class 1 Piping Inspection is a new one-time inspection that will be implemented prior to the period of extended operation. The inspection activities will be conducted within the 10-year period prior to the period of extended operation.

3.14.2.45 Structures Monitoring Program (NUREG 1931, Section 3.0.3.2.17)

The Structures Monitoring Program is an existing program that manages age-related degradation of plant structures and structural components within its scope to ensure that each structure or structural component retains the ability to perform its intended function. Aging effects are detected by visual inspection of external surfaces prior to the loss of the structure's or component's intended function.

This program implements provisions of the Maintenance Rule, 10 CFR 50.65, that relate to structures, masonry walls, and water-control structures. Concrete and masonry walls that perform a fire barrier intended function are also managed by the Fire Protection Program.

Prior to the period of extended operation, the Structures Monitoring Program will be enhanced to include structures within the scope of license renewal. The program will be enhanced to specify inspections of a below grade structural wall or structural component that becomes accessible through excavation. The program will be enhanced to clarify the component types included as "structural components". The program will be enhanced to specify degradation mechanisms for elastomer and earthen embankment inspection. The program will be enhanced to include RG 1.127 inspection elements for water-control structures. The program will be enhanced to include requirements for review of site groundwater and raw water parameters. The program will also be enhanced to specify inspection requirements for masonry walls. The program will be enhanced to include direction for quantifying, monitoring and trending of inspection results, guidance for inspection reporting, data collection and documentation, acceptance criteria and critical parameters to monitor degradation and to trigger level of inspection and initiation of corrective action; and better alignment with referenced Industry codes, standards and guidelines. The program will be enhanced to include specific qualification requirements for the inspector.

3.14.2.46 Supplemental Piping/Tank Inspection (NUREG 1931, Section 3.0.3.1.16)

The Supplemental Piping/Tank Inspection detects and characterizes the condition of carbon steel, stainless steel and cast iron components that are exposed to moist air environments, particularly the aggressive alternate wet/dry environment that exists at air-water interfaces, and for internal surfaces of diesel exhaust components due to periodic exposure to exhaust gases containing moisture and contaminants. The inspection provides direct evidence as to whether, and to what extent, a loss of material or cracking (of stainless steel exposed to diesel exhaust) has occurred or is likely to occur that could result in a loss of intended function.

The Supplemental Piping/Tank Inspection is a new one-time inspection that will be implemented prior to the period of extended operation. The inspection activities will be conducted within the 10-year period prior to the period of extended operation.

3.14.2.47 System Walkdown Program (NUREG 1931, Section 3.0.3.2.14)

The System Walkdown Program is an existing program that manages the following aging effects for the external surfaces, and in some cases the internal surfaces, of mechanical components within the scope of license renewal:

a) Loss of material for metals that are exposed to indoor air, outdoor air, or ventilation environments, including both the HVAC-type internal environments and ambient air internal environments, such as that found in the upper portion of a vented tank.

b) Cracking and/or change in material properties for elastomers (neoprene and rubber) and polymers (Teflon) that are exposed to indoor air or ventilation environments.

The System Walkdown Program is a condition monitoring program, consisting of observation and surveillance activities to detect aging and age-related degradation.

Prior to the period of extended operation, the System Walkdown Program will be enhanced to include the license renewal systems that contain mechanical components whose external surfaces require aging management during the period of extended operation. The program will also be enhanced to address opportunistic inspections of normally inaccessible components (e.g., those that are insulated), and those that are accessible only during refueling outages. The program will also be enhanced by addition of a routine activity to inspect elastomers and polymers for cracking and/or change in material properties and to include inspection of other metals, copper alloy and stainless steel. The program will also be enhanced to sample normally inaccessible components in underground vaults, pits, and manholes. In addition, the program will be enhanced to include a visual and ultrasonic inspection of the external surfaces of piping passing into structures through penetrations (underground piping) for those penetrations with a history of leakage.

3.14.2.48 Thermal Aging and Neutron Embrittlement of Cast Austenitic Stainless Steel (CASS) Program (NUREG 1931, Section 3.0.3.1.6)

The Thermal Aging and Neutron Embrittlement of Cast Austenitic Stainless Steel (CASS) Program augments the visual inspection of the reactor vessel internals done in accordance with the ASME Code, Section XI, Subsection IWB, Category B-N-2. The inspection is augmented to detect the effects of loss of fracture toughness due to thermal aging and neutron irradiation embrittlement of cast austenitic stainless steel (CASS) reactor vessel internal components. The aging management program includes (a) identification of susceptible components determined to be limiting from the standpoint of thermal aging susceptibility (i.e., ferrite and molybdenum contents, casting process, and operating temperature) and/or neutron irradiation embrittlement (neutron fluence), and (b) for each potentially susceptible component, aging management is accomplished through either a supplemental examination of the affected component based on the neutron fluence to which the component has been exposed as part of the SSES 10-year Inservice Inspection (ISI) Program during the license renewal term, or a component-specific evaluation to determine its susceptibility to loss of fracture toughness.

The Thermal Aging and Neutron Embrittlement of Cast Austenitic Stainless Steel (CASS) Program is a new aging management program that will be implemented prior to the period of extended operation.

3.14.2.49 Fatigue Monitoring Program (NUREG 1931, Section 3.0.3.2.19)

The Fatigue Monitoring Program manages fatigue for all Class 1 components, including the reactor pressure vessel. In order not to exceed the design basis limit on fatigue usage, the aging management program monitors and tracks the number and severity of critical thermal and pressure transients and calculates the fatigue usage for the limiting locations of the reactor coolant pressure boundary.

Prior to the period of extended operation the program will be enhanced by adding the following required actions:

a) The program will verify that components which have satisfied ASME Section III, Paragraph N-415.1 requirements (i.e., RPV nozzles N6A, N6B, and N7) continue to satisfy these requirements prior to and during the period of extended operation, thereby allowing fatigue to be continued to be addressed under N-415.1.

b) The program will review Class 1 valve fatigue analyses and other fatigue-related TLAA, such as flued head analyses and high energy line break evaluations, when sufficient fatigue accumulation has occurred, to determine if additional actions are required to address fatigue-related concerns.

c) The program will define specific fatigue usage values for all monitored locations, including those locations that account for the effect of the reactor water environment that, if reached, will require further action. These fatigue usage values shall be conservatively set to values that will allow for not less than 4 years of additional plant operation before the actual fatigue usage at any location would reach the design basis limit. Upon reaching the defined usage at a location, the program will require an action request to be generated. The action request will require further engineering evaluation to resolve the issue.

d) The program will implement one or more of the following actions, if fatigue usage at a monitored location, including any location that accounts for the effect of the reactor water environment, is projected to reach the design basis limit prior to the end of the period of extended operation:

   1) Further refinement of the fatigue analyses to lower the CUFs to less than the allowable;

   2) Repair of the affected components;

   3) Replacement of the affected components;

   4) Management by an inspection program that has been reviewed and approved by the NRC.

3.14.2.50 Preventive Maintenance Activities – Main Turbine Casing (NUREG 1931, Section 3.0.3.3.4)

The Preventive Maintenance Activities – Main Turbine Casing is an existing program that manages loss of material due to flow-accelerated corrosion on the internal surfaces of the high pressure casing for the main turbine.

The Preventive Maintenance Activities – Main Turbine Casing is a condition monitoring program consisting of inspections performed on a nominal 10-year (12-year maximum) frequency to detect aging and age-related degradation.

Prior to the period of extended operation, the Preventative Maintenance Activities – Main Turbine Casing will be enhanced to specify that the inspection of the high pressure turbine shell will consist of a VT-3 or equivalent visual inspection of accessible surfaces and an ultrasonic examination of selected locations for wall thickness.

3.14.2.51 Fuse Holders Program (NUREG 1931, Section 3.0.3.2.20)

The Fuse Holders Program is a new aging management program that manages increased connection resistance due to fatigue of fuse holder clamps. The program provides for periodic inspection of fuse holder clamps within the scope of license renewal that are not in enclosures containing active components and whose fuses are scheduled for removal once every 12 months, or more frequently.

The Fuse Holders Program is a condition monitoring program consisting of inspections performed on a 10-year frequency to detect aging and age-related degradation.

The Fuse Holders Program is a new aging management program that will be implemented prior to the period of extended operation.

3.14.3 EVALUATION OF TIME-LIMITED AGING ANALYSES (TLAA)

In accordance with 10 CFR 54.21(c), an application for a renewed operating license requires an evaluation of time-limited aging analyses (TLAA) for the period of extended operation. The license renewal evaluation of time-limited aging analyses (TLAA) identified existing aging management programs necessary to provide reasonable assurance that components within the scope of license renewal will continue to perform their intended functions consistent with the current licensing basis (CLB) for the period of extended operation. This section identifies the TLAA, summarizes the evaluation of each, and, where necessary, describes the aging management programs that will be required. These aging management programs will be implemented during the period of extended operation. One-time inspections will be conducted within the 10-year period prior to beginning the period of extended operation.

Three elements of an effective aging management program that are common to all aging management programs are corrective actions, confirmation process, and administrative controls. These elements are included in the SSES Operational Quality Assurance (OQA) Program, which implements the requirements of 10 CFR 50, Appendix B. Prior to the period of extended operation, the elements of corrective action, confirmation process, and administrative controls in the SSES OQA Program will be applied to required aging management programs for both safety-related and nonsafety-related structures and components determined to require aging management during the period of extended operation.

3.14.3.1 Reactor Vessel Neutron Embrittlement

The ferritic materials of the reactor vessel are subject to embrittlement due to high energy neutron exposure. Reactor vessel neutron embrittlement is a TLAA. Neutron fluence, upper

shelf energy, adjusted reference temperature, and vessel pressure-temperature limits are time-dependent items that must be investigated to evaluate vessel embrittlement.

3.13.3.1.1 Neutron Fluence

High energy (>1 MeV) neutron fluence for the welds and shells of the reactor pressure vessel beltline region was calculated using the RAMA fluence methodology. The RAMA methodology was developed for the Electric Power Research Institute and the Boiling Water Reactor Vessel and Internals Project and is approved by the NRC for use at SSES Units 1 and 2. Use of this methodology for evaluations of fluence for the SSES units was performed in accordance with guidelines presented in Regulatory Guide 1.190. The evaluations determined values for neutron fluence for extended power uprate (EPU) conditions and for extended operation out to 54 effective full power years (EFPY), i.e., at the end of 60 years of operation. Using actual reactor core power histories to-date and conservative estimates of future core designs for each unit, extended operation to 60 years will be bounded by 54 EFPY.

Neutron fluence is not a TLAA, it is a time-limited assumption used in various neutron embrittlement TLAA.

3.14.3.1.2 Upper Shelf Energy Evaluation

10 CFR 50, Appendix G requires that upper shelf energy (USE) values for reactor pressure vessel materials include the effects of neutron radiation. It states that USE for the beltline materials including plates and welds be maintained at no less than 50 ft-lb for the life of the reactor vessel. Calculated fluence values for extended power uprate (EPU) and extended operation to 54 EFPY exceed previously determined fluence values based on materials surveillance program information for Units 1 and 2. Therefore, projections of changes in USE for the period of extended operation are required in accordance with 10 CFR 50, Appendix G.

The projections of changes in USE for the period of extended operation for the RPV beltline plates and welds for Units 1 and 2 were determined in accordance with Regulatory Guide (RG) 1.99, Revision 2. For the plates and welds with projected USE values of 50 ft-lb or greater at 54 EFPY, the criterion of 10 CFR 50, Appendix G, has been met and no further evaluation is required.

For plates and welds that do not meet the 50 ft-lb criterion, the equivalent margin analyses (EMA) documented in BWRVIP-74-A were used to demonstrate that the 54 EFPY USE values remain in compliance with 10 CFR 50, Appendix G. As prescribed in BWRVIP-74-A, the predicted decrease in USE from RG 1.99, Figure 2 was compared to the decrease assumed in the EMA for each vessel beltline plate and weld that fails to meet the 50 ft-lb criterion. The results demonstrate that all evaluated plates and welds are bounded by the BWRVIP-74-A equivalent margin analyses.

Therefore, the effects of neutron radiation have been evaluated, and all RPV beltline materials for Units 1 and 2 have been demonstrated to remain in compliance with Appendix G of 10 CFR 50 for the period of extended operation.

3.14.3.1.3 Adjusted Reference Temperature (ART) Analysis

In addition to USE, the other key parameter that characterizes the fracture toughness of a material is the reference temperature for nil-ductility transition (RT_{NDT}). This reference temperature will change as its exposure to neutron radiation increases. The effects of neutron fluence on RT_{NDT} are reflected in the change in this reference temperature, ΔRT_{NDT}, and the resulting adjusted reference temperature, ART, is calculated by adding ΔRT_{NDT} to RT_{NDT} along with appropriate margin to account for uncertainties.

The methodology used to calculate ART for the vessel beltline plates and welds is provided in Regulatory Guide 1.99. The ART values projected to 54 EFPY are used to develop Pressure-Temperature (P-T) limit curves. There are no limits or specific acceptance criteria for the projected ART values.

3.14.3.1.4 Pressure-Temperature (P-T) Limits

To assure that adequate margins of safety are maintained for various modes of reactor operation, 10 CFR 50, Appendix G specifies pressure and temperature requirements for affected materials for the service life of the reactor vessel. The basis for these fracture toughness requirements is ASME Section XI, Appendix G. The ASME Code requires P-T limits be established for hydrostatic pressure tests and leak tests, for operation with the core not critical during heatup and cooldown, and for core critical operation.

Calculations were performed to develop P-T limit curves for SSES Units 1 and 2 for the period of extended operation (54 EFPY). The calculations were performed for the bounding regions of the reactor vessel to account for 54 EFPY fluence projections, which include the effects of EPU conditions. The P-T curves were developed in accordance with 10 CFR 50, Appendix G and the methodology in ASME Section XI, Appendix G. PPL will submit future P-T curve data to the NRC as necessary to comply with 10 CFR 50 Appendix G,

3.14.3.1.5 Reactor Vessel Circumferential Weld Examination Relief

BWRVIP-74-A reiterated the recommendation of BWRVIP-05 that reactor pressure vessel circumferential welds could be exempted from examination. The NRC safety evaluation report for BWRVIP-74 agreed, but required that plants apply for this relief request individually. The relief request should demonstrate that at the expiration of the current license, the circumferential welds satisfy the limiting conditional failure probability for circumferential welds in the (BWRVIP-05) evaluation. This evaluation of circumferential weld parameters is a TLAA.

PPL requested and received relief from circumferential vessel shell weld volumetric examinations. The SSES submittal included an analysis that showed that the reactor vessel parameters at 32 EFPY were within the bounding parameters for Chicago Bridge & Iron (CBI) vessels from the BWRVIP-05 safety evaluation report. As such, there is a lower conditional probability of failure for circumferential welds at SSES than that stated in the NRC's Final Safety Evaluation Report of BWRVIP-05.

The SSES reactor pressure vessel circumferential weld parameters at 54 EFPY will remain within the bounding parameters for CBI vessels at 64 EFPY from the BWRVIP-05 safety evaluation report. As such, the conditional probability of failure for circumferential welds remains below that stated in the safety evaluation report for BWRVIP-05.

PPL will process a relief request for circumferential vessel shell weld volumetric examinations for the period of extended operation in the same manner that has been the practice during the original licensing period.

3.14.3.1.6 Reactor Vessel Axial Weld Failure Probability

The NRC safety evaluation report for BWRVIP-74-A evaluated the failure frequency of axially oriented welds in BWR reactor pressure vessels, and determined failure frequency acceptance criteria for 40 years of reactor operation. Applicants for license renewal must evaluate axially oriented RPV welds to show that their failure frequency remains below the acceptance criteria calculated in the safety evaluation report for BWRVIP-74. An acceptable way to do this is to show that the mean RT_{NDT} of the limiting axial beltline weld at the end of the period of extended operation is less than the values specified in the safety evaluation report for BWRVIP-74.

The SSES axial weld mean RT_{NDT} at 54 EFPY is projected to be well below that in the SER, and thus the SSES axial weld failure frequency is well within the acceptable criteria.

3.14.3.1.7 Reflood Thermal Shock Analysis

FSAR Section 3.13.1, Regulatory Guide 1.2 “Thermal Shock to Reactor Pressure Vessels” addresses the possibility of brittle fracture of the reactor vessel resulting from reflooding of the vessel following a postulated loss of coolant accident. This concern is addressed in NEDO-10029, “An Analytical Study on Brittle Fracture of GE-BWR Vessels Subject to the Design Basis Accident,” in which a conservative analysis is documented. That document provides an upper bound limit on brittle fracture failure for the materials and concludes that catastrophic failure is not possible. However, the analysis performed in NEDO-10029 is only valid for 40 years of operation.

A more recent analysis provides a technical basis for addressing brittle fracture of BWR vessels due to vessel reflood following a design basis Loss of Coolant Accident (LOCA) during the period of extended operation. The analysis is documented in a paper by S. Ranganath of General Electric, entitled “Fracture Mechanics Evaluation of a Boiling Water Reactor Vessel Following a Postulated Loss of Coolant Accident,” which was presented as Paper G1/5 at the Fifth International Conference on Structural Mechanics in Reactor Technology in Berlin, Germany, in August 1979. While the analysis was performed for BWR-6 vessels, it is applicable to the SSES (BWR-4) vessels, based on the following:

a) It evaluated the main steam line break LOCA event, which is bounding for the evaluation of thermal stresses and brittle fracture in the vessel beltline region.

b) It applied to 251-inch, 238-inch, and 201-inch diameter BWR-6 vessels, since the structural details and operating conditions are similar. The SSES vessels are 251-inch diameter and the SSES response to a main steam line break LOCA is equivalent to that of the BWR-6 vessel (i.e., rapid depressurization and blowdown, immediately followed by ECCS injections to reflood the vessel).

c) It analyzed a 6-inch thick wall of a BWR-6 vessel. The SSES vessels are 6.1875 inches thick. A critical parameter of the fracture mechanics analysis is the wall temperature at a depth of 1/4 of the total thickness, 1/4T. The 1/4T location of the thinner BWR-6 vessel

will cool faster than the 1/4T location of the SSES vessels. Thus, the BWR-6 analysis is conservative for the SSES vessels, since a lower temperature at the 1/4T location is worse for brittle fracture concerns.

The analysis presented in the Ranganath paper assumes end-of-life material toughness, which in turn depends on end-of-life adjusted reference temperature (ART). The analysis determined that the peak stress intensity for the LOCA event at the 1/4T location on the BWR-6 vessel would be 100 ksi√inches and that the available fracture toughness at the 1/4T location that coincides with the peak stress intensity would be a minimum of 200 ksi√inches. Thus, since the available toughness exceeds the applied stress intensity, an existing 1/4T flaw in the vessel wall would not propagate following a LOCA.

The BWR-6 analysis conclusion applies to the SSES vessels because the end-of-life (54 EFPY) ART values for the SSES vessels are such that the available material toughness at the 1/4T location of the SSES vessels would remain on the upper shelf of 200 ksi√inches, which exceeds the peak stress intensity of 100 ksi√inches for the analyzed event. Therefore, brittle fracture of the SSES vessels due to reflood thermal shock following a design basis LOCA is not possible during the period of extended operation.

3.14.3.2 Metal Fatigue

Fatigue evaluations for mechanical components are identified as TLAA; therefore, the effects of fatigue must be addressed for license renewal.

PPL monitors fatigue of the ASME Class 1 reactor coolant pressure boundary via the Fatigue Monitoring Program, which uses a computer program, FatiguePro, to count transient cycles and calculate fatigue usage.

Calculation of fatigue usage values is not required for non-Class 1 SSCs. Instead, stress intensification factors and lower stress allowables are used to ensure components are adequately designed for fatigue.

Certain components enveloped by the Primary Containment are also required to be evaluated for fatigue. These include penetrations, hatches, the drywell head, downcomer vents, safety relief valve (SRV) discharge piping, and SRV quenchers.

3.14.3.2.1 Reactor Pressure Vessel Fatigue Analysis

The design transients for the reactor pressure vessel (RPV) assembly are reported in FSAR Table 3.9-1. Design cumulative usage factors (CUFs) for the limiting reactor pressure vessel assembly locations are obtained from applicable design reports. These CUFs were calculated based on applicable design transients.

Metal fatigue for all reactor pressure vessel assembly components is managed by the existing SSES Fatigue Monitoring Program. This program includes requirements for continued monitoring and periodic updates to current and projected CUFs for the limiting reactor pressure vessel locations. The program will be enhanced to include an approach to address CUFs that will exceed the allowable before the end of the period of extended operation. The aging

management approach will include one or more of the following, which is similar to the approach documented in ASME Code Section III Non-mandatory Appendix L:

a) Further refinement of the fatigue analyses to lower the CUFs to less than the allowable

b) Repair of the affected components

c) Replacement of the affected components

d) Management by an inspection program that has been reviewed and approved by the NRC (e.g., periodic non-destructive examination of the affected locations at intervals determined by a method accepted by the NRC)

The original RPV design report was not required to provide an explicit fatigue analysis for nozzles N6A, N6B, and N7, since the nozzles satisfied all requirements of ASME Section III, Paragraph N-415.1. As such, design CUFs were not calculated for these nozzles. The SSES Fatigue Monitoring Program will be enhanced to include a requirement to periodically determine if the requirements of N-415.1 remain satisfied, such that fatigue evaluations are not required for these nozzles prior to entering and during the period of extended operation.

The Fatigue Monitoring Program is credited for managing the effects of fatigue during the period of extended operation. Therefore, the TLAA associated with reactor pressure vessel fatigue are dispositioned in accordance with 10 CFR 54.21(c)(1)(iii).

3.14.3.2.2 Reactor Vessel Internals Fatigue Analyses

The Reactor Internals and Core Support Structures at SSES were designed in accordance with ASME Section III, Subsection NG. The fatigue evaluations performed to demonstrate the design adequacy of the internals for 40 years are TLAA.

Most recently, structural evaluations were performed to address the effects of operation under extended power uprate conditions and the extended period of plant operation to 60 years. The evaluations determined that the fatigue usage factors for all reactor pressure vessel internals remain within the ASME Section III Subsection NG allowable limits.

PPL also monitors the design transients using FatiguePro, as described above under Reactor Pressure Vessel Fatigue Analysis. This monitoring allows PPL to continually assess the potential for plant operating anomalies that could impact the assumptions made in the fatigue evaluations of plant components. In addition to plant transient monitoring, PPL has effectively implemented the inspection requirements of the BWRVIP program at SSES, as described in Section 3.14.2.10 above. These inspections provide further assurance that the effects of aging due to fatigue of the RPV internals will be managed during the period of extended operation.

Structural evaluations have demonstrated that fatigue usage will remain within design limits to the end of the period of extended operation. Also, the BWR Vessel Internals Program is credited for managing the aging effects of the reactor vessel internals during the period of extended operation. Therefore, the TLAA associated with fatigue of the reactor vessel internals are dispositioned in accordance with 10CFR54.21 (c)(1)(ii) and 10CFR54.21(c)(1)(iii).

3.14.3.2.3 Effects of Reactor Coolant Environment on Fatigue Life of Components and Piping (Generic Safety Issue 190)

Applicants for license renewal are required to address the reactor coolant environmental effects on fatigue of plant components. The minimum set of components is suggested to be the six (6) components defined in NUREG/CR-6260, as follows:

a) Reactor vessel shell and lower head

b) Reactor vessel feedwater nozzle

c) Reactor recirculation piping (including inlet and outlet nozzles)

d) Core spray line reactor vessel nozzle and associated Class 1 piping

e) Residual heat removal return line Class 1 piping

f) Feedwater line Class 1 piping

Calculation of a fatigue life adjustment factor, F_{en}, is determined for each fatigue-sensitive component. The environmental fatigue life adjustment factors are applied to the appropriate component CUFs to verify acceptability of the components for the period of extended operation.

Using fatigue data projected by the SSES Fatigue Monitoring Program and methodology accepted by the NRC, as noted above, PPL evaluated the limiting locations (a total of eleven component locations corresponding to the six NUREG/CR-6260 components), as appropriate for the material for each component location. Seven of the eleven locations evaluated have an environmentally adjusted CUF of greater than 1.0.

Prior to entering the period of extended operation, for each location that may exceed a CUF of 1.0 when considering environmental effects, SSES will implement one or more of the following:

a) Further refinement of the fatigue analyses to lower the CUFs to less than the allowable

b) Repair of the affected components

c) Replacement of the affected components

d) Management by an inspection program that has been reviewed and approved by the NRC (e.g., periodic non-destructive examination of the affected locations at intervals determined by a method accepted by the NRC)

Should PPL select the option to manage environmentally-assisted fatigue during the period of extended operation, details of the aging management program such as scope, qualification, method, and frequency will be provided to the NRC prior to the period of extended operation.

The effects of environmentally-assisted fatigue for the limiting locations identified in NUREG/CR-6260 have been evaluated. The effects of environmentally-assisted fatigue for these locations is addressed using one of the four approaches identified above.

The Fatigue Monitoring Program is credited for managing the effects of the reactor coolant environmental effects on fatigue during the period of extended operation. Therefore, the TLAA associated with environmentally-assisted fatigue has been dispositioned in accordance with 10 CFR 54.21(c)(1)(iii).

3.14.3.2.4 Reactor Coolant Pressure Boundary Piping and Component Fatigue Analyses

The Class 1 boundary encompasses all reactor coolant pressure boundary piping (pipe and fittings) and in-line components subject to ASME Section XI, Subsection IWB, inspection requirements.

FSAR Section 3.9 provides details on the design transients to be considered in the fatigue analyses of reactor coolant pressure boundary (RCPB) components.

The SSES Fatigue Monitoring Program tracks the fatigue usage at the limiting locations throughout the RCPB. The use of FatiguePro and the SSES Fatigue Monitoring Program ensure that the fatigue of RCPB components is maintained below the ASME Code design limits.

All Class 1 valves are required to have a fatigue analysis. A review of a representative sample of Class 1 valve stress reports found the fatigue analyses to be conservatively simplistic, and the predicted fatigue was extremely low (less than 0.1). The simplified analyses for the valves do not provide the detailed information required to track fatigue usage by cycle counting or similar means. As an alternative, since the fatigue usage is typically much higher on the associated piping systems, and fatigue monitoring is performed for the limiting piping locations, the fatigue usage on the Class 1 valves is assumed to be bounded by the Class 1 piping locations. The fatigue on the valves will be managed indirectly by monitoring fatigue on the piping. If a piping system accumulates sufficient fatigue usage to indicate that design values are being approached, the Fatigue Monitoring Program will require a review of the valve fatigue analyses and other fatigue-related TLAA (such as flued head analyses and high energy line break evaluations) to determine if additional actions are required to address any of these additional fatigue-related concerns on the affected piping system.

Metal fatigue for all Class 1 reactor coolant pressure boundary piping and in-line components is managed by the SSES Fatigue Monitoring Program. This program includes requirements for continued monitoring and periodic updates to current and projected CUFs for the limiting piping locations. The program will be enhanced to include an approach to address CUFs that will exceed the allowable before the end of the period of extended operation. The aging management approach will include one or more of the following, which is similar to the approach documented in ASME Code Section III Non-mandatory Appendix L:

a) Further refinement of the fatigue analyses to lower the CUFs to less than the allowable

b) Repair of the affected components

c) Replacement of the affected components

d) Management by an inspection program that has been reviewed and approved by the NRC (e.g., periodic non-destructive examination of the affected locations at intervals determined by a method accepted by the NRC)

The Fatigue Monitoring Program is credited for managing the effects of fatigue during the period of extended operation. Therefore, the TLAA associated with fatigue of the reactor coolant pressure boundary piping and components have been dispositioned in accordance with 10 CFR 54.21(c)(1)(iii).

3.14.3.3 Non-Class 1 Component Fatigue Analyses

Calculation of cumulative fatigue usage, i.e., CUFs, is not required for non-Class 1 components designed in compliance with the codes and standards for non-Class 1 components. For non-Class 1 components stresses due to thermal expansion and anchor movement, which are important for fatigue evaluations, are analyzed using stress intensification factors and stress allowables. Allowable stresses are defined for 7000 full temperature cycles with reductions in allowable stresses as cycles increase beyond 7000. In addition, temperature thresholds above which fatigue should be considered for carbon steel and austenitic stainless steel are established.

The fatigue evaluation of non-Class 1 components determined whether the associated operating temperature exceeded threshold values for the affected materials and, if so, evaluated the number of transient cycles expected. In every case, the number of projected cycles for 60 years was found to be less than 7000 for piping and in-line components whose temperatures exceed threshold values. Therefore, fatigue for non-Class 1 piping and in-line components remains valid for the period of extended operation.

None of the non-Class 1 vessels, heat exchangers, storage tanks, or pumps were designed to ASME Section VIII, Division 2 or ASME Section III, Subsection NC-3200. Therefore, there is no fatigue TLAA for these components.

The TLAA associated with the fatigue of non-Class 1 components have been dispositioned in accordance with 10 CFR 54.21(c)(1)(i).

3.14.3.4 Environmental Qualification of Electric Equipment (NUREG 1931, Section 3.0.3.1.28)

Environmental Qualification analyses for those components with a qualified life of 40 years or greater are identified as TLAA for SSES. NRC regulation 10 CFR 50.49, “Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants” requires licensees to identify electrical equipment covered under this regulation and to maintain a qualification file demonstrating that the equipment is qualified for its application and will perform its safety function up to the end of its qualified life.

10 CFR 50.49 requires EQ components that are not qualified for the current license term to be refurbished, replaced, or have their qualifications extended prior to reaching the aging limits established in the aging evaluation. Reanalysis of aging evaluations to extend the qualifications of components is performed on a routine basis as part of the EQ Program. Important attributes for the reanalysis of aging evaluations include analytical methods, data collection and reduction methods, underlying assumptions, acceptance criteria, corrective actions (if acceptance criteria are not met), and the time remaining to the end of qualified life.

The SSES EQ Program is an existing program that implements the requirements of 10 CFR 50.49 and will be used to manage the effects of aging on the intended function(s) of the components associated with EQ TLAA for the period of extended operation.

3.14.3.5 Fatigue of Primary Containment, Attached Piping, and Components

3.14.3.5.1 ASME Class MC Components

FSAR Section 3.8.2.3.2.4 states the design thermal cycles for containment ASME Class MC stainless steel components, which includes the containment penetrations, hatches, and drywell head, to be 500 cycles for plant startup and shutdown and one cycle for a design basis accident. The reactor pressure vessel assembly and internal components are designed for 117 startups and 111 shutdowns for a combined total of 228 events. The maximum projected cycles for extended life to 60 years includes 148 startups and 148 shutdowns for a total of 296 events. Therefore, the Class MC component design value of 500 cycles for startups and shutdowns remains well above the projected value. Also, the one cycle allowed for a design basis accident is a value assumed in the design for a faulted condition for the life of the plant, whether that is 40 years or 60 years. Hence, the performance of these components will not be impacted by extending the life of the plant to 60 years.

The TLAA associated with thermal cycles on the ASME Class MC components have been dispositioned in accordance with 10 CFR 54.21(c)(1)(i).

3.14.3.5.2 Downcomer Vents and Safety Relief Valve Discharge Piping

Downcomer vents and safety relief valve (SRV) discharge piping penetrate the drywell / suppression pool diaphragm slab with the purpose of transporting steam and non-condensable gases to the suppression pool from the reactor and from the drywell during SRV lifts and under accident conditions. To ensure the integrity of the downcomers and SRV discharge piping for the original 40-year life of the plant, extensive analyses were performed. These analyses satisfy the definition for TLAA.

The significant area analyzed for the downcomers in the suppression pool air space was the downcomer penetration through the diaphragm slab. Structural analyses of all the SRV discharge lines from the diaphragm slab penetration to the quencher were performed, including flued head connections, elbows, and three-way restraint attachments.

The design rules, as set forth in the ASME Section III, Subsection NB were used for the fatigue assessment. The downcomers and SRV discharge lines were analyzed for the appropriate load combinations and their associated number of cycles. The combined stresses and corresponding equivalent stress cycles were computed to obtain the fatigue usage factors in accordance with the equations of Subsection NB-3600 of the ASME Code. The maximum cumulative usage factors for the downcomers and SRV discharge lines for the 40-year plant lifetime were determined from these analyses.

The minimum number of SRV actuations assumed in any of the fatigue analyses was 1100. The projected number of events for 60 years is less than the number assumed in the design basis (40 year) analysis. Therefore, the design basis analysis remains valid for the period of extended operation

The TLAA associated with stress cycles on the downcomer vents and safety relief valve discharge piping have been dispositioned in accordance with 10 CFR 54.21(c)(1)(i).

3.14.3.5.3 Safety Relief Valve Quenchers

Quenchers provide proper dispersion of reactor steam into the suppression pool upon lifts of SRVs and discharge of the steam through the SRV discharge piping.

Analyses for fatigue of the quenchers satisfy 10 CFR 54.3 criteria as TLAA. Fatigue evaluations for the original 40-year life of the plant list 7000 cycles as the expected number of cycles for each quencher component analyzed. The evaluations calculate the number of allowable cycles for the components and give the expected CUF for each analysis.

Since a quencher can experience up to seven cycles each time its associated SRV actuates (lifts), the worst case number of cycles is seven times the number of actuations projected for 40 years and for 60 years. These projected cycles were compared with analysis data results.

The design cycles exceed the number of cycles projected to 60 years for all components which were analyzed for the quencher. Therefore, the CUFs calculated in the fatigue evaluation remain valid for the period of extended operation.
The TLAA associated with stress cycles on the safety relief valve quenchers have been dispositioned in accordance with 10 CFR 54.21(c)(1)(i).

3.14.3.6 Other Plant-Specific Time-Limited Aging Analyses

3.14.3.6.1 Main Steam Flow Restrictor Erosion Analyses

A flow restrictor is incorporated in each main steam line to limit flow to 200 percent of rated flow in the event that a main steam line ruptures outside containment. Erosion of a flow restrictor is a safety concern since it could impair the ability of the flow restrictor to limit vessel blowdown following a main steam line break.

FSAR Section 5.4.4.4 discusses an evaluation of the effect of potential erosion of main steam line flow restrictors on radiological dose resulting from a main steam line break accident.

Operating for another 20 years will allow further erosion and, therefore, increased opening in the throat of the flow restrictors. The erosion can be linearly extrapolated from 40 to 60 years. This is conservative since the rate of erosion would be expected to decrease as the throat area of the restrictor increased due to erosion. (It has been determined that operation at Extended Power Uprate conditions will not significantly affect the erosion rate.) Therefore, it can be concluded that erosion for the 20 years of extended operation will be no more than half the erosion for the first 40 years, and the corresponding increase in steam flow will be no more than half of the increase in steam flow due to erosion at the end of 40 years (namely, 5 percent). This means that by the end of 60 years, the increase in flow compared to flow at the beginning of life will be no more than 7.8 percent. Therefore, the released dose for the accident case at 60 years would be no more than a 7.8 percent increase. Such an increase in dose over the analyzed case remains within regulatory limits, as indicated in FSAR Section 15.6.4.5.3.

Hence, the performance of the main steam line flow restrictors is not significantly impacted by the additional erosion during the period of extended operation.

3.14.3.6.2 High Energy Line Break Cumulative Fatigue Usage Factors

High energy line breaks have been postulated and analyzed for potential effects on surrounding equipment and systems. FSAR Section 3.6 provides criteria for determining break locations and types of breaks that could occur, descriptions of analysis methodologies, and results for significant attached piping showing where breaks could develop and where restraints were to be installed. Cumulative fatigue usage factors (CUFs) for the high energy lines are included in the criteria to determine postulated breaks. The CUFs, as calculated in the design fatigue analyses, account for the design transients assumed for the original 40-year life of the plant.

The postulated breaks are in piping for systems important to safety and integrity of the reactor coolant pressure boundary. The restraints designed for these potential breaks are significant for protection of systems and equipment important to plant safety. Therefore, the CUF calculations used in the selection of postulated high energy line break locations are TLAA.

Since these breaks are postulated to occur only once in the lifetime of the plants and restraints were installed appropriately to mitigate these potential breaks, the results of analyses for the potential breaks and the restraints installed in the plants remain unchanged for the extended life of 60 years. However, it is possible that other locations that had 40-year CUFs below the criteria for postulated breaks, could exceed that CUF criteria in 60 years. The possibility of these additional postulated breaks will need to be managed based on the actual fatigue accumulation encountered as the plant ages.

Presently, SSES utilizes the EPRI FatiguePro software to monitor fatigue at the critical bounding locations of piping systems in the plant. The SSES Fatigue Monitoring Program will identify when piping systems are approaching their original 40-year design CUFs. Prior to any piping system exceeding its' original maximum design CUF, the pertinent design calculations for the affected system will be reviewed to determine if any additional locations should be designated as postulated high energy line breaks, under the original criteria of FSAR Section 3.6. If other locations are determined to require consideration as postulated break locations, appropriate actions will be taken to address the new break locations.

The Fatigue Monitoring Program is credited for managing the effects of fatigue during the period of extended operation. Therefore, the TLAA associated with high energy line break cumulative fatigue have been dispositioned in accordance with10CFR54.21(c)(1)(iii).

3.14.3.6.3 Core Plate Rim Hold-Down Bolts

The NRC safety evaluation report that references BWRVIP-25, "BWR Core Plate Inspection and Flaw Evaluation Guidelines," for license renewal identifies loss of preload on the core plate rim hold-down bolts as one of the TLAA that must be addressed by applicants seeking license renewal.

PPL will address the loss of preload on the core plate rim hold-down bolts by one of the following two actions:

a) PPL will perform a SSES plant-specific evaluation consistent with BWRVIP-25 to demonstrate that the core plate rim hold-down bolts will be capable of preventing lateral displacement of the core plate for the period of extended operation. The evaluation will determine the maximum expected reduction in the bolt preload at the end of the period of

extended operation, considering all applicable parameters (i.e., operating temperature, operating loads, and irradiation effects) and demonstrate the acceptability of the final preload at the end of the period of extended operation. Using the methodology of BWRVIP-25 Appendix A, the evaluation will also determine the primary membrane and bending stresses for the limiting bolt(s) to demonstrate that ASME Code allowables are not exceeded as a result of the reduction in bolt preload at the end of the period of extended operation. The evaluation will also provide either a) justification for not inspecting the core plate hold-down bolts, or b) an inspection strategy to ensure an adequate number of bolts are intact to prevent lateral displacement of the core plate. The evaluation will be submitted to the NRC for review no less than two years prior to the period of extended operation.

b) PPL will install core plate wedges to structurally replace the lateral load resistance provided by the hold-down bolts. With wedges installed, any loss of preload on the core plate rim hold-down bolts during the period of extended operation will have no effect on the lateral stability of the core plate. The wedges will be installed prior to entering the period of extended operation.

If the evaluation described as Action 1 above is unable to demonstrate acceptable bolt preload or bolt stress values at the end of the period of extended operation, appropriate corrective action will be taken prior to entering the period of extended operation. The installation of core plate wedges described as Action 2 above is considered an appropriate and acceptable corrective action.

3.14.3.6.4 Irradiation Assisted Stress Corrosion Cracking (IASCC)

Austenitic stainless steel reactor internal components exposed to a neutron fluence of greater than 5×10^{20} n/cm^2 (E > 1 MeV) are susceptible to irradiation assisted stress corrosion cracking (IASCC) in the BWR environment. Analyses were performed to determine neutron fluence for extended power uprate (EPU) conditions and for extended operation out to 54 effective full power years (EFPY). The projected fluence values are used to identify the components that exceed the threshold fluence for IASCC.

The following reactor internal components have been identified as being susceptible to IASCC for the period of extended operation for SSES Units 1 and 2:

a) Top Guide

b) Core Shroud

c) In-Core Flux Monitoring Dry Tubes

d) Core Plate

The components identified as being susceptible to IASCC require aging management to identify and address potential degradation (crack initiation and growth) prior to any loss of intended function.

All identified components have been evaluated for IASCC by the BWRVIP, as described in the inspection and evaluation guideline reports for each component: BWRVIP-26-A for the Top

Guide; BWRVIP-76 for the Core Shroud; BWRVIP-47-A for the In-Core Flux Monitoring Dry Tubes; and BWRVIP-25 for the Core Plate. The inspection and evaluation guidelines of the identified BWRVIP reports are implemented by the BWR Vessel Internals Program for SSES.

3.14.4 LICENSE RENEWAL COMMITMENT LIST

The listing of commitments identified for SSES license renewal is provided in Table 3.14-1. These commitments are tracked within PPL's regulatory commitment management program.

3.14.5 NEWLY IDENTIFIED ITEMS (10 CFR 54.37 (b))

After the renewed license is issued, the FSAR update required by 10 CFR 50.71(e) must include any systems, structures, and components newly identified that would have been subject to an aging management review or evaluation of time-limited aging analyses in accordance with 10 CFR 54.21. This FSAR update must describe how the effects of aging will be managed such that the intended function(s) in 10 CFR 54.4(b) will be effectively maintained during the period of extended operation. These newly identified items are listed in Table 3.14-2.

Table 3.14-1
SSES License Renewal Commitments

Item Number	Commitment	FSAR Description Location	Enhancement or Implementation Schedule
1. Inservice Inspection (ISI) Program	Existing program is credited.	3.14.2.23	Ongoing
2. BWR Water Chemistry Program	Existing program is credited.	3.14.2.11	Ongoing
3. Reactor Head Closure Studs Program	Existing program is credited.	3.14.2.40	Ongoing
4. BWR Vessel ID Attachment Welds Program	Existing program is credited.	3.14.2.9	Ongoing
5. BWR Feedwater Nozzle Program	Existing program is credited.	3.14.2.6	Ongoing
6. BWR CRD Return Line Nozzle Program	Existing program is credited. • PPL will implement weld overlay repairs in accordance with ASME Section XI and NRC-approved Code Cases. If no NRC-approved Code Case exists for the weld overlay, PPL will obtain NRC approval prior to implementing the repair in accordance with 10 CFR 50.55a.	3.14.2.5	Ongoing
7. BWR Stress Corrosion Cracking (SCC) Program	Existing program is credited.	3.14.2.8	Ongoing

Table 3.14-1

SSES License Renewal Commitments

Item Number	Commitment	FSAR Description Location	Enhancement or Implementation Schedule
8. BWR Penetrations Program	Existing program is credited.	3.14.2.7	Ongoing
9. BWR Vessel Internals Program	Existing program is credited. • PPL will continue to perform inspections on at least 10% of the top guide grid beam cells containing control rod drives/blades every twelve years during the period of extended operation. Inspections on at least 5% of the top guide locations will be performed within the first six years of each twelve year interval. The top guide locations to be inspected are those subject to neutron fluence levels that exceed the IASCC threshold of 5.0E+20 n/cm^2. The inspections will be performed using the enhanced visual inspection technique, EVT-1.	3.14.2.10	Ongoing
10. Thermal Aging and Neutron Embrittlement of Cast Austenitic Stainless Steel (CASS) Program	Program is new. The new program for SSES will be consistent with the program described in NUREG-1801 Section XI.M13, Thermal Aging and Neutron Embrittlement of Cast Austenitic Stainless Steel (CASS) Program. The SSES program will identify susceptible components, evaluate those components to determine their susceptibility to loss of fracture toughness, and examine those components that are evaluated to be susceptible.	3.14.2.48	Prior to the period of extended operation.
11. Flow-Accelerated Corrosion (FAC) Program	Existing program is credited.	3.14.2.20	Ongoing
12. Bolting Integrity Program	Existing program is credited with the following enhancement: • Include specific precautions against the use of sulfur (sulfide) containing compounds as a lubricant for bolted connections.	3.14.2.2	Prior to the period of extended operation.

Table 3.14-1

SSES License Renewal Commitments

Item Number	Commitment	FSAR Description Location	Enhancement or Implementation Schedule
13. Piping Corrosion Program	Existing program is credited with the following enhancements: • Include the Standby Gas Treatment System loop seals within the scope of the program. • Incorporate performance, documentation and trending of opportunistic visual inspections (during normal maintenance/repair activities) in addition to existing Piping Corrosion Program inspections.	3.14.2.38	Prior to the period of extended operation.
14. Crane Inspection Program	Existing program is credited.	3.14.2.17	Ongoing
15. Fire Protection Program	Existing program is credited.	3.14.2.18	Ongoing
16. Buried Piping Surveillance Program	Program is new. The scope of the Buried Piping Surveillance Program includes only the portions of the buried piping in the Residual Heat Removal Service Water (RHRSW) and Emergency Service Water (ESW) common return header known to have damaged coatings. The program is credited for managing loss of material due to crevice, general, and pitting corrosion and microbiologically influenced corrosion (MIC) for buried steel piping components with damaged coatings.	3.14.2.4	Prior to the period of extended operation.

Table 3.14-1

SSES License Renewal Commitments

Item Number	Commitment	FSAR Description Location	Enhancement or Implementation Schedule
17. Condensate and Refueling Water Storage Tanks Inspection	Program is a new one-time inspection. The scope of the Condensate and Refueling Water Storage Tanks Inspection includes the base (bottom surface and foundation pad interface) of the Condensate Storage Tanks (CSTs) and Refueling Water Storage Tank (RWST) that are in the scope of license renewal and included in the Condensate Storage and Transfer and the Refueling Water Storage and Transfer systems. An appropriate combination of volumetric (including thickness measurement) and visual examinations will be conducted, for a unit's CST (or RWST), to detect evidence of a loss of material due to crevice, general, or pitting corrosion or to confirm a lack thereof. Results will be applied to the other unit's tank(s) based on engineering evaluation.	3.14.2.14	Within the 10-year period prior to the period of extended operation.
18. Reactor Vessel Surveillance Program	Existing program is credited with the following enhancement: • Address the additional requirements specified in the NRC safety evaluation dated March 1, 2006, for BWRVIP-116. The program will include a requirement that, if a standby capsule is removed from either of the SSES Unit 1 or Unit 2 reactor vessels without the intent to test it, the capsule will be stored in a manner which maintains it in a condition which would permit its future use, including during the period of extended operation if necessary.	3.14.2.41	Prior to the period of extended operation.

Table 3.14-1

SSES License Renewal Commitments

Item Number	Commitment	FSAR Description Location	Enhancement or Implementation Schedule
19. Chemistry Program Effectiveness Inspection	Program is a new one-time inspection. The Chemistry Program Effectiveness Inspection includes the internal surfaces of aluminum, copper and copper alloy, carbon and low alloy steel, cast iron, stainless steel, and nickel alloy components in systems that contain treated water or fuel oil. A representative sample of components in low flow and stagnant areas (i.e., locations that are isolated from the flow stream and possibly prone to gradual accumulation/concentration of contaminants) will be examined for evidence of a loss of material (due to crevice, galvanic, general, or pitting corrosion or to erosion, and to MIC in fuel oil), or to confirm a lack thereof, and the results applied to the rest of the system(s) based on engineering evaluation.	3.14.2.12	Within the 10-year period prior to the period of extended operation.
20. Cooling Units Inspection	Program is a new one-time inspection. The Cooling Units Inspection activities focus on a representative sample population of subject components at susceptible locations, to be defined in the implementing documents. These inspection activities provide symptomatic evidence of cracking, loss of material, or reduction in heat transfer at all other susceptible locations due to the similarities in materials and environmental conditions.	3.14.2.16	Within the 10-year period prior to the period of extended operation.
21. Heat Exchanger Inspection	Program is a new one-time inspection. The Heat Exchanger Inspection detects and characterizes conditions to determine whether, and to what extent a loss of heat transfer due to fouling is occurring (or likely to occur) for heat exchangers within the scope of license renewal. The Heat Exchanger Inspection is also credited for managing cracking due to stress corrosion cracking / inter-granular attack in the treated water (internal) environment of the admiralty brass tubes.	3.14.2.22	Within the 10-year period prior to the period of extended operation.
22. Not Used		3.14.2.30	

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Table 3.14-1

SSES License Renewal Commitments

Item Number	Commitment	FSAR Description Location	Enhancement or Implementation Schedule
23. Monitoring and Collection System Inspection	Program is a new one-time inspection. The scope of the Monitoring and Collection System Inspection includes the internal surfaces of subject carbon steel (and low alloy steel) and cast iron piping and valve bodies that are exposed to potentially radioactive drainage water (untreated water) and potentially other contaminants/fluids during normal plant operations. A representative sample of components in the system, to be defined in the implementing documents, and to include containment isolation piping and/or valve bodies, will be examined for evidence of a loss of material (due to crevice, general, or pitting corrosion or to MIC), or to confirm a lack thereof, and the results applied to the rest of the system based on engineering evaluation.	3.14.2.33	Within the 10-year period prior to the period of extended operation.
24. Supplemental Piping/Tank Inspection	Program is a new one-time inspection. The Supplemental Piping/Tank Inspection is credited for managing loss of material due to crevice and pitting corrosion on carbon steel surfaces at air-water interfaces. The inspection is also credited for managing loss of material due to microbiologically influenced corrosion (MIC) at the air-water interface with the mist eliminator loop seal, which is filled with raw water from the Service Water System, and galvanic corrosion at points of contact between the mist eliminator housing and the SGTS filter enclosure, where condensation and water pooling may occur. Additionally, the Supplemental Piping/Tank Inspection detects and characterizes whether, and to what extent, a loss of material due to crevice and pitting corrosion is occurring (or is likely to occur) for stainless steel surfaces at air-water interfaces. The Supplemental Piping/Tank Inspection also detects and characterizes loss of material due to crevice, galvanic, general, and pitting corrosion on internal carbon steel surfaces within the scram discharge volume (piping and valve bodies) of the Control Rod Drive Hydraulic System, within the air space of the condensate storage tanks and within the Diesel Generator	3.14.2.46	Within the 10-year period prior to the period of extended operation.

Table 3.14-1

SSES License Renewal Commitments

Item Number	Commitment	FSAR Description Location	Enhancement or Implementation Schedule
	starting air receiver tanks and E diesel compressor skid air receiver tanks to determine whether, and to what extent, degradation is occurring (or is likely to occur). In addition, the Supplemental Piping/Tank Inspection is credited to detect and characterize loss of material due to general, crevice, and pitting corrosion on the internal surfaces of carbon steel and cast iron diesel exhaust piping, piping components, and turbocharger casings. The inspection is also credited to detect and characterize cracking and loss of material due to crevice and pitting corrosion on the internal surfaces of stainless steel diesel exhaust piping components.		
25. Selective Leaching Inspection	Program is a new one-time inspection. The Selective Leaching Inspection detects and characterizes conditions to determine whether, and to what extent a loss of material due to selective leaching is occurring (or likely to occur) for susceptible components including piping and tubing, valve bodies, pump and turbocharger casings, heat exchanger, cooler, and chiller components, hydrants, sprinkler heads, strainers, level gauges, orifices, and heater sheaths. The components within the scope of the program are formed of cast iron, brass, bronze, and copper alloy materials. The components are subject to raw water, treated water, groundwater (buried), indoor air with condensation, outdoor air, and fuel oil environments. The components within the scope of this program are located in twenty-six different plant systems.	3.14.2.43	Within the 10-year period prior to the period of extended operation.

Table 3.14-1

SSES License Renewal Commitments

Item Number	Commitment	FSAR Description Location	Enhancement or Implementation Schedule
26. Buried Piping and Tanks Inspection Program	Program is new. The scope of the Buried Piping and Tanks Inspection Program includes buried components that are within the scope of license renewal for SSES. The program is credited for managing loss of material due to crevice, general, and pitting corrosion and microbiologically influenced corrosion (MIC) for buried steel piping components. In addition, the program is credited with managing loss of material for buried stainless steel piping components. The Buried Piping and Tanks Inspection Program is also credited for managing loss of material due to crevice, general, and pitting corrosion and microbiologically influenced corrosion (MIC) for buried steel tanks in the Diesel Fuel Oil System.	3.14.2.3	Prior to the period of extended operation.
27. Small-Bore Class 1 Piping Inspection	Program is a new one-time inspection. The SSES program will include measures to verify that cracking is not occurring in Class 1 small-bore piping, thereby validating the effectiveness of the Chemistry Program to mitigate cracking and confirming that no additional aging management programs are needed for the period of extended operation.	3.14.2.44	Within the 10-year period prior to the period of extended operation.

Table 3.14-1

SSES License Renewal Commitments

Item Number	Commitment	FSAR Description Location	Enhancement or Implementation Schedule
28. System Walkdown Program	Existing program is credited with the following enhancements: • The governing procedure for the System Walkdown Program must be revised to add the listing of systems crediting the program for license renewal and to explicitly include inspection of other metals, copper alloy and stainless steel. ○ It may be determined by engineering evaluation that these components do not require monitoring every two weeks, and the basis for a different walkdown frequency must be documented on the appropriate procedure form. • The governing procedure for the System Walkdown Program must be enhanced to address the license renewal requirement for opportunistic inspections of normally inaccessible components (e.g., those that are insulated), and those that are accessible only during refueling outages. For underground vaults/pits/manholes, an initial sample of at least one vault/pit/manhole from each grouping of components with identical material and environment combinations will be inspected prior to entering the period of extended operation. A representative sample of the entire population will be inspected within the first 6 years of the period of extended operation. Results of the inspection activities that require further engineering evaluation/resolution (e.g., sample expansion and inspection frequency changes if degradation is detected), if any, will be evaluated using the SSES corrective action process. • The governing procedure for the System Walkdown Program will be enhanced to include a visual and ultrasonic inspection of the external surfaces of piping passing into structures through penetrations (underground piping) for those penetrations with a history of leakage. These inspections will be focused on penetrations that are leaking at that time and will include a representative population of each material, environment	3.14.2.47	Prior to the period of extended operation.

Table 3.14-1

SSES License Renewal Commitments

Item Number	Commitment	FSAR Description Location	Enhancement or Implementation Schedule
	combination from those piping systems within the scope of license renewal (which includes those for the RHRSW, ESW, and Fire Protection systems) that enter structures below grade. • A routine activity to supplement the existing plant program will be generated, and based at least in part on EPRI 1007933, “Aging Assessment Field Guide,” to inspect elastomers and polymers for cracking and/or change in material properties. ○ Evidence of surface degradation, such as cracking or discoloration, as well as physical manipulation and/or prodding, will be used as a measure of the material condition. ○ A representative sample will be determined by engineering evaluation with a focus on components considered to be most susceptible to aging, such as due to their time in service, the severity of conditions during normal plant operations, and any pertinent design margins.		
29. Inservice Inspection (ISI) Program – IWE	Existing program is credited.	3.14.2.24	Ongoing
30. Inservice Inspection (ISI) Program – IWF	Existing program is credited.	3.14.2.25	Ongoing
31. Inservice Inspection (ISI) Program - IWL	Existing program is credited.	3.14.2.26	Ongoing

Table 3.14-1

SSES License Renewal Commitments

Item Number	Commitment	FSAR Description Location	Enhancement or Implementation Schedule
32. Containment Leakage Rate Test Program	Existing program is credited.	3.14.2.15	Ongoing
33. Masonry Wall Program	Existing program is credited with the following enhancement: • Specify that for each masonry wall, the extent of observed masonry cracking and/or degradation of steel edge supports/bracing is evaluated to ensure that the current evaluation basis is still valid. Corrective action is required if the extent of masonry cracking and steel degradation is sufficient to invalidate the evaluation basis.	3.14.2.31	Prior to the period of extended operation.
34. Structures Monitoring Program	Existing program is credited with the following enhancements: • Include additional structures requiring aging management for license renewal to the scope of the inspections. • Specify that if a below grade structural wall or structural component becomes accessible through excavation; a follow-up action is initiated for the responsible engineer to inspect the exposed surfaces for age-related degradation. • Clarify “structural component” for inspection includes each of the component types identified as requiring aging management. • Include degradation mechanisms for elastomer and earthen embankment inspection. • Include RG 1.127 inspection elements for water-control structure. • Specify that the responsible engineer shall review site groundwater and raw water pH, chlorides, and sulfates results prior to inspection to validate that the below-grade or raw water	3.14.2.45	Prior to the period of extended operation.

Table 3.14-1

SSES License Renewal Commitments

Item Number	Commitment	FSAR Description Location	Enhancement or Implementation Schedule
	environment remains non-aggressive during the period of extended operation. • Specify that for each masonry wall, the extent of observed masonry cracking and/or degradation of steel edge supports/bracing is evaluated to ensure that the current evaluation basis is still valid. Corrective action is required if the extent of masonry cracking and steel degradation is sufficient to invalidate the evaluation basis. • Include additional direction for quantifying, monitoring and trending of inspection results; Include additional guidance for inspection reporting, data collection and documentation; Specify acceptance criteria and critical parameters to monitor degradation and to trigger level of inspection and initiation of corrective action; and provide better alignment with referenced Industry codes, standards and guidelines. • Include specific qualification requirements for the inspector.		

Table 3.14-1

SSES License Renewal Commitments

Item Number	Commitment	FSAR Description Location	Enhancement or Implementation Schedule
35. RG 1.127 Water-Control Structures Inspection	Existing program is credited with the following enhancements: • Add the Spray Pond (including concrete liner, emergency spillway, riser encasements and earthen embankments) to its scope for inspection. • Include RG 1.127 Revision 1 Section C.2 inspection elements and degradation mechanisms for water-control structure inspection. • Include acceptance criteria as delineated in NUREG-1801 Section XI.S7 for water-control structures. Evaluation criteria provided in Chapter 5 of ACI 349.3R-96 provides acceptance criteria (including quantitative criteria) for determining the adequacy of observed aging effects and specifies criteria for further evaluation.	3.14.2.42	Prior to the period of extended operation.
36. Non-EQ Electrical Cables and Connections Visual Inspection Program	Program is new. The Non-EQ Electrical Cables and Connections Visual Inspection Program is credited with detecting aging effects from adverse localized environments in non-EQ cables and connections at SSES. The program is applicable to non-EQ cables and connections found in the Reactor Buildings, Circulating Water Pumphouse and Water Treatment Building, Control Structure, Diesel Generator Buildings, Turbine Building, Engineered Safeguards Service Water Pumphouse, and various yard structures (manholes, duct banks, valve vaults, instrument pits, etc.). This program is also applicable to the cables and connections within the scope of license renewal located in the yard areas and control cubicles of the T10 230 kV Switchyard, the 500 kV Switchyard, and the 230 kV Switchyard.	3.14.2.35	Prior to the period of extended operation.

Table 3.14-1

SSES License Renewal Commitments

Item Number	Commitment	FSAR Description Location	Enhancement or Implementation Schedule
37. Non-EQ Cables and Connections Used in Low-Current Instrumentation Circuits Program	Program is new. The Non-EQ Cables and Connections Used in Low-Current Instrumentation Circuits Program is credited with identifying aging effects for sensitive, high-voltage, low-current signal applications that are in-scope for license renewal at SSES. These sensitive circuits are potentially subject to reduction in insulation resistance (IR) when found in adverse localized environments.	3.14.2.34	Prior to the period of extended operation.
38. Non-EQ Inaccessible Medium-Voltage Cables Program	Program is new. The Non-EQ Inaccessible Medium-Voltage Cables Program involves two parts: first, the actions to inspect the applicable plant manholes (and to drain them, if necessary) on a periodic basis; and second, the development of a testing program to confirm that the conductor insulation on the applicable cables is not degrading. This program applies to six cables associated with the offsite power supply for SSES. These are the only inaccessible medium-voltage cables at SSES that are within the scope of license renewal and are exposed to significant moisture simultaneously with significant voltage.	3.14.2.36	Prior to the period of extended operation.
39. Metal-Enclosed Bus Inspection Program	Program is new. The Metal-Enclosed Bus Inspection Program is credited with detecting aging effects for in-scope metal-enclosed bus at SSES. The applicable components for the metal-enclosed bus will be listed in the program implementing document(s), with their locations specified, as appropriate. The in-scope bus is limited to non-segregated metal-enclosed bus in the 13.8 kV and 4 kV electrical systems associated with the off-site power supply at SSES.	3.14.2.32	Prior to the period of extended operation.

Table 3.14-1

SSES License Renewal Commitments

Item Number	Commitment	FSAR Description Location	Enhancement or Implementation Schedule
40. Area-Based NSAS Inspection	Program is a new one-time inspection. The Area-Based NSAS Inspection includes confirming the environmental and/or internal surfaces conditions of subject nonsafety-related carbon steel (includes low alloy steel), cast iron, copper alloy and stainless steel components in systems that (frequently or continuously during normal plant operations) contain raw water, potable water, non-radioactive equipment/area drainage water, or in some select cases, treated water. The program is plant-specific.	3.14.2.1	Within the 10-year period prior to the period of extended operation.
41. Leak Chase Channel Monitoring Activities	The existing program is credited. The program is plant-specific.	3.14.2.27	Ongoing
42. Preventive Maintenance Activities – RCIC/HPCI Turbine Casings	Existing program is credited with the following enhancements: • Include a specific step to perform a visual inspection of the RCIC turbine casing. • Add requirements to have inspections performed by qualified personnel using VT-3 or equivalent inspection methods, and to document and trend inspection results. • Establish specific acceptance criteria for inspection results. The program is plant-specific.	3.14.2.39	Prior to the period of extended operation.
43. Fatigue Monitoring Program	Existing program is credited with the following enhancements: • Provisions will be made in the Fatigue Monitoring Program to validate that components which have satisfied ASME Section III, Paragraph N-415.1 requirements (i.e., RPV nozzles N6A, N6B, and N7) continue to satisfy these requirements prior to and during the period of extended operation, thereby allowing fatigue to be	3.14.2.49	Prior to the period of extended operation.

Table 3.14-1

SSES License Renewal Commitments

Item Number	Commitment	FSAR Description Location	Enhancement or Implementation Schedule
	continued to be addressed under N-415.1. • The Fatigue Monitoring Program will be enhanced to ensure that the fatigue usage at all monitored locations, including those locations that account for the effect of the reactor water environment, is managed such that an adequate margin against fatigue cracking is maintained. PPL will implement one or more of the following actions, if fatigue usage at a monitored location, including any location that accounts for the effect of the reactor water environment, is projected to reach the design basis limit prior to the end of the period of extended operation: 1. Further refinement of the fatigue analyses to lower the CUFs to less than the allowable; 2. Repair of the affected components; 3. Replacement of the affected components; 4. Management by an inspection program that has been reviewed and approved by the NRC. • The Fatigue Monitoring Program will be enhanced to include the review of Class 1 valve fatigue analyses and other fatigue-related TLAA, such as flued head analyses and high energy line break evaluations, when sufficient fatigue accumulation has occurred, to determine if additional actions are required to address fatigue-related concerns. • The Fatigue Monitoring Program will be enhanced to include fatigue monitoring of the additional locations required to bound the limiting locations applicable to SSES, as identified in NUREG/CR-6260. • The Fatigue Monitoring Program will be enhanced to establish		

Table 3.14-1

SSES License Renewal Commitments

Item Number	Commitment	FSAR Description Location	Enhancement or Implementation Schedule
	monitoring criteria to ensure that the fatigue usage at all monitored locations, including those locations that account for the effect of the reactor water environment, is managed such that design basis limits are not exceeded during the period of extended operation. The Fatigue Monitoring Program will define specific fatigue usage values for all monitored locations that, if reached, will require further action. These fatigue usage values shall be conservatively set to values that will allow for not less than 4 years of additional plant operation before the actual fatigue usage at any location would reach the design basis limit. Upon reaching the defined usage at a location, the Fatigue Monitoring Program will require an action request to be generated. The action request will require further engineering evaluation to resolve the issue.		
44. Environmental Qualification (EQ) Program	Existing program is credited. For those EQ components that do not show a minimum 60-year life, the EQ Program will ensure qualified life is not exceeded by directing refurbishment, replacement, or reanalysis to extend the qualification.	3.14.3.4	Ongoing
45. Closed Cooling Water Chemistry Program	Existing program is credited.	3.14.2.13	Ongoing.

Table 3.14-1

SSES License Renewal Commitments

Item Number	Commitment	FSAR Description Location	Enhancement or Implementation Schedule
46. Fire Water System Program	Existing program is credited with the following enhancements: • The Fire Water System Program will be revised to incorporate sprinkler head sampling/replacements, in accordance with NFPA 25. • The Fire Water System Program will be revised to incorporate ultrasonic testing of representative above ground portions of water suppression piping that are exposed to water but which do not normally experience flow, are associated with a dry-pipe sprinkler system and may contain stagnant water, or is pre-action or deluge piping that is normally dry but may have been wetted and not completely dried. • Perform at least one visual inspection (opportunistic or focused) of the internal surface of buried fire water piping, within the 10 year period prior to the period of extended operation. • Perform at least one inspection per year of 'wet' fire protection piping for wall thickness and pipe blockage, if no opportunistic inspection has been completed.	3.14.2.19	Prior to the period of extended operation.
47. Fuel Oil Chemistry Program	Existing program is credited.	3.14.2.21	Ongoing

Table 3.14-1

SSES License Renewal Commitments

Item Number	Commitment	FSAR Description Location	Enhancement or Implementation Schedule
48. Lubricating Oil Analysis Program	Existing program is credited with the following enhancements: • The Lubricating Oil Analysis Program will be enhanced to include sampling of the lubricating oil from the Control Structure Chiller and Diesel Engine Driven Fire Pump when the oil is changed. The oil will be tested for water and for particle count. • The Lubricating Oil analysis Program will be revised to include sampling of the lubricating oil from the Reactor Building Chiller when the oil is changed. The oil will be tested for water and particle count.	3.14.2.28	Prior to the period of extended operation.
49. Lubricating Oil Inspection	Program is a new one-time inspection. The Lubricating Oil Inspection detects and characterizes the condition of materials in systems and components for which the Lubricating Oil Analysis Program is credited with aging management. The inspection provides direct evidence as to whether, and to what extent, a loss of material or a reduction in heat transfer due to fouling has occurred.	3.14.2.29	Within the 10-year period prior to the period of extended operation.
50. Non-EQ Electrical Cable Connections Program	Program is new. The Non-EQ Electrical Cable Connections Program manages the aging for the metallic parts of non-EQ electrical cable connections within the scope of license renewal. The program addresses cable connections that are used to connect cable conductors to other cables or electrical devices. Aging management for the metallic parts of the non-EQ electrical cable connections that are subject to aging stressors will be provided by testing.	3.14.2.37	Prior to the period of extended operation.
51. New P-T Curves	Revised Pressure-Temperature (P-T) limits will be submitted to the NRC for approval when necessary to comply with 10 CFR 50 Appendix G.	3.14.3.1.4	Ongoing
52. OE Review at	Perform an Operating Experience (OE) review for the period of operation at EPU conditions and its impact on aging management	---------	Prior to the period of extended

Table 3.14-1

SSES License Renewal Commitments

Item Number	Commitment	FSAR Description Location	Enhancement or Implementation Schedule
EPU Conditions	programs for systems, structures and components (SSCs).		operation.
53. Incorporate FSAR Supplement	Incorporate FSAR Supplement into the SSES FSAR as required by 10 CFR 54.21(d).	3.14.1	Following issuance of the renewed operating licenses.
54. Re-apply for relief request	PPL will process a relief request for circumferential vessel shell weld volumetric examinations for the period of extended operation.	3.14.3.1.5	Prior to the period of extended operation.
55. Core Plate Hold Down Bolts	PPL will either: (1) obtain NRC approval of a SSES plant-specific evaluation consistent with BWRVIP-25 to demonstrate that the core plate rim hold-down bolts will be capable of preventing lateral displacement of the core plate for the period of extended operation (the plant-specific evaluation will be submitted for NRC review no less than 2 years prior to the period of extended operation and will address the inspection strategy for the hold-down bolts); or (2) install core plate wedges to structurally replace lateral load resistance provided by the bolts.	3.14.3.6.3	Prior to the period of extended operation.
56. BWRVIP-76	PPL will address any future conditions, requirements, or limitations imposed by the NRC's safety evaluation for license renewal for BWRVIP-76.	3.14.3.6.4	Prior to the period of extended operation.
57. Preventative Maintenance Activities-Main Turbine Casing	Existing program is credited with the following enhancement: • Specify that the inspection of the high pressure turbine shell will consist of a visual inspection (VT-3 or equivalent) of accessible surfaces and an ultrasonic examination of selected locations for wall thickness. The program is plant specific.	3.14.2.50	Prior to the period of extended operation.

Table 3.14-1

SSES License Renewal Commitments

Item Number	Commitment	FSAR Description Location	Enhancement or Implementation Schedule
58. Activities in Response to NRC Generic Letter 88-14	Activities credited in the SSES response to NRC Generic Letter 88-14 will be continued throughout the period of extended operation.	---------	Ongoing
59. Fuse Holders Program	Program is new. The Fuse Holders Program is credited with identifying increased connection resistance between the fuse holder metallic clamp and fuse due to fatigue of the metallic clamp. The program provides for periodic inspection of fuse holder clamps within the scope of license renewal that are not in enclosures containing active components and whose fuses are scheduled for removal once every 12 months, or more frequently.	3.14.2.51	Prior to the period of extended operation.
60. Activities in Response to NRC Concerns Regarding Fatigue Analyses	PPL will either (1) implement fatigue monitoring software that satisfactorily addresses all issues raised in Regulatory Information Summary (RIS) 2008-30, “Fatigue Analysis of Nuclear Power Plant Components”, or (2) perform a confirmatory ASME Code, Section III fatigue evaluation for the SBF-monitored locations to justify the existing FatiguePro methodology used at SSES Units 1 and 2.	-----------	Prior to the period of extended operation.
61. Boral Coupon Testing	Spent fuel pool Boral coupon testing will be continued in the period of extended operation with one set of coupons being tested during the tenth or eleventh year after Unit 1 enters the period of extended operation. SSES FSAR section 9.1.2.3.3 Inservice Inspection will be revised to identify the coupon testing schedule during the period of extended operation. SSES FSAR section 9.1.2.3.3.2 Test Coupon Inspection will be revised to require neutron attenuation testing as part of the inspection of test coupons removed from the spent fuel pool.	-----------	Revise the FSAR prior to the period of extended operation, with coupon testing ongoing as indicated.

Table 3.14-2

Newly Identified Structures, Systems, and Components (SSC)

No.	FSAR Revision No.	SSC & Description	Aging Management Review Conclusion	Aging Management Program
1	69	Backup Fire Protection System: • The original Backup Fire Protection System was installed in the plant at the time of the license renewal review, in accordance with the CLB at the time. The Backup Fire Protection System should have been included in the scope of license renewal per 10 CFR 54.4(a); however, the system was not identified as in scope until after issuance of the renewed license. The Backup Fire Protection System (both the original and 2017 modification) is credited to perform a function meeting the requirements of 10 CFR 50.48.† † The backup diesel fire pump, backup jockey pump and other components were replaced with a modified design after the renewed license was issued and are therefore not subject to the provisions of 10 CFR 54.37(b). As such, these SSCs are not in LR scope.	Components added to scope because of this change: • Pipe • Valves • Well Water Storage Tank Materials of Construction: • Carbon Steel • Cast Iron Environments: • Buried • Outdoor (uncontrolled) • Sheltered Aging effects for the material/environment combinations: • Loss of material in carbon steel/cast iron in Potable Water, Outdoor, and Buried environments	There are no changes required to any of the AMPs for fire protection as identified in LRA Table 3.3.2-13 for aging management of the newly identified SSCs. The material/environment combination is the same as the material/environment combination previously evaluated for the fire protection SSCs in the License Renewal Application, therefore, the aging effects and aging management programs previously credited are being used to manage the newly added backup fire protection system SSCs.