SECTION 3

DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT AND SYSTEMS

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SECTION 3

DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

3.1 CONFORMANCE WITH NRC GENERAL DESIGN CRITERIA

3.1.1 Summary Description

This section contains an evaluation of the design bases of the Hope Creek Generating Station as measured against the NRC GDC for Nuclear Power Plants, Appendix A of 10CFR50. For each of the 64 criteria, a specific assessment of the plant design has been made. In addition, a list of sections where further information pertinent to each criterion is included is also given.

Based on the content of this section, the applicant concludes that the Hope Creek Generating Station is in compliance with the GDC.

3.1.2 Criterion Conformance

3.1.2.1 Group I - Overall Requirements

3.1.2.1.1 Criterion 1 - Quality Standards and Records

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

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

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

3.1.2.1.1.1 Design Conformance to Criterion 1

Structures, systems, and components important to safety are designed, fabricated, erected, tested, and operated under a quality assurance (QA) program that satisfies the requirements of Appendix B of 10CFR50. Chapter 17 of the FSAR discusses the QA program during station operation, which is designed and implemented to ensure that HCGS is tested and operated in conformance with the regulatory requirements and design bases outlined in the license application.

Design requirements and other information regarding implementation of the QA program are described in various sections of the FSAR. Codes and standards that apply to safety-related, pressure retaining piping and equipment are discussed in Section 3.2. Building codes and standards are discussed in Section 3.8. Detailed seismic design is outlined in Section 3.7.

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

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

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

The QA program developed by the applicant and its contractors satisfies the requirements of Criterion 1.

For further information, see Section 1.2.

3.1.2.1.2 Criterion 2 - Design Bases for Protection Against Natural Phenomena

Structures, systems, and components important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. The design bases for these structures, systems, and components shall reflect:

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.

3. The importance of the safety functions to be performed.

3.1.2.1.2.1 Design Conformance to Criterion 2

The design basis for protection against natural phenomena is in accordance with GDC 2. Structures, systems, and components important to safety are designed to withstand the effects of natural

phenomena such as earthquakes, tornadoes, hurricanes and floods without loss of the capability to perform required safety functions, with 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, 3.4, 3.5, 3.7, and 3.8. Those combinations of natural phenomena and plant-originated accidents that are considered in the design are identified in Sections 3.6, 3.8, 3.9, 3.10, and 3.11.

3.1.2.1.3 Criterion 3 - Fire Protection

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

3.1.2.1.3.1 Design Conformance to Criterion 3

Structures, systems, and components important to safety are designed to minimize the probability and effects of fires and explosions. Noncombustible and heat resistant materials are used wherever practicable throughout the plant, particularly in the primary containment, main control room, and areas containing engineered safety features.

Appropriate equipment and facilities for fire protection, including the detection, alarm, and extinguishing of fires, are provided to

protect plant equipment and personnel from fire, explosions, and the resultant release of toxic vapors. Automatic and manual types of fire protection equipment are provided.

Two 100 percent capacity fire pumps provide an adequate supply of water to the hydrants, sprinkler systems, and hose stations located throughout the plant. Water is the primary firefighting agent; chemical systems are employed whenever the type of hazard does not favor the use of water spray. Portable fire extinguishers are provided throughout the plant. A detailed description of the Fire Protection System, its design bases, and fire hazards analysis is provided in Section 9.5.1.

Early warning of incipient fires is provided by a fire detection system using smoke detectors and/or heat responsive devices located in areas of the plant where significant fire potential exists.

The Fire Protection System is designed, fabricated, and installed in accordance with the requirements of the National Fire Protection Association (NFPA), American Nuclear Insurers (ANI), Nuclear Mutual Limited (NML), Occupational Safety and Health Act (OSHA), and applicable local codes and regulations listed in Section 9.5.1. Components used within these systems are Underwriters Laboratories listed and/or Factory Mutual approved, where available.

The Fire Protection System is inspected and tested prior to plant operation. The fire suppression systems are provided with test valves and facilities for periodic testing. All equipment is accessible for periodic inspection.

Although it can be postulated that failure or inadvertent operation of the Fire Suppression System may incapacitate some safety-related systems or components, such failure or inadvertent operation does not prevent safe shutdown from being achieved through the use of redundant safety-related systems.

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

Fire barriers are provided in areas where fire separation is required. These barriers are rated for 1, 2, or 3 hours, depending on location. All penetrations through fire barriers are sealed with a sealant consistent with the fire rating of the barrier.

3.1.2.1.4 Criterion 4 - Environmental and Missile Design Bases

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

3.1.2.1.4.1 Design Conformance to Criterion 4

Structures, systems and components important to safety are designed to accommodate the effects of, and to be compatible with, the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including LOCAs.

These structures, systems, and components are protected against dynamic effects and discharging fluids that may result from equipment failures. Normal and postulated accident effects and load combinations are given in Sections 3.6, 3.8, 3.9 and 3.10. Section 3.11 contains information on environmental conditions.

Special attention has been given to the effects of pipe movement, jet forces, and missiles within the primary containment. The structures, systems, and components important to safety are

protected from dynamic effects by separating redundant counterparts so that no single event can prevent a required safety action. The means used to preserve the independence of redundant counterparts of safety-related systems are discussed in Chapter 6.

The electrical equipment, instrumentation, and associated cables of protection and Engineered Safety Feature (ESF) Systems are discussed in the sections listed below.

Dynamic effects external to the plant, caused by natural phenomena, e.g., tornado produced missiles, are discussed in Section 3.5.

Environmental and missile design bases are in accordance with GDC 4.

For further discussion, see the following sections:

1. Meteorology - Section 2.3

2. Hydrology - Section 2.4

3. Geology and seismology - Section 2.5

4. Classification of structures, components, and systems - Section 3.2

5. Wind and tornado loadings - Section 3.3

6. Water level (flood) design - Section 3.4

7. Missile protection - Section 3.5

8. Protection against dynamic effects associated with a postulated rupture of piping - Section 3.6

9. Seismic design - Section 3.7

10. Design of Seismic Category I structures - Section 3.8

11. Mechanical systems and components - Section 3.9

12. Seismic qualification of Seismic Category I instrumentation and electrical equipment - Section 3.10

13. Environmental design of mechanical and electrical equipment - Section 3.11

14. Integrity of reactor coolant pressure boundary - Section 5.2.

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

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

3.1.2.1.5.1 Design Conformance to Criterion 5

There is no sharing of safety-related structures, systems, or components.

3.1.2.2 <u>Group II - Protection by Multiple Fission Product Barriers</u>

3.1.2.2.1 Criterion 10 - Reactor Design

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

3.1.2.2.1.1 Design Conformance to Criterion 10

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

An Oscillation Power Range Monitor (OPRM) subsystem is also installed. This subsystem provides detection and suppression of reactor core power oscillations which could result from core thermal-hydraulic instabilities. When operating at power and flow conditions where oscillations could occur, the OPRM initiates reactor trip signals to the RPS to prevent power oscillations from exceeding preset limits.

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 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 - Section 1.2

2. Plant description - Section 1.2

3. Fuel mechanical design - Section 4.2

4. Nuclear design - Section 4.3

5. Thermal and hydraulic design - Section 4.4

6. Reactor Recirculation System - Section 5.4

7. Reactor Core Isolation Cooling (RCIC) System - Section 5.4

8. Residual Heat Removal (RHR) System - Section 5.4

9. Accident analysis - Section 15.

3.1.2.2.2 Criterion 11 - Reactor Inherent Protection

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

3.1.2.2.2.1 Design Conformance to Criterion 11

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

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

1. Fuel temperature or Doppler coefficient

2. Moderator void coefficient

3. 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, contributing to system stability. Since the Doppler reactivity opposes load changes, it is desirable to maintain a large ratio of moderator void coefficient to Doppler coefficient for optimum load following capability. The boiling water reactor (BWR) has an inherently large moderator-to-Doppler coefficient ratio, which permits use of coolant flow rate for load following.

BWR nuclear design requires the moderator void coefficient inside the fuel channel to be negative during operation. The negative void reactivity coefficient provides an inherent negative feedback during power transients. Because of the large negative moderator coefficient of reactivity, the BWR has a number of inherent advantages, such as:

1. Use of the coolant flow, as opposed to control rods, for load following

2. Self flattening of the radial power distribution

3. Ease of control

4. Spatial xenon stability.

The reactor is designed so that the moderator temperature coefficient is small and positive and 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 -0.03 $(\Delta k/k)/(\Delta P/P)$ at 10,000 MWd/t. These values are well within

the range required for adequate damping of power and spatial xenon disturbances.

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

For further discussion, see the following sections:

1. Principal design criteria - Section 1.2

2. Nuclear design - Section 4.3

3. Thermal and hydraulic design - Section 4.4.

3.1.2.2.3 Criterion 12 - Suppression of Reactor Power Oscillations

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

3.1.2.2.3.1 Design Conformance to Criterion 12

The reactor core is designed to ensure that no power oscillation causes fuel design limits to be exceeded. The power reactivity coefficient is the composite simultaneous effect of the fuel temperature or Doppler coefficient, moderator void coefficient, and moderator temperature coefficient to the change in power level. It is negative and well within the range required for adequate damping of power and spatial xenon disturbances. Analytical studies indicate that for large BWRs, under damped, unacceptable power distribution behavior could only be expected to occur with power coefficients more positive than about $-0.01\ (\Delta k/k)/(\Delta P/P)$. Operating experience has shown large BWRs to be inherently stable

against xenon induced power instability. The large negative operating coefficients provide:

1. Good load following with well dampened behavior and little undershoot or overshoot in the heat transfer response

2. Load following with recirculation flow control

3. Strong damping of spatial power disturbances.

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

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

An Oscillation Power Range Monitor (OPRM) subsystem is also provided. This system detects power oscillations which can result from thermal-hydraulic reactor core instabilities, and provides alarms which alert the Control Room operator to their occurrence. The OPRM subsystem can also suppress these oscillations by providing trip signals to the Reactor Protection System (RPS) trip logic to shut down the reactor. The OPRM subsystem is described in Section 7.6.1.4.

For further discussion, see the following sections:

1. Principal design criteria - Section 1.2
2. Fuel mechanical design - Section 4.2
3. Nuclear design - Section 4.3
4. Thermal and hydraulic design - Section 4.4
5. Integrity of RCPB - Section 5.2
6. RPS - Section 7.2

7. All other instrumentation systems required for safety - Section 7.6

8. Accident analysis - Section 15.

3.1.2.2.4 Criterion 13 - Instrumentation and Control

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

3.1.2.2.4.1 Design Conformance to Criterion 13

The neutron flux in the reactor core is monitored by five subsystems. The source range monitor (SRM) subsystem measures the flux from startup through criticality. The intermediate range monitor (IRM) subsystem overlaps the SRM subsystem and extends well into the power range. The power range is monitored by many detectors that make up the local power range monitor (LPRM) subsystem. The output from these detectors is used in many ways. The output of selected, core wide sets of detectors is averaged to provide a core average neutron flux. This output is called the average power range monitor (APRM) subsystem. The traversing incore probe (TIP) subsystem provides a means for calibrating the LPRM subsystem. Both the IRM and APRM subsystems generate scram trips to the reactor trip system. All subsystems but the TIP subsystem generate rod block trips. Additional information on the neutron monitoring system is given in Section 7.

The Reactor Protection System protects the fuel barriers and the nuclear process barrier by monitoring plant parameters and causing a reactor scram when predetermined setpoints 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 RCPB, the primary containment and reactor vessel isolation control systems initiate 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 RCPB. Nuclear system leakage rates are classified as identified and unidentified, which corresponds respectively to the flow to the equipment drain and to the floor drain sumps. The permissible total leakage rate limit to these sumps is based upon the makeup capabilities of various reactor component systems. High pump fillup rate and pumpout rate are alarmed in the main control room. The unidentified leakage rate, as established in Section 5.2.5, is less than the value that has been conservatively calculated to be a minimum leakage from a crack large enough to propagate rapidly, but which still allows time for identification and corrective action before integrity of the process barrier is threatened.

The Process Radiation Monitoring System monitors radiation levels of various processes and provides trip signals to the reactor protection system, the primary containment and reactor vessel isolation control systems whenever pre-established limits are exceeded.

As noted above, adequate instrumentation has been provided to monitor system variables in the reactor core, RCPB and primary 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 an abnormal operational occurrence or an accident.

3.1.2.2.5 Criterion 14 - Reactor Coolant Pressure Boundary

The RCPB 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.

3.1.2.2.5.1 Design Conformance to Criterion 14

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. 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.

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. Section 5.2.3 describes the methods used to control toughness properties. Materials are impact-tested in accordance with ASME B&PV 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 joints. Welding procedures employed produce welds of complete fusion, 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 B&PV Code for the materials to be welded. Qualification records are maintained, including the results of procedure and performance qualification tests and identification symbols assigned to each welder.

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 the evaluation against Criterion 30 of the GDC.

The design, fabrication, erection, and testing of the RCPB ensure an extremely 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 - Section 1.2

2. Design of structures, components, equipment, and systems - Section 3

3. Overpressurization protection - Section 5.2

4. Reactor vessel and appurtenances - Section 5.3

5. Reactor Recirculation System - Section 5.4

6. Accident analysis - Section 15

7. QA program - Section 17.

3.1.2.2.6 Criterion 15 - Reactor Coolant System (RCS) Design

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

3.1.2.2.6.1 Design Conformance to Criterion 15

The RCS consists of the reactor vessel and appurtenances, the Reactor Recirculation System, the Pressure Relief System, the main steam lines, the RCIC system, and the RHR system. These systems are designed, fabricated, erected, and tested to stringent quality requirements and appropriate codes and standards that ensure high integrity of the RCPB throughout the plant lifetime. The RCS is designed and fabricated to the requirements indicated in Section 3.2.2.

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

The automatic initiation of the Pressure Relief System upon receipt of an overpressure condition is an example of the integrated protective action scheme that provides sufficient margin to ensure that the design conditions of the RCPB are not exceeded. To accomplish overpressure protection, a number of pressure operated relief valves are provided that can discharge steam from the nuclear system to the suppression pool. The 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 cooling water to adequately cool the core. In a similar manner, other auxiliary, control, and protection systems ensure that the design conditions of the RCPB are

not exceeded during any conditions of normal operation, including anticipated operational occurrences.

The application of appropriate codes, standards, and high quality requirements to the RCS 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 - Section 1.2

2. Design of structures, components, equipment, and systems - Section 3

3. Overpressurization protection - Section 5.2.2

4. RCPB Leakage Detection System - Section 5.2.5

5. Reactor vessel - Section 5.3

6. Reactor Recirculation System - Section 5.4

7. Accident analysis - Section 15.

3.1.2.2.7 Criterion 16 - Containment Design

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

3.1.2.2.7.1 Design Conformance to Criterion 16

The Primary Containment System, which includes the drywell and suppression chamber, is designed, fabricated, and erected to accommodate the pressures and temperatures resulting from double ended rupture or equivalent failure of any coolant pipe within the primary containment. The Reactor Building, encompassing the primary containment, provides secondary containment. To provide protection against the consequences of accidents involving the release of radioactive materials from the fuel and nuclear system process barrier, the primary containment and reactor vessel isolation control systems initiate automatic isolation of appropriate pipelines that penetrate the primary containment whenever monitored variables exceed preselected operational limits. The two containment systems and their associated safety systems are designed and maintained so that offsite doses, which would result from postulated design basis accidents (DBAs), remain below the guideline values stated in 10CFR50.67 when calculated by the methods of Regulatory Guide 1.183 (Revision 0, July 2000). The following referenced sections provide detailed information that demonstrates compliance with Criterion 16:

1. Containment systems - Section 6.2

2. Deleted

3. ESF systems - Section 7.3

4. Primary Containment Ventilation System - Section 9.4.5

5. Accident analysis - Section 15.

3.1.2.2.8 Criterion 17 - Electrical Power Systems

An Onsite Electric Power System and an Offsite Electric Power System shall be provided to permit functioning of structures, systems, and components important to safety. The safety function for each system, assuming the other system is not functioning, shall be to provide sufficient capacity and capability to ensure that:

1. Specified acceptable fuel design limits and design conditions of the RCPB are not exceeded as a result of anticipated operational occurrences.

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 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 supply and the other offsite electric power circuit to assure that specified acceptable fuel design limits and design conditions of the RCPB are not exceeded. One of these circuits shall be designed to be available within a few seconds following a LOCA to assure that the core cooling, containment integrity, and other vital safety functions are maintained.

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

3.1.2.2.8.1 Design Conformance to Criterion 17

The 500-kV Hope Creek switchyard is the offsite power source for the generating unit. There are two physically independent connections from the 500-kV switchyard to a 13.8-kV ring bus. Each connection supplies power to the 13.8-kV ring bus through two 500-kV/14.4-kV station power transformers. The physical and operating arrangement of the breakers in the 500-kV switchyard and 13.8-kV ring bus is such that it minimizes the possibility of a simultaneous failure of both the offsite power supplies.

The 13.8-kV ring bus is the preferred source of all auxiliary power during startup, normal operation, shutdown, and post-shutdown. In the event of total offsite power loss, four independent diesel generators provide the standby power for all Class 1E loads and some selected non-Class 1E loads that are important to the integrity of the power generating equipment.

Each of the ties between the 500-kV switchyard and the 13.8-kV ring bus is capable of supplying the total auxiliary power requirement of the plant under all operating conditions. In case of the unavailability of auxiliary power from the preferred source, three out of four standby diesel generators and their associated dc batteries are capable of supplying power for shutting down the unit and maintaining it in a safe shutdown condition.

There are four independent ac load group channels (A, B, C, and D) provided to ensure 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 load requirements to safely shut down the unit.

For each of the four ac load groups, there is an independent 125 V dc system that uses batteries upon failure of primary power from battery chargers. Two of the four load group channels have dedicated dc power provided from independent 250 V dc sources. These sources use batteries upon failure of preferred power.

The reactor protection instrumentation is powered from two independent ride through ac power sources.

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

For further information, see the following sections:

1. Offsite power systems - Section 8.2

2. Onsite ac power systems - Section 8.3.1

3. Onsite dc power systems - Section 8.3.2.

3.1.2.2.9 Criterion 18 - Inspection and Testing of Electric Power Systems

Electric power systems important to safety shall be designed to permit appropriate periodic inspection and testing of important areas and features, such as wiring, insulation, connections, and switchgear, in order to assess the continuity of the systems and the condition of their components. The systems shall be designed with a capability to periodically test:

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

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

3.1.2.2.9.1 Design Conformance to Criterion 18

The onsite power systems, consisting of the standby diesel generators, with their associated switchgear assemblies supplying power to safety-related equipment, and the associated battery/inverter systems, are designed and arranged for periodic testing of each system independently.

Each standby diesel generator can be full load tested while the plant is at power by manually starting each standby diesel generator (SDG) and manually synchronizing them to the normal power supply. These tests demonstrate the operability of the electric power systems, under conditions as close to design as practical, to assess the continuity of these systems and the condition of the components.

Inspection and testing of electric power systems described in Sections 8 and 16 conform to Criterion 18.

3.1.2.2.10 Criterion 19 - Control Room

A control room shall be provided from which the nuclear power unit can be operated safely under normal conditions and maintained in a safe condition under all postulated accident conditions. 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 total effective dose equivalent (TEDE) for the duration of the accident. This limitation is consistent with the requirements of 10CFR50.67, "Accident Source Term".

Equipment at appropriate locations outside the control room shall be provided:

1. With 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.

2. With a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures.

3.1.2.2.10.1 Design Conformance to Criterion 19

A main control room is provided with appropriate controls and instrumentation to permit personnel to operate the unit safely under normal and accident conditions, including LOCAs. The main control room and associated post-accident ventilation systems are designed in accordance with Seismic Category I requirements.

The design of the main control room permits access and occupancy during accident conditions. Shielding and ventilation are provided to permit occupancy of the main control room for a period of 30 days following a LOCA without receiving more than a 5 rem integrated whole body dose, or its equivalent to any part of the body.

The capability for prompt hot shutdown, including instrumentation and controls to maintain the unit in a safe condition during hot shutdown and subsequent cold shutdown of the reactor through suitable procedures from locations outside the main control room, is provided by the remote shutdown system if the main control room becomes inaccessible.

The main control room and the Remote Shutdown System conform to Criterion 19.

For further discussion, see the following sections:

1. Systems required for safe shutdown - Section 7.4

2. Air conditioning, heating, cooling, and ventilation systems - Section 9.4

3. Habitability systems - Section 6.4.

3.1.2.3 Group III - Protection and Reactivity Control Systems

3.1.2.3.1 Criterion 20 - Protection System Functions

The protection system shall be designed:

1. To automatically initiate 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.

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

3.1.2.3.1.1 Design Conformance to Criterion 20

The RPS is designed to provide timely protection against the onset and consequences of conditions that threaten the integrity of the fuel barrier and the RCPB. Fuel damage is prevented by initiation of an automatic reactor shutdown if monitored nuclear system variables exceed pre-established limits of anticipated operational occurrences. Scram trip settings are selected and verified to be far enough above or below operating levels to provide proper protection but to not be subject to spurious scrams. The RPS includes a high inertia motor generator power system, sensors, bypass circuitry, and switches that signal the control rod drive (CRD) system to scram and thereby shut down the reactor. The scrams initiated by neutron monitoring system variables, nuclear system high pressure, turbine stop valve closure, turbine control valve

(TCV) fast closure, MSIV closure, and reactor vessel low water level prevent fuel damage following abnormal operational transients. Specifically, these process parameters initiate a shutdown 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.

In addition to the RPS, 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. Following a LOCA, the ECCS, primary containment isolation, and RCIC system are initiated automatically to limit the extent of fuel damage and 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. Fuel system design - Section 4.2

2. ECCS - Section 6.3

3. RPS - Section 7.2

4. ESF systems - Section 7.3

5. All other instrumentation systems required for safety - Section 7.6

6. Accident analysis - Section 15.

3.1.2.3.2 Criterion 21 - Protection System Reliability and Testability

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

1. No single failure results in loss of the protection function

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.

3.1.2.3.2.1 Design Conformance to Criterion 21

The RPS is designed for high functional reliability and inservice testability. The protection system design fulfills the single failure criterion by providing redundant channels. No single component failure, intentional bypass, maintenance operation, calibration operation, or test to verify operational availability impairs the ability of the system to perform its intended safety function. The system design ensures that when a scram trip setpoint is exceeded, there is a high probability of successful completion of the required safety function. Electrical and physical separation between channels and between logics monitoring the same variable prevent environmental factors, electrical transients, and physical

events from impairing the ability of the system to respond correctly.

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

The Reactor Protection System initiates an automatic reactor shutdown if the monitored plant variables exceed pre-established limits. This system is arranged as two separately powered trip systems. Each trip system has two trip 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 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.

CRD operability can be tested during normal reactor operation. Drive position indicators and incore 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 signficantly disturbing 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. The hydraulic control unit (HCU) scram accumulator and the scram discharge volume level are continuously monitored.

The MSIVs 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.

The ESFs are designed to be operable for test purposes during normal operation of the nuclear system. The high functional reliability, redundancy, independence, and inservice testability of the protection system satisfy the requirements specified in Criterion 21.

For further information, see the following sections:

1. Component and subsystem design - Section 5.4

2. Containment systems - Section 6.2

3. ECCS - Section 6.3

4. RPS - Section 7.2

5. ESF systems - Section 7.3

6. Systems required for safe shutdown - Section 7.4

7. All other instrumentation systems required for safety - Section 7.6

8. Accident analysis - Section 15.

3.1.2.3.3 Criterion 22 - Protection System Independence

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

principles of operation, shall be used to the extent practical to prevent loss of the protection function.

3.1.2.3.3.1 Design Conformance to Criterion 22

The components of protection systems are designed so that the mechanical and thermal environmental conditions resulting from any potential accident in which the components are required to function does not interfere with that function. Wiring for the RPS outside of the main control room enclosures is run in rigid conduits or enclosed raceways segregated from all other wiring. Only one trip actuator logic circuit from each trip system may be run in the same conduit.

The system sensors are electrically and physically separated to provide protection from loss of function caused by a DBA. In general, redundant sensors have separate process taps. Where common process taps are used, failure of the common process taps does not interfere with the protection function. The wires from duplicate sensors on a common process tap are run in separate wireways.

The design uses multiple trip logics so that an intentional bypass, maintenance operation, calibration operation, or test does not prevent completion of a protection function when required.

The RPS is designed to permit maintenance and diagnostic work while the reactor is operating without restricting the plant operation or hindering safety functions.

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

For further information, see the following sections:

1. Component and subsystem design - Section 5.4

2. ECCS - Section 6.3

3. RPS - Section 7.2

4. ESF systems - Section 7.3

5. All other instrumentation systems required for safety - Section 7.6

6. Accident analysis - Section 15.

3.1.2.3.4 Criterion 23 - Protection System Failure Modes

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

3.1.2.3.4.1 Design Conformance to Criterion 23

The RPS and the normally energized portion of the Primary Containment and Reactor Vessel Isolation Control Systems are designed to fail into a safe state on disconnection or loss of energy supply.

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 causes a channel trip. Only one trip channel in each trip system must be actuated to initiate a scram. A failure of any one Reactor Protection (trip) System input or subsystem component produces a trip in one of two channels; and therefore one trip system. This condition is insufficient to produce a reactor scram, but the system is ready to perform its protective function upon another channel trip in the other trip system.

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

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

For further information, see the following sections:

1. Environmental design of mechanical and electrical equipment - Section 3.11

2. ECCS - Section 6.3

3. RPS - Section 7.2

4. ESF systems - Section 7.3

5. All other instrumentation systems required for safety - Section 7.6.

3.1.2.3.5 Criterion 24 - Separation of Protection and Control Systems

The protection system shall be separated from control systems to the extent that failure of any single control system component or channel, or failure or removal from service of any single protection system component or channel that 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 to assure that safety is not significantly impaired.

3.1.2.3.5.1 Design Conformance to Criterion 24

The RPS and the process control systems are separated. Sensors, trip channels, and trip logics of the RPS are not used directly for automatic control of process systems. Therefore, failure in the controls and instrumentation of process systems cannot induce failure of any portion of the protection system. High scram reliability is designed into the Reactor Protection (trip) System and the hydraulic control unit for the CRD system. 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 systems 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 line isolation.

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

For further information, see the following sections:

1. ECCS - Section 6.3

2. RPS - Section 7.2

3. ESF systems - Section 7.3

4. All other instrumentation systems required for safety - Section 7.6.

3.1.2.3.6 Criterion 25 - Protection System Requirements for Reactivity Control Malfunctions

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

3.1.2.3.6.1 Design Conformance to Criterion 25

The RPS provides protection against the onset and consequences of conditions that threaten the integrity of the fuel barrier and the RCPB. Any monitored variable that exceeds the scram setpoint initiates an automatic scram and does not impair the remaining variables from being monitored. If one channel fails, the remaining portions of the RPS continue to 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 Reactor Manual Control System is completely independent of the circuitry controlling the scram valves. This separation of the scram and normal rod control functions prevents failures in the reactor manual control circuitry from affecting the scram circuitry. Because each control rod is controlled as an individual unit, a failure that results in energizing any of the insert or withdraw solenoid valves can affect only one control rod. The effectiveness of a reactor scram is not impaired by the malfunctioning of any one control rod.

The design of the protection system 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 information, see the following sections:

1. Fuel system design - Section 4.2

2. Nuclear design - Section 4.3

3. Thermal and hydraulic design - Section 4.4

4. RPS - Section 7.2

5. All other instrumentation systems required for safety - Section 7.6

6. Control systems not required for safety - Section 7.7

7. Accident analysis - Section 15.

3.1.2.3.7 Criterion 26 - Reactivity Control System Redundancy and Capability

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

3.1.2.3.7.1 Design Conformance to Criterion 26

Two independent reactivity control systems of different design are provided. The normal method of reactivity control employs control rod assemblies that contain boron carbide powder. Positive insertion of these control rods is provided by the CRD hydraulic system. The control rods are capable of reliably controlling reactivity changes during normal operation, e.g., power changes, power shaping, xenon burnout, and normal startup and shutdown, via operator-controlled insertions and withdrawals. The control rods are also capable of maintaining the core within acceptable fuel design limits during anticipated operational occurrences via the automatic scram function. The unlikely occurrence of a limited number of rods stuck during a scram would 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. Accumulator pressure and reactor vessel pressure, two sources of scram energy, provide needed scram performance over the entire range of reactor pressure, i.e., from operating conditions to cold shutdown.

The design of the Control Rod System includes appropriate margin for malfunctions, such as stuck rods in the highly unlikely event that they do occur. Control rod withdrawal sequences and patterns are selected prior to operation to achieve optimum core performance, and simultaneously, low individual rod worths. The operating procedures to accomplish these patterns are supplemented by the rod worth minimizer, which prevents the rod withdrawals from yielding a rod worth greater than permitted by the preselected rod withdrawal pattern. Because of the carefully planned and regulated rod withdrawal sequence, prompt shutdown of the reactor can be achieved with the insertion of a small number of the many independent control rods. In the event that a reactor scram is necessary, the unlikely

occurrence of a limited number of stuck rods does not hinder the capability of the control rod system to render the core subcritical.

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

The Control Rod System is capable of holding the reactor core subcritical under cold conditions, even when the control rod of highest worth is assumed to be stuck in the fully withdrawn position. This shutdown capability of the control rod system is made possible by designing the fuel with burnable poison (Gd_2O_2) to control the high reactivity of fresh fuel.

In addition to the Control Rod System, the Standby Liquid Control (SLC) System, containing a neutron-absorbing sodium pentaborate solution, is available as an independent backup system. This system has the capability to shut down the reactor from full power and maintain it in a subcritical condition at any time during the core life. The reactivity control provided to reduce reactor power from rated power to shutdown condition with the control rods withdrawn in the power pattern accounts for the following:

1. Reactivity effects on xenon decay

2. Elimination of steam voids

3. Change in water density due to the reduction in water temperature

4. Doppler effect in uranium

5. Change in neutron leakage due to moderator change from boiling to cold

6. Change in rod worth as boron affects the neutron migration length.

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

For further information, see the following sections:

1. Fuel mechanical design - Section 4.2

2. RPS - Section 7.2

3. ESF system - Section 7.3

4. Systems required for safe shutdown - Section 7.4

5. All other instrumentation systems required for safety - Section 7.6

6. Control systems not required for safety - Section 7.7.

3.1.2.3.8 Criterion 27 - Combined Reactivity Control Systems Capability

The reactivity control systems shall be designed to have a combined capability, in conjunction with poison addition by the ECCS 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.

3.1.2.3.8.1 Evaluation for Criterion 27

There is no credible event that requires combined capability of the control rod system and poison additions by the SLC system. The primary reactivity control system for the BWR during postulated accident conditions is the control rod system. Abnormalities are sensed, and, if protection system limits are reached, corrective action is initiated through an automatic scram. High integrity of the protection system is achieved through the combination of logic arrangement, 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 CRD system. Response by the RPS is prompt, and the total scram time is short.

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

The reactor core design assists in maintaining the stability of the core under accident conditions, as well as during power operation. The fuel temperature or Doppler coefficient, moderator void coefficient, and moderator temperature coefficient are reactivity

coefficients in the power range that contribute to system stability. The overall power reactivity coefficient is negative and provides a strong negative reactivity feedback under severe power transient conditions.

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

For further discussion, see the following sections:

1. Fuel mechanical design - Section 4.2

2. RPS - Section 7.2

3. Systems required for safe shutdown - Section 7.4

4. All other instrumentation systems required for safety - Section 7.6

5. Control systems not required for safety - Section 7.7.

3.1.2.3.9 Criterion 28 - Reactivity Limits

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

1. Result in damage to the RCPB greater than limited local yielding, nor

2. Sufficiently disturb the core, its support structures or other reactor pressure vessel (RPV) internals to significantly impair the capability to cool the core.

These postulated reactivity accidents shall include consideration of rod ejection (unless prevented by positive means), rod dropout, steam line rupture, changes in reactor coolant temperature and pressure, and cold water addition.

3.1.2.3.9.1 Design Conformance to Criterion 28

The Control Rod System design incorporates appropriate limits on the potential amount and rate or reactivity increase. Control rod withdrawal sequences and patterns are selected to achieve optimum core performance and low individual rod worths. The rod worth minimizer limits withdrawal by other than the preselected rod withdrawal pattern. The Rod Worth Minimizer 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 that prevents rapid rod ejection. This engineered safeguard protects against a high reactivity insertion rate by limiting the control rod velocity to less than 5 ft/sec. Normal rod movement is limited to 6-inch increments, and the rod withdrawal rate is limited through the hydraulic valve to 3 in./sec.

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

for any of the postulated reactivity accidents described in Section 15.

The design features of the Reactivity Control System that limit the potential amount and rate of reactivity increase ensure that Criterion 28 is satisfied for all postulated reactivity accidents.

For further information, see the following sections:

1. Design of structures, components, equipment, and systems - Section 3

2. Fuel system design - Section 4.2

3. Nuclear design - Section 4.3

4. Reactor materials - Section 4.5

5. Integrity of RCPB - Section 5.2

6. Reactor vessel - Section 5.3

7. Component and subsystem design - Section 5.4

8. RPS - Section 7.2

9. All other instrumentation systems required for safety - Section 7.6

10. Accident analysis - Section 15.

3.1.2.3.10 Criterion 29 - Protection Against Anticipated Operational Occurrences

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

3.1.2.3.10.1 Design Conformance to Criterion 29

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

An extremely high reliability of timely response to anticipated operational occurrences is maintained by a thorough program of inservice testing and surveillance. Active components can be tested or removed from service for maintenance during reactor operation without compromising the protection or reactivity control functions.

Components important to safety such as CRDs, MSIVs, RHR pumps, etc, are tested during normal reactor operation. Functional testing and calibration schedules are developed using available failure rate, data, reliability analyses, and operating experience. These schedules represent an optimization of protection and reactivity control system reliability by considering the reliability effects during individual component testing on the portion of the system not being tested. The capability for inservice testing ensures the high functional reliability of protection and reactivity control systems should a reactor variable exceed the corrective action setpoint.

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

For further information, see the following sections:

1. Fuel system design - Section 4.2

2. Component and subsystem design - Section 5.4

3. Containment systems - Section 6.2

4. ECCS - Section 6.3

5. RPS - Section 7.2

6. ESF systems - Section 7.3

7. All other instrumentation systems required for safety - Section 7.6

8. Accident analysis - Section 15.

3.1.2.4 Group IV - Fluid Systems

3.1.2.4.1 Criterion 30 - Quality of Reactor Coolant Pressure Boundary

Components that are part of the RCPB 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.

3.1.2.4.1.1 Design Conformance to Criterion 30

By using conservative design practices and detailed quality control procedures, the pressure retaining components of the RCPB are designed and fabricated to retain their integrity during normal operation 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 Section 5 and Table 3.2-1. Furthermore, product and process quality planning is provided as described in Section 17 to ensure conformance with the applicable codes and standards and to retain appropriate documented evidence verifying compliance. Because this criterion deals with aspects of the RCPB, further discussion appears in the response to Criterion 14.

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 potential hazardous leaks before predetermined limits are exceeded. Small leaks are detected by temperature and pressure changes, increased frequency of sump pump operation, condensate flow in the drywell unit coolers, 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 loss of feedwater supply, makeup capabilities are provided by the RCIC system.

While the Leak Detection System provides protection from small leaks, the ECCS network provides protection for the complete range of discharges from ruptured pipes. Thus, protection is provided for the full spectrum of possible discharges.

The RCPB and the Leak Detection System are designed to meet the requirements of Criterion 30.

For further information, see the following sections:

1. Design of structures, components, equipment, and systems - Section 3

2. Integrity of RCPB - Section 5.2

3. Reactor vessel - Section 5.3

4. Component and subsystem design - Section 5.4

5. Reactor Recirculation System - Section 5.4

6. ESF systems - Section 7.3

7. Systems required for safe shutdown - Section 7.4

8. Reactor vessel instrumentation - Section 7.7

9. All other instrumentation systems required for safety - Section 7.6

10. Accident analysis - Section 15

11. QA - Section 17.

3.1.2.4.2 Criterion 31 - Fracture Prevention of RCPB

The RCPB 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

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:

    a. Material properties

    b. The effects of irradiation on material properties

    c. Residual, steady state, and transient stresses

d. Size of flaws.

3.1.2.4.2.1 Design Conformance to Criterion 31

Brittle fracture control of pressure retaining ferritic materials is provided to ensure protection against nonductile fracture. To minimize the possibility of brittle fracture failure of the RPV, the RPV is designed to meet the requirements of the ASME B&PV Code, Section III, as described in Section 5.

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 1×10^{17} 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 attenuates the fast neutron flux incident upon the reactor vessel wall. This annular volume contains the core shroud, jet pump assemblies, and reactor coolant.

The effect of neutron radiation on the fracture toughness of the RPV materials has been considered in the design. Plant operation will be modified as necessary to accommodate the small change in the initial reference transition temperature that will occur.

The RCPB is designed, maintained, and tested such that adequate assurance is provided that the boundary behaves in a nonbrittle manner throughout the life of the plant. Therefore, the RCPB is in conformance with Criterion 31.

For further information, see the following sections:

1. Design of structures, components, equipment, and systems - Section 3

2. Integrity of RCPB - Section 5.2

3. Reactor vessel - Section 5.3.

3.1.2.4.3 Criterion 32 - Inspection of RCPB

Components that are part of the RCPB shall be designed to permit:

1. Periodic inspection and testing of important areas and features to assess their structural and leaktight integrity

2. An appropriate material surveillance program for the RPV.

3.1.2.4.3.1 Design Conformance to Criterion 32

The RCPB design meets the requirements of the ASME B&PV Code, Section XI, which requires access for all mandatory inspections. Section 5.2.4 outlines additional details of these features.

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

Vessel material surveillance samples are located within the RPV to enable periodic monitoring of material properties with exposure. The program includes specimens of the base metal, weld metal, and heat affected zone metal.

The portion of the feedwater system that forms the RCPB is hydrostatically tested in accordance with Section III of the ASME B&PV Code.

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

For further information, see the following sections:

1. Design of structures, components, equipment, and systems - Section 3

2. RCPB - Section 5.2

3. Reactor vessel - Section 5.3

4. Component and subsystem design - Section 5.4.

3.1.2.4.4 Criterion 33 - Reactor Coolant Makeup

A system to supply reactor coolant makeup for protection against small breaks in the RCPB 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 RCPB and rupture of small piping or other small components that 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.

3.1.2.4.4.1 Design Conformance to Criterion 33

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 potential hazardous leaks before predetermined limits are exceeded. Small leaks are detected by temperature and pressure changes, increased frequency of sump pump operation, sump level measurement, 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 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 accompanied by a loss of feedwater supply, makeup capabilities are provided by the RCIC system. Thus, protection is provided to ensure that fuel clad temperature limits are not exceeded.

The plant is designed to provide ample reactor coolant makeup for protection against small leaks in the RCPB. The design of these systems meets the requirements of Criterion 33.

For further information, see the following sections:

1. Integrity of RCPB - Section 5.2

2. ECCS - Section 6.3

3. Systems required for safe shutdown - Section 7.4

4. All other instrumentation systems required for safety - Section 7.6.

3.1.2.4.5 Criterion 34 - Residual Heat Removal

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

Suitable redundancy in components, features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that, assuming a single failure: 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.

3.1.2.4.5.1 Design Conformance to Criterion 34

The RHR system provides the means to:

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

The RHR system consists of four motor driven pumps, two heat exchangers, and associated piping, valves, and instrumentation. It is divided into four loops. Each loop has a pump, and loops A and B each have a heat exchanger. The pump in each loop is physically separated and protected from all the other pumps to minimize the possibility of a single accident causing the loss of more than one loop.

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

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

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

For further information, see the following sections:

1. RHR system - Section 5.4.7

2. ECCS - Section 6.3

3. ESF systems - Section 7.3

4. Systems required for safe shutdown - Section 7.4

5. Onsite power systems - Section 8.3

6. Water systems - Section 9.2

7. Accident analysis - Section 15.

3.1.2.4.6 Criterion 35 - Emergency Core Cooling

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

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

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, assuming a single failure: 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.

3.1.2.4.6.1 Design Conformance to Criterion 35

The ECCS consists of the following:

1. High pressure coolant injection (HPCI)

2. Automatic Depressurization System (ADS)

3. Core Spray System

4. Low pressure coolant injection (LPCI), an operating mode of the RHR system.

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

The HPCI system consists of a turbine driven pump, system piping, valves, controls, and instrumentation. It 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 (CST) or the suppression pool.

The ADS reduces the reactor pressure so that flow from LPCI and core spray enters the reactor vessel in time to cool the core and prevent excessive fuel clad temperature. The ADS uses several of the

nuclear system pressure relief valves to relieve the high pressure steam to the suppression pool.

Two independent loops are provided as a part of the core spray system. Each loop consists of two centrifugal pumps that can be powered by normal auxiliary power or by the standby ac 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. If there is low water level in the reactor vessel or high pressure in the drywell, the core spray system automatically sprays water onto the top of the fuel assemblies in time and at a sufficient flow rate to cool the core and prevent excessive fuel temperature. The LPCI system starts from the same signals that initiate the core spray system and operates independently to achieve the same objective by flooding the reactor vessel.

If there is low water level in the reactor or high pressure in the drywell, the LPCI mode of operation of the RHR system pumps water into the reactor vessel in time to flood the core and prevent excessive fuel temperature. Protection provided by LPCI and core spray also extends to a small break where the ADS has operated to lower the RPV.

The RHR and core spray systems are powered from the safety-related buses. The design of the safety-related buses, as described in the evaluation for Criterion 17, ensures that emergency core cooling can be provided, assuming a single failure, when onsite electric power is available, assuming offsite power is not available, and when offsite electric power is available, assuming onsite power is not available.

Results of the performance of the ECCS for the entire spectrum of liquid breaks are discussed in Section 6.3. Also provided in Section 6.3.3 is an analysis to show that the ECCS conforms to 10CFR50, Appendix K. This analysis shows complete compliance with the final acceptance criteria.

The ECCS is 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 power supply, meets the requirements of Criterion 35.

For further information, see the following sections:

1. RHR system - Section 5.4.7
2. ECCS - Section 6.3
3. ESF systems - Section 7.3
4. Onsite power systems - Section 8.3
5. Water systems - Section 9.2
6. Accident analysis - Section 15.

3.1.2.4.7 Criterion 36 - Inspection of Emergency Core Cooling System

The ECCS shall be designed to permit appropriate periodic inspection of important components, such as spray rings in the RPV, water injection nozzles, and piping to assure the integrity and capability of the system.

3.1.2.4.7.1 Design Conformance to Criterion 36

The ECCS discussed in Criterion 35 include inservice inspection considerations.

The core spray spargers within the reactor vessel are accessible for inspection during each refueling outage. Removable plugs in the reactor shield and/or panels in the insulation provide access for examination of nozzles. Removable insulation is provided on the ECCS piping out to and including the first isolation valve outside

the drywell. Inspection of the ECCS is in accordance with Section XI of the ASME B&PV Code.

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 inservice inspection to detect defects that might affect the cooling performance. The design of the reactor vessel and internals for inservice inspection, and the plant testing and inspection program ensures that the requirements of Criterion 36 are met.

See Section 5.2.4 for further information on ECCS inservice inspection.

3.1.2.4.8 Criterion 37 - Testing of Emergency Core Cooling System

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

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.

3.1.2.4.8.1 Design Conformance to Criterion 37

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

The HPCI, LPCI, core spray systems 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 are tested periodically to verify operability. Flow rate tests are conducted on the HPCI, LPCI, and core spray systems.

Each system of the ECCS is capable of being tested under conditions as close to design as practicable, to verify the performance of the full operational sequence that brings each system into operation, including the transfer between normal and emergency power sources. The operation of the associated cooling water systems is discussed in the evaluation of Criterion 46. It is concluded that the requirements of Criterion 37 are met.

For further information, see the following sections:

1. Overpressurization protection - Section 5.2

2. ECCS inspection and testing - Section 6.3

3. ECCS instrumentation and controls - Section 7.3

4. Standby ac power system - Section 8.3

5. Technical specifications

3.1.2.4.9 Criterion 38 - Containment Heat Removal

A system to remove heat from the reactor containment shall be provided. The system safety function shall be to rapidly reduce, consistent with the functioning of other associated systems, the containment pressure and temperature following any LOCA 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, assuming a single failure: 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.

3.1.2.4.9.1 Design Conformance to Criterion 38

The containment heat removal function is accomplished by the Residual Heat Removal (RHR) System. Following a LOCA, one or both of the following operating modes of the RHR system would be initiated:

1. Containment spray - Condenses steam within the containment

2. Suppression pool cooling - Limits the temperature within the containment by removing heat from the suppression pool water via the RHR heat exchangers. Either or both redundant RHR heat exchangers can be manually actuated.

The redundancy and capability of the Offsite and Onsite Electrical Power System for the RHR system is presented in the evaluation Criterion 34.

For further information, see the following sections:

1. RHR system - Section 5.4.7

2. Containment systems - Section 6.2

3. Onsite power systems - Section 8.3.1

4. Water systems - Section 9.2

5. Accident analysis - Section 15.

3.1.2.4.10 Criterion 39 - Inspection of Containment Heat Removal System

The Containment Heat Removal System shall be designed to permit appropriate periodic inspection of important components, such as the suppression chamber, sumps, spray nozzles, and piping to assure the integrity and capability of the system.

3.1.2.4.10.1 Design Conformance to Criterion 39

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 drywell can be visually inspected at any time and are inspected periodically. The testing frequencies of most components are correlated with the component inspection.

The suppression chamber is designed to permit appropriate periodic inspection. Space is provided outside the drywell 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 information, see the following sections:

1. RHR system - Section 5.4.7

2. Containment systems - Section 6.2

3. ECCS - Section 6.3

4. ESF systems - Section 7.3

5. Water systems - Section 9.2.

3.1.2.4.11 Criterion 40 - Testing of Containment Heat Removal System

The Containment Heat Removal System shall be designed to permit appropriate periodic pressure and functional testing to assure:

1. The structural and leaktight integrity of its components

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

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.

3.1.2.4.11.1 Design Conformance to Criterion 40

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 system are periodically operated to verify operability. The cooling mode is not automatically

initiated, but operation of the components is periodically verified. The operation of associated cooling water systems is discussed in the evaluation of GDC 46. It is concluded that the requirements of Criterion 40 are met.

3.1.2.4.12 Criterion 41 - Containment Atmosphere Cleanup

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

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

3.1.2.4.12.1 Design Conformance to Criterion 41

Fission products, hydrogen, oxygen, and other substances released from the reactor during a postulated accident are contained within the primary containment. Any fission leakage from the primary containment during this postulated accident enters the reactor building, where it is removed by the Filtration, Recirculation, and Ventilation System (FRVS).

The FRVS is actuated automatically by separate, redundant trip circuits that sense high radioactivity in the Reactor Building. The system is also actuated simultaneously with Reactor Building

isolation and shutdown of the normal Reactor Building Ventilation System based on a LOCA signal.

The FRVS consists of six 25 percent capacity recirculation units and two 100 percent capacity vent units, connected in parallel to the RBVS supply and exhaust ducts within the reactor building. The filtered recirculation and low flow ventilation of the FRVS, in conjunction with low leakage building construction, act jointly to confine, control, and collect radioactive airborne contamination that might be released within the reactor building during a postulated accident.

A Combustible Gas Control System, consisting of two 100 percent capacity hydrogen recombiners, maintains hydrogen and oxygen concentrations in the primary containment below flammable limits following a beyond design basis accident. Each hydrogen recombiner has its own primary containment penetration.

The FRVS and the Combustible Gas Control System meet the requirements of GDC 41. The redundancy and capability of the Offsite and Onsite Electrical Power Systems for these systems ensure that the systems safety functions can be accomplished, assuming there is a single failure, for Onsite Electric Power System operation, assuming that offsite power is not available, and for Offsite Electric Power System operation, assuming that onsite power is not available.

For further information, see the following sections:

1. ESF filter systems - Section 6.5.1

2. FRVS - Section 6.8

3. Combustible gas control - Section 6.2.5.

3.1.2.4.13 Criterion 42 - Inspection of Containment Atmosphere Cleanup Systems

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

3.1.2.4.13.1 Design Conformance to Criterion 42

Inspection of the internal structures of the FRVS is facilitated by access doors installed in each unit to allow entry for visual inspection of structural members and filter faces.

All active components of the Combustible Gas Control System are located external to the primary containment and are accessible for inspection during normal operation of the plant.

The design of the containment atmosphere cleanup systems meets the requirements of Criterion 42.

For further information, see Section 6.5.

3.1.2.4.14 Criterion 43 - Testing of Containment Atmosphere Cleanup Systems

The containment atmosphere cleanup systems shall be designed to permit appropriate periodic pressure and functional testing to assure:

1. The structural and leaktight integrity of its components.

2. The operability and performance of the active components of the systems such as fans, filters, dampers, pumps, and valves.

3. The operability of the systems as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the systems into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of associated systems.

3.1.2.4.14.1 Design Conformance to Criterion 43

The FRVS is operated periodically to verify the operability and performance of major active components, such as fans, filters, dampers, motors, pumps, and valves, as well as the structural integrity of the unit. See Section 8.3.1 for a discussion of the testing of the auxiliary power system.

The leaktightness of the high efficiency particulate air (HEPA) filters is measured by the dioctyl-phthalate (DOP) test, and is done in accordance with Section 10 of ANSI N510. For further discussion of this testing, see Section 6.5 and 6.8.

Each loop of the Combustible Gas Control System is designed for periodic pressure and operability testing. For a further discussion of the combustible gas control system, see Section 6.2.5.

The design of the containment atmosphere cleanup systems meets the requirements of GDC 43.

3.1.2.4.15 Criterion 44 - Cooling Water

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

Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that, assuming a single failure: 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.

3.1.2.4.15.1 Design Conformance to Criterion 44

The Station Service Water System (SSWS) provides cooling water from the Delaware River to the Safety Auxiliaries Cooling System (SACS) and the Reactor Auxiliaries Cooling System (RACS) heat exchangers for the removal of excess heat from all structures, systems, and components that are necessary to maintain safety during all normal and accident conditions. These include the SDGs and room coolers; the fuel pool heat exchangers; the RHR pump seal and motor bearing coolers; the main control room chillers; the core spray pump compartment unit coolers; the RCIC pump compartment unit coolers; the HPCI pump compartment unit coolers; the RHR heat exchangers and pump compartment unit coolers; and the FRVS cooling coils; the Reactor Water Cleanup (RWCU) System pumps; the RWCU system non-regenerative heat exchangers; reactor recirculation pump seal coolers; the reactor recirculation pump motor oil coolers; the CRD pump seal coolers; reactor building equipment drain sump coolers; the feed gas cooler condensers; the concentrated waste tanks; the waste evaporator condensers**; the recombiner cooler condenser; the phase separator coolers; the feed gas compressor aftercoolers; the gaseous radwaste compressor aftercoolers; and the emergency air compressor heat exchanger.

The SSWS and SACS are designed to Seismic Category I requirements. Redundant safety-related components served by the systems are supplied through redundant supply headers and returned through redundant discharge or return lines. Electric power for operation of redundant safety-related components is supplied from separate independent offsite and redundant onsite standby power sources. No

** NOTE: Waste Evaporator Condensers are abandoned in place.

single failure prevents these systems from performing their safety function.

The SSWS and the SACS meet the requirements of GDC 44. For further information, see the following sections:

1. AC power systems - Section 8.3.1

2. Station Service Water System - Section 9.2.1

3. Cooling systems for reactor auxiliaries - Section 9.2.2

4. Ultimate heat sink - Section 9.2.5

5. SACS - Section 9.2.2

6. RACS - Section 9.2.8.

3.1.2.4.16 Criterion 45 - Inspection of Cooling Water System

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

3.1.2.4.16.1 Design Conformance to Criterion 45

The SSWS and the SACS are designed to permit appropriate periodic inspection of important components, such as heat exchangers and piping, to ensure the integrity and capability of the system. For further discussion of the SSWS and SACS, see Section 9.2.1 and 9.2.2, respectively.

3.1.2.4.17 Criterion 46 - Testing of Cooling Water System

The Cooling Water System shall be designed to permit appropriate periodic pressure and functional testing to assure:

1. The structural and leaktight integrity of its components

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

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

3.1.2.4.17.1 Design Conformance to Criterion 46

The SSWS and SACS are in operation during all plant shutdowns. They are periodically tested when the SDGs are tested. This testing includes transfer between the normal offsite power supply and the Emergency Onsite Power System. These systems are designed, to the extent practicable, to permit demonstration of operability of the systems and structural and leaktight integrity of cooling water components as required for operation during a LOCA or a loss of offsite power (LOP). Thus, the SSWS and SACS meet the requirements of GDC 46.

For further information, see the following sections:

1. AC power systems - Section 8.3.1

2. SSWS - Section 9.2.1

3. SACS - Section 9.2.2.

3.1.2.5 Group V - Reactor Containment

3.1.2.5.1 Criterion 50 - Containment Design Basis

The reactor containment structure, including access openings, penetrations, and the Containment Heat Removal System shall be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and, with sufficient margin, the calculated pressure and temperature conditions resulting from any LOCA. This margin shall reflect consideration of:

1. The effects of potential energy sources that have not been included in the determination of the peak conditions, such as energy in steam generators and energy from metal-water and other chemical reactions that may result from degraded emergency core cooling functioning

2. The limited experience and experimental data available for defining accident phenomena and containment responses

3. The conservatism of the calculational model and input parameters.

3.1.2.5.1.1 Design Conformance to Criterion 50

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 occurring during the postulated DBA. More detailed information demonstrating compliance with Criterion 50 can be found in Section 6.2.

3.1.2.5.2 Criterion 51 - Fracture Prevention of Containment Pressure Boundary

The containment pressure 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

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

3. Size of flaws.

3.1.2.5.2.1 Design Conformance to Criterion 51

The primary containment boundary is designed to the load combinations 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 established conditions above.

The weld procedure qualification ensures that the toughness of the weld metal and the heat affected zones meet 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.

Sufficient margin is inherent in the design to account for the various uncertainties involved in design and fabrication. The design of the containment pressure boundary meets the requirements of GDC 51.

3.1.2.5.3 Criterion 52 - Capability for Containment Leakage Rate Testing

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

3.1.2.5.3.1 Design Conformance to Criterion 52

The primary containment and all other equipment that may be subjected to containment test conditions are designed to permit Type A, integrated leak rate testing, as described in Appendix J, Option B, of 10CFR50. The design of the primary containment thus meets the requirements of GDC 52. A more complete discussion can be found in Section 6.2.6 and Section 16.

3.1.2.5.4 Criterion 53 - Provisions for Containment Testing and Inspection

The reactor containment shall be designed to permit:

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

2. An appropriate surveillance program

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

3.1.2.5.4.1 Design Conformance to Criterion 53

The primary containment is designed to optimize the accessibility of important areas to permit required inspection and surveillance.

All penetrations with resilient seals and expansion bellows are designed to permit local leak rate testing, as described in Appendix J, Option B, of 10CFR50. This is discussed further in Section 6.2.6.

The primary containment meets the requirements of GDC 53.

3.1.2.5.5 Criterion 54 - Piping Systems Penetrating Containment

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

3.1.2.5.5.1 Design Conformance to Criterion 54

Piping systems that penetrate the primary containment have been provided with isolation and leak detection capabilities. These penetrations are discussed in Section 6.2.4. Both the isolation valves and the system that initiates isolation use components whose quality maximizes reliability. Sufficient independence and redundancy is provided to ensure effective isolation. Primary

containment isolation is discussed in Section 6.2.4, and the system that initiates isolation is discussed in Section 7.3.

Piping systems that penetrate the primary containment are designed to permit Type C local leak rate testing as described in Section 6.2.6. The operability of the isolation valves and associated equipment can be verified during the leak rate testing program. Primary containment leakage testing is discussed further in Section 6.2.6.

Piping systems that penetrate the primary containment meet GDC 54.

3.1.2.5.6 Criterion 55 - RCPB Penetrating Containment

Each line that is part of the RCPB, 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.

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

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.

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 inservice inspection, protection against more severe natural phenomena, and additional isolation valves and containment shall include consideration of the population density, use characteristics, and physical characteristics of the site environs.

3.1.2.5.6.1 Design Conformance to Criterion 55

The lines of the RCPB that penetrate the primary containment have suitable isolation valves capable of isolating the primary containment, thereby precluding any significant release of radioactivity. Similarly, for lines that do not penetrate the primary containment but form a portion of the RCPB, the design assures that isolation from the RCPB can be achieved.

Each line that is part of the RCPB, and that penetrates the primary containment, meets the requirements outlined in GDC 55.

For further information, see the following sections:

1. Integrity of RCPB - Section 5.2

2. Containment isolation systems - Section 6.2.4

3. Instrumentation and controls - Section 7

4. Accident analysis - Section 15

5. Technical specifications

3.1.2.5.7 Criterion 56 - Primary Containment Isolation

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

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

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

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.

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.

3.1.2.5.7.1 Design Conformance to Criterion 56

Each line that primary connects directly to the containment atmosphere and penetrates primary containment conforms to the requirements of GDC 56, as described in the following sections:

1. Containment isolation systems - Section 6.2.4

2. Instrumentation and controls - Section 7

3. Accident analyses - Section 15.

4. Technical specifications

3.1.2.5.8 Criterion 57 - Closed System Isolation Valves

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

3.1.2.5.8.1 Design Conformance to Criterion 57

Each line that penetrates the primary containment, and is neither part of the RCPB nor primary connected directly to the primary containment atmosphere, conforms to the requirements of GDC 57, as described in Section 6.2.4.

3.1.2.6 <u>Group VI - Fuel and Radioactivity Control</u>

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

The nuclear power unit design shall include means to suitably control 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.

3.1.2.6.1.1 Design Conformance to Criterion 60

Waste handling systems have been incorporated in the plant design for processing and/or retention of radioactive wastes from normal plant operations, to ensure that the effluent releases to the environment are as low as is reasonably achievable and within the limits of 10CFR20, 10CFR50, and applicable regulations for normal operations and any transient situation that might reasonably be anticipated. The plant is also designed with provisions to prevent radioactive releases during accidents from exceeding the limits of 10CFR50.67 dosage level guidelines for potential accidents of exceedingly low probability of occurrence. Consistent with Regulatory Guide 1.21, all releases will be reported.

The principal gaseous effluents from the plant during normal operation are the noncondensable gases from the condenser air ejectors. The activity level of waste gas effluents is substantially reduced by holdup of noble gases from the gaseous radwaste system in ambient temperature charcoal decay beds and subsequent release at plant exhaust ducts. The effluent from this system is continuously monitored and controlled and the system is shut down and isolated in the event of abnormally high radiation levels.

Liquid radioactive wastes are collected in waste collector tanks, treated on a batch basis through filters and demineralizers or evaporators depending on stream chemistry, and then either returned to the plant systems or released in a controlled manner to the environment. Radioactive liquid waste system tankage and evaporator capacity is sufficient to handle any expected transient in processing liquid waste volume. All discharges to the environment are routed through a process monitor and a monitoring station that continuously monitor and record the activity of the waste, rate of flow, and provide an alarm to the operator in the event of high

activity level. The process monitor also isolates the system and terminates the discharge.

The Turbine Building Circulating Water Dewatering Sump (CWDWS) collects condensation through drains from certain ventilation units. These drains may contain low levels of tritium. The CWDWS pumps discharge to the cooling tower. Discharges are routed through a radiation monitor to continuously monitor discharges. The radiation monitor trips the sump pumps on high gamma activity levels to terminate the discharge. A composite sampler is used to obtain samples during discharge in accordance with UFSAR Table 11.5-3.

Solid wastes, including spent resins, filter sludges, filter cartridges, evaporator bottoms, and contaminated tools and equipment, are collected, packaged, and shipped offsite in shielded and reinforced containers that meet applicable NRC and Department of Transportation requirements.

The design of the Waste Disposal System meets the requirements of Criterion 60.

For further discussion, see the following sections:

1. General plant description - Section 1.2

2. Detection of leakage through RCPB - Section 5.2.5

3. Containment systems - Section 6.2

4. Liquid waste systems - Section 11.2

5. Gaseous waste systems - Section 11.3

6. Solid waste system - Section 11.4

7. Process and effluent radiological monitoring and sampling systems - Section 11.5

8. Accident analysis - Section 15

9. Technical specifications

3.1.2.6.2 Criterion 61 - Fuel Storage and Handling and Radioactivity Control

The fuel storage and handling, radioactive waste, and other systems that may contain radioactivity shall be designed to ensure 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.

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

3.1.2.6.2.1 Design Conformance to Criterion 61

The fuel storage pool has adequate water shielding for stored spent fuel. Adequate shielding for transporting fuel is also provided. Liquid level sensors are installed in the surge tank and the fuel pool to detect low pool water level. The Reactor Building is designed to meet Regulatory Guide 1.13 criteria as follows:

1. New fuel storage - New fuel can be placed in dry storage in the new fuel storage vault located inside the reactor building. The storage vault provides adequate shielding for radiation protection. The geometry and administratively controlled loading pattern of the storage racks preclude accidental criticality mentioned in the Criterion 62 evaluation. The new fuel storage racks do

not require any special inspection and testing for nuclear safety purposes. New fuel can also be stored in the spent fuel pool.

2. Spent fuel handling and storage - Irradiated fuel is stored submerged in the spent fuel storage pool located in the reactor building in high density racks. 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 rack geometry and materials preclude accidental criticality mentioned in the Criterion 62 evaluation.

3. Radioactive waste systems - The radioactive waste systems provide all equipment necessary to collect, process, and prepare for disposal of all radioactive liquid, gas, and solid wastes 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, evaporation, and dilution. The accumulated wet solid wastes are dewatered and the concentrates evaporated prior to being solidified and packaged in steel drums. Compactible dry solid radwastes are compressed and packaged in steel drums or other suitable containers. Gaseous radwastes are processed, delayed, monitored, recorded, controlled, and released in such a way 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 areas have sufficient shielding to maintain dose rates within the limits set forth in 10CFR20 and 10CFR50. The radwaste area is designed to preclude accidental

release to the environment of radioactive materials that exceed the limits allowed by these regulations.

The radwaste systems are used on a routine basis and do not require specific testing to ensure operability. Radiation monitors check performance 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. RHR system - Section 5.4.7

2. Containment systems - Section 6.2

3. Fuel storage and handling - Section 9.1

4. Heating Ventilating, and Air Conditioning (HVAC) Systems - Section 9.4

5. Radioactive waste management - Chapter 11

6. Radiation protection - Chapter 12.

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

Criticality in the Fuel Storage and Handling System shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations.

3.1.2.6.3.1 Design Conformance to Criterion 62

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 geometric configuration and loading patterns of the storage rack. Criticality in the spent fuel pool is prevented by geometric configuration and material construction of the storage racks. Fuel elements are limited by rack design to only top loading and fuel assembly positions. The new and spent fuel racks are Seismic Category I components.

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 reactor building and the administratively controlled loading patterns are designed to prevent an accidental critical array, even if the vault becomes flooded or subjected to seismic loadings.

The new fuel vault assembly spacing in the administratively controlled loading pattern limits the effective multiplication factor of the fuel array so that it will not exceed 0.90 for dry conditions and 0.95 for flooded conditions. New fuel storage is discussed in Section 9.1.1. New fuel can also be stored in the spent fuel pool.

Spent fuel is stored underwater in the spent fuel pool. The racks in which spent fuel assemblies are placed are designed and arranged to ensure subcriticality in the storage pool. The effective multiplication factor of the fuel array will not exceed 0.95. Spent fuel storage is discussed in Section 9.1.2.

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

The use of geometrically safe configurations for new and spent fuel storage and the design of fuel handling systems precludes accidental criticality in accordance with Criterion 62.

3.1.2.6.4 Criterion 63 - Monitoring Fuel and Waste Storage

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

1. Detect conditions that may result in loss of RHR capability and excessive radiation levels

2. Initiate appropriate safety actions.

3.1.2.6.4.1 Design Conformance to Criterion 63

Appropriate systems have been provided to meet the requirements of GDC 63. A malfunction of the fuel pool cooling and cleanup system that could result in loss of RHR capability and excessive radiation levels is alarmed in the main control room. Alarmed conditions include surge tank low-low level alarm, low pump suction pressure, low pump discharge flow, and low fuel pool level. System temperature is also continuously monitored and alarmed in the main control room. Area radiation monitors initiate an alarm in the main control room if radiation is abnormal. Spent fuel storage is discussed in Section 9.1.2, and fuel pool cooling and cleanup are discussed in Section 9.1.3.

Area radiation, tank, and sump levels are monitored and alarmed to give indication of conditions that may result in excessive radiation levels in Radioactive Waste System areas. Area radiation monitors are discussed in Sections 9.4 and 11.5. These systems satisfy the requirements of Criterion 63.

For further information, see the following sections:

1. Fuel storage and handling - Section 9.1

2. Liquid Radwaste System - Section 11.2

3. Gaseous Radwaste System - Section 11.3

4. Solid Radwaste System - Section 11.4

5. Process radiation monitoring - Section 11.5.

3.1.2.6.5 Criterion 64 - Monitoring Radioactivity Releases

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

3.1.2.6.5.1 Design Conformance to Criterion 64

Appropriate means have been provided for monitoring radioactivity releases resulting from normal and anticipated operational occurrences.

A fission product monitoring system is provided to sample the containment atmosphere, including the drywell and suppression chamber, for radioactive particulates, noble gases, and iodine during normal operation.

Means are provided to monitor radioactive effluent discharge paths and the site environment for radioactivity released.

For further information, see the following sections:

1. Detection of leakage through RCPB - Section 5.2.5

2. ESF systems - Section 7.3

3. All other instrumentation systems required for safety - Section 7.6

4. Control systems not required for safety - Section 7.7

5. Radioactive waste management - Section 11

6. Airborne radioactivity monitoring - Section 12.3.

3.2 CLASSIFICATION OF STRUCTURES, COMPONENTS, AND SYSTEMS

3.2.1 Seismic Classification

General Design Criterion (GDC) 2, of Appendix A to 10CFR50, requires 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 necessary safety functions. Appendix A to 10CFR100 requires that all nuclear power plants be designed so that, if a safe shutdown earthquake (SSE) occurs, certain structures, systems, and components important to safety remain functional. These plant features are those necessary to ensure:

1. The integrity of the reactor coolant pressure boundary (RCPB)

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

3. The capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the guideline exposures of 10CFR50.67.

Regulatory Guide 1.29, Revision 3, describes an acceptable method for identifying and classifying those plant features that should be designed to withstand the effects of an SSE.

Plant structures, systems, and components, including their foundations and supports, designed to remain functional in the event of an SSE, are designated as Seismic Category I, as indicated in Table 3.2-1.

Those portions of structures, systems, or components whose continued function is not required but whose failure could reduce the functioning of any Seismic Category I plant feature to an unacceptable safety level are verified so that an SSE would not

cause such a failure. These items are classified as Seismic Category II/I, as indicated in Table 3.2-1.

The boundaries of each Seismic Category I portion of the systems are shown on the piping and instrument diagrams (P&IDs) in the appropriate sections of the FSAR. A cross reference of system to FSAR figure number is provided in Table 1.7-2. The Seismic Category I boundaries are indicated by the Q-flags.

GE supplied analyses, design, and/or equipment in this facility are in compliance with the intent of Regulatory Guide 1.29, which describes an acceptable method of identifying and classifying those features of light water cooled nuclear power plants that should be designed to withstand the effects of the SSE. It is used as a basis for identifying the systems and components that must meet Seismic Category I requirements.

The seismic classifications of Non-nuclear Steam Supply System (NSSS) analysis, design, and/or equipment indicated in Table 3.2-1 meet the intent of Regulatory Guide 1.29, except as noted in Section 1.8.1.29.

3.2.2 System Quality Group Classifications

System quality group classifications, as defined in Regulatory Guide 1.26, have been determined for components containing, water, steam, and radioactive waste in fluid systems relied upon to:

1. Prevent or mitigate the consequences of accidents and malfunctions originating within the reactor coolant pressure boundary (RCPB)

2. Permit shutdown of the reactor and maintain it in the safe shutdown condition

3. Contain radioactive material.

A tabulation of quality group classification for each component is shown in Table 3.2-1. Interfaces between components and piping of different classifications are indicated on the system piping and instrumentation diagrams (P&IDs), which are found in pertinent sections of the FSAR. For information on instrument and electrical equipment classification, see Section 3.10. A cross-reference of system to FSAR figure number is provided in Table 1.7-2. The code requirements applicable to each quality group classification are identified in Tables 3.2-2 and 3.2-3. Quality group classifications have been maintained during design and construction and are actively maintained during plant operations and system modifications commensurate with the safety functions performed by the safety-related components, except where later requirements allow alternative quality group classifications. Table 3.2-2 is intended to indicate design basis minimum code requirements for general categories of NSSS components. Code requirement information for specific components is provided in Table 3.2-1.

The plant design complies with Regulatory Guide 1.26, with clarifications as discussed in Section 1.8.

Portions of the Radioactive Waste Management System which are within the boundaries delineated by the direction of the "R" flags shown in the figures of Chapters 9 and 11, including piping, valves, vessels, tanks, and equipment, are classified as quality group R. Quality assurance controls for group R items comply with applicable requirements of Regulatory Guide 1.143. The plant design complies with Regulatory Guide 1.143, with clarifications as discussed in Section 1.8.1.143.

3.2.2.1 SRP Rule Review

In SRP Section 3.2.2, Subsection II, reference is made to Regulatory Guide 1.26 for determining quality group classifications of components that are important to safety. Section A and B of this guide imply that all components under the quality groups shown are safety-related, including those listed under Quality Group D.

On HCGS, the Quality Group D items are not considered to be "safety-related" or "important to safety" in the same sense that these terms are used in other guides and regulations. All Quality Group D items are only subject to Section VIII of the ASME B&PV Code, or to other non-nuclear industrial codes and standards. For this reason, the quality assurance criteria of 10 CFR 50, Appendix B, are not considered for these items.

See Table 3.2-1 for a detailed discussion of the implementation of the requirements of Regulatory Guide 1.26 at the Hope Creek Generating Station.

The Acceptance Criteria of SRP 3.2.2 requires that Regulatory Guide 1.26 be used for establishing a quality group classification system for Quality Group B, C, and D components.

At HCGS, certain components, as referenced in Table 3.2-1, are classified as Quality Group B, C, or D and do not meet the applicable quality group standards as specified in Regulatory Guide 1.26. The components purchased in this period were designed and fabricated to the required codes and standards in effect at that time.

TABLE 3.2-1

HCGS CLASSIFICATION OF STRUCTURES, SYSTEMS, AND COMPONENTS

Principal Components(57)	FSAR Section	Source of Supply (1)	Location (2)	Quality Group Classification (3)	Principal Construction Codes and Standards (5)	Seismic Category (6)	QA Requirements (7)	Comments
I. <u>Reactor System</u>	4.1							
a. Reactor vessel and head		GE	A	A	III-A(9)	I	Y	(9)(66)
b. Reactor vessel support skirt		GE	A	NA	III-A(9)	I	Y	(9)
c. Reactor vessel appurtenances, pressure retaining portions		GE	A	A	III-A(9)	I	Y	
d. CRD housing supports		GE	A	NA	AISC	I	Y	
e. Reactor internal structures, engineered safety features		GE	A	NA	None	I	Y	(13)
f. Reactor internal structures, other		GE	A	NA	None	NA	N	(13)(55)
g. Control rods		GE/ABB	A	NA	None	I	Y	
h. Control rod drives		GE	A	NA	III-A(9)	I	Y	
i. Core support structure		GE	A	NA	None	I	Y	
j. Power range detector hardware		GE	A	B	III-2	I	Y	(10)
k. Fuel assemblies		GE/ABB	A	NA	None	I	Y	
l. Reactor vessel stabilizer		GE	A	NA	III-NF	I	Y	
II. <u>Nuclear Boiler System</u>	5.1							
a. Vessels, level instrumentation condensing chambers		GE	A	A	III-1	I	Y	
b. Vessels, air accumulators		P	A,C	C	III-3	I	Y	
c. Air supply check valves and piping downstream of air supply check valves		P	A,C	C	III-3	I	Y	
d. Piping, safety relief valve discharge		P	A	C	III-3	I	Y	
e. Piping, main steam, within outboard isolation valves		GE/P	A,C	A	III-1	I	Y	
f. Piping, feedwater, within outboard isolation valves		P	A,C	A	III-1	I	Y	
g. Piping, main steam, between outboard and outermost isolation valves		P	C	B	III-2	I	Y	(13)

TABLE 3.2-1 (Cont)

Principal Components[57]	FSAR Section	Source of Supply (1)	Location (2)	Quality Group Classification (3)	Principal Construction Codes and Standards (5)	Seismic Category (6)	QA Requirements (7)	Comments
h. Piping, feedwater, between outboard and outermost isolation valves		P	C	B	III-2	I	Y	(13)
i. Pipe supports, main steam, within outboard isolation valves		GE/P	A	NA	III-NF	I	Y	
j. Piping, other, within outermost isolation valves		P	A,C	A	III-1	I	Y	(10)
k. Piping, instrumentation beyond outermost isolation valves		P	C	B	III-2	I	Y	(10)
l. Safety/relief valves		GE	A	A	III-1	I	Y	
m. Valves, main steam isolation valves		GE	A,C	A	P&V-I	I	Y	(48)
n. Valves, feedwater isolation		P	A,C	A	III-1	I	Y	(48)
o. Valves, main steam outermost isolation (main steam stop valves)		P	C	B	III-2	I	Y	(69)
p. Valves, feedwater outermost isolation		P	C	B	III-2	I	Y	(13)(48)
q. Valves, other, isolation valves and within		P	A,C	A	III-1	I	Y	(10)(48)
r. Valves, instrumentation beyond outermost isolation valves		P	C	B	III-2	I	Y	(10)(48)
s. Mechanical modules with safety function[27]		GE	C	NA	None	I	Y	
t. Electrical modules with safety function[27]		GE	C	NA	IEEE-279/323	I	Y	
u. Cable with safety function		P	A,C	NA	IEEE-279/323	NA	Y	(15)
v. Pipe whip restraints, main steam		P	A,C	NA	AISC	I	Y	(41)
w. Quenchers		P	A	C	III-3	I	Y	
x. Quencher supports		P	A	C	III-3/III-NF	I	Y	
III. <u>Recirculation System</u>	5.4.1							
a. Piping		GE	A	A	III-1/B31.7	I	Y	(10)(53)
b. Pipe supports, recirculation line		GE/P	A	NA	III-NF	I	Y	
c. Pipe whip restraints, recirculation line		GE/P	A	NA	AISC	I	Y	(41)
d. Pumps		GE	A	A	P&V-I	I	Y	(13)
e. Valves		GE	A	A	P&V-I	I	Y	(10)(48)
f. Pump motors		GE	A,C	NA	None	I	Y	

TABLE 3.2-1 (Cont)

Principal Components (57)	FSAR Section	Source of Supply (1)	Location (2)	Quality Group Classification (3)	Principal Construction Codes and Standards (5)	Seismic Category (6)	QA Requirements (7)	Comments
g. Electrical modules with safety function (27)		GE	A	NA	IEEE-279/323	I	Y	
h. Cable with safety function		P	A,C	NA	IEEE-279/323	NA	Y	(15)
IV. CRD Hydraulic System	4.6.1							
a. Piping and valves, reactor building penetration		P	C	D	B31.1.0	II/I		(50)
b. Valves, scram discharge volume lines		P/GE	C	B	III-2	I	Y	(10)
c. Valves, insert and withdraw lines		P/GE	A,C	B	III-2	I	Y	(8)(48)
d. Valves, other		P/GE	C	D	B31.1.0	NA	N	
e. Pipe cap, water return line		GE	A	A	III-1	I	Y	
f. Piping, scram discharge volume lines		P	C	B	III-2	I	Y	
g. Piping, insert and withdraw lines		P	A,C	B	III-2	I	Y	
h. Piping, other		P	C	D	B31.1.0	NA	N	(13)
i. Hydraulic control unit including scram accumulator		GE	C	Special	(44)	I	Y	(12)
j. Electrical modules with safety function (27)		GE	C	NA	IEEE-279/323	I	Y	
k. Cable with safety function		P	C	NA	IEEE-279/323	NA	Y	(15)
l. Pumps		GE	C	D	None	NA	N	
m. Pump motors		GE	C	NA	None	NA	N	
V. Engineered Safety Features								
a. RHR System:	6.3/5.4.7							
1. Heat exchangers, primary side (shutdown cooling, suppression pool cooling)		GE	C	B	III-C & TEMA C (9)	I	Y	
2. Heat exchangers, secondary side		GE	C	C	VIII-1 TEMA C (9)	I	Y	
3. Piping, within outermost containment isolation valves (LPCI, shutdown cooling)		P	C,A	A	III-1	I	Y	(10)

Principal Components[57]	FSAR Section	Source of Supply (1)	Location (2)	Quality Group Classification (3)	Principal Construction Codes and Standards (5)	Seismic Category (6)	QA Requirements (7)	Comments
4. Piping, beyond outermost containment isolation valves (LPCI, shutdown cooling, suppression pool cooling, containment spray)		P	C	B	III-2	I	Y	(10)
5. Piping and spray nozzles, containment spray lines within outermost isolation valves		P	A	B	III-2	I	Y	
6. Deleted								
7. Pumps (LPCI, shutdown cooling, suppression pool cooling, containment spray)		GE	C	B	P&V-II[9]	I	Y	
8. Pump motors		GE	C	NA	NEMA MG-1	I	Y	
9. Valves, inboard isolation, LPCI line & shutdown return line		GE	A	A	III-1	I	Y	(48)
10. Valves, isolation and within (shutdown suction)		P	C,A	A	III-1	I	Y	(10)(48)
11. Valves, beyond isolation valves (LPCI, shutdown cooling, suppression pool cooling, containment spray)		P	C	B	III-2	I	Y	(10)(48)
12. Mechanical modules with safety function[27]		GE	C	NA	None	I	Y	
13. Electrical modules with safety function[27]		GE	C	NA	IEEE-279/323	I	Y	
14. Cable with safety function		P	C	NA	IEEE-279/323	NA	Y	(15)
15. ECCS jockey pumps		P	C	B	III-2	I	Y	
16. Piping and valves, reactor building penetration and isolation		P	C	D	B31.1.0	II/I		(50)
17. ECCS jockey pump motors		P	C	NA	IEEE-323/344	I	Y	
b. Core Spray System:	6.3							
1. Piping, within outermost isolation valves		P	A,C	A	III-1	I	Y	(10)

Principal Components[57]	FSAR Section	Source of Supply (1)	Location (2)	Quality Group Classification (3)	Principal Construction Codes and Standards (5)	Seismic Category (6)	QA Requirements (7)	Comments
2. Piping, beyond outermost isolation valves		P	C	B	III-2	I	Y	(10)
3. Pumps		GE	C	B	P&V-II[9]	I	Y	(48)
4. Pump motors		GE	C	NA	NEMA MG-1[9]	I	Y	(48)
5. Valves, inboard isolation		GE	A	A	III-1	I	Y	(48)
6. Valves, outboard isolation and within		P	C	A	III-1	I	Y	
7. Valves, beyond outermost containment isolation valves		P	C	B	III-2	I	Y	(10)
8. Electrical modules with safety function[27]		GE	A,C	NA	IEEE-279/323	I	Y	
9. Cable with safety function		P	A	NA	IEEE-279/323	NA	Y	(15)
10. ECCS jockey pump		P	C	B	III-2	I	Y	
11. ECCS jockey pump motors		P	C	NA	IEEE-323/344	I	Y	
c. High Pressure Coolant Injection (HPCI) System:	6.3							
1. Piping, within outermost containment isolation valves		P	A,C	A	III-1	I	Y	(10)
2. Piping, test return line to condensate storage tank up to second isolation valve		P	C	B	III-2	I	Y	
3. Pumps (main and booster)		GE	C	B	P&V-II[9]	I	Y	(11)(59)
4. HPCI turbine		GE	C	NA	VIII-1	I	Y	
5. HPCI barometric condenser		GE	C	NA	(44)	NA	N	
6. HPCI vacuum pump & condensate pump		GE	C	NA	None	NA	N	
7. Vacuum pump & condensate pump motors		GE	C	NA	None	NA	N	
8. Piping, valve leakoff and cooling lines to barometric condenser		P	C	B	III-2	I	Y	
9. Piping, other		P	C	B	III-2	I	Y	(10)(48)
10. Valves, containment isolation and within		P	A,C	A	III-1	I	Y	(10)(48)
11. Valves, other		P	C	B	III-2	I	Y	(10)(48)
12. Electrical modules with safety function		GE	C	NA	IEEE-279/323	I	Y	(27)(60)

Principal Components (57)	FSAR Section	Source of Supply (1)	Location (2)	Quality Group Classification (3)	Principal Construction Codes and Standards (5)	Seismic Category (6)	QA Requirements (7)	Comments
13. Electrical auxiliary equipment		GE	C	NA	None	I	Y	
14. Cable with safety function		P	A,C,O	NA	IEEE-279/323	NA	Y	(15)
15. ECCS jockey pump		P	C	B	III-2	I	Y	
16. ECCS jockey pump motor		P	C	NA	IEEE-323/344	I	Y	
d. Containment Atmosphere Control System:	6.2.5							
1. Piping and valves, containment penetration and isolation		P	A,C	B	III-2	I	Y	(48)
2. Containment/drywell monitoring								
a. H /O analyzer		P	C	B	III-2	I	Y	(10)
b. H /O analyzer bottle station supply lines		P	C,O	NA	B31.1.0	II/I	N	(10)(71)
3. Piping and valves, reactor building penetrations and isolation		P	C	C	III-3	I	Y	
4. Nitrogen system (containment inerting):								
a. Vessels		P	O	D	VIII-1	NA	N	
b. Piping & valves, reactor building penetration & isolation		P	C,R	D	B31.1.0	II/I		(50)
c. Piping & valves, other		P	O,C	D	B31.1.0	NA	N	
d. Heat exchangers		P	R	D	VIII-1	NA	N	
5. Containment Hydrogen Recombiner System:								
a. Motors		P	C	NA	NEMA MG-1	I	Y	
b. Blowers		P	C	NA	None	I	Y	
c. Reaction chambers and spray cooler		P	C	B	III-2	I	Y	
d. Hydrogen recombiner heaters		P	C	NA	NEMA IEEE-279/323	I	Y	
e. Deleted								
f. Deleted								
g. Piping, containment penetration	P	A	B	III-2	I	Y		
h. Valves, containment isolation		P	C	B	III-2	I	Y	(48)
i. Piping and valves, other		P	C	B	III-2	I	Y	(48)

Principal Components[57]	FSAR Section	Source of Supply (1)	Location (2)	Quality Group Classification (3)	Principal Construction Codes and Standards (5)	Seismic Category (6)	QA Requirements (7)	Comments
e. Primary Containment Leakage Rate Testing System:	6.2.6							
1. Piping and valves, containment penetration & isolation		P	C	B	III-2	I	Y	(48)
VI. <u>Reactor Core Isolation Cooling (RCIC) System:</u>	5.4.6							
1. Piping, within outermost containment isolation valves		P	A,C	A	III-1	I	Y	(10)
2. Piping, beyond outermost containment isolation valves		P	C	B	III-2	I	Y	(10)
3. Piping, test return to condensate storage tank up to second isolation valve		P	C	B	III-2	I	Y	
4. Piping, valve leakoff & cooling lines to barometric condenser		P	C	B	III-2	I	Y	
5. RCIC pump		GE	C	B	P&V-II[9]	I	Y	
6. RCIC barometric condenser		GE	C	NA	VIII-1[9]	NA	N	
7. RCIC condensate pump and vacuum pump		GE	C	NA	None	NA	N	
8. Condensate and vacuum pump motors		GE	C	NA	None	NA	N	
9. Valves, containment isolation and within		P	A	A	III-1	I	Y	(10)(48)
10. Valves, other		P	C	B	III-2	I	Y	(10)(48)
11. RCIC turbine		GE	C	NA	(44)	I	Y	(11)(59)
12. Electrical modules with safety function		GE	C	NA	IEEE-279/323	I	Y	(27)(60)
13. Cable with safety function		P	C	NA	IEEE-279/323	NA	Y	(15)

Principal Components(57)	FSAR Section	Source of Supply (1)	Location (2)	Quality Group Classification (3)	Principal Construction Codes and Standards (5)	Seismic Category (6)	QA Requirements (7)	Comments
14. ECCS Jockey pump		P	C	B	III-2	I	Y	
15. ECCS jockey pump motor		P	C	NA	IEEE-323/344	I	Y	
VII. Reactor Water Cleanup System (RWCU)	5.4.8							
1. Vessels, filter/demineralizer		GE	C	C	III-3	NA	N	
2. Heat exchangers, nonregenerative, reactor water side		GE	C	C	III-C/TEMA R	II/I(9)		(50)
3. Heat exchangers, nonregenerative, cooling water side		GE	C	D	VIII-1/TEMA R	II/I(9)		(50)
4. Heat exchanger, regenerative		GE	C	C	III-C/TEMA R	II/I(9)		(50)
5. Piping, within outermost isolation valves		P/GE	A,C	A	III-1	I	Y	(10)
6. Piping, between outermost feedwater isolation valve and flow element		P/GE	C	C	III-3	I	Y	(10)
7. Piping, beyond outermost isolation valves or beyond flow element		P/GE	C	C	III-3	II/I		(10)(50)
8. Pumps		P/GE	C	C	P&V-III(9)	II/I		(50)
9. Pumps, filter/demineralizer		GE	C	C	P&V-III(9)	NA	N	
10. Valves, isolation and within		P	A,C	A	III-1	I	Y	(10)(48)
11. Valves, beyond isolation valves		P/GE	C	C	III-3/P&V-III	II/I(9)		(50)
12. Valves, filter/demineralizer		P/GE	C	C	III-3/P&V-III	NA(9)	N	
13. Mechanical modules(27)		GE	C	NA	None	II/I		(50)
14. Piping, Reactor Building penetration		P	C,R	D	B31.1.0	II/I		(50)
15. Valves, Reactor Building isolation		P	C	D	B31.1.0	II/I		(50)
16. Cable with safety function		P	C	NA	IEEE-279/323	NA	Y	(15)
17. Tank, precoat, filter/demineralizer		GE	C	C	API-650	NA	N	
18. Electrical modules with safety function(27)		GE	C	NA	IEEE-279/323	I	Y	

TABLE 3.2-1 (Cont)

Principal Components(57)	FSAR Section	Source of Supply (1)	Location (2)	Quality Group Classification (3)	Principal Construction Codes and Standards (5)	Seismic Category (6)	QA Requirements (7)	Comments
VIII. <u>Fuel Handling and Storage</u>								
a. Fuel servicing equipment	9.1.4							
1. Fuel preparation machines		GE	C	NA	None	I	Y	
2. General purpose grapples		GE	C	NA	None	I	Y	
3. Jib crane		P	C	NA	None	II/I		(50)
4. ABB J-Hook		ABB	C	NA	None	I	Y	
b. Reactor vessel servicing equipment	9.1.4							
1. Steam line plugs		P/GE	C	NA	None	NA	Y	
2. Dryer - separator sling		GE	C	NA	None	NA	Y	(23)(38)(42)
3. RPV and drywell head strongback		GE	C	NA	None	NA	Y	(23)(38)(42)
4. Reactor building polar crane	9.1.5	P	C	NA	(20)	I	Y	
c. In-vessel service equipment	9.1.4							
1. Control rod grapple		GE or NES	C	NA	None	NA	Y	(23)(39)(42)
2. Combined CRB/FSP Grapple		GE	C	NA	None	NA	Y	(23)(39)(42)
d. Refueling equipment	9.1.4							
1. Refueling platform		GE	C	NA	AISC	I	Y	
2. Shielded fuel transfer chute	9.1.4	GE	C	NA	None	II/I	Y	(50)
3. Shielded fuel transfer chute strongback	9.1.4	P/GE	C	NA	None	NA	Y	(23)(38)(42)
e. Storage equipment	9.1							
1. Spent fuel/defective fuel storage racks		P	C	NA	III-NF	I	Y	
2. Defective fuel storage containers		P	C	NA	III-NF	I	Y	(43)

TABLE 3.2-1 (Cont)

Principal Components(57)	FSAR Section	Source of Supply (1)	Location (2)	Quality Group Classification (3)	Principal Construction Codes and Standards (5)	Seismic Category (6)	QA Requirements (7)	Comments
3. New fuel storage rack		GE	C	NA	None	I	Y	
4. In-vessel racks		GE	C	NA	None	I	Y	
5. Channel storage racks		GE	C	NA	None	NA	Y	
f. Undervessel service equipment	9.1.4							
1. Equipment handling platform		GE	A	NA	None	NA	N	

Principal Components (57)	FSAR Section	Source of Supply (1)	Location (2)	Quality Group Classification (3)	Principal Construction Codes and Standards (5)	Seismic Category (6)	QA Requirements (7)	Comments
IX. <u>Fuel Pool Cooling and Cleanup System and Torus Water Cleanup System</u>:	9.1.3							
a. Vessels, filter/demineralizer		P	R	D	VIII-1	NA	N	
b. Precoat tank		P	R	D	API-650	NA	N	
c. Heat exchangers		P	C	C	III-3	I	Y	
d. Fuel pool cooling pumps		P	C	C	III-3	I	Y	
e. Valves and piping, cooling loop		P	C	C	III-3	I	Y	(48)
f. Valves, other		P	C,R	D	B31.1.0	NA	N	
g. Piping, makeup		P	C	C	III-3	I	Y	
h. Skimmer surge tanks		P	C	D	API-650	NA	N	(46)
i. Piping, other		P	C	D	B31.1.0	NA	N	
j. Torus water cleanup pump, fuel pool cleanup holding pumps		P	R	D	(44)	NA	N	
k. Piping and valves, torus water cleanup containment penetration and isolation		P	C	B	III-2	I	Y	
l. RHR connection (emergency cooling)		P	C	B	III-2	I	Y	
m. Fuel pool cooling pump motors		P	C	NA	IEEE-323/344	I	Y	
n. Valves and piping, reactor building isolation and penetration		P	C	D	B31.1.0	II/I		(50)
X. <u>Radioactive Waste Systems</u>								
a. Liquid Radwaste System:	11.2							
1. Spent resin storage tank		P	R	R	API-620	NA	N	(22)
2. Tanks, atmospheric		P	R	R	API-650	NA	N	(22)
3. Heat exchangers		P	R	R	VIII-1/TEMA C	NA	N	(22)
4. Piping		P	C,R	R	B31.1.0	NA	N	(10) (22)
5. Pumps		P	R	R	B31.1.0/ Hyd.I	NA	N	(22) (24)
6. Valves		P	C,R	R	B31.1.0	NA	N	(10) (22)
7. Vessels		P	R	R	VIII-1	NA	N	(22)
8. Waste evaporator (68)		GE	R	R	VIII-1	NA	N	(22)
9. Mechanical modules		P/GE	R	R	B31.1.0	NA	N	(22)
10. Instrument and control boards		GE	R	NA	NEMA12	NA	N	(22)
11. Recontamination solution evaporator		GE	R	R	VIII-1	NA	N	(22)

TABLE 3.2-1 (Cont)

Principal Components[57]	FSAR Section	Source of Supply (1)	Location (2)	Quality Group Classification (3)	Principal Construction Codes and Standards (5)	Seismic Category (6)	QA Requirements (7)	Comments
12. Valves, flow control and filter system		P/GE	R	R	B31.1.0	NA	N	(22)
13. Piping and valves, reactor building penetration and isolation		P	C	D	B31.1.0	II/I		(50)
b. Gaseous Radwaste System:	11.3							
1. Tank, atmospheric		P	R	R	API-650	NA	N	(22)
2. Heat exchangers		P	R	R	VIII-1/ TEMA C	NA	N	(22)
3. Piping		P	R	R	B31.1.0	NA	N	(10) (22)
4. Valves, flow control		P	R	R	B31.1.0	NA	N	(22)
5. Valves, other		P	R	R	B31.1.0	NA	N	(10)
6. HEPA filters		P	R	R	VIII-1	NA	N	(22)
7. Adsorber units		P	R	R	VIII-1	NA	N	(22)
8. Charcoal guard bed		P	R	R	VIII-1	NA	N	(22)
c. Solid Radwaste System:	11.4							
1. Piping		P	R	R	B31.1.0	NA	N	(22)
2. Valves		P	R	R	B31.1.0	NA	N	(22)
3. Pumps		P	R	R	B31.1.0/ Hyd.I/(44)	NA	N	(22)(24)
4. Tanks, atmospheric		P	R	R	API-650/ D100/VIII-1	NA	N	(22)
5. Vessels		P	R	R	VIII-1	NA	N	(22)
6. Compressors		P	R	R	(44)	NA	N	(22)
7. Blowers		P	R	R	(44)	NA	N	(22)
8. Piping and valves, reactor building penetration and isolation		P	D	D	B31.1.0	II/I		(50)
d. Process and Effluent Radiological Monitoring and Sampling System:	11.5							
1. Main steam line RMS		GE/P	C	NA	IEEE-323/344	I	Y	

Principal Components (57)	FSAR Section	Source of Supply (1)	Location (2)	Quality Group Classification (3)	Principal Construction Codes and Standards (5)	Seismic Category (6)	QA Requirements (7)	Comments
2. Gaseous effluent stream monitors								
a) Refueling floor exhaust RMS		P	C	NA	IEEE-323/344	I	Y	
b) Reactor building exhaust RMS		P	C	NA	IEEE-323/344	I	Y	
3. Control room ventilation RMS		P	B	NA	IEEE-323/344	I	Y	
4. Drywell atmosphere post-accident RMS		P	A	NA	IEEE-323/344	I	Y	
XI. Water Systems								
a. Station Service Water System:	9.2.1							
1. Emergency cross-connect piping to RHR system		P	C	C	III-3	I	Y	
2. Piping and valves, chemical treatment		P	W	D	B31.1.0	NA	N	
3. Piping, safety-related, other		P	O,C	C	III-3	I	Y	
4. Service water pumps		P	W	C	III-3	I	Y	
5. Pump motors		P	W	NA	IEEE-323/344	I	Y	
6. Valves, isolation		P	C,W	C	III-3	I	Y	(48)
7. Valves, other		P	C	C	III-3	I	Y	(48)
8. Electrical modules with safety function (27)		P	W	NA	IEEE-279/ 323/344	I	Y	
9. Traveling screens		P	W	C	None	I	Y	
10. Trash racks		P	W	NA	None	II/I		(50)
11. Trash rake		P	W	NA	None	NA	N	
12. Cable with safety function		P	W	NA	IEEE-279/323	NA	Y	(15)
13. Piping and valves, reactor building penetration and isolation		P	C	C	III-3	I	Y	(48)
b. Safety Auxiliaries Cooling System (SACS):	9.2.2							
1. Piping and valves, reactor building penetration and isolation		P	C	C	III-3	I	Y	(48)
2. Piping and valves, other		P	C,G	C	III-3	I	Y	(48)

Principal Components (57)	FSAR Section	Source of Supply (1)	Location (2)	Quality Group Classification (3)	Principal Construction Codes and Standards (5)	Seismic Category (6)	QA Requirements (7)	Comments
3. Expansion tanks		P	C	C	III-3	I	Y	
4. Heat exchangers		P	C	C	III-3/TEMA R	I	Y	
5. Pumps		P	C	C	III-3	I	Y	
6. Pump motors		P	C	C	NEMA MG-1	I	Y	
7. Hydropneumatic accumulators (operated Water solid)		P	G	C	III-3	I	Y	
8. Electrical modules with safety function		P	C	NA	IEEE-279/323	I	Y	
c. Reactor Auxiliaries Cooling System (RACS):	9.2.8							
1. Piping and valves forming part of containment boundary		P	A,C	B	III-2	I	Y	(48)
2. Piping and valves, reactor building penetration and isolation		P	C	D	B31.1.0	II/I		(50)
3. Piping and valves, other		P	A,C,R	D	B31.1.0	NA	N	(13)
4. Heat exchangers		P	C	D	VIII-1/ TEMA R	NA	N	
5. Pumps		P	C	D	B31.1.0/ Hyd.I	NA	N	(24)
6. Expansion tank		P	C	D	API-620	NA	N	
d. Turbine Auxiliaries Cooling System (TACS):	9.2.2							
1. Piping and valves		P	T	D	B31.1.0	NA	N	
e. Condensate and Refueling Water Storage and Transfer System:	9.2.6							
1. Tank, condensate storage		P	O	D	D100	NA	N	
2. Piping and valves, reactor building penetration and isolation		P	C	D	B31.1.0	II/I		(50)
3. Pumps		P	T	D	Hyd.I	NA	N	(24)
4. Piping and valves, HPCI, RCIC, and core spray pump suction		P	C	B	III-2	I	Y	(48)

Principal Components[57]	FSAR Section	Source of Supply (1)	Location (2)	Quality Group Classification (3)	Principal Construction Codes and Standards (5)	Seismic Category (6)	QA Requirements (7)	Comments
5. Piping and valves, HPCI, RCIC, and CRD return line		P	O,C	C	III-3	I	Y	(48)
6. Piping and valves, level instrumentation		P	O,C	C	III-3	I	Y	(48)
7. Piping and valves, dike penetrations		P	O	C	III-3	I	Y	(48)
8. Piping and valves, other		P	O,T,R	D	B31.1.0	NA	N	
f. Turbine Building Chilled Water System:	9.2.7.1							
1. Tanks		P	T	D	VIII-1	NA	N	
2. Chillers		P	T	D	VIII-1	NA	N	
3. Pumps		P	T	D	VIII-1/Hyd.I	NA	N	(24)
4. Piping & valves, containment penetration & isolation		P	A,C	B	III-2	I	Y	(48)
5. Piping & valves, reactor building penetration & isolation		P	C	D	B31.1.0	II/I		(50)
6. Piping, other		P	T,A,C,R	D	B31.1.0	NA	N	
7. Valves, other		P	T,A,C,R	D	B31.1.0	NA	N	
8. Cooling coils		P	T,A,C	NA	ARI-410	NA	N	
9. Motors		P	T	NA	NEMA MG-1	NA	N	
10. Side Stream Filters		P	T	D	VIII-1	NA	N	(14)
11. Carbon Filter		P	T	D	VIII-1	NA	N	
12. Mixed-Bed Demineralizer		P	T	D	VIII-1	NA	N	
g. Auxiliary Building Control Area Chilled Water System:	9.2.7.2							
1. Chillers		P	B,G	C	III-3	I	Y	
2. Cooling coils		P	B,C,G	C	III-3/ARI-410	I	Y	
3. Pumps		P	B,G	C	III-3	I	Y	
4. Motors		P	B,G	NA	IEEE-323/344	I	Y	
5. Piping and valves		P	B,G,C	C	III-3	I	Y	(14)(48)
6. Tank, head		P	G	C	VIII-1	I	Y	
h. Potable & Sanitary Water System:	9.2.4							
1. Pumps		P	O,G,B,R	D	B31.1.0/ Hyd.I	NA	N	(24)

Principal Components (57)	FSAR Section	Source of Supply (1)	Location (2)	Quality Group Classification (3)	Principal Construction Codes and Standards (5)	Seismic Category (6)	QA Requirements (7)	Comments
2. Motors		P	O,G,B,R	NA	NEMA MG-1	NA	N	
3. Piping and valves		P	O,G,B,R	D	B31.1.0/NSPC	NA	N	
i. Demineralized Water Makeup Storage & Transfer System:	9.2.3							
1. Tanks		P	T	D	API-620	NA	N	(24)
2. Pumps		P	T	D	Hyd.I	NA	N	
3. Motors		P	T	NA	NEMA MG-1	NA	N	(50)
4. Piping and valves, reactor building penetration & isolation		P	C	D	B31.1.0	II/I		
5. Piping and valves, other		P	All	D	B31.1.0	NA	N	
XII. <u>Standby Diesel Generator and Auxiliary Systems</u>								
a. Fuel Oil Storage and Transfer System:	9.5.4							
1. Storage tanks		P	G	C	III-3, N195	I	Y	
2. Day tanks		P	G	C	III-3, N195	I	Y	(48)
3. Piping and valves, fuel oil system		P	G	C	III-3, N195	I	Y	
4. Pumps, motor-driven fuel oil transfer		P	G	C	III-3, N195	I	Y	
5. Motors, motor-driven fuel oil transfer pump and standby fuel oil pump		P	G	NA	IEEE-323/344	I	Y	
6. Pump, motor-driven standby fuel oil		P	G	C	III-3	I	Y	
7. Strainers, fuel oil		P	G	C	III-3	I	Y	
8. Pump, engine-driven fuel oil		P	G	NA	Hyd.I	I	Y	
b. Diesel generators		P	G	NA	IEEE-387/(44)	I	Y	
c. Electrical modules with safety functions (27)		P	G	NA	IEEE-279/323	I	Y	
d. Cable with safety functions		P	G	NA	IEEE-279/323	NA	Y	(15)

Principal Components(57)	FSAR Section	Source of Supply (1)	Location (2)	Quality Group Classification (3)	Principal Construction Codes and Standards (5)	Seismic Category (6)	QA Requirements (7)	Comments
e. Lubrication System:	9.5.7							
1. Piping and valves		P	G	C	III-3	I	Y	(48)
2. Heat exchangers		P	G	C	III-3	I	Y	
3. Lube oil filter housing		P	G	C	III-3	I	Y	
4. Lube oil strainer housing		P	G	C	III-3	I	Y	
5. Heater		P	G	C	III-3	I	Y	
6. Lube oil makeup tanks		P	G	C	III-3	I	Y	
7. Pump, motor driven prelube keepwarm		P	G	C	III-3	I	Y	
8. Pump motor driven prelube		P	G	C	III-3	I	Y	
9. Pump, engine driven lube oil		P	G	NA	(44)	I	Y	
10. Suction strainer, engine driven lube oil pumps		P	G	NA	(44)	I	Y	
11. Motors, prelube/keepwarm and prelube pumps		P	G	NA	IEEE-323/344	I	Y	
f. Starting and Control Air System:	9.5.6							
1. Piping and valves from receiver to diesel and from receiver to receiver inlet check valve		P	G	C	III-3	I	Y	(48)
2. Piping and valves, other		P	G	D	B31.1.0	NA	N	
3. Receiver tanks		P	G	C	III-3	I	Y	
4. Compressor, starting air		P	G	NA	None	II/I		(50)
5. Dryer, starting air		P	G	NA	None	II/I		(50)
g. Cooling Water System:	9.5.5							
1. Expansion tanks		P	G	C	III-3	I	Y	
2. Heat exchangers		P	G	C	III-3	I	Y	(48)
3. Piping and valves		P	G	C	III-3	I	Y	
4. Pump, jacket water		P	G	NA	(44)	I	Y	
5. Pump, jacket water keep warm		P	G	C	III-3	I	Y	

TABLE 3.2-1 (Cont)

Principal Components[57]	FSAR Section	Source of Supply (1)	Location (2)	Quality Group Classification (3)	Principal Construction Codes and Standards (5)	Seismic Category (6)	QA Requirements (7)	Comments
h. Combustion Air Intake and Exhaust System:	9.5.8							
1. Piping and valves		P	G	C	III-3	I	Y	(48)
1. Intake and exhaust silencers		P	G	NA	(44)	I	Y	
XIII. <u>Heating, Ventilating, and Air Conditioning Systems</u>								
a. Main Control Room and Control Building HVAC Systems:	9.4.1							
1. Control Room Supply System, Control Area Exhaust System, and Control Room Emergency Filter System:								
a) Motors		P	B	NA	IEEE-323/344	I	Y	
b) Fans		P	B	NA	AMCA	I	Y	
c) Prefilters & afterfilters		P	B	NA	UL 900	I	Y	
d) HEPA filters		P	B	NA	HSI-306	I	Y	
e) Adsorber units		P	B	NA	CS-8T	I	Y	
f) Valves/dampers, isolation		P	B	C/NA	III-3/AMCA	I	Y	(48)
g) Dampers, flow distribution		P	B	NA	AMCA	I	Y	
h) Duct work		P	B	NA	AISI/SMACNA	I	Y	
i) Coils, cooling		P	B	C	III-3	I	Y	
j) Coils, electric heating		P	B	NA	IEEE-323/344	I	Y	
k) Humidifier		P	B	NA	ARI	I	Y	
2. Control Equipment Room Supply System and Control Area Battery Exhaust System:								
a) Motors		P	B, G	NA	IEEE-323/344	I	Y	
b) Fans		P	B, G	NA	AMCA	I	Y	
c) Coils, cooling		P	B, G	C	III-3	I	Y	
d) Coils, heating, electric		P	B, G	NA	IEEE-323/344	I	Y	
e) Dampers		P	B, G	NA	AMCA	I	Y	
f) Duct work		P	B, G	NA	AISI/SMACNA	I	Y	

TABLE 3.2-1 (Cont)

Principal Components (57)	FSAR Section	Source of Supply (1)	Location (2)	Quality Group Classification (3)	Principal Construction Codes and Standards (5)	Seismic Category (6)	QA Requirements (7)	Comments
b. Primary Containment Ventilation Systems:	9.4.5							
1. Drywell Air Cooling System:								
a) Motors		P	A	NA	NEMA MG-1	II/I		(23)(50)
b) Fans		P	A	NA	AMCA	II/I		(23)(50)
c) Coils, cooling		P	A	NA	ARI-410	II/I		(23)(50)
d) Duct work		P	A	NA	SMACMA	II/I		(23)(50)
e) Dampers		P	A	NA	AMCA	II/I		(23)(50)
c. Reactor Building Heating, Ventilating, and Air Conditioning (HVAC) System:	9.4.2							
1. Reactor Building Ventilation System (RBVS):	9.4.2							
a) Motors		P	R	NA	NEMA MG-1	NA	N	
b) Fans		P	R	NA	AMCA	NA	N	
c) Prefilters		P	R	NA	UL 900	NA	N	
d) HEPA filters		P	R	NA	HSI-306	NA	N	
e) Coils, cooling		P	R	NA	ARI-410	NA	N	(47)
f) Duct work		P	R	NA	SMACNA	NA	N	
g) Dampers, isolation		P	R	C	AMCA	I	Y	
h) Piping & valves		P	R	D	B31.1.0	NA	N	
2. Equipment Area Cooling System (EACS):	9.4.2							
a) Motors		P	C	NA	IEEE-323/344	I	Y	
b) Fans		P	C	NA	AMCA	I	Y	
c) Duct work		P	C	NA	AISI/SMACNA	I	Y	
d) Dampers		P	C	NA	AMCA	I	Y	

TABLE 3.2-1 (Cont)

Principal Components(57)	FSAR Section	Source of Supply (1)	Location (2)	Quality Group Classification (3)	Principal Construction Codes and Standards (5)	Seismic Category (6)	QA Requirements (7)	Comments
3. Filtration, Recirculation, and Ventilation System (FRVS):	6.8							
a) Motors		P	C	NA	IEEE-323/344	I	Y	
b) Fans		P	C	NA	AMCA	I	Y	
c) Prefilters		P	C	NA	UL 900	I	Y	(16)
d) Demisters		P	C	NA	None	I	Y	(16)
e) HEPA filters		P	C	NA	HSI-306	I	Y	(16)
f) Adsorber units		P	C	NA	CS-8T	I	Y	(16)
g) Duct work		P	C	NA	AISI/SMACNA	I	Y	(16)
h) Dampers		P	C	NA	AMCA	I	Y	(16)
i) Piping and coils		P	C	C	III-3	I	Y	(16)
j) Valves		P	C	C	III-3	I	Y	(16)(48)
4. Containment Prepurge Cleanup System (CPCS):	9.4.2							
a) Piping & valves, containment penetration and isolation		P	C	B	III-2	I	Y	(48)
b) Piping and dampers, pipe chase isolation		P	C	C	III-3	I	Y	(48)
d. Auxiliary Building Service and Radwaste Area Ventilation Systems:	9.4.3							
1. Motors		P	R	NA	NEMA MG1	NA	N	
2. Fans		P	R	NA	AMCA	NA	N	
3. Prefilters		P	R	NA	UL 900	NA	N	
4. HEPA filters		P	R	NA	HSI-306	NA	N	
5. Coils, cooling & heating		P	R	NA	ARI-410	NA	N	
6. Adsorber units		P	R	NA	CS-8T	NA	N	
7. Duct work		P	R	NA	SMACNA	NA	N	
8. Dampers, isolation		P	R	NA	AMCA	NA	N	
9. Dampers, flow distribution		P	R	NA	AMCA	NA	N	

Principal Components[57]	FSAR Section	Source of Supply (1)	Location (2)	Quality Group Classification (3)	Principal Construction Codes and Standards (5)	Seismic Category (6)	QA Requirements (7)	Comments
e. Standby Diesel Generator Area Ventilation Systems (safety-related battery room exhaust, switchgear room cooling, Class 1E panel room supply, SDG room recirculation, safety-related battery room):	9.4.6							
1. Motors		P	G	NA	IEEE-323/344	I	Y	
2. Fans		P	G	NA	AMCA	I	Y	
3. Filters		P	G	NA	UL 900	I	Y	
4. Coils, cooling		P	G	C	III-3	I	Y	
5. Duct work		P	G	NA	AISI/SMACNA	I	Y	(48)
6. Valves		P	G	C	III-3	I	Y	
7. Dampers		P	G	NA	AMCA	I	Y	
f. Turbine Building Ventilation System:	9.4.4							
1. Motors		P	T	NA	NEMA MG-1	NA	N	
2. Fans		P	T	NA	AMCA	NA	N	
3. Filters		P	T	NA	UL 900	NA	N	
4. Coils, cooling & heating		P	T	NA	ARI-410	NA	N	
5. Duct work		P	T	NA	SMACNA	NA	N	
6. Dampers		P	T	NA	AMCA	NA	N	
g. Service Water Intake Structure Ventilation Systems:	9.4.7							
1. Motors		P	W	NA	IEEE-323/344	I	Y	
2. Fans		P	W	NA	AMCA	I	Y	
3. Deleted								
4. Duct work		P	W	NA	AISI/SMACNA	I	Y	
5. Dampers		P	W	NA	AMCA	I	Y	
h. Miscellaneous Structures Ventilation Systems:	9.4.8							
1. Motors		P	O	NA	NEMA MG-1	NA	N	
2. Fans		P	O	NA	AMCA	NA	N	

Principal Components(57)	FSAR Section	Source of Supply (1)	Location (2)	Quality Group Classification (3)	Principal Construction Codes and Standards (5)	Seismic Category (6)	QA Requirements (7)	Comments
3. Prefilters		P	O	NA	UL 900	NA	N	
4. Dampers		P	O	NA	AMCA	NA	N	
5. Coils, cooling & heating		P	O	NA	ARI-410	NA	N	
6. Duct work		P	O	NA	SMACNA	NA	N	
XIV. <u>Main Steam and Power Conversion Systems</u>								
a. Main Steam Supply System:	10.3							
1. Piping, main steam, from main steam stop valve to main stop valve		P	C,T	C	III-3	NA	N	(13)(14)(18)
2. Piping and valves, other, main steam		P	C,T	D	B31.1.0	NA	N	
b. Main Condenser Evacuation System:	10.4.2							
1. Piping and valves		P	T,R	D	B31.1.0	NA	N	(19)
2. Heat exchangers		P	T	D	HEI	NA	N	
3. Air ejectors		P	T	D	B31.1.0	NA	N	
4. Mechanical vacuum pumps		P	T	D	None	NA	N	
c. Feedwater and Condensate System:	10.4.7							
1. Piping and valves, other, feedwater		P	C,T	D	B31.1.0	NA	N	(13)
2. Steam piping to feedwater pump turbines								
a) Crossover (low pressure) piping		P	T	D	B31.1.0	NA	N	
b) Bypass (high pressure) piping, downstream of first isolation valve		P	T	D	B31.1.0	NA	N	
3. Heat exchangers		P	T	D	VIII-1/HEI	NA	N	
4. Pressure vessels		P	T	D	VIII-1/ API-620	NA	N	
5. Pumps, feedwater and condensate		P	T	D	Hyd.I	NA	N	(24)
6. Piping and valves, other, condensate		P	T	D	B31.1.0	NA	N	

Principal Components[57]	FSAR Section	Source of Supply (1)	Location (2)	Quality Group Classification (3)	Principal Construction Codes and Standards (5)	Seismic Category (6)	QA Requirements (7)	Comments
d. Condensate Cleanup System:	10.4.6							
1. Piping and valves		P	T	D	B31.1.0	NA	N	
2. Pressure vessels (filter/ demineralizers)		P	T	D	VIII-1	NA	N	
e. Circulating Water System:	10.4.5							
1. Piping		P	O,T	D	B31.1.0	NA	N	
2. Piping, cooling tower		P	O	NA	C504	NA	N	
3. Condenser		P	T	D	VIII-1/HEI	NA	N	(24)
4. Pumps		P	O	D	Hyd.I	NA	N	
5. Valves		P	O,T	D	B31.1.0	NA	N	
6. Cooling tower		P	O	NA	CTI	NA	N	
7. Head tanks		P	O	D	VIII-1	NA	N	
8. Expansion joints		P	C	NA	None	NA	N	
f. Steam Seal System:	10.4.3							
1. Gland steam condenser		P	T	D	VIII-1/TEMA C	NA	N	
2. Piping and valves		P	T	D	B31.1.0	NA	N	
3. Steam seal evaporator		P	T	D	VIII-1/TEMA C	NA	N	
g. Lube Oil System:	10.2							
1. Tanks, reservoirs		P	T	D	API-620	NA	N	(24)
2. Pumps		P	T	D	VIII-1/Hyd.I	NA	N	
3. Motors		P	T	NA	NEMA MG-1	NA	N	
4. Centrifuges		P	T	NA	None	NA	N	
5. Heat exchangers		P	T	D	VIII-1/TEMA C	NA	N	
6. Piping and valves		P	T	D	B31.1.0	NA	N	
7. Transfer piping and valves		P	T	D	B31.1.0	NA	N	
h. Generator Hydrogen & Carbon Dioxide Purge System:	10.2							
1. Vessels		P	T	D	VIII-1	NA	N	
2. Piping		P	T	D	B31.1.0	NA	N	
3. Valves		P	T	D	B31.1.0	NA	N	

TABLE 3.2-1 (Cont)

Principal Components (57)	FSAR Section	Source of Supply (1)	Location (2)	Quality Group Classification (3)	Principal Construction Codes and Standards (5)	Seismic Category (6)	QA Requirements (7)	Comments
i. Turbine Bypass System:	10.4.4							
1. Piping from main steam line to bypass valve chest		P	T	C	III-3	NA	N	(13)(14)(18)
2. Bypass valve chest		P	T	C	None	NA	N	(13)(14)(18)
3. Piping from bypass valve chest to condenser		P	T	D	B31.1.0	NA	N	
XV. Instrumentation and Control Systems								
a. Reactor Trip System:								
1. Reactor Protection System (RPS):	7.2							
a) Electrical modules (27)		GE	C,A,T	NA	IEEE-279/323	I	Y	
b) Cable with safety function		P	C,A,T	NA	IEEE-279/323	I	Y	(15)
b. Engineered Safety Features Systems (controls and instrumentation required for safety, associated with each actuated system):	7.3							
1. Emergency Core Cooling System (HPCI, ADS, core spray, LPCI)		GE	C,A	NA	IEEE-279	I	Y	
2. Primary Containment and Reactor Vessel Isolation control system (PCRVICS)		GE/P	C	NA	IEEE-279	I	Y	
3. Deleted								
4. RHR containment spray cooling mode		GE	A,C	NA	IEEE-279	I	Y	
5. RHR suppression pool cooling mode		GE	A,C	NA	IEEE-279	I	Y	
6. Containment Atmosphere Control System		P	A,B,C	NA	IEEE-279	I	Y	
7. DELETED								
8. Filtration, Recirculation, and Ventilation System		P	B,C	NA	IEEE-279	I	Y	

Principal Components(57)	FSAR Section	Source of Supply (1)	Location (2)	Quality Group Classification (3)	Principal Construction Codes and Standards (5)	Seismic Category (6)	QA Requirements (7)	Comments
9. Reactor Building Ventilation Isolation System		P	C	NA	IEEE-279	I	Y	
10. Main Control Room Habitability and Isolation System		P	B	NA	IEEE-279	I	Y	
11. Essential auxiliary supporting systems for engineered safety features control		P	All	NA	IEEE-279	I	Y	
c. Controls and instrumentation associated with safe shutdown systems:	7.4							
1. Reactor Core Isolation Cooling System (RCIC)		GE	C	NA	IEEE-279	I	Y	
2. Standby Liquid Control (SLC) System		GE	C	NA	IEEE-279	I	Y	
3. RHR, reactor shutdown cooling mode		GE	C	NA	IEEE-279	I	Y	
4. Remote shutdown systems		P	R	NA	IEEE-279	I	Y	
5. Essential auxiliary supporting systems for the safe shutdown systems		GE/P	All	NA	IEEE-279	I	Y	
d. Safety-related display instrumentation								
1. Control rod position indication system (CRPIS) (Nonsafety-related)	7.5	GE	B	NA		I	N	(49)
2. Bypass and Inoperable Status Indication System (BISIS) (safety-related)	7.5	P	B	NA	IEEE-279	I	Y	(49)
- Isolation device	7.5	P	B	NA	IEEE-279	I	Y	
- CRIDS Computer	7.5	P	B	NA		NA	N	
3. Plant Computer System (PCS)								
a. NSSS (nonsafety-related)	7.5	GE	B	NA		NA	N	(51)
b. CRIDS (nonsafety-related)	7.5	P	B	NA		NA	N	

TABLE 3.2-1 (Cont)

Principal Components (57)	FSAR Section	Source of Supply (1)	Location (2)	Quality Group Classification (3)	Principal Construction Codes and Standards (5)	Seismic Category (6)	QA Requirements (7)	Comments
c. RMS (nonsafety-related) Note: safety-related portion of the RMS system is found in section XVe of this table.	7.5	P	B	NA		NA	N	
d. ERF DAS (safety-related)								
Isolation device	7.5	P	B	NA	IEEE-279	I	Y	(49)
Data concentrator portion	7.5	P	B					(51)
4. Startup Transient Monitoring System (STMS) (nonsafety-related)	7.5	P	B	NA		I	N	
5. Safety Relief Valve Position Indication System (SRV PIS) (nonsafety-related)	7.5	P	B	NA		I	N	
6. DELETED								
7. Post accident monitoring instrumentation (PAMI) (safety-related)	7.5	P	B	NA	IEEE-279	I	Y	(49)
- Isolation device	7.5	P	B	NA	IEEE-279	I	Y	(49)
- CRIDS computer	7.5	P	B	NA		NA	N	
e. Controls and instrumentation associated with other systems required for safety:	7.6							
1. Process Radiation Monitoring System		GE/P	A,B, C,R,T	NA	IEEE-279	I	Y	(30)
2. Leak Detection System (RCIC, RWCU, HPCI)		GE/P	C	NA	IEEE-279	I	Y	(30)
3. Recirculation pump trip (RPT) controls & instrumentation		GE	C,T	NA	IEEE-279	I	Y	
4. High pressure-low pressure system interlocks		GE	C	NA	IEEE-279	I	Y	
5. Neutron monitoring system								
a) Flanges, TIP probe and purge, containment penetration		GE	A,C	E	NONE	I	Y	(64)(65)
b) Valves, isolation, TIP purge		P	A,C	E	NONE	I	Y	(64)(65)

Principal Components[57]	FSAR Section	Source of Supply (1)	Location (2)	Quality Group Classification (3)	Principal Construction Codes and Standards (5)	Seismic Category (6)	QA Requirements (7)	Comments
c) Electrical modules, IRM and APRM[27]		GE	C	NA	IEEE-279/323 IEEE-603-1991(APRM)	I	Y	
d) Cable, IRM and APRM, with safety function		P	A,C	NA	IEEE-279/323	NA	Y	(15)
e) Valve, TIP probe, isolation		GE	C	NA	(44)	I	Y	(64)
f) Tubing, TIP probe, instrument penetration		GE	C	NA	(44)	I	Y	(64)
g) Tubing, TIP purge, containment penetration		P	A,C	E	NONE	I	Y	(64) (65)
6. Redundant Reactivity Control System (RRCS)		GE	C	NA	IEEE-279	I	Y	
7. Main steam SRV relief function		GE	A,B,C	NA	IEEE-279	I	Y	
8. Safety system/nonsafety system isolation		P	A,B,C	NA	IEEE-279	I	Y	
f. Controls & instrumentation associated with systems not required for safety:	7.7							
1. Reactor Manual Control System		GE	A,C	NA	None	NA	N	
2. Recirculation Flow Control System		GE	T	NA	None	NA	N	
3. Feedwater Control System		GE/P	T	NA	None	NA	N	
4. Refueling interlocks		GE	C	NA	None	NA	N	
5. Deleted								
6. Deleted								
7. Pressure Regulator & Turbine Generator System		P	T	NA	None	NA	N	
8. Deleted								
9. Area Radiation Monitoring Systems		P	All	NA	None	NA	N	(51)
10. Deleted								
11. Reactor Water Cleanup System		GE	C	NA	None	NA	N	
12. Radwaste systems		P	R	NA	None	NA	N	
13. Fuel Pool Cooling System		P	C	NA	None	NA	Y	(51)
14. Fuel Pool Cleanup System		P	C	NA	None	NA	N	

Principal Components(57)	FSAR Section	Source of Supply (1)	Loca-tion (2)	Quality Group Classi-fication (3)	Principal Construc-tion Codes and Standards (5)	Seismic Category (6)	QA Require-ments (7)	Comments
g. Control complex panels								
1. Electrical modules with safety function(27)		GE/P	B	NA	IEEE-279/323	I	Y	(15)
2. Cable with safety function		P	B	NA	IEEE-279/323	NA	Y	
h. Local panels and racks								
1. Electrical modules with safety function		GE/P	All	NA	IEEE-279/323	I	Y	(15)
2. Cable with safety function		P	All	NA	IEEE-279/323	NA	Y	
XVI. Electric System								
a. Engineered safety features (Class 1E) ac equipment:	8.3							
1. 4.16-kV switchgear		P	G	NA	IEEE-308/ 323/344	I	Y	
2. 480 V unit substations		P	G	NA	IEEE-308/ 323/344	I	Y	
3. 480 V motor control centers		P	B,C,G,W	NA	IEEE-308/ 323/344	I	Y	
b. Engineered safety features (Class 1E) dc equipment:	8.3							
1. 125 V and 250 V station batteries and racks, battery chargers, & distribution bus		P	B,G	NA	IEEE-450/ 323/344/650	I	Y	
2. 125 V switchgear and distribution panels		P	B	NA	IEEE-308/ 323/344	I	Y	
3. 250 V motor control center & switchgear		P	B,C	NA	IEEE-308/ 323/344	I	Y	

TABLE 3.2-1 (Cont)

Principal Components[57]	FSAR Section	Source of Supply (1)	Location (2)	Quality Group Classification (3)	Principal Construction Codes and Standards (5)	Seismic Category (6)	QA Requirements (7)	Comments
c. 120 V vital (Class 1E) ac system equipment:	8.3							
1. Static inverters		P	B,G	NA	IEEE-308/ 323/344/650	I	Y	
2. 120 V distribution panels		P	B,G	NA	IEEE-308/ 323/344	I	Y	
d. Electric cables for ESF equipment:	8.3							
1. 5-kV power cables		P	All	NA	IEEE-323/383	NA	Y	(15)
2. 600 V power cables		P	All	NA	IEEE-323/383	NA	Y	(15)
3. Control and instrumentation cables		P	All	NA	IEEE-323/383	NA	Y	(15)
e. Miscellaneous electrical:	8.3							
1. Reactor Building and primary containment penetration assemblies		P	C	NA	IEEE-317/344	I	Y	
2. Conduit and supports, safety-related		P	All	NA	IEEE-344	I	Y	(15)
3. Tray supports, safety-related		P	All	NA	IEEE-344	I	Y	(15)
4. Emergency lighting systems		P	All	NA	IEEE-344	NA		(52)
5. Emergency communications systems		P	All	NA	None	NA		(52)
6. Conduit and supports		P	All[54]	NA	None	II/I		(23)(50)
7. Tray and supports		P	All[54]	NA	None	II/I		(23)(50)
XVII. <u>Auxiliary Systems</u>								
a. Compressed Air (service and instrument) Systems:	9.3.1							
1. Compressors		P	T	NA	None	II/I		(50)
2. Pressure vessels, not for safety-related equipment		P	All	D	VIII-1	II/I		(50)
3. Piping and valves, containment penetration and isolation		P	C,A	B	III-2	I	Y	(48)

TABLE 3.2-1 (Cont)

Principal Components (57)	FSAR Section	Source of Supply (1)	Location (2)	Quality Group Classification (3)	Principal Construction Codes and Standards (5)	Seismic Category (6)	QA Requirements (7)	Comments
4. Piping and valves, reactor building penetration and isolation		P	C	D	B31.1.0	II/I		(50)
5. Piping and valves, other		P	All	D	B31.1.0	II/I		(50)
b. Primary Containment Instrument Gas System:	9.3.6							
1. Compressors		P	C	B	III-2	I	Y	(45)
2. Filter housings, dryers, & coolers (air side)		P	C	B	III-2	I	Y	(67)
3. Coolers (water side)		P	C	C	III-3	I	Y	(62)
4. Receiver tanks		P	C	B	III-2	I	Y	
5. Piping and valves, air with safety function		P	C	B	III-2	I	Y	(48)(63)(70)
6. Piping and valves, cooling water		P	C	C	III-3	I	Y	(48)(63)
7. Piping and valves, air with safety function (inside drywell)		P	A	C	III-3	I	Y	(48)
8. Piping and valves, containment penetration and isolation		P	A,C	B	III-2	I	Y	(48)
9. Piping and valves, air, other		P	A,C	D	B31.1.0	II/I		(50)
10. Motors, compressors		P	C	N/A	IEEE-323/344	I	Y	
c. Process Sampling Systems:	9.3.2							
1. Sample coolers		P	C,A,T,R	D	VIII-1/TEMA C	NA	N	
2. Piping and valves on III-1 systems		P	C	A	III-1	I	Y	(10)(48)
3. Piping and valves on III-2 systems		P	C,A	B	III-2	I	Y	(10)(48)
4. Piping and valves on III-3 systems		P	A,T,R	C	III-3	I	Y	(10)(48)
5. Piping and valves, other		P	A,T,R	D	B31.1.0	NA	N	(10)
d. Standby Liquid Control System:	9.3.5							
1. Standby liquid control tank		GE	C	B	API-650	I	Y	
2. Pumps		GE	C	B	P&V-II (9)	I	Y	
3. Pump motors		GE	C	NA	NEMA MG-1	I	Y	

TABLE 3.2-1 (Cont)

Principal Components[57]	FSAR Section	Source of Supply (1)	Location (2)	Quality Group Classification (3)	Principal Construction Codes and Standards (5)	Seismic Category (6)	QA Requirements (7)	Comments
4. Valves, explosive		GE	C	B	P&V-II[9]	I	Y	(48)
5. Valves, isolation and within		P	A,C	A	III-1	I	Y	(10)(48)
6. Valves, beyond isolation valves		P	C	B	III-2	I	Y	(10)(48)
7. Piping, within isolation valves		P	A,C	A	III-1	I	Y	(10)(48)
8. Piping, beyond isolation valves		P	C	B	III-2	I	Y	(10)(48)
9. Electrical equipment/devices with safety function		GE/P	C/B	NA	IEEE-279/ 323/344	I/NA	Y/N	(29)
10. Cable with safety function		P	C	NA	IEEE-279/323	NA	Y	(15)
11. Test tank		GE	C	D	None	NA	N	
e. Fire Protection System:	9.5.1							
1. Tanks		P	O	D	D100/NML	NA	N	(61)
2. Pumps, piping and water system components		P	All	NA	NFPA/NML	NA		(52)
3. Gas system components (carbon dioxide and Halon 1301)		P	All	NA	NFPA/NML	NA		(52)
4. Fire and smoke detection and alarm system		P	All	NA	NFPA/NML	NA		(52)
f. Auxiliary Boiler System:	9.5.9							
1. Tanks, blowdown, chemical feed, fuel oil storage and fuel oil day		P	O	D	VIII-1	NA	N	
2. Boilers		P	O	D	P&V-I	NA	N	
3. Deaerator		P	O	D	VIII-1	NA	N	
4. Pumps		P	O	D	VIII-1	NA	N	
5. Piping and valves, reactor building penetration and isolation		P	C	D	B31.1.0	II/I		(48) (50)
6. Piping & valves, other		P	All	D	B31.1.0	NA	N	
g. Equipment and Floor Drainage System:	9.3.3							
1. Piping, radioactive		P	C,A,T,R	R/D	B31.1.0	NA	N	
2. Piping, nonradioactive		P	All	D	B31.1.0	NA	N	

Principal Components(57)	FSAR Section	Source of Supply (1)	Location (2)	Quality Group Classification (3)	Principal Construction Codes and Standards (5)	Seismic Category (6)	QA Requirements (7)	Comments
3. Piping and valves, primary containment isolation boundary		P	A,C	B	III-2	I	Y	(48)
4. Piping and valves, reactor building penetration and isolation		P	C	D	B31.1.0	II/I		(50)
h. Post-Accident Liquid and Gas Sample System (PASS):	9.3.2							
1. Piping and valves, primary containment isolation and reactor coolant pressure boundary		P	A,C	B	III-2	I	Y	(48)
2. Tubing, Reactor Building penetration and isolation		P	C,R	D	B31.1.0	NA	N	
3. Piping and valves, other		P	C,R	D	B31.1.0	NA	N	
4. Piping station		GE	R	D	B31.1.0	NA	N	
5. Post-accident sampler		GE	R	D	B31.1.0	NA	N	
i. Breathing air:	9.5.10							
1. Piping and valves, reactor building penetration & isolation		P	C	D	B31.1.0	II/I		(50)
2. Piping and valves, containment penetration and isolation		P	A,C	B	III-2	I	Y	(48)
j. Lighting systems:	9.5.3							
1. Components located in safety-related areas		P	All	NA	None	II/I		(50)
XVIII. <u>Buildings</u>	3.8							
a. Primary containment:	3.8.2							
1. Access hatches/locks/doors		P	C	B	III-MC	I	Y	
2. Vessel and head		P	C	B	III-MC	I	Y	
3. Penetration assemblies-pipes		P	C	B	III-2 MC	I	Y	
4. Vent piping		P	C	B	III-MC	I	Y	
5. Vacuum relief valves	6.2.1	P	A,C	B	III-2	I	Y	(48)

Principal Components[57]	FSAR Section	Source of Supply (1)	Location (2)	Quality Group Classification (3)	Principal Construction Codes and Standards (5)	Seismic Category (6)	QA Requirements (7)	Comments
6. Monorail supports		P	C	NA	AISC	I	Y	
7. Biological shield		P	A	NA	AISC/ACI-318	I	Y	(25)
8. Coating		P	A,C	NA	None	NA	Y	
9. ECCS suction strainers		P	A	B	None	I	Y	
b. Auxiliary Building - diesel area		P	G	NA	AISC/ACI-318	I	Y	
c. Auxiliary Building - control area		P	B	NA	AISC/ACI-318	I	Y	
d. Auxiliary Building - radwaste area		P	R	NA	AISC/ACI-318	I	Y	(22)
e. Turbine Building		P	T	NA	AISC/ACI-318	II/I		(21)(50)
f. Administration facility		P	O	NA	AISC/ACI-318	II/I		(50)
g. Circulating water pumphouse		P	O	NA	AISC/ACI-318	NA	N	
h. Reactor Building/including pressure-retaining doors		P	C	NA	AISC/ACI-318	I	Y	
i. Plant cancelled area		P	All	NA	AISC/ACI-318	I	Y	
j. Shore protection at intake structure (including sheet pile retaining wall)		P	O,W	NA	None	II/I		(50)
XIX. <u>Structures</u>[58]	3.8							(28)
a. Station service water intake structure		P	O,W	NA	AISC/ACI-318	I	Y	
b. Deleted								
c. Diesel generator fuel tank room		P	G	NA	None	I	Y	
d. Station battery rooms		P	B	NA	None	I	Y	
e. Spent fuel pool, reactor well, new fuel vault, dryer separator pool, and cask pit	9.1.1, 9.1.2	P	C	NA	None	I	Y	
f. Deleted								
g. Unit vent stack, North & South		P	O	NA	AISC/SMACNA	NA	N	
h. Condensate storage tank dike		P	O	NA	ACI-318	I	Y	

Principal Components[57]	FSAR Section	Source of Supply (1)	Location (2)	Quality Group Classification (3)	Principal Construction Codes and Standards (5)	Seismic Category (6)	QA Requirements (7)	Comments
i. Spent fuel pool liner	9.1.2	P	C	NA	None	NA	N	(56)
j. Skimmer surge tanks (concrete structure)	9.1.1	P	C	NA	ACI-318	I	Y	(46)
k. Missile/jet barriers		P	A,B,C R,W,G	NA	AISC/ACI-318	I	Y	
l. Structural backfill		P	O	NA	None	I	Y	
m. Post accident shielding		P	A,B,C R,T,G	NA	ACI-318	I	Y	
n. Seismic Category I electrical duct bank manholes		P	O	NA	ACI-318	I	Y	

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(1) GE = General Electric
P = Public Service Electric and Gas Company/Bechtel Power Corporation
ABB = ASEA Brown Boveri

(2) A = drywell
B = Auxiliary Building: control area
C = reactor building
G = Auxiliary Building: standby diesel generator area
L = offsite locale
O = outdoors onsite
R = Auxiliary Building: radwaste area
T = Turbine Building
W = station service water intake structure.

(3) A, B, C, D - NRC quality group classification as defined in Regulatory Guide 1.26.
R - Quality Group R is comprised of the requirements provided in Regulatory Guide 1.143.
E - GE. QUASI - Code Group Classification satisfies intent of Reg Guide 1.11 and meets the requirements of GDC 56
NA - quality group classification not applicable to this equipment.

(4) Deleted

(5) Notations for principal construction codes:

III- A, B, C - ASME Boiler and Pressure Vessel Code, Section III, Class A, B, or C

III-	1, 2, 3, MC, NG, NF - ASME Boiler and Pressure Vessel Code, Section III, Class 1, 2, 3, or MC or Subsections NG or NF.
P&V-	I, II, & III - ASME Pump and Valve for Nuclear Power, Class I, II, & III
VIII-1	ASME Boiler and Pressure Vessel Code, Section VIII, Division 1
XI	ASME Boiler and Pressure Vessel Code, Section XI
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
C504	American Water Works Association, AWWA 504-70; Section 2 through 19
D100	American Water Works Association, AWWA-D100, Standard for Steel Tanks, Standpipes, Reservoirs and Elevated Tanks for Water Storage
B.31.7	ANSI B31.7, Nuclear Power Piping
B31.1.0	ANSI B31.1.0, Code for Pressure Piping
N195	ANSI N195, Fuel Oil Systems for Standby Diesel Generators
SMACNA	Sheet Metal & Air Conditioning Contractors National Association, Inc
HEI	Heat Exchange Institute
TEMA C&R	Tubular Exchanger Manufacturers Assoc, Class C & R
HYD.I	Hydraulic Institute
AISC	American Institute of Steel Construction, Specification for Design Fabrication, and Erection of Structural Steel for Buildings
AISI	American Iron and Steel Institute, Specification for the Design of Cold-Formed Steel Structural Members; Design of Light-Gauge Cold-Formed Stainless Steel Structural Members
ACI-307	American Concrete Institute, Specifications for the Design and Construction of Reinforced Concrete Chimneys
ACI-318	Building Code Requirements for Reinforced Concrete
AMCA	Air Moving and Conditioning Association (AMCA) 210, Test Codes for Air Moving Devices AMCA 211 A, AMCA Certified Ratings Program for Air Performance
CS-8T	American Association for Contamination Control, AACC CS-8T, Tentative Standard for High Efficiency Gas Phase Adsorber Cells
NEMA	National Electrical Manufacturers' Association
NEMA MG-1	National Electrical Manufacturers' Association, NEMA-MG-1, Motors and Generators
NML	Nuclear Mutual Limited
IEEE-279	IEEE-279, Criteria for Protection Systems for Nuclear Power Generating Stations
IEEE-308	IEEE-308, Standard Criteria for Class 1E Electric Systems for Nuclear Power Generating Stations
IEEE-317	IEEE-317, Standard for Electrical Penetration Assemblies in Containment Structures for Nuclear-Fueled Power Generating Stations
IEEE-323	IEEE-323, Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations
IEEE-344	IEEE-344, Guide for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations
IEEE-383	IEEE-383, Type Test of Class 1E Electrical Cables, Field Splices, and Connections for Nuclear Power Generating Stations
IEEE-387	IEEE-387, Criteria for Diesel Generator Units Applied as Standby Power Supplies for Nuclear Power Generating Stations
IEEE-450	IEEE-450, Recommended Practice for Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Generating Stations and Substations

IEEE-603	1991, Standard Criteria for Safety Systems for Nuclear Power Plants
IEEE-650	IEEE-650, Standard for Qualification of Class 1E Static Battery Charges and Inverters for Nuclear Power Generation Station.
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
ARI-410	Air Conditioning and Refrigeration Institute, 410, Forced-Circulation Air Cooling and Air-Heating Coils.
UL 900	Standards for Air Filter Units
CTI	Cooling Tower Institute.
NSPC	National Standard Plumbing Code

(6) I = The equipment is constructed in accordance with the requirements of Seismic Category I structures and equipment as described in Section 3.7

II/I = The equipment is constructed so that it cannot adversely affect plant safety features during and/or after the SSE

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

(7) Y = The equipment meets the quality assurance requirements of 10 CFR 50, Appendix B, in accordance with the quality assurance program described in Chapter 17

N = Quality assurance requirements not applicable to this equipment, except as noted.

(8) The control rod drive insert and withdraw lines from the drive flange, up to and including the first valve on the hydraulic control unit, are Quality Group B.

(9) Built to existing codes at the time of manufacture.

(10) Instrument and sampling lines quality group, seismic category, and quality assurance requirements are as follows:

1. Lines 1 inch and smaller that are part of the reactor coolant pressure boundary are Quality Group B and Seismic Category I.

2. All instrument lines that are connected to the RCPB and used to actuate and/or monitor safety systems are Quality Group B from the outer isolation valve or the process shutoff valve (excess flow check valve) to the sensing instrumentation.

3. All instrument lines that are connected to the RCPB and are not used to actuate and/or monitor safety systems are Quality Group D from the outer isolation valve or the process shutoff valve (excess flow check valve) to the sensing instrumentation.

4. All other instrument lines:

   (a) Through the root valve are of the same classification as the system to which they are attached.

(b) Beyond the root valve, if used to actuate a safety system, are of the same classification as the system to which they are attached.

(c) Beyond the root valve, if not used to actuate a safety system, may be Quality Group D.

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

(11) The HPCI and RCIC turbines do not fall within the applicable design codes. To ensure that these turbines are fabricated to the standards commensurate with their safety and performance requirements, GE has established specific design requirements for these components, which are as follows:

1. All welding is qualified in accordance with ASME B&PV Code, Section IX

2. All pressure-containing castings and fabrications are hydrotested at 1.5 times design pressure

3. All high pressure castings are radiographed according to ASTM E-94, E-142, maximum feasible volume; E-71, E-186 or E-280, severity level 3

4. As-cast surfaces are magnetic-particle or liquid-penetrant tested according to ASME B&PV Code, Section III, Paragraph N-3234 or N-323.3.

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

6. Butt-welds are radiographed and magnetic particle- or liquid penetrant-tested according to ASME B&PV Code Section III, Paragraph N-626 or N-627, respectively

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

8. Record system and traceability are according to ASME B&PV Code, Section III, Appendix IX, Paragraph IX-225

9. Control and identification are according to ASME B&PV Code, Section III, Appendix IX, Paragraph IX-226

10. Procedures conform to ASME B&PV Code, Section III, Appendix IX, Paragraph, IX-300

11. Inspection personnel are qualified according to ASME B&PV Code, Section III, Appendix IX, Paragraph IX-400.

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

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

The design and construction specifications for the HCU do invoke such codes and standards as can be reasonably applied to individual parts in developing required quality levels, but these codes and standards are supplemented with additional requirements for these parts and for the remaining parts and details. For example:

1. All welds are liquid penetrant 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

Quality Group D is generally applicable because the codes and standards invoked by that group contain clauses that permit the use of manufacturer's standards and proven design techniques. This is supplemented by the quality control techniques described above.

(13) See Section 3.2.1 for discussion of conformance to Regulatory Guide 1.29.

(14) See Section 3.2.2 for discussion of conformance to Regulatory Guide 1.26.

(15) The conduit, trays, and supports for safety-related cables are Seismic Category I and Q-listed.

(16) FRVS safety classification is at variance with ANSI N212, which has upgraded this system to Quality Group B.

(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 is seismically analyzed to ensure that it does not fail under loadings normally associated with Seismic Category I.

(19) Impact testing of carbon or low-alloy steels is in accordance with ASME B&PV Code, Section VIII, Division 1, Paragraph UCS 66. Low temperature criteria for carbon or low alloy steels is defined as -20°F or below.

(20) Build to ANSI B30.2 and New Jersey Administrative Code for Overhead and Gantry Cranes, Title 12.

(21) The power conversion system structures are constructed in accordance with applicable codes for steam power plants.

(22) Portions of the radwaste systems are built to Quality Group R Standards, which are those specified in the NRC Regulatory Guide 1.143. The equipment, piping, and components are fabricated with a mandatory pressure test and welded construction wherever possible. Regulatory Guide 1.143 reduces the seismic design requirements from Seismic Category I to a simplified seismic analysis. For further information, refer to NRC Regulatory Guide 1.143.

Note: For this project, the radwaste area shares a building that includes control and diesel generator areas, and therefore is required to be Seismic Category I.

(23) These components and associated supporting structures must be designed to retain structural support and/or pressure integrity during and after a Seismic Category I event, 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 that are Seismic Category I.

(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 that, while intended for other applications, are used for guidance and recommendations in determining Quality Group D and R 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.

(25) Regulatory Guide 1.54 applies.

(26) These devices are supported and analyzed to remain functional up to an SSE.

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

(28) Refer to Sections 3.7 and 3.8 for discussions of seismic design and Seismic Category I structure design, respectively.

(29) Electrical devices include components such as switches, controllers, fuses, junction boxes, and relays, which are discrete components of a larger subassembly/module. Safety-related devices are Seismic Category I; fail-safe devices are not Seismic Category I.

(30) Only equipment associated with a safety action, e.g., isolation, need conform to a safety function.

(31) Deleted.

(32) Deleted.

(33) Deleted.

(34) Deleted.

(35) Deleted.

(36) Deleted.

(37) Deleted.

(38) Dynamic analysis methods for seismic loading are not applicable, because this equipment is supported by the reactor building crane when it performs its function. It is designed with a minimum safety factor of 5 and is proof-tested after fabrication.

(39) Dynamic analysis methods for seismic loading are not applicable, because this equipment is supported by the refueling platform when it performs its function.

(40) The standby diesel generator and auxiliary systems are in compliance in NUREG/CR0660, Enhancement of Onsite Emergency Diesel Generator Reliability.

(41) Pipe-whip restraints are not required to restrain the piping during an earthquake. These restraints are designed to withstand an SSE without loss of functional capability.

(42) These items are classified as seismic NA (exempt from seismic evaluation) because they are suspended from cables that dampen out the transmission of floor response spectra.

(43) A defective fuel storage container is seismic category NA, because it is isolated from the seismic excitation.

(44) Built to manufacturer's standards.

(45) Instrument gas compressors are designed, fabricated, and tested per ASME B&PV Code, Section III, Class 2, but are not stamped. The ASME B&PV material requirements apply to the compressor cylinder heads only.

(46) The skimmer surge tanks are of non-Seismic Category I design, but they are embedded in a Seismic Category I concrete structure.

(47) Duct work is of non-Seismic Category I design, but is installed and supported as Seismic Category I.

(48) Valve operators on safety related valves that must function are Q-listed and Seismic Category I

(49) Equipment is classified in accordance with the conformance statements made in Sections 7.2, 7.3, 7.4, 7.5 and 7.6 in reference to IEEE 279 paragraph 4.4 and IEEE-323.

(50) The QA Program controls applicable to equipment classified as Seismic II/I are in accordance with Regulatory Guide 1.29 commitments contained in FSAR Section 1.8

(51) No QA Program controls applied during Design and Construction Phase. QA Programs controls during operation are applied to an extent consistent with the items importance to safety.

(52) QA program controls for the fire protection program ('F' program), including emergency lighting and communications, are applied to the extent of the 10 quality assurance criteria of Branch Technical Position CMEB 9.5-1 and to an extent consistent with the item's/activity's importance to safety. Fire protection QA program ('F' program) was formally implemented effective July 1, 1978 for the fire protection systems in safety-related areas.

(53) The recirculation system piping was built to both ASME Section III and B31.7 codes as required by the GE design specification. The ASME Section III NPP-1 report requires signatures by a qualified inspector and also indicates that the pipe was built to the requirements of B31.7.

(54) Except north radwaste area of auxiliary building, since there are no seismic Category I components in this area.

(55) The reactor pressure vessel internal structures which are accessible are included in the ISI program, which is covered by the operational QA program.

(56) Any modifications, repair, or rework to the liner will be conducted under the operational QA program.

(57) Containment isolation valves that are required per GDC 54-56 and are not part of the principal components shown are subject to the pertinent provisions of 10CFR50 Appendix B.

(58) Modifications to roof parapet and openings of Q structures will be conducted under the operational QA program.

(59) The governor valves for HPCI and RCIC turbines are part of the operational QA program.

(60) These modules comply with IEEE 279-1971, as applicable, i.e., redundancy and separation within the modules is inapplicable.

(61) 'F' program is not retroactive to components purchased and installed prior to July 1, 1978. Fire water storage tanks, the tank heaters and associated controls, and the valve pit unit heaters are excluded from the 'F' program during the design and construction phase but will be included in the QA program for fire protection during the operations phase.

(62) The water/glycol side of the containment instrument gas thermo-siphon is noncoded.

(63) The CVI installed portion of the gas and cooling water piping system on the containment instrument gas compressor skids, which is designed and fabricated in accordance with Section III of the ASME Code, is not stamped as an installed nuclear piping system and is not covered in an N-5 Code data package.

(64) The TIP probe instrument penetration and isolation valves are built to GE's standards for safety-related instrumentation.

(65) Portions of the FLG Assembly, TIP Probe and Purge Assembly components and certain valves, which are currently installed in the Neutron Monitoring System (SE) have had their quality group classification and principal construction code standard categories downgraded from "B" and "Sect. III CL 2" designations to an "E" and "None". This design criteria has been set forth in GE Documents with NRC Approval.

(66) Safe end to N5B Core Spray nozzle repair performed in October 1997 which was designed in accordance with 1989 Edition ASME Section XI, IWB-3641, NUREG 0313, Revision 2 and installed using ASME Code Cases N432, N504-1, 2142, 2143, N416-1 as modified for use in accordance with USNRC SER (TAC # M99755). Safe end to N2K Reactor Recirculation nozzle repair performed in December 2004 which was designed in accordance with ASME Code Section XI 1998 Edition, including Addenda through 2000, IWB-3640, NUREG 0313, Revision 2 (which was implemented by Generic Letter 88-01) and installed using ASME Code Cases N504-2 and N638 as modified for use in accordance with USNRC Safety Evaluation Report (SER) as tracked by SAP operations 228 and 279 for DCP 80076353. Safe end to N2A Reactor Recirculation nozzle repair performed in October 2007 via DCP 80094209, which was designed in accordance with ASME Code Section XI 1998 Edition, including Addenda through 2000, IWB-3640, NUREG 0313, Revision 2 (which was implemented by Generic Letter 88-01) and installed using ASME Code Cases N504-3 and N638-1 as modified for use in accordance with USNRC Safety Evaluation Report (SER).

(67) The inter and after cooler moisture separators installed on the PCIG skid assembly have been replaced. The replacement separators are ASME Sec. VIII and have been installed I.A.W. NRC Generic Letter 89-09, DCP 4EC-3274.

(68) Abandoned in place.

(69) The main steam line stop valves do not have an active safety function due to the deletion of the MSIV Sealing System.

(70) The inter and after cooler moisture separator drain traps installed on the PCIG skid assembly have been replaced. The replacement traps have been installed I.A.W. NRC Generic Letter 89-09, (DCPs 80005324 & 80007625).

(71) The HOAS bottles were reduced in quantity and location changed to the Yard area via DCP 80087681. This was possible due to the declassification of the HOAS to a non-safety related system by Amendment No. 160 to the HCGS Operating License.

TABLE 3.2-2

CODE GROUP DESIGNATION - INDUSTRY CODES AND STANDARDS FOR MECHANICAL COMPONENTS (NSSS SCOPE)

Quality Group Classification	ASME B&PV Sect. III Code Classes 1968 Ed.	/1971 Ed.	Components Ordered Prior to July 1, 1971 (3)	Components Ordered on or After July 1, 1971 and Prior to July 1, 1974	Components Ordered After July 1, 1974
A	A	1	ASME III, A ANSI B31.7 I NP&VC I TEMA C See Note (4)	ASME III, 1 NA & NB Subsections B31.7 I (2) TEMA C See Note (6)	ASME III, 1 NA, NB, & NF TEMA C See Note (6)
B	B(1),C	2,MC(1)	ASME III, A(1), C ANSI B31.7, II NP & VC, II TEMA C Tanks API 620/650 See Notes (4) & (8)	ASME III, 2 & MC(1) NA & NC Subsections NA & NE Subsections TEMA C Tanks See Notes (6) & (8)	ASME III, 2 & MC(1) NA, NC, NF, & NG NA, NE, NF, & NG TEMA C Tanks See Notes (6) & (8)
C	-	3	ASME VIII, Div. 1 ANSI B31.7, III NP&VC III TEMA C Tanks API 620/650 See Notes (5) & (8)	ASME III, 3 NA & ND subsections TEMA C Tanks See Notes (6) & (8)	ASME III, 3 NA, ND, & NF TEMA C Tanks See Notes (6) & (8)
D	-	-	ASME VIII, Div. 1 ANSI B31.1.0 TEMA C Tanks API 620/650 See Notes (5) & (8)	ASME VIII, Div. 1 ANSI 31.1.0 TEMA C Tanks API 620/650 See Notes (5) & (8)	ASME VIII, Div. 1 ANSI B31.1.0 TEMA C Tanks API 620/650 See Notes (7) & (8)
E	Special engineered equipment with codes and standards as specified in notes and comments in Table 3.2-1				

(1) Metal containment vessel only.

(2) Section III - 71 Ed. requires design of pipe supporting elements to be in accordance with the requirement of ANSI 631.7-6a, Divisions 1-720 and 1-721.

(3) No piping procured prior to Jan. 1, 1970.

(4) Pumps Classified A and B

The requirements of ASME Section III, C, Boiler and Pressure Vessel Code, are used as a guide in calculating the thickness of pressure retaining portions of the pump and in sizing cover bolting.

(5) Pumps Classified C or D and Operating Above 150 psig or 212°F
The requirements of ASME Section VIII, Div. 1, Boiler and Pressure Vessel Code are used as a guide in calculating the thickness of pressure retaining portions of the pump and in sizing cover bolting. Pumps classified D and operating below 150 psig and 212°F use manufacturer's standard pump for service intended.

(6) Pumps Classified A, B, and C
Use applicable ASME Section III Subsections N, NB, NC or ND respectively for vessel design as a guide in calculating the thickness of pressure retaining portions of the pump and in sizing over boiling.

(7) Pumps Classified D and Operating Above 150 psi and 212°F
The requirements of ASME VIII, Div. 1 are used as a guide in calculating the thickness of pressure retaining portions of the pump and in sizing over boiling. Pumps operating below 150 psig and 212°F use manufacturer's pump for service intended.

(8) Tanks are not fully covered by ASME codes. Groups B and C tanks ordered on or after July 1, 1972, apply Winter 1971 Addenda of ASME Section III, 1971 Edition.

Other tanks are designed, constructed and tested to meet the intent of API standards 620/650, AWWA Standard D100 or ANSI B96.1. Standard for Aluminum Tanks.

TABLE 3.2-3

CODE REQUIREMENTS FOR COMPONENTS AND QUALITY GROUPS FOR
PUBLIC SERVICE ELECTRIC & GAS COMPANY/BECHTEL-PROCURED COMPONENTS[1]

Component	Quality Group Classifications: Group A	Group B	Group C	Group D
Pressure vessels	ASME B&PV Code, Section III, Class 1	ASME B&PV Code, Section III, Class 2	ASME B&PV Code, Section III, Class 3	ASME B&PV Code, Section VIII, Division 1
Piping	ASME B&PV Code Section III, Class 1	ASME B&PV Code Section III, Class 2	ASME B&PV Code Section III, Class 3	ANSI B31.1
Piping supports	ASME B&PV Code Section III, Class 1	ASME B&PV Code Section III, Class 2	ASME B&PV Code Section III, Class 3	ANSI B31.1
Pumps	ASME B&PV Code Section III, Class 1	ASME B&PV Code Section III, Class 2	ASME B&PV Code Section III, Class 3	Manufacturer's standards
Valves	ASME B&PV Code Section III, Class 1	ASME B&PV Code Section III, Class 2	ASME B&PV Code Section III, Class 3	ANSI B31.1
Heat ex-changers	ASME B&PV Code Section III, Class 1	ASME B&PV Code, Section III, Class 2, TEMA-C	ASME B&PV Code, Section III, Class 3, TEMA-C	ASME B&PV Code, Section VIII Division 1
Atmospheric storage tanks	ASME B&PV Code Section III, Class 1	ASME B&PV Code, Section III, Class 2	ASME B&PV Code, Section III, Class 3	API-650, AWWAD 100, ANSI B96.1[2]

(1) Code effective date for construction of piping systems is 1974, including the 1974 Winter Addenda, with the following exceptions: (a) Installation of welded attachments to ASME Section III, Class 2 and 3 piping after hydrostatic testing is in accordance with ASME Section III, 1980 Edition, through 1981 Winter Addenda, Paragraph NC-4436 and ND-4436, respectively; (b) Installation of welded attachments to B31.1 piping after hydrostatic testing is in accordance with ANSI B31.1, 1980 Edition through 1982 Winter Addenda, Paragraph 137.1.2(D); (c) The provisions of NB-2510 in the 1983 Edition, Summer 1983 Addenda of ASME Section III apply to pipe, tubes and fittings 1" nominal pipe size and less. Code effective date for procurement of vessels, pumps, valves, heat exchangers, and storage tanks is that in effect upon award of the contract; (d) NC-3652, 3653, 3654, and 3655 of the 1983 Edition through Summer1984 Addenda or ND-3652, 3653, 3654 and 3655 of the 1983 Edition, Summer 1984 Addenda may be used for piping system analysis during As-Built Reconciliation (ABR); (e) ASME Code effective dates for piping supports vary between the 1974 and 1983 Editions through the Summer 1984 Addendum, as allowed by HCGS compliance with 10CFR50.55a. The applicable Code editions and addenda shall be specified on the purchase orders.

(2) 0-15 psig tanks may also be built to API-620.

3.3 WIND AND TORNADO LOADINGS

3.3.1 Wind Loadings

Design wind loads for all exposed structures are based on Bechtel Topical Report BC-TOP-3A, American National Standards Institute ANSI A58.1, and American Society of Civil Engineers Paper No. 3269, References 3.3-1, 3.3-2, and 3.3-3. Table 3.3-2 lists structures, systems, and components to which extreme wind design criteria are applied.

3.3.1.1 Design Wind Velocity

The design wind velocities are 108 mph (including a gust factor of approximately 1.3) at 30 feet above ground for Seismic Category I structures and 100 mph at 30 feet above ground for non-Seismic Category I structures. The recurrence interval of this wind velocity is estimated to be at least 100 years.

3.3.1.2 Determination of Applied Forces

The effective velocity pressure, as defined in Reference 3.3-2 takes into account the vertical velocity profiles and gust factors for different types of exposure and dynamic response characteristics of structures.

The force distribution and shape coefficients are based on the procedures in References 3.3-1, 3.3-2, and 3.3-3. Design wind loads on Seismic Category I structures are provided in Table 3.3-1.

3.3.2 Tornado Loadings

Seismic Category I structures exposed to the design basis tornado wind, or missiles associated with this wind, are designed so that they are not affected by these conditions to the extent that radioactivity releases result or capability of structures to protect

vital, functional equipment is reduced. Table 3.3-2 lists the systems and components that are protected against tornadoes, and the structures that provide this protection for safe shutdown purposes. Tornado loading is determined using methods and procedures outlined in Reference 3.3-1. The design parameters are defined below.

3.3.2.1 Applicable Tornado Design Parameters

Structures required to be tornado resistant are designed for the following effects of a design basis tornado:

1. Dynamic wind loadings - These are the external pressure or suction forces on a structure due to the passage of a tornado funnel. The design basis tornado has a maximum wind speed of 360 mph, a maximum rotational speed of 290 mph with a radius of 150 feet, a maximum translational speed of 70 mph, and a minimum translational speed of 5 mph.

2. Differential pressures - When the low pressure within a tornado funnel engulfs a structure, a rapid depressurization occurs and produces differential pressures between the inside and outside of the structure and between the compartments inside the structure depending on the available vent paths. The pressure transient caused by the design basis tornado is assumed to be a 3.0 psi pressure drop at a rate of 2.0 psi/s.

3. Tornado missiles - The types of missiles postulated to be generated by a tornado are discussed in Section 3.5.1.

3.3.2.2 Determination of Forces on Structures

The procedures used to transform the tornado loading into effective loads on structures are in accordance with Reference 3.3-1.

Combinations of extreme wind loads or tornado loads with other loads and maximum allowable values of stress are specified in Section 3.8.4.

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

Structures not designed for tornado loads are checked to ensure that during a tornado they do not generate missiles that have more severe effects than the tornado missiles discussed in Section 3.5.1. Miscellaneous yard buildings such as the Auxiliary Boiler Building, Circulating Water Chemical Control Building, and the cooling tower are at a minimum of 300 feet away from the main power block.

The modes of failure of these structures are analyzed to verify that their failure due to tornado loading cannot prevent structures, systems, and components needed for safe shutdown from performing their intended functions.

3.3.3 References

3.3-1 "Tornado and Extreme Wind Design Criteria for Nuclear Power Plants," BC-TOP-3A, Revision 3, August 1974.

3.3-2 "American National Standards Institute, American National Standard Building Code Requirements for Minimum Design Loads in Buildings and Other Structures," A58.1, 1972.

3.3-3 American Society of Civil Engineers, "Wind Forces on Structures," Paper No. 3269, 1961.

TABLE 3.3-1

DESIGN WIND LOADS ON SEISMIC CATEGORY I STRUCTURES

Height Zone (ft)	Dynamic Pressure, $q^{(1)}$ (psf)	WALL LOAD(2) Windward Pressure, 0.8 q (psf)	Leeward Suction, 0.5 q (psf)	Roof Load Suction 0.7 q (psf)
0-50	46	37	23	32
50-150	58	46	29	41
150-400	72	58	36	50
> 400	79	63	40	55

(1) Pressures are based on values from Table 5 (Exposure C) from Reference 3.3-3 and on a basic wind speed of 110 mph (including gust factors).

(2) The wall loadings presented in this table are adjusted by the following multiplication factors when applied to structure design:

Square or rectangular structures - 1.00

Round or elliptical structures - 0.60.

TABLE 3.3-2

SECURITY-RELATED INFORMATION
WITHHELD UNDER 10 CFR 2.390

TABLE 3.3-2 (Cont)

SECURITY-RELATED INFORMATION
WITHHELD UNDER 10 CFR 2.390

3.4 WATER LEVEL (FLOOD) DESIGN

3.4.1 Flood Protection

3.4.1.1 Flood Protection Measures for Seismic Category I Structures

The safety-related systems and components for which flood protection is provided are listed in Section 1.2 and identified in response to Position C.1 of Regulatory Guide 1.29, as described in Section 1.8. Flood protection of safety-related systems and components is provided for all postulated flood levels and conditions described in Section 2.4.

Structures that house safety-related equipment and offer flood protection to this equipment are identified on Figure 3.4-1. They include the Reactor Building, Auxiliary Building, and Station Service Water System (SSWS) intake structure.

A description of these structures is provided in Sections 3.8.4 and 3.8.5. Design basis flood elevations are identified in Table 3.4-1.

Seismic Category I structures that may be affected by design basis floods are designed to withstand the floods postulated in Section 2.4. The "hardened" flood protection approach is used to incorporate structural provisions into the design of the plant for protection of safety-related structures, systems, and components from the combined static and dynamic effects of a flood.

Safety-related systems and components are not affected by a flood when they are located above the postulated maximum flood level. When located below flood level, these systems and components are enclosed in reinforced concrete Seismic Category I structures that have:

1. Exterior wall thicknesses below flood level of not less than 2 feet.

2. Waterstops provided in exterior wall construction joints and seismic separation joints below flood level.

3. A minimum number of openings in exterior walls and slabs below flood level (these openings are designed to prevent intrusion of flood water.).

4. Water pressure tight doors installed in exterior walls below flood level.

5. Exposed equipment hatches installed above flood level; those below flood level installed behind exterior walls designed to prevent intrusion of water. One exception to this condition is the exterior hatch located at grade level in the north Radwaste Building. This hatch is designed to be water pressure tight.

6. Continuous waterproofing systems applied to the underside of base slabs and on exterior walls to grade, as discussed below.

Doors and hatches in exterior walls below flood elevation (including wave effects), are either provided with a sensor to alarm in the main control room or will be administratively controlled.

Except for the intake structure, the HCGS safety-related structures are provided with roof drainage systems capable of handling a maximum rainfall rate of 4 inches per hour for a period of 20 minutes.

The roof drainage system consists of roof drains and 6-inch diameter scuppers located 6 inches above the roof drain elevations. Supplementing the roof drain system is a series of openings in the parapets of the roofs of the buildings. The 6 hour, local, all

season PMP was used to size these openings. The PMP, which is 27.5 inches, is distributed into 5-minute increments such that the maximum amounts for durations of one hour, 30 minutes, 15 minutes and 5 minutes are 18.1, 13.7, 9.5 and 6 inches respectively. Roof elevations, subdrainage areas, and the dimension of parapet openings are shown in Table 3.4-3. A schematic of the roof drainage is shown on Figure 3.4-4.

The routing of the PMP assumes: no losses, the Roof Drain System is plugged, and ponding is allowed up to the limiting elevation of the top of the curb of each roof hatch within each roof drainage area system. Prior to the PMP, an initial level of ponding at the invert elevation of the parapet openings is assumed (invert elevation is 6 inches above the roof drain elevation).

Rectangular parapet openings are analyzed as a broad crested weir for upstream water surface elevations below the top of the opening. For unsubmerged conditions the rating curve for each rectangular parapet opening is derived using the weir equation:

$$Q = 3.0\ LH^{1.5}$$

where:

Q is the discharge in cubic feet per second,

L is the length of the parapet opening in feet,

H is the head in feet of water above the invert of the parapet opening.

For conditions when the upstream water surface elevation is higher than the top of the opening, the orifice equation is used:

$$Q = 0.6\ A\sqrt{2gh}$$

where:

Q is the discharge in cubic feet per second,

A is the area of the opening in square feet,

h is the head measured from the centerline of the opening in feet,

g is the acceleration of gravity (32.2 feet/second^2).

The flow capacity of the 8-inch diameter openings is derived using the following short-culvert equations:

Inlet control flow for unsubmerged inlets:

$$\frac{H}{D} = \frac{H}{D} c + K \left(1.273 \frac{Q}{D^{5/2}}\right) m$$

Inlet control flow for submerged inlets:

$$\frac{H}{D} = \frac{h}{D} 1 + K_1 \left(\frac{Q}{D^{5/2}}\right) 2$$

where:

H is the total head above the invert of the opening in feet,

H_c is the specific energy,

Q is the discharge in cubic feet per second,

D is the opening diameter in feet,

$k, m, \frac{h}{d}1$ and k_1 are inlet control performance coefficients. The experimentally determined values for a square-edged entrance are:

$k = 0.0098$

$m = 2.0$

$\frac{h}{d}1 = 0.67$

$k_1 = 0.0645$

Since the limiting water depths are greater than the ponding levels resulting from the PMP (as shown in Table 3.4-3), the ponding levels do not effect safety-related facilities.

The intake structure roof is designed without parapets or other continuous obstructions and is sloped to shed the water. Accordingly, no significant ponding will occur.

To prevent seepage into any Seismic Category I structure all roof openings are watertight and provided with either metal sleeves or concrete curbs of sufficient height to exceed any possible ponding levels.

As an additional margin of safety, all Seismic Category I roofs are designed to withstand a loading of 150 lb/ft^2, which is greater than the loading resulting from the maximum ponding on the roofs.

Doors and penetrations in exterior walls of the Auxiliary and Reactor Buildings are protected against water inflow up to

Elevation 127 feet for parts of the south exterior walls and up to Elevation 121 feet of other exterior walls. Interior drains from the radwaste areas are independently piped to the liquid waste disposal system and are not connected to the yard drainage system. Wall penetrations above Elevations 121 feet and 127 feet are designed to prevent roof spillage or heavy rain from seeping inside the building.

All personnel access closures, to areas where flood protection must be provided, are provided with submarine type doors which are watertight and open outward, with the exception of Doors 31B and 15B at Elevation 102'-0" (Col lines H, 35 and 25) these two doors swing inward. However, both doors have been designed for specified unseating pressure of 19 feet of water. This design ensures the required watertight integrity will be provided. To assure that these doors will not be inadvertently open during a flood event both doors are locked closed and administratively controlled.

Penetrations in exterior walls and slabs of the SSWS intake structure are protected against water inflow up to Elevation 121 feet for the north and east exterior walls and up to Elevation 128.5 feet for other exterior walls and slabs. As described in Section 2.4.2, the SSWS intake structure may be subjected to hurricane produced waves which could overtop the roof of the western portion of the structure at Elevation 128 feet. However, worst case water levels will not exceed the top of the wall at the air intake screens at Elevation 128.5 feet. Therefore, flood water will not enter into the dry area of the SSWS intake structure.

All exterior doors in Seismic Category I structures are designed to withstand the static and dynamic effects from postulated floods and the associated wave action. The design of each door is based on the more severe of the following two conditions:

1. The static head due to the maximum still water level plus the water rise from the wave effect at the location of the door (Table 2.4-10 and 2.4-10a).

2. The static head due to the combined still water and wave height plus the dynamic effects associated with a breaking wave, utilizing the fetch that maximizes the total loading on the door.

In the event the capacity of the yard drainage system is exceeded as a result of an unusually severe rainstorm, the excess water accumulates in puddles in the vicinity of catch basins and runs off as the storm subsides. No significant barriers exist such as road crowns, dikes or mounds that could appreciably increase the ponding levels adjacent to Seismic Category I structures. Water does not enter any safety-related structure, since the structures are watertight up to Elevations 121 feet or 127 feet. Therefore, no safety-related equipment is adversely affected as the result of severe rainstorms. Additional details on yard drainage and grading are provided in Plant Drawings C-0043-0, C-0044-0, C-0045-0 and C-0046-0.

The failure of non-Seismic Category I and non-tornado protected tanks, vessels, and major pipes located outside of buildings (Table 3.4-4) will not adversely affect safety-related structures, systems and components by flooding, as discussed below:

Failure of Tanks

The locations of tanks in the yard area are shown on Plant Drawing C-0001-0. Failure of the condensate storage tank, located on the south side of the power block (Table 3.4-4, Item 1), will not cause flooding. Any spillage due to failure of this tank will be contained within a reinforced concrete dike designed to be Seismic Category I, as discussed in Section 3.8.4.1.6.

The tanks located on the north and west sides of the power block (Table 3.4-4, Items 2 through 7) do not have Seismic Category I dikes around them. Failure of these tanks could cause local flooding. However, this flooding would not adversely affect safety-related facilities for the following reasons:

1. Any spillage will be conveyed to the Delaware River by means of overland surface runoff without adversely affecting any safety-related structures, systems or components by flooding. There is a clear path to the river from the building which will assure that any surface water will not enter the building. In addition, storm drainage is provided to facilitate conveyance of runoff to the river which will further minimize the potential for any local ponding.

2. Seismic Category I electrical cables and duct banks located in the vicinity of these tanks are protected against flooding. Electrical cables for HCGS have been tested for moisture absorption per ICEA Standard S-19-81 1979 paragraph 6.16.3. These cables in manholes and ductbanks will continue to perform properly even if the manholes and ductbanks are flooded. In addition, all buried electrical conduits that travel to electrical manholes outside the structures are sealed to prevent water from entering the structures through the electrical ductbank.

Failure of Cooling Tower Basin Wall (Table 3.4-4, Item 8)

The failure of the cooling tower basin wall would not adversely affect safety-related structures, systems and components, as discussed below:

The operating water level within the cooling tower basin is elevation 102.5 feet. The slabs and walls are conservatively designed for 3 feet of freeboard, allowing the water level to rise
to elevation 105.5 feet. The grade around the basin well is at elevation 104.5 which is 2 feet above the operating water level in the basin.

The worst case flooding could result from the unlikely "wash-off" of the soil on the south side of the tower. For this case, the run-off

would be dispersed and intercepted by the storm drainage system before it could reach the power block area or the water would flow overland to the Delaware River as discussed for tanks (Items 2 through 7). The Seismic Category I duct banks located between the intake structure and the power block will not be affected as they are not located in the flow path of the water.

Failure of Circulating Water Pipes (Table 3.4-4, Item 9)

Failure of these pipes within the yard area between the cooling tower basin and the Turbine Building will cause flooding of this area. Water from the damaged pipes will erode the soil cover and flood the yard. No Seismic Category I equipment or components are located in this area of possible erosion. The storm drainage system would eventually drain the water to the Delaware River or the water would flow overland to the Delaware River as discussed for tanks (Items 2 through 7).

In the most sever case, all the water from the cooling tower basin could drain through the damaged pipe into the yard area between the circulating water pumphouse and the Turbine Building. This could cause flooding of the lower level of the Turbine Building. However, safety-related systems and components would not be damaged, as discussed in Section 10.4.1.3.3.

Failure of Major Yard Piping

Failure of any of the pipes identified in Table 3.4-4, Items 10 to 14, may cause local flooding. However, the intensity and volume of water discharge from any of these pipes is less than that of the circulating water pipes discussed above and would not cause damage to any safety-related facilities. Soil erosion caused by failure of these pipes is discussed in Section 9.2.1.

In addition, buildings in the power block complex have a multi-ply waterproofing system between the leveling mat and the concrete

topping below the foundation mat, as shown on Figure 3.4-2. The system extends upward to grade on outside walls.

The concrete topping mat and base mat subgrade exterior construction joints are also treated with an additional waterproofing system.

Vertical and horizontal construction joints are provided with waterstops to Elevation 121 feet. Waterstops are provided to Elevation 124 feet and 127 feet for a portion of the south exterior walls of the auxiliary building and reactor building, respectively.

Seismic joints have additional backup waterstops from within the base mat up to grade level.

The waterstop materials are selected and designed to resist possible deterioration due to potential environmental effects. The waterstop material is styrene-butadiene synthetic rubber.

The waterstops are designed and tested to satisfy design conditions which are as follows:

1. Hydrostatic head of 90 feet.

2. Temperatures between -35°F and 200°F.

3. Radiation exposure of 2×10^6 rads cumulative over a period of 40 years.

4. Where waterstops cross seismic gaps between buildings, extensibility shall be sufficient to accommodate a relative displacement of at least 1 inch in each of the three principal axes simultaneously. (See Table 3.7-6 for computed relative movements between adjacent structures.)

5. Conform to Corps of Engineer's Standard CRD C513, except that elongation is 450 percent minimum and tensile strength is 2500 psi minimum.

6. The material of the water stop is selected to resist the soil and water chemistry present for the service life of the plant.

As an additional safety measure, any water that might conceivably enter a Seismic Category I structure is controlled by a system of floor drains to sumps equipped with automatic sump pumps discharging to holding tanks.

Safety-related structures are designed to resist overturning, sliding, or flotation.

In the unusual occurrence of a flood, the flood levels take time to develop, as discussed in Section 2.4. This development period provides time to perform the necessary emergency actions to secure all openings and maintain leaktightness.

The worst single access door left open is postulated to be the exterior door at Elevation 100 feet-6 at the east wall of Service Water Intake Structure. If the water could enter this door, it might render the Station Service Water System inoperative and prevent a continuous safe shutdown.

To preclude this event, procedures will be provided detailing guidelines to the plant personnel to ensure that all exterior doors are closed prior to any postulated flood-producing event, as listed in Table 2.4-6.

This procedure will also include the limiting values of the flood warning system indicators located at the site, at which value, a plant safe shutdown will be initiated by the plant personnel on their own volition.

The communications systems and procedures which will be utilized to alert both onsite and offsite Company and offsite public officials that an emergency action level has been reached due to potential

flood conditions, will be the same communications used to make notifications of emergency classifications (unusual events, alerts, etc.). These communications consist of direct pickup and ring telephones between the site and the States of Delaware and New Jersey and appropriate EPZ counties and as a backup, the normal Bell Telephone system. It should be noted that the decision to shutdown the Plant based on flood conditions can be made independent of offsite communications by the Shift Manager.

3.4.1.2 Means of Protecting Safety-Related Components From Internal Flood

Floor drainage systems are provided in each isolated compartment that houses a safety-related component or system. The lines are, in general, 4 inches in diameter and sloped a minimum of 1/8 in./ft.

These floor drainage systems are capable of handling potential normal leakage due to leaking pipe joints, valves, minor breaks, concrete cracking, etc. Potential flooding due to postulated failure of non-Seismic Category I tanks, vessels, and other process equipment, will be isolated in the respective compartments or an analysis has been performed to determine that the flooding consequences resulting from failures of such liquid carrying systems will not preclude required functions of safety systems. These compartments are designed to accommodate the water loads. In addition, high water level detectors and alarms are provided in compartments, shown on Plant Drawing M-25-1, where flooding may occur. When the level detectors are actuated by water level rising to a given elevation, they trip alarms in the main control room for follow-up action.

3.4.1.3 Permanent Dewatering Systems

Permanent dewatering systems are not required because safety-related structures extending below the design basis groundwater level are provided with waterproofing systems, as described in Section 3.4.1.1, and are designed to withstand the resulting

hydrostatic loading combined with other design loadings. The safety-related structures, systems, and components are protected from the effects of groundwater.

3.4.1.4 Internal Flooding Protection

Protection provided for the safety-related systems and components that may be subjected to internal flooding due to postulated piping failure is described in Section 3.6.

3.4.1.5 SRP Rule Review

The Hope Creek design for flood protection meets the intent of Section 3.4.1 of NUREG-0800. See Reference 3.4-2 regarding the probability of floating missiles impacting plant operation.

3.4.2 Analytical And Test Procedures

3.4.2.1 Design Parameters

The design parameters of the flood are described in Section 2.4.

3.4.2.2 Groundwater Effects

Under normal operating conditions, the effects of groundwater have been considered.

3.4.2.3 Flood Loading

The flood loading is treated as the extreme environmental load.

The lateral load due to flooding is developed using the method as delineated in Reference 3.4-1. The loads also accommodate the dynamic effect of the breaking waves, as in the case of the reactor building, or the nonbreaking waves at the SSWS intake structure, as appropriate.

The buoyancy force used for the uplift evaluation of the entire structure is based on the flood level, excluding wave action. However, the upward hydrostatic pressure on the foundation slab is based on the total head including wave action.

3.4.2.4 Load Combination

The appropriate load combinations using the effects of the flood are described in Sections 3.8.4 and 3.8.5.

3.4.3 Reference

3.4-1 U.S. Coastal Engineering Research Center, "Shore Protection Manual," 3rd edition, 1977.

3.4-2 R.L. Mittl, PSE&G, to A. Schwencer, NRC, "Floating Missiles", dated July 27 and September 17, 1984, and January 31, and February 22, 1985. R.L. Mittl, PSE&G, to W. Butler, NRC, "Floating Missiles", dated May 8 and September 16, 1985.

TABLE 3.4-1

FLOOD LEVELS AT SAFETY-RELATED STRUCTURES[6]

	Structure	Safety-Related System or Components Housed in Structure	Design Flood Elevation[1]	Elevation of Lowest Exterior Access Opening[2]	Type of Protection
1.	Reactor Building	Primary containment, ECCS, misc safety-related systems & components	119.0 ft[3]	102 ft	Water-pressure-tight doors
2.	Diesel Generator Building[4]	Diesel generators & misc safety-related systems and components	119.0 ft	102 ft	Water-pressure-tight doors
3.	Control Building[4]	Control systems & misc safety-related systems & components	-	-	-
4.	Radwaste Building[4]	Misc safety-related systems & components	119.0 ft[3]	102 ft	Water-pressure-tight doors
5.	SSWS intake structure	SSWS	119.0 ft[5]	102 ft	Water-pressure-tight doors

(1) Elevation shown is maximum design wave height, except per [3] below. Maximum design still-water height is at elevation 113.8 feet at all locations.
(2) Penetrations below the design flood elevation typically include electrical and piping penetrations that are designed to prevent intrusion of water.
(3) Maximum design wave height on south wall of Reactor and Radwaste Buildings, between 20 and 120 feet west of turbine building, is 124.4 feet.
(4) This is actually housed in the Auxiliary Building as shown on Figure 3.4-1.
(5) Maximum design wave height on south and west wall of the SSWS intake structure is 134.4 feet.
(6) Exterior walls of safety-related structures shall be designed to accommodate the wave loading conditions as summarized in Table 2.4-11a.

TABLE 3.4-2

THIS TABLE IS DELETED

TABLE 3.4-3

MAXIMUM PONDING DEPTHS ON ROOFS
OF SAFETY-RELATED STRUCTURES
FOR LOCAL 6 HOUR PMP

Roof No.(2)	Min. Roof Elevation (ft)	Sub-Drainage Area (ft^2)	Number of 8-inch Diameter Openings	Width of 8-inch High Slot (ft)	Width of Parapet Opening (ft)	Limiting Water Depth Over Roof Drain Elevation (in.)	Max. Water Depth Over Roof Drain Elevation (in.)
1	159	2720	-	2.5	-	12.0	11.5
2	137	2570	2	-	-	28.8	18.0
3	172	1530	2	-	-	15.0	13.6
4	153	1930	1	-	-	28.8	16.1
5	155.25	3700	-	-	50	12.0	11.9
6	172	38850	-	-	25	13.0	12.6
7	198	18420	-	-	35	10.0	9.8
8	155.25	3490	-	3.0	-	12.0	11.7
9	158.33	7380	-	2.5	-	19.0	18.1
10	172	5220	1	0.83	-	18.0	15.8
11	124	5030	2	-	-	18.0	17.5
12	132	33500	-	-	14	18.0	17.6

Notes:

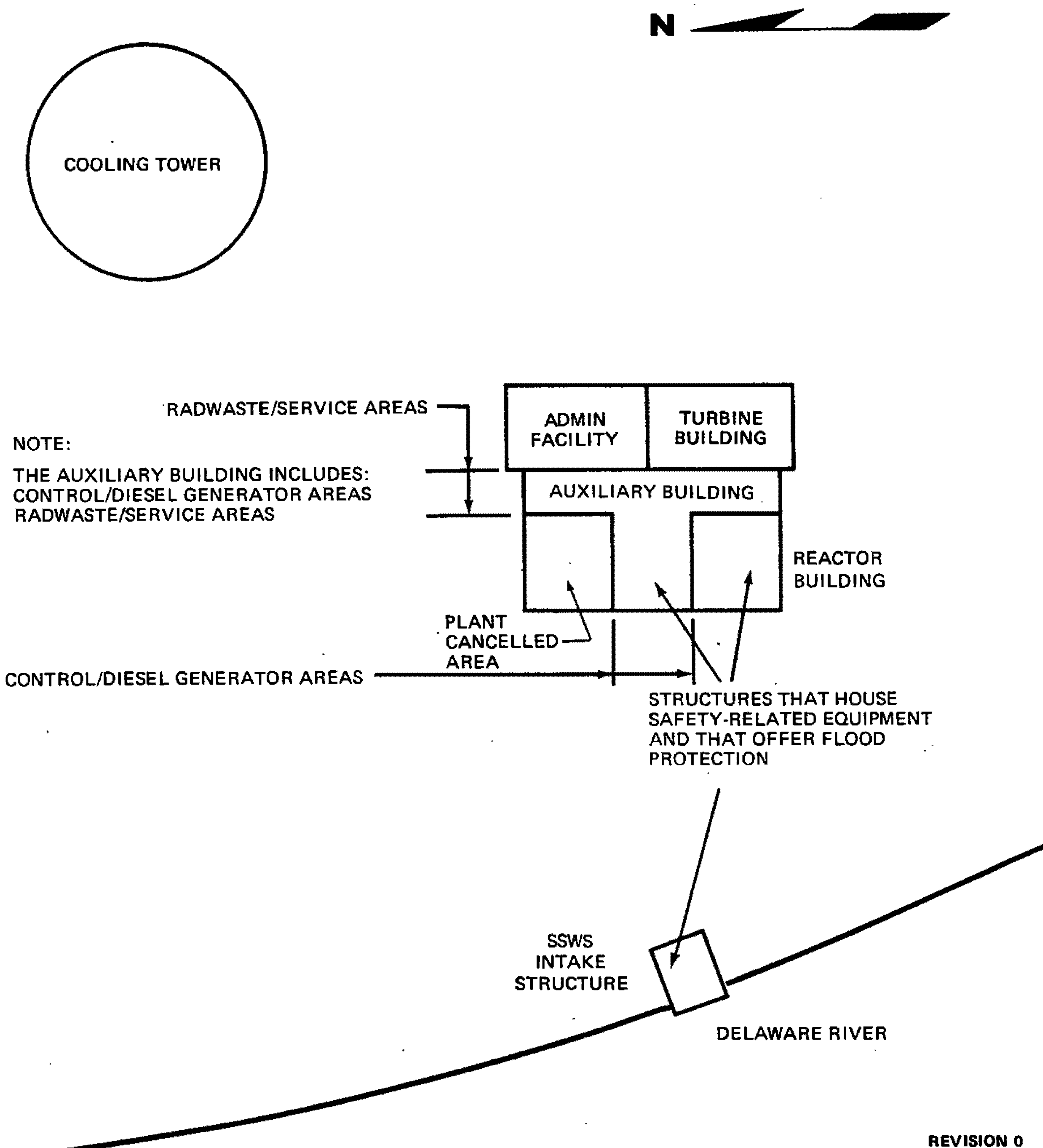
(1) The invert elevation of openings and the crest elevation of slots and parapet openings are 6 inches above the roof drain elevation.

(2) See Figure 3.4-4.

TABLE 3.4-4

YARD TANKS AND MAJOR PIPING (NON-SEISMIC)

Item No.	Tank or Pipe Description	Capacity or Flow	Location	Type of Containment	Tornado Protection
1	Condensate Storage Tank	500,000 gal	South of power plant complex	Seismic Cat. I Reinforced Conc. walls	None
2	Fire Water Tanks (2)	300,000 gal ea	North of power plant complex	None	None
3	Asphalt Storage Tank	9,000 gal	North of power plant complex	Concrete unit Masonry walls	None
4	Fuel Oil Day Tank	18,000 gal	North of power plant complex	Reinforced Conc. walls	None
5	Chemical Treatment Tanks				
	2 Sodium Hypochlorite	30,000 gal ea	North of power plant complex	Reinforced	None
	1 Sulfuric Acid	20,000 gal	North of power plant complex	Concrete	None
	2 Sodium Hypochlorite	15,000 gal ea	West of power plant complex	Walls	None
6	Sewage Treatment Plant				
	1 Equalization Tank	20,000 gal	North of power plant complex	Buried	None
	2 Treatment Tanks	8,000 gal ea	North of power plant complex	Buried	None
	1 Treatment Tank	35,000 gal	North of power plant complex	Earth berm	None
7	Fuel Oil Storage Tank	1,000,000 gal	North of power plant complex	Earth dike	None
8	Cooling Tower Basin	6,500,000 gal	North of power plant complex	Reinforced Conc. wall	None
9	144"0 Circulating Water Pressure	552,000 gpm	Between cooling tower and turbine building	Underground	Soil cover
10	48"0 Makeup Water Pressure Pipe	30,000 gpm	Reactor building to cooling tower	Underground	Soil cover
11	36"0 Makeup Water Pressure Pipe	21,000 gpm	Reactor building to cooling tower	Underground	Soil cover
12	48"0 Blowdown Water Gravity Pipe	15,400 gpm	Cooling tower to Delaware River	Underground	Soil cover
13	36"0 Deicing Water Pressure Pipe	12,000 gpm	Circulating water pipe to intake structure	Underground	Soil cover
14	12"0 Fire Water Loop	2,500 gpm	Around plant complex	Underground	Soil cover



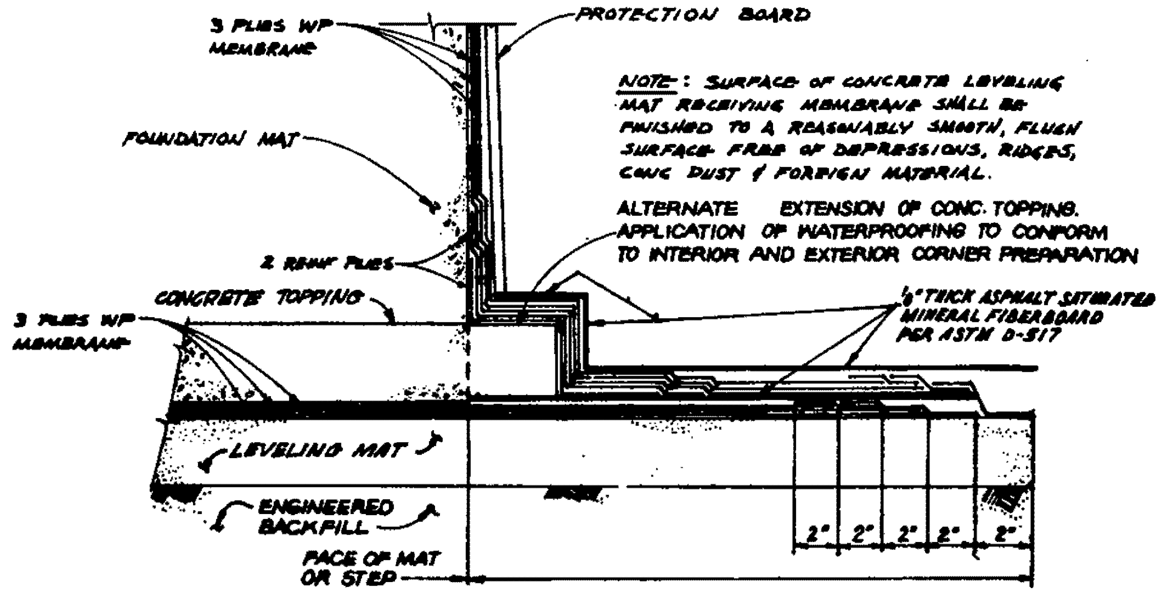
REVISION 0
APRIL 11, 1988

PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

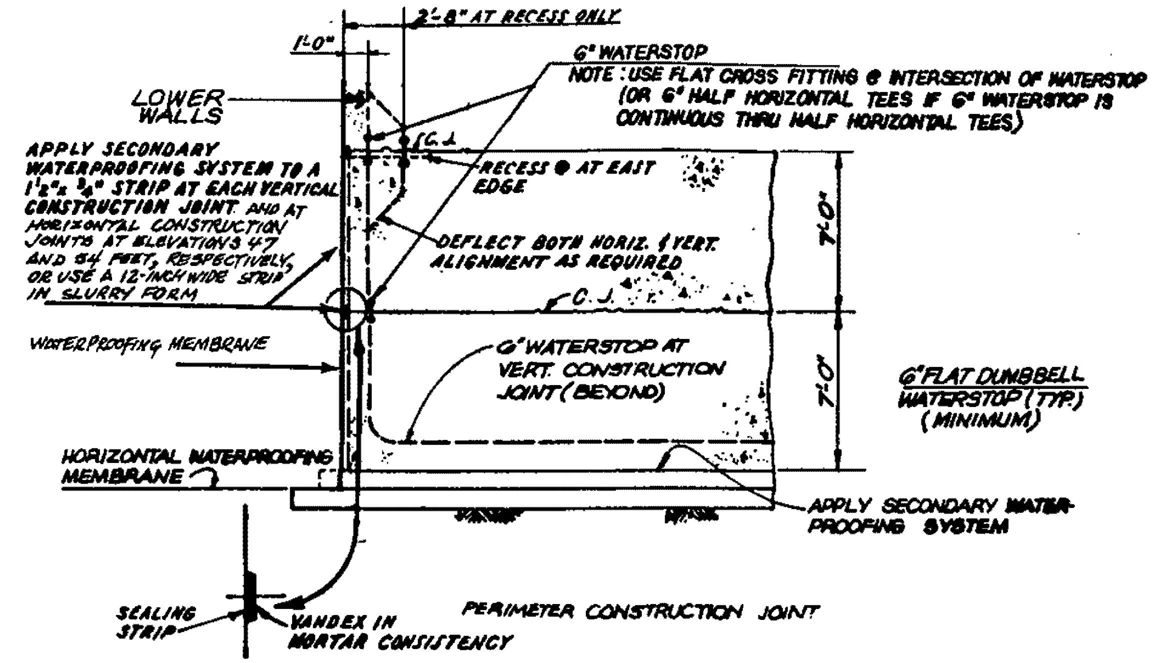
STRUCTURES THAT PROVIDE FLOOD PROTECTION

UPDATED FSAR

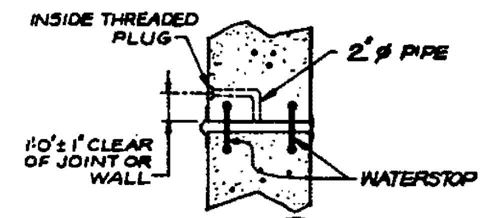
FIGURE 3.4-1



TYPICAL HORIZONTAL WATERPROOFING APPLICATION & EXTERIOR TERMINATION



TYPICAL CONSTRUCTION JOINT DETAILS



SECTION A
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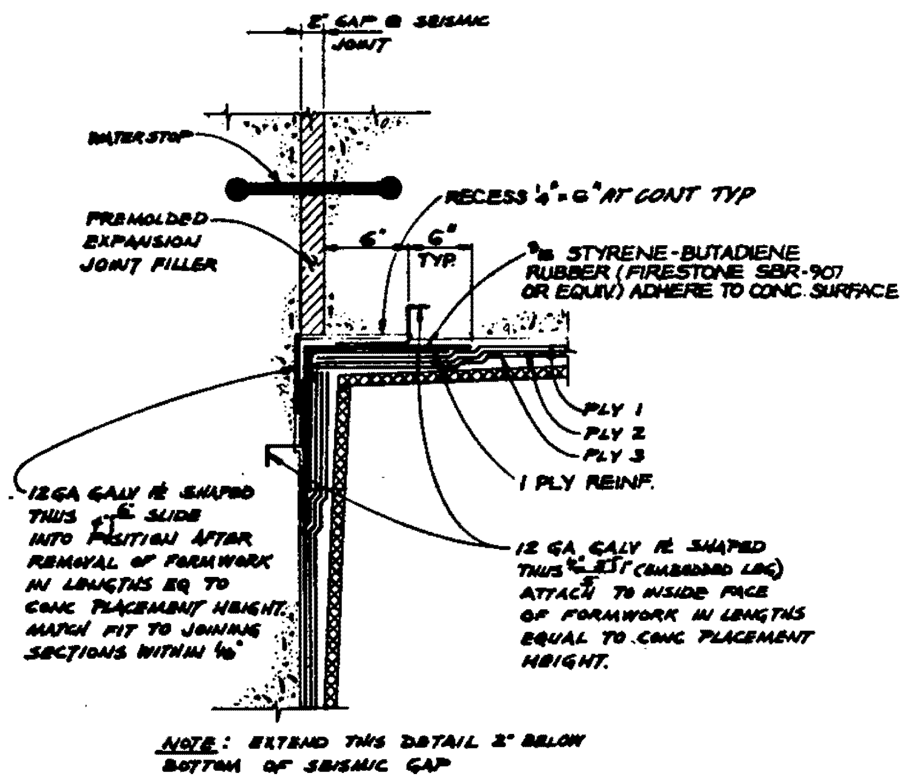
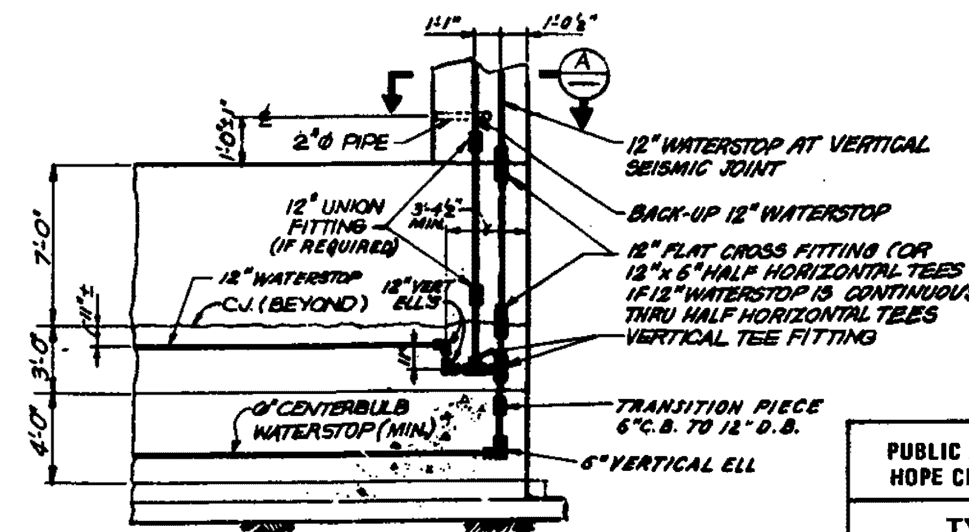


SECTION IN PLAN OF SEISMIC JOINT IN LOWER WALLS OR BASE MAT ©



TYPICAL SEISMIC JOINT DETAILS

REVISION 0
APRIL 11, 1988

PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

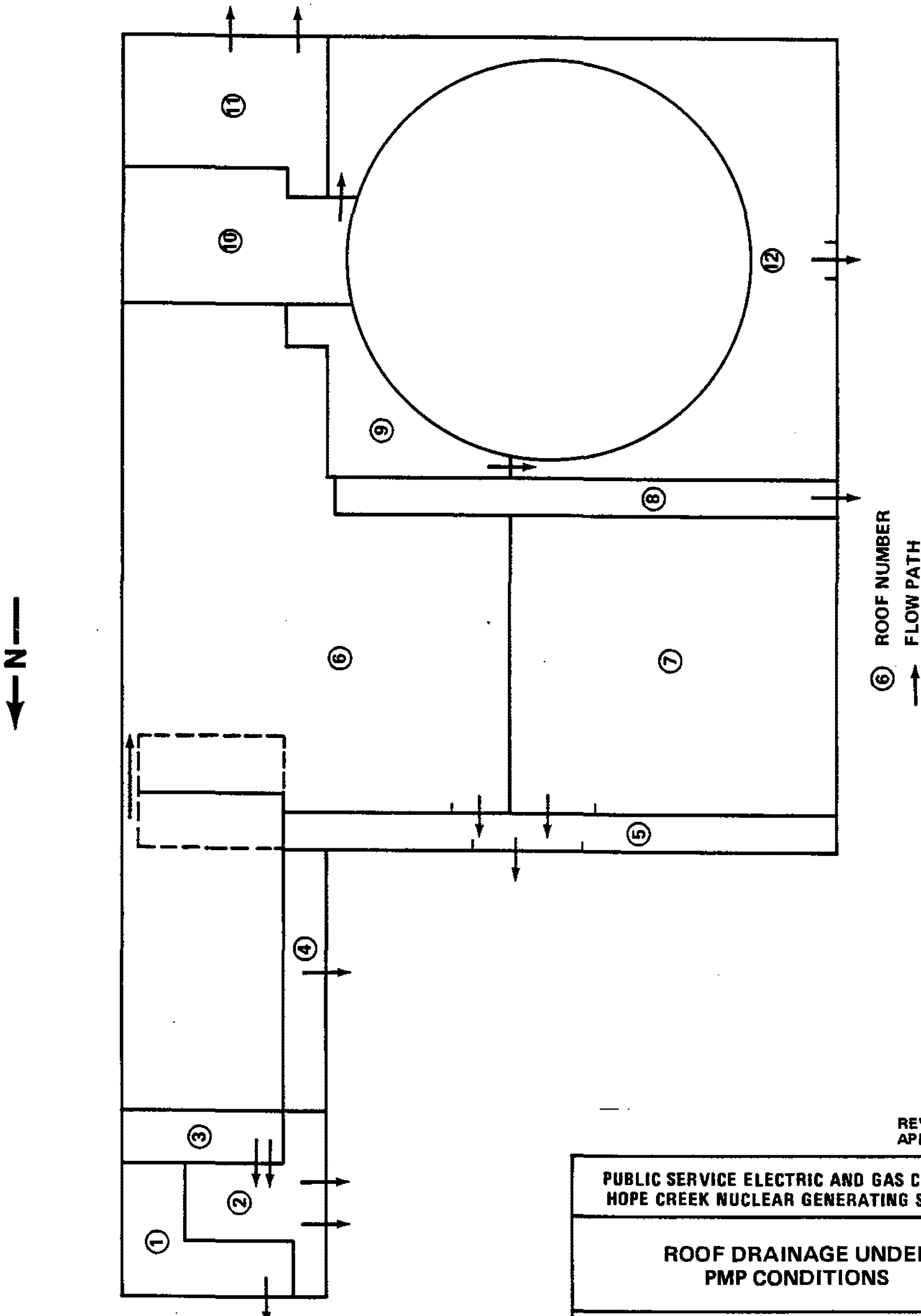
TYPICAL WATERPROOFING DETAILS FOR FLOOD PROTECTION

UPDATED FSAR

FIGURE 3.4-2

Figure F3.4-3 SH 1-4 intentionally deleted.

Refer to Plant Drawing C-0043-0 for sheet 1 in DCRMS
Refer to Plant Drawing C-0044-0 for sheet 2 in DCRMS
Refer to Plant Drawing C-0045-0 for sheet 3 in DCRMS
Refer to Plant Drawing C-0046-0 for sheet 4 in DCRMS



REVISION 0
APRIL 11, 1988

PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

ROOF DRAINAGE UNDER
PMP CONDITIONS

UPDATED FSAR

FIGURE 3.4-4

3.5 MISSILE PROTECTION

The Seismic Category I and safety-related structures, equipment, and systems are protected from postulated missiles through basic plant arrangement so that a missile does not cause the failure of systems that are required for safe shutdown or whose failure could result in a significant release of radioactivity. Where it is impossible to provide protection through plant layout, suitable physical barriers are provided to shield the critical system or component from credible missiles. Redundant safety-related Seismic Category I components are arranged so that a single missile cannot simultaneously damage a critical system component and its backup system.

A tabulation of safety-related structures, systems, and components, their locations, seismic category, quality group classification, and the applicable FSAR sections is given in Table 3.2-1. General arrangement drawings are included as Plant Drawings P-0001-0 and P-0072-0.

3.5.1 Missile Selection and Description

3.5.1.1 Internally Generated Missiles (Outside Primary Containment)

The systems located outside the primary containment have been examined to identify and classify potential missiles. These systems and missiles are listed in Tables 3.5-1 and 3.5-13. Redundant systems are normally located in different areas of the plant or separated by missile proof walls so that a single missile can not damage both systems.

The diesel generator structures provide protection from missiles generated by internal rotating or pressurized mechanisms to prevent any credible missiles from damaging more than one engine. Any postulated missiles from a crankcase explosion are expected to be of low energy and incapable of penetrating the concrete barrier into

any other cell. Failure of one or all of the starting air receivers by explosion is not expected to produce any credible missiles capable of penetrating the 18 inches thick cell walls.

The residual heat removal (RHR) and core spray pumps, are located in separate missile proof compartments and their impellers are enclosed in a concrete structure. Therefore they are not considered a potential missile source or hazard to other systems.

Refer to Section 3.5.3 for barrier design procedure.

There are three general sources of postulated missiles:

1. Rotating component failure

2. Pressurized component failure

3. Gravitationally generated missiles.

3.5.1.1.1 Rotating Component Failure Missiles

Catastrophic failure of rotating equipment having synchronous motors, e.g., pumps, fans, and compressors that could lead to the generation of missiles is not considered probable. Massive and rapid failure of these components is improbable because of the

conservative design, material characteristics, inspections, and quality control during fabrication and erection. Also, the rotational speed is limited to the design speed of the motor, thereby precluding component failures due to runaway speeds.

Similarly, it is concluded that the high pressure coolant injection (HPCI) and reactor core isolation cooling (RCIC) pumps and turbines cannot generate credible missiles. These pumps are not in continuous use, but are periodically tested and otherwise operate only in the unlikely event of a postulated accident. They are classified as moderate energy systems. Overspeed tripping devices ensure that the turbines do not reach runaway speed, where failure leading to the ejection of a missile could take place.

A tabulation of missiles generated by postulated failures of rotating components, their sources and characteristics, and a safety evaluation are provided in Table 3.5-13.

The evaluation identified one instance where a postulated missile, which could penetrate through the flexible connection of a vane axial fan, could have the potential to damage safe shutdown equipment in the room. In order to prevent the postulated missile from damaging safe shutdown equipment, a missile shield has been added to the design to withstand the impact of the postulated fan blade missile.

The formulas used to predict the penetration resulting from missile impact are provided in Reference 3.5-4. The penetration and perforation formulas assume that the missile strikes the target normal to the surface, and the axis of the missile is assumed parallel to the line of flight. The rotating component is assumed to fail at 120 percent overspeed. These assumptions result in a conservative estimate of local damage to the target.

3.5.1.1.2 Pressurized Component Failure Missiles

The following are potential internal missiles from pressurized equipment:

1. Valve bonnets

2. Valve stems

3. Temperature detectors

4. Nuts and bolts

5. Blind flanges

6. Accumulators

7. Gas (O_2, N_2, etc) bottles

8. Unrestrained sections of piping.

Pressurized components in systems where service pressure exceeds 275 psig are evaluated as to their potential for becoming missiles, except for the systems which are down graded and evaluated as moderate energy systems per FSAR Section 3.6.

Safety/relief valves (SRVs) and valve headers are not considered 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 break of the header, no missile would result.

All valves constructed in accordance with Section III of the ASME B&PV Code, and with an ANSI rating of 900 psig and above, are pressure seal bonnet valves. Pressure seal bonnet valves are prevented from becoming missiles by the retaining ring, which would

have to fail in shear, and by the yoke, which either captures the bonnet or reduces the bonnet energy. Because of the highly conservative design of the retaining ring of these valves, bonnet ejection is highly improbable, and therefore the bonnets are not considered credible missiles.

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 as set forth in Section III of the ASME B&PV Code, 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.

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 are effectively restrained by the valve operators.

Temperature or other detectors installed on high energy piping or in wells are evaluated as potential missiles if a single circumferential weld failure could 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, cannot hit any redundant safety-related equipment.

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

Bolted blind flanges are not considered credible missiles because of the extremely unlikely occurrence of all bolts experiencing simultaneous complete severance failure.

Pressurized vessels, such as SRV and main steam isolation valve (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. Therefore, any failures would be slot type and not considered a credible source of missile generation.

Pressurized bottles containing noncondensable gases, e.g., air, nitrogen, CO_2, etc, are seismically supported and therefore do not present a hazard as falling objects. The valve on top of the bottle is the only source for a postulated missile. No safety-related equipment is located in the postulated direction of a possible missile ejection. CO_2 fire extinguishers are not seismically supported, but have been evaluated for seismic II/I interaction and do not present a missile hazard from falling off its support during a seismic event.

Unrestrained sections of piping, such as vents, drains, and test connections, are evaluated as potential missiles if they are part of a high energy piping system, and the failure of a single circumferential weld could cause their ejection. The effects of these missiles are considered minimal due to the full separation of the high energy piping systems by compartments.

3.5.1.1.3 Gravitationally Generate Missiles

Equipment and components installed in safety-related plant areas outside primary containment are designed and installed so that they do not present gravitational missile hazards to safety-related structures, systems, or components during or after a SSE. This is achieved for safety-related equipment by Seismic Category I design and installation (in accordance with Regulatory Guide 1.29 criteria) and for non-safety-related equipment by Seismic II/I design and installation (Section 3.2.1). Non-permanently installed equipment

is either:

- removed from the safety-related areas,
- secured in place, or
- evaluated (if not secured)

before reactor operation to ensure that it does not become dislodged and present a missile hazard.

3.5.1.2 Internally Generated Missiles (Inside Primary Containment)

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

1. Rotating component failure
2. Pressurized component failure
3. Gravitationally generated missiles.

3.5.1.2.1 Rotating Component Failure Missiles

The most substantial pieces of Nuclear Steam Supply System (NSSS) rotating equipment are the recirculation pumps and motors. This potential missile source is covered in detail in Reference 3.5-1.

It is concluded in Reference 3.5-1 that destructive pump overspeed can result in certain types of potential missiles, but that no damage is possible to any safety-related equipment because these missiles cannot escape from the interior of either the pump or the motor.

With regard to evaluation of the probabilistic consequences of pump impeller missiles ejected from pipe breaks, it is concluded in Attachment 3 of Reference 3.5-1 that no damage is possible to the primary containment, any major piping system, or an inboard MSIV. No damage would occur because trajectories of postulated missiles do not intersect these systems.

Other rotating components inside the primary containment, such as fans, do not have sufficient energy to move the masses of their

rotating parts through the housings in which they are contained, and therefore are not considered missile hazards.

3.5.1.2.2 Pressurized Component Failure Missiles

It is concluded that potential internal missiles generated by pressurized components inside the primary containment are not considered credible for the reasons given in Section 3.5.1.1.2. Thermowells and unrestrained piping are not considered potential missiles.

3.5.1.2.3 Gravitationally Generated Missiles

Equipment and components installed in the primary containment are designed so that they do not present gravitational missile hazards to safety-related structures, systems or component during or after a SSE. This is achieved for safety-related equipment by Seismic Category I design and for nonsafety-related equipment by Seismic Category II/I design, as discussed in Section 3.2. Temporary equipment is either removed from the containment, or is secured in place before reactor operation to ensure that it does not become dislodged and present a missile hazard.

3.5.1.3 Turbine Missiles

The original LP rotors on the Hope Creek turbine generator set were replaced with monoblock rotor forgings. Therefore, the missile analysis issued previously considering an SCC failure mechanism no longer applies. In the monoblock rotor, the stress levels at the design point are conservative and the stress concentration associated with wheel keys no longer exists.

If the unit trips, valves fail to operate and full flow steam remains, the maximum possible speed the rotors can obtain is about 220% running speed, assuming that all steam path components on the rotor remain in place. This is the point at which the driving forces in the steam path are countered by the drag forces and can no longer accelerate the rotors. The rotor overspeed capability, with the assumption that all buckets remain in place, is 225% for typical rotor strengths. Therefore, rotor missiles will not be generated. A complete failure of the control and safety systems is required for this to occur and is very unlikely. The probability of a control failure of this nature is approximately 10^{-8} per year. In conclusion, given the low stress levels monoblock rotors and the elimination of the wheel SCC mechanism, the probability of generating rotor missiles is not present.

See References 3.5-22 and 23 for information regarding the Hope Creek turbine system maintenance program.

3.5.1.3.1 Turbine Placement and Orientation

Figure 3.5-1 shows the locations of turbines in the vicinity of the HCGS site. These include the Salem Generating Station turbines as well as the HCGS turbine.

HCGS structures are located well outside the low trajectory missile strike cone (~25°) for the Salem turbines. Based on Regulatory Guide 1.115, Protection Against Low Trajectory Turbine Missiles, it is concluded that HCGS is adequately protected from missiles that may be generated by the Salem turbines. Consequently, these potential missiles are excluded from the probability analysis. The HCGS turbine has an in-line arrangement. Figure 3.5-1 historically shows the low trajectory strike zone for the turbine.

Structures, systems, and components considered in the analysis are those required to ensure the integrity of the reactor coolant pressure boundary (RCPB), the capability to shutdown the reactor and maintain it in a cold shutdown condition, and the capability to prevent accidents that could result in offsite releases that violate 10CFR50.67 guidelines. Redundant safety-related systems and components that are separated to preclude damage to both trains are considered adequately protected against turbine missiles. The following structures, systems, and components are identified for consideration in the analysis:

1. Main control room

2. Fuel pool

3. Hydraulic control units (HCUs)

4. Drywell

5. Suppression chamber

6. Standby diesel generators (SDGs)

7. Plant areas containing cables for safety-related systems with insufficient separation to prevent damage to both trains by a single missile. No areas outside of those listed in 1. through 6. have been identified.

A detailed list of turbine missile targets and their relevant parameters is historically given in Table 3.5-2. The target locations are historically shown on Figures 3.5-6 and 3.5-10 and Plant Drawings P-0001-0 through P-0004-0, P-0006-0, P-0007-0 and P-0011-0.

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3.5.1.4 Missiles Generated by Natural Phenomena

Only tornado generated missiles have been considered. Tornado generated missiles considered are listed in Table 3.5-12. Refer to Section 3.5.3 for tornado generated missiles.

3.5.1.5 Missiles Generated by Events Near the Site

As discussed in Section 2.2.3, Evaluation of Potential Accidents, there is no credible basis for anticipating site proximity missiles.

3.5.1.6 Aircraft Hazards

3.5.1.6.1 General

Aircraft operations near the HCGS site are described in Section 2.2.2.5. This information indicates that detailed investigation is needed to consider the potential hazards from: operations at the Greater Wilmington Airport, along three federal airways; miscellaneous flights conducted under visual flight rules near the plant; and the PSE&G helicopter operations to the HCGS plant site.

A detailed study of the probability of potentially unacceptable impact of aircraft on the safety-related structures at HCGS has been performed. In general, the probability of impact is given by:

$$P = \sum_{i=1}^{T} \sum_{i=1}^{J} N_{ij} \quad C_i \quad A_i \quad P_{ij} \tag{3.5-74}$$

where:

P = Probability of an unacceptable impact, per year

N_{ij} = Operation of aircraft of type i along airway or from airport j, per year

C_i = Crash rate for aircraft of type i, per mile

A_i = Effective impact area for aircraft of type i, mi^2

ρ_{ij} = Aircraft crash density at plant site for aircraft of type i operating along airway or from airport j, per mile.

The values of various parameters used in the analysis, and the bases for their determination, are discussed in the following paragraphs.

3.5.1.6.2 Number of Aircraft Operations

The total number of aircraft operations near the HCGS site includes every category of aircraft that flies near the site. Because the potential for damage to the plant is a function of the aircraft gross weight, speed, etc, and the crash frequency is a function of the type of aircraft, the number of operations is divided into categories by type of aircraft and/or type of flying.

The number of normal aircraft operations in this analysis is divided into five categories, with the PSE&G company helicopter as a sixth category. The categories of aircraft operations are:

1. General aviation, single engine

2. General aviation, multi-engine

3. Commuter air carrier and on-demand air taxi (less than 12,500 lbs)

4. Air carrier (over 12,500 lbs)

5. Military

6. PSE&G helicopter

3.5.1.6.2.1 Airports - Greater Wilmington

The Greater Wilmington Airport, because of its past and forecast number of operations, as reported in Section 2.2.2.5, must be considered. Each operation at an airport is either a takeoff or a landing, and can be either local or itinerant, as discussed in Reference 3.5-11. Local operations are those performed by aircraft that: 1) operate in the local traffic pattern or within sight of

Security Related Information
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IFR operations on the low level airways. This number of operations is shown in Table 3.5-8. The number of VFR operations near the HCGS site, as determined from the radar survey, are proportioned by the relative number of hours flown by single and multiengine general aviation aircraft nationally, as given in Reference 3.5-15; or 75 percent by single engine and 25 percent by multi-engine, as shown in Table 3.5-8. The total number of operations shown in Table 3.5-8 by the other categories of aircraft, including commuter air carrier and on-demand air taxi, air carrier, and military, are based on a count of the FAA Philadelphia TRACON operations below 9000 feet, found in Reference 3.5-14, and on the Washington and New York ARTCC flight progress strips above 9000 feet. The annual number of PSE&G helicopter VFR operations to the HCGS site is as estimated by PSE&G.

3.5.1.6.3 Crash Rates

The crash rates used are based on National Transportation Safety Board (NTSB) crash data and FAA flight data. Accident rates which might cause significant damage are based on historical records of crashes that result in fatalities. Nonfatal accidents are not considered severe enough to cause significant damage to the plant structures. In-flight or enroute fatal crash rates are used, since the plant is more than 5 miles from the nearest airport. The PSE&G helicopter operations to the site are evaluated using NTSB fatal crashes that occur during takeoff and landing operations. The crash rates per mile for each category of aircraft considered, as well as the helicopter crashes per operation, are summarized in Table 3.5-9.

3.5.1.6.3.1 General Aviation Small Fixed Wing Single and Multi-Engine Aircraft

For all general aviation operations in 1979, the NTSB reported, in Reference 3.5-17, fatal crash rates of 1.45×10^{-7} per hour for single engine aircraft and 9.9×10^{-8} per hour for multi-engine aircraft. Of these, approximately 47 percent were enroute. Using average speeds estimated from FAA data in Reference 3.5-15, the resulting fatal in-flight crash rates are 6.82×10^{-8} crashes per

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3.5.1.6.3.5 Helicopter

The NTSB reported five fatal helicopter crashes on takeoff and landing during 1979, and the FAA estimated 2.55 million helicopter hours for 1979, as indicated by Reference 3.5-17. The average flight time for helicopter operations is conservatively estimated to be 30 minutes, based on Reference 3.5-18. These two operations (takeoff and landing) per half-hour helicopter flight results in 1.02×10^{7} takeoffs or landings for 1979 (2.56×10^{6} flight hours x 4 operations/hour). This produces a fatal crash rate of 4.88×10^{-7} per takeoff or landing for 1979 (five divided by 1.02×10^{7}).

[redacted]

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V = crash decay rate = 2 mi^{-1} for general aviation small fixed-wing aircraft

These crash density values have been used in determining overall aircraft impact probability and are shown in Table 3.5-10.

3.5.1.6.6 Helicopter Impact Methodology

Public Service Electric and Gas Company expects to conduct approximately 700 helicopter operations annually at the HCGS and Salem Generating Station heliport, as indicated by Reference 3.5-20. A helicopter is very maneuverable and has other flight characteristics different from the conventional aircraft. Therefore, the probability of a helicopter impacting a safety-related structure on the HCGS is treated separately.

[redacted]

[redacted]

The rotors provide the only lift mechanism for a helicopter (vis-à-vis a wing for a conventional aircraft). Any failure of the rotor lift device would limit the lateral distance a helicopter could travel out of control away from the traffic pattern path(s).

The possible deviation from the bounding traffic pattern for landing or takeoff path is represented by an exponential density function, noted in Reference 3.5-19, of the form:

$$f(x;\lambda) = \lambda e^{-\lambda x} \qquad (3.5\text{-}77)$$

Where: $f(x;\lambda)dx$ gives the probability that the deviation is in the range $(x,x+dx)$; x is the perpendicular distance (deviation) from the flight path; and λ is the exponential distribution parameter. λ is

the inverse of the expected value of x, i.e., λ is equal to 1/x. x has been chosen as the mean distance which the helicopter can travel from an altitude h. Thus x equals h (tan A), where A is an angle of 45°. The expected altitude, h, of the helicopter on an approach or departure from the helipad is h = a(Tan B), where a is the distance from the helipad and B is the approach angle of 15°, as shown on Figure 3.5-19.

On the basis of the above:

$$\lambda = 1/ [a(\tan A)(\tan B)] = 1/ [.268\ a] \qquad (3.5\text{-}78)$$

The probability of impacting at a distance greater than d perpendicular to the flight path on one side of the flight path, is then given by the cumulative distribution function:

$$P(>d) = 1/2e^{-\lambda|d|} = 1/2e^{-|d|/.268a} \qquad (3.5\text{-}79)$$

[redacted]

$$P(>700) = 1/2e^{-700/.268(250)} = 1.45\times10^{-5} \qquad (3.5\text{-}80)$$

This is conservative because the conditional probability of impacting the Auxiliary Building, which is greatest at a distance of 250 feet from the helipad, is used to determine the probability of an impact by a helicopter from any point on the takeoff and departure path. Using this conservative approximation, the expected annual frequency of a fatal crash of a helicopter on takeoff or landing, leading to an impact on the Auxiliary Building which could cause unacceptable damage, may be estimated by the product of the

number of operations per year, multiplied by the national average rate of fatal crashes per takeoff and landing, multiplied by the conditional probability of crashing into the auxiliary building. The expected frequency is less than:

$$(700) \times (4.88\times10^{-7}) \times (1.45\times10^{-5}) = 4.95 \times 10^{-9}/\text{year} \quad (3.5\text{-}81)$$

Therefore, helicopter operations to the HCGS heliport are no credible hazard to the plant.

3.5.1.6.7 Conclusions

The parameters used to calculate impact probability are summarized in Tables 3.5-8 through 3.5-10. The results of the probability calculation are summarized in Table 3.5-11. The probability of an aircraft strike with a potential for causing radiological consequences in excess of the exposure guidelines of 10CFR100 is 6.7×10^{-8} per year. Therefore, aircraft accidents do not constitute design basis events.

3.5.2 Structures, Systems, and Components to be Protected from Externally Generated Missiles

Structures and barriers designed to provide protection from postulated external missiles are discussed in Section 3.3.2. Table 3.3-2 lists systems and components that are protected, together with the protecting structures.

All external doors which have safety-related equipment or cables behind them are capable of withstanding the effects of external missiles generated by natural phenomena.

The Seismic Category I electrical manhole is designed to withstand the effects of tornado missile impact. The manhole cover is an 18 inch thick concrete slab held in place by anchor bolts. The cover

is designed without a personnel manhole access. Access to the Seismic Category I manhole is provided by removing the concrete cover with a crane.

For missiles generated by events near the site, refer to Section 3.5.1.5.

The Station Service Water System, which supplies cooling water from the ultimate heat sink, is protected from externally generated missiles as described in Section 3.3. Yard piping from the intake structure to the plant is protected by being buried under at least 10 feet of earth cover. The routing of the buried service water pipes and the profile and details are shown in Plant Drawings C-0091-0 and C-0094-0. The location of the safety-related equipment and its relative arrangement is shown on Plant Drawings P-0001-0 through P-0007-0, P-0010-0, P-0011-0 and P-0012-0.

Openings through which missiles could potentially pass exist in the roofs and exterior walls of Seismic Category I structures. Safety-related systems and components are protected from direct impact by interior walls, slabs or other structures, which are designed to resist the effects of externally generated missile impacts. Grating or metal decking has been provided to prevent falling debris from damaging safety-related systems or components.

[redacted]

3.5.3 Barrier Design Procedures

Structures and barriers are designed to resist internal and external missile impact effects based on the following postulated missiles striking exposed surfaces:

1. Internal missiles - see Table 3.5-1

2. External missiles (tornado missiles) - see Table 3.5-12.

Missile barrier design is in accordance with the procedures detailed in Reference 3.5-4. The procedures include:

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

2. Estimation of barrier thickness required to prevent perforation

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

[redacted]

Typical details for missile protection of openings in exterior walls and roofs are shown in Figures 3.5-20 through 3.5-28.

3.5.3.1 SRP Rule Review

See Section 3.8.4.8 for a discussion of compliance with SRP Section 3.5.3.

3.5.4 References

3.5-1 General Electric, "Analysis of the Recirculation Pump Under Accident Conditions," Revision 2, Letter Report, March 30, 1979.

3.5-2 Deleted

3.5-3 Deleted

3.5-4 Bechtel, "Design of Structures for Missile Impact," Revision 2, BC-TOP-9A, September 1974.

3.5-5 Deleted

3.5-6 R.B. Barber, "Steel Rod/Concrete Slab Impact Test (Experimental Simulation)," Bechtel, October 1973.

3.5-7 F.A. Vasallo, "Missile Impact Testing of Reinforced Concrete Panels," prepared for Bechtel, Calspan Corp, January 1975.

3.5-8 Deleted

3.5-9 National Transportation Safety Board Annual Review of Aircraft Accident Data, U.S. Air Carrier Operations, Calendar Year 1979, Report No. NTSB-ARC-81-1, November 16, 1981.

3.5-10 National Transportation Safety Board Briefs of Accidents Involving Commuter Air Carrier/On-Demand Air Taxi Operations, U.S. General Aviation, Calendar Year 1979, NTSB-AMM-81-11, October 16, 1981.

3.5-11 T. F. Henry, "Terminal Area Forecasts Department of Transportation," Federal Aviation Administration, FAA-APD-80-10, February 1981.

3.5-12 Personal communication with J. Graham, FAA Flight Standards Field Unit, Greater Wilmington Airport and K. Toth, NUS Corporation, September 8, 1982.

3.5-13 The Defense Mapping Agency Aerospace Center, "United States Government Flight Information Publications (FLIP)," June 1982.

3.5-14 FAA TRACON Flight Strips, 7/18/82, 7/19/82, 7/21/82, and 8/2/82 from John Furling Philadelphia Approach Control, Terminal Radar Control (TRACON).

3.5-15 FAA Statistical Handbook of Aviation, Calendar Year 1979, U.S. Department of Transportation, Federal Aviation Administration, December 1981.

3.5-16 Federal Aviation Administration (FAA) Air Traffic Activity FY 1981, September 30, 1981.

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3.5-18 H. E. P. Krug, "Testimony on Aircraft Operations in Response to a Request from the Board," D69E Docket 50-275 and 50-323.

3.5-19 K. A. Salomon, "Analysis of Ground Due to Aircraft and Missiles," Hazard Prevention, Journal of the System Safety Society, Vol 12 number 4, March/April 1976.

3.5-20 Personal communication with J. James, PSE&G pilot, and K. Toth, NUS Corporation, September 13, 1982.

3.5-21 Department of Transportation, "Helicopter Design Guide, Federal Aviation Administration, ACNo: 150/5390-1B, August 22, 1977."

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3.5-23 NUREG-1048, HCGS SSER No. 6, Appendix U, "NRC Safety Evaluation regarding the Probability of Missile Generation in General Electric Nuclear Turbines", July 1986.

3.5-24 General Electric, “Thermal Kit for LP Monoblock Steam Path Replacement Only,” Revision 0, February 17, 2004.

3.5-25 General Electric, GE-NE 0000-0000-5735-01, “Task T0700, Turbine-Generator Performance Evaluation – Final Task Report,” Revision 0, Project Task Report, March 2003.

3.5-26 General Electric, “Missile Probability Assessment, Nuclear BWR Controls Retrofit,” Revision A, March 19, 2004.

TABLE 3.5-1

INTERNALLY GENERATED PRESSURIZED COMPONENT MISSILES OUTSIDE PRIMARY CONTAINMENT

System	FSAR Section	Missile Description	Protection Evaluation Codes(1)
HPCI	6.3	Test connection	c
		Startup flange	c
		Pressure indicator (PI-R003)	c
Main steam (portion inside Reactor Building only)	5.1	Test connections	c
		Temperature elements (TE-N040)	c
		Pressure indicators (PP-3632 A, B, C, D)	c
Main steam drain	5.1	Temperature elements (TE-N057 A, B, C, D, E)	c
		Pressure transmitter (PT-5838 A, B)	c
		Blind flange or Y-strainer	c
		Test connection	c
		Temperature element (TE-N060)	c
Feedwater (portion inside Reactor Building only)	5.1	Test connections	c
		Drains	c
		Vent	c
		Blind flange	c
		Temperature sensors/elements (TE-N007, TE-N019, TE-N006, TE-N015, TE-N004, TS-169, TS-170, TS-242 A, B)	c
		Pressure transmitter (PT-N005)	c
		Pressure point (PP-3876 A, B; PP-3875 A, B; PP-3916 A, B; PP-3917 A, B)	c
		Pressure indicators (PI-3877 A, B; PI-R009; PI-R003; PI-R004; PI-R008; PDIS-3987 A, B; PDIS-3988 A, B; PDIS-3993)	c
		Pressure switches (PSL-N013)	
		Flow elements (FE-3986 A, B)	c
RCIC	6.3	Test connection	c
		Pressure indicator (PI-R003)	c

---

(1) Protection evaluation codes:

a. Wall or floor mounted missile barrier
b. Pipe-mounted missile barrier
c. Missile is contained within a subcompartment
d. Consequences of the missile are acceptable
e. Vertically-mounted integral missile barrier.

TABLE 3.5-2

SECURITY-RELATED INFORMATION
WITHHELD UNDER 10 CFR 2.390

SECURITY-RELATED INFORMATION
WITHHELD UNDER 10 CFR 2.390

TABLE 3.5-3

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TABLE 3.5-4

THIS TABLE INTENTIONALLY DELETED

TABLE 3.5-5

SECURITY-RELATED INFORMATION
WITHHELD UNDER 10 CFR 2.390

TABLE 3.5-6

SECURITY-RELATED INFORMATION
WITHHELD UNDER 10 CFR 2.390

TABLE 3.5-7

COMPUTED PROBABILITIES

Target Description[1]	Design Overspeed P2	Design Overspeed P2xP3	Design Overspeed P3	Design Overspeed P4	Destructive Overspeed P2	Destructive Overspeed P2xP3	Destructive Overspeed P3	Destructive Overspeed P4	Total P2	Total P2xP3	Total P3	Total P4
A Drywell	1.9-2[2]	0	0	0	1.6-2	0	0	0	1.8-2	0	0	0
B Spent fuel pool	7.2-3	1.7-4	2.3-2	1.0-8	6.1-3	1.2-4	2.0-2	4.9-9	6.8-3	1.5-4	2.2-2	1.5-8
C Control structure	2.6-3	1.5-3	5.8-1	8.9-8	1.8-3	1.2-3	6.6-1	4.7-8	2.3-3	1.4-3	6.0-1	1.4-7
D Standby diesel generator	2.7-3	2.3-3	9.5-1	1.4-7	2.3-3	2.0-3	8.4-1	7.8-8	2.6-3	2.2-3	8.8-1	2.2-7
E South HCU	1.8-3	2.2-4	1.3-1	1.3-8	1.6-3	3.8-4	2.4-1	1.5-8	1.7-3	2.9-4	1.7-1	2.8-8
F North HCU	1.8-3	5.9-4	3.2-1	3.5-8	1.5-3	6.1-4	4.0-1	2.4-8	1.7-3	5.9-4	3.5-1	5.9-8
G Suppression chamber compartment	2.0-2	7.7-4	3.8-2	4.6-8	1.7-2	1.2-3	7.4-1	4.8-8	1.9-2	9.4-4	5.0-2	9.4-8
TOTAL	5.5-2	5.6-3	1.0-1	3.3-7	4.6-2	5.4-3	1.2-1	2.2-7	5.2-2	5.5-3	1.1-1	5.5-7

(1) Target locations are shown on Figures 3.5-2 through 3.5-10.
(2) Read as 1.9×10^{-2}.

TABLE 3.5-8

SUMMARY NUMBER OF OPERATIONS

	IFR		VFR			
	V123-312 & J150	V29-157	Within 1 Mile	1-2 Miles	2-3 Miles	3-5 Miles
GA-SFW[1] Single Engine	2200	5000	150	430	150	525
GA-SFW Multi-Engine	4800	2900	50	145	50	175
Commuter Air Carrier/On Demand Air Taxi (<12,500 lbs)	3700	8800	0	0	0	0
Air Carrier (>12,500 lbs)	116200	3100	0	0	0	0
Military (>12,500 lbs)	1600	1700	200	0	0	0
Helicopter (PSE&G)			700			

(1) GA-SFW - General aviation small fixed wing

TABLE 3.5-9

CRASH RATES PER MILE AND EFFECTIVE IMPACT AREA
BY CATEGORY OF AIRCRAFT

Category of Aircraft	Crash Rate, $mi.^{-1}$	Effective Impact Area, $mi.^2$
GA-SFW[1] Single Engine	6.82×10^{-8}	4.82×10^{-4}
GA-SFW Multi-Engine	4.66×10^{-8}	8.97×10^{-4}
Commuter Air Carrier/On Demand Air Taxi (<12,500 lbs)	1.75×10^{-8}	8.97×10^{-4}
Air Carrier (>12,500 lbs)	5.27×10^{-10}	0.0212
Military (>12,500 lbs)	2.64×10^{-9}	0.0227
Helicopter (Crashes per takeoff and landing operation)	4.88×10^{-7}	6.3×10^{-4}

(1) GA-SFW = General aviation small fixed wing

TABLE 3.5-10

AIRCRAFT CRASH DENSITY BY LOCATION/ROUTE/ALTITUDE

Location or Route & Altitude	Crash Density, mi^{-1}
VFR Within 1 Mile of Site	0.135
VFR 1-2 miles	0.018
VFR 2-3 miles	$2.5x10^{-3}$
VFR 3-5 miles	$4.3x10^{-5}$
VFR military over site	0.296
V123-312 - 7000'	0.048
V29-157 - 9000'	0.037
V123-312 & J150 - 20,000'	0.017
V29-157 - 16,000'	0.021

TABLE 3.5-11

PROBABILITY SUMMARY

	C (Crashes/Mi)	N (No. of Operations)	A_2 (Mi^2)	ρ (Mi^{-1})	P	P (Totals)
GA-SFW(1) Single Engine						
VFR						
w/in 1 mile	$6.82x10^{-8}$	150	$4.82x10^{-4}$	.1353	$6.67x10^{-10}$	
2 mile		430		.0183	$2.59x10^{-10}$	$9.39x10^{-10}$
3 mile		150		.0025	$1.23x10^{-11}$	
5 mile		525		$4.54x10^{-5}$	$7.89x10^{-13}$	
IFR						
TRACON-V123-312, J150 @ 7000 ft	$6.82x10^{-8}$	616	$4.82x10^{-4}$	.0480	$9.72x10^{-10}$	
TRACON-V29-157 @ 9000		4448		.0373	$5.45x10^{-9}$	$9.43x10^{-9}$
ARTCC-V123-312, J150 @ 7000		1551		.0480	$2.45-10^{-9}$	
ARTCC-V29-157 @ 9000		456		.0373	$5.59x10^{-10}$	
GA-SFW Multi Engine						
VFR						
w/in 1 mile	$4.66x10^{-8}$	50	$8.97x10^{-4}$	.1353	$2.83x10^{-10}$	
2 mile		145		.0183	$1.11x10^{-10}$	
3 mile		50		.0025	$5.22x10^{-12}$	$4.0x10^{-10}$
5 mile		175		$4.54x10^{-5}$	$3.32x10^{-13}$	
IFR						
TRACON-V123-312, J150 @ 7000 ft	$4.66x10^{-8}$	205	$8.97x10^{-4}$	.048	$4.11x10^{-10}$	
TRACON-V29-157 @ 9000		1483		.0373	$2.31x10^{-9}$	$1.42x10^{-8}$
ARTCC-V123-312, J150 @ 7000		4654		.048	$9.34x10^{-9}$	
ARTCC-V29-157 @ 9000		1369		.0373	$2.13x10^{-9}$	
IFR Commuter Air Carrier and On-Demand Air Taxi (<12,500 lbs)						
TRACON-V123-312, J150 @ 7000 ft	$1.75x10^{-8}$	1916	$8.97x10^{-4}$	.048	$1.44x10^{-9}$	
TRACON-V29-157 @ 9000		7026		.0373	$4.11x10^{-9}$	
ARTCC-V123-312, J150 @ 16000		1825		.021	$6.02x10^{-10}$	$6.75x10^{-9}$
ARTCC-V29-157 @ 16000		1825		.021	$6.02x10^{-10}$	
IFR Air Carrier (>12,500 lbs)						
TRACON-V123-312, J150 @ 7000 ft	$5.27x10^{-10}$	182	.0212	.048	$9.76x10^{-11}$	
TRACON-V29-157 @ 16000		1642		.021	$3.85x10^{-10}$	
ARTCC-V123-312, J150 @ 20000		116070		.017	$2.2x10^{-8}$	$2.28x10^{-8}$
ARTCC-V29-157 @ 16000		1460		.021	$3.42x10^{-10}$	

TABLE 3.5-11 (Cont)

VFR Military Aircraft (>12,500 lbs)	2.64×10^{-9}	200	.0227	.296	3.55×10^{-9}	
IFR Military Aircraft						
TRACON-V123-312, J150 @ 7000 ft	2.64×10^{-9}	91	.0227	.048	2.62×10^{-10}	
TRACON-V29-157 @ 16000		1004		.021	1.26×10^{-9}	
ARTCC-V123-312, J150 @ 20000		1460		.017	1.49×10^{-9}	7.48×10^{-9}
ARTCC-V29-157 @ 16000		730		.021	9.19×10^{-10}	
			Conventional aircraft total		=	6.2×10^{-8}/yr
	Crashes/Operation	(Operation/year	Conditional P		P	
Helicopter	4.88×10^{-7}	700	1.45×10^{-5}		4.95×10^{-9}	4.95×10^{-9}
				Total	=	6.7×10^{-8}/yr

(1) GA-SFW = General aviation small fixed wing

TABLE 3.5-12

TORNADO MISSILES

Missile[1]	Dimensions	Weight, lbs	Velocity, ft/s
a. Wood plank	3-5/8 in. x 11-5/8 in. x 12 ft 0 in.	114.6	272.3
b. 6-in schedule 40 pipe	6-5/8 in. diameter x 15 ft 0 in.	286.6	170.6
c. 1-in steel rod	1 in. diameter x 3 ft 0 in.	8.8	167.3
d. Utility pole	13.5 in. diameter x 35 ft 1/2 in.	1124.4	180.4
e. 12-inch schedule 40 pipe	12.6 in. diameter x 15 ft 0 in.	749.6	154.2
f. Automobile	16 ft 4-7/8 in. x 6 ft 6-3/4 in. x 4 ft 3-1/4 in.	3990.4	193.6

(1) Velocities are horizontal velocities. For vertical velocities, 70 percent of the horizontal velocities are acceptable, except for missile c. above, which is 100 percent in all directions. Missiles a., b., c., and e. are to be considered at all elevations, and missiles d. and f. at elevations up to 30 feet above all grade elevations within 0.5 miles of the facility structure.

Table 3.5-13

Security Related Information Text Withheld under 10 CFR 2.390

TABLE 3.5-13 (Cont)

MISSILE IDENTI-FICATION	SOURCE OF MISSILE	LOCATION	MISSILE CHARACTERISTICS VELOCITY (FT/S)	DIA. (IN.)	WEIGHT (LBs)	CALCULATED MAX. STEEL PERF. DEPTH (IN.)	CASING THICKNESS	REMARKS

Security Related Information
Text Withheld under 10 CFR 2.390

TABLE 3.5-13 (Cont)

MISSILE IDENTIFICATION	SOURCE OF MISSILE	LOCATION	MISSILE CHARACTERISTICS VELOCITY (FT/S)	DIA. (IN.)	WEIGHT (LBs)	CALCULATED MAX. STEEL PERF. DEPTH (IN.)	CASING THICKNESS	REMARKS

Security Related Information
Text Withheld under 10 CFR 2.390

TABLE 3.5-13 (Cont)

MISSILE IDENTI-FICATION	SOURCE OF MISSILE	LOCATION	MISSILE CHARACTERISTICS VELOCITY (FT/S)	DIA. (IN.)	WEIGHT (LBs)	CALCULATED MAX. STEEL PERF. DEPTH (IN.)	CASING THICKNESS	REMARKS

Security Related Information
Text Withheld under 10 CFR 2.390

TABLE 3.5-13 (Cont)

MISSILE IDENTI- FICATION	SOURCE OF MISSILE	LOCATION	MISSILE CHARACTERISTICS VELOCITY (FT/S)	DIA. (IN.)	WEIGHT (LBs)	CALCULATED MAX. STEEL PERF. DEPTH (IN.)	CASING THICKNESS	REMARKS

Security Related Information
Text Withheld under 10 CFR 2.390

TABLE 3.5-13 (Cont)

MISSILE IDENTI- FICATION	SOURCE OF MISSILE	LOCATION	MISSILE CHARACTERISTICS VELOCITY (FT/S)	DIA. (IN.)	WEIGHT (LBs)	CALCULATED MAX. STEEL PERF. DEPTH (IN.)	CASING THICKNESS	REMARKS

Security Related Information
Text Withheld under 10 CFR 2.390

TABLE 3.5-13 (Cont)

MISSILE IDENTI-	SOURCE OF	LOCATION	MISSILE CHARACTERISTICS VELOCITY	DIA.	WEIGHT	CALCULATED MAX. STEEL PERF. DEPTH	CASING THICKNESS	REMARKS

Security Related Information
Text Withheld under 10 CFR 2.390

TABLE 3.5-13 (Cont)

MISSILE IDENTI-FICTION	SOURCE OF MISSILE	LOCATION	MISSILE CHARACTERISTICS VELOCITY (FT/S)	DIA. (IN.)	WEIGHT (Lbs)	CALCULATED MAX. STEEL PERF. DEPTH (IN.)	CASING THICKNESS	REMARKS

Security Related Information
Text Withheld under 10 CFR 2.390

TABLE 3.5-13 (Cont)

MISSILE IDENTI-FICATION	SOURCE OF MISSILE	LOCATION	MISSILE CHARACTERISTICS VELOCITY (FT/S)	DIA. (IN.)	WEIGHT (LBs)	CALCULATED MAX. STEEL PERF. DEPTH (IN.)	CASING THICKNESS	REMARKS

Security Related Information
Text Withheld under 10 CFR 2.390

Security Related Information Figure
Withheld under 10 CFR 2.390

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APRIL 11, 1988

PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

LOW TRAJECTORY STRIKE ZONES

UPDATED FSAR

FIGURE 3.5-1

Figure F3.5-2 intentionally deleted.

Refer to Plant Drawing P-0001-0 in DCRMS

Figure F3.5-3 intentionally deleted.

Refer to Plant Drawing P-0002-0 in DCRMS

Figure F3.5-4 intentionally deleted.

Refer to Plant Drawing P-0003-0 in DCRMS

Figure F3.5-5 intentionally deleted.

Refer to Plant Drawing P-0004-0 in DCRMS

SECURITY - RELATED
INFORMATION WITHHELD
UNDER 10 CFR 2.390

REVISION 14, JULY 26, 2005

PSEG Nuclear, LLC HOPE CREEK NUCLEAR GENERATING STATION	Hope Creek Nuclear Generating Station TARGET AND BARRIER GEOMETRY
	Updated FSAR Figure 3.5-6

Figure F3.5-7 intentionally deleted.

Refer to Plant Drawing P-0006-0 in DCRMS

Figure F3.5-8 intentionally deleted.

Refer to Plant Drawing P-0007-0 in DCRMS

Figure F3.5-9 intentionally deleted.

Refer to Plant Drawing P-0011-0 in DCRMS

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INFORMATION WITHHELD
UNDER 10 CFR 2.390

PEER REV

Ref. Dwg. P-0012-0 sh.1 Rev. 6

PSEG NUCLEAR, L.L.C. HOPE CREEK GENERATING STATION	
TARGET AND BARRIER GEOMETRY	
Updated FSAR REVISION 10, MAY 30, 1999	Sheet 1 of 1 Fig. 3.5-10

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Revision 14, July 26, 2005

PSEG Nuclear, LLC HOPE CREEK NUCLEAR GENERATING STATION	Hope Creek Nuclear Generating Station Updated FSAR Figure 3.5-11

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Revision 14, July 26, 2005

PSEG Nuclear, LLC HOPE CREEK NUCLEAR GENERATING STATION	Hope Creek Nuclear Generating Station Updated FSAR Figure 3.5-12

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Revision 14, July 26, 2005

PSEG Nuclear, LLC HOPE CREEK NUCLEAR GENERATING STATION	Hope Creek Nuclear Generating Station Updated FSAR Figure 3.5-14

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Revision 14, July 26, 2005

PSEG Nuclear, LLC HOPE CREEK NUCLEAR GENERATING STATION	Hope Creek Nuclear Generating Station
	Updated FSAR Figure 3.5-15

THIS FIGURE HAS BEEN DELETED

Revision 14, July 26, 2005

PSEG Nuclear, LLC HOPE CREEK NUCLEAR GENERATING STATION	Hope Creek Nuclear Generating Station Updated FSAR Figure 3.5-16

SECURITY - RELATED
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PEER
REV

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PSEG NUCLEAR, L.L.C.
HOPE CREEK GENERATING STATION

HELICOPTER GLIDE PATHS

Updated FSAR

Fig. 3.5-17

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UNDER 10 CFR 2.390

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PSEG NUCLEAR, L.L.C.
HOPE CREEK GENERATING STATION

HELICOPTER GLIDE DISTANCES

Updated FSAR Fig. 3.5-18

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INFORMATION WITHHELD
UNDER 10 CFR 2.390

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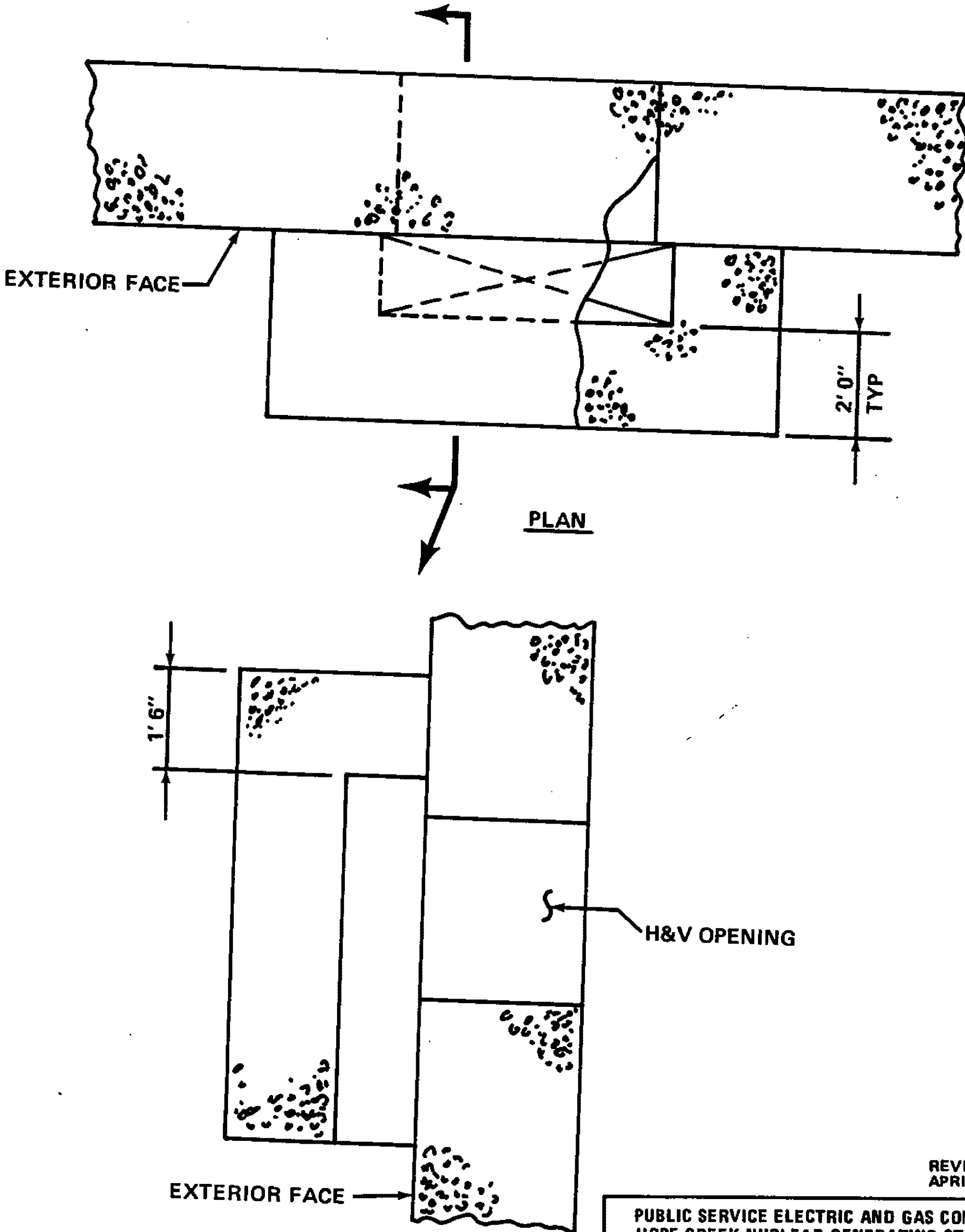
PSEG NUCLEAR, L.L.C. HOPE CREEK NUCLEAR GENERATING STATION
SCHEMATIC OF HELIPORT FLIGHT PATH, AND BUILDING OF CONCERN
Updated FSAR Fig. 3.5-19



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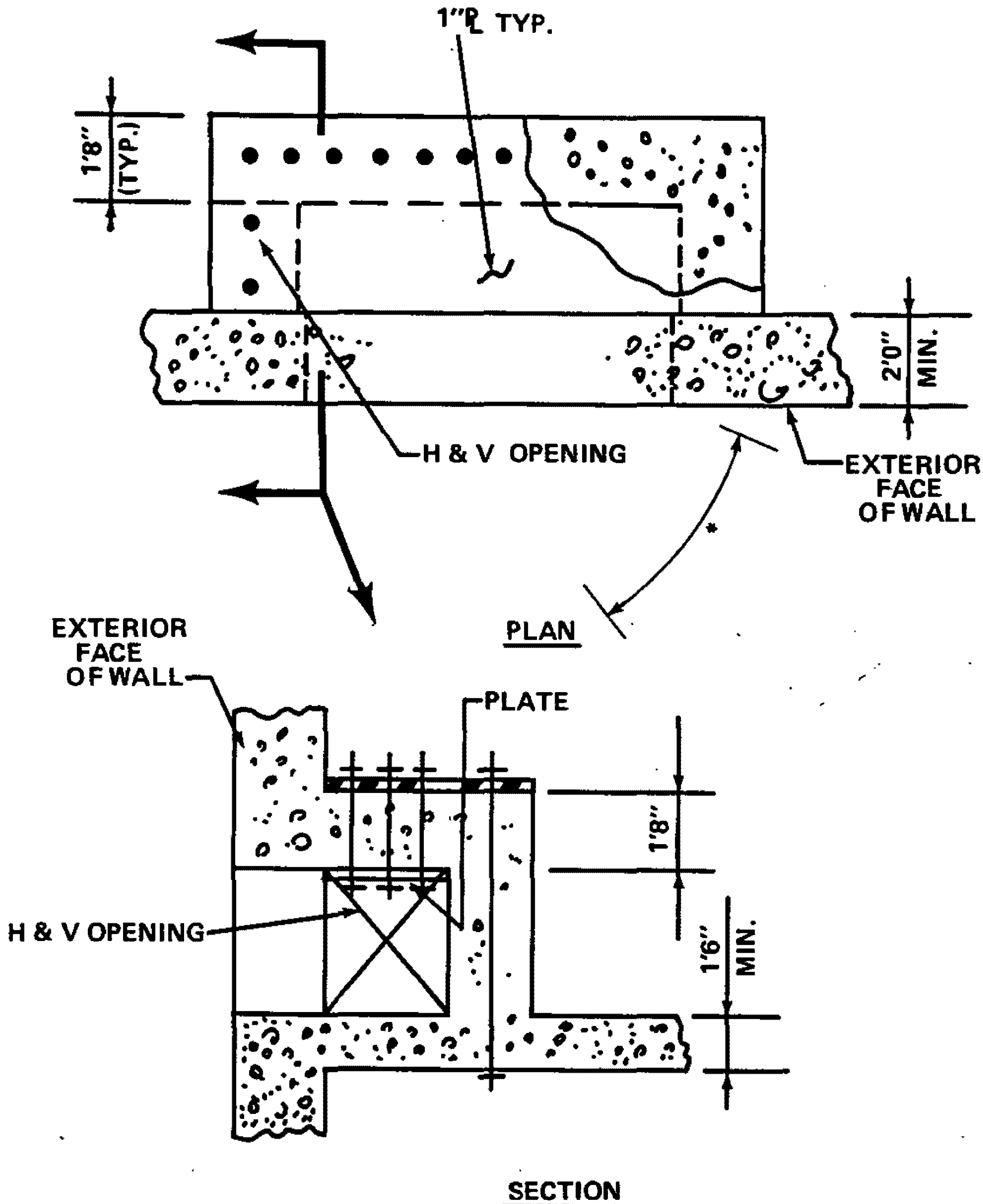
HOPE CREEK NUCLEAR GENERATING STATION

WALL OPENING
MISSILE BARRIER
TYPE – W1

UPDATED FSAR

FIGURE 3.5-20



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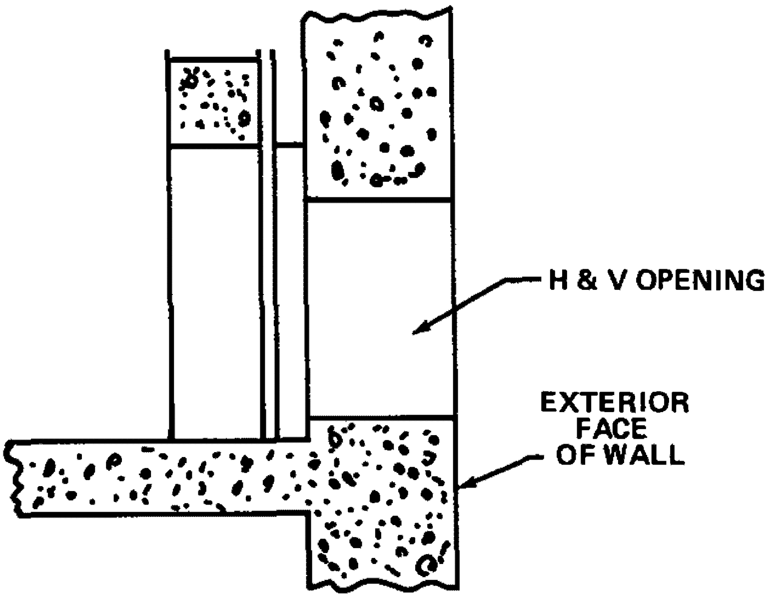
* POTENTIAL MISSILES IN THIS ZONE ARE SHIELDED BY ADJACENT STRUCTURES

PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

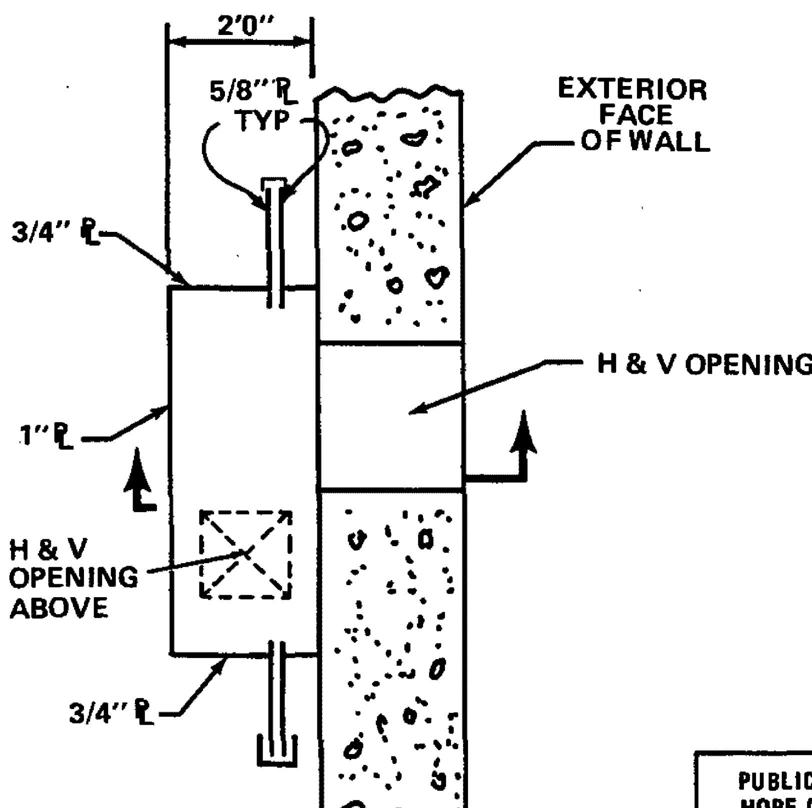
WALL OPENING
MISSILE BARRIER
TYPE – W2

UPDATED FSAR

FIGURE 3.5-21



ELEVATION



PLAN

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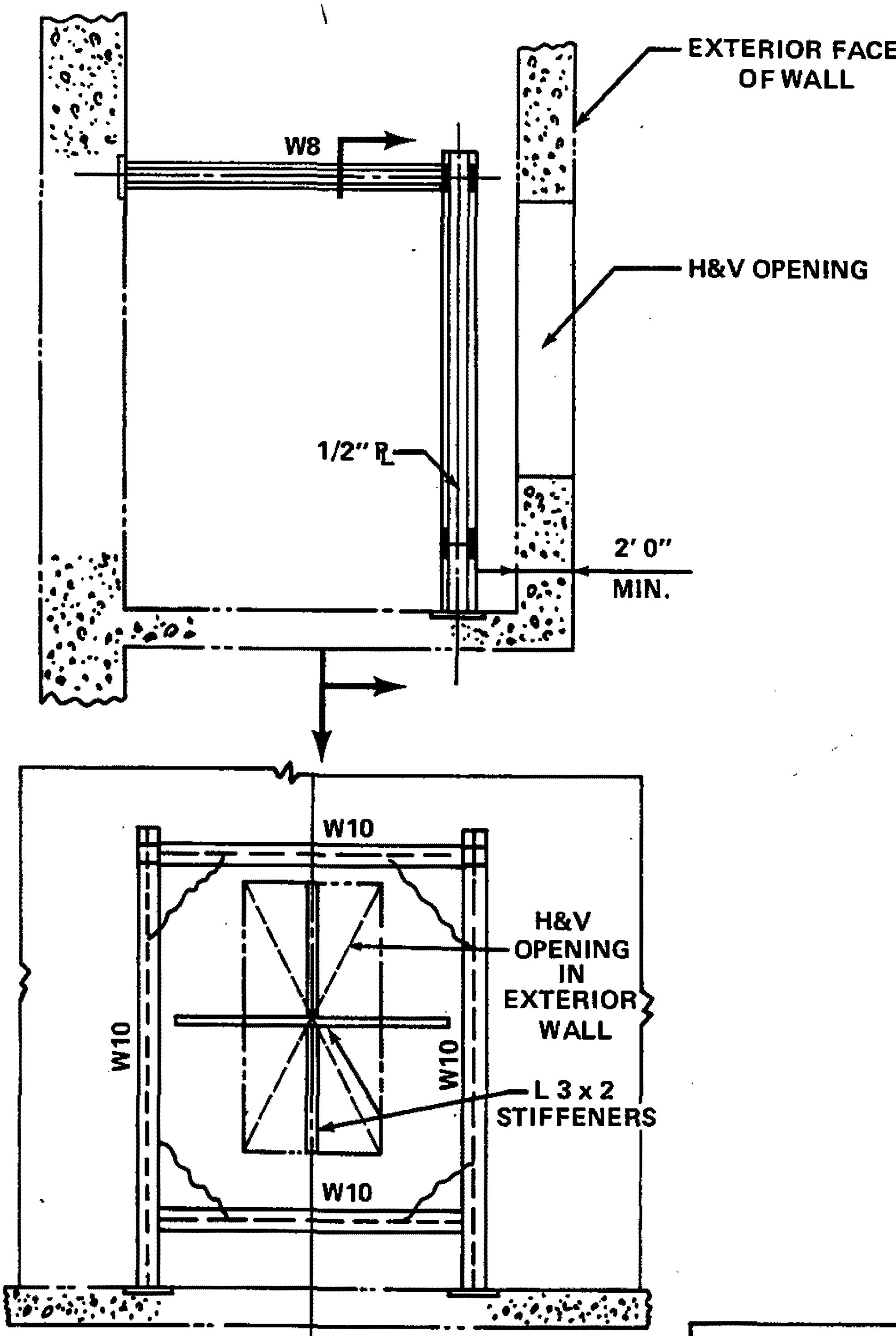
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

WALL OPENING
MISSILE BARRIER
TYPE – W3

UPDATED FSAR

FIGURE 3.5-22



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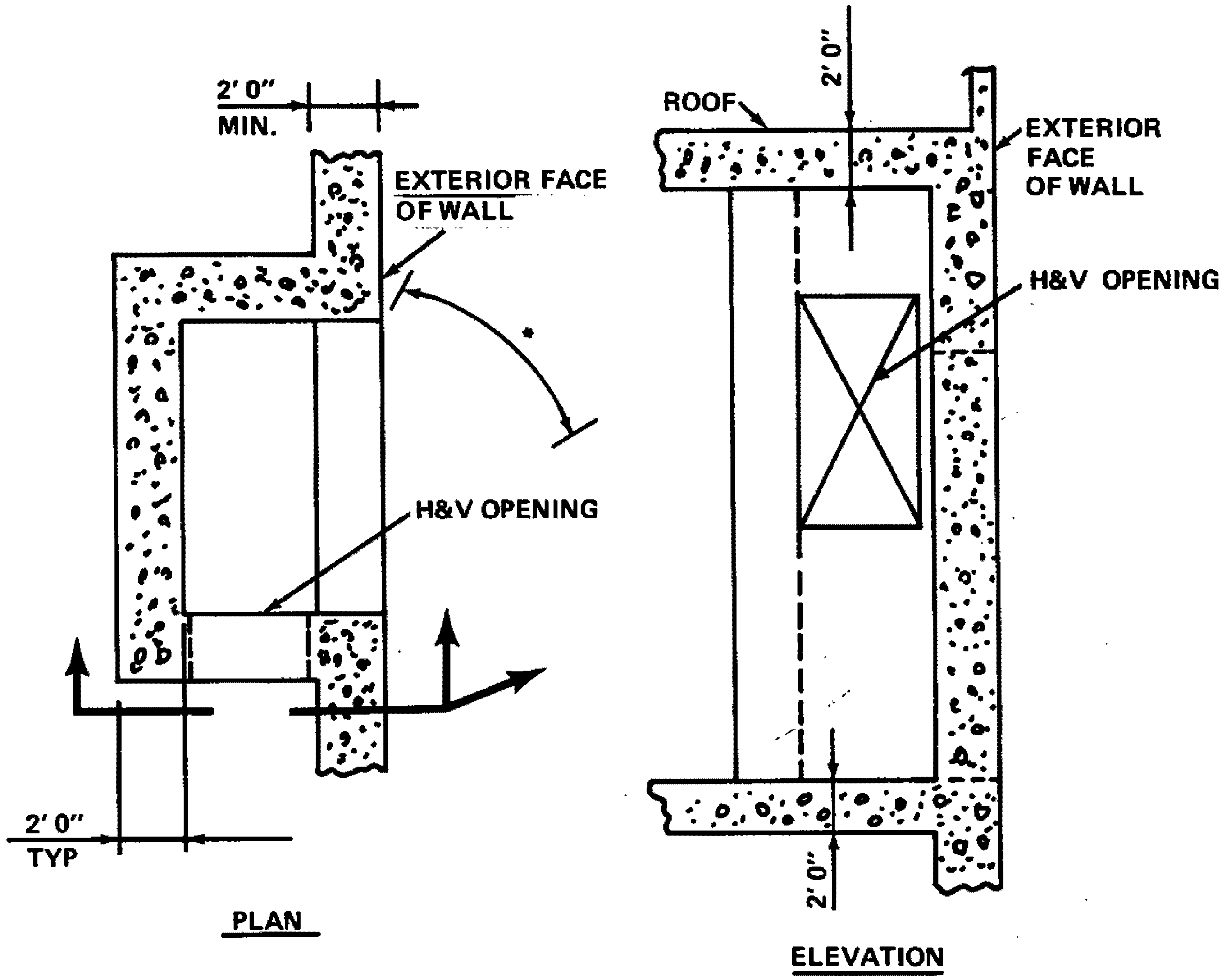
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

WALL OPENING
MISSILE BARRIER
TYPE – W4

UPDATED FSAR

FIGURE 3.5-23



*POTENTIAL MISSILES IN THIS ZONE ARE SHIELDED BY ADJACENT STRUCTURES

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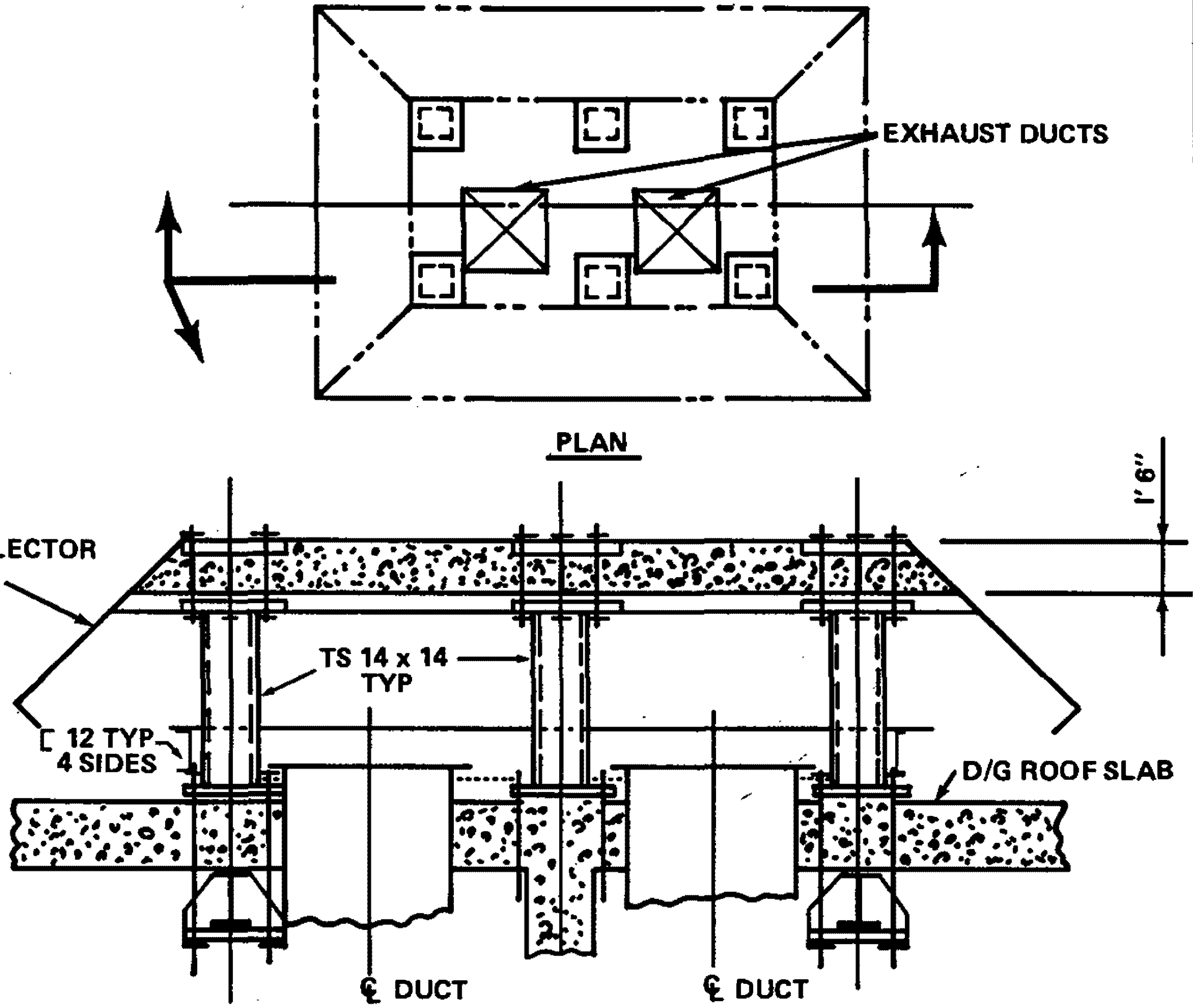
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

WALL OPENING
MISSILE BARRIER
TYPE – W5

UPDATED FSAR

FIGURE 3.5-24



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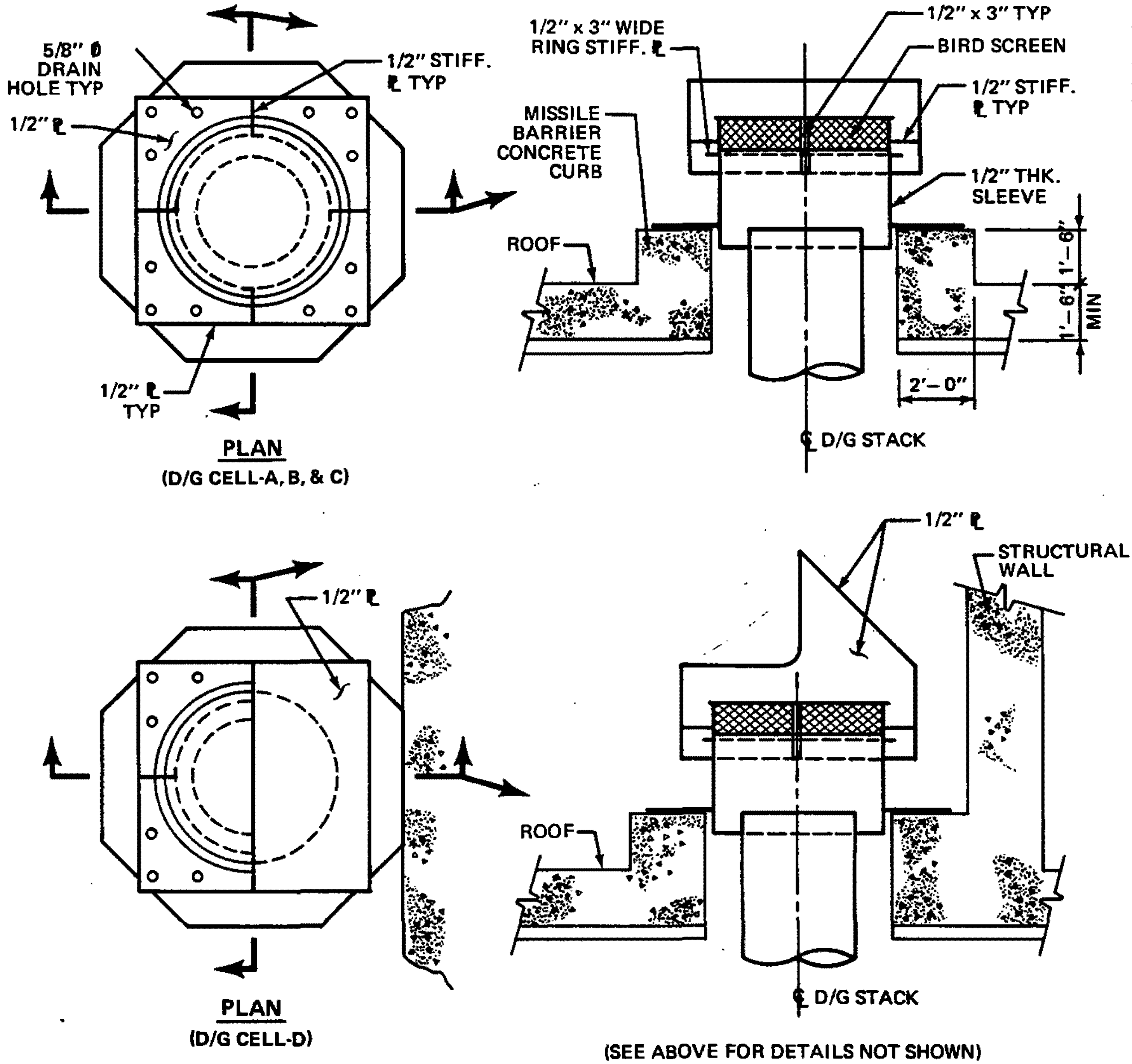
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

ROOF OPENING
MISSILE BARRIER
TYPE – R1

UPDATED FSAR

FIGURE 3.5-25



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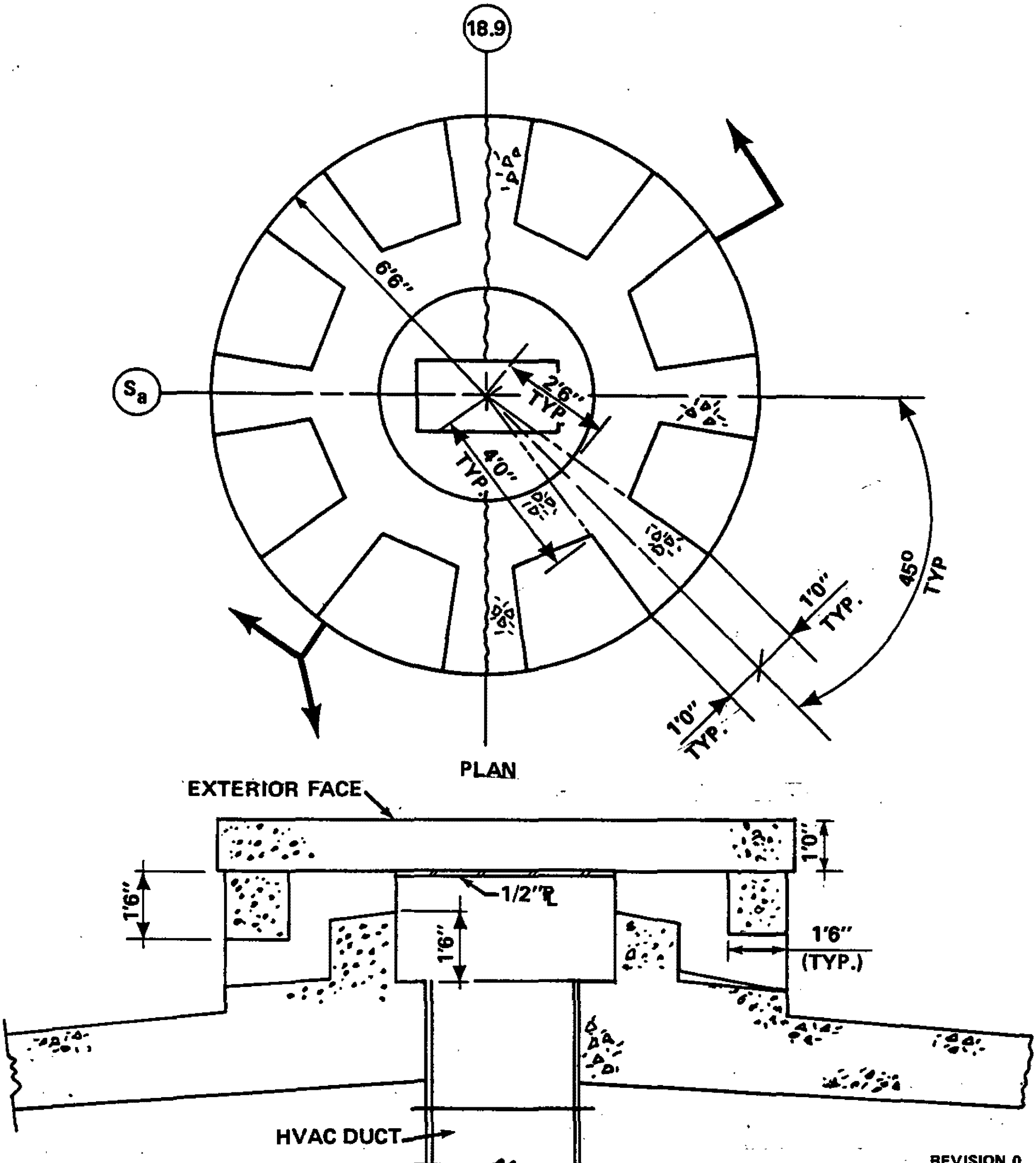
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

ROOF OPENING
MISSILE BARRIER
TYPE – R2

UPDATED FSAR

FIGURE 3.5-26



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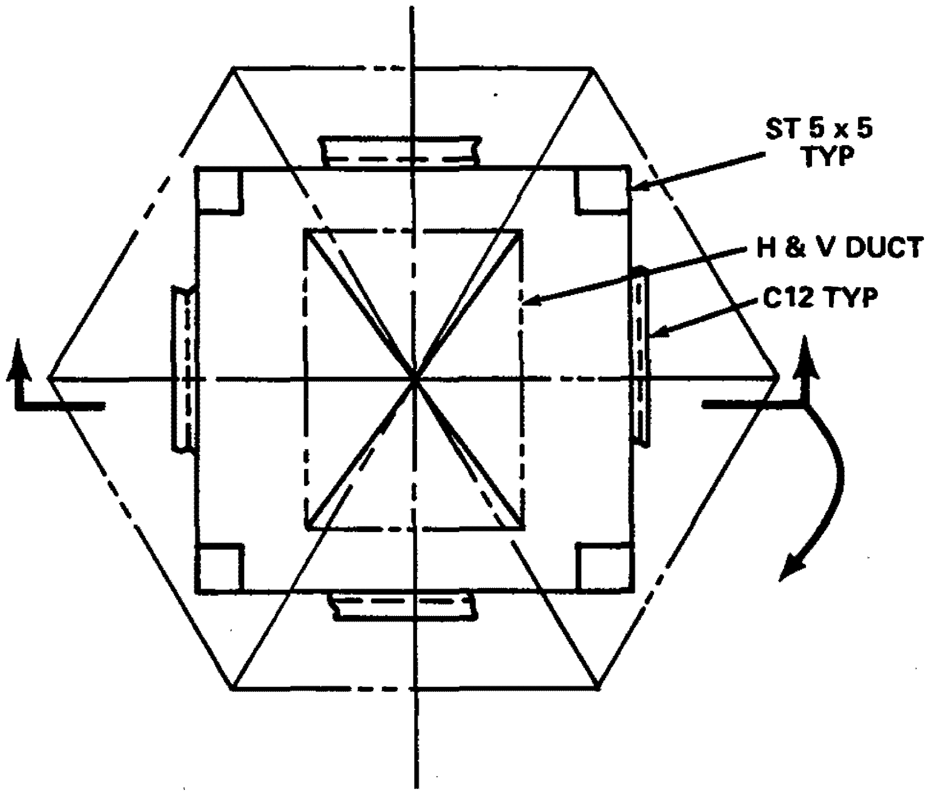
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

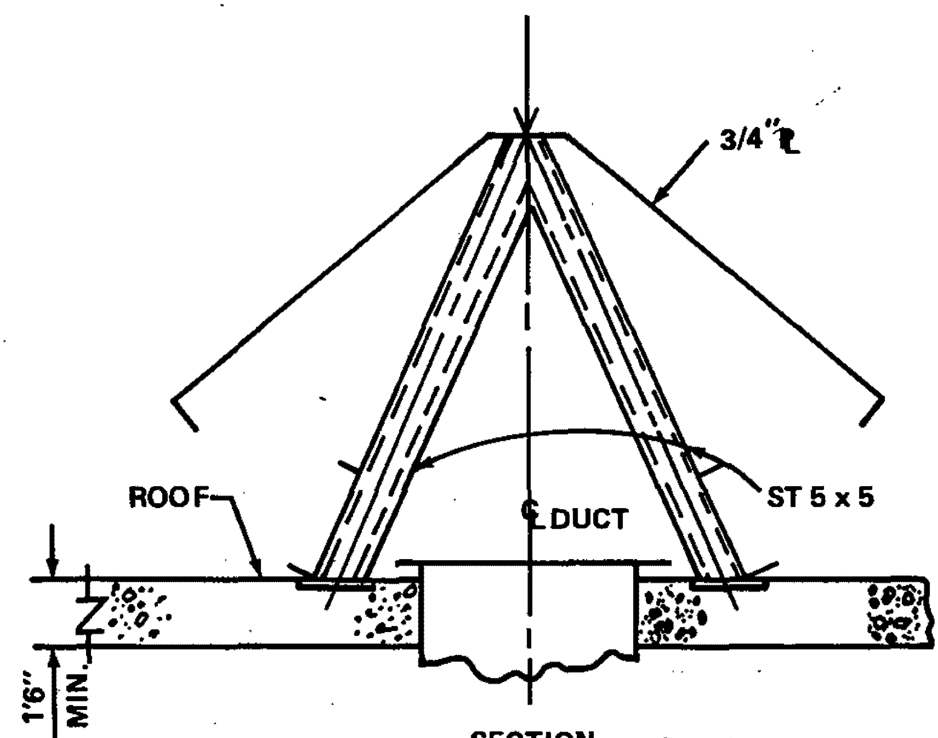
ROOF OPENING
MISSILE BARRIER
TYPE – R3

UPDATED FSAR

FIGURE 3.5-27



PLAN



SECTION

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PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

ROOF OPENING
MISSILE BARRIER
TYPE – R4

UPDATED FSAR

FIGURE 3.5-28

Figure F3.5-29 SH 1-2 intentionally deleted.

Refer to Plant Drawing C-0091-1 for sheet 1 in DCRMS
Refer to Plant Drawing C-0094-0 for sheet 2 in DCRMS

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

This section describes the method of protection against dynamic effects associated with postulated ruptures in high energy and moderate energy piping located both inside and outside of the primary containment, as defined in Section 3.6.3. The methods used to determine pipe rupture locations and to analyze the results of the ruptures - which include jet thrust forces, jet impingement forces, piping dynamic responses, and compartment pressure temperature transients - are also described. A description is also provided for the design measures implemented to ensure that no function necessary to mitigate the consequences of a pipe rupture is lost during any postulated rupture.

The definitions of terms used in this section are provided in Section 3.6.3.

3.6.1 Postulated Piping Failures In Fluid Systems

The failure of high or moderate energy piping could cause damage to surrounding structures, systems, and components. Nuclear safety-related systems are designed to ensure that components required for the safe shutdown and isolation of the reactor do not fail as a result of a failure in a high or moderate energy piping system. Depending on the fluid system involved and the rupture location, postulated piping failures can result in one or more of the following effects: pipe whip, jet impingement, environmental effects, i.e., pressure, temperature, and humidity; water spray; and flooding.

Essential systems and components are protected from the effects listed below, unless it can be demonstrated that their function is not impaired:

1. Pipe whip - Pipe whip is the unrestrained movement of a pipe due to the reaction force imposed on the pipe by

fluid discharging from a rupture. Protection against pipe whip can be provided by interposing structural members between high energy piping and the essential systems and components, by providing pipe whip restraints on the high energy piping, or by locating essential systems and components sufficiently distant from high energy piping. Through wall cracks in moderate energy systems do not cause pipe whip. Examples of typical pipe whip restraints are shown on Figure 3.6-1.

A whipping pipe is assumed to cause functional failure of an impacted pipe of smaller nominal pipe size. The whipping pipe could also lead to the development of a through wall crack in an impacted pipe of the same nominal pipe size with thinner wall thickness. The whipping pipe is assumed to have sufficient energy to cause the failure of impacted electrical cable and instrumentation, unless the equipment is shown to be sufficiently strengthened or protected. High energy piping is located away from the essential safety-related systems wherever practical. Otherwise, piping is provided with pipe whip restraints.

2. Jet impingement - Jet impingement loads, resulting from postulated pipe failures, are considered for equipment and safety-related systems. The blowdown of fluid from the rupture of a high energy pipe can exert forces on nearby equipment sufficient to damage the equipment. Protection against jet impingement can be provided by installing jet impingement barriers to deflect the blowdown jet, or by locating essential systems and components a sufficient distance from high energy piping.

   Jets from postulated pipe breaks in the drywell are analyzed in detail. All equipment located within the potential impingement zone of a jet is identified as a target. A target is not considered impacted when structural members or equipment shield it from the jet

effects. After all the targets are identified, they are tabulated by pipe break location. Each target is then further organized into one of four safety categories:

a. Nonsafety-related equipment - This equipment consists of items that are not required either for safe shutdown or to mitigate the effects of any pipe break in question.

b. Essential safety-related equipment - This equipment consists of items that must remain in operation or be available for operation in order to accomplish safe shutdown or to mitigate the effects of the particular pipe break being examined.

c. Nonessential safety-related equipment - This equipment consists of items that are required for some postulated pipe breaks, but are not required for safe shutdown or to mitigate the effects of the particular pipe break being examined.

d. Redundant, essential safety-related equipment - This equipment consists of items that are designed to mitigate the effects of a postulated pipe break but, due to sufficient system redundancy and/or separation, may not actually be required for safe shutdown of the plant.

Nonsafety-related equipment, and redundant, essential safety-related and nonessential safety-related equipment are reviewed and identified only, whereas essential safety-related equipment is analyzed for functional as well as structural integrity. The jet impingement loads are reduced by accounting for the frictional effects and target shape factors that reduce the total force on the target. Where structural integrity of equipment or the function of essential safety-related equipment is exceeded

by the calculated jet impingement force, protection is provided by spatial separation or by the addition of barriers or enclosures.

Jet impingement loads in the Reactor Building are reviewed along with other pipe break effects on a compartment by compartment basis. Structures designed to enclose and separate high energy piping from essential safety-related equipment are designed to sustain the predicted jet impingement and pipe whip loads. Loss of the impacted safety-related systems occupying the compartment where the postulated pipe break occurs is considered in the evaluation of the plant's ability to shut down, cool down, or isolate.

3. Environmental - Ruptures in high energy piping result in the release of fluid that can increase temperature, pressure, humidity, and radiation levels in the vicinity of the pipe failure and also in remote areas that communicate with the local atmosphere. Essential systems and components may be exposed to abnormal conditions that could degrade the capability of that equipment to perform its function. Safety-related equipment is qualified to meet the above environmental conditions resulting from postulated breaks.

   Piping systems whose failure could generate hazardous environmental conditions are generally located in compartments that are capable of being isolated from required safety-related systems. Isolation of compartments that enclose high energy lines is provided by maintaining normally closed accessways; by sealing penetrations through walls and slabs; and by providing automatic isolation of other communication paths, such as ventilation ductwork, except where the design provides for steam venting through an adjacent compartment. Compartments are designed to withstand the maximum

internal pressurization that can develop as a result of a pipe failure, and are provided with vent capability to the atmosphere or adjacent areas where these effects would not escalate the event. Essential systems and components are either located in areas not affected by pipe ruptures or are qualified for operation under the maximum environmental conditions that they may be subjected to as a result of pipe ruptures.

Pressure rise analysis and verification of structural adequacy of enclosures used to provide protection are discussed in Sections 3.8.2, 3.8.3, and 3.8.4. Transport of a steam environment that could affect the habitability of the main control room is discussed in Section 6.4.2.

Radiation is an additional environmental consequence of some pipe failures. Essential equipment is designed to tolerate integrated exposure resulting from normal plant operations. Essential equipment inside and outside the primary containment is designed for the additional exposure resulting from a design basis accident (DBA). Equipment qualification is discussed in Section 3.11, with other radiological considerations discussed in Section 12.1.

4. Water spray - Water itself is a hazard to certain equipment, particularly electrical equipment. In most cases, spatial separation and intermediate obstructions are adequate to prevent spray from reaching the equipment. Essential equipment, i.e., equipment that is required to operate under and/or mitigate the accident condition and that can potentially be subjected to water spray, is either designed to operate when wetted, or is protected from water sprays where necessary by barriers or equipment enclosure.

5. Flooding - Any significant failure of a steam or fluid system may result in flooding in the vicinity of the rupture and in the compartments into which the released fluid drains. The flooding rate and the total fluid volume released are based on the pipe break configuration, the service of the system, and the time required to isolate the system. The plant drainage system handles minor releases of fluid with no adverse effects on essential systems and components.

   Compartments containing safety-related equipment are designed with features that permit rapid detection and isolation of flooding resulting from major line breaks, except where it can be demonstrated that flooding would not affect the performance of that equipment or its redundant counterparts.

   Because of the high degree of equipment and system separation in the plant, flooding of an Emergency Core Cooling System (ECCS) equipment room is limited to one division of equipment.

3.6.1.1 Design Bases

Pipe breaks are postulated to occur in all high energy fluid system piping, or a portion of the system, in accordance with the criteria in Section 3.6.2. Pipe cracks are postulated to occur in all moderate energy fluid system piping, in accordance with the criteria in Section 3.6.2.1.3.

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:

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

2. Containment integrity

3. That a postulated pipe break on a line that is not part of the reactor coolant pressure boundary (RCPB) will not cause a loss of reactor coolant

4. Resultant radiation exposures are below the guideline values in 10CFR50.67.

In analyzing the effects of postulated pipe ruptures, the following assumptions are made:

1. Each pipe break in high energy fluid system piping or crack in moderate energy fluid system piping is considered separately as a single, postulated, initial event occurring during normal plant conditions.

2. Offsite power is assumed to be unavailable if a trip of the turbine generator or the Reactor Protection System (RPS) is a direct consequence of the postulated piping rupture.

3. A single, active component failure is assumed to occur in systems used to mitigate the consequences of the postulated piping rupture and to shut down the reactor, except as noted in item 4 below.

4. Where the postulated piping rupture is assumed to occur in one of the two redundant trains of either the Residual Heat Removal (RHR) System, Safety Auxiliaries Cooling System (SACS), Station Service Water System (SSWS), Auxiliary Building control Area chilled water system, a single active component failures in the other

train is not assumed, as discussed in NRC Branch Technical Position (BTP) ASB 3-1, Paragraph B.3.b.(3). These dual purpose, moderate energy systems are powered from both onsite and offsite sources and are designed, constructed, and inspected to standards appropriate for nuclear safety-related systems.

5. All available systems, including those actuated by operator actions, may be employed to mitigate the consequences of the postulated piping rupture. In judging the availability of systems, both the postulated piping rupture and its direct consequences, and the assumed single active component failure and its direct consequences, are considered. The feasibility of carrying out operator actions is judged on the availability of ample time and adequate access to equipment for the required actions.

6. An unrestrained whipping pipe is considered capable of causing functional failure in impacted piping of smaller nominal pipe size. It could also lead to the development of through wall leakage cracks in impacted piping of equal or larger nominal pipe size with thinner wall thickness.

A postulated pipe break inside the primary containment, up to and including a rupture of the recirculation piping, in conjunction with a safe shutdown earthquake (SSE) and a single active component failure, will not prevent the plant from achieving and maintaining reactor shutdown, maintaining containment integrity, and maintaining dose levels within 10CFR50.67 guidelines. Outside the primary containment, the single failure is qualified to BTP ASB 3-1, Paragraph B.3.b.

3.6.1.2 Description

A listing of high energy fluid system piping is provided in Table 3.6-1. All other piping in the plant that is pressurized above atmospheric pressure is considered to be moderate energy piping. The routing of high energy fluid system piping within the Reactor Building and the primary containment is shown on the isometric drawings referenced in Section 3.6.1.2.1.

For each pipe rupture location determined in accordance with the criteria of Section 3.6.2.1, an analysis is performed using the assumptions of Section 3.6.1.1 to verify that the consequences of the pipe rupture are acceptable. These analyses are summarized below for high energy and moderate energy fluid systems.

Proximity of the essential systems and components to the high and moderate energy fluid system piping is reviewed and the essential systems and components are located with acceptable separation, unless the effects of pipe failure can be withstood.

3.6.1.2.1 High Energy Fluid Systems

All high energy fluid system piping is described in the following paragraphs. The discussion of each high energy fluid system includes a general system description and discussion of pipe break locations, the compartment pressure temperature transient, and a verification of the reactor shutdown capability.

3.6.1.2.1.1 Main Steam System

The four 26-inch main steam lines are routed as shown on Figure 3.6-2 for the portion inside the primary containment, and on Figure 3.6-3 for the portion outside the primary containment. The A and B steam lines are connected to the south side of the reactor vessel and the C and D steam lines are connected to the north side of the vessel. All four steam lines penetrate the east side of the

primary containment. The portion of the Reactor Building through which the main steam lines are routed (between the primary containment and the Turbine Building) is referred to as the main steam tunnel and is separated from other areas of the reactor building by concrete walls and slabs. Only piping, HVAC, valves, and associated instrumentation are located in the main steam tunnel.

Figure 3.6-4 shows an elevation view of the main steam tunnel penetration chamber.

The following features are incorporated into the design of the main steam line and nearby structures to mitigate the consequences of a main steam line break or to minimize the probability of its occurrence:

1. A venturi type flow restrictor is located in each main steam line inside the primary containment. The flow restrictor reduces the rate of loss of reactor coolant from a main steam line break downstream of the restrictor. The flow restrictors are described in Section 5.4.4.

2. Each main steam line is provided with three or four main steam safety/relief valves (SRVs) that reduce the probability of breaks by protecting the steam line against overpressurization. The SRVs are described in Section 5.2.2.

3. Each main steam line is provided with two fast acting main steam isolation valves (MSIVs), one upstream and one downstream of the primary containment penetration. These valves close automatically upon receipt of signals indicating high steam flow or high temperature in the vicinity of the main steam piping outside the primary containment, as well as upon receipt of other initiating signals discussed in Section 7.3. This is done to

terminate blowdown through breaks outside the primary containment. The MSIVs are described in Section 5.4.5.

4. Moment limiting pipe restraints are located upstream of the inboard MSIVs and downstream of the outboard MSIVs in order to ensure the operability of these valves in the event of a main steam line break in the general vicinity of the valves. The piping between the containment inboard MSIV and the outboard MSIV is designed to the stress limit criteria of Section 3.6.2.1.1.1 so that no break is postulated in this region.

The main steam lines are provided with pipe whip restraints inside the primary containment and in the main steam tunnel. Typical restraints inside the primary containment are shown on Figure 3.6-1. Figure 3.6-4 shows the locations of the restraints in the main steam tunnel. As shown on Figures 3.6-5 and 3.6-6, the anchor and restraint upstream and downstream of the outboard MSIVs span between the north and south walls of the tunnel and restrain all four steam lines. A built-up member, shown on Figures 3.6-4 and 3.6-7, extending out from the north wall to the south wall of the main steam tunnel limits the possible upward movement of the upper elbow of each steam line in the tunnel. Additionally, the vertical portion of the steam line run in the tunnel is restrained against the east wall of the tunnel at two separate locations, as shown on Figure 3.6-7.

After entering the Turbine Building from the main steam tunnel, the main steam lines are routed along the west side of the turbine building before turning eastward and running to the turbines. This arrangement is shown on Figure 3.6-3.

In reviewing the potential consequences of jet impingement resulting from main steam line breaks, it was determined that some breaks inside the primary containment could result in impingement on the

control rod drive (CRD) withdrawal piping. Analysis shows no significant increase in scram times resulting from withdrawal line crimping under jet impingement loads, as discussed in Reference 3.6-3. Electrical cabling associated with essential systems and components is either routed to avoid jet impingement, shown to be redundant, or has barriers, as necessary, to provide protection from postulated breaks of main steam piping.

1. Pipe break locations - The postulated pipe break locations and the pipe whip restraint locations for the main steam piping are shown on Figures 3.6-2 and 3.6-3. The calculated stress levels, usage factors, and postulated break types are listed in Tables 3.6-2 and 3.6-3.

2. Compartment pressure temperature transients - the pressure temperature transient in the primary containment resulting from a complete circumferential break of one main steam line is discussed in Section 6.2.1.

   Protection against overpressurization of the main steam tunnel in the event of a main steam line break in the tunnel is provided by two sets of blowout panels. One set of blowout panels is located in the east wall of the penetration chamber of the main steam tunnel and vents to the auxiliary building main steam tunnel. The second set of blowout panels is located in the steam vent that leads upward from the tunnel and discharges to the atmosphere above the top of the Auxiliary Building. This steam vent is shown on Figures 3.6-8 and 3.6-39.

   A pressure temperature transient analysis for a main steam line break in the tunnel was performed using the analytical technique described in Reference 3.6-1 and the blowdown data provided in Table 3.6-4. The flow schematic diagram used is shown on Figures 3.6-40 and 3.6-9, and the results of the analysis are listed in Table 3.6-5 and

shown graphically on Figures 3.6-41 through 3.6-44. The main steam tunnel is designed to withstand the maximum pressure developed, and the MSIVs are qualified to operate under environmental conditions more severe than those calculated to occur.

Analysis shows that gross structural failure of the Turbine Building floors, steel framework and operating deck as a result of a main steam line break within the Turbine Building will not occur. In this analysis, credit is taken for structural failure (blowout) of the insulated metal siding on the south and east exterior walls of the Turbine Building, above Elevation 125 feet 6 inches.

3. Verification of reactor shutdown capability - Breakage of a main steam line inside the primary containment would result in a nonisolable blowdown of the reactor vessel. The sequence of events that would occur automatically to shut down the reactor and cool the core is discussed in Section 6.3.2.

   For a main steam line break outside the primary containment, the MSIVs will be closed automatically because of high steam flow, low reactor water level, or high temperature in the vicinity of the main steam lines. Closure of the MSIVs will cut off steam flow to the feedwater pump turbines, causing the pumps to coast down and stop. The reactor will be tripped by low reactor water level or closure of the MSIVs. After isolation of the reactor vessel, the Reactor Coolant System pressure will increase until the setpoint of the SRVs is reached. Steam will then be automatically discharged to the suppression chamber to limit the pressure rise.

   Low reactor water level will initiate operation of the High Pressure Coolant injection (HPCI) and Reactor Core

Isolation Cooling (RCIC) Systems to maintain reactor water level. If both the HPCI and RCIC systems are unavailable, the Automatic Depressurization System (ADS) will be automatically initiated to depressurize the reactor vessel so that the Low Pressure Coolant Injection (LPCI) and Core Spray Systems can inject water into the vessel. These latter two systems are initiated automatically by low reactor water level and will provide sufficient flow to restore reactor water level and cool the core.

After reactor water level has been restored, the operator will manually initiate a reactor shutdown. When the Reactor Coolant System pressure and temperature have decreased sufficiently, the shutdown cooling mode of the RHR system can be manually initiated to bring the reactor to cold shutdown.

A combination of pipe whip restraints, jet impingement barriers, and separation by distance or intervening structures is used to ensure the availability of essential systems and components in the event of a main steam line break in either the primary containment or the main steam tunnel. Essential systems and components located in these areas are qualified to operate under the environmental conditions resulting from such a break. Since no essential systems and components are located in the turbine building, no special provisions are necessary to provide protection for equipment in this area from the effects of pipe breaks. Where turbine building interactions may have indirect impact on essential systems, such as TACS impacts that could affect SACS, special separation, pipe whip pathways, concrete barriers and stress evaluations of pipe break locations can be considered as mitigating or protection factors.

3.6.1.2.1.2 Reactor Recirculation System

The two reactor recirculation loops are located entirely within the primary containment and are arranged on opposite sides of the reactor pedestal and biological shield. Pipe whip restraints anchored in the reactor pedestal and biological shield are

provided for the recirculation loops and are arranged as shown on Figure 3.6-10. This system prevents unrestrained pipe whip resulting from a postulated rupture at any of the identified break locations. The restraints are of two different designs: a U-strap design, and a frame type design. The U-strap restraints consist of two basic components: the frame attached to a support member, and two straps attached to each frame. Stainless steel bars or wire ropes are used as straps. A schematic detail of a U-strap restraint is shown on Figure 3.6-11. The frame type restraints are located at azimuths 90° and 270° at an elevation of approximately 98 feet and are fabricated of steel plate.

The postulated break locations on the reactor recirculation system lead to significant jet impingement loads on the CRD withdrawal lines. However, analysis shows no significant increase in scram times resulting from CRD withdrawal line crimping under jet impingement loads, as discussed in Reference 3.6-3. Complete severance of CRD withdrawal piping does not prevent the associated control rods from being inserted into the reactor core. In addition, electrical cabling associated with essential systems and components is protected by routing to avoid jet impingement or is protected by adding jet impingement barriers.

1. Pipe break locations - The postulated pipe break locations for the recirculation loop piping are shown on Figure 3.6-12. The calculated stress levels, usage factors, and postulated break types are listed in Table 3.6-6. The blowdown time history for a recirculation system pipe break is provided in Table 3.6-7.

2. Compartment pressure temperature transient - The pressure temperature transient in the primary containment resulting from a complete circumferential rupture of one recirculation loop is discussed in Section 6.2.1.

3. Verification of reactor shutdown capability - The automatic sequence of events that shuts down the reactor and cools the core in the event of a recirculation loop rupture is discussed in Section 6.3.2. A combination of pipe whip restraints, jet impingement barriers, and separation by distance is used to ensure the availability of sufficient equipment to accomplish these functions.

3.6.1.2.1.3 Feedwater System

The discharge lines from the three reactor feedwater pumps are routed into a common mixing header in the Turbine Building. From this header, two parallel 24-inch feedwater lines enter the main steam tunnel and then penetrate the primary containment. Inside the primary containment, the two lines diverge to form symmetrical headers on opposite sides of the reactor vessel. Each header splits into three 12-inch risers that attach to the reactor vessel nozzles. The routing of the feedwater lines in the main steam tunnel and the primary containment is shown on Figures 3.6-13 and 3.6-14.

Each feedwater containment penetration is provided with three check valves as containment isolation valves: one inside primary containment, and two in the main steam tunnel. In the event of a feedwater line break outside the primary containment, these check valves close to prevent backflow from the reactor vessel. Thus, flow from the break comes from the feedwater pump side only.

Moment limiting pipe restraints ensure the operability of the outboard containment isolation valve closest to the primary containment in the event of a feedwater line break. The inboard valve and the second outboard valve need not be protected from overstress due to pipe whip, since only one of these valves may be damaged by a single pipe break and the two remaining isolation valves provide adequate redundancy to ensure the containment isolation function. See Reference 3.6-11 for the results of the analyses of the feedwater check valves.

The feedwater lines are provided with pipe whip restraints inside the primary containment, and in the main steam tunnel. Typical restraints inside the primary containment are shown on Figure 3.6-1. The restraint in the tunnel, RS-1, spans between the north and south walls of the tunnel and limits the motion of both feedwater lines on pipe break, as shown on Figure 3.6-6. A built-up member from the north wall to the south wall of the steam tunnel limits the possible upward movement of the upper elbows of the feedwater lines, as shown on Figure 3.6-4.

In reviewing the potential consequences of jet impingement resulting from feedwater line breaks, it was determined that breaks near the first elbow inside the primary containment would result in impingement on the MSIV operators. Steel plate barriers have been provided to protect the operability of the MSIVs from this source of impingement. Electrical cabling associated with essential systems and components is routed to avoid jet impingement, or jet impingement barriers have been provided for protection, from postulated breaks of feedwater piping.

1. Pipe break locations - The postulated pipe break locations and the pipe whip restraint locations for the feedwater piping are shown on Figures 3.6-13 and 3.6-14. The calculated stress levels, usage factors, and postulated break types are listed in Tables 3.6-8 and 3.6-9.

2. Compartment pressure temperature transients - The pressure temperature transient in the primary containment resulting from a break of any of the feedwater lines in the drywell is exceeded in severity by the transients resulting from recirculation loop and main steam line breaks, which are discussed in Section 6.2.1. The pressure temperature transient resulting from a feedwater line break in the main steam tunnel or the Turbine Building is exceeded in severity by the transient

resulting from a main steam line break, which is discussed in Section 3.6.1.2.1.1.

3. Verification of reactor shutdown capability - A feedwater line break inside the primary containment would result in a nonisolable blowdown of the reactor vessel. The automatic sequence of events that shuts down the reactor and cools the core is discussed in Section 6.3.2.

An analysis to determine the feedwater check valve dynamics and stresses following a double ended break of the feedwater line outside containment has been performed. The RELAP 5/ MOD 1 computer code was used to predict the maximum valve disc angular velocity and the peak pressures upstream and downstream of the valve disc following closure. In addition, a sensitivity analysis was performed to select the feedwater check valve that yields the most conservative stress results.

An inelastic stress analysis was performed on the valve body, disc, hinge arm, and valve seat with the calculated stresses determined to be less than stress allowables. In addition, the maximum displacement of the hinge arm before and after valve disc closure was determined and deemed acceptable. Stresses were evaluated for the faulted condition based on the methods for analysis and design limits contained in Appendix F of the ASME B&PV Code, Section III.

For a feedwater line break outside the primary containment, differential pressure across the containment isolation check valves in the reverse direction causes these valves to close rapidly, isolating the reactor vessel from the break. The loss of feedwater flow causes the reactor water level to drop, initiating a reactor

scram when the reactor water level 3 trip point is reached. Water level continues to drop because of steam generation from decay heat, causing closure of the MSIVs when the level 1 trip point is reached, as well as initiation of the RCIC and HPCI systems when the reactor water level 2 trip point is reached. Once the reactor has been scrammed and the Reactor Coolant System isolated, the sequence of events is similar to that for a main steam line break outside the primary containment.

A combination of pipe whip restraints, jet impingement barriers, and separation by distance or intervening structures is used to ensure the availability of essential systems and components in the event of a feedwater line break in either the drywell or the main steam tunnel. Since no essential systems and components are located in the Turbine Building, no special provisions are necessary to provide protection for equipment in this area from the effects of pipe breaks.

3.6.1.2.1.4 Condensate System

The Condensate System is located entirely within the Turbine Building. No pipe whip restraints are provided for the condensate piping.

1. Pipe break locations - Since the condensate system consists of non-nuclear class piping, breaks are postulated to occur at each location of potential high stress, such as pipe fittings, valves, and welded attachments. Breaks need not be postulated at locations that are subject to low stresses, consistent with Section 3.6.2.1.1.

2. Compartment pressure temperature transients - Since the normal fluid temperature in the Condensate System is less than 135°F, no significant pressure temperature transient results from a condensate line break.

3. Verification of reactor shutdown capability - In the event of a condensate line break, low suction pressure causes the feedwater pumps to trip. A subsequent loss of feedwater flow results in closure of the containment isolation check valves, thus preventing reactor blowdown through the break. The sequence of events that occur from this point on is the same as for breakage of a feedwater line outside the primary containment.

   Because no essential systems and components are located in the Turbine Building, special provisions are not necessary to protect equipment in this area from the effects of pipe breaks. The flooding effects of a condensate line break are exceeded by the effects of a circulating water line expansion joint rupture in the Turbine Building, which is discussed in Sections 10.4.1 and 10.4.5.

3.6.1.2.1.5 Reactor Water Cleanup System

The Reactor Water Cleanup (RWCU) System takes suction from each reactor recirculation loop within the primary containment. Two 4-inch RWCU suction lines converge into one 6-inch RWCU suction line that is routed up to Elevation 150 feet 6 inches, at which point it penetrates the primary containment. The RWCU piping is then routed through the various RWCU equipment compartments at Elevations 132 feet, 145 feet, and 162 feet in the Reactor Building, including the pipe chase compartment, the regenerative and nonregenerative heat exchanger compartment, two filter/demineralizer compartments, and two RWCU recirculation pump compartments. The RWCU return piping is routed from the pipe chase compartment directly into the main steam tunnel, where the piping branches into two 4-inch lines. One line connects to the A feedwater line, and the other connects to the B feedwater line. Both of these connections to the feedwater lines are located between the two outboard containment isolation valves in the feedwater lines.

The RWCU suction line is provided with two fast acting isolation valves, one upstream and one downstream of the primary containment penetration. These valves close automatically upon receipt of signals indicating high differential flow (ratio of RWCU suction to return) or high temperature in the vicinity of the RWCU piping outside the drywell, as well as upon receipt of other initiating signals discussed in Section 7.3. This is done in order to terminate blowdown through breaks outside the drywell. To ensure the operability of these valves in the event of a RWCU suction line break, moment limiting pipe restraints are located upstream of the inboard containment isolation valve and downstream of the outboard containment isolation valve. Whip restraints are also located on the portion of the RWCU suction line within the drywell.

In reviewing the potential consequences of jet impingement resulting from RWCU line breaks, it was determined that breaks in the RWCU return lines in the main steam tunnel do not impinge on any essential equipment. It was also determined that RWCU line breaks in the pipe chase compartment do not result in unacceptable impingement on the RWCU outboard containment isolation valve, since the containment isolation valve is far from the closest break location. However, a RWCU line break in this compartment will result in impingement on the containment spray system containment isolation valves. These isolation valves are motor operated and "fail as is." Also, this system is not needed to mitigate the effects of the RWCU line break. Therefore, protective barriers are not needed.

1. Pipe break locations - The postulated pipe break locations and the pipe whip restraint locations for the RWCU piping inside and outside the primary containment are shown on Figures 3.6-15 and 3.6-16, respectively. The calculated stress levels, usage factors, and postulated break types for these portions of the RWCU piping are listed in Tables 3.6-10 and 3.6-11.

2. Compartment pressure temperature transients - The pressure temperature transient in the primary containment resulting from a break in the drywell portion of the RWCU suction line is exceeded in severity by the transients resulting from recirculation loop, main steam line, and intermediate size breaks, which are discussed in Section 6.2.1.

   Protection against overpressurization of the RWCU equipment compartments in the reactor building as a result of RWCU line breaks in these areas is provided by interconnecting steam venting paths between the various compartments and by blowout panels leading to the outside atmosphere.

   The RWCU heat exchanger compartment vents directly into the pipe chase compartment. Two of the four pump rooms vent directly to the pipe chase compartment. These two pump rooms serve as part of the vent path for the remaining pump rooms. The pipe chase compartment is vented to the suppression chamber compartment, and then to steam vent located on the west side of the reactor building. The steam vent is open to the atmosphere at its upper end via a set of blowout panels. This venting pathway is shown on Figure 3.6-45.

   Pressure temperature transient analyses for cases involving RWCU line breaks in the RWCU equipment compartments are performed using the analytical technique described in Reference 3.6-16 and with the blowdown data provided in Table 3.6-4. These blowdown data are developed using Reference 3.6-2. The flow schematic diagram used for breaks in the RWCU equipment compartments is shown on Figure 3.6-17, and the results of the analyses are listed in Table 3.6-5. Pressure and temperature

transient curves for a RWCU filter demineralizer line break in the filter demineralizer room are shown on Figures 3.6-46 and 3.6-47, respectively.

The RWCU equipment compartments are designed to withstand the maximum pressure due to a pipe break. The outboard containment isolation valves for the RWCU system and other systems located in the pipe chase compartment are designed to operate under environmental conditions more severe than those calculated to occur due to a pipe break.

3. Verification of reactor shutdown capability - A RWCU suction line break inside the drywell results in a nonisolable blowdown of the reactor vessel. The automatic sequence of events that shuts down the reactor and cools the core is discussed in Section 6.3.2.

   For a RWCU line break outside the drywell, the RWCU containment isolation valves are closed automatically due to high differential flow in the RWCU system or high temperature in the RWCU equipment compartments. Backflow from the feedwater lines into the RWCU return line is prevented by closure of the two check valves in the return line. If the break occurs in the RWCU piping within the main steam tunnel, the MSIVs will close automatically due to high temperature in the tunnel. Once MSIV isolation has occurred, the sequence of events is similar to that for a main steam line break outside the primary containment. If the break occurs in the RWCU equipment compartments, however, no reactor scram or MSIV closure results, due to the rapid termination of blowdown. After investigation of the cause of the RWCU system isolation, the operator initiates a normal shutdown of the reactor.

A combination of pipe whip restraints, jet impingement barriers, and separation by distance or intervening structures ensures the availability of essential systems and components in the event of an RWCU line break occurring in the drywell, the main steam tunnel, torus compartment, or the RWCU equipment compartments. Essential systems and components located in these areas and the suppression chamber compartment are designed to operate under the environmental conditions resulting from the break. Among the RWCU equipment compartments, only the pipe chase compartment contains safety-related equipment: the primary containment purge line, the containment spray lines, HVAC isolation instrumentation, and the RWCU outboard containment isolation valve. The containment isolation valves in the purge containment spray lines are normally closed during reactor operation and are not required to operate after a RWCU line break outside primary containment. Therefore, they require no protection. RWCU pipe breaks in the torus compartment were reviewed to verify that no essential component would be adversely affected.

3.6.1.2.1.6 HPCI Steam Supply Line

The HPCI steam supply piping has a nominal diameter of 10 inches for the portion inside the drywell and 12 inches for most of the portion outside the drywell. The supply line connects to main steam line C inside the drywell. The line penetrates the drywell at Elevation 106 feet 1-1/4 inches, entering the HPCI pipe chase compartment located at Elevation 99 feet 9 inches in the Reactor Building. It then penetrates the floor of the HPCI pipe chase compartment and runs toward the HPCI pump compartment located at Elevation 54 feet. The routing of this line is shown on Figures 3.6-18 and 3.6-19. During normal reactor operation, the supply line is pressurized from main steam line C up to HPCI turbine steam supply valve HV-F001.

The HPCI steam supply line is provided with two fast acting isolation valves, one upstream and one downstream of the primary containment penetration. These valves close automatically upon receipt of signals indicating high steam flow or high temperature in the vicinity of the HPCI piping outside the drywell, as well as upon receipt of other initiating signals discussed in Section 7.3. This is done in order to terminate blowdown through breaks outside the drywell.

Moment limiting pipe restraints are located upstream of the inboard containment isolation valve and downstream of the outboard containment isolation valve in order to ensure the operability of these valves in the event of a break in the HPCI steam supply line near the valves. Pipe whip restraints are also located on the HPCI steam supply line in the drywell, in the HPCI pipe chase compartment at Elevation 99 feet 9 inches, in the HPCI pump compartment and the torus compartment. Typical restraints inside the drywell are shown on Figure 3.6-1.

1. Pipe break locations - The postulated pipe break locations and the pipe whip restraint locations for the HPCI steam supply line are shown on Figure 3.6-18 for the portion of the line inside the drywell and on Figure 3.6-19 for the portion of the line outside the drywell. The calculated stress levels, usage factors, and the postulated break types are listed in Tables 3.6-12 and 3.6-13.

2. Compartment pressure temperature transients - The pressure temperature transient in the primary containment resulting from a break in the portion of the HPCI steam supply line in the drywell is exceeded in severity by the transients resulting from recirculation loop breaks and main steam line breaks, which are discussed in Section 6.2.1.

Protection against overpressurization of the HPCI pump compartment and of the HPCI pipe chase compartment at Elevation 99 feet 9 inches in the Reactor Building, as a result of HPCI steam supply line breaks in these areas, is provided by steam venting paths and blowout panels leading to the outside atmosphere. The HPCI pipe chase compartment is vented to the atmosphere via blowout panels located on the west side of the Reactor Building, as shown on Plant Drawings P-0044-1 and P-0045-1. The HPCI pump compartment is vented to the suppression chamber compartment via hinged, metal plate blowout panels located in the HPCI pump compartment wall at Elevation 71 feet. These hinged panels open to relieve pressure in the HPCI pump compartment, but do not allow pressurization of the suppression chamber compartment to result in steam flow back into the HPCI pump compartment.

Pressure transient analyses for cases involving HPCI steam supply line breaks in the HPCI pump compartment and in the HPCI pipe chase compartment area are performed using the analytical technique described in Reference 3.6-1 and with the blowdown data provided in Table 3.6-4. The flow schematic diagram used for breaks in these two compartments is shown on Figure 3.6-17, and the results of the analyses are listed in Table 3.6-5. Pressure and temperature transient curves for a HPCI steam supply line break in the HPCI pump compartment are shown on Figures 3.6-20 and 3.6-21, respectively.

The HPCI pump compartment, the torus compartment, and the pipe chase compartment are designed to withstand the maximum pressures due to a pipe break. No equipment located in the HPCI pump compartment is required to operate following a break of the HPCI steam supply line, except the sensing lines leading to PDSH 9434-1 and

9434-2. These sensing lines are not impacted by the jets caused by postulated breaks in the HPCI pump compartment. All equipment located in the HPCI pipe chase compartment and suppression chamber compartment that is required to operate following a break of the HPCI steam supply line is qualified to operate under environmental conditions more severe than those calculated to occur due to pipe break.

3. Verification of reactor shutdown capability - A HPCI steam supply line break inside the drywell results in a nonisolable blowdown of the reactor vessel. The automatic sequence of events that shuts down the reactor and cools the core is discussed in Section 6.3.2.

   For a HPCI steam supply line break outside the drywell, the steam supply line containment isolation valves are closed automatically, terminating reactor vessel blowdown. No reactor scram occurs, due to the rapid termination of blowdown. After investigating the cause of the HPCI steam supply line isolation, the operator initiates a normal shutdown of the reactor.

   A combination of pipe whip restraints and separation by distance or intervening structures ensures the availability of essential systems and components in the event of a HPCI steam supply line break occurring in the drywell, the HPCI pipe chase compartment, or the HPCI pump compartment. Essential systems and components located in these areas are designed to operate under the environmental conditions resulting from the break. Electrical cabling associated with essential systems and components is either routed to avoid jet impingement, or jet impingement barriers are provided.

One pipe whip restraint is provided for the portion of the HPCI steam supply line located within the HPCI pump compartment. This whip restraint is provided for the protection of penetrations and SACS cooling water lines from the consequences of pipe whip due to a HPCI line break adjacent to the turbine.

3.6.1.2.1.7 RCIC Steam Supply Line

The RCIC steam supply line has a nominal diameter of 4 inches for the portion inside the drywell and 6 inches for the portion outside the drywell. The supply line connects to main steam line A inside the drywell. The line penetrates the drywell and enters the RCIC pipe chase compartment at Elevation 106 feet. At Elevation 99 feet 9 inches in the Reactor Building, the line enters the suppression chamber compartment. It then enters the RCIC pump compartment located at elevation 54 feet. The routing of this line is shown on Figures 3.6-22 and 3.6-23. During normal reactor operation, the line is pressurized from the main steam line up to RCIC turbine steam supply valve HV-F045.

The RCIC steam supply line is provided with two fast acting isolation valves, one upstream and one downstream of the primary containment penetration. These valves close automatically upon receipt of signals indicating high steam flow or high temperature in the vicinity of the RCIC piping outside the drywell, as well as upon receipt of other initiating signals discussed in Section 7.3. This is done in order to terminate blowdown through breaks outside the drywell.

Moment limiting pipe restraints are located upstream of the inboard containment isolation valve and downstream of the outboard containment isolation valve in order to ensure the operability of these valves in the event of a break in the RCIC steam supply line near the valves. Pipe whip restraints are also located on the RCIC

steam supply line inside the drywell, in the RCIC pipe chase compartment, and in the suppression chamber compartment in the Reactor Building. Typical restraints inside the drywell are shown on Figure 3.6-1.

1. Pipe break locations - The postulated pipe break locations and the pipe whip restraint locations for the RCIC steam supply line are shown on Figure 3.6-22 for the portion of the line inside the drywell and on Figure 3.6-23 for the portion of the line outside the drywell. The calculated stress levels, usage factors, and postulated break types are listed in Tables 3.6-14 and 3.6-15.

2. Compartment pressure temperature transients - The pressure temperature transient in the primary containment resulting from a break in the portion of the RCIC steam supply line in the drywell is exceeded in severity by the transients resulting from recirculation loop breaks and main steam line breaks, which are discussed in Section 6.2.1.

   Protection against overpressurization of the RCIC pump compartment and of the RCIC pipe chase compartment at Elevation 99 feet 9 inches in the Reactor Building as a result of RCIC steam supply line breaks in these areas is provided by steam venting paths to the suppression chamber compartment, and then to a steam vent containing blowout panels leading to the outside atmosphere. The steam vent and blowout panels are located on the west side of the Reactor Building, as shown on Plant Drawings P-0044-1 and P-0045-1. The RCIC pump compartment is vented to the suppression chamber compartment via hinged, metal plate blowout panels located in the RCIC pump compartment wall at Elevation 71 feet. These hinged panels open to relieve pressure in the RCIC pump compartment, but do not allow pressurization of the suppression chamber compartment to result in steam flow back into the RCIC pump compartment.

Pressure temperature transient analyses for cases involving RCIC steam supply line breaks in the RCIC pump compartment and the RCIC pipe chase compartment were performed using the analytical technique described in Reference 3.6-1 and the blowdown data provided in Table 3.6-4. The flow schematic diagram used for breaks in these two compartments is shown on Figure 3.6-17, and the results of the analyses are listed in Table 3.6-5. Pressure and temperature transient curves for a RCIC steam supply line break in the RCIC pump room are shown on Figures 3.6-24 and 3.6-25, respectively.

The RCIC pump compartment is designed to withstand the maximum pressure due to pipe break. No equipment located in the RCIC pump compartment is required to operate following a RCIC steam supply line break, except the sensing lines to PDSH 9435-1 and 9435-2. These sensing lines do not receive jet impingement or pipe whip loading. The pressure temperature transient in the suppression chamber compartment at the 54-foot elevation, resulting from a break in the portion of the RCIC steam supply line within that compartment, is exceeded in severity by the transient resulting from a HPCI steam supply line rupture in the same compartment. All equipment located in the suppression chamber compartment that is required to operate following a break of the RCIC steam supply line is qualified to operate under environmental conditions more severe than those calculated to occur due to a break of the HPCI steam supply line.

3. Verification of reactor shutdown capability - A RCIC steam supply line break inside the drywell results in a nonisolable blowdown of the reactor vessel. The automatic sequence of events that shuts down the reactor and cools the core is discussed in Section 6.3.2.

For a RCIC steam supply line break outside the drywell, the steam supply line containment isolation valves are closed automatically on high steam flow, terminating reactor vessel blowdown. No reactor scram occurs, due to the rapid termination of blowdown. After investigation of the cause of the steam supply line isolation, the operator initiates a normal shutdown of the reactor.

A combination of pipe whip restraints and separation by distance or intervening structures ensures the availability of essential systems and components in the event of a RCIC steam supply line break occurring in the drywell, the RCIC pipe chase compartment, the suppression chamber compartment, or the RCIC pump compartment. Essential systems and components located in these areas are designed to operate under the environmental conditions resulting from the break. Electrical cabling associated with essential systems and components is either routed so as to avoid jet impingement, or jet impingement barriers are provided.

No pipe whip restraints are provided for the portion of the RCIC steam supply line located within the RCIC pump compartment, because no essential systems or component are impacted by a jet or a pipe whip.

3.6.1.2.1.8 Main Steam Drain Lines

The main steam drain lines connect to the four main steam lines, both inside and outside the drywell. Inside the drywell, 2-inch drain lines that connect to each of the main steam lines are headered together into a single 3-inch line, which then penetrates the primary containment. This 3-inch drain header has two normally open containment isolation valves, one upstream and one

downstream of the primary containment penetration. The inboard and outboard isolation valves are provided with valve operability restraints. The routing of the main steam drain lines is shown on Figures 3.6-26 and 3.6-27.

1. Pipe break locations - The postulated pipe break locations and the pipe whip restraint locations for the main steam drain lines are shown on Figures 3.6-26 and 3.6-27. The calculated stress levels, usage factors, and postulated break types are listed in Tables 3.6-16 and 3.6-17.

2. Compartment pressure temperature transients - The pressure transient in the primary containment resulting from a break in a main steam drain line within the drywell is exceeded in severity by transients resulting from recirculation loop breaks and main steam line breaks. The temperature transient in the primary containment resulting from a main steam drain line break is exceeded in severity by the transient resulting from an intermediate size break. These design basis transients are discussed in Section 6.2.1.

   The pressure temperature transient in the main steam tunnel resulting from a main steam drain line break within the tunnel is exceeded in severity by the transient resulting from a main steam line break.

3. Verification of reactor shutdown capability - A main steam drain line break inside the drywell results in a nonisolable blowdown of steam from the reactor vessel through the broken line. The automatic sequence of events that shuts down the reactor and cools the core is discussed in Section 6.3.2.

For the case of a main steam drain line break inside the main steam tunnel, the resultant temperature rise in the tunnel causes both the MSIVs and the main steam drain line containment isolation valves to close automatically, thereby terminating steam blowdown through the break. After the isolation valves have been closed, the sequence of events is similar to that for a main steam line break outside the drywell.

A combination of pipe whip restraints and separation by distance or intervening structures is used to ensure the availability of essential systems and components in the event of a main steam drain line break in either the drywell or the main steam tunnel.

3.6.1.2.1.9 RPV Head Vent Line

The RPV head vent line is a 2-inch line located entirely within the drywell. From its connection point to a flanged nozzle on the RPV top head, the line is routed horizontally and then generally downward to a penetration through the containment seal plate. From this point, the line continues downward to its connection with the 26-inch main steam line A.

1. Pipe break locations - The postulated pipe break locations for the RPV head vent line are shown on Figure 3.6-28.

   The calculated stress levels, usage factors, and postulated break types are listed in Table 3.6-18.

2. Compartment pressure temperature transients - Since the RPV head vent line is located entirely within the drywell, breakage of this line has no effect on plant areas outside the primary containment. The pressure transient in the primary containment resulting from a break in the RPV head

vent line is exceeded in severity by transients resulting from recirculation loop breaks and main steam line breaks. The temperature transient in the primary containment resulting from a break in the RPV head vent line is exceeded in severity by the transient resulting from an intermediate size break. These design basis transients are discussed in Section 6.2.1.

3. Verification of reactor shutdown capability - An RPV head vent line break results in a nonisolable blowdown of steam into the drywell. The automatic sequence of events that shuts down the reactor and cools the core is discussed in Section 6.3.2.

   Separation by distance or intervening structures ensures the availability of essential systems and components in the event of an RPV head vent line break.

3.6.1.2.1.10 Standby Liquid Control Injection Line

The discharge lines from the two standby liquid control (SLC) injection pumps are headered together outside the drywell and penetrate the drywell as a single 1-1/2 inch line. This line traverses the lower part of the drywell, and connects to the A core spray injection loop line. The SLC injection line is provided with three containment isolation valves: two parallel stop check valves outside the drywell, and a simple check valve inside the drywell. Only that portion of the line between the reactor vessel nozzle and the inboard check valve is considered high energy during periods when the reactor is pressurized.

1. Pipe break locations - The postulated pipe break locations for the SLC injection line are shown on Figure 3.6-29. The calculated stress levels, usage factors, and postulated break types are listed in Table 3.6-19.

2. Compartment pressure temperature transients - Since the high energy portion of the SLC injection line is located entirely within the drywell, a break of this line has no effect on plant areas outside the primary containment. The pressure temperature transient inside the primary containment resulting from a break in the SLC injection line is exceeded in severity by the transients resulting from recirculation loop breaks and main steam line breaks, which are discussed in Section 6.2.1.

3. Verification of reactor shutdown capability - A SLC injection line break between the RPV and the first check valve in that line results in a nonisolable blowdown from the reactor vessel into the drywell. The automatic sequence of events that shuts down the reactor and cools the core is discussed in Section 6.3.2.

   Separation by distance and/or intervening structures ensures the availability of essential systems and components in the event of a SLC injection line break.

3.6.1.2.1.11 RHR Shutdown Cooling Suction Line

The RHR shutdown cooling suction line is a 20-inch line connected to reactor recirculation loop B. The line is routed horizontally and vertically to its containment penetration at Elevation 106 feet. The line is provided with two normally closed containment isolation valves, one inboard and one outboard of the primary containment penetration. Thus, only that portion of the line between the recirculation loop and the inboard containment isolation valve is considered high energy. The routing of the line is shown on Figure 3.6-30.

The RHR shutdown cooling suction line is provided with pipe whip restraints on the portion of the line inside the drywell. Examples of these restraints are shown on Figure 3.6-1.

1. Pipe break locations - The postulated pipe break locations and the pipe whip restraint locations for the RHR shutdown cooling suction line are shown on Figure 3.6-30. The calculated stress levels, usage factors, and postulated break types are listed in Table 3.6-20.

2. Compartment pressure temperature transients - Since the high energy portion of the RHR shutdown cooling suction line is located entirely within the drywell, a break in the line has no effect on plant areas outside the primary containment. The pressure temperature transient in the primary containment resulting from a break in the RHR shutdown cooling suction line is exceeded in severity by the transients resulting from recirculation loop breaks and main steam line breaks, which are discussed in Section 6.2.1.

3. Verification of reactor shutdown capability - An RHR shutdown cooling suction line break results in a nonisolable blowdown of the reactor vessel into the drywell. The automatic sequence of events that shuts down the reactor and cools the core is discussed in Section 6.3.2.

   A combination of pipe whip restraints and separation by distance or intervening structures ensures the availability of essential systems and components in the event of a break in the RHR shutdown cooling suction line.

3.6.1.2.1.12 RHR Shutdown Cooling Return Lines

One 12-inch RHR shutdown cooling return line is associated with RHR loop A and a second return line is associated with RHR loop B. The two lines are routed almost symmetrically on opposite sides of the drywell. Each RHR shutdown cooling return line from the discharge

of the associated RHR pump and heat exchanger penetrates the drywell at Elevation 106 feet. Each line is then routed to its connection with the discharge riser of the reactor recirculation loop at Elevation 114 feet 4-1/2 inches. The routing of this piping is shown on Figure 3.6-31.

Each RHR shutdown cooling return line is provided with a motor operated, normally closed, globe type containment isolation valve outside the drywell. There is also a check valve in each return line inside the drywell. Only that portion of the line between the reactor recirculation line and the inboard check valve is considered high energy.

Each RHR shutdown cooling return line is also provided with pipe whip restraints on the portion of the line inside the drywell. Examples of these restraints are shown on Figure 3.6-1.

1. Pipe break locations - The postulated pipe break locations and the pipe whip restraint locations for the RHR shutdown cooling return lines are shown on Figure 3.6-31. The calculated stress levels, usage factors, and postulated break types are listed in Table 3.6-21.

2. Compartment pressure temperature transients - Since the high energy portion of the RHR shutdown cooling return lines is located entirely within the drywell, a break in one of the lines has no effect on plant areas outside the primary containment. The pressure temperature transient in the primary containment resulting from a break in an RHR shutdown cooling return line is exceeded in severity by the transients resulting from recirculation loop breaks and main steam line breaks, which are discussed in Section 6.2.1.

3. Verification of reactor shutdown capability - An RHR shutdown cooling return line break results in a nonisolable blowdown of the reactor vessel. The automatic sequence of events that shuts down the reactor and cools the core is discussed in Section 6.3.2.

   A combination of pipe whip restraints, and separation by distance or intervening structures ensures the availability of essential systems and components in a event of a break in the RHR shutdown cooling return line.

3.6.1.2.1.13 LPCI Injection Lines

There are four 12-inch LPCI injection lines; one is associated with each of the four RHR pumps. The four lines are routed symmetrically inside the drywell, with the A and C injection lines entering the north side of the drywell and the B and D lines entering the south side of the drywell. Each LPCI injection line penetrates the drywell at Elevation 106 feet and is routed up to Elevation 146 feet 3-1/2 inches, where it connects to a reactor vessel nozzle. The routing of this piping is shown on Figure 3.6-32.

Each LPCI injection line is provided with a motor operated, normally closed, gate type containment isolation valves outside the drywell. There is also a check valve in each injection line inside the drywell. Only that portion of the line between the reactor vessel nozzle and the inboard check valve is considered high energy.

Each LPCI injection line is restrained to prevent pipe whip inside the drywell. Typical pipe whip restraints are shown on Figure 3.6-1.

1. Pipe break locations - The postulated pipe break locations and the pipe whip restraint locations for the LPCI

injection lines are shown on Figure 3.6-32. The calculated stress levels, usage factors, and postulated break types are listed in Table 3.6-22.

2. Compartment pressure temperature transients - Since the high energy portion of the LPCI injection lines is located entirely within the drywell, a break in one of the lines has no effect on plant areas outside the primary containment. The pressure temperature transient in the primary containment resulting from a break in a LPCI injection line is exceeded in severity by the transients resulting from recirculation loop breaks and main steam line breaks, which are discussed in Section 6.2.1.

3. Verification of reactor shutdown capability - A LPCI injection line break results in a nonisolable blowdown of the reactor vessel into the drywell. The automatic sequence of events that shuts down the reactor and cools the core is discussed in Section 6.3.2.

   A combination of pipe whip restraints and separation by distance or intervening structures ensures the availability of essential systems and components in the event of a break in a LPCI injection line.

3.6.1.2.1.14 Core Spray Injection Lines

There are two core spray injection lines: one associated with core spray pumps A and C, and one associated with core spray pumps B and D. The two lines are routed symmetrically within the drywell. Each core spray injection line penetrates the drywell at Elevation 106 feet 9 inches and is routed up to Elevation 156 feet 7-5/8 inches before connecting to a RPV nozzle. The routing of the piping is shown on Figure 3.6-33.

Each core spray injection line is provided with a containment isolation valve outside the drywell. There is also a check valve in each injection line inside the drywell. Only that portion of the line between the reactor vessel nozzle and the inboard check valve is considered high energy.

Each core spray injection line is restrained to prevent pipe whip inside the drywell. Typical pipe whip restraints are shown on Figure 3.6-1.

1. Pipe break locations - The postulated pipe break locations and the pipe whip restraint locations for the core spray injection lines are shown on Figure 3.6-33. The calculated stress levels, usage factors, and postulated break types are listed in Table 3.6-23.

2. Compartment pressure temperature transients - Since the high energy portion of the core spray injection line is located entirely within the drywell, a break in one of the lines has no effect on plant areas outside the primary containment. The pressure temperature transient in the primary containment resulting from a break in a core spray injection line is exceeded in severity by the transients resulting from recirculation loop breaks and main steam line breaks, which are discussed in Section 6.2.1.

3. Verification of reactor shutdown capability - A core spray injection line break results in a nonisolable blowdown of the reactor vessel. The automatic sequence of events that shuts down the reactor and cools the core is discussed in Section 6.3.2.

   A combination of pipe whip restraints and separation by distance or intervening structures ensures the availability of essential systems and components in the event of a break in a core spray injection line.

3.6.1.2.1.15 Control Rod Drive Hydraulic System

The two CRD drive water pumps are located in the reactor building. The high energy discharge pipes from the two pumps are headered together into a single 2-inch pipe that is routed in the reactor building to the CRD hydraulic system master control station at Elevation 102 feet. From this master control station, a 2-inch cooling water header and a 2-inch charging water header are routed to the hydraulic control units (HCUs) on the north side and south side of the drywell.

1. Pipe break locations - Since the CRD pump discharge line is incapable of maintaining a high energy flow stream, pipe whip and jet impingement are not credible, and no whip restraints are provided. Through wall leakage cracks are examined in accordance with Section 3.6.2.1.2.

   Separation by distance and/or intervening structures ensures the availability of essential systems and components in the event of a through wall leakage crack in the control rod drive hydraulic system piping.

2. Compartment pressure temperature transients - Since the normal fluid temperature in the CRD hydraulic system is less than 120°F, no significant pressure temperature transient results from postulated breaks.

3. Verification of reactor shutdown capability - Loss of water pressure due to a break in the CRD pump discharge line, cooling water header, or charging water header does not prevent the control rods from being inserted into the reactor core. Reactor pressure alone is sufficient to fully insert the control rods. At lower reactor pressures, the scram accumulators assist in supplying the energy necessary to insert the control rods.

3.6.1.2.1.16 Auxiliary Steam Lines

Auxiliary steam from the auxiliary boiler is distributed via a 10- and a 16-inch header to the various steam consuming components in the Turbine Building, the radwaste area of the Auxiliary Building, and the Reactor Building. This auxiliary steam header traverses the Auxiliary Building at Elevation 96 feet, passing into the Turbine Building. A 6-inch steam supply line enters the Reactor Building to provide steam for HPCI and RCIC turbine testing at Elevation 73 feet 7-3/8 inches.

1. Pipe break locations - Since the auxiliary steam lines consist of nonnuclear class piping, breaks are postulated to occur at each location of potential high stress, such as pipe fittings, valves, and welded attachments.

2. Compartment pressure temperature transients - The portions of the auxiliary steam lines routed through the Auxiliary Building and the Reactor Building are unpressurized during normal reactor operation. Pressure temperature transient analyses were not performed.

3. Verification of reactor shutdown capability - An auxiliary steam line break has no effect on operation of the reactor, since reactor systems, including electrical cabling, are not located in areas through which the auxiliary steam lines are routed.

3.6.1.2.1.17 Reactor Vessel Drain Line

The 2-inch reactor vessel drain connects to the reactor vessel at Elevation 114 feet 0-1/2 inches. The drain line is routed vertically and horizontally until it penetrates the west side of the reactor pedestal at Elevation 102 feet 9 inches as a 4-inch line. The drain line is routed horizontally south through control and bypass valves and then vertically to Elevation 120 feet 6 inches,

where it ties into the reactor water clean up system, without penetrating the primary containment.

1. Pipe break locations - The postulated pipe break locations for the reactor vessel drain line is shown on Figure 3.6-34. The calculated stress levels, usage factors, and postulated break types are listed in Table 3.6-24.

2. Compartment pressure temperature transients - The pressure transient in the primary containment resulting from a break in the reactor vessel drain line within the drywell is exceeded in severity by transients resulting from recirculation loop breaks and main steam line breaks. The temperature transient in the primary containment resulting from a reactor vessel drain line break is exceeded in severity by the transient resulting from an intermediate size break. These design basis transients are discussed in Section 6.2.1.

3. Verification of reactor shutdown capability - A reactor vessel drain line break inside the drywell results in a nonisolable blowdown of the reactor vessel through the broken line. The automatic sequence of events that shuts down the reactor and cools the core is discussed in Section 6.3.2.

   A combination of pipe whip restraints and separation by distance and/or intervening structures is used to ensure the availability of essential systems and components in the event of a reactor vessel drain line break in the drywell.

3.6.1.2.1.18 Reactor Vessel Head Spray Line

The head spray line has been removed. However, the analysis of a postulated head spray line break discussed in Appendix 6B bounds the effects of an RPV head vent line break. Therefore, the discussion regarding a postulated head spray line break is still valid.

Compartment pressure temperature transients - Since the high energy portion of the reactor vessel head spray line was located entirely within the drywell, a postulated break in the line has no effect on the plant areas outside the containment. The drywell head region is pressurized by postulated breaks on the head spray line above the containment seal plate. The pressure temperature transient for the drywell head region is discussed in Appendix 6B. The pressure temperature transient in the

remainder of the primary containment resulting from a postulated break in the reactor head spray line is exceeded in severity by transients resulting from recirculation loop breaks and main steam line breaks, which are discussed in Section 6.2.1.

3.6.1.2.1.19 Standby Diesel Generator Starting Air System

The 3-inch standby diesel generator starting air line connects to the diesel generator from the starting air skid. There are 4 starting air skids (1 per diesel generator). Each skid is located in the respective diesel generator compartment.

1. Pipe break locations - The postulated pipe break locations and the pipe whip restraint locations for the starting lines are shown on Figure 3.6-48. The calculated stress levels, usage factors, and postulated break types are listed in Table 3.6-29.

   For the purposes of pipe break and jet impingement analysis the emergency diesel generator and its associated auxiliaries are considered a single system. As a single system a single failure is only required to be postulated

in one system. Separation of the diesel generator rooms by 18 inch reinforced concrete walls protect other diesel generator units and auxiliaries from damage due to a pipe break in adjacent diesel generator rooms. Therefore, a pipe break in any one of the diesel generator rooms will not affect the remaining diesel generator units and their associated auxiliaries.

All of the air start piping, valves and receivers from the check valve on the air receiver inlet (including the check valve) to the air start solenoid valve on the engine are designed to Seismic Category I, ASME Section III, Class 3, requirements. Refer to Vendor Technical Document PM018Q-0048 for component descriptions.

The compressor, air dryer, and piping up to the air receiver inlet check valve are not built to meet ASME code requirements because they do not serve a safety-related function. The air start valves, air distributors, and the diesel engine cylinders are all pressure retaining parts downstream of the air start solenoid valves which do serve a safety-related function and are non-ASME code items built to Seismic Category I requirements. The air start solenoid pilot valves reduce the starting air pressure to approximately 250 psi, therefore, these components, which are downstream of the air start solenoid pilot valves, are actually located in a moderate energy portion of the system. The non-ASME III pipe in the air-start system is designed to Seismic Category I requirements. These are specialty items that are not available as ASME components but which are built to the SDG manufacturers own critical specifications (see Table 3.2-1, Item XII).

2. Compartment pressure - temperature transients - This high energy line contains compressed air. Therefore, the mass momentum of the high energy line will be lower than the comparable steam system of same pressure and temperature.

Also, the doors of each compartment are not pressure tight and they do not communicate directly with other compartments. As a result, the additional air mass discharged into the compartment will be vented to the corridors on both ends of the compartments and through the HVAC ducts. Therefore, the compartment will not experience room pressurization due to starting air line break. Also, the safety-related commodities in the room are qualified to operate under the environmental conditions specified in Table 3.11-1C.

3. Verification of reactor shutdown capability - The starting air line break does not directly cause a trip of the turbine generator. Therefore, offsite power is assumed to be available (See Section 3.6.1.1). As a result, normal shutdown sequence will be followed to achieve cold shutdown and the standby diesel generators will not be started or required. The starting air skids are located in the respective diesel generator compartment. Therefore, the effects of starting air line breaks are confined to the diesel generator compartment which is in the diesel generator area of Auxiliary Building. A combination of pipe whip restraints and separation by distance or intervening structures ensures the availability of essential systems and components in the event of a break in the starting air line. As a result, the safe shutdown equipment located in the Reactor Building will not be affected.

3.6.1.2.2 Moderate Energy Fluid Systems

Through wall leakage crack locations are postulated in areas containing essential systems and components in accordance with the criteria stated in Section 3.6.2.1.2. When moderate energy fluid systems share a compartment with safety-related components, the effects of water spray and flooding are reviewed, although pipe

crack locations may not be postulated. When moderate energy fluid systems share a compartment with high energy fluid systems, the water spray and flooding effects resulting from a postulated moderate energy line leakage crack are considered if the effects exceed those resulting from the postulated high energy line break effects.

The moderate energy piping failure review was conducted in the Reactor Building, and diesel and control areas of the Auxiliary Building. The postulated failure of a moderate energy line can at most affect only the operations of one train of a redundant safety-related system due to the provisions for physical separation of redundant trains. Further, for the purpose of this evaluation, it was conservatively assumed that the impacted train failed to perform its safety function. The crack sizes postulated, and the nominal pipe sizes in which moderate energy pipe cracks are postulated to occur, are discussed in Section 3.6.2.1.3.3. Additional criteria used in the moderate energy fluid systems analysis are discussed in Section 3.6.1.1.

A list of moderate energy systems with the normal operating temperature and pressure is provided in Table 3.6-28. The definition of a moderate energy fluid system is stated in Section 3.6.3.3.

Operation of the Standby Diesel Generators (SDG) is not required during the normal plant operating conditions defined in SRP 3.6.1, however, the fuel oil transfer line is pressurized by the static head of the fluid in the line while the SDG is not in operation. During SDG operation, the fuel oil transfer line is pressurized to approximately 47 psig. It is routed from the fuel oil storage tank at elevation 54' through the recirculation ventilation room (see Section 9.4.6) on elevation 77' to the respective fuel oil day tank on elevation 102'. Any cracks in this line would only effect systems associated with the diesel being served by that transfer

line because of SDG compartmentalization. However, a review of the potential fire hazard created by the fluid spray was performed. The fuel oil would have to be heated above its flash point of 125° by any potential ignition source. The fuel oil transfer pumps at elevation 54' are canned pumps. The ventilation fans are direct drive and completely contained within the distribution ductwork. These units contain no heating coils that could act as potential ignition sources.

3.6.1.3 Safety Evaluation

The analyses of postulated pipe ruptures summarized in Section 3.6.2 verify that the consequences of any single rupture of fluid system piping in the plant do not prevent safe shutdown of the reactor.

The offsite radiological consequences of piping ruptures are enveloped by a Reactor Recirculation System break inside the primary containment, and by main steam system and feedwater system breaks outside the primary containment. The radiological consequences of these breaks are presented in Sections 15.6.5, 15.6.4, and 15.6.6, respectively.

Special consideration has been given to separation of areas in the Reactor Building containing essential systems and components from high energy pipe break compartments and the effects of postulated pipe ruptures. HVAC ducts penetrating high energy pipe break compartment walls are equipped with backpressure dampers, while other types of penetrations through the walls are designed as steam tight.

3.6.2 Determination of Pipe Failure Locations and Dynamic Effects Associated With Postulated Piping Failures

Information concerning break and crack location criteria and methods of analysis is presented in this section. The break location criteria and methods of analysis are needed to evaluate the dynamic effects associated with postulated ruptures of high and moderate energy piping inside and outside the primary containment.

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

3.6.2.1.1 Break Locations in High Energy Fluid System Piping

The consequences of high energy line cracks have been considered during the review of high energy line breaks. Jet impingement pressure and temperature, pipe whip, environmental effects, etc., for high energy piping system line breaks have been evaluated in accordance with Sections B.1.a and B.1.c of BTP MEB 3-1 (SRP 3.6.2). Due to this review, any pipe failure consequence that could adversely affect the safety of the plant has been considered. This conclusion is based on the following:

1. The criteria in Section B.1.a are invoked whenever possible to separate essential equipment from high energy piping. In this case, breaks are arbitrarily postulated without consideration of stress levels.

2. When it is not possible to separate high energy piping from essential equipment, redundancy is provided or an evaluation is performed to ensure that the equipment will remain operable.

3. In areas in which high energy pipe is routed a sufficient number of breaks have been postulated such that the effects of jet impingement, pipe whip, environment, etc., envelop any postulated leakage crack effects.

The following discussion shows that for all areas of the plant the existing criterion used to postulate high energy line breaks encompasses the effects of high energy line cracks:

<u>High energy piping outside the reactor building</u>

The separation review program ensures that high energy pipes are not routed near systems, components, or structures essential to safe shutdown in areas other than the reactor building. The piping in this area meets the criteria of Section B.1.a where breaks are arbitrarily postulated to ensure separation of high energy piping and essential equipment. It is therefore concluded that cracks in this area will not degrade the safety of the plant.

<u>High energy piping in the reactor building (excluding primary containment and containment penetration areas)</u>

Excluding the main steam tunnel piping, the systems which qualify as high energy piping in the reactor building are the RWCU, CRD, RCIC, and HPCI systems. Routing of the RWCU, RCIC, and HPCI systems has been controlled so that they are located in well defined areas (e.g., pipe chases, pump rooms, torus compartment, etc.). These compartments have also been evaluated for environmental, flood, pressure, etc., effects using the worst-case pipe break condition. Breaks in these areas are postulated as described in this section, to thoroughly encompass the effects of high energy pipe cracks. The CRD system analysis is described in Section 3.6.1.2.1.15. Cracks in the CRD system have been examined at every fitting and change of direction in accordance with Section 3.6.2.1.2. It is therefore concluded that cracks in this area will not degrade the safety of the plant.

The main steam tunnel has numerous postulated pipe breaks. Several other lines also have breaks postulated at every fitting and change of direction. The effects of these postulated pipe breaks in the tunnel will therefore encompass the effects of any pipe cracks that may have been postulated.

Primary containment: During the pipe break review, in excess of 360 high energy line jets were examined for their consequences. In light of the separation between the high energy systems in the primary containment, it is reasonable to assume that these high energy breaks will always govern. Any equipment, system, or structure in primary containment must be designed for the extreme environment regardless of its particular location. The combination of separation and redundancy (the preferred method of protection) is also integral to components and piping routed in the primary containment. This is verified in the jet impingement evaluation where breaks are postulated at various elevations and azimuths. It is therefore concluded that the effects of pipe cracks in this area are less severe than the effects of high energy pipe breaks, and therefore acceptable.

See Reference 3.6-12 regarding the postulation of intermediate pipe breaks. See Reference 3.6-13 regarding acceptability for elimination of arbitrary intermediate pipe breaks. See References 3.6-14 and 3.6-15 regarding the elimination of additional intermediate pipe break locations.

3.6.2.1.1.1 Piping in Containment Penetration Areas

Except for the feedwater system, high energy pipes penetrating the primary containment are provided with valve operability restraints that are located reasonably close to the containment isolation valves and are designed to withstand the loadings resulting from a pipe break either inboard of the inboard isolation valve restraints or outboard of the outboard isolation valve restraints so that neither isolation valve operability nor leaktight integrity of the containment penetration would be impaired as a result of such pipe breaks. Terminal ends of the piping runs extending beyond these portions of high energy piping are considered to originate at a point adjacent to the required valve operability restraints and outboard of the outboard isolation valve operability restraints or inboard of the inboard isolation valve operability restraints.

Valve operability restraint configuration and break locations of the feedwater system are explained in Section 3.6.1.2.1.3.

Breaks are not postulated in these portions of high energy piping in containment penetration areas provided that the following design stress and fatigue limits are satisfied:

1. For ASME B&PV Code, Section III, Class 1 Piping:

   a. The maximum stress range, S_n, calculated by equation 10 of Paragraph NB-3653 of the ASME B&PV Code, Section III, does not exceed 2.4 S_m for those loads and conditions for which normal and upset stress limits have been specified, including an operating basis earthquake (OBE) transient.

   b. If the maximum stress range of equation 10 exceeds 2.4 S_m, the stress ranges calculated by both equation 12 and equation 13 of Paragraph NB-3653 do not exceed 2.4 S_m.

   c. The cumulative usage factor associated with normal, upset, and testing conditions is less than 0.1.

   d. The loading resulting from a postulated pipe break beyond these portions of the piping does not cause the stress as calculated by equation 9 of Paragraph NB-3652 to exceed 2.25 S_m, except for the portion of piping between the isolation valve and the adjacent restraints protecting the operability of the valve. For this latter portion of piping, higher stresses are permitted, provided that a plastic hinge is not formed and that the operability of the isolation valve is ensured.

2. For ASME B&PV Code, Section III, Class 2 and 3 Piping:

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

   b. The maximum stress, as calculated by equation 9 of Paragraph NC-3652, under the loadings resulting from a postulated rupture of fluid system piping beyond these portions of piping, does not exceed 1.8 S_h. Higher stresses are permitted in pipe between the outboard isolation valve and the adjacent restraints protecting the operability of the valve, provided that:

      (1) All circumferential and longitudinal welds in that pipe region are fully radiographed

      (2) Analysis shows that a plastic hinge is not formed and that the operability of the valve is ensured.

In addition to these stress and fatigue criteria, high energy piping in containment penetration areas must meet the following requirements:

1. Welded pipe support attachments are avoided to eliminate stress concentrations.

2. The number of circumferential and longitudinal pipe welds and branch connections is minimized.

3. The length of the piping run is minimized, consistent with requirements to keep stress levels low and to provide access for inservice inspection.

4. The design at points of pipe anchors, welded connections, and containment penetrations does not require welding directly to the outer surface of the piping (flued, integrally forged pipe fittings are acceptable), except where such welds are 100 percent volumetrically examinable inservice and a detailed stress analysis is performed to demonstrate compliance with the limits of the stress and fatigue criteria stated above.

5. To the extent practicable, the inservice examination completed during each inspection interval will provide volumetric examination of circumferential and longitudinal pipe welds within these portions of piping, as required by ASME B&PV Code, Section XI. See Sections 5.2.4 and 6.6 for additional information.

6. When a no-break region is established, the terminal end for piping in the region is consequently shifted away from the containment anchor. The terminal end is located adjacent to the pipe whip restraints that limit the bending and torsion moments exerted on the isolation valve as a consequence of pipe break. These restraints are:

   a. Located reasonably close to the isolation valves

   b. Capable of withstanding the loadings resulting from postulated pipe rupture beyond this portion of the piping such that neither valve operability nor the leaktight integrity of the primary containment is impaired.

7. Operability of the isolation valve is ensured for pipe break events where it is required to ensure primary containment integrity.

The main steam and the main feedwater lines are conservatively designed in accordance with the criteria stated above. However, for additional conservatism, the enclosure for the main steam and main feedwater lines is also designed for the effects of pressurization, flooding, and the environment resulting from the single area crack of either the main steam or main feedwater lines. The nonmechanistic single area pipe crack of the main steam or main feedwater lines is postulated to occur either upstream or downstream of the outboard containment isolation valves. Safety-related equipment in the enclosure is environmentally qualified for the pressure, temperature, radiation, humidity, and flooding resulting from the worst case, single area pipe crack.

A mechanistic double-ended break of the largest branch line of the feedwater system is also postulated to occur within this enclosure. The effects of pipe whip, jet impingement, pressurization, and flooding due to the branch line break are considered in the evaluation of the enclosure design adequacy.

3.6.2.1.1.2 Recirculation System Piping

See Section 3.6.2.6 for a discussion of recirculation system piping.

3.6.2.1.1.3 Class 1 Piping (Other Than Recirculation System Piping and Piping in Containment Penetration Areas)

Breaks in high energy Class 1 piping (ASME B&PV Code, Section III) are postulated to occur at the following locations:

1. At terminal ends of piping runs or branch runs

2. At intermediate locations between terminal ends, as determined by one of the two following criteria:

a. The maximum range of stress intensity as calculated by ASME B&PV Code equation 10 and either equation 12 or 13 exceeds 2.4 S_m.

b. The cumulative usage factor exceeds 0.1.

When the above stress and usage factor criteria are not exceeded, the minimum of two intermediate breaks based on highest stress, as calculated by Equation 10 of Paragraph NB-3653, are not postulated unless the break location is in the proximity of a welded attachment.

Intermediate pipe break locations are initially based upon committed design piping stress calculations in accordance with the above criteria. As a result of piping reanalysis, the highest stress locations may be shifted. An initially determined pipe break location will not be changed as a consequence however unless one of the following conditions exist:

1. Reanalysis shows that the maximum stress range or the cumulative usage factor at another location not only exceeds that for the initial pipe break location but also exceeds the above pipe break criteria. In addition, the break at the new location results in more serious consequences to safety-related systems than the initial break.

2. Significant changes are made in the routing, size, or wall thickness of the pipe after the initial pipe break determination.

3.6.2.1.1.4 Class 2 and 3 Piping (Other Than Recirculation System Piping and Piping in Containment Penetration Areas)

Breaks in high energy Class 2 and 3 piping (ASME B&PV Code, Section III) are postulated to occur at the following locations:

1. At terminal ends of piping runs or branch runs

2. At intermediate locations between terminal ends, as determined by one of the two following criteria:

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

   b. At each location where the maximum stress range, as calculated by the sum of equations 9 and 10 of Paragraph NC-3652, considering normal and upset plant conditions, exceeds $0.8(1.2\ S_h + S_A)$.

   When the above stress criteria are not exceeded, the minimum of two intermediate breaks based on highest stress, as calculated by the sum of Equations 9 and 10 of Paragraph NC-3652, are not postulated unless the break location is in the proximity of a welded attachment.

   Intermediate pipe break locations are initially based upon committed design piping stress calculations in accordance with the above criteria. As a result of piping reanalysis, the highest stress locations may be shifted. An initially determined pipe break location will not be changed as a consequence, however, unless one of the following conditions exist:

   1. Reanalysis shows that the maximum stress range at another location not only exceeds that for the initial pipe break location but also exceeds the

above pipe break criteria. In addition, the break at the new location results in more serious consequences to safety-related systems than the initial break.

2. Significant changes are made in the routing, size, or wall thickness of the pipe after the initial pipe break determination.

3.6.2.1.1.5 Nonnuclear Class Piping

Breaks in high energy nonnuclear class piping are postulated to occur at the following locations:

1. At terminal ends of piping runs or branch runs

2. At each intermediate location of potential high stress, such as pipe fittings with elbows, tees, reducers, etc; valves; and welded attachments.

Alternatively, the break locations for nonnuclear class piping can be selected according to the same criteria used for Class 2 and 3 piping, provided that all necessary analyses are made.

3.6.2.1.2 Crack Locations in Moderate Energy Fluid System Piping

Through wall leakage cracks are postulated to occur in moderate energy piping located in areas containing essential systems and components. Cracks are postulated to occur in accordance with either of the two following criteria:

1. At locations of potential high stress, such as pipe fittings with elbows, tees, reducers, etc; valves; and welded attachments

2. For Class 1 piping (ASME B&PV Code, Section III), at locations where the maximum stress range, as calculated by equation 10 of Paragraph NB-3652, exceeds 1.2 S_m, and for

Class 2 or 3 piping (ASME B&PV Code, Section III) or nonnuclear piping, at locations where the maximum stress range, as calculated by the sum of equations 9 and 10 of Paragraph NC-3652, exceeds 0.4 $(1.2\ S_h + S_A)$.

The above criteria notwithstanding, cracks are not postulated in those portions of moderate energy piping located in the following areas:

1. Areas in which high energy pipe breaks are postulated, provided that moderate energy piping cracks would not result in more severe environmental conditions than the high energy pipe breaks.

2. Between containment isolation valves, provided that:

   a. The piping meets the requirements of Subarticle NE-1120 of the ASME B&PV Code, Section III

   b. The maximum stress range for Class 1 piping (ASME B&PV Code, Section III), as calculated by equation (10) of Paragraph NB-3652, does not exceed 1.2 S_m, and the maximum stress range for Class 2 and 3 (ASME B&PV Code, Section III) or nonnuclear piping, as calculated by the sum of equations 9 and 10 of Paragraph NC-3652, does not exceed 0.4 $(1.2\ S_h + S_A)$.

3.6.2.1.3 Types of Breaks and Cracks in Fluid System Piping

3.6.2.1.3.1 Circumferential Breaks

A circumferential break is assumed to result in both:

1. Severance of a high energy pipe on a plane perpendicular to the pipe axis

2. Separation amounting to at least a one-diameter lateral displacement of the ruptured piping ends, unless physically limited by piping restraints, structural members, or piping stiffness. 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.

Circumferential breaks are postulated in high energy fluid system piping of nominal pipe size greater than 1 inch, at the locations determined by the criteria listed in Section 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 a longitudinal break is postulated in pipes of nominal pipe sizes 4 inches and larger.

3.6.2.1.3.2 Longitudinal Breaks

A longitudinal break is assumed to result in an axial split parallel to the pipe axis, without causing pipe severance. The break opening area is assumed to be equal to the effective cross-sectional flow area of the pipe at the break location. The split is assumed to be oriented so that the jet reaction force causes out of plane bending of the piping configuration. Piping movement is assumed to occur in the direction of the jet reaction unless limited by piping restraints, structural members, or piping stiffness.

Longitudinal breaks are postulated in high energy fluid system piping of nominal pipe sizes 4 inches and larger, at the locations determined by the criteria listed in Section 3.6.2.1.1, with the following exceptions. Longitudinal breaks are not postulated:

1. At terminal ends

2. At intermediate break locations chosen to satisfy the criterion for a minimum number of break locations

3. At locations where the criteria of Section 3.6.2.1.1 are not satisfied, but it is shown that the maximum stress is in the longitudinal direction and is at least 1.5 times the circumferential stress, in which case only circumferential breaks are postulated.

3.6.2.1.3.3 Through Wall Leakage Cracks

Through wall leakage cracks are postulated to occur in moderate energy fluid system piping exceeding a nominal pipe size of 1 inch, at the locations determined by the criteria listed in Section 3.6.2.1.2. A crack is assumed to occur at any orientation about the circumference of a pipe. Fluid flow from a crack is based on a circular opening with an area equal to that of a rectangle one half the pipe diameter in length and one half the pipe wall thickness in width.

3.6.2.2 <u>Analytical Models to Define Forcing Functions and Response Models</u>

3.6.2.2.1 Recirculation Piping System

See Section 3.6.2.6.2 for a discussion relating to the Recirculation Piping System.

3.6.2.2.2 Piping Systems Other Than The Recirculation Piping System

Analysis to determine the jet impingement effects and the piping and restraint displacements resulting from a pipe break are performed in general accordance with Reference 3.6-7. Analysis of jet thrust forces is described in Section 2.2 of Reference 3.6-7. Fluid jet impingement forces are discussed in Section 2.3 of Reference 3.6-7.

Impulsive loading and impact combined with impulsive loadings are described in Sections 3.2 and 3.3, respectively, of Reference 3.6-7.

Piping response to pipe break loads are analyzed by various methods, applied in the appropriate circumstances. These methods include energy balance, lumped parameter and non-linear time history analysis models. The forcing function used in piping dynamic analysis is obtained using Reference 3.6-8. A typical pipe break forcing function and piping system model used for the dynamic response analysis are provided on Figure 3.6-38. Pipe rebound effects are also considered in this analysis.

Two different types of pipe break whip design problems are addressed. The first type of problem is to ensure the operability of containment isolation valves and the leaktight integrity of the primary containment following any postulated pipe break. The second type of problem is to ensure that pipe whip resulting from postulated breaks is controlled sufficiently to prevent damage to adjacent safety-related systems.

Valve operability and primary containment integrity is verified by dynamic analysis of the piping system in the containment penetration area under the conditions imposed by pipe break outside that region, beyond the moment limiting restraints near the isolation valve. The bases for ensuring valve operability are as follows:

1. For the postulated pipe break, pipe stress at the junction with the isolation valve does not exceed 1.1 times the static minimum yield strength.

2. Rigid restraints intended for ensuring valve operability do not exceed their pipe break design load.

3. Piping in the containment penetration area exceeds the limits of neither Section 3.6.2.1.1.1.(1.d) for ASME

Class 1 pipe nor Section 3.6.2.1.1.1.(2.a) for Class 2 or 3 pipe.

4. Valve operator peak acceleration, in multiples of gravity, does not exceed the fundamental frequency of the acceleration response, in hertz.

5. Pipe restraints designed for normal operation design load events are assumed to contribute no pipe break restraint.

Protection of essential systems against uncontrolled pipe whip resulting from postulated breaks is ensured by pipe whip restraints, which are located so as to most effectively limit abnormal pipe movement. Pipe whip restraints are designed to permit free pipe movement during normal design events, but to limit the pipe break whip to acceptable movements. The pipe whip restraints are designed to provide the strength, stiffness, and pipe whip energy absorption capacity needed to limit pipe motion. The bases for design and dynamic analysis to control pipe break whipping motion are:

1. The postulated pipe break is permitted to cause neither pipe whip restraint failure nor pipe whip motion threatening to an essential system.

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

3. Dynamic analytical methods used for calculating the piping/restraint system response to the pipe break forces adequately account for the effects of the following:

   a. Pipe mass, stiffness, and resistance to dynamic plastic hinge formation and propagation

b. Pipe whip restraint resistance to pipe impact in terms of stiffness, yield strength, and impact energy absorption

c. Transient time history of the pipe break blowdown forces acting on the exit leg of the pipe rupture

d. The requirement for clearance between pipe and pipe whip restraint during any normal design event.

4. Plastic deformation design limit for structural members of pipe whip restraints is limited to the ductility ratio of the system. A ductility ratio limit equal to 20 is used for compression, flexure, and shear. A review of the design of structural steel beams in flexure, for loads other than tornado, indicates that the demands for ductility ratios are less than 10. The ductility ratio of 50 percent of ultimate strain divided by yield strain is used for tension members. The members are proportioned to preclude lateral and local buckling.

3.6.2.2.2.1 RELAP4/MOD5

A lumped parameter model simulates the ruptured piping system for input into the computer code RELAP4/MOD5, as discussed in Reference 3.6-2. The code computes time, varying pressure, momentum flux, and mass acceleration throughout a system containing water, steam, and/or a two phase mixture. From these data, the blowdown reaction load is computed by a postprocessor REPIPE, discussed in Reference 3.6-10, using the following relations:

(3.6-1)

$$F_M = \frac{\delta}{\delta t}\iiint \vec{u}\,.\,\vec{m}\,(\rho dV + \iiint \rho\vec{g}\,.\,\vec{m}dV - \iint [P_O\vec{n}_O+\vec{u}_O(\rho_O\vec{u}_O\vec{n}_O)]\,.\,\vec{m}dS_O +$$

$$\iint_{S_i}[P_i\ \vec{n}_i\ +\vec{u}_i\ (\rho_i\ \vec{u}_i.\vec{n}_i)]\,.\,\vec{m}dS_i$$

where:

F = total resultant force acting on pipe
m = unit vector in the direction of force
V = volume of RELAP4 control volume
u = unit vector in direction of local fluid velocity
q = mass density of fluid
S = control volume surface
o = subscript for outlet
n = unit vector in the direction of positive flow
i = subscript for inlet
p = pressure.

3.6.2.3 Dynamic Analysis Methods to Verify Integrity and Operability

3.6.2.3.1 Recirculation Piping System

See Section 3.6.2.6.3 for a discussion of the Recirculation Piping System.

3.6.2.3.2 Piping Systems Other Than The Recirculation Piping System

The pipe break restraints provided for protection from high energy pipe breaks are of two basic types: pipe whip restraints and valve operability restraints. Pipe whip restraints are provided solely to protect nearby structures and essential equipment from damage due to whipping pipes and are designed so that a gap is maintained between the pipe and the restraint during normal plant conditions. Valve operability restraints are provided near primary containment isolation valves whose operability is required following a break of the pipe in which they are installed. These operability restraints are designed to limit the stress in the piping near the valve to below the dynamic yield strength of the material in order to ensure operability of the valve. To accomplish this function, it is normally necessary to minimize the gap between the pipe and the restraint so that contact occurs during normal plant conditions.

3.6.2.3.2.1 Design Loading Combinations

The design loading combinations applied in the design of restraints are categorized with respect to the plant operating conditions, which are identified as normal, upset, emergency, and faulted, as described in Section 3.9.3. Pipe break is considered as a faulted plant condition for those piping systems remaining intact. For the high energy piping system in which the break has occurred, the ASME B&PV Code, Section III categorization of operating conditions no longer controls the design.

3.6.2.3.2.2 Design Stress Limits

3.6.2.3.2.2.1 Valve Operability Restraints

When restraints for piping are designed so that contact between pipe and restraint will occur during normal plant conditions, the design loading combinations for normal, upset, emergency, and faulted

conditions are applicable to pipe without rupture. In evaluating the supports and restraints for normal operation of pipe Classes 1, 2, and 3 (ASME B&PV Code, Section III), the design stress limits applied in evaluating loading combinations for normal, upset, emergency, and faulted (except for pipe rupture) conditions are those given in Tables 3.9-9 and 3.9-13. After rupture of the supported pipe occurs, the piping system is no longer within the jurisdiction of the ASME B&PV Code, Section III, because the pressure boundary has been breached. The restraints are evaluated for pipe rupture loads as described in Section 3.6.2.2.2.

3.6.2.3.2.2.2 Pipe Whip Restraints

When restraints are designed solely to control pipe whip 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 pipe whip restraints 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 safe shutdown earthquake (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, shims are provided to permit adjustment of the gap size during hot functional testing.

Pipe whip restraints are evaluated for the pipe rupture loads as described in Section 3.6.2.2.2.

3.6.2.4 <u>Guard Pipe Assembly Design Criteria</u>

Guard pipe assemblies are not used at HCGS.

3.6.2.5 Material to be Submitted for the Operating License Review

Pipe break and crack locations were obtained in accordance with the criteria of Section 3.6.2.1. High energy piping break locations as well as break types, circumferential or longitudinal, are identified on piping isometric drawings provided in Section 3.6.1 and referenced in Section 3.6.1.2.1.

The augmented inservice inspection requirement is described in Section 6.6. Pipe whip restraints were designed as discussed in Section 3.6.2.3. The restraint locations and orientations are shown on various figures referenced in Section 3.6.1.2.1. Jet thrust and impingement forces were determined in accordance with Sections 3.6.2.2 and 3.6.2.3.

The effects of breaks and cracks are discussed in detail in Section 3.6.1. The results are based on the protection evaluation criteria provided in Section 3.6.1. Any protection measures to ensure safe shutdown, i.e, barriers, separation, and restraints, are also discussed.

3.6.2.6 Determination of Break Locations and Dynamic Effects Associated with the Postulated Rupture of Recirculation System Piping (NSSS)

3.6.2.6.1 Criteria Used to Define Break Location and Configuration

The following section establishes the criteria for the location and configuration of postulated breaks.

3.6.2.6.1.1 Definition of High Energy Fluid System

High energy fluid systems are defined to be those systems, or portions of systems, that during normal plant conditions are either in operation or are maintained pressurized under conditions where either one or both of the following are met:

1. Maximum operating temperature exceeds 200°F

2. Maximum operating pressure exceeds 275 psig.

The recirculation piping system is a high energy fluid system designed to the ASME Boiler and Pressure Vessel (B&PV) Code, Section III, Class 1 requirements. No portion of the recirculation piping system is a moderate energy fluid system.

Normal plant conditions are defined as the plant operating conditions during reactor startup, operation at power, hot standby, or reactor cooldown to a cold shutdown condition.

3.6.2.6.1.2 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 the development of a sudden, longitudinal break and is postulated for high energy fluid systems only.

The following high energy piping systems or portions of piping systems are considered to have a potential for initiation of a postulated pipe break during normal plant conditions and are analyzed for potential damage due to dynamic effects:

1. All piping that is part of the RCPB and subject to reactor pressure continuously during plant operation

2. All piping that is beyond the second isolation valve, but that is subject to reactor pressure continuously during plant operation

3. In addition to piping under 1. and 2., all other piping systems or portions of piping systems considered high energy systems.

Portions of piping systems that are isolated from the source of the high energy fluid during normal plant conditions are exempted from consideration of postulated pipe breaks. This would include portions of piping systems beyond a normally closed valve. Pump and valve bodies are also exempted from consideration of pipe break because of their greater wall thickness.

A high energy piping system break is neither postulated to occur simultaneously with a moderate energy piping system break nor is any pipe break outside primary containment postulated to occur concurrently with a postulated pipe break inside primary containment.

3.6.2.6.1.3 Exemptions from Pipe Whip Protection Requirements

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

1. 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.

2. Piping for which the internal energy level associated with whipping is insufficient to impair the safety function of any structure, system, or component to an unacceptable level. Any line restrictions, e.g., flow limiters, between the pressure source and break location, and the effects of either a single ended or double ended flow condition, are accounted for in the determination of the internal fluid energy level associated with the postulated pipe break reaction. The energy level in a whipping pipe is considered as insufficient to rupture an impacted pipe of equal or greater nominal pipe size and equal or heavier wall thickness.

3.6.2.6.1.4 Location for Postulated Pipe Breaks

Postulated pipe break locations are selected in accordance with the intent of Regulatory Guide 1.46, the U.S. Nuclear Regulatory Commission (NRC) Branch Technical Position (BTP) APCSB 3-1, Appendix B, and as expanded in NRC Branch Technical Position MEB 3-1. For ASME B&PV Code, 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 pipe run. Terminal ends are extremities of piping runs that connect to structures, equipment, or pipe anchors that act as rigid constraints to piping motion and thermal expansion. A branch connection to a main piping run is a terminal end for a branch run, except when the branch and main run is modeled as a common piping system during the piping stress analysis.

2. At intermediate locations between the terminal ends where the maximum stress range between any two load sets, including the zero load set, according to Subarticle NB-3600 of the ASME B&PV Code, Section III, for upset plant conditions and an independent OBE event transient, exceeds the following:

   a. If the stress range, as calculated using equation 10 of the ASME B&PV Code exceeds 2.4 S_m but is not greater than 3 S_m, no breaks are postulated unless the cumulative usage factor exceeds 0.1.

   b. The stress ranges, as calculated by equations 12 or 13 of the ASME B&PV Code, exceed 2.4 S_m or if the cumulative usage factor exceeds 0.1 when equation 10 exceeds 3 S_m.

3. When the above stress and usage factor criteria are not exceeded, the minimum of the two intermediate breaks based on highest stress, as calculated by Equation 10 of Paragraph NB-3653, are not postulated, unless the break location is in the proximity of a welded attachment.

3.6.2.6.1.5 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 1 inch.

2. Circumferential breaks are postulated only in piping exceeding a 1-inch nominal pipe diameter.

3. Longitudinal breaks 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. At each of the intermediate postulated break locations identified to exceed the stress and usage factor limits of the criteria in Section 3.6.2.6.1.4 for Class 1 piping systems, either a circumferential or a longitudinal break, or both, are postulated per the following:

    a. Circumferential breaks are postulated at fitting joints.

    b. Longitudinal breaks are 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 is 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 the maximum stress range in the circumferential direction is greater than 1.5 times the stress range in the longitudinal direction, only the longitudinal break may be postulated. If no significant difference between the circumferential and longitudinal stresses is determined, then both types of breaks are considered.

5. For design purposes, a longitudinal break area is 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 or longitudinal break, the dynamic force of the jet discharge at the break location is 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 are taken into account, as applicable, in the reduction of the jet discharge.

3.6.2.6.2 Analytical Methods to Define Blowdown Forcing Functions and Response Models

3.6.2.6.2.1 Analytical Methods to Define Blowdown Forcing Functions

Rupture of a pressurized pipe causes the flow characteristics of the system to change, creating reaction forces that can dynamically excite the piping system. The reaction forces are a function of time and space and depend upon the fluid state within the pipe prior to the rupture, the break flow area, frictional losses, plant system characteristics, piping system, and other factors. The methods used to calculate the reaction forces are presented in the following sections.

3.6.2.6.2.1.1 Criteria

The following criteria are used for calculation of fluid blowdown forcing functions:

1. Circumferential breaks are assumed to result in pipe severance and separation amounting to at least a one pipe diameter lateral displacement of the ruptured piping sections, unless physically limited by piping restraints, structural members, or piping stiffness, as may be demonstrated by the inelastic pipe whip analysis discussed in Section 3.6.2.2.2.

2. For circumferential breaks, the dynamic force of the jet discharge at the break location is based on the effective cross-sectional flow area of the pipe and on a calculated fluid pressure as modified by an analytically or

experimentally determined thrust coefficient. Justifiable line restrictions, flow limiters, positive pump controlled flow, and the absence of energy reservoirs are taken into account, as applicable, in the reduction of the jet discharge.

3. A rise time not exceeding 1 millisecond is used for the initial pulse, unless longer crack propagation times or rupture opening times are substantiated by experimental data or analytical theory.

3.6.2.6.2.1.2 Forcing Functions

The predicted blowdown forces on pipes fed by a pressurized vessel can be described by transient (time dependent) and steady state forcing functions. The forcing functions used are based on methods described in Reference 3.6-4. These may be simply described as follows:

1. The transient forcing functions occur at points along the pipe from the propagation of waves (wave thrust) along the pipe and, at the broken end, from the reaction force due to the momentum of the fluid leaving the end of the pipe (blowdown thrust).

2. The waves cause various sections of the pipe to be loaded with time dependent forces. It 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 occur at the break end and at the pressure vessel end until a steady flow condition is established. Free space and vessel conditions are used as boundary conditions. The blowdown thrust causes a time dependent reaction force

perpendicular to the pipe break that reaches a final steady state value.

3. The initial blowdown force on the pipe is taken as the sum of the wave and blowdown thrusts and is equal to the vessel pressure (P_0) 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_0A.

4. Time histories of transient pressure, flow rate, and other thermodynamic properties of the fluid can be used to calculate the blowdown force on the pipe using the following equation:

$$F = \left[(P - P_a) + \frac{\rho u^2}{g_c}\right]A \qquad (3.6.2)$$

where in any consistent set of units:

F = blowdown force
P = pressure at exit plane
P = ambient pressure
u = velocity at exit plane
q = density at exit plane
A = area of break
g_C = Newton's gravitational constant

5. Following the transient period, a steady state period is assumed to exist. Steady state blowdown forces are calculated considering frictional effects. For the recirculation system, these effects reduce the blowdown forces from the theoretical maximum of 1.26 P_0A.

The

method of accounting for these effects is presented in Reference 3.6-4. For subcooled water, a reduction from the theoretical maximum of 2.0 P_0A is found through the use of Bernoulli's equation and standard equations, such as Darcy's equation, which account for friction.

3.6.2.6.2.2 Pipe Whip Dynamic Response Analyses

The prediction of time dependent and steady thrust reaction loads, caused by the blowdown of subcooled, saturated, and two phase fluid from a ruptured pipe, is used in the 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 Section 3.6.2.6.2.1.

The following criteria are used for performing the pipe whip dynamic response analyses:

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 components 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 contained within the broken loop is no longer considered part of the RCPB. Plastic deformation in the pipe is considered as a potential energy absorber. A limit of strain is imposed on the pipe material.

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 B&PV Code imposed limits for essential components under faulted loading. However, if these components are required for safe shutdown or if they serve a safety function to protect the structural integrity of an essential component, limits to meet the ASME B&PV Code requirements for faulted conditions and limits to ensure operability, if required, are met.

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

Shear deformation is also neglected. The pipe bending moment deflection (or rotation) relation used for these locations is obtained from a static, nonlinear, cantilever beam analysis. Using the moment rotation relation, nonlinear equations of motion of the pipe are formulated using an energy consideration, and the equations

are numerically integrated in small time steps to yield time-history information of the deformed pipe.

A comprehensive verification program has been performed to demonstrate the conservatisms inherent in the PDA pipe whip computer program and the analytical methods used. Part of this verification program included an independent analysis by Nuclear Services Corporation (NSC), under contract to General Electric Company (GE), 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 on Figure 3.6-36. 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 3.6-26.

A comparison of the NSC analysis with the PDA analysis, as presented in Table 3.6-27 and on Figure 3.6-37, shows that PDA predicts higher loads in 15 of the 18 restraints analyzed. This is due to the NSC model including energy absorbing effects in secondary pipe elements and structural members. However, PDA predicts higher restraint deflections in 50 percent of the restraints. The higher deflections predicted by NSC for the lower loads are caused by the linear approximation used for the force deflection curve rather than by differences in computer techniques. This comparison demonstrates that the simplified modeling system used in PDA is adequate for pipe

rupture loading, restraint performance, and pipe movement predictions within the meaningful design requirements for these low probability postulated accidents.

3.6.2.6.3 Dynamic Analysis Methods to Verify Integrity and Operability

This section provides the criteria and methods used to evaluate the effects of pipe displacements on safety-related structures, systems, and components following a postulated pipe rupture.

3.6.2.6.3.1 Pipe Whip Effects Following a Postulated Pipe Rupture.

Pipe whip (displacement) effects on safety-related structures, system, and components can be placed in two categories:

1. Pipe displacement effects on components, e.g., nozzles, valves, tees, etc, that are in the same piping run in which the break occurred.

2. Pipe whip or controlled displacements onto external components, e.g., building structure, other piping systems, cable trays and conduits, etc.

The criteria which are used for determining the effects of pipe displacements on in-line components are as follows:

1. Components such as vessel safe ends and valves that 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 the limits for essential components under faulted loading imposed by the ASME B&PV Code, Section III.

2. 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 ASME B&PV Code requirements for faulted conditions and limits to ensure operability, if required, are met.

The methods used to calculate the pipe whip loads on piping components in the same run as the postulated break are described in Section 3.6.2.6.2.2.

3.6.2.6.3.2 Loading Combinations and Design Criteria for Pipe Whip Restraints

Pipe whip restraints, as differentiated from piping supports, are designed to function and carry the load from an extremely low probability gross failure in a piping system carrying high energy fluid. The piping integrity does not depend on the pipe whip restraints for any loading combination. If 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 a once in a lifetime loading. The pipe break event is considered to be an accident condition for the ruptured pipe, its restraints, and structure to which the restraint is attached. The design and analysis of these components for this event are performed specifically as described in Section 3.6.2.6.2 and as described in the following paragraphs.

The pipe whip restraints used for the recirculation system consist of straps (either carbon steel ropes or stainless steel bars) attached to a steel frame. The analytical methods used in the design of these restraints have been improved by incorporation of the latest force deflection data available for wire rope and by

using GE's PDA code for the dynamic analysis. Load capacities for the restraint frames were developed by using a finite element structural analysis program code (SAP) and were confirmed by a test series using slowly applied loading methods to determine restraint load deflection data in the tangential direction (parallel to the restraint base). The results of this test program are presented in Reference 3.6-9.

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 pipe failure (gross loss of piping integrity) without allowing either damage to critical components or missile development.

3. The restraints shall provide minimum hindrance to inservice inspection of the process piping.

For the purposes of design, the pipe whip restraints are designed for the following dynamic loads:

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

2. Dynamic inertia loads of the moving pipe section that is accelerated by the blowdown thrust and 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 Section 3.6.2.6.2.2.

4. Since the pipe whip restraints are not contacted during normal plant operation, the postulated pipe rupture event is the only design loading condition.

For non-NSSS pipe whip restraints, an evaluation of the impact of seismic loads on pipe whip restraints has been performed. The results of this evaluation demonstrated that seismic stresses in pipe whip restraints are extremely low. Therefore, the pipe whip restraints will not fail during a seismic event and collapse onto safety-related components.

The postulated pipe rupture loads are the only design loading conditions for the NSSS pipe whip restraints because other loads are negligible in relationship to the pipe rupture loads.

The recirculation loop pipe whip restraints are composed of two parts: the straps and the restraint frame. Both parts of the restraining device function as load carrying members and will deflect under load. The load configurations for a restraint are shown on Figure 3.6-11. The components of the restraints are categorized as Type I and II, as described below:

1. Type I, radial load carrying members - These members consisting of cables or bars, will absorb energy when loaded in the direction perpendicular to the restraint base by elastic and plastic deformation as shown on Figure 3.6-11.

2. Type II, tangential load carry members - These members, consisting of restraint frames, will absorb energy when loaded in the direction parallel to the base by plastic deformation as shown in Figure 3.6-11.

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 - straps

   a. For carbon steel wire ropes, the maximum acceptable load is 90 percent of the load carrying capacity of the cable in the restraint configuration. This limit takes into consideration the efficiency reduction experienced when a cable is wrapped around a pipe.
      This means that the design load is limited to about 5 percent of the minimum certified load carrying capacity of the cable in tension.

   b. For stainless steel bars, the design limit base is 50 percent of the minimum uniform ultimate tensile elongation.

2. Type 2, Restraint frames - Design limits for the ASTM A36 restraint frames are as follows:

   a. 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 the 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 of 58,000 psi specified for ASTM A36 material.

b. 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. This corresponds to 50 percent of the minimum ultimate elongation of 20 percent as specified for ASTM A36 material. The ultimate deflection of the beam is based on a 20 percent ultimate elongation of the diagonal plate.

3.6.2.6.4 Material to be Submitted for the Operating License Review

3.6.2.6.4.1 Implementation of Criteria for Pipe Break Location and Orientation

The criteria for selection of postulated pipe breaks in the recirculation piping system are provided in Section 3.6.2.6.1. The postulated breaks and types, recirculation pipe breaks selected in accordance with these criteria are shown on Figure 3.6-12. Conformation with the criteria is demonstrated in Table 3.6-6.

3.6.2.6.4.2 Implementation of Special Protection Criteria

The location of pipe whip restraints provided for the recirculation piping systems are also shown in Figure 3.6-12. Using the analysis methods of Section 3.6.2.6.2.2, this system of restraints was found to prevent unrestrained pipe whip at the break locations, postulated in Section 3.6.2.6.1.

3.6.2.7 <u>Standard Review Plan Rule Review</u>

3.6.2.7.1 Acceptance Criterion II.1

Acceptance criterion II.1 of Standard Review Plan (SRP) Section 3.6.2 provides that postulated pipe rupture locations in containment should meet BTP MEB 3-1, which imposes new limits of 2.4 S for Class 1 pipe, in equations 10 and 12 of Paragraph NB-3653 of the ASME B&PV Code, Section III, for which pipe breaks must be postulated.

The HCGS NSSS design meets the intent of MEB 3-1, Revision 1, with the following clarifications:

1. GE meets the requirements of criterion B.1.d, B.3.a (2-5), and B.3.b, as described in Sections 3.6.2.6.1.5 and 3.6.2.6.2.1.1.

2. GE has taken the following positions on the remaining items of BTP MEB 3-1, Revision 1, criteria within GE scope:

   a. Criterion B.1.c(1) - GE uses criteria from SRP Section 3.6.2, Revision 0, which requires no break postulation if equation 10 is less than 3 S and the cumulative usage factor is less than 0.1. Section 3.6.2.6.1.4 discusses this criterion in detail.

The HCGS non-NSSS design meets the intent of MEB 3-1, Revision 1, with the following clarifications:

1. For Class 1 piping, when the stress and usage factor criteria in Section 3.6.2.1.1.3.b are not exceeded, the minimum of two intermediate breaks based on highest stress, as calculated by Equation 10 of Paragraph NB-3653, are not postulated unless the break location is in the proximity of a welded attachment.

2. For Class 2 and 3 piping, when the stress criteria of Section 3.6.2.1.1.4.b are not exceeded, the minimum of two intermediate breaks based on highest stress, as calculated by the sum of Equations 9 and 10 of Paragraph NC-3652, are not postulated unless the break location is in the proximity of a welded attachment.

In addition to limiting the stress and usage factor values for Class 1 piping and limiting the stress values for Class 2 and 3 piping, the following criteria are all required to be met when considering deletion of arbitrary intermediate breaks:

1. The piping systems are not susceptible to Intergranular Stress Corrosion Cracking (IGSCC) nor to unanticipated waterhammer/thermal transient events.

2. The piping system is included in the piping startup testing program for steady state vibrations.

3. Safety-related equipment in the vicinity of the deleted intermediate break remains environmentally qualified to the non-dynamic effects of the pipe break with the greatest consequences on the equipment.

4. The deleted intermediate break is not in the vicinity of a welded attachment.

3.6.2.7.2 Acceptance Criterion II.3

Acceptance criterion II.3 of SRP Section 3.6.2 provides criteria for initial conditions used in the dynamic analysis of postulated pipe break of the pressurized non-NSSS piping during operation at power. The initial condition to be used is the greater of the contained energy at hot standby or at 102 percent power.

On HCGS, the dynamic analysis of postulated pipe break is based on the initial condition of 100 percent power in the pressurized pipe. It is recognized that, for short periods of time, the pressure and enthalpy in some systems may be higher for some modes than for 100 percent power operation. From a safe and realistic protection point of view, 100 percent power represents the high energy condition of most likely occurrence, due to the relatively short time period of operation at the higher energy modes.

3.6.3 Definitions

Certain terms used in Sections 3.6.1 and 3.6.2 have specific meanings, as described below:

1. Essential systems and components - Systems and components required to shut down the reactor, maintain it in a safe shutdown mode, and mitigate the consequences of a postulated piping failure, without offsite power.

2. 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.

3. 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 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 qualifies as a high energy fluid system for only short periods and qualifies as a moderate-energy fluid system for the majority of the time is classified as a moderate energy fluid system, provided that the total time the system operates within high energy pressure/temperature conditions is less than either of the following:

a. 2 percent of the time that the system operates as a moderate energy fluid system

b. 1 percent of the normal operating life span of the plant.

4. Normal plant conditions - Plant operating conditions during reactor startup, operation at power, hot standby, or reactor cooldown to cold shutdown condition.

5. Upset plant conditions - Plant operating conditions during system transients, which may occur with moderate frequency during plant service life and are anticipated operational occurrences, but not during system testing.

6. S_h and S_A - Allowable stresses at maximum (hot) temperature and allowable stress range for thermal expansion, respectively, as defined in Article NC-3600 of the ASME B&PV Code, Section III.

7. S_m - Design stress intensity, as defined in Article NB-3600 of the ASME B&PV Code, Section III.

8. S_n - Primary plus secondary stress intensity range for normal and upset conditions, as defined in Paragraph NB-3653 of the ASME B&PV Code, Section III.

9. 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.

10. 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:

    a. The nominal size of the branch run is at least half that of the main run

    b. The intersection is not rigidly constrained to the building structure

    c. The branch run and main run are included together in the same piping stress analysis model.

    For piping in containment penetration areas, terminal ends are selected at points located immediately beyond the required valve operability restraints inside and outside primary containment.

    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.

11. Cumulative usage factor - The sum of all contributions to fatigue damage by every stress cycle during the life of the component. (ASME Section III, Subsection NB-3222.4).

3.6.4 References

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3.6-5 General Electric, PDA - "Pipe Dynamic Analysis Program for Pipe Rupture Movement" (proprietary filing), NEDE-10313.

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3.6-11 R.L. Mittl, PSE&G, to W. Butler, NRC, "Safety Evaluation Report Confirmatory Issue 1; Feedwater Isolation Check Valve Analysis", dated July 26, 1985 and September 9, 1985.

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3.6-13 NUREG-1048, HCGS SSER No. 3, Appendix 0, "NRC Safety Evaluation for the Elimination of Arbitrary Intermediate Pipe Breaks", October 1985.

3.6-14 C. McNeill, PSE&G, to E. Adensam, NRC, "Elimination of Additional Intermediate Pipe Break Locations", dated December 18, 1985.

3.6-15 NUREG-1048, HCGS SSER No. 5, Appendix O, "NRC Safety Evaluation for the Elimination of Arbitrary Intermediate Pipe Breaks", April 1986.

3.6-16 COMPARE-MOD 1 “A Computer Code for Transient Analysis of Volumes With Heat Sinks, Flowing Vents and Doors”, LA-7199-MS.

TABLE 3.6-1

HIGH ENERGY FLUID SYSTEM PIPING

Fluid System	Extent of High Energy Piping
Reactor recirculation	From reactor vessel suction nozzles to recirculation pumps to reactor vessel discharge nozzles, as shown on Figure 5.4-2
Main steam	From reactor vessel nozzles to main steam stop valves, as shown on Figures 5.1-3
Feedwater	From condensate filter/demineralizers through feedwater heaters and reactor feedwater pumps to reactor vessel nozzles, shown on Figures 10.4-4, 10.4-5, 10.4-6, and 5.1-3
Condensate	From condensate pumps through steam jet air ejector condensers, steam packing exhauster, and condensate filter/demineralizers, as shown on Figures 10.4-4 and 10.4-4
RWCU	From reactor recirculation loops through RWCU pumps, regenerative and nongenerative heat exchangers, and cleanup filter/demineralizers to feedwater lines, as shown on Figures 5.4-2, 5.4-17, 5.4-19, and 5.1-3
Reactor vessel drain	From reactor vessel bottom head nozzle to RWCU line inside primary containment, as shown on Figures 5.4-2 and 5.4-17

Fluid System	Extent of High Energy Piping
HPCI steam supply	From main steam line C to HPCI turbine steam supply valve HV-F001 as shown on Figure 6.3-1
RCIC steam supply	From main steam line A to RCIC turbine steam supply valve HV-F045, as shown on Figures 5.4-8 and 5.4-9
Main steam drain lines	From main steam lines inside drywell and from main steam lines outside drywell to the condenser, as shown on Figure 5.1-3
RPV head vent line	From reactor vessel top head nozzle to main steam line A, as shown on Figure 5.1-3
Standby liquid control injection	From core spray injection line A to inboard check valve, as shown on Figures 9.3-8 and 5.1-3
RHR shutdown cooling suction	From reactor recirculation loop to inboard containment isolation valve, as shown on Figure 5.4-13
RHR shutdown cooling return	From reactor recirculation loops to inboard check valves, as shown on Figure 5.4-13
LPCI injection	From reactor vessel nozzles to inboard check valves, as shown on Figure 5.4-13
Core spray injection	From reactor vessel nozzles to inboard check valves, as shown on Figure 6.3-7

Fluid System	Extent of High Energy Piping
CRD hydraulic	From CRD drive water pumps to master control station to HCUs, as shown on Figures 4.6-5 and 4.6-6
Auxiliary steam	From auxiliary boiler to various steam consuming components, as shown on Figure 9.5-30
Emergency diesel generator starting air line	From starting air skid to emergency diesel generator as shown on Figure 9.5-28

(Historical Information)

TABLE 3.6-2

FINAL MAIN STEAM SYSTEM PIPING STRESS LEVELS
AND PIPE BREAK DATA
(PORTION INSIDE PRIMARY CONTAINMENT)

Node Point(1)	Node Type(2)	Stress By EQ. 10 (ksi)	Cumulative Usage Factor	Pipe Break Stress Limit 2.4 S_m (ksi)	Break Type(3)	Basis for Break Selection(4)
Line A						
1	TTJ	27.78	0.010	42.5	C	TE
45	EL	56.58	0.010	42.5	C	TE
Line B						
1	TTJ	25.93	0.010	42.5	C	TE
49	EL	49.83	0.010	42.5	C	TE
Line C						
1	TTJ	26.88	0.010	42.5	C	TE
42	EL	56.15	0.010	42.5	C	TE
Line D						
1	TTJ	28.10	0.010	42.5	C	TE
39	EL	61.99	0.020	42.5	C	TE

THIS TABLE CONTAINS HISTORICAL DATA ONLY AND IS NO LONGER UPDATED.
FOR CURRENT INFORMATION, SEE THE LATEST APPLICABLE STRESS CALCULATIONS.

(Historical Information)

TABLE 3.6-2 (Cont)

(1) Locations of the nodes are shown in Figure 3.6-2

(2) Symbols used to denote the node type are as follows:

TTJ - Tapered transition joint

EL - Elbow

(3) Break types are indicated as follows:

C - Circumferential

(4) Symbols used to denote the basis for break selection are as follows:

TE - Terminal end

THIS TABLE CONTAINS HISTORICAL DATA ONLY AND IS NO LONGER UPDATED.
FOR CURRENT INFORMATION, SEE THE LATEST APPLICABLE STRESS CALCULATIONS.

(Historical Information)

TABLE 3.6-3

FINAL MAIN STEAM SYSTEM PIPING STRESS LEVELS AND PIPE BREAK DATA (PORTION OUTSIDE PRIMARY CONTAINMENT)

Node Point(1)	Node Type(2)	Total Stress EQ.9+EQ.10 ksi	Pipe Break Stress Limit $0.8(1.2S_h+S_A)$ ksi	Break Type(3)	Basis for Break Selection
45	BW	22.76	37.8	C	TE
215	BW	23.24	37.8	C	TE
385	BW	23.80	37.8	C	TE
565	BW	25.90	37.8	C	TE
75	EL	24.26	37.8	C	MBL
245	EL	25.09	37.8	C	MBL

(1) Locations of the nodes are shown in Figure 3.6-3

(2) Symbols used to denote the node type are as follows

EL - Elbow

BW - Buttweld

(3) Break types are indicated as follows

C - Circumferential

(4) Symbols used to denote the basis for break selection are as follows:

TE - Terminal end

MBL - Intermediate break locations selected to satisfy the requirements for a minimum number of break locations where such locations are in the proximity of welded attachments.

THIS TABLE CONTAINS HISTORICAL DATA ONLY AND IS NO LONGER UPDATED.
FOR CURRENT INFORMATION, SEE THE LATEST APPLICABLE STRESS CALCULATIONS.

TABLE 3.6-4

BLOWDOWN TIME HISTORIES FOR HIGH ENERGY PIPE BREAKS OUTSIDE PRIMARY CONTAINMENT

High Energy Line	Blowdown Time After Break (s)	Blowdown Mass Flow Rate (lbm/s)	Blowdown Enthalpy (Btu/lbm)	Isolation Valves	Isolation Valve Closure[1] Valve Closing Time(s)	Isolation Valve Closure[1] Signal Delay Time(s)	Isolation Valve Closure[1] Total Internal Time(s)
Main steam line	0.0	7812.0	1192.0	HVF022A,B,C,D	5.0	0.5	5.5
(auxiliary building section	0.16	7812.0	1192.0	HVF028A,B,C,D	5.0	0.5	5.5
of main steam tunnel)	0.161	18395.0	591.0				
	5.5	0.0	591.0				
Main steam line	0.0	13144.0	1192.0	HVF022A,B,C,D	5.0	0.5	5.5
(penetration champer	0.092	13144.0	1192.0	HVF028A,B,C,D	5.0	0.5	5.5
of the MST)	0.093	8539.0	1192.0				
	1.306	8539.0	1192.0				
	1.307	11205.0	962.0				
	2.297	11205.0	962.0				
	2.298	18533.0	591.0				
	5.50	18533.0	591.0				
	5.51	0.0	591.0				
RWCU 6" line (Cases 4&5)	0.0	4454.5	493.6	HVF001	30.0[2]		
(South pipe case)	0.182	4454.5	493.6	HVF004	30.0[2]		
FWTR	0.183	256.8	493.6	HVF039	check valve		
	30.0	256.8	493.6				
RWCU pump discharge	0.0	0.0	494.4	HVF001	35.0[2]	5.0	40.0
line at the check valve	0.001	297.91	494.4	HVF004	35.0[2]	5.0	40.0
(RWCU pump room)Case 1	0.005	795.05	494.0	HVF039	check value		
MPS w/FWTR	0.48	581.31	461.7				
	1.78	402.05	464.6				
	8.98	373.55	467.9				
	100.0	373.55	467.9				

TABLE 3.6-4 (Cont)

High Energy Line	Blowdown: Time After Break (s)	Blowdown: Mass Flow Rate (lbm/s)	Blowdown: Enthalpy (Btu/lbm)	Isolation Valves	Isolation Valve Closure[1]: Valve Closing Time(s)	Signal Delay Time(s)	Total Internal Time(s)
RWCU valve and pump room	0.0	0.0	420.3	HVF001	40.9[2]		
Case 7 (MPS w/FWTR)	0.005	416.59	420.3	HVF004	40.9[2]		
	0.025	1194.71	437.0	HVF039	check valve		
	0.480	659.65	439.9				
	1.380	404.46	423.6				
	3.980	408.41	379.8				
	10.230	265.61	413.3				
	100.000	265.61	413.3				
RWCU line at the heat	0.0	0.0	420.3	HVF001	35.0[2]	5.0	40.0
exchangers (RWCU heat exchanger room) Case 2 (MPS w/FWTR)	0.005	416.7	420.3	HVF004	35.0[2]	5.0	40.0
	0.025	1194.8	437.0	HVF039	check valve		
	0.480	659.7	439.9				
	1.380	404.5	423.6				
	3.980	408.5	379.8				
	10.230	265.6	413.3				
	100.000	265.6	413.3				
RWCU discharge line at the	0.0	821.7	88.3	HVF001	35.0[2]	5.0	40.0
inlet nozzle to the filter/ demineralizer vessel (RWCU filter/demineralizer room) Case 6	0.0001	1670.2	88.3	HVF004	35.0[2]	5.0	40.0
	0.005	572.2	97.0	HVF039	check valve		
	0.98	315.2	175.3				
	2.58	281.7	254.3				
	6.58	236.9	325.0				
	10.03	218.5	367.7				
	41.0	218.5	367.7				

TABLE 3.6-4 (Cont)

High Energy Line	Blowdown: Time After Break (s)	Blowdown: Mass Flow Rate (lbm/s)	Blowdown: Enthalpy (Btu/lbm)	Isolation Valves	Isolation Valve Closure[(1)]: Valve Closing Time(s)	Signal Delay Time(s)	Total Internal Time(s)
HPCI steam supply line (HPCI pump room)	0.0	1088.4	1192.2	HVF002	35.0		
	0.282	1088.4	1192.2	HVF003	35.0		
	0.283	414.0	1192.2				
	35.0	414.0	1192.2				
	35.1	0.0	1192.0				
HPCI steam supply line (HPCI pipe chase)	0.0	1088.36	1192.2	HVF002	35.0		
	0.0651	1088.36	1192.2	HVF003	35.0		
	0.0652	414.0	1192.2				
	35.0	414.0	1192.2				
	35.1	0.0	1192.2				
RCIC steam supply line (RCIC pump room)	0.0	164.68	1192.2	HVF007	11.0		
	11.0	164.68	1192.2	HVF008	11.0		
	11.1	0.0	1192.2				

(1) These values are the assumed valve closure times in the pressure temperature transient analysis. These analyses are insensitive to small variations in actual valve closing times.

(2) The required closing time for this valve, which was specified for containment isolation purposes, is actually 45 seconds. Since the peak pressure and temperature occur long before valve closure, the analysis is insensitive to the actual valve closing.

(3) All RWCU line breaks evaluated at the following conditions: Normal feedwater temperature, reduced feedwater temperature, Increased Core Flow (ICF), Minimum Pump Speed (MPS), and MPS with reduced feedwater temperature. The mass and energy release resulting in the peak break node pressure/temperature is provided.

TABLE 3.6-5

PRESSURE-TEMPERATURE TRANSIENT ANALYSIS RESULTS FOR HIGH-ENERGY PIPE BREAKS OUTSIDE PRIMARY CONTAINMENT

			Calculated Peak[5]		Initial Condition	
Case	Break	Room	Pressure (psig)	Temp. (°F)	Temp. (°F)	Relative Humid.[1,2] (%)
1	RWCU Pump Disch. Line Break	4405: RWCU Pump Rm.	1.7	217	104	50
		4403: RWCU Pump Rm.	1.7	217	104	50
2	RWCU Hx Line Break (4" Line Break)	4506: RWCU Hx Rm.	2.1	217	120	50
3	HPCI Steam Supply Line (Chase)	4102: Torus Rm.	1.5	302	90	90
		4327: HPCI Pipe Chase	2.0	302	75	90
		4329: North Pipe Chase	1.5	302	90	90
		4409: Steam Vent	1.5	302	90	90
4[1]	RWCU 6" Line Break	4319: RCIC Pipe Chase	1.7	302[3]	95	90
		4321: South Pipe Chase	1.6	218	95	90
		4402: South Pipe Chase	1.7	218	95	90
5[1]	RWCU 6" Line Break	4505: South Pipe Chase	1.8	218	95	90
6	RWCU F/D Line Break	4620: RWCU F/D Rm.	6.4	231	104	50
		4621: RWCU F/D Rm.	6.4	231	104	50
7	RWCU F/D Line Break	4502: RWCU Valve & Pump Rm.	1.9	218	115	50
		4503: RWCU Valve & Pump Rm.	1.9	218	120	50
8	HPCI Steam Supply Line (Pump Rm.)	4111: HPCI Pump Rm.	2.8	301	85	90
9	RCIC Steam Supply Line (Pump Rm.)	4110: RCIC Pump Rm.	2.8	301[4]	80	90
10	Main Steam Line Break in the Penetration Chamber of the Main Steam Tunnel	4316: MST Penetration Rm.	16.3	315	120	30
		4518: MST Unit Cooler Rm.	16.3	315	120	30
11	Main Steam Line Break in the Main Steam Tunnel	MST & Emergency Vent Stack	10.0	297	120	30

(1) The initial pressure in all of the rooms was 14.7 psia.
(2) Outside atmospheric conditions were assumed to be 14.7 psia, 70°F, and 90% relative humidity.
(3) Peak temperature is based on Case 3.
(4) Peak temperature is based on Case 8.
(5) Calculated peak pressure and temperature values bound those determined at 3952 MWth.

TABLE 3.6-6

FINAL RECIRCULATION SYSTEM PIPING STRESS RATIOS
AND PIPE BREAK DATA - LOOP A (4)

Break Ident (1)	Stress Ratio per ASME Equation EQ 10 2.4 S_m	EQ 12 2.4 S_m	EQ 13 2.4 S_m	Usage Factor	Break Type (2)	Break Basis Section No.
S1	0.452	0.072	0.414	0.00	CRCMF	3.6.2.6
F1	0.534	0.060	0.420	0.00	CRCMF	3.6.2.6
F2	0.664	0.187	0.425	0.00	CRCMF	3.6.2.6
F3	0.557	0.093	0.445	0.00	CRCMF	3.6.2.6
F4	0.603	0.104	0.440	0.00	CRCMF	3.6.2.6
F5	0.639	0.177	0.433	0.00	CRCMF	3.6.2.6
F6(3)	1.008	0.088	0.758	0.010	CRCMF	3.6.2.6
F7(3)	1.142	0.130	0.748	0.010	CRCMF	3.6.2.6
F8(3)	1.002	0.089	0.613	0.00	CRCMF	3.6.2.6
F9(3)	1.335	0.329	0.750	0.010	CRCMF	3.6.2.6
F10(3)	1.159	0.264	0.747	0.010	CRCMF	3.6.2.6

(1) Location at the nodes are shown in Figure 3.6-12.

(2) CRCMF = Circumferential

(3) Postulated Arbitrary Intermediate Breaks (AIB) are not required and stress ratios are not applicable.

(4) Reference PSE&G Calculation C-0142

NOTE: THIS TABLE CONTAINS HISTORICAL DATA ONLY AND IS NO LONGER UPDATED. FOR CURRENT INFORMATION, SEE THE LATEST APPLICABLE STRESS CALCULATION.

TABLE 3.6-6A

FINAL RECIRCULATION SYSTEM PIPING STRESS RATIOS
AND PIPE BREAK DATA - LOOP B (4)

Break Ident (1)	Stress Ratio per ASME Equation: EQ 10 2.4 S_m	EQ 12 2.4 S_m	EQ 13 2.4 S_m	Usage Factor	Break Type (2)	Break Basis Section No.
S1	0.450	0.046	0.414	0.00	CRCMF	3.6.2.6
F1	0.492	0.062	0.427	0.00	CRCMF	3.6.2.6
F2	0.687	0.215	0.422	0.00	CRCMF	3.6.2.6
F3	0.565	0.092	0.429	0.00	CRCMF	3.6.2.6
F4	0.648	0.169	0.427	0.00	CRCMF	3.6.2.6
F5	0.655	0.203	0.425	0.00	CRCMF	3.6.2.6
F6(3)	0.912	0.080	0.753	0.01	CRCMF	3.6.2.6
F7(3)	1.060	0.152	0.753	0.01	CRCMF	3.6.2.6
F8(3)	1.045	0.164	0.617	0.00	CRCMF	3.6.2.6
F9(3)	1.325	0.377	0.757	0.01	CRCMF	3.6.2.6
F10(3)	1.153	0.295	0.742	0.01	CRCMF	3.6.2.6

(1) Location at the nodes are shown in Figure 3.6-12.

(2) CRCMF = Circumferential.

(3) Postulated Arbitrary Intermediate Breaks (AIB) are not required and stress ratios are not applicable.

(4) Reference PSE&G Calculation C-0142.

NOTE: THIS TABLE CONTAINS HISTORICAL DATA ONLY AND IS NO LONGER UPDATED. FOR CURRENT INFORMATION, SEE THE LATEST APPLICABLE STRESS CALCULATION.

TABLE 3.6-7

RECIRCULATION SYSTEM BLOWDOWN TIME HISTORY

Time, s	Rate, lbm/s	Enthalpy, Btu/lbm
0	0	544.5
0.00255	1210	544.5
0.00496	3600	544.5
0.00804	8410	544.5
0.00924	10,810	544.5
0.01180	16,400	544.5
0.01580	24,500	544.5
0.01880	30,190	544.5
0.01910	30,780	544.5
0.01911	11,660	544.5
0.01980	12,140	544.5
0.02580	16,340	544.5
0.03380	21,860	544.5
0.04180	26,880	544.5
0.05480	31,300	544.5
0.05890	32,060	544.5
5.00000	32,060	544.5

(Historical Information)

TABLE 3.6-8

FINAL FEEDWATER SYSTEM PIPING STRESS LEVELS

AND PIPE BREAK DATA

(PORTION INSIDE PRIMARY CONTAINMENT)

AE-035

Node Point[(1)]	Node Type[(2)]	Stress By EQ. 10 (ksi)	Cumulative Usage Factor	Pipe Break Stress Limit 2.4 S_m (ksi)	Break Type[(3)]	Basis for Break Selection[(4)]
200	TTJ	67.60	0.6017	47.34	C	TE
315	TTJ	65.93	0.5445	47.34	C	TE
265	TTJ	67.69	0.5376	47.34	C	TE
130	TEE	80.14	0.4595	47.34	C&L	SFL
95	TEE	114.42	0.4304	47.34	C&L	SFL
70	TTJ	57.50	0.2105	47.34	C&L	SFL
60	TTJ	57.30	0.2107	47.34	C&L	SFL
25	TTJ	61.66	0.2498	47.34	C&L	SFL
15	TTJ	57.62	0.2144	47.34	C	TE
178	LUG	17.78	0.3399	47.34	S	SFL
288	LUG	21.57	0.3438	47.34	S	SFL
250	LUG	14.52	0.3252	47.34	S	SFL
225	LUG	18.47	0.1514	47.34	S	SFL
108	LUG	22.39	0.8374	47.34	S	SFL

(1) Locations of the nodes are shown in Figure 3.6-13

(2) Symbols used to denote the node type are as follows:
TTJ - Tapered transition joint
BRA - Branch Connection
TEE - Tee
LUG - Shear Lug

(3) Break types are indicated as follows:
C - Circumferential
L - Longitudinal
S - Slot break at welded attachment

(4) Symbols used to denote the basis for break selection are as follows:
TE - Terminal end
SFL - Stress and fatigue limits established in Section 3.6.2.1.1.3 are not met.

THIS TABLE CONTAINS HISTORICAL DATA ONLY AND IS NO LONGER UPDATED.
FOR CURRENT INFORMATION, SEE THE LATEST APPLICABLE STRESS CALCULATIONS.

(Historical Information)

TABLE 3.6-8A

FINAL FEEDWATER SYSTEM PIPING STRESS LEVELS
AND PIPE BREAK DATA
(PORTION INSIDE PRIMARY CONTAINMENT)
AE-036

Node Point[1]	Node Type[2]	Stress By EQ. 10 (ksi)	Cumulative Usage Factor	Pipe Break Stress Limit 2.4 S_m (ksi)	Break Type[3]	Basis for Break Selection[4]
200	TTJ	68.87	0.6099	47.34	C	TE
315	TTJ	64.44	0.5359	47.34	C	TE
265	TTJ	69.02	0.5463	47.34	C	TE
130	TEE	86.37	0.6775	47.34	C&L	SFL
95	TEE	93.96	0.8276	47.34	C&L	SFL
70	TTJ	57.25	0.2101	47.34	C&L	SFL
60	TTJ	57.11	0.2105	47.34	C&L	SFL
25	TTJ	61.65	0.2497	47.34	C&L	SFL
15	TTJ	57.65	0.2145	47.34	C	TE
180	BRA	53.19	0.2765	47.34	S	SFL
178	LUG	16.90	0.3388	47.34	S	SFL
288	LUG	21.87	0.3434	47.34	S	SFL
250	LUG	15.15	0.3260	47.34	S	SFL
225	LUG	19.36	0.1518	47.34	S	SFL
108	LUG	24.16	0.8539	47.34	S	SFL

(1) Locations of the nodes are shown in Figure 3.6-13

(2) Symbols used to denote the node type are as follows:
TTJ - Tapered transition joint
BRA - Branch Connection
TEE - Tee
LUG - Shear Lug

(3) Break types are indicated as follows:
C - Circumferential
L - Longitudinal
S - Slot break at welded attachment

(4) Symbols used to denote the basis for break selection are as follows:
TE - Terminal end
SFL - Stress and fatigue limits established in Section 3.6.2.1.1.3 are not met.

THIS TABLE CONTAINS HISTORICAL DATA ONLY AND IS NO LONGER UPDATED.
FOR CURRENT INFORMATION, SEE THE LATEST APPLICABLE STRESS CALCULATIONS.

(Historical Information)

TABLE 3.6-9

FINAL FEEDWATER SYSTEM PIPING STRESS LEVELS AND
PIPE BREAK DATA
(PORTION OUTSIDE PRIMARY CONTAINMENT)

Node Point[1]	Node Type[2]	Total Stress (ksi)	Pipe Break Stress Limit $0.8(1.2\ S_h+S_A)$ (ksi)	Break Type[3]	Basis for Break Selection[4]
Feedwater lines:					
70	BW	11.39	32.40	C	TE
630	BW	11.36	32.40	C	TE
HPCI pump discharge to FW:					
A05	BW	19.94	32.40	C	TE
A10	BW	19.68	32.40	C	TE
RCIC pump discharge to FW:					
60	BW	9.30	32.40	C	TE
958	BW	23.53	32.40	C	TE
RWCU discharge to FW:					
40	BW	9.82	32.40	C	TE
665	BW	9.95	32.40	C	TE

(1) Locations of the nodes are shown in Figure 3.6-14

(2) Symbols used to denote the node type are as follows:
BW - Butt weld

(3) Break types are indicated as follows:
C - Circumferential

(4) Symbols used to denote the basis for break selection are as follows:
TE - Terminal end

THIS TABLE CONTAINS HISTORICAL DATA ONLY AND IS NO LONGER UPDATED.
FOR CURRENT INFORMATION, SEE THE LATEST APPLICABLE STRESS CALCULATIONS.

TABLE 3.6-10

FINAL RWCU SYSTEM PIPING STRESS LEVELS AND PIPE BREAK DATA
(PORTION INSIDE PRIMARY CONTAINMENT)

Node(5) Point(1)	Node Type(2)	Stress By EQ 10 (ksi)	Cumulative Usage Factor	Pipe Break Stress Limit 2.4 Sm (ksi)	Break Type(3)	Basis for Break Selection(4)
90	BW	15.65	0.0002	43.60	C	TE
101	BW	67.941	0.8869	43.60	C&L	SFL
480	BW	12.86	0.0000	43.60	C	TE
518	BW	11.64	0.0000	43.60	C	TE
760	RED	69.73	0.1853	43.60	C	SFL
799	BW	15.01	0.0003	43.60	C	TE
108	TTJ	76.288	0.5386	43.60	C&L	SFL
109	DSW	51.813	0.1346	43.60	C&L	SFL
570	SW	61.26	0.6088	43.60	C	SFL
575	SW	61.58	0.6386	43.60	C	SFL
819	SW	25.59	0.0056	43.60	C	TE
705	TTJ	42.66	0.0059	34.64	C	TE
710	TTJ	69.42	0.3583	34.64	C&L	SFL
910	RED	48.52	0.0154	43.60	C	SFL
920	SW	9.14	0.0003	43.60	C	TE
855	BW	14.68	0.0001	43.60	C	TE
902	TTJ	43.17	0.0075	34.64	C	TE
905	TTJ	69.97	0.4372	34.64	C&L	SFL
984	RED	48.52	0.0155	43.60	C	SFL
988	SW	9.19	0.0003	43.60	C	TE
968	BW	10.25	0.0000	43.60	C	TE

---

(1) Locations of the nodes are shown in Figure 3.6-15

(2) Symbols used to denote the node type are as follows:

TTJ - Tapered transition joint
BW - Butt weld
RED - Reducer
SW - Socket weld

(3) Break types are indicated as follows:

C - Circumferential
L - Longitudinal

(4) Symbols used to denote the basis for break selection are as follows:

TE - Terminal end
SFL - Stress and fatigue limits established in Section 3.6.2.1.1.3 are not met.

(5) Node points 101, 108, and 109 are branch connections on flow element No35 which are not within the snubber reduction program scope; therefore, values listed are not revised to reflect the snubber reduction configuration.

TABLE 3.6-11

FINAL RWCU SYSTEM PIPING STRESS LEVELS AND PIPE BREAK DATA
(PORTION OUTSIDE PRIMARY CONTAINMENT)

Node Point(1)	Node Type(2)	Total Stress EQ.9+EQ.10 (ksi)	Pipe Break Stress Limit $0.8(1.2\ S_h+S_A)$ (ksi)	Break Type(3)	Basis for Break Selection(4)
E	ANCH	38.83	32.4	C	TE
D	ANCH	18.95	32.4	C	TE
250	FL	26.76	32.4	C	TE
370	FL	12.69	32.4	C	TE
255	FL	12.66	32.4	C	TE
380	FL	17.43	32.4	C	TE
B	ANCH	12.32	32.4	C	TE
5	BW	15.26	32.4	C	TE
640	BW	16.58	32.4	C	TE
50	BW	13.50	32.4	C	TE
850	ANCH	16.76	32.4	C	TE

(1) Locations of the nodes are shown on Figure 3.6-16.

(2) Symbols used to denote the node type are as follows:
- FL - Flange
- BW - Butt weld
- ANCH - Anchor

(3) Break type is indicated as follows:
- C - Circumferential

(4) Symbol used to denote the basis for break selection is as follows:
- TE - Terminal end

(Historical Information)

TABLE 3.6-12

FINAL HPCI SYSTEM PIPING STRESS LEVELS AND PIPE BREAK DATA
(PORTION INSIDE PRIMARY CONTAINMENT)

Node Point[1]	Node Type[2]	Stress By EQ. 10 (ksi)	Cumulative Usage Factor	Pipe Break Stress Limit 2.4 S_m (ksi)	Break Type[3]	Basis for Break Selection[4]
402	TTJ	31.3	0.0026	42.48	C	TE
420	BW	33.2	0.0015	42.48	C	TE

(1) Locations of the nodes are shown in Figure 3.6-18

(2) Symbols used to denote the node type are as follows:
TTJ - Tapered transition joint
BW - Buttweld

(3) Break types are indicated as follows:
C - Circumferential

(4) Symbols used to denote the basis for break selection are as follows:
TE - Terminal end

THIS TABLE CONTAINS HISTORICAL DATA ONLY AND IS NO LONGER UPDATED.
FOR CURRENT INFORMATION, SEE THE LATEST APPLICABLE STRESS CALCULATIONS.

TABLE 3.6-13

FINAL HPCI SYSTEM PIPING STRESS LEVELS AND PIPE BREAK DATA
(PORTION OUTSIDE PRIMARY CONTAINMENT)

Node Point(1)	Node Type(2)	Total Stress (ksi)	Pipe Break Stress Limit $0.8(1.2\ S_h+S_A)$ (ksi)	Break Type(3)	Basis for Break Selection(4)
Pump Discharge					
(see Feedwater and Core Spray)					
Turbine Steam Supply					
79	BW	23.20	32.40	C	TE
120	BW	10.55	32.40	C	TE
C	ANCH	22.86	32.40	C	TE
182	BW	16.41	32.40	C	TE
110	BW	8.604	32.40	C	TE

(1) Locations of the nodes are shown in Figure 3.6-19

(2) Symbols used to denote the node type are as follows:

BW - Butt weld

ANCH - Anchor

(3) Break types are indicated as follows:

C - Circumferential

(4) Symbols used to denote the basis for break selection are as follows:

TE - Terminal end

(Historical Information)

TABLE 3.6-14

FINAL RCIC SYSTEM PIPING STRESS LEVELS AND PIPE BREAK DATA
(PORTION INSIDE PRIMARY CONTAINMENT)

Node Point(1)	Node Type(2)	Stress By EQ. 10 (ksi)	Cumulative Usage Factor	Pipe Break Stress Limit $2.4\ S_m$ (ksi)	Break Type(3)	Basis for Break Selection(4)
405	TTJ	53.666	0.0028	42.14	C	TE
420	DMW	28.934	0.0023	33.72	C	MBL
455	BW	49.756	0.0028	42.14	C	TE

(1) Locations of the nodes are shown in Figure 3.6-22

(2) Symbols used to denote the node type are as follows:
TTJ - Tapered transition joint
BW - Butt weld
DMW - Dissimilar Metal Weld

(3) Break types are indicated as follows:
C - Circumferential

(4) Symbols used to denote the basis for break selection are as follows:
TE - Terminal end
MBL - Intermediate break locations selected to satisfy the requirements for a minimum number of break locations where such locations are in the proximity of welded attachments.

THIS TABLE CONTAINS HISTORICAL DATA ONLY AND IS NO LONGER UPDATED.
FOR CURRENT INFORMATION, SEE THE LATEST APPLICABLE STRESS CALCULATIONS.

TABLE 3.6-15

FINAL RCIC SYSTEM PIPING STRESS LEVELS AND PIPE BREAK DATA
(PORTION OUTSIDE PRIMARY CONTAINMENT)

Node Point(1)	Node Type(2)	Total Stress EQ.9+EQ.10 (ksi)	Pipe Break Stress Limit $0.8(1.2\ S_h+S_A)$ (ksi)	Type Break(3)	Basis for Break Selection(4)
A	ANCH	41.82	32.4	C	TE
85	BW	11.13	32.4	C	TE
44	BW	7.61	32.4	C	TE

(1) Locations of the nodes are shown in Figure 3.6-23

(2) Symbols used to denote the node type are as follows:
BW - Butt weld
ANCH- Anchor

(3) Break types are indicated as follows:
C - Circumferential

(4) Symbols used to denote the basis for break selection are as follows:
TE - Terminal end

TABLE 3.6-16

MAIN STEAM DRAIN PIPING STRESS LEVELS AND PIPE BREAK DATA
(PORTION INSIDE PRIMARY CONTAINMENT)

Breaks are postulated at every fitting and change of direction. Refer to Figure 3.6-26

TABLE 3.6-17

FINAL MAIN STEAM DRAIN PIPING STRESS LEVELS AND
PIPE BREAK DATA
(PORTION OUTSIDE PRIMARY CONTAINMENT)

Node Point (1)	Node Type (2)	Total Stress (ksi)	Pipe Break Stress Limit 0.8(1.2S +S) (ksi)	Break Type (3)	Basis for Break Selection (4)
120B	TE	13.33	32.40	C	TE
765	BW	17.35	32.40	C	TE
680	BW	20.43	32.40	C	TE
610	BW	22.01	32.40	C	TE
540	BW	23.92	32.40	C	TE
274	BW	20.72	32.40	C	TE
75	BW	9.05	32.40	C	TE

---

(1) Locations of the nodes are shown in Figure 3.6-27.

(2) Symbols used to denote the node type are as follows:

EL - Elbow

TEE - Tee

BW - Butt weld

(3) Break types are indicated as follows:

C - Circumferential

(4) Symbols used to denote the basis for break selection are as follows:

TE - Terminal end

SFL - Stress and fatigue limits established in Section 3.6.2.1.1.3 are not met.

TABLE 3.6-18

RPV HEAD VENT PIPING STRESS LEVELS AND PIPE BREAK DATA

Breaks are postulated at every fitting and change of direction. Refer to Figure 3.6-28.

TABLE 3.6-19

STANDBY LIQUID CONTROL INJECTION PIPING
STRESS LEVELS AND PIPE BREAK DATA

Breaks are postulated at every fitting and change of direction. Refer to Figure 3.6-29.

TABLE 3.6-20

FINAL RHR SHUTDOWN COOLING SUCTION PIPING STRESS LEVELS AND PIPE BREAK DATA

Node Point(1)	Node Type(2)	Stress By EQ. 10 (ksi)	Cumulative Usage Factor	Pipe Break Stress Limit 2.4 S_m (ksi)	Break Type(3)	Basis for Break Selection(4)
500	TTJ	42.5	0.013	34.05	C	TE
530	TTJ	24.2	0.0303	42.375	C	TE

(1) Locations of the nodes are shown in Figure 3.6-30

(2) Symbols used to denote the node type are as follows:
TTJ - Tapered transition joint

(3) Break types are indicated as follows:
C - Circumferential

(4) Symbols used to denote the basis for break selection are as follows:
TE - Terminal end

TABLE 3.6-21

FINAL RHR SHUTDOWN COOLING RETURN PIPING STRESS LEVELS
AND PIPE BREAK DATA

Node Point(1)	Node Type(2)	Stress By EQ. 10 (ksi)	Cumulative Usage Factor	Pipe Break Stress Limit 2.4 S_m (ksi)	Break Type(3)	Basis for Break Selection(4)
LOOP A 12"-CCA-116(SS) 12"-DLA-069(CSS)						
600	TTJ	17.1	0.00	34.05	C	TE
622	TTJ	21.7	0.0088	42.375	C	TE
LOOP B 12"-CCA-115(SS) 12"-DLA-021(CSS)						
600	TTJ	17.4	0.00	34.05	C	TE
625	TTJ	23.4	0.0092	42.375	C	TE

(1) Locations of the nodes are shown in Figure 3.6-31

(2) Symbols used to denote the node type are as follows:
TTJ - Tapered transition joint

(3) Break types are indicated as follows:
C - Circumferential

(4) Symbols used to denote the basis for break selection are as follows:
TE - Terminal end

TABLE 3.6-22

FINAL LPCI INJECTION PIPING STRESS LEVELS
AND PIPE BREAK DATA

Node Point(1)	Node Type(2)	Stress By EQ. 10 (ksi)	Cumulative Usage Factor	Pipe Break Stress Limit 2.4 S_m (ksi)	Break Type(3)	Basis for Break Selection(4)
Line 12"-DLA-014						
80	TTJ	64.01	0.0335	42.48	C	TE
25	TTJ	36.81	0.0027	42.48	C	TE
Line 12"-DLA-015						
180	TTJ	64.04	0.0334	42.48	C	TE
125	TTJ	31.36	0.0014	42.48	C	TE
Line 12"-DLA-055						
495	TTJ	67.05	0.044	42.48	C	TE
425	TTJ	28.96	0.0005	42.48	C	TE
Line 12"-DLA-056						
395	TTJ	67.98	0.0482	42.48	C	TE
335	EL	50.02	0.0022	42.48	C	MBL
325	TTJ	19.34	0.0001	42.48	C	TE
393	EL	76.55	0.0465	42.48	C	MBL

---

(1) Locations of the nodes are shown in Figure 3.6-32

(2) Symbols used to denote the node type are as follows:

TTJ - Tapered transition joint

EL - Elbow

(3) Break types are indicated as follows:

C - Circumferential

(4) Symbols used to denote the basis for break selection are as follows:

TE - Terminal end

MBL - Intermediate break locations selected to satisfy the requirements for a minimum number of break locations where such locations are in the proximity of welded attachments.

TABLE 3.6-23

FINAL CORE SPRAY INJECTION PIPING STRESS LEVELS
AND PIPE BREAK DATA

Node Point(1)	Node Type(2)	Stress By EQ. 10 (ksi)	Cumulative Usage Factor	Pipe Break Stress Limit 2.4 S_m (ksi)	Break Type(3)	Basis for Break Selection(4)
Line 12" - DLA-001						
150	RED	53.366	0.0327	42.48	C	TE
35	TTJ	25.073	0.0004	42.48	C	TE
Line 12" - DLA-023						
140	RED	48.745	0.034	42.48	C	TE
35	TTJ	24.409	0.002	42.48	C	TE

(1) Locations of the nodes are shown in Figure 3.6-33

(2) Symbols used to denote the node type are as follows:
TTJ - Tapered transition joint
RED - Reducer

(3) Break types are indicated as follows:
C - Circumferential

(4) Symbols used to denote the basis for break selection are as follows:
TE - Terminal end

TABLE 3.6-24

REACTOR VESSEL DRAIN PIPING STRESS LEVELS
AND PIPE BREAK DATA

Shown as part of RWCU.

Refer to Table 3.6-10 and Figure 3.6-15.

TABLE 3.6-25

THIS TABLE INTENTIONALLY DELETED

TABLE 3.6-26

PDA VERIFICATION RESTRAINT DATA[1]

Pipe Size in.	Rest Load Direction	C_2[2]	N[2]	Limit Restraint[2]	Initial Clearance in.	Effective Clearance in.	Total Clearance in.
12	0°	27,733	0.24	6.129	4	1.941	5.941
12	90°	14,795	0.401	9.063	4	12.247	16.247
16	0°	109,265	0.24	6.728	4	1.934	5.934
16	90°	62,599	0.377	8.978	4	12.187	16.187
24	0°	102,228	0.24	8.222	4	1.984	5.984
24	90°	55,531	0.375	11.972	4	13.685	17.685
24	38°[3]	109,888	0.24	5.588	4	5.698	9.698
24	52°[3]	109,835	0.24	5.473	4	8.462	12.462

(1) The restraint data listed applies to one bar of a restraint.
(2) $F = C_2\ (\Delta\text{restraint})^N$
where F is the resistance force for one bar of a restraint and
where $(\Delta\text{restraint}) = (\delta\text{pipe})$ - total clearance
(3) Applies to restraint RCR 3 only. See Figure 3.6-38.

TABLE 3.6-27

COMPARISON OF PDA AND NSC CODE

Break Designation(1)	Restraint Designation(1)	No. of Bars PDA	No. of Bars NSC	Load (Kips) PDA	Load (Kips) NSC	Restraint Deflection (in.) PDA	Restraint Deflection (in.) NSC	Fraction of Design Restraint Deflection (%) PDA	Fraction of Design Restraint Deflection (%) NSC	Pipe Deflection (in.) PDA	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	85.4	88.1	41.48	43.0
$RC4_{LL}$	RHR3	5	5	846.0	838.4	8.2	8.05	92.95	97.98	18.87	16.43
$RC4_{LL}$	RCR3	8	8	1319.0	1073.9	5.43	4.2	99.23	76.85	23.28	17.25
$RC4C_V$	RCR3	8	8	1260.7	1275.0	4.49	5.58	80.37	99.89	22.56	18.73
$RC6A_V$	RCR3	8	8	928.5	722.5	1.22	1.77	22.46	31.7	23.68	95.39
$RC7_J$	RCR7	6	6	953.3	801.6	6.28	5.76	76.4	70.12	16.46	21.63
$RC8_{LL}$	RCR6 RCR7	4 6	4 6	599.0 895.0	N/A N/A	8.28 8.16	N/A N/A	64.2 68.2	N/A N/A	26.76 29.316	N/A N/A
$RC9C_V$	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
RC12	RCR9	6	6	N/A	832.9	N/A	6.3	N/A	76.9	N/A	15.7
RC13	RCR10	4	4	668.4	478.4	5.87	3.66	93.5	58.39	13.37	10.44
RC16	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	22.12	23.56

(1) Break designations and restraints designations are shown on Figure 3.6-38.

TABLE 3.6-28

MODERATE ENERGY FLUID SYSTEM PIPING

Fluid System	Pressure (psig)	Temperature (°F)
Demineralized Water	100	70
Condensate & Refueling Water Storage & Refueling	190	108
Station Service Water	65	89 (2) (3)
Safety Auxiliaries Cooling	110	95 (4)
Reactor Auxiliaries Cooling	110	95
Fire Protection	125	70
CRD Hydraulics (Pump Suction Only)	10	100
Standby Liquid Control (Pump Suction Only)	10	80
Reactor Core Isolation Cooling (excluding the steam supply line)[(1)]	50	120
Residual Heat Removal	125	120
Core Spray	125	120
Fuel Pool Cooling & Torus Water Cleanup	135	90
High Pressure Coolant Injection	110	140
Chilled Water System - Reactor Building and Drywell, and Auxiliary Building Control Area	65	60

---

(1) Evaluated as a moderate energy line.

(2) 89°F maximum UHS Temperature permitted by Plant Technical Specifications

(3) 95°F System Design (Supply)
120°F System Design (Discharge)

(4) 100°F maximum post-accident

TABLE 3.6-29

FINAL STARTING AIR PIPING STRESS LEVELS AND PIPE BREAK DATA

Node Point(1)	Node Type(2)	Total Stress EQ.9+EQ.10 ksi	Pipe Break Stress Limit $0.8(1.2S_h+S_A)$ ksi	Break Type(3)	Basis for Break Selection
C50	SW	14.3	33.91	C	TE
320	SW	13.7	33.91	C	TE
450	SW	8.2	33.91	C	TE
D50	SW	16.0	33.91	C	TE

(1) Locations of the nodes are shown in Figure 3.6-3

(2) Symbols used to denote the node type are as follows:

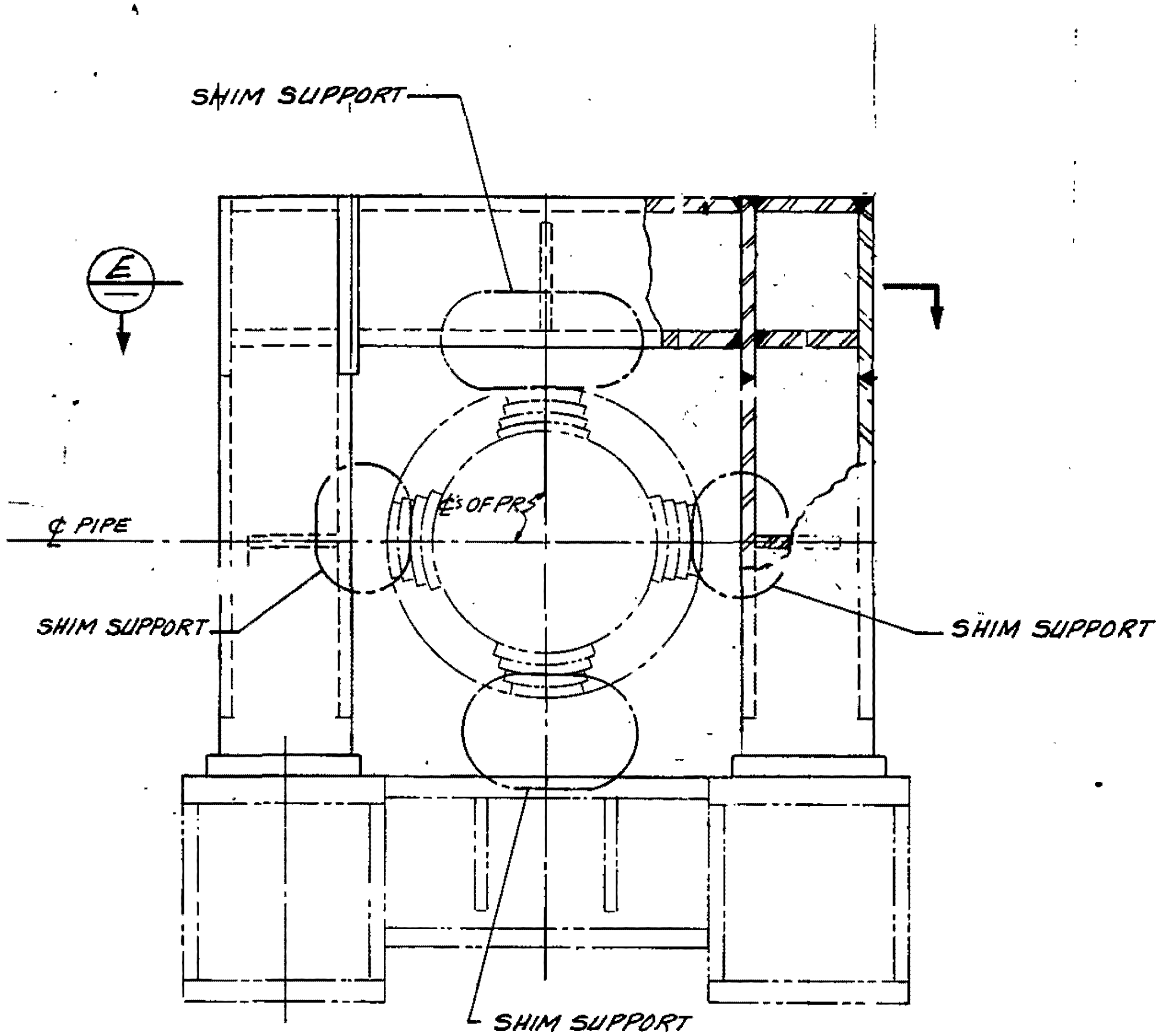
SW - Socket weld

(3) Break types are indicated as follows:

C - Circumferential

(4) Symbols used to denote the basis for break selection are as follows:

TE - Terminal end



PIPE WHIP RESTRAINTS TYPE I

ELEVATION

FRAME LOCATED
IN HORIZONTAL PLANE OR
VERTICAL PLANE SIMILAR

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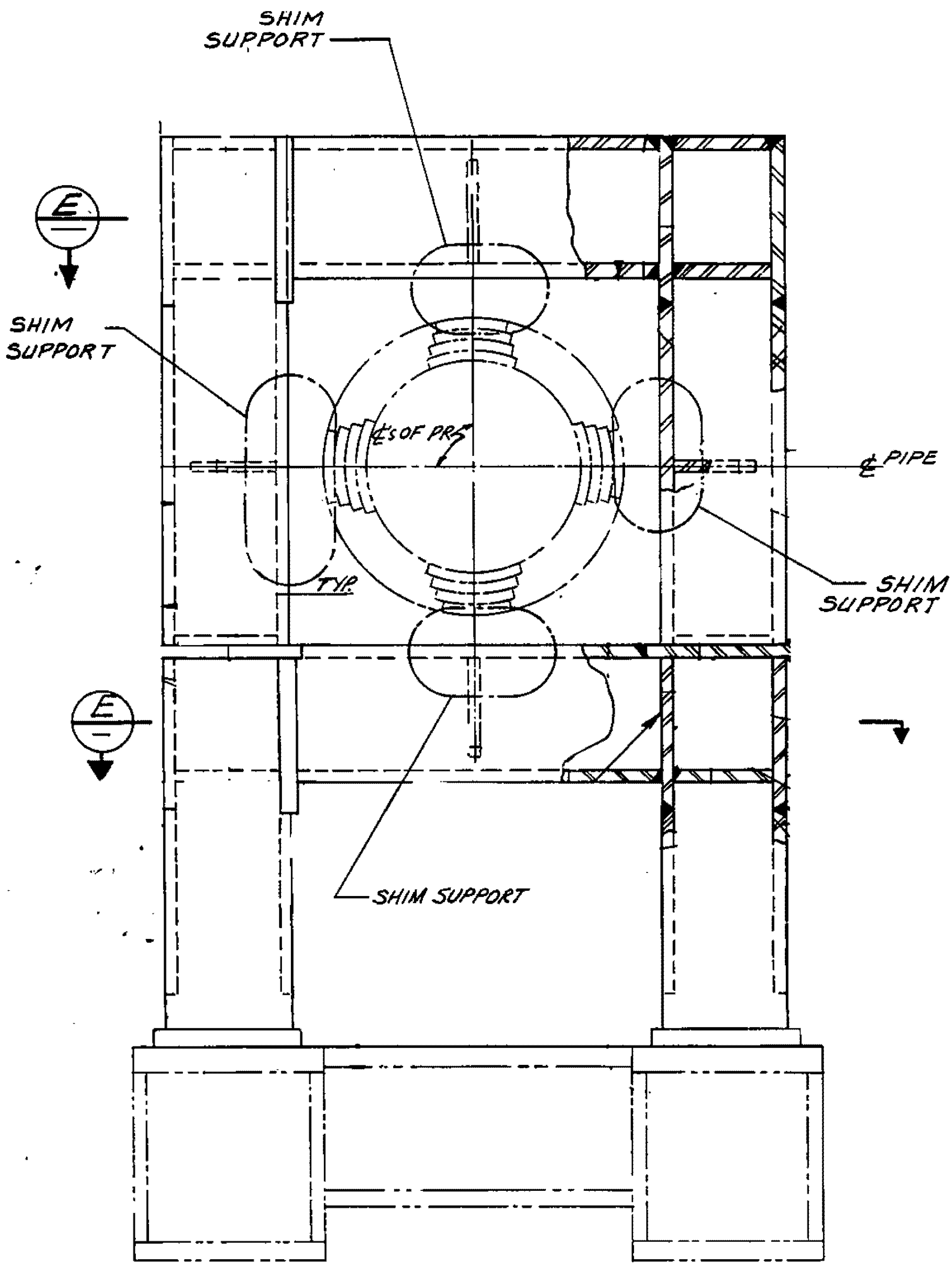
TYPICAL PIPE WHIP
RESTRAINT DETAIL

UPDATED FSAR

Sheet 1 of 7
FIGURE 3.6-1



PIPE WHIP RESTRAINT TYPE II

ELEVATION

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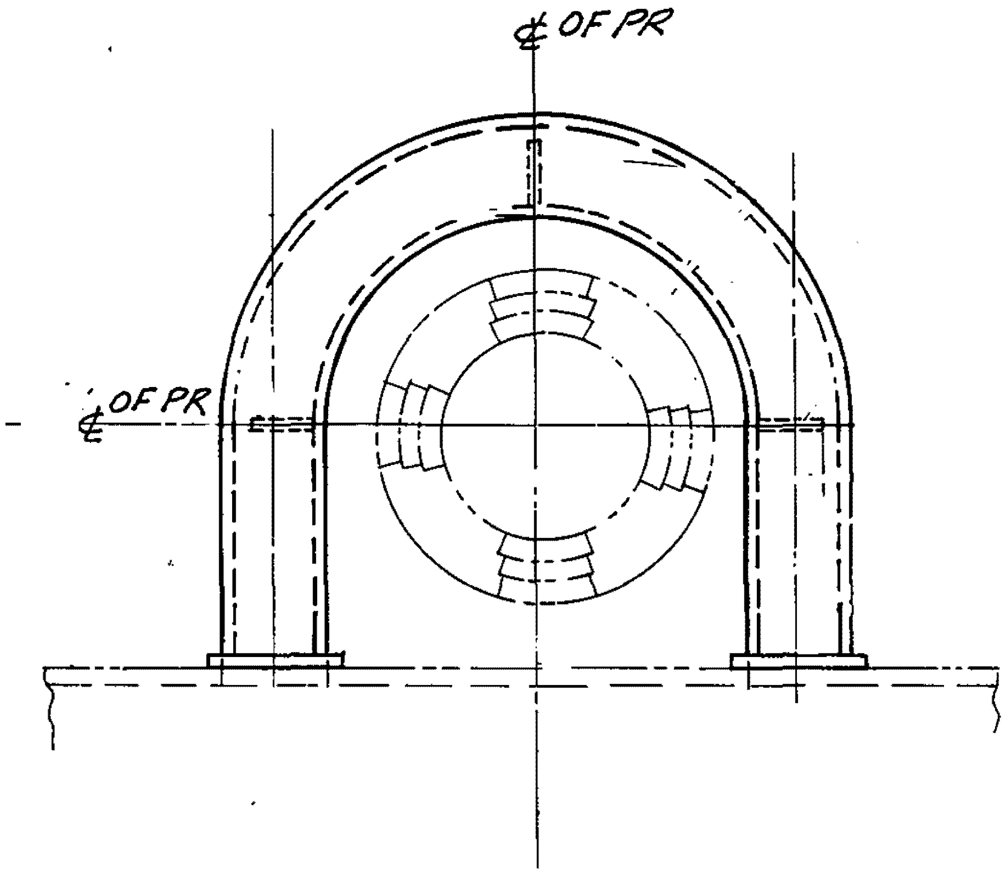
TYPICAL PIPE WHIP
RESTRAINT DETAIL

UPDATED FSAR

Sheet 2 of 7
FIGURE 3.6-1



PIPE WHIP RESTRAINT TYPE III

ELEVATION

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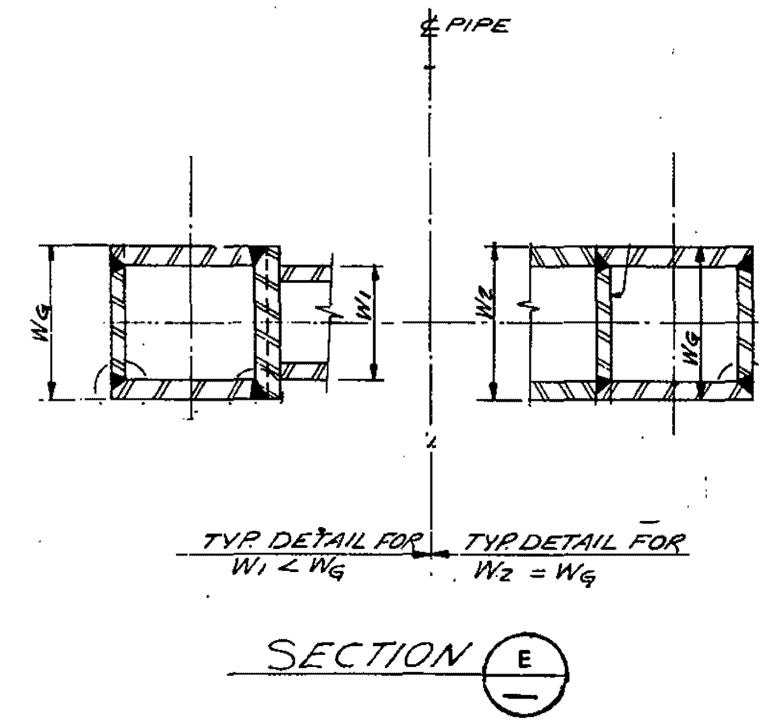
TYPICAL PIPE WHIP
RESTRAINT DETAIL

UPDATED FSAR

Sheet 3 of 7
FIGURE 3.6-1



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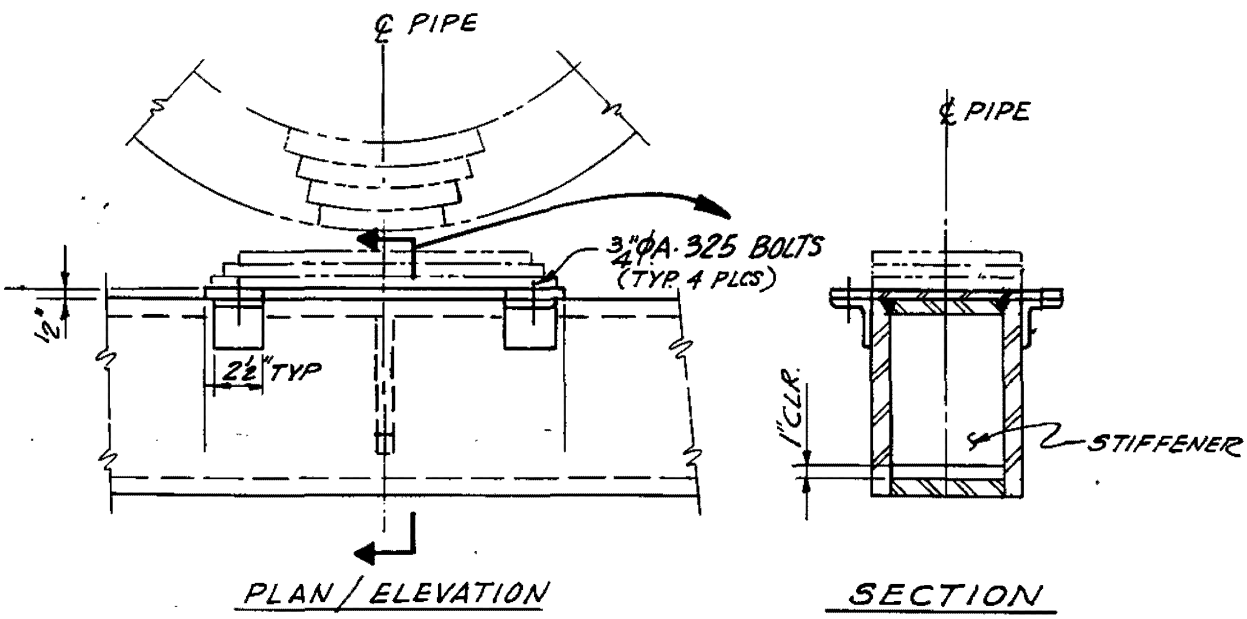
TYPICAL PIPE WHIP
RESTRAINT DETAIL

UPDATED FSAR

Sheet 4 of 7
FIGURE 3.6-1



SHIM SUPPORT DETAIL

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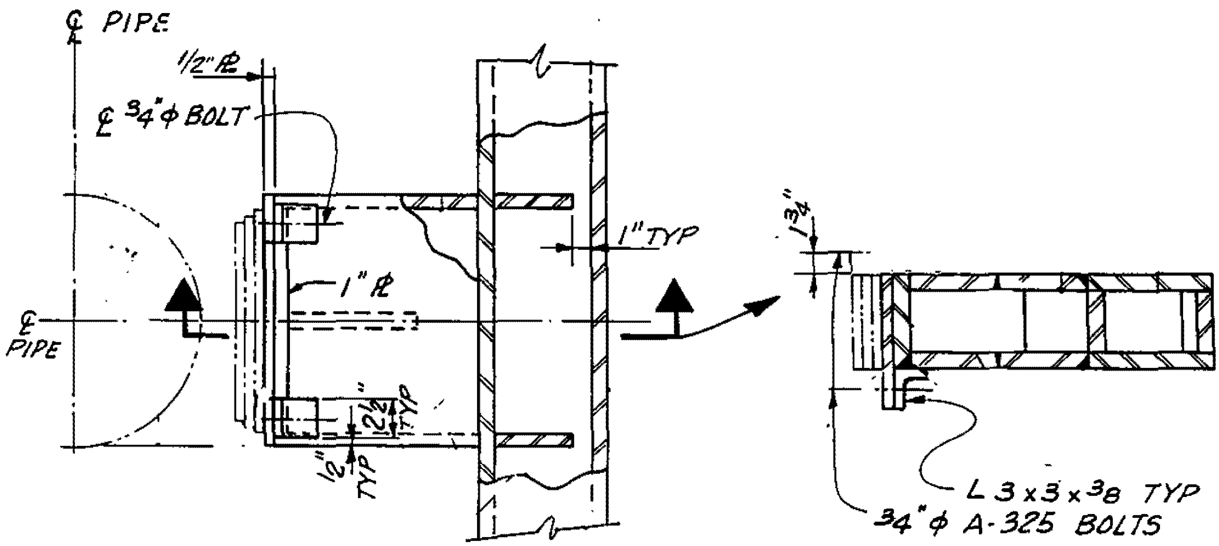
TYPICAL PIPE WHIP
RESTRAINT DETAIL

UPDATED FSAR

Sheet 5 of 7
FIGURE 3.6-1



SHIM SUPPORT DETAIL

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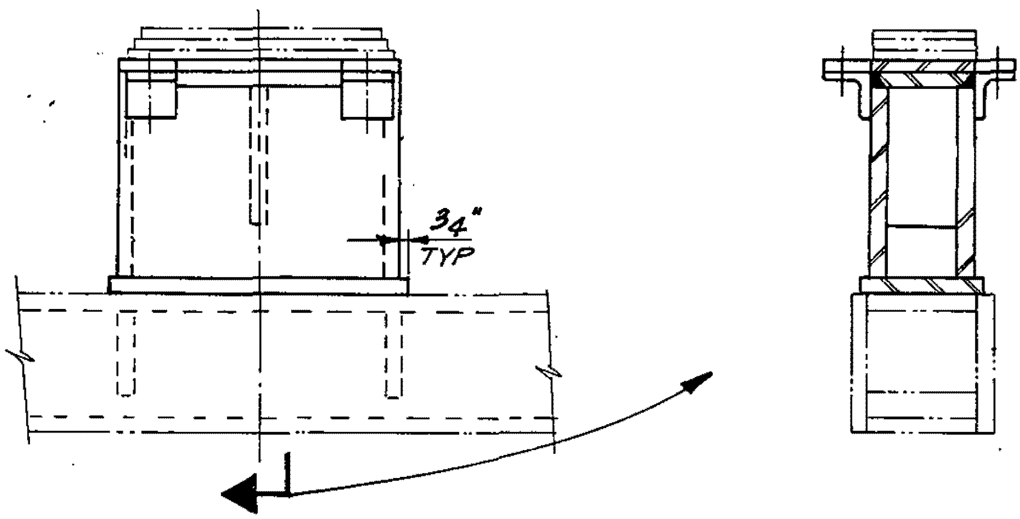
HOPE CREEK NUCLEAR GENERATING STATION

TYPICAL PIPE WHIP
RESTRAINT DETAIL

UPDATED FSAR

Sheet 6 of 7
FIGURE 3.6-1



SHIM SUPPORT DETAIL

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HOPE CREEK NUCLEAR GENERATING STATION

TYPICAL PIPE WHIP
RESTRAINT DETAIL

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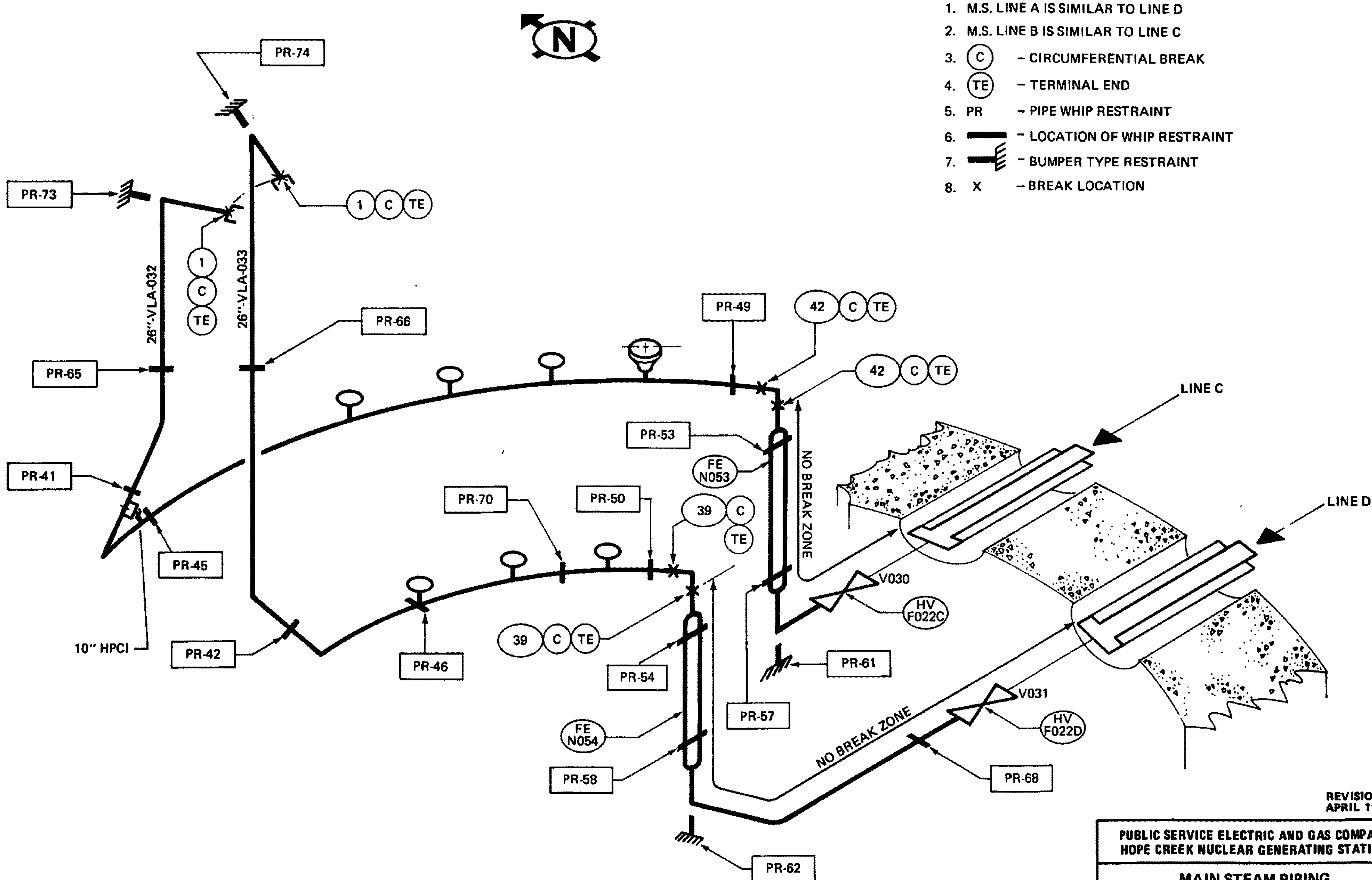
Sheet 7 of 7
FIGURE 3.6-1

NOTES:

1. M.S. LINE A IS SIMILAR TO LINE D
2. M.S. LINE B IS SIMILAR TO LINE C
3. (C) – CIRCUMFERENTIAL BREAK
4. (TE) – TERMINAL END
5. PR – PIPE WHIP RESTRAINT
6. – LOCATION OF WHIP RESTRAINT
7. – BUMPER TYPE RESTRAINT
8. X – BREAK LOCATION



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PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

MAIN STEAM PIPING
ISOMETRIC (PORTION INSIDE
PRIMARY CONTAINMENT)

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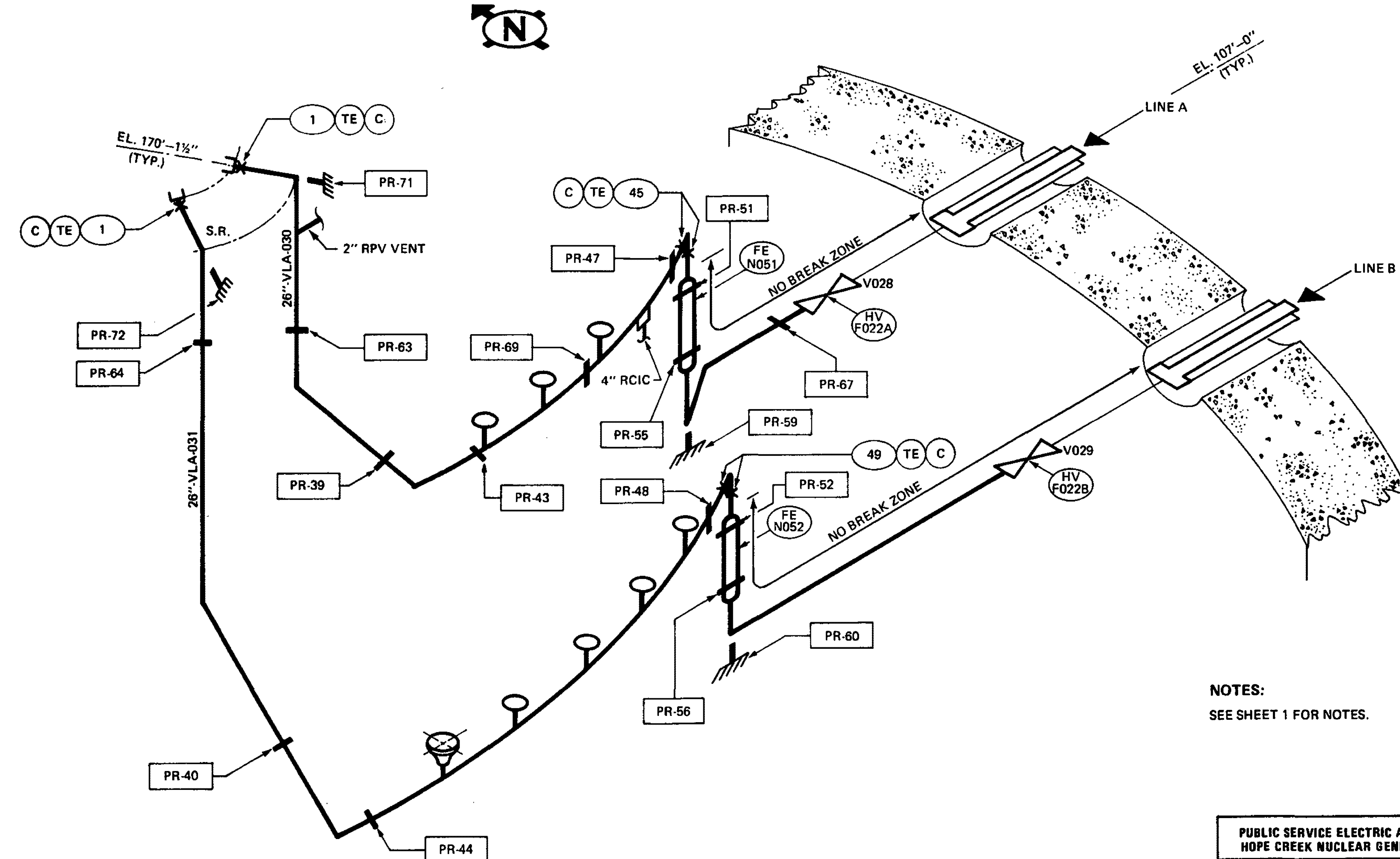
Sheet 1 of 2
FIGURE 3.6-2



NOTES:

SEE SHEET 1 FOR NOTES.

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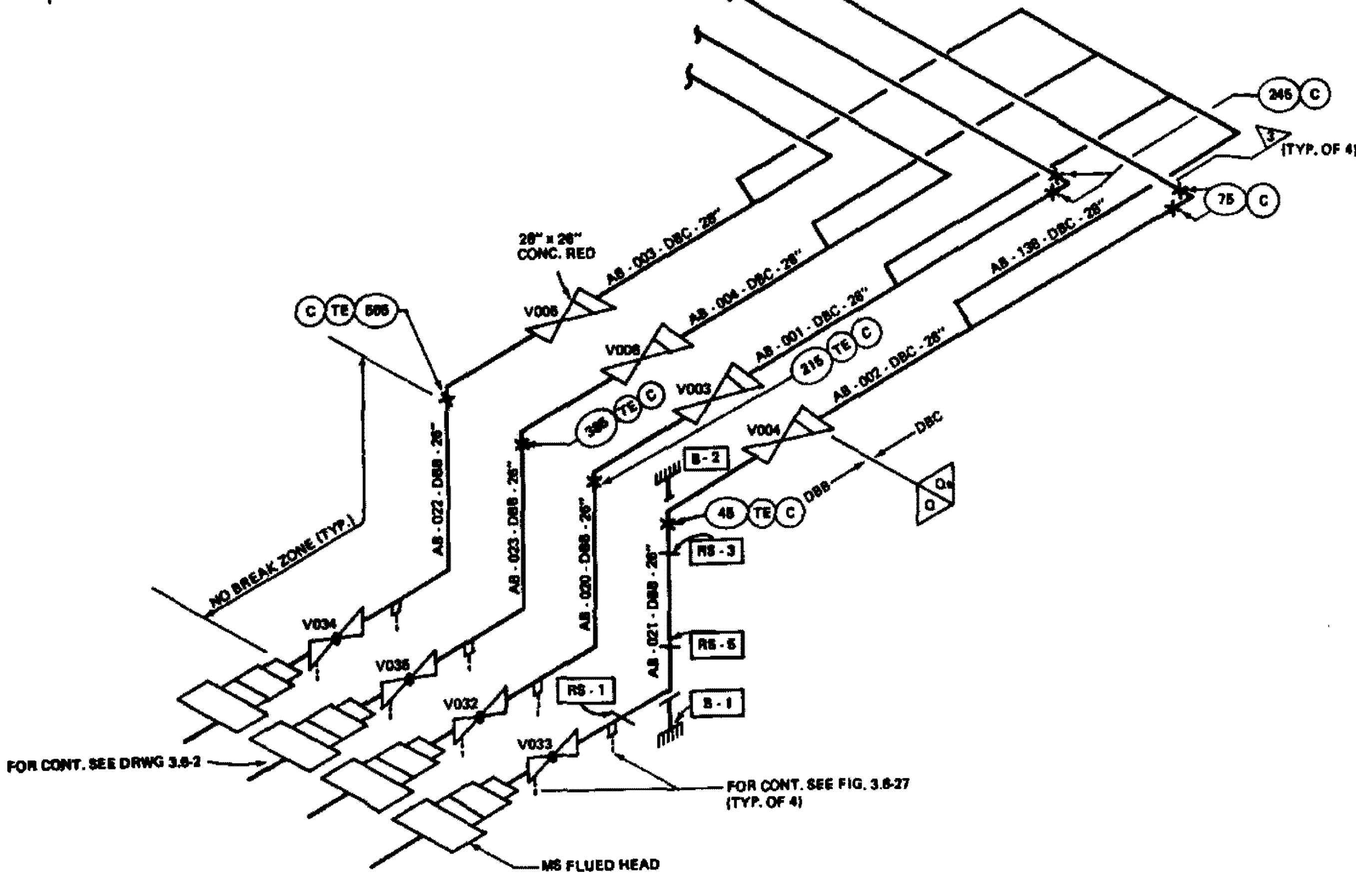
MAIN STEAM PIPING
ISOMETRIC (PORTION INSIDE
PRIMARY CONTAINMENT)

UPDATED FSAR

Sheet 2 of 2
FIGURE 3.6-2

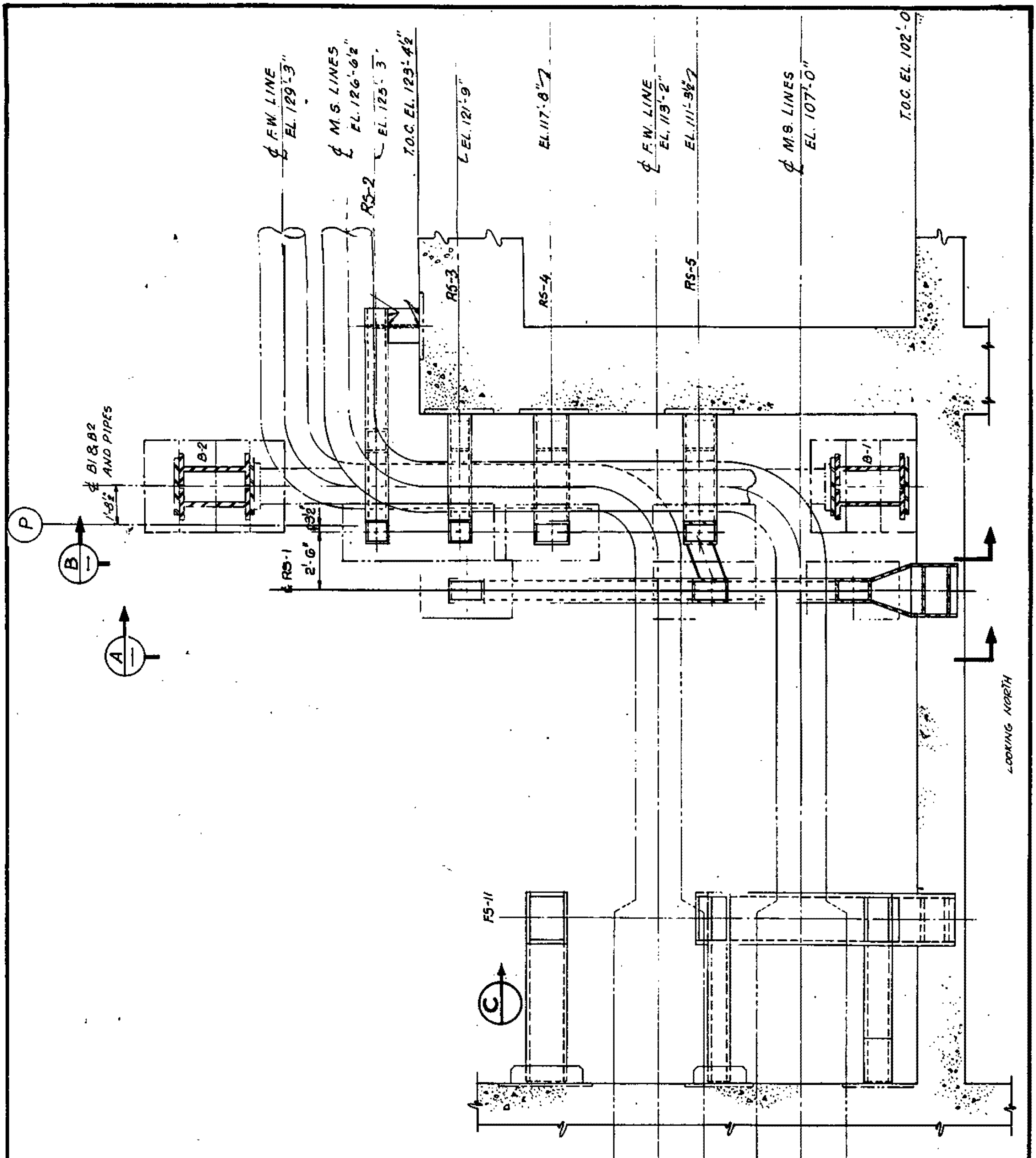
NOTES:

1. BREAKS, WHIP RESTRAINTS SHOWN ARE TYPICAL OF ALL FOUR MAIN STEAM LINES.
2. RS – PIPE WHIP RESTRAINT
3. (TE) – TERMINAL END
4. (C) – CIRCUMFERENTIAL BREAK
5. ▬ – LOCATION OF PR
6. X – BREAK LOCATION
7. B – BUMPER RESTRAINT
8. ALL WHIP RESTRAINT LOCATIONS ARE TYPICAL FOR THE FOUR MAIN STEAM LINES.
9. 2 PIPING BEYOND THIS POINT IS MODERATE ENERGY
10. 3 – BREAKS IN HIGH ENERGY PIPING BEYOND THIS POINT ARE ASSUMED AT EVERY FITTING.



Revision 12, May 3, 2002

PSEG Nuclear, LLC HOPE CREEK NUCLEAR GENERATING STATION	Hope Creek Nuclear Generating Station MAIN STEAM PIPING ISOMETRIC (PORTION OUTSIDE PRIMARY CONTAINMENT)
	Updated FSAR Figure 3.6-3



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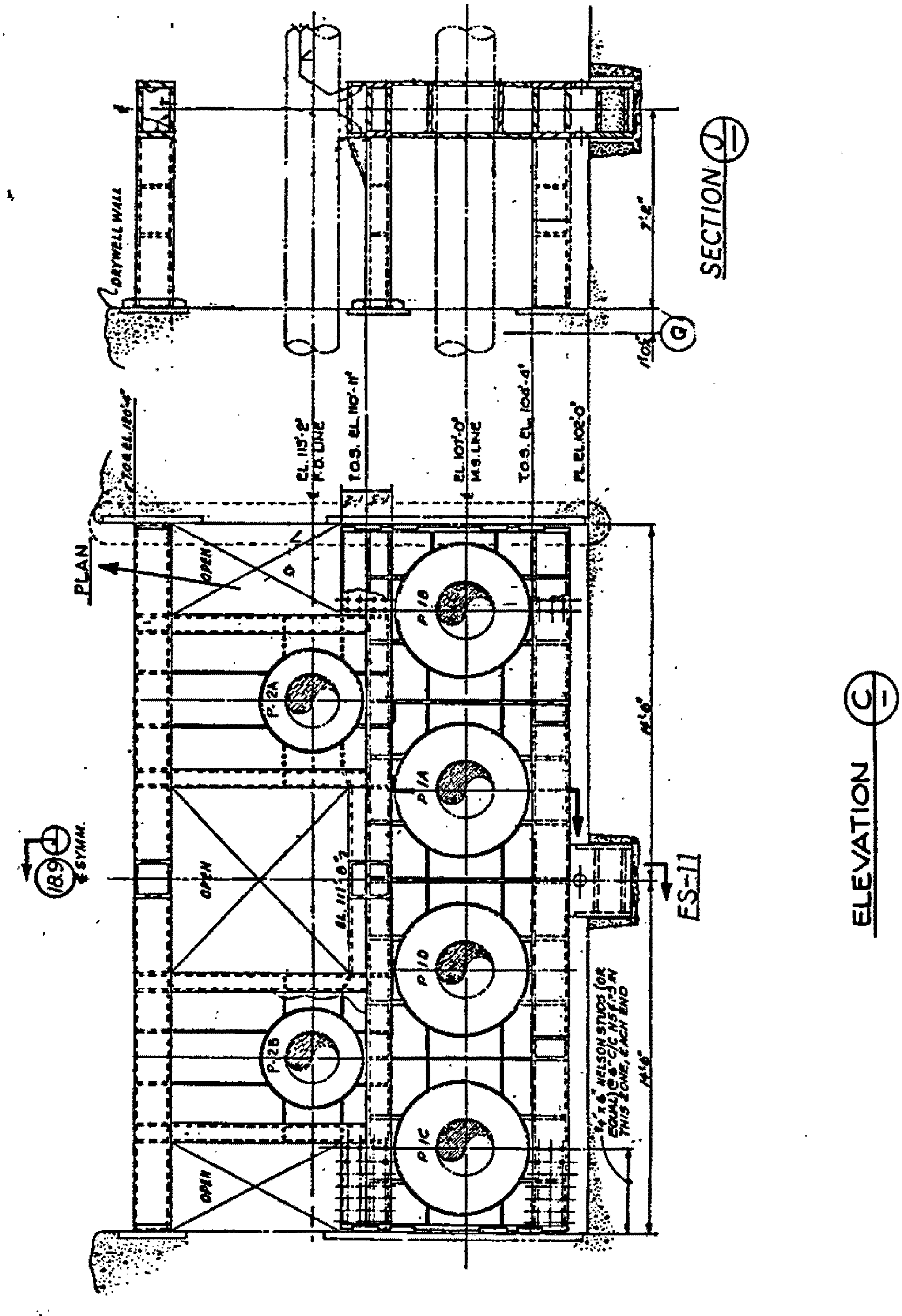
HOPE CREEK NUCLEAR GENERATING STATION

MAIN STEAM TUNNEL
ELEVATION VIEW

UPDATED FSAR

FIGURE 3.6-4



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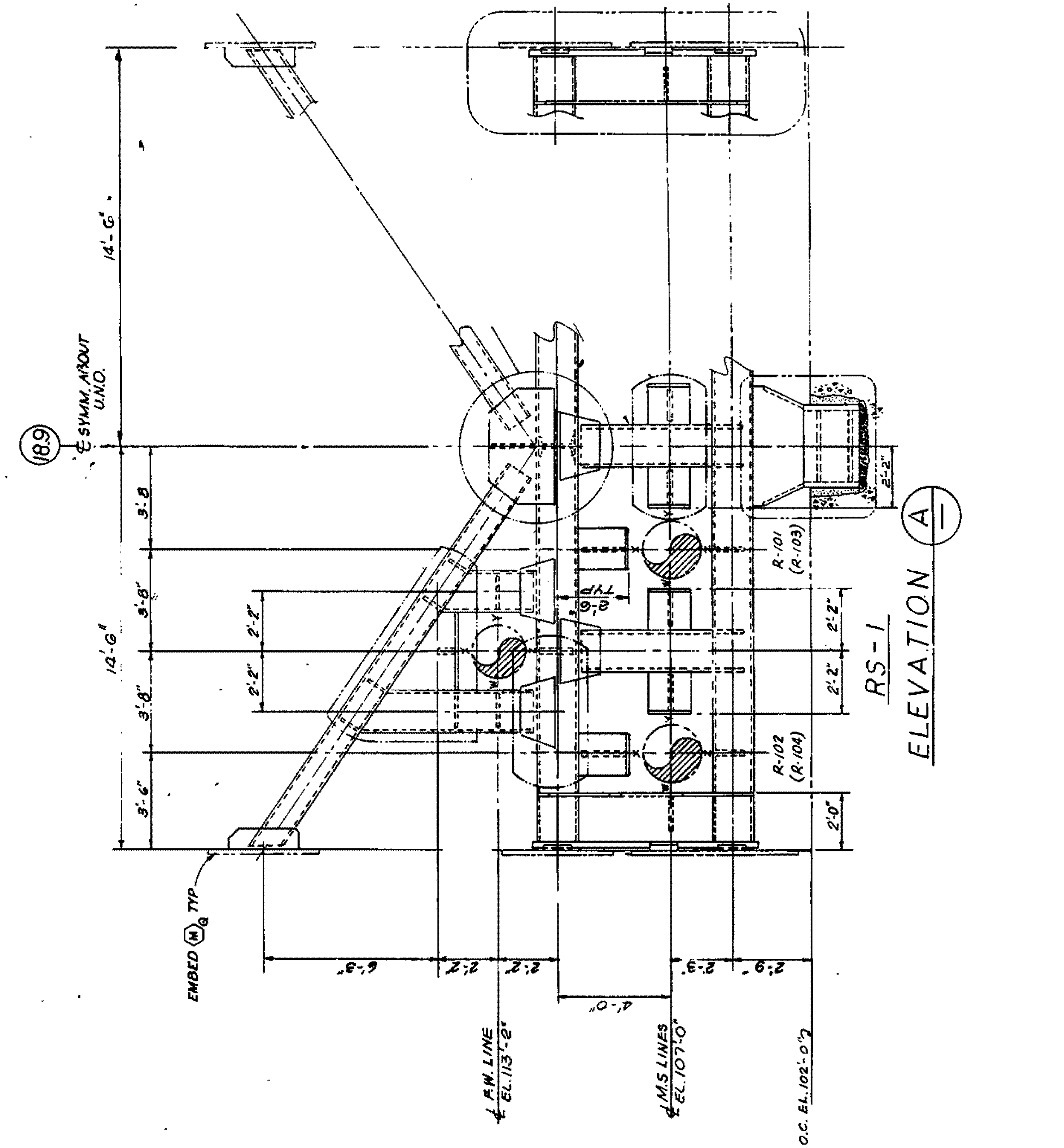
APRIL 11, 1988

PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

MAIN STEAM PIPING AND FEEDWATER PIPING ANCHOR FS–11 INSIDE MAIN STEAM TUNNEL

UPDATED FSAR

FIGURE 3.6-5



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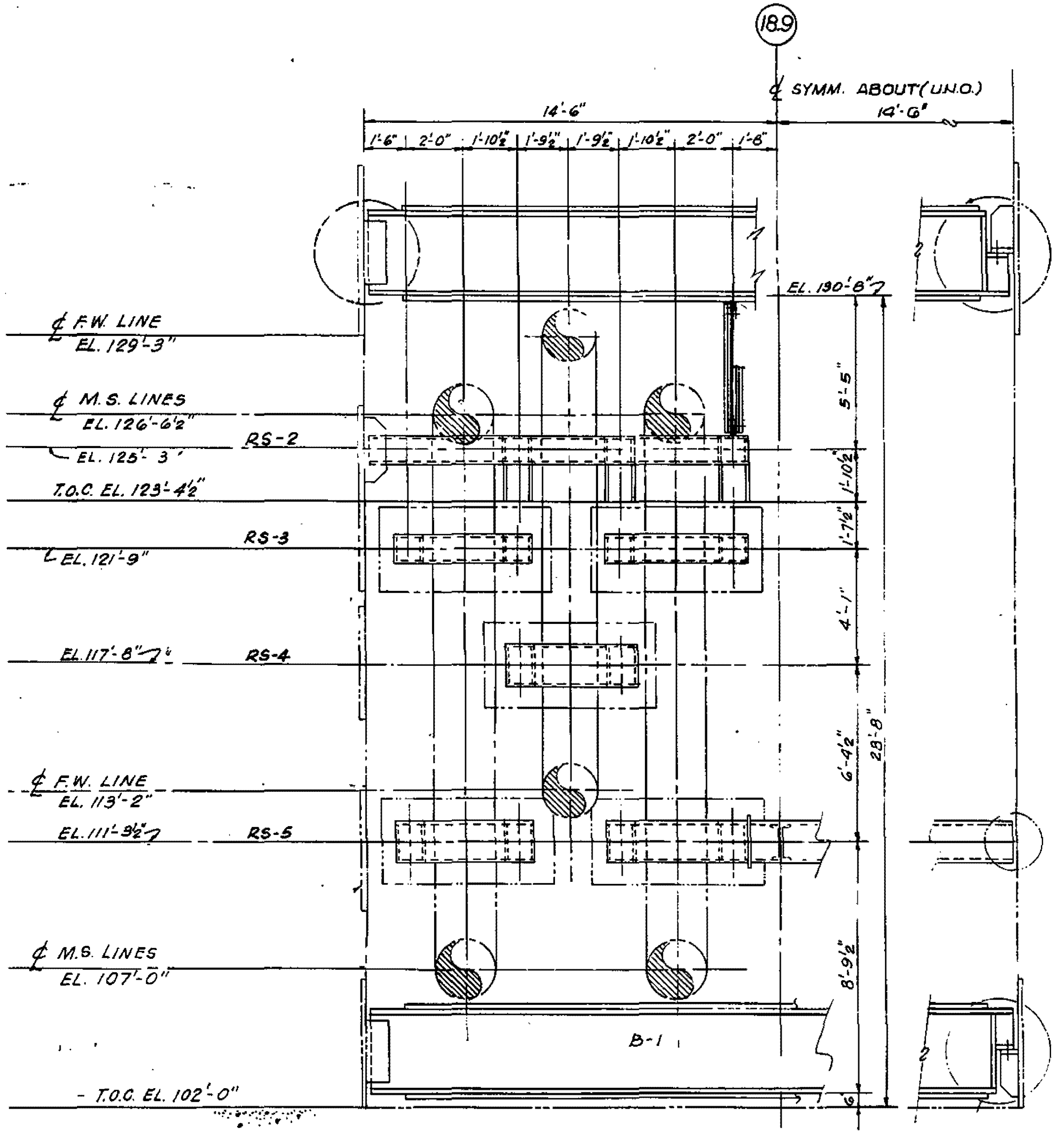
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

MAIN STEAM PIPING AND FEEDWATER PIPING RESTRAINT RS–1 INSIDE MAIN STEAM TUNNEL

UPDATED FSAR

FIGURE 3.6-6



ELEVATION (B)

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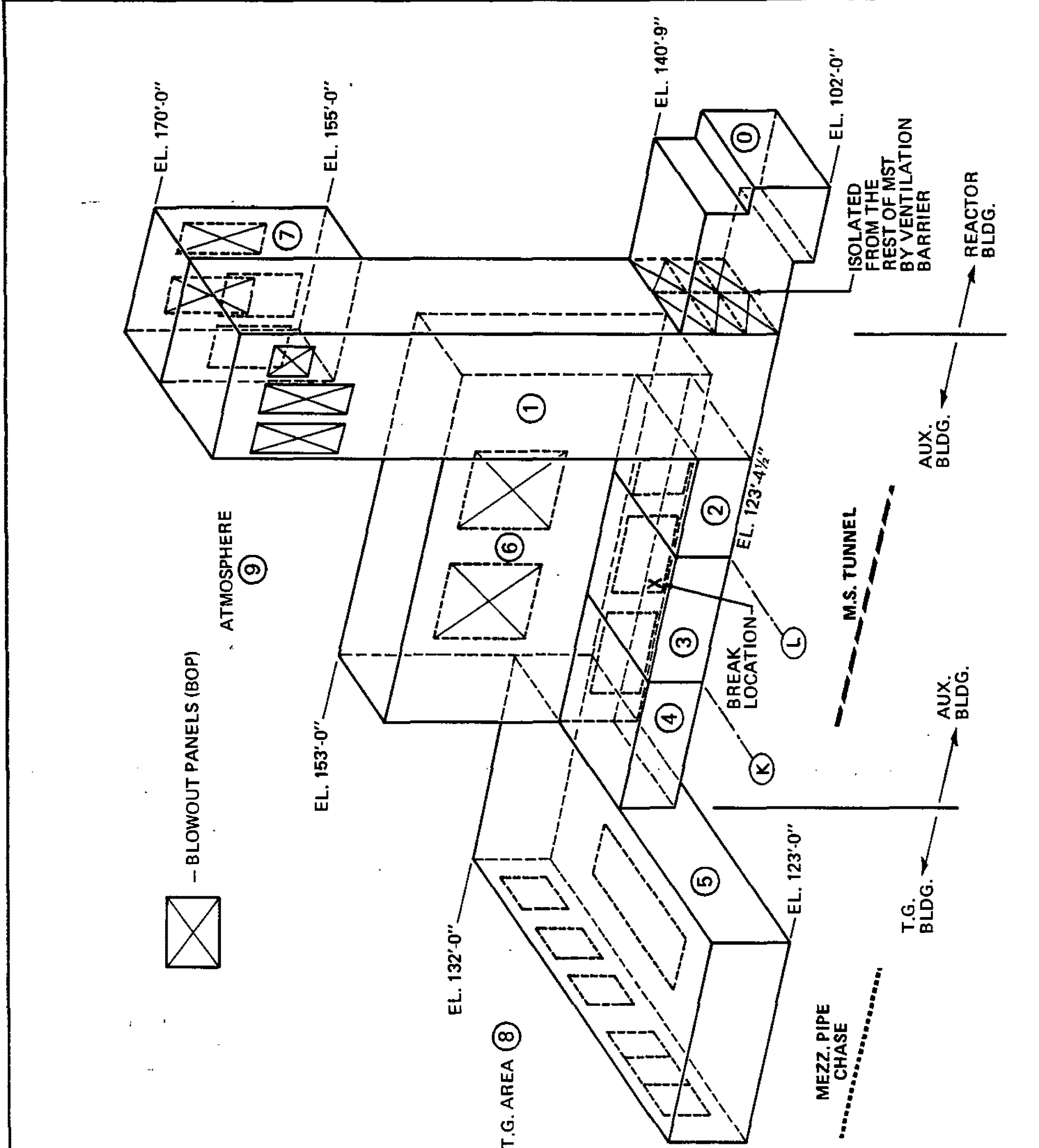
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

MAIN STEAM PIPING AND FEEDWATER PIPING RESTRAINTS INSIDE MAIN STEAM TUNNEL

UPDATED FSAR

FIGURE 3.6-7



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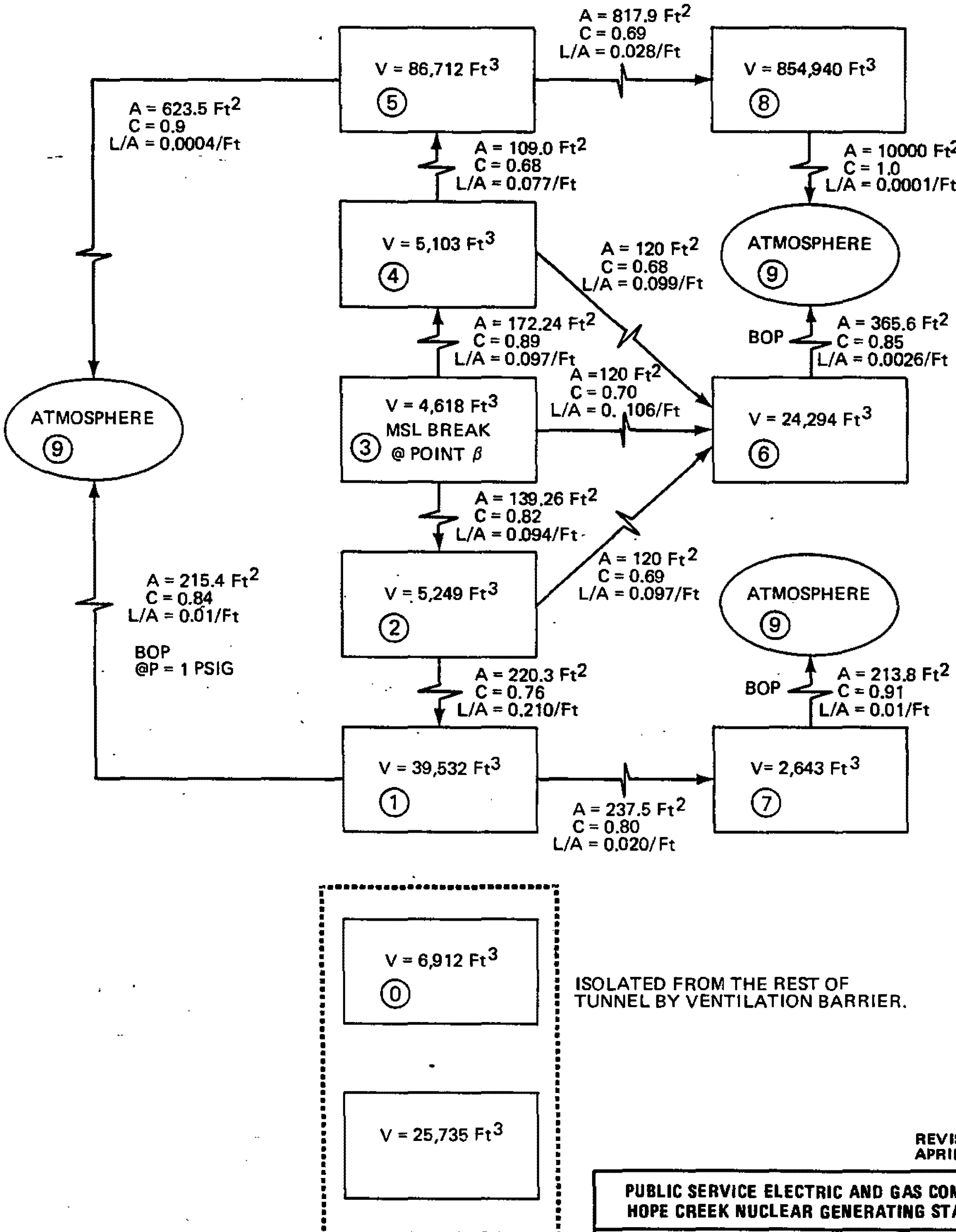
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

PRESSURE – TEMPERATURE
TRANSIENT ANALYSIS MODEL FOR A
MAIN STEAM LINE BREAK IN THE
MAIN STEAM TUNNEL

UPDATED FSAR

FIGURE 3.6-8



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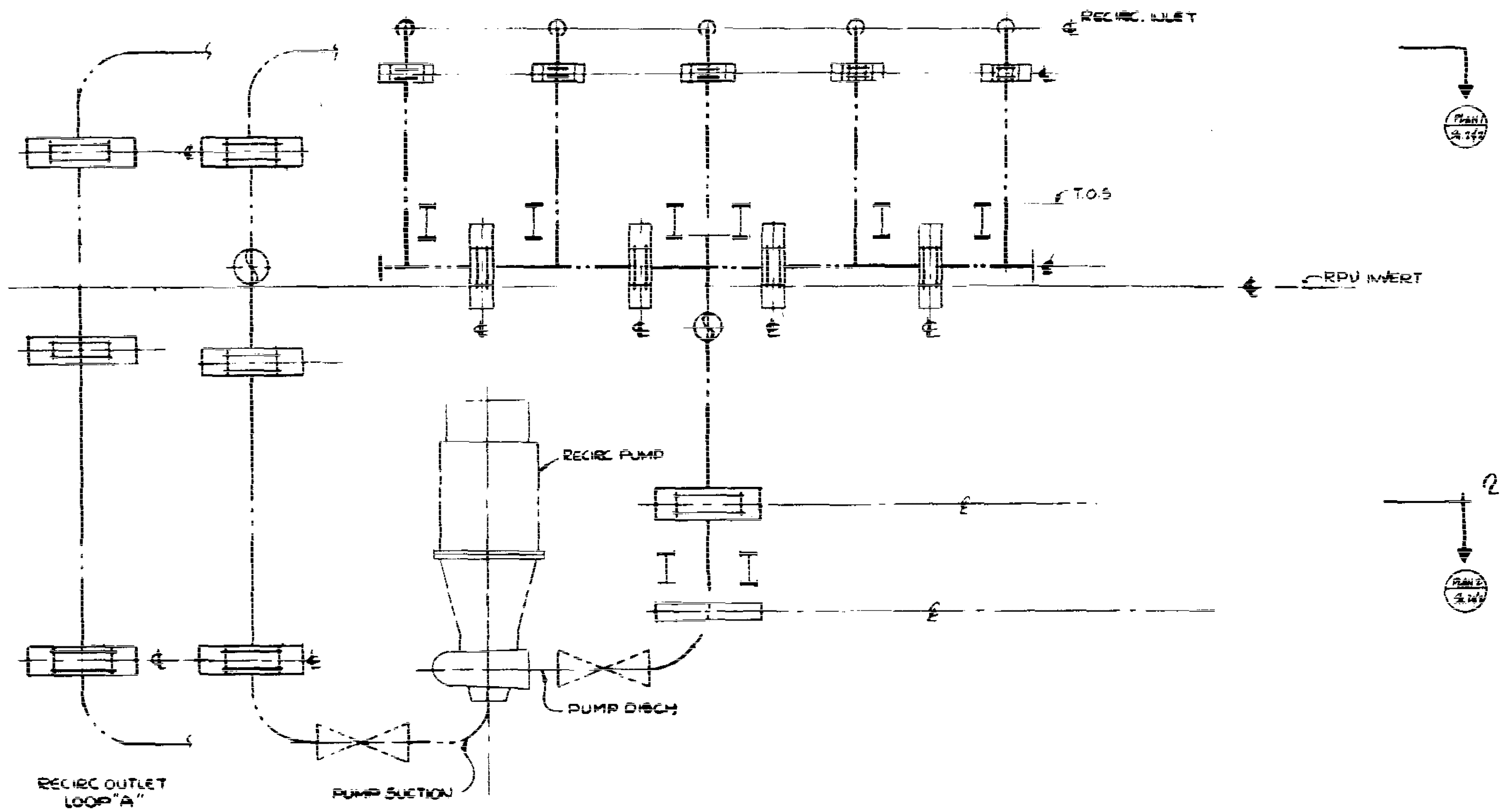
APRIL 11, 1988

PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

PRESSURE – TEMPERATURE
TRANSIENT SCHEMATIC DIAGRAM
FOR A MAIN STEAM LINE BREAK
IN MAIN STEAM TUNNEL

UPDATED FSAR

FIGURE 3.6-9



ELEVATION A-A
LOOP "B"
LOOP A SAME AS LOOP B UNLESS OTHERWISE SPECIFIED

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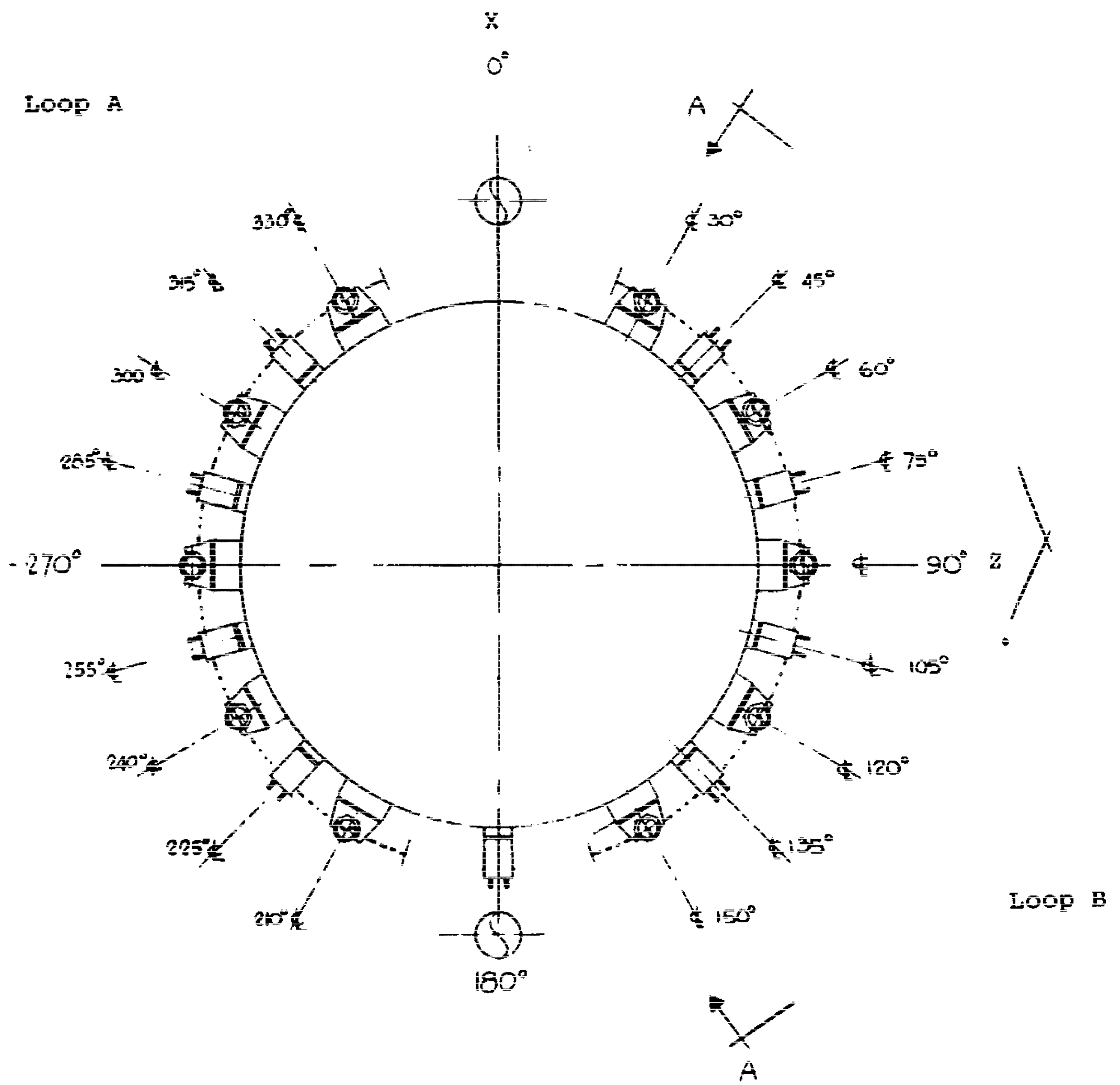
HOPE CREEK NUCLEAR GENERATING STATION

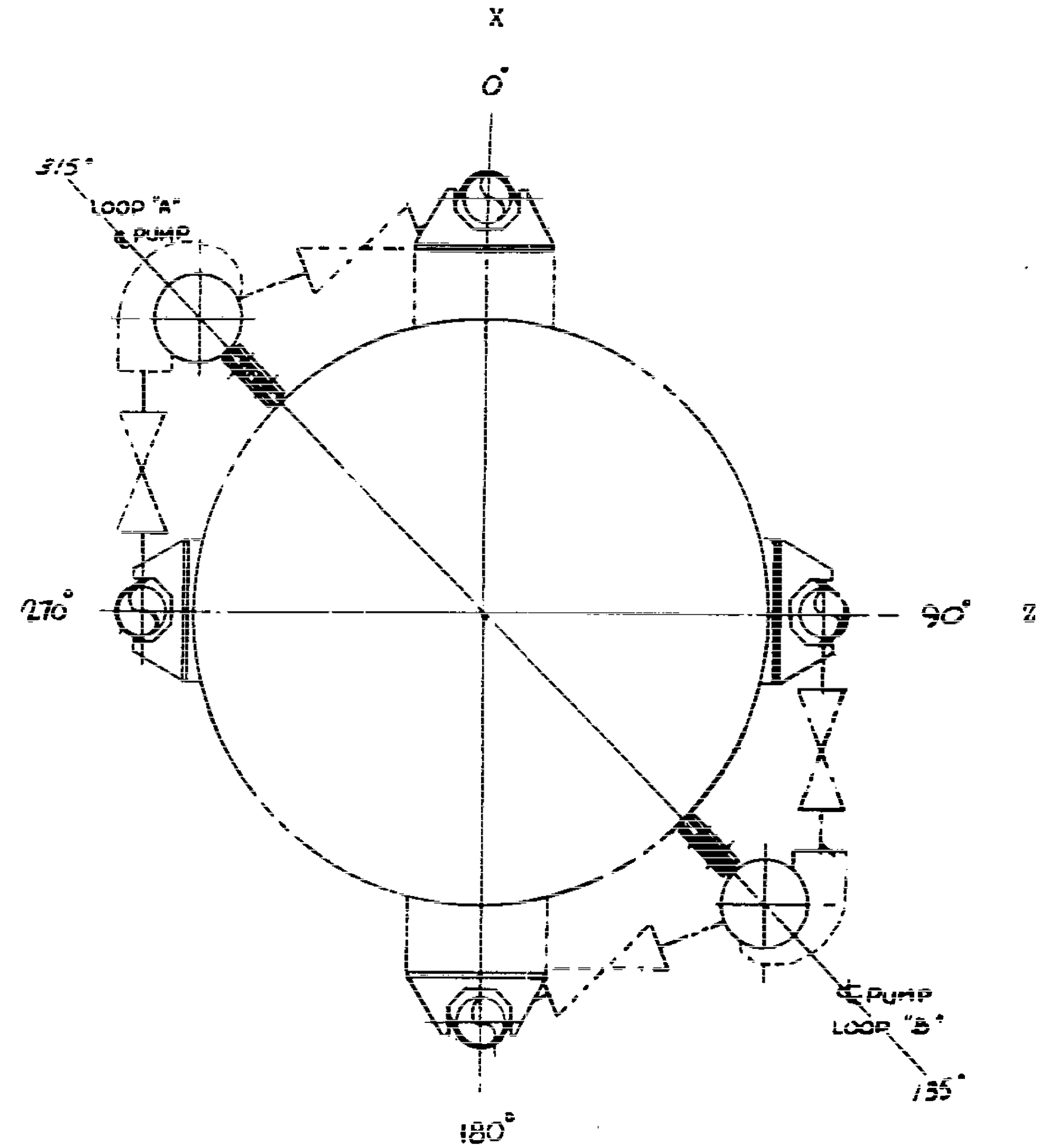
ARRANGEMENT OF RECIRCULATION LOOP PIPE WHIP RESTRAINTS

UPDATED FSAR

Sheet 1 of 2
FIGURE 3.6-10



PLAN 1



PLAN 2

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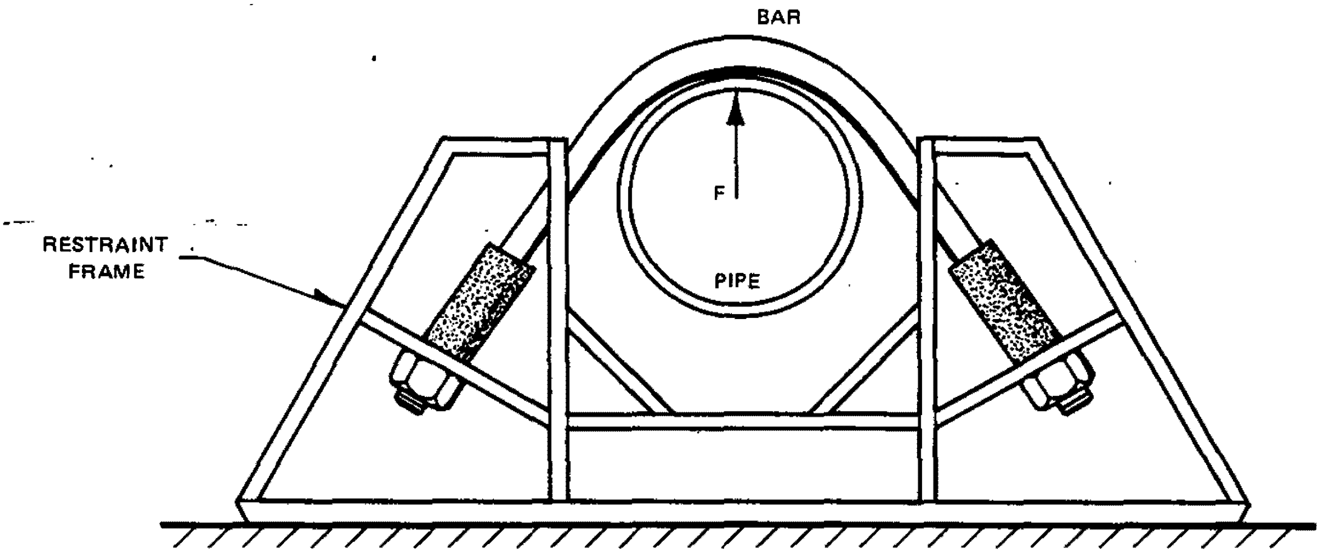
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

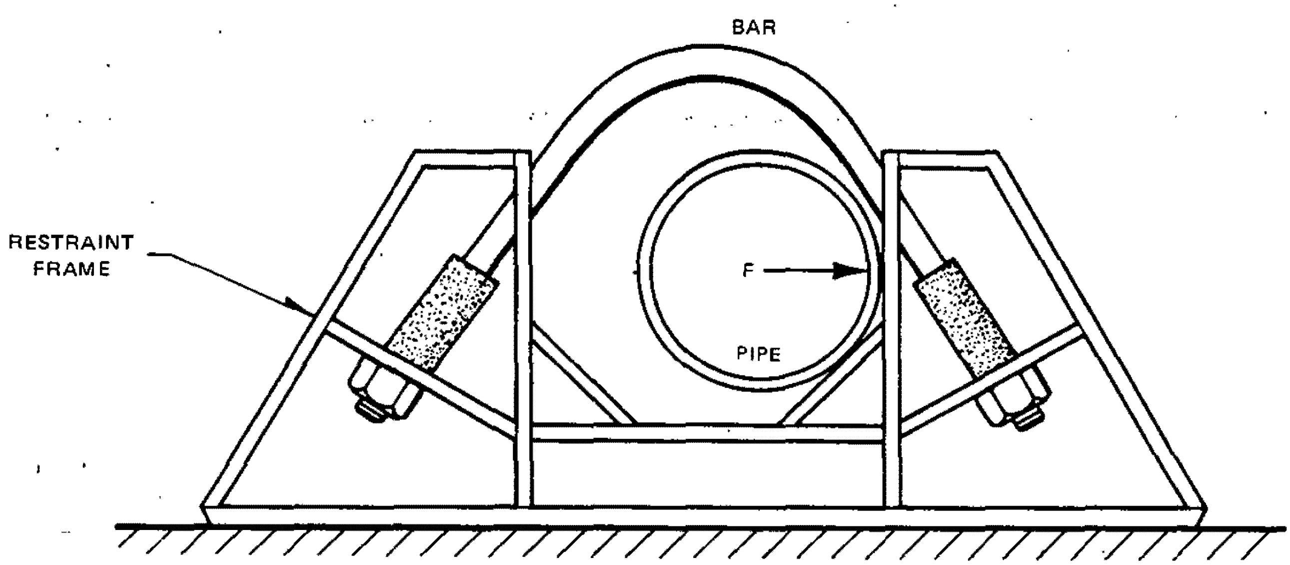
ARRANGEMENT OF RECIRCULATION LOOP PIPE WHIP RESTRAINTS

UPDATED FSAR

Sheet 2 of 2
FIGURE 3.6-10



(a) Load Applied Perpendicular to Restraint Base Against Cables



(b) Load Applied Parallel to Frame Base Against One Side of Restraint Frame

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HOPE CREEK NUCLEAR GENERATING STATION

TYPICAL RECIRCULATION SYSTEM
PIPE WHIP RESTRAINT

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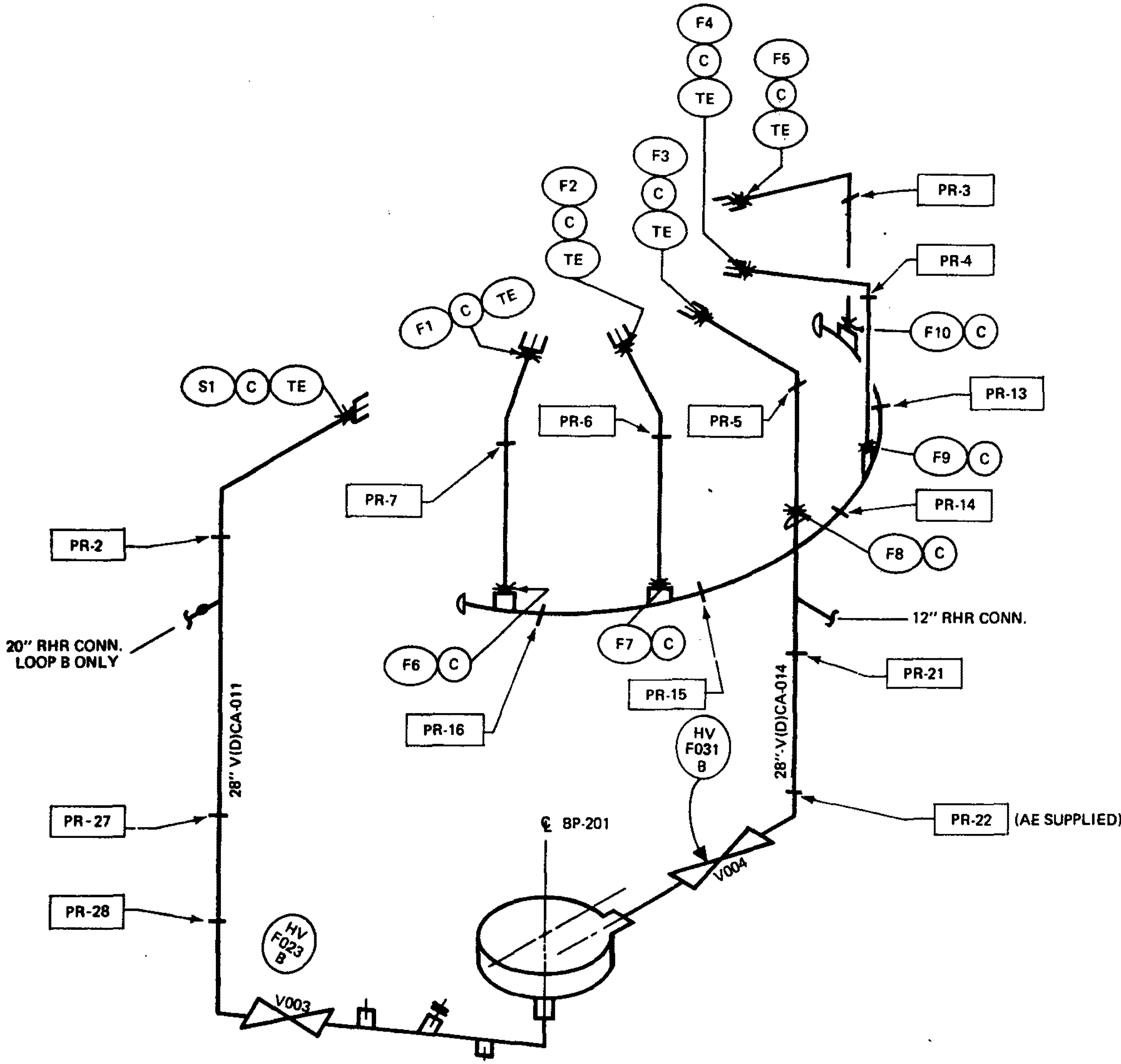
FIGURE 3.6-11



NOTE:

1. FIGURE SHOWN IS LOOP B.
   LOOP A SAME AS LOOP B UNLESS SPECIFIED
2. PR INDICATES WHIP RESTRAINT IDENTIFICATION
3. C - CIRCUMFERENTIAL BREAK
4. TE - TERMINAL END
5. X - BREAK LOCATION
6. PIPE WHIP RESTRAINTS PR 21,22,27, AND 28 ON LOOPS A AND B ARE INACTIVE.

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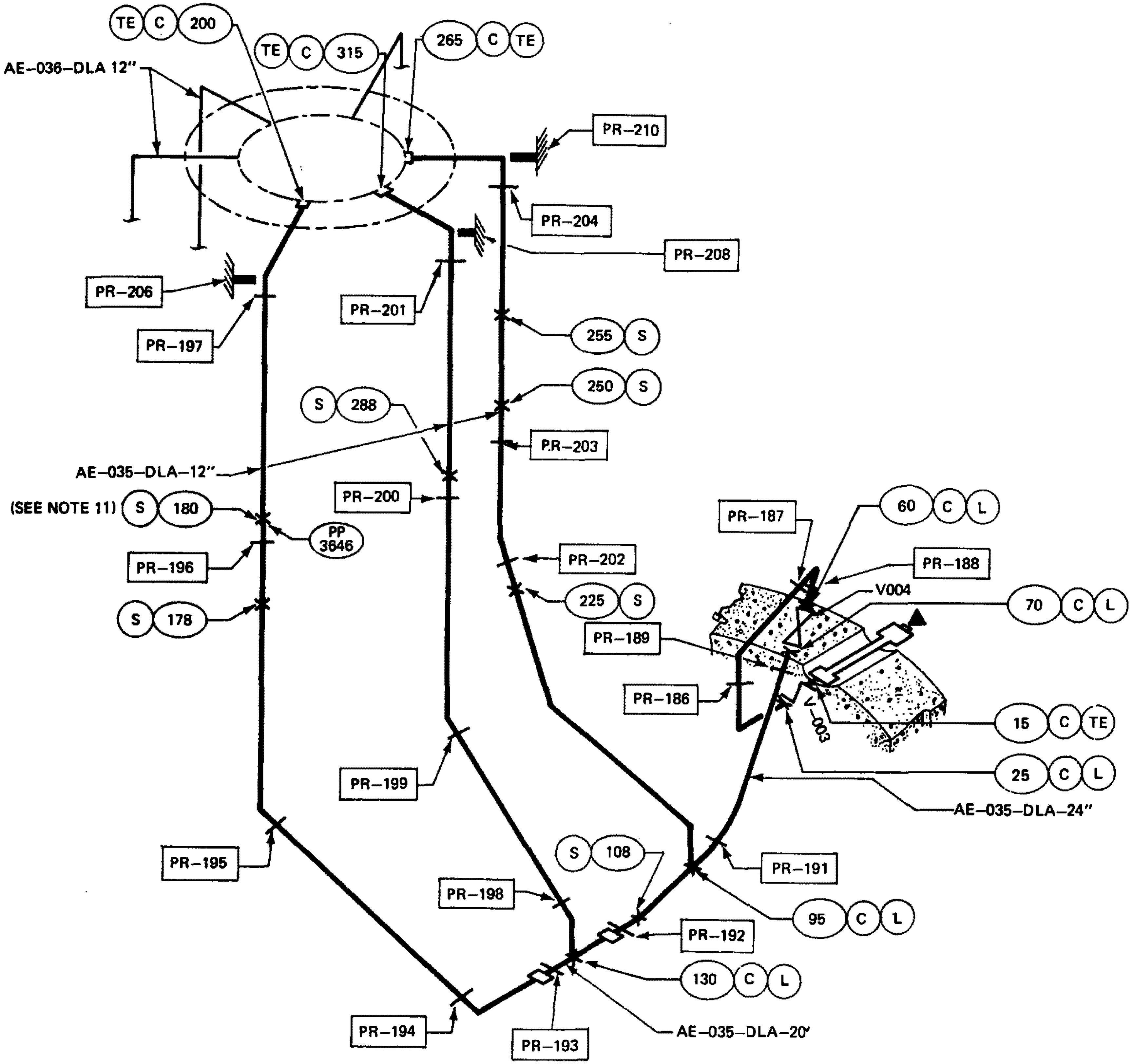
HOPE CREEK NUCLEAR GENERATING STATION

RECIRCULATION SYSTEM ISOMETRIC

UPDATED FSAR

FIGURE 3.6-12



NOTES:

1. FW LINE AE-036 IS SIMILAR TO LINE AE-035
2. (C) CIRCUMFERENTIAL BREAK
3. (L) LONGITUDINAL BREAK
4. (TE) TERMINAL END
5. PR PIPE WHIP RESTRAINT
6. — LOCATION OF PR
7. BUMPER TYPE RESTRAINT
8. ▲ ANCHOR
9. ✕ BREAK LOCATION
10. (S) SLOT BREAK, DEFINED AS A LONGITUDINAL BREAK IN THE RUN PIPING WITH FLOW AREA EQUAL TO THE CROSS SECTIONAL AREA OF THE WELDED ATTACHMENT.
11. LOCATED ON AE-036-DLA-12" ONLY WITH FLOW AREA EQUAL TO CROSS-SECTIONAL AREA BASED ON INSIDE DIAMETER OF BRANCH PIPE.

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PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

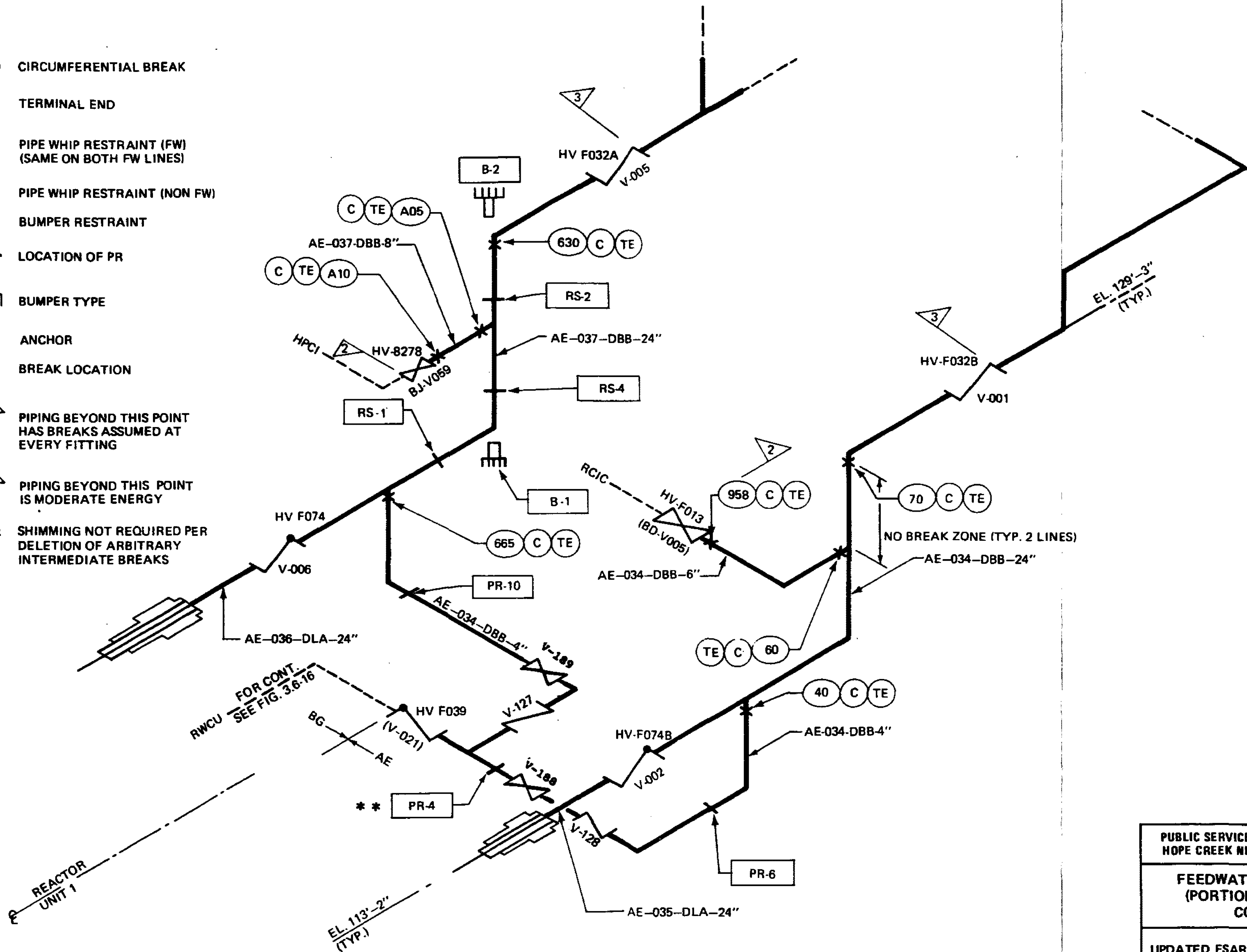
FEEDWATER PIPING ISOMETRIC
(PORTION INSIDE PRIMARY CONTAINMENT)

UPDATED FSAR

FIGURE 3.6-13

NOTES:

1. (C) CIRCUMFERENTIAL BREAK
2. (TE) TERMINAL END
3. RS PIPE WHIP RESTRAINT (FW) (SAME ON BOTH FW LINES)
4. PR PIPE WHIP RESTRAINT (NON FW)
5. B BUMPER RESTRAINT
6. — LOCATION OF PR
7. BUMPER TYPE
8. ▲ ANCHOR
9. ✕ BREAK LOCATION
10. 3 PIPING BEYOND THIS POINT HAS BREAKS ASSUMED AT EVERY FITTING
11. 2 PIPING BEYOND THIS POINT IS MODERATE ENERGY
12. ✱✱ SHIMMING NOT REQUIRED PER DELETION OF ARBITRARY INTERMEDIATE BREAKS



✱✱

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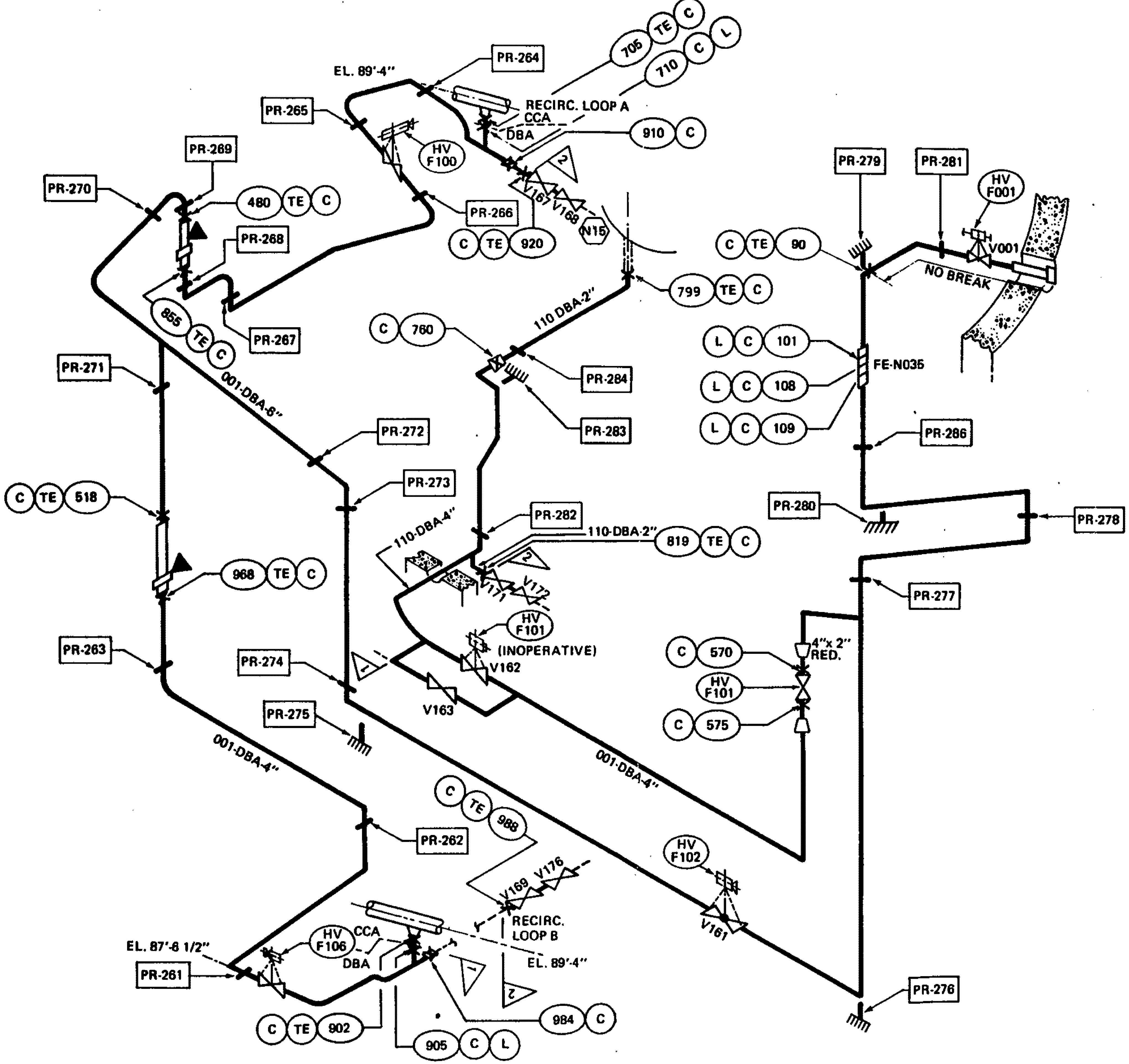
September 25, 1996

PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

FEEDWATER PIPING ISOMETRIC (PORTION OUTSIDE PRIMARY CONTAINMENT)

UPDATED FSAR

FIGURE 3.6-14



NOTES:

1. NODE POINTS (799) AND (760) (2"-DBA-110) ARE PART OF THE REACTOR VESSEL DRAIN.
2. (C) – CIRCUMFERENTIAL BREAK
3. (TE) – TERMINAL END
4. PR – PIPE WHIP RESTRAINT
5. – LOCATION OF WHIP RESTRAINT
6. – BUMPER TYPE RESTRAINT
7. – ANCHOR
8. X – BREAK LOCATION
9. 1 – PIPING BEYOND THIS POINT IS 1" NOMINAL DIAMETER
10. 2 – PIPING BEYOND THIS POINT IS MODERATE ENERGY
11. (L) – LONGITUDINAL BREAK

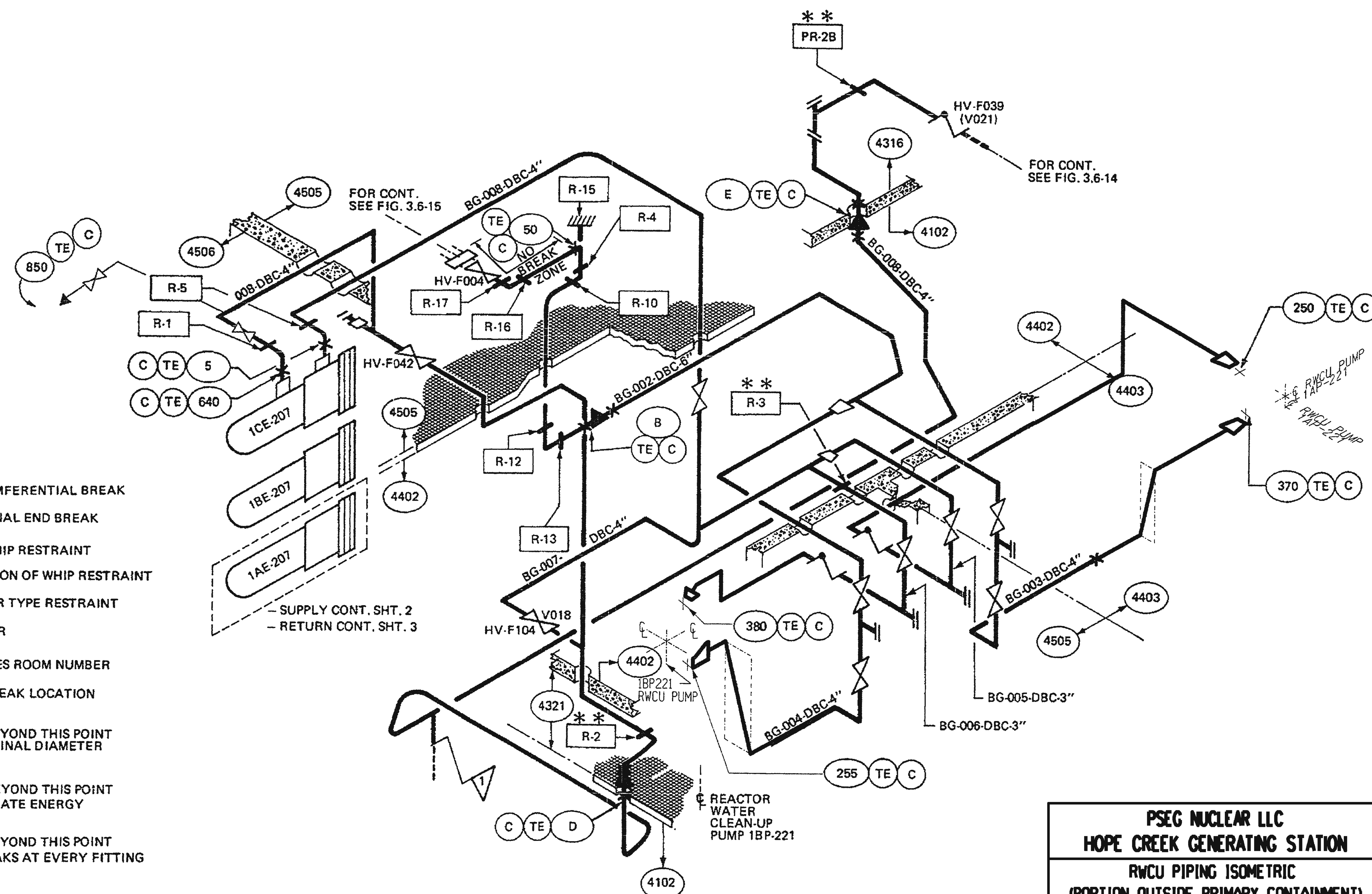
Revision 7
December 29, 1995

PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

RWCU PIPING ISOMETRIC
(PORTION INSIDE PRIMARY CONTAINMENT)

UPDATED FSAR

FIGURE 3.6-15



NOTES:

1. (C) – CIRCUMFERENTIAL BREAK
2. (TE) – TERMINAL END BREAK
3. R – PIPE WHIP RESTRAINT
4. – LOCATION OF WHIP RESTRAINT
5. – BUMPER TYPE RESTRAINT
6. ▲ – ANCHOR
7. (XXXX) – DENOTES ROOM NUMBER
8. X – PIPE BREAK LOCATION
9. 1 PIPING BEYOND THIS POINT IS 1" NOMINAL DIAMETER
10. 2 PIPING BEYOND THIS POINT IS MODERATE ENERGY
11. 3 PIPING BEYOND THIS POINT HAS BREAKS AT EVERY FITTING
12. ✻✻ – SHIMMING NOT REQUIRED PER DELETION OF ARBITRARY INTERMEDIATE BREAKS

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HOPE CREEK GENERATING STATION

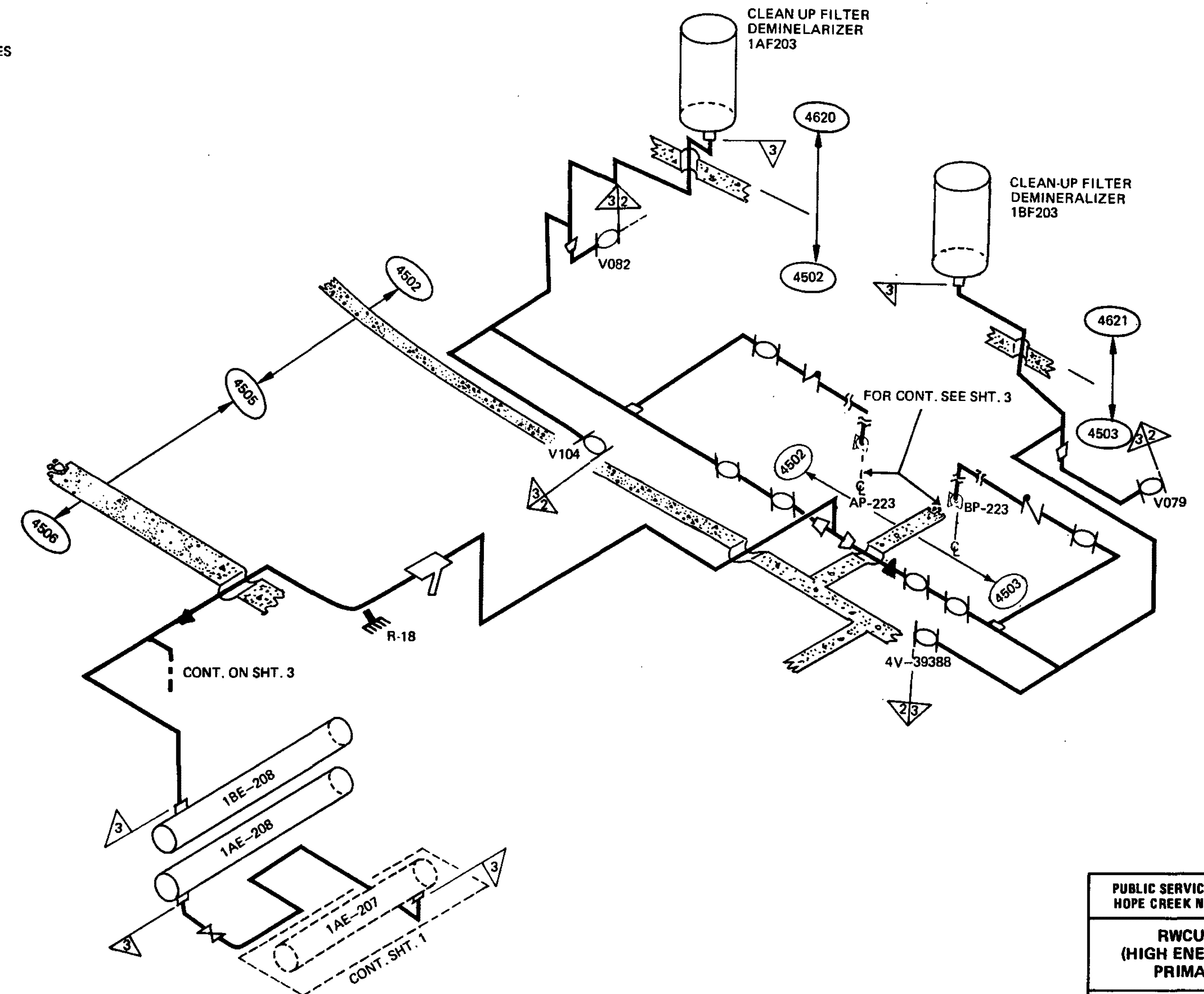
RWCU PIPING ISOMETRIC
(PORTION OUTSIDE PRIMARY CONTAINMENT)

Updated FSAR — Sheet 1 of 3

Revision 21, Nov 9, 2015 — Fig. 3.6-16

NOTES:

SEE SHT. 1 FOR NOTES



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APRIL 11, 1988

PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

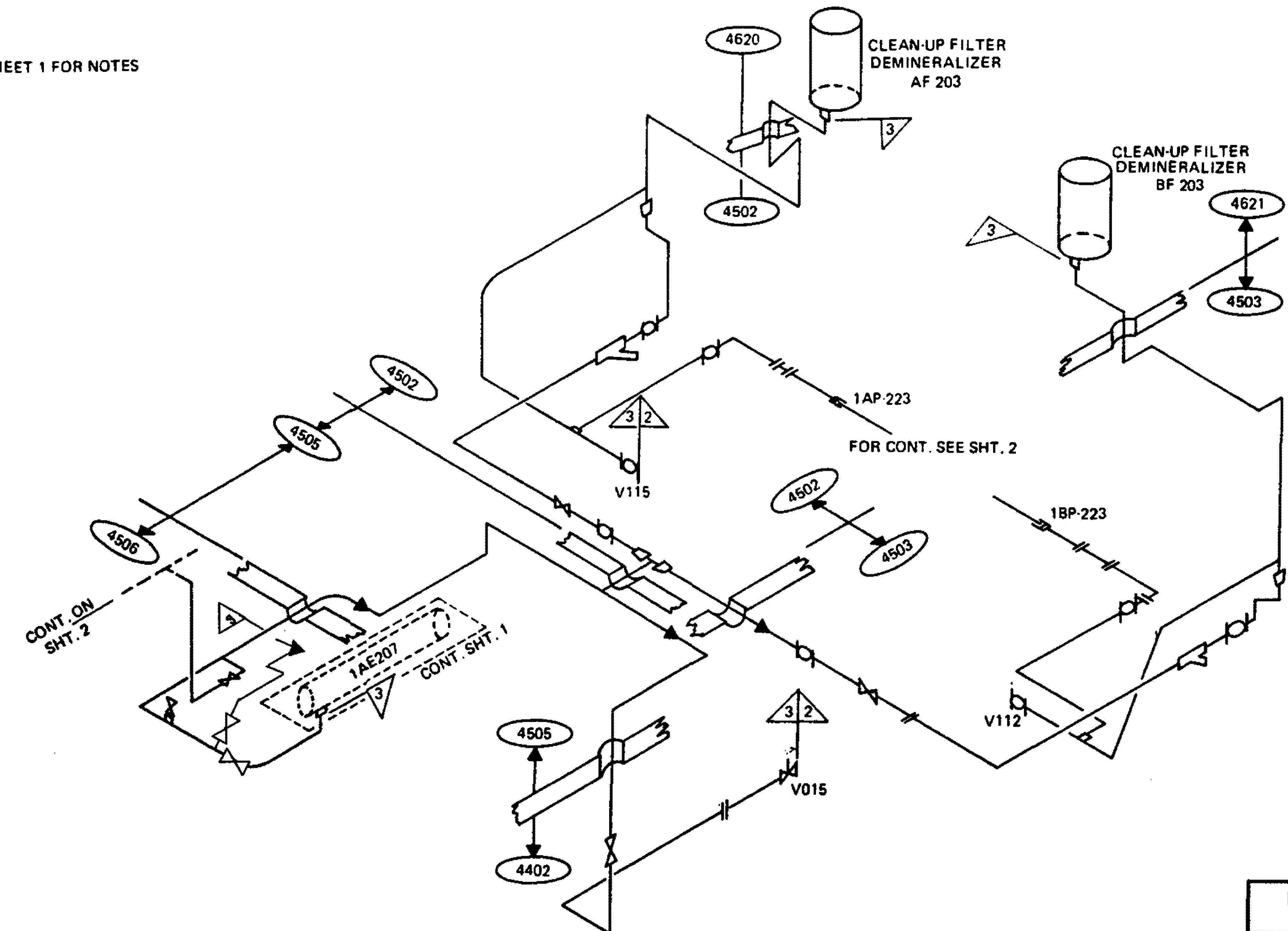
RWCU PIPING ISOMETRIC
(HIGH ENERGY PORTION OUTSIDE PRIMARY CONTAINMENT)

UPDATED FSAR

Sheet 2 of 3
FIGURE 3.6-16

NOTES:

SEE SHEET 1 FOR NOTES

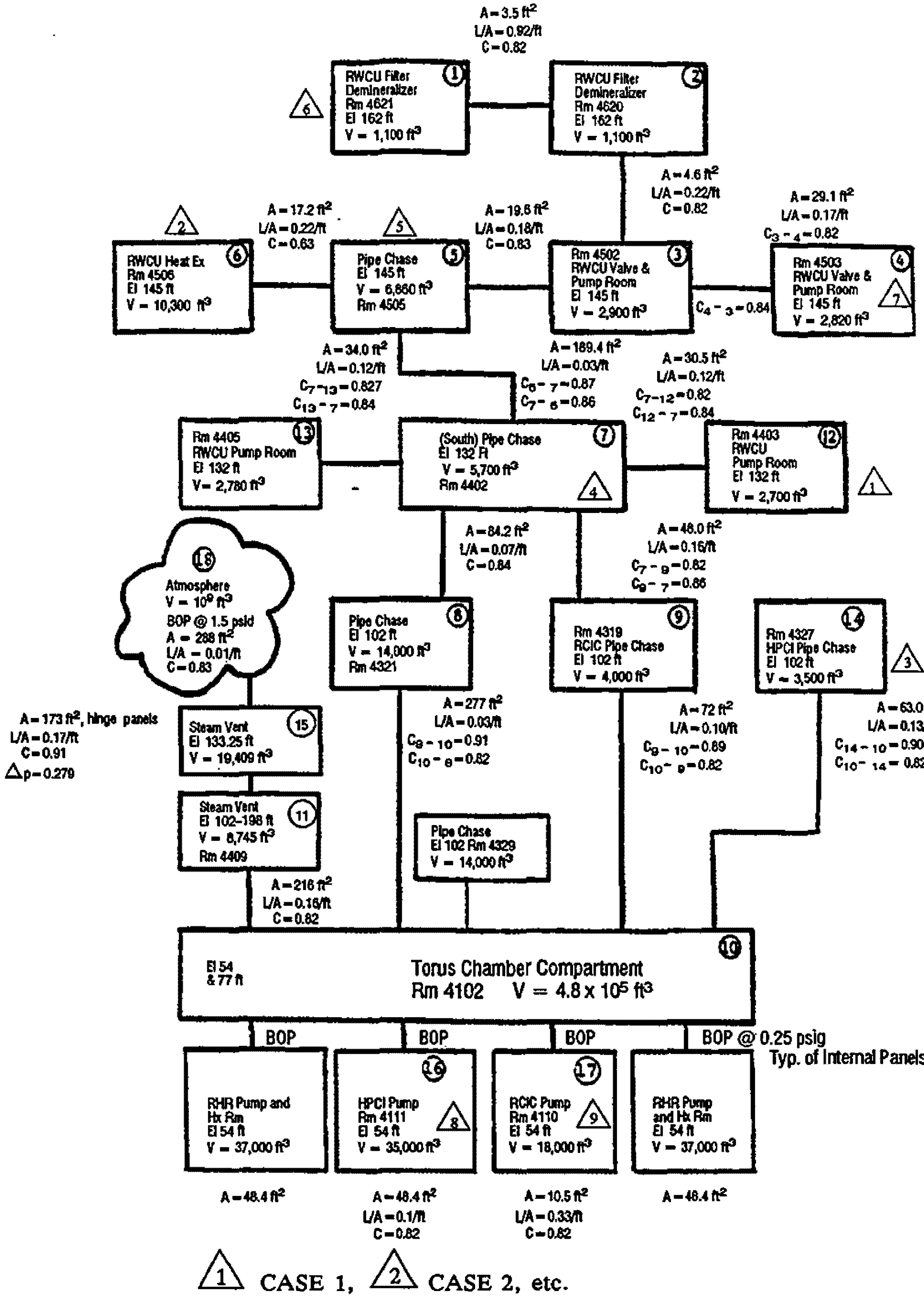


PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK GENERATING STATION

RWCU PIPING ISOMETRIC
(HIGH ENERGY PORTION OUTSIDE PRIMARY CONTAINMENT)

Updated FSAR
Revision 2, April 11, 1990

Sheet 3 of 3
Fig. 3.6-16



NOTES:

1. The volumes shown on this figure have been reduced by the volume of equipment as estimated on Sheet 13.
2. Room 4329 has no high-energy lines and is conservatively neglected.
3. C is same in both directions unless noted otherwise.

PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK GENERATING STATION

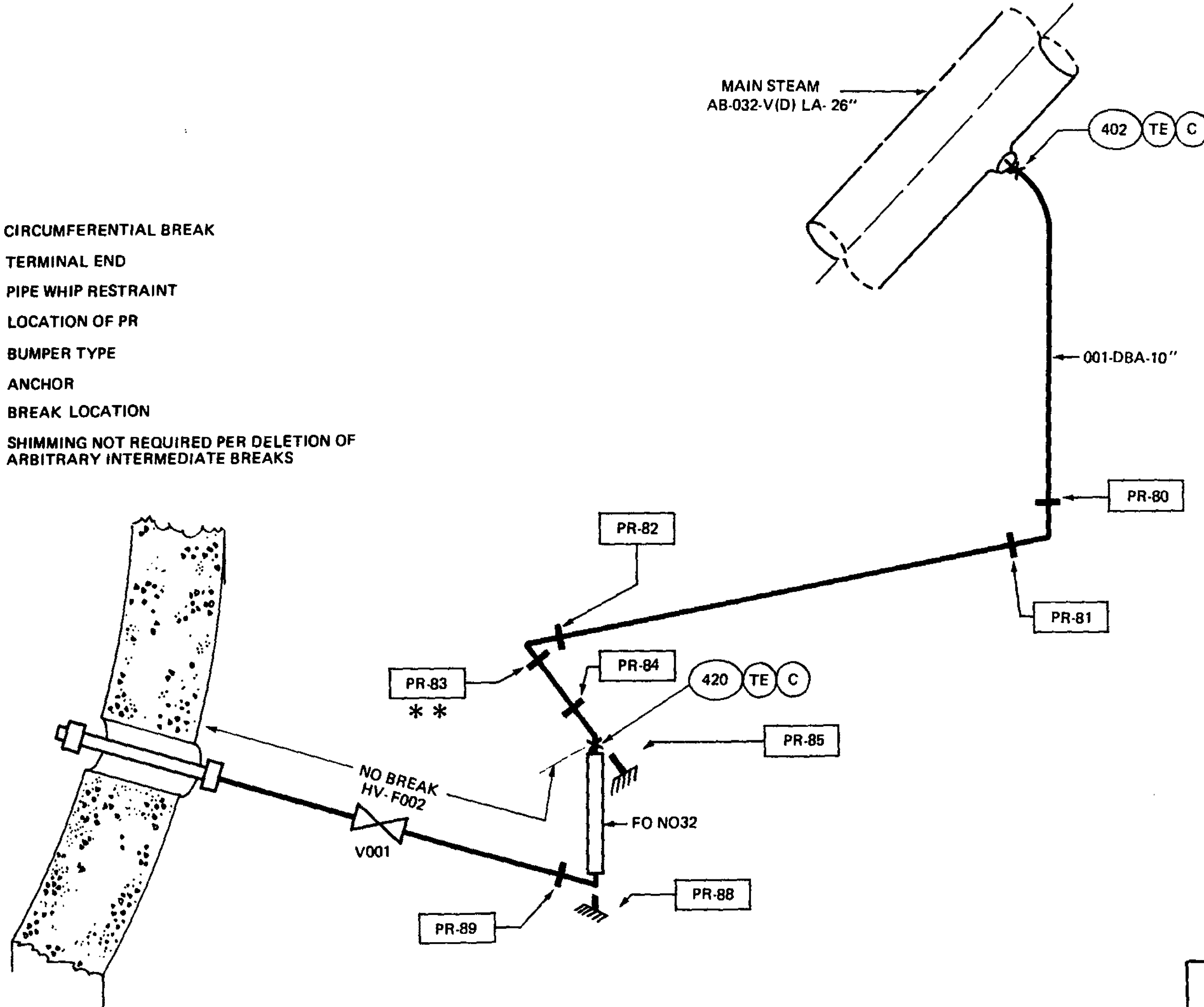
PRESSURE – TEMPERATURE TRANSIENT SCHEMATIC DIAGRAM FOR HPCI, RCIC, AND RWCU LINE BREAKS OUTSIDE CONTAINMENT

Updated FSAR
Revision 2, April 11, 1990

Sheet 1 of 1
Fig. 3.6-17

NOTES:

1. (C) – CIRCUMFERENTIAL BREAK
2. (TE) – TERMINAL END
3. PR – PIPE WHIP RESTRAINT
4. ▬ – LOCATION OF PR
5. – BUMPER TYPE
6. ▲ – ANCHOR
7. X – BREAK LOCATION
8. * * – SHIMMING NOT REQUIRED PER DELETION OF ARBITRARY INTERMEDIATE BREAKS



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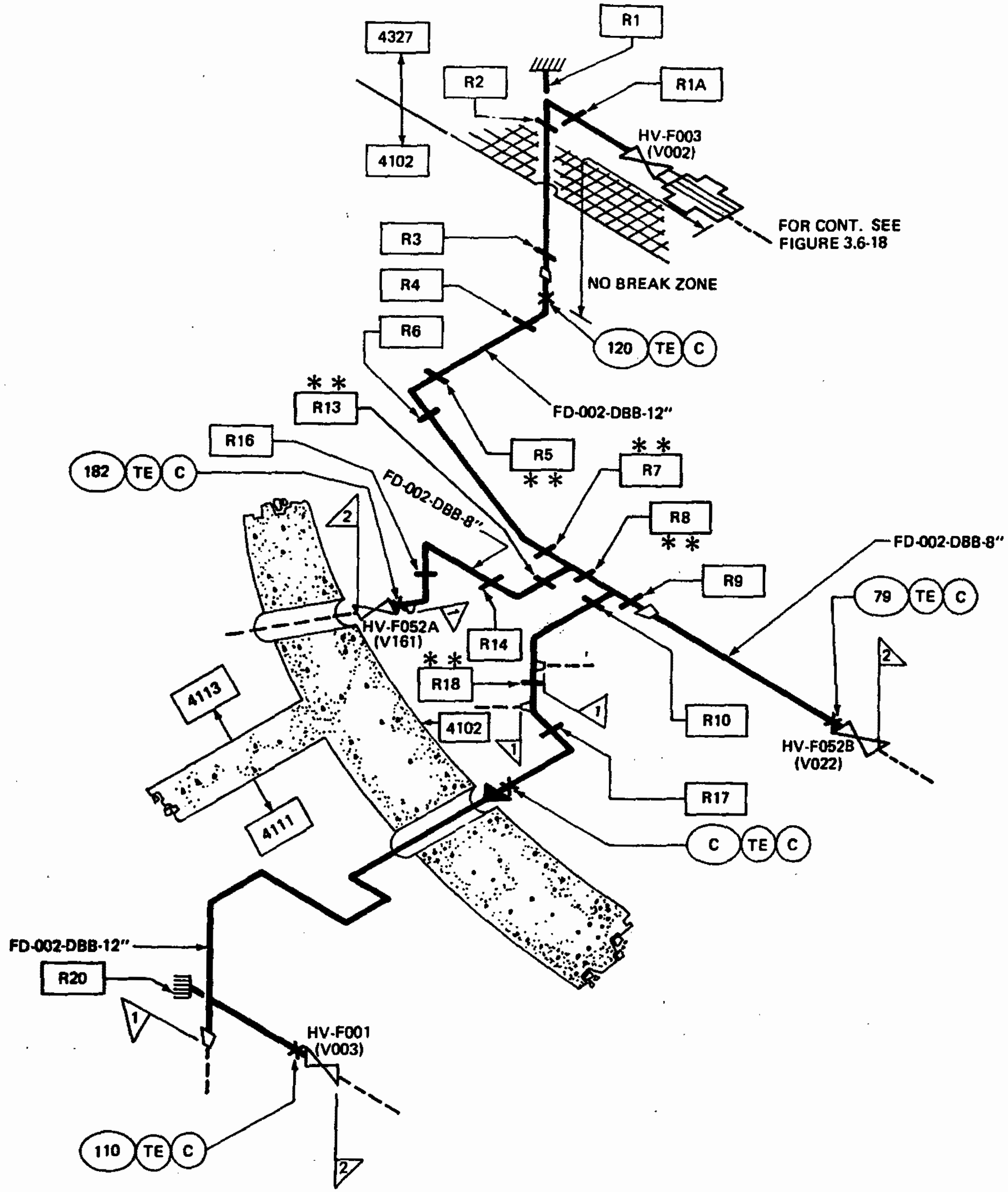
APRIL 11, 1988

PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

HPCI STEAM SUPPLY PIPING ISOMETRIC (PORTION INSIDE PRIMARY CONTAINMENT)

UPDATED FSAR

FIGURE 3.6-18



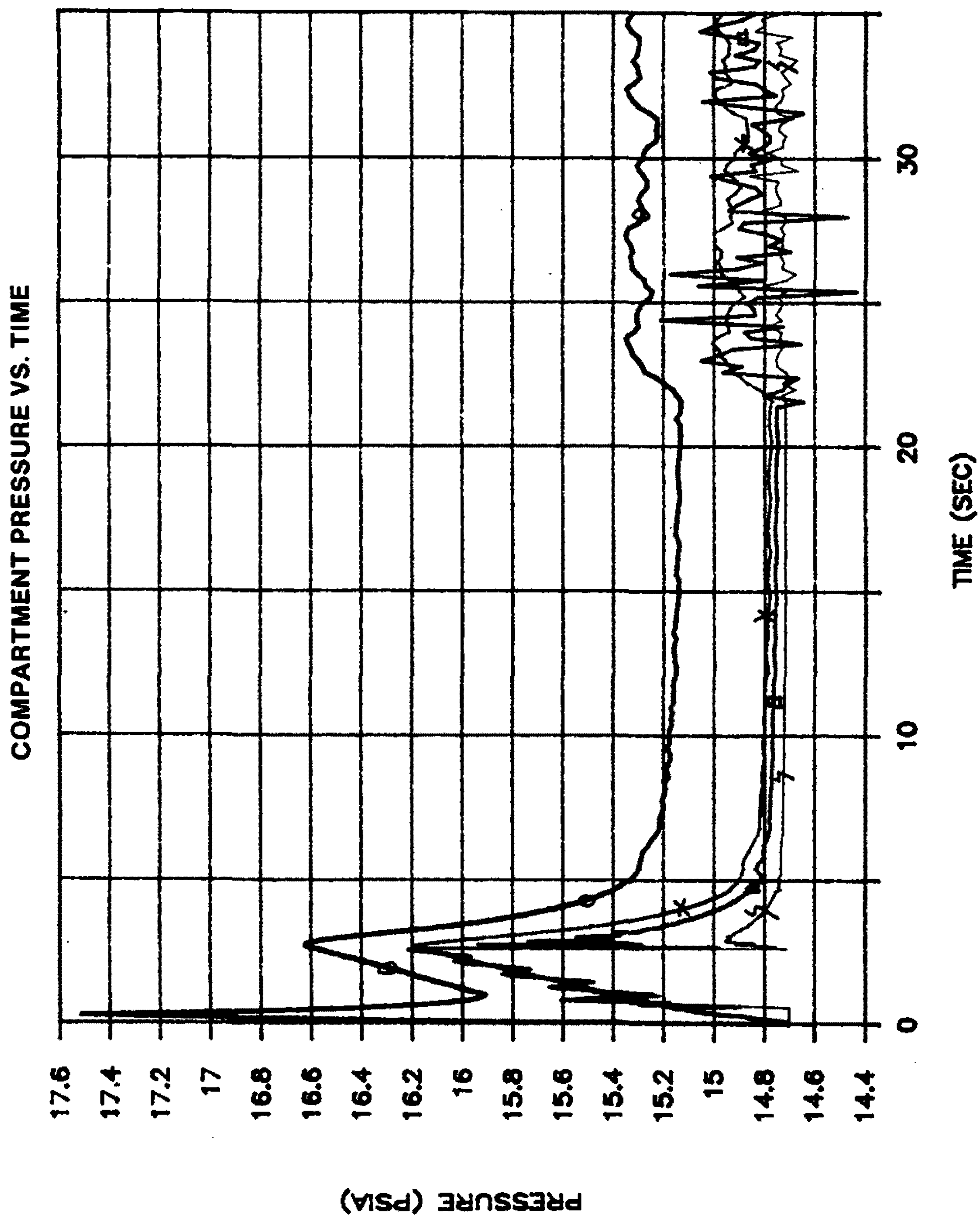
NOTES:

1. (TE) TERMINAL END
2. X BREAK LOCATION
3. XXXX ROOM NUMBER
4. 1 PIPING BEYOND THIS POINT IS 1" NOMINAL PIPE SIZE
5. 2 PIPING BEYOND THIS POINT IS MODERATE ENERGY
6. ▲ ANCHOR
7. (C) CIRCUMFERENTIAL BREAK
8. R PIPE WHIP RESTRAINT
9. BUMPER TYPE RESTRAINT
10. PIPE WHIP LOCATIONS
11. * * SHIMMING NOT REQUIRED PER DELETION OF ARBITRARY INTERMEDIATE BREAKS

PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK GENERATING STATION

HCPI STEAM SUPPLY PIPING ISOMETRIC (PORTION OUTSIDE PRIMARY CONTAINMENT)

UPDATED FSAR
REVISION 1, APRIL 11, 1989 FIGURE 3.6-19

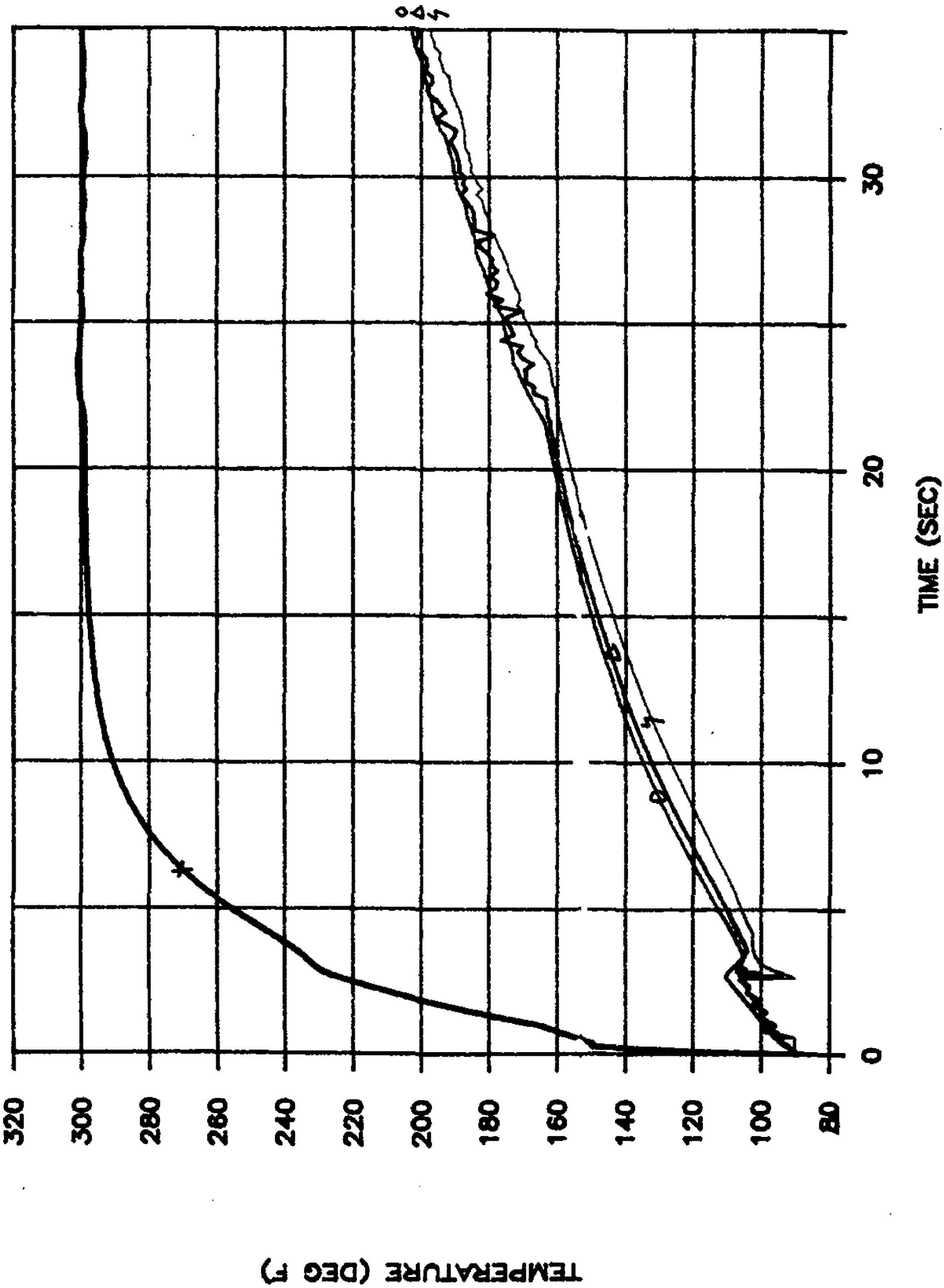


PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK GENERATING STATION

PRESSURE TRANSIENT ANALYSIS FOR A HPCI STEAM SUPPLY LINE BREAK IN THE HPCI PUMP ROOM

UPDATED FSAR
REVISION 1, APRIL 11, 1989 FIGURE 3.6-20

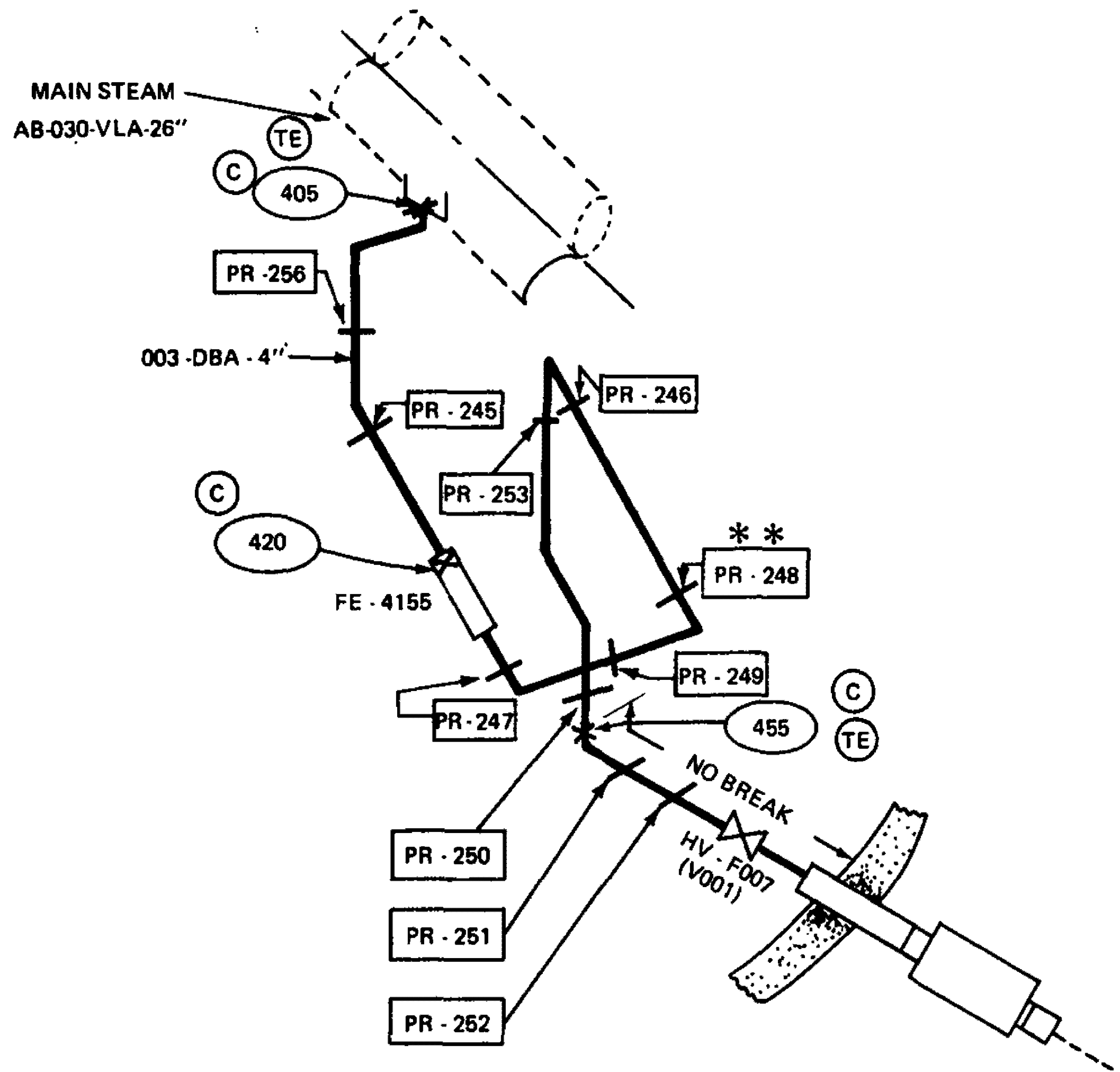
COMPARTMENT TEMPERATURE VS. TIME



PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK GENERATING STATION

TEMPERATURE TRANSIENT ANALYSIS FOR HPCI STEAM SUPPLY LINE BREAK IN THE HPCI PUMP ROOM

UPDATED FSAR
REVISION 1, APRIL 11, 1989 FIGURE 3.6-21



NOTES:

1. (C) – CIRCUMFERENTIAL BREAK
2. (TE) – TERMINAL END
3. PR – PIPE WHIP RESTRAINT
4. ▬ – LOCATION OF PR
5. – BUMPER TYPE
6. ▲ – ANCHOR
7. X – BREAK LOCATION
8. * * – SHIMMING NOT REQUIRED PER DELETION OF ARBITRARY INTERMEDIATE BREAKS

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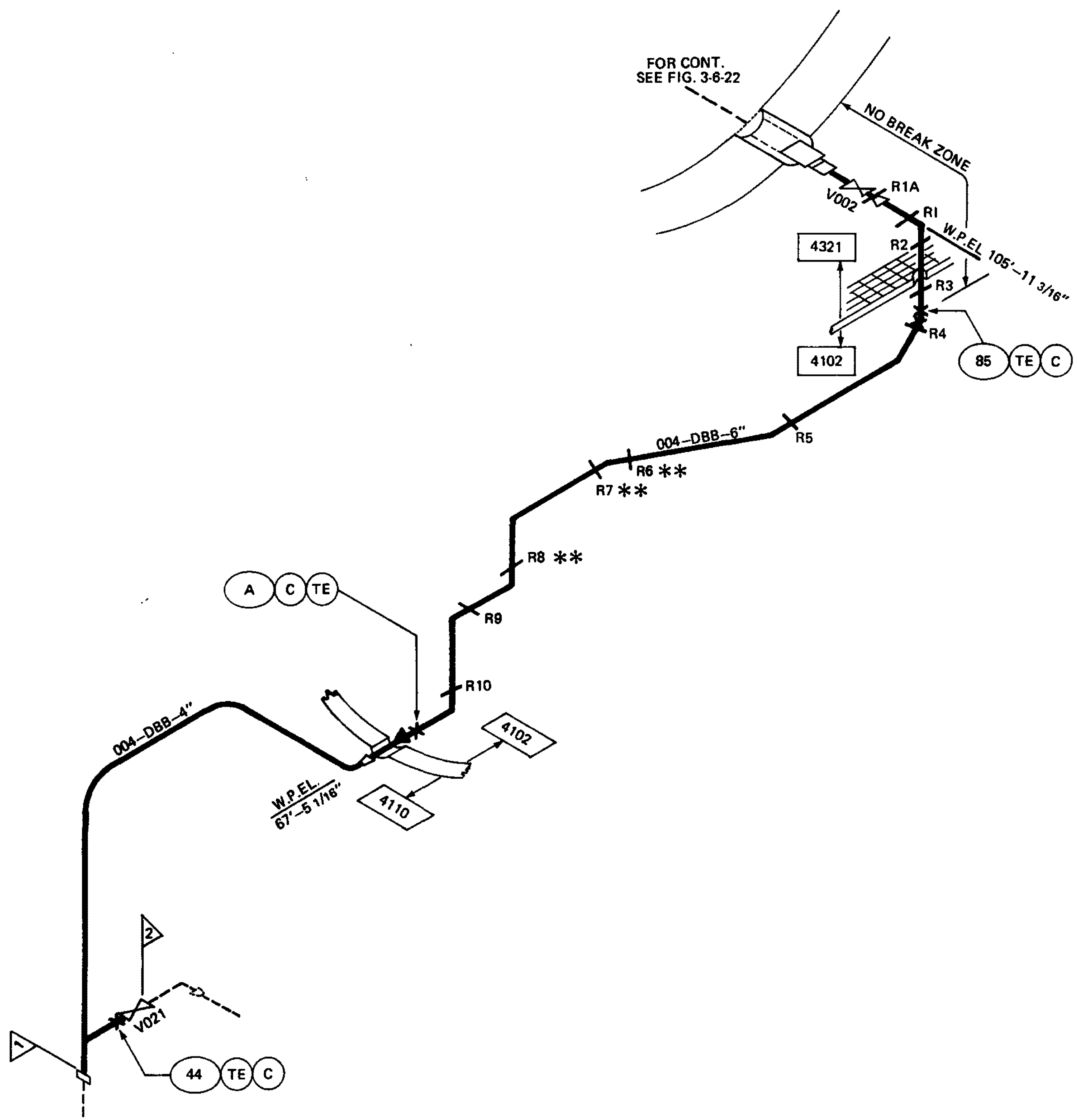
APRIL 11, 1988

PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

RCIC STEAM SUPPLY PIPING
ISOMETRIC (PORTION INSIDE
PRIMARY CONTAINMENT)

UPDATED FSAR

FIGURE 3.6-22



NOTES:

1. –R– PIPE WHIP RESTRAINT
2. (C) CIRCUMFERENTIAL BREAK
3. (TE) TERMINAL END
4. —— LOCATION OF RESTRAINT
5. X BREAK LOCATION
6. ▲ ANCHOR
7. [flag 1] PIPING BEYOND THIS POINT IS 1" NOMINAL PIPE DIAMETER
8. [flag 2] PIPING BEYOND THIS POINT IS MODERATE ENERGY
9. DELETED
10. [XXXX] DENOTES ROOM NUMBER
11. ✱✱ SHIMMING NOT REQUIRED PER DELETION OF ARBITRARY INTERMEDIATE BREAKS

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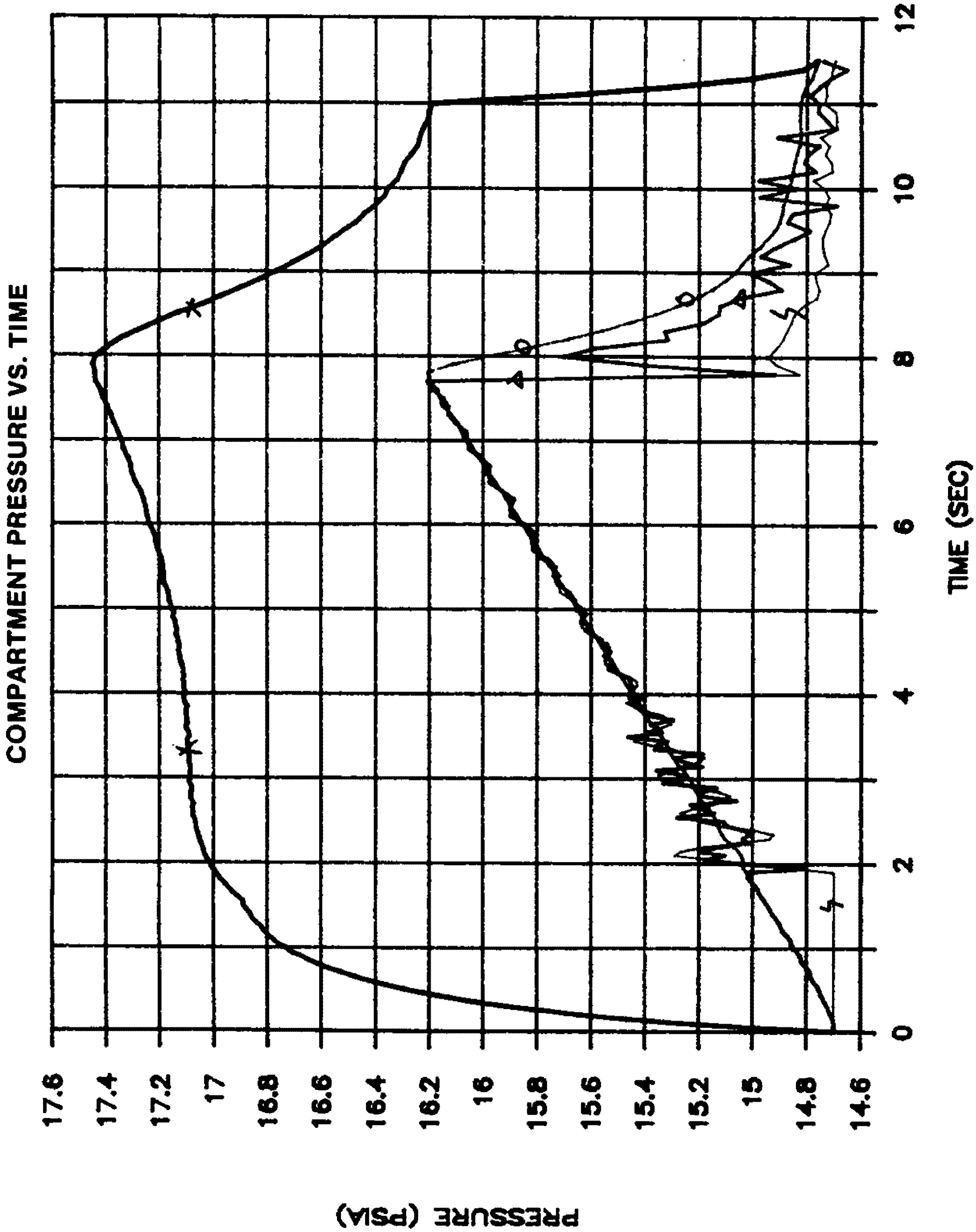
APRIL 11, 1988

PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

RCIC STEAM SUPPLY PIPING ISOMETRIC (PORTION OUTSIDE PRIMARY CONTAINMENT)

UPDATED FSAR

FIGURE 3.6-23

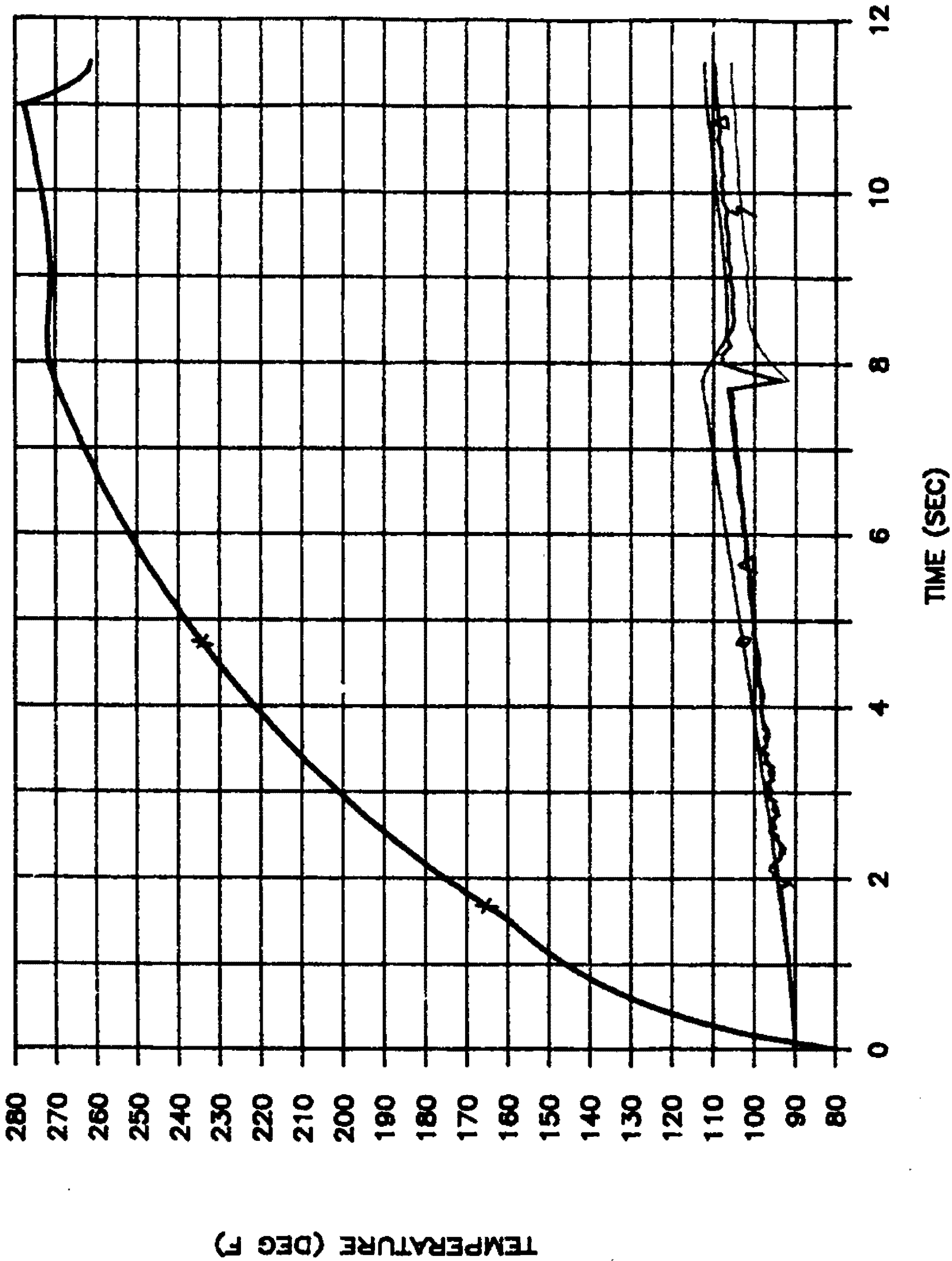


PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK GENERATING STATION

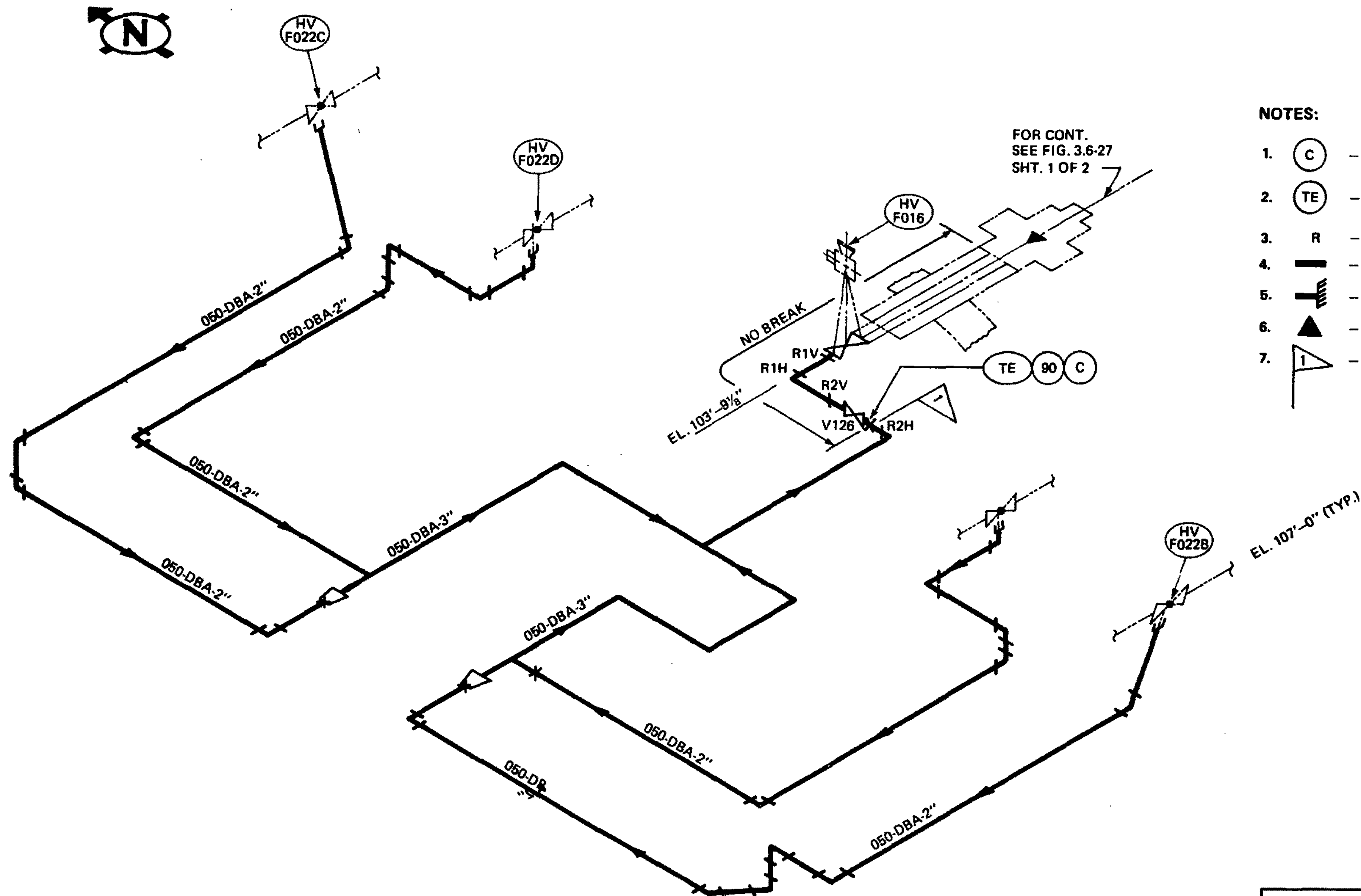
PRESSURE TRANSIENT ANALYSIS FOR A RCIC STEAM SUPPLY LINE BREAK IN THE RCIC PUMP ROOM

UPDATED FSAR
REVISION 1, APRIL 11, 1989 FIGURE 3.6-24

COMPARTMENT TEMPERATURE VS. TIME



PUBLIC SERVICE ELECTRIC AND GAS COMPANY HOPE CREEK GENERATING STATION
TEMPERATURE TRANSIENT ANALYSIS FOR RCIC STEAM SUPPLY LINE BREAK IN THE RCIC PUMP ROOM
UPDATED FSAR REVISION 1, APRIL 11, 1989 FIGURE 3.6-25



NOTES:

1. (C) – CIRCUMFERENTIAL BREAK
2. (TE) – TERMINAL END
3. R – PIPE WHIP RESTRAINT
4. – LOCATION OF RESTRAINT
5. – BUMPER TYPE
6. – ANCHOR
7. – BREAKS POSTULATED AT EVERY FITTING

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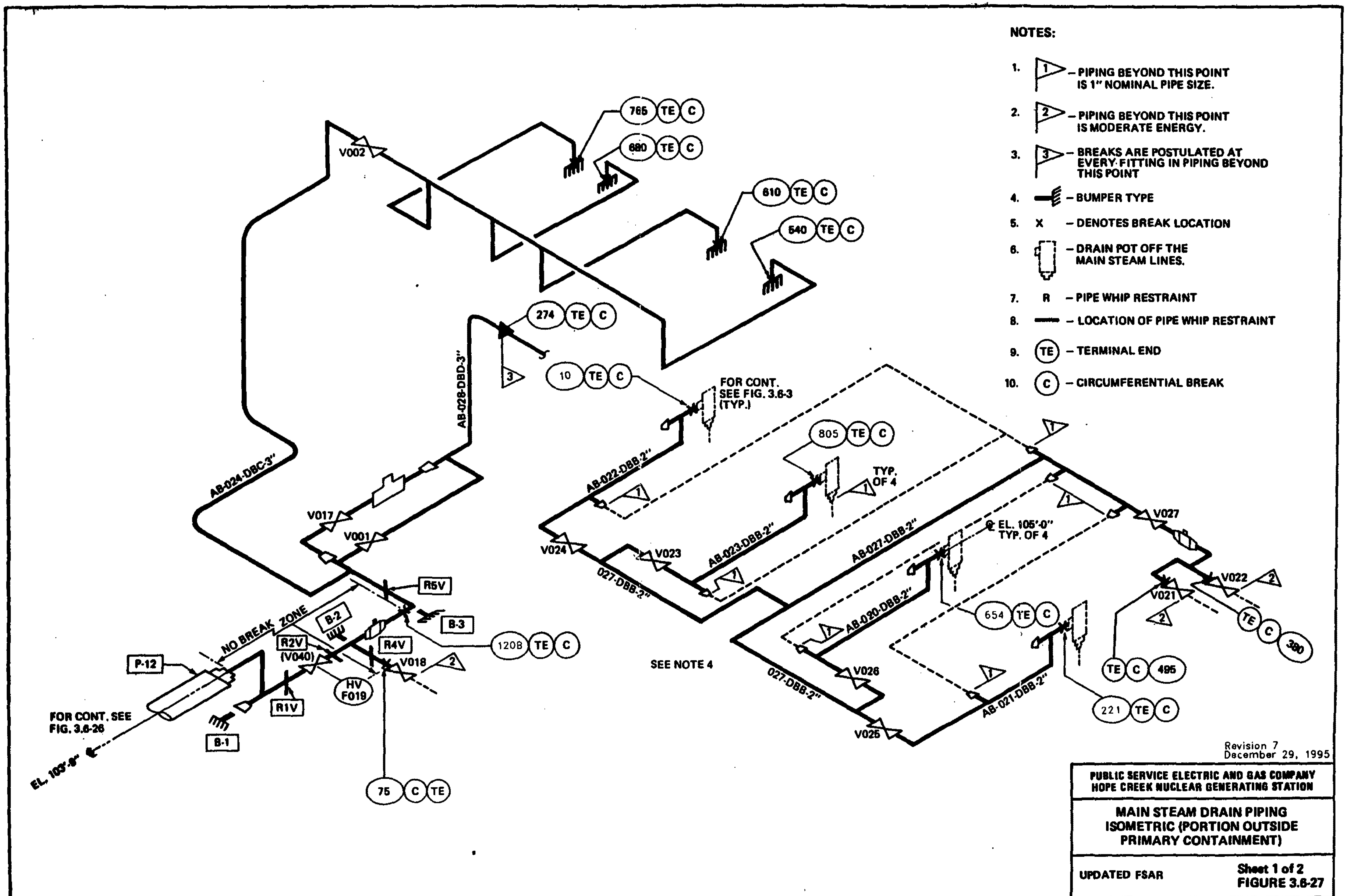
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

MAIN STEAM DRAIN PIPING ISOMETRIC (PORTION INSIDE PRIMARY CONTAINMENT)

UPDATED FSAR

FIGURE 3.6-26



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December 29, 1995

PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

MAIN STEAM DRAIN PIPING ISOMETRIC (PORTION OUTSIDE PRIMARY CONTAINMENT)

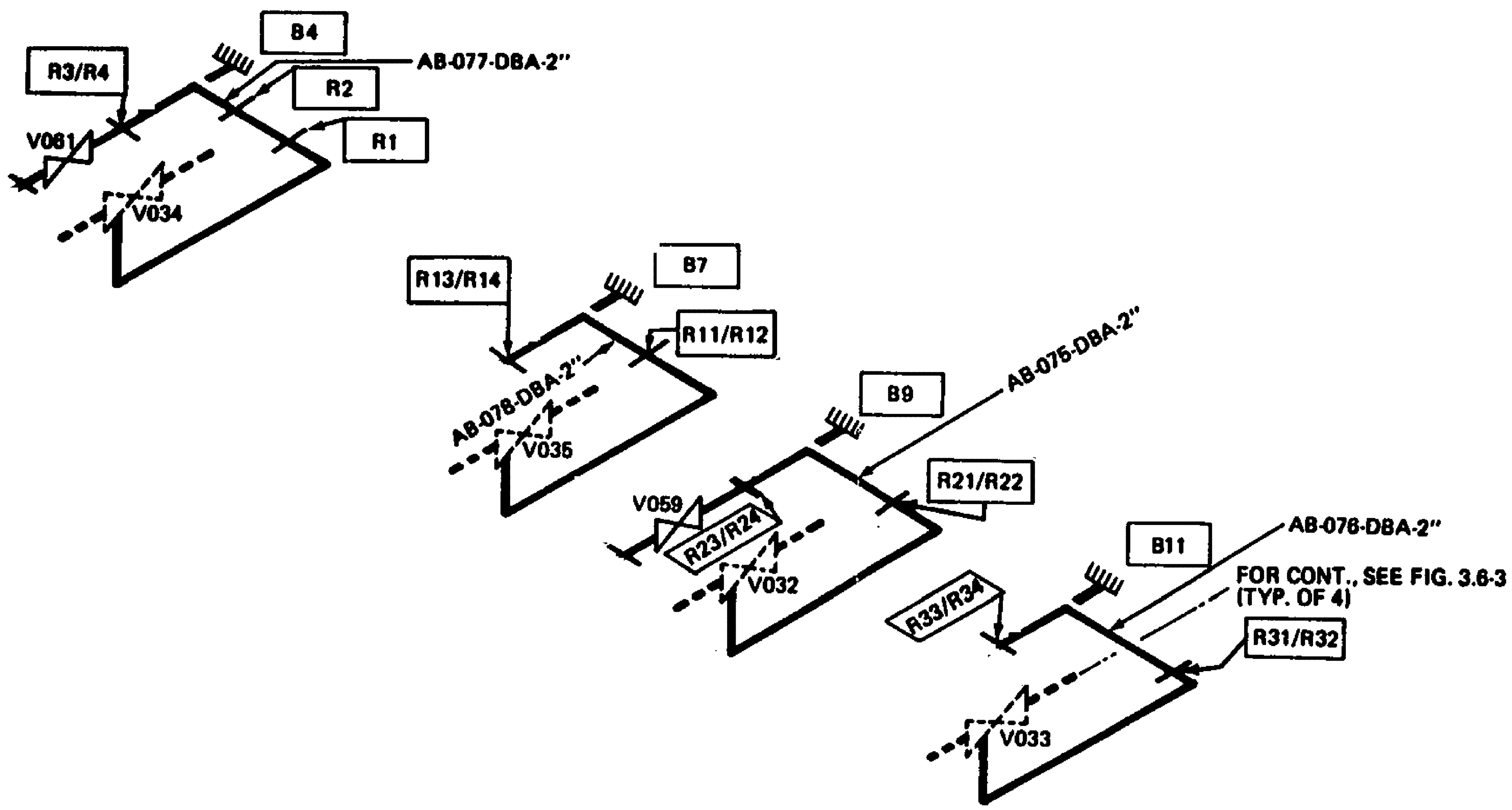
UPDATED FSAR

Sheet 1 of 2

FIGURE 3.6-27

NOTES:

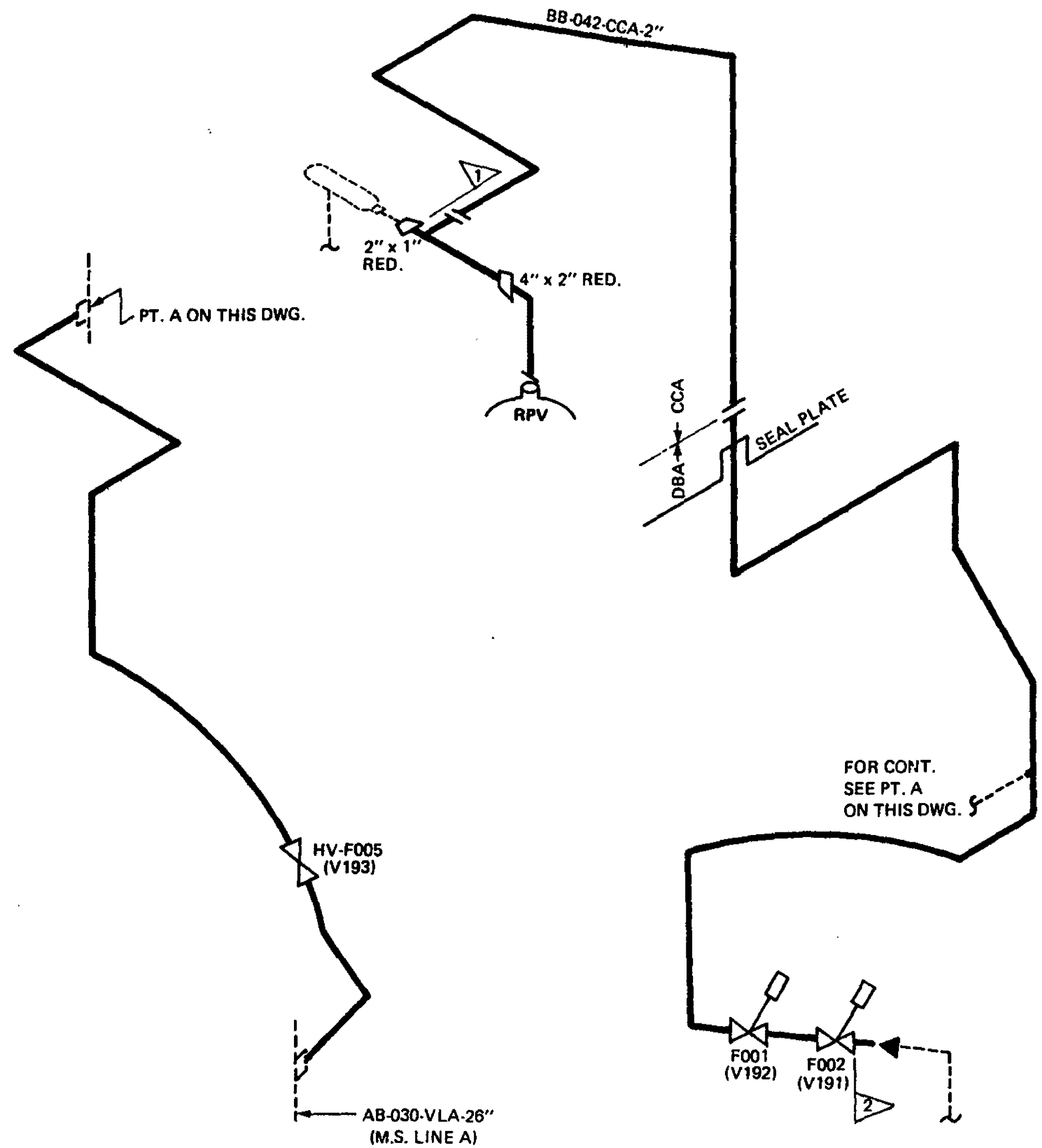
1. B – BUMPER RESTRAINT
2. R – PIPE WHIP RESTRAINT
3. (TE) – TERMINAL END
4. (C) – CIRCUMFERENTIAL BREAK
5. ▬ – LOCATION OF PIPE WHIP RESTRAINT
6. X – BREAK LOCATION
7. 1 – PIPING BEYOND THIS POINT IS 1" NOMINAL PIPE SIZE
8. 2 – PIPING BEYOND THIS POINT IS MODERATE ENERGY.
9. 3 – BREAKS ARE POSTULATED AT EVERY FITTING IN PIPING BEYOND THIS POINT



PSEG NUCLEAR, L.L.C.
HOPE CREEK GENERATING STATION

MAIN STEAM DRAIN PIPING ISOMETRIC
(PORTION OUTSIDE PRIMARY CONTAINMENT)

Updated FSAR Sheet 2 of 2
Revision 12, May 3, 2002 Fig.3.6-27



NOTES:

1) BREAKS POSTULATED AT EVERY FITTING AND CHANGE OF DIRECTION.

2) ▷1 PIPING BEYOND THIS IS 1" NOMINAL DIAMETER.

3) ▷2 PIPING BEYOND THIS IS MODERATE ENERGY.

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PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

RPV HEAD VENT PIPING ISOMETRIC

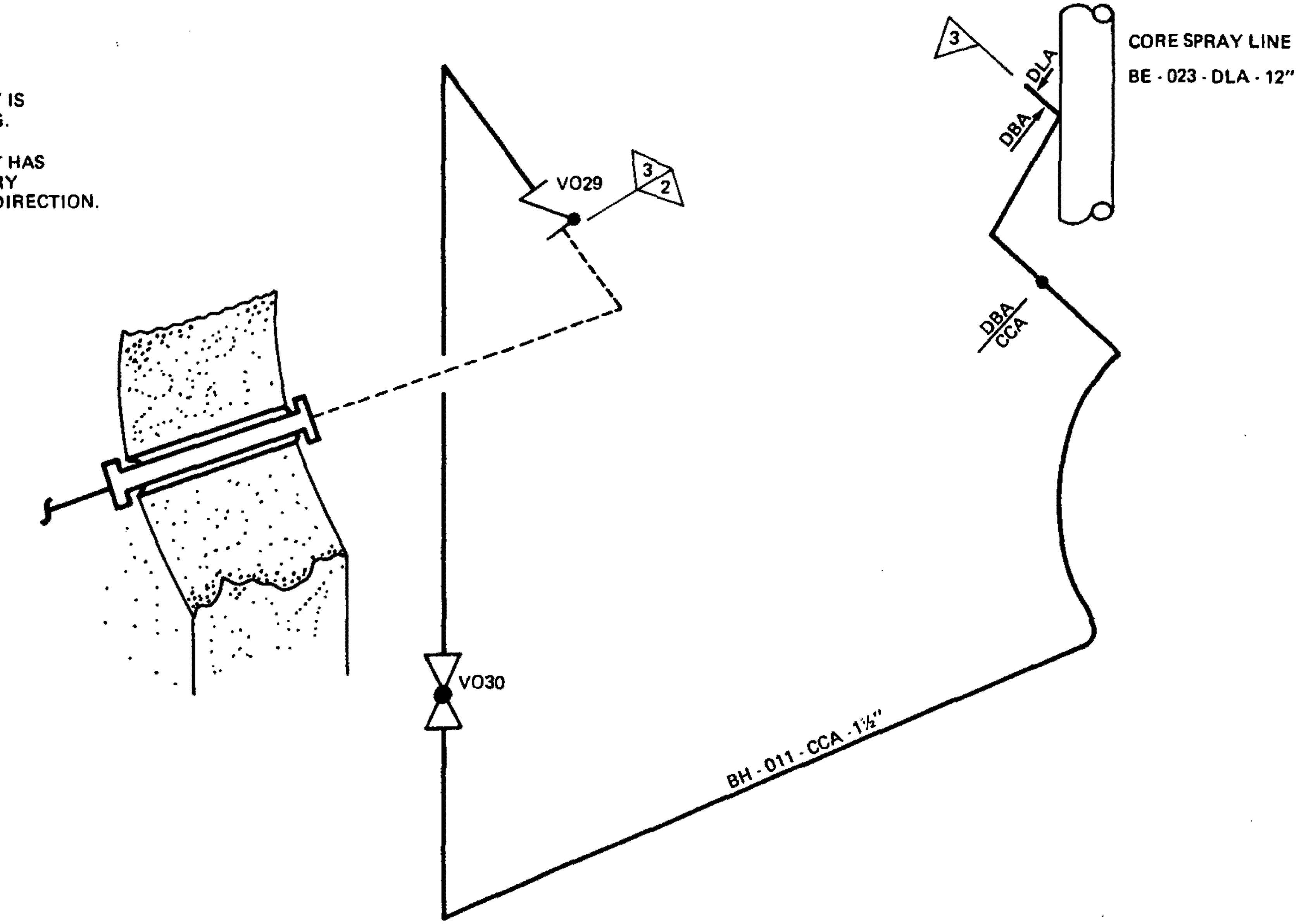
UPDATED FSAR

FIGURE 3.6-28

NOTES:

1) 2 PIPING BEYOND THIS POINT IS MODERATE ENERGY PIPING.

2) 3 PIPING BEYOND THIS POINT HAS BREAKS ASSUMED AT EVERY FITTING AND CHANGE OF DIRECTION.



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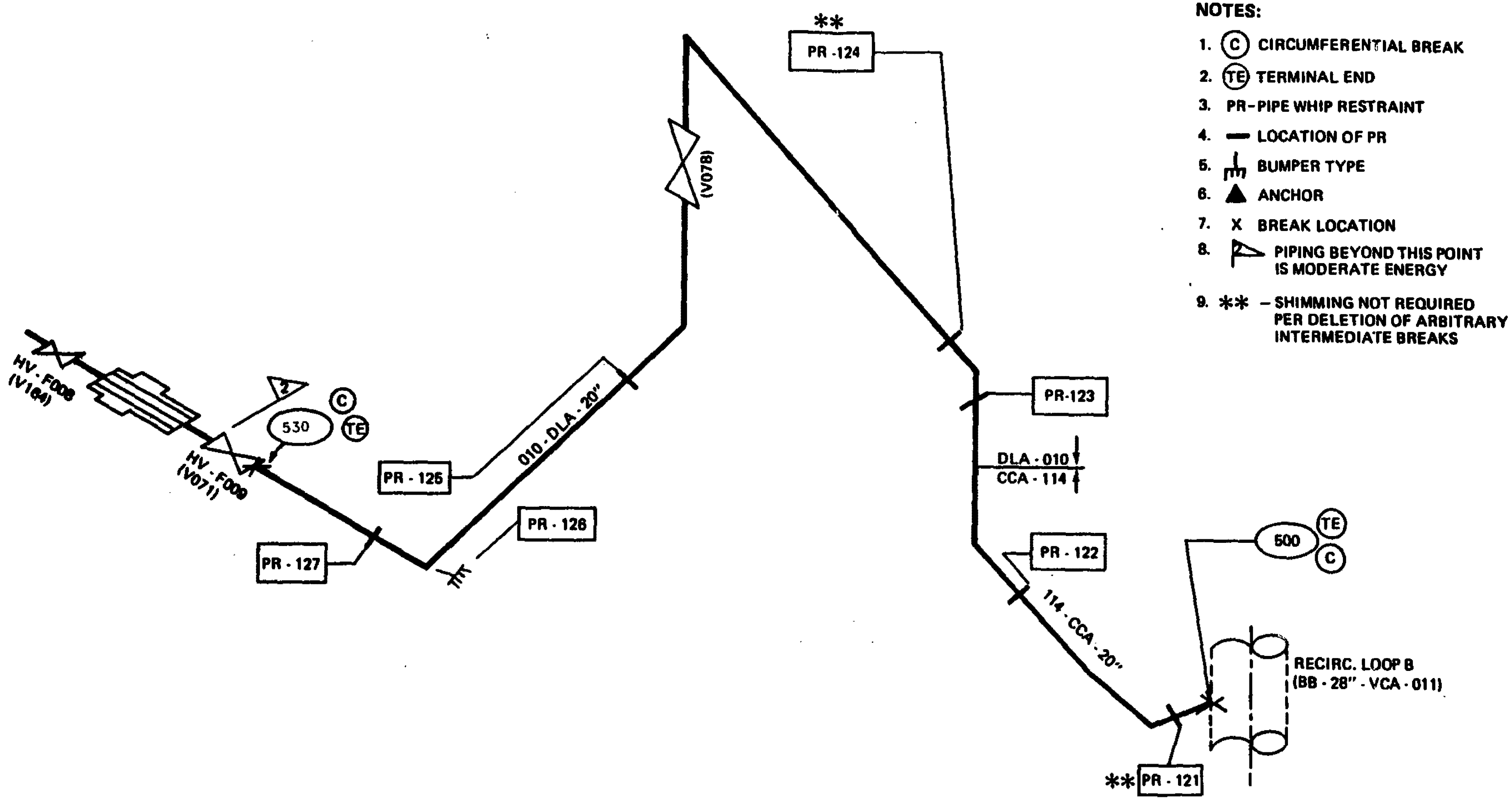
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

STANDBY LIQUID CONTROL INJECTION PIPING ISOMETRIC

UPDATED FSAR

FIGURE 3.6-29



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December 29, 1995

PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

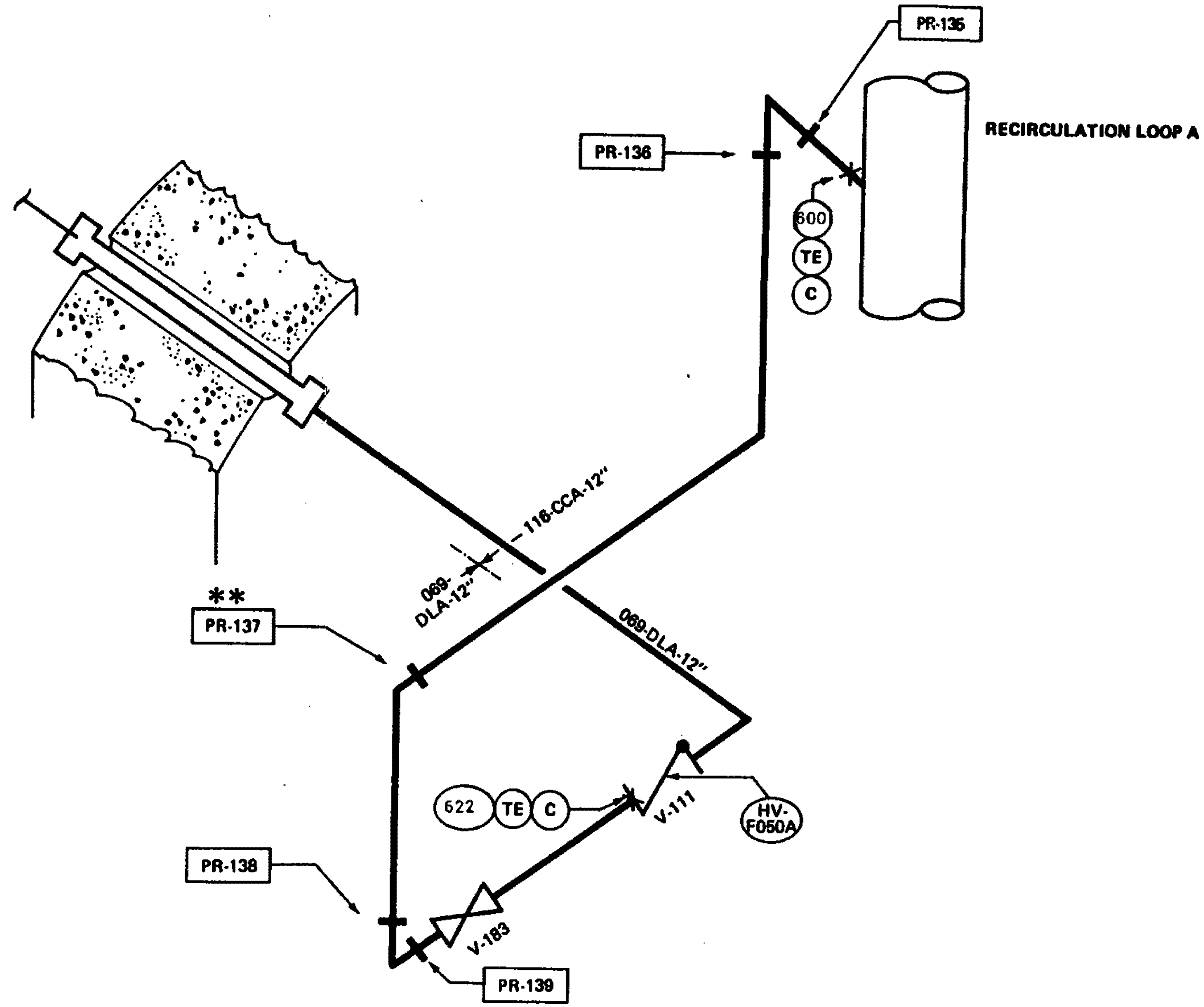
RHR SHUTDOWN COOLING
SUCTION PIPING ISOMETRIC

UPDATED FSAR

FIGURE 3.6-30

NOTES:

1. (C) – CIRCUMFERENTIAL BREAK
2. (TE) – TERMINAL END
3. PR – PIPE WHIP RESTRAINT
4. ▬ – LOCATION OF PR
5. X – BREAK LOCATION
6. ** – SHIMMING NOT REQUIRED PER DELETION OF ARBITRARY INTERMEDIATE BREAKS



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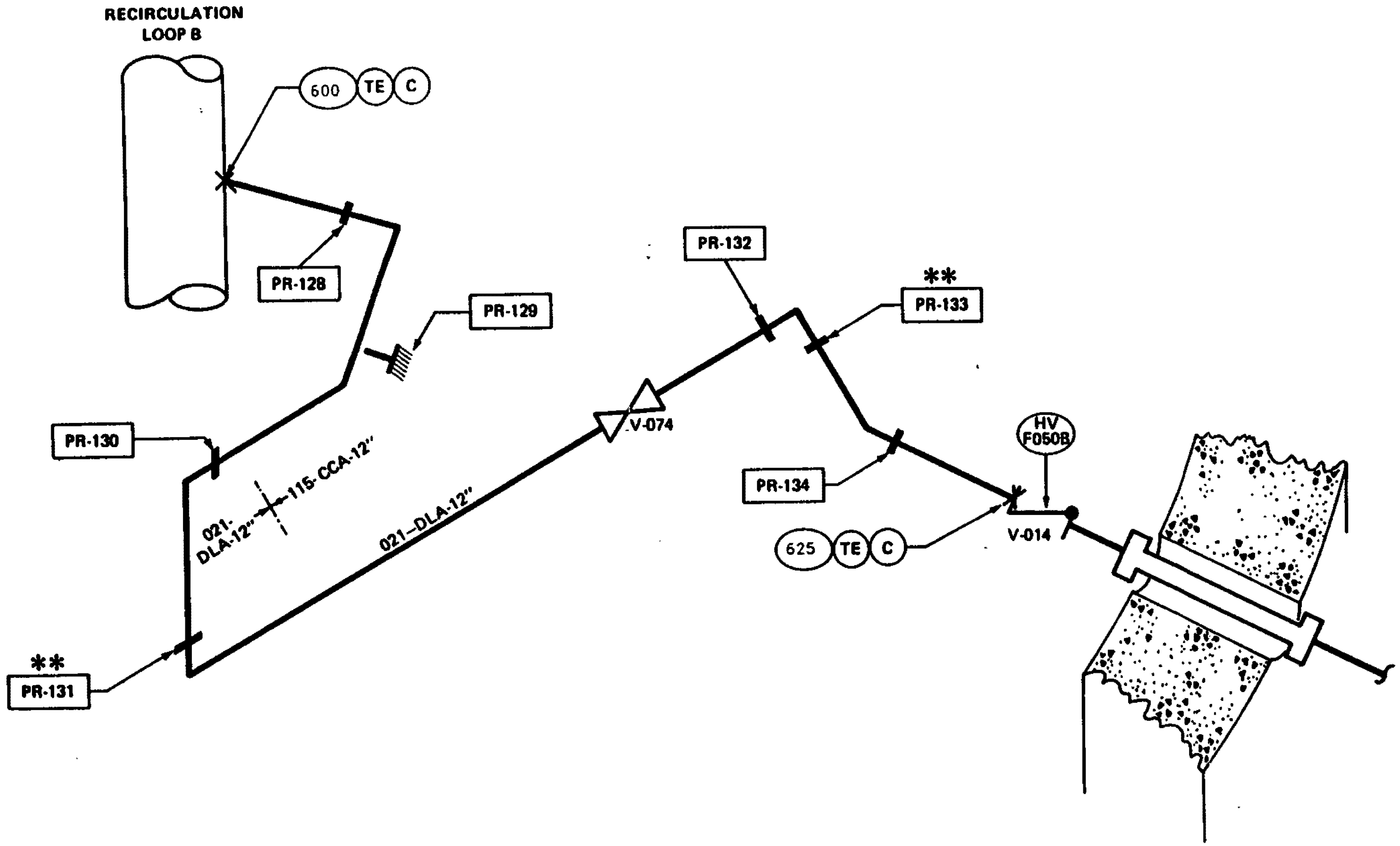
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

RHR SHUTDOWN COOLING RETURN PIPING ISOMETRIC LOOP A

UPDATED FSAR

Sheet 1 of 2
FIGURE 3.6-31



NOTES:

1. (C) – CIRCUMFERENTIAL BREAK
2. (TE) – TERMINAL END
3. PR – PIPE WHIP RESTRAINT
4. ▬ – LOCATION OF PR
5. X – BREAK LOCATION
6. ** – SHIMMING NOT REQUIRED PER DELETION OF ARBITRARY INTERMEDIATE BREAKS

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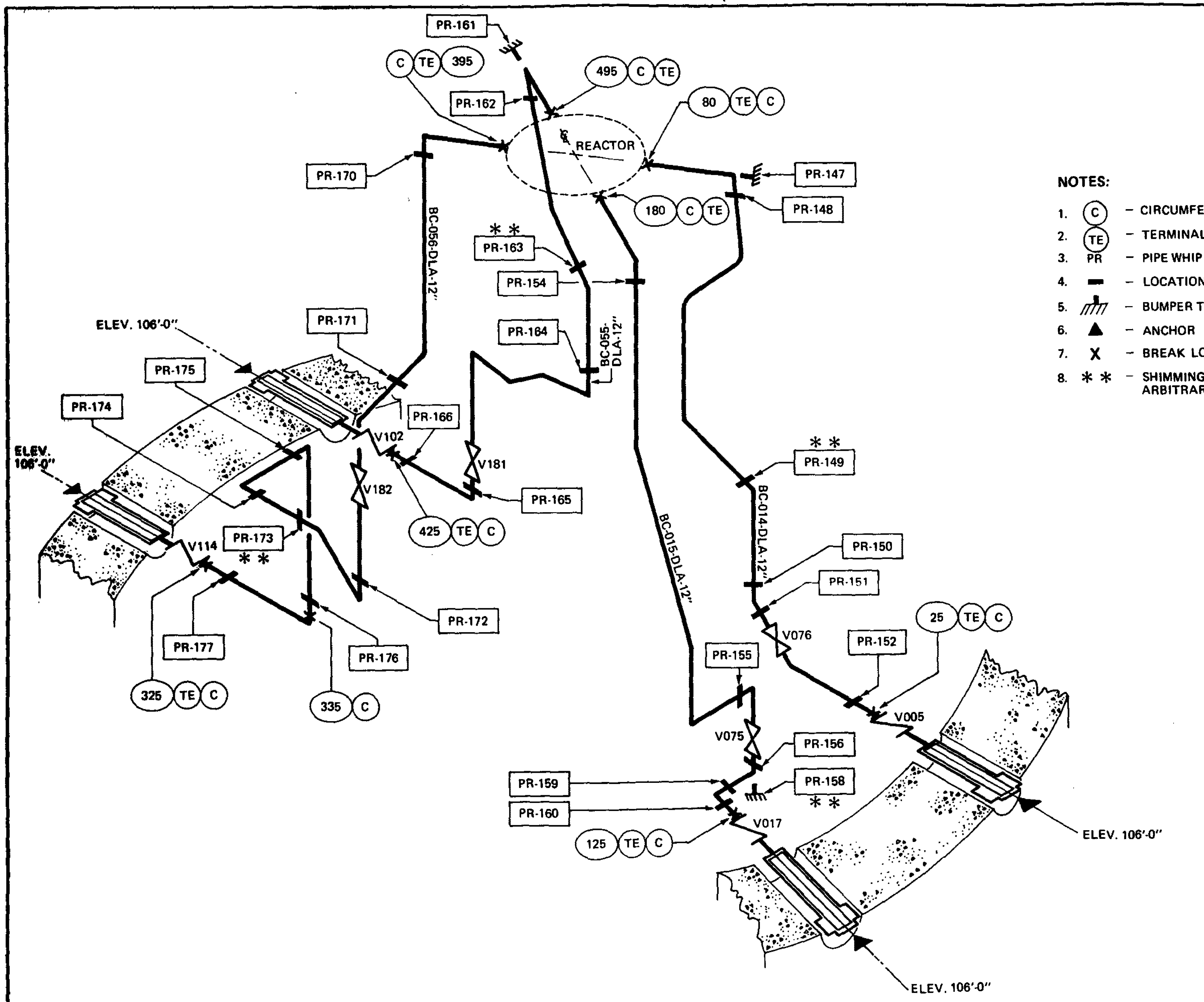
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

RHR SHUTDOWN COOLING
RETURN PIPING ISOMETRIC
LOOP B

UPDATED FSAR

Sheet 2 of 2
FIGURE 3.6-31



NOTES:

1. (C) – CIRCUMFERENTIAL BREAK
2. (TE) – TERMINAL END
3. PR – PIPE WHIP RESTRAINT
4. ▬ – LOCATION OF PR
5. – BUMPER TYPE
6. ▲ – ANCHOR
7. X – BREAK LOCATION
8. * * – SHIMMING NOT REQUIRED PER DELETION OF ARBITRARY INTERMEDIATE BREAKS

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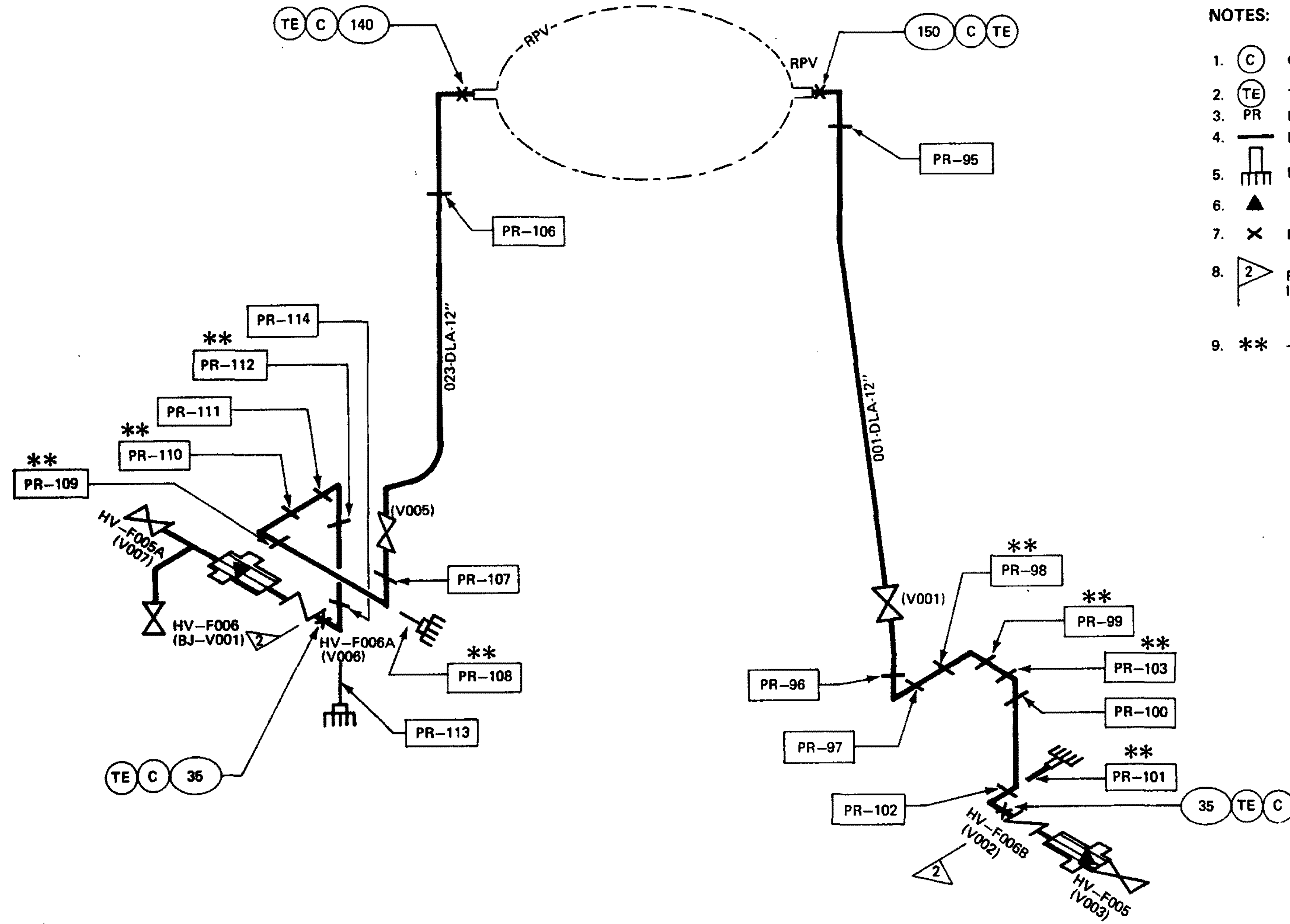
HOPE CREEK NUCLEAR GENERATING STATION

LPCI INJECTION PIPING ISOMETRIC

UPDATED FSAR

FIGURE 3.6-32



NOTES:

1. (C) CIRCUMFERENTIAL BREAK
2. (TE) TERMINAL END
3. PR PIPE WHIP RESTRAINT
4. LOCATION OF PR
5. BUMPER TYPE
6. ▲ ANCHOR
7. ✕ BREAK LOCATION
8. 2 PIPING BEYOND THIS POINT IS MODERATE ENERGY.
9. ** – SHIMMING NOT REQUIRED PER DELETION OF ARBITRARY INTERMEDIATE BREAKS

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CORE SPRAY INJECTION
PIPING ISOMETRIC
INSIDE CONTAINMENT

UPDATED FSAR

FIGURE 3.6-33

REACTOR VESSEL DRAIN PIPING ISOMETRIC INCLUDED AS PART RWCU PIPING ISOMETRIC (PORTION INSIDE PRIMARY CONTAINMENT), FIGURE 3.6-15

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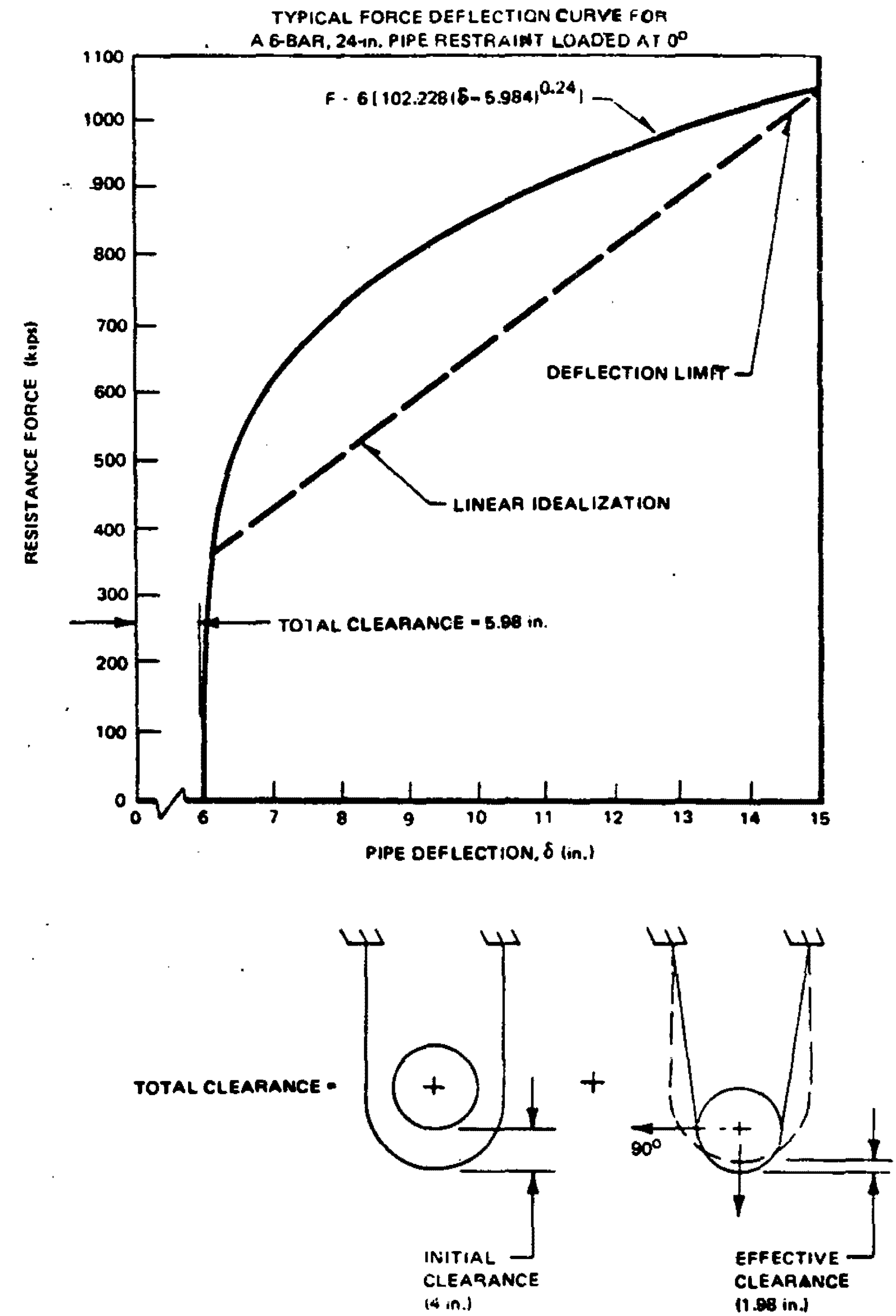
PUBLIC SERVICE ELECTRIC AND GAS COMPANY HOPE CREEK NUCLEAR GENERATING STATION	
REACTOR VESSEL DRAIN PIPING ISOMETRIC	
UPDATED FSAR	FIGURE 3.6-34

THIS FIGURE HAS BEEN DELETED

PSEG NUCLEAR L.L.C. HOPE CREEK GENERATING STATION	
HOPE CREEK UFSAR - REV 14	SHEET 1 OF 1
July 26, 2005	F3.6-35



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PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

TYPICAL FORCE-DEFLECTION CURVE FOR RECIRCULATION SYSTEM PIPE WHIP RESTRAINT

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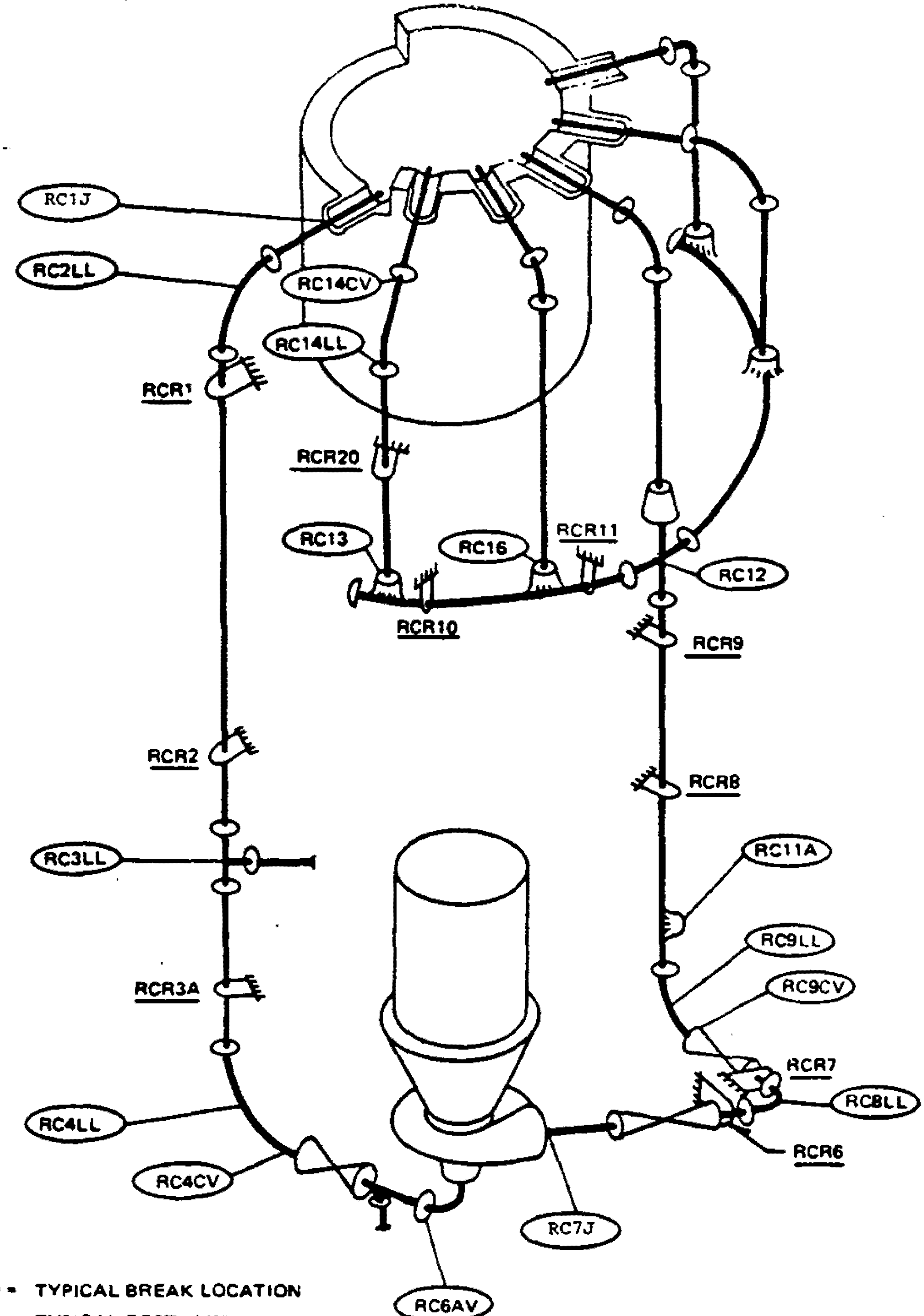
FIGURE 3.6-36



KEY:
RC4CV - TYPICAL BREAK LOCATION
RCR3A - TYPICAL RESTRAINT DESIGNATION

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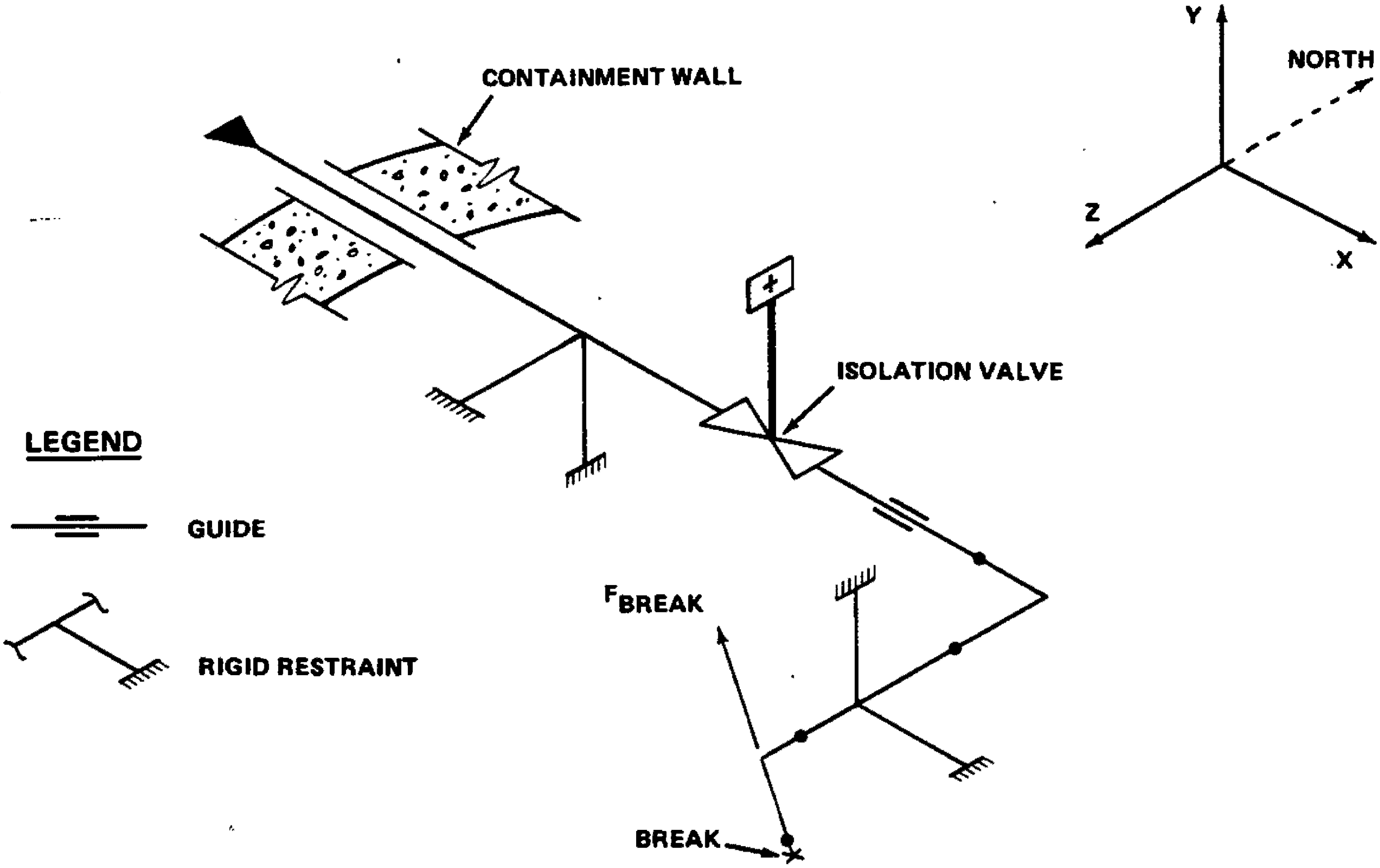
December 29, 1995

PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

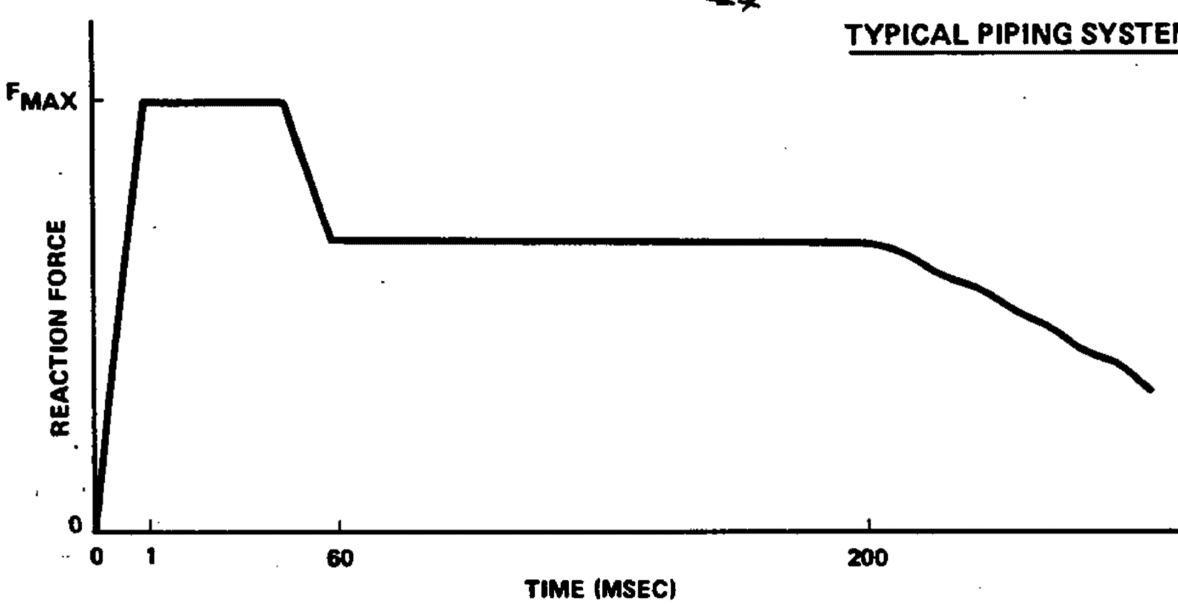
BREAK LOCATIONS AND RESTRAINTS ANALYZED, PDA VERIFICATION PROGRAM

UPDATED FSAR

FIGURE 3.6-37



TYPICAL PIPING SYSTEM MODEL



TYPICAL PIPE BREAK FORCING FUNCTION

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PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

TYPICAL PIPE BREAK VALVE OPERABILITY ANALYSIS MODELS

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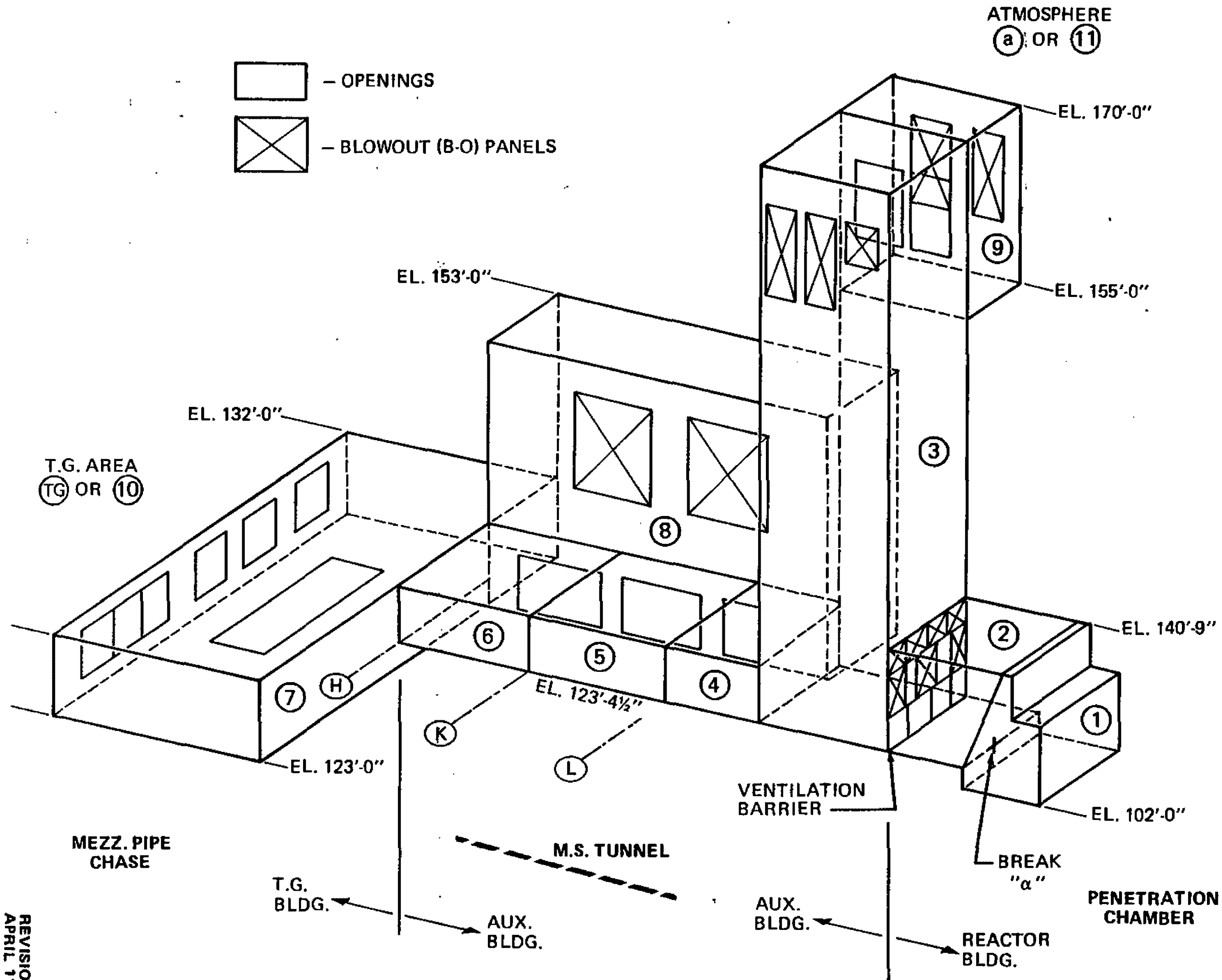
FIGURE 3.6-38



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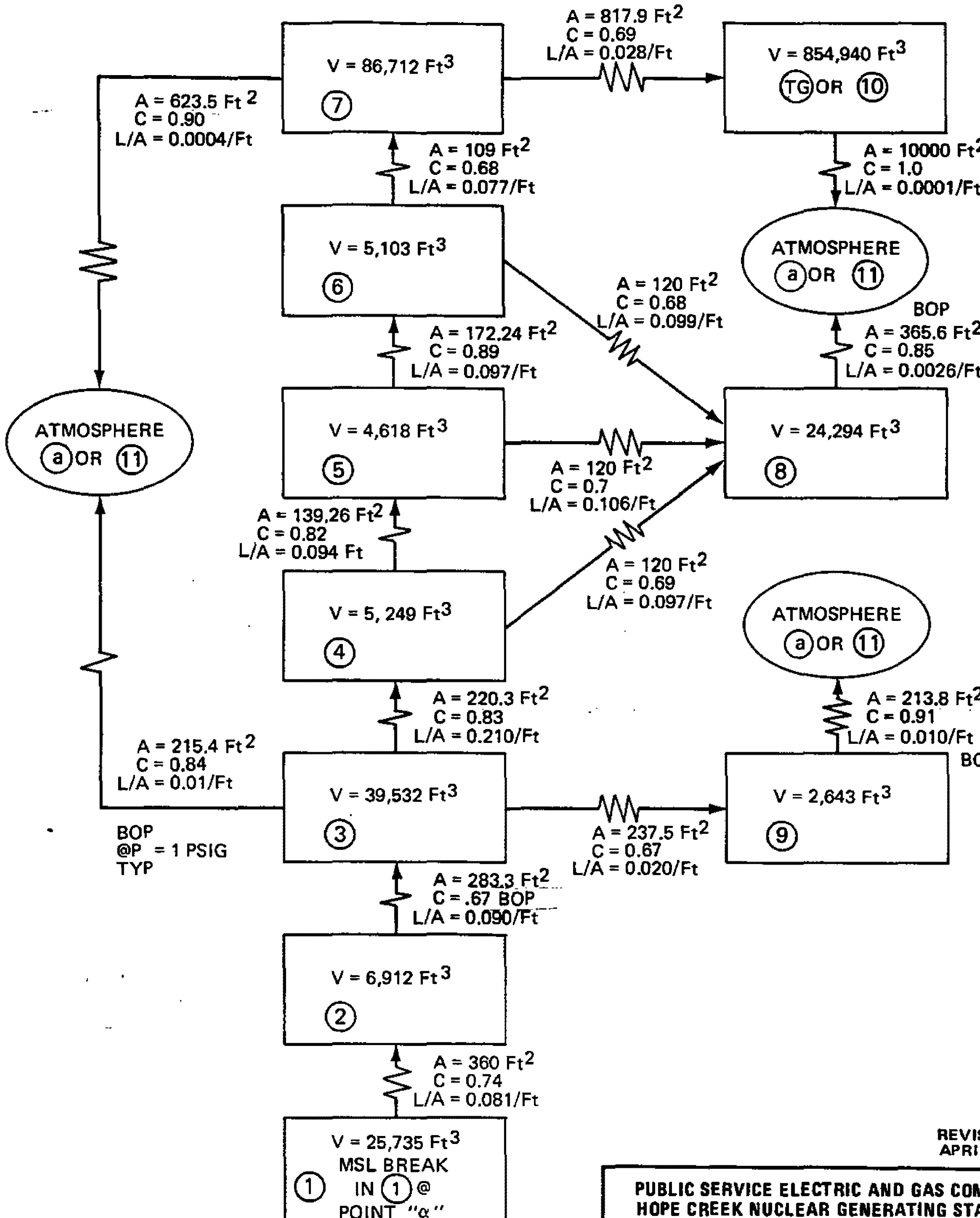
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

PRESSURE – TEMPERATURE TRANSIENT ANALYSIS MODEL FOR A MAIN STEAM LINE BREAK IN THE PENETRATION CHAMBER OF THE MAIN STEAM TUNNEL

UPDATED FSAR

FIGURE 3.6-39



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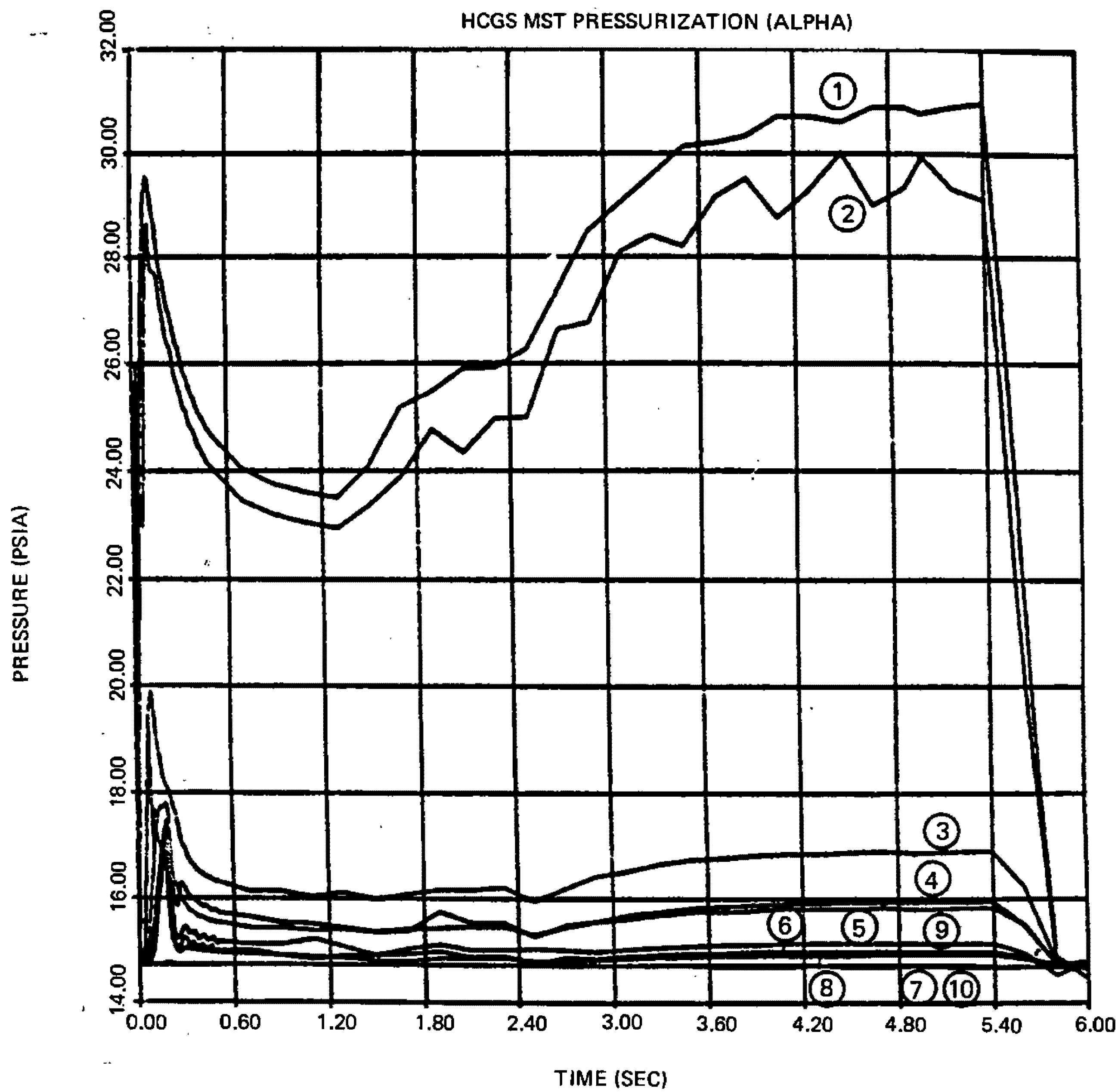
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

PRESSURE – TEMPERATURE
SCHEMATIC DIAGRAM FOR A MAIN STEAM
LINE BREAK IN THE PENETRATION
CHAMBER OF THE MAIN STEAM TUNNEL

UPDATED FSAR

FIGURE 3.6-40

COMPARTMENT PRESSURE vs. TIME



NOTE:

SEE FIGURE 3.6-41 FOR IDENTIFICATION OF NODES.

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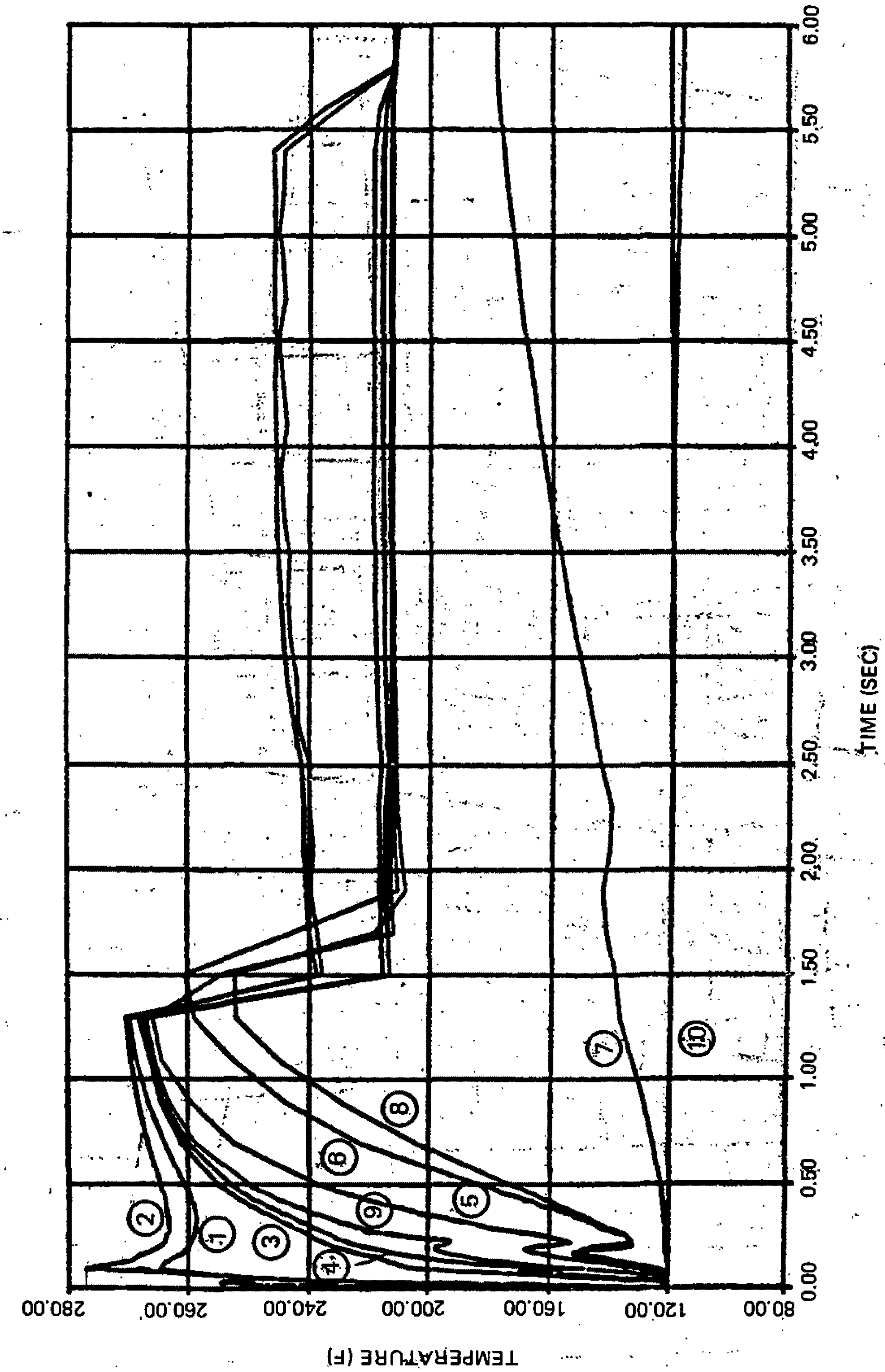
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

PRESSURE TRANSIENT ANALYSIS FOR A MAIN STEAM LINE BREAK IN THE PENETRATION CHAMBER OF THE MAIN STEAM TUNNEL

UPDATED FSAR

FIGURE 3.6-41

COMPARTMENT TEMPERATURE vs. TIME



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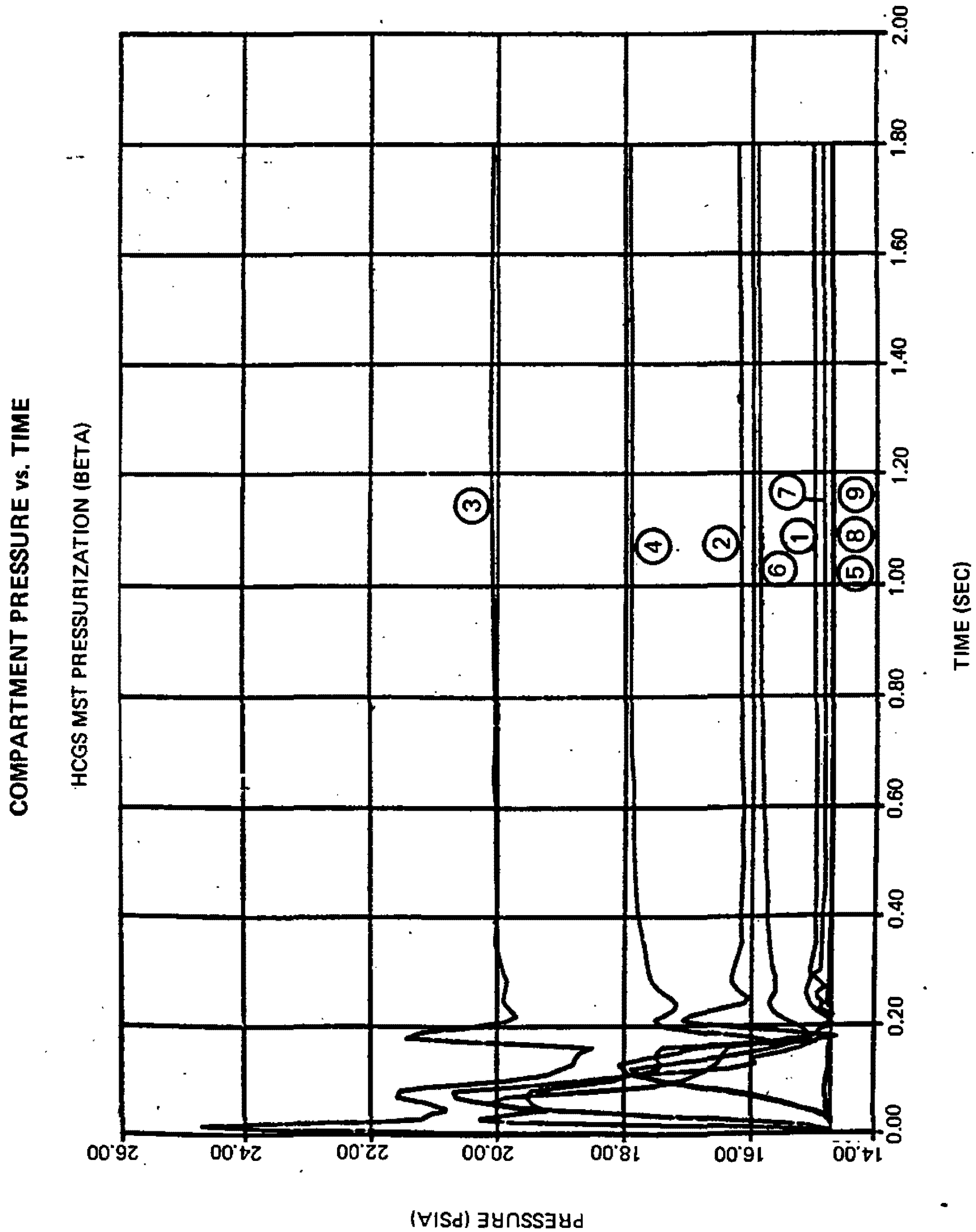
NOTE:

SEE FIGURE 3.6-41 FOR IDENTIFICATION OF NODES.

PUBLIC SERVICE ELECTRIC AND GAS COMPANY HOPE CREEK NUCLEAR GENERATING STATION
TEMPERATURE TRANSIENT ANALYSIS FOR A MAIN STEAM LINE BREAK IN THE PENETRATION CHAMBER OF THE MAIN STEAM TUNNEL
UPDATED FSAR — FIGURE 3.6-42



NOTE:

SEE FIGURE 3.6-9 FOR IDENTIFICATION OF NODES.

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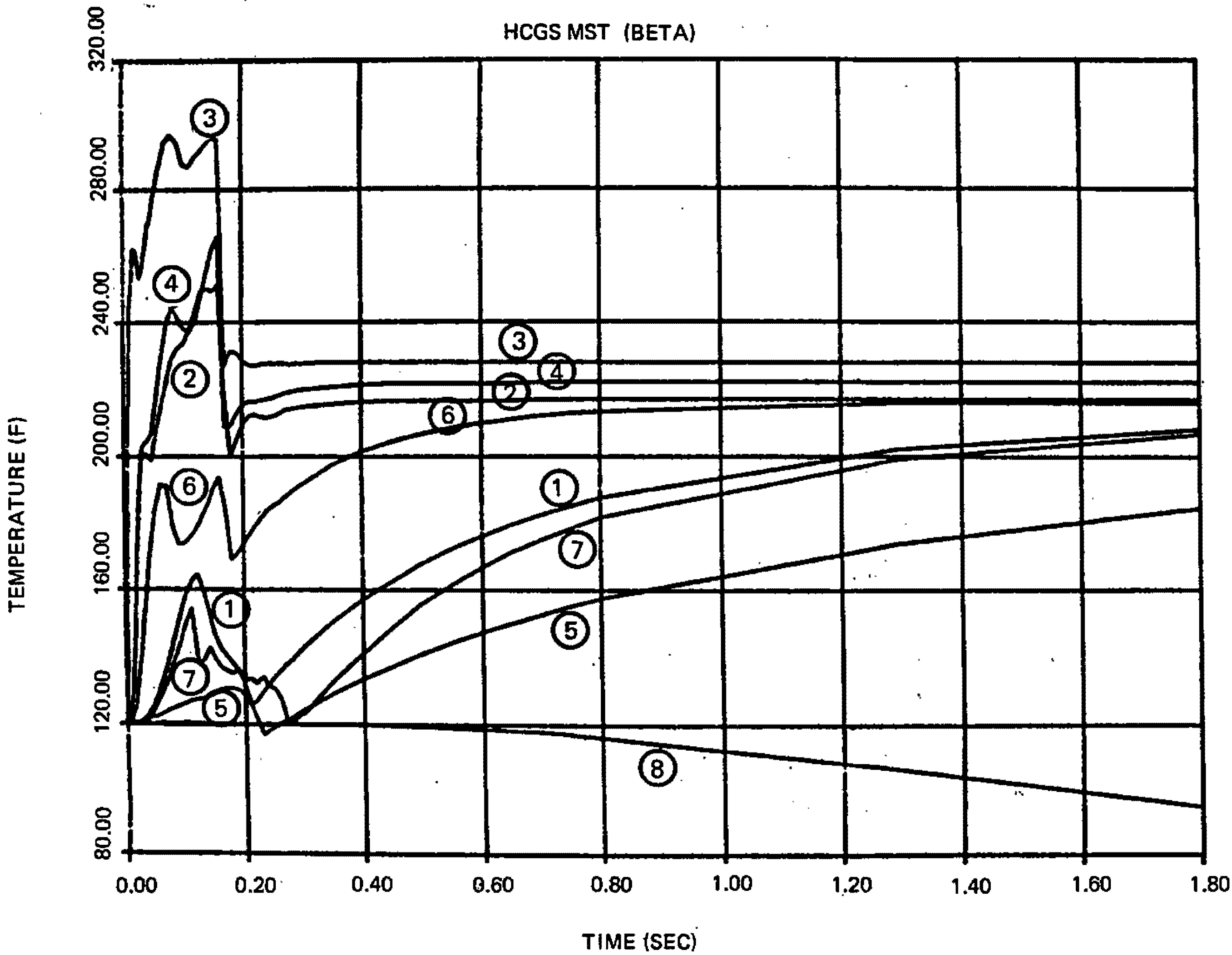
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

PRESSURE TRANSIENT ANALYSIS FOR A MAIN STEAM LINE BREAK IN THE MAIN STEAM TUNNEL

UPDATED FSAR

FIGURE 3.6-43

COMPARTMENT TEMPERATURE vs. TIME



NOTE:
SEE FIGURE 3.6-9 FOR
IDENTIFICATION OF NODES

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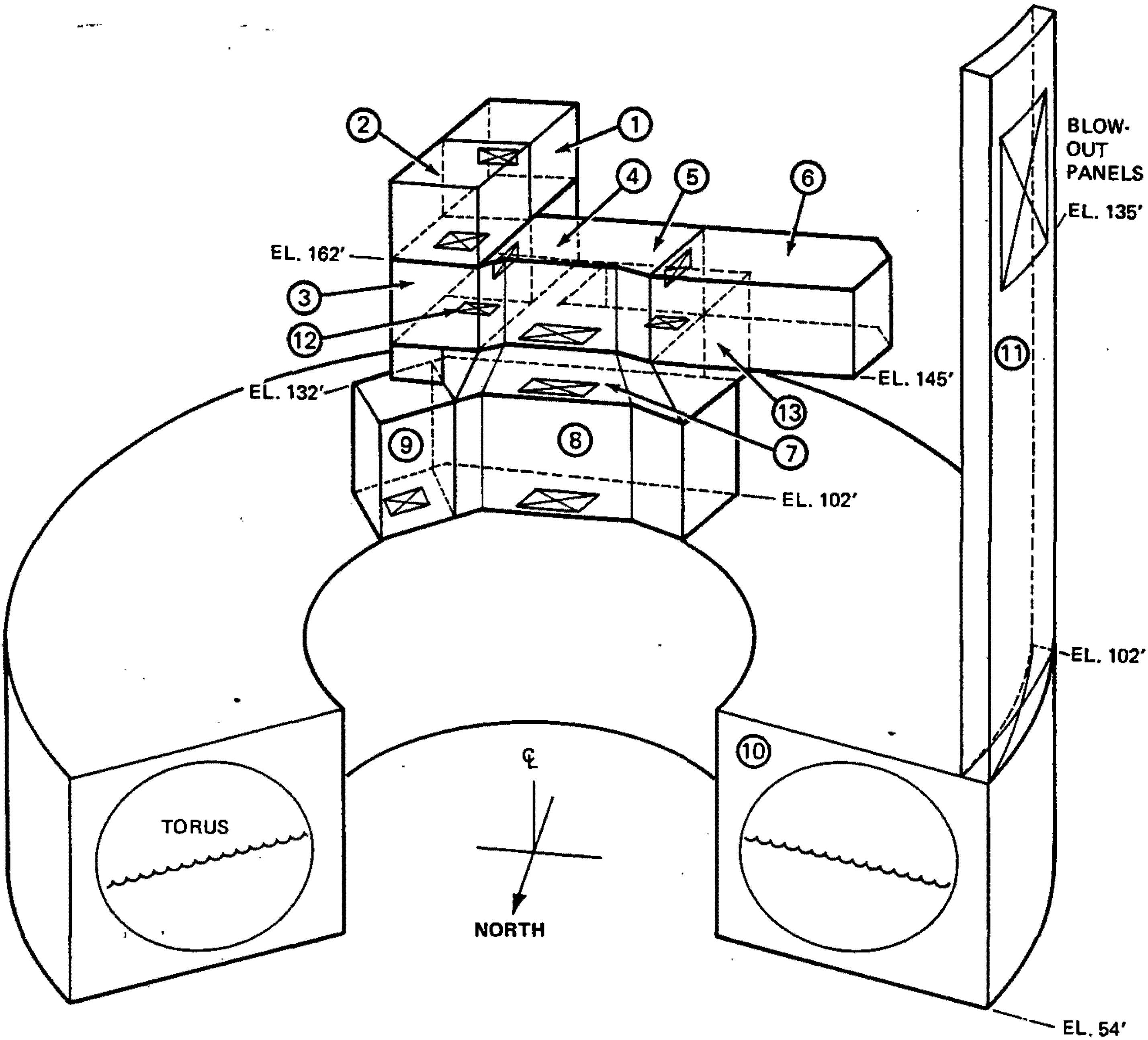
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

TEMPERATURE TRANSIENT ANALYSIS
FOR A MAIN STEAM LINE BREAK
IN THE MAIN STEAM TUNNEL

UPDATED FSAR

FIGURE 3.6-44



ONLY RWCU VENTING COMPARTMENTS ARE SHOWN FOR CLARITY.

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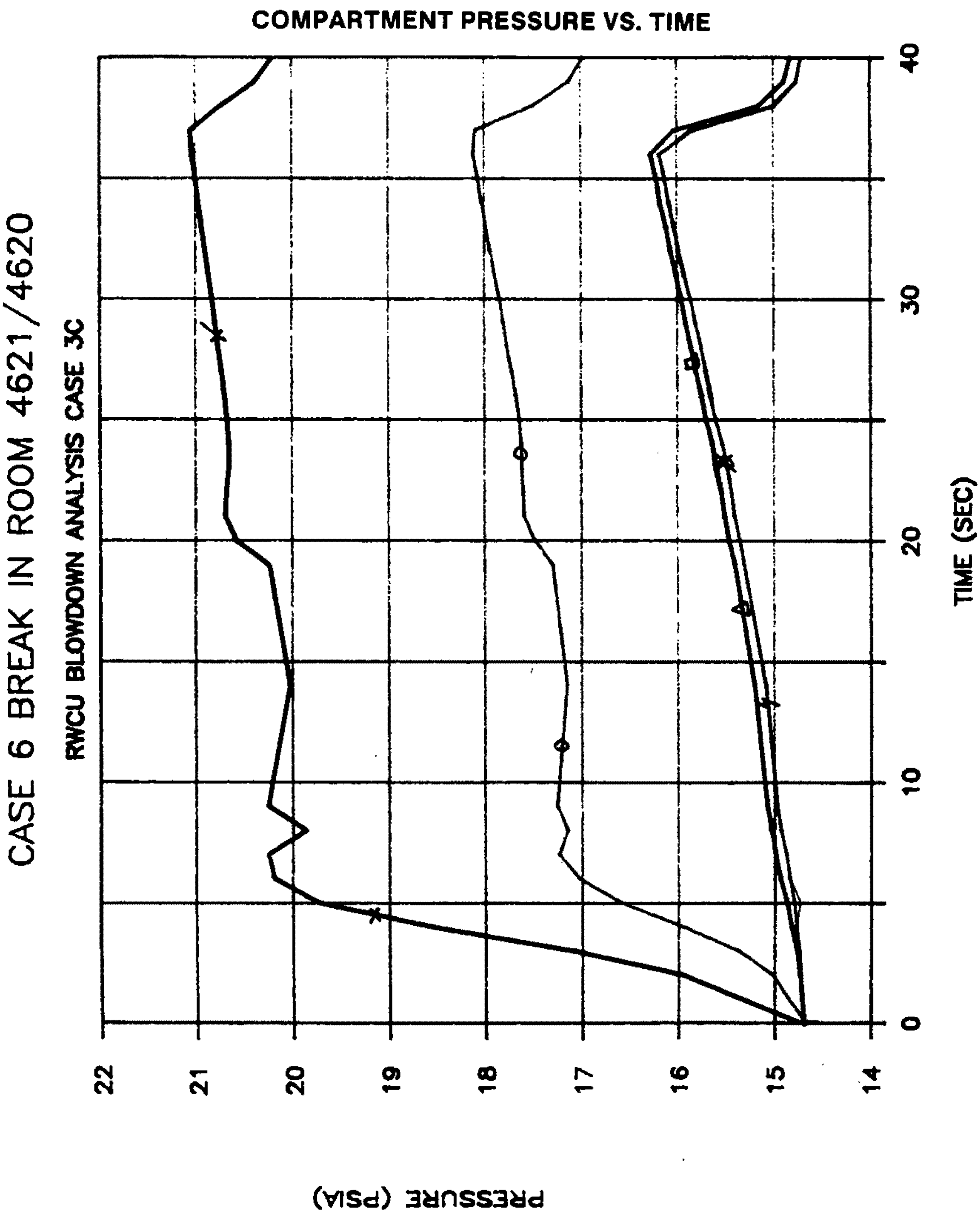
HOPE CREEK NUCLEAR GENERATING STATION

PRESSURE – TEMPERATURE TRANSIENT ANALYSIS MODEL FOR HPCI, RCIC AND RWCU LINE BREAKS OUTSIDE CONTAINMENT

UPDATED FSAR

FIGURE 3.6-45



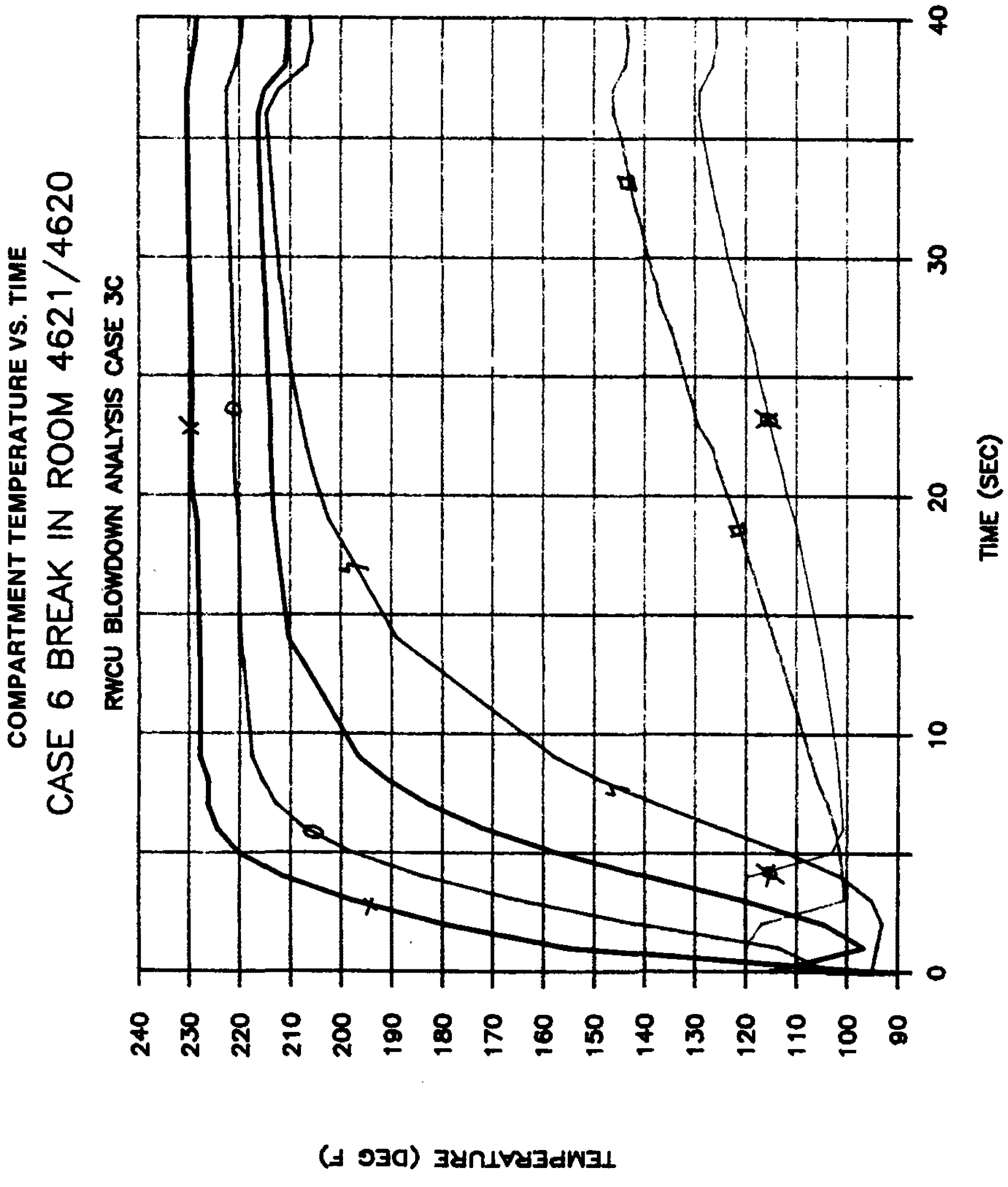
NOTE:
SEE FIGURE 3.6-17 FOR
IDENTIFICATION OF NODES

PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK GENERATING STATION

PRESSURE TRANSIENT ANALYSIS
FOR A RWCU F/D LINE
BREAK IN THE F/D ROOM

UPDATED FSAR
REVISION 1, APRIL 11, 1989

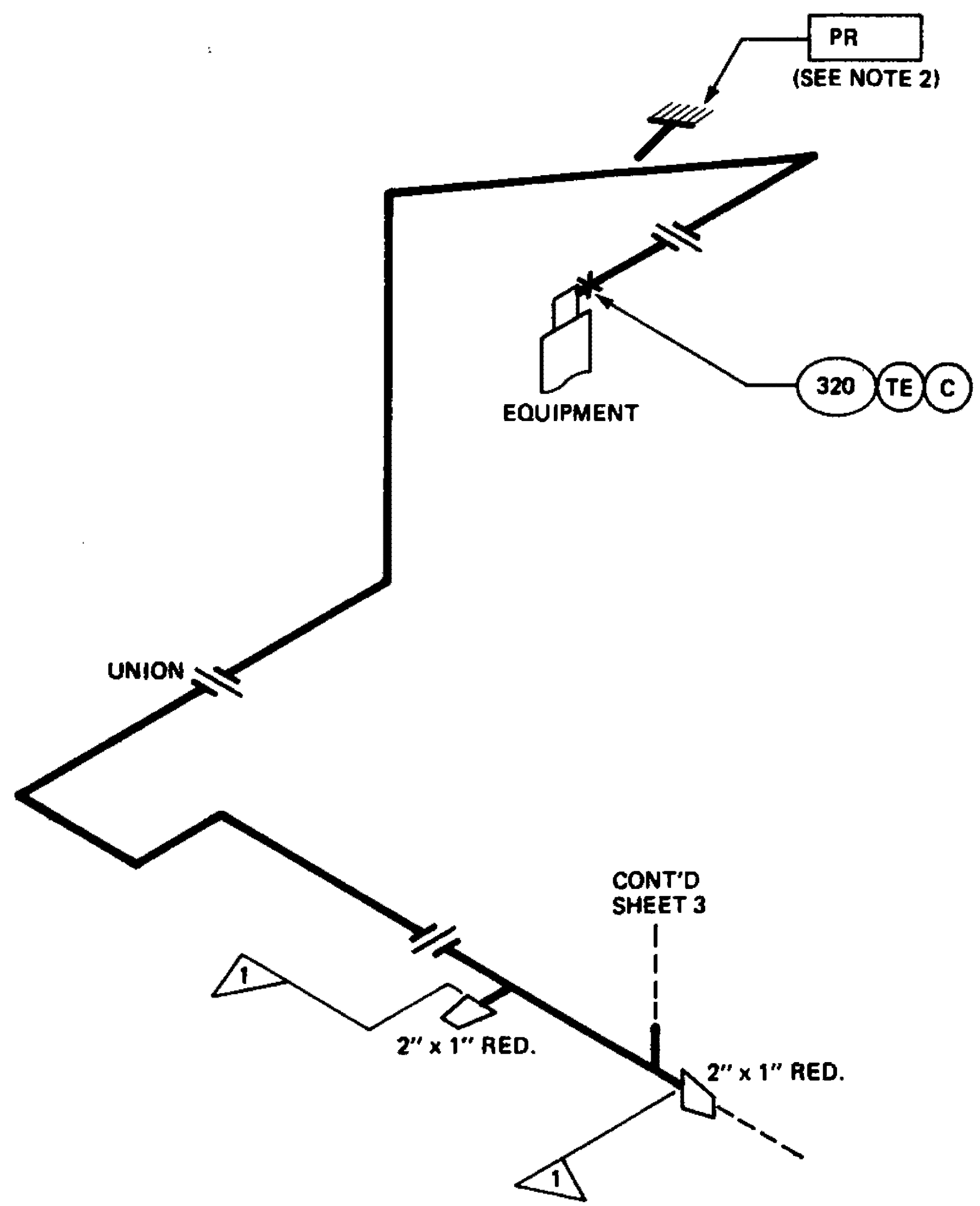
FIGURE 3.6-46



PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK GENERATING STATION

TEMPERATURE TRANSIENT ANALYSIS FOR RWCU F/D LINE BREAK IN THE F/D ROOM

UPDATED FSAR
REVISION 1, APRIL 11, 1989 FIGURE 3.6-47



NOTES:

1. 1 PIPING BEYOND THIS POINT IS 1" NOMINAL DIAMETER

2.

RESTRAINT NO.	D.G. COMP.
RIA	A
RIB	B
RIC	C
RID	D

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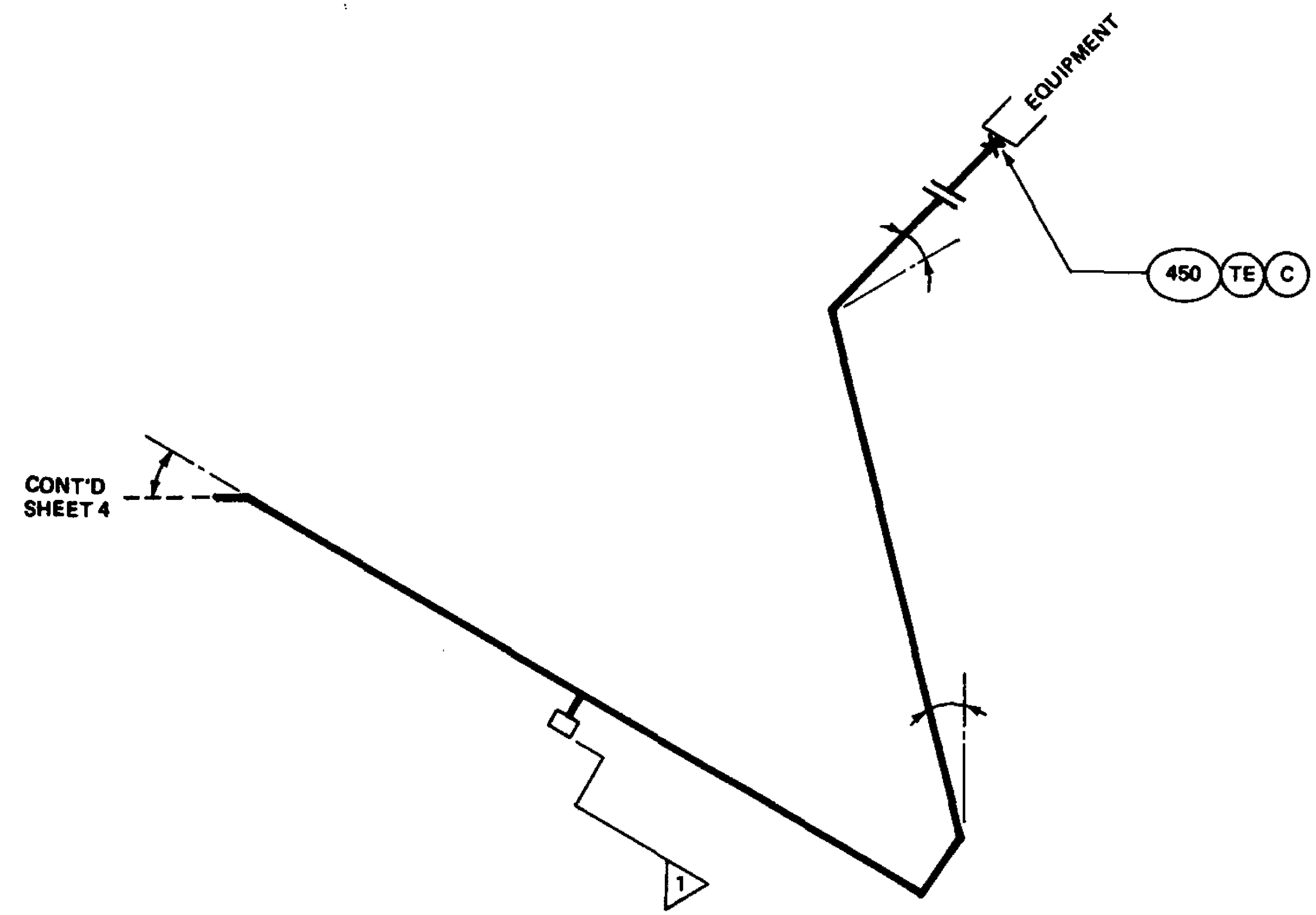
HOPE CREEK NUCLEAR GENERATING STATION

DIESEL STARTING AIR
PIPING ISOMETRIC

UPDATED FSAR

Sheet 1 of 6
FIGURE 3.6-48



NOTE:

1 PIPING BEYOND THIS POINT IS 1" NOMINAL DIAMETER

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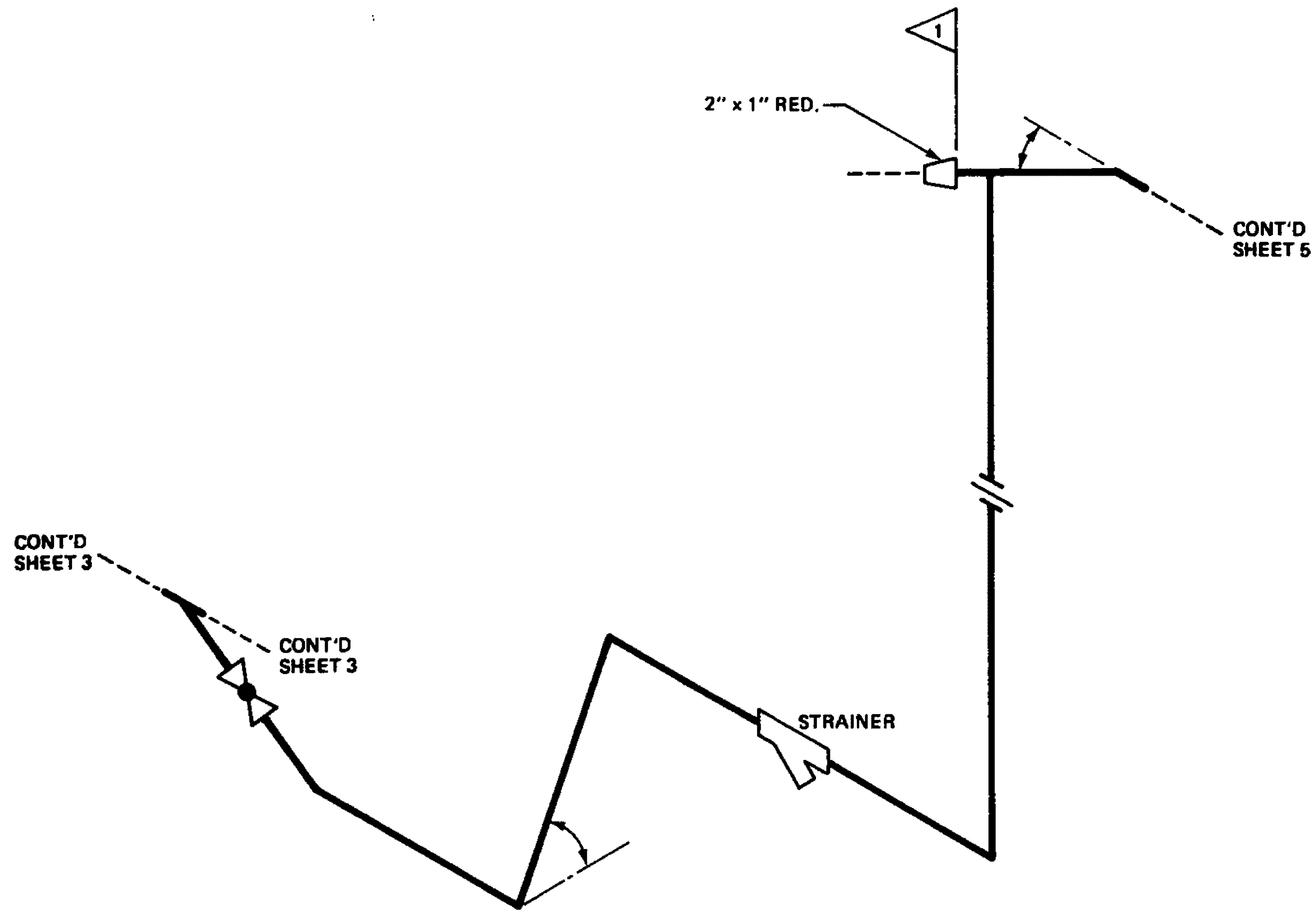
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

DIESEL STARTING AIR
PIPING ISOMETRIC

UPDATED FSAR

Sheet 2 of 6
FIGURE 3.6-48



NOTE:

1 PIPING BEYOND THIS POINT IS 1" NOMINAL DIAMETER.

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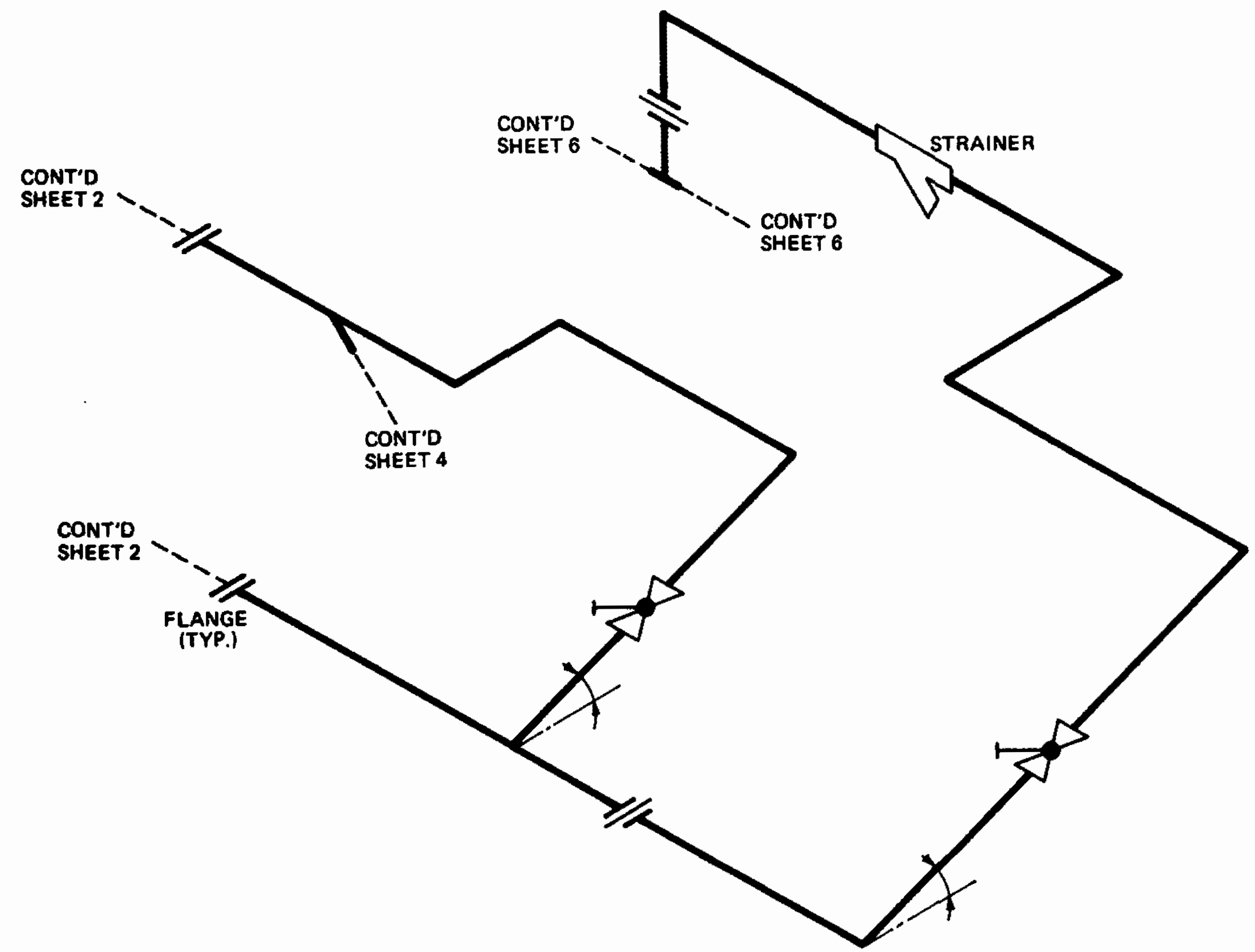
PUBLIC SERVICE ELECTRIC AND GAS COMPANY HOPE CREEK NUCLEAR GENERATING STATION	
DIESEL STARTING AIR PIPING ISOMETRIC	
UPDATED FSAR	Sheet 3 of 6 FIGURE 3.6-48



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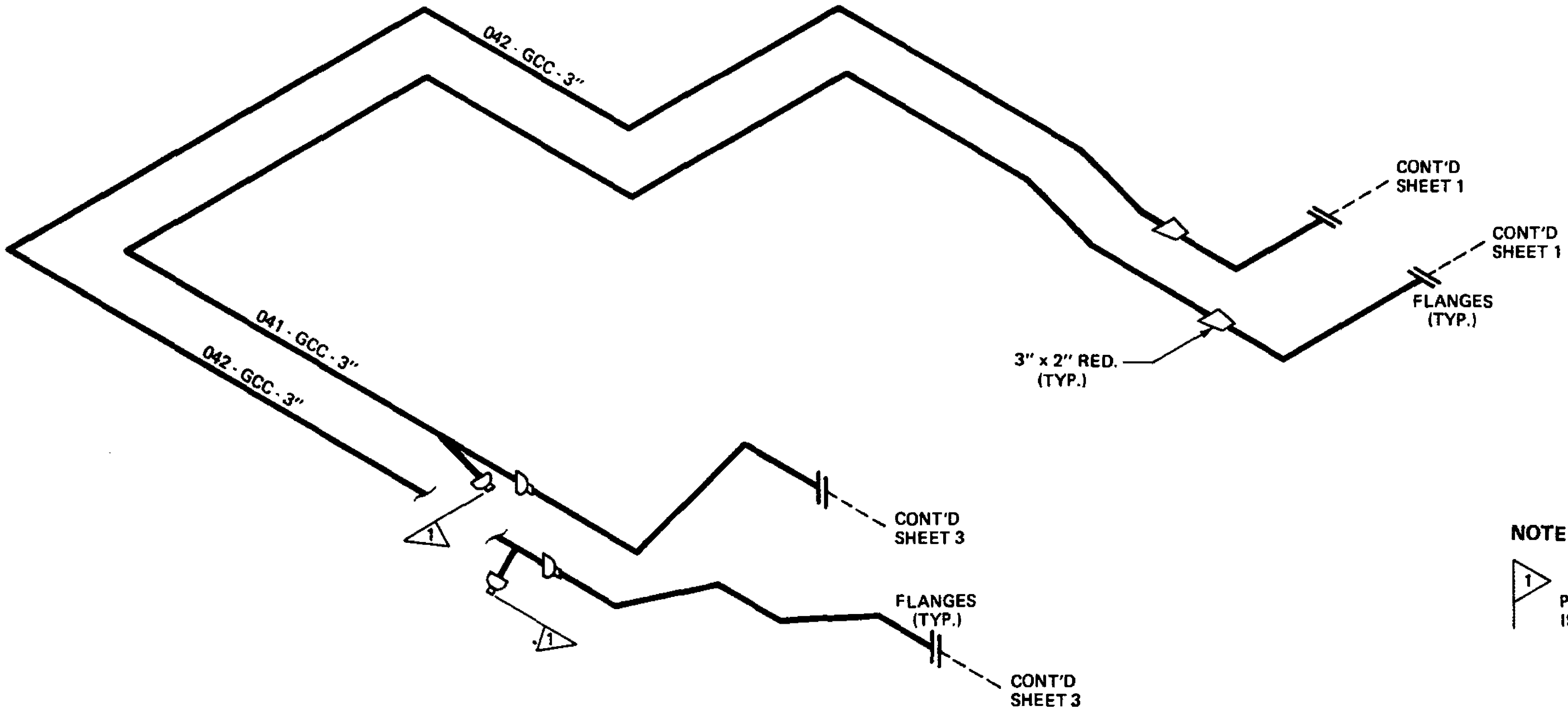
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

DIESEL STARTING AIR
PIPING ISOMETRIC

UPDATED FSAR

Sheet 4 of 6
FIGURE 3.6-48



NOTE:

1 PIPING BEYOND THIS POINT IS 1" NOMINAL DIAMETER.

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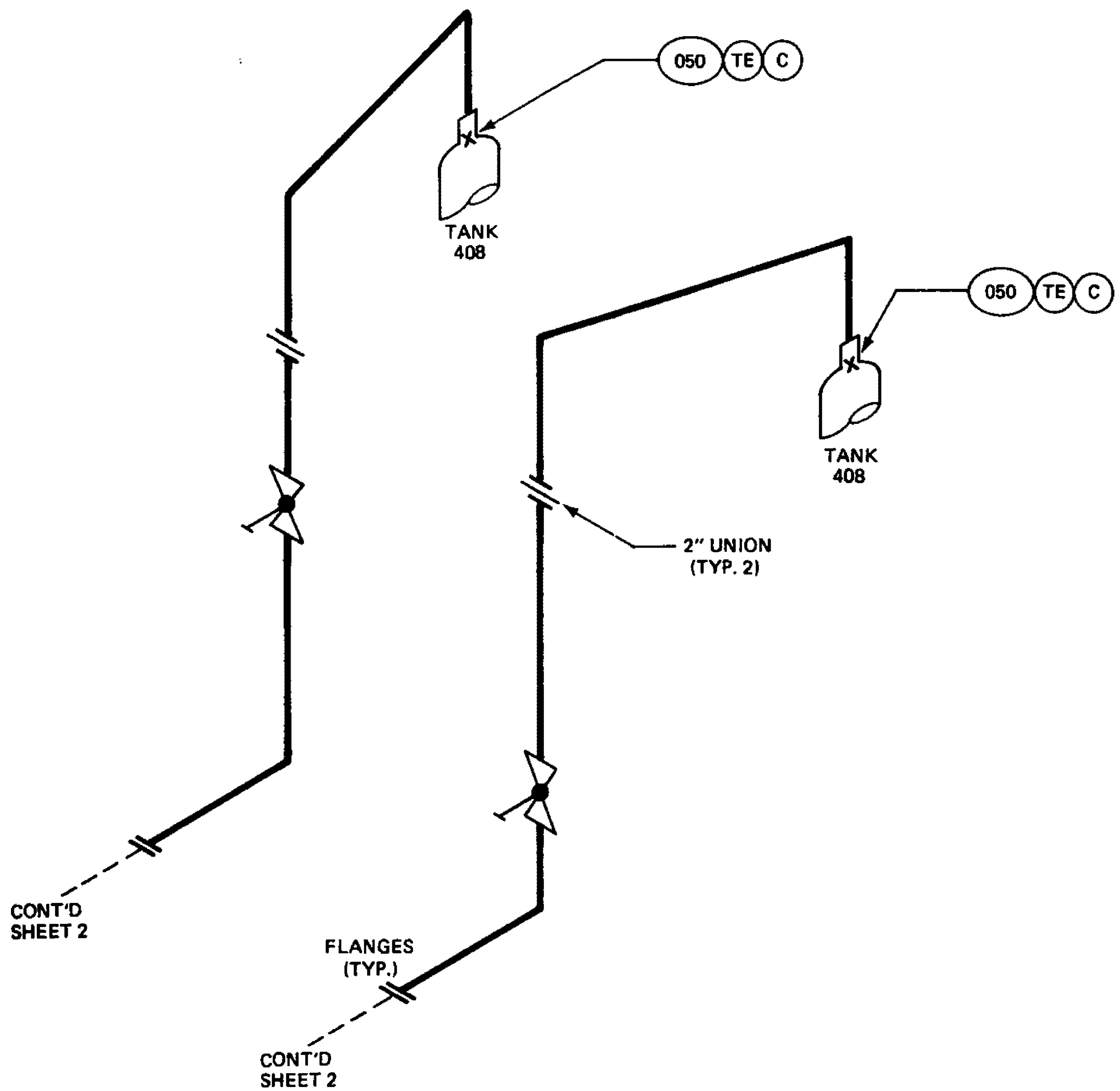
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

DIESEL STARTING AIR
PIPING ISOMETRIC

UPDATED FSAR

Sheet 5 of 6
FIGURE 3.6-48



NOTES:

1. (TE) – TERMINAL END
2. (C) – CIRCUMFERENTIAL BREAK
3. (XXX) – DATA POINT NO.
4. TYPICAL ROUTING FOR FOUR SYSTEMS

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HOPE CREEK NUCLEAR GENERATING STATION

DIESEL STARTING AIR
PIPING ISOMETRIC

UPDATED FSAR

Sheet 6 of 6
FIGURE 3.6-48

3.7 SEISMIC DESIGN

All structures, systems, and components are defined as either Seismic Category I or non-Seismic Category I. The requirements for Seismic Category I identification are given in Section 3.2 with a list of the qualified structures, systems, and components.

All structures, systems, and components important to plant safety are designed to withstand a safe shutdown earthquake (SSE) and an operating basis earthquake (OBE).

The SSE is based on an evaluation of the maximum earthquake potential considering the regional and local geology, seismology, and specific characteristics of local subsurface material. The SSE produces the maximum vibratory ground motion for which Seismic Category I structures, systems, and components are designed to remain functional. These structures, systems, and components are those necessary to ensure:

1. The integrity of the reactor coolant pressure boundary (RCPB).

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 that could result in potential offsite exposures comparable to the guideline exposures of 10CFR50.67.

The OBE is an earthquake that, considering the regional and local geology, seismology, and specific characteristics of local subsurface material, could reasonably be expected to affect the plant site during the operating life of the plant. Seismic Category I structures, systems, and components 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

and within applicable stress and deformation limits during the vibratory ground motion produced by an OBE.

3.7.1 Seismic Input

3.7.1.1 Design Response Spectra

The design response spectra, which comply with the requirements of Regulatory Guide 1.60, are shown on Figures 3.7-1 and 3.7-2, in both the horizontal and vertical directions, respectively, for the SSE. For the OBE, the design response spectral values are taken as half of the SSE values.

Based on geological and seismological information, as discussed in Sections 2.5.2.6 and 2.5.2.7, the maximum ground acceleration values for both horizontal and vertical components of the earthquake are 10 percent and 20 percent of gravity for an OBE and SSE, respectively.

The vibratory ground motion (free field) produced by the seismic motion, as defined by the design response spectra, and the design time history are conservatively applied at the elevation that corresponds to the bottom of the structural foundation in the free field without the effect of the structures.

3.7.1.2 Design Time History

Two synthetic time history motions of 24 seconds duration have been generated by modifying the 1952 Taft earthquake according to the techniques described in Reference 3.7-1. The synthetic time history motions have been generated because the response spectra of available recorded earthquake time histories do not adequately match the site design response spectra. Figures 3.7-3 and 3.7-4 show the synthetic time history motions for the SSE in horizontal and vertical directions, respectively. These time history motions are the same time histories shown in BC-TOP-4A, Reference 3.7-1. For

the OBE, the values of the synthetic time history are taken as half of the SSE values.

Figures 2-13, 2-14, 2-17, and 2-18, in Reference 3.7-1, show that the response spectra of the synthetic time-history motions for the horizontal and vertical directions envelop the corresponding design spectra for 1 percent, 2 percent, 5 percent, 7 percent, and 10 percent damping values. The response spectra are computed at 71 frequencies identified in Section 2.5.1 of Reference 3.7-1.

The design time histories are used in the structural seismic analysis. The acceleration time-history from the result of the structural seismic analysis is used for the generation of floor response spectra.

3.7.1.3 Critical Damping Values

3.7.1.3.1 Critical Damping Values (NSSS)

The damping factors indicated in Table 3.7-1 are used in the response analysis of various Nuclear Steam Supply System (NSSS) systems, components, and equipment, and in preparation of floor response spectra used as forcing inputs for piping and equipment analysis or testing.

For a general compliance or alternate approach assessment, see the Regulatory Guide commitment matrix in Section 1.8.1 for commitment, revision number, and scope.

GE supplied NSSS analysis, design, and/or equipment used in this facility is in compliance with the intent of Regulatory Guide 1.61, which delineates damping values that should be applied to modal dynamic seismic analysis of Seismic Category I structures and components. The damping values used in the seismic analysis conform to the data available on the subject when the analysis was performed, which was the practice accepted by industry and the NRC at the time of the design.

The damping values shown for NSSS materials in Table 3.7-1 are less than those given by Regulatory Guide 1.61; therefore, the calculated responses are conservative.

3.7.1.3.2 Critical Damping Values (Non-NSSS)

For non-NSSS Seismic Category I structures, systems, and components, critical damping, values, expressed as a percentage of critical damping are shown in Table 3.7-2 and comply with Regulatory Guide 1.61. For cable tray support systems, the damping value is 15 percent of critical for the SSE. As discussed in Section 3.10.3, the testing of cable tray systems clearly demonstrates that a substantial amount of vibration energy is dissipated by friction between cables and by friction between cables and the cable tray.

The strain dependent soil damping values used in the seismic analysis are based upon measured values, as shown in Section 2.5.4.

3.7.1.4 Supporting Media for Seismic Category I Structures

All Seismic Category I structures, including plan dimensions of foundation, foundation embedment depth, and total structural height are listed in Table 3.7-3.

The structures in the power block area and the Station Service Water System (SSWS) intake structure rest on the Vincentown Formation, as identified in the soil profiles of Figures 3.7-5 and 3.7-6. However, there is approximately a 10-foot layer of engineered backfill between the foundation and the Vincentown Formation. A description of the supporting media and their properties is provided in Section 2.5.4.

3.7.1.5 SRP Rule Review

Acceptance Criterion II.1(b) of SRP Section 3.7.1 addresses design time history calculations for seismic ground motion. Specifically, spectral values calculated from the design time history should have

frequency ranges in agreement with Table 3.7.1-1 of the SRP, or selection of a set of frequencies should be such that each frequency is within 10 percent of the previous one. In addition, no more than five points of the spectra obtained from the design time history should fall below the design response spectra.

On Hope Creek, in the chosen set of frequencies for the 28-33 Hz range, each frequency is generally not within 10 percent of the previous one. In addition, the spectra obtained from the design time history have more than eight points that fall below the design response spectra for 1 percent, 2 percent, 5 percent, and 7 percent damping.

The design time histories used on Hope Creek are taken from Bechtel Topical Report BC-TOP-4A, Revision 3. These time histories that encompass the SRP deviations have been reviewed and approved by the NRC Staff.

3.7.2 Seismic System Analysis

3.7.2.1 Seismic Analysis Methods

3.7.2.1.1 Seismic Analysis Methods (NSSS)

Seismic Category I NSSS systems and components are under the category of a seismic subsystem and are discussed in Section 3.7.3.

3.7.2.1.2 Seismic Analysis Methods (Non-NSSS)

The two analytical methods utilized for the structural response analysis and the seismic soil structure interaction of Category I structures are discussed in the following sections. The Seismic Category I structures are supported by separate foundation base mats. The relative motions between the base mats are calculated in the soil structure interaction analyses, and the seismic joints between the base mats are designed to accommodate twice the maximum relative displacement between the adjacent base mats. Methods to

account for relative displacement effects on Seismic Category I systems and components are discussed in Section 3.7.3.

3.7.2.1.2.1 Soil Structure Interaction

Two methods used to analyze the soil structure interaction effects are the finite elements method and the impedance (half-space) approach. The finite element method is used for the design base and the independent verification analysis, whereas the impedance approach is used to reconcile the results of the finite element analysis.

3.7.2.1.2.1.1 Finite Element Method

In the finite element method of analysis, the design earthquake motion is defined at the foundation level in the free field. This motion is deconvolved in a one dimensional free field analysis of the site soil deposits to determine the bedrock motion at the base of the soil column model. When this bedrock motion is applied at the base of the soil column, it produces the design earthquake motion at the control point. One dimensional amplification theory is used for this purpose. The computed bedrock motion is then used as input to a finite element model of the soil structure system to compute the structural responses.

The analysis is performed iteratively to account for the strain dependent nature of the nonlinear soil properties. In each iteration the analysis is linear but the soil properties are adjusted from iteration to iteration until the computed soil strains are compatible with the soil properties used in the analysis. The soil structure interaction analysis models are composed of two dimensional, plane strain finite elements representing the structure foundation mats, nonlinear strain dependent soil medium, and lumped mass beam elements representing the structures. The direct integration time history method is used for the soil structure interaction analysis of the intake structure and the East-West and vertical directions of the power block structures.

The complex frequency response analysis method is used for the soil structure interaction of the North-South direction of the power block structures. The building base mat motions, including translation and rocking components obtained from the interaction analysis, are in general used as input for subsequent seismic analysis of the more detailed structural models.

3.7.2.1.2.1.2 Impedance (Half-Space) Approach

In the soil structure interaction analysis, using the impedance approach, the effect of the foundation medium is represented by the foundation impedances, which are functions of the base mat dimensions, embedment depth, elastic properties of the foundation medium, and forcing frequencies. With the foundation impedances known, the structure foundation system is modeled by coupling the fixed-base structural model with the foundation impedances through the basemat. The method of coupling and the equation of motion, is described in Appendix D of BC-TOP-4 (Reference 3.7-1). The technique used to determine the composite modal damping of the interaction system is also given in Appendix D of Reference 3.7-1. The effects of embedment which increase both damping and stiffness of the soil structure systems are considered.

3.7.2.1.2.2 Structural Response

Seismic structural responses of Category I structures are calculated using the modal superposition time history technique for independent earthquake components in the vertical and two horizontal directions. Both the OBE and the SSE are considered in all directions for the power block. For the intake structure, the SSE is considered in all directions, and the OBE is considered using the above procedures for the North-South and Vertical directions only. The intake structure OBE load condition in the East-West direction is considered by scaling the appropriate SSE East-West direction conditions by a 70 percent factor. This 70 percent factor is chosen because it is observed from the North-South response spectra that a value of

70 percent of SSE conservatively envelopes the OBE response. The resulting OBE and SSE response data include time-histories of floor acceleration and associated floor response spectra, maximum displacements, and member forces.

Seismic analysis of the structures consider all modes with frequencies up to 33 cps. Consideration of modes higher than 33 cps does not result in more than a 10 percent increase in response.

3.7.2.2 Natural Frequencies and Response Loads

Natural frequencies of the significant modes of the Reactor Building, the Auxiliary Building, and the SSWS intake structure are shown in Table 3.7-4. The significant mode shapes of the Reactor Building and the Auxiliary Building are shown on Figures 3.7-10 through 3.7-57 for each of three orthogonal directions: east-west, north-south, and vertical.

Figures 3.7-58 through 3.7-119 show the structural responses, i.e., displacements, accelerations, shear forces, bending moments, and axial forces, of the Reactor Building, the Auxiliary Building, and the SSWS intake structure for each of the three orthogonal directions.

In-structure floor response spectra at critical locations are shown on Figures 3.7-120 through 3.7-155. The curves are shown for each of the three orthogonal directions at the damping values used for each design earthquake. A brief description of the location of each series of response spectrum curves is provided below with the corresponding figure numbers:

1. Figures 3.7-120 through 3.7-125 - Reactor Building at Elevation 102 feet 0 inches

2. Figures 3.7-126 through 3.7-131 - Reactor Building at Elevation 201 feet 0 inches

3. Figures 3.7-132 through 3.7-137 - Auxiliary Building diesel generator area at Elevation 130 feet 0 inches

4. Figures 3.7-138 through 3.7-143 - Auxiliary Building diesel generator area at Elevation 178 feet 0 inches

5. Figures 3.7-144 through 3.7-149 - Auxiliary Building control area at Elevation 137 feet 0 inches

6. Figures 3.7-150 through 3.7-155 - Intake structure at Elevation 122 feet 0 inches.

3.7.2.3 <u>Procedure Used for Modeling (Non-NSSS)</u>

Section 3.2 identifies the Seismic Category I structures, systems, and components. This section discusses Seismic Category I structures. Section 3.7.3 discusses Seismic Category I subsystems and components.

Procedures for development of the building mathematical models are discussed in this section, and those for development of the finite element soil models are discussed in Section 3.7.2.5.

The mathematical models of the Reactor Building, Auxiliary Building, and intake structure are shown on Figures 3.7-7 through 3.7-9. The building mathematical models consist of lumped masses connected by massless, elastic beam members. Masses are located at floor elevations and elevations of major mass concentration. Masses are computed by considering the weights of the floor, floor framing, structural walls, and columns above and below the floor level, nonstructural walls above the floor level, and all equipment (except the reactor vessel), components, and piping systems. The number and location of the mass points are chosen so that all significant degrees of freedom have been incorporated in the models to ensure that an accurate representation of the dynamic response is obtained. Since the mass and stiffness distributions in the buildings are generally unsymmetric, the effects of torsional rotation are

included by using three dimensional models having six dynamic degrees of freedom at each mass point and by including the computed eccentricities between the centers of mass and centers of structural rigidity. The beam elements representing structural walls and columns connecting two adjacent floors are located at the center of rigidity of the cross section. The elastic properties of the beam members include the effects of bending, shear, axial, and torsional structural stiffnesses. Material damping characteristics are defined in accordance with Table 3.7-2, and are incorporated into modal superposition time history analyses, using a strain energy weighting technique.

Separate mathematical models for vertical floor flexibility analysis are formulated for each building. In these models, the floor diaphragms are modeled to allow for vertical floor flexibility effects. The floors are modeled using horizontal elements connected between vertical resisting elements. The horizontal elements are tuned in frequency to match the floor vertical frequencies, as calculated in separate analyses using detailed finite element meshes. The vertical elements represent the concrete walls and steel columns of the building.

Criteria used for decoupling subsystems from Seismic Category I structures are discussed in Section 3.2 of Reference 3.7-1.

3.7.2.4 Soil Structure Interaction

Three categories of seismic soil structure interaction analyses are performed for the major plant structures. The design base analyses are performed using the finite element method. Independent finite element soil structure interaction analyses are subsequently performed to verify the "design base" analyses. The impedance approach (the half-space) soil-structure interaction analyses are performed to evaluate the adequacy of the finite element soil structural interaction analysis results, used in the plant design.

3.7.2.5 Design Base Analysis

The design base seismic soil structure interaction analyses are performed to determine the response time histories at the base mats of all Category I and major non-Category I structures for use in subsequent seismic analyses of the individual structures, and to evaluate maximum dynamic soil pressures beneath the base mats and against exterior walls of the buildings during a seismic event. The maximum dynamic responses (displacements, accelerations, and member forces) induced in the structure due to the building base motions obtained from the soil structure interaction analyses, are determined, and the floor response spectra at selected elevations in each structure are developed for use in subsequent analyses of the structural components, mechanical equipment, and attached piping systems. These analyses are performed for the following structures:

1. Reactor Building

2. Auxiliary Building

3. Turbine Generator Buildings

4. Service Water Intake Structure

The soil structure interaction analyses are performed by constructing two dimensional finite element mathematical models of soil and structures at the site. These models are subjected to seismic excitations at the base of the soil model, determined by deconvolution analyses.

For the intake structure and the East-West and vertical analysis of the power block area the computer code DECON is used in the free field soil column deconvolution analysis to generate the bedrock motion. The interaction analyses employ a direct time integration procedure in which the time history of responses of the soil structure system are calculated using a step by step

integration of the coupled equations of motion. The computer code EDSGAP is used for these analyses.

For the North-South analyses of the power block both the free field soil column deconvolution and the soil structure interaction analyses are performed using the computer code FLUSH, Reference 3.7-6.

Figures 3.7-156 and 3.7-159 show the coupled soil and structure models constructed along the north-south and east-west directions for both the power block area and the intake structure. The soil structure interaction analysis models are used for the following analyses:

1. Figure 3.7-156 is the horizontal model used for the North-South analysis of the Reactor Building and the Auxiliary Building.

2. Figure 3.7-156a is the vertical model used for the vertical analysis of the Reactor Building and the Auxiliary Building.

3. Figure 3.7-157 is the horizontal model used for the East-West analysis of the Reactor Building.

4. Figure 3.7-157a is the horizontal model used for the East-West analysis of the Auxiliary Building.

5. Figure 3.7-158 is the model used for the horizontal North-South and Vertical analyses of the intake structure, and

6. Figure 3.7-159 is the horizontal model used for the East-West analysis of the intake structure.

Each model consists of a vertical section of plane strain isotropic quadrilateral elements representing the soil and foundation mats,

and lumped mass beam simplified stick models representing the Seismic Category I and major non-Seismic Category I structures. The simplified models used in the soil structure interaction analyses are two dimensional and are developed based on the detailed three dimensional building models, which are discussed in Section 3.7.2.3. The simplified models have significant mode shapes and frequencies closely matching those of the detailed models.

For the modeling of supporting soil, the lateral boundaries are either simulated by transmitting boundaries or located far enough from the building to minimize the effect due to wave reflections from the model boundary. All significant interaction effects occur within 300 feet of the power block area and 200 feet of the intake structure. An evaluation of soil shear strains and accelerations shows a return to free field conditions within these distances.

Parametric studies are performed for the power block area and the intake structure area to evaluate the depth of soil structure interaction model, variations in soil damping, and variations in soil modulus.

The criterion used in the depth parametric studies is that further increases in the depth of soil structure interaction models would not alter the interactive response of the structures. For the power block area, this criterion is met using a soil structure interaction model with a depth of 402 feet below grade. For the intake structure area, this criterion is met using a soil-structure interaction model with a depth of 300 feet below grade.

The results of the soil structure interaction analysis for the power block indicate that the average soil strain of the foundation soil is about 5.0×10^{-4} in/in. Based on the available information on soil properties (Figure 2.5-41) it is concluded that the soil properties could vary by approximately ±50 percent from the final iterated average soil properties. A soil variation study of the power block area was performed using the above bases. The results indicate that the major response spectral peak frequency shifts ±22 percent and

minor peaks have insignificant frequency shifts. For conservatism, the computed horizontal response spectral peaks are broadened by ±25 percent for the major peak and ±15 percent for secondary peaks.

No explicit soil variation study was performed for the vertical direction. Due to the high ground water table, the effective compressional wave velocity of the saturated soil is controlled by the compressional wave velocity of the ground water. Accordingly, the vertical effective compressional wave velocity is not sensitive to variations in the soil shear modules. Therefore, it is concluded that the peak broadening of ±15 percent for the major peak and ±10 percent for secondary peaks are adequate.

Because dynamic soil properties underlying the power block area and the intake structure are essentially the same, the broadening criteria for the power block area response spectra are also applicable to the response spectra of the intake structure. However, the design of the intake structure was originally based on preliminary requirements which called for ±50 percent broadening of the response spectra. The preliminary ±50 percent spectral peak broadening criteria are maintained as the final broadening criteria for the intake structure response spectra since they are conservative.

The variation studies for soil damping showed that, for the HCGS site, changes from upper bound to lower bound damping properties result in frequency shifts of ±2 percent in the building base mat spectra.

The vertical layer depth dimension of the soil finite element mesh is selected based on the procedure outlined in Reference 3.7-6 and Section 3.3 of Reference 3.7-1 so that the soil model is able to pass an adequately high frequency of the interaction system.

The boundary conditions on the side boundaries of the finite element soil model depend on the direction of input motion. When the input motion is horizontal, the nodal points on the vertical boundaries

are restrained from moving vertically. In the case of vertical input motion, the nodal points on the vertical side boundary of the model are restrained from moving horizontally.

The soil structure interaction analysis is performed by applying the deconvoled bedrock motion at the base of the finite element model developed above. The base rock input motion is generated through deconvolution analysis. Input spectra to the deconvolution analyses are the design response spectra described in Section 3.7.1.1. The input elevations to the deconvolution analyses are the foundation levels of the embedded structures. The technique used for the deconvolution analysis is discussed in Reference 3.7-4.

Separate time history analyses are performed for the vertical and two orthogonal horizontal earthquakes. The analyses for horizontal earthquake excitation are used to determine horizontal base mat as well as rocking acceleration time histories. The analyses for vertical earthquake excitation are used to determine vertical base mat acceleration time histories.

The effective soil shear strain levels, equal to 60 percent of the maximum shear strains, are determined throughout the soil region of the models (For the North-South analysis of the power block, the computer code FLUSH calculates effective shear strain levels as 65 percent of the maximum shear strains). Effective shear strain levels are used to determine strain compatible soil properties. Nonlinear, strain dependent characteristics of the soil are treated using an iterative linear approach. During each step of the iterative analytical process, the assumed soil properties used in the model are evaluated for compatibility with induced strain levels. The properties are revised and the analyses repeated until compatibility within 10 percent is obtained.

For the power block area, the effect of structure-soil structure interaction is considered in the soil structure interaction analyses by including multiple buildings in each model, as shown on Figures 3.7-156 and 3.7-157.

The initial design base seismic soil-structure interaction analysis was performed using the assumption that construction of both units would be completed prior to start of operation of Unit 1. However, with the cancellation of Unit 2, the Unit 2 Reactor Building is terminated at elevation 132 feet 0 inches, resulting in a non-symmetric configuration. The North-South analysis is revised to include the effects of the Unit 2 cancellation. For the East-West direction, a parametric analysis has been performed to evaluate the effects of the partially completed Unit 2 on the seismic structural responses and other major plant structures. It is concluded that the cancellation of Unit 2 has no significant impact on the seismic structural responses of other plant structures.

3.7.2.5.1 Independent Finite Element Verification Analysis

Independent soil structure interaction verification analyses are performed to verify the accuracy of the results of the design base analyses. These analyses are performed using the finite element method and take foundation embedment into account. The computer code FLUSH is used in this study.

In this independent analysis, the NRC broad band design response spectra are specified at the foundation elevation in the free field. Simplified seismic structural models for the Reactor Building, the Auxiliary Building, and the Turbine Building are developed for use in the FLUSH soil structure interaction analyses. The dynamic behavior of the simplified structural models compared reasonably well with that of the structural models used in the design base analysis. The horizontal North-South, the horizontal East-West, and the vertical seismic soil structure interaction analyses are performed for both the SSE and the OBE cases. The results of the independent analyses are determined to be in reasonable agreement with those of the design base analyses.

3.7.2.5.2 Impedance (Half-Space) Approach Analyses

Seismic soil structure interaction analyses of all Seismic Category I structures are performed using the impedance approach with strain independent soil properties for the North-South, East-West and the vertical excitations.

The impedance approach analysis is performed to assess the adequacy of the results of the finite element analysis. The impedance analysis results are used to confirm the adequacy of the plant design.

3.7.2.6 Development of Floor Response Spectra

3.7.2.6.1 Floor Response Spectra (NSSS)

Floor response spectra for NSSS equipment are developed considering three components of earthquake motion. The individual floor response spectra in each orthogonal direction are then obtained as the square root of the sum of the squares (SRSS) combination of the collinear contributions, due to the three directions of earthquake motion. These are used to predict the total floor response spectra at each frequency.

3.7.2.6.2 Floor Response Spectra (Non-NSSS)

Time history analyses for independent excitation in one vertical and two horizontal excitations are performed to develop the floor response spectra. The mathematical models and the analytical method used are as described in Sections 3.7.2.1, 3.7.2.3, and 3.7.2.4.

The floor response spectrum at a given location and direction is developed considering the three components of the earthquake motion. The response spectral values for each frequency at a given location and direction are combined by taking the square root of the sum of the squares (SRSS) of the co-directional response spectral values from each of the three components of earthquake motion at critical

locations. These response spectra are compared to the single directional response spectra, resulting in a difference of less than 5 percent. Therefore, single directional response spectra are used. The effect of rocking causes amplification in floor response spectra. For horizontal floor spectra, rocking effects are directly included due to incorporation of both translational and rocking base mat time-histories in the seismic structural analyses. For vertical floor spectra, rocking effects are calculated from seismic structural analyses using vertical base mat time histories and rocking time histories separately, and combining the response using the SRSS method.

3.7.2.7 Three Components of Earthquake Motion (Non-NSSS)

The time history analysis method is employed for the seismic analysis of all Seismic Category I structures. Maximum structural responses, displacements, accelerations, and member forces, due to each of the three components of earthquake motion, are obtained.

In accordance with Regulatory Guide 1.92, parametric studies were performed to evaluate the significance of an out of plane response from an in plane base excitation. These studies verified that the out of plane structural responses added no significant contribution to the in plane structural responses.

3.7.2.8 Combination of Modal Responses (Non-NSSS)

When the response spectrum method is used in seismic analysis of structures, systems, and components, the modal responses, i.e., displacements, accelerations, and member forces, are combined in accordance with Regulatory Guide 1.92.

3.7.2.9 Interaction Between Adjacent Structures

3.7.2.9.1 Interaction of Non-Seismic Category I Structures with Seismic Category I Structures

The Turbine Building and the administration facility are the only non-Seismic Category I structures located near Seismic Category I structures. They are designed to withstand an SSE in accordance with Section 3.8.4. Dynamic analyses of these structures are performed using the time history method. Structure to structure interaction between the Turbine Building and the administration facility, and the Seismic Category I buildings is accounted for by including all buildings in the soil structure interaction model shown on Figure 3.7-157.

Structural separation is provided to ensure that physical contact between Seismic Category I and non-Seismic Category I structures does not occur. Considering the variability and uncertainties associated with parameters in the analysis, the minimum separation between the structures is maintained at twice the absolute sum of the predicted maximum displacements (due to seismic loadings) of the adjacent structures. Table 3.7-6 compares the actual structural gaps with the worst computed gaps.

3.7.2.9.2 Interaction Between Adjacent Seismic Category I Structures

Structural separation is provided to ensure that physical contact between adjacent Seismic Category I structures does not occur. Considering the variability and uncertainties associated with the parameters in the analysis, the minimum separation between the structures is maintained at twice the absolute sum of the predicted maximum displacements (due to seismic loadings) of the adjacent structures. Table 3.7-6 compares the actual structural gaps with the worst computed gaps.

3.7.2.10 Effects of Parameter Variations On Floor Response Spectra

3.7.2.10.1 Effects of Parameter Variations on Floor Response Spectra (NSSS)

To account for potential variations in the primary structure response frequencies due to uncertainties in material properties of the soil and structure, to the soil structure interaction techniques, to approximation of damping, and to approximations in dynamic modeling, the computed floor response spectra are peak broadened as shown in Table 3.7-7.

3.7.2.10.2 Effects of Parameter Variations on Floor Response Spectra (Non-NSSS)

To account for variations in the structural frequencies, owing to uncertainties associated with the soil modulus, damping, and structural properties, and also 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.

Variation studies in soil damping, shear moduli, and the depth of soil models were performed to evaluate the effects on the response spectra at the foundation level. These studies showed some variations in frequency of in-structure response spectra. The overall effect of the shift in the peak frequency of the spectral acceleration on the in-structure response spectra was determined by the SRSS of the individual variations.

The amounts of peak widening associated with the structural frequencies used at HCGS are ±25 percent for the dominant spectral peaks and ±15 percent for all other responses for the North-South and East-West directions of the power block area, ±15 percent for the dominant spectral peaks and ±10 percent for all other responses for the vertical direction of the power block area, and ±50 percent for the intake structure.

3.7.2.11 Use of Constant Vertical Static Factors

Equivalent static load 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.

3.7.2.12 Method Used to Account for Torsional Effects (Non-NSSS)

Torsional response in the seismic analysis of the Seismic Category I structures, resulting from eccentricity between center of mass and center of rigidity, is explicitly included in the analytical procedure, as discussed in Section 3.7.2.3.

In addition to the torsional responses discussed above, the shear resisting elements, such as concrete walls, are capable of resisting an additional torsional moment assumed to be equivalent to the story shear acting with an additional eccentricity of 5 percent of the maximum building dimension at that level.

3.7.2.13 Comparison of Responses

A comparison between the response spectrum and time history method of dynamic analysis is not applicable, because only the time history method of analysis is used on major Seismic Category I structures.

3.7.2.14 Methods for Seismic Analysis of Dams

Dams are not provided on HCGS.

3.7.2.15 Determination of Seismic Category I Structure Overturning Moments

The overturning moment for Seismic Category I structures is the absolute sum of the moments at the level of the base mat of each stick of the mathematical model. For each stick, the moment at the base is determined by the method discussed in Section 3.7.2.1.

The components of the earthquake motion used are the same as those discussed in Section 3.7.2.6.

Section 3.8.5 discusses the factor of safety against overturning for several loadings, including seismic loads.

3.7.2.16 Analysis Procedure for Damping (Non-NSSS)

When the time history analysis with the modal superposition technique is used in the seismic analysis of the structures, the equivalent modal damping ratio for each mode is calculated based on the use of the stiffness as weighting function.

When the time history analysis with direct integration technique (EDSGAD) is used in the finite element soil structure interaction analysis, the damping matrix C is assumed to consist of the following linear combination of the mass M and stiffness K matrices:

$$C = \alpha M + \beta K \qquad (3.7\text{-}1)$$

The damping coefficients, α and β, are determined such that a reasonable approximation of the soil strain compatible damping over the frequency range of interest is provided.

For the finite element soil-structure interaction analysis using the computer code FLUSH, the stiffness matrix or the complex equation of motion are formed using the complex shear moduli.

$$G^* = G\,(1 - 2\,\xi^2 + 2i\xi\,\sqrt{1 - \xi^2})$$

Where ξ is the fraction of the critical damping which may vary from element to element, G is the shear modulus and $i = \sqrt{-1}$. The damping is included in the analysis by the use of complex shear modulus.

For the impedance (half-space) approach soil structure interaction analysis, the damping is included in the composite modal damping of

the interaction system. The technique to determine the composite modal damping of an interaction system is given in Appendix D of Reference 3.7-1.

3.7.3 Seismic Subsystem Analysis

This section discusses the seismic analysis of equipment, piping, and supports for Seismic Category I heating, ventilating, and air conditioning (HVAC) ducts, cable trays, conduits, and the NSSS components.

3.7.3.1 Seismic Analysis Methods

3.7.3.1.1 Seismic Analysis Methods (NSSS)

Analysis of Seismic Category I NSSS systems and components is accomplished, where applicable, using the response spectrum or time history approach. Both use the natural period, mode shapes, and appropriate damping factors of the particular system. Certain pieces of equipment that have very high natural frequencies are analyzed statically if the fundamental frequency of the component is greater than the zero period acceleration (ZPA) frequency of the excitation. In some cases, dynamic testing of equipment is used for seismic qualification.

The time history analyses involve the solution of the equations of dynamic equilibrium discussed in Section 3.7.1.1.1 by means of the method discussed in Section 3.7.3.1.1.2. 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 discussed in Section 3.7.3.1.1.1 by the method discussed in Section 3.7.3.1.1.3. The method of combining responses for the three components of an earthquake motion is described in Section 3.7.3.6 for NSSS systems, components, and equipment. Seismic and dynamic analysis methods include the investigation of a

sufficient number of modes to ensure participation of all significant modes. All modal responses that cumulatively contribute to at least 90 percent of the total response are included. This meets the criterion that the inclusion of any additional modes does not result in more than a 10 percent increase in the overall response.

3.7.3.1.1.1 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:

$$[M]\ \{\ddot{u}(t)\} + [C]\ \{\dot{u}(t)\} + [K]\ \{u(t)\} = 0 \qquad (3.7\text{-}2)$$

$$\text{and } u_t(t) = u(t) + u_s(t) \qquad (3.7\text{-}3)$$

or

$$[M]\{\ddot{u}(t)\} + [C]\{\dot{u}(t)\} + [K]\{u(t)\} = P(t) \equiv -[M]\{\ddot{u}_s(t)\} \qquad (3.7\text{-}4)$$

where:

$\{u(t)\}$ = time dependent displacement vector (1xN) of nonsupport points relative to the base support displacement $u_s(t)$

$\{\dot{u}(t)\}$ = time dependent velocity vector (1xN) of nonsupport points relative to the base support velocity $u\ (t)$

$\{\ddot{u}(t)\}$ = time dependent acceleration vector (1xN) of nonsupport points relative to the base support acceleration $u_s(t)$

$\{u_t(t)\}$, $\{\ddot{u}f_t(t)\}$ - total displacement and acceleration, respectively

[M] - diagonal matrix of lumped masses

[C] - damping matrix (NxN)

[K] - stiffness matrix (NxN)

P(t) - time dependent inertial force vector acting at nonsupport points.

The manner in which a distributed mass, distributed stiffness system is idealized into a lumped mass, distributed stiffness system of the NSSS component is shown on Figure 3.7-160, along with a schematic representation of relative acceleration, support acceleration, and total acceleration.

3.7.3.1.1.2 Solution of Equations of Motion by Mode Superposition

The technique used for the solution of the equations of motion is the method of mode superposition, in which the equations of motion are decoupled by the eigen transformation.

The set of homogenous equations represented by the undamped free vibration of the system is:

$$[M]\ \{\ddot{u}(t)\} + [K]\ \{u(t)\} = \{0\} \qquad (3.7\text{-}5)$$

Since the free oscillations are assumed to be harmonic, the displacement vector $\{u(t)\}$ can be written as:

$$\{u(t)\} = \{\phi\}\ e^{i\omega t} \qquad (3.7\text{-}6)$$

where:

$\{\phi\}$ = column matrix of the amplitude of displacements $\{u\}$

ω = circular frequency of oscillation

t = time

i = $\sqrt{-1}$

Substituting Equation 3.7-6 and its derivatives into Equation 3.7-5, and noting that $e^{i\omega t}$ is unequal to zero for all values of ωt, yields:

$$(-\omega^2 [M] + [K]) \{\phi\} = \{0\} \qquad (3.7\text{-}7)$$

Equation 3.7-7 is the characteristic equation for the classical eigenvalue problem, in which the eigenvalues are the frequencies of vibrations, ω_i , and the eigenvectors are the mode shapes, $\{\phi_i\}$, $(i=1,2,\ldots.n)$.

For each frequency ω_i , there is a corresponding solution vector $\{\phi_i\}$. It can be shown that the mode shape vectors are orthogonal with respect to the weighted stiffness matrix [K] in the n-dimensional vector space.

The eigenvectors are also orthogonal with respect to the weighted mass matrix [M].

The orthogonality of the eigenvectors is used to effect a coordinate transformation to the generalized coordinate system, in which the governing equations of motion are decoupled. Thus, the problem becomes one of solving n independent differential equations rather than n simultaneous differential equations, and because the system is linear, the principle of superposition holds, and the total response of the system oscillating simultaneously in n modes is determined by direct addition of the responses in the individual modes.

3.7.3.1.1.3 Analysis by the Response Spectrum Method

The response spectrum method is based on the fact that the modal responses can be expressed in terms of a set of convolution integrals 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 that 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 Section 3.7.3.7.

3.7.3.1.1.4 Dynamic Analysis of Seismic Category I Systems and Components

The time history and the response spectrum techniques are used as applicable for the dynamic analysis of Seismic Category I NSSS systems and components.

Dynamic analysis of piping systems, equipment, and interconnected components is as follows:

1. 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. Included are the effects of torsion, bending, shear, and axial deformations, as well as the change in stiffness due to curved members. Next, the mode shapes and the undamped natural frequencies are obtained. When

the piping system is anchored and supported at points with different excitations, the response spectrum analysis is performed using the envelope response spectrum of all attachment points. Alternately, the multiple excitation analysis methods may be used where acceleration time-histories or response spectra are applied to all piping system attachment points.

The maximum relative displacement between anchors is determined from the dynamic analysis of the structures, and the results are used for a static analysis to determine the additional stresses due to relative anchor point displacements.

2. Equipment - Each component of equipment is idealized as a mathematical model consisting of lumped masses connected by elastic members or springs.

   When the equipment is supported at more than two points located at different elevations in the building, the response spectrum analysis is performed using the enveloped response spectrum of all attachment points. Alternately, the multiple excitation analysis methods may be used where individual acceleration time-histories or response spectra are applied at each of the equipment attachment points.

   The maximum relative displacements between supports are determined from the dynamic analysis of the structures and are used for a static analysis to determine the secondary stresses due to support displacements.

3. Differential seismic movement of interconnected components - The procedure for considering differential displacements for equipment anchored and supported at points with different input motion is as follows:

a. The maximum relative displacements between the supporting points induce additional stresses in the equipment supported at these points. These stresses can be evaluated by performing a static analysis where each of the supporting points is displaced a prescribed amount. The time history of displacement at each supporting point is obtained from the corresponding acceleration time history, which is provided as input for the dynamic analysis of the total component. These displacements are used to calculate stresses by determining the peak nodal responses.

b. 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 the maximum displacement vector of its supporting points obtained from the modal displacements. This procedure is repeated for the significant modes of the structure, i.e., those that contribute most to the total displacement response at the supporting

c. point. The total stresses due to relative displacement are obtained by combining the modal results using the square root of the sum of the squares (SRSS) method. Because the maximum displacements for different modes do not occur at the same time, the SRSS method is a reliable method.

d. When a component is covered by the ASME B&PV Code, the stresses due to relative displacement as obtained above are treated as secondary stresses.

3.7.3.1.1.5 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 that, under the specified conditions, was subjected to dynamic loads equal to or greater than those to be experienced under the specified seismic inservice conditions.

2. Test data from previously tested comparable equipment that, under similar conditions, was 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 3.9.2.2 and 3.10.

3.7.3.1.2 Seismic Analysis Methods (Non-NSSS)

3.7.3.1.2.1 Equipment

Seismic qualification of equipment is performed by using either analysis or dynamic testing, or a combination of both.

3.7.3.1.2.1.1 Analysis

Seismic qualification of equipment is performed by analysis when the equipment can be adequately idealized as a system of lumped masses and stiffnesses, and when the analysis can determine its structural and functional adequacy.

The seismic analysis methods used for equipment are similar to methods used for seismic systems, and are described in Section 3.7.2. They include:

1. Response spectrum analysis

2. Time history analysis.

3.7.3.1.2.1.2 Dynamic Testing

Seismic adequacy can also be established by means of dynamic testing or previous dynamic environmental performance data that demonstrate that the equipment meets the seismic design criteria. Acceptable test methods are as follows:

1. Continuous sinusoidal test, sine beat test, or decaying sinusoidal test when the floor acceleration spectrum is a narrow band response spectrum

2. Random motion test, or equivalent, when the floor response spectra have broad-band frequency content.

3.7.3.1.2.1.3 Combination of Analysis and Dynamic Testing

Some types of equipment cannot be practically qualified by analysis or testing alone. This may be because of the size of the equipment, its complexity, or the large number of similar configurations. Experimental methods are used to aid in the formulation of the mathematical model for any piece of equipment. Mode shapes and frequencies are determined experimentally and incorporated into a mathematical model of the equipment.

3.7.3.1.2.2 Piping Systems

Reference 3.7-2 describes the methods used for seismic analysis of piping systems.

3.7.3.1.2.3 Supports for Seismic Category I Heating, Ventilating, and Air Conditioning Ducts, Cable Trays, and Conduits

These supports are qualified as follows:

1. Supports for HVAC ducts are analyzed by the response spectrum method

2. Analysis of supports for cable trays and conduits is based on the response spectrum method and/or experimental data acquired from actual seismic testing performed on cable tray and support systems.

3.7.3.2 Determination of Number of Earthquake Cycles

3.7.3.2.1 Determination of Number of Earthquake Cycles (NSSS)

3.7.3.2.1.1 Piping Systems

A total of 50 peak OBE stress cycles are postulated for fatigue evaluation.

3.7.3.2.1.2 Other Equipment and Components.

To evaluate the number of cycles engendered by a given earthquake, a typical BWR building-reactor dynamic model was excited by three different recorded time histories:

1. May 18, 1940, El Centro NS component, 29.4 seconds

2. 1952, Taft N 69° W component, 30 seconds

3. March 1957, Golden Gate S 80° E component, 13.2 seconds.

The modal response was truncated so that the response of three different frequency bandwidths could be studied: 0 to 10 hertz, 10 to 20 hertz, and 20 to 50 hertz. This was done to give a good approximation of the cyclic behavior expected from structures with different frequency content.

Enveloping the results from the three earthquakes and averaging the results from several different points of the dynamic model, the cyclic behavior described in Table 3.7-5 was formed.

Independent of earthquake or component frequency, 99.5 percent of the stress reversals occur below 75 percent of the maximum stress level, and 95 percent of the reversals occur below 50 percent of the maximum stress level.

In summary, the cyclic behavior number of fatigue cycles of a component during an earthquake is determined in the following manner:

1. The fundamental frequency and peak seismic loads are determined by a standard seismic analysis, i.e., from eigenvalue extractions and a forced response analysis.

2. The number of cycles that the component experiences are determined using Table 3.7-5, according to the frequency range within which the fundamental frequency lies.

3. For fatigue evaluation, 0.5 percent (0.005) of these cycles are conservatively assumed to be at the peak load and 4.5% (.045) at or above the three-quarter peak. The remainder of the cycles have negligible contribution to fatigue usage.

The SSE has the highest level of response. However, the encounter probability of an SSE is so small that it is not necessary to postulate the possibility of more than one SSE occurring during the 40-year life of a plant. Fatigue evaluation due to the SSE is not necessary because it is a faulted condition and thus not required by the ASME B&PV Code, Section III.

The OBE is an upset condition and therefore must be included in fatigue evaluations, according to the ASME B&PV Code, Section III.

Investigation of seismic histories of many plants shows that, during a 40-year life, it is probable that five earthquakes will occur with 10 percent of the proposed SSE intensity, and one earthquake will occur with approximately 20 percent of the proposed SSE intensity. To cover the combined effects of these earthquakes and the cumulative effects of even lesser earthquakes, ten peak OBE stress cycles are postulated for fatigue evaluation.

3.7.3.2.2 Determination of Number of Earthquake Cycles (Non-NSSS)

In general, the design of the equipment is not fatigue controlled because the equipment is designed to remain 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 on tested equipment are accounted for by five OBEs and one SSE. The minimum duration for each condition is 20 seconds. Consequently, the number of cycles of the earthquake has been accounted for.

To conduct a fatigue evaluation for nuclear Class I piping, the number of cycles for a given load set is obtained by considering ten maximum stress cycles per earthquake, and assuming five OBEs and one SSE to occur within the life of the plant.

3.7.3.3 Procedure Used for Modeling

3.7.3.3.1 Procedure Used for Modeling (NSSS)

An important step in the seismic analysis of Seismic Category I systems, components, or structures is the procedure used for modeling. The techniques currently being used are represented by lumped masses and a set of spring dashpots idealizing both the inertial and stiffness properties of the system. The details of the mathematical models are determined by the complexity of the actual system and the information required for the analysis.

The modeling procedure uses an adequate number of masses or degrees of freedom to determine the response of all Seismic Category I and applicable non-Seismic Category I structures and equipment. The refinement of all dynamic models is sufficient for analysis up through at least 60 hertz. The number of masses or degrees of freedom is adequate so that any further refinement does not result in more than a 10 percent increase in the final response. This means that the number of degrees of freedom is equal to at least twice the number of modes with frequencies up through 60 hertz.

3.7.3.3.1.1 Modeling of Reactor Pressure Vessel and Internals

The seismic loads on the reactor pressure vessel (RPV) and its internals are based on a dynamic analysis of the Reactor Building with the appropriate forcing function supplied at ground level. For this analysis, the models shown on Figure 3.7-161 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, and the effects of bending and shear are included. Mass points are located at all points of critical interest, such as anchor points, supports, and points of discontinuity. In addition, mass points are chosen such that the total mass of the structure is generally uniformly distributed over all the mass points, and the full range of frequency of response of interest is adequately represented. Furthermore, to facilitate hydrodynamic mass calculations, several mass points, (fuel, shroud, vessel) are selected at the same elevation. The various lengths of control rod drive (CRD) housings are grouped into the two representative lengths shown on Figure 3.7-161. 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 frequencies of the CRD housings result in very small seismic loads. Furthermore, the small frequency differences

between the various housings due to the length differences result in negligible differences in dynamic response. Hence, the modeling of intermediate length members becomes unnecessary. Not included in the mathematical model are the stiffness of light components, such as jet pumps, in-core guide tubes and housings, spargers, and their supply headers. This is done to reduce the complexity of the dynamic model and is justified because dynamic interaction is not significant. Floor response spectra generated from the analysis are used for the seismic responses of these components.

The presence of water and other structural components, e.g., fuel within the RPV, introduces a dynamic coupling effect. Dynamic effects of water enclosed by the RPV are accounted for by introduction of a hydrodynamic mass matrix, which serves to link the acceleration terms of the equations of motion of points at the same elevation in concentric cylinders with a fluid entrapped in the annulus. Details of the hydrodynamic mass derivation are given in Reference 3.7-3. The seismic model of the RPV and its internals has two horizontal coordinates for each mass point considered in the analysis. The remaining translational coordinate (vertical) is excluded because the vertical frequencies of the RPV and internals are well above the significant horizontal frequencies. Furthermore, all support structures, the building, and the primary containment walls have a common centerline; therefore, 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 the components, whichever is more conservative. The two rotational coordinates about each node point are excluded because the contribution of rotating 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 in its own plane is extremely stiff and is 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.

3.7.3.3.1.2 Modeling of Piping Systems

The continuous piping system is modeled as an assemblage of three dimensional straight or curved pipe elements. The mass of each pipe element is lumped at nodes connected by a weightless elastic member, representing the physical properties of each segment. The pipe lengths between mass points are no greater than a length that would have a natural frequency of 33 hertz when calculated as a simply supported beam. In addition, mass points are located at the beginning and end of each elbow, tee, and other such components as main valves, relief valves, and pumps. The torsional effects of the valve operators and other equipment with offset centers of gravity with respect to the center line of the pipe are included in the analytical model.

The criteria employed for decoupling the main steam and recirculation piping systems, thus establishing the analytical models necessary to perform seismic analyses, are given below:

1. The small branch lines, 6-inch diameter and less, are decoupled from the main steam and recirculation piping systems and analyzed separately.

2. The stiffness of all the anchors and the supporting steel is large enough to effectively decouple the piping on either side of the anchor for analytic and code jurisdictional boundary purposes. The RPV is very stiff and massive compared to the piping system; thus, during normal operating conditions, the RPV is also assumed to act as an anchor. Penetration assemblies (head fittings) are also very stiff compared to the piping system and are assumed to act as an anchor. The stiffness matrix at the attachment location of the process pipe head fitting, i.e., main steam, reactor core isolation cooling (RCIC),

residual heat removal (RHR) supply, or RHR return, is sufficiently high to decouple the penetration assembly from the process pipe. GE analysis indicates that a satisfactory minimum stiffness for this attachment point is the stiffness in bending and torsion of a cantilever that is equal to a pipe section of the same size as the process pipe, and equal in length to three times the outer diameter of the process pipe.

3.7.3.3.1.3 Modeling of Equipment

For dynamic analyses, Seismic Category I equipment is represented by lumped mass systems that 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 significant if the corresponding natural frequencies are less than 33 hertz and the stresses calculated from these modes are greater than 10 percent of the total stresses obtained from lower modes.

2. Mass is lumped at any point of significant concentrated weight, e.g., the motor in the analysis of the pump motor stand and the impeller in the analysis of the pump shaft.

3. If the equipment has a free end overhang span with flexibility that 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 lower the natural frequencies of the equipment. This results in a conservative analysis, because the equipment frequencies are in the higher

spectral range of response spectra. 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 ensures conservative dynamic loads, since the equipment frequencies are always higher than the frequencies at which the spectral peaks occur. If such is not the case, the model is adjusted to give more conservative results.

3.7.3.3.1.4 Field Location of Supports and Restraints

The final location of seismic supports and restraints for Seismic Category I piping, piping system components, and equipment, including the placement of snubbers, is checked against the drawings and instructions issued by the engineer. An additional examination of these as-built supports and restraining devices is made to ensure that the location and characteristics of these supports and restraining devices are consistent with the dynamic and static analyses of the systems. The final analyses of the as-built systems are performed as necessary, and the final, certified, as-built design reports are issued.

3.7.3.3.2 Procedure Used for Modeling (Non-NSSS)

Mathematical models are used that describe the mass and stiffness properties of the equipment. The models define the dynamic behavior of the equipment within the frequency range of interest.

The boundary conditions are modeled to reflect the actual 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 dimensional, lumped mass, plane frame models. The masses are lumped at the center of the ducts. Sections 2 and 3 of Reference 3.7-2 discuss the techniques and procedures used to model piping other than buried piping.

3.7.3.4 Basis for Selection of Frequencies

3.7.3.4.1 Basis for Selection of Frequencies (NSSS)

All frequencies in the range of 0.25 to 33 hertz are considered in the analysis and testing of systems, components, and equipment. These frequencies are excited under the seismic excitation.

If the fundamental frequency of a component is greater than or equal to 33 hertz, it is treated as seismically rigid and analyzed accordingly. Frequencies less than 0.25 hertz are not considered, as they represent very flexible structures not encountered in this plant.

The frequency range between 0.25 hertz and 33 hertz covers the range of the broad band response spectrum used in the design.

3.7.3.4.2 Basis for Selection of Frequencies (Non-NSSS)

The natural frequencies of components are calculated. Only those modes that have natural frequencies less than 33 hertz are considered in the dynamic analysis. If a component has a fundamental frequency equal to or greater than 33 hertz, it is considered seismically rigid. In this case, the acceleration response of the component equals the floor acceleration. If the natural frequency of the component falls within the broadened peak of the response spectrum curve, then it is designed to take the peak response load.

3.7.3.5 Use of Equivalent Static Load Method of Analysis

3.7.3.5.1 Use of Equivalent Static Load Method of Analysis (NSSS)

When the natural frequency of a structure or component is unknown, it may be analyzed by applying a static force at the center of mass. To account conservatively for the possibility of more than one significant dynamic mode, the static force is calculated as 1.5

times the mass times the maximum spectral acceleration from the floor response spectra of the points of attachment of multi supported structures. The factor of 1.5 is adequate for a simple beam type of structure. For simply supported structures, the peak spectral acceleration is used. For other, more complicated structures, the factor used is justified.

3.7.3.5.2 Use of Equivalent Static Load Method of Analysis (Non-NSSS)

The equivalent static load method 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 weight of the equipment times the peak value of the response spectrum curve. Seismic acceleration coefficients for multiple degree of freedom systems, which may be in the resonance region of the amplified response spectra curves, are increased by 50 percent to account conservatively for the increased modal participation.

Appendix D of Reference 3.7-2 discusses the use of the equivalent static load method of analysis, as applicable to piping.

3.7.3.6 Three Components of Earthquake Motion

3.7.3.6.1 Three Components of Earthquake Motion (NSSS)

The simultaneous use of three components of earthquake motion was not a design basis requirement of the construction permit for this plant. However, the NSSS components and equipment are evaluated to the requirements of Regulatory Guide 1.92.

3.7.3.6.1.1 Response Spectrum Method

Response spectra generated by GE are developed considering three components of earthquake motion. The individual floor response

spectra in each orthogonal direction are obtained by the SRSS combination of the collinear contribution due to the three directions of earthquake motion. These are used to predict the total response at each frequency.

3.7.3.6.1.2 Time History Method

When the time history method of analysis is used, the time history responses from each of the three components of the earthquake motion are combined algebraically at each time step. The maximum response is obtained from this combined time solution.

3.7.3.6.2 Three Components of Earthquake Motion (Non-NSSS)

For equipment, cable trays, supports for cable trays, and HVAC ducts, the three spatial components of the earthquake are considered in the same manner as structures described in Section 3.7.2.6.

The criteria used for combining the results of horizontal and vertical seismic responses of piping systems are described in Section 5.1 of Reference 3.7-2.

3.7.3.7 Combination of Modal Responses

3.7.3.7.1 Combination of Modal Responses (NSSS)

All piping and equipment analyzed or supplied by GE is evaluated to the requirements of Regulatory Guide 1.92.

When the response spectrum method of modal analysis is used, all modes except the closely spaced modes (i.e., the difference between any two natural frequencies is equal to or less than 10 percent) are combined by the square root of the sum of the squares (SRSS), as described in Section 3.7.3.7.1.1. Closely spaced modes are combined by the double sum method described in Section 3.7.3.7.1.2.

In the time history method of dynamic analysis, the vector sum at every step is used to calculate the combined response. The use of the time history method precludes the need to consider modal spacing.

3.7.3.7.1.1 Square Root of the Sum of the Squares

The square root of the sum of the squares (SRSS) combination of modal responses is defined mathematically as:

$$R = \left[\sum_{i=1}^{n} (R_i)^2 \right]^{0.5} \qquad (3.7\text{-}8)$$

where:

R = combined response

R_i = response due to the i^{th} mode

n = number of modes considered in the analysis

3.7.3.7.1.2 Procedure for Combining Closely Spaced Modal Responses

The double sum method is used to combine the responses of closely spaced modes when the response spectrum method of modal dynamic analysis is used. This method is defined mathematically as:

$$R = \left[\sum_{k=1}^{N} \sum_{s=1}^{N} \left| R_k R_s \right| \left| E_{ks} 0 \right. \right]^{0.5} \qquad (3.7\text{-}9)$$

where R is the representative maximum value of a particular response of a given element to a given component of excitation, R_k is the peak value of the response of the element due to the k^{th} mode, and N is the number of significant modes considered in the modal response

combination. In addition, R_s is the peak value of the response of the element attributed to the s^{th} mode. Also,

$$E_{ks} = \left[1 + \left[\frac{\omega'_k \quad \omega'_s}{\beta'_k \; \omega_k \quad \beta'_s \; \omega_s} \right]^2 \right]^{-1} \qquad (3.7\text{-}10)$$

in which :

$$\omega'_k = \left[\omega_k \quad 1 - \beta_k^2 \right] \qquad (3.7\text{-}11)$$

and

$$\beta'_k = \beta_k + \frac{2}{t_d \omega_k} \qquad (3.7\text{-}12)$$

where: ω_k and β_k are the modal frequency and the damping ration in the k^{th} mode, respectively, and t_d is the duration of the earthquake.

3.7.3.7.2 Combination of Modal Responses (Non-NSSS)

The modal responses of equipment are combined by the SRSS method. The absolute values of two closely spaced modes are added before being combined with the other modes by the SRSS method. Two consecutive modes are defined as closely spaced when their frequencies differ from each other by 10 percent or less.

3.7.3.8 Analytical Procedure for Piping

3.7.3.8.1 Analytical Procedure for Piping (NSSS)

The analytical procedures for piping analyses are described in Section 3.7.3.1.1.4. Methods to include differential piping support movements at different support points are also described.

3.7.3.8.2 Analytical Procedure for Piping (Non-NSSS)

The design criteria and the analytical procedures applicable to piping systems are as described in Section 2 of Reference 3.7-2. The methods used to consider differential piping support movements at different support points are described in Section 4 of Reference 3.7-2.

3.7.3.9 Multiple Supported Equipment Components With Distinct Inputs

3.7.3.9.1 Multiple Supported Equipment Components With Distinct Inputs (NSSS)

The procedure and criteria for the analysis of multiple supported equipment components with distinct inputs are described in Section 3.7.3.1.1.4.

3.7.3.9.2 Multiple Supported Equipment Components With Distinct Inputs (Non-NSSS)

For piping systems, cable trays, and HVAC ducts of which the supports have two or more distinct inputs, a response spectrum curve envelops the curves at all support locations. Section 4 of Reference 3.7-2 discusses the methods used for analysis of multiple supported piping systems due to seismic anchor differential movement.

When a piping system is subjected to a distinct input consisting of significantly varying amplitudes at supports, nozzles, or anchors, a more realistic approach using the multiple support excitation analysis method is considered. Either time history or multiple response spectrum analysis methods are used. The same analytical procedures and criteria described in Section 3.7.3.8.2 are applied for this type of analysis, with the additional parameters coming from the unit vector r, which is computed as the response due to static support displacement. The time-history method uses the $U = \phi\eta + rU_r$ in the equation of motion for the unsupported degree of freedom, U (partitioned dynamic equilibrium matrix). ϕ is the orthonormal mode shapes, η is the matrix of modal amplitudes, and U is the support displacement amplitudes. The multiple support response spectrum method uses the unit r vector to calculate a modal participation factor for each distinct group of support input that directly corresponds to a modal spectra value. After all spectra participation, absolute summation of supports due to each distinct group, the total response is calculated by performing modal and spatial combination in accordance to Regulatory Guide 1.92.

3.7.3.10 Use of Constant Vertical Static Factors

3.7.3.10.1 Use of Constant Vertical Static Factors (NSSS)

Constant vertical static factors are not used by General Electric.

3.7.3.10.2 Use of Constant Vertical Static Factors (Non-NSSS)

Constant vertical static factors are not used in the seismic design of subsystems.

3.7.3.11 Torsional Effects of Eccentric Masses

3.7.3.11.1 Torsional Effects of Eccentric Masses (NSSS)

Torsional effects of eccentric masses for the piping systems are discussed in Section 3.7.3.3.1.2.

The RPV is an axisymmetric model with no built-in eccentricity. Therefore, the torsional effects on the RPV are only those associated with the Reactor Building model and are accounted for in the Reactor Building model, as described in Section 3.7.2.3.

3.7.3.11.2 Torsional Effects of Eccentric Masses (Non-NSSS)

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 3.7-2.

3.7.3.12 Buried Seismic Category I Piping Systems and Tunnels (Non-NSSS)

Buried Seismic Category I piping that is connected between the intake structure and the Reactor Building is analyzed and designed for seismic effects in accordance with Section 6 of Reference 3.7-1 and Part III of Reference 3.7-7.

Additional requirements for design and manufacture of the prestressed concrete pipe are in accordance with American Water Works Association (AWWA) Standard C-301. The flexible joints are designed to accommodate the maximum postulated axial and rotational movements induced by a seismic event, thereby allowing the system to follow the displacements that take place in the surrounding soil. The piping is also founded on engineered backfill, as discussed in Section 3.8.6.1. This base will essentially eliminate any soil settlement or arching so that the joint integrity and water tightness can be maintained during and after a seismic event.

In addition, the service water pipeline has been evaluated using the recommended guidelines provided in NUREG/CR-1161 (Reference 3.7-8). In all cases, the final design meets or exceeds the allowables required by NUREG/CR-1161.

There is no buried tunnel at HCGS.

3.7.3.13 Interaction of Other Piping With Seismic Category I Piping

3.7.3.13.1 Interaction of Other Piping With Seismic Category I Piping (NSSS)

When other non-Seismic Category I piping is attached to Seismic Category I piping, the other piping is analytically coupled sufficiently so as not to degrade significantly the accuracy of the analysis of the Seismic Category I piping. Furthermore, other piping is designed to withstand an SSE to prevent failure of Seismic Category I piping.

3.7.3.13.2 Interaction of Other Piping With Seismic Category I Piping (Non-NSSS)

The techniques used to consider the interaction of Seismic Category I piping with non-Seismic Category I piping are discussed in Section 3.4 of Reference 3.7-2.

3.7.3.14 Seismic Analysis for Reactor Internals (NSSS)

The modeling of the RPV and the internals is discussed in Section 3.7.3.3.1.1. The damping values are given in Table 3.7-1. The seismic model is shown on Figure 3.7-161, and a summary of loading conditions, evaluation criteria, calculated maximum stresses in the selected locations, and the allowable stresses are given in Table 3.9-5.

3.7.3.15 Analysis Procedures for Damping

3.7.3.15.1 Analysis Procedures for Damping (NSSS)

In a linear dynamic analysis, the procedure used to account properly 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 element that contributes to the complete stiffness of the system.

2. Perform a modal analysis of the linear system model. This will result in the eigenvector matrices (ϕ) normalized such that:

$$[\phi^T] \; [K] \; [\phi] = [W^2] \qquad (3.7\text{-}13)$$

and

$$[\phi^T] \; [M] \; [\phi] = [I] \qquad (3.7\text{-}14)$$

where [K] is the stiffness matrix, $[W^2]$ is the circular natural frequency, and $[\phi^T]$ is the transpose of the mode shape matrix ϕ, which contains all translational and rotational coordinates.

3. Using the strain energy of the individual components as a weighting function, the following equation is used to obtain a suitable damping ratio (β_i) for the mode i:

$$\beta = \frac{\sum_{j=1}^{N} [\beta_j (\phi_i^T \; K \; \phi_i)_j]}{W_i^2} \qquad (3.7\text{-}15)$$

where:

N = total number of structural elements

ϕ_i = components of the i mode eigenvector corresponding to the j beam element

β_j = percent damping associated with element j

K - stiffness contribution of element j

W_i^2 - circular natural frequency of mode i

ϕ_i^T - transpose of ϕ defined above

β - percent critical damping associated with element j

3.7.3.15.2 Analysis Procedure for Damping (Non-NSSS)

In general, a single damping value, as shown in Table 3.7-2, is used for the analysis of Seismic Category I subsystems.

3.7.3.16 SRP Rule Review

Acceptance criterion II.2(b) of SRP Section 3.7.3 establishes the number of OBE cycles to be assumed for component fatigue analysis. It requires 50 OBE peak stress cycles for the life of the plant. In the HCGS design, 50 peak OBE stress cycles are postulated for the NSSS piping in accordance with the SRP.

However, 10 peak OBE cycles are postulated for other NSSS equipment and components. This 10-cycle approach has been approved by the NRC-MEB on the grounds of "equivalent level of safety." This approval was contingent upon GE's presentation of the fatigue calculation of the most limiting component in the BWR 4 product line; see Reference 3.7-5. This presentation took place during the Limerick Generating Station licensing process. The results of the fatigue calculation for the most limiting RPV Component for BWR/4 product line were presented and were found acceptable by NRC-MEB.

3.7.4 Seismic Instrumentation

3.7.4.1 Comparison With Regulatory Guide 1.12, Revision 1

The seismic instrumentation system of Hope Creek Generating Station (HCGS) complies with Regulatory Guide 1.12, Revision 1, except as discussed below.

The response spectrum recorders required by Position 1.C of Regulatory Guide 1.12 are not supplied as discrete instruments. Instead, triaxial time-history accelerographs are provided at the required locations. Together with a multichannel magnetic tape recorder and a response spectrum analyzer, this system yields more complete information than that from response-spectrum recorders. Following a seismic event, recorded acceleration data from the triaxial time history accelerographs are fed into a response spectrum analyzer, one channel at a time, to produce seismic response spectra. Permanent records of the response spectra are provided by an x-y plotter. This system meets the intent of Position 1.C of Regulatory Guide 1.12.

Position 3 of the Regulatory Guide does not apply, since the HCGS safe shutdown earthquake (SSE) is 0.2 g.

3.7.4.2 Location and Description of Instrumentation

3.7.4.2.1 Triaxial Time History Accelerographs (T/A)

One T/A is provided at each of the following locations:

1. Free field, 60 feet below grade in the northwest quadrant of the plant site, 500 feet from the reactor building, on the Vincentown Formation

2. Primary containment foundation, Elevation 54 feet, in the northeast quadrant of the Reactor Building

3. Reactor Building, Elevation 201 feet, in the northwest quadrant on the refueling floor

4. Reactor piping (core spray piping entering the reactor), Elevation 115 feet, in the north quadrant of the drywell along the reactor centerline

5. Auxiliary Building foundation, Elevation 54 feet, in the northwest quadrant of the Auxiliary Building.

T/As produce a record of the time varying acceleration at the sensor location. The signal amplitude is proportional to the instantaneous acceleration value of the point on the structure/component to which the T/A is attached.

Each T/A contains three accelerometers mounted in a mutually orthogonal array. All T/As have their principal axes oriented identically, with one horizontal axis parallel to the major horizontal axis assumed in the plant seismic analysis.

A triaxial seismic trigger (S/T), sensitive in north-south, east-west, and vertical directions, is provided at the primary containment foundation to start the T/A sensor recording system. The multichannel magnetic tape recorders for the system are housed in the system control panel, which is located in the control complex. The tape recording system is capable of simultaneously recording signals from all of the T/As.

3.7.4.2.2 Triaxial Peak Recording Accelerographs (P/A)

One P/A is provided at a location on the reactor equipment (reactor support lateral truss), a location on the reactor piping (core spray piping entering the reactor), and on Seismic Category I equipment at the most significant location outside of the primary containment structure (on the station service water pump piping, in the intake structure).

P/As record the actual peak response at the sensor location. Each P/A contains three accelerometers mounted in a mutually orthogonal array. All P/As have their principal axes oriented identically, with one horizontal axis parallel to the major horizontal axis assumed in the seismic analysis. Data from the P/As are manually retrieved following an earthquake.

3.7.4.2.3 Triaxial Response Spectrum Recorder (R/R)

One R/R is provided at the primary containment foundation. Immediate main control room indication is provided by the system response spectrum annunciator.

The R/R is comprised of three R/Rs mounted in a mutually orthogonal array, with one horizontal R/R mounted parallel to the major horizontal axis assumed in the seismic analysis.

Each R/R consists of a spring mass system with 16 vibratory reeds responsive to 16 discrete frequencies. Each reed is fitted with a stylus that inscribes a mark on a permanent recording plate. This scribe mark is proportional to the maximum acceleration to which its respective reed has been subjected. These data are manually retrieved following an earthquake.

Each R/R also contains 16 response spectrum switches (R/S), integrally related to each of the 16 vibratory reeds. When any reed senses acceleration greater than the preset level, the R/S contacts close, providing main control room annunciation at the response spectrum annunciator. Annunciation is provided independently for each of the 16 discrete frequencies.

3.7.4.2.4 Triaxial Response Spectrum Annunciator (R/A)

One R/A assembly is provided in the main control room, mounted as an insert in the main vertical board.

The R/A provides visual annunciation that predetermined acceleration limits, both warning and design, within the response spectrum have been exceeded. The R/A contains three banks of indicator lamps, one bank for each of three mutually perpendicular axes. Each bank contains a double row of 16 colored indicator lamps per row. Amber lamps indicate acceleration levels approaching design limits, while red lamps indicate that design limits have been exceeded. Each pair of indicator lamps per bank is associated with a single vibratory reed.

3.7.4.2.5 Triaxial Seismic Switch (S/S)

One S/S is provided on the primary containment foundation.

The S/S is provided to monitor the occurrence of an operating basis earthquake (OBE) acceleration at the mounting location. The S/S contains three sensor relay modules mounted in a mutually orthogonal array, with one horizontal axis parallel to the major horizontal axis assumed in the seismic analysis. When any axis is excited beyond the OBE acceleration level, the S/S contacts close, providing immediate main control room annunciation.

3.7.4.2.6 System Control Panel

The system control panel is located in the upper control equipment room of the control complex. It houses the recording and playback units used in conjunction with the T/A sensors to produce a time history and frequency amplitude record of a seismic event. The panel also contains signal conditioning equipment associated with the R/R, R/A, S/S, and the system power supply modules.

A cassette tape deck is provided to record the output signal of each T/A.

Each tape deck records four channels of information, i.e., the three output signals from each accelerometer within the respective T/A and a timing mark signal.

A cassette tape playback unit is included to provide an immediate analog record of acceleration versus time on the integral strip chart recorder. Any one signal channel of a tape may be selected along with timing marks to generate a recording on the strip chart recorder. In addition, all four channels are available simultaneously for use with the auxiliary spectrum analyzers.

3.7.4.3 Control Room Operator Notification

Activation of the S/S causes audible and visual annunciation in the main control room to alert the main control room operator that OBE acceleration levels have been exceeded.

The peak acceleration levels experienced by the primary containment foundation and the free field are determined following a seismic event by playing back the recorded T/A data from the respective sensor tape, and reading the peak value from the strip chart recorder located in the system control panel discussed in Section 3.7.4.2.6. Alternatively, the sensor tape can be examined on a spectrum analyzer and the results plotted on an x-y plotter located in the control complex.

The R/R located on the primary containment foundation provides immediate indication in the main control room of any design response spectra values for discrete frequencies that have been exceeded following a seismic event.

3.7.4.4 Comparison of Measured and Predicted Responses

Initial determination of the seismic event level is performed immediately following a seismic event. A comparison of the measured response spectra from the primary containment foundation and the calculated OBE response spectra is made. If either the response spectra from the R/R or the peak acceleration from the T/A, experienced at the primary containment foundation or the free field, exceed the allowable OBE acceleration level, the plant is immediately placed in shutdown condition. To resume operation, a

detailed seismic analysis will be performed to compare the calculated response spectra with measured response spectra at locations where sensors were installed. Time history records from free field triaxial time history accelerographs shall be used as input ground motions for the detailed soil structure interaction analysis. Newly calculated structural response spectra will also be compared with design floor response spectra to determine the seismic effects on structures, components and equipment. A report on the analysis results will be submitted to the NRC for permission to restart plant operation. An outline of the post-seismic event plant procedures is provided on Figure 3.7-162.

3.7.4.5 Inservice Surveillance

Each seismic instrument is periodically demonstrated operable in accordance with Section 7.7.1.10. The seismic instruments are designed to ensure that channel checks, channel calibration, and channel functional tests can be performed to frequencies identified in Section 7.7.1.10.

3.7.5 References

3.7-1 Bechtel Power Corporation, "Seismic Analysis of Structures and Equipments for Nuclear Power Plants, Bechtel Topical Report" BC-TOP-4A, Revision 3, November 1974.

3.7-2 Bechtel Power Corporation, "Seismic Analysis of Piping Systems," BP-TOP-1, Revision 3, January 1976.

3.7-3 L. K. Liu, "Seismic Analysis of the 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.

3.7-4 R. B. Reimer, et al, "Evaluation of Pacoima Dam Accelerogram," Proceedings of the Fifth World Conference on Earthquake Engineering, Rome, Italy, 1973.

3.7-5 R. Bosnak, Letter to R. Artigas, "Number of OBE Fatigue Cycles in the BWR NSSS Design," February 18, 1973.

3.7-6 J. Lysmer, et al, "FLUSH A Computer Program for Approximate 3-D Analysis of Soil-Structure Interaction Problems," Report No. EERC 75-30, University of California, Berkeley, 1975.

3.7-7 Public Service Electric and Gas, "Additional Site Stability Evaluation Hope Creek Generation Station" December, 1976.

3.7-8 NUREG/CR-1161, Recommended Revisions to Nuclear Regulatory Commission Seismic Design Criteria, prepared by Lawrence Livermore Laboratory for U.S. Nuclear Regulatory Commission, May 1980.

TABLE 3.7-1

CRITICAL DAMPING RATIOS FOR ANALYSIS OF NSSS MATERIALS

Item	Percent Critical Damping OBE Condition	SSE Condition
Welded structural assemblies	1.0	2.0
Steel frame structures	2.0	3.0
Equipment	2.0[(2)]	3.0
Bolted or riveted structural assemblies	4.0	7.0
Vital piping systems:		
Diameter greater than 12 in.	2.0	3.0
Diameter less than or equal to 12 in.	1.0	2.0
Reactor pressure vessel, shroud support, support skirt, shroud head, and separator	2.0	4.0
Control rod drive (CRD) housings and guide tubes	1.0	2.0
Fuel assemblies	4.0 (vert.) 6.0 (horiz.)	6.0
CRD restraint bellows and stabilizer	2.0	4.0
Primary containment, RPV pedestal, and biological shield wall	4.0	7.0

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(1) In the dynamic analysis of active components as defined in Regulatory Guide 1.48, this value is also used for SSE.

TABLE 3.7-2

DAMPING VALUES FOR SEISMIC CATEGORY I STRUCTURES, SYSTEMS, AND COMPONENTS[1] (NON-NSSS)

Structure or Component	Modal Damping Values, Percent of Critical Damping OBE	SSE
Equipment and large diameter piping systems (pipe diameter in excess of 12 inches)[2]	2[3]	3
Small diameter piping systems (pipe diameter equal to or less than 12 inches)	1	2
Welded steel structures	2	4
Bolted steel structures	4	7
Prestressed concrete structures	2	5
Reinforced concrete structures	4	7

(1) Damping values for foundation material and for foundation structure interaction analysis are not included in this table.

(2) This includes both material and structural damping. If the piping system consists of only one or two spans with little structural damping, the values for small diameter piping are used.

(3) In the dynamic analysis of active components as defined in Regulatory Guide 1.48, this value is also used for SSE.

TABLE 3.7-3

FOUNDATION DATA FOR MAJOR STRUCTURES

Structure	Plan Dimensions of Foundation	Structural Height(1), ft	Foundation Depth(2), ft
Reactor Building and South Radwaste Building	192 ft 6 in. x 312 ft 0 in.	261	62
North Radwaste Building and plant cancelled area	192 ft 6 in. x 312 ft 0 in.	132	62
Diesel generator, control and central radwaste buildings	164 ft 8 in. x 312 ft 0 in.	158	62
Turbine Building	364 ft 4 in. x 194 ft 10 in.	158	62
Administration facility	265 ft 6 in. x 194 ft 10 in.	158	62
SSWS intake structure	103 ft 6 in. x 94 ft 0 in.	69.5	34.5

(1) These heights are measured from the bottom of the base mat to the highest point of the structure.

(2) These depths are measured from the bottom of the base mat to grade elevation.

TABLE 3.7-4

SIGNIFICANT NATURAL FREQUENCIES FOR SEISMIC CATEGORY I STRUCTURES

	Reactor Building		Auxiliary Building		Intake Structure	
Direction	Mode No.	Freq, Hz	Mode No.	Freq, Hz	Mode No.	Freq, Hz
North-			4	6.20	5	9.5
south	4	4.09	8	15.28	6	10.0
	10	8.87	-	-	7	10.0
	11	8.97	-	-	19	26.5
	17	12.34	-	-	-	-
East-	5	4.25	5	6.48	11	13.5
west	7	4.45	9	16.54	22	36.7
	10	8.87	11	19.80	-	-
	11	8.97	-	-	-	-
	18	12.74	-	-	-	-
Vertical	15	11.42	7	13.24	21	35.3
	17	12.34	9	16.54	-	-
	18	12.74	10	17.54	-	-
	27	18.35	11	19.80	-	-
	45	28.24	-	-	-	-
Torsion	3	2.50	6	12.36	-	-

TABLE 3.7-5

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
Number of seismic cycles - 0.5 percent cycles between 75 percent and 100 percent of peak loads	0.8	1.8	3.2
Number of seismic cycles - 4.5 percent cycles between 50 percent and 75 percent of peak loads	7.5	16.2	28.9

TABLE 3.7-6

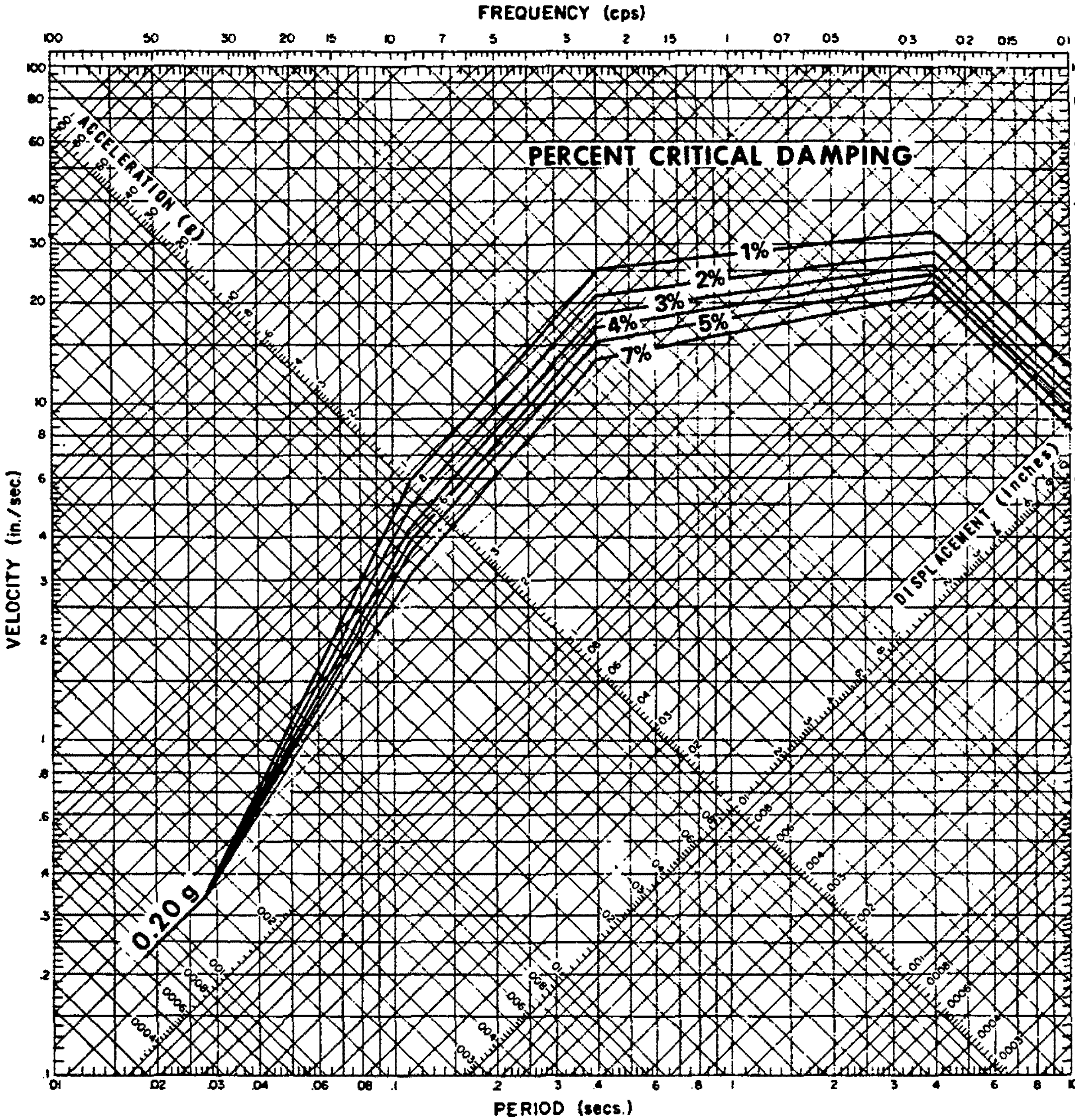
ACTUAL AND THE WORST COMPUTED STRUCTURAL GAPS FOR CATEGORY I STRUCTURES

Category I and Non-Category I Structures	Actual Structural Gaps Provided (in.)	The Worst Computed Structural Gaps Required (in.)
Reactor Building and Auxiliary Building	1.75	0.74
Reactor Building and Turbine Building	2.00	0.92
Auxiliary Building and Plant Cancelled Area	1.75	0.74
Auxiliary Building and Turbine Building	2.00	0.91
Plant Cancelled Area and Administration Facility	2.00	0.88
Auxiliary Building and Administration Facility	2.00	0.87

TABLE 3.7-7

FLOOR RESPONSE SPECTRA PEAK BROADENING CRITERIA (NSSS)

Direction	Response Spectra Peak Broadening Criteria
North-South and East-West	±25 percent for any spectral peak between approximately 3.5 and 6 Hz (Peak response in this frequency range is due to soil response).
	±15 percent for all other responses
Vertical	±15 percent for all responses



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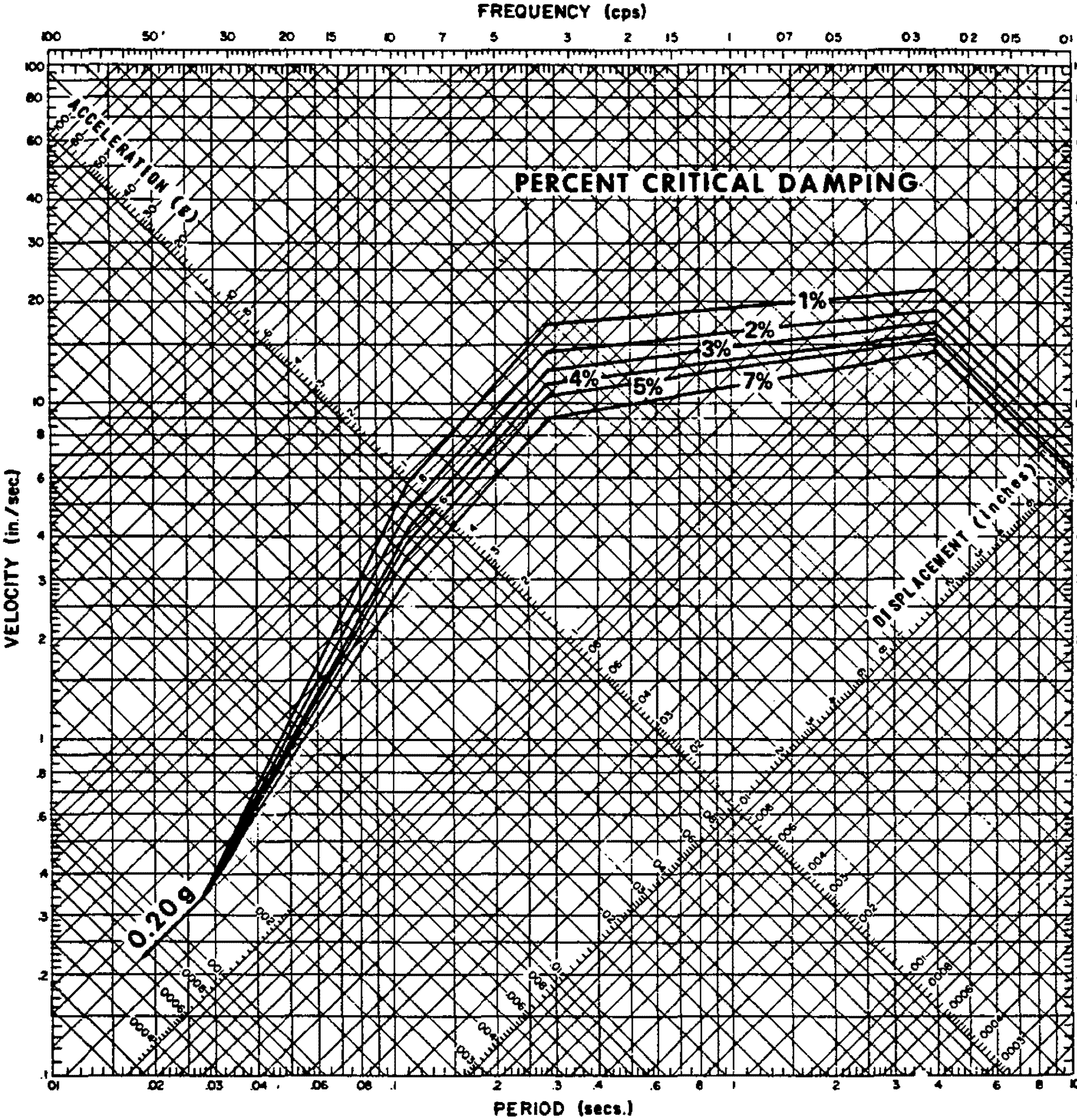
APRIL 11, 1988

PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

SSE HORIZONTAL
GROUND SPECTRA
0.20g

UPDATED FSAR

FIGURE 3.7-1



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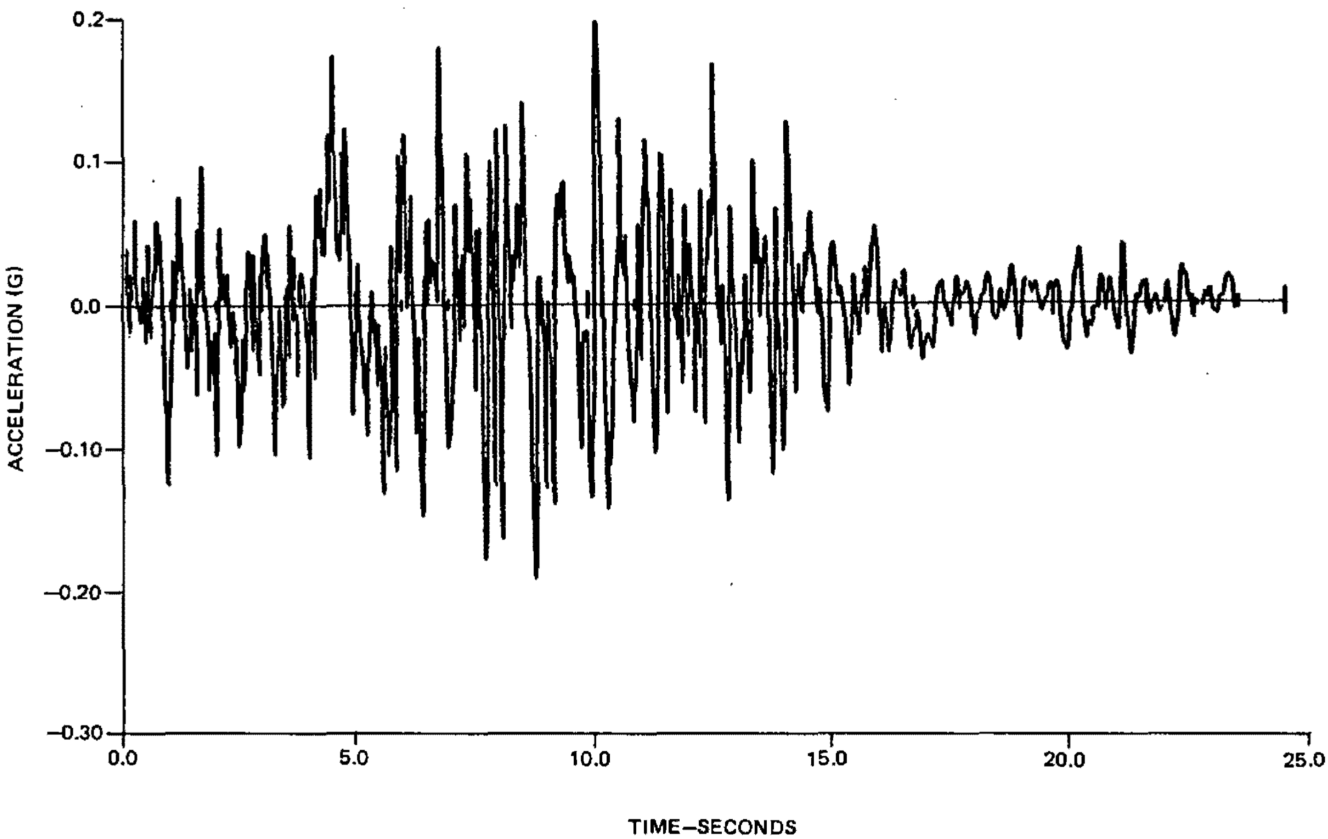
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

SSE VERTICAL
GROUND SPECTRA
0.20g

UPDATED FSAR

FIGURE 3.7-2



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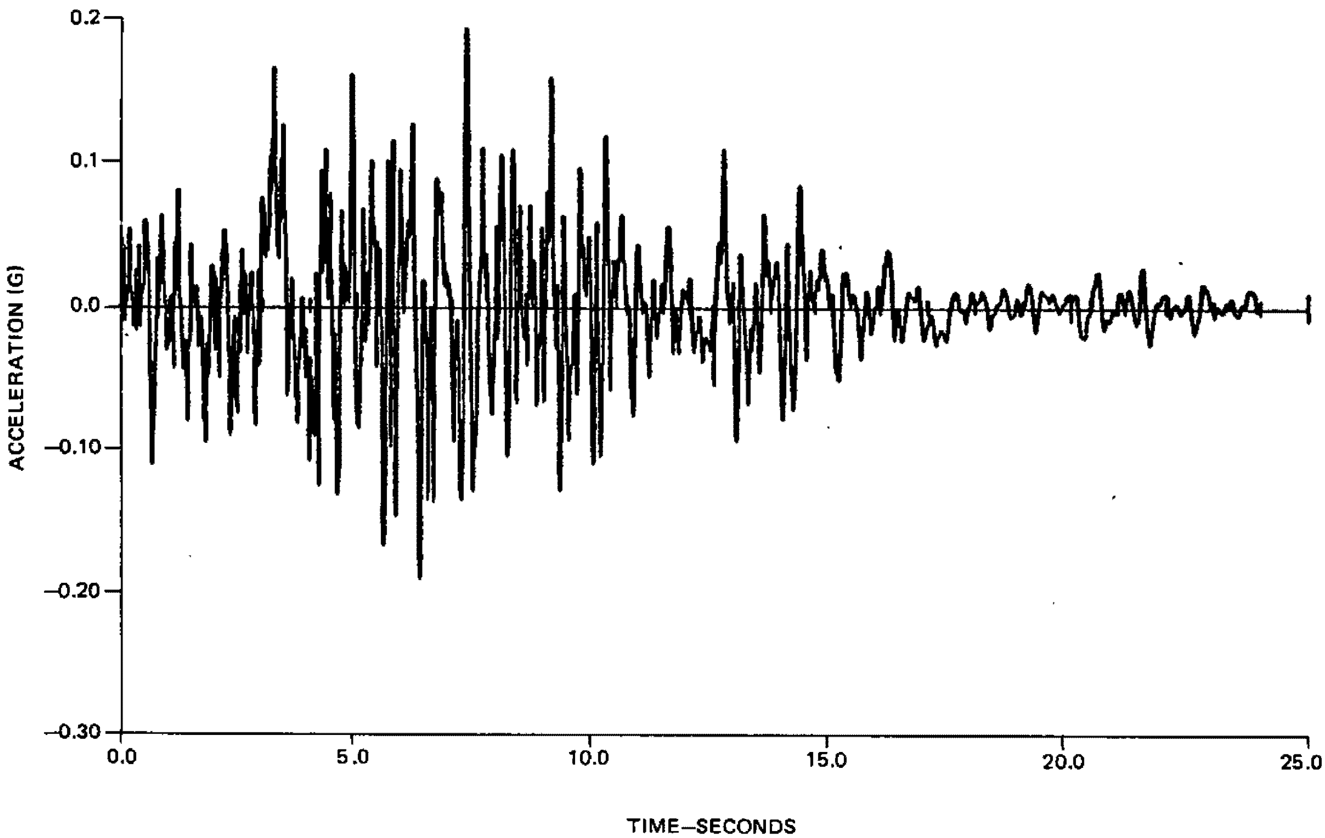
HOPE CREEK NUCLEAR GENERATING STATION

SYNTHETIC TIME-HISTORY OF THE HORIZONTAL COMPONENT OF THE DESIGN EARTHQUAKE FOR SSE

UPDATED FSAR

FIGURE 3.7-3



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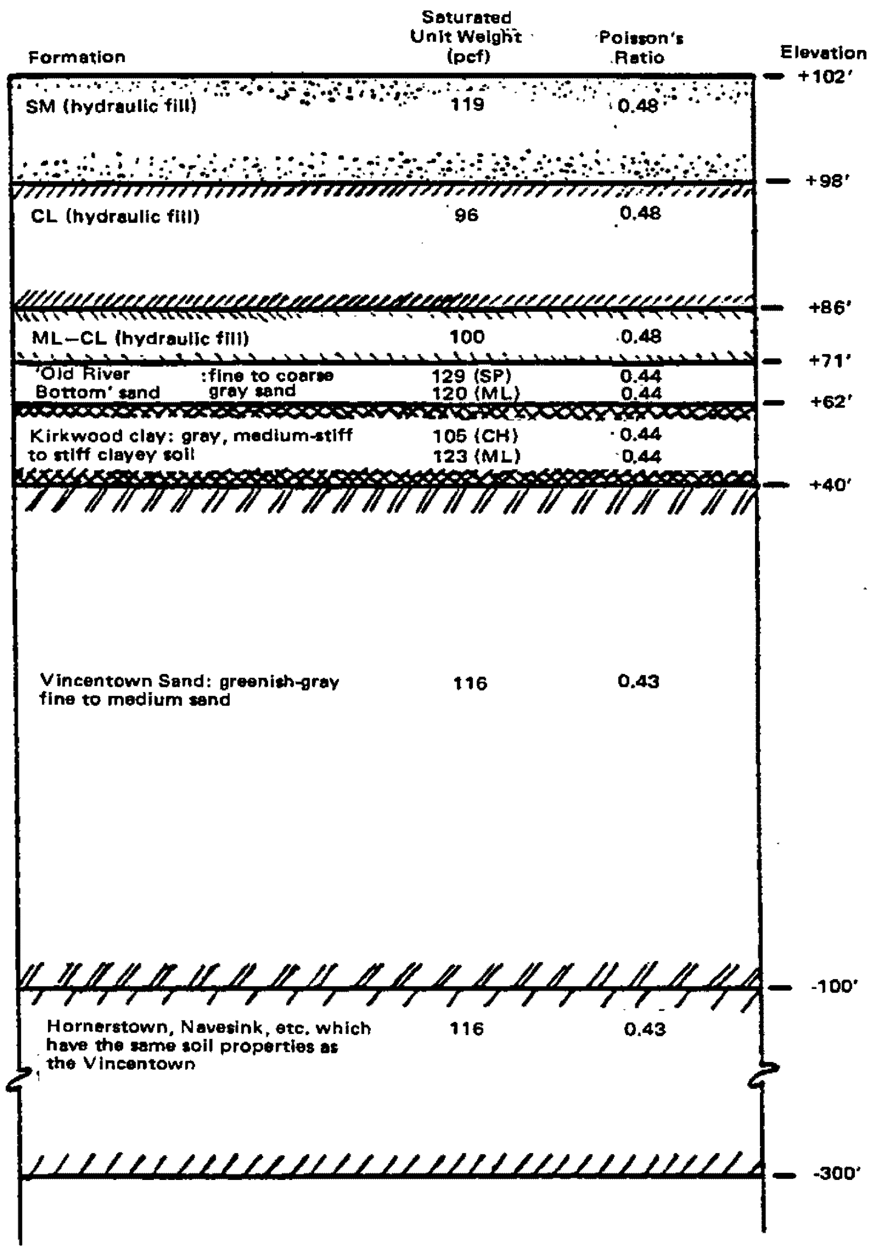
HOPE CREEK NUCLEAR GENERATING STATION

SYNTHETIC TIME-HISTORY OF THE VERTICAL COMPONENT OF THE DESIGN EARTHQUAKE FOR SSE

UPDATED FSAR

FIGURE 3.7-4



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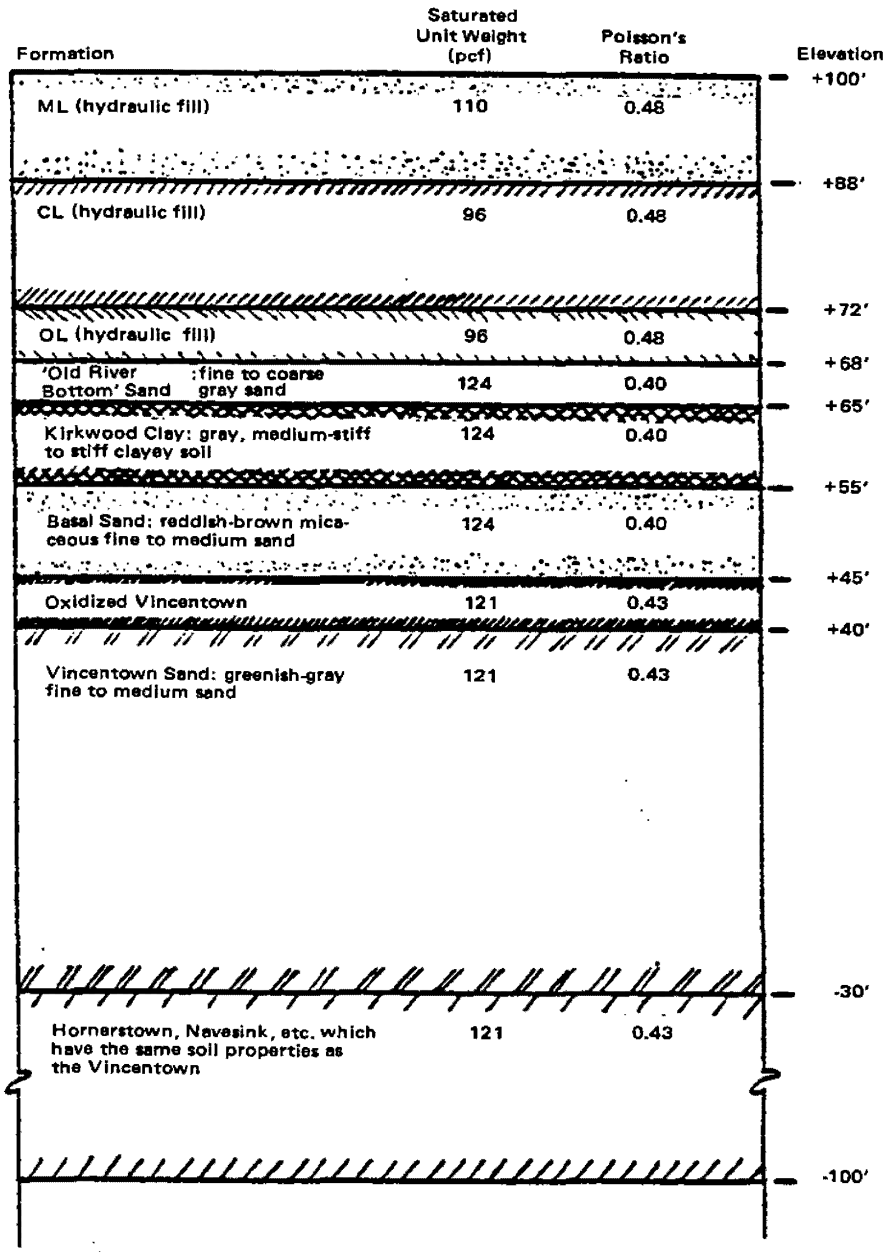
HOPE CREEK NUCLEAR GENERATING STATION

IDEALIZED SOIL PROFILE
FOR THE POWER BLOCK AREA

UPDATED FSAR

FIGURE 3.7-5



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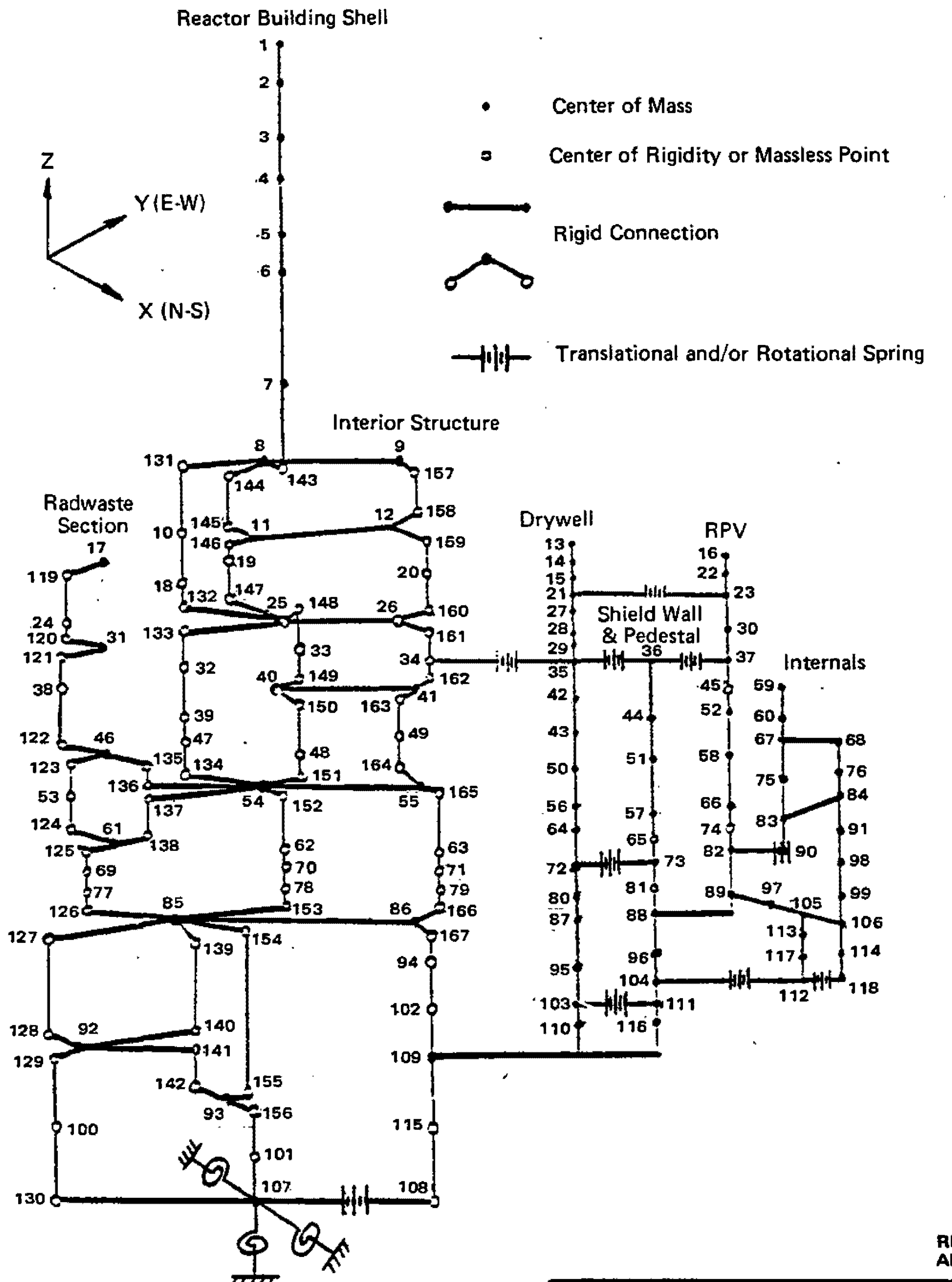
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

IDEALIZED SOIL PROFILE
FOR THE INTAKE STRUCTURE AREA

UPDATED FSAR

FIGURE 3.7-6



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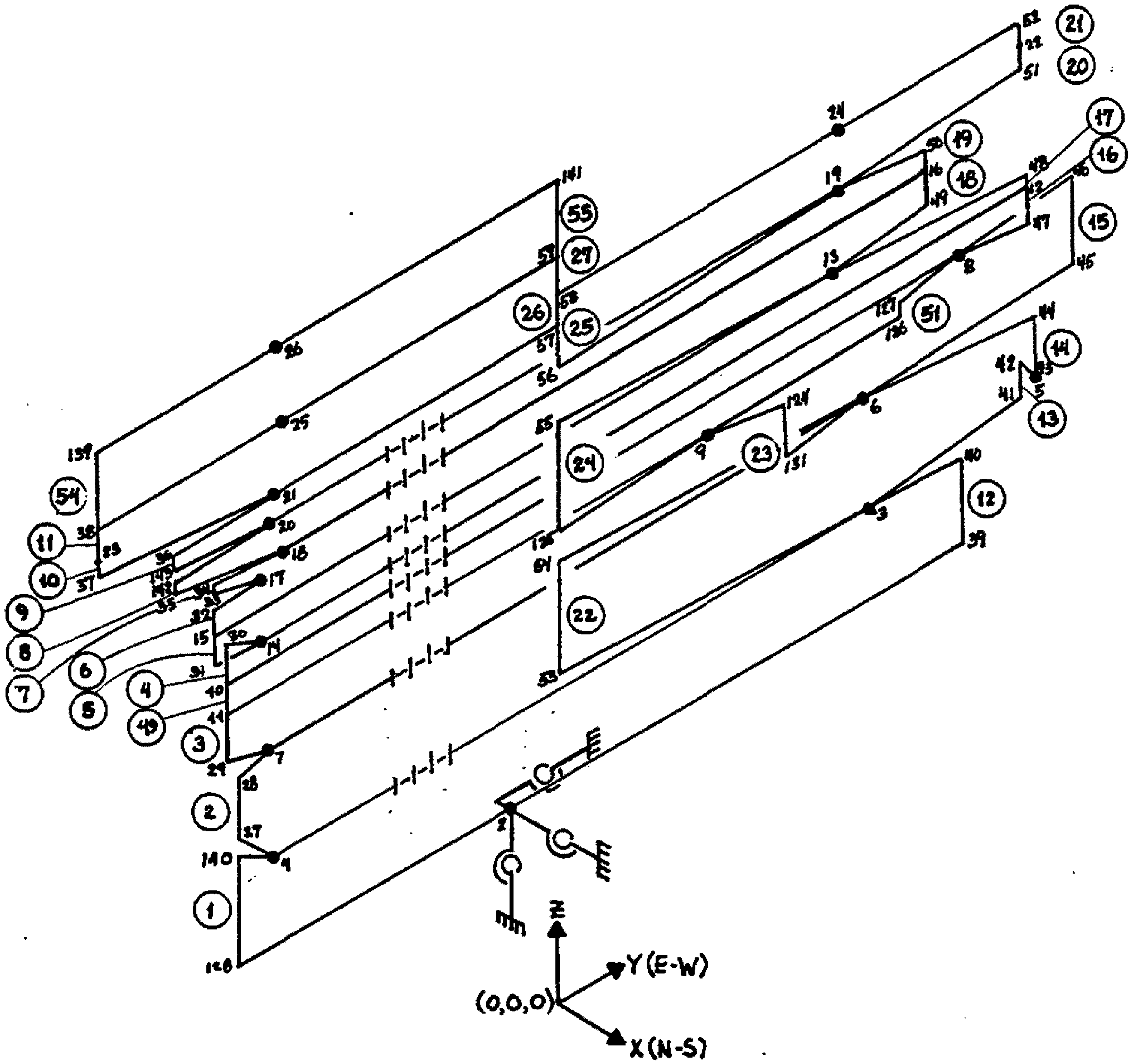
HOPE CREEK NUCLEAR GENERATING STATION

REACTOR BUILDING
MATHEMATICAL MODEL

UPDATED FSAR

FIGURE 3.7-7



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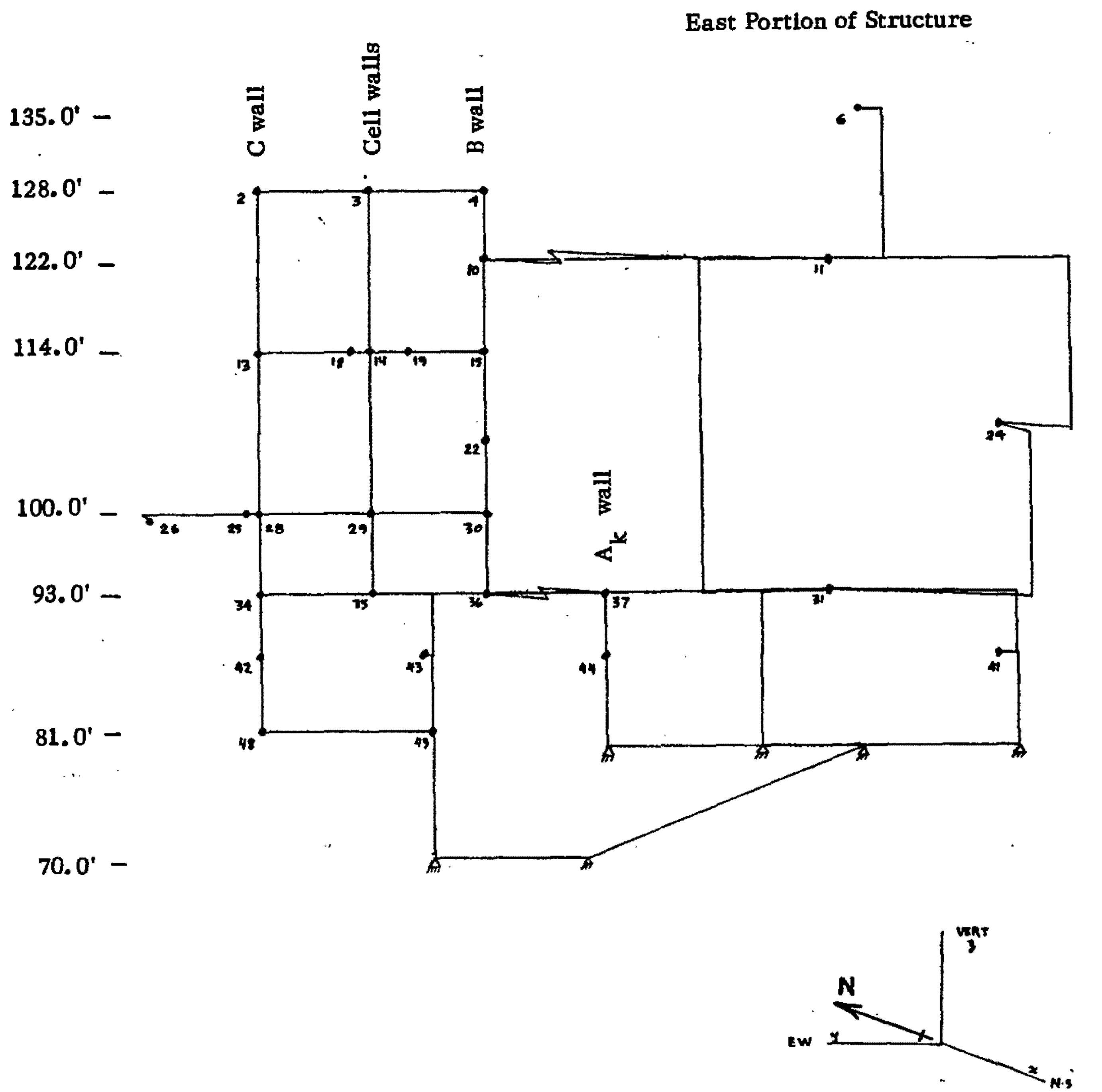
HOPE CREEK NUCLEAR GENERATING STATION

AUXILIARY BUILDING
MATHEMATICAL MODEL

UPDATED FSAR

FIGURE 3.7-8



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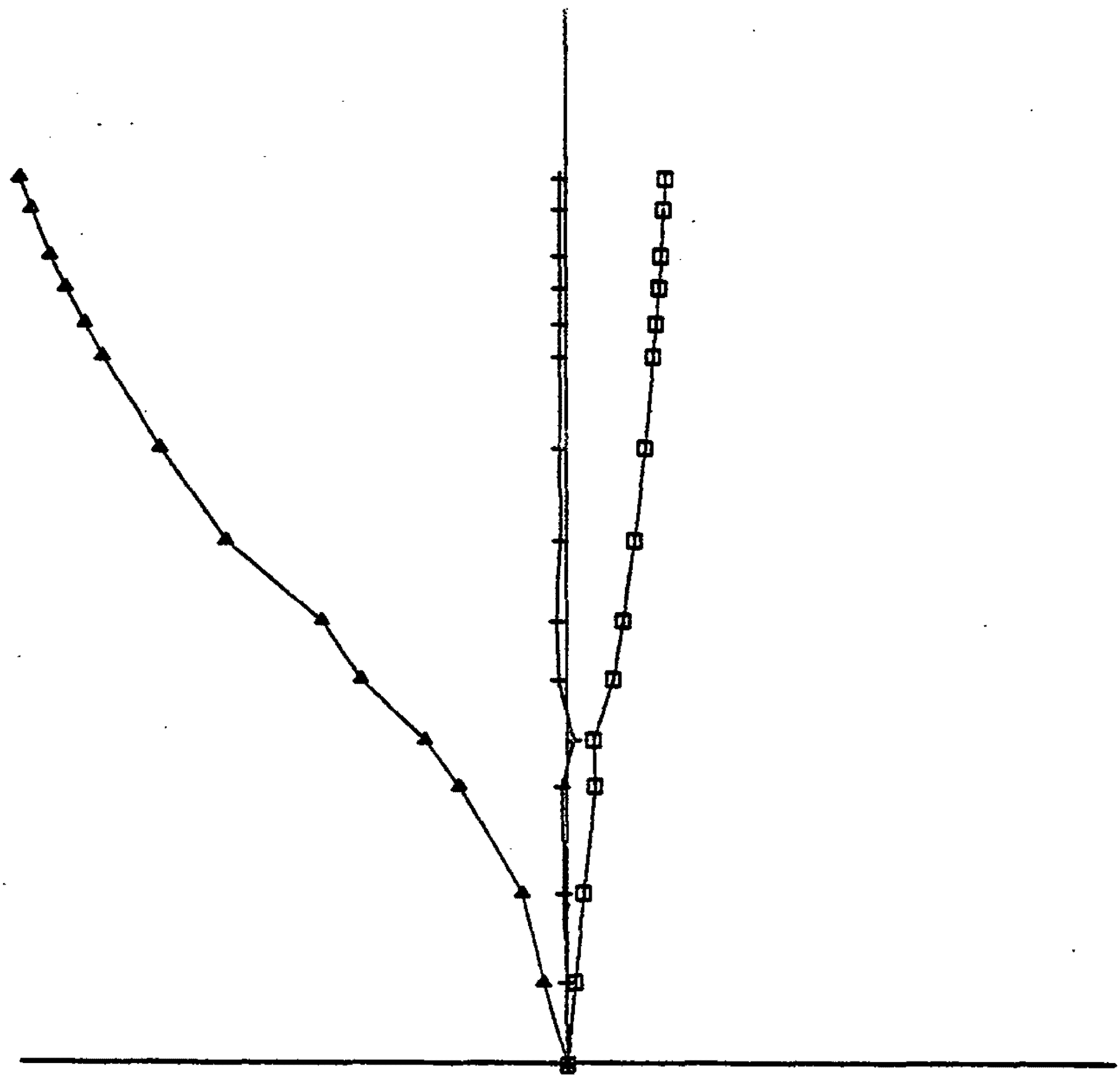
HOPE CREEK NUCLEAR GENERATING STATION

INTAKE STRUCTURE
MATHEMATICAL MODEL

UPDATED FSAR

FIGURE 3.7-9



Mode Number 4
Frequency (CPS) = 4.09

Direction N-S = Triangle
Direction E-W = Square
Vertical = Cross

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APRIL 11, 1988

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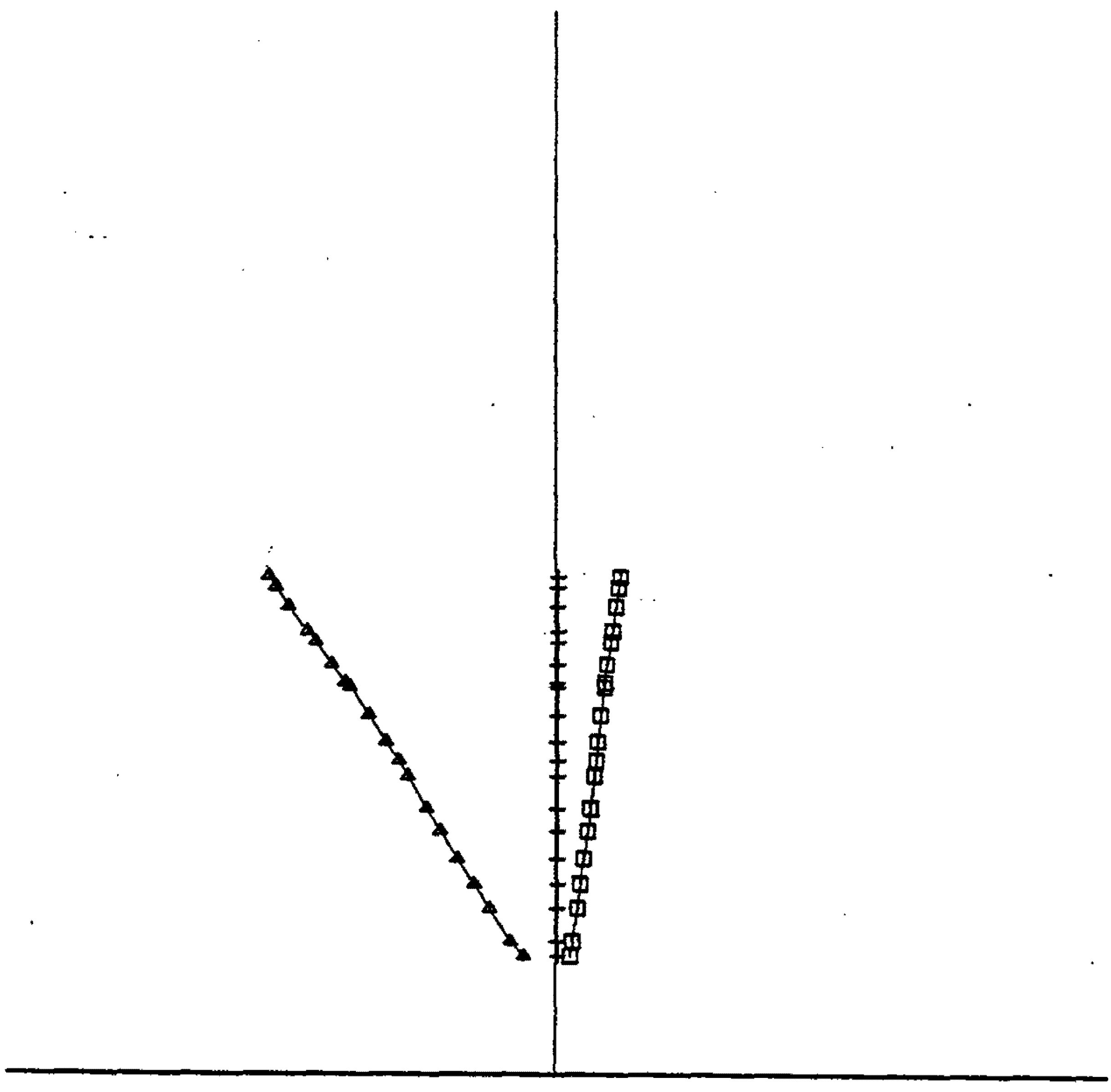
HOPE CREEK NUCLEAR GENERATING STATION

REACTOR BUILDING SHELL – MODE SHAPES, MODE 4

UPDATED FSAR

FIGURE 3.7-10



Mode Number 4
Frequency (CPS) = 4.09

Direction N-S = Triangle
Direction E-W = Square
Vertical = Cross

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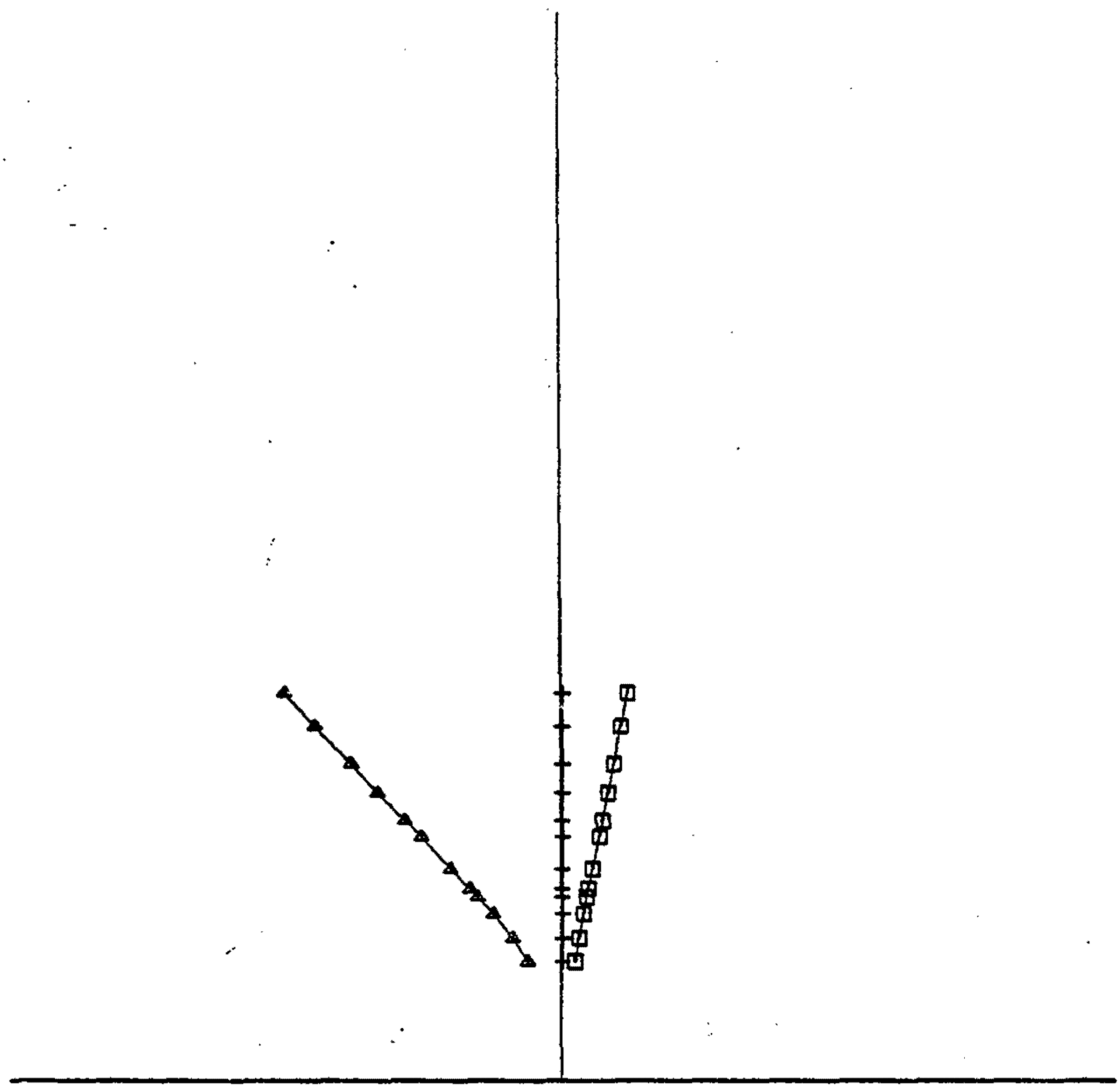
HOPE CREEK NUCLEAR GENERATING STATION

REACTOR BUILDING DRYWELL — MODE SHAPES, MODE 4

UPDATED FSAR

FIGURE 3.7-11



Mode Number 4
Frequency (CPS) = 4.09

Direction N-S = Triangle
Direction E-W = Square
Vertical = Cross

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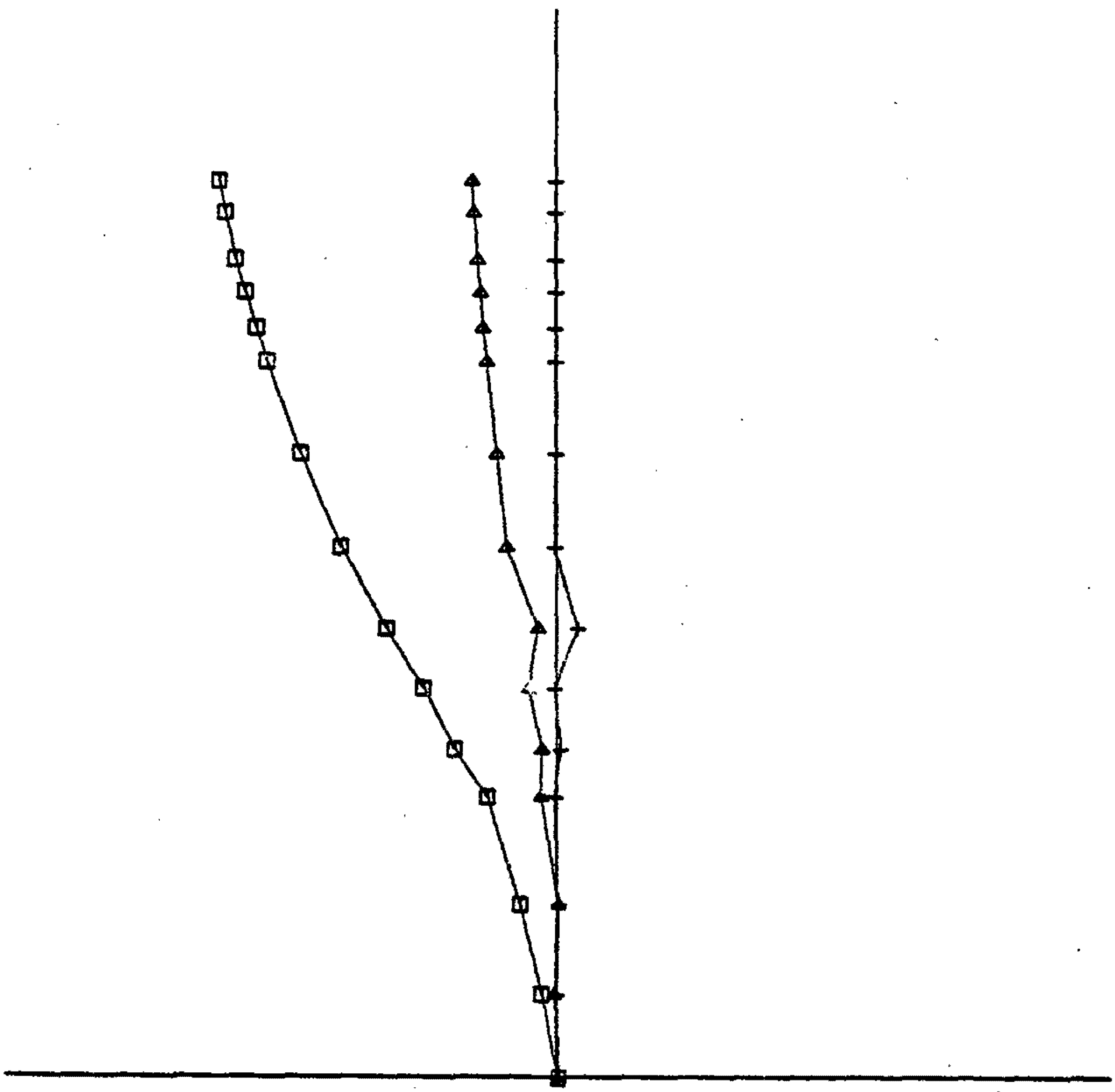
HOPE CREEK NUCLEAR GENERATING STATION

REACTOR BUILDING BIOLOGICAL SHIELD WALL AND RPV PEDESTAL – MODE SHAPES, MODE 4

UPDATED FSAR

FIGURE 3.7-12



Mode Number 5
Frequency (CPS) = 4.25

Direction N-S = Triangle
Direction E-W = Square
Vertical = Cross

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APRIL 11, 1988

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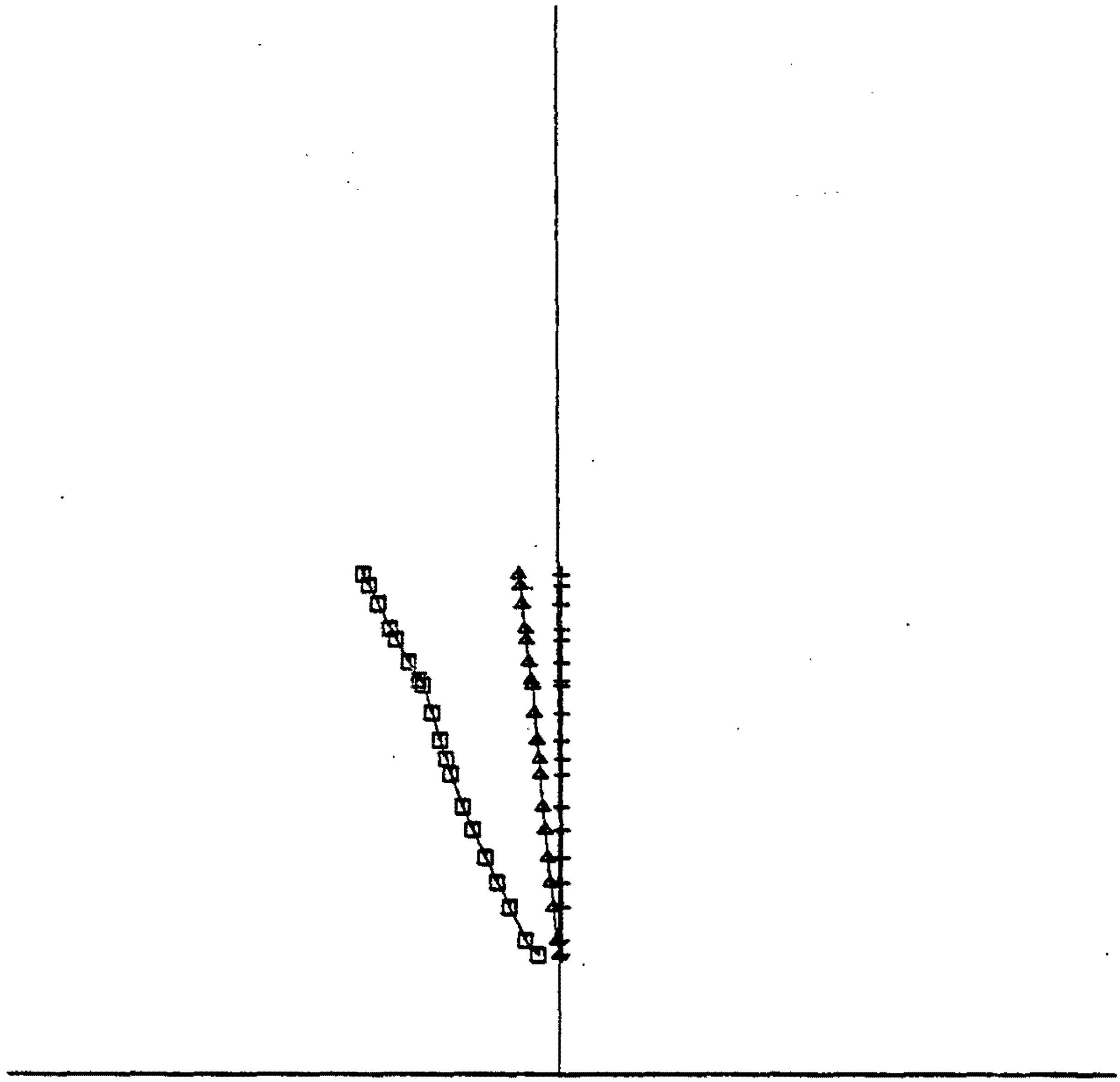
HOPE CREEK NUCLEAR GENERATING STATION

REACTOR BUILDING SHELL – MODE SHAPES, MODE 5

UPDATED FSAR

FIGURE 3.7-13



Mode Number 5
Frequency (CPS) = 4.25

Direction N-S = Triangle
Direction E-W = Square
Vertical = Cross

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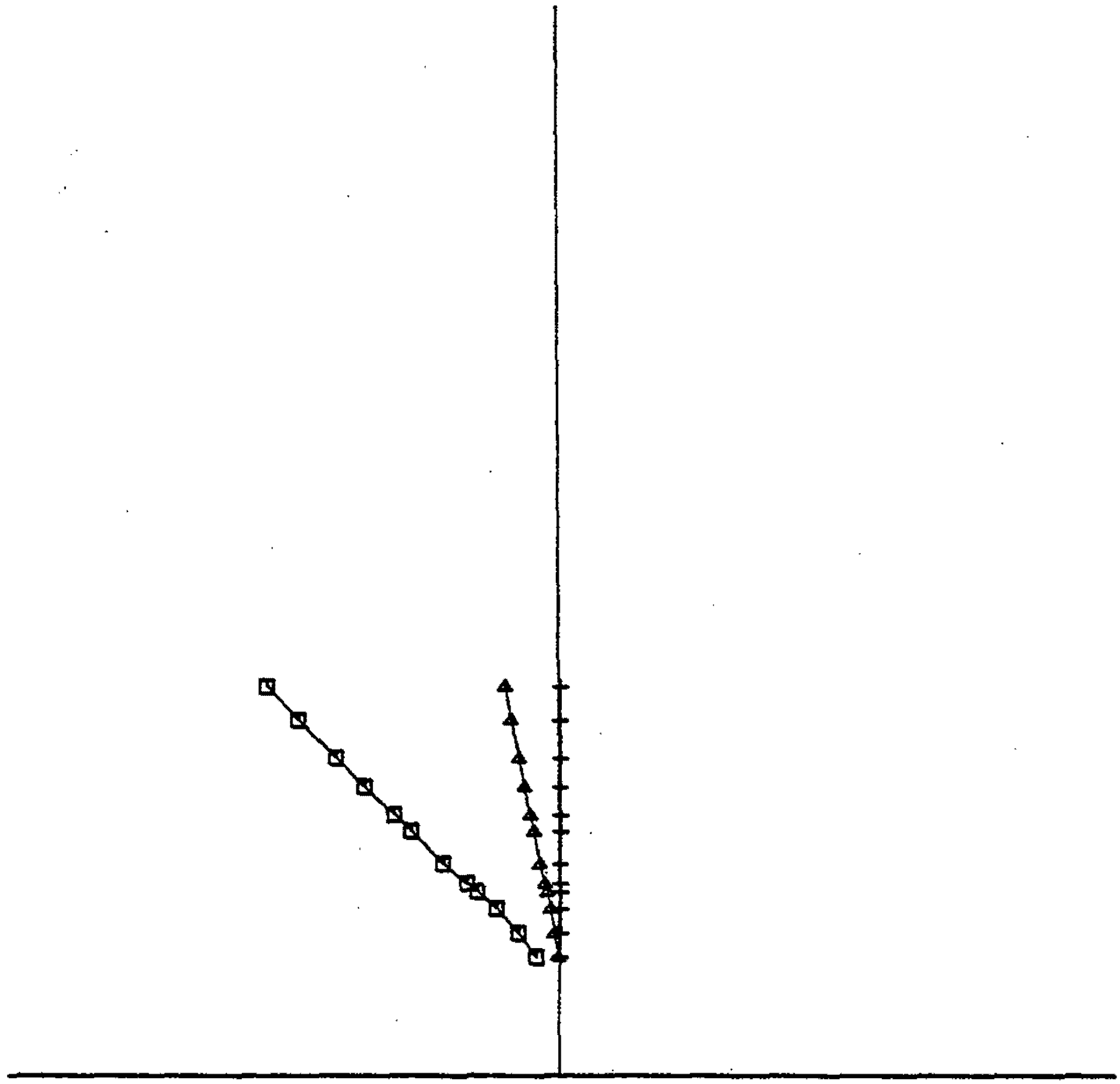
HOPE CREEK NUCLEAR GENERATING STATION

REACTOR BUILDING DRYWELL –
MODE SHAPES, MODE 5

UPDATED FSAR

FIGURE 3.7-14



Mode Number 5
Frequency (CPS) = 4.25

Direction N-S = Triangle
Direction E-W = Square
Vertical = Cross

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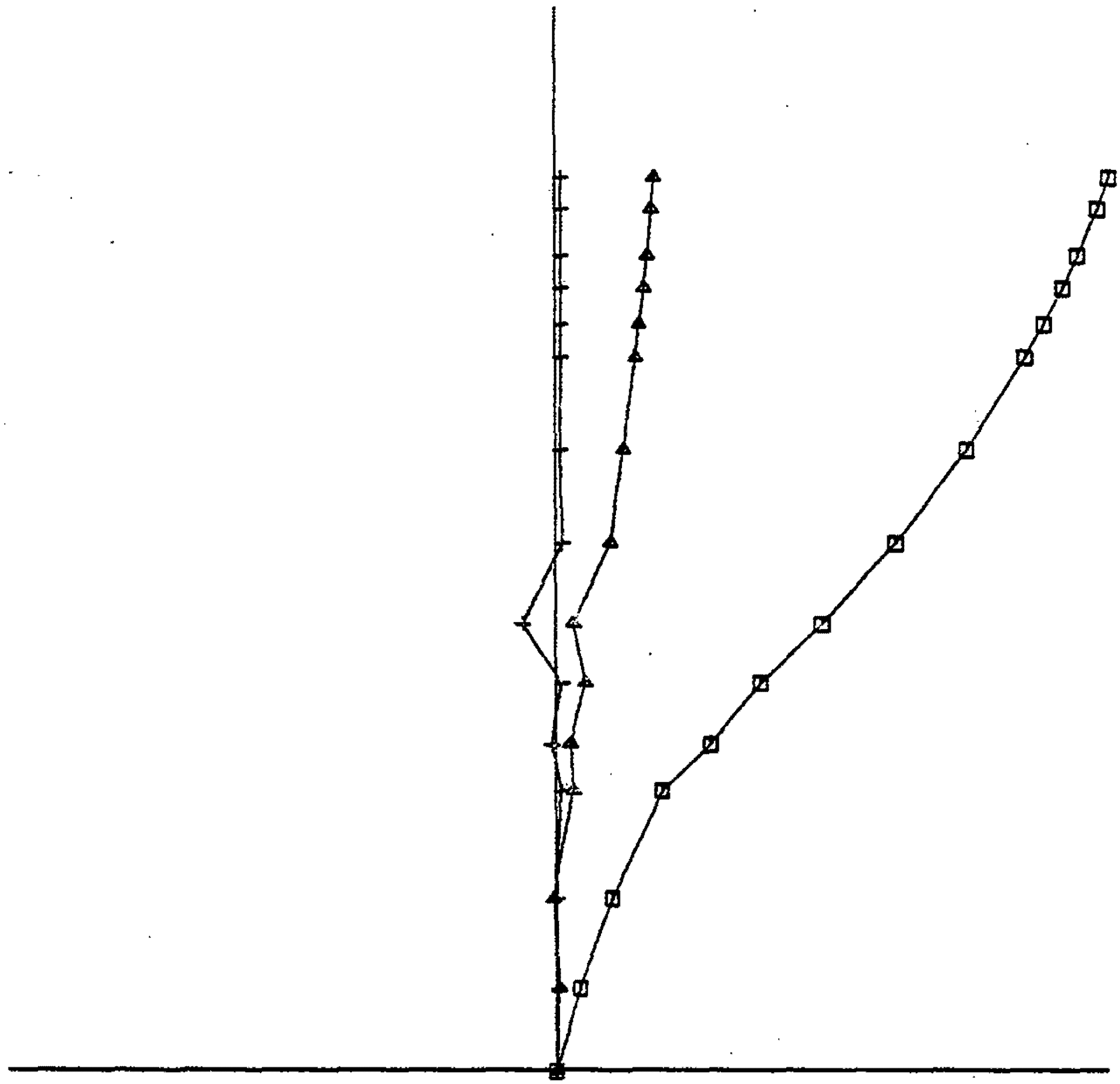
HOPE CREEK NUCLEAR GENERATING STATION

REACTOR BUILDING BIOLOGICAL SHIELD WALL AND RPV PEDESTAL – MODE SHAPES, MODE 5

UPDATED FSAR

FIGURE 3.7-15



Mode Number 7
Frequency (CPS) = 4.45

Direction N-S = Triangle
Direction E-W = Square
Vertical = Cross

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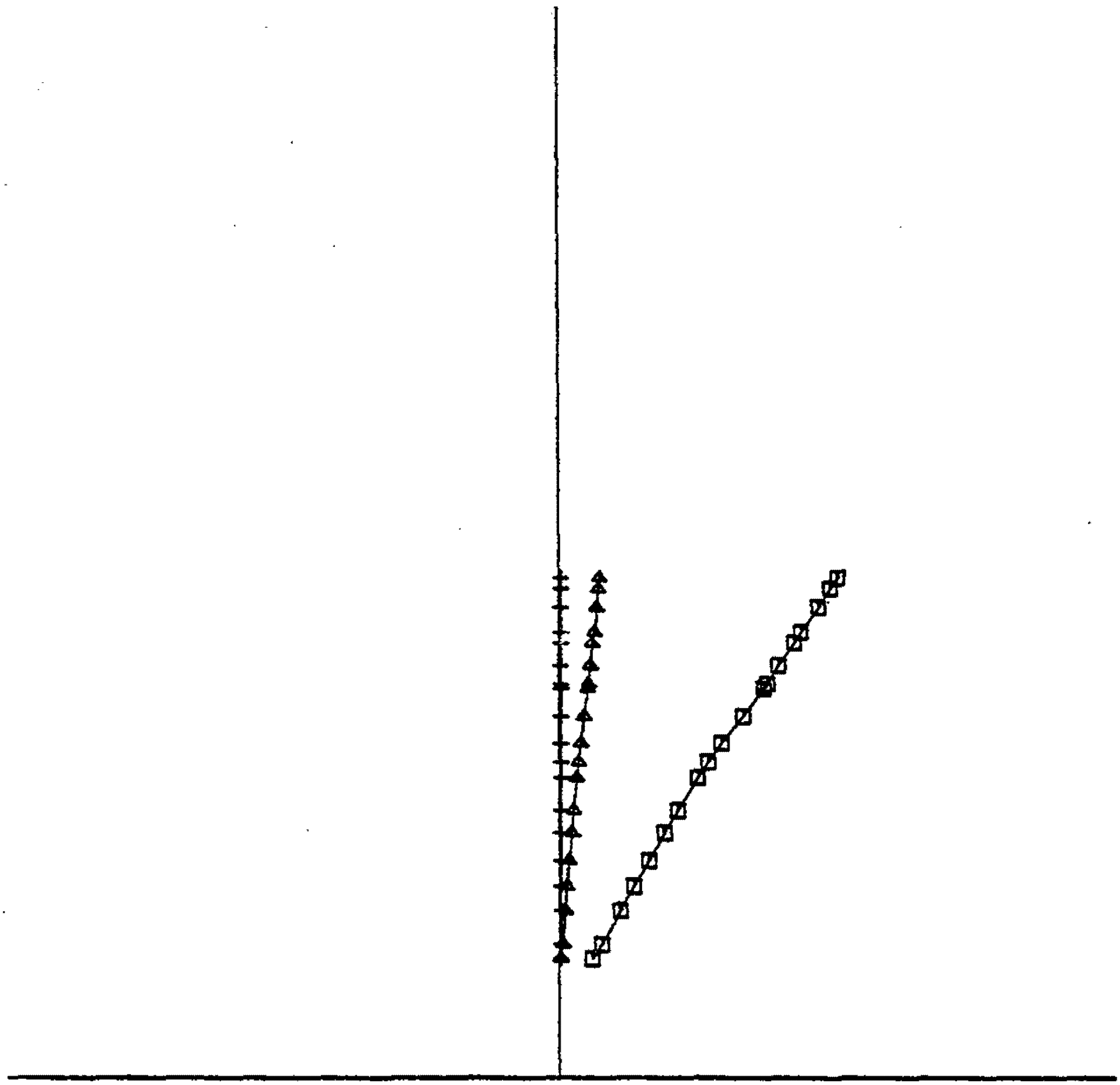
HOPE CREEK NUCLEAR GENERATING STATION

REACTOR BUILDING SHELL –
MODE SHAPES, MODE 7

UPDATED FSAR

FIGURE 3.7-16



Mode Number 7
Frequency (CPS) = 4.45

Direction N-S = Triangle
Direction E-W = Square
Vertical = Cross

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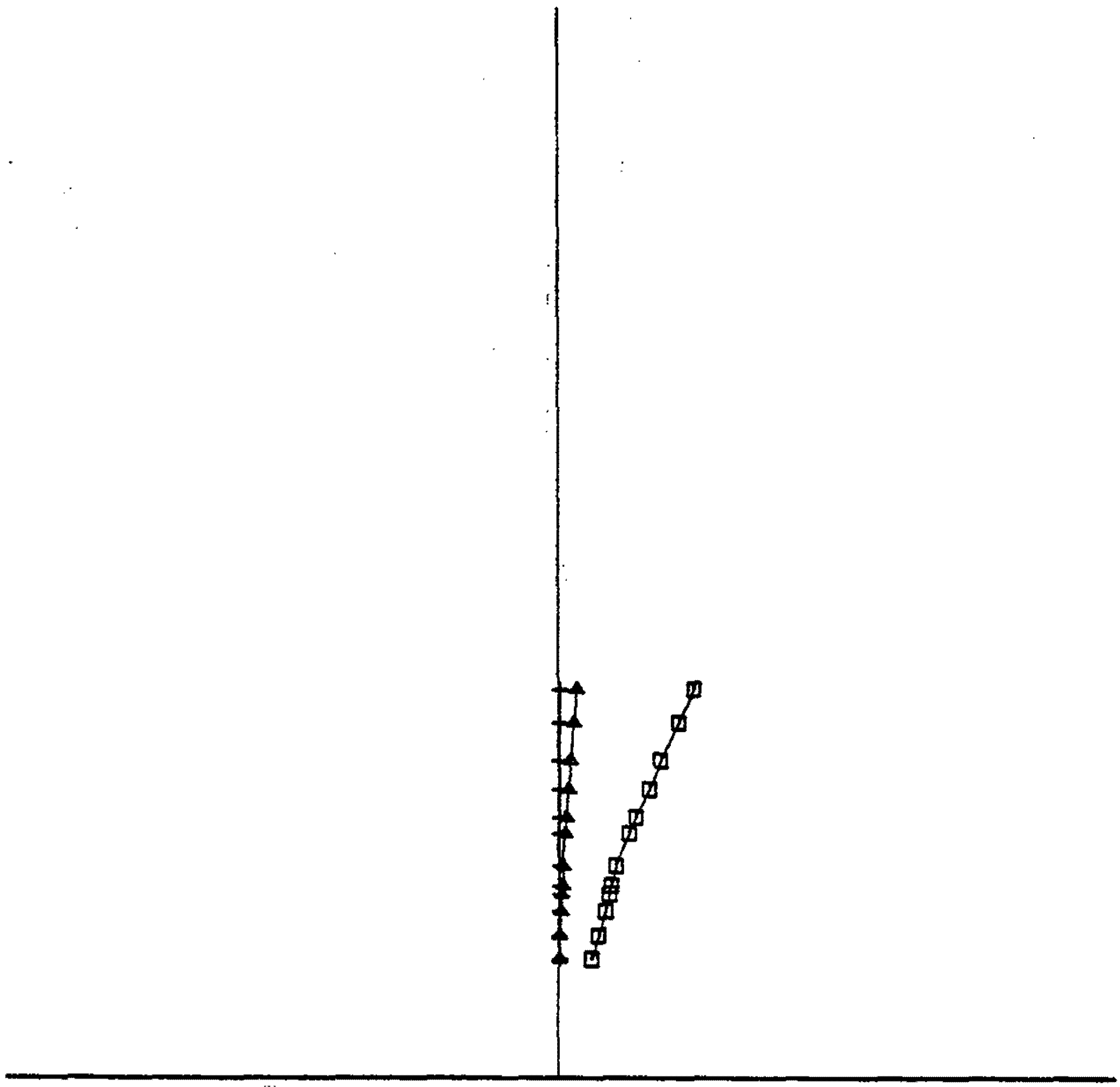
HOPE CREEK NUCLEAR GENERATING STATION

REACTOR BUILDING DRYWELL – MODE SHAPES, MODE 7

UPDATED FSAR

FIGURE 3.7-17



Mode Number 7
Frequency (CPS) = 4.45

Direction N-S = Triangle
Direction E-W = Square
Vertical = Cross

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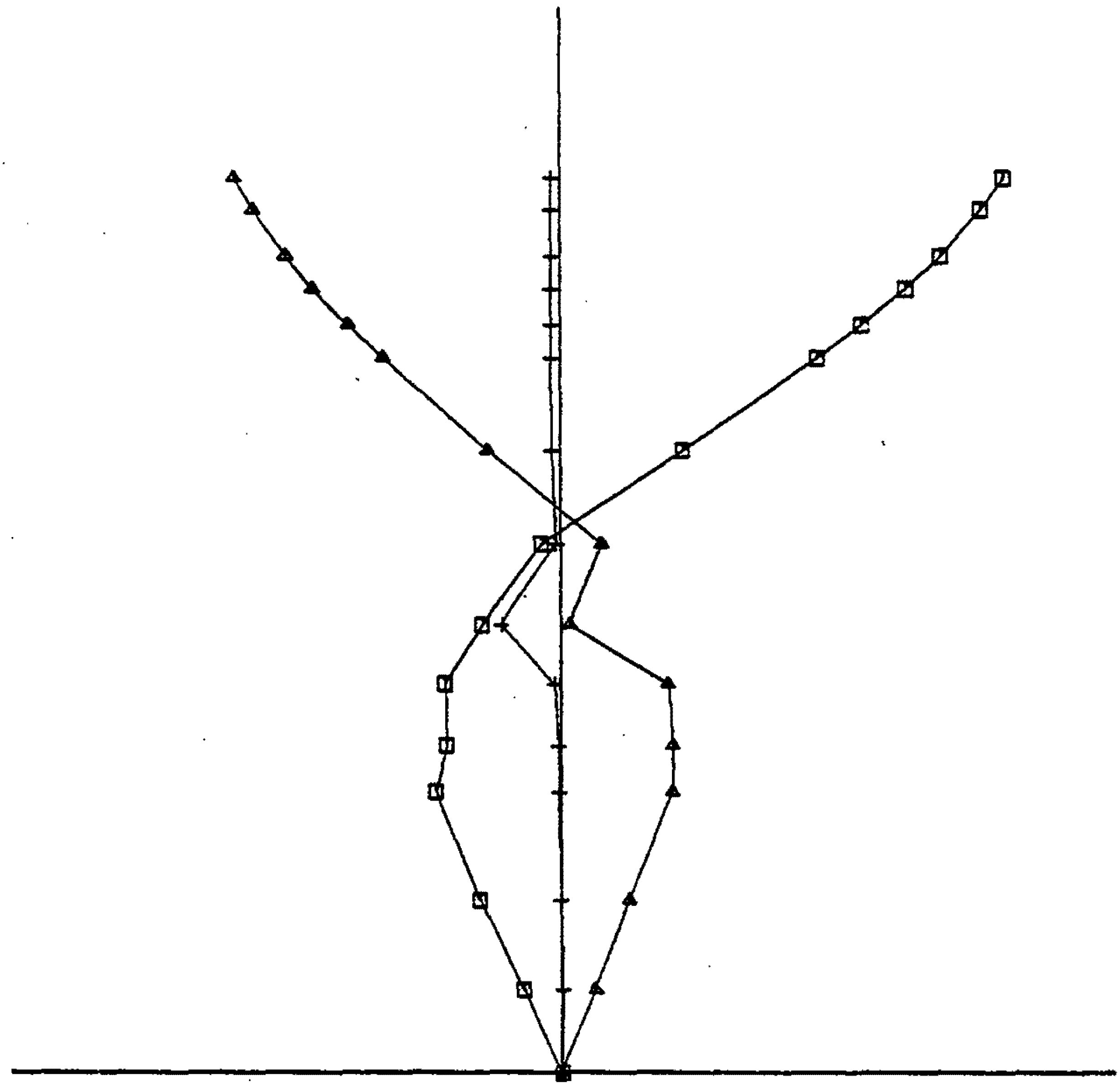
HOPE CREEK NUCLEAR GENERATING STATION

REACTOR BUILDING BIOLOGICAL SHIELD WALL AND RPV PEDESTAL – MODE SHAPES, MODE 7

UPDATED FSAR

FIGURE 3.7-18



Mode Number 10
Frequency (CPS) = 8.87

Direction N-S = Triangle
Direction E-W = Square
Vertical = Cross

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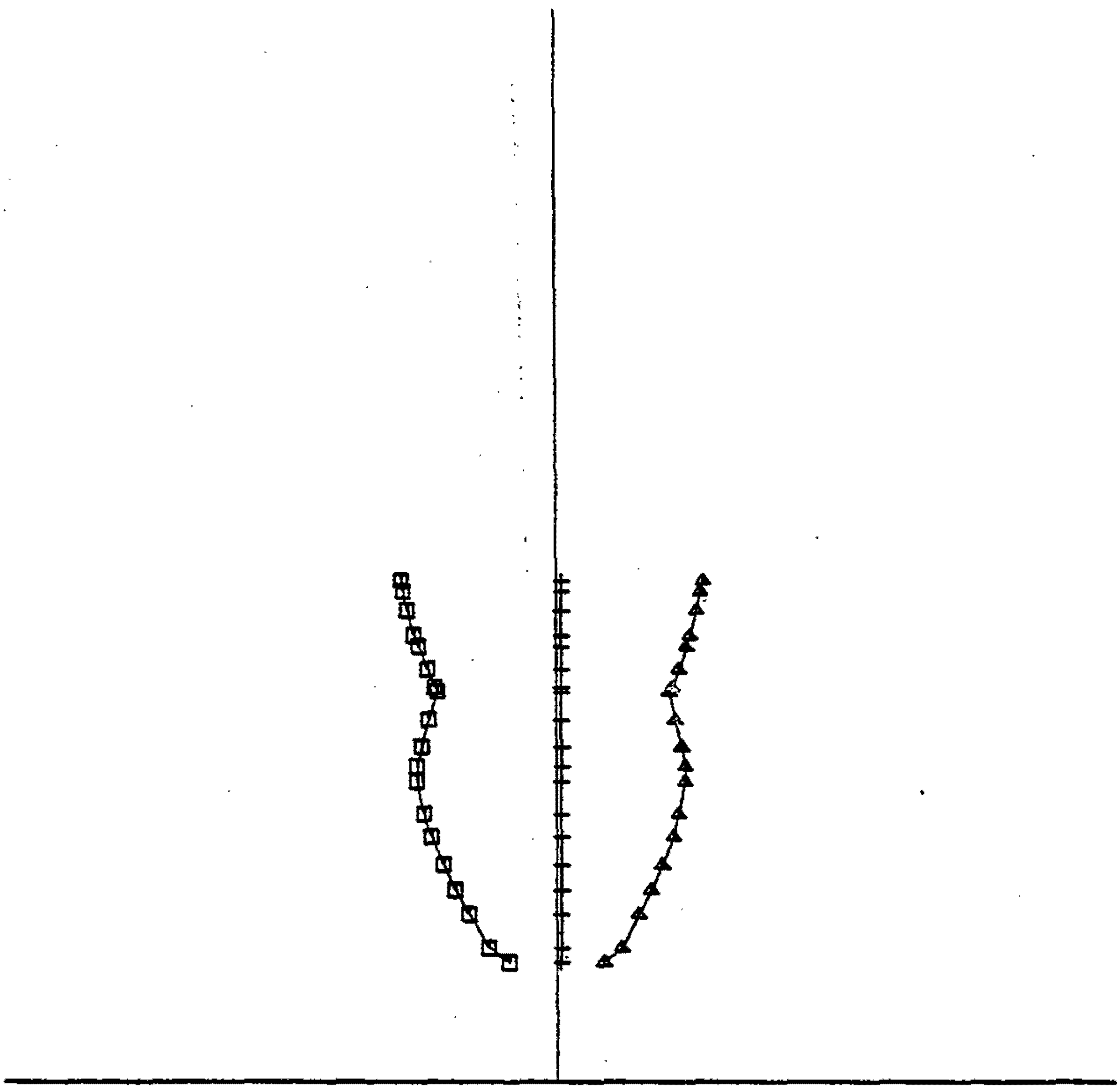
HOPE CREEK NUCLEAR GENERATING STATION

REACTOR BUILDING SHELL – MODE SHAPES, MODE 10

UPDATED FSAR

FIGURE 3.7-19



Mode Number 10
Frequency (CPS) = 8.87

Direction N-S = Triangle
Direction E-W = Square
Vertical = Cross

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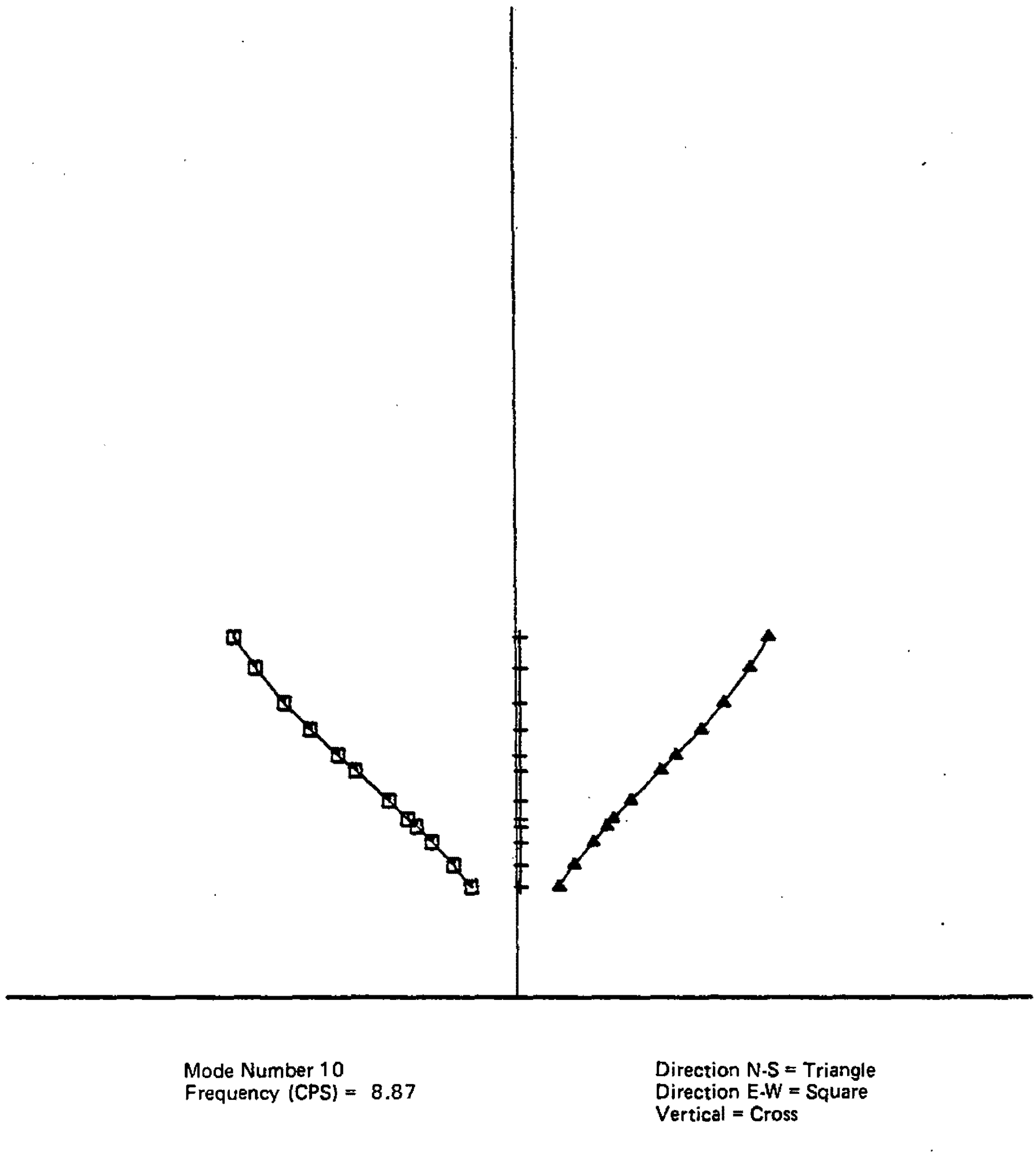
HOPE CREEK NUCLEAR GENERATING STATION

REACTOR BUILDING DRYWELL — MODE SHAPES, MODE 10

UPDATED FSAR

FIGURE 3.7-20



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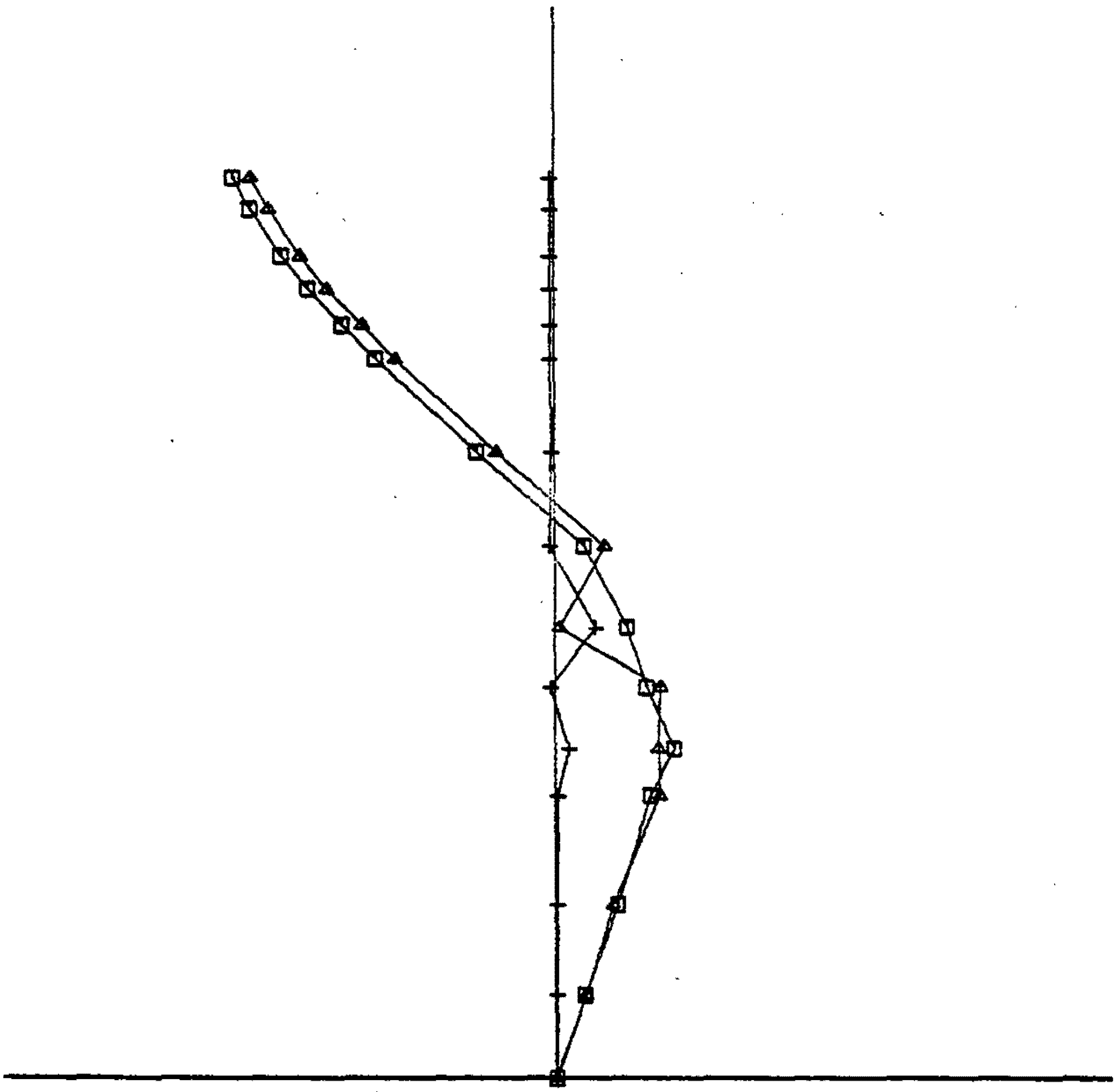
HOPE CREEK NUCLEAR GENERATING STATION

REACTOR BUILDING BIOLOGICAL SHIELD WALL AND RPV PEDESTAL – MODE SHAPES, MODE 10

UPDATED FSAR

FIGURE 3.7-21



Mode Number 11
Frequency (CPS) = 8.97

Direction N-S = Triangle
Direction E-W = Square
Vertical = Cross

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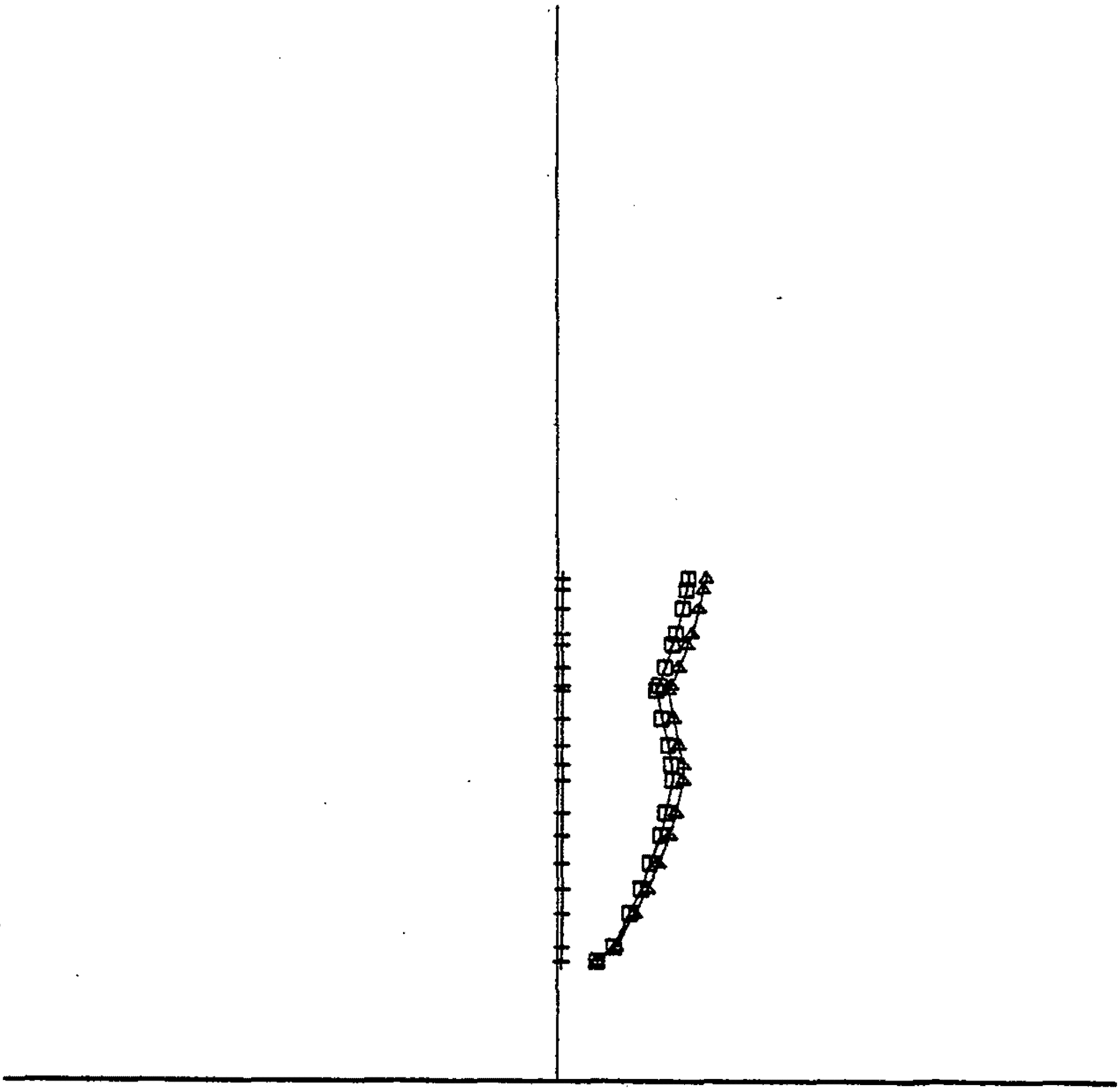
APRIL 11, 1988

PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

REACTOR BUILDING SHELL –
MODE SHAPES, MODE 11

UPDATED FSAR

FIGURE 3.7-22



Mode Number 11
Frequency (CPS) = 8.97

Direction N-S = Triangle
Direction E-W = Square
Vertical = Cross

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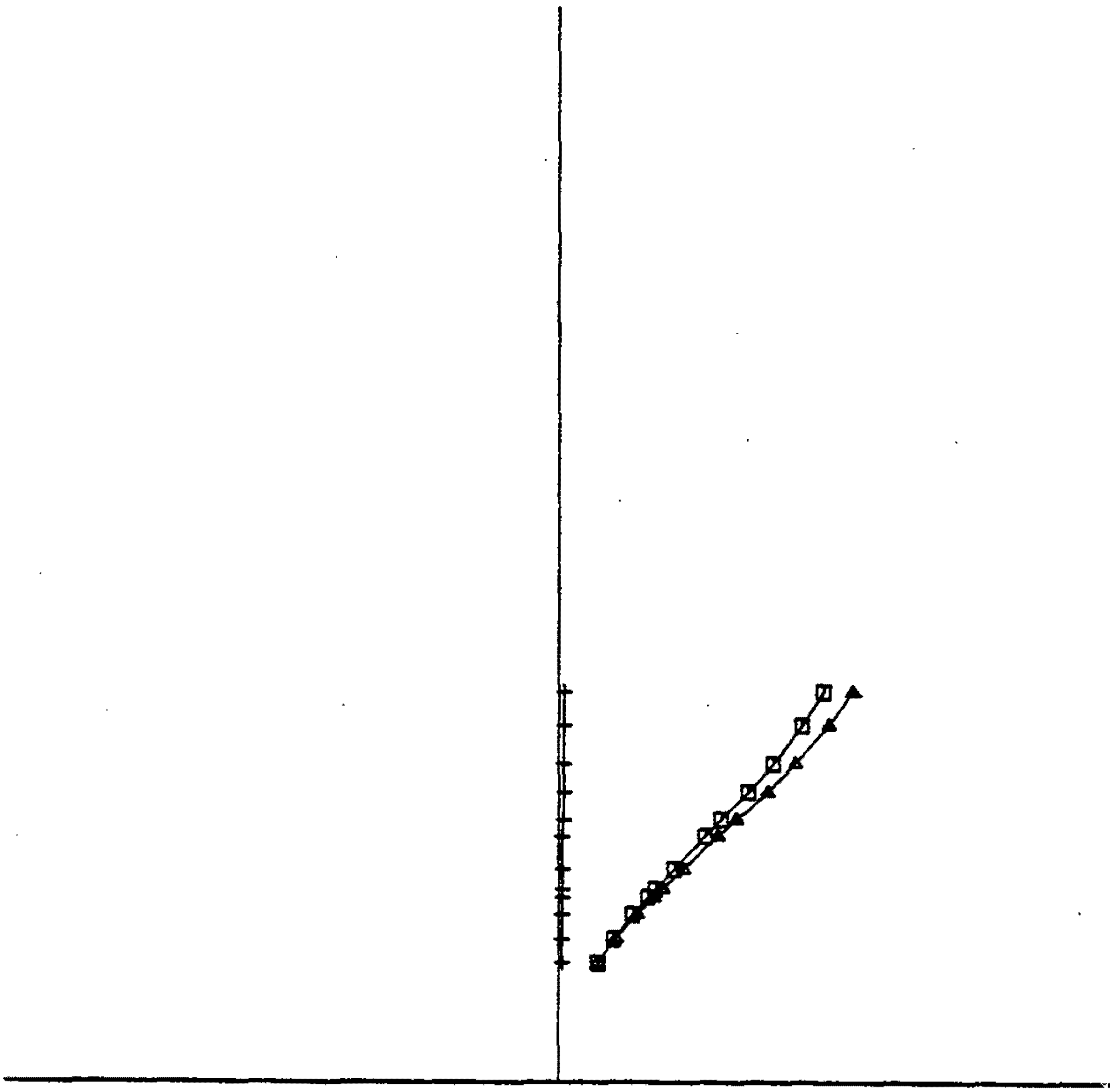
APRIL 11, 1988

PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

REACTOR BUILDING DRYWELL – MODE SHAPES, MODE 11

UPDATED FSAR

FIGURE 3.7-23



Mode Number 11
Frequency (CPS) = 8.97

Direction N-S = Triangle
Direction E-W = Square
Vertical = Cross

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APRIL 11, 1988

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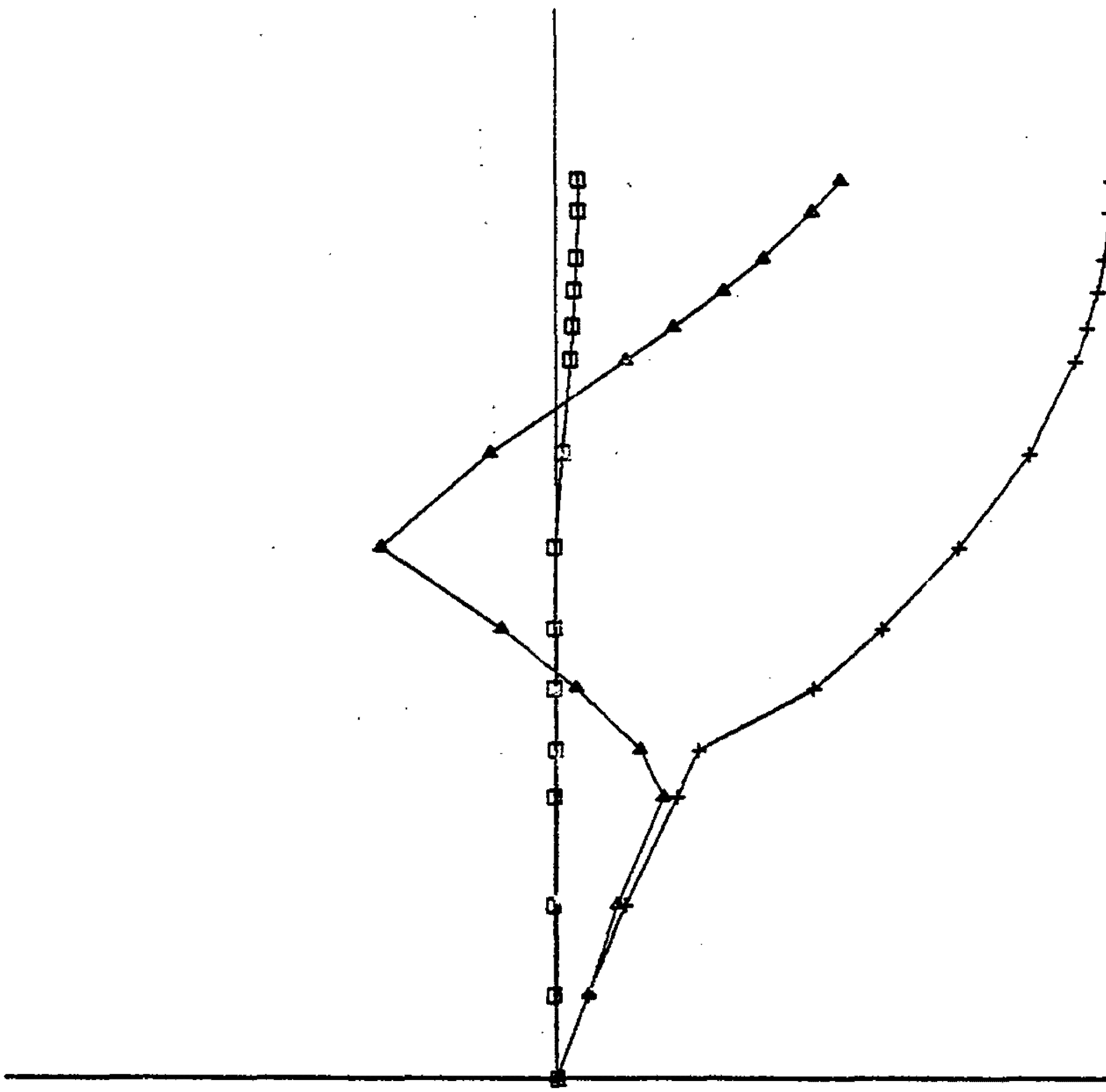
HOPE CREEK NUCLEAR GENERATING STATION

REACTOR BUILDING BIOLOGICAL SHIELD WALL AND RPV PEDESTAL – MODE SHAPES, MODE 11

UPDATED FSAR

FIGURE 3.7-24



Mode Number 15
Frequency (CPS) = 11.42

Direction N-S = Triangle
Direction E-W = Square
Vertical = Cross

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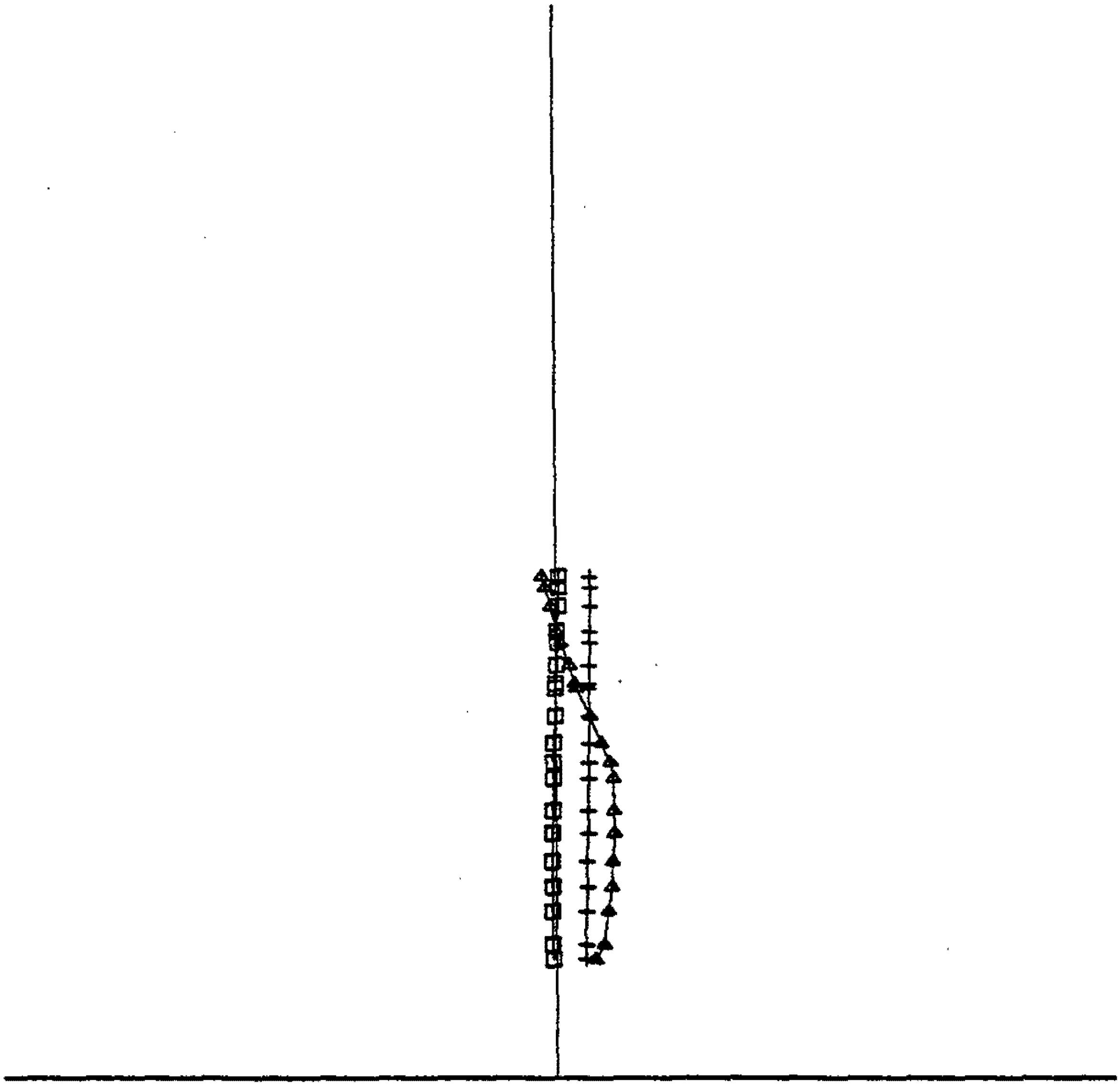
HOPE CREEK NUCLEAR GENERATING STATION

REACTOR BUILDING SHELL – MODE SHAPES, MODE 15

UPDATED FSAR

FIGURE 3.7-25



Mode Number 15
Frequency (CPS) = 11.42

Direction N-S = Triangle
Direction E-W = Square
Vertical = Cross

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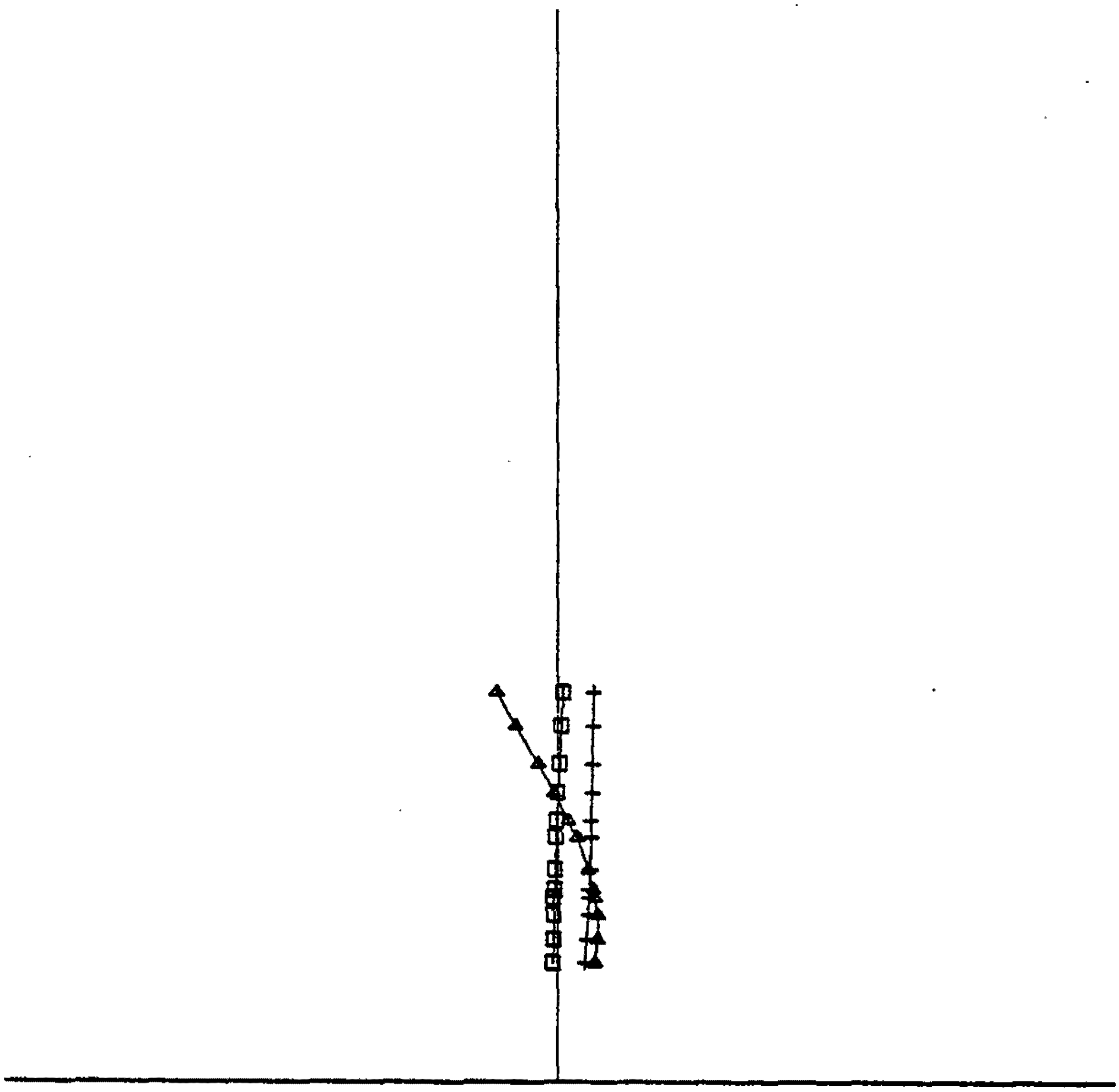
HOPE CREEK NUCLEAR GENERATING STATION

REACTOR BUILDING DRYWELL – MODE SHAPES, MODE 15

UPDATED FSAR

FIGURE 3.7-26



Mode Number 15
Frequency (CPS) = 11.42

Direction N-S = Triangle
Direction E-W = Square
Vertical = Cross

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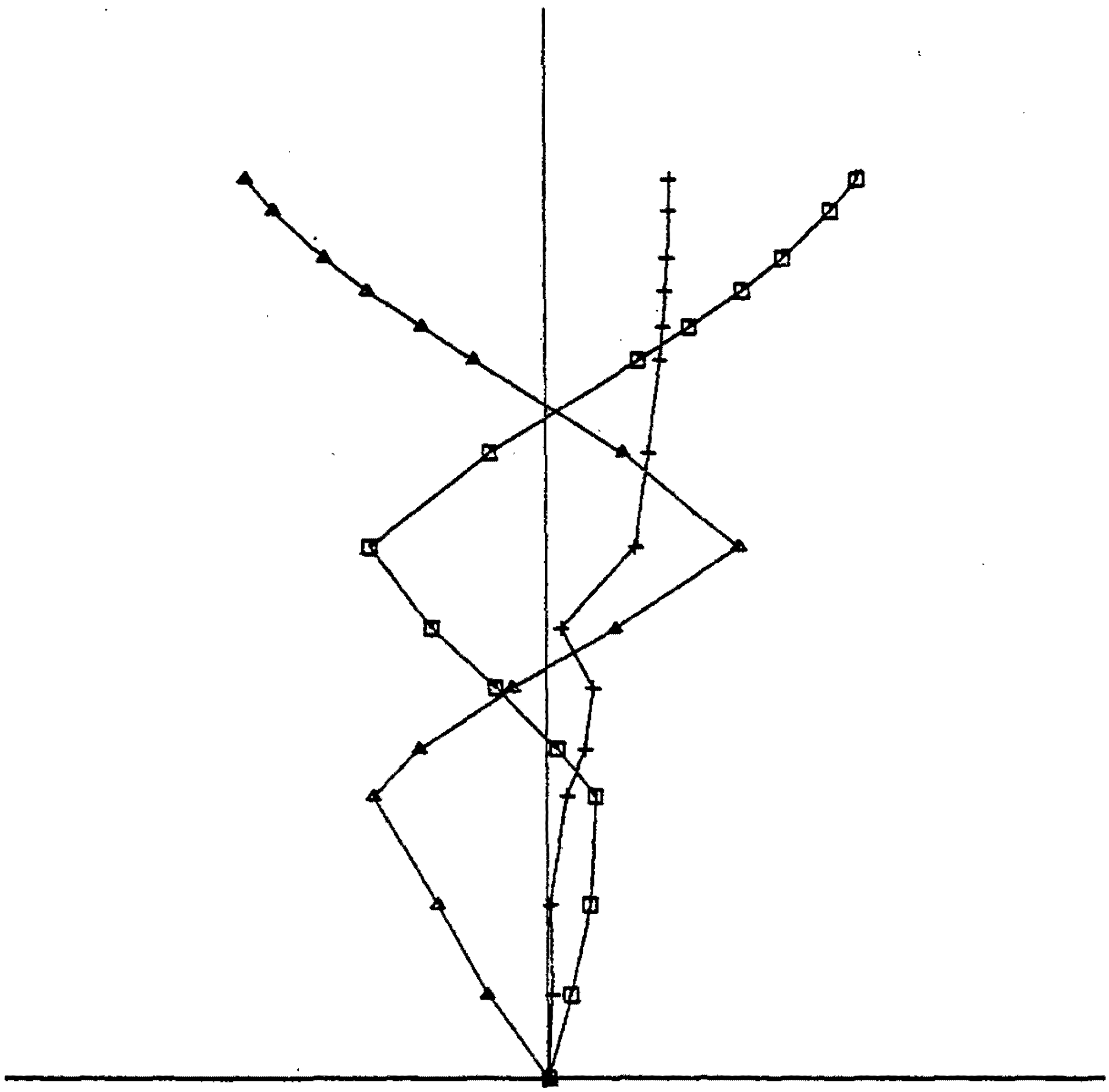
HOPE CREEK NUCLEAR GENERATING STATION

REACTOR BUILDING BIOLOGICAL SHIELD WALL AND RPV PEDESTAL – MODE SHAPES, MODE 15

UPDATED FSAR

FIGURE 3.7-27



Mode Number 17
Frequency (CPS) = 12.34

Direction N-S = Triangle
Direction E-W = Square
Vertical = Cross

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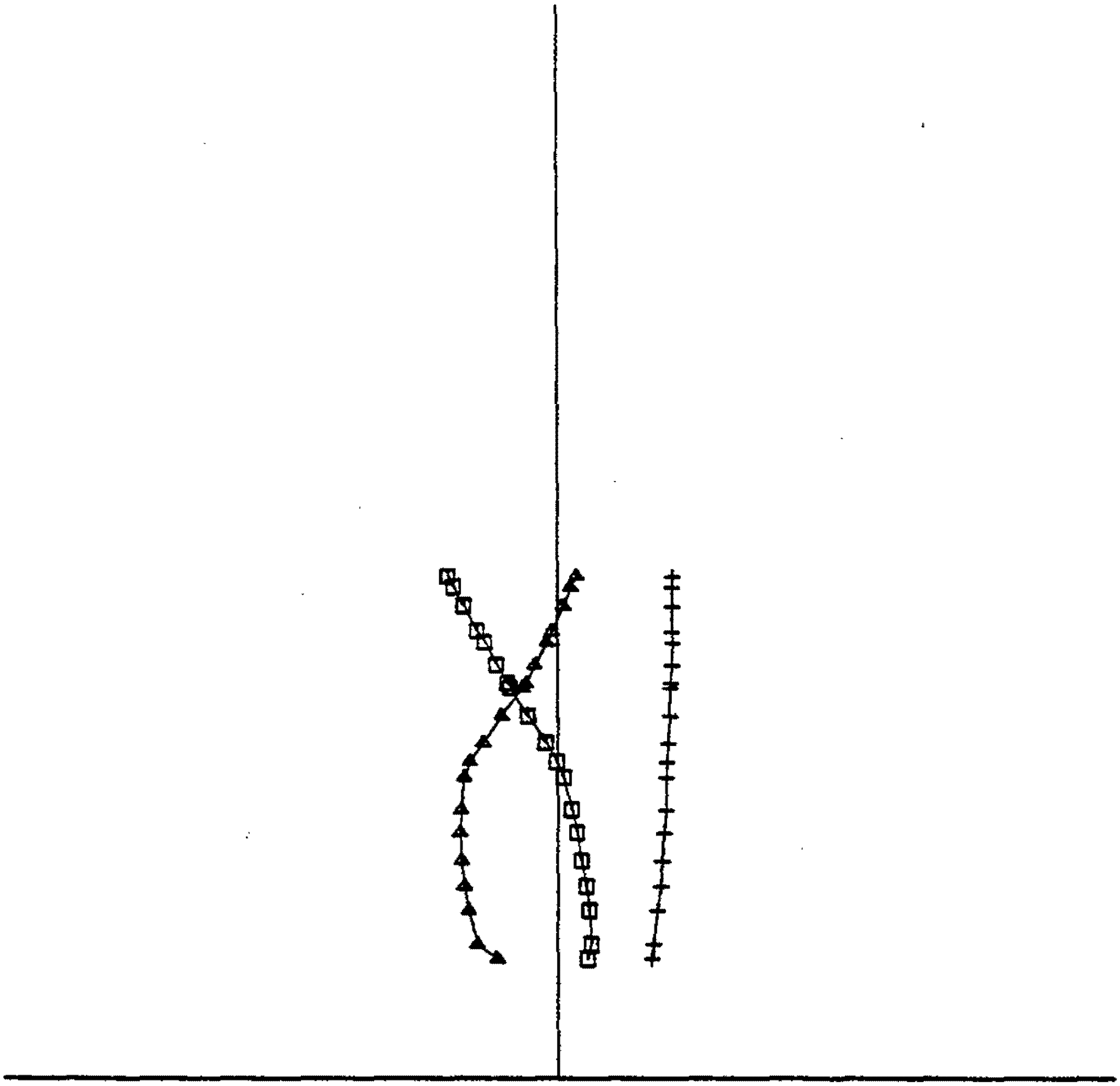
APRIL 11, 1988

PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

REACTOR BUILDING SHELL – MODE SHAPES, MODE 17

UPDATED FSAR

FIGURE 3.7-28



Mode Number 17
Frequency (CPS) = 12.34

Direction N-S = Triangle
Direction E-W = Square
Vertical = Cross

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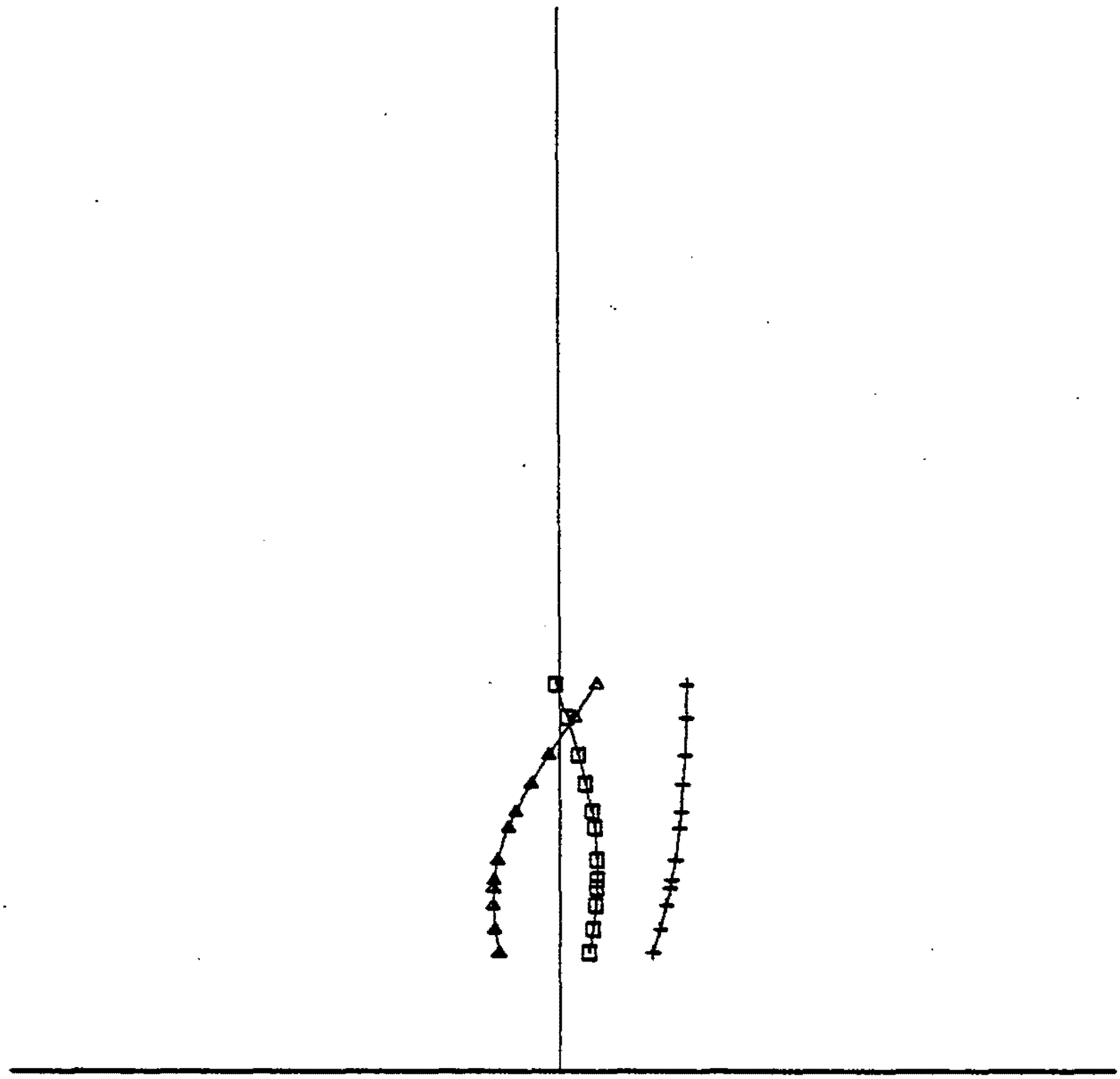
APRIL 11, 1988

PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

REACTOR BUILDING DRYWELL – MODE SHAPES, MODE 17

UPDATED FSAR

FIGURE 3.7-29



Mode Number 17
Frequency (CPS) = 12.34

Direction N-S = Triangle
Direction E-W = Square
Vertical = Cross

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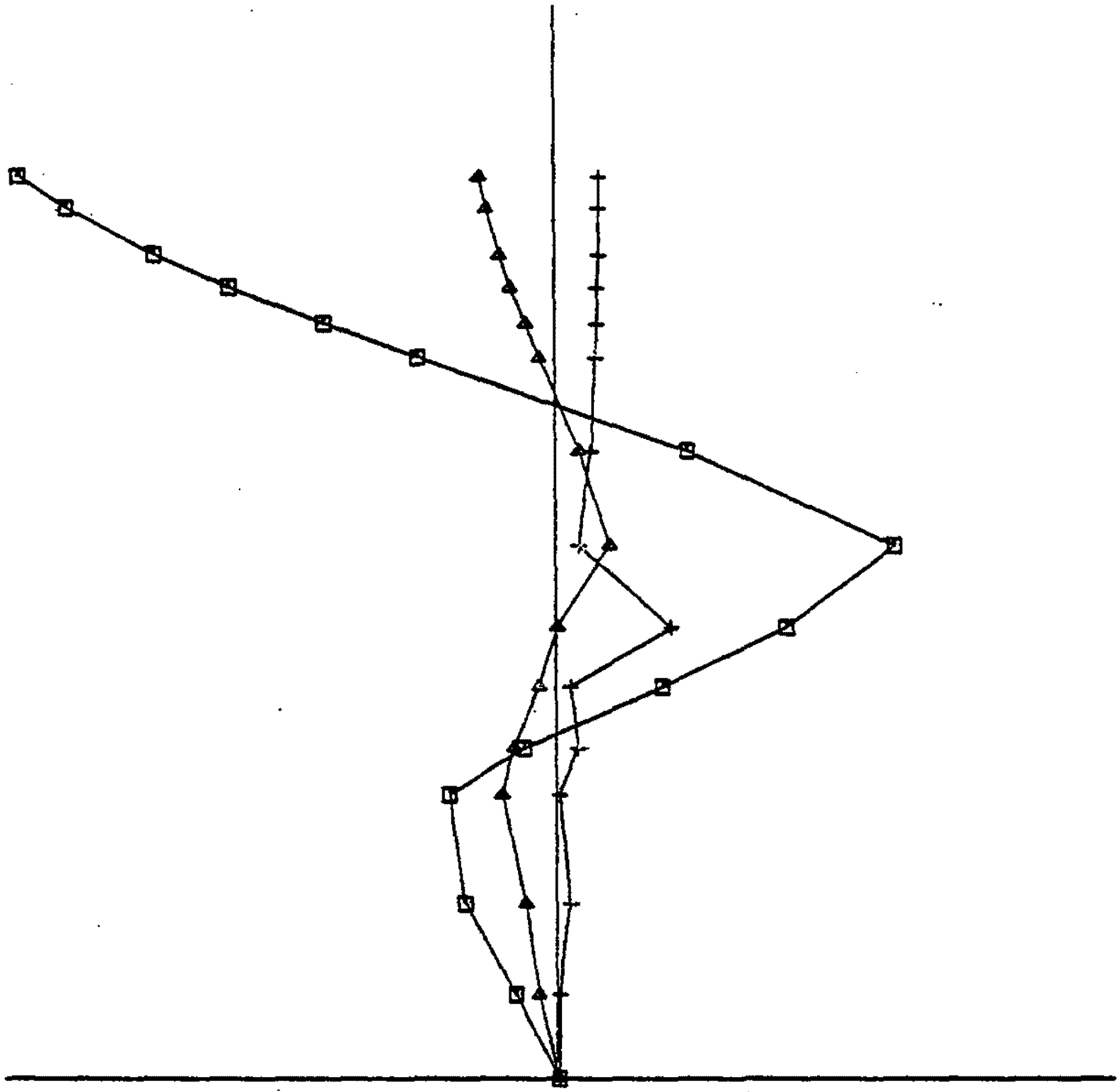
HOPE CREEK NUCLEAR GENERATING STATION

REACTOR BUILDING BIOLOGICAL SHIELD WALL AND RPV PEDESTAL – MODE SHAPES, MODE 17

UPDATED FSAR

FIGURE 3.7-30



Mode Number 18
Frequency (CPS) = 12.74

Direction N-S = Triangle
Direction E-W = Square
Vertical = Cross

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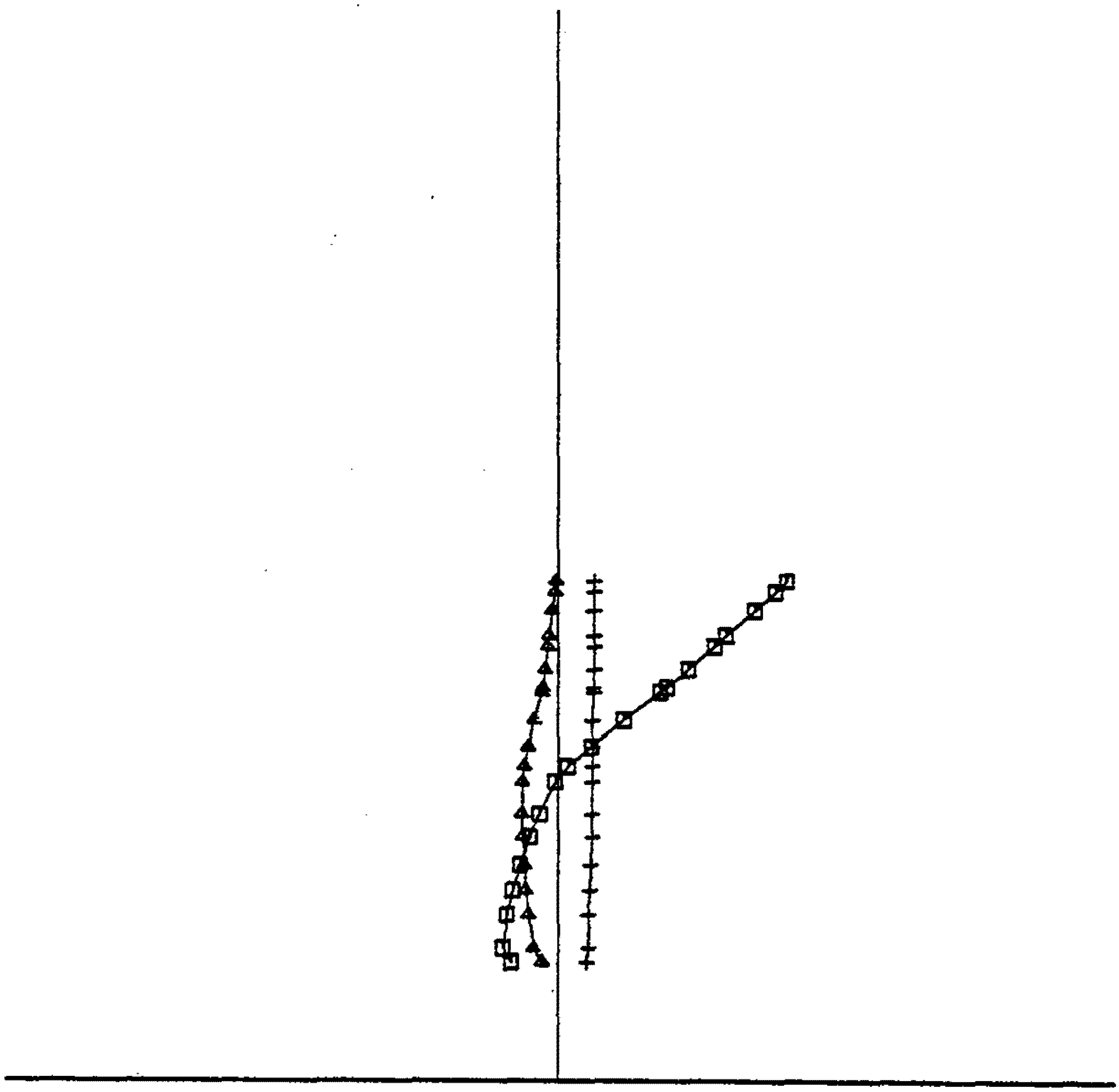
HOPE CREEK NUCLEAR GENERATING STATION

REACTOR BUILDING SHELL –
MODE SHAPES, MODE 18

UPDATED FSAR

FIGURE 3.7-31



Mode Number 18
Frequency (CPS) = 12.74

Direction N-S = Triangle
Direction E-W = Square
Vertical = Cross

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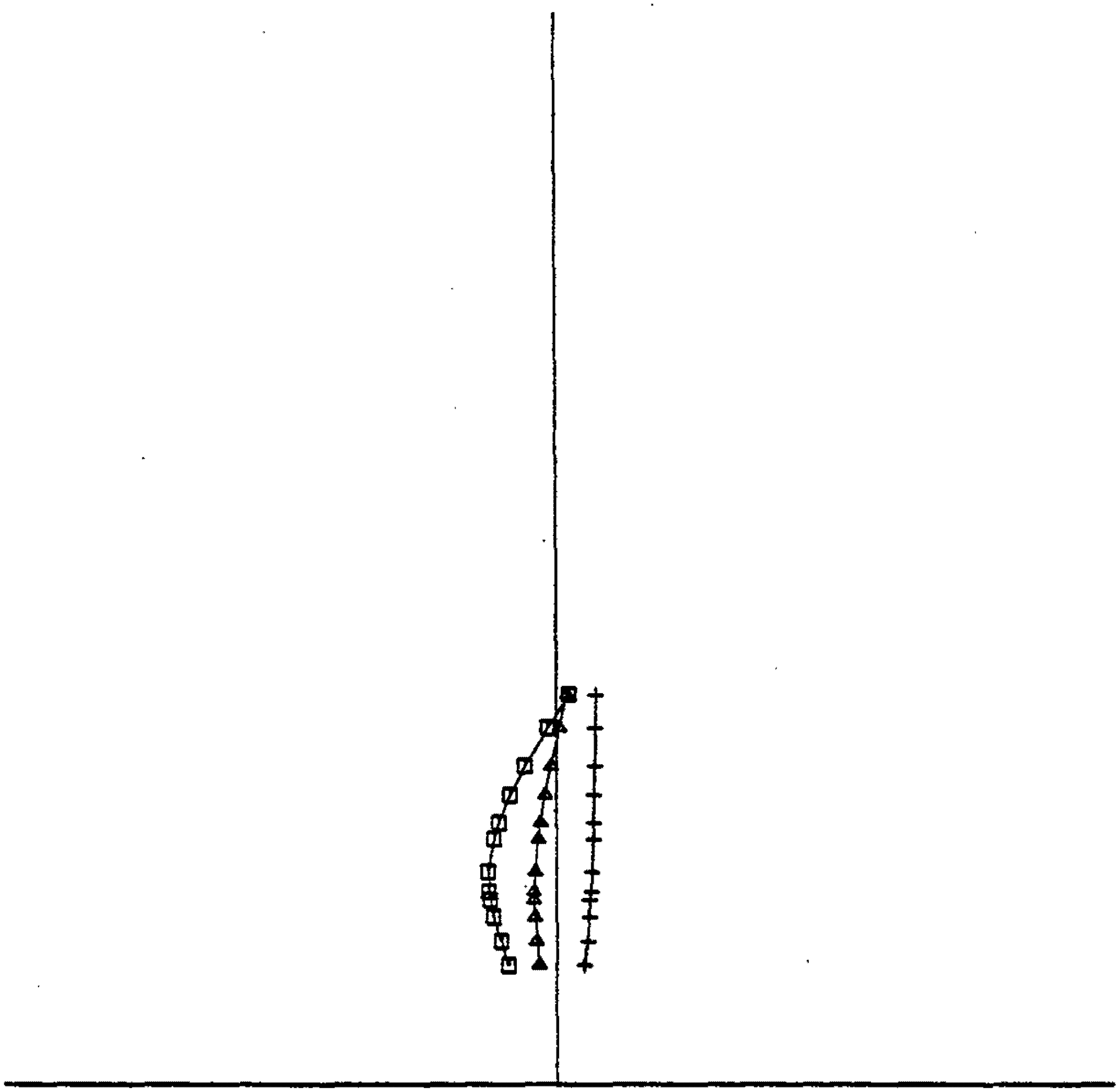
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

REACTOR BUILDING DRYWELL — MODE SHAPES, MODE 18

UPDATED FSAR

FIGURE 3.7-32



Mode Number 18
Frequency (CPS) = 12.74

Direction N-S = Triangle
Direction E-W = Square
Vertical = Cross

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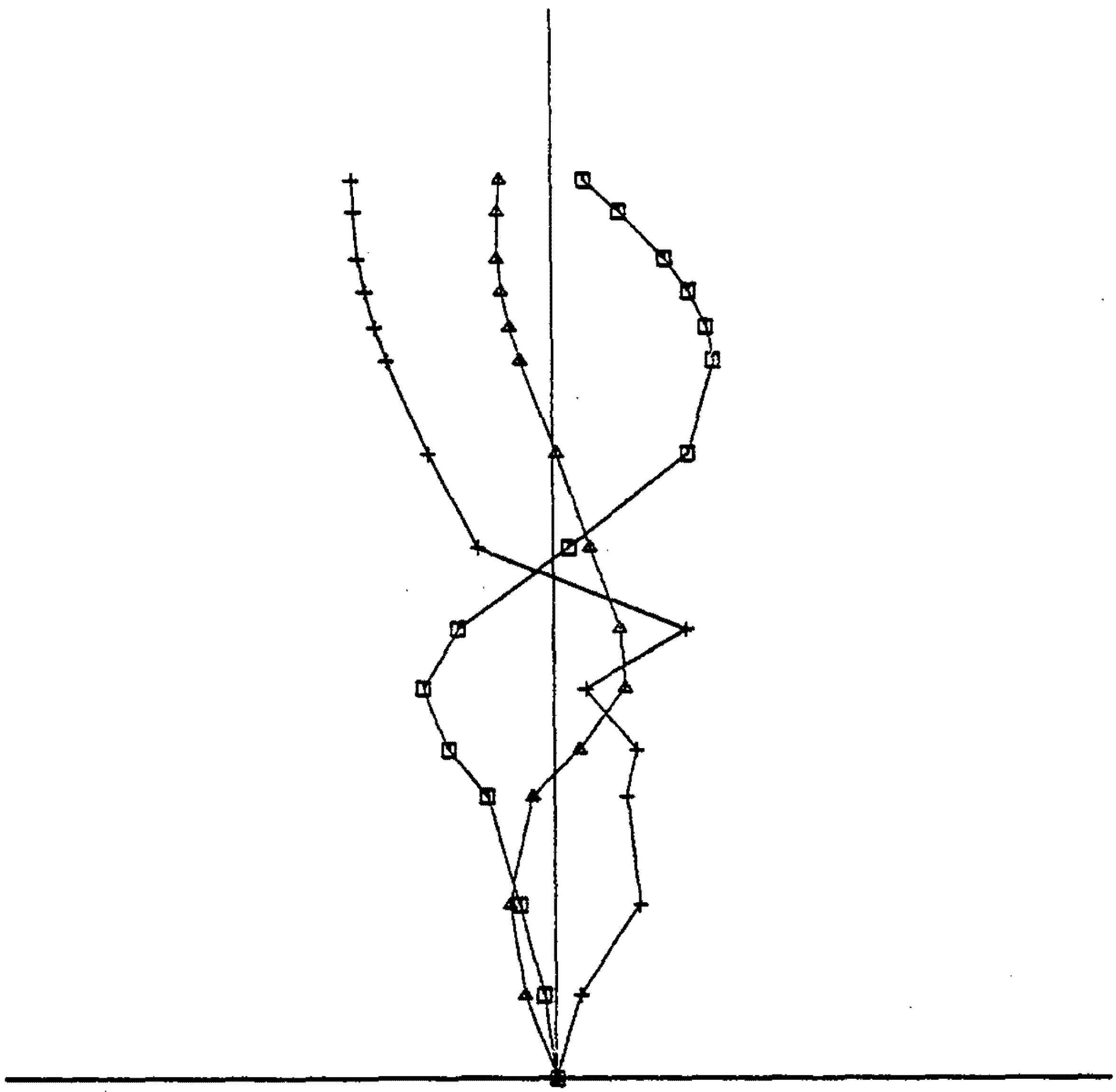
HOPE CREEK NUCLEAR GENERATING STATION

REACTOR BUILDING BIOLOGICAL SHIELD WALL AND RPV PEDESTAL — MODE SHAPES, MODE 18

UPDATED FSAR

FIGURE 3.7-33



Mode Number 27
Frequency (CPS) = 18.35

Direction N-S = Triangle
Direction E-W = Square
Vertical = Cross

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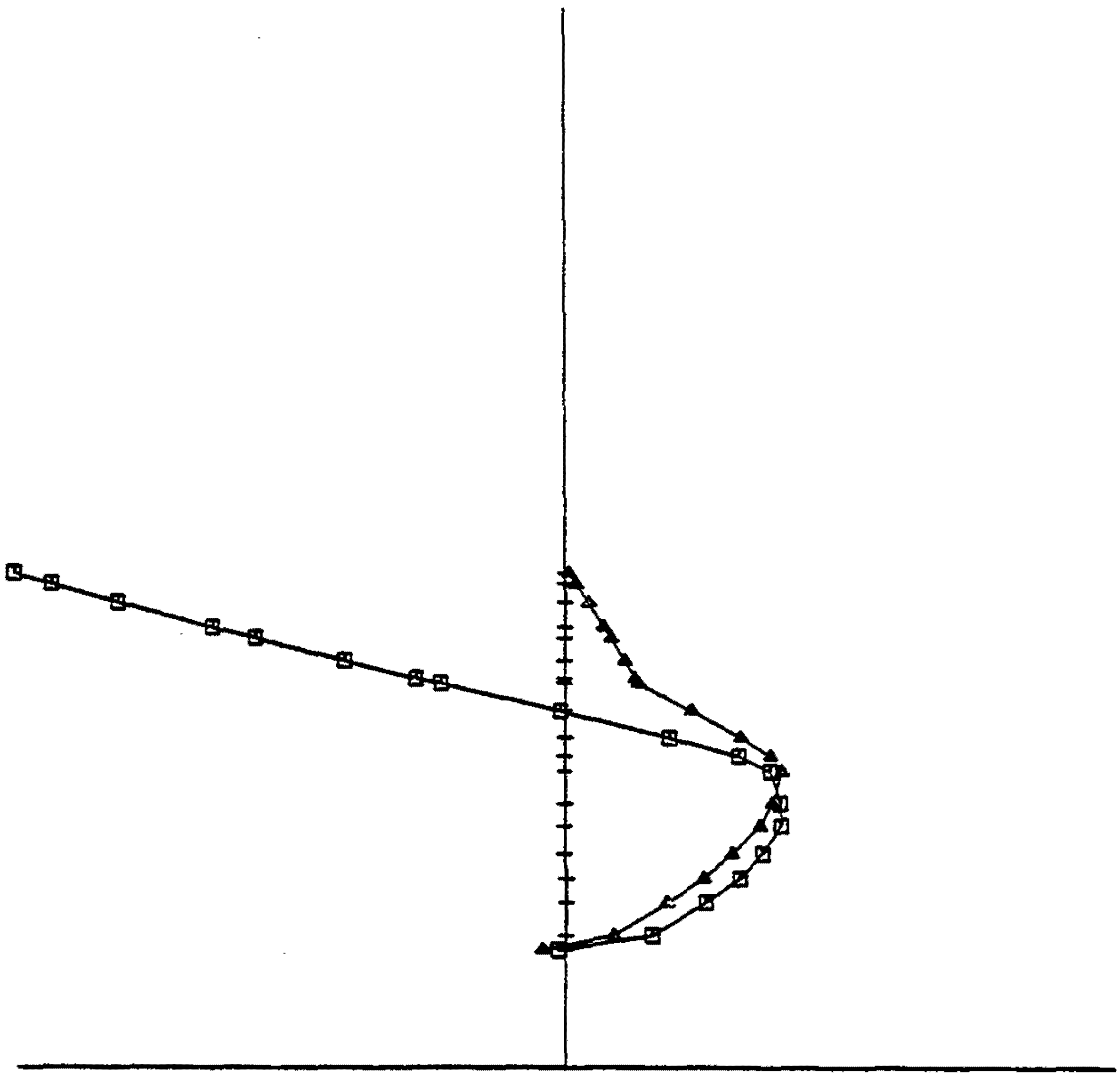
HOPE CREEK NUCLEAR GENERATING STATION

REACTOR BUILDING SHELL –
MODE SHAPES, MODE 27

UPDATED FSAR

FIGURE 3.7-34



Mode Number 27
Frequency (CPS) = 18.35

Direction N-S = Triangle
Direction E-W = Square
Vertical = Cross

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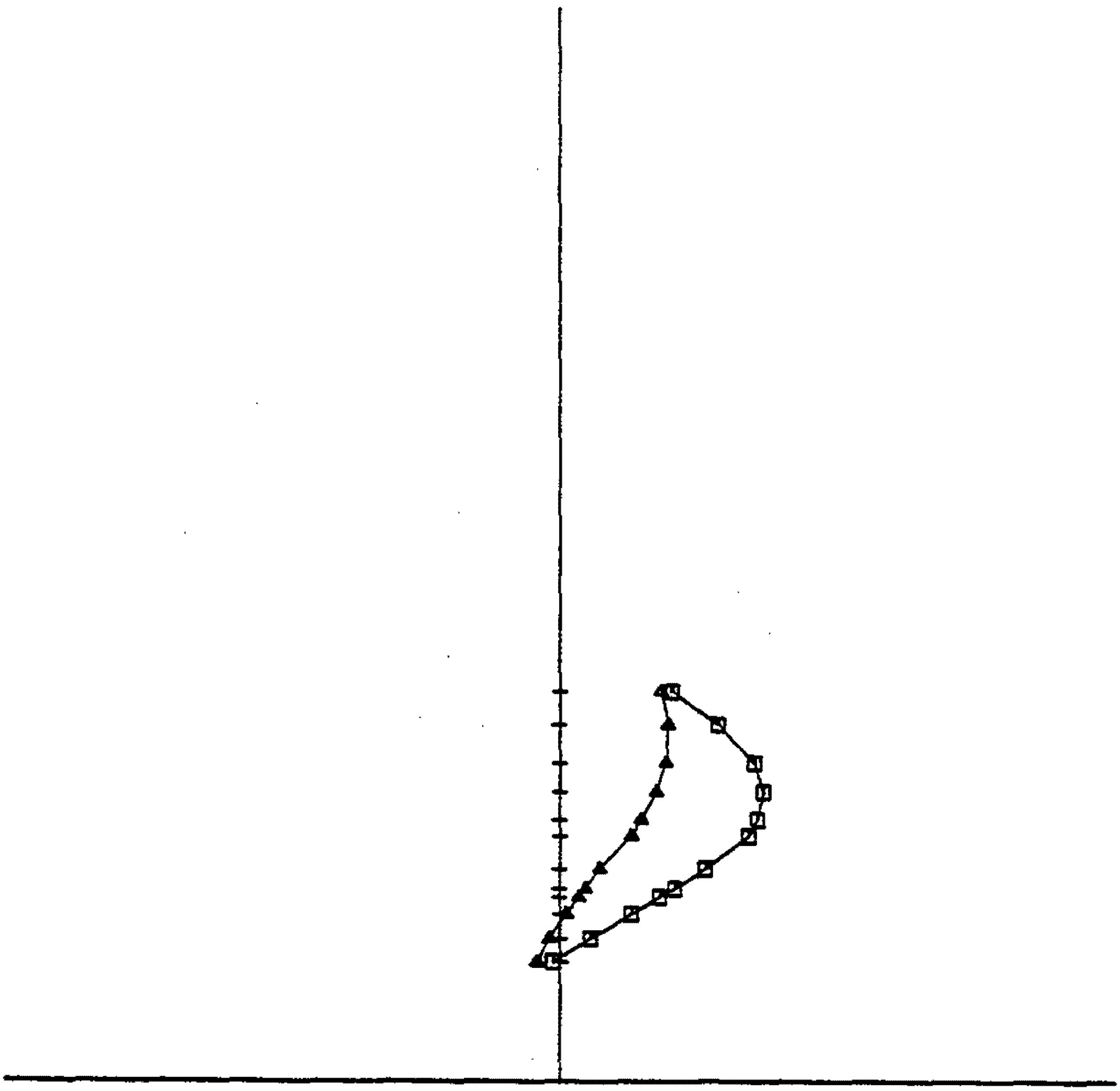
APRIL 11, 1988

PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

REACTOR BUILDING DRYWELL – MODE SHAPES, MODE 27

UPDATED FSAR

FIGURE 3.7-35



Mode Number 27
Frequency (CPS) = 18.35

Direction N-S = Triangle
Direction E-W = Square
Vertical = Cross

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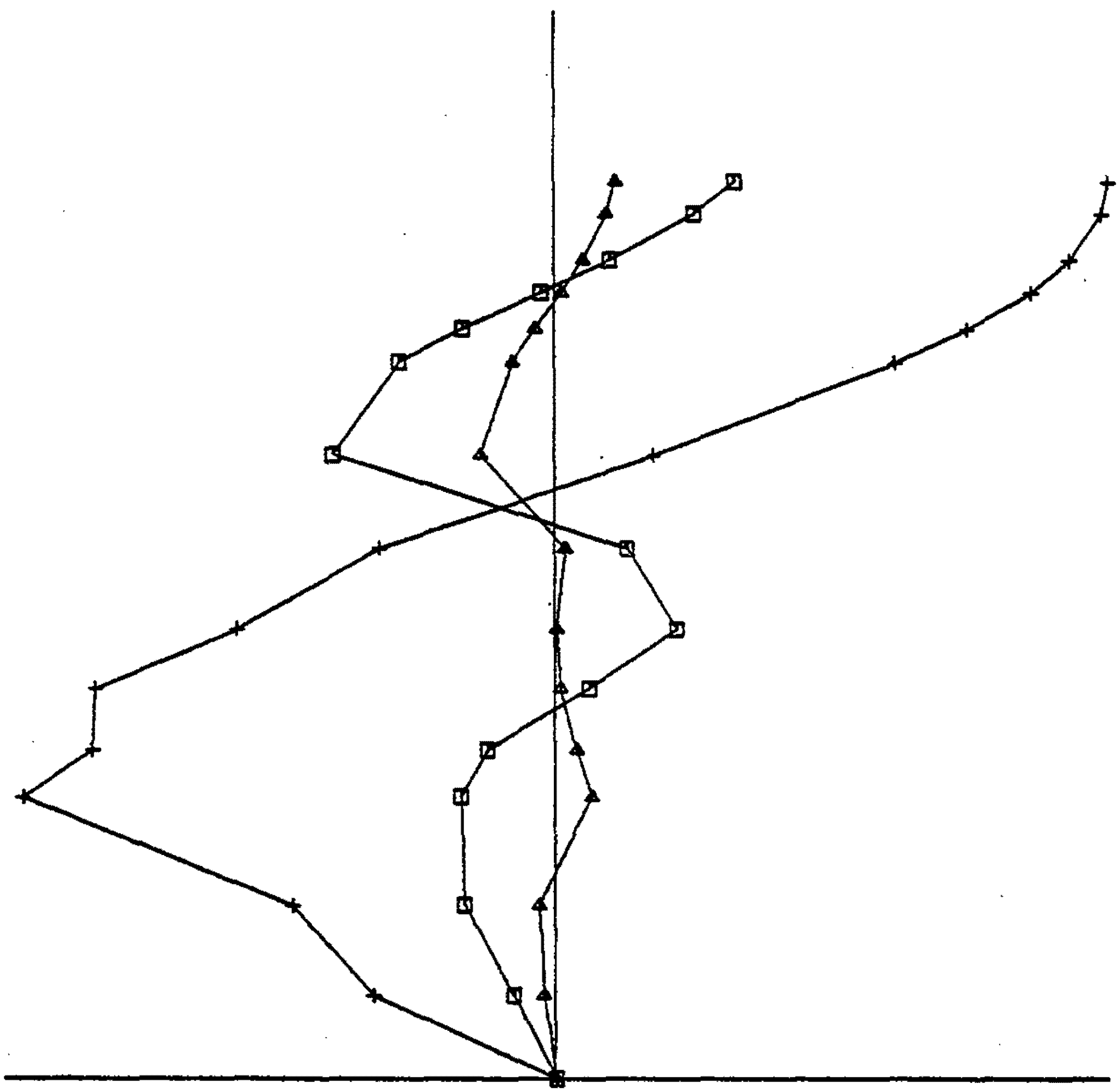
HOPE CREEK NUCLEAR GENERATING STATION

REACTOR BUILDING BIOLOGICAL SHIELD WALL AND RPV PEDESTAL — MODE SHAPES, MODE 27

UPDATED FSAR

FIGURE 3.7-36



Mode Number 45
Frequency (CPS) = 28.24

Direction N-S = Triangle
Direction E-W = Square
Vertical = Cross

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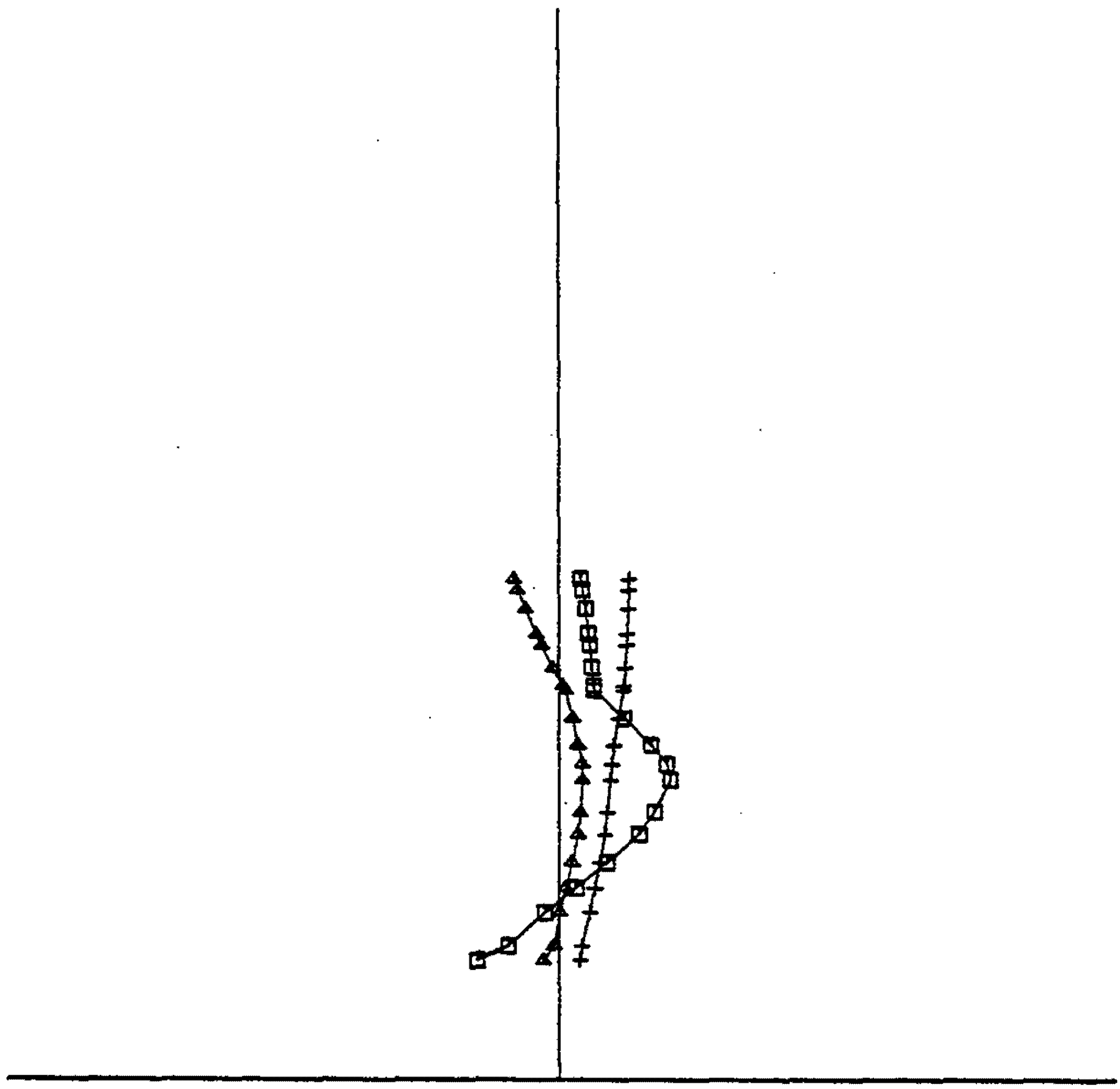
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

REACTOR BUILDING SHELL — MODE SHAPES, MODE 45

UPDATED FSAR

FIGURE 3.7-37



Mode Number 45
Frequency (CPS) = 28.24

Direction N-S = Triangle
Direction E-W = Square
Vertical = Cross

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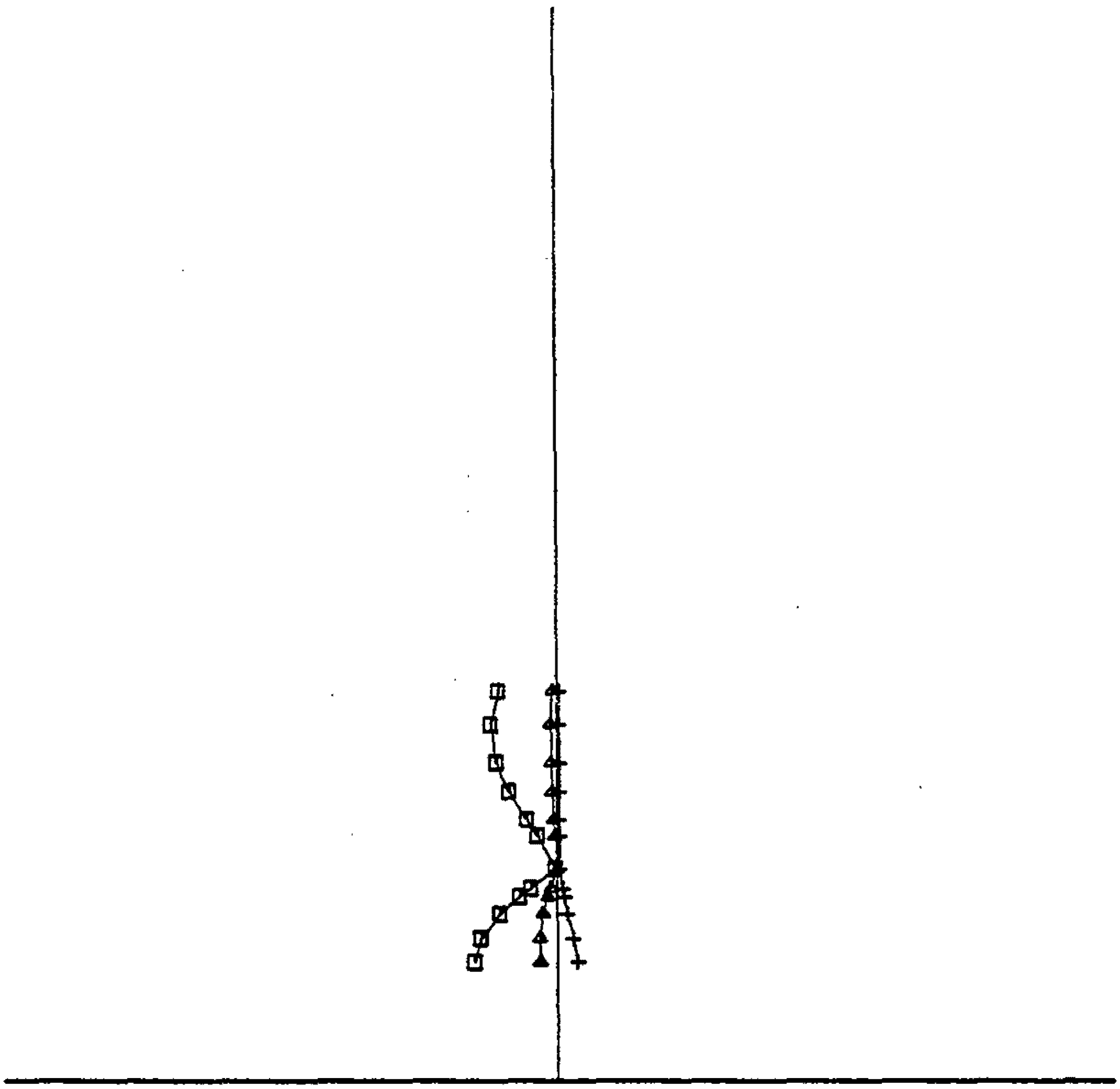
APRIL 11, 1988

PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

REACTOR BUILDING DRYWELL –
MODE SHAPES, MODE 45

UPDATED FSAR

FIGURE 3.7-38



Mode Number 45
Frequency (CPS) = 28.24

Direction N-S = Triangle
Direction E-W = Square
Vertical = Cross

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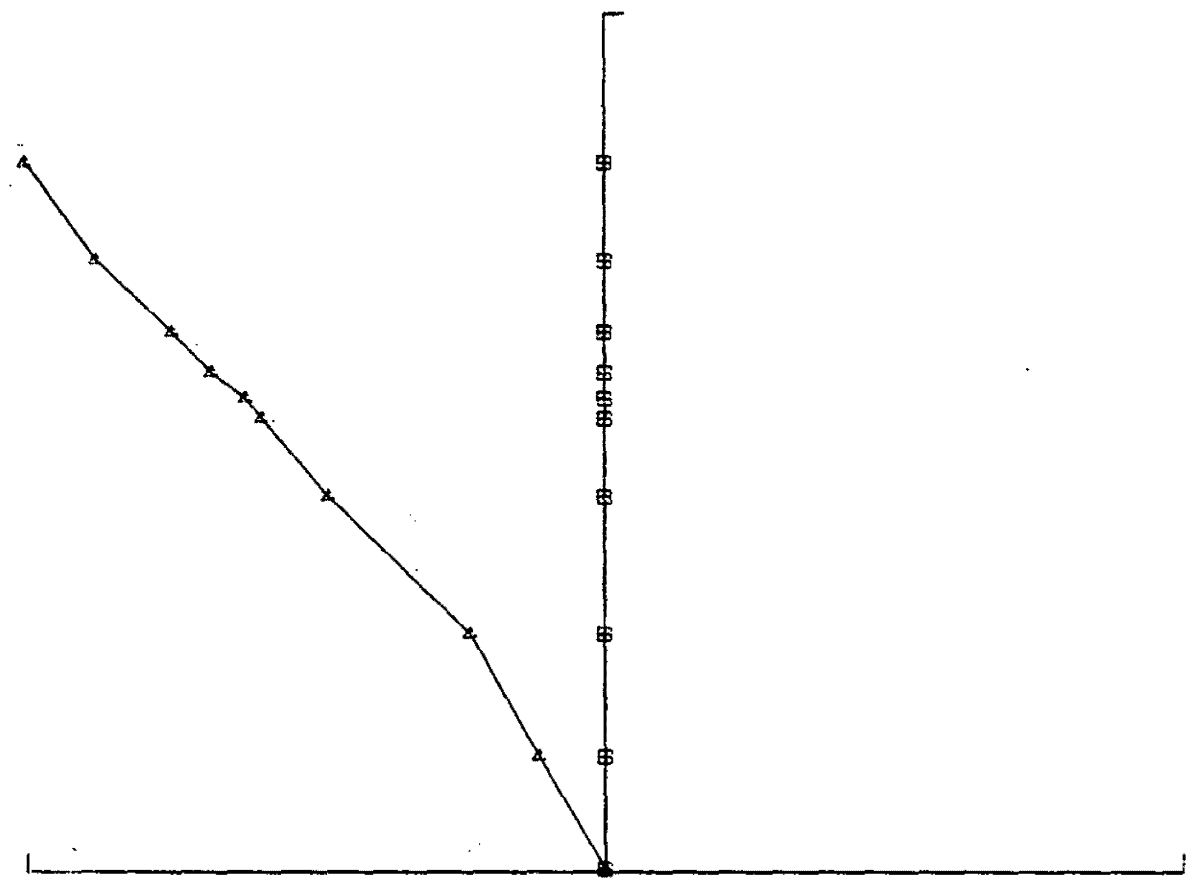
HOPE CREEK NUCLEAR GENERATING STATION

REACTOR BUILDING BIOLOGICAL SHIELD WALL AND RPV PEDESTAL – MODE SHAPES, MODE 45

UPDATED FSAR

FIGURE 3.7-39



Mode Number 4
Frequency (CPS) = 6.20

Direction N-S = Triangle
Direction E-W = Square
Vertical = Cross

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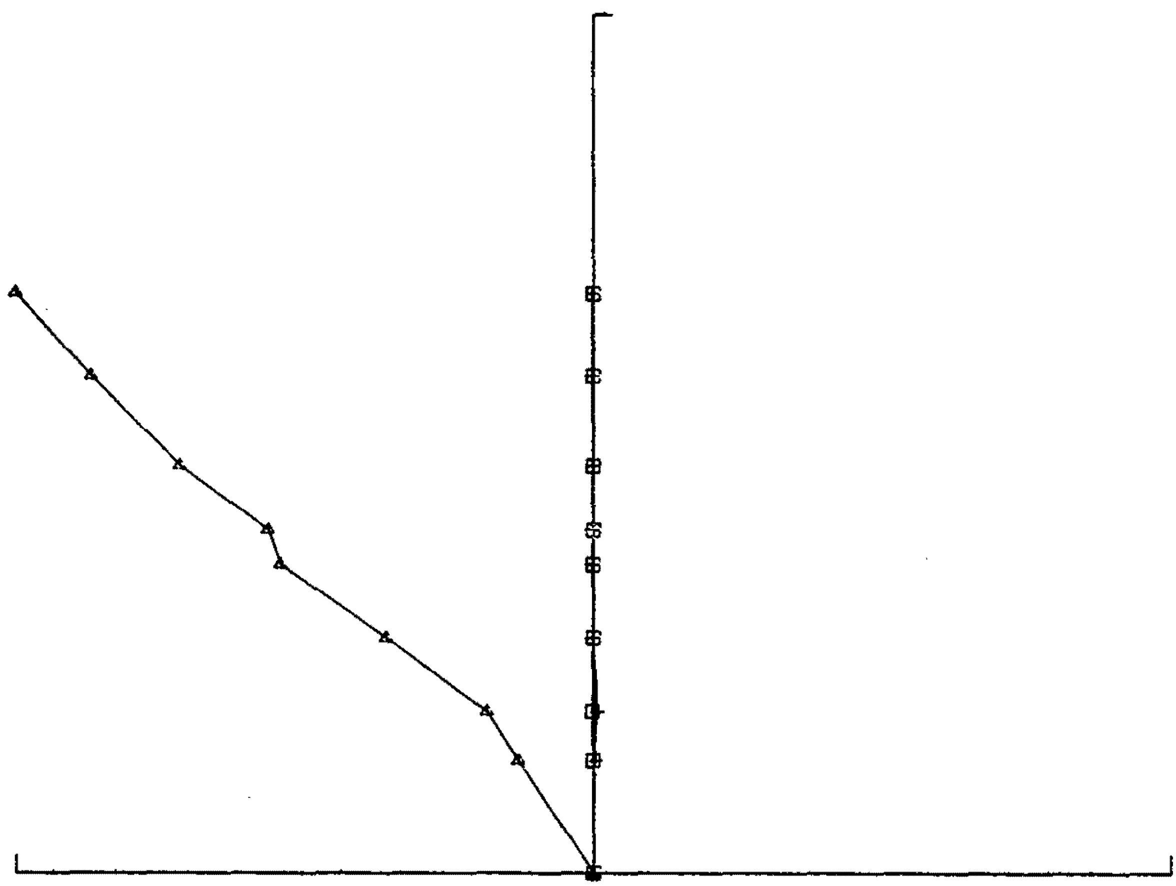
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

AUXILIARY BUILDING
DIESEL GENERATOR AREA –
MODE SHAPES, MODE 4

UPDATED FSAR

FIGURE 3.7-40



Mode Number 4
Frequency (CPS) = 6.20

Direction N-S = Triangle
Direction E-W = Square
Vertical = Cross

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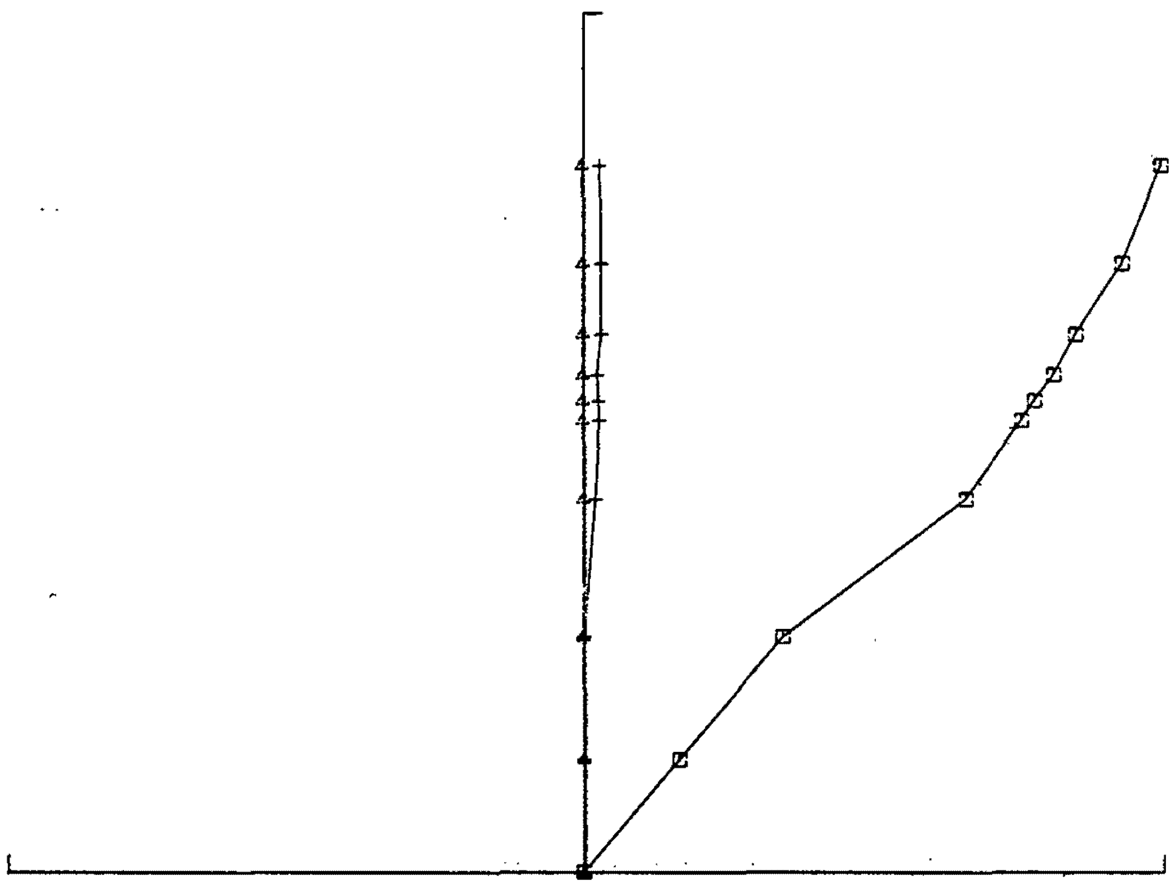
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

AUXILIARY BUILDING CONTROL AND RADWASTE AREAS – MODE SHAPES, MODE 4

UPDATED FSAR

FIGURE 3.7-41



Mode Number 5
Frequency (CPS) = 6.48

Direction N-S = Triangle
Direction E-W = Square
Vertical = Cross

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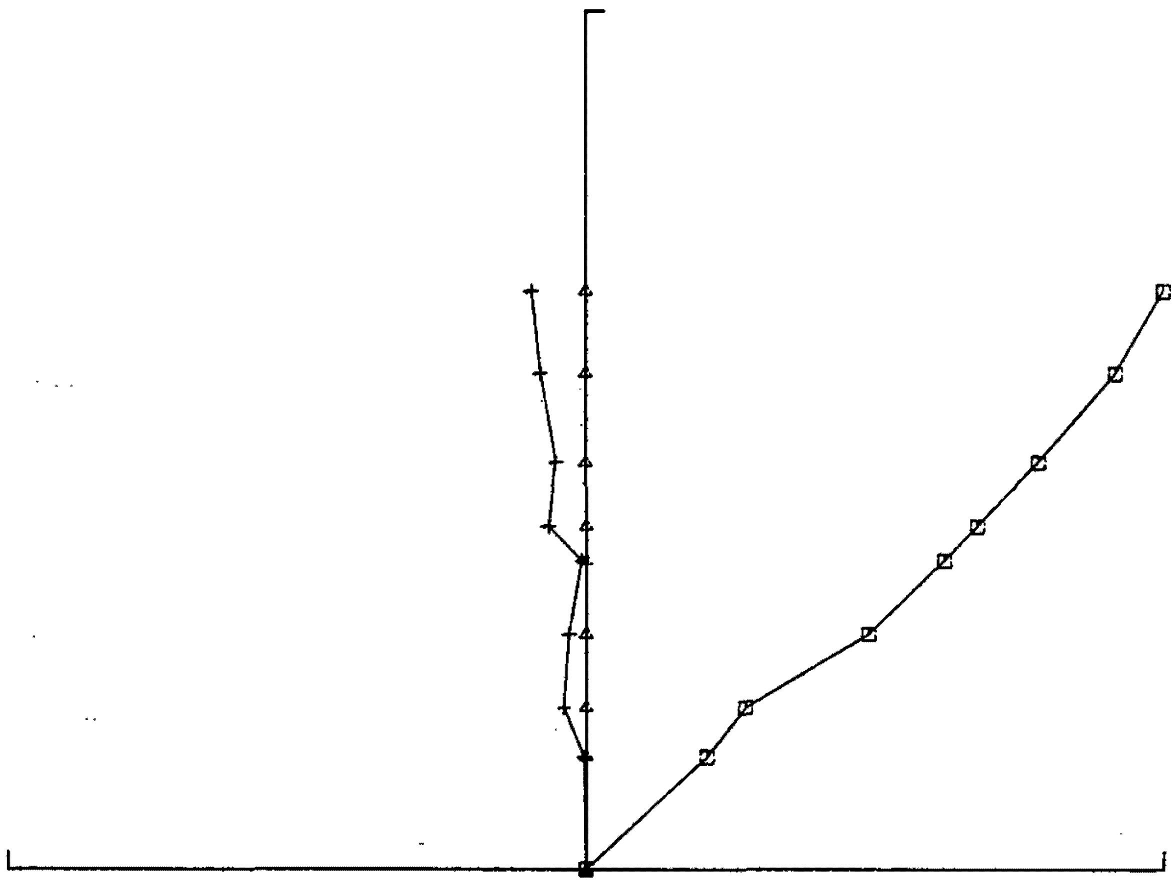
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

AUXILIARY BUILDING
DIESEL GENERATOR AREA –
MODE SHAPES, MODE 5

UPDATED FSAR

FIGURE 3.7-42



Mode Number 5
Frequency (CPS) = 6.48

Direction N-S = Triangle
Direction E-W = Square
Vertical = Cross

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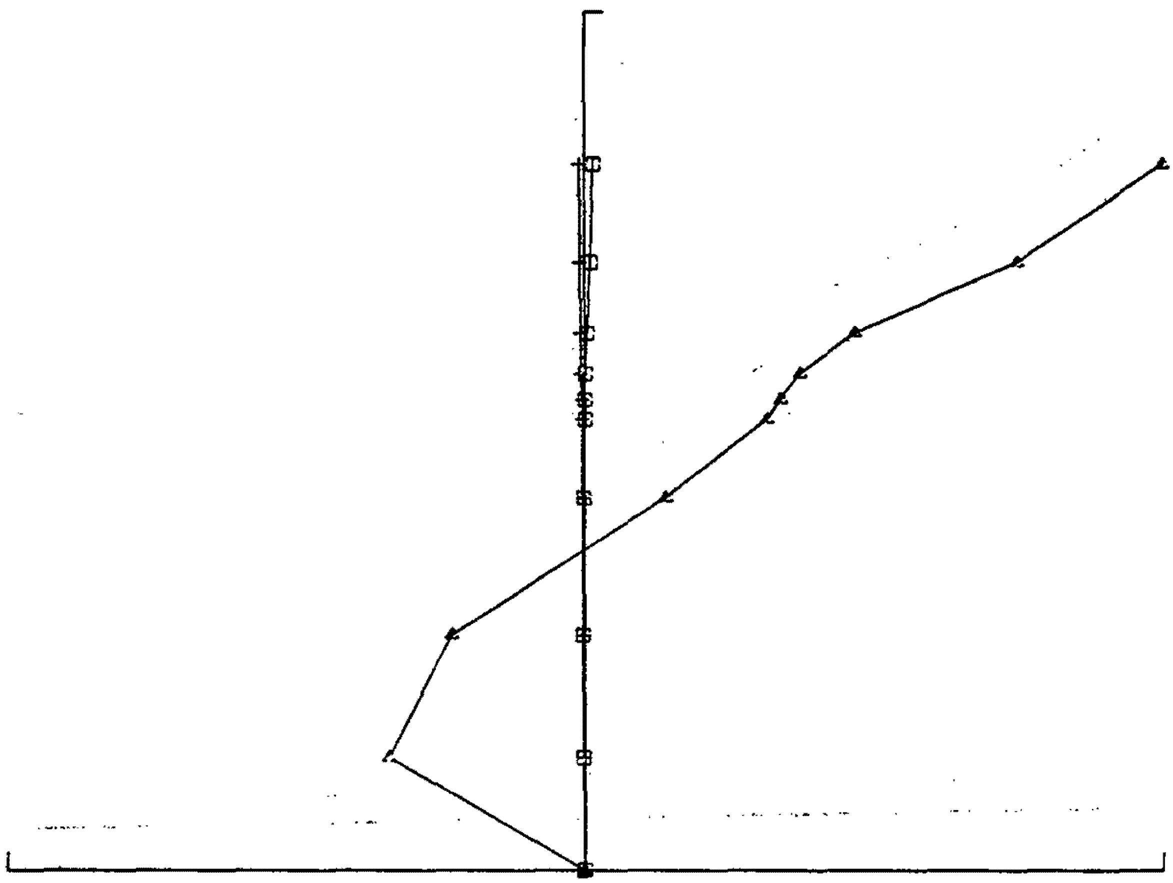
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

AUXILIARY BUILDING CONTROL AND RADWASTE AREAS — MODE SHAPES, MODE 5

UPDATED FSAR

FIGURE 3.7-43



Mode Number 6
Frequency (CPS) = 12.36

Direction N-S = Triangle
Direction E-W = Square
Vertical = Cross

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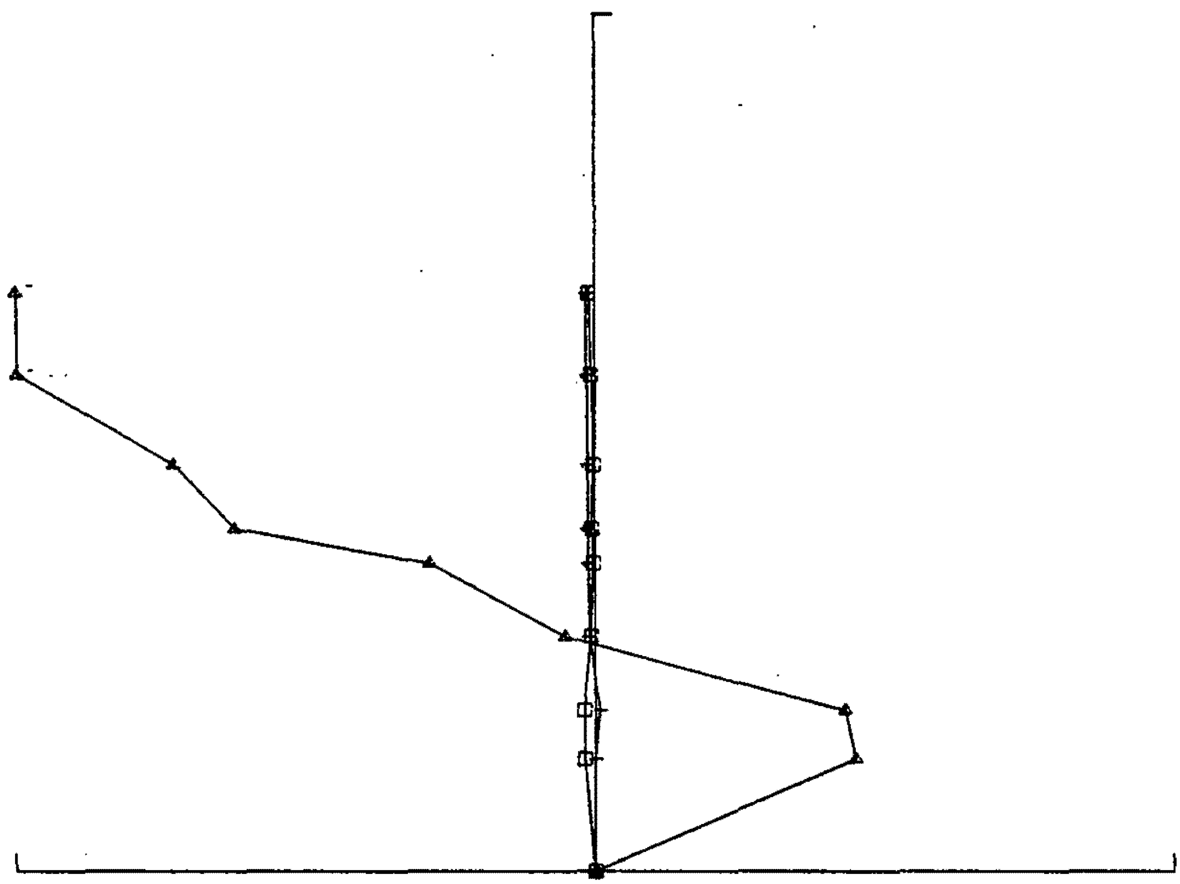
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

AUXILIARY BUILDING
DIESEL GENERATOR AREA –
MODE SHAPES, MODE 6

UPDATED FSAR

FIGURE 3.7-44



Mode Number 6
Frequency (CPS) =12.36

Direction N-S = Triangle
Direction E-W = Square
Vertical = Cross

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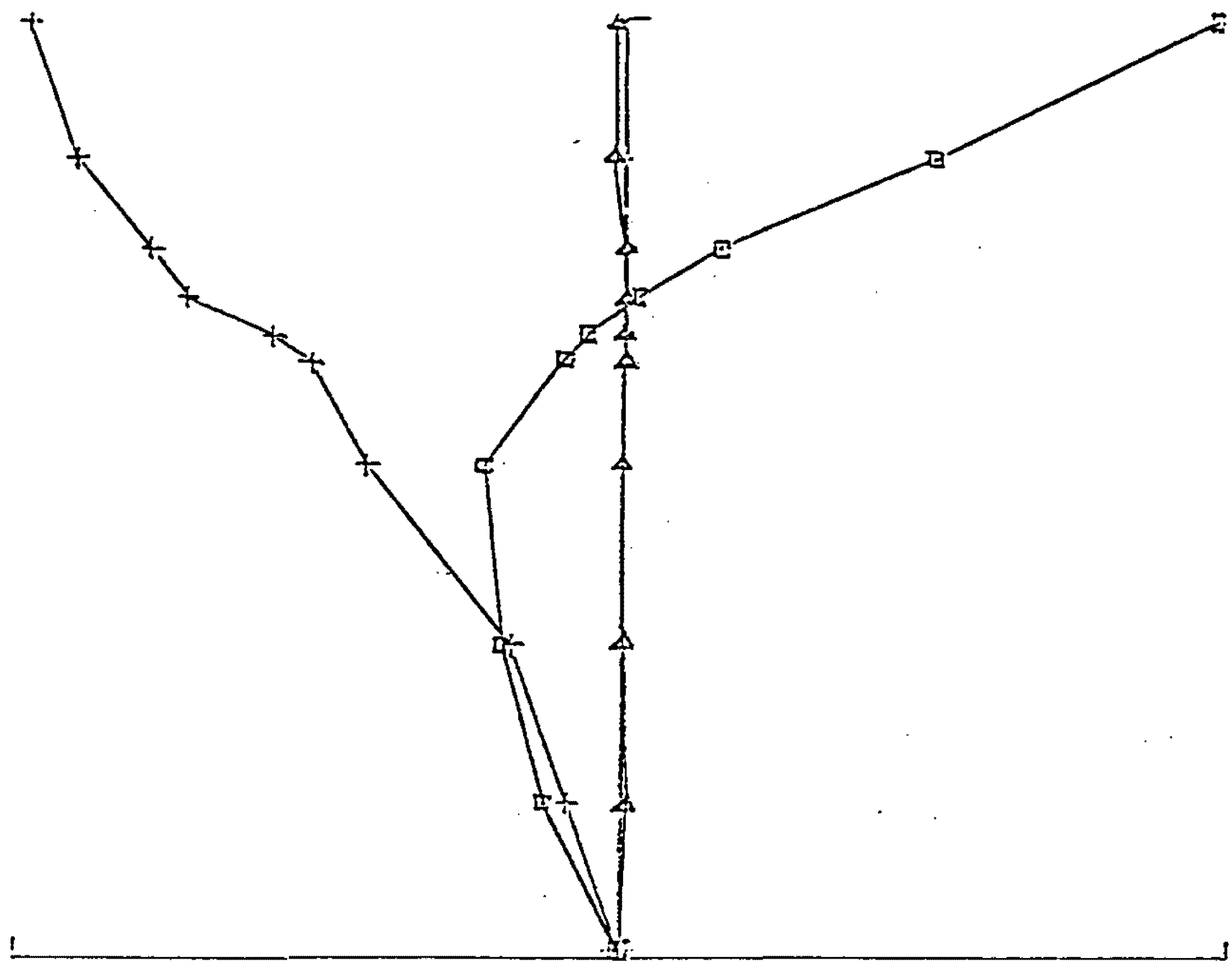
APRIL 11, 1988

PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

AUXILIARY BUILDING CONTROL AND RADWASTE AREAS – MODE SHAPES, MODE 6

UPDATED FSAR

FIGURE 3.7-45



Mode Number 7

Frequency = 13.24 cps

Direction X = TRIANGLE

Direction Y = SQUARE

Direction Z = CROSS

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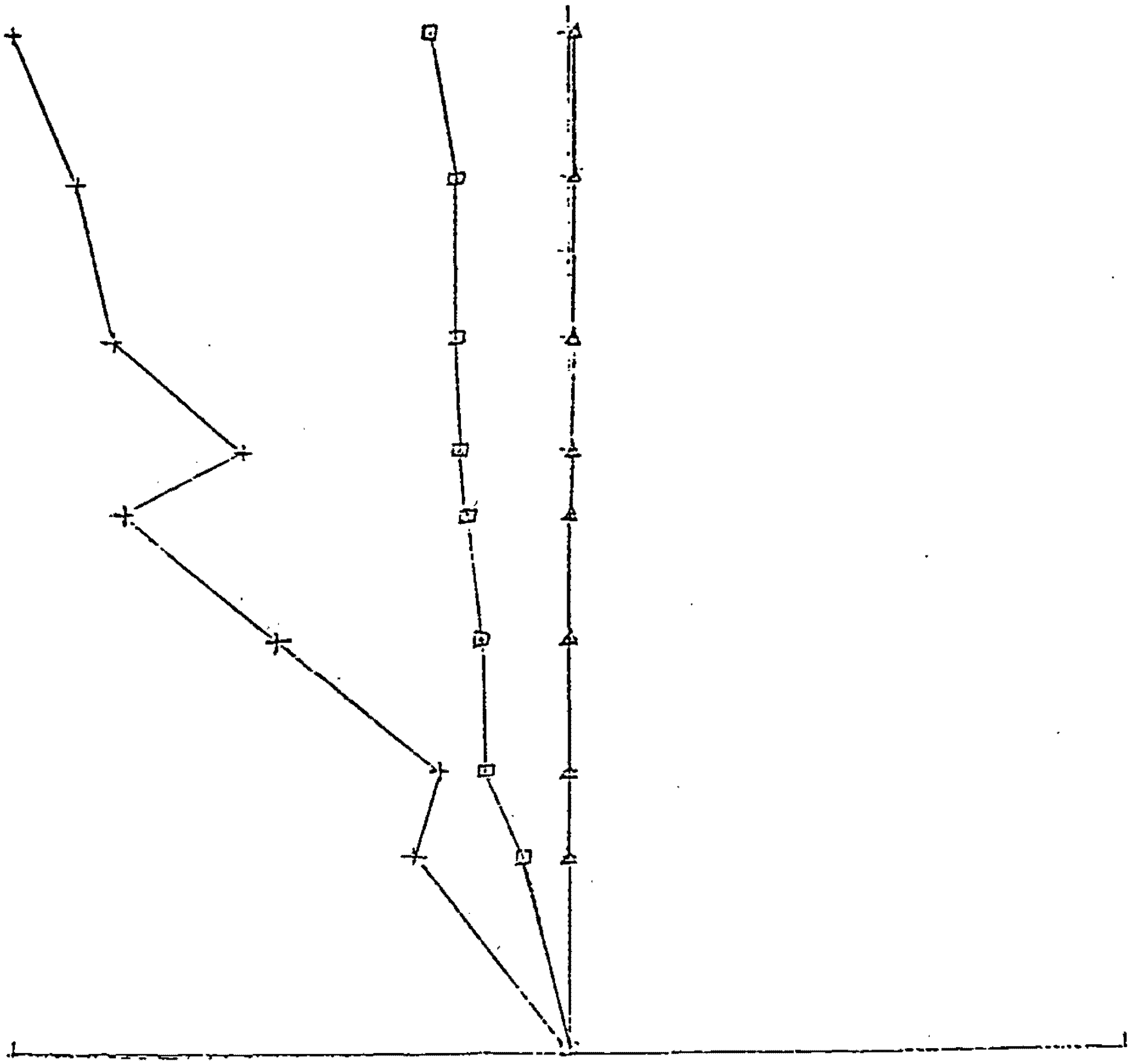
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

AUXILIARY BUILDING
DIESEL GENERATOR AREA –
MODE SHAPES, MODE 7

UPDATED FSAR

FIGURE 3.7-46



Mode Number 7
Frequency = 13.24 cps

Direction X = TRIANGLE

Direction Y = SQUARE

Direction Z = CROSS

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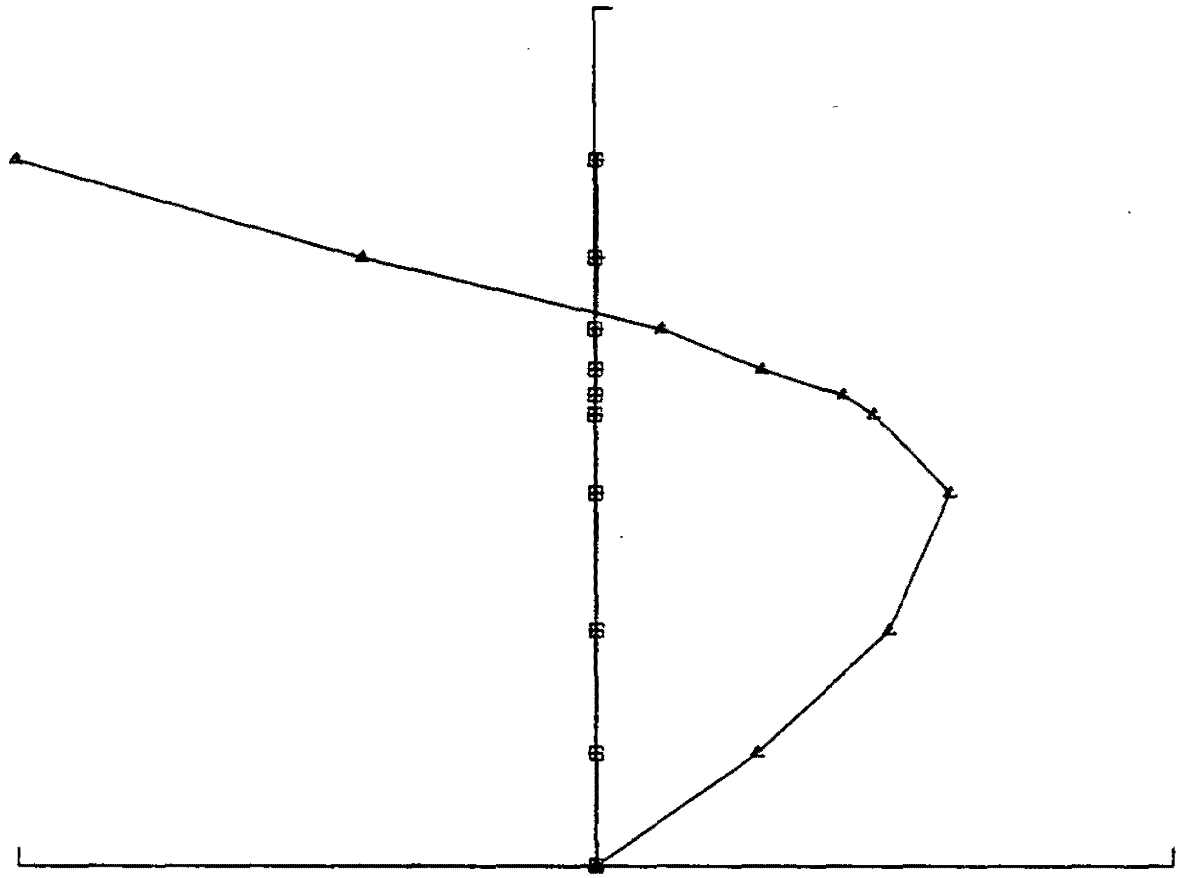
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

AUXILIARY BUILDING CONTROL AND RADWASTE AREAS – MODE SHAPES, MODE 7

UPDATED FSAR

FIGURE 3.7-47



Mode Number 8
Frequency (CPS) = 15.28

Direction N-S = Triangle
Direction E-W = Square
Vertical = Cross

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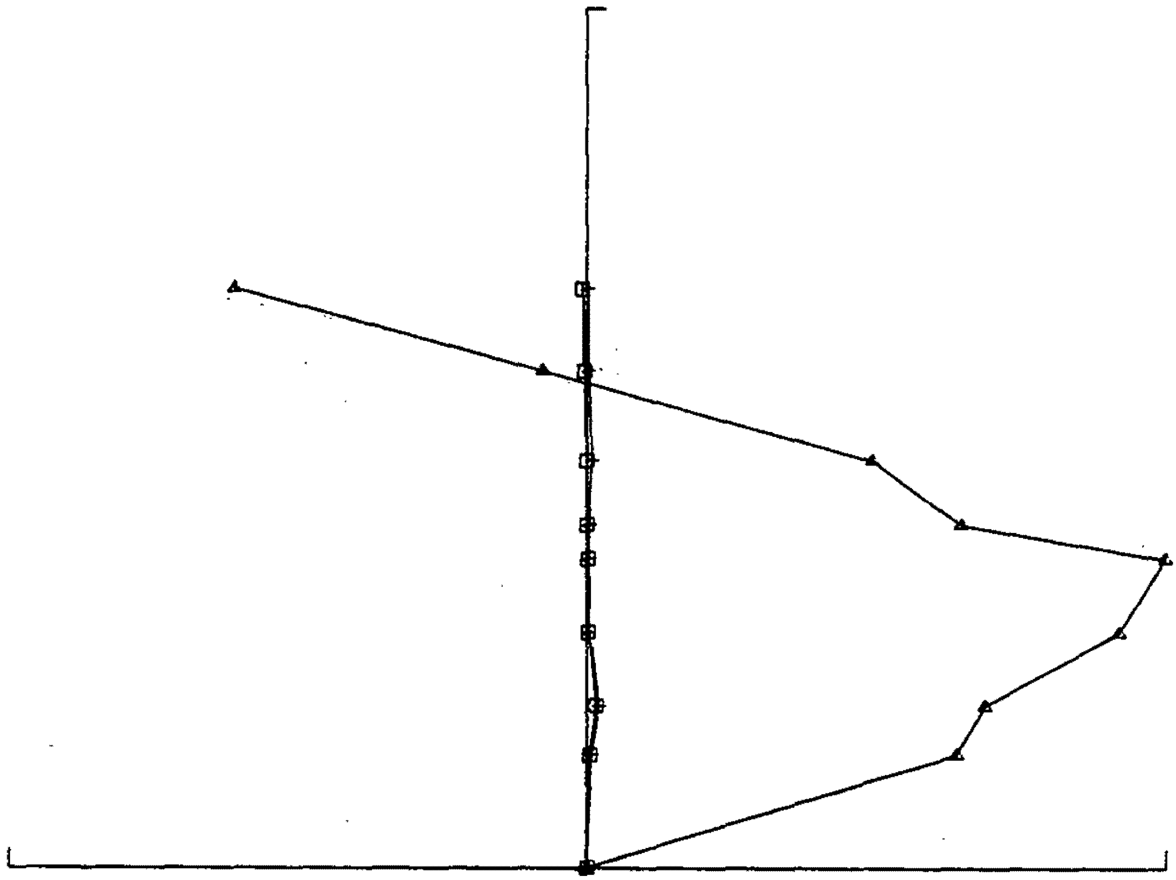
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

AUXILIARY BUILDING
DIESEL GENERATOR AREA –
MODE SHAPES, MODE 8

UPDATED FSAR

FIGURE 3.7-48



Mode Number 8
Frequency (CPS) = 15.28

Direction N-S = Triangle
Direction E-W = Square
Vertical = Cross

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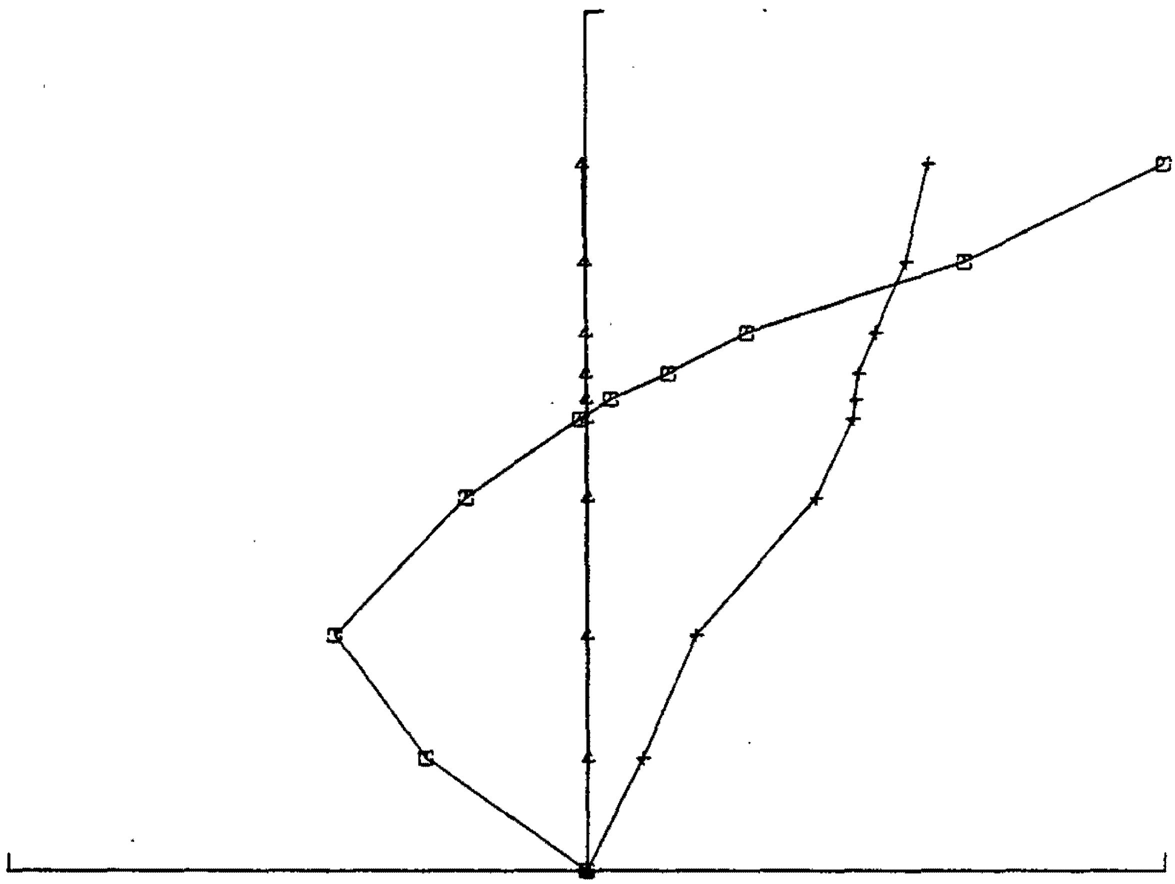
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

AUXILIARY BUILDING CONTROL AND RADWASTE AREAS – MODE SHAPES, MODE 8

UPDATED FSAR

FIGURE 3.7-49



Mode Number 9
Frequency (CPS) = 16.54

Direction N-S = Triangle
Direction E-W = Square
Vertical = Cross

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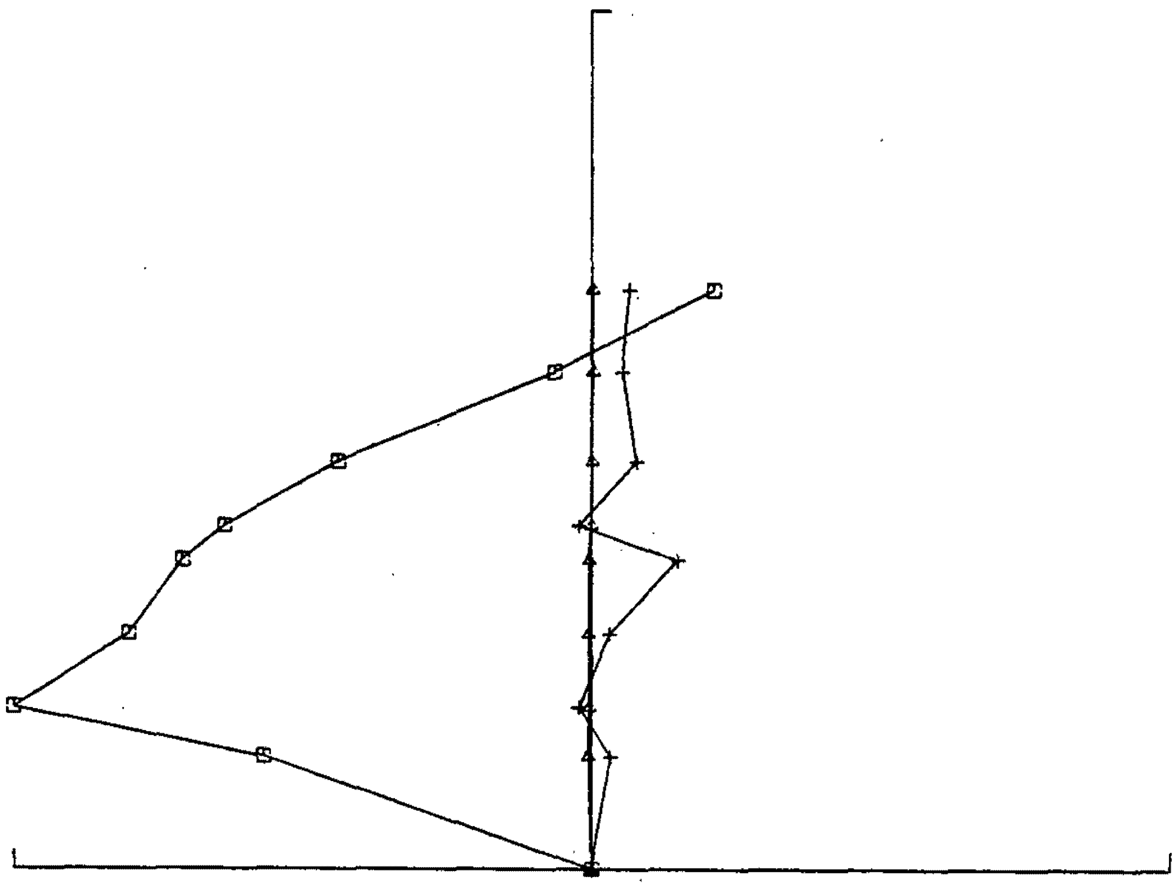
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

AUXILIARY BUILDING
DIESEL GENERATOR AREA –
MODE SHAPES, MODE 9

UPDATED FSAR

FIGURE 3.7-50



Mode Number 9
Frequency (CPS) = 16.54

Direction N-S = Triangle
Direction E-W = Square
Vertical = Cross

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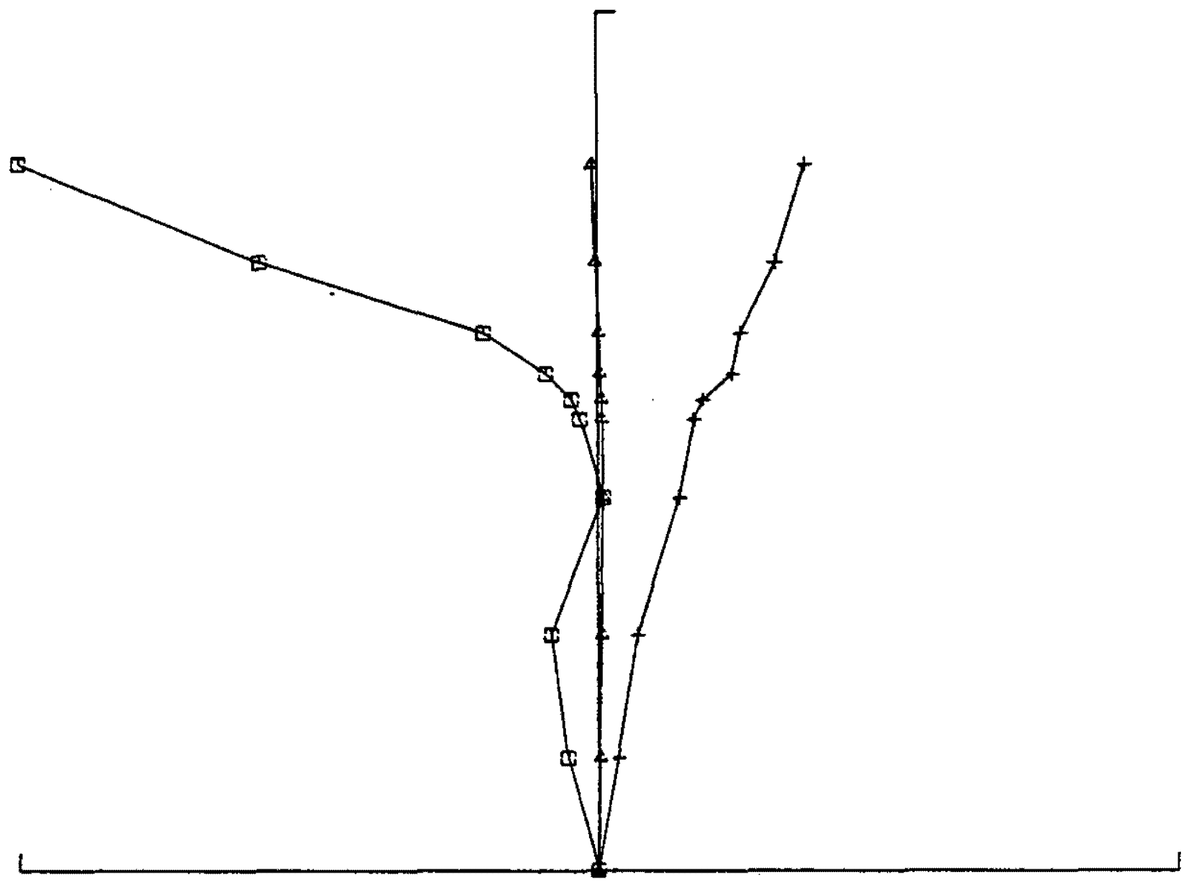
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

AUXILIARY BUILDING CONTROL AND RADWASTE AREAS – MODE SHAPES, MODE 9

UPDATED FSAR

FIGURE 3.7-51



Mode Number 10
Frequency (CPS) = 17.54

Direction N-S = Triangle
Direction E-W = Square
Vertical = Cross

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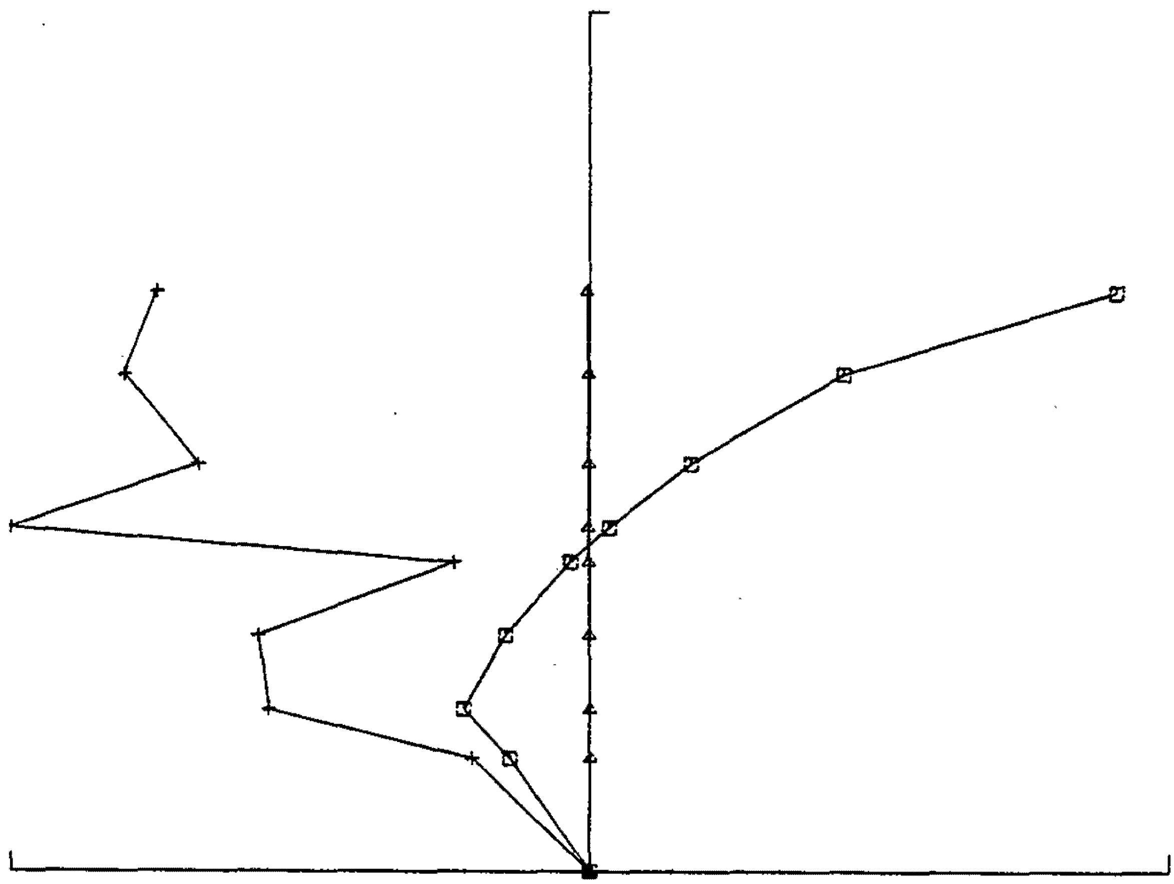
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

AUXILIARY BUILDING
DIESEL GENERATOR AREA –
MODE SHAPES, MODE 10

UPDATED FSAR

FIGURE 3.7-52



Mode Number 10
Frequency (CPS) = 17.54

Direction N-S = Triangle
Direction E-W = Square
Vertical = Cross

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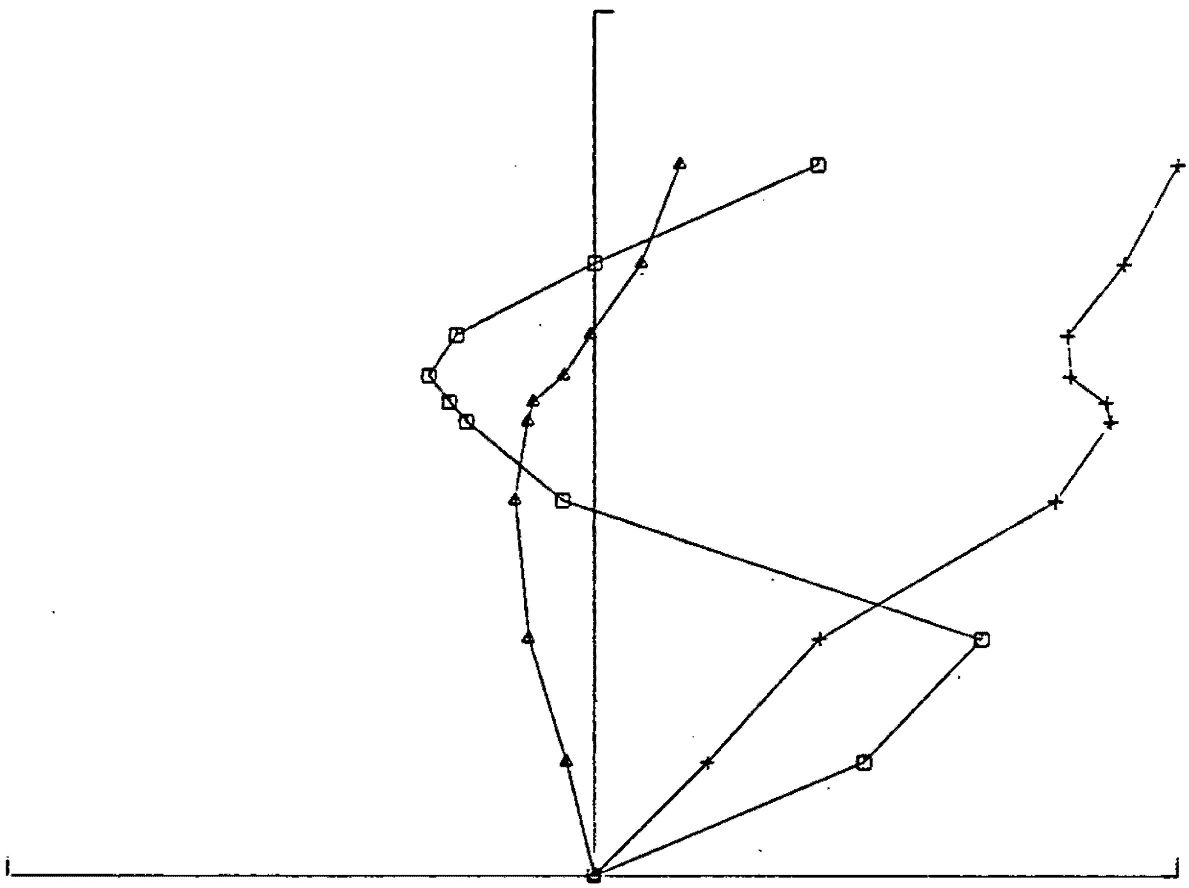
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

AUXILIARY BUILDING CONTROL AND RADWASTE AREAS – MODE SHAPES, MODE 10

UPDATED FSAR

FIGURE 3.7-53



MODE NUMBER 11
FREQUENCY (CPS) = 19.80

DIRECTION X = TRIANGLE
DIRECTION Y = SQUARE
DIRECTION Z = CROSS

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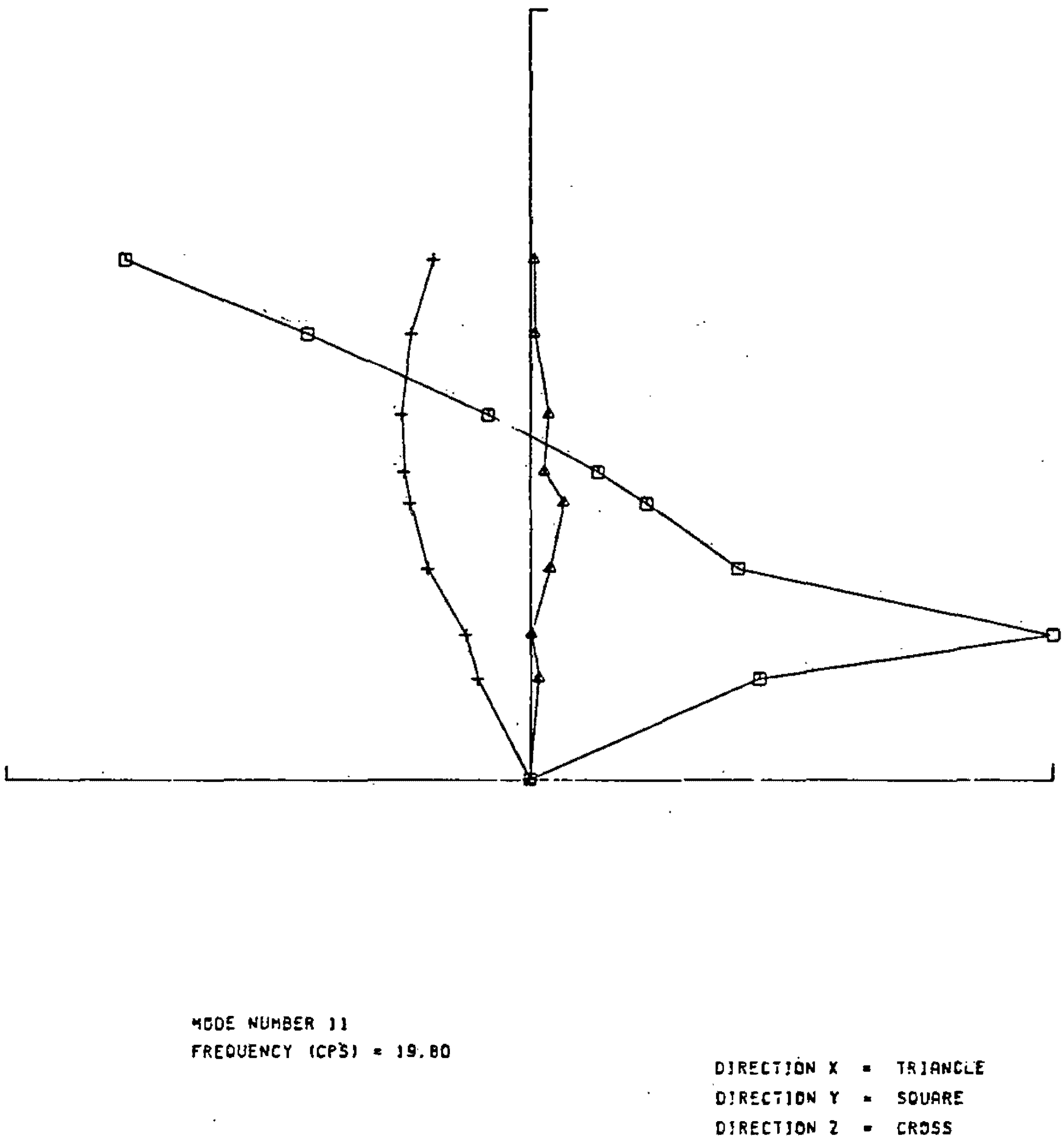
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

AUXILIARY BUILDING
DIESEL GENERATOR AREA –
MODE SHAPES, MODE 11

UPDATED FSAR

FIGURE 3.7-54



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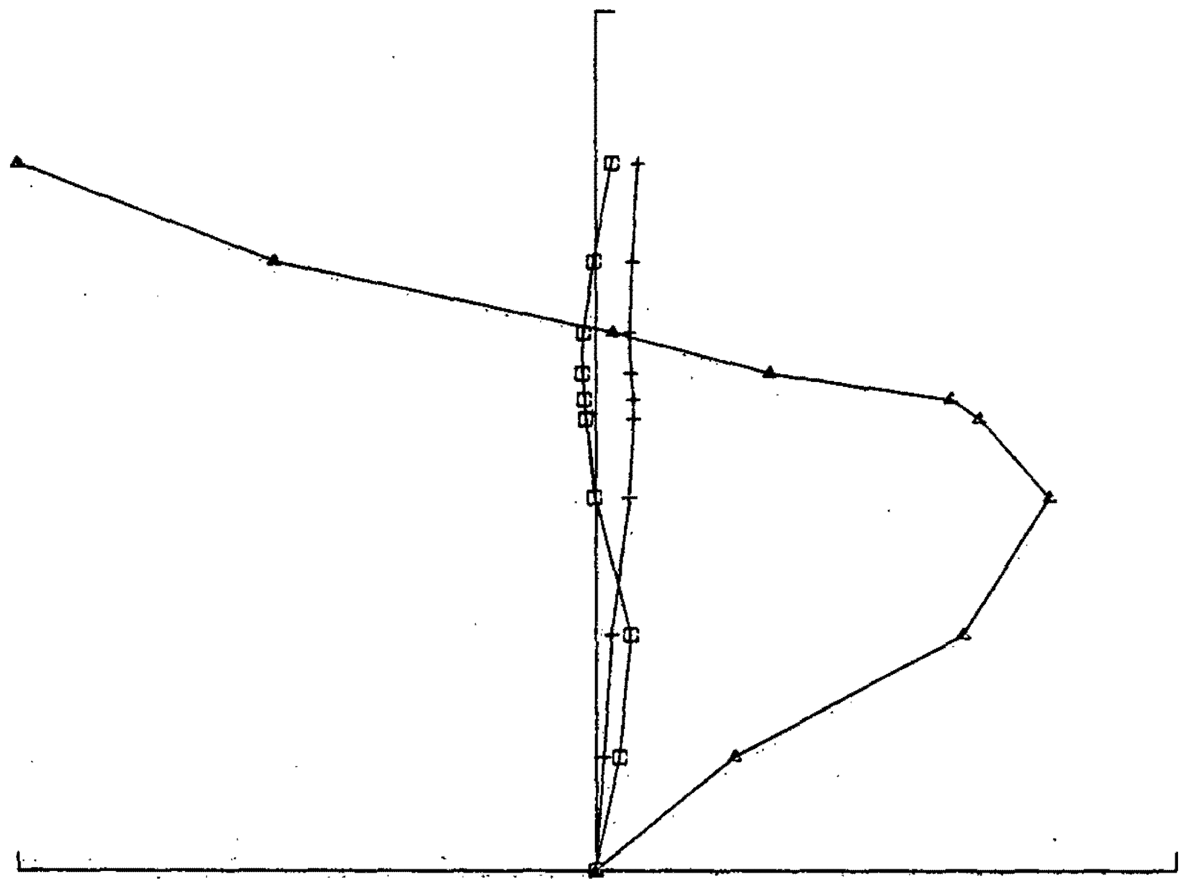
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

AUXILIARY BUILDING CONTROL AND RADWASTE AREAS – MODE SHAPES, MODE 11

UPDATED FSAR

FIGURE 3.7-55



Mode Number 12
Frequency (CPS) = 20.10

Direction N-S = Triangle
Direction E-W = Square
Vertical = Cross

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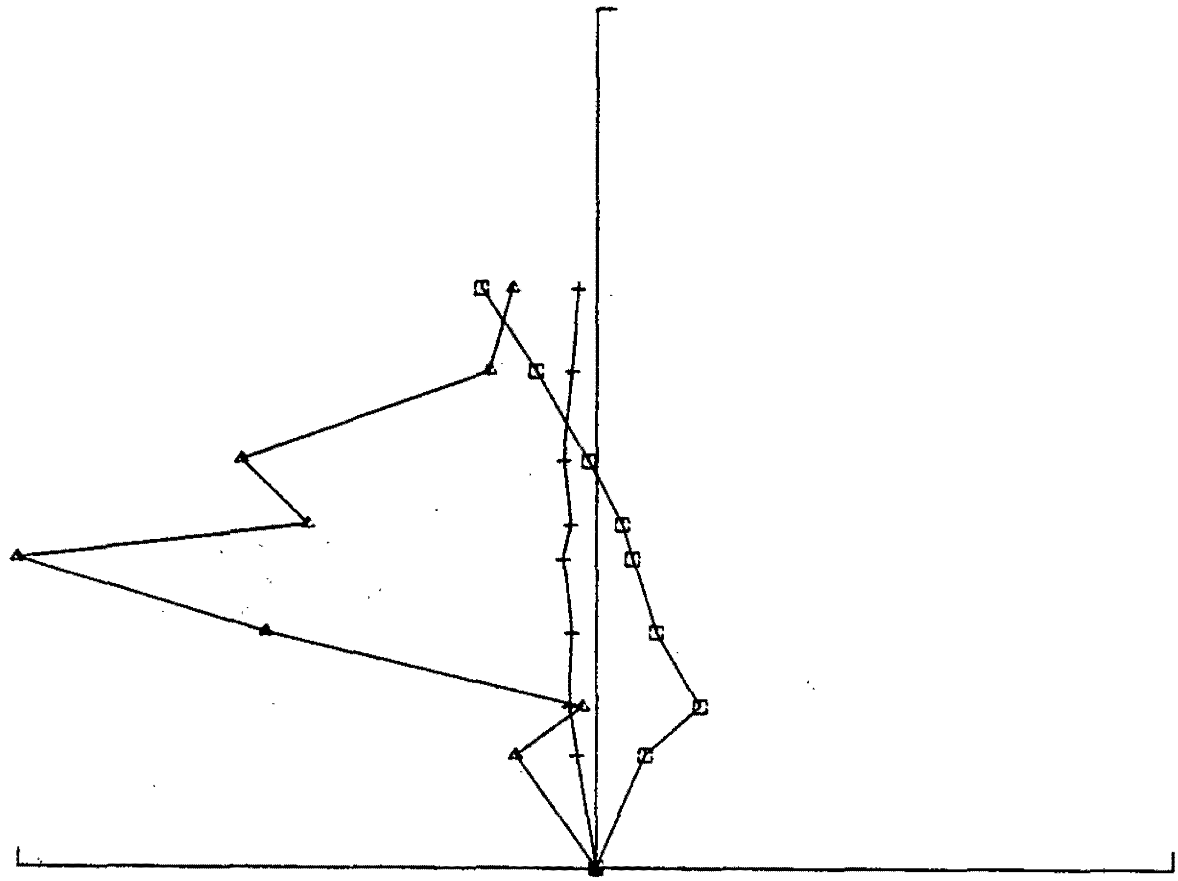
HOPE CREEK NUCLEAR GENERATING STATION

AUXILIARY BUILDING
DIESEL GENERATOR AREA –
MODE SHAPES, MODE 12

UPDATED FSAR

FIGURE 3.7-56



Mode Number 12
Frequency (CPS) = 20.10

Direction N-S = Triangle
Direction E-W = Square
Vertical = Cross

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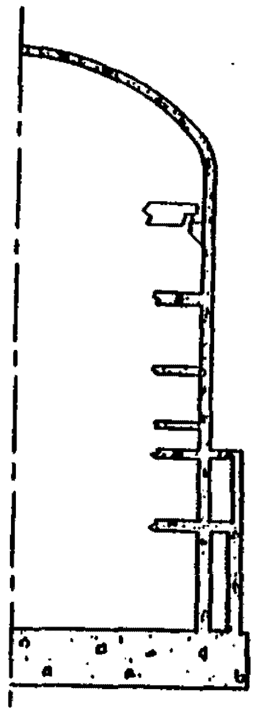
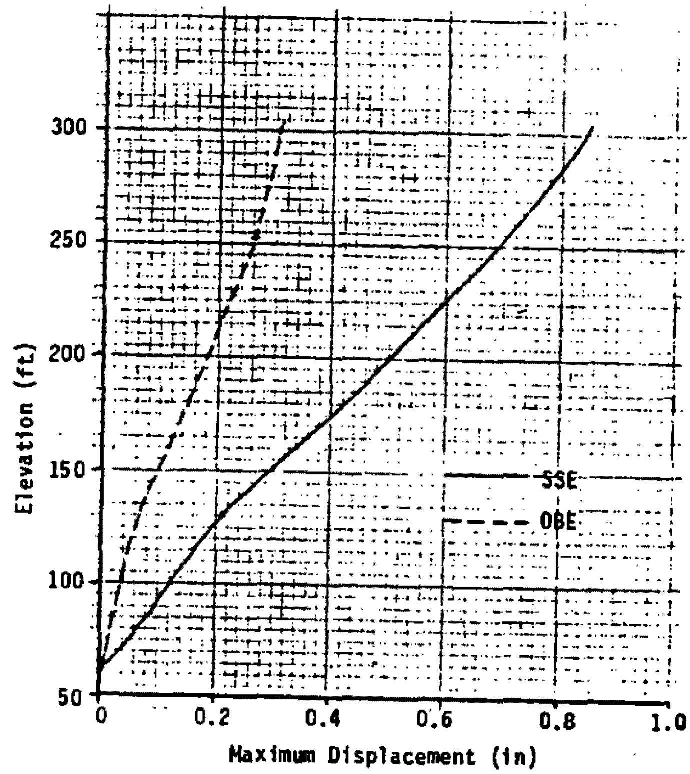
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

AUXILIARY BUILDING CONTROL AND RADWASTE AREAS – MODE SHAPES, MODE 12

UPDATED FSAR

FIGURE 3.7-57





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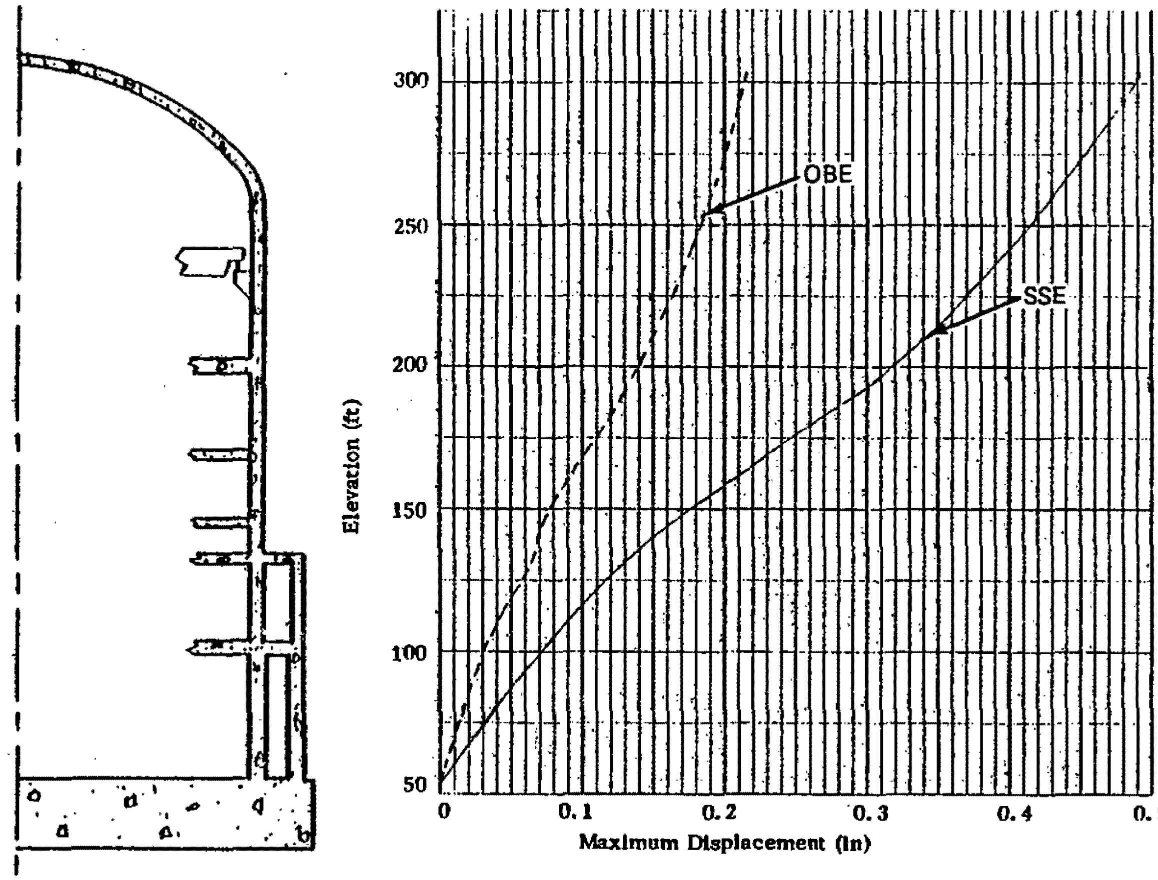
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

REACTOR BUILDING SHELL
N-S DISPLACEMENT

UPDATED FSAR

FIGURE 3.7-58



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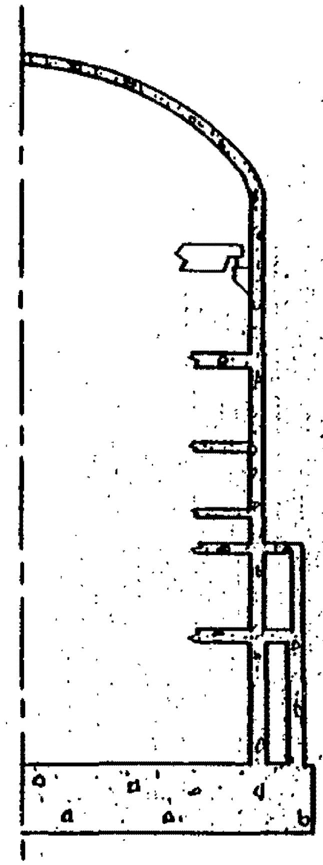
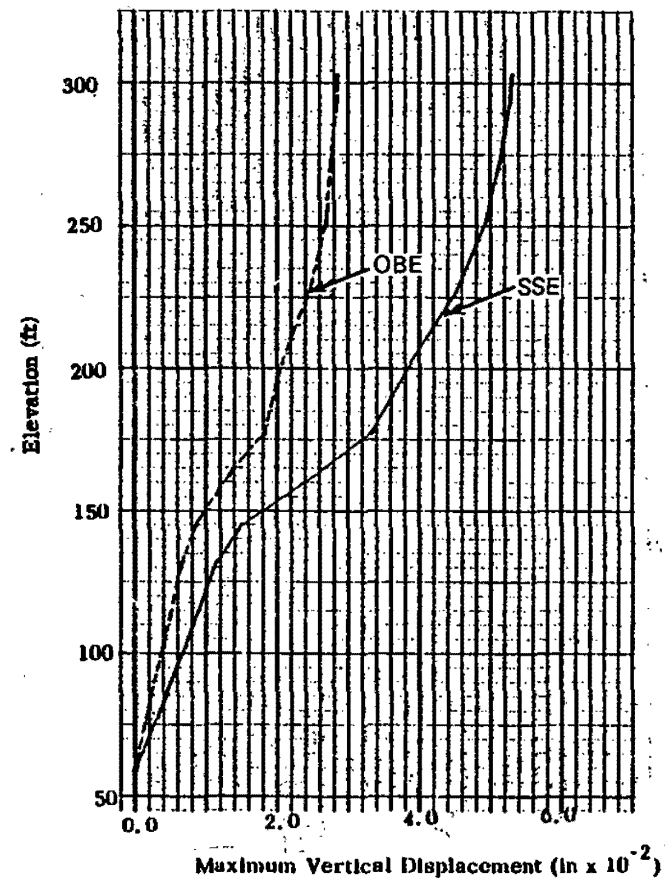
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

REACTOR BUILDING SHELL
E-W DISPLACEMENT

UPDATED FSAR

FIGURE 3.7-59





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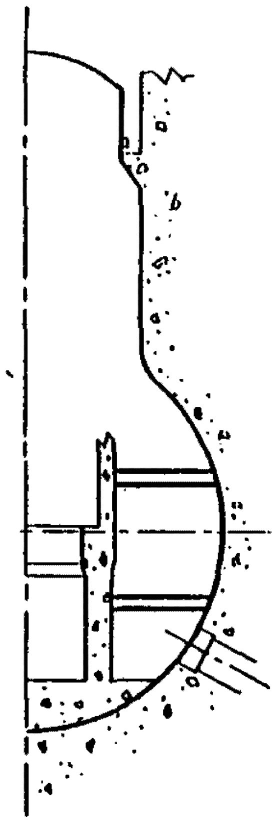
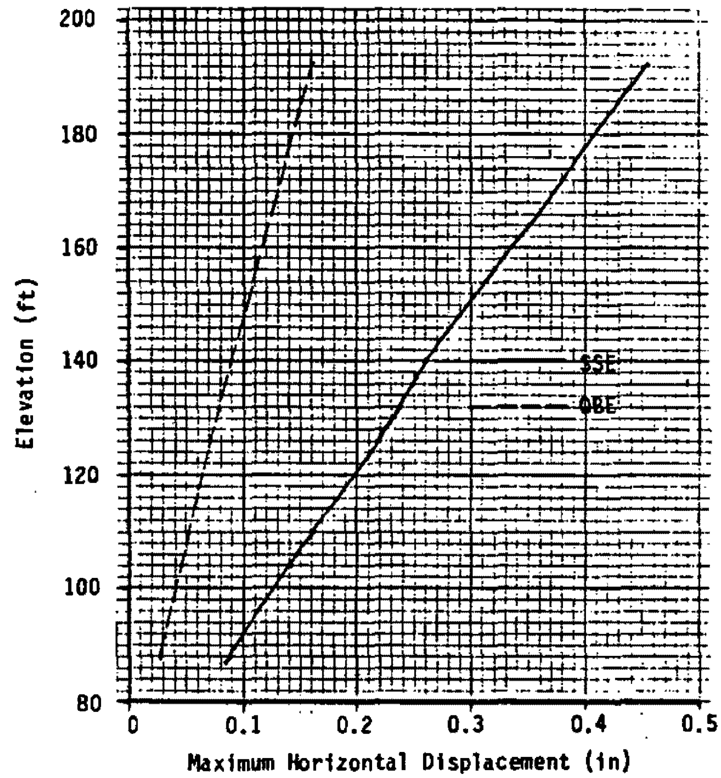
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

REACTOR BUILDING SHELL
VERTICAL DISPLACEMENT

UPDATED FSAR

FIGURE 3.7-60





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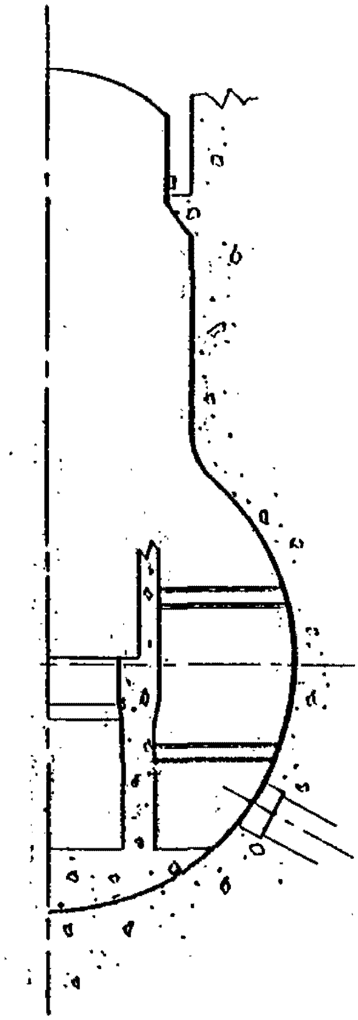
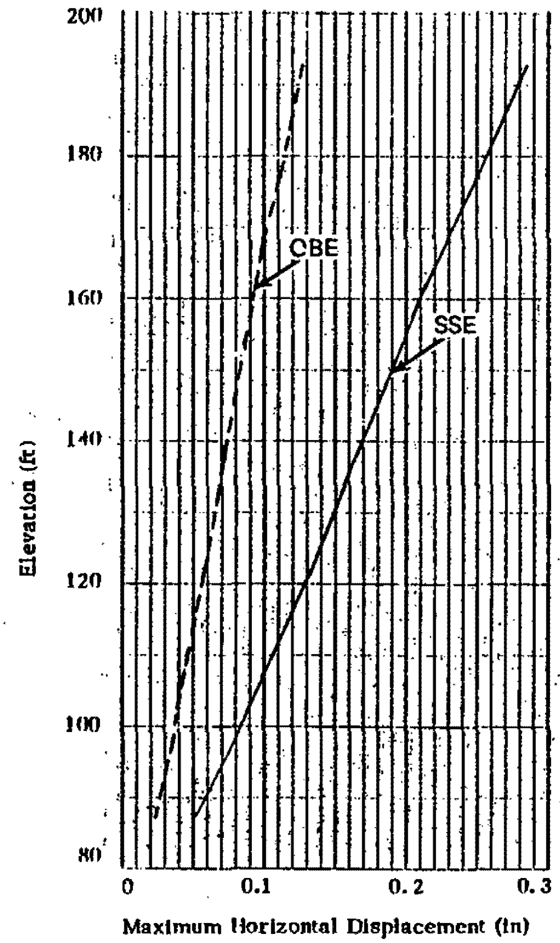
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

REACTOR BUILDING DRYWELL
N-S DISPLACEMENT

UPDATED FSAR

FIGURE 3.7-61





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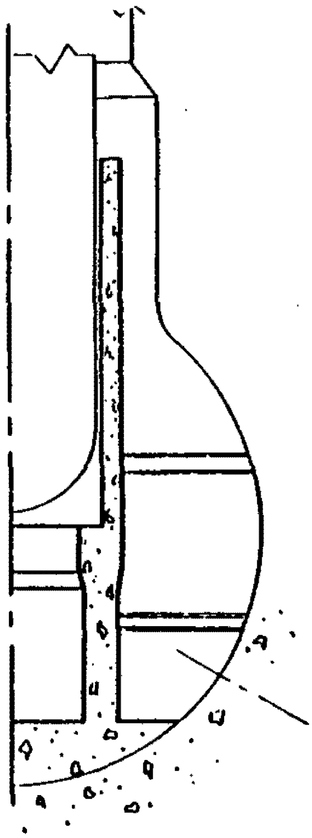
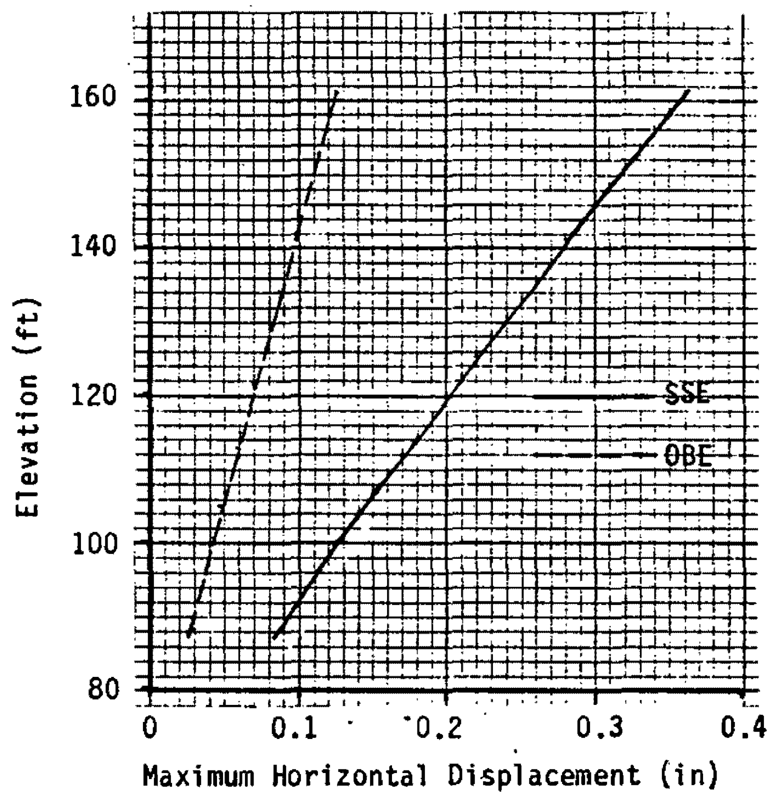
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

REACTOR BUILDING DRYWELL
E-W DISPLACEMENT

UPDATED FSAR

FIGURE 3.7-62





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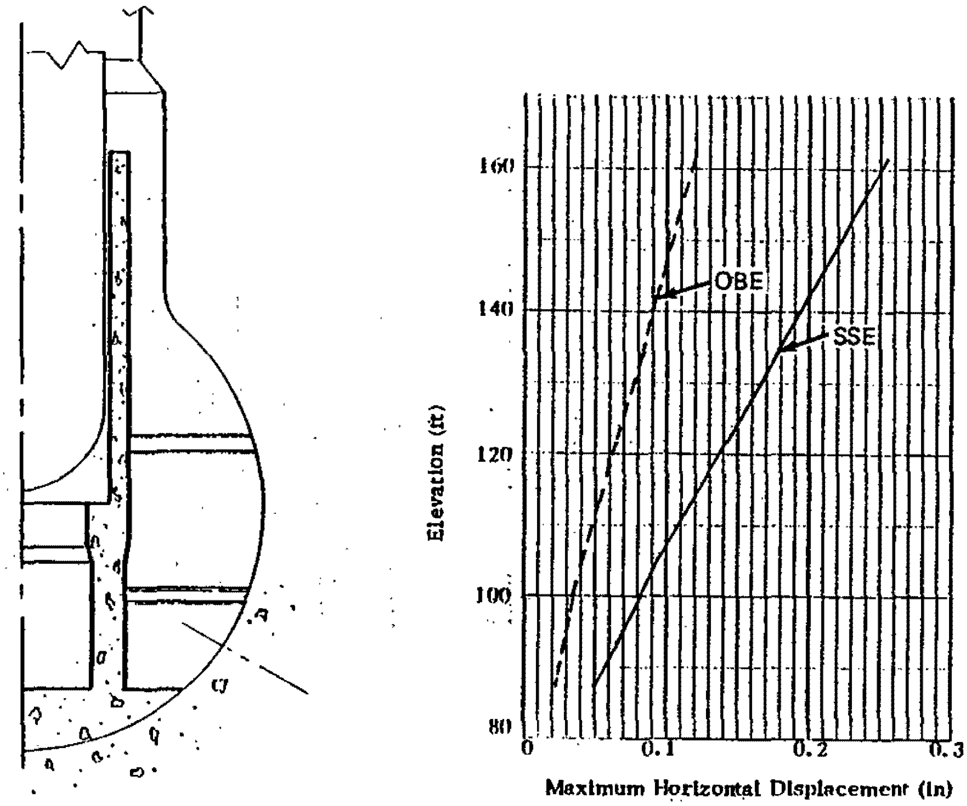
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

REACTOR BUILDING SHIELD WALL
AND RPV PEDESTAL
N-S DISPLACEMENT

UPDATED FSAR

FIGURE 3.7-63



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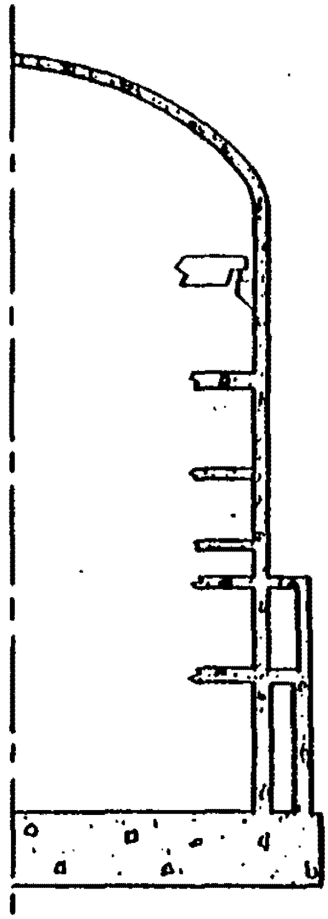
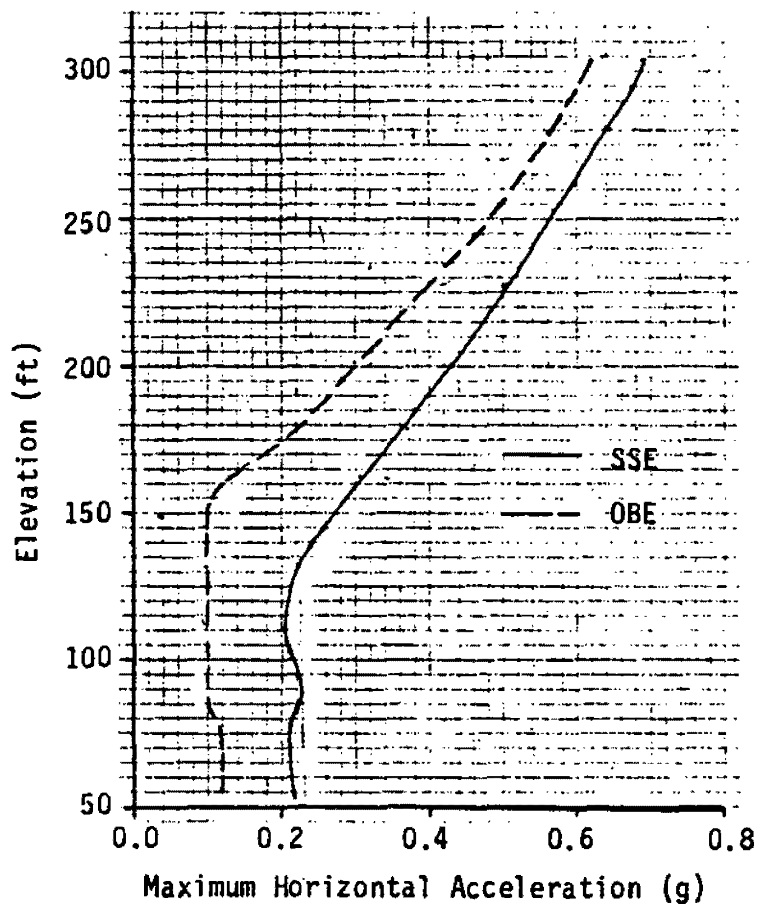
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

REACTOR BUILDING SHIELD WALL
AND RPV PEDESTAL
E-W DISPLACEMENT

UPDATED FSAR

FIGURE 3.7-64





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PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

REACTOR BUILDING SHELL
N-S ACCELERATION

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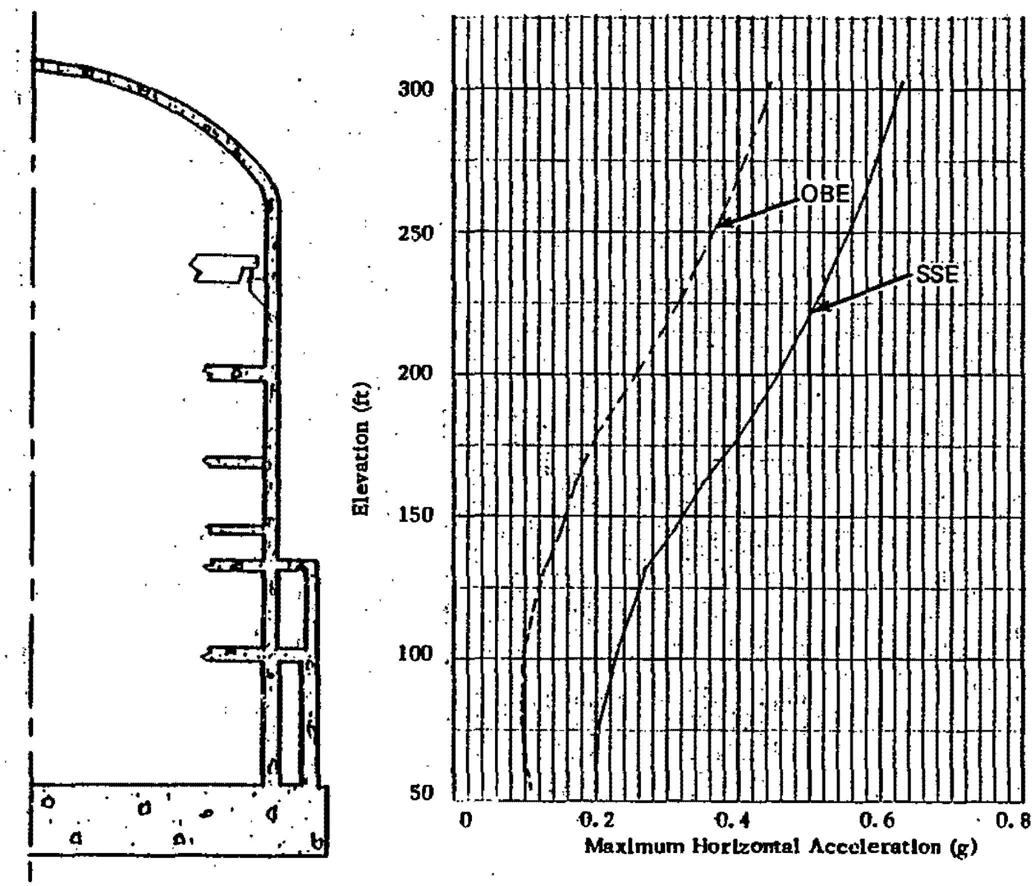
FIGURE 3.7-65



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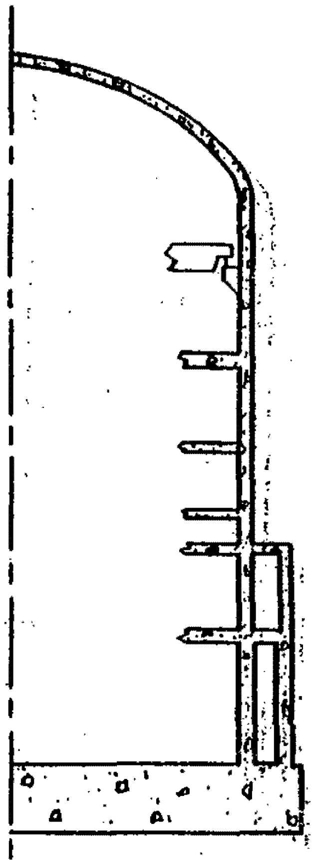
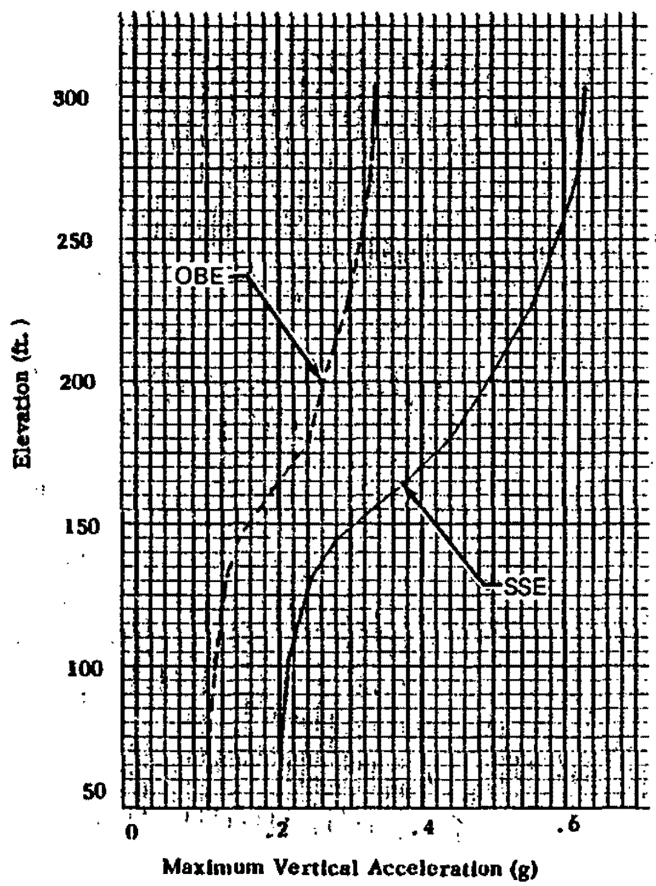
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

REACTOR BUILDING SHELL
E-W ACCELERATION

UPDATED FSAR

FIGURE 3.7-66





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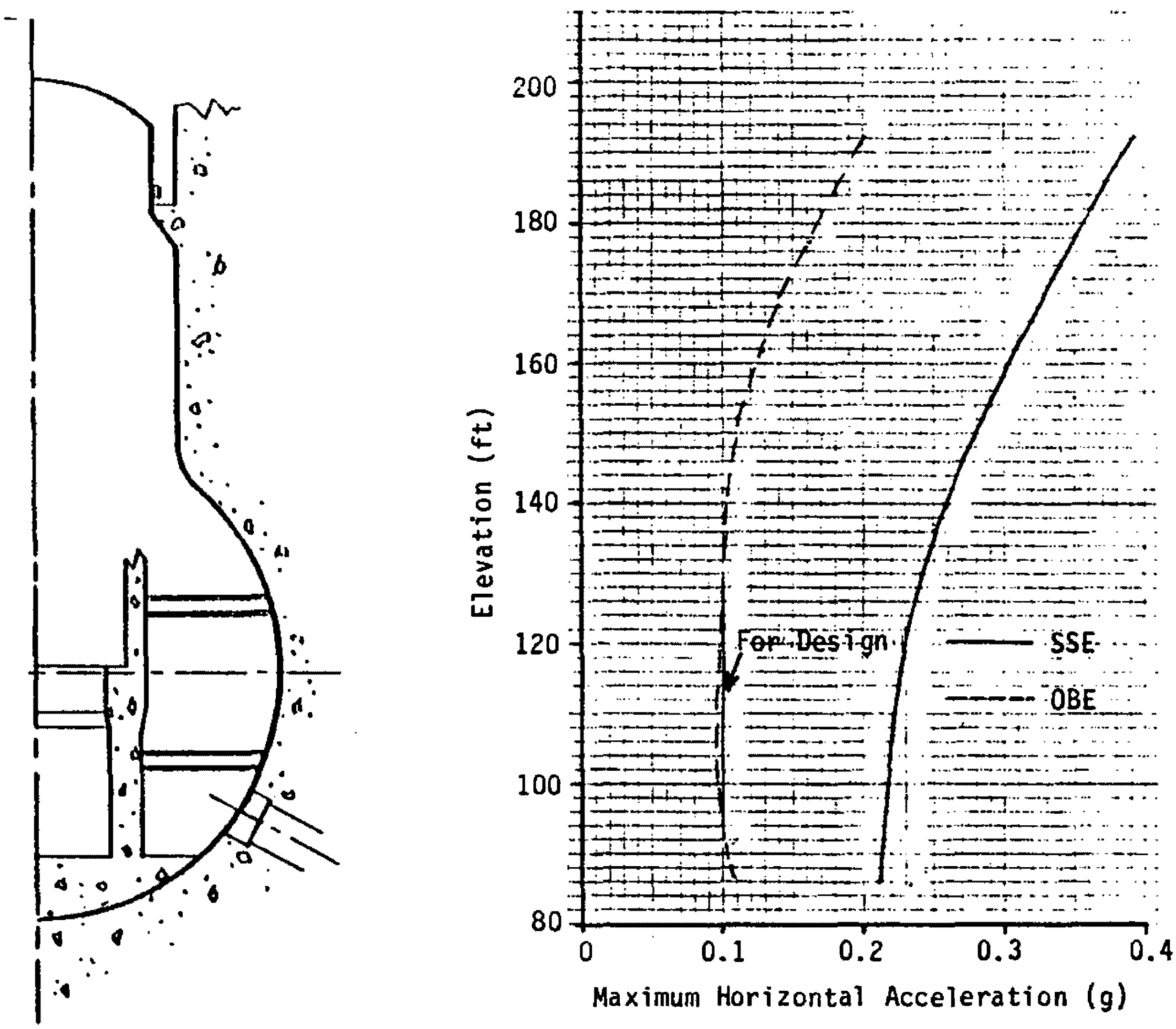
HOPE CREEK NUCLEAR GENERATING STATION

REACTOR BUILDING SHELL
VERTICAL ACCELERATION

UPDATED FSAR

FIGURE 3.7-67



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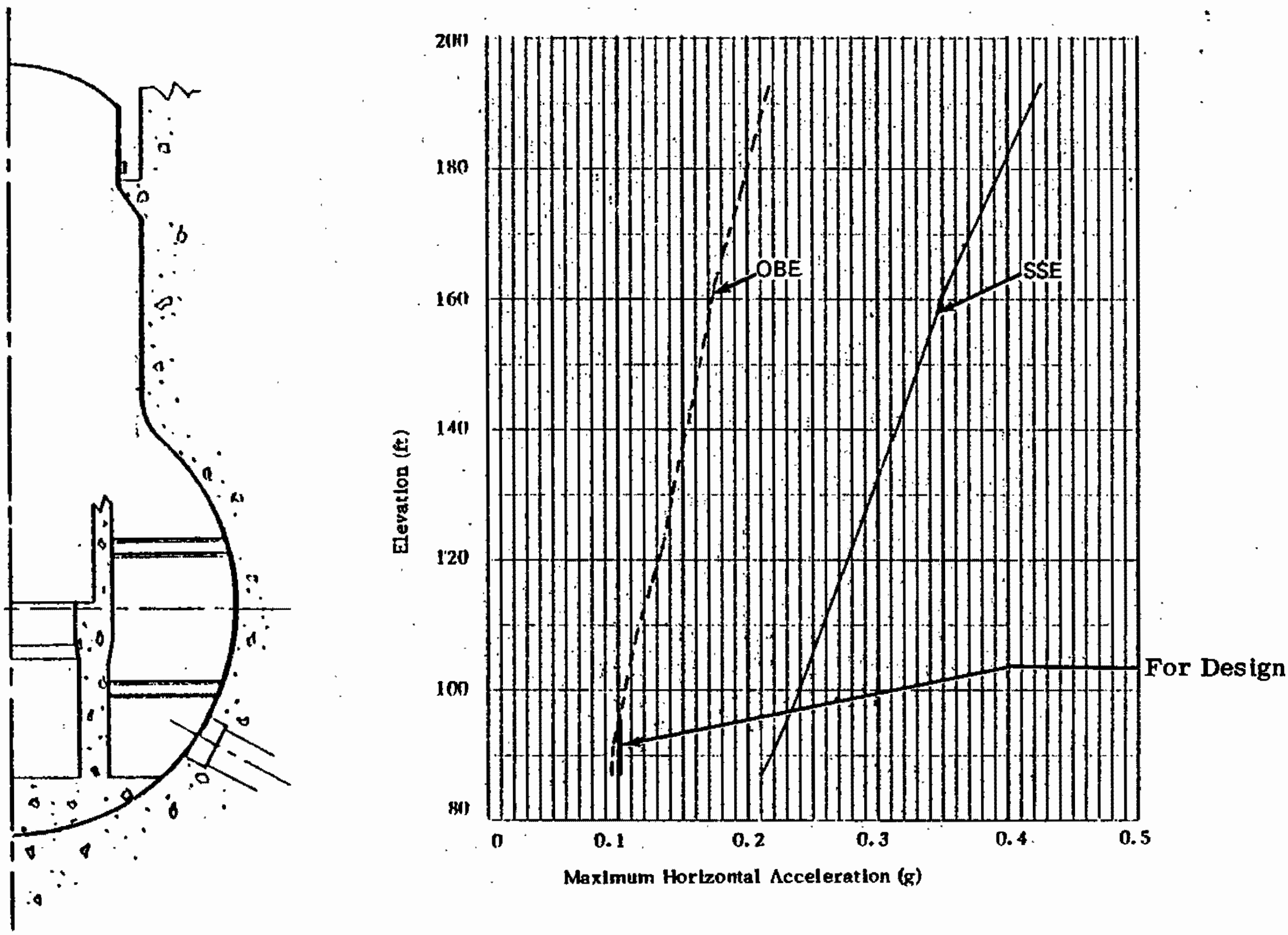
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

REACTOR BUILDING DRYWELL
N-S ACCELERATION

UPDATED FSAR

FIGURE 3.7-68



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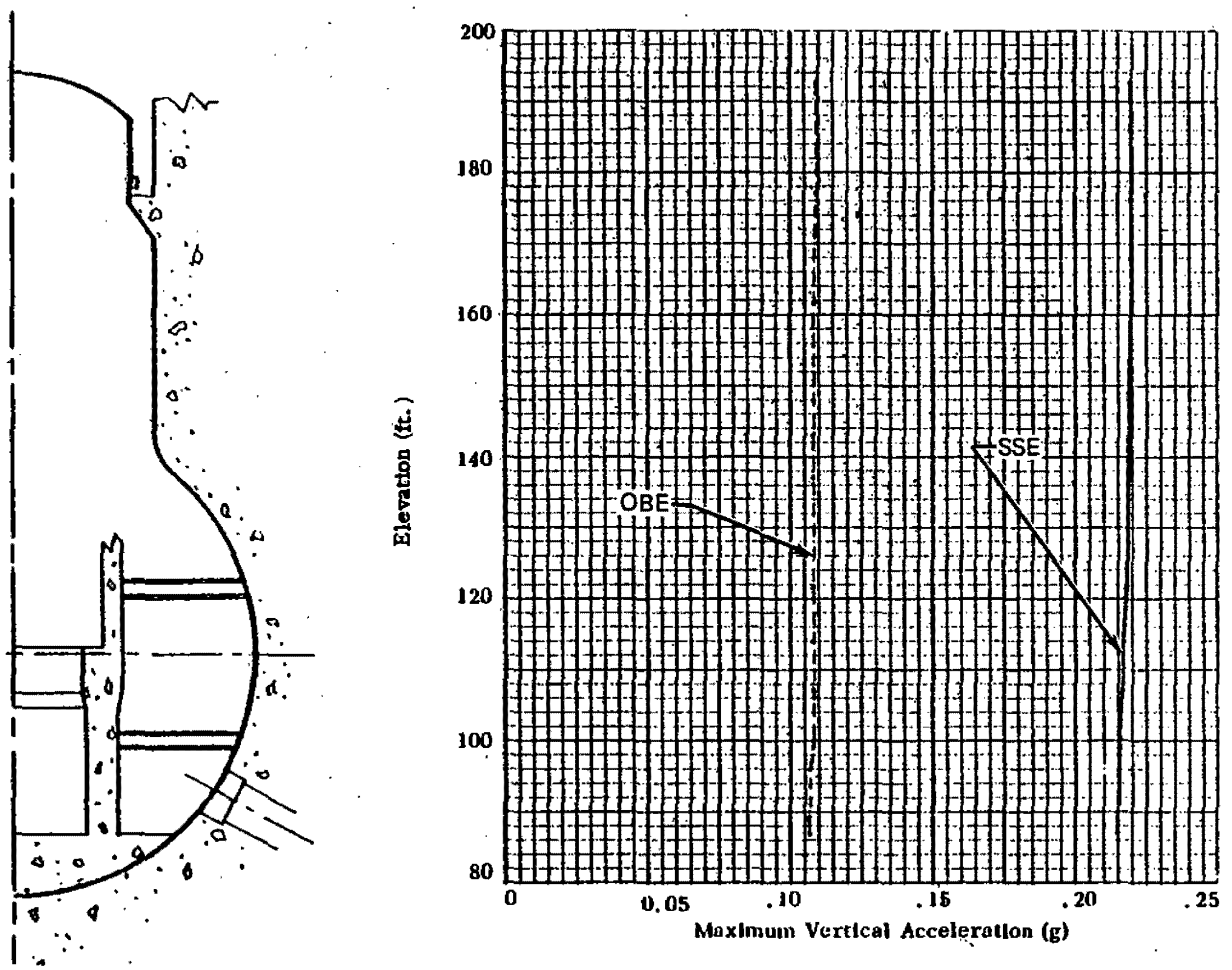
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

REACTOR BUILDING DRYWELL
E-W ACCELERATION

UPDATED FSAR

FIGURE 3.7-69



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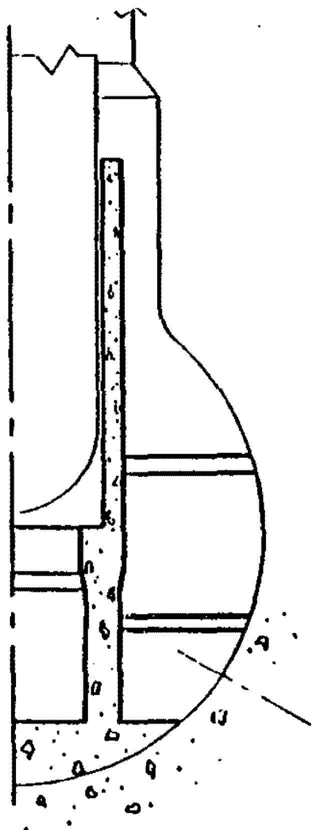
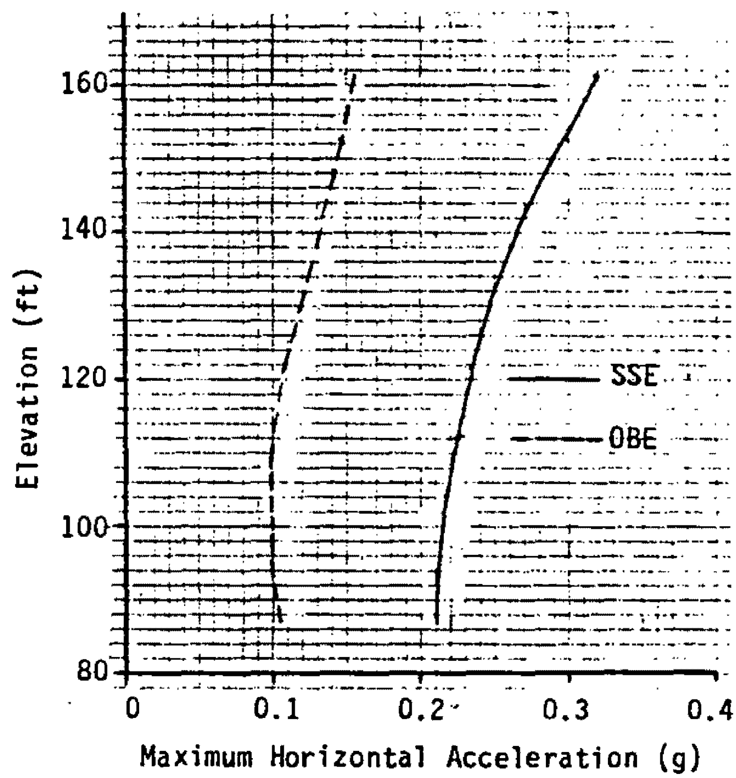
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

REACTOR BUILDING DRYWELL
VERTICAL ACCELERATION

UPDATED FSAR

FIGURE 3.7-70





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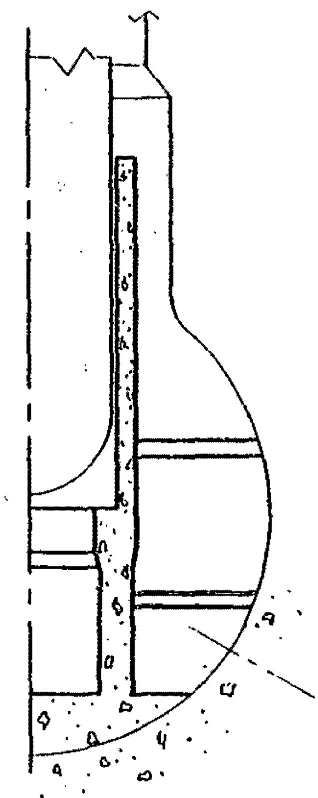
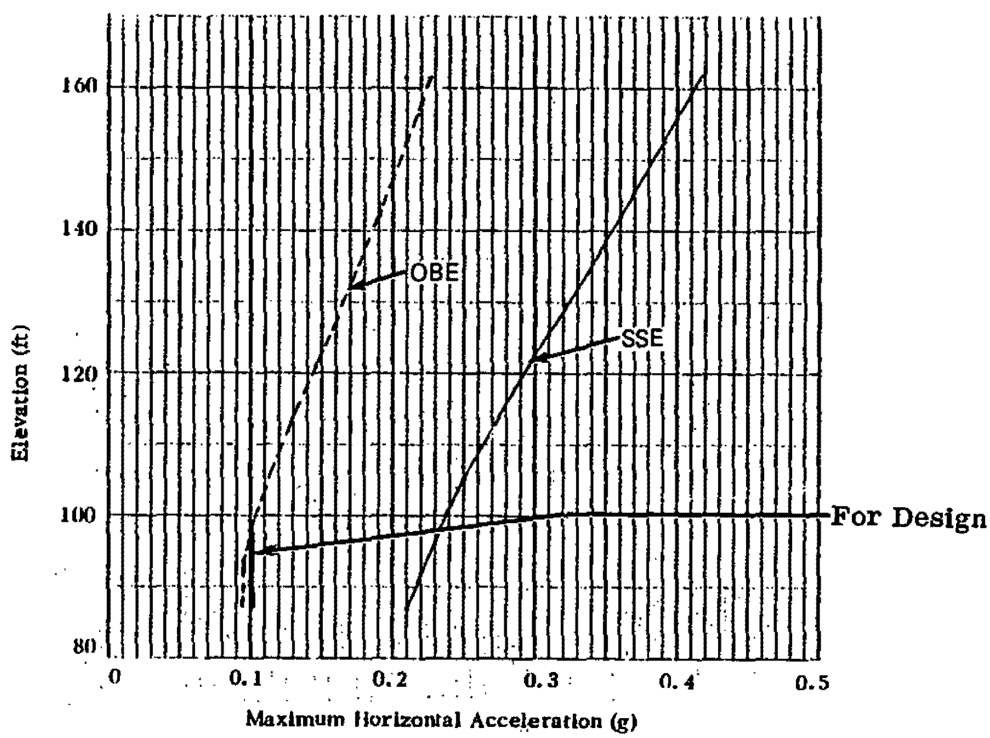
REACTOR BUILDING
SHIELD WALL AND RPV PEDESTAL
N-S ACCELERATION

UPDATED FSAR

FIGURE 3.7-71





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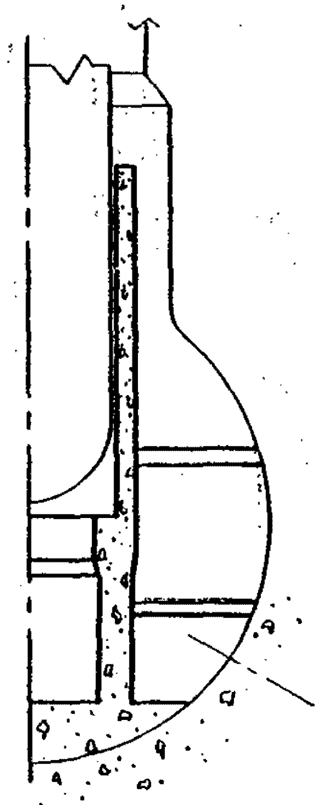
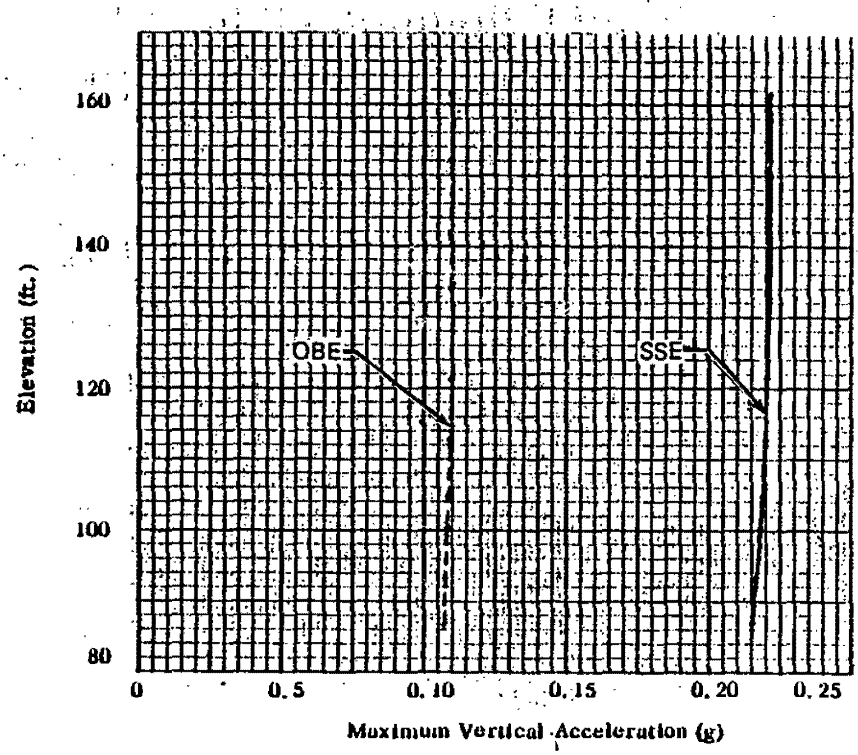
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

REACTOR BUILDING
SHIELD WALL AND RPV PEDESTAL
E-W ACCELERATION

UPDATED FSAR

FIGURE 3.7-72





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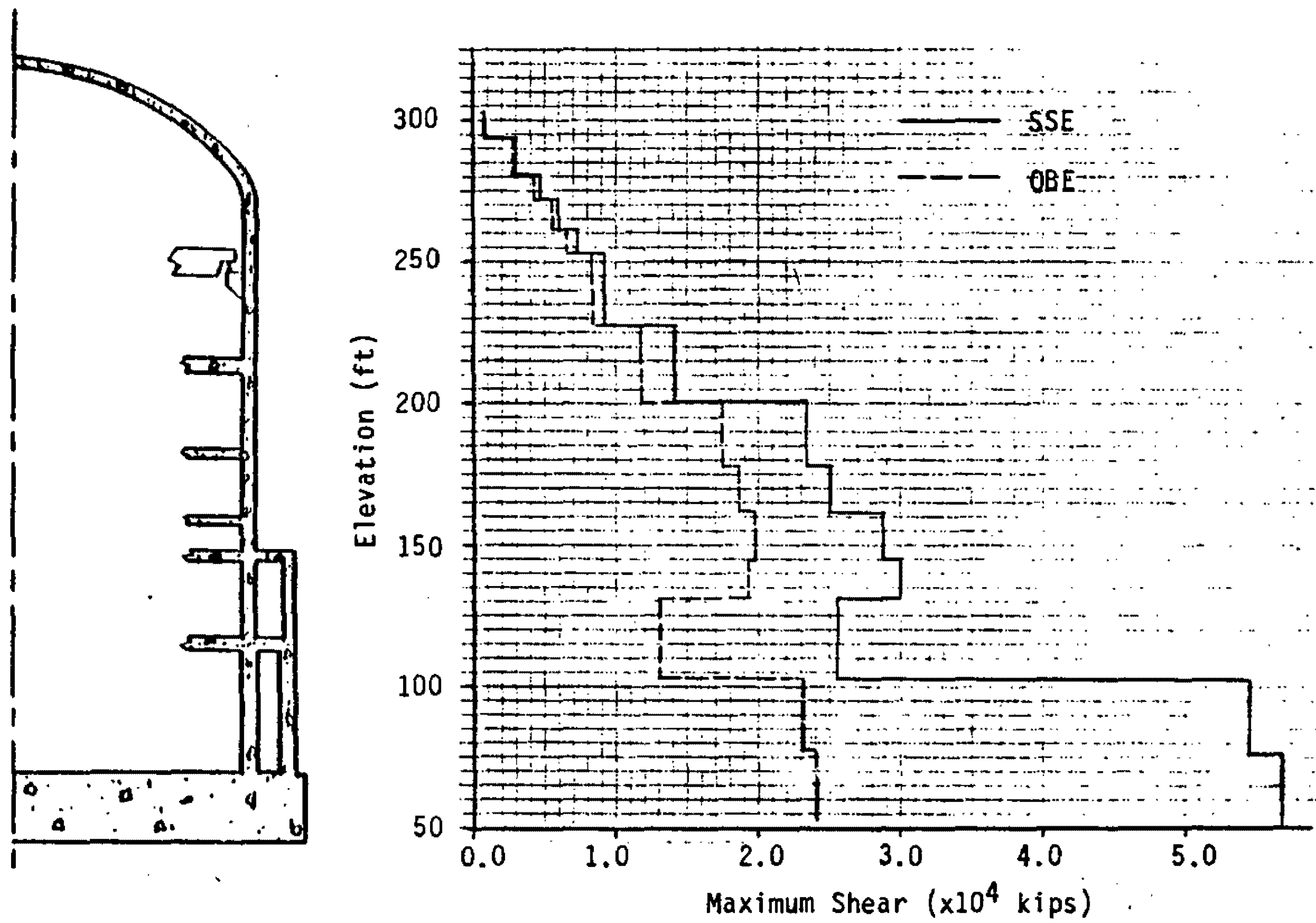
REACTOR BUILDING
SHIELD WALL RPV PEDESTAL
VERTICAL ACCELERATION

UPDATED FSAR

FIGURE 3.7-73



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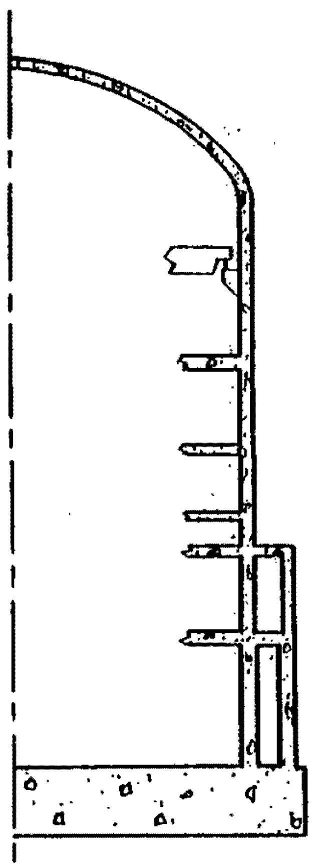
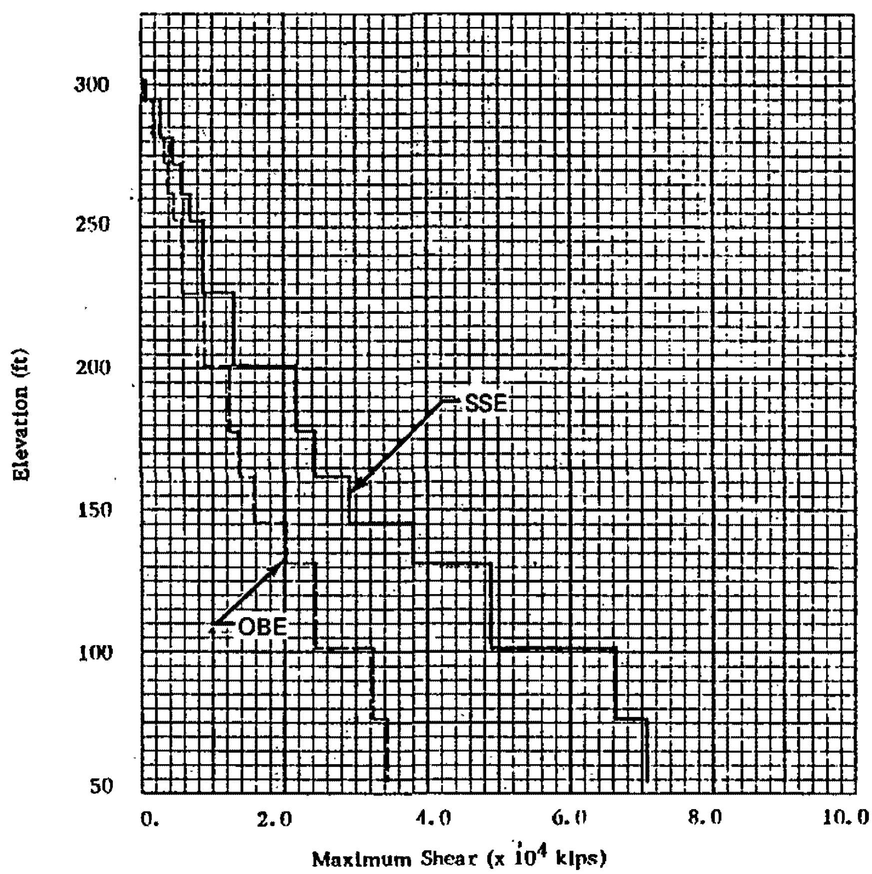
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

REACTOR BUILDING SHELL
N-S SHEAR FORCES

UPDATED FSAR

FIGURE 3.7-74





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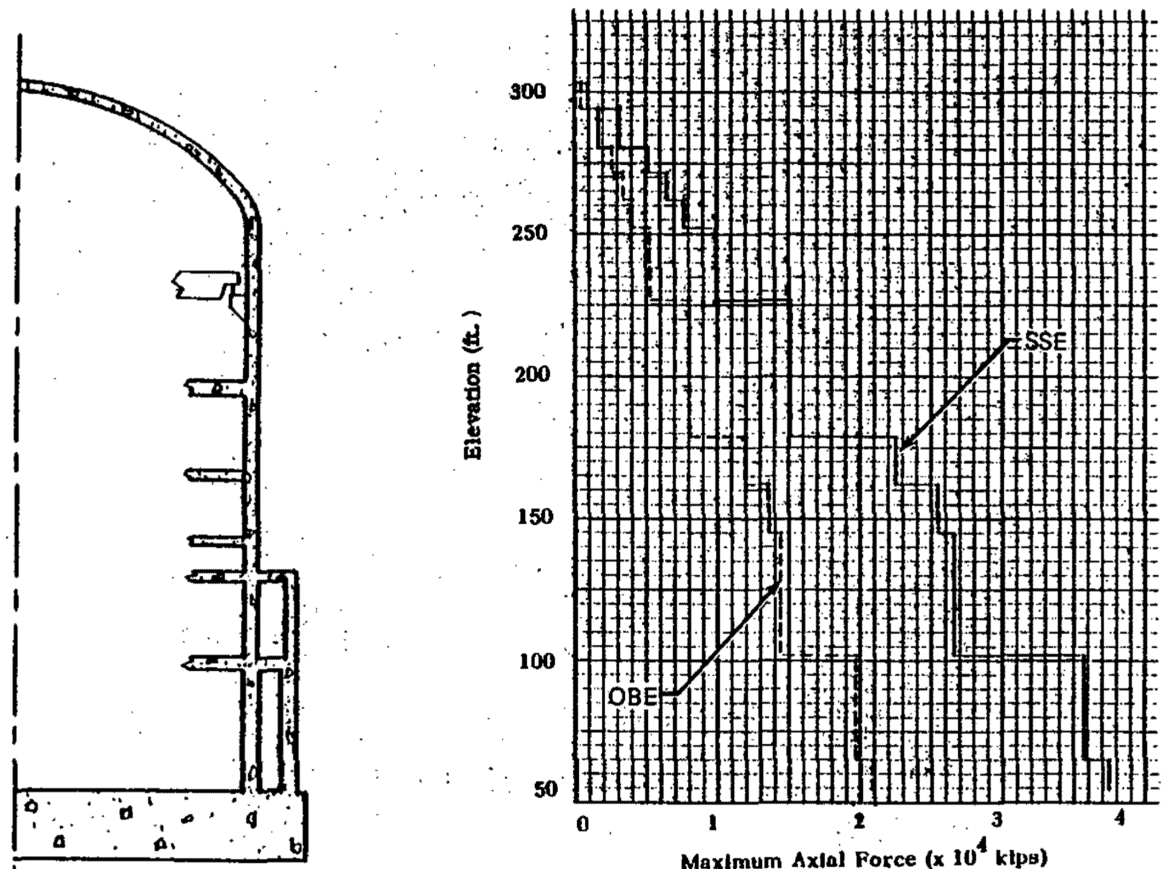
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

REACTOR BUILDING SHELL
E-W SHEAR FORCES

UPDATED FSAR

FIGURE 3.7-75



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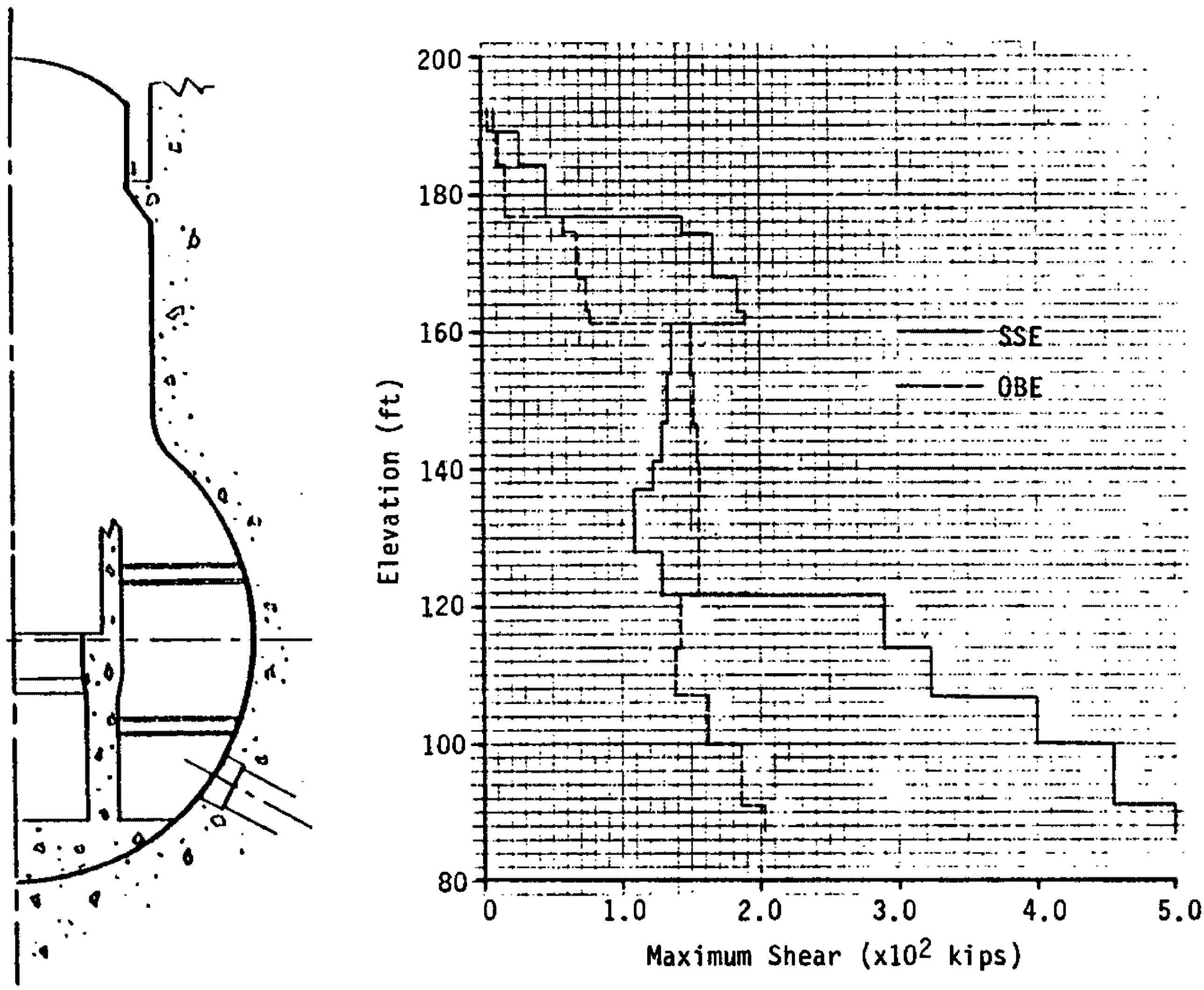
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

REACTOR BUILDING SHELL
VERTICAL AXIAL FORCES

UPDATED FSAR

FIGURE 3.7-76



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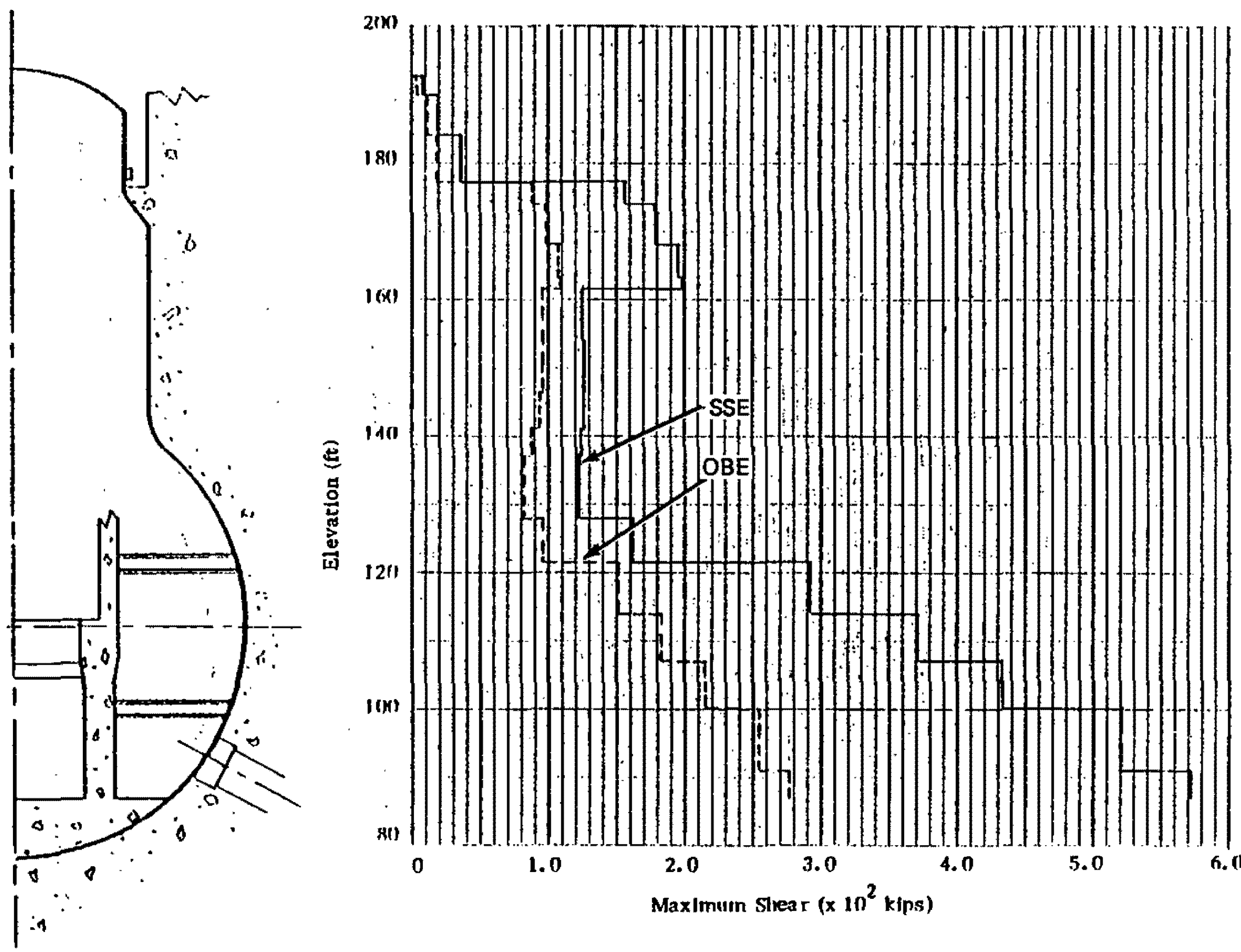
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

REACTOR BUILDING DRYWELL
N-S SHEAR FORCES

UPDATED FSAR

FIGURE 3.7-77



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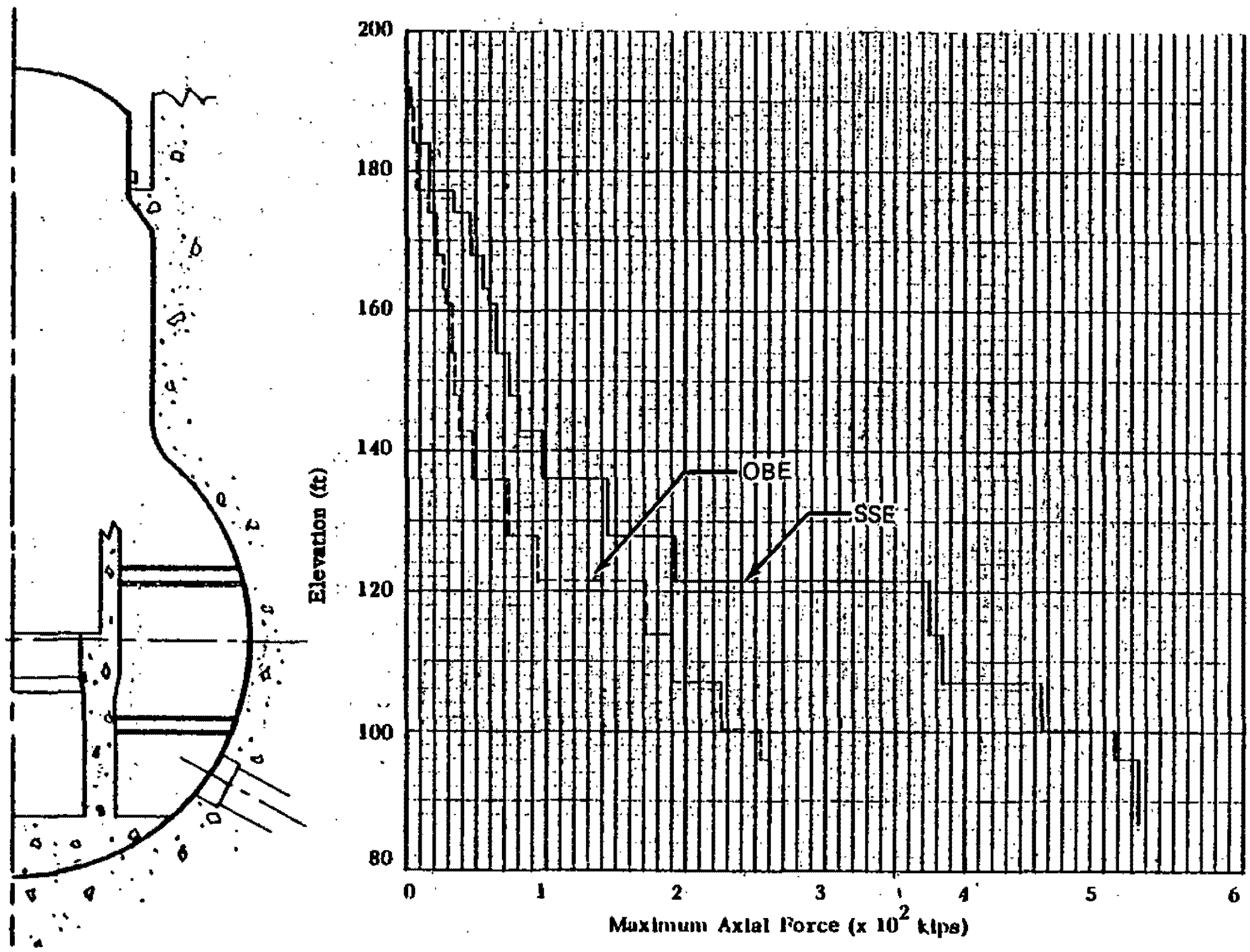
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

REACTOR BUILDING DRYWELL
E-W SHEAR FORCES

UPDATED FSAR

FIGURE 3.7-78



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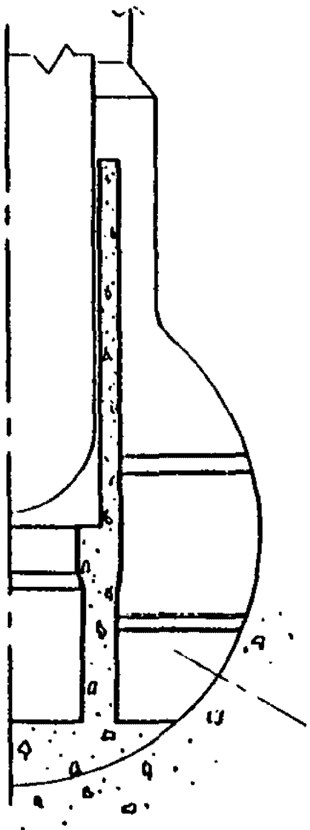
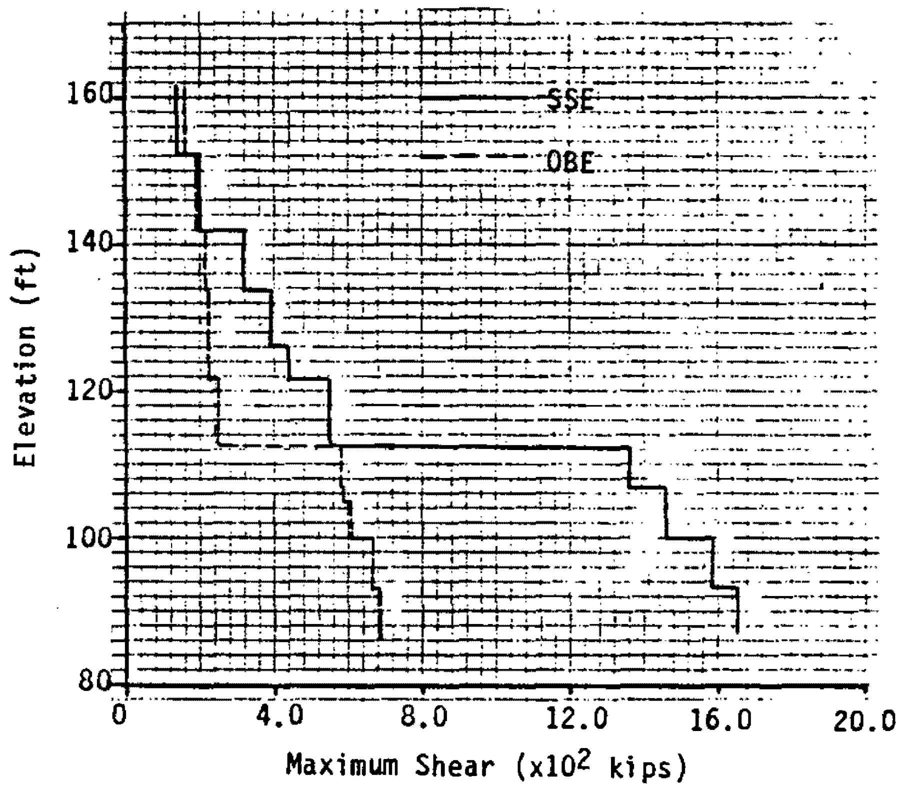
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

REACTOR BUILDING DRYWELL
VERTICAL AXIAL FORCES

UPDATED FSAR

FIGURE 3.7-79





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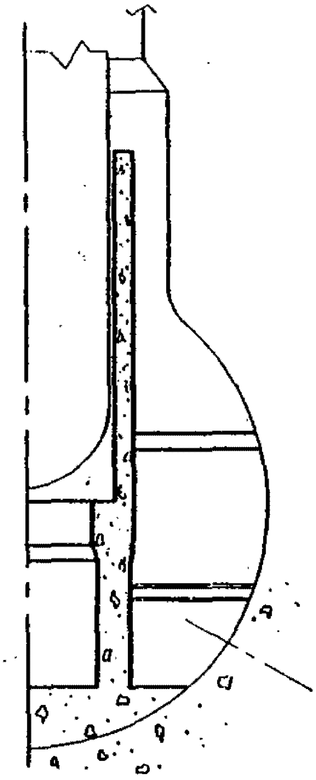
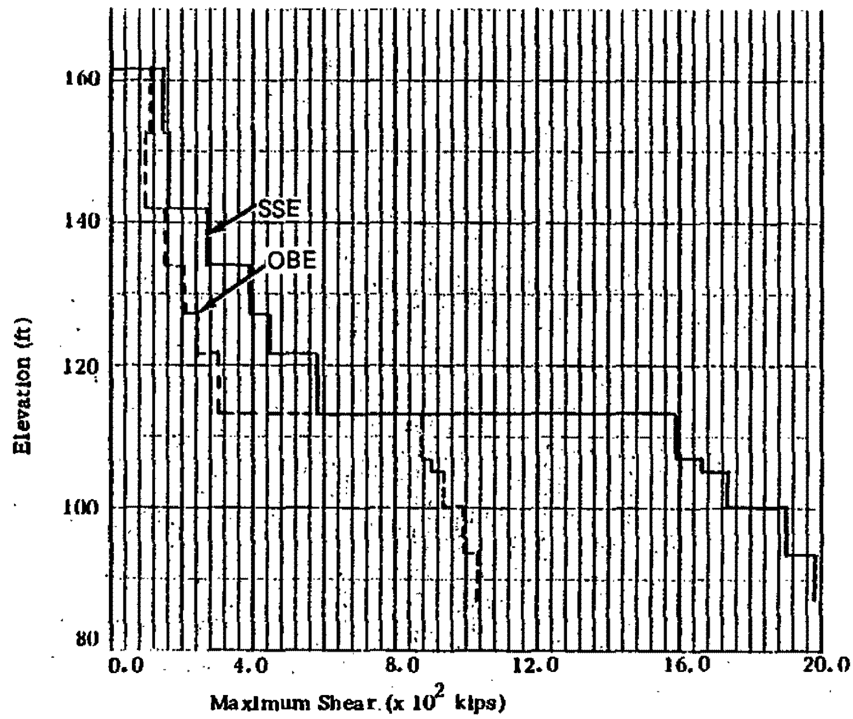
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

REACTOR BUILDING
SHIELD WALL AND RPV PEDESTAL
N-S SHEAR FORCES

UPDATED FSAR

FIGURE 3.7-80





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PUBLIC SERVICE ELECTRIC AND GAS COMPANY
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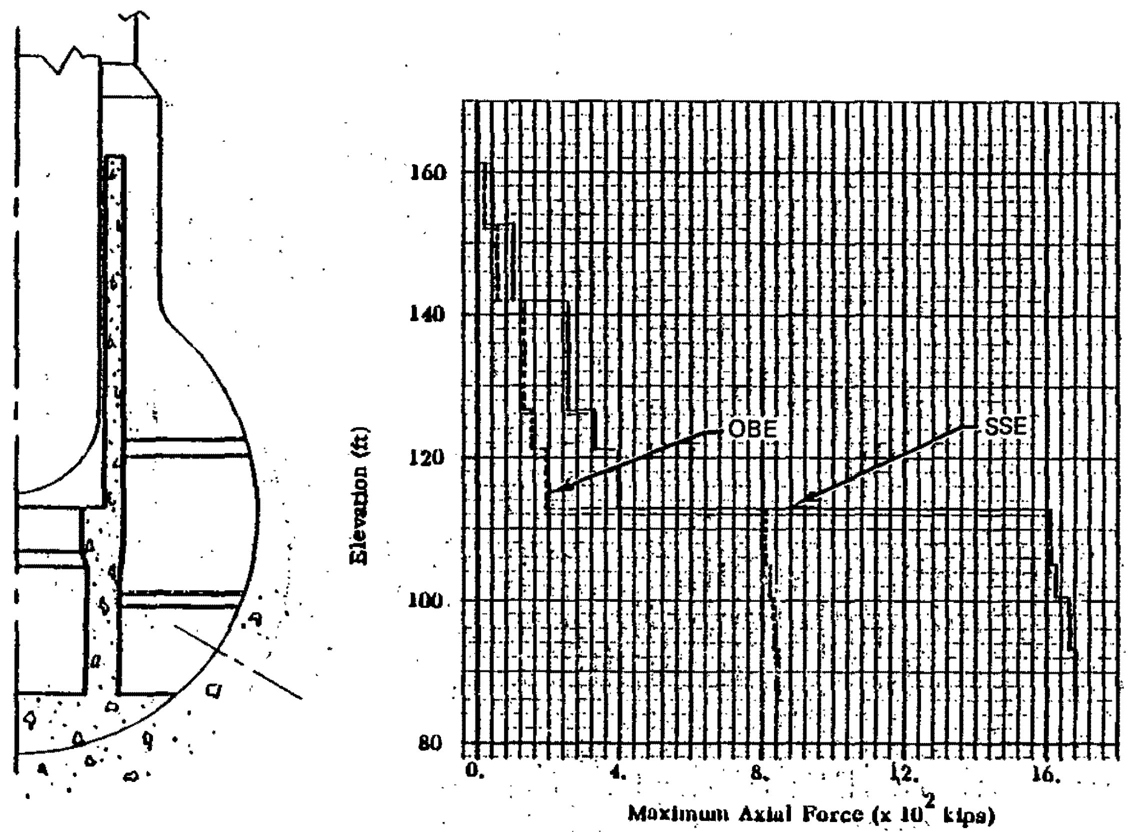
REACTOR BUILDING
SHIELD WALL AND RPV PEDESTAL
E-W SHEAR FORCES

UPDATED FSAR

FIGURE 3.7-81



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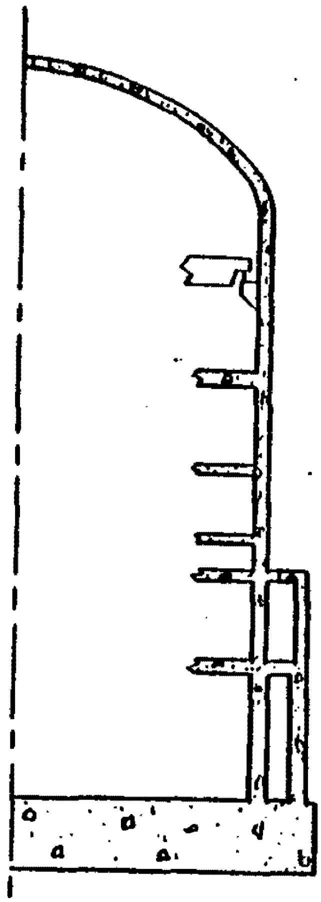
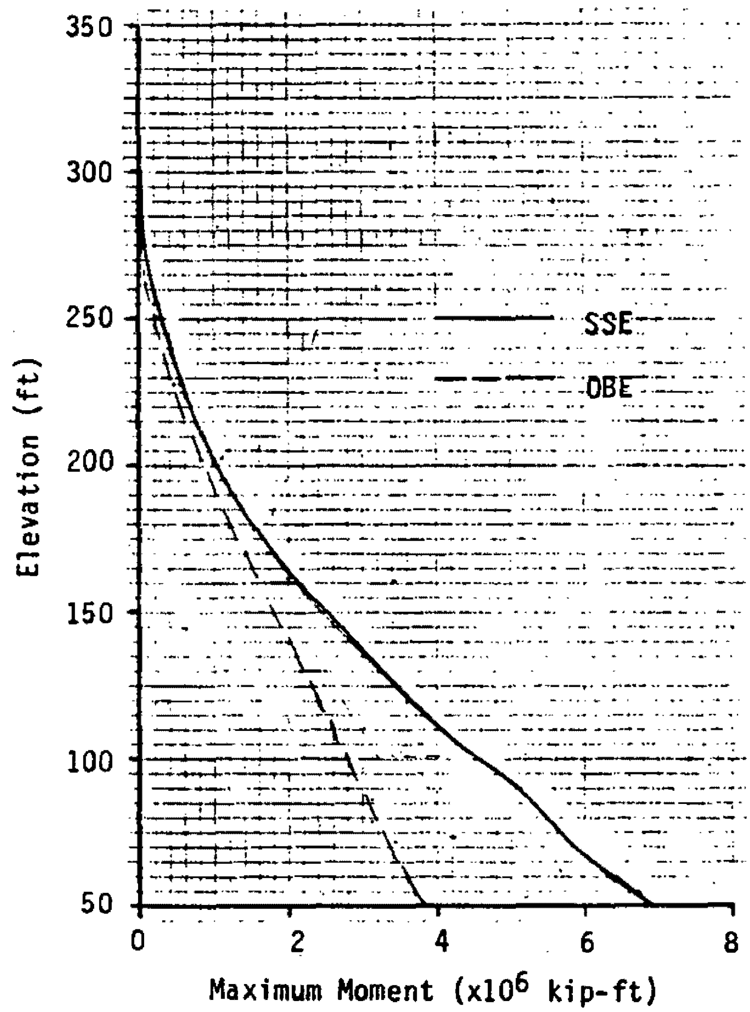
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

REACTOR BUILDING
SHIELD WALL AND RPV PEDESTAL
VERTICAL AXIAL FORCES

UPDATED FSAR

FIGURE 3.7-82





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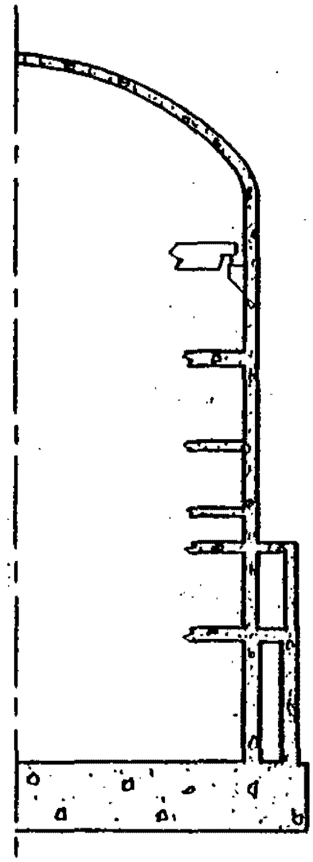
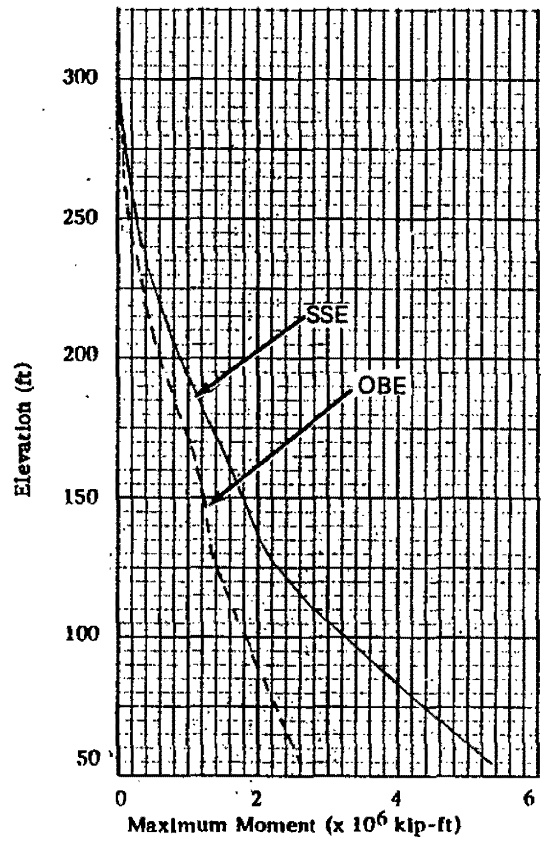
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

REACTOR BUILDING SHELL
N-S MOMENT

UPDATED FSAR

FIGURE 3.7-83





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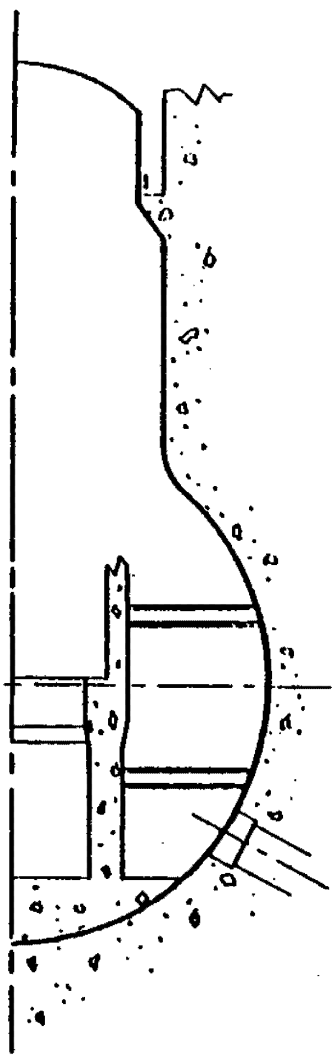
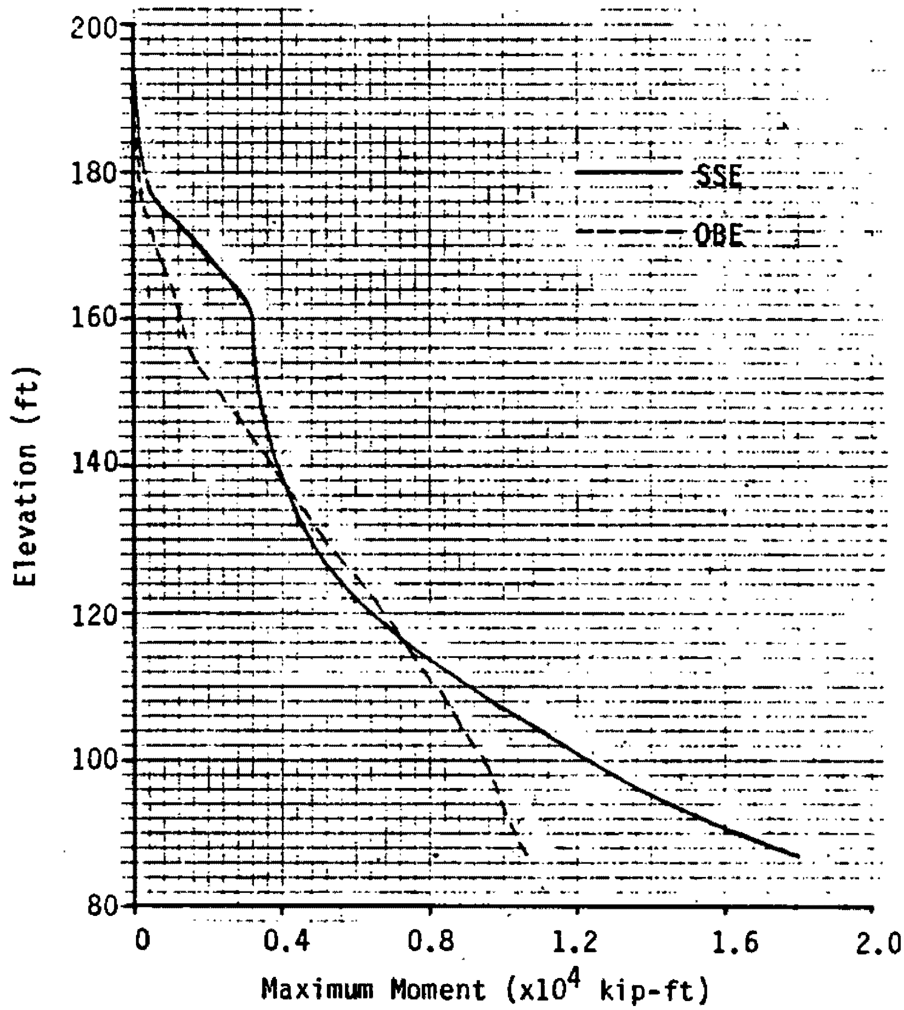
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

REACTOR BUILDING SHELL
E-W MOMENT

UPDATED FSAR

FIGURE 3.7-84





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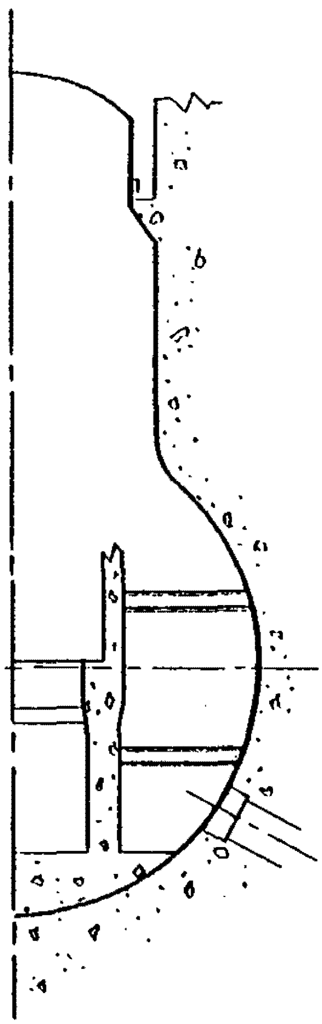
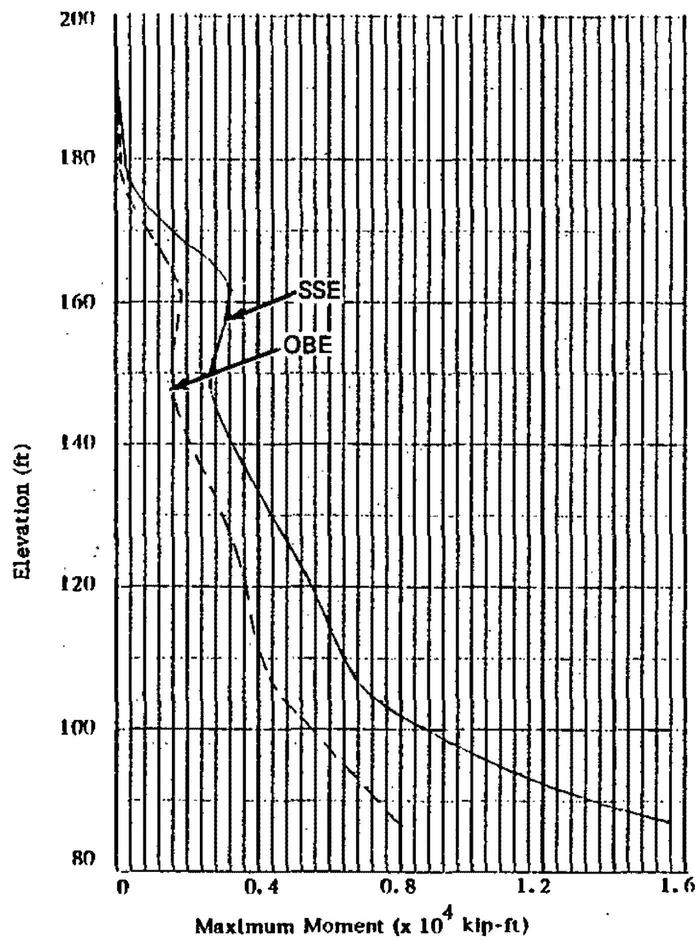
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

REACTOR BUILDING DRYWELL
N-S MOMENT

UPDATED FSAR

FIGURE 3.7-85





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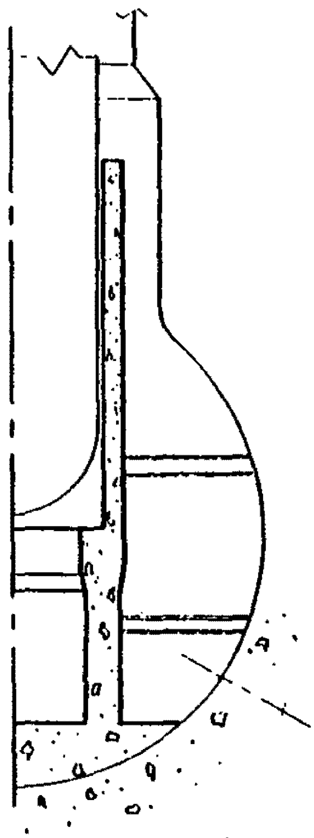
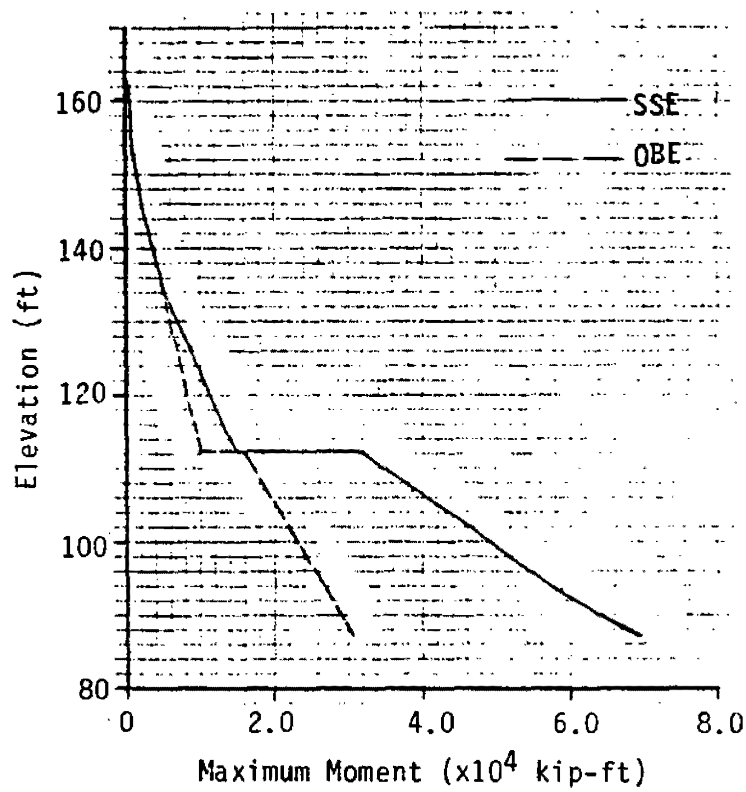
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

REACTOR BUILDING DRYWELL
E-W MOMENT

UPDATED FSAR

FIGURE 3.7-86





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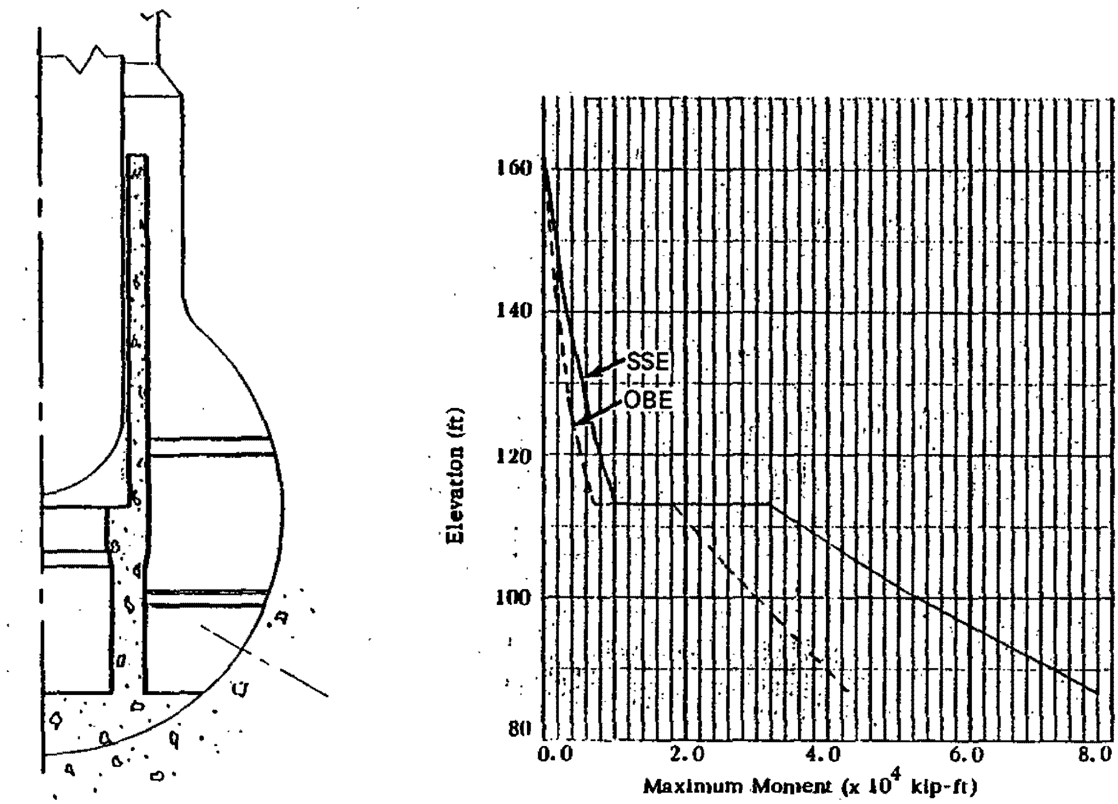
REACTOR BUILDING
SHIELD WALL AND RPV PEDESTAL
N-S MOMENT

UPDATED FSAR

FIGURE 3.7-87



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PUBLIC SERVICE ELECTRIC AND GAS COMPANY
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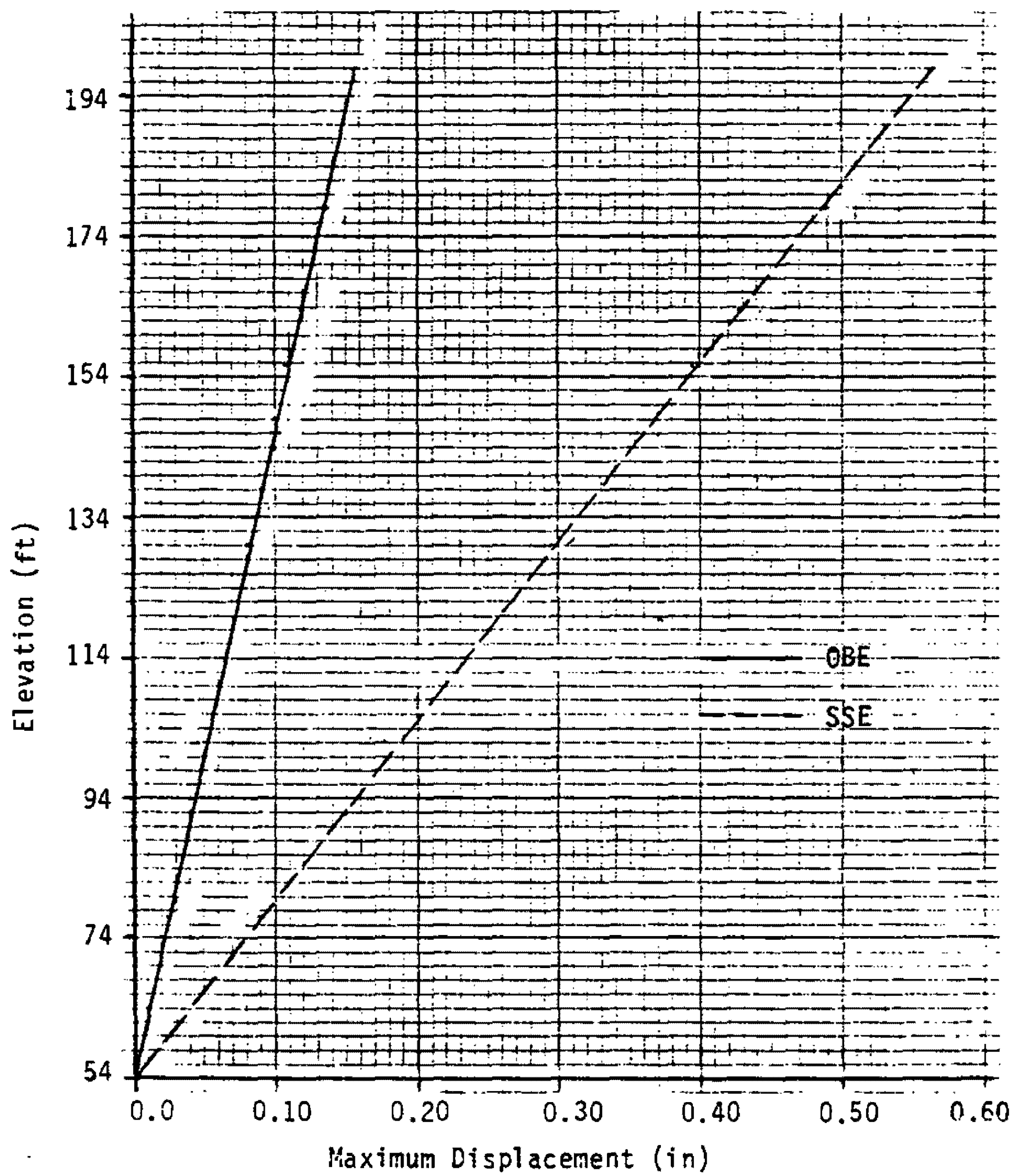
REACTOR BUILDING
SHIELD WALL AND RPV PEDESTAL
E-W MOMENT

UPDATED FSAR

FIGURE 3.7-88



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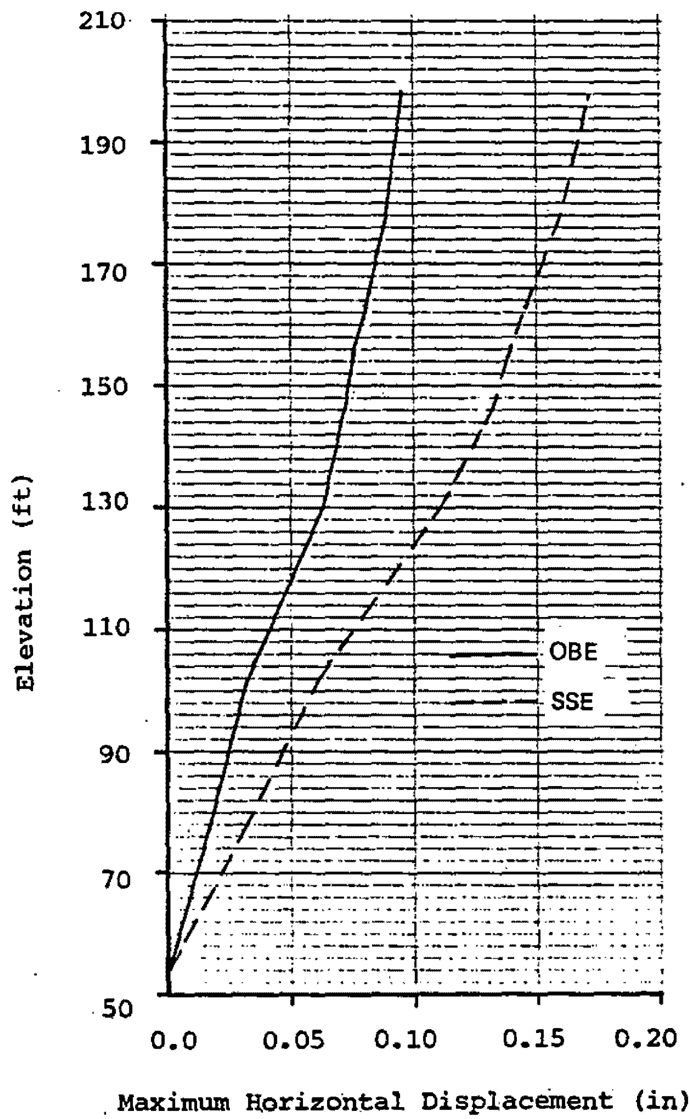
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

AUXILIARY BUILDING
DIESEL GENERATOR AREA
N-S DISPLACEMENT

UPDATED FSAR

FIGURE 3.7-89



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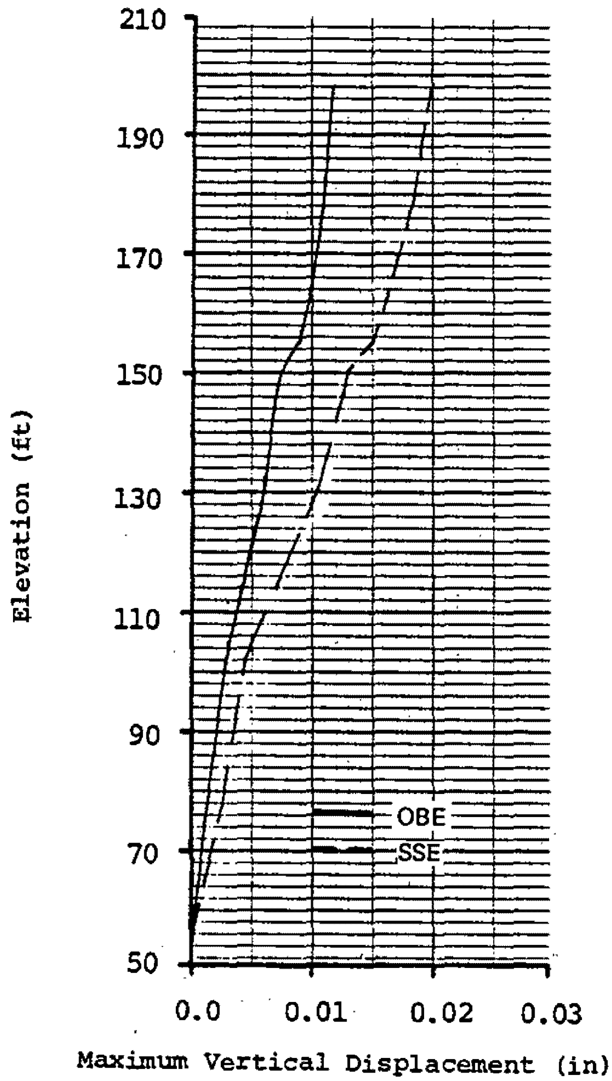
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

AUXILIARY BUILDING
DIESEL GENERATOR AREA
E-W DISPLACEMENT

UPDATED FSAR

FIGURE 3.7-90



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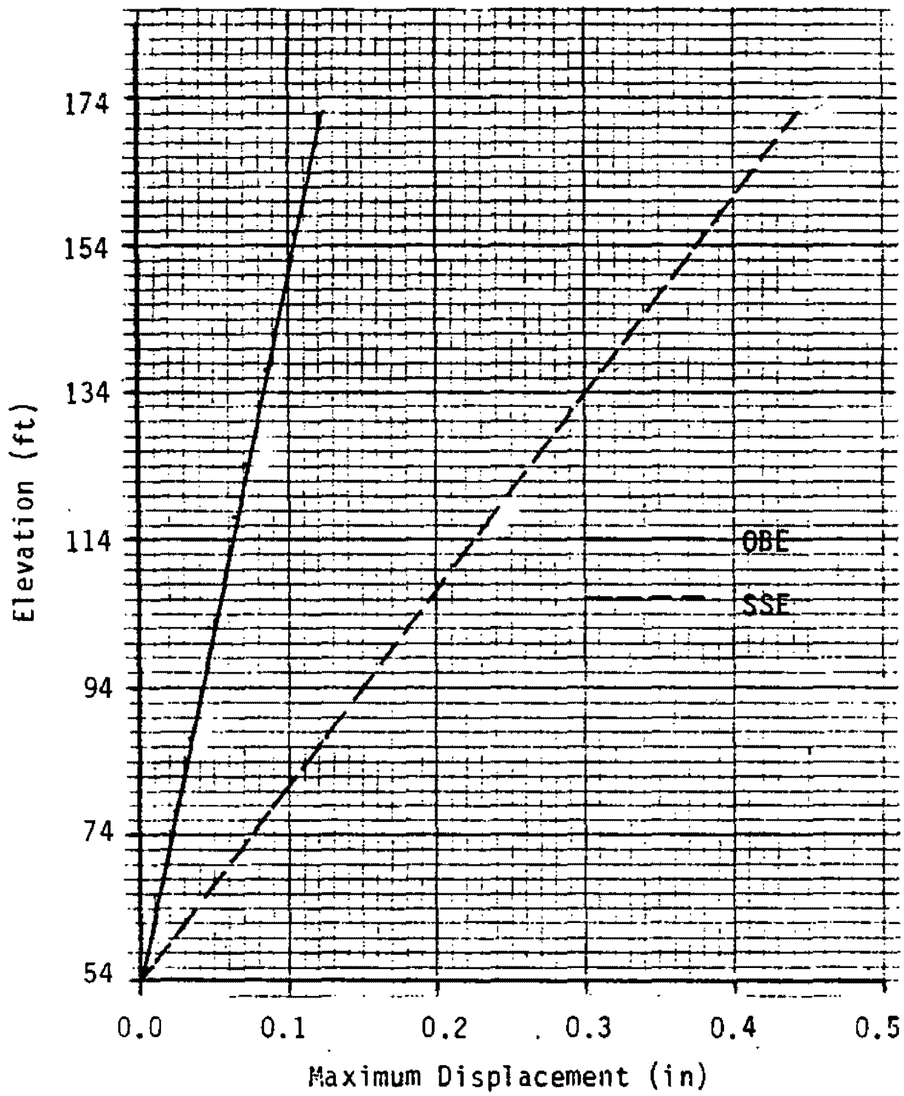
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

AUXILIARY BUILDING
DIESEL GENERATOR AREA
VERTICAL DISPLACEMENT

UPDATED FSAR

FIGURE 3.7-91



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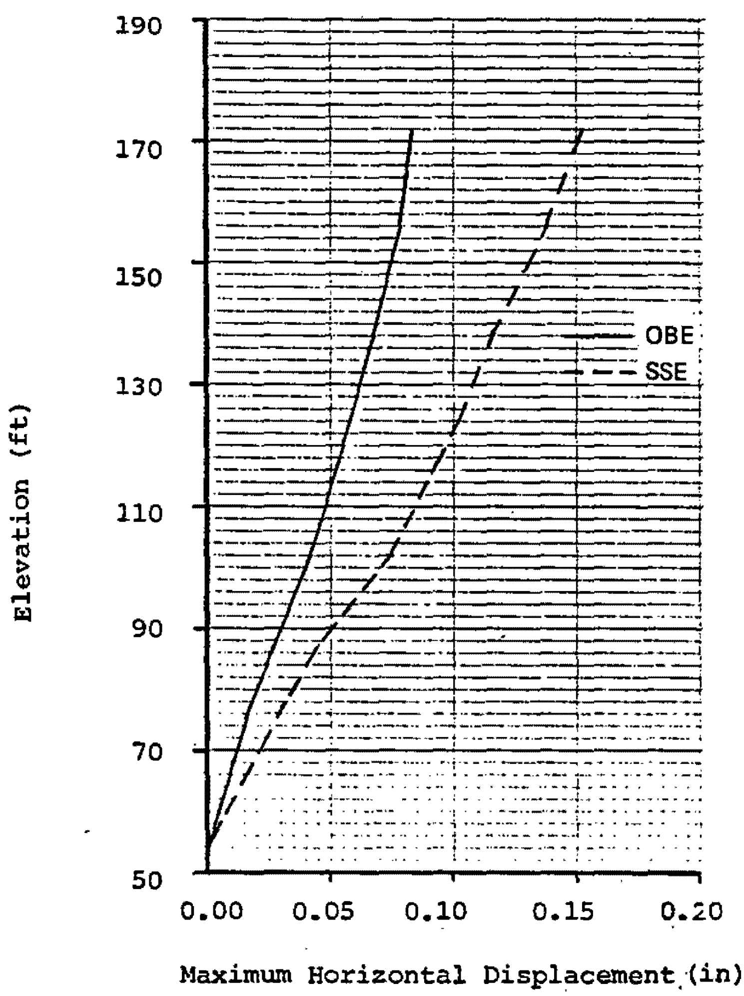
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

AUXILIARY BUILDING
CONTROL AND RADWASTE AREAS
N-S DISPLACEMENT

UPDATED FSAR

FIGURE 3.7-92



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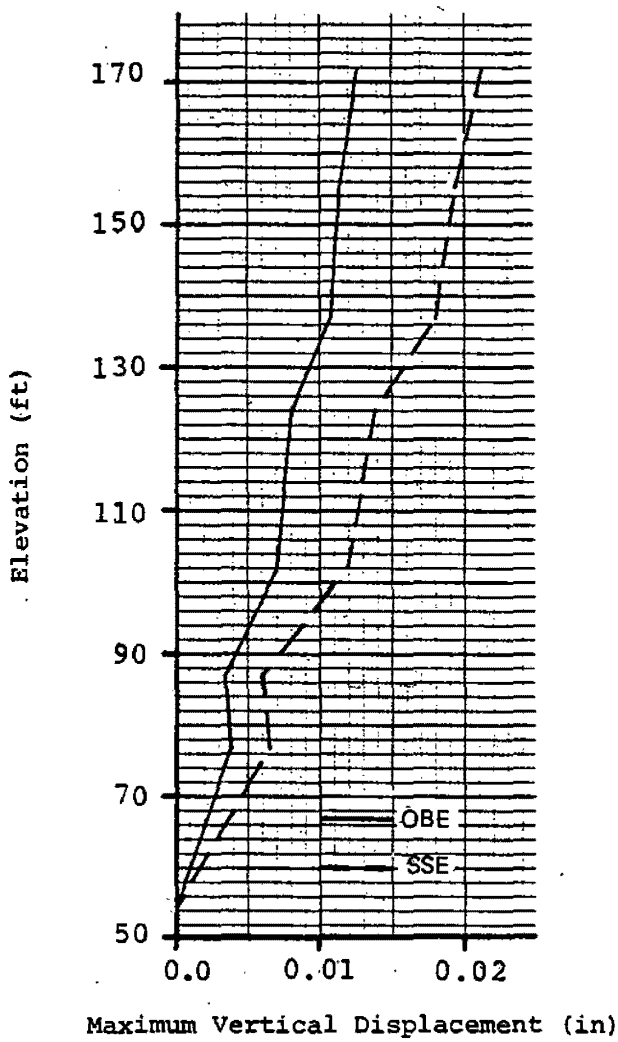
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

AUXILIARY BUILDING
CONTROL AND RADWASTE AREAS
E-W DISPLACEMENT

UPDATED FSAR

FIGURE 3.7-93



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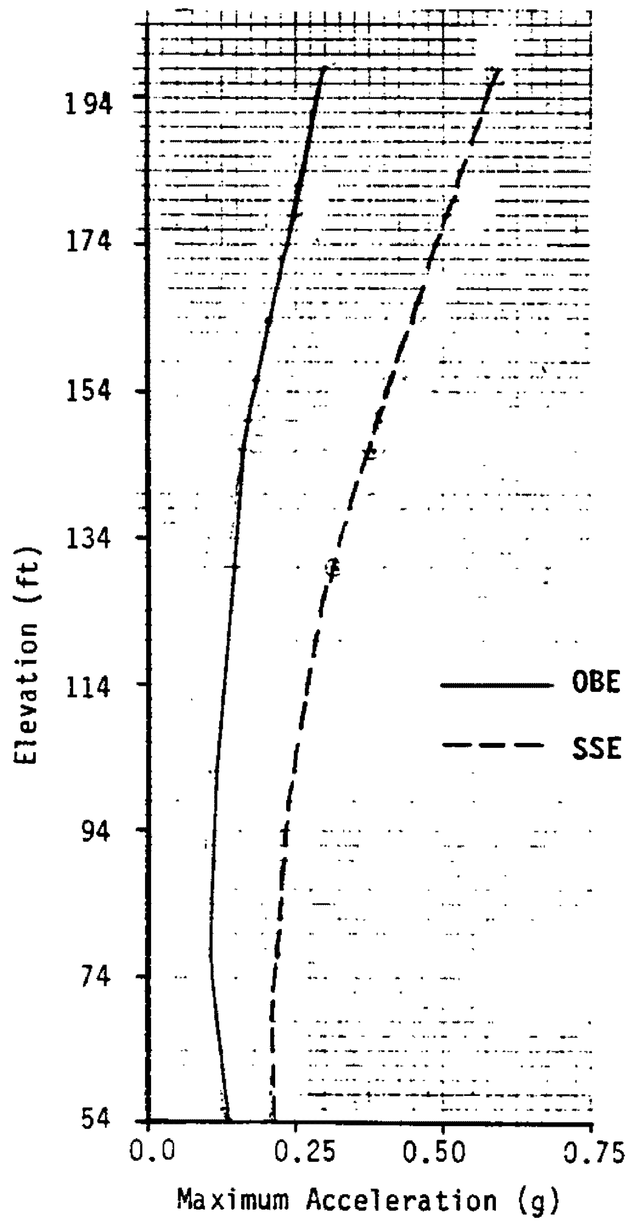
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

AUXILIARY BUILDING
CONTROL AND RADWASTE AREAS
VERTICAL DISPLACEMENT

UPDATED FSAR

FIGURE 3.7-94



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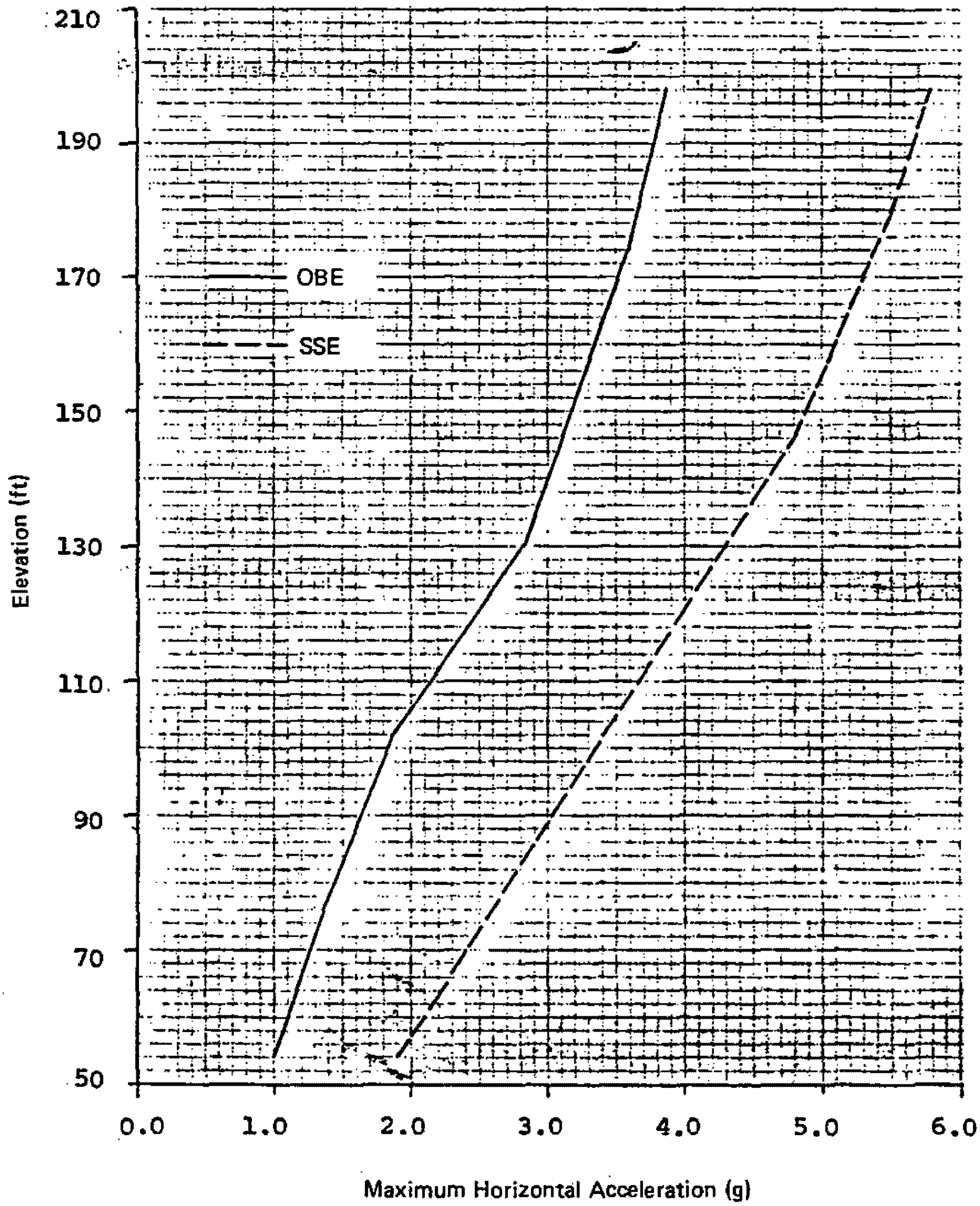
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

AUXILIARY BUILDING
DIESEL GENERATOR AREA
N-S ACCELERATION

UPDATED FSAR

FIGURE 3.7-95



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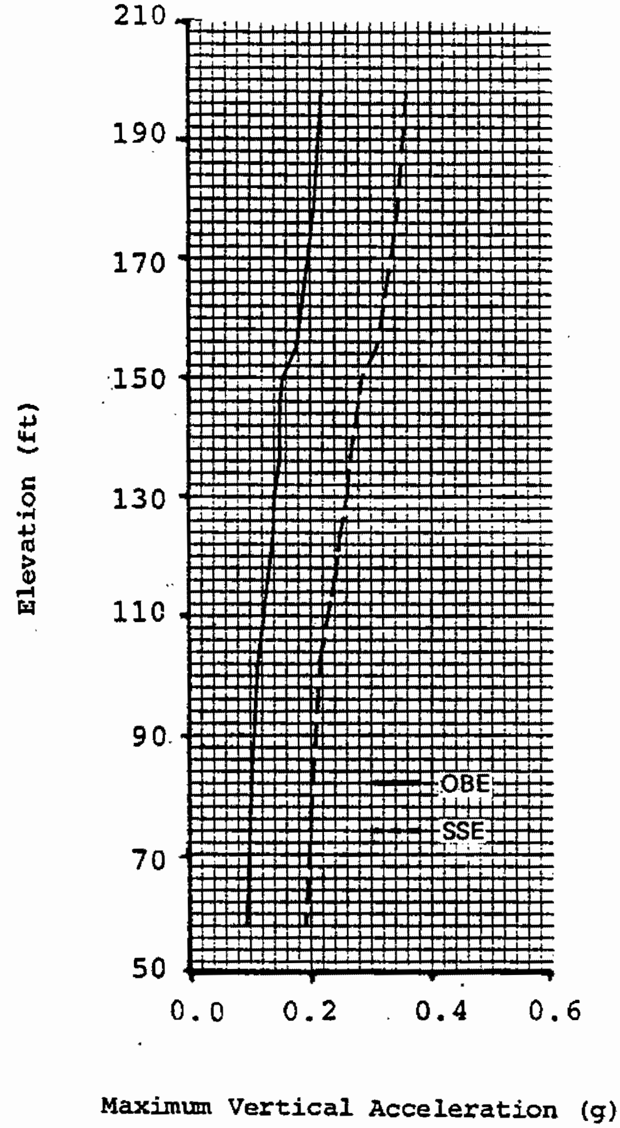
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

AUXILIARY BUILDING
DIESEL GENERATOR AREA
E-W ACCELERATION

UPDATED FSAR

FIGURE 3.7-96



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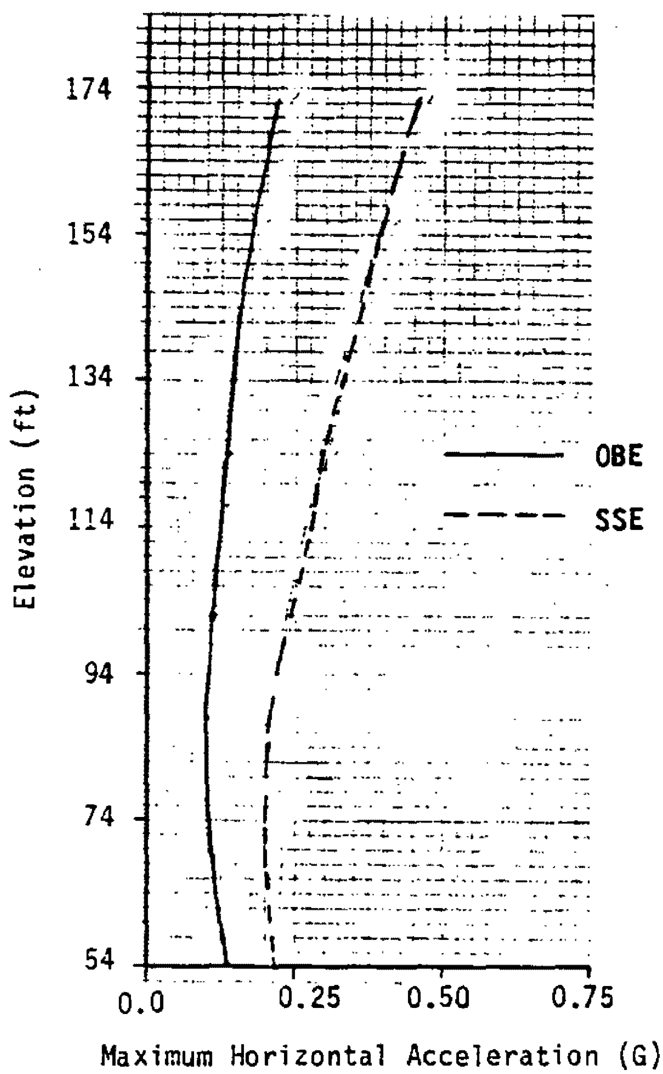
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

AUXILIARY BUILDING
DIESEL GENERATOR AREA
VERTICAL ACCELERATION

UPDATED FSAR

FIGURE 3.7-97



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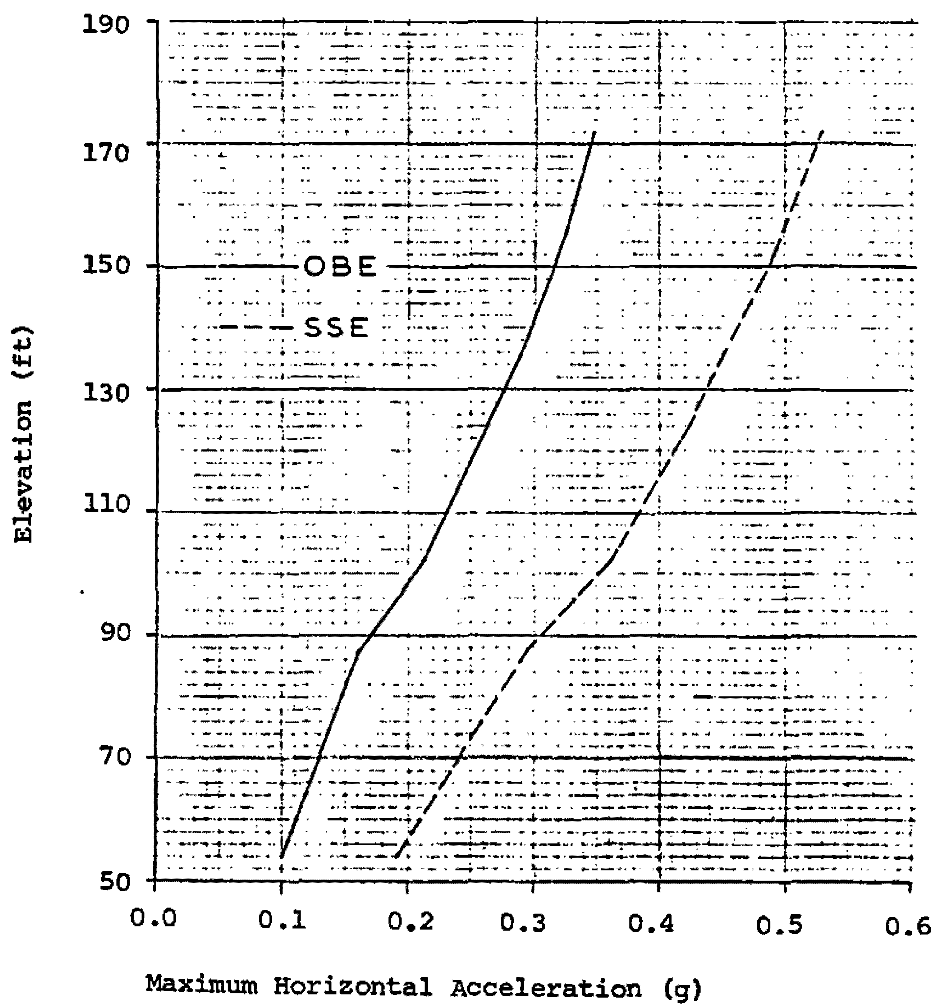
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

AUXILIARY BUILDING
CONTROL AND RADWASTE AREAS
N-S ACCELERATION

UPDATED FSAR

FIGURE 3.7-98



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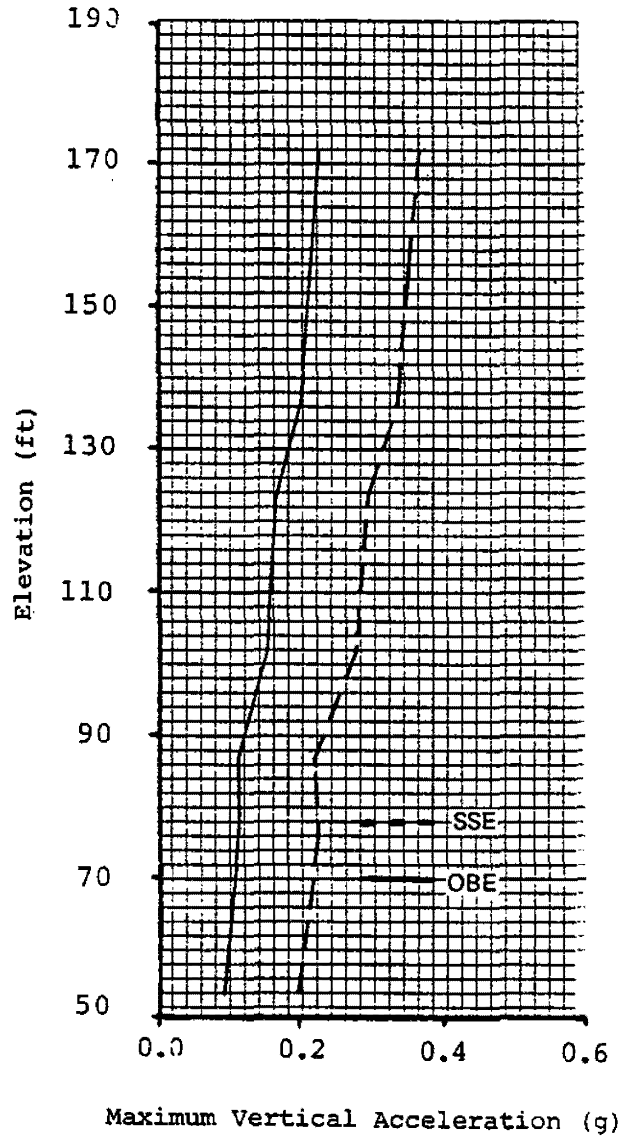
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

AUXILIARY BUILDING
CONTROL AND RADWASTE AREAS
E-W ACCELERATION

UPDATED FSAR

FIGURE 3.7-99



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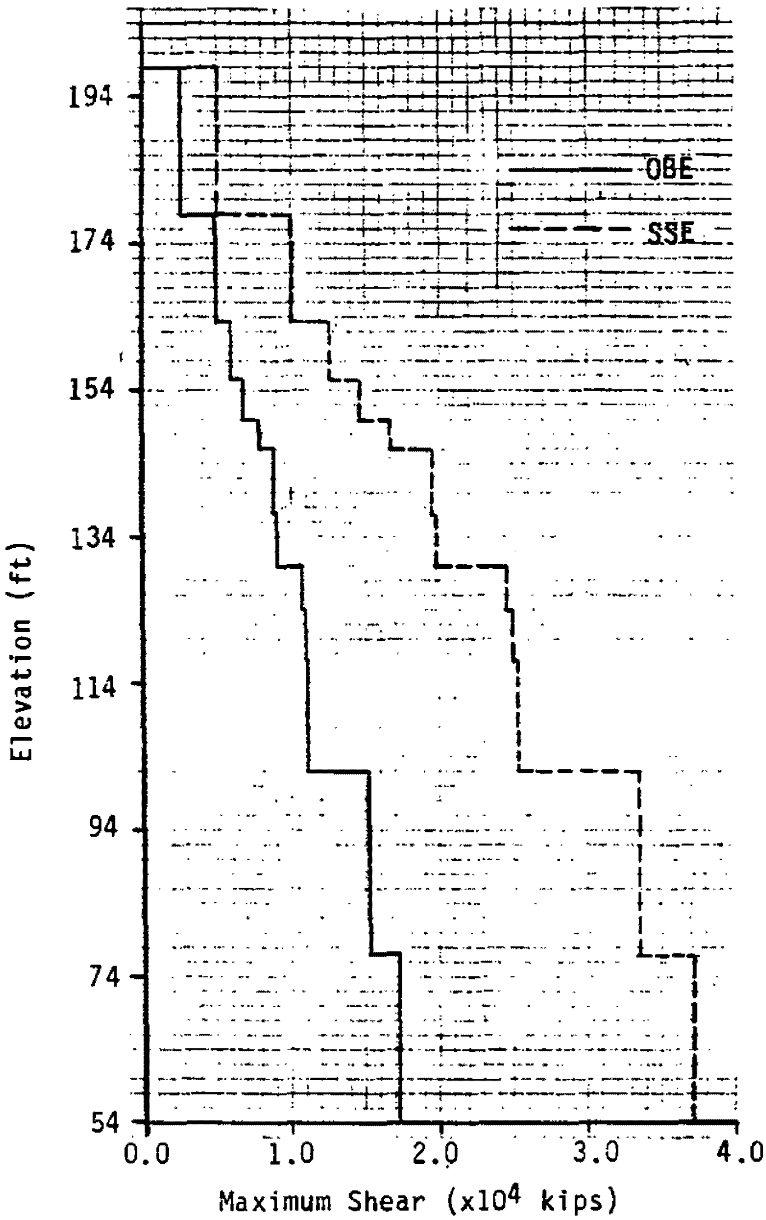
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

AUXILIARY BUILDING
CONTROL AND RADWASTE AREAS
VERTICAL ACCELERATION

UPDATED FSAR

FIGURE 3.7-100



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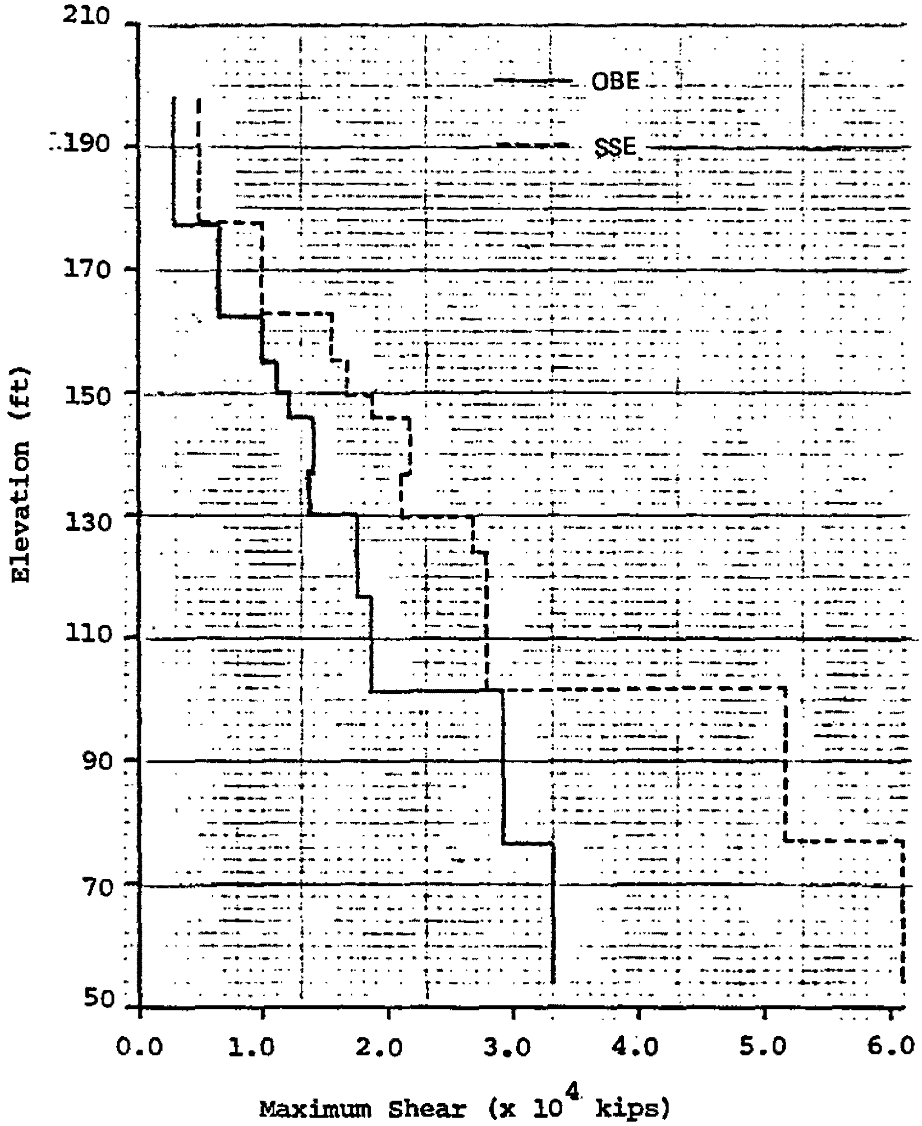
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

AUXILIARY BUILDING
DIESEL GENERATOR AREA
N-S SHEAR FORCES

UPDATED FSAR

FIGURE 3.7-101



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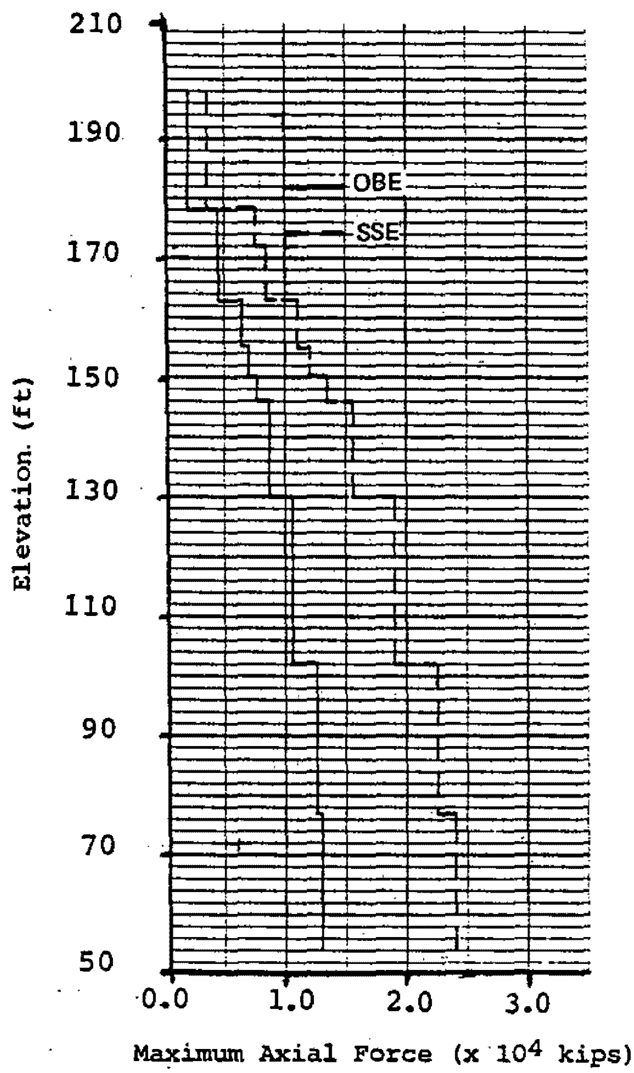
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

AUXILIARY BUILDING
DIESEL GENERATOR AREA
E-W SHEAR FORCES

UPDATED FSAR

FIGURE 3.7-102



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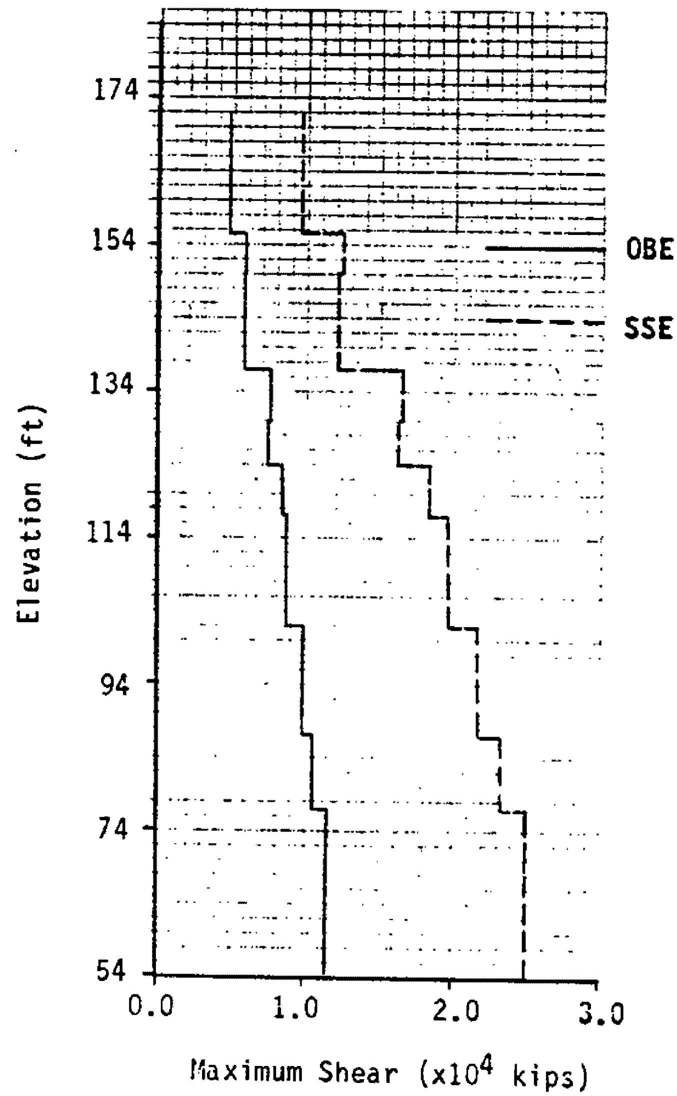
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

AUXILIARY BUILDING
DIESEL GENERATOR AREA
VERTICAL AXIAL FORCES

UPDATED FSAR

FIGURE 3.7-103



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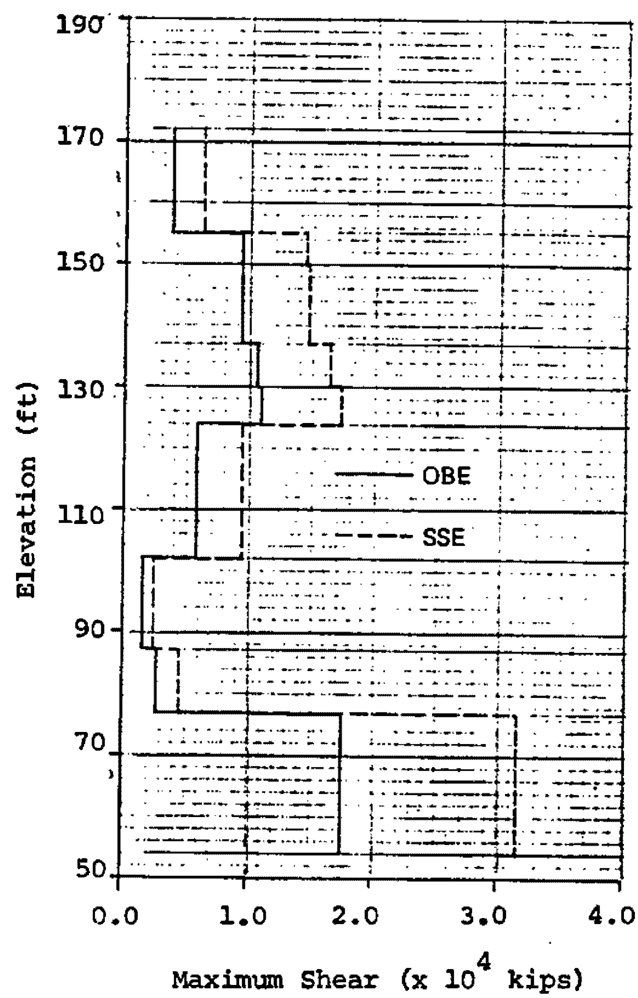
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

AUXILIARY BUILDING
CONTROL AND RADWASTE AREAS
N-S SHEAR FORCES

UPDATED FSAR

FIGURE 3.7-104



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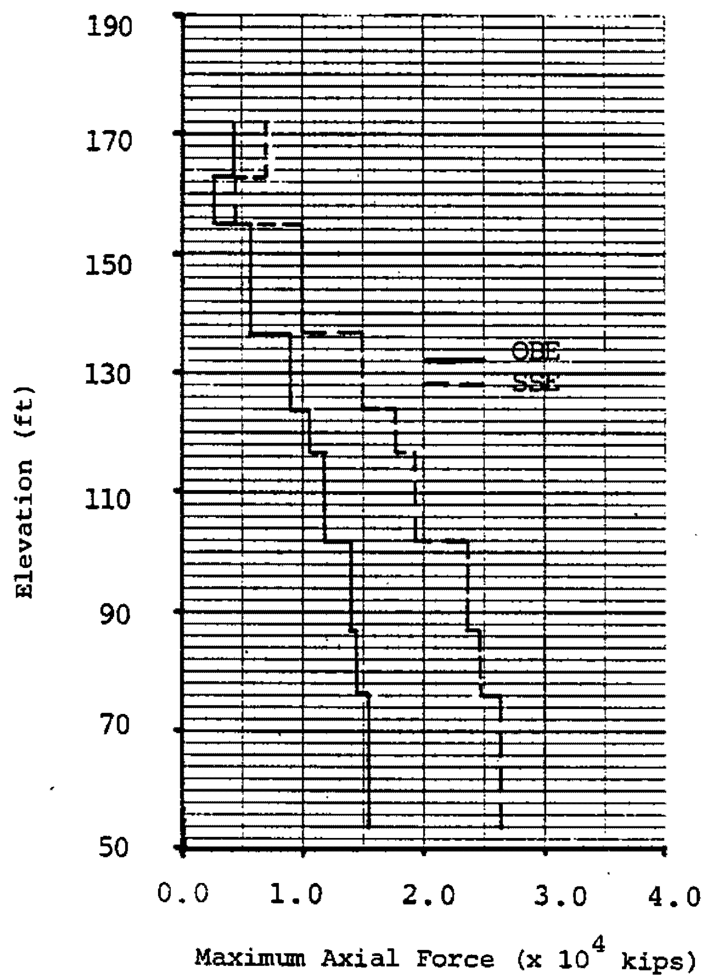
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

AUXILIARY BUILDING
CONTROL AND RADWASTE AREAS
E-W SHEAR FORCES

UPDATED FSAR

FIGURE 3.7-105



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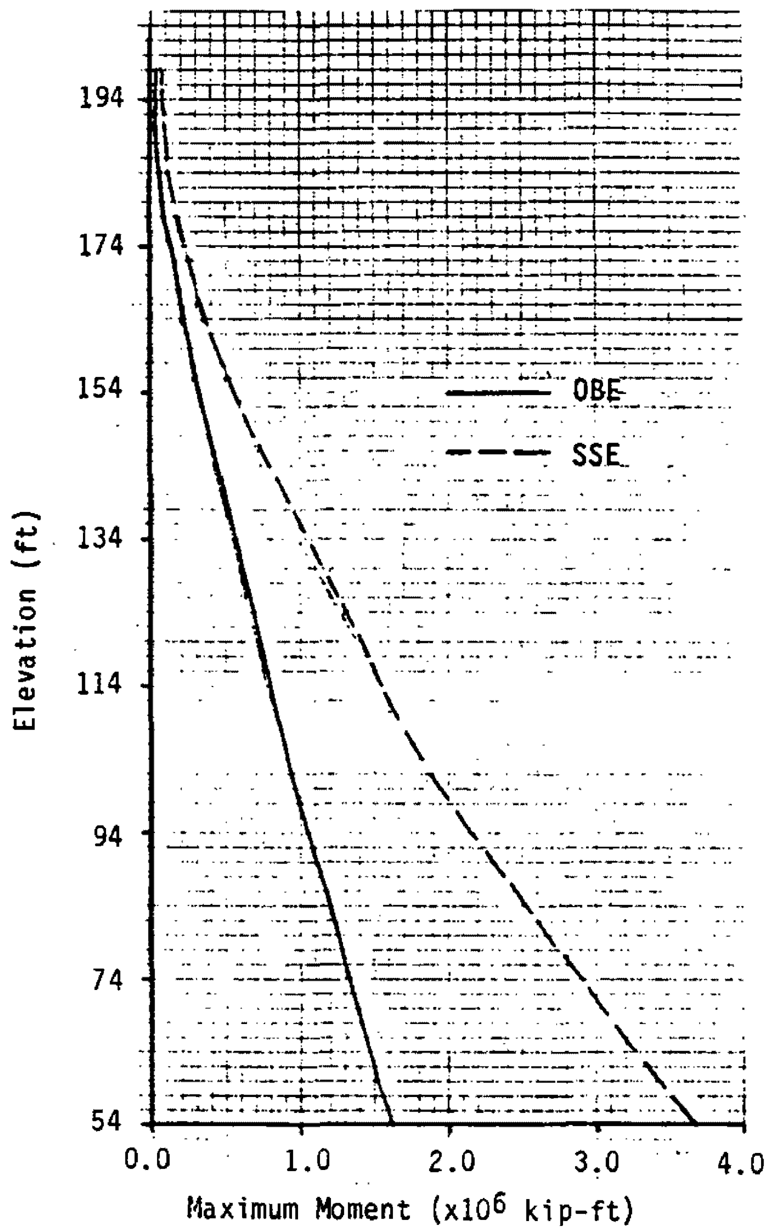
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

AUXILIARY BUILDING
CONTROL AND RADWASTE AREAS
VERTICAL AXIAL FORCES

UPDATED FSAR

FIGURE 3.7-106



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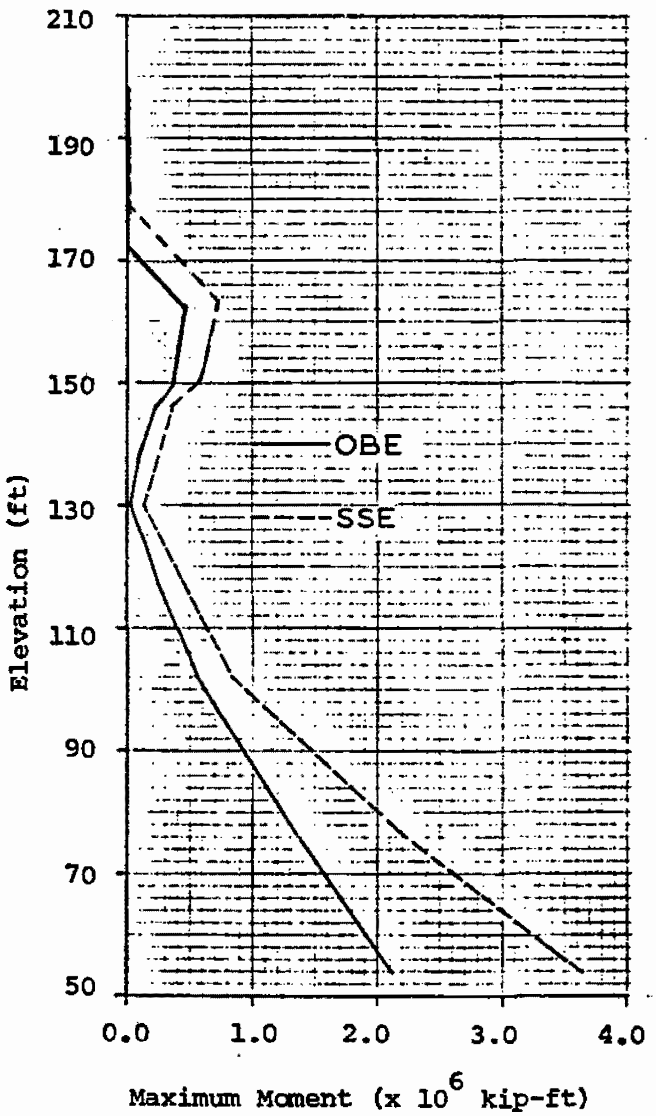
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

AUXILIARY BUILDING
DIESEL GENERATOR AREA
N-S MOMENT

UPDATED FSAR

FIGURE 3.7-107



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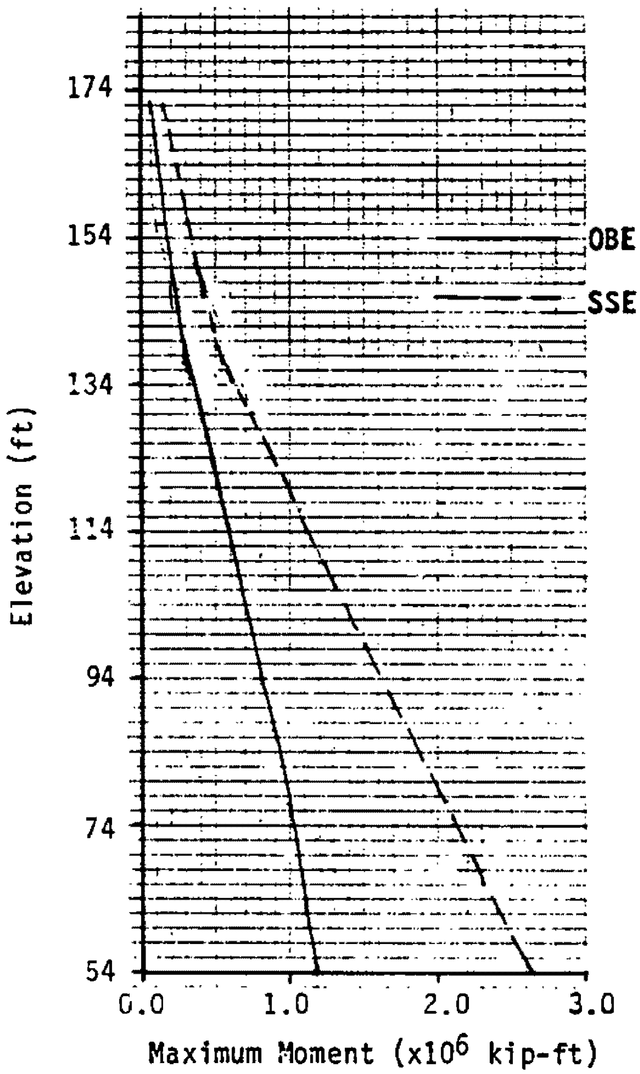
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

AUXILIARY BUILDING
DIESEL GENERATOR AREA
E-W MOMENT

UPDATED FSAR

FIGURE 3.7-108



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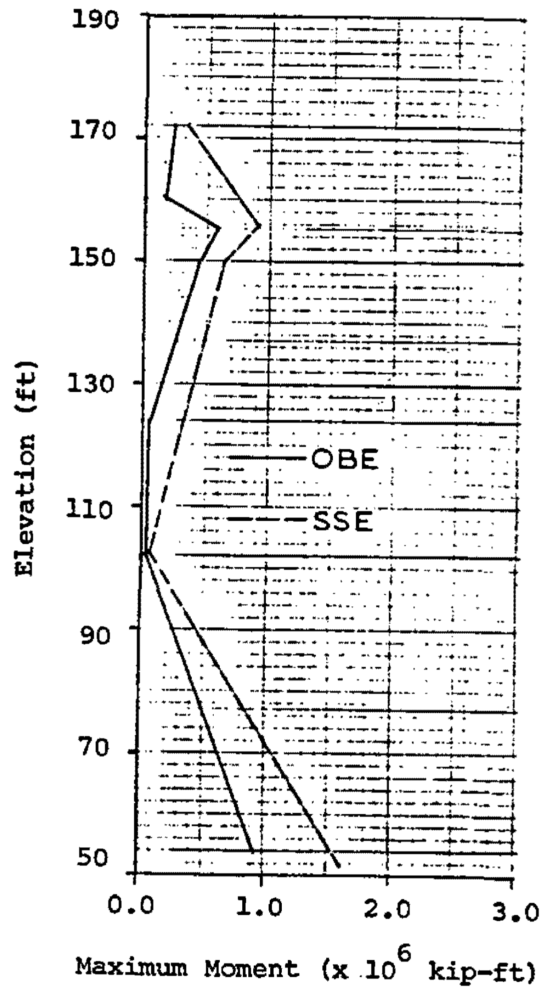
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

AUXILIARY BUILDING
CONTROL AND RADWASTE AREAS
N-S MOMENT

UPDATED FSAR

FIGURE 3.7-109



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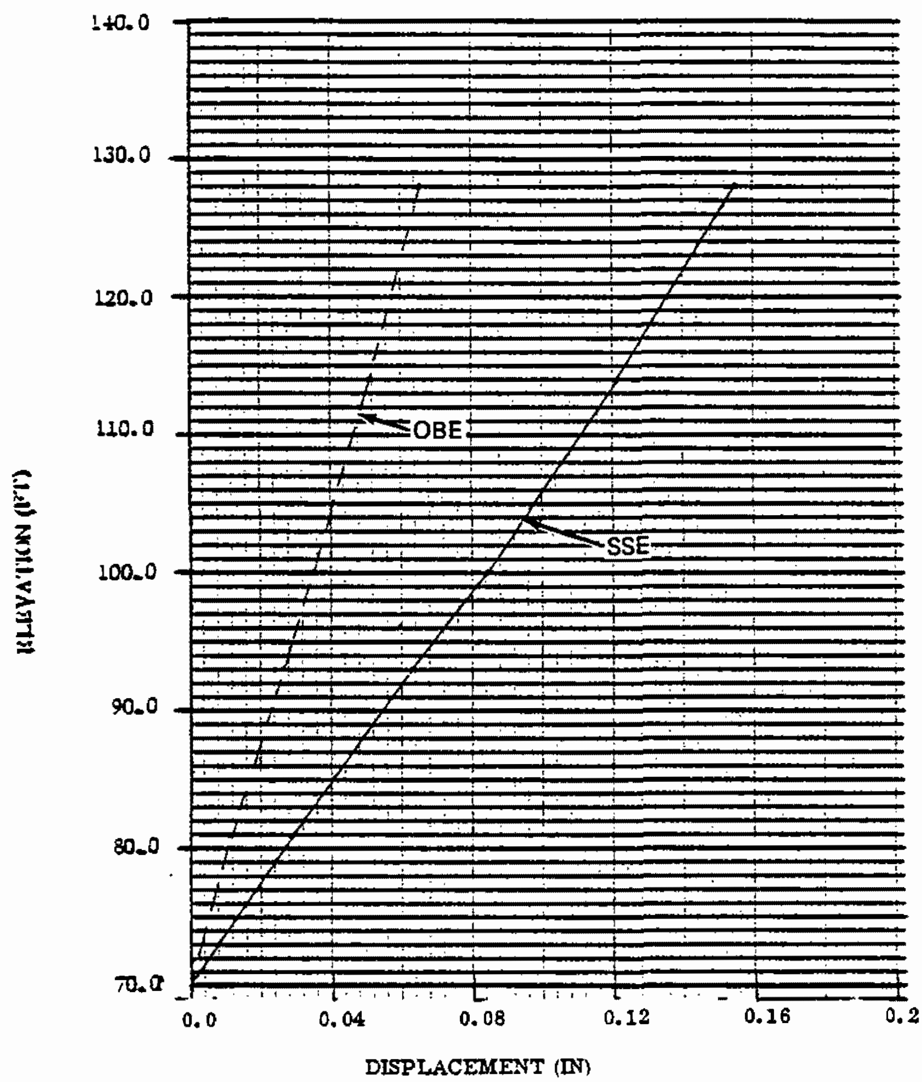
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

AUXILIARY BUILDING
CONTROL AND RADWASTE AREAS
E-W MOMENT

UPDATED FSAR

FIGURE 3.7-110



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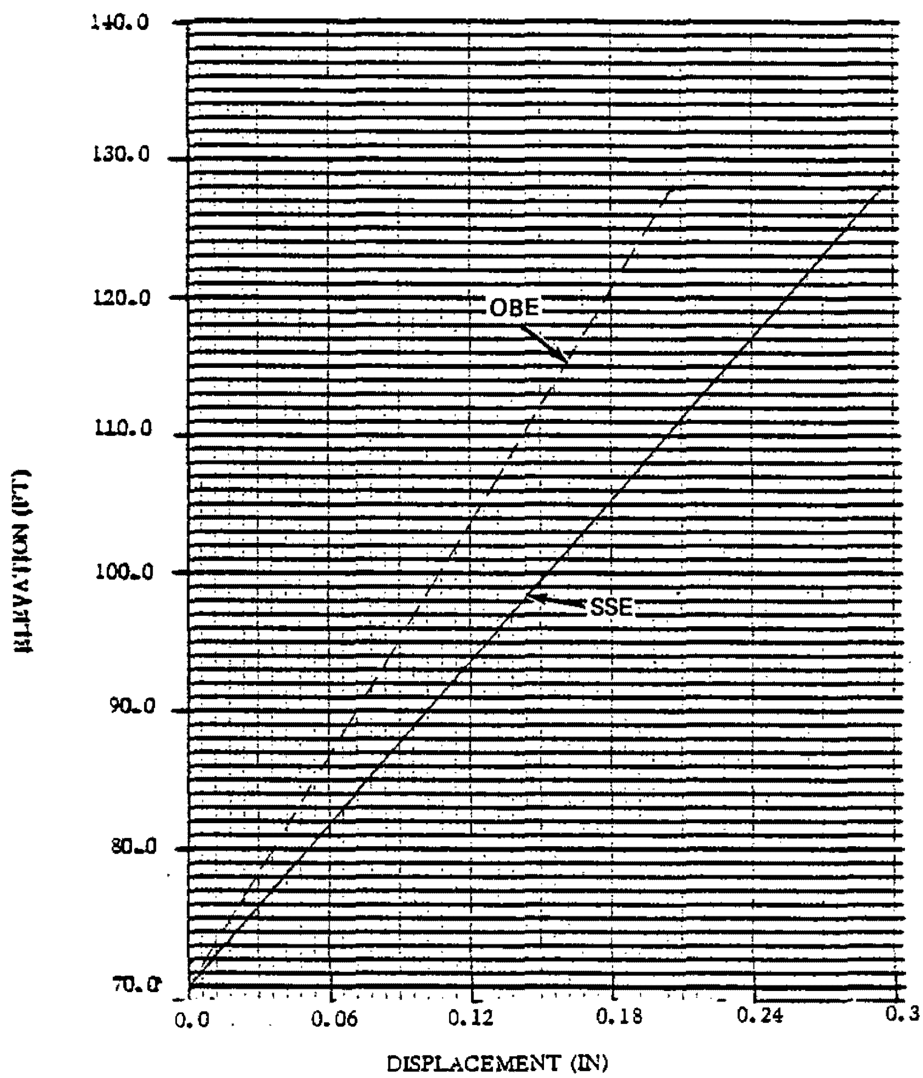
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

SERVICE WATER INTAKE STRUCTURE
CELL WALLS
N-S DISPLACEMENT

UPDATED FSAR

FIGURE 3.7-111



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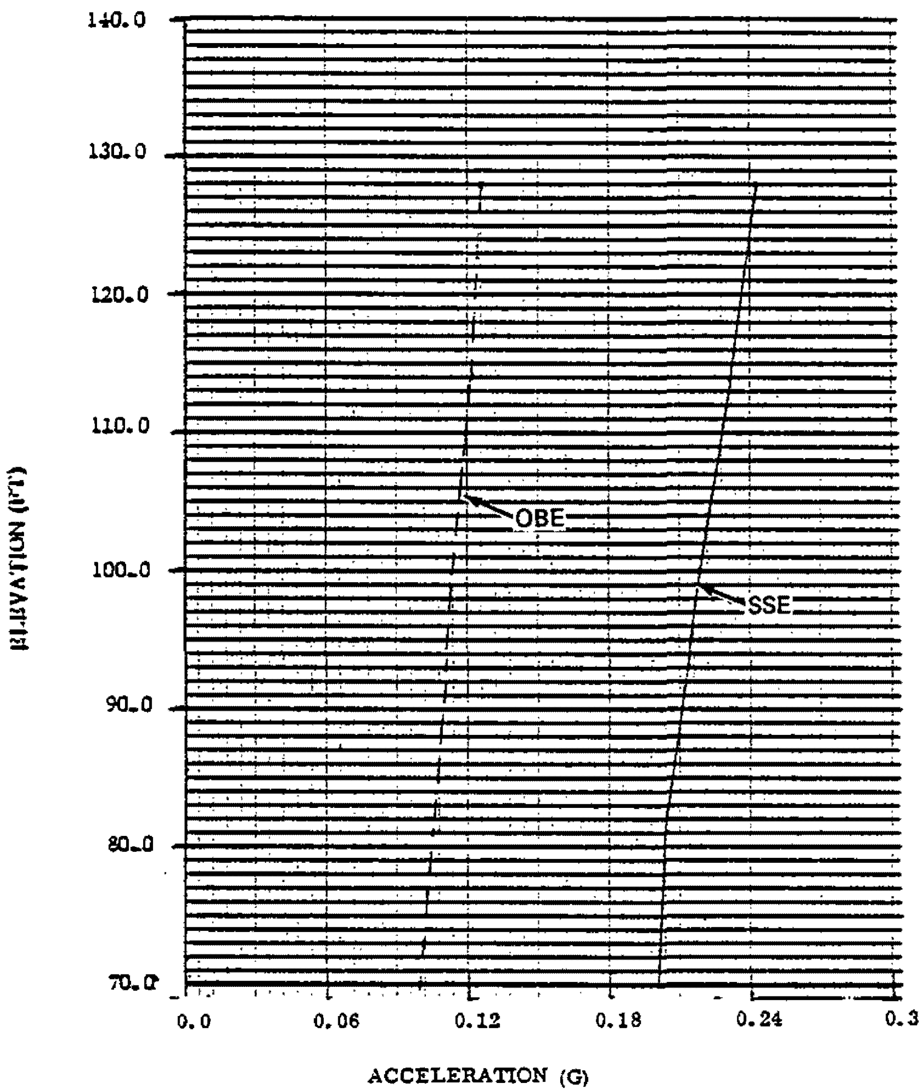
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

SERVICE WATER INTAKE STRUCTURE
CELL WALLS
E-W DISPLACEMENT

UPDATED FSAR

FIGURE 3.7-112



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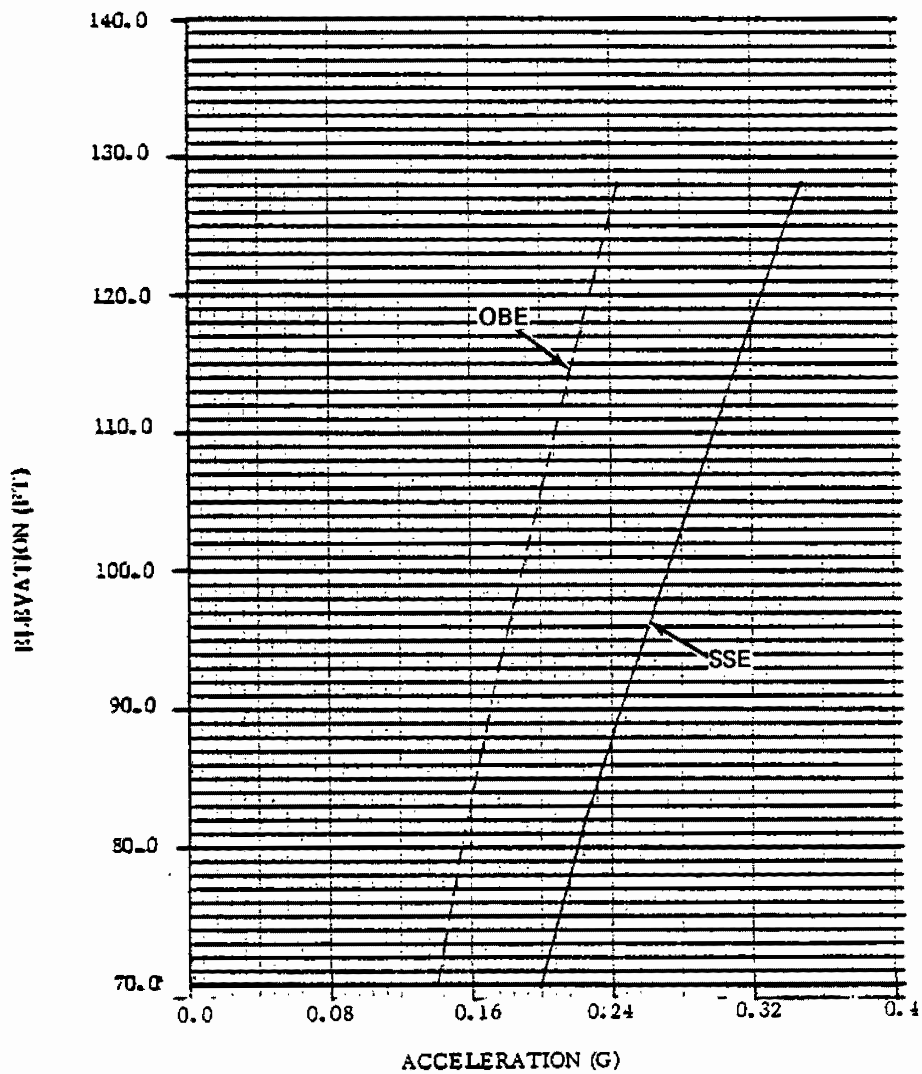
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

SERVICE WATER INTAKE STRUCTURE
CELL WALLS
N-S ACCELERATION

UPDATED FSAR

FIGURE 3.7-113



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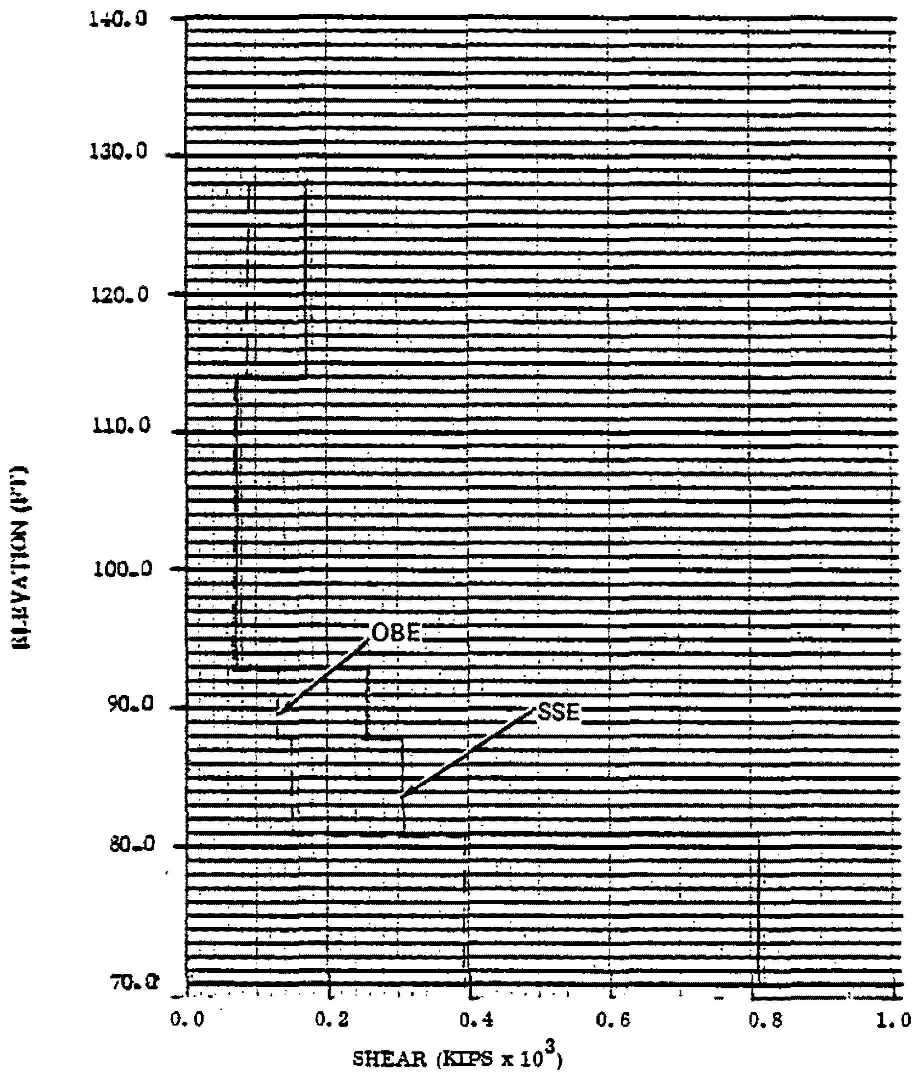
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

SERVICE WATER INTAKE STRUCTURE
CELL WALLS
E-W ACCELERATION

UPDATED FSAR

FIGURE 3.7-114



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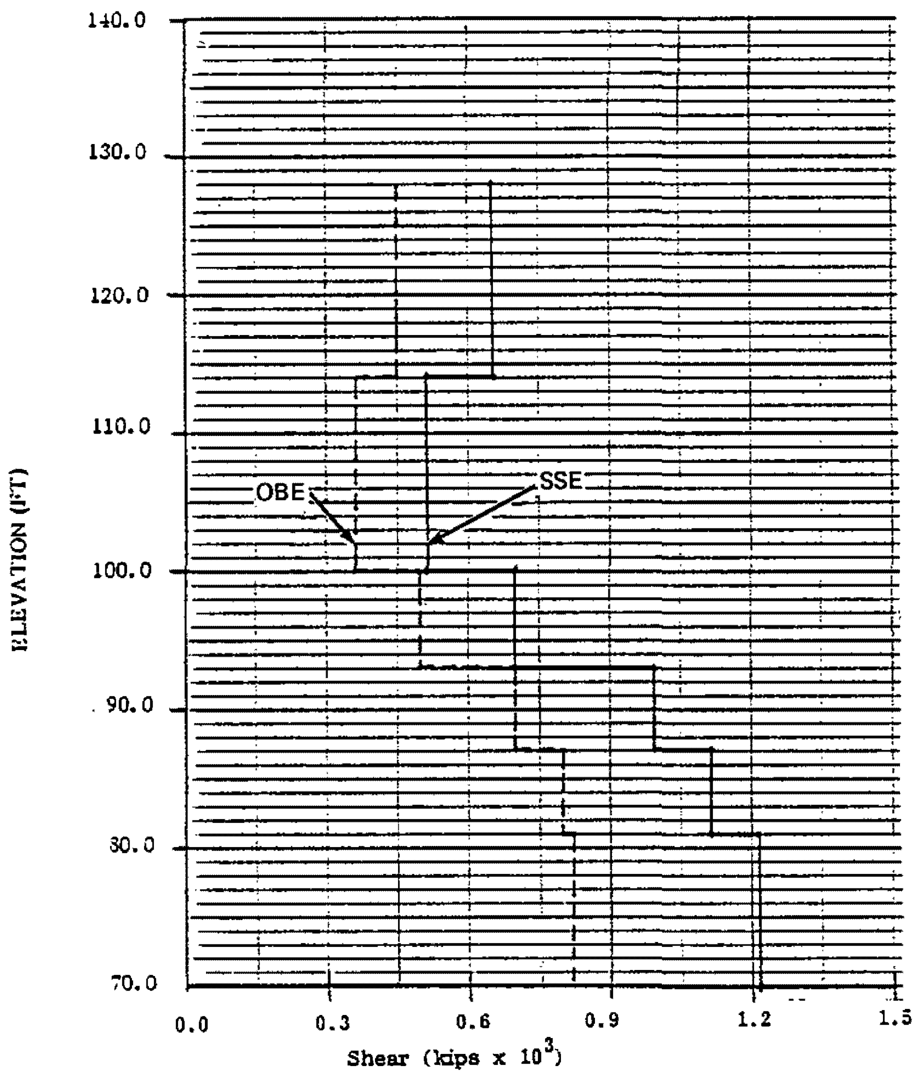
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

SERVICE WATER INTAKE STRUCTURE
CELL WALLS
N-S SHEAR FORCES

UPDATED FSAR

FIGURE 3.7-115



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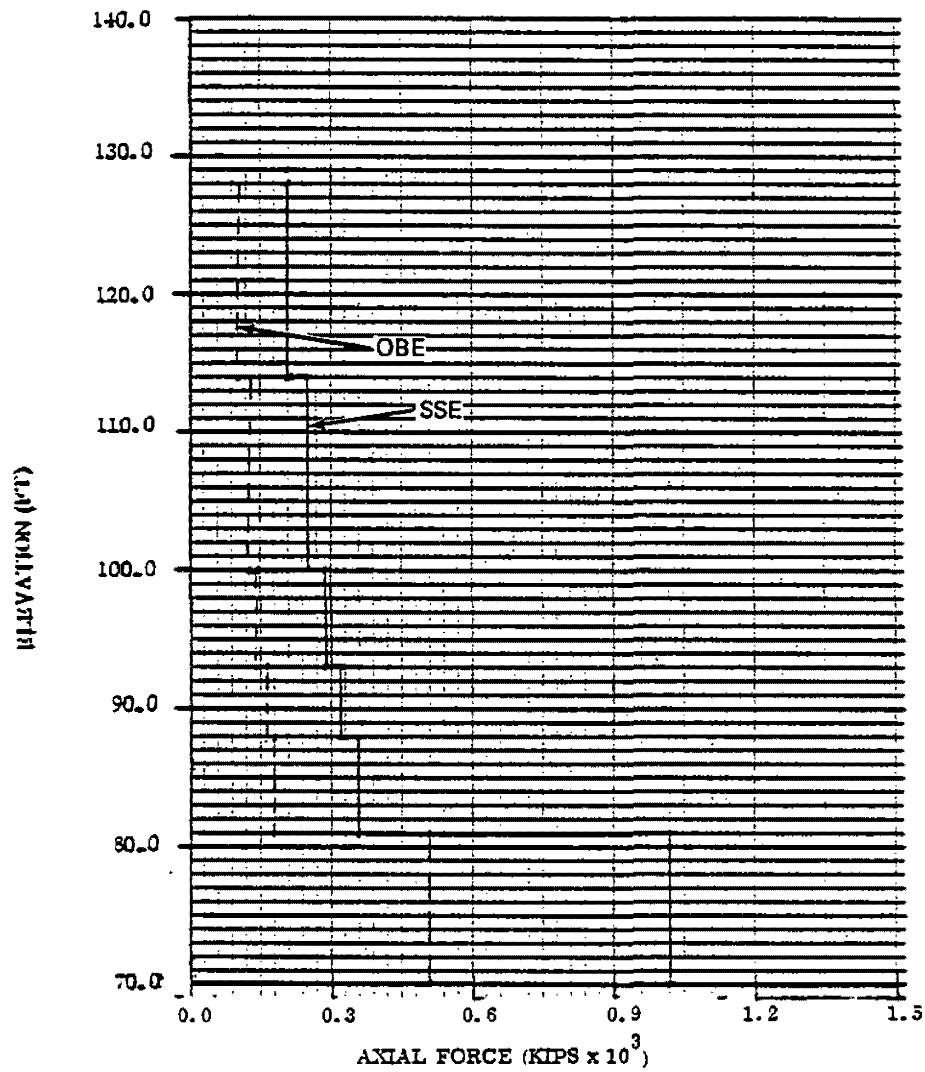
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

SERVICE WATER INTAKE STRUCTURE
CELL WALLS
N-S SHEAR FORCES

UPDATED FSAR

FIGURE 3.7-116



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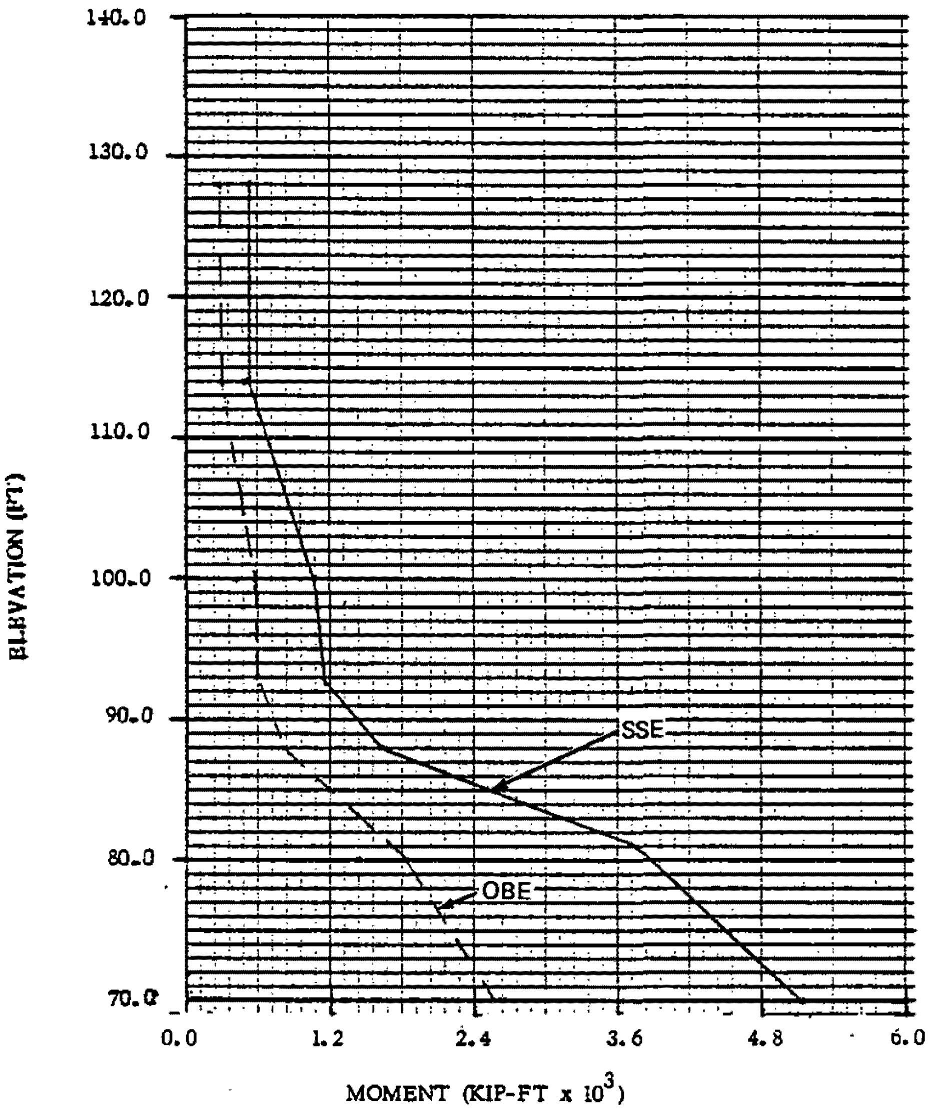
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

SERVICE WATER INTAKE STRUCTURE
CELL WALLS
VERTICAL AXIAL FORCES

UPDATED FSAR

FIGURE 3.7-117



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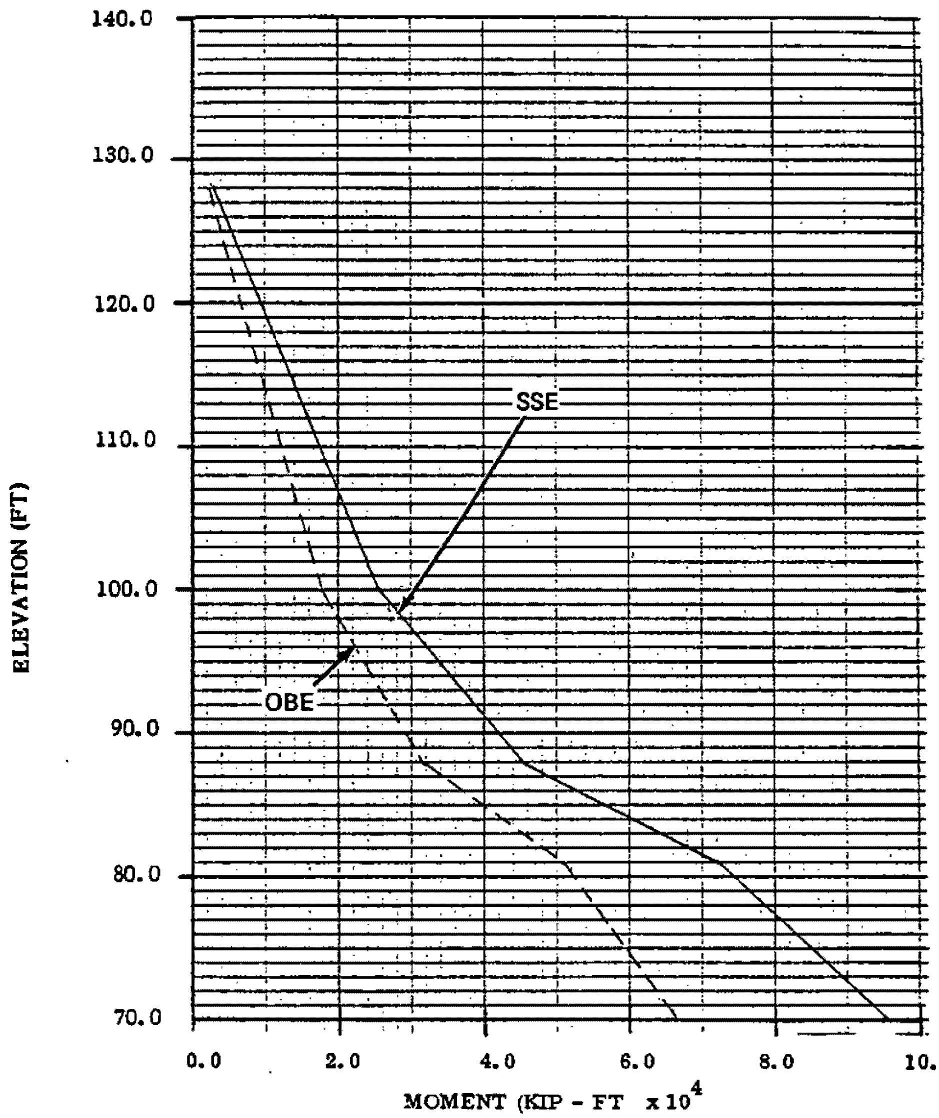
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

SERVICE WATER INTAKE STRUCTURE
CELL WALLS
N-S MOMENT

UPDATED FSAR

FIGURE 3.7-118



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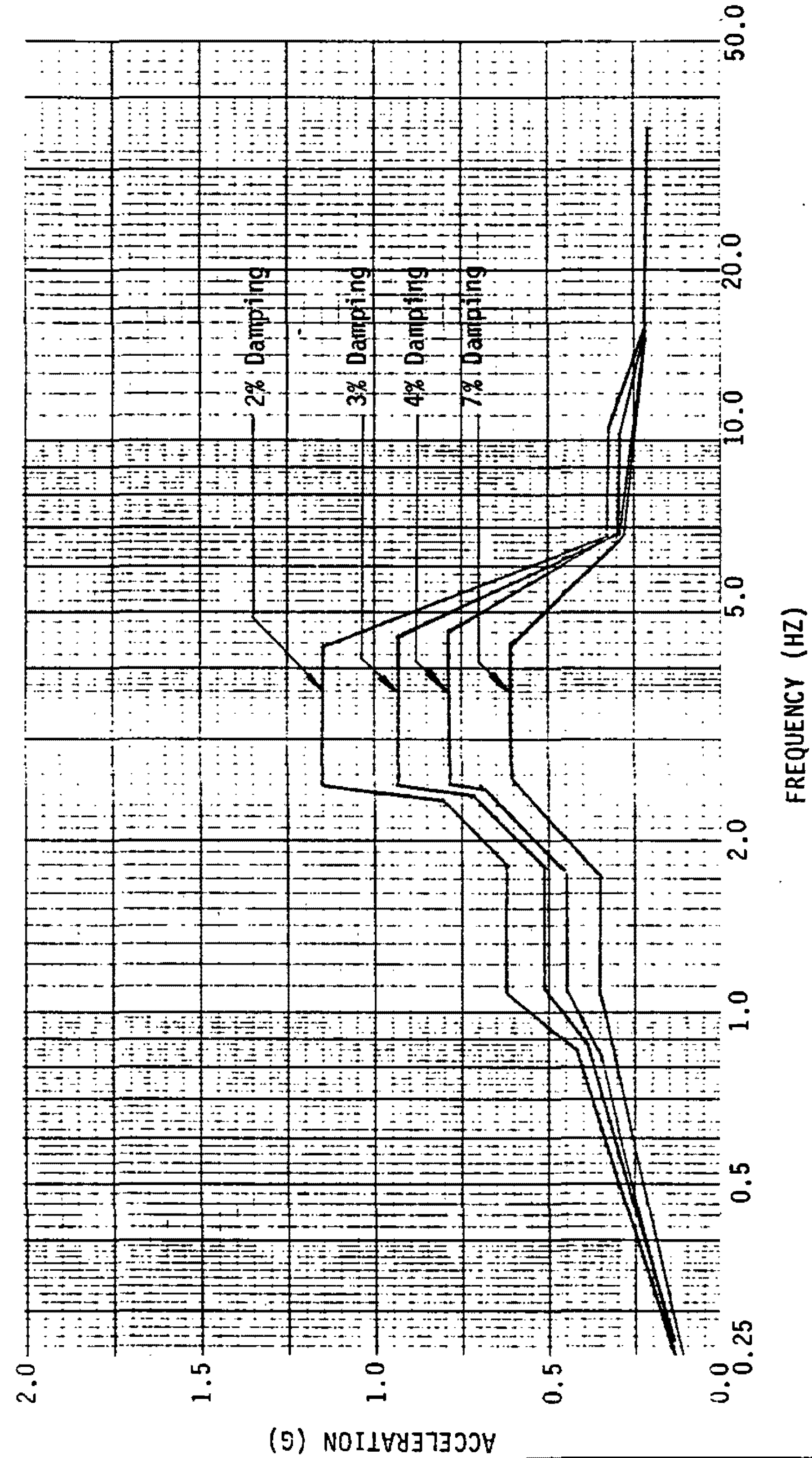
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

SERVICE WATER INTAKE STRUCTURE
CELL WALLS
E-W MOMENT

UPDATED FSAR

FIGURE 3.7-119



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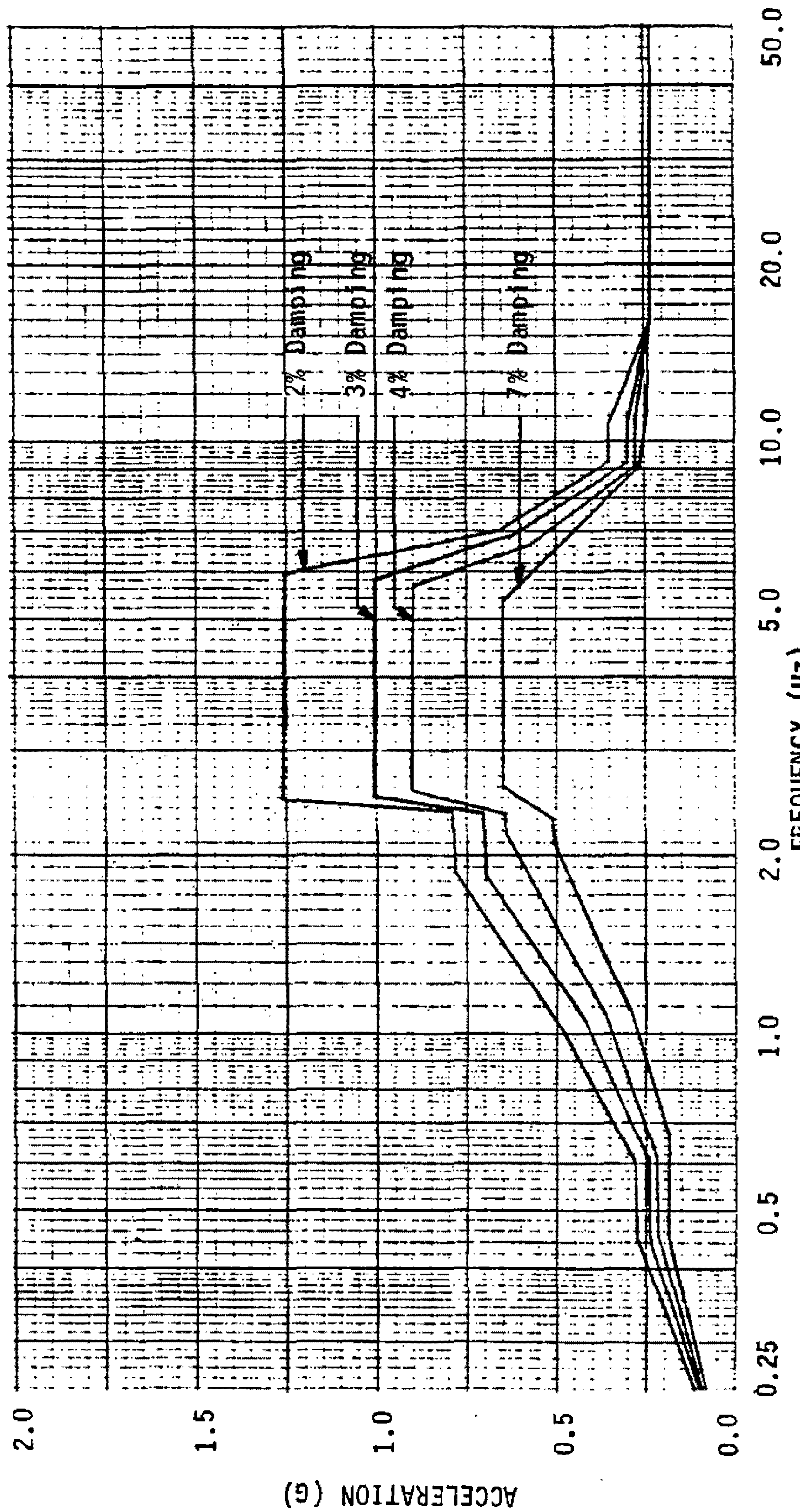
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

FLOOR RESPONSE SPECTRA
REACTOR BUILDING AT EL 102'-0"
N–S HORIZONTAL SSE CASE

UPDATED FSAR

FIGURE 3.7-120



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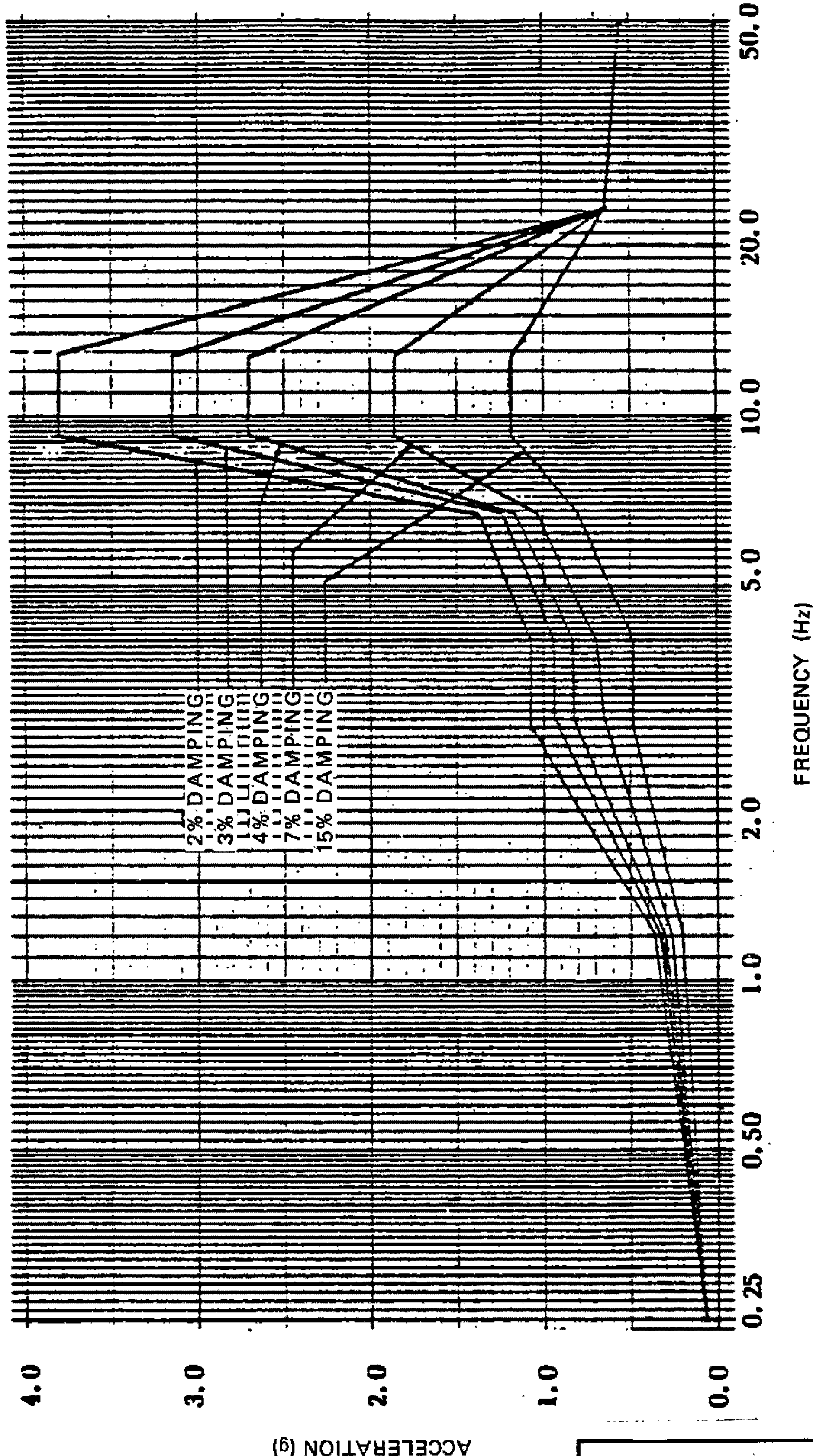
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

FLOOR RESPONSE SPECTRA
REACTOR BUILDING AT EL 102'-0"
E–W HORIZONTAL SSE CASE

UPDATED FSAR

FIGURE 3.7-121



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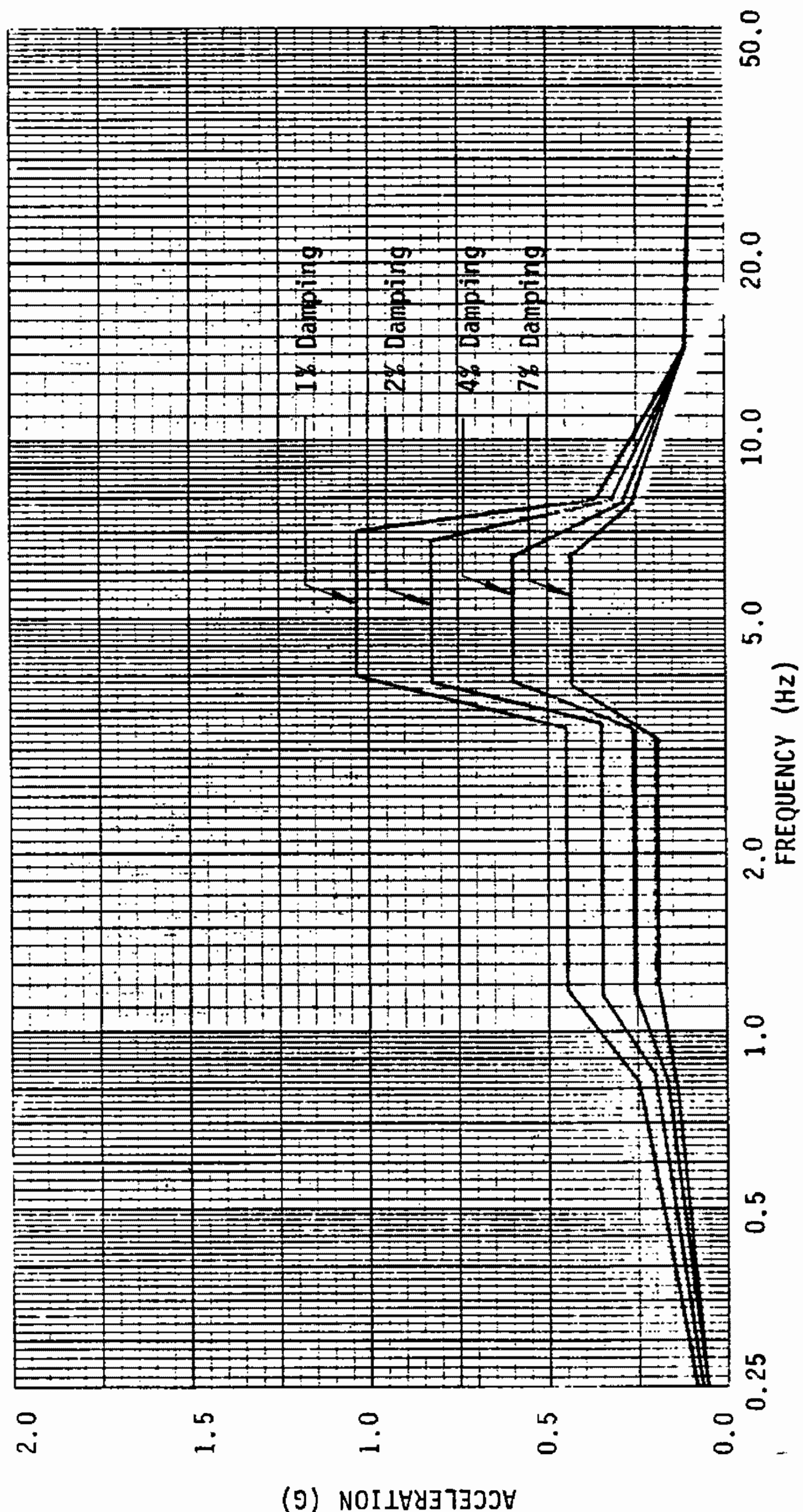
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

FLOOR RESPONSE SPECTRA
REACTOR BUILDING AT EL 102'-0"
VERTICAL SSE CASE

UPDATED FSAR

FIGURE 3.7-122



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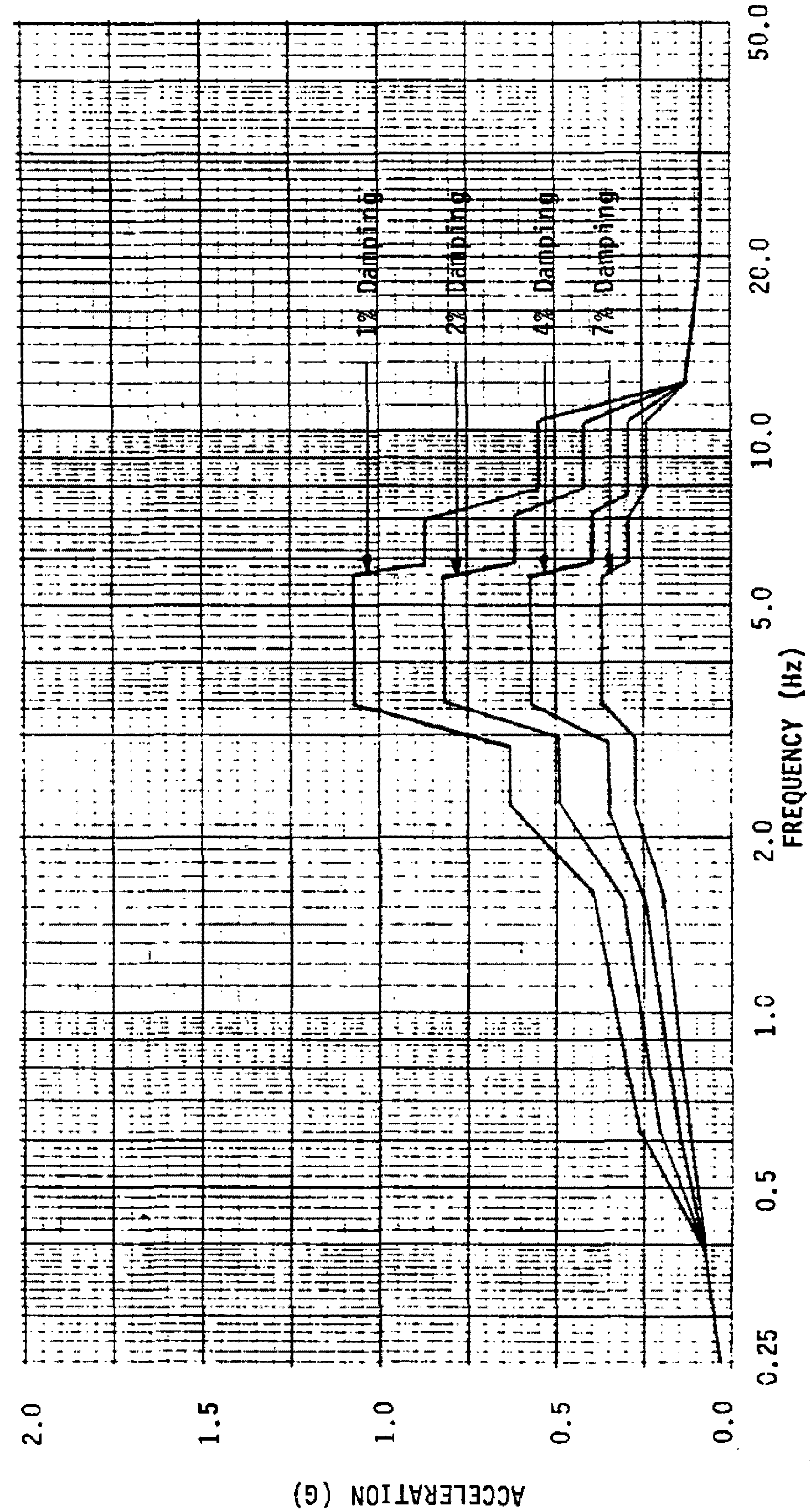
PUBLIC SERVICE ELECTRIC AND GAS COMPANY HOPE CREEK NUCLEAR GENERATING STATION	
FLOOR RESPONSE SPECTRA REACTOR BUILDING AT EL 102'-0" N–S HORIZONTAL OBE CASE	
UPDATED FSAR	FIGURE 3.7-123



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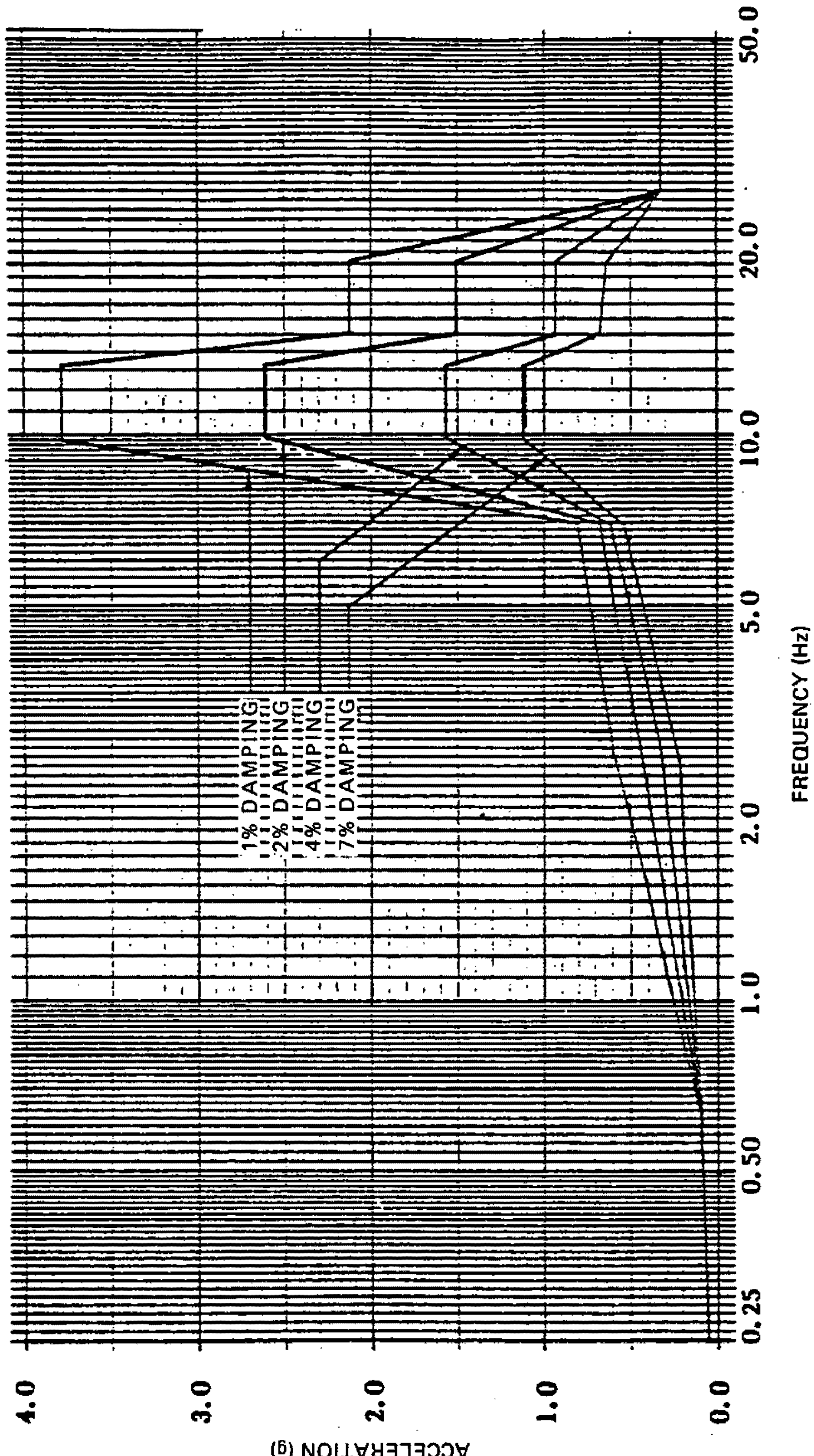
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

FLOOR RESPONSE SPECTRA
REACTOR BUILDING AT EL 102'-0"
E–W HORIZONTAL OBE CASE

UPDATED FSAR

FIGURE 3.7-124



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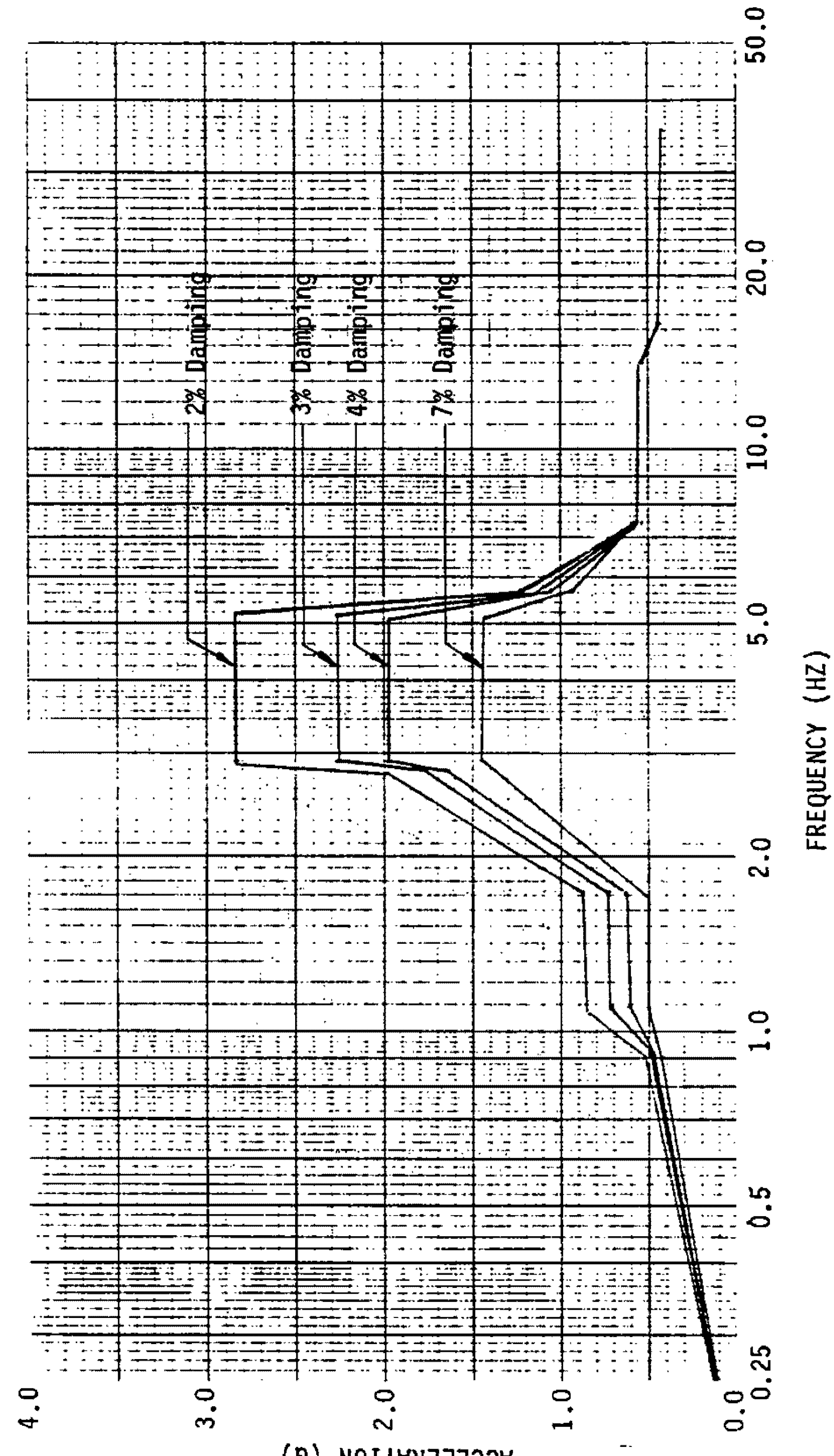
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

FLOOR RESPONSE SPECTRA
REACTOR BUILDING AT EL 102'-0"
VERTICAL OBE CASE

UPDATED FSAR

FIGURE 3.7-125



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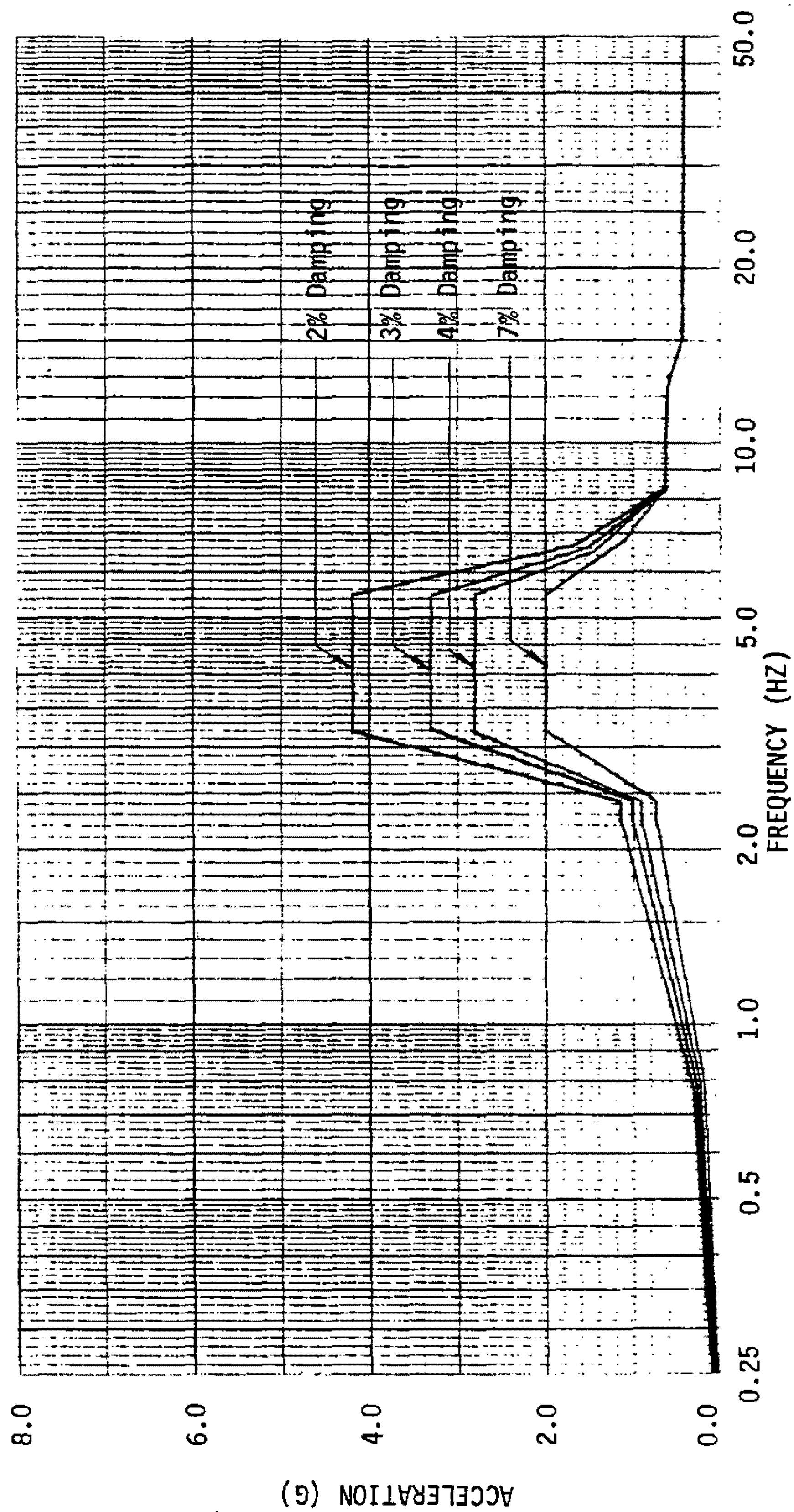
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

FLOOR RESPONSE SPECTRA
REACTOR BUILDING AT EL 201'-0"
N–S HORIZONTAL SSE CASE

UPDATED FSAR

FIGURE 3.7-126



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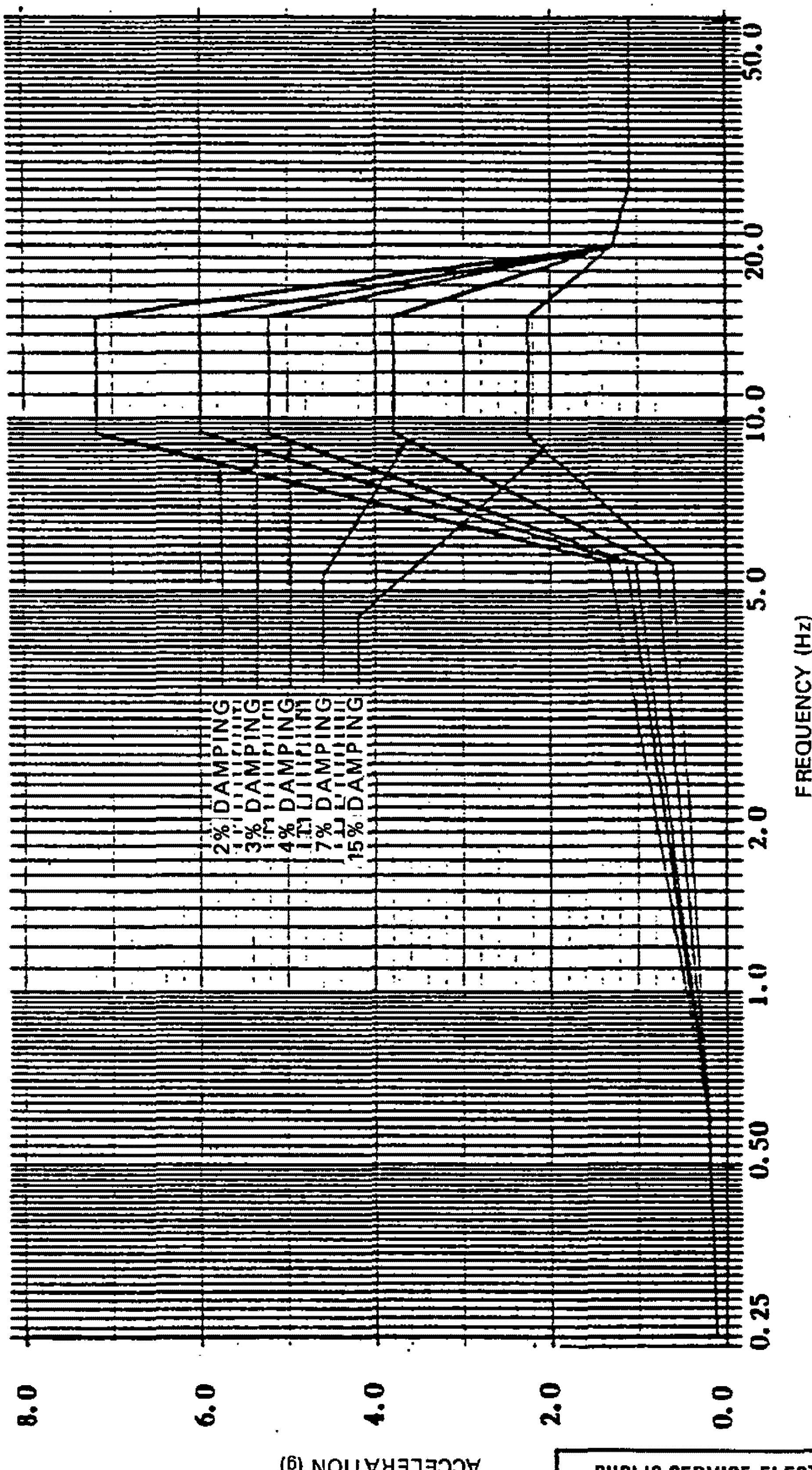
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

FLOOR RESPONSE SPECTRA
REACTOR BUILDING AT EL 201'-0"
E–W HORIZONTAL SSE CASE

UPDATED FSAR

FIGURE 3.7-127



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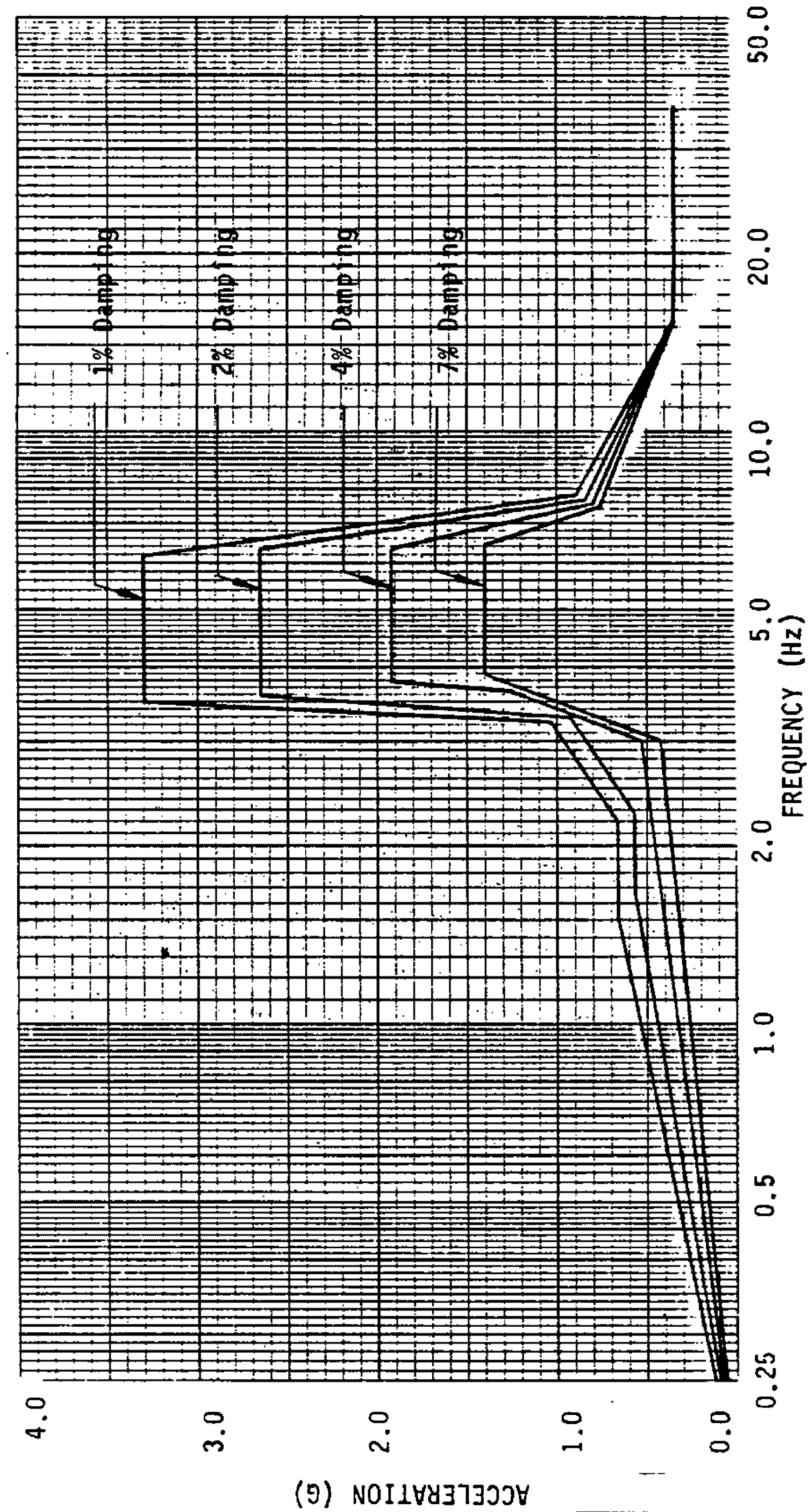
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

FLOOR RESPONSE SPECTRA
REACTOR BUILDING AT EL 201'-0"
VERTICAL SSE CASE

UPDATED FSAR

FIGURE 3.7-128



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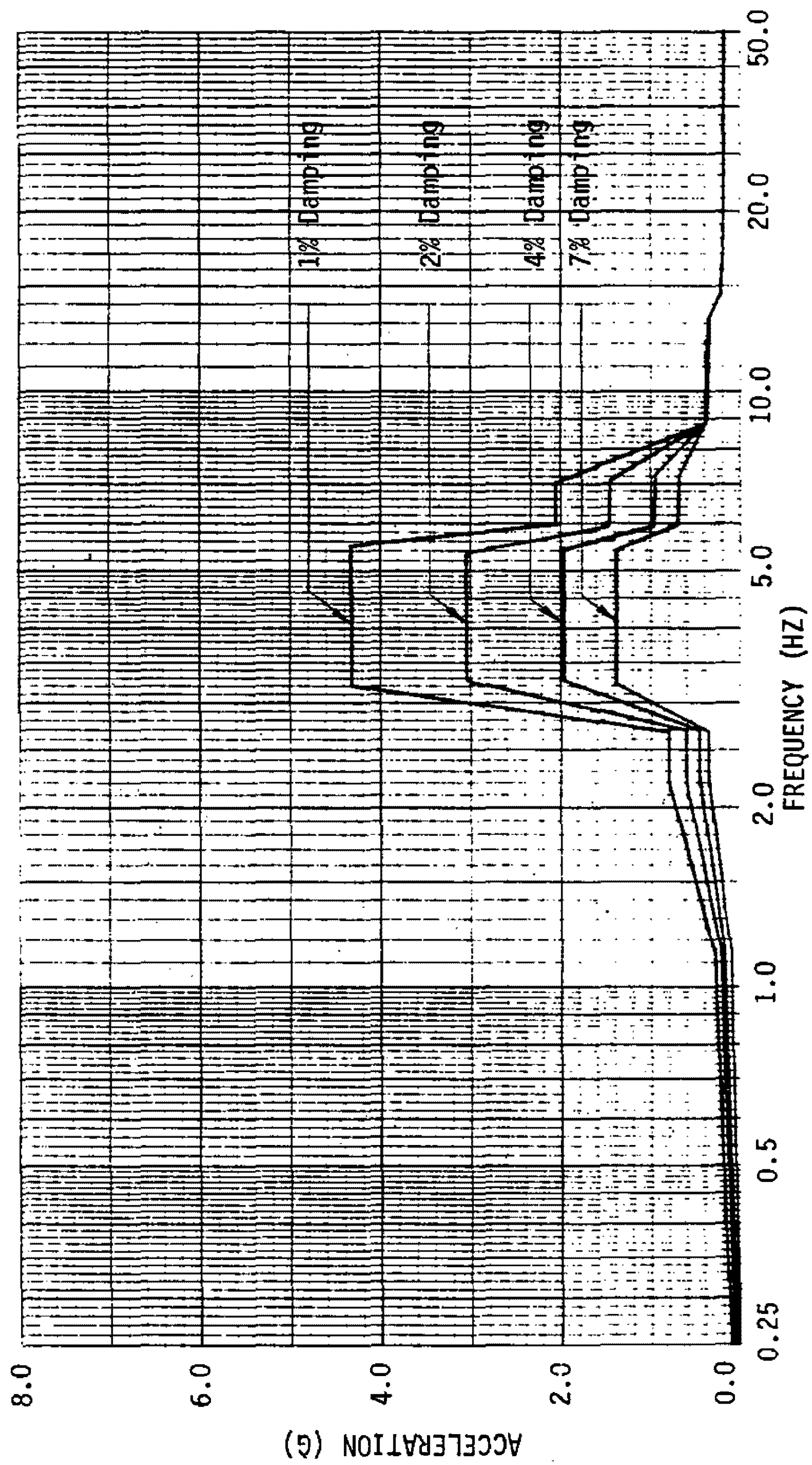
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

FLOOR RESPONSE SPECTRA
REACTOR BUILDING AT EL 201'-0"
N-S HORIZONTAL OBE CASE

UPDATED FSAR

FIGURE 3.7-129



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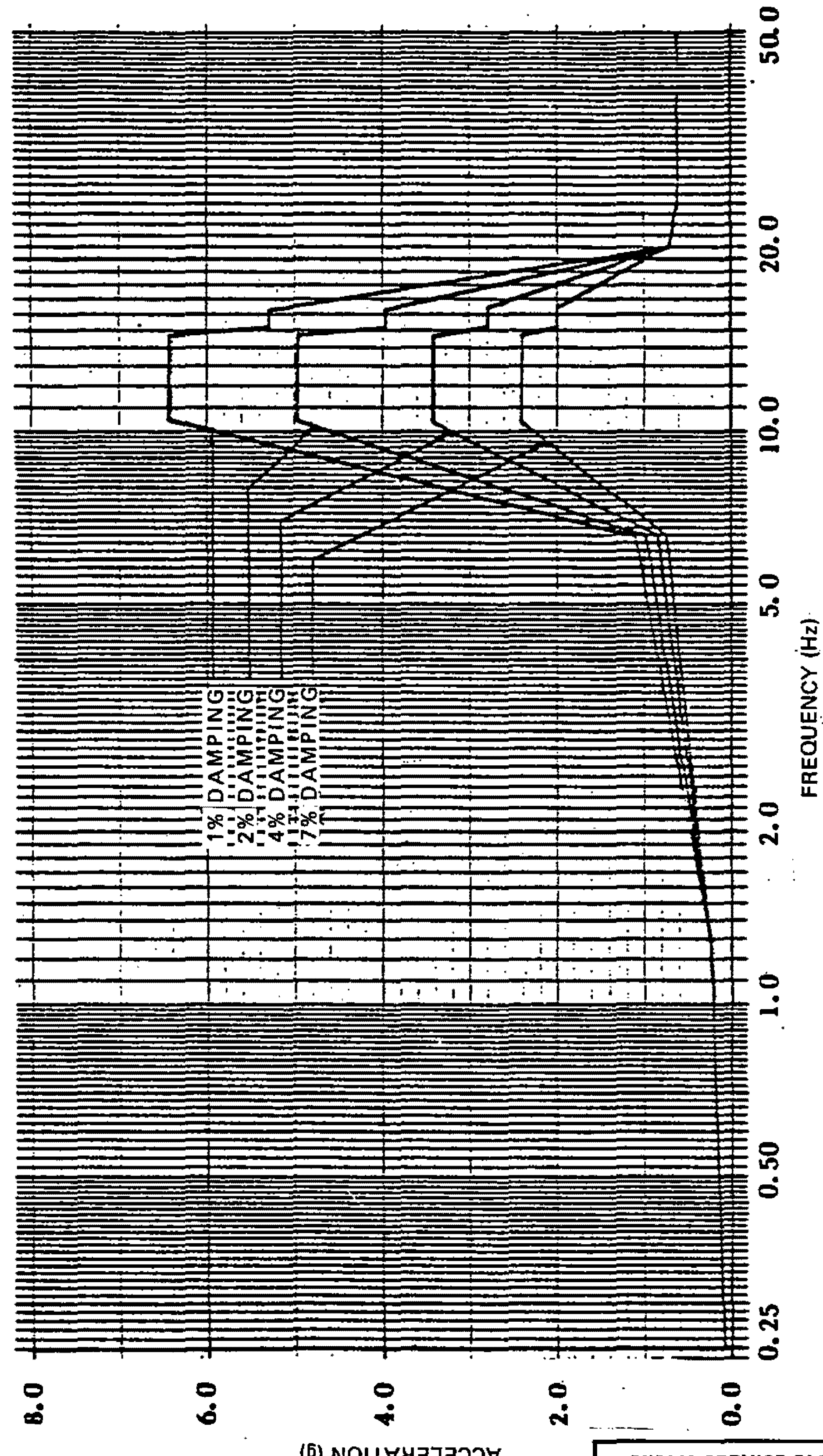
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

FLOOR RESPONSE SPECTRA
REACTOR BUILDING AT EL 201'-0"
E–W HORIZONTAL OBE CASE

UPDATED FSAR

FIGURE 3.7-130



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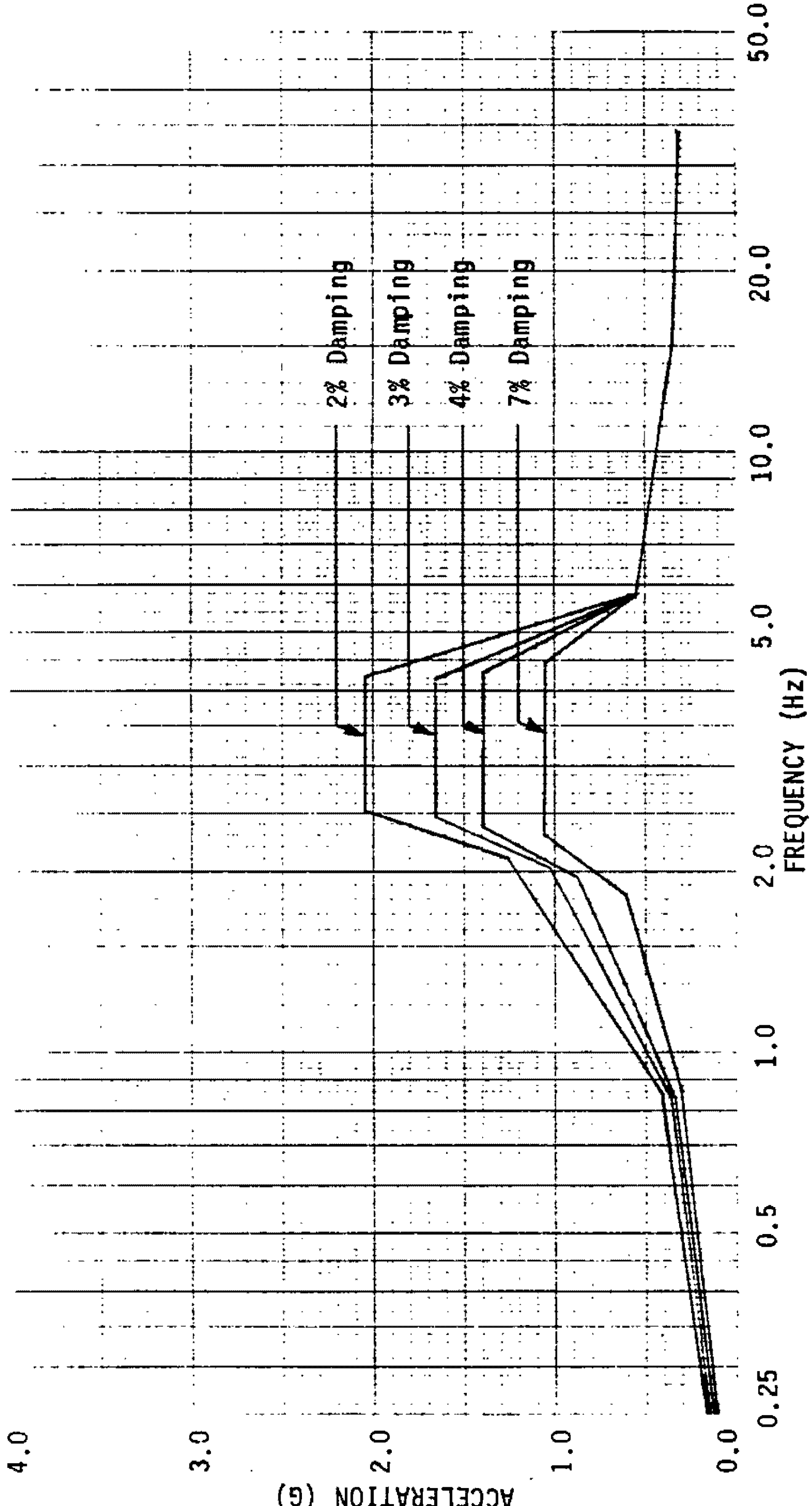
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

FLOOR RESPONSE SPECTRA
REACTOR BUILDING AT EL 201'-0"
VERTICAL OBE CASE

UPDATED FSAR

FIGURE 3.7-131



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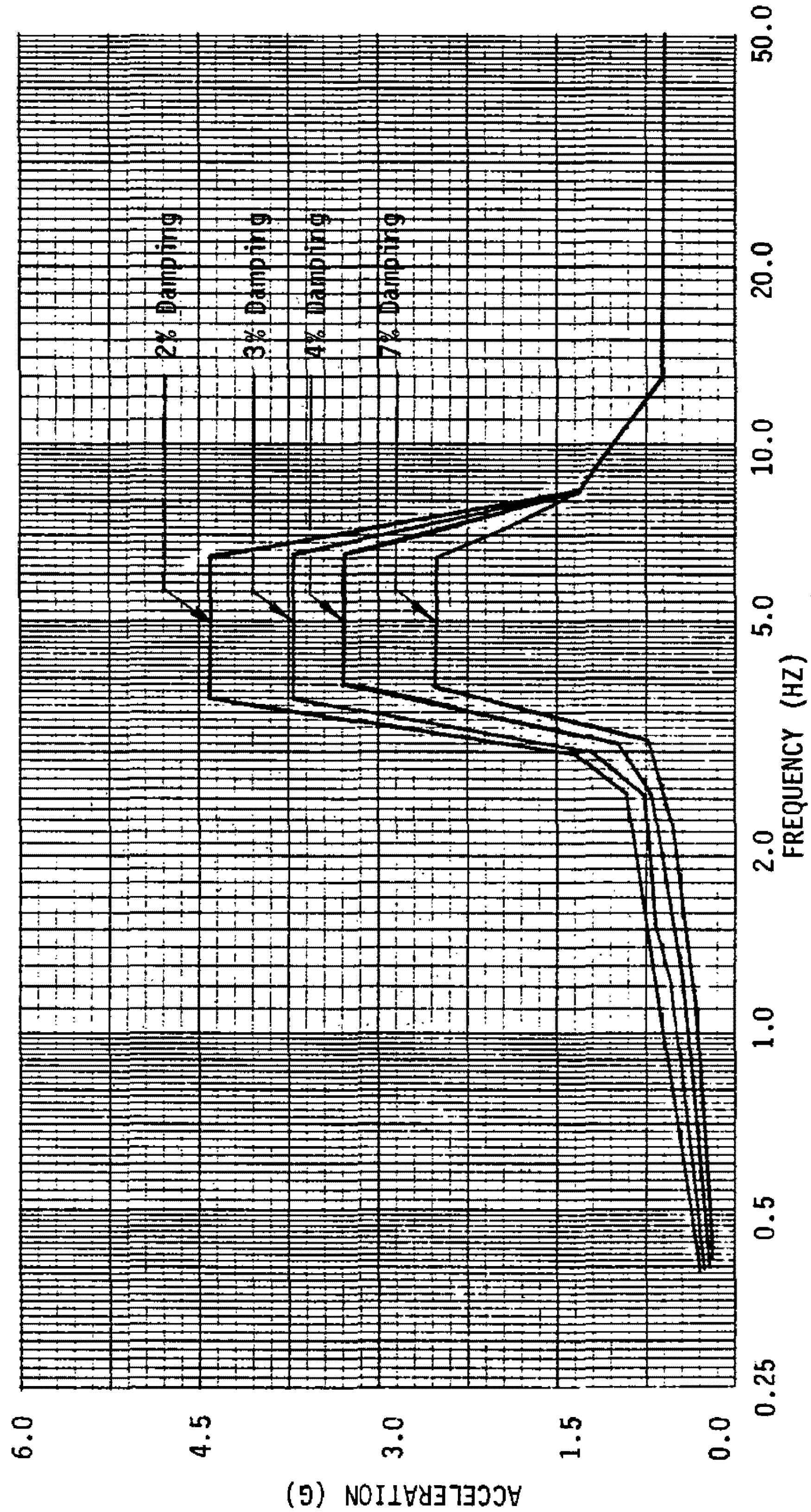
PUBLIC SERVICE ELECTRIC AND GAS COMPANY HOPE CREEK NUCLEAR GENERATING STATION
FLOOR RESPONSE SPECTRA AUXILIARY BUILDING, DIESEL GENERATOR AREA AT EL 130'-0" N–S HORIZONTAL SSE CASE
UPDATED FSAR — FIGURE 3.7-132



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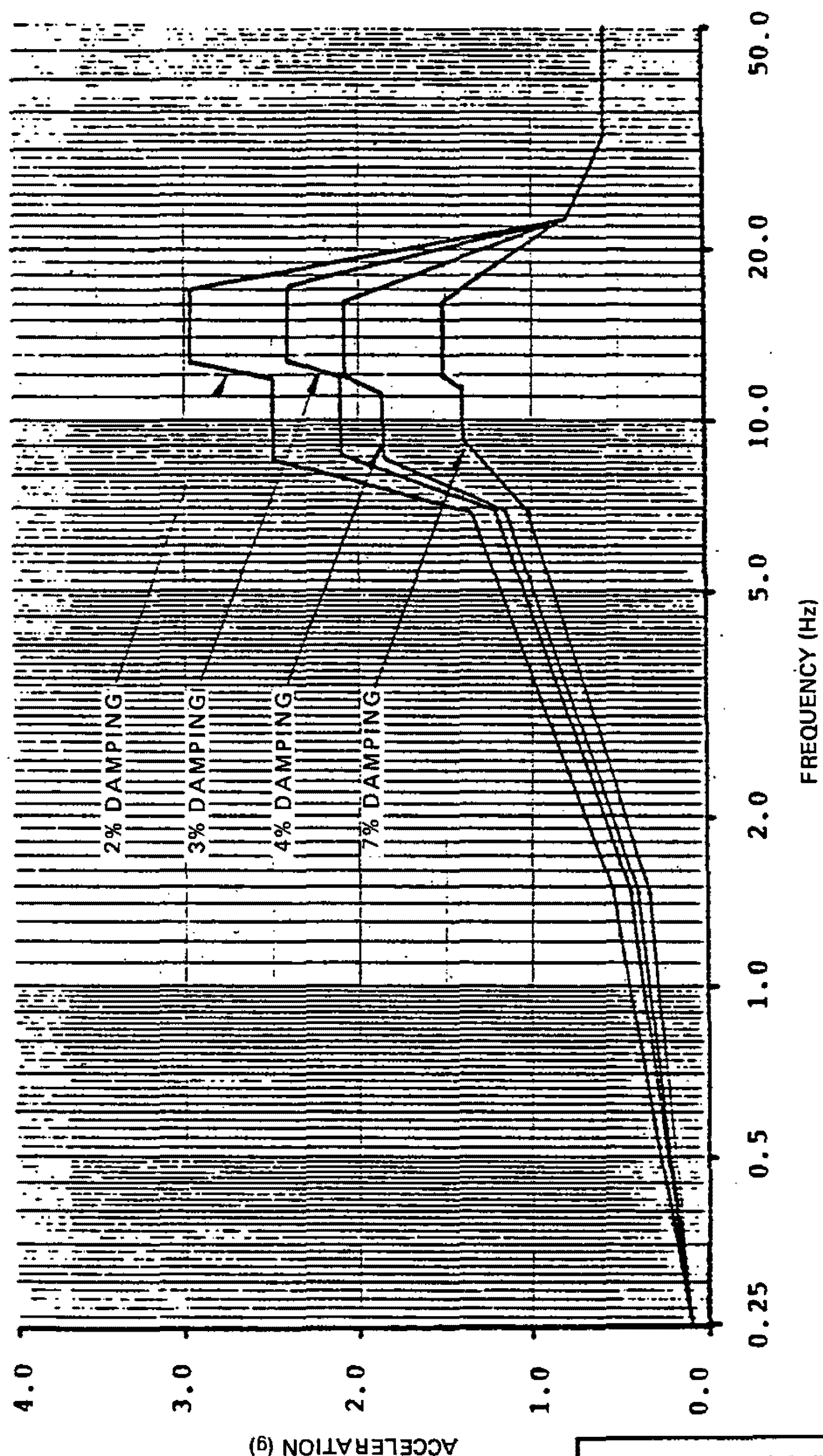
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

FLOOR RESPONSE SPECTRA
AUXILIARY BUILDING, DIESEL
GENERATOR AREA AT EL 130'-0"
E–W HORIZONTAL SSE CASE

UPDATED FSAR

FIGURE 3.7-133



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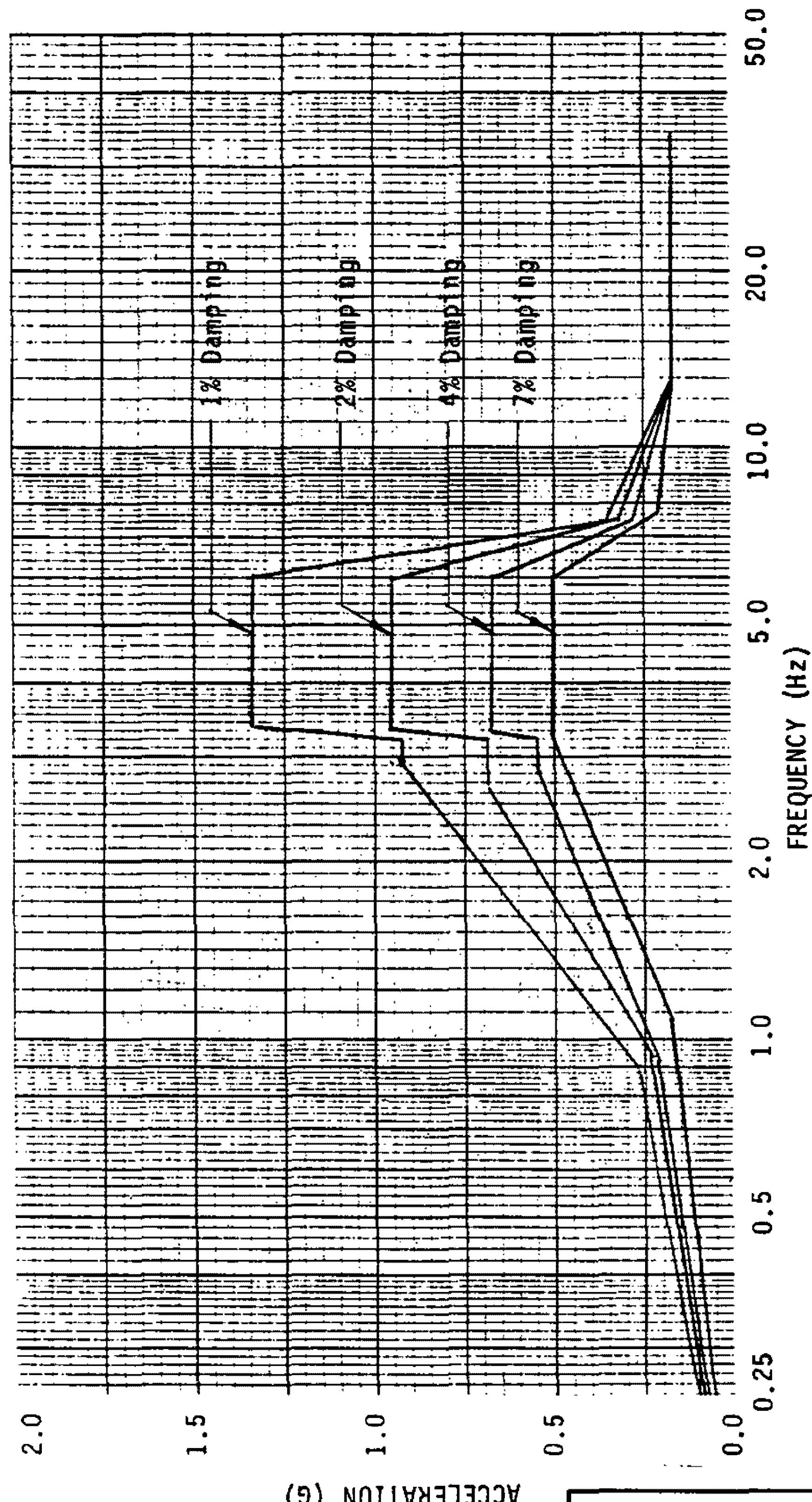
PUBLIC SERVICE ELECTRIC AND GAS COMPANY HOPE CREEK NUCLEAR GENERATING STATION
FLOOR RESPONSE SPECTRA AUXILIARY BUILDING, DIESEL GENERATOR AREA AT EL 130'-0" VERTICAL SSE CASE
UPDATED FSAR   FIGURE 3.7-134



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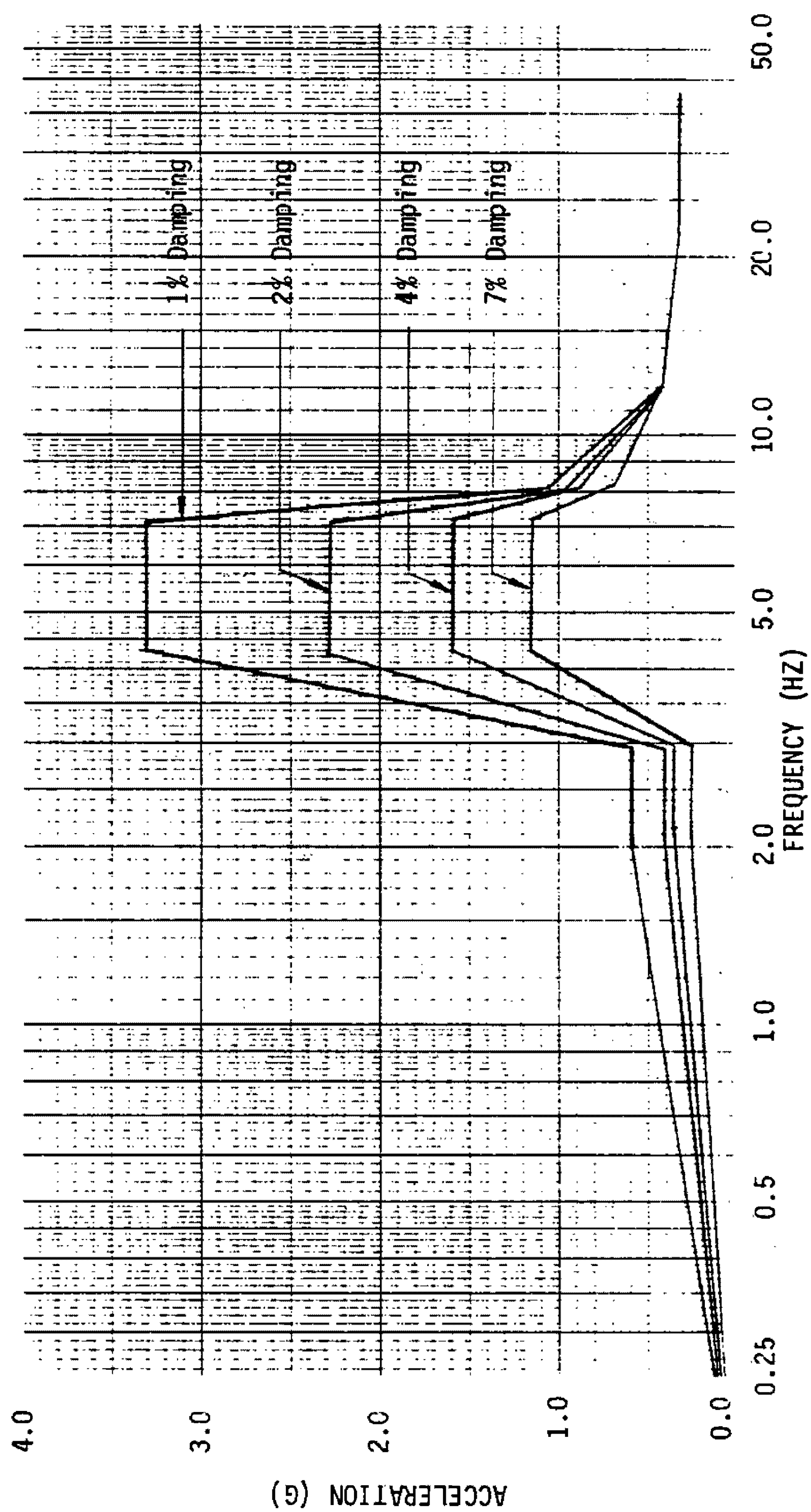
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

FLOOR RESPONSE SPECTRA
AUXILIARY BUILDING, DIESEL
GENERATOR AREA AT EL 130'-0"
N–S HORIZONTAL OBE CASE

UPDATED FSAR

FIGURE 3.7-135



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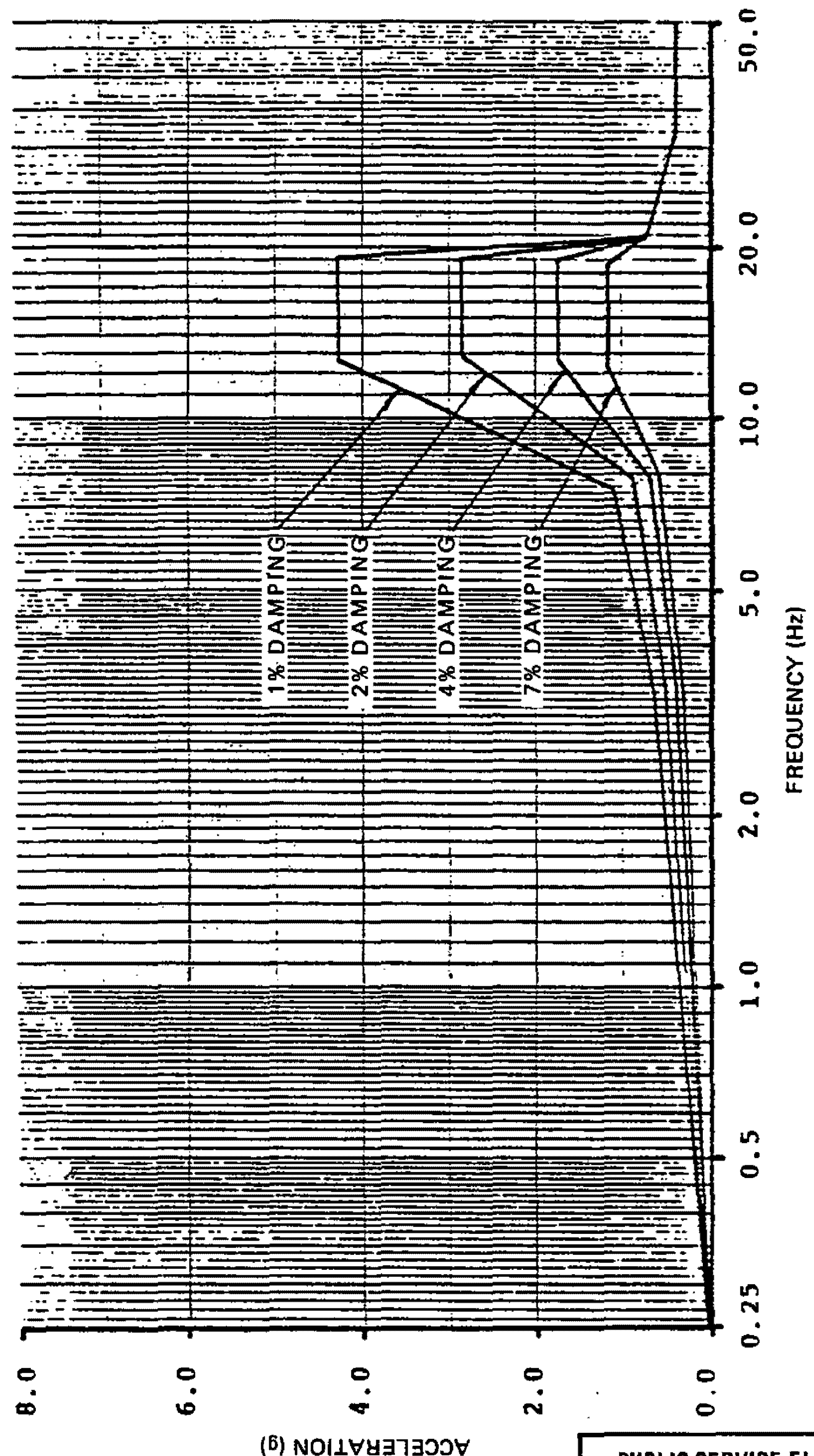
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

FLOOR RESPONSE SPECTRA
AUXILIARY BUILDING, DIESEL
GENERATOR AREA AT EL 130'-0"
E–W HORIZONTAL OBE CASE

UPDATED FSAR

FIGURE 3.7-136



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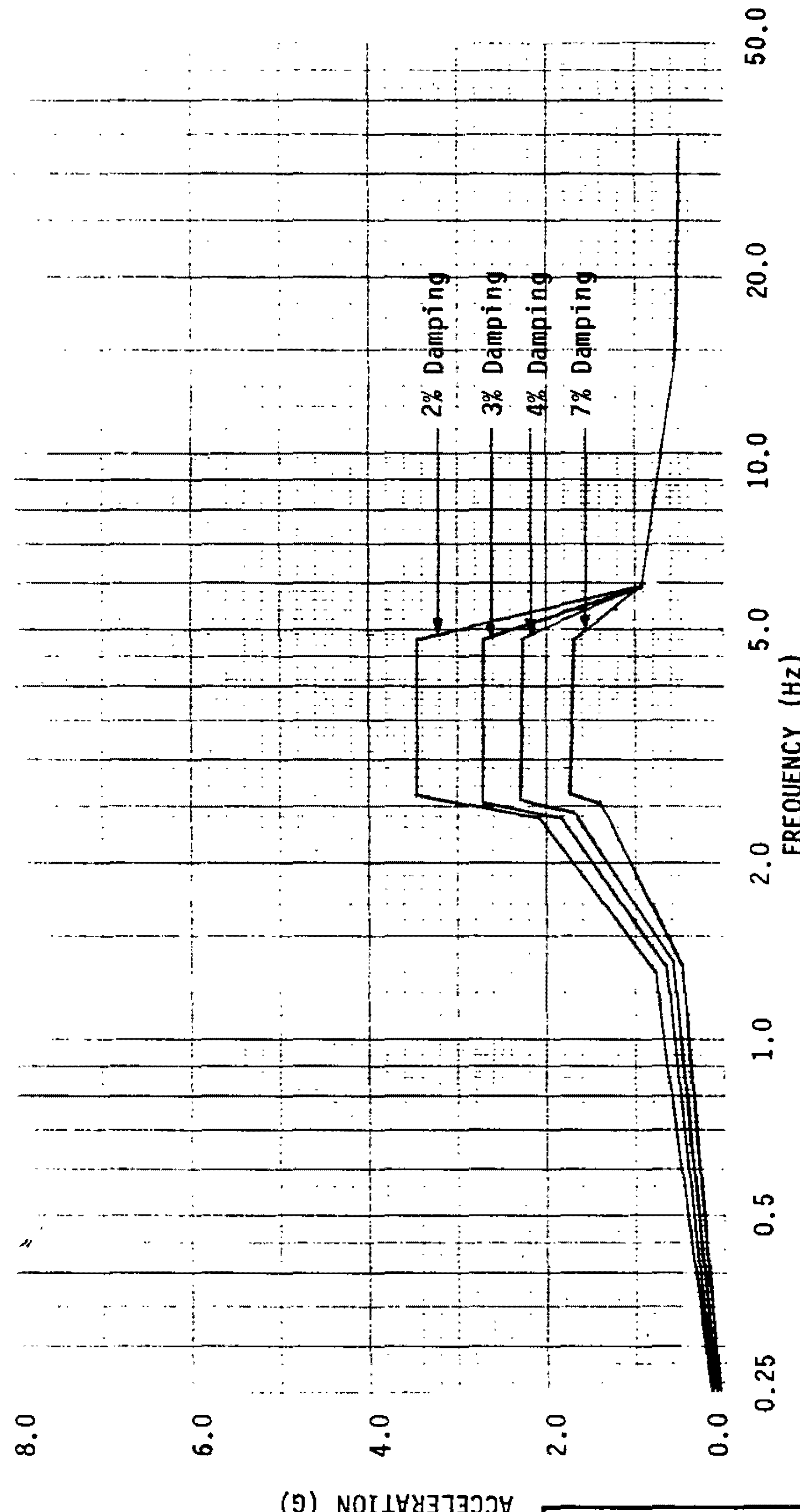
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

FLOOR RESPONSE SPECTRA
AUXILIARY BUILDING, DIESEL
GENERATOR AREA AT EL 130'-0"
VERTICAL OBE CASE

UPDATED FSAR

FIGURE 3.7-137



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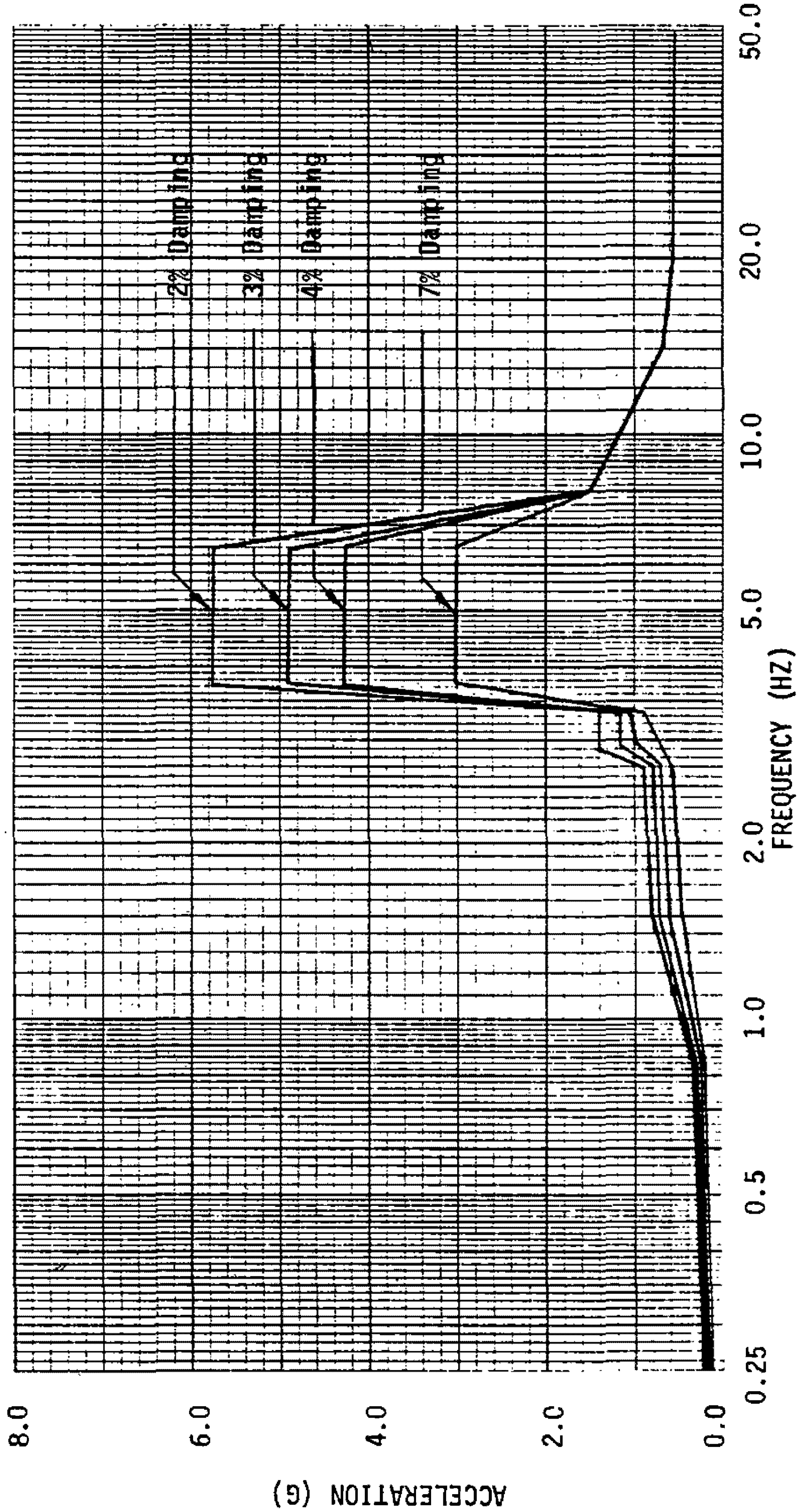
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

FLOOR RESPONSE SPECTRA
AUXILIARY BUILDING, DIESEL
GENERATOR AREA AT EL 178'-0"
N–S HORIZONTAL SSE CASE

UPDATED FSAR

FIGURE 3.7-138



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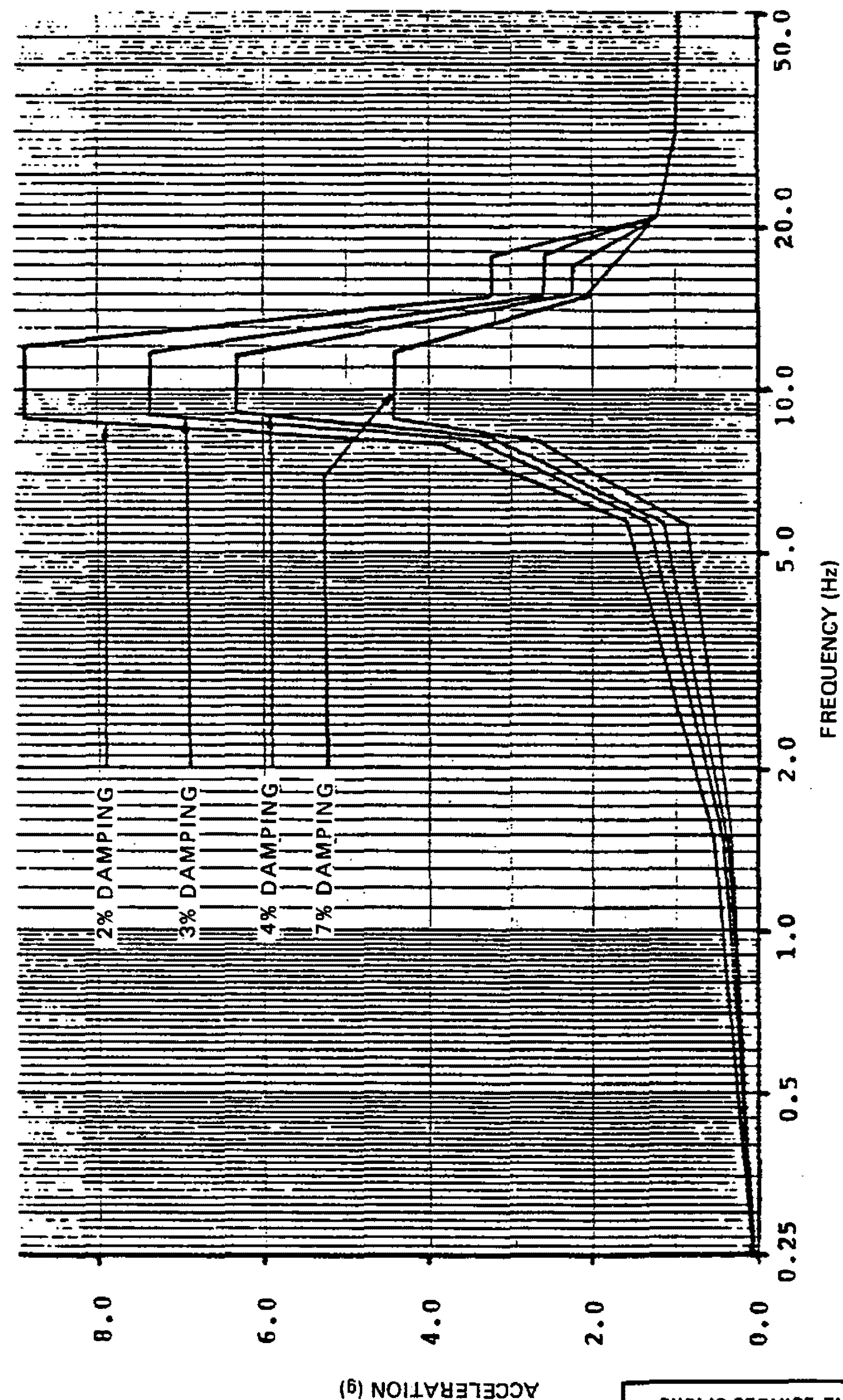
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

FLOOR RESPONSE SPECTRA
AUXILIARY BUILDING, DIESEL
GENERATOR AREA AT EL 178'-0"
E–W HORIZONTAL SSE CASE

UPDATED FSAR

FIGURE 3.7-139



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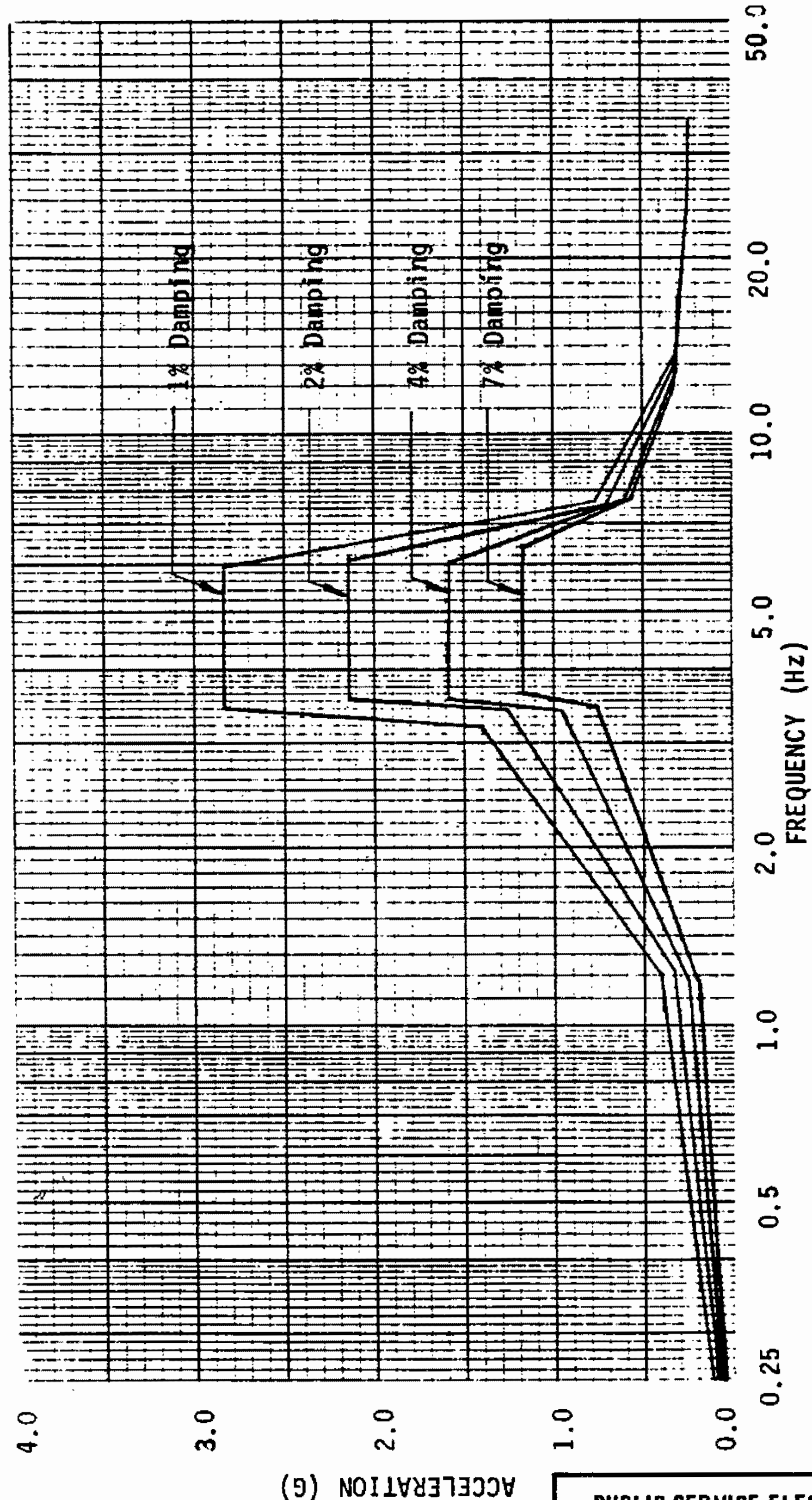
APRIL 11, 1988

PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

FLOOR RESPONSE SPECTRA
AUXILIARY BUILDING, DIESEL
GENERATOR AREA AT EL 178'-0"
VERTICAL SSE CASE

UPDATED FSAR

FIGURE 3.7-140



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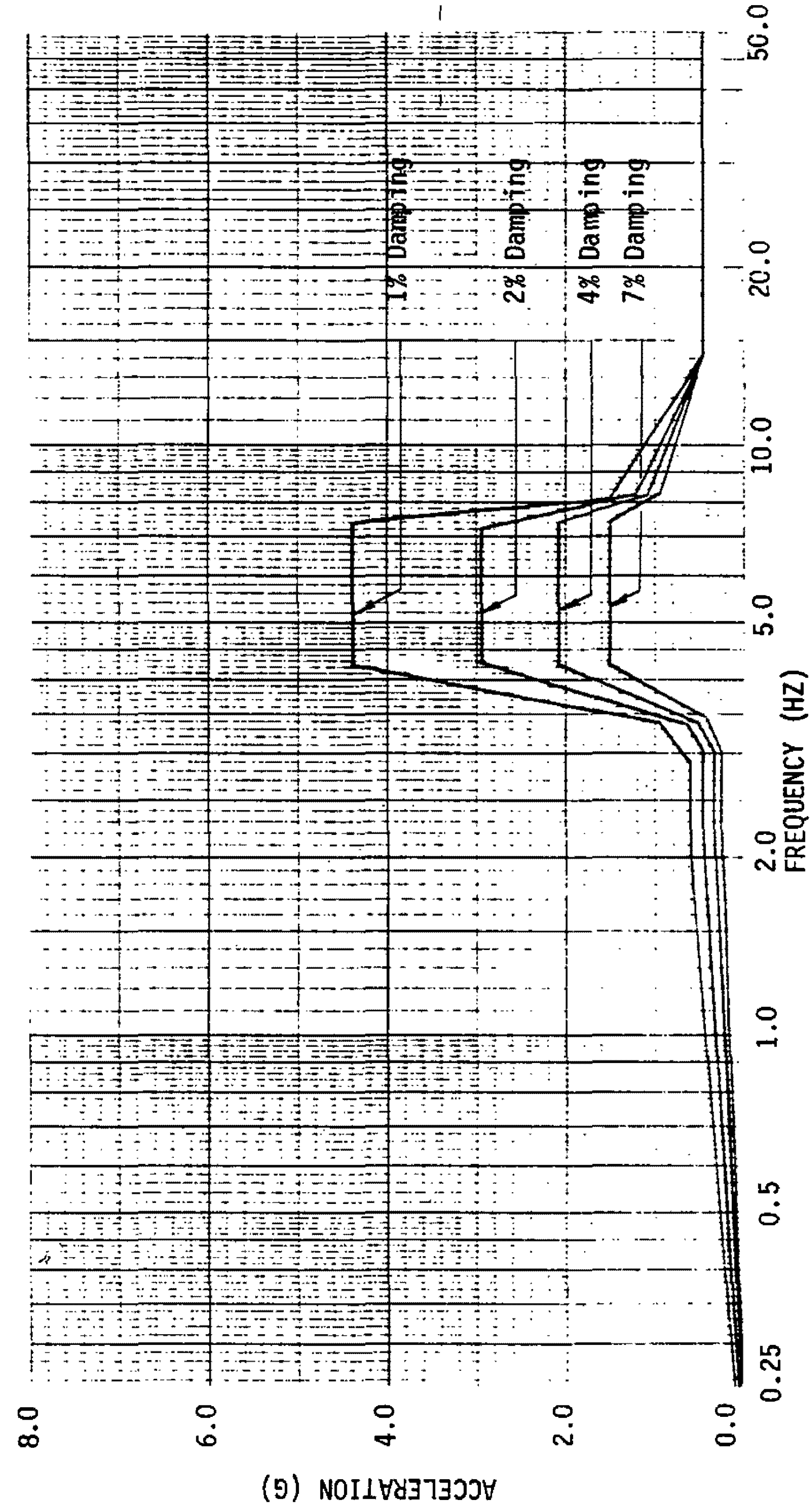
APRIL 11, 1988

PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

FLOOR RESPONSE SPECTRA
AUXILIARY BUILDING, DIESEL
GENERATOR AREA AT EL 178'-0"
N–S HORIZONTAL OBE CASE

UPDATED FSAR

FIGURE 3.7-141



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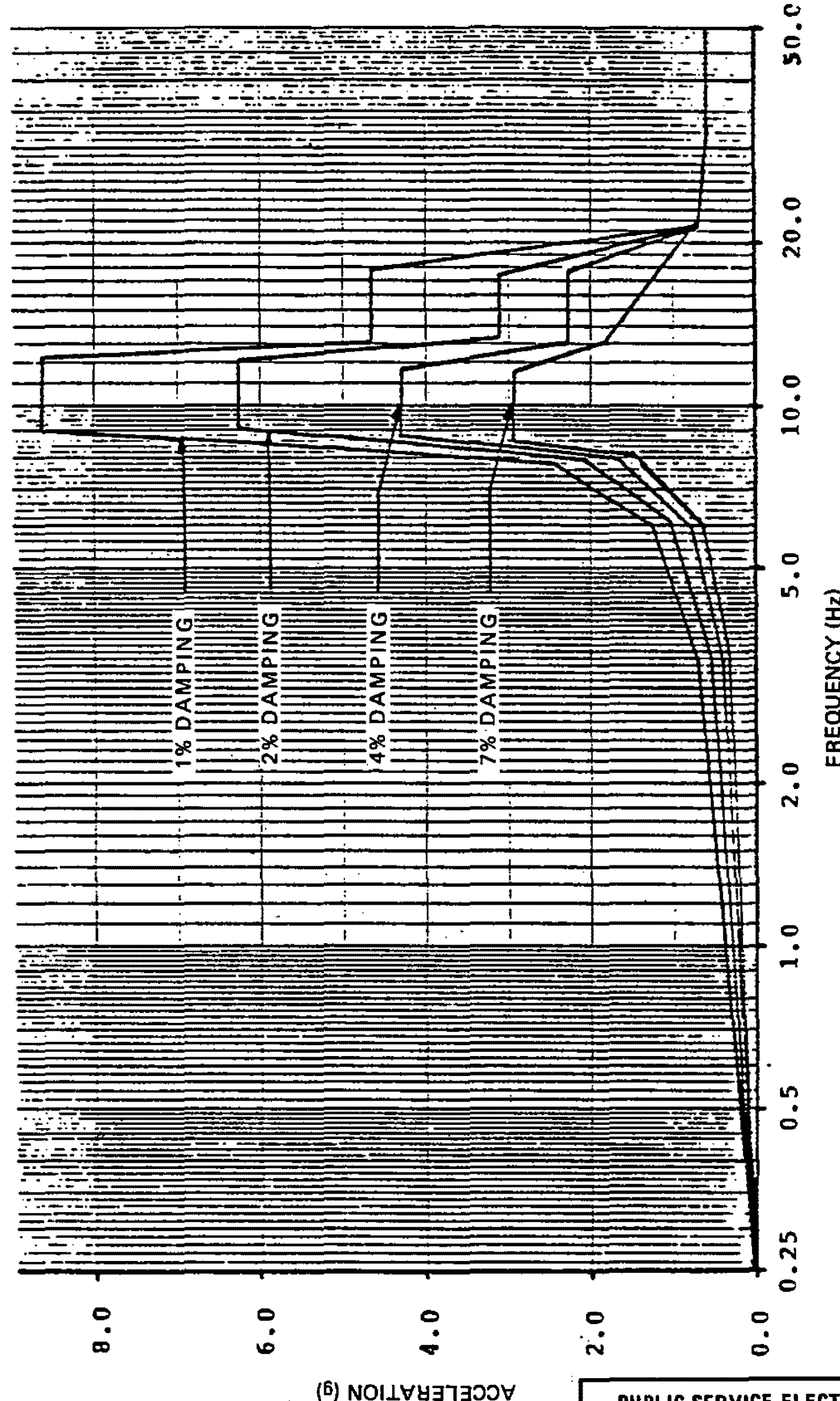
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

FLOOR RESPONSE SPECTRA
AUXILIARY BUILDING, DIESEL
GENERATOR AREA AT EL 178'-0"
E–W HORIZONTAL OBE CASE

UPDATED FSAR | FIGURE 3.7-142



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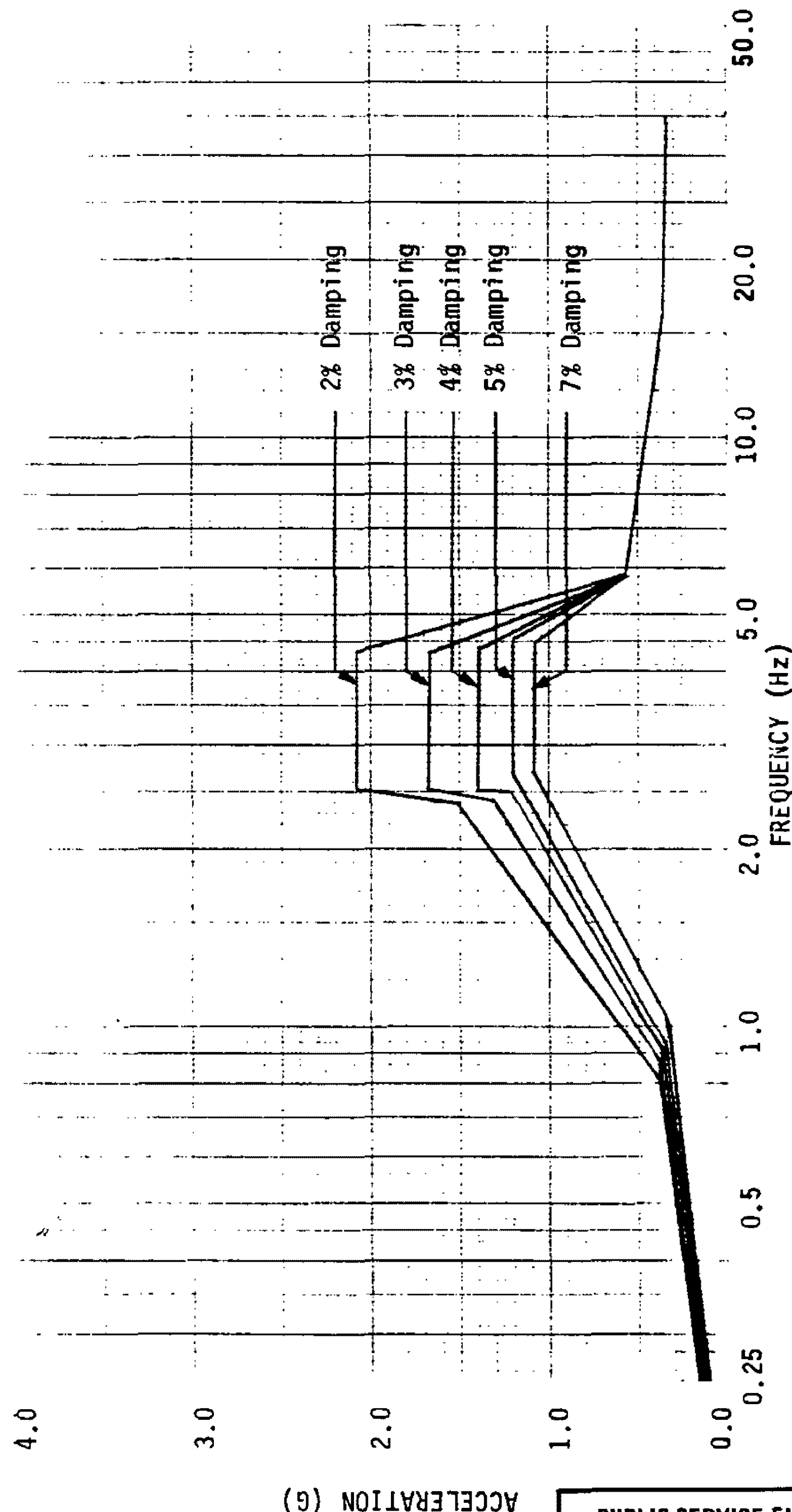
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

FLOOR RESPONSE SPECTRA
AUXILIARY BUILDING, DIESEL
GENERATOR AREA AT EL 178'-0"
VERTICAL OBE CASE

UPDATED FSAR

FIGURE 3.7-143



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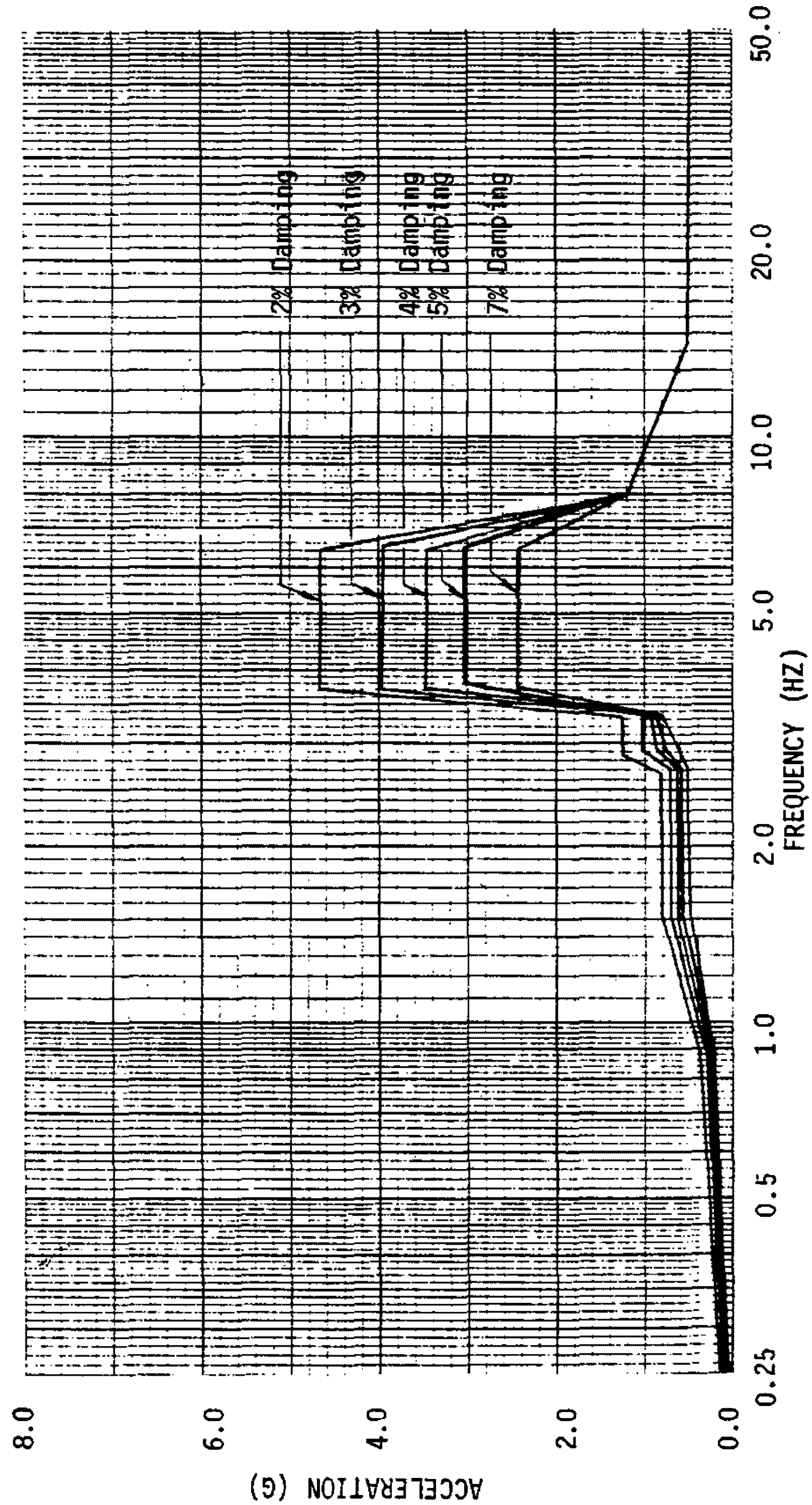
APRIL 11, 1988

PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

FLOOR RESPONSE SPECTRA
AUXILIARY BUILDING, CONTROL
AREA AT EL 137'-0"
N–S HORIZONTAL SSE CASE

UPDATED FSAR

FIGURE 3.7-144



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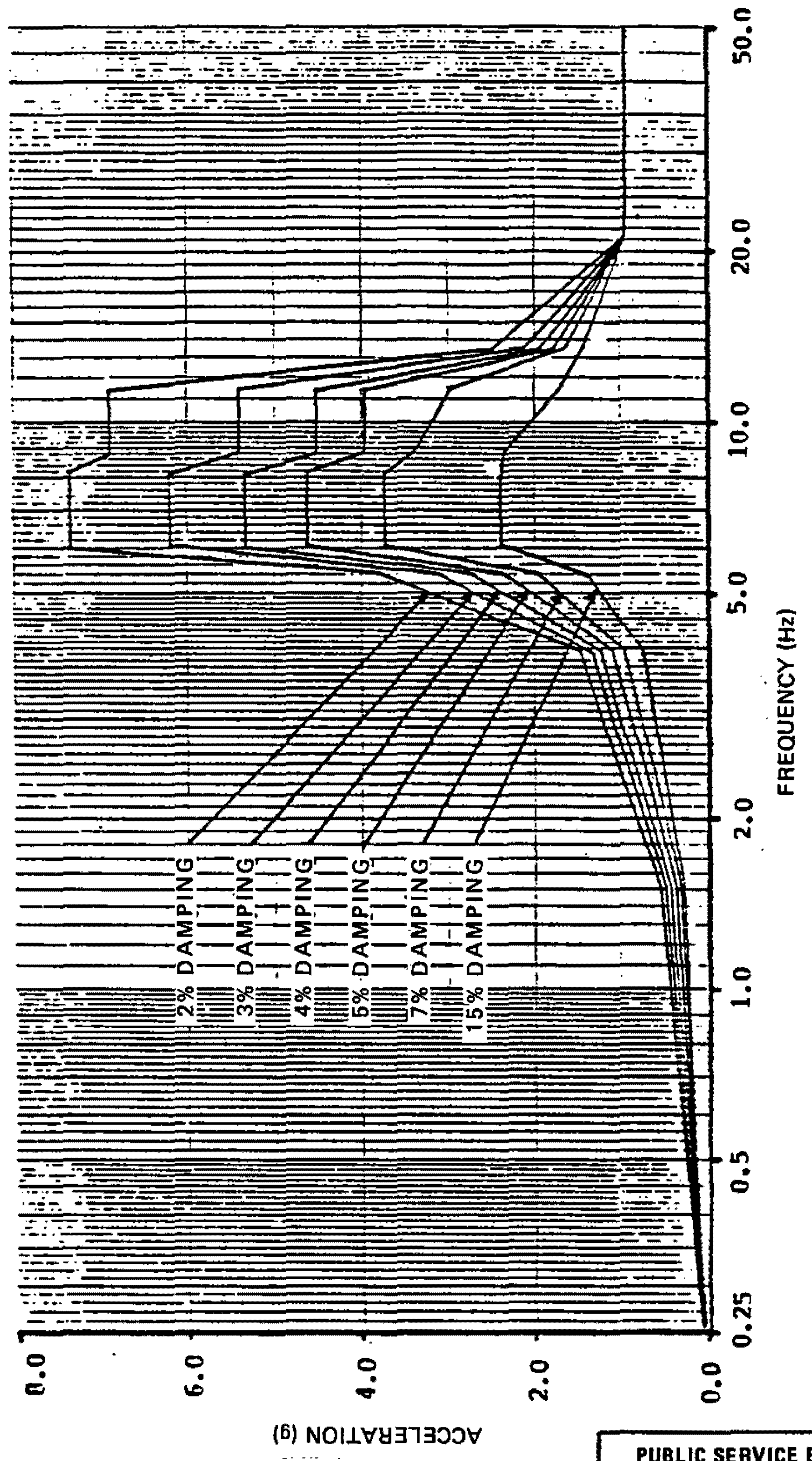
APRIL 11, 1988

PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

FLOOR RESPONSE SPECTRA
AUXILIARY BUILDING, CONTROL
AREA AT EL 137'-0"
E–W HORIZONTAL SSE CASE

UPDATED FSAR

FIGURE 3.7-145



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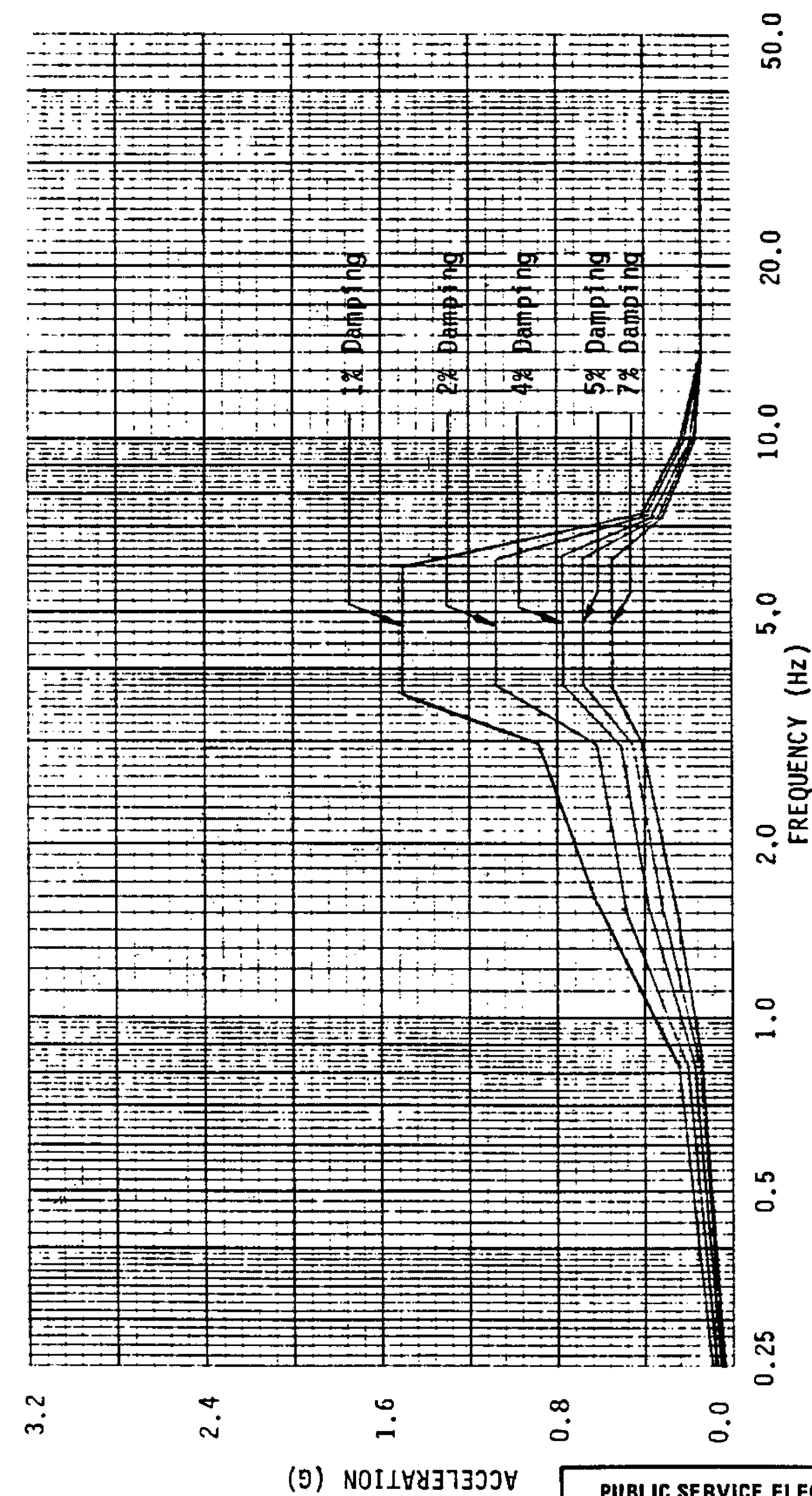
APRIL 11, 1988

PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

FLOOR RESPONSE SPECTRA
AUXILIARY BUILDING, CONTROL
AREA AT EL 137'-0"
VERTICAL SSE CASE

UPDATED FSAR

FIGURE 3.7-146



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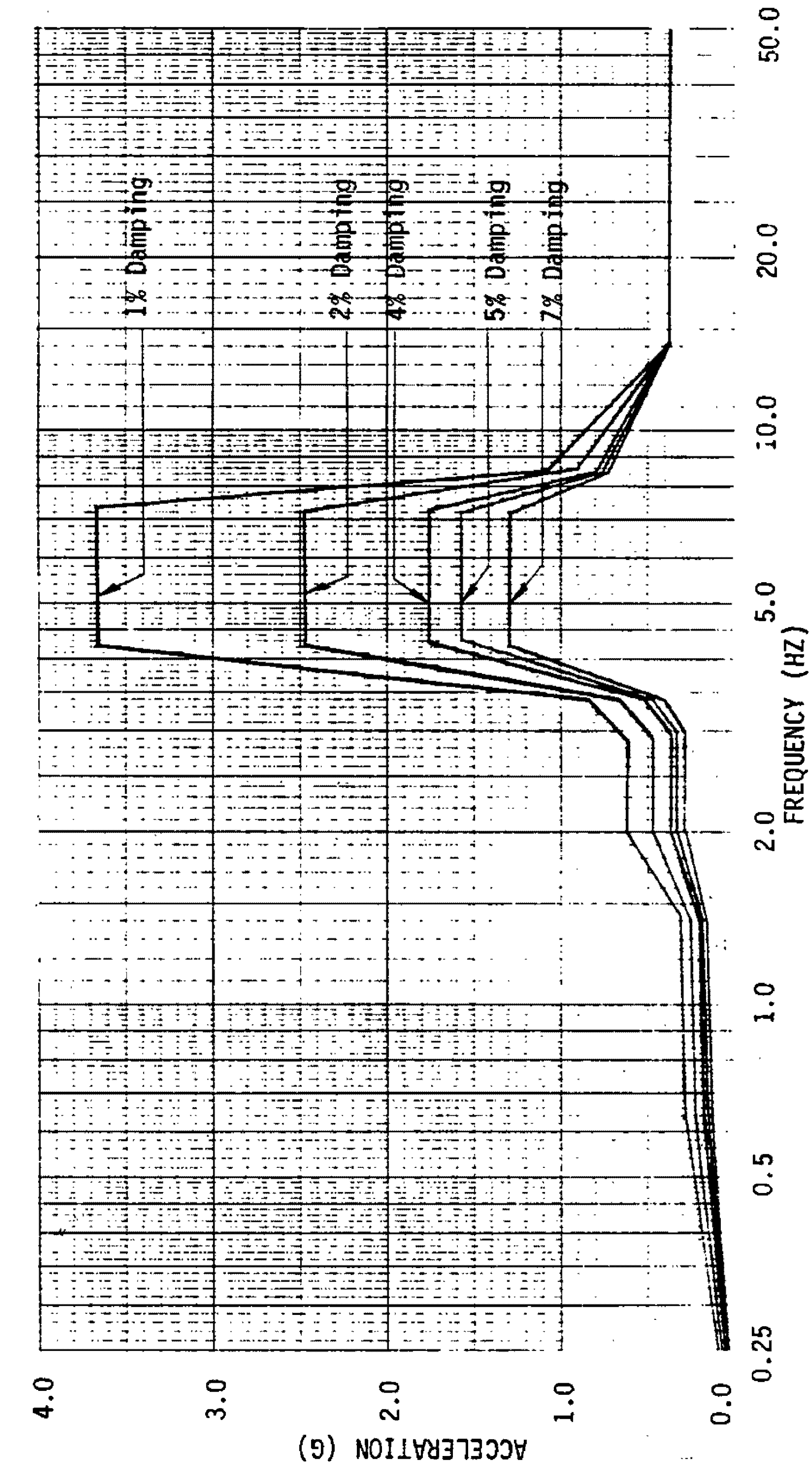
APRIL 11, 1988

PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

FLOOR RESPONSE SPECTRA
AUXILIARY BUILDING, CONTROL
AREA AT EL 137'-0"
N–S HORIZONTAL OBE CASE

UPDATED FSAR

FIGURE 3.7-147



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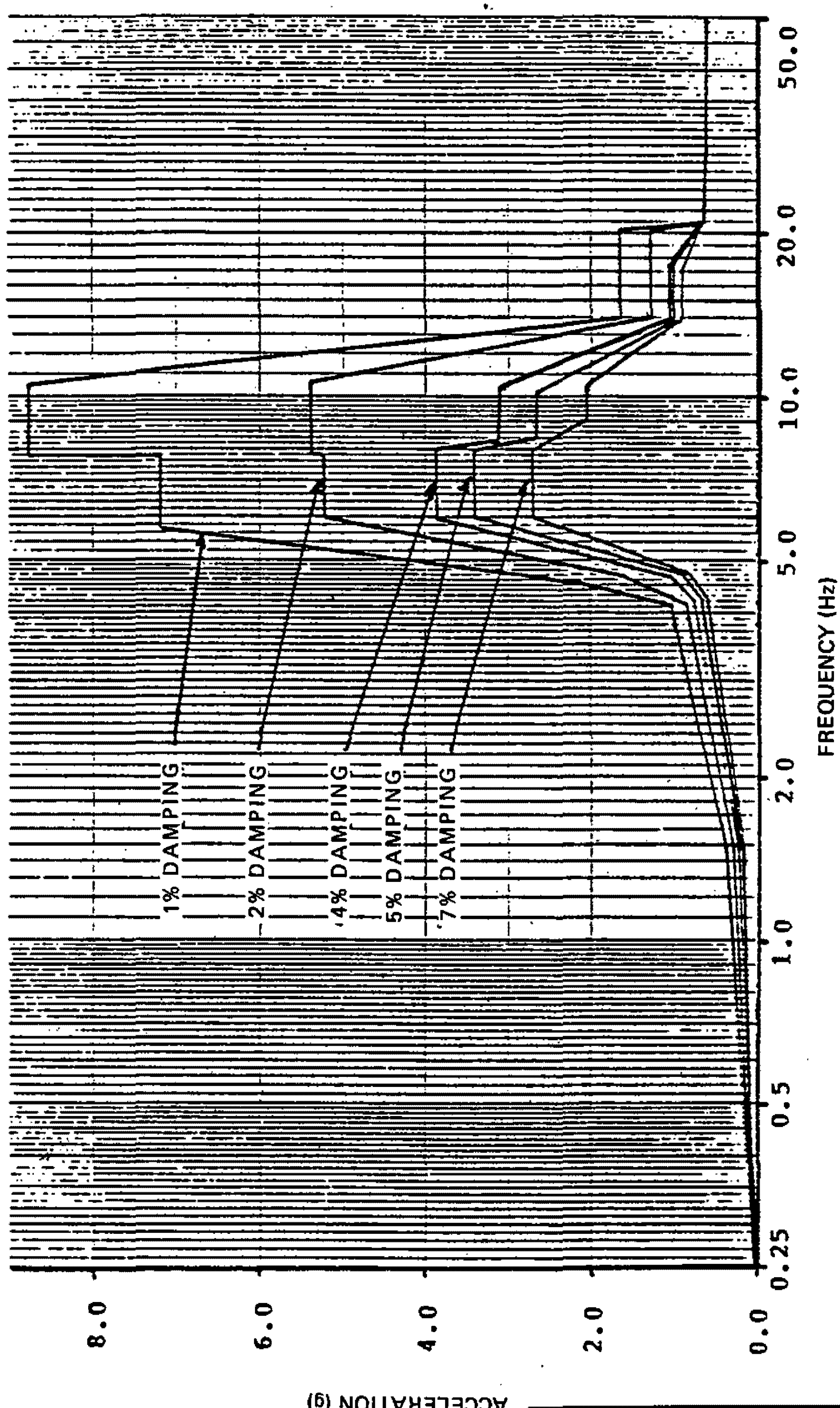
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

FLOOR RESPONSE SPECTRA
AUXILIARY BUILDING, CONTROL
AREA AT EL 137'-0"
E–W HORIZONTAL OBE CASE

UPDATED FSAR

FIGURE 3.7-148



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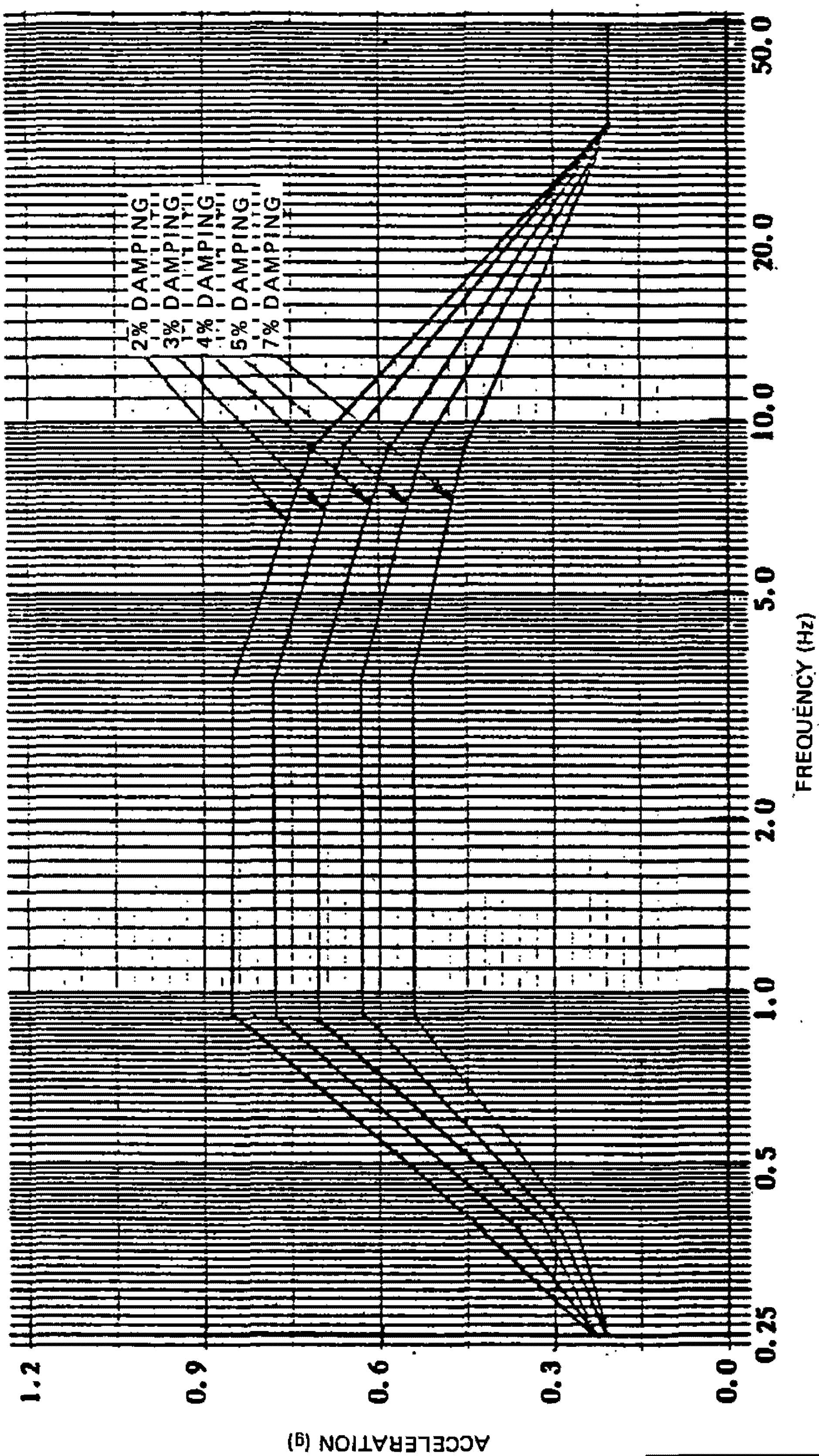
APRIL 11, 1988

PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

FLOOR RESPONSE SPECTRA
AUXILIARY BUILDING, CONTROL
AREA AT EL 137'-0"
VERTICAL OBE CASE

UPDATED FSAR

FIGURE 3.7-149



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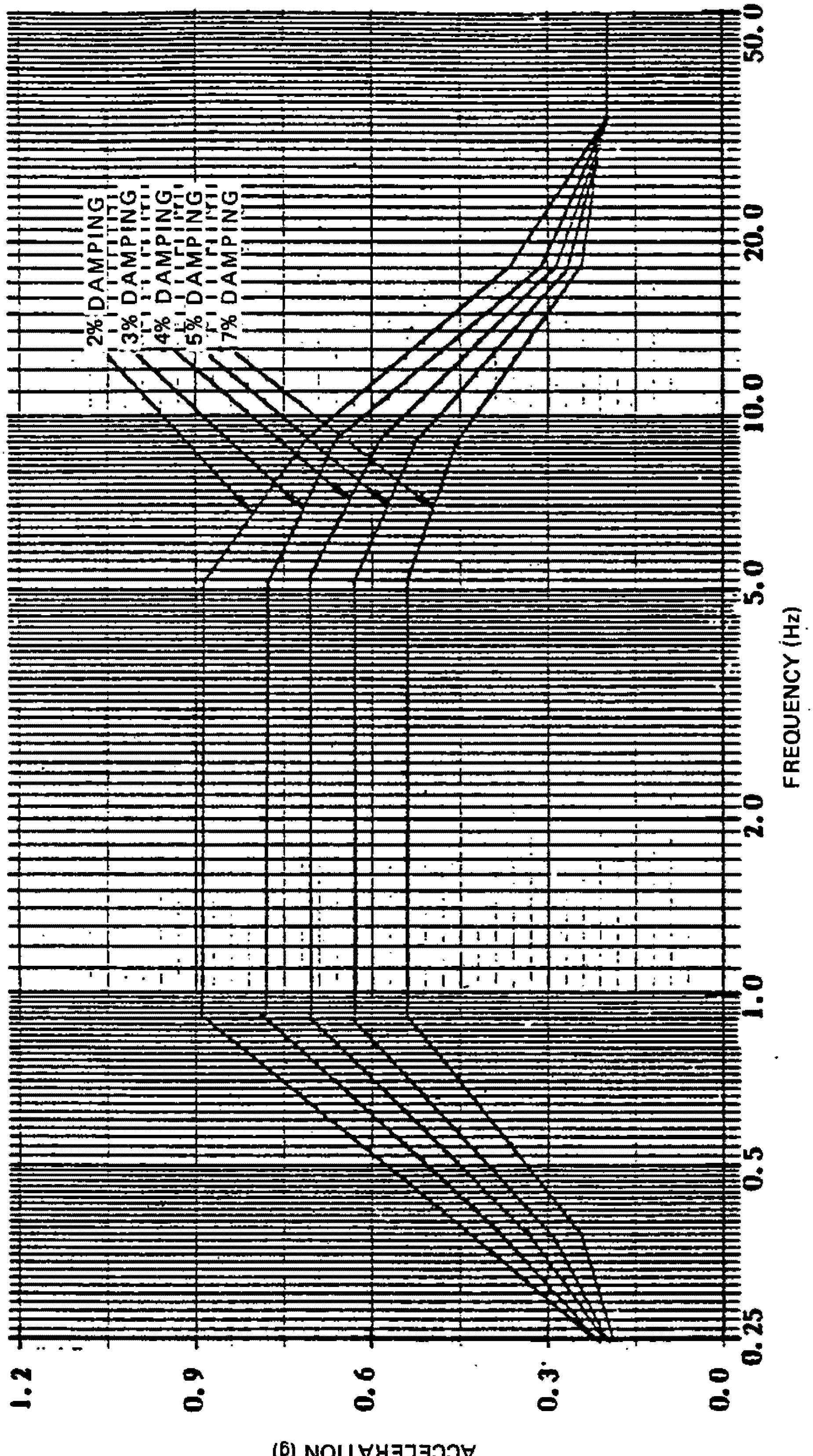
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

FLOOR RESPONSE SPECTRA
INTAKE STRUCTURE AT EL 122'-0"
N–S HORIZONTAL SSE CASE

UPDATED FSAR

FIGURE 3.7-150



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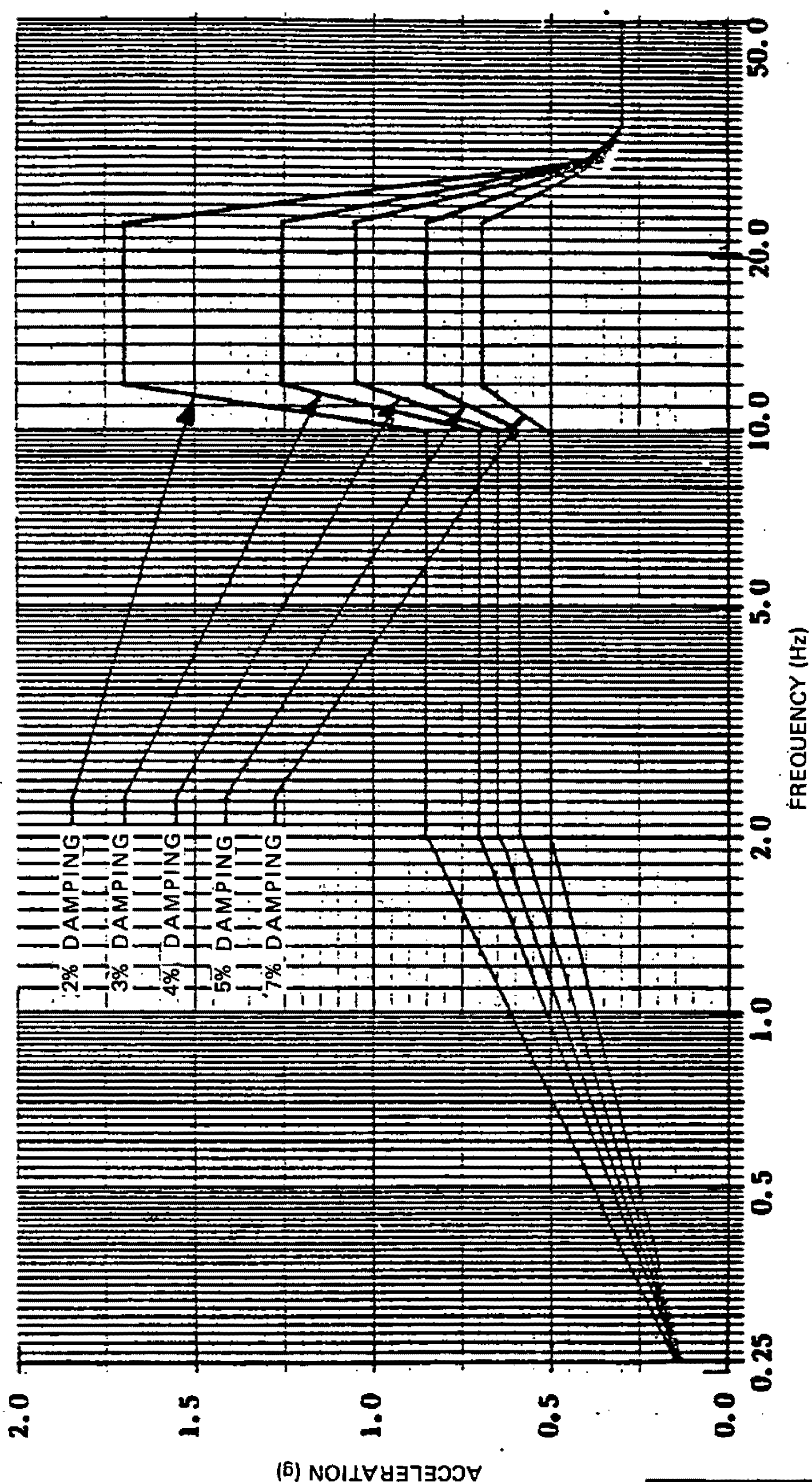
APRIL 11, 1988

PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

FLOOR RESPONSE SPECTRA
INTAKE STRUCTURE AT EL 122'-0"
E–W HORIZONTAL SSE CASE

UPDATED FSAR

FIGURE 3.7-151



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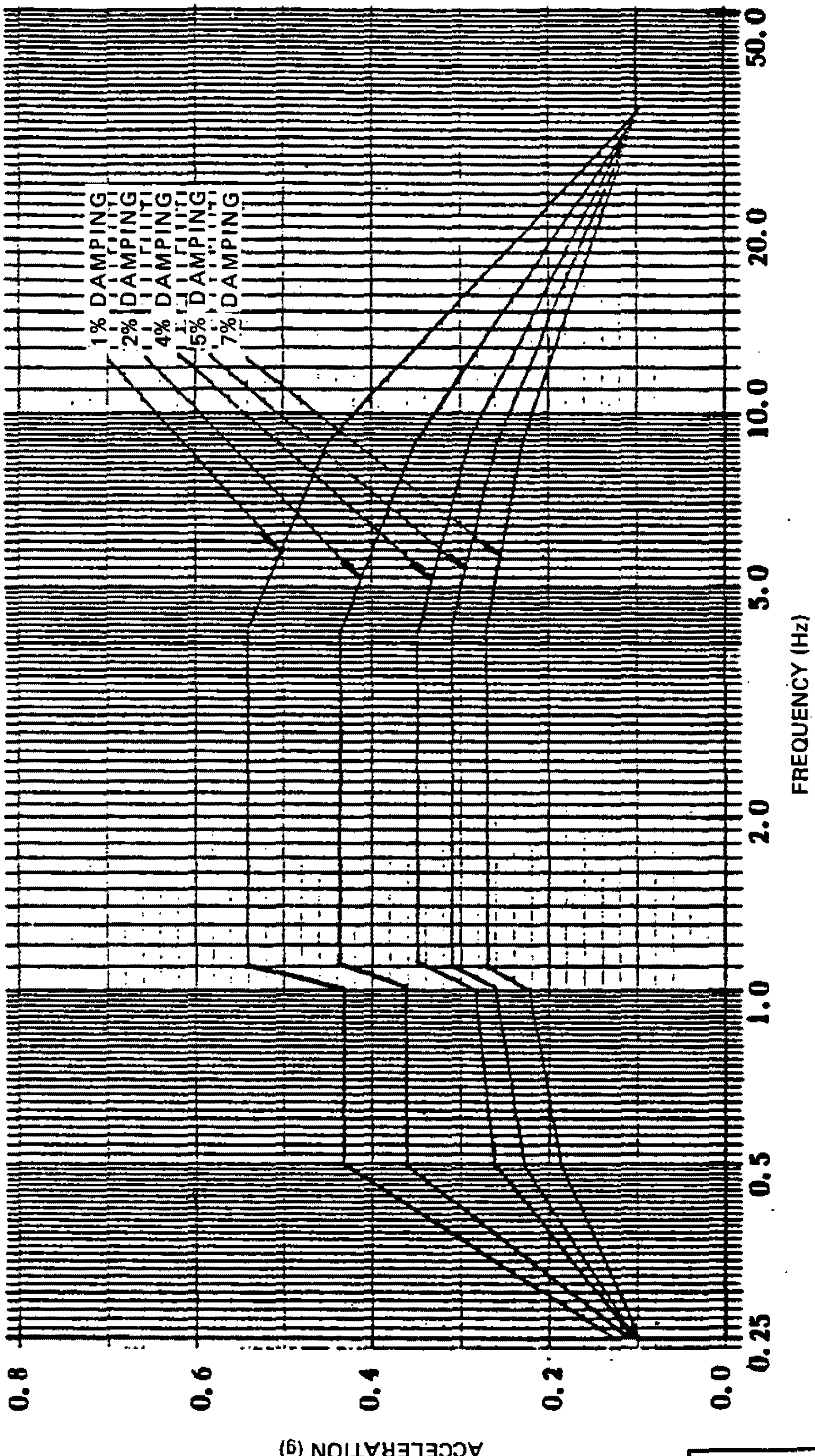
APRIL 11, 1988

PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

FLOOR RESPONSE SPECTRA
INTAKE STRUCTURE AT EL 122'-0"
VERTICAL SSE CASE

UPDATED FSAR

FIGURE 3.7-152



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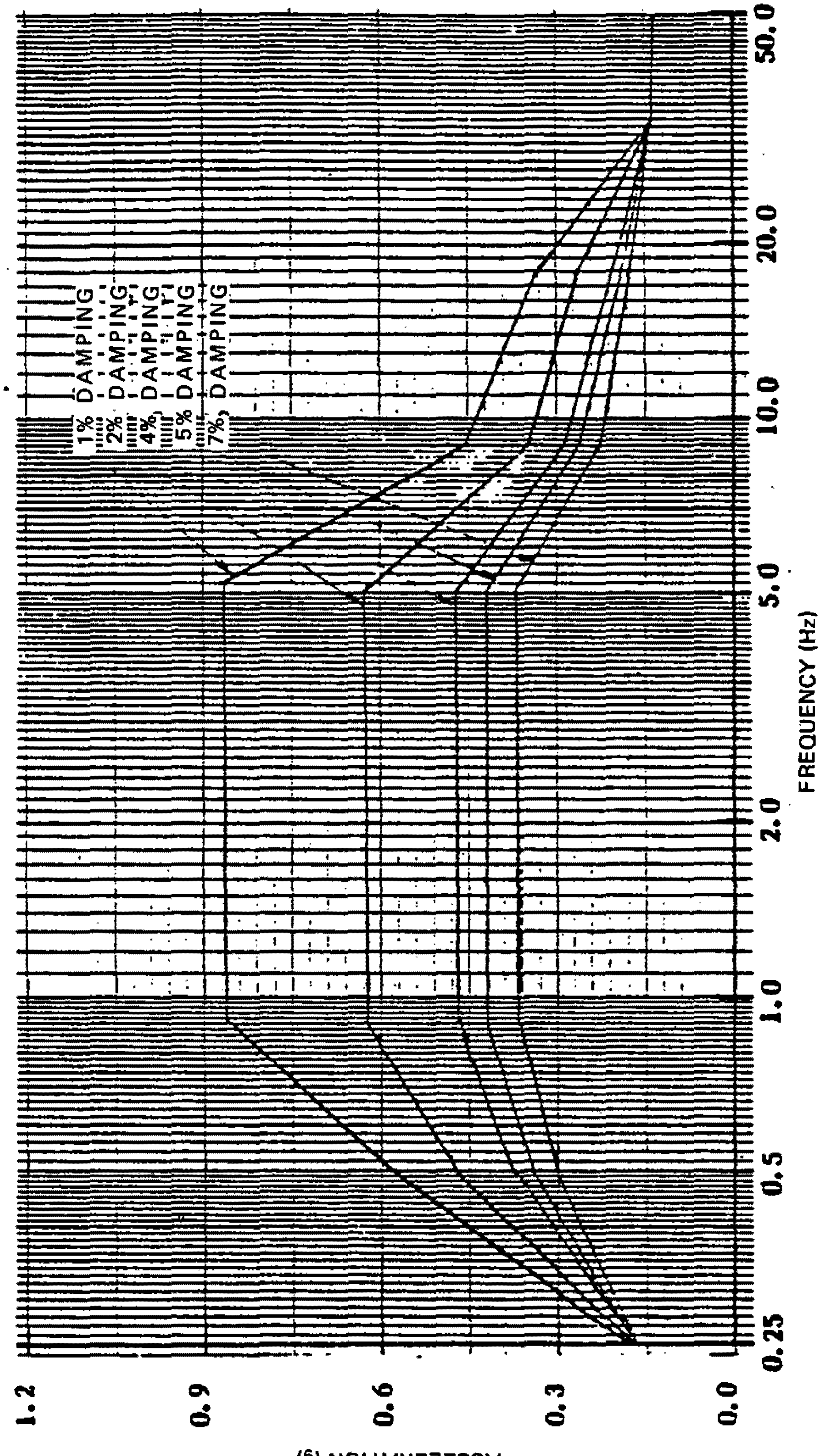
APRIL 11, 1988

PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

FLOOR RESPONSE SPECTRA
INTAKE STRUCTURE AT EL 122'-0"
N–S HORIZONTAL OBE CASE

UPDATED FSAR

FIGURE 3.7-153



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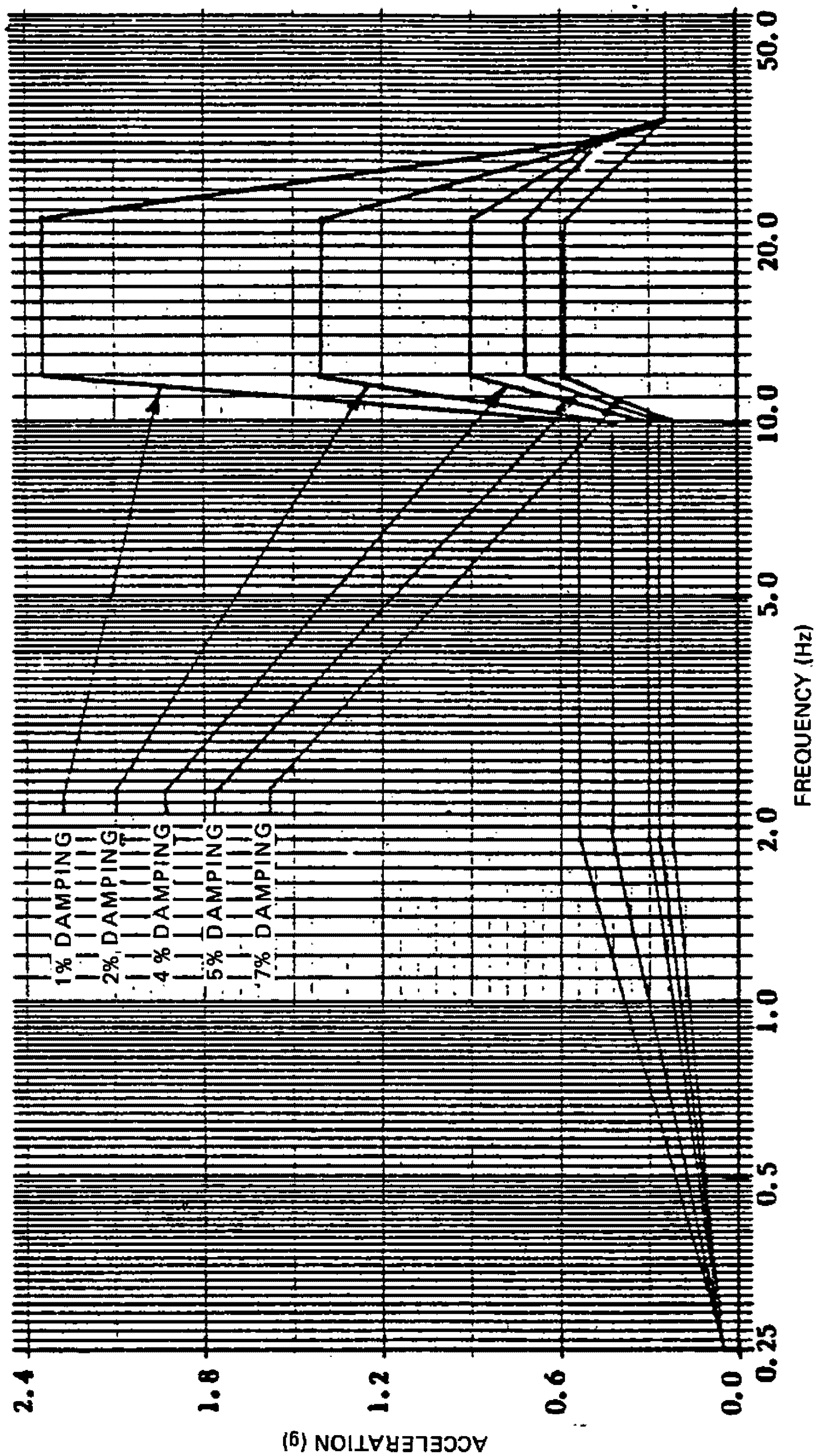
APRIL 11, 1988

PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

FLOOR RESPONSE SPECTRA
INTAKE STRUCTURE AT EL 122'-0"
E–W HORIZONTAL OBE CASE

UPDATED FSAR

FIGURE 3.7-154



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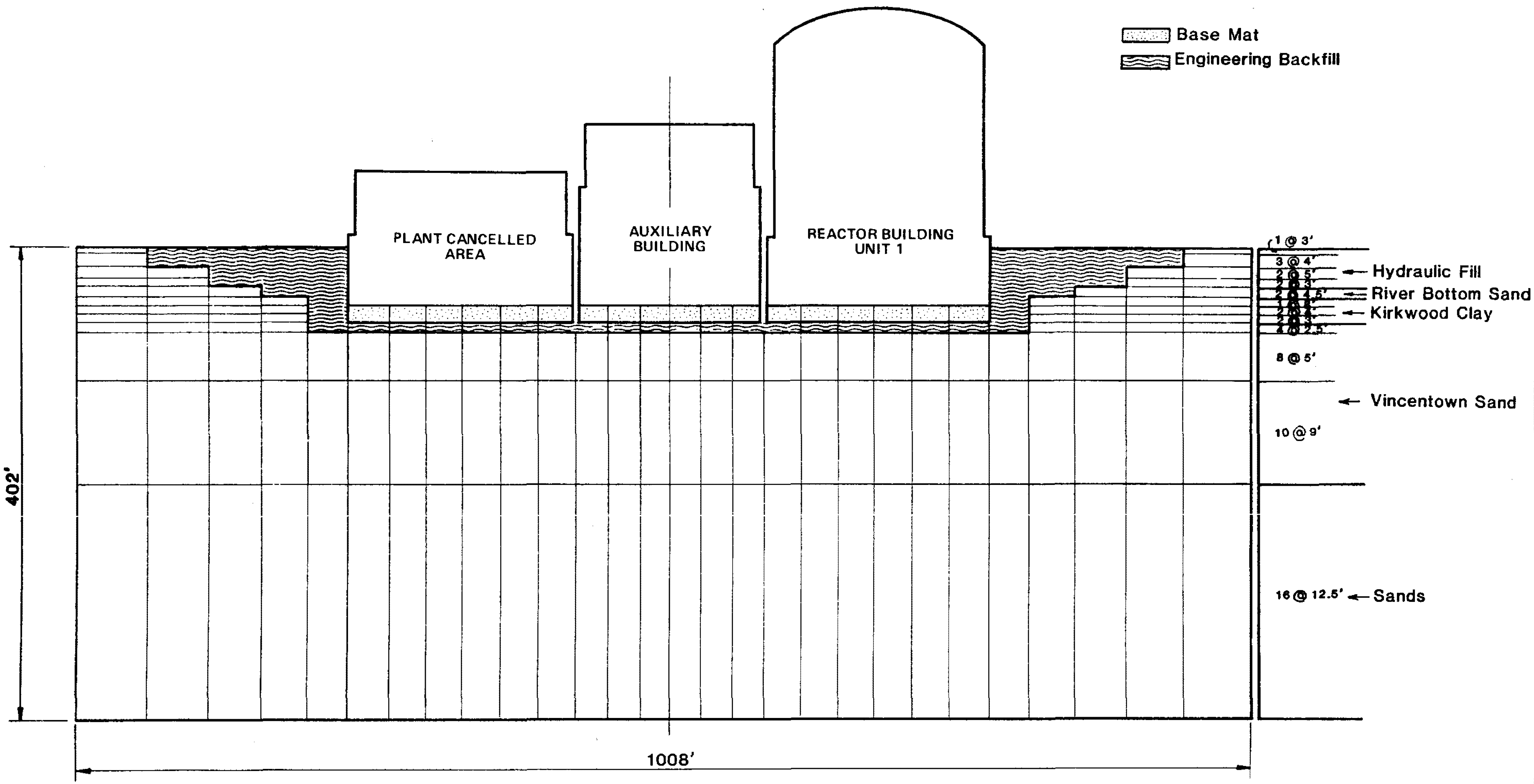
APRIL 11, 1988

PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

FLOOR RESPONSE SPECTRA
INTAKE STRUCTURE AT EL 122'-0"
VERTICAL OBE CASE

UPDATED FSAR

FIGURE 3.7-155



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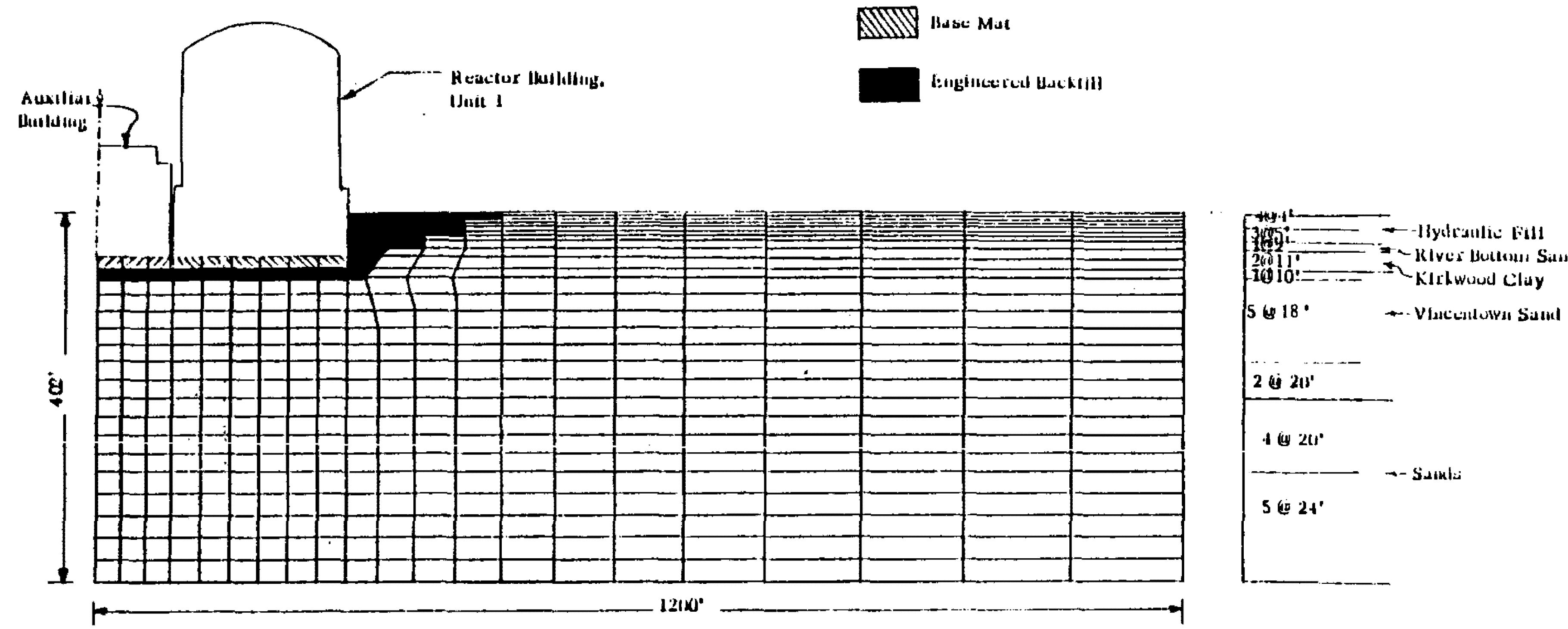
APRIL 11, 1988

PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

HORIZONTAL SOIL-STRUCTURE INTERACTION ANALYSIS MODEL N-S CROSS SECTION POWER BLOCK AREA

UPDATED FSAR

FIGURE 3.7-156



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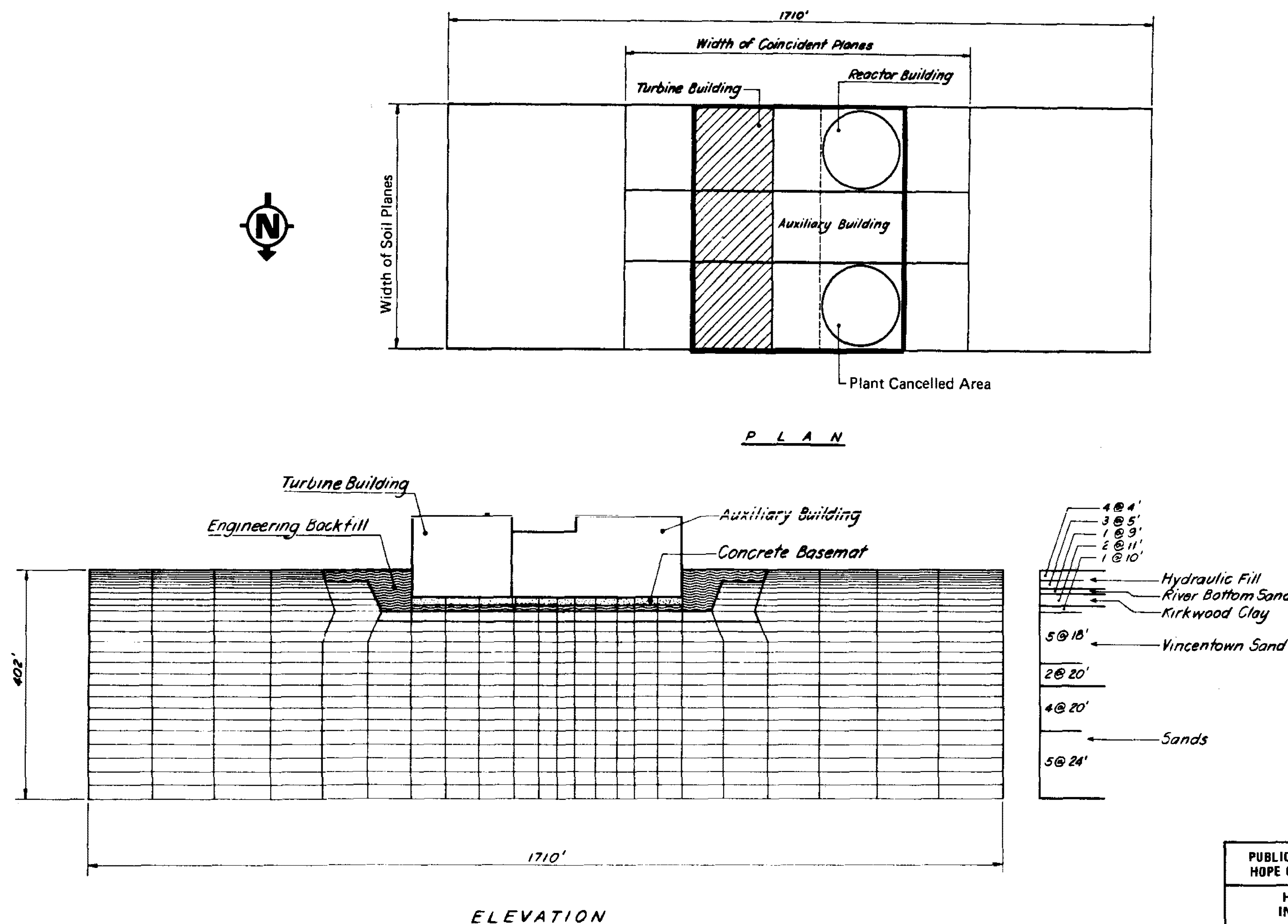
APRIL 11, 1988

PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

VERTICAL SOIL-STRUCTURE
INTERACTION ANALYSIS MODEL
N-S CROSS SECTION
POWER BLOCK AREA

UPDATED FSAR

FIGURE 3.7-156a



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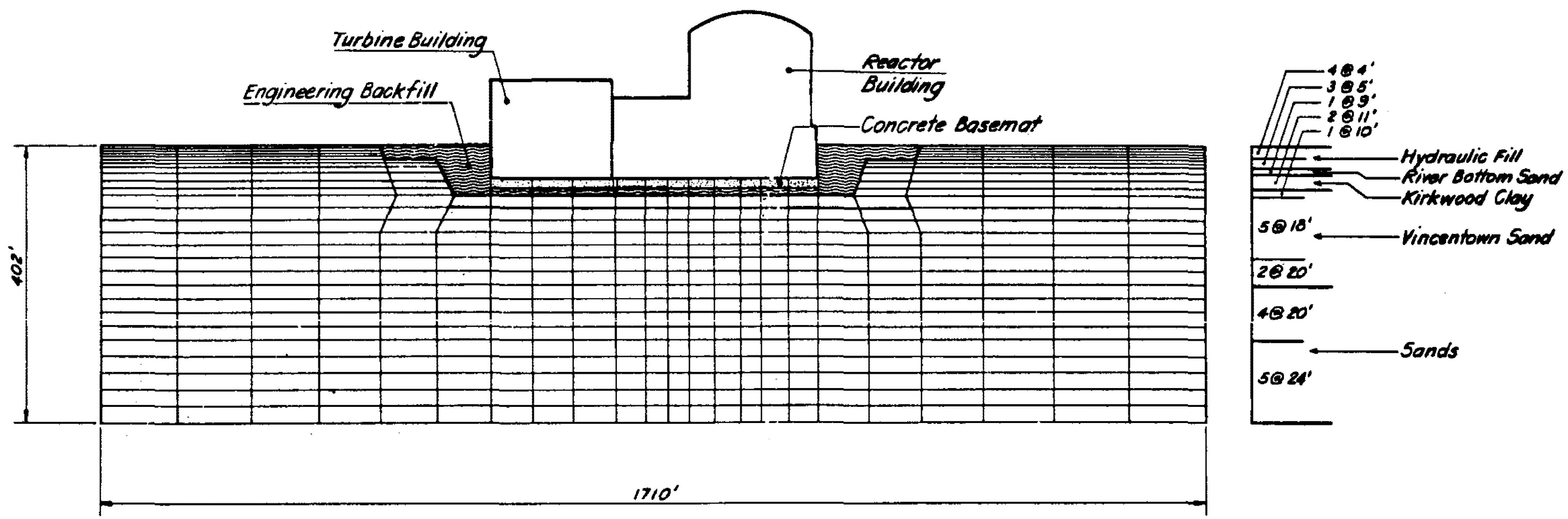
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

HORIZONTAL SOIL-STRUCTURE
INTERACTION ANALYSIS MODEL
E-W CROSS SECTION
FOR AUXILIARY BUILDING ANALYSIS

UPDATED FSAR

FIGURE 3.7-157



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APRIL 11, 1988

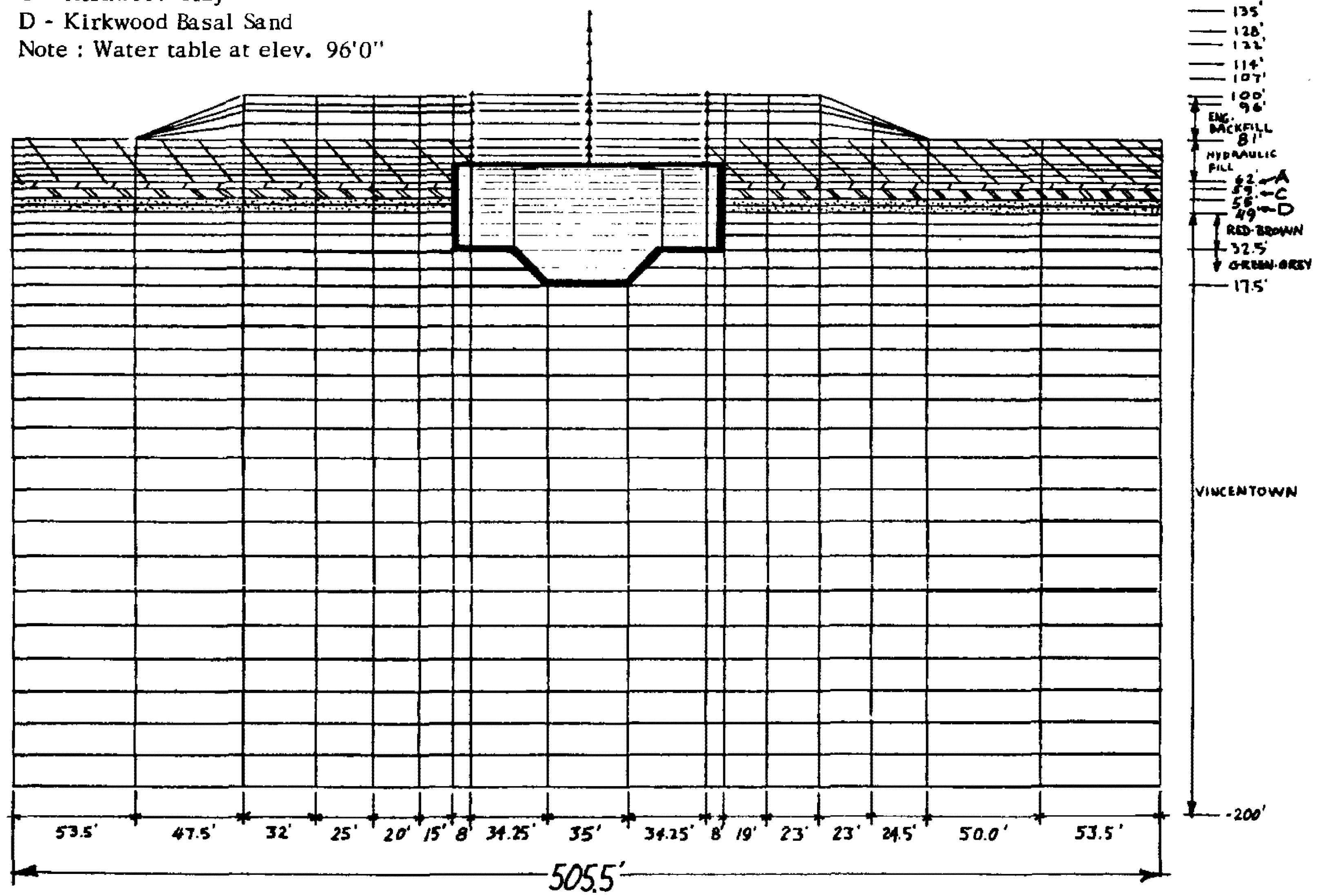
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

HORIZONTAL SOIL-STRUCTURE
INTERACTION ANALYSIS MODEL
E-W CROSS SECTION
FOR REACTOR BUILDING ANALYSIS

UPDATED FSAR

FIGURE 3.7-157a

A - Recent Delaware River Bottom
C - Kirkwood Clay
D - Kirkwood Basal Sand
Note : Water table at elev. 96'0"



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APRIL 11, 1988

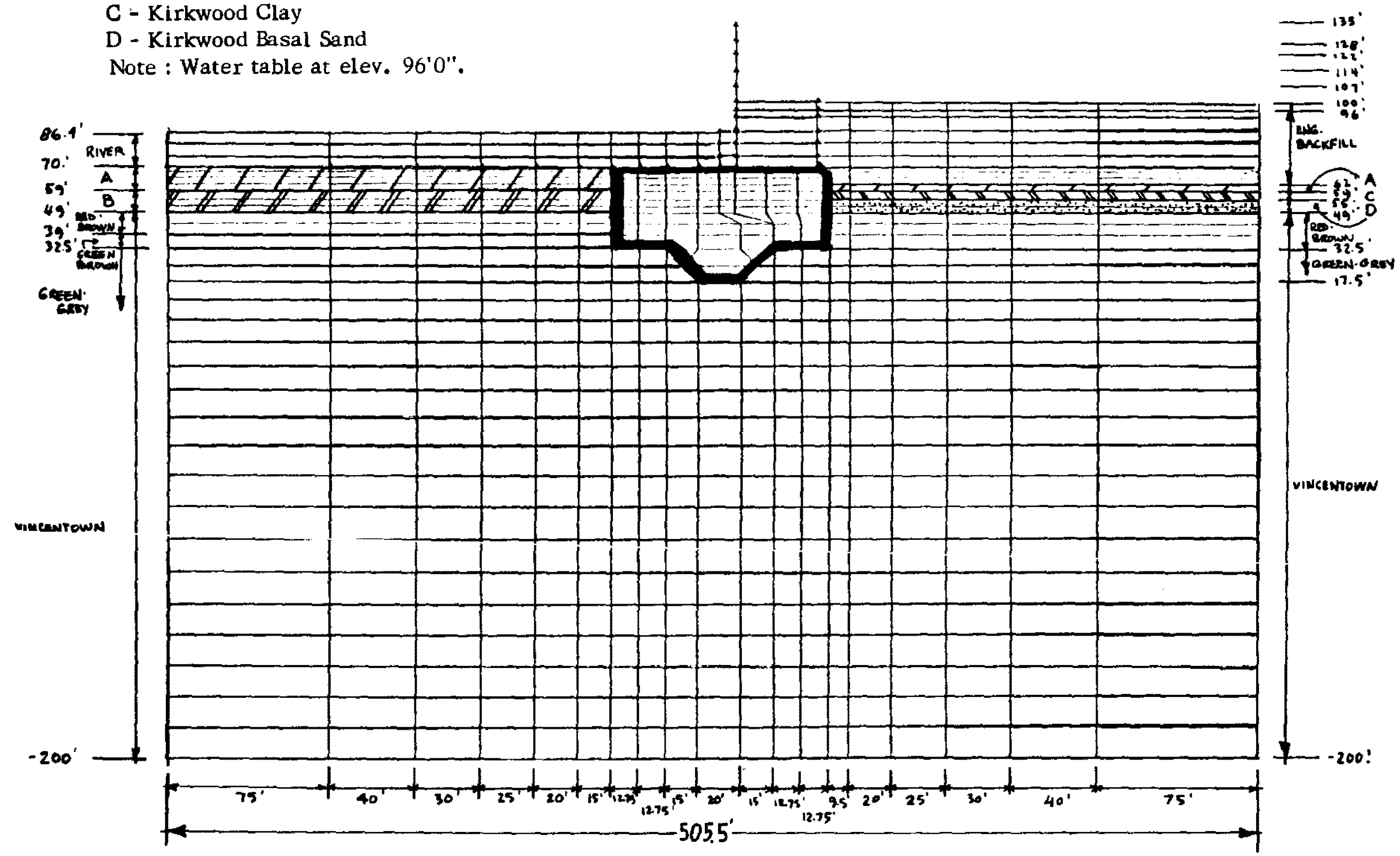
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

SOIL-STRUCTURE INTERACTION
ANALYSIS MODEL
N-S CROSS SECTION,
INTAKE STRUCTURE

UPDATED FSAR

FIGURE 3.7-158

A - Recent Delaware River Bottom
B - Older Delaware River Bottom
C - Kirkwood Clay
D - Kirkwood Basal Sand
Note : Water table at elev. 96'0".



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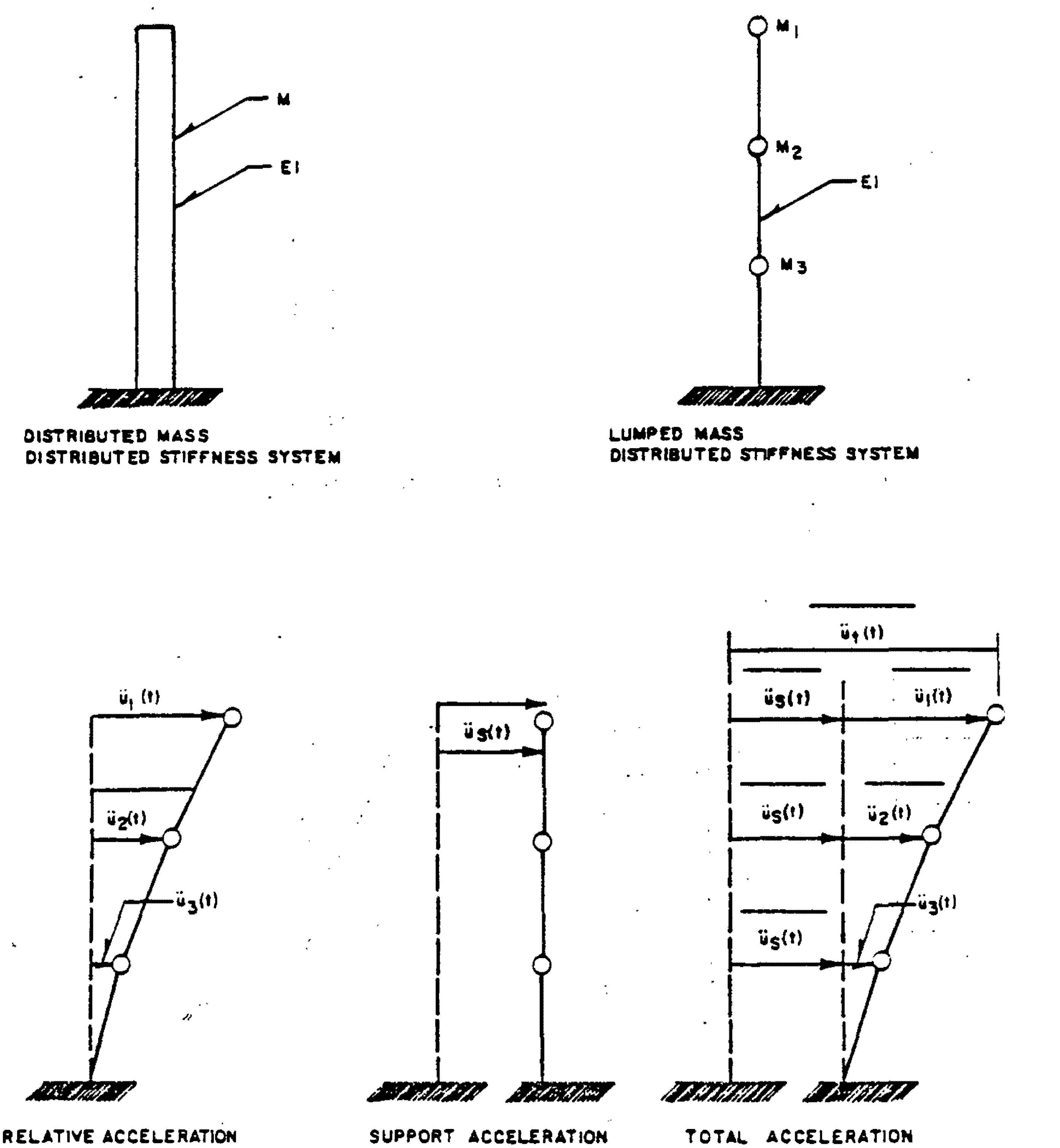
APRIL 11, 1988

PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

SOIL-STRUCTURE INTERACTION ANALYSIS MODEL E-W CROSS SECTION, INTAKE STRUCTURE

UPDATED FSAR

FIGURE 3.7-159



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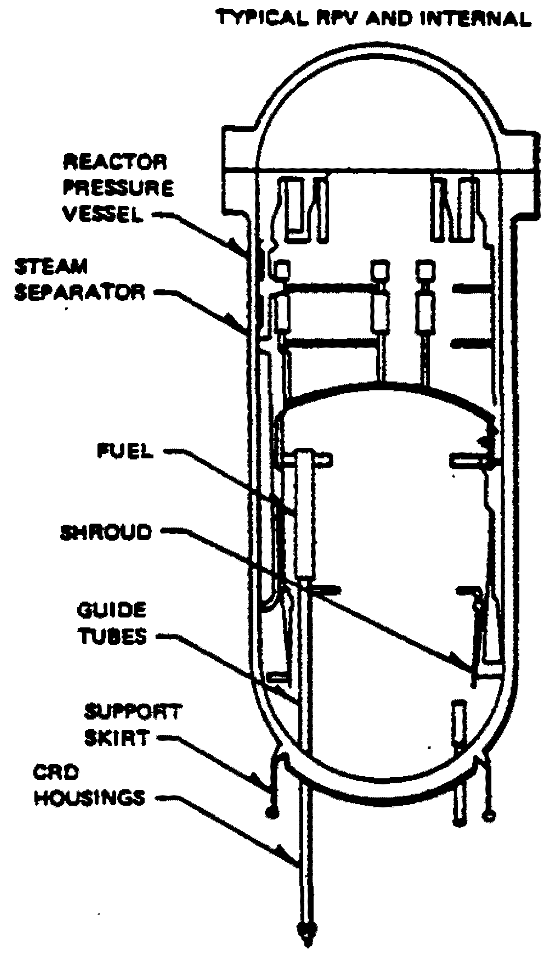
APRIL 11, 1988

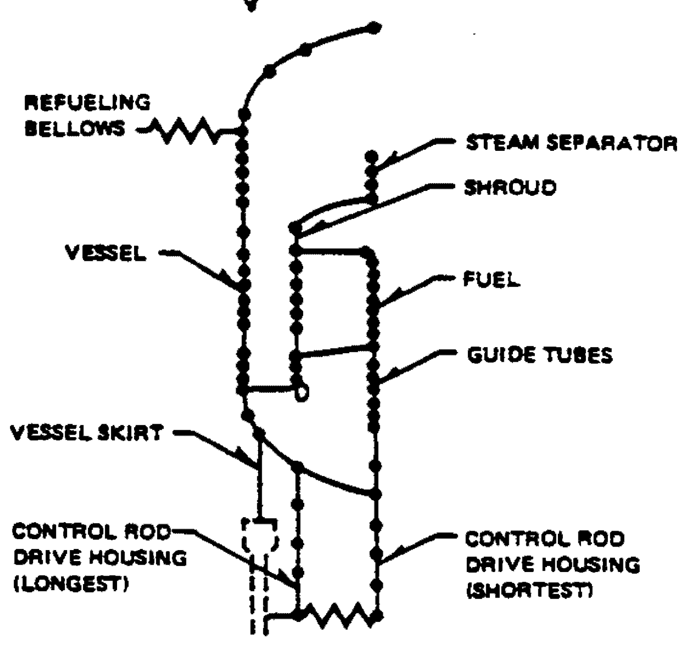
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

LUMPED MASS DISTRIBUTED STIFFNESS SYSTEM

UPDATED FSAR

FIGURE 3.7-160





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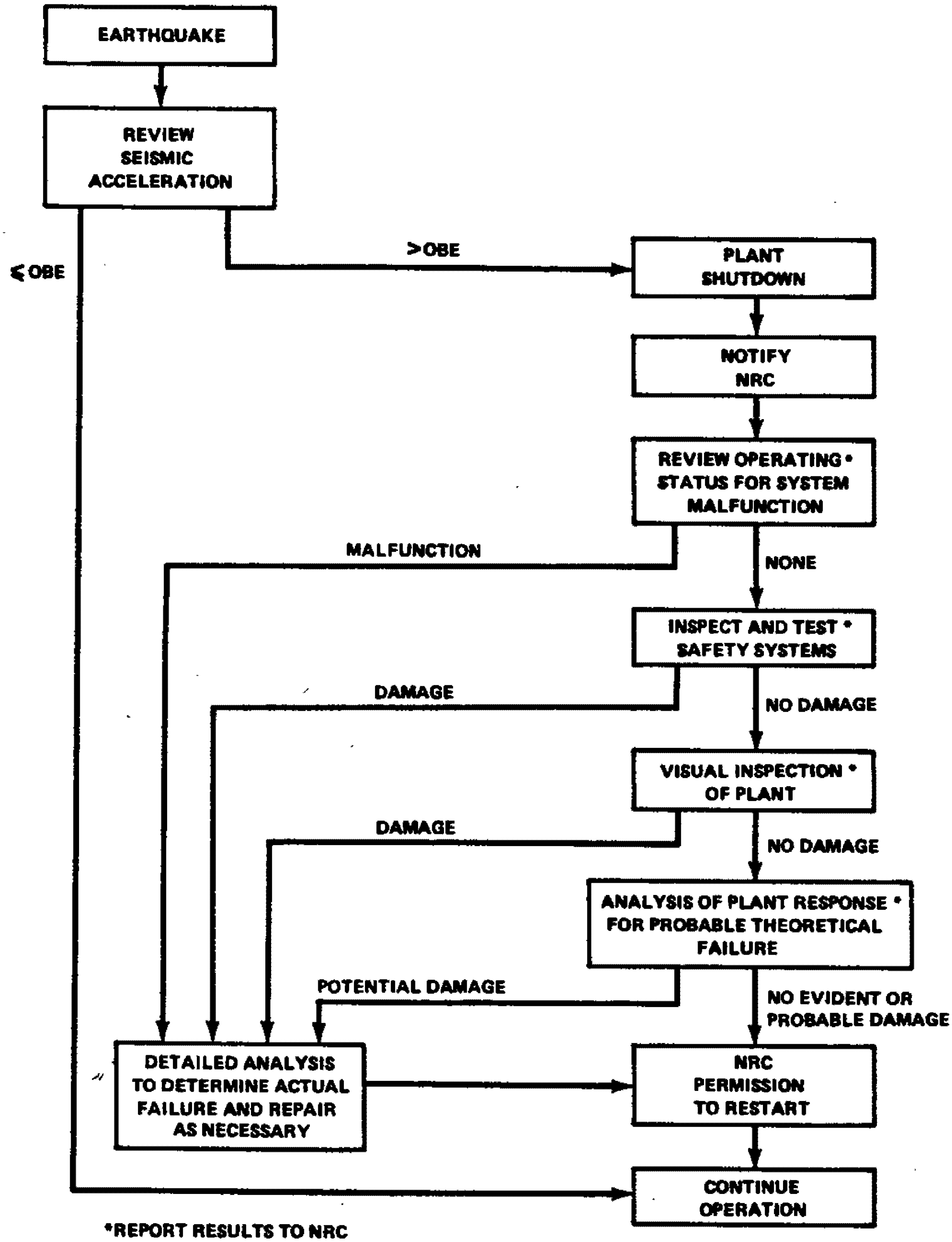
APRIL 11, 1988

PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

REACTOR PRESSURE VESSEL AND INTERNALS SEISMIC MODEL

UPDATED FSAR

FIGURE 3.7-161



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PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

POST SEISMIC EVENT PLANT PROCEDURES

UPDATED FSAR

FIGURE 3.7-162

3.8 DESIGN OF CATEGORY I STRUCTURES

3.8.1 Concrete Containment

This section is not applicable to HCGS because it has a steel containment.

3.8.2 Steel Containment

The steel containment is an ASME B&PV Code Class MC vessel designed to house the Nuclear Steam Supply System (NSSS). The steel containment is a part of the Primary Containment System, which limits the postulated release of radioactivity from the NSSS. This section describes the structural design considerations for the primary containment and includes information that provides the bases for design, construction, and testing of the steel containment, except as modified by the plant unique analysis report, submitted to the NRC under separate cover (letter from R.L. Mittl to Albert Schwencer, dated February 10, 1984.)

The primary containment consists of a drywell, a pressure suppression chamber, and an interconnecting vent system. The drywell is a steel pressure vessel with the shape of a light bulb. The pressure suppression chamber is a torus shaped steel pressure vessel located below and encircling the drywell. A vertical section of the drywell and suppression chamber is shown on Figure 3.8-1.

3.8.2.1 Containment Description

3.8.2.1.1 Drywell

The drywell, shown on Figure 3.8-1, is a steel pressure vessel with a spherical lower portion 68 feet inside diameter, a cylindrical upper portion 40 feet 6 inches inside diameter, and a removable, flanged, hemi-ellipsoidal top head, 33 feet 2 inches inside diameter. Its overall height is 114 feet 9 inches. The bottom elevation of the spherical portion is 77 feet 10 inches. Inner and

outer steel cylindrical skirts that are encased in concrete and anchored to a concrete pedestal support the drywell. The outer skirt is designed to transfer the drywell loads at the bottom of the drywell into the foundation and is the primary support for the drywell during construction. The inner skirt extends into the drywell and transfers reactor pressure vessel (RPV) loads into the foundation. The inside of the drywell is filled with concrete up to Elevation 86 feet 11 inches. The drywell is enclosed by the concrete drywell shield wall. An air gap of nominally 2 inches separates the drywell vessel from the concrete drywell shield wall. The air gap permits displacement of the vessel, but the size of the gap is limited to allow transfer of postulated jet impingement forces into the drywell shield wall without rupturing the vessel.

There are a few, very localized areas, below Elevation 100 feet-0 inches, where the air gap is reduced to as narrow as 0.5 inches. Generally, the reduced gap permits unrestrained displacement of the drywell vessel. Where restraint occurs, the structural effects have been included in the shell analysis. Additionally, a few localized areas exist where no concrete backing is provided. These areas have been evaluated to verify that the vessel alone can satisfactorily resist postulated jet impingement forces without the added resistance of the shield wall.

The drywell is supported laterally by the drywell shield wall near the top of the cylindrical portion of the vessel. The lateral supports are designed to permit vertical and radial displacement of the vessel.

Beam supports are provided for the drywell structural steel framing at Elevations 100 feet 2 inches and 121 feet 7-1/2 inches. The supports are designed to permit differential radial movement between the beams and the shell.

Weld pads are provided on the drywell shell for the attachment of pipe supports, pipe restraints, and similar items.

Containment spray headers at Elevations 129 feet 2 inches and 137 feet 3 inches and a monorail at Elevation 135 feet 6 inches are supported by the drywell shell.

The drywell water seal plate is supported by the drywell at elevation 176 feet 11 inches.

Access to the drywell is provided through a bolted equipment hatch at Elevation 107 feet and another bolted equipment hatch with a double-door air lock at Elevation 107 feet.

3.8.2.1.2 Drywell Head Assembly

The drywell head provides a flanged removable closure at the top of the drywell for RPV access during refueling operations. The drywell head assembly consists of a hemi-ellipsoidal head held in place to the drywell flange by bolts, as shown on Figure 3.8-2. The head is made of 1-1/2-inch thick plate with a 4-inch thick flange and is secured with 180, 2-1/2-inch diameter bolts to the 4-inch thick drywell flange. The head to drywell flanged connection is made leaktight by two replaceable compression seals. Test connections are provided between the seals to allow pneumatic testing from a remote location, outside the steel containment. A personnel access manhole with double, testable seals is provided in the drywell head. Figure 3.8-2 shows details of the drywell head assembly.

3.8.2.1.3 Drywell Equipment Hatches and Personnel Air Lock

Two 12-foot inside diameter equipment hatches in the drywell, at elevation 107 feet, permit the transfer of equipment and components. One hatch, at azimuth 135°, consists of a hatch barrel and a bolted cover with double, testable seals. The other hatch, at azimuth 315°, with similar seals, is furnished with a personnel air lock welded to the removable cover. The personnel air lock is an 8-foot 10-1/2-inch inside diameter cylindrical pressure vessel with inner and outer bulkheads. Interlocked doors, 3-feet 9-inches wide by 7-feet 1-inch high, with double, testable compression seals are

furnished in each bulkhead. The doors are mechanically interlocked to ensure that at least one door is locked to maintain the primary containment integrity. Figures 3.8-3 and 3.8-4 show details of the equipment hatch and the equipment hatch with personnel air lock, respectively.

3.8.2.1.4 Control Rod Drive Removal Hatch

One 3-foot inside diameter control rod drive (CRD) removal hatch at Elevation 103 feet 6 inches in the drywell permits transfer of the CRD assemblies. The hatch is furnished with double, testable seals and a bolted cover. Figure 3.8-5 shows details of the CRD removal hatch.

3.8.2.1.5 Drywell Penetrations

Two general types of process pipe penetrations are provided: those that must accommodate thermal movement as shown by type A on Figure 3.8-6, and those that experience insignificant thermal stress as shown by types B and C on Figure 3.8-6.

The bellows used in Type A triple flued head containment penetrations are designed, fabricated, tested, and examined in accordance with the requirements for Class 2 components of ASME B&PV Section III Code.

Non-NSSS:

The list of non-NSSS systems and their penetration identification numbers that use a Type A triple flued head are shown in Table 3.8-20.

The design considerations for Nuclear Class 1 flued heads consist of evaluation of the loads transmitted to the flued head by the piping from both sides due to:

1. Thermal expansion

2. Seismic reactions

3. Dead weight loads

4. Internal pressure

5. Dynamic loads

6. Thermal gradient effects through the flued head body

7. Thermal transient effects as a result of temperature and pressure changes in the system

8. Discontinuity effects resulting from dissimilar metal welds, if any

9. Fatigue analysis using cumulative usage approach (NB-3653.5 of Section III).

Nuclear Class 2 flued heads are evaluated to the loads listed above with the exception of items 6., 7., and 9.

The type A triple flued head containment penetrations are anchored to the building steel as shown in revised Figure 3.8-6. In the connecting piping analyses, the flued head is considered a rigid anchor. Piping reaction loads (forces, bending moments and torsion) are evaluated as stated above. Fatigue is considered per item(9) above and includes evaluation of the flued head and the butt weld between the flued head and the process pipe.

NSSS:

The main steam piping and the head fittings are designed and fabricated to the requirements of the 1971 edition of Section III of the ASME B&PV Code with addenda through and including those of

Summer 1972. The main steam head fittings are analyzed to the requirements of NB-3200 of the 1977 editions of Section III of the ASME B&PV Code and are evaluated to more restrictive stress limits of BTP MEB 3-1 in SRP 3.6.2. The design report for main steam head fittings includes the evaluation of fatigue and the effect of pipe rupture loads. The head fitting is modeled as a pipe element with rigid stiffness, and its effect on the main steam piping is evaluated to the requirements of NB-3600 of Section III of the ASME B&PV Code.

A typical instrumentation penetration, a typical electrical penetration, and a typical traversing in-core probe (TIP) penetration are shown on Figures 3.8-7, 3.8-8, and 3.8-9, respectively.

The maximum allowable temperature of the drywell shield wall concrete in the areas around the drywell penetrations is 200°F.

3.8.2.1.6 Suppression Chamber

The suppression chamber consists of 16 mitered cylindrical shell segments joined together to form a torus shaped pressure vessel located below and encircling the drywell, as shown on Figure 3.8-10. The suppression chamber has a major diameter of 112 feet 8 inches, a minor or chamber diameter of 30 feet 8 inches, and contains water to an approximate depth of 14 feet. Vertical sections of the suppression chamber are shown on Figures 3.8-11 and 3.8-12.

The 1-inch thick suppression chamber shell is reinforced by full 360° ring beams located 3-1/2 inches from each mitered joint and by partial ring beams at each midcylinder location, which extend a short distance beyond the suppression chamber equator, as shown on Figure 3.8-11. The ring beams provide stiffening for the suppression chamber shell and also allow for transfer of shell pressure loads and support reactions from the vent system, piping, spray header, and monorail and catwalk to the suppression chamber support columns.

The suppression chamber is supported on columns symmetrically arranged in two concentric rings. These columns consist of 2-1/4-inch thick flange plates connected by a 1-inch thick web. The columns are pinned to the base plate assembly at the bottom and to the column connection assembly at the top (Figure 3.8-11), thus carrying only axial loads. Horizontal loads on the suppression chamber are transferred into the drywell foundation pedestal by a horizontal restraint system. The horizontal restraint has pinned connections and slotted holes to allow for thermal expansion of the suppression chamber. Details of the suppression chamber columns and horizontal restraint system are shown on Figures 3.8-11 through 3.8-13.

Attachments to the suppression chamber include vent system supports; penetrations; access hatches; supports for the spray header, monorail and catwalk; pipe supports; and weld pads.

3.8.2.1.7 Suppression Chamber Access Hatches

Four 4-foot inside diameter access hatches in the suppression chamber permit personnel access and the transfer of equipment and components. Each hatch is furnished with double, testable seals. See Figure 3.8-14 for details of the suppression chamber access hatches.

3.8.2.1.8 Vent System

The drywell and the suppression chamber are connected by eight equally spaced vent pipes, each with an internal diameter of 6 feet 2 inches. These vent pipes are connected to a common mitered header within the suppression chamber with a major diameter of 112 feet 8 inches and a minor diameter of 4 feet 3 inches.

Connected to the header are 80 downcomers that terminate at Elevation 68 feet 0-1/2 inch, below the normal water level of the suppression pool at Elevation 71 feet 2-1/2 inches. The downcomers have a 2-foot nominal diameter. At the drywell end, the vent line

openings are protected by jet deflectors to prevent damage to the vent system from postulated jet impingement loadings originating in the drywell. A vacuum breaker assembly is located at the suppression chamber end of each vent line to limit differential pressure between the drywell and suppression chamber. The vent lines are provided with two-ply testable expansion bellows assemblies at the suppression chamber penetrations to accommodate differential movement between the drywell and suppression chamber.

The vent system is supported in the suppression chamber by columns, an upper truss, and a downcomer bracing system. The columns transfer vent system loads into the suppression chamber ring girders. The upper truss connects the vent line and vent header to the ring girder above.

Details of the vent system components and supports are shown on Figures 3.8-15 through 3.8-17.

3.8.2.2 Applicable Codes, Standards, and Specifications

The codes, standards, and specifications used in the design and construction of the primary containment are listed in Table 3.8-1.

Structural specifications are prepared to cover the areas related to design and construction of the primary containment. These specifications emphasize important points of industry standards for design and construction of the primary containment and reduce options that otherwise would be permitted by the industry standards. The following areas are covered in the specifications:

1. Design loads, loading combinations, and allowable stresses for the drywell, suppression chamber, vent system, penetrations, and accessories

2. Materials for primary containment components

3. Fabrication methods, including welding requirements

4. Nondestructive examination requirements

5. Test requirements.

Section 1.8 provides references to regulatory guides discussed in the FSAR. Regulatory Guides specific to this section are discussed in Sections 3.8.2.4 and 3.8.2.5.

3.8.2.3 Loads and Loading Combinations

Table 3.8-2 lists the loading combinations used for the design and analysis of the primary containment.

3.8.2.3.1 Dead Load

The dead load includes the weight of the primary containment structure and appurtenances plus any other permanent loads, such as concrete and hydrostatic loads.

3.8.2.3.2 Live Load

The live load includes the weight superimposed by the use and occupancy of the steel containment, such as moveable equipment and monorail and personnel loading.

3.8.2.3.3 Design Basis Accident Pressure Load

Transients resulting from the design basis accident (DBA) and other lesser accidents are presented in Section 6.2.1, and serve as the basis for the primary containment internal design pressure of 62 psig.

3.8.2.3.4 Thermal Loads

The operating and postulated DBA temperatures inside the primary containment used for the structural analysis are as follows:

Condition	Temperature (°F) Drywell	Suppression Chamber
Operating	130 to 150	50 to 150
Design Basis Accident	340	310

3.8.2.3.5 Earthquake Loads

Earthquake loads are in accordance with those discussed in Section 3.7.

3.8.2.3.6 Wind and Tornado Loads

Wind and tornado loads are not considered during plant operation, because the primary containment is enclosed by the Reactor Building.

3.8.2.3.7 External Pressure Loading

An external to internal differential pressure of 3 psi, as described in Section 6.2.1, is considered in the design of the primary containment.

3.8.2.3.8 Pipe Rupture Loads

The drywell and appurtenances are designed for local pipe rupture effects. Section 3.6 contains a detailed discussion of postulated pipe ruptures and their effects.

3.8.2.3.9 Pool Swell and Main Steam Relief Valve Discharge Loads

The suppression chamber and vent system are designed for pool swell loads resulting from a loss-of-coolant accident (LOCA) and for safety/relief valve discharge loads.

3.8.2.3.10 Post-Accident Containment Flooding

During the period after a LOCA, the entire primary containment, including the suppression chamber, vent system, and drywell, may be flooded up to Elevation 201 feet. This condition is considered in the primary containment design.

3.8.2.3.11 Test Pressure Load

Upon completion of erection, the primary containment vessel, penetrations, and appurtenances undergo an overpressure test at 71.5 psig, followed by a leak rate test at 62 psig.

3.8.2.4 Design and Analysis Procedures

This section describes the procedures used by the primary containment manufacturer and engineer, Pittsburgh-Des Moines Corporation (PDM), and its subcontractor, NUTECH Engineers Incorporated, for the design and analysis of the primary containment. All computer programs referenced are described in Appendix 3A.

The ASME B&PV Code Class MC components and Class MC component supports, described in Section 3.8.2.1, are designed and analyzed in accordance with Article NE-3000 of Subsection NE and Article NF-3000 of Subsection NF, respectively, of the ASME B&PV Code, Section III, Division 1, and as augmented by the applicable provisions of Regulatory Guide 1.57.

3.8.2.4.1 Drywell

The BOSOR4 computer program is used to analyze the drywell shell.

The BOSOR4 mathematical model is shown on Figure 3.8-18. The ASME B&PV Code provides compressive stress requirements for spherical shells subjected to external pressure loads. It does not, however, address specific requirements for compressive stresses in spherical shells that are produced from loads other than external pressure. Therefore, the following procedure is used to demonstrate the adequacy of the drywell shell when subjected to compressive loads:

1. The critical buckling pressure for the drywell spherical shell is determined for buckling of a thin shell sphere under uniform external pressure and is then used to compute a critical buckling stress in the shell.

2. The ASME B&PV Code allowable compressive stress for the drywell shell under external pressure is determined.

3. The factor of safety, which includes an allowance for shell imperfections, against drywell shell buckling under external pressure is established by dividing the critical buckling stress by the ASME B&PV Code allowable compressive stress.

4. The critical buckling stress for the drywell shell, when subjected to external pressure and other compressive loads, is determined from a BOSOR4 analysis.

5. The factor of safety obtained in 3. above is applied to the critical buckling stress determined in 4. above to obtain the theoretical allowable compressive stress for the drywell shell when subjected to a particular loading combination.

The procedure above meets the intent of Regulatory Guide 1.57, since the method maintains a factor of safety that includes an allowance for shell imperfections as established by the ASME B&PV Code for the external pressure loads.

For additional information on the drywell buckling analysis see Appendix 3E.

All computed shell stresses are within the allowable values developed by applying ASME B&PV Code safety factors to the computed critical buckling stresses.

The drywell shell is analyzed for internal pressure using the BOSOR4 computer program. The analysis includes the local effects of jet impingement using localized finite element models of the drywell shell.

The air space between the outside surface of the drywell shell and the adjacent concrete drywell shield wall is modeled with gap elements. An incremental analysis procedure is used where the total load is applied in small steps until the gap is closed. The results show that all stresses are within their respective allowable values.

3.8.2.4.2 Drywell Head Assembly

Stresses in the drywell head are determined for dead load, seismic load, and internal and external pressure using linear elastic theory for thin shells. Stresses in the shell resulting from jet impingement are computed using Welding Research Council Bulletin 107, Reference 3.8-1. Resulting stresses are combined and compared with ASME B&PV Code allowable values for specified loading combinations. The shell is analyzed for external pressure using the ASME B&PV Code, Section III, Paragraph NE-3133.

A BOSOR4 model of the drywell head and flange area is used to examine the flanges and bolts under jet impingement load in combination with the internal pressure.

3.8.2.4.3 Drywell Equipment Hatches and Personnel Air Lock

3.8.2.4.3.1 Equipment Hatch

The equipment hatch is designed and analyzed in accordance with Section III, Subsection NE of the ASME B&PV Code. The cover plate is modeled as a simply supported circular flat plate and analyzed inelastically by means of yield line theory for jet impingement loads. The hatch barrel and shell junction are analyzed using the computer program BOSOR4. The stress intensities computed are compared with the ASME B&PV Code allowable values.

3.8.2.4.3.2 Drywell Equipment Hatch and Personnel Air Lock

The equipment hatch and personnel air lock are designed in accordance with the ASME B&PV Code, Section III, Subsection NE. Reinforcement requirements for the opening in the drywell shell for the hatch barrel is determined by area replacement in accordance with the ASME B&PV Code, Section III, Paragraph NE-3332. Stresses resulting from external forces were computed manually in accordance with Welding Research Council Bulletin 107, Reference 3.8-1. The stress analysis of the hatch cover plate and air lock barrel is accomplished using the computer program ANSYS.

3.8.2.4.4 Drywell and Suppression Chamber Penetrations

Design and analysis requirements of the drywell and suppression chamber penetrations include the following:

1. Ensure that reinforcing around the penetration complies with area replacement requirements of the ASME B&PV Code, Section III, Subarticle NE-3330.

2. Calculate stresses and stress intensities in the penetration nozzle for specified loading combinations. The calculated stress intensities are compared to ASME B&PV Code allowable values.

3. Calculate stresses in the nozzles to insert plate weld for specified loading combinations. These stresses are compared to ASME B&PV Code allowable values. The weld is also checked to ensure that it meets the ASME B&PV Code minimum weld size requirements.

4. Determine stresses in the insert plate at the nozzle to insert plate junction for specified loading combinations, by the method described in Reference 3.8-1. Calculated stresses are compared to ASME B&PV Code allowable values.

5. Determine stresses in the vessel shell at the insert plate to shell junction for specified loading combinations, by the method described in Reference 3.8-1. Calculated stresses are compared to ASME B&PV Code allowable values.

3.8.2.4.5 Suppression Chamber

The seismic analysis of the suppression chamber by the response spectra method uses a 360° finite element beam model, as shown on Figure 3.8-19.

The suppression chamber stress analysis uses a typical 1/32 segment, finite element beam and shell model, as shown on Figure 3.8-20, and the STARDYNE computer program.

The suppression chamber horizontal restraint system is analyzed using a finite element model of a 1/32 segment of the torus to compute shell and ring beam stresses, and manual calculations to compute stresses in other parts of the support system.

Stress intensities are calculated for specified loading combinations and compared to ASME B&PV Code allowable values, and found to be acceptable.

3.8.2.4.6 Vent System

A finite element beam and shell model of a 1/16 segment of the vent system and suppression chamber, as shown on Figure 3.8-21, is used to compute the response of the vent system for all loads except seismic and certain downcomer lateral loads. A 360° beam model, as shown on Figure 3.8-22, is used to compute the response of the vent system for seismic and certain downcomer lateral loads. Finite element models shown on Figures 3.8-23 and 3.8-24 are used to determine stresses at the vent line vent header and vent header downcomer intersections. The resultant stress intensities are compared with ASME B&PV Code allowable values.

3.8.2.4.7 Plant-Unique Analysis

A corroborative analysis is performed for the suppression chamber and vent system for applicable load combinations, including hydrodynamic loads resulting from main steam relief valve discharge and LOCA phenomena, in accordance with the GE Mark I Containment Load Definition Report, Reference 3.8-2; the Hope Creek Plant Unique Load Definition Report, Reference 3.8-3; and appropriate GE Mark I Containment Program Application Guides. Appendix 3B includes a summary description of the confirmatory analysis methods used for stress assessment and identifies modifications to the suppression chamber and vent system.

3.8.2.4.8 Ultimate Capacity of Steel Containment

An analysis was performed to determine the ultimate capacity of the containment. The results of this analysis are summarized in Appendix 3I.

3.8.2.5 Structural Acceptance Criteria

Structural acceptance criteria for the ASME B&PV Code Class MC components and Class MC component supports, which form the bases for establishing allowable stress values, deformation limits, and

factors of safety, are established by Section III, Subsection NE and Subsection NF, respectively, of the ASME B&PV Code, as augmented by the requirements of Regulatory Guide 1.57.

The allowable stress criteria for ASME B&PV Code Class MC components and Class MC component supports are listed in Tables 3.8-2 and 3.8-3 for various loading conditions.

3.8.2.6 Materials, Quality Control, and Special Construction Techniques

3.8.2.6.1 Materials

All materials for Class MC components and component supports meet the requirements of Subsections NE and NF, as applicable, of Section III of the ASME B&PV Code. The primary containment components, other than stainless steel items, have been painted to protect against corrosion.

3.8.2.6.1.1 Drywell Shell

Materials used in construction of the drywell shell assembly include the following:

Item	ASME Specification
Drywell shell	SA-516, Grade 70
Beam seats pad plate and stiffeners	SA-516, Grade 70

3.8.2.6.1.2 Drywell Head

Materials used in construction of the drywell head assembly include the following:

Item	ASME Specification
Drywell head and lower flange	SA-516, Grade 70
Bolts	SA-320, Grade L43
Nuts	SA-194, Grade 7

3.8.2.6.1.3 Drywell Support Skirts

Materials used in construction of the drywell support skirts include the following:

Item	ASME Specification
Inner and outer skirts	SA-516, Grade 70
Base plates	SA-516, Grade 70
Anchor bolts	SA-354, Grade BC

3.8.2.6.1.4 Drywell Access Hatches

Materials used in construction of the drywell access hatches include the following:

Item	ASME Specification
Sleeve and cover	SA-516, Grade 70
Bolts	SA-320, Grade L43 or SA-193, Grade B7

Item	ASME Specification
Nuts	SA-194, Grade 7

3.8.2.6.1.5 Penetrations

Materials used in construction of piping and electrical penetrations include the following:

Item	ASME Specification
Insert plates	SA-516, Grade 70
Nozzles	SA-516, Grade 70
	SA-155, Grade KCF 70
	SA-333, Grade 6
	SA-333, Grade 1
	SA-312, Type 304L

3.8.2.6.1.6 Suppression Chamber

Materials used in construction of the suppression chamber and its supports include the following:

Item	ASME Specification
Shell, ring beams, and ring beam stiffeners	SA-516, Grade 70
Support columns	SA-537, Class 2
Base plates	SA-537, Class 2
Bolting material	SA-540, Grade B21, Class 1
Pins	SA-540, Grade B21, Class 5
Horizontal restraint system:	
Struts	SA-36
Connecting plates	SA-537, Class 2
Bolting material	FA-540, Grade B21, Class 1

3.8.2.6.1.7 Vent System

Materials used in construction of the vent system include the following:

Item	ASME Specification
Vent line	SA-516, Grade 70
Vent header	SA-516, Grade 70
Downcomers	SA-516, Grade 70
Bellows ring	SA-240, Type 304L

3.8.2.6.2 Welding

Welding conforms to the requirements of the ASME B&PV Code, Section III, Subsections NE and NF, as applicable. All butt seam welds in the shell of the primary containment vessel are full penetration, double bevel welds. All welders and weld procedures are qualified in accordance with Section IX of the ASME B&PV Code.

Post-weld heat treatment for pressure-retaining components is in accordance with the ASME B&PV Code, Section III, Subsection NE and NF, as applicable.

3.8.2.6.3 Materials Testing

The pressure retaining parts and attachments to the pressure retaining parts of the primary containment vessel are impact tested, in accordance with the applicable Subsections of the ASME B&PV Code, Section III. The impact specimens were tested at +5°F or below.

3.8.2.6.4 Nondestructive Examination of Welds

Nondestructive examination of all pressure retaining welds is in accordance with the ASME B&PV Code, Section III, Subsections NE and NF, as applicable.

3.8.2.6.5 Quality Control

The quality assurance provisions of the applicable parts of Articles in Sections NA-4000, NE-4000, NE-5000, NF-4000, and NF-5000 of Section III of the ASME B&PV Code, including Code Case N-242, were followed in all phases of design, procurement, shop fabrication, and field installation of the primary containment.

3.8.2.6.6 Erection Tolerances

Erection tolerances for the primary containment vessel meet the requirements of Section III of the ASME B&PV Code. In addition, the specified erection tolerances include the following:

1. The top head flange is within 2 inches of the design elevation and is level within 1/2 inch.

2. Penetrations are within 1/2 inch of their specified elevation and azimuth at their intersection with the vessel.

3. Alignment of penetrations are within 1/2 degree of the design alignment.

Actual deviations from the above are evaluated in accordance with procedures covered in Section 3.8.2.6.5.

3.8.2.6.7 Corrosion

The thickness of pressure boundary elements and other critical components of the primary containment has been increased beyond

minimum design thickness to include a corrosion allowance as follows:

1. Drywell shell: 1/16 inch
2. Suppression chamber shell and ring girders: 1/8 inch
3. Vent lines and vent header: 1/16 inch
4. Downcomers: 1/8 inch
5. Vent header and downcomer supports: 1/8 inch
6. Pipe supports and related items: 1/8 inch for submerged items and 1/16 inch for portions above water.

3.8.2.6.8 Special Construction Techniques

Erection of the primary containment was performed by PDM using methods, tools, and equipment generally accepted in the industry.

3.8.2.7 <u>Testing and Inservice Surveillance</u>

3.8.2.7.1 Preoperational Testing

3.8.2.7.1.1 Structural Acceptance Test

The primary containment is pneumatically tested to 1.15 times the design pressure during the containment overpressure test, in accordance with Article NE-6000 of Subsection NE, Section III of the ASME B&PV Code.

The personnel air lock is pneumatically tested to 1.15 times the design pressure, following shop fabrication and field erection, to verify its structural integrity.

3.8.2.7.1.2 Leak Rate Testing

The leaktight status of the primary containment is verified during the integrated leak rate test performed in accordance with 10CFR50, Appendix J, Option B. See Section 6.2.6 for a description of the primary containment integrated leak rate test.

3.8.2.7.2 Inservice Leak Rate Testing

Inservice leak rate testing is discussed in Section 6.2.6.

3.8.2.8 SRP Rule Review

3.8.2.8.1 Deleted

3.8.2.8.2 Acceptance Criterion II.4(f)

Acceptance Criteria II.4(f) of SRP Section 3.8.2 and the relevant acceptance criteria of SRP Sections 3.8.3, 3.8.4 and 3.8.5 require that a design report be prepared and is considered acceptable if it contains the information specified in Appendix C of SRP Section 3.8.4

Sufficient information is provided in the HCGS FSAR to outline the structural design of the Seismic Category I structures. This information includes such items as structural description and geometry, load combinations, materials used, applicable codes and standards, and computer codes, as required by Regulatory Guide 1.70, Revision 3. As required by 10CFR50 Appendix B, information is also available to enable an audit of these Seismic Category I structures to inspect and verify their structural integrity. The information available for such an audit is consistent with the information requested in Appendix C to SRP 3.8.4.

3.8.2.8.3 Acceptance Criterion II.5

Acceptance Criterion II.5 of SRP Section 3.8.2 refers to Table 3.8.2-1 of the SRP for allowable stresses for the loading combinations given in SRP Section 3.8.2.II.3(b). The design requirements for HCGS deviate from the SRP requirements for two loading combinations: testing, and post-accident containment flooding.

For these two loading conditions, the specified allowable stress for HCGS is higher than that specified in Table 3.8.2-1 of the SRP. However, the calculated stresses for the Class MC components are less than the SRP stress limits for these two loading conditions.

3.8.3 Primary Containment Internal Structures

3.8.3.1 Description of the Internal Structures

The functions of the primary containment internal structures include support and shielding of the reactor pressure vessel (RPV) and support of piping and equipment. The primary containment internal structures are constructed of concrete and structural steel and include the following:

1. RPV pedestal

2. Biological shield

3. Platforms and pipe restraints

4. Biological shield lateral truss and RPV stabilizer.

Figure 3.8-1 shows the general arrangement of the primary containment, including the internal structures.

3.8.3.1.1 Reactor Pressure Vessel Pedestal

The RPV pedestal, approximately 26-feet high, is a vertical, cylindrical, reinforced concrete structure that rests on the drywell floor/pedestal mat and supports the RPV, biological shield, drywell platforms, and pipe restraints.

The RPV pedestal has an outside diameter of 29 feet 11 inches and a wall thickness of 4 feet 10 inches. The thickness at the top of the pedestal increases to 5 feet 9 inches, where it supports both the RPV and the biological shield. Figure 3.8-25 shows the connection of the RPV pedestal to the base foundation.

The biological shield is supported on the RPV pedestal, as described in Section 3.8.3.1.2. The RPV is supported on the pedestal through a ring girder, as described in Section 5.3.3.1. The ring girder is attached to the RPV pedestal by 120, 3-1/4-inch diameter high strength anchor bolts, as shown on Figure 3.8-26.

Figures 3.8-27 and 3.8-28 show reinforcement details for the RPV pedestal.

Openings are provided to allow access for personnel, piping, and equipment into the pedestal cavity, with additional reinforcement furnished at the openings. Embedded transfer girders are provided to transfer loads around the CRD penetrations. A carbon steel liner plate on the inside of the RPV pedestal acts as a concrete form during construction.

3.8.3.1.2 Biological Shield

The biological shield is a 49-foot high, vertical, cylindrical shell that provides primary radiation shielding, as well as support for pipe restraints and drywell platforms. It is designed as a composite steel-concrete structure and is constructed of carbon steel inner and outer plates with concrete and shear ties between the two plates.

The biological shield has an inside diameter of 26 feet 5 inches and a wall thickness of 1 foot 9 inches, as shown on Figure 3.8-29. The outer steel plate is 1-1/2 inches thick, and the inner steel plate is 3/4-inch thick. These inner and outer plates are connected with shear ties spaced on approximately 4-degree centers in the circumferential direction and 12-inch centers in the vertical direction to provide adequate shear transfer. Internal stiffeners are provided to withstand local loads transferred through pipe restraints and drywell platform attachments. The annular space between the inner and outer plates is filled with concrete. The upper section, above Elevation 125 feet 5-1/2 inches, contains high density concrete for radiation shielding in the reactor core area. The biological shield is connected to the top of the RPV pedestal by 60, 3-1/4-inch diameter, high strength anchor bolts embedded in the pedestal, as shown on Figure 3.8-26.

The biological shield lateral truss and RPV stabilizer, which provide lateral support to the biological shield and RPV, are attached to the top of the biological shield. The biological shield has penetrations with hinged doors or removable plugs to accommodate piping connections to the RPV, and also to provide access for inservice inspection. All doors are bolted to penetration sleeves, and the inner section of certain doors are filled with boron concrete where required for radiation shielding.

3.8.3.1.3 Platforms and Pipe Whip Restraints

Two major platforms are furnished in the drywell to provide access and support to piping and equipment. The platforms consist of structural steel framing with steel grating. Built up box shapes are used for beams that must resist significant biaxial loading. Beams that span between the RPV pedestal or the biological shield and the drywell shell are provided with sliding connections at the drywell shell. Thus, no significant thermal axial loads are developed in the beams, and no significant thermal radial loads are imposed on the pedestal, biological shield, or primary containment

shell. Figures 3.8-31 and 3.8-32 show details of the drywell platforms.

Pipe whip restraints are provided inside the drywell to prevent pipe whip due to a high energy pipe break. The restraints are of two different designs: a U-strap design, and a frame type design. Typical restraints inside the drywell are shown on Figure 3.6-1.

3.8.3.1.4 Biological Shield Lateral Truss and RPV Stabilizer

The lateral truss and the RPV stabilizer provide lateral support for the biological shield and the RPV during seismic and pipe break loading. The lateral truss spans horizontally between the primary containment and the biological shield. It is shaped like an eight-point star and is fabricated from steel plate and pipe sections. Figure 3.8-33 shows details of the lateral truss. The truss transfers lateral forces from the RPV and the biological shield through the drywell shell to the concrete drywell shield wall by eight shear lugs attached to the drywell shell. The shear lugs are designed to permit vertical and radial thermal expansion of the drywell shell. The RPV stabilizer spans horizontally between the biological shield and the RPV.

3.8.3.2 Applicable Codes, Standards, and Specifications

The codes, standards, and specifications used in the design and construction of the primary containment internal structures are listed in Table 3.8-7.

Specifications were prepared specifically to cover the areas related to design and construction of the primary containment internal structures. These specifications supplement the industry standards for the primary containment internal structures, and reduce options that would otherwise be permitted by the industry standards. They cover the following major areas:

1. Furnishing and delivering concrete,

2. Forming, placing, finishing, and curing concrete,

3. Furnishing, detailing, fabricating, delivering, and placing reinforcing steel,

4. Splicing reinforcing steel,

5. Furnishing, detailing, fabricating, delivering, and erecting structural steel,

6. Coating of steel and concrete surfaces.

Section 1.8 provides references to Regulatory Guides discussed in the FSAR.

3.8.3.3 Loads and Loading Combinations

Tables 3.8-4 through 3.8-6 list the load combinations used for the design and analysis of the primary containment internal structures.

3.8.3.3.1 RPV Pedestal

Table 3.8-4 lists the load combinations used for the design of the RPV pedestal. Descriptions of the loads are as follows:

1. Dead load, live load, and seismic loads - For a description of dead load, live load, and seismic loads, see Section 3.8.2.3.

2. Thermal loads - The RPV pedestal is designed for the temperature gradient resulting from the postulated design accident condition.

3. Pipe break loads - The RPV pedestal is designed to withstand pipe break loads due to a postulated break of

any high energy pipe, including a 28-inch diameter recirculation loop pipe. The analysis considers the effects of jet impingement, pipe whip, and pipe reaction. An equivalent static load of 1860 kips is considered, which includes an appropriate dynamic load factor to account for the dynamic nature of the load. Section 3.6 contains a detailed discussion of postulated pipe breaks and their effects.

4. Additional loads - For conservatism, a subcompartment pressurization is postulated due to a nonmechanistic break inside the bioshield. This loading is resisted by the pedestal. For additional information, see Appendix 6B.1.2.

3.8.3.3.2 Biological Shield

Tables 3.8-4 and 3.8-5 list the load combinations used for the design of the biological shield. The most severe loading condition combines the DBA loads with the maximum seismic loads. Descriptions of the loads are as follows:

1. Dead load, live load, and seismic loads - For a description of dead load, live load, and seismic loads, see Section 3.8.2.3.

2. Abnormal pressure load - The biological shield is designed for internal pressurization due to a postulated break of any high energy pipe, including a 28-inch diameter recirculation loop pipe. The following two pressure conditions are considered:

   a. Maximum unbalanced pressure, which is a pressure condition occurring shortly after pipe break that produces a net lateral load on the biological shield.

b. Maximum uniform pressure, which is a pressure condition after pipe break that produces a uniform 150 psig internal pressure.

3. Thermal loads - The biological shield is designed for the temperature gradient resulting from the postulated design accident condition.

4. Pipe break loads - The biological shield is designed to withstand the pipe break effects due to a postulated break of any high energy pipe. The analysis considers the effects of jet impingement, pipe whip, and pipe reaction. Equivalent static loads are considered, including an appropriate dynamic load factor to account for the dynamic nature of the load. Section 3.6 contains a detailed discussion of postulated pipe breaks and their effects.

5. Additional loads - For conservatism, a subcompartment pressurization is postulated due to a nonmechanistic break inside the bioshield. This loading is resisted by the pedestal. For additional information, see Appendix 6.B.1.2.

3.8.3.3.3 Platforms and Pipe Whip Restraints

The drywell platforms are designed using the AISC working stress design methods, except for pipe whip restraints supported by the platforms. The pipe whip restraints are designed to undergo local inelastic deformations due to postulated pipe break loads. The inelastic deformations do not cause loss of function of the pipe whip restraints. The built-up beams that support the pipe whip restraints are designed to withstand all postulated pipe break loads.

Design accident pressure, operating and design accident thermal, and seismic loads have been considered in the design of the drywell platforms. The uniform design live load for the grating and framing

beam is 200 lb/ft^2. The design load for the framing beams also includes the gravity load, thermal reaction load, and seismic reaction load of all piping and equipment supported on the beams. Table 3.8-5 lists the load combinations used to design the drywell platforms and pipe whip restraints.

3.8.3.3.4 Biological Shield Lateral Truss

The lateral truss is designed using the AISC working stress design methods. It is designed for lateral loads, including seismic and postulated pipe break effects.

Design accident pressure, and operating and design accident thermal loads have been considered in the design of the lateral truss. Table 3.8-5 lists the load combinations used to design the lateral truss.

3.8.3.4 Design and Analysis Procedures

This section describes the procedures used for the design and analysis of the primary containment internal structures. All computer programs referenced are described in Appendix 3A.

3.8.3.4.1 RPV Pedestal

The RPV pedestal is designed for axisymmetric loads, which include dead load and design accident temperature load, using the FINEL computer program. Both concrete and reinforcing steel materials are included in the model. The operating and design accident temperature gradients are computed. For transient loads, such as design accident pressure and thermal loads, the most critical combination of these loads is considered.

The RPV pedestal is also designed for nonaxisymmetric loads, which include seismic loads, design accident pressure and pipe break loads, and RPV and biological shield loads, using the STRUDL and ASHSD computer programs.

Figure 3.8-34 shows a vertical section through the finite-element model used to analyze the RPV pedestal. The model includes the RPV pedestal, the foundation anchorage, and the biological shield.

Concrete and reinforcing steel stresses, due to axisymmetric and nonaxisymmetric loads, are combined where applicable to determine the total stress and are compared with allowable values.

3.8.3.4.2 Biological Shield

The biological shield is analyzed as an axisymmetric structure. The FINEL computer program is used in analysis of axisymmetric loads, which include dead load, design accident thermal load, and design accident uniform pressure load. The temperature gradient across the thickness of the wall is computed. For nonaxisymmetric loads, which include design accident unbalanced pressure load, seismic load, and pipe break load, the ASHSD computer program is used. Figure 3.8-34 shows a vertical section through the finite element model used to analyze the biological shield. Total stresses in the biological shield are determined by summing the stresses resulting from axisymmetric and nonaxisymmetric loads and are compared with allowable values.

Openings in the biological shield are analyzed locally to determine reinforcement requirements using the ASME B&PV Code area replacement method. Local stiffening of the shell is provided by thick walled penetration sleeves and reinforcing rings.

3.8.3.4.3 Platforms and Pipe Whip Restraints

The drywell platforms are designed using conventional elastic design methods, in accordance with the AISC Specification, Part I. Members that are impacted as a result of postulated pipe break are designed using elasto-plastic methods to determine energy absorption capacity, as described in Bechtel Topical Report BN-TOP-2, Reference 3.8-4.

3.8.3.4.4 Biological Shield Lateral Truss

Seismic forces in the lateral truss are calculated using the methods described in Section 3.7. Axial force, shear force, and moment in the lateral truss due to postulated pipe break are calculated using the STRUDL computer program.

3.8.3.4.5 Plant Unique Analysis

A confirmatory analysis is performed for the suppression chamber and internal structures for all applicable loads, including hydrodynamic loads resulting from main steam relief valve discharge and loss-of-coolant accident (LOCA) phenomena, in accordance with NUREG 0661; NUREG 0763; the GE Mark I Containment Load Definition Report, Reference 3.8-2; the Hope Creek Plant-Unique Load Definition Report, Reference 3.8-3; and appropriate GE Mark I Containment Program Application Guides. Appendix 3B includes a summary description of the confirmatory analysis methods used for stress assessment and identifies potential modifications to the suppression chamber internal structures.

3.8.3.5 Structural Acceptance Criteria

3.8.3.5.1 Concrete

The RPV pedestal and the biological shield are designed for the factored load combinations listed in Table 3.8-4, in accordance with the strength method in American Concrete Institute (ACI) 318.

3.8.3.5.2 Structural Steel

Structural steel portions of the containment internal structures include the biological shield, platforms, pipe whip restraints, and lateral truss. For normal loading conditions, the allowable stresses are in accordance with the AISC Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings.

For extreme environmental and abnormal loading conditions, the allowable stresses are given in Table 3.8-5.

For members that are impacted as a result of postulated pipe break effects, energy absorption is determined and compared to the energy input in order to verify that energy absorption capacity exceeds energy input.

3.8.3.6 Materials, Quality Control, and Special Construction Techniques

The criteria of ACI 349, Code Requirements for Nuclear Safety-Related Concrete Structures, applicable to this section are not used by HCGS. This section discusses the alternate criteria used.

3.8.3.6.1 Concrete Containment Internal Structures

The concrete and reinforcing steel materials for the primary containment internal structures are discussed in Section 3.8.6.

3.8.3.6.2 Biological Shield and Biological Shield Lateral Truss

3.8.3.6.2.1 Materials

Materials used in the construction of the biological shield and the lateral truss include the following standard specifications:

Item	Specification
Biological shield outer plate	ASTM A537, Class 1, electric furnace doubleslagged plus vacuum degassed in accordance with supplementary requirements S-1 of ASTM A20

Item	Specification
Biological shield inner plate	ASTM A537, Class 1
Horizontal & vertical stiffener plate	ASTM A537, Class 1
Bars for shear ties	ASTM A321
Bolts for shear ties	ASTM A490
Top plate and bottom plate	ASTM A537, Class 1
Lateral truss pipe members	ASTM A618
Biological shield anchor bolts	ASTM A540, Class 3, Grade B23
Normal weight concrete	See Section 3.8.6.2.4
High density concrete	See Section 3.8.6.2.5.

3.8.3.6.2.2 Welding Procedure and Qualifications

The biological shield is fabricated using welding procedures prepared and qualified in accordance with the requirements of the ASME B&PV Code, Section III, Subsection ND, Article ND-4000, and Section IX. Nondestructive examination of welds, including radiographic examination, ultrasonic examination, magnetic particle examination, and liquid penetrant examination, is in accordance with the ASME B&PV Code, Section III, Article ND-5000, and Section V.

3.8.3.6.2.3 Materials Testing

The biological shield outer liner plate is ultrasonically tested in accordance with ASTM A-578-75, including supplemental requirements 5 through 8.

3.8.3.6.2.4 Tolerances

The specified erection tolerances include the following:

1. Each of the two concentric cylinders of the biological shield is plumb within 1:500 of the height.

2. The radial dimension to any point on the biological shield plates does not vary by more than ~1/4-inch from the centerline established by the design.

3. The clear distance between the two steel biological shield inner and outer plates does not vary more than ~1/4-inch from the theoretical distance at any point.

4. The penetration sleeve centerlines are within ~1/4-inch of specified elevations and azimuths of the RPV nozzles.

5. The elevation of the top of the biological shield is within ~1/4 inch of that shown on the design drawings.

Actual deviations from the above are handled in accordance with procedures covered in Section 17.

3.8.3.6.3 Drywell Platforms

3.8.3.6.3.1 Materials

Materials used in construction of the drywell platforms include the following standard specifications:

Item	Specification
Structural shapes (less than 30 pounds per linear foot)	ASTM A36

Item	Specification
Structural shapes (more than 30 pounds per linear foot)	ASTM A441 or ASTM A588
Box beams and built-up wide flange beams	ASTM A537, Class 1.

3.8.3.6.3.2 Welding

The drywell platforms are fabricated using welding procedures in accordance with the American Welding Society (AWS) Structural Welding Code D1.1. (See Table 3.8-7).

3.8.3.6.3.3 Nondestructive Examination

Nondestructive examination of welds for the drywell platforms, including radiographic examination, magnetic particle examination, ultrasonic examination, and liquid penetrant examination, is in accordance with Sections 6 and 8 of AWS D1.1.

3.8.3.6.3.4 Erection Tolerances

Erection tolerances for the drywell platforms are in accordance with the AISC specification. Actual deviation from the specification is evaluated in accordance with procedures covered in Section 17.

3.8.3.6.4 Quality Control

Quality control requirements during construction are discussed in Section 17.

3.8.3.7 Testing and Inservice Inspection Requirements

The internal structures are not directly related to the functioning of the containment concept. Therefore, no testing or inspection is required.

3.8.4 Other Seismic Category I Structures

This section discusses all Seismic Category I structures, except the primary containment and its internals, which are described in Sections 3.8.2 and 3.8.3. This section also describes certain related non-Seismic Category I structures that could affect safety-related systems, components, or structures.

Specific structures included are:

1. Seismic Category I structures - Reactor Building; auxiliary Building, including control/diesel generator area and radwaste/service area; Station Service Water System (SSWS) intake structure; plant cancelled area, the former Unit 2 Reactor Building; and condensate storage tank dike.

2. Non-Seismic Category I structures - Turbine building; and administration facility, the former Unit 2 turbine building.

All of these structures and their physical interrelationships are shown on Figure 3.8-35.

3.8.4.1 Description of Structures

3.8.4.1.1 Reactor Building

The Reactor Building, as shown on Plant Drawings P-0014-1 and P-0042-1 through P-0047-1, is a reinforced concrete enclosure that consists of a cylindrical containment structure topped by a toroid spherical dome, with a rectangular lower section enclosing the base of the cylinder. The cylindrical portion completely encloses both the reactor and the pressure suppression primary containment system. It also houses fuel storage and handling facilities and engineered safety features (ESFs). It is located in the southwest quadrant of the power

complex adjacent to the Auxiliary Building, which is to the north and east.

The Reactor Building bearing/shear walls are designed to resist lateral loads and transmit them to the reinforced concrete foundation mat, where all loads are dissipated into the Vincentown Formation. The reinforced concrete floors are generally supported by structural steel framing systems that are in turn supported by the walls. Floor systems are designed to act as diaphragms that transmit lateral loads to the shear walls. Radial framing is used within the cylindrical portion, while framing in the rectangular area is laid out on east-west and north-south lines.

At the north wall of the Reactor Building, where it interfaces with the auxiliary building control/standby diesel generator (SDG) area, a seismic separation joint extends from the foundation mat through the roof. The steel primary containment is isolated from the reinforced concrete drywell shield wall by an air gap.

The refueling facility is located above the primary containment. This facility is supported by steel girders and by the reinforced concrete slabs and walls of the pools that span between the drywell shield wall and the cylindrical wall.

Interior surfaces of walls and slabs of the spent fuel pool, cask loading pit, reactor well, and steam dryer and separator storage pool are lined with stainless steel plate. The entire refueling facility meets radiation shielding requirements, as discussed in Section 12.3.2.

All reinforced concrete walls and floors meet both structural and radiation shielding requirements, as discussed in Section 12.3.2. There are no concrete masonry unit walls used in the reactor building.

The Reactor Building foundation mat, described in Section 3.8.5, extends eastward beyond the Reactor Building to support the southern section of the auxiliary building radwaste/service area.

The 150 ton capacity polar crane, as described in Section 9.1.5, is supported by a continuous, circular corbel constructed integrally with the cylindrical wall.

3.8.4.1.2 Auxiliary Building Control/Standby Diesel Generator Area

The control/SDG area, as shown on Plant Drawings P-0051-0 through P-0057-0, is located in the Auxiliary Building.

The control area houses the controls for both the reactor and the balance of plant (BOP) elements that constitute HCGS.

The SDG area houses systems that provide operating power for HCGS in case of loss of the primary power source.

The control/SDG area is separated from the Reactor Building and the plant cancelled area by seismic separation joints, extending in the east-west direction from the bottom of the foundation mats through the roofs, at their respective interfaces. The area is bounded on the east by the radwaste/service area reinforced concrete isolation wall, and on the west by an exterior reinforced concrete wall. The area is separated into individual utility areas by interior reinforced concrete walls, the eastern areas of which constitute the control area and the western SDG area.

The reinforced concrete foundation mat, described in Section 3.8.5, extends eastward beyond the control/SDG area to support the central section of the radwaste/service area.

The control/SDG area is a structurally integrated reinforced concrete structure that has bearing/shear walls designed to resist lateral loads and transmit them to the foundation mat, where all loads are dissipated into the Vincentown Formation. The reinforced

concrete floors are supported by structural steel framing systems that are in turn supported by the walls and structural steel columns. All floor systems are designed to act as diaphragms that transmit lateral loads to the shear walls. All reinforced concrete walls and floors meet both structural and radiation shielding requirements, as discussed in Section 12.3.2. There are no concrete masonry unit walls in the control/SDG area.

3.8.4.1.3 Auxiliary Building Radwaste/Service Area

The Auxiliary Building radwaste/service area, as shown on Plant Drawings P-0031-0 through P-0037-0, houses radwaste treatment and storage facilities, cable tray runs, main steam line tunnels, heating and ventilating equipment, machine shops, decontamination equipment, and personnel facilities.

The Auxiliary Building radwaste/service area is separated into three sections by seismic separation joints extending in the east-west direction from the bottom of the foundation mats through the roofs. A similar north-south seismic joint separates the east interface of the radwaste/service area from the Turbine Building and administration facility. The west side of the radwaste/service area is bounded by the Reactor Building, the Auxiliary Building control area, and the plant cancelled area. The northern section of the radwaste/service area is structurally continuous with the plant cancelled area, the central section with the Auxiliary Building control/SDG area, and the southern section with the reactor building.

The structural foundations consist of the eastern portions of three separate reinforced concrete mats, isolated by seismic separation joints. Each foundation continuously projects to the west as founding support for the reactor building, the control/SDG area, and the plant cancelled area, described in Sections 3.8.4.1.1, 3.8.4.1.2, and 3.8.4.1.4, respectively.

The radwaste/service area is a reinforced concrete structure that has bearing/shear walls designed to resist lateral loads and transmit them to the reinforced concrete foundation mat, where all loads are dissipated into the Vincentown Formation. The reinforced concrete floors are supported by structural steel beam and column framing systems and are designed as diaphragms to resist lateral loads and transmit them to the shear walls. All reinforced concrete walls and floors meet both structural and radiation shielding requirements, as discussed in Section 12.3.2. There are no concrete masonry unit walls in the radwaste/service area.

3.8.4.1.4 Plant Cancelled Area

The plant cancelled area, formerly a portion of the Unit 2 Reactor Building, is shown on Plant Drawings P-0001-0 through P-0004-0, and P-0011-0. It is a reinforced concrete enclosure that is rectangular in shape and is located in the northwest quadrant of the power complex, adjacent to the auxiliary building, which is to the south and east. The facility does not house any safety-related equipment and is not occupied, except for periodic surveillance.

The plant cancelled area bearing/shear walls are designed to resist lateral loads and transmit them to the reinforced concrete foundation mat, where all loads are dissipated into the Vincentown Formation. The reinforced concrete floors are supported by structural steel framing systems that are in turn supported by the walls. Floor systems are designed to act as diaphragms that transmit lateral loads to the shear walls. Radial framing is used within the cylindrical portion, while framing in the rectangular area is laid out on east-west and north-south lines. The central portion of the roof consists of cellular metal decking and built-up roofing material.

A seismic separation joint extends from the foundation mat through the roof at the south wall of the facility, where it interfaces with the Auxiliary Building control/SDG area. The plant cancelled area foundation mat, described in Section 3.8.5, extends eastward beyond

the facility to support the northern section of the Auxiliary Building radwaste/service area.

3.8.4.1.5 Station Service Water System (SSWS) Intake Structure

The Station Service Water System (SSWS) intake structure, as shown on Plant Drawings P-0071-0 and P-0072-0, houses four service water pumps and associated equipment, such as ice barriers, trash racks, traveling screens, and oil skimmer walls.

The SSWS intake structure is a reinforced concrete structure supported on a reinforced concrete foundation mat, as described in Section 3.8.5. The mat is founded on top of a tremie concrete plug, which is in turn founded on, and keyed into, the Vincentown Formation.

Bearing walls are designed as shear walls to resist and transfer lateral loads to the foundation mat, and thus through the tremie plug into the Vincentown Formation. All floors and the roof of the intake structure are of reinforced concrete and are designed to act as diaphragms that transmit lateral loads to the shear walls. There are no concrete masonry unit walls in the SSWS intake structure.

3.8.4.1.6 Condensate Storage Tank Dike

The condensate storage tank dike, as shown on Figure 3.8-36, is located in the yard adjacent to the Reactor Building. It is designed to contain the total volume of the condensate storage tank. The dike walls and foundation slab are provided with waterstops to prevent spillage from infiltrating into the surrounding soil.

3.8.4.1.7 Non-Seismic Category I Structures

3.8.4.1.7.1 Turbine Building

The Turbine Building design is shown on Plant Drawings N-1011 and P-0012-1 through P-0016-1.

The building houses the turbine generator unit and its attendant auxiliary equipment, including condensers, condensate pumps, moisture separators, air ejectors, feedwater heaters, reactor feed pumps, motor generator sets for reactor recirculation pumps, recombiners, interconnecting piping and valves, and switchgear. Two 220 ton overhead cranes, described in Section 9.1.5, are provided above the operating floor to service the turbine generator unit.

The building enclosure consists of exterior walls of reinforced concrete to Elevation 102 feet. Except where shielding is required, the enclosure above Elevation 102 feet is accomplished with precast concrete panels to Elevation 125 feet 6 inches and with insulated metal siding from Elevation 125 feet 6 inches to the roof. The roof has a nominal Elevation of 200 feet and consists of cellular metal decking, insulating board, and built-up roofing material.

Vertical loads are supported by reinforced concrete walls and structural steel columns. Generally, interior reinforced concrete walls and structural steel columns extend from the top of the base mat to Elevation 137 feet.

Floor slabs are reinforced concrete supported by structural steel framing. They are designed to act as diaphragms to resist lateral loads and transfer them to the shear walls. The reinforced concrete shear walls transfer the lateral loads to the reinforced concrete foundation mat, which dissipates them into the Vincentown Formation.

In the turbine generator bay, structural steel rigid frames spanning the east-west direction support roof loads, east-west lateral loads, and crane loads. North-south lateral loading is generally resisted by steel bracing and transferred into the shear walls at elevation 137 feet.

The turbine generator is supported by a free standing, reinforced concrete pedestal founded on the base mat and flush with the operating floor at Elevation 137 feet. The operating floor framing is supported on vibration damping pads that are in turn supported by

the pedestal. Separation joints are provided between the pedestal and walls and other Turbine Building floors to prevent transfer of vibration to the building.

The Turbine Building is isolated from the auxiliary building radwaste/service area by a seismic separation joint extending from the basemat through the roof in the north-south direction, and from the administration facility by a similar seismic separation joint extending in the east-west direction.

Some interior walls, required for separation, radiation shielding, or fire protection, are constructed of fully grouted, reinforced concrete masonry units.

3.8.4.1.7.2 Administration Facility

The administration facility, formerly the Unit 2 Turbine Building, is shown on Plant Drawings P-0001-0 through P-0005-0, and P-0010-0. The building houses the administrative offices and warehouse facility in support of plant operation. In addition, the former Unit 2 Turbine Building operating floor is accessible from the adjacent operating floor for use as a laydown area. The two overhead cranes provided to service the turbine generator unit can also operate over this laydown area.

The facility enclosure consists of exterior walls of reinforced concrete to Elevation 102 feet. The enclosure above elevation 102 feet is accomplished with precast concrete panels and window walls to Elevation 125 feet 6 inches, and with insulated metal siding and window walls from Elevation 125 feet 6 inches to the roof. The roof has a nominal Elevation of 200 feet and consists of cellular metal decking, insulating board, and built-up roofing material.

Vertical loads are supported by reinforced concrete walls and structural steel columns. Generally, reinforced concrete walls and structural steel columns extend from the top of the base mat to Elevation 137 feet.

Floor slabs are reinforced concrete supported by structural steel framing. They are designed to act as diaphragms to resist lateral loads and transfer them to the shear walls. The reinforced concrete shear walls transfer the lateral loads to the reinforced concrete foundation mat, described in Section 3.8.5, which dissipates them into the Vincentown Formation.

In the laydown area, structural steel rigid frames spanning the east-west direction support roof loads, east-west lateral loads, and crane loads. North-south lateral loading is generally resisted by steel bracing and transferred into the shear walls at elevation 137 feet.

The administration facility is isolated from the Auxiliary Building radwaste/service area by a seismic separation joint, extending from the basemat through the roof in the north-south direction, and from the Turbine Building by a similar seismic separation joint extending in the east-west direction.

Some interior walls, required for radiation shielding, area isolation, or fire protection, are constructed of fully grouted, reinforced concrete masonry units.

3.8.4.2 Applicable Codes, Standards, and Specifications

Table 3.8-7 lists the codes, standards, and specifications used in designing, fabricating, and constructing non-Seismic Category I structures discussed in Section 3.8.4.1, and Seismic Category I structures other than the primary containment and its internals, which are discussed in Sections 3.8.2 and 3.8.3. Applicable regulatory guides are discussed in Section 1.8.

3.8.4.3 Loads and Load Combinations

The following loads and load combinations are considered in the design of non-Seismic Category I structures discussed in Section 3.8.4.1, and Seismic Category I structures other than the

primary containment and its internals. Structures that directly affect the integrity of the reactor coolant pressure boundary (RCPB), the capability to shut down the reactor and maintain it in a safe shutdown condition, or the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the guideline exposures of 10CFR50.67, are designed to withstand the effects of these loads and load combinations.

3.8.4.3.1 Definition of Loads

3.8.4.3.1.1 Normal Loads

Normal loads are those encountered during normal plant startup, operation, and shutdown. They include dead loads, live loads, operating thermal loads, and operating pipe reaction loads.

3.8.4.3.1.1.1 Dead Loads (D)

Dead loads include the weight of framing, roofs, floors, walls, partitions, platforms, and all permanent equipment and materials. The vertical and lateral pressures of groundwater and liquids are also treated as dead loads.

Floors are designed for major equipment loads. For permanently attached small equipment, piping, conduit, and cable trays, a minimum of 50 psf is added.

Where piping is supported from platforms or walkway beams, actual loads are determined and used. After pipe hanger locations and loads for main piping are fully established, all structural members, including those already designed, are reviewed for structural adequacy and, if loads exceed design allowables, the members are reinforced to withstand the established loads.

3.8.4.3.1.1.2 Live Loads (L) and Operating Live Loads (L_o)

Live loads include any movable equipment loads and other loads that vary with intensity and occurrence, such as soil pressures, snow loads, pressure difference due to variation in heating and cooling, outside atmospheric changes, and the dynamic effects of operating equipment.

The design live loads designated as "L" include floor area loads, laydown loads, nuclear fuel and fuel transfer cask loads, equipment handling loads, and lateral earth pressure loads. The floor area live load is omitted from areas occupied by equipment whose weight is specifically included in dead load. Live load is not omitted under equipment where access is provided.

In load combinations including earthquake motions, the live loads are limited to the designation "L_o," which is defined as the live load expected to be present when the plant is operating. The L_o loads are applied simultaneously with the seismic forces. In the laydown areas, the actual weight of the equipment, as spread out on the floor, is considered L_o.

3.8.4.3.1.1.3 Operating Thermal Loads (T_o)

Operating thermal loads are based on the most critical transient or steady state condition to occur during normal operation.

3.8.4.3.1.1.4 Operating Pipe Reaction Loads (R_o)

Operating pipe reaction loads are based on the most critical transient or steady state condition.

3.8.4.3.1.2 Severe Environmental Loads

Severe environmental loads are those that could infrequently be encountered during the plant life and include operating basis earthquake seismic loads and severe wind loads. Components are

designed to remain within appropriately defined allowable stress limits when subjected to severe environmental loads.

3.8.4.3.1.2.1 Operating Basis Earthquake Seismic Loads (E_o)

The free field ground acceleration at the bottom of the foundation mat for the OBE is 0.1g. Refer to Sections 3.7.1, 3.7.2, and 3.7.3 for a detailed discussion of seismic requirements.

3.8.4.3.1.2.2 Severe Wind Loads (W)

Severe wind loads are as described in Section 3.3.

3.8.4.3.1.3 Extreme Environmental Loads

Extreme environmental loads are those that are credible but highly improbable and include safe shutdown earthquake seismic loads, tornado loads, and extreme wind and flood loads.

3.8.4.3.1.3.1 Safe Shutdown Earthquake Seismic Loads (E_s)

The free field ground acceleration at the bottom of the foundation mat for the SSE is 0.2g. Refer to Sections 3.7.1, 3.7.2, and 3.7.3 for a detailed discussion of seismic requirements.

3.8.4.3.1.3.2 Tornado Loads (W_t)

Tornado loads include wind velocity pressure loads (W_{tq}) and differential pressure loads (W_{tp}) as described in Section 3.3, and tornado generated missile impact loads (W_{tm}), as described in Section 3.5.3.

3.8.4.3.1.3.3 Extreme Wind and Flood Loads (W_e)

Loads under extreme wind conditions are based on the probable maximum hurricane (PMH) and concurrent flood resulting from wind and tidal action, as discussed in Sections 2.4.2 and 3.3.1.

3.8.4.3.1.4 Abnormal Loads

Abnormal loads are those generated by a postulated high energy pipe break.

3.8.4.3.1.4.1 Abnormal Pressure Loads (P_a)

Abnormal pressure loads within or across a compartment and/or structure result from postulated pipe rupture. The time dependent nature of the load and the ability of the structure to deform beyond yield is considered in establishing the structural capacity necessary to resist the effects of P_a.

3.8.4.3.1.4.2 Abnormal Thermal Effects (T_a)

Abnormal thermal effects result from thermal conditions generated by the postulated pipe rupture. T_a includes the effects of T_o.

3.8.4.3.1.4.3 Abnormal Pipe Reaction Loads (R_a)

Abnormal pipe reaction loads result from the thermal conditions generated by the postulated pipe rupture and include R_o.

3.8.4.3.1.4.4 Abnormal Local Effects (R_r)

Abnormal local effects on structures are due to postulated pipe rupture. The local effects include the following:

1. R_{rr} - Load on the structure generated by the reaction of a ruptured high energy pipe during the postulated event. The time dependent nature of the load and the ability of the structure to deform beyond yield is considered in establishing the structural capacity necessary to resist the effects of R_{rr}.

2. R_{rj} - Load on the structure generated by the jet impingement from a ruptured high energy pipe during the

postulated event. The time dependent nature of the load and the ability of the structure to deform beyond yield is considered in establishing the structural capacity necessary to resist the impact.

3. R_{rm} - The energy resulting from the impact of a ruptured high energy pipe on a structure or pipe restraint during the postulated event. The type of impact, together with the ability of the structure to deform beyond yield, is considered in establishing the structural capacity necessary to resist the impact.

The jet forces used to evaluate R are determined using methods and procedures discussed in Section 3.6.2.

3.8.4.3.2 Load Combinations

Tables 3.8-8 through 3.8-11 list the load combinations, applicable load factors, and allowable limits used in the design of the applicable structure. Table 3.8-12 summarizes the symbols used in the load combinations.

Maximum effects of P_a, T_a, R_a, and R_r are combined, unless a time history analysis is performed to justify lower combined values.

In addition to the combinations listed in Tables 3.8-8 through 3.8-11, the following combinations for W_t and R_r are also a design requirement, where applicable, for Seismic Category I structures:

1. Tornado effects W_t:

   a. W_{tq}, W_{tp} or W_{tm} acting independently

   b. $W_{tp} + 0.5\ W_{tp}$

   c. $W_{tq} + W_{tm}$

d. $W_{tq} + 0.5\ W_{tp} + W_{tm}$

2. Local effects of pipe rupture R_r:

a. R_{rj} or R_{rr} acting independently

b. $R_{rr} + R_{rm}$

c. $R_{rr} + R_{rm} + R_{rj}$

The central portion of the roof of the plant cancelled area is not designed to withstand the tornado effects of item 1. above.

3.8.4.4 Design and Analysis Procedures

3.8.4.4.1 Seismic Category I Structures

The Seismic Category I structures described in Section 3.8.4.1 are designed to maintain elastic behavior for the loads and load combinations described in Section 3.8.4.3, except for dynamic loads generated by abnormal pressure and abnormal local effects. All reinforced concrete components of the structure are designed by the strength method per ACI 318, as listed in Table 3.8-7. Generally, all structural steel components are designed by the working stress method per AISC specifications listed in Table 3.8-7, except for dynamic loads generated by abnormal pressure and abnormal local effects. For dynamic loads generated by abnormal pressure and abnormal local effects, the structural members are allowed to exceed the yield strain and displacement values, since the impulse loads are short term and missile impact has a defined input energy limit.

Seismic design of structures is described in Sections 3.7.1, 3.7.2, and 3.7.3. The structures are analyzed dynamically.

Design of structures for missile protection is covered in Section 3.5.3.

There are no concrete masonry unit walls in Seismic Category I structures.

Appendix 3F discusses the design and analysis procedures for the fuel pool liner and slab.

3.8.4.4.2 Non-Seismic Category I Structures

The non-Seismic Category I structures described in Section 3.8.4.1 are designed to maintain elastic behavior for the loads and load combinations described in Tables 3.8-9 and 3.8-11. In addition, the Turbine Building and administration facility are checked to verify that they do not collapse on, or interact with, adjacent Seismic Category I structures for certain abnormal and extreme environmental conditions, as described in Tables 3.8-9 and 3.8-11. All reinforced concrete components of the structure are designed by the strength method per ACI 318, as listed in Table 3.8-7. Structural steel components are designed by the working stress method per AISC specifications listed in Table 3.8-7.

The Turbine Building and administration facility are designed in accordance with the criteria established by the UBC, as listed in Table 3.8-7, for structures in Seismic Zone No. 1, together with any additional requirements stated herein.

To provide assurance that the turbine building and administration facility will not collapse due to SSE ground motions, they are analyzed using dynamic techniques. These structures are designed to accommodate an SSE event by the following methods:

1. Reinforced concrete elements are designed for ductile behavior in accordance with UBC or for elastic-plastic behavior provided its ductility factor does not exceed 3 and structural resistance is based on Section Strength (U) for concrete.

2. Structural steel elements are designed by the working stress method or for elastic plastic behavior provided its ductility factor does not exceed 3.

Concrete masonry unit walls in the non-Seismic Category I turbine building and administration facility are used only for radiation shielding, fire separation, and miscellaneous supports, and are designed for vertical loading and seismic loading in accordance with the UBC, as listed in Table 3.8-7.

3.8.4.4.3 Computer Programs

Computer programs used in the design and analysis of the Seismic Category I and non-Seismic Category I structures described in Section 3.8.4.1 are discussed in Appendix 3A.

3.8.4.5 Structural Acceptance Criteria

3.8.4.5.1 Reinforced Concrete

The reinforced concrete structural components are designed by the strength method in accordance with ACI 318, as listed in Table 3.8-7, for loads and load combinations described in Section 3.8.4.3. The margins of safety are contained in the capacity reduction factors (u) specified in the code. Table 3.8-18 provides the allowable ductility ratios used in design for impactive and impulsive loading. A review of the design of flexural beams and slabs indicates that the actual ductility ratios are less than the allowable ductility ratios in Regulatory Guide 1.142.

3.8.4.5.2 Structural Steel

Generally, structural steel components are designed by the working stress method in accordance with AISC specifications, as listed in Table 3.8-7, for loads and load combinations described in Section 3.8.4.3. The allowable stresses for different load combinations are also indicated in Tables 3.8-10 and 3.8-11. The

margins of safety are contained in the allowable design stresses. Table 3.8-19 provides the allowable ductility used for impactive and impulsive loading.

3.8.4.5.3 Concrete Masonry Unit Walls

Concrete masonry unit walls are used only for radiation shielding, fire separation, or miscellaneous supports in the non-Seismic Category I Turbine Building and administration facility. They are not shear walls and are designed to the working stress method of UBC, as listed in Table 3.8-7.

3.8.4.6 Materials, Quality Control, and Special Construction Techniques

Materials, quality control, and special construction techniques are discussed in Section 3.8.6.

3.8.4.7 Testing and Inservice Inspection Requirements

Testing and inservice inspection are not required for Seismic Category I structures other than the primary containment and its internals.

3.8.4.8 SRP Rule Review

3.8.4.8.1 Concrete Design

Acceptance Criteria II.2 of SRP 3.8.3 and 3.8.4 requires that Category I structures be designed in accordance with Specification ACI 349 as augmented by Regulatory Guide 1.142. The HCGS design was based on the requirements of Specification ACI 318-71.

The Category I structures concrete design for HCGS began prior to the issue of Specification ACI 349 (1976). As a result, all concrete design is based on using Specification ACI 318-71 with the following clarifications:

A review of the design of the HCGS Seismic Category I structures indicates that there is no impact due to differences in the structural acceptance criteria between ACI 318-71 and ACI 349-76 as augmented by Regulatory Guide 1.142.

The load combinations used are in conformance with the following SRP sections except that the 0.9 load factor on dead load as required by ACI 349-76 was not used:

Structures	SRP Section
Primary Containment Internal Concrete Structures	3.8.3.II.3.b.
Other Seismic Category I Concrete Structures	3.8.4.II.3.b.

Based on parametric analyses, an adequate design margin exists to compensate for the effects of the reduced dead load factor.

Table 3.8-18 provides a comparison of the allowable ductility ratios used for design of the concrete structural components subjected to impactive and impulsive loadings and the criteria outlined in Appendix C of ACI 349 as modified by Regulatory Guide 1.142. The criteria in Appendix C of ACI 349 as modified by Regulatory Guide 1.142 is referenced in Appendix A of NUREG-0800, SRP Section 3.5.3.

Except for flexural beams and slabs subjected to impactive loads, the allowable ductility ratios used in the design are less than or equal to those in the Regulatory Guide. The allowable ductility ratios for beams and slabs used in design are based on the evaluation of test data reported in References 3.8-5 and 3.8-6 and tests performed by the Architect/Engineer.

The test results consistently demonstrate that actual ductility ratios in excess of 50 are reached prior to failure. Therefore, by limiting the values to 10 for beams and 30 for slabs, the design is

conservative. Furthermore, the flexural members are designed to meet additional reinforcing requirements (See Table 3.8-18) to ensure ductile behavior. A review of the design of flexural beams and slabs indicates that the actual ductility ratios are less than the allowable ductility ratios in Regulatory Guide 1.142.

3.8.4.8.2 Structural Steel Design

Table 3.8-19 provides a comparison of the allowable ductility ratios used for design of structural steel subjected to impactive and impulsive loading, and the criteria outlined in Appendix A of NUREG-0800, SRP Section 3.5.3. Except for flexure in beams subjected to impactive loads (other than the tornado missiles) and axial tension members subject to impulsive loads, the ductility ratios are essentially identical. Based on the recommendations provided in References 3.8-5 and 3.8-6 and tests performed by the Architect/Engineer, it has been demonstrated that steel members under flexural loads can sustain higher ductility ratios (on the order of 30) without collapse. Therefore, a limiting value of 20 used in the design is conservative. Furthermore, additional design and fabrication features (such as box sections, lateral bracings, NDE, etc.) are incorporated in the flexural members to preclude buckling and to ensure material quality.

As a follow-up of the NRC Structural Audit, all flexural beams subjected to impactive loads (other than tornado missiles) have been reevaluated utilizing final design parameters. This reevaluation revealed that the actual ductility ratios are less than or equal to the allowable ductility ratios in Appendix A of NUREG-0800, SRP Section 3.5.3.

Regarding the ductility ratio for axial tension members subject to impulsive loads, the HCGS limit of 3 is always conservative for the types of steel used.

3.8.4.8.3 Spent Fuel Rack Design

Acceptance Criterion II.4.f requires that the spent fuel racks be designed in compliance with Appendix D of SRP 3.8.4, which requires that construction materials should conform to Section III, Subsection NF of the ASME Code.

The design, analysis and fabrication of the spent fuel racks conforms with the applicable provisions of Subsection NF. See Appendix 9B for a description of the design, analysis and construction of the racks.

The spent fuel racks are constructed of ASTM A-240 and ASTM A-564 stainless steel. The A-240 and A-564 material specifications are identical to the ASME SA-240 and SA-564 material specifications. All rack steel is supplied with certified material test reports.

The rack materials are procured under a Q.A. Program that is intended to comply with:

1. 10CFR50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants".

2. ANSI/ASME N45.2, "Quality Assurance Program Requirements for Nuclear Facilities", and

3. ANSI/ASME NQA-1, "Quality Assurance Program Requirements for Nuclear Power Plants".

3.8.5 Foundations

Foundations for all Seismic Category I structures and the turbine building and the administration facility, which are non-Seismic Category I structures, are described in this section.

3.8.5.1 Description of the Foundations

The configuration of the foundation mats for the various structures is shown on Figure 3.8-37.

Reinforced concrete mat foundations are provided for all structures. Except for the Station Service Water System (SSWS) intake structure, the mats rest either on the Vincentown Formation or on engineered structural backfill placed on the Vincentown Formation. The mat and the lean concrete leveling course for the intake structure rest on a tremie concrete plug supported by the Vincentown Formation.

Bearing walls of the structures are rigidly connected to the foundations. Steel columns are attached to the foundation 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 roof and floor diaphragms. The shear walls in turn transfer the horizontal shears to the foundation mats. The mats transfer all loads to the Vincentown Formation through friction and/or direct bearing.

All mats, except that for the SSWS intake structure, are 14 feet thick and are constructed in two lifts. Additional shear reinforcement is provided at the horizontal joints where necessary. The thickness of the mat for the SSWS intake structure varies between 6 feet and 4 feet 6 inches and is constructed in one lift. Each concrete pour is placed in a "checkered-board" pattern to minimize the effects of concrete shrinkage and heat of hydration.

In the power block area, a leveling mud mat, an unreinforced concrete layer, is provided beneath the concrete topping mat to facilitate construction and installation of the waterproofing membrane. A multiple waterproofing membrane is provided on the leveling mat and on the outside face of the peripheral walls below grade. In the case of the SSWS intake structure, the exterior walls

and the bottom of the structural mat are protected with a waterproofing system.

Each main foundation mat is separated from the others by a seismic joint at least 2-inches wide. Piping and conduit crossing seismic joints are provided with sufficient flexibility to accommodate a 3/4-inch post-earthquake differential settlement. The basis for estimating post-earthquake differential settlement, as indicated in Section 2.5.4.8.3, is based on analytical procedures developed by Lee and Albaisa (Reference 2.5-110). This settlement is considered to have an insignificant effect on the structural design of the base mat.

Piping which crosses a seismic joint is analyzed for building settlement effects assuming a 3/4 inch relative vertical displacement between the first vertical rigid support on both sides of the seismic joint. The stresses generated in the pipe as a result of differential settlement is evaluated against ASME B&PV Section III code allowables equal to 3S (Ref: NC/ND-3652.3) where S is the basic material allowable stress value at room temperature. The loads on supports are accounted for in the design of pipe supports. Electrical conduit crossing a seismic joint are provided with flexible couplings and fittings as shown in Figures 3.8-45 and 3.8-46.

Peripheral subterranean walls are designed to resist lateral pressures due to backfill, groundwater, flood, and surcharge loads in addition to dead loads, live loads, and seismic loads.

Figures 3.8-38 through 3.8-43 show details and Table 3.8-13 summarizes descriptions of the foundations.

3.8.5.1.1 Reactor Building and Southern Section of the Radwaste/Service Area of the Auxiliary Building

The foundation mat for the Reactor Building and the southern section of the radwaste/service area is poured to act as a single slab, as

shown for Mat 3 on Figures 3.8-37 and 3.8-40. The mat is typically reinforced with No. 18 bars at 26-inch centers on the top, and No. 18 bars at 13-inch centers on the bottom. A second layer of the same reinforcement is provided in the area supporting the primary containment. Vertical shear reinforcement is provided with No. 10 bars typically located in a 26-inch by 52-inch grid pattern.

3.8.5.1.2 Auxiliary Building Control/Diesel Generator Area and Central Section of the Radwaste/Service Area

The foundation mat for the control/diesel generator area and the central section of the radwaste/service area is poured to act as a single slab, as shown for Mat 5 on Figures 3.8-37 and 3.8-42.

The mat is typically reinforced with No. 18 bars at 26-inch centers at both the top and the bottom. No. 11 bars are provided for additional reinforcement, where required. Vertical reinforcement is provided by No. 10 bars, where required.

3.8.5.1.3 Plant Cancelled Area and Northern Section of the Auxiliary Building Radwaste/Service Area

The foundation mat for the plant cancelled area and the northern section of the radwaste/service area is poured to act as a single slab, as shown for Mat 4 on Figures 3.8-37 and 3.8-41. The mat is typically reinforced with No. 18 bars at 26-inch centers on both top and the bottom in multiple layers. Vertical shear reinforcement is provided with No. 10 bars typically located in a 26-inch by 52-inch grid pattern.

3.8.5.1.4 SSWS Intake Structure

The foundation mat for the SSWS intake structure is 4 feet 6 inches thick on the waterfront side and 6 feet thick on the landward side. It is placed in four blocks to act as a single slab, as shown for Mat 6 on Figures 3.8-37 and 3.8-43. Typical reinforcement consists

of No. 10 bars at 12-inch centers. No. 7 bars are provided for additional reinforcement, where required.

3.8.5.1.5 Turbine Building and Administration Facility

The foundation mat for the Turbine Building is shown as Mat 1 and the administration facility as Mat 2 on Figure 3.8-37, and on Figures 3.8-38 and 3.8-39, respectively. Both mats are typically reinforced with No. 18 bars at 26-inch centers on both top and bottom. No. 11 bars are provided for additional reinforcement, where required.

3.8.5.2 Applicable Codes, Standards, and Specifications

Codes, standards, and specifications used in the design, fabrication, and construction of foundations of the structures are listed in Table 3.8-15.

3.8.5.3 Loads and Load Combinations

Loads and load combinations used in the foundation mat design are described in Section 3.8.4. In addition, the following load combinations are considered in order to determine the factor of safety against sliding and overturning due to winds, tornadoes, seismic loads, and against flotation due to groundwater and design basis flood:

1. $D + H + W$
2. $D + H + W_t$
3. $D + H + E_o$
4. $D + H + E_s$
5. $D + F$

where D, E_o, W, E_s and W_t are defined in Section 3.8.4, and H and F are the lateral earth pressure and buoyant force due to design basis flood, respectively.

3.8.5.4 Design and Analysis Procedures

The foundations are designed to maintain elastic behavior under different loads and load combinations. Loads and load combinations are described in Sections 3.8.4 and 3.8.5.3. The design and analysis of the reinforced concrete foundations are carried out in accordance with ACI 318. Appendix 3D contains critical sections, loads, and a discussion of how these loads are accommodated in the Reactor Building and the southern section of the radwaste/service area, and the Auxiliary Building control/diesel generator and central section of radwaste/service area basement designs.

Bearing walls and steel columns carry all the vertical loads from the structure to the foundation mat. Lateral loads are transferred to the shear walls by the roof and floor diaphragms. The shear walls then transmit the loads to the foundation mat.

The loads on the mats are determined using finite element analysis program BSAP, as discussed in Appendix 3A. The adjacent mats, the supporting and surrounding soils, and the stiff load bearing walls are included in the model to determine their effects.

Settlement of the foundations of the Seismic Category I structures is considered in the design. Estimated settlement is discussed in Section 2.5.4.10.

Stability against sliding is ensured by dead weight of the structures, the subgrade soil friction, and lateral soil resistance to the foundations. The SSWS intake structure is provided with additional anchorage by having a shear key installed at the bottom of the tremie concrete. Stability against overturning is ensured by the dead weight and lateral soil resistance.

Appendix 3G contains a discussion of the intake structure stability analysis.

A detailed description of the foundation bearing stratum is given in Section 2.5. The calculated bearing pressure is within allowable limits, as discussed in Section 2.5.4.11. Summaries of stability calculations are provided in Appendixes 3G and 3H.

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 Section 3.8.4. In addition, for the additional load combinations a. through e. delineated in Section 3.8.5.3, the calculated factor of safety against overturning, sliding, and flotation exceeds the following minimum values:

Load Combination		Minimum Factors of Safety		
		Overturning	Sliding	Flotation
1.	D + H + W	1.5	1.5	-
2.	D + H + W	1.1	1.1	-
3.	D + H + E	1.5	1.5	-
4.	D + H + E	1.1	1.1	-
5.	D + F	-	-	1.1

3.8.5.6 Materials, Quality Control, and Special Construction Techniques

3.8.5.6.1 Materials and Quality Control

The foundation mats of the Seismic Category I structures are constructed of reinforced concrete. Concrete and reinforcing steel materials are discussed in Section 3.8.6. The concrete design strength is generally 4000 psi, except for the 2500 psi tremie concrete fill beneath the base mat of the SSWS intake structure.

3.8.5.6.2 Construction Techniques

3.8.5.6.2.1 Power Block Complex

In the general area where the reactor building, the control/diesel generator area and radwaste/service area of the Auxiliary Building, the administration facility, and the Turbine Building are located, an open cut excavation is made by the hydraulic dredge method to a depth where competent Vincentown Formation is exposed.

The excavation is controlled so that the integrity of the bearing stratum is maintained. Because of the high water table, a dewatering system is installed and operated to facilitate excavation. The groundwater level is maintained below the water level in the excavation pool at all times so there is never an upward flow of groundwater within the excavated area.

Groundwater levels and dewatering discharge are monitored periodically to ensure proper functioning of the dewatering system.

Upon completion of the excavation, the excavated area is dewatered, and final cleanup is performed. Adequate protection of the foundation stratum from frost and construction equipment is provided by engineered backfill and/or lean concrete cover.

To reach the base mat construction level, engineered backfill, as required, is placed on the Vincentown Formation. Material descriptions and the placing requirements for engineered backfill are given in Section 3.8.6.

Requirements for decommissioning the dewatering system are shown in Table 3.8-14. The dewatering system is decommissioned, since sufficient dead load is provided to protect the structure under construction from the effects of overturning, sliding, or flotation.

3.8.5.6.2.2 SSWS Intake Structure

The SSWS intake structure is constructed by installing a steel sheet cofferdam around the perimeter of the required structural excavation. The soil within the cofferdam is excavated underwater to predetermined elevations and line. Upon completion, the excavation is inspected by a diver trained in geology. Finally, a tremie concrete plug is placed in the cofferdam up to the bottom levels of the structural base mat. Following the placement and curing of the tremie concrete, the cofferdam is dewatered and the superstructure is constructed.

3.8.5.7 Testing and Inservice Inspection Requirements

Foundation testing and inservice inspection involves monitoring the structures to detect any settlement that might occur.

During construction, the base mats are checked periodically for any settlement. They are also checked for settlement when significant loads are added to the mats. Actual settlement readings must compare reasonably with predicted values.

Settlement is monitored periodically during operation of the plant unless settlement is stabilized prior to startup.

3.8.6 Materials of Construction

This section discusses the materials of construction, workmanship, and quality control used to construct the Seismic Category I structures of HCGS.

3.8.6.1 Engineered Backfill

The engineered backfill is installed adjacent to and underneath Seismic Category I structures. These backfill areas and related cross sections are shown on Figure 3.8-44. The engineered backfill

also supports the Seismic Category I service water pipes above the Kirkwood Formation, located outside the main excavation area.

3.8.6.1.1 Material Selection

The material used for engineered backfill consists of pit run sand, silty sand, sandy gravel, or gravelly sand with not more than 20 percent by weight passing the No. 200 U.S. standard sieve size. The particles of the backfill material consist of sound, dense, and durable material.

The sources of backfill material are the Oldman's Pit and Hitchner Borrow. Representative samples of the material from each source are tested prior to use, to determine both static and dynamic properties. The results of testing are reported in:

1. Dames and Moore Report - Additional Site Stability Evaluation, Hope Creek Generating Station, Appendix IV-B, Evaluation of Structural Backfill - Oldman's Borrow Source, December 1976.

2. Dames and Moore Report - Liquefaction Potential Analysis for Backfill, Power Block Area, Hope Creek Generating Station, April 1977.

3. Supplementary Borrow Area Investigation for Structural Backfill, October 1980.

A program of testing and inspection is carried out prior to and during placement to confirm that backfill material is in conformance with the approved borrow material and placement requirements.

3.8.6.1.2 Installation and Compaction

The engineered backfill is spread in uniform lifts not exceeding 8 inches in loose thickness and compacted to an average of

98 percent and a minimum of 95 percent of the maximum dry density, as determined by ASTM D 1557, Method D.

Test embankments are constructed and tested for use in establishing final placing, compaction procedures, and techniques for the backfill operation. A new test embankment is constructed for each new type of equipment and/or different source of borrow material.

The test embankment is used to establish the lift thickness, moisture conditioning procedure, and the number of passes for each type of compaction equipment required to achieve the specified degree of compaction. The type of equipment used for the backfill is the same type used in the test embankment.

In areas where field density test results are less than the required degree of compaction, the backfill is replaced and/or recompacted to attain the required degree of compaction. Similarly, any areas that are previously accepted and later become disturbed or loosened, are replaced and/or recompacted to the required degree of compaction.

The excavated subgrade is thoroughly proof rolled prior to the installation of the initial backfill lift to recompact any areas of its surface that have become disturbed during construction operations. Successive lifts of backfill are not installed upon frozen subgrade or backfill soils. Any such frozen material is removed and suitably thawed or broken up before it is considered for reuse.

The moisture content of the backfill is adjusted as required to facilitate compaction. All backfill areas are appropriately graded during installation to facilitate surface drainage and to prevent any local ponding.

3.8.6.1.3 Testing of Backfill Material

3.8.6.1.3.1 Field Density Tests

The degree of compaction attained during backfill operations is verified by performing field density tests in accordance with ASTM D 1556 and/or ASTM D 2922.

If nuclear methods are employed for the above test procedures, the ratio of density tests performed in accordance with ASTM D 2922 procedures to ASTM D 1556 procedures is a maximum of three to one.

For self-propelled vibratory compactors, the minimum number of tests is one for every 250 cubic yards of backfill placed, or one test per lift covering 10,000 square feet of surface area, or one test for every day of compaction operation, whichever is more frequent.

For hand compactors, the minimum number of tests is one for each day of backfill placement, or 50 cubic yards in-place, whichever is more frequent.

3.8.6.1.3.2 Laboratory Tests

A minimum of one Proctor test in accordance with ASTM D 1557, Method D, is performed for every 1000 cubic yards of backfill placed. Material for this test is obtained from the field density test performed to ASTM D 1556.

3.8.6.1.3.3 Gradation Tests

Gradation tests for the backfill material are performed in accordance with ASTM D 422, with the exception of the hydrometer test (Paragraph 9, ASTM D 422 and ASTM D 1140) at the minimum rate of one test for every 1000 cubic yards, or one test per day during backfilling operations, whichever is more frequent.

3.8.6.2 Concrete and Concrete Materials

The codes, standards, and recommendations used for construction are listed in Table 3.8-15. Some of these documents are modified to suit the particular conditions of design and construction associated with nuclear power plants without compromising structural adequacy. The extent of application and the principal exceptions are indicated herein.

3.8.6.2.1 ACI 301

ACI 301 is modified as follows:

1. Section 8.3.4 - ACI 309 is used in lieu of Section 8.3.4.

2. Section 8.4.3 - ACI 305 and ACI 306 are used in lieu of Section 8.4.3.

3. Section 12.2.1 is revised to state:

   "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 for a localized area may be interrupted as necessary, for a period not to exceed 8 hours provided that requirements for weather protection are maintained. Such curing process may be interrupted provided that the local surface area has received a minimum of 48 hours of continuous curing prior to the interruption."

4. Section 12.2.3 is revised to state:

   "Curing in accordance with Section 12.2.1 or 12.2.2 shall be continued 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. The required period of initial curing need not be greater than the lesser of the two periods. If one of the curing procedures of Sections 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 1 day old, provided the concrete is not permitted to become surface dry during the transition. Curing during periods of cold weather shall be in accordance with Section 12.3.1."

5. Section 12.3.1 is replaced with:

   "Initial curing and protection measures for the concrete during periods of cold weather shall be in accordance with the recommendations of ACI 306."

6. Section 14.4.1 is replaced with:

   "The slump of the concrete as placed shall be 3 inches or less, unless indicated otherwise, except that a tolerance of up to 2 inches above this maximum shall be allowed for occasional batches, provided that the concrete supplier is notified to reduce the slump. Failure to comply shall be cause for rejection of the concrete. Concrete of lower than usual slump may be used provided that it is properly placed and consolidated."

7. Section 14.4.3 The first sentence is revised to state:

   "Concrete shall be placed in layers approximately 24 inches thick."

8. Section 14.5.1 is revised to state:

"The minimum curing period shall be 14 days after the concrete has been placed, when the mean daily air temperature is 50°F or more. During cold weather, the curing period shall be 7 days."

9. Section 14.5.4 is replaced with:

"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."

3.8.6.2.2 ACI 318

ACI 318 is modified as follows:

1. Section 3.5.1(a) is revised as follows:

"Specification for Deformed Billet-Steel Bars for Concrete Reinforcement" (ASTM A 615). No. 14 and No. 18 bars shall be subject to the bend test of supplementary requirement, S1 of ASTM A 615. Full section bars shall be bent 90 degrees, at a minimum temperature of 60°F, around an eight-bar-diameter pin without cracking transverse to the axis of the bar."

2. Section 5.4.4 is revised as follows:

"Starter mixes, defined as concrete with 3/4-inch maximum size aggregate and a slump of 6 to 8 inches, shall be used as an alternative to mortar."

3. Section 5.5 is modified by the addition of a new section:

"5.5.3. The curing requirements described in Sections 5.5.1 and 5.5.2 may be interrupted, as necessary, for a period not to exceed 8 hours, provided that requirements for weather protection are maintained. Such curing may be interrupted, provided the local surface has received a minimum of 48 hours of continuous curing prior to the interruption."

4. Section 6.3.2.4 is replaced with:

"Embedded piping joints, temporary or permanent, except as noted in Section 6.3.2.5, shall be leak tested prior to the placement of concrete. Leak-testing shall be in accordance with the requirements of the code governing that piping system e.g., ASME Boiler and Pressure Vessel Code, ANSI B31.1, state or local plumbing codes, etc.

5. Section 6.3.2.5 is replaced with:

"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. Section 6.4.1 is replaced with:

"Joints not indicated on the plans shall be made and located so as not to significantly impair the strength of the structure. Where a joint is to be made, the surface of the concrete shall be thoroughly cleaned and all laitance and standing water removed. Vertical construction joints shall be cleaned and roughened by waterblasting, sandblasting, or bush hammering after the concrete reaches its final set. Prior to receiving additional concrete, vertical construction joints shall be wetted."

7. Section 7.3.2 and related sections are replaced with:

"7.3.2 Tolerances - Reinforcement shall be placed within the following tolerances:

7.3.2.1 For clear concrete protection from the surface of the reinforcement to the concrete surface in flexural members, walls, and compression members where member thickness is:

Member Thickness	Reduction in Nominal Cover	Increase in Nominal Cover
Less than 12 inches	3/8 inch	3/8 inch
12 inches or more	1 inch	1 inch
Base mat (nominal 14-feet thick)	1 inch	1-1/2 inch

7.3.2.2 For longitudinal location of bends and ends of bars: ~3 inches, provided that specified nominal cover at the ends of members shall not be reduced by more than 1/2 inch."

8. Section 7.4.1 is replaced with:

"Bar spacing: ~4 bar diameters for No. 8 bars and less, ~2 bar diameters for other bar sizes, except that the minimum clear distance between parallel bars in a layer shall be not less than the nominal diameter of the bar, nor less than 1 inch. Where parallel reinforcement is placed in two or more layers, the bars in the upper layer shall be placed directly above those in the bottom layer, with the clear distance between the bars not less than 1 inch, unless specifically shown otherwise on the design drawings. The total number of bars shall be maintained."

3.8.6.2.3 Cement

Cement is Type II, Portland cement conforming to ASTM C 150. Certified copies of material test reports showing chemical composition of the cement and verification that the cement complies with requirements are furnished by the manufacturer for each load of cement delivered.

For every 6000 bbl of cement delivered, or for each silo of cement certified at the mill, confirmatory tests consisting of complete chemical and physical analyses are performed by the concrete supplier.

3.8.6.2.4 Normal Density Aggregates

Fine and coarse aggregates conform to ASTM C 33. Aggregate source acceptability is based on the following test requirements:

Method of Test	Designation
Organic impurities in sands	ASTM C 40
Effect of organic impurities in fine aggregate on strength of mortar	ASTM C 87
Soundness of aggregates	ASTM C 88
Materials finer than No. 200 sieve	ASTM C 117
Specific gravity and absorption of coarse aggregate	ASTM C 127
Specific gravity and absorption of fine aggregate	ASTM C 128
L.A. abrasion	ASTM C 131

Method of Test	Designation
Sieve or screen analysis of fine and coarse aggregates	ASTM C 136
Clay lumps and friable particles	ASTM C 142
Potential reactivity of aggregates	ASTM C 289
Petrographic examination	ASTM C 295

Coarse aggregate grading is for size numbers 2, 4, 8, and 67, as defined in ASTM C 33, and the quantity of flat and elongated particles is limited to 15 percent by weight in any nominal size group.

Coarse aggregate loss from the L.A. abrasion test (ASTM C 131) using Grading A is limited to 40 percent by weight at 500 revolutions.

3.8.6.2.5 High Density Aggregates

The requirements for high density aggregates are the same as for normal density aggregates, except as noted below.

Fine and coarse aggregates conform to ASTM C 637, except that grading is as follows:

3/8-Inch Maximum Size Aggregate

Sieve Size U.S. Std. Square Mesh	Percentage Passing
3/8 inch	100
No. 4	75-95
No. 8	55-85
No. 16	30-60

Sieve Size U.S. Std. Square Mesh	Percentage Passing
No. 30	15-45
No. 50	10-30
No. 100	5-15

The fineness modulus is not less than 3.0, nor more than 3.8.

3/4-Inch Maximum Size Aggregate

Sieve Size	Percentage Passing	
U.S. Std. Square Mesh	Fine Aggregate, Sand	Coarse Aggregate, 3/4 inch
1 inch	-	100
3/4 inch	-	90-100
3/8 inch	100	20-55
No. 4	95-100	2-15
No. 8	75-100	0-8
No. 16	50-85	-
No. 30	25-60	-
No. 50	10-30	-
No. 100	5-15	-

The fineness modulus of the fine aggregate is not less than 2.4 nor more than 2.9.

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 C 117 and specific gravity requirements when tested per ASTM C 127 and C 128.

3.8.6.2.6 Pozzolan

Pozzolan, when used, conforms to ASTM C 618 for Class F, except that the maximum loss on ignition is 6 percent. Prior to use, a minimum of one sample is taken and tested for physical and chemical properties listed in ASTM C 311, except for pozzolanic activity with lime, tests for air entrainment, and for alkalies.

3.8.6.2.7 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. Such water and ice do not contain impurities that would cause either a change in the setting time of Portland cement of more than 25 percent, as determined in accordance with ASTM C 266, or a reduction in compressive strength of mortar of more than 10 percent, compared with results obtained with distilled water. The water and ice do not contain more than 250 ppm of chlorides as Cl, or more than 1000 ppm of sulphates as SO_4. The pH range is between 5.0 and 9.75.

3.8.6.2.8 Admixtures

Air entraining admixtures, when used, conform to ASTM C 260. Water reducing and retarding admixtures, when used, conform to ASTM C 494 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.

3.8.6.2.9 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 C 39
Unit weight	ASTM C 138
Slump	ASTM C 143
Air content	ASTM C 231

The following additional properties of selected mix designs are determined to ascertain material compatibility with design assumptions:

Property	Test Designation
Modulus of elasticity	ASTM C 469
Modulus of rupture	ASTM C 78
Heat content:	
Heat of hydration	ASTM C 186
Specific heat (heat capacity)	
Density	ASTM C 138

3.8.6.2.10 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. Two cylinders are tested for each mix design and age as follows:

Pozzolan Mix	Nonpozzolan Mix
7 days	7 days
28 days	28 days
90 days	

Concrete mixes for limited use, such as in radiation-sensitive facilities and high density concrete, do not contain pozzolan. All other concrete mixes have approximately 15 to 30 percent pozzolan by weight as cement replacement. Concrete mixes, except limited application use, such as high density concrete, are based on 3 to 6% air entrainment for both 3/4 and 1-1/2 inch nominal maximum size aggregate mixes. These measures provide a concrete with 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.

3.8.6.2.11 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 C 109.

3.8.6.2.12 Starter Mix

Starter mixes are used in such applications as at the bottom of foundation slabs or in lieu of construction grout and are proportioned from the same materials as for concrete. These mixes are generally proportioned for "working limit" slump 3 inches greater than the associated concrete mix. Trial mixes are prepared and tested for strength as described for general concrete mixes.

3.8.6.2.13 Nonshrink Grout

Nonshrink grout is prepared from proprietary materials. The grout is proportioned in accordance with the manufacturer's recommendations and is tested for expansion, compressive strength, and flow characteristics with maximum water content recommended by the manufacturer, prior to use.

3.8.6.2.14 Storage and Handling

Storage and handling of aggregates, cement, pozzolan, and admixtures is in accordance with the recommendations of ACI 301, Chapters 2, 3, 7, and 14, and ACI 304, Chapters 1, 2, 3, 4, and 6. Additionally, storage of cement and admixtures is in accordance with ANSI N45.2.2, Section 6.2.

3.8.6.2.15 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 of all concrete conform to ASTM C 94, except as otherwise noted.

3.8.6.2.16 Conveying and Placing

Conveying and placing of concrete is in accordance with the recommendations of ACI 301, Chapters 6 and 8 through 15, and ACI 304, Chapters 5 and 6.

3.8.6.2.17 Consolidation

Consolidation of concrete is in accordance with the recommendations of ACI 309.

3.8.6.2.18 Curing

Curing of concrete is in accordance with the recommendations of ACI 301, as modified herein.

3.8.6.2.19 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.

3.8.6.2.20 Concrete and Concrete Materials-Construction Testing

An independent concrete and concrete materials testing laboratory is 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 ANSI N45.2.6. Procedures and tests for accomplishing such work are reviewed and accepted by Bechtel prior to use.

Production testing for concrete and concrete materials is as shown in Table 3.8-16.

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 remaining concrete is not used in the construction.

Concrete cylinder test results are reviewed for compliance with Section 4.3.3 of ACI 318 and are evaluated in accordance with ACI 214.

3.8.6.2.21 Formwork and Construction Joints

Formwork is designed and constructed so that the final structure conforms within the specified tolerances to the shape, lines, and dimensions required by the design drawings. The design includes consideration of the following:

1. Rate and method of placing concrete

2. Density of concrete

3. Construction loads, including vertical, horizontal, and impact loads.

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 are then covered with grout or a layer of starter mix which is approximately 4 to 6 inches deep.

3.8.6.3 Reinforcing Steel

3.8.6.3.1 Material

Reinforcing steel for concrete structures conforms to ASTM A 615, Grade 60, including Supplementary Requirements S-1 for bar sizes 14 and 18. Certified copies of material test reports indicating chemical composition, physical properties, and dimensional compliance for each heat are furnished by the manufacturer.

Each bundle of reinforcing steel is tagged to ensure 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.

Prior to installation, all reinforcing steel is subjected to a test program meeting the requirements of Regulatory Guide 1.15. Reinforcing steel that does not meet these requirements is not used.

3.8.6.3.2 Fabrication

Hooks and bends are fabricated in accordance with ACI 318, Chapter 7.1. Bending of partially embedded bars is subject to the following conditions:

The minimum distance from concrete surface to the beginning of bend and the minimum inside diameter of bend is:

Bar Size	Minimum Distance from Surface to Beginning of Bend	Minimum 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

Bar sizes No. 3, 4, and 5 may be bent cold once. Heating is required for subsequent straightening or bending. Bar sizes No. 6 through 9 may be bent and straightened, provided that heating is used. These bars may be bent over and straightened once cold when the bend does not exceed 30 degrees.

Heat is applied uniformly over a length of bar equal to 10 bar diameters, centered at the middle of the arc of the completed bend. The maximum bar temperature is maintained between 1100 and 1200°F until bending is completed.

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 cooling 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.

Bar sizes No. 10 and larger are bent only if approved by the responsible field engineer. Heating is required except when bending of bars does not exceed an offset of 1:6 where required to put bars

in proper position. When only cold bend is applied, the distance between the beginning of the bend and the existing concrete may be decreased to a minimum of 0.5 bar diameter.

3.8.6.3.3 Splices

Lapped splices, used for No. 11 and smaller bars, are in accordance with Sections 7.5, 7.6, and 7.7 of ACI 318.

Mechanical (Cadweld) splices are used for all No. 14 and No. 18 splices. These bars are located in the basemat and fuel pool's walls and slab. Cadweld splices are also used locally on other bar sizes in lieu of standard hooks and to minimize rebar congestion at plate anchorages and construction openings. To obtain effective quality control, a qualification, inspection, testing, and acceptance program in accordance with NRC Regulatory Guide 1.10 is used (See Section 1.8.1.10). Welding of splice sleeves to liners, plates, and shapes is in accordance with AWS D1.1.

Welded splices have not been used.

The requirements of Regulatory Guide 1.136, Revision 2, are not implemented in the design, since significant concrete work using cadwelds was completed prior to issuance of this guide. As a result both sister and production splicing have been used for testing of cadweld splices. Where sufficient bar length exists future cadwelding will be in accordance with Regulatory Guide 1.136, Revision 2.

3.8.6.3.4 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 design requirements.

3.8.6.3.5 Spacing of Reinforcement

Spacing of reinforcement is in accordance with Sections 3.3.2 and 7.4 of ACI 318, except as modified herein.

3.8.6.3.6 Surface Conditions of Reinforcement

Surface conditions of reinforcement at the time of concrete placement are in compliance with Section 7.2 of ACI 318.

3.8.6.3.7 Reinforcement Placing Tolerance

The section strength for reinforced concrete is based on the strength design method of ACI 318 and on the reduced effective depth (d) resulting from specified rebar placing tolerances in excess of those specified in ACI 318, Section 7.3.2.1. The effective depth is reduced by an additional 1/2 inch for all members greater than 12 inches but not greater than 24 inches and by an additional 1 inch for all members greater than 24 inches. The cover used in calculating the effective depth is the "nominal cover" as set forth in Table 3.8-17.

The above design considerations allow the reinforcement placing tolerances, as defined in Section 3.8.6.2.2.

3.8.6.4 Structural Steel

3.8.6.4.1 Materials

The various structural steel components conform to the following ASTM specifications:

Item	ASTM Specification
Beams, girders, and plates	A 36 or A 441
Structural tubing	A 500

Item	ASTM Specification
Anchor bolts	A 36 or A 307
High strength bolts	A 325 or A 490

3.8.6.4.2 Fabrication and Erection

The fabrication and erection of structural steel conforms to the AISC specifications.

3.8.6.4.3 Welding and Nondestructive Testing

Welding and nondestructive testing is performed in accordance with either AWS D1.1 or Section IX of the ASME B&PV Code, as shown in Table 3.8-15.

3.8.7 References

3.8-1 K.R. Wichman, A.G. Hopper, and J.L. Mershon, "Local Stresses in Spherical and Cylindrical Shells Due to External Loadings," Welding Research Council Bulletin 107, August 1965, third revised printing, April 1972.

3.8-2 General Electric, "Mark I Containment Program Load Definition Report," Revision 2, November 1981.

3.8-3 General Electric, "Mark I Containment Program Plant Unique Load Definition, Hope Creek Generating Station: Unit 1," Revision 1, January 1982.

3.8-4 Bechtel Power Corporation, "Design for Pipe Break Effects," BN-TOP-2, Revision 2, May 1974.

3.8-5 N.M. Newmark, et al., "Air Force Design Manual Principles and Practices for Design of Hardened Structures," December, 1962

3.8-6 Norris, Hansen, Holley, Biggs, Namyet and Minami, "Structural Design for Dynamic Loads"

3.8-7 Bechtel Power Corporation, "Containment Building Liner Plate Design Report," BC-TOP-1, Revision 1, December, 1972.

TABLE 3.8-1

CODES, STANDARDS, AND SPECIFICATIONS USED IN DESIGN AND CONSTRUCTION OF THE PRIMARY CONTAINMENT

Designation	Title	Edition
American Society of Mechanical Engineers (ASME)		
ASME	ASME B&PV Code	
	a. Section III, Subsection NA, General Requirements	1974 with Winter 1974 Addenda
	b. Section III, Subsection NE, Class MC Components	1974 with Winter 1974 Addenda
	c. Section III, Subsection NF, Component Supports	1974 with Winter 1974 Addenda
	d. Code Case 1567, Testing Lots of Carbon and Low Alloy Steel Covered Electrodes, Section III	Approved by Council March 1973
	e. Code Case 1557, Steel Products Refined by Secondary Remelting. Section III and VIII, Divisions 1 and 2	Approved by Council December 17, 1973
	f. Code Case 1644, Additional Materials for Component Supports and Alternate Design Requirements for Bolted Joints, Section III, Division 1, Subsection NF, Class 1, 2, 3 and MC Construction	Approved by Council March 3, 1976
	g. Code Case 1648, SA537 Plates for Section III, Class 2, 3 and MC Components	Approved by Council August 12, 1974
	h. Code Case N-236, Repair and Replacement of Class MC Vessels, Section XI, Division 1	Approved by Council January 21, 1982
	i. Code Case N-242, Material Certification, Section III, Division 1, Class 1, 2, 3, MC and CS Construction	Approved by Council April 12, 1979
	j. Code Case N-252, Low Energy Capacitive Discharge Welding Method for Temporary or Permanent Attachments to Components and Supports, Section III, Division 1, and Section XI	Approved by Council November 19, 1979
	k. Code Case N-284, Metal Containment Shell Buckling Design Methods, Section III, Division 1, Class MC	Approved by Council August 25, 1980
	l. Code Case N-308, Documentation of Repairs and Replacement of Components in Nuclear Power Plants	Approved by Council September 30, 1981
	m. Code Case N-313, Alternative Rule for Half-Coupling Branch Connections, Section III, Division 1	Approved by Council May 11, 1981

Designation	Title	Edition
	n. Section IX, Welding and Brazing Qualification	1974 with Winter 1974 Addenda
	o. Section XI, Rules for Inservice Inspection of Nuclear Power Plant Components, Division 1	1980 with Winter 1981 Addenda
	p. Code Case N-362-2, Pressure Testing of Containment Items, Section III, Division 1, Classes 1, 2 and MC	Approval date: July 12, 1984
American Concrete Institute (ACI)		
ACI 318	Building Code Requirements for Reinforced Concrete	1971
The Institute of Electrical and Electronic Engineers, Inc (IEEE)		
IEEE 317-72	IEEE Standard for Electric Penetration Assemblies in Containment Structures for Nuclear Power Generating Stations	1972
IEEE 383-74	IEEE Standard for Type Test of Class 1E Electric Cables, Field Splices, and Connections for Nuclear Power Generating Stations	1974
American National Standards Institute (ANSI)		
ANSI A58.1-1972	Building Code Requirements for Minimum Design Loads in Buildings and Other Structures	1972
ANSI N45.4-1972	Leakage Rate Testing of Containment Structures for Nuclear Reactors	1972
ANSI N45.2	Quality Assurance Program Requirements for Nuclear Power Plants	1971
American Society of Testing and Materials (ASTM)		
ASTM A 572	Standard Specification for High Strength Low Alloy Columbium-Vanadium Steels of Structural Quality	1973

TABLE 3.8-2

LOADING COMBINATIONS AND ALLOWABLE STRESSES FOR PRIMARY CONTAINMENT

Combination (1)	ASME Section III Table / Paragraph S_m or S	S_y	S_c	Region	General (2) Membrane P_m	Local (2) Membrane P_l	Local (2) Membrane & Bending $P_l + P$	Primary & (2) Secondary Stresses $P_l + P_b + Q$	Buckling (3)
$D+L+P_t+T_t+E$	-	I-2.0	NE-3133	All	0.85 S_y	1.25 S_y	1.25 S_y	-	1.25 S_c
$D+L+F_l+T_o+Ro$	I-10.0	-	NE-3133	All	1.5 S_m	Greater of 1.5 S_y or 1.8 S_m	Greater of 1.5 S_y or 1.8 S_m	-	1.20 S_c
$D+L+T_o+R_o$ $D+L+T_o+R_o+E+P_o$	I-10.0	-	NE-3133	All	S_m	1.5 S_m	1.5 S_m	3.0 S(4)	S_c
$D+L+T_a+R_a+P_a+E$ $D+L+T_e+R_e+P_e+E$ $D+L+T_s+R_s+P_s+E$	I-10.0	-	NE-3133	All	S_m	1.5 S_m	1.5 S_m	-	S_c
$D+L+T_a+R_a+P_a+E'$ $D+L+T_e+R_e+P_e+E'$	I-10.0	-	NE-3133	Nonintegral & noncontinuous	S	1.5 S	1.5 S	-	S_c
$D+L+T_s+R_s+P_s+E'$	I-10.0	I-2.0	NE-3133	Integral & continuous	Greater of S_y or 1.2 S_m	Greater of 1.5 S_y or 1.8 S_m	Greater of 1.5 S_y or 1.8 S_m	- -	1.2 S_c
$D+L+T_a+R_a+P_a+Y_r+Y_j+Y_m+E'$ $D+L+T_s+R_s+P_s+Y_r+Y_j+Y_m+E'$	I-10.0	I-2.0	NE-3133	Nonintegral & noncontinuous	Greater of S_y or 1.2 S_m	Greater of 1.5 S_y or 1.8 S_m	Greater of 1.5 S_y or 1.8 S_m	-	1.2 S_c
$D+L+T_a+R_a+P_a+Y_r+Y_j+Y_m+E'$ $D+L+T_s+R_s+P_s+Y_r+Y_j+Y_m+E'$	F-1324	-	NE-3133	Integral & continuous	0.85 S	1.28 S	1.28 S	-	1.28 S_c
D+L+F +E +T +R	I-10.0	I.2.0	NE-3133	All	1.5 S_m	Greater of 1.5 S_y or 1.8 S_m	Greater of 1.5 S_y or 1.8 S_m	-	1.2 S_c

(1) Load definitions:

D	Dead loads
L	Live loads
P_t	Test pressure
T_t	Test temperature
T_o	Thermal effects and loads during startup, normal operating, or shutdown conditions, based on the most critical transient or steady state condition
R_o	Pipe reactions during startup, normal operating, or shutdown conditions, based on the most critical transient or steady state condition
P_o	Pressure loads during startup, normal operating, or shutdown conditions
P_e	Design external pressure
T_e	Thermal loads under thermal conditions during event causing external pressure
R_e	Pipe reactions under thermal conditions during event causing external pressure
E	Loads generated by the operating basis earthquake, including sloshing effects, if applicable
E'	Loads generated by the safe shutdown earthquake, including sloshing effects, if applicable
P_a	Pressure equivalent static load generated by the postulated design basis accident, including P_o, pool swell, and subsequent hydrodynamic loads
T_a	Thermal loads under thermal conditions generated by the postulated pipe break accident, including T_o, pool swell, and subsequent hydrodynamic reaction loads
R_a	Pipe reactions under thermal conditions generated by the postulated pipe break accident, including R_o, pool swell, and subsequent hydrodynamic reaction loads
P_s	All pressure loads which are caused by the actuation of safety/relief valve discharge
T_s	All thermal loads which are generated by the actuation of safety/relief valve discharge
R_s	All pipe reaction loads which are generated by the actuation of safety/relief valve discharge
Y_r	Equivalent static load on the structure generated by the reaction on the broken pipe during the design basis accident

Y_j Jet impingement equivalent static load on the structure generated by the broken pipe during the design basis accident

Y_m Missile impact equivalent static load on the structure generated by or during the design basis accident, such as pipe whipping

F Loads generated by the post-LOCA flooding of the primary containment, if any

(2) For definitions of P_m, P_l, P_b and Q, see Figure NE-3222-1 of the ASME B&PV Code.

(3) A factor of two shall exist between the critical buckling stress and the applied stress where a rigorous buckling analysis is performed that considers inelastic behavior.

(4) Fatigue analysis shall be performed for loading combinations listed, except post-LOCA flooding.

TABLE 3.8-3

ASME CODE CLASS MC COMPONENT SUPPORTS

Combination[4]	ASME Section III Table / Paragraph S	ASME Section III Table / Paragraph S_c	Plate and Shell Type Component Supports σ_1 Tension	σ_1 Comp	$\sigma_1+\sigma_2$	Notes
$D+L+P_t+T_t+E$ $D+L+T_o+R_o$ $D+L+T_o+R_o+E+P$ $D+L+T_a+R_a+P_a+E$ $D+L+T_e+R_e+P_e+E$ $D+L+T_s+R_s+P_s+E$	I-7.0 or I-8.0 or I-12.0 or Code Case 1644	NC-3133 or ND-3133 Or NE-3133	S	S_c	1.5 S	(1)
$D+L+T_a+R_a+P_a+E'$ $D+L+T_e+R_e+P_e+E'$ $D+L+T_s+R_s+P_s+E'$	I-7.0 or I-8.0 or I-12.0 or Code Case 1644	NC-3133 or ND-3133 or NE-3133	1.2 S	1.2 S_c	1.8 S	(2)
$D+L+T_a+R_a+P_a+Y_r+Y_j+Y_m+E'$ $D+L+T_s+R_s+P_r+Y_j+Y_m+Y+E'$ $D+L+T_o+R_o+F_1+E$	I-7.0 or I-8.0 or I-12.0 or Code Case 1644	NC-3133 or ND-3133 or NE-3133	2.0 S	1.28 S_c	2.4 S	(3)

(1) Linear type component supports - Stress limits given by Appendix XVII

(2) Linear type component supports - Appendix XVII allowables increased by a factor of 1.33

(3) Linear type component supports - Paragraph F-1370, Appendix F

(4) Load definitions:

- D Dead loads
- L Live loads
- P_t Test pressure
- T_t Test temperature

TABLE 3.8-3 (Cont)

Combination[4]	ASME Section III Table / Paragraph		Plate and Shell Type Component Supports			Notes
			σ_1			
	S	S_c	Tension	Comp	$\sigma_1+\sigma_2$	

T_o Thermal effects and loads during startup, normal operating, or shutdown conditions, based on the most critical transient or steady state condition

R_o Pipe reactions during startup, normal operating, or shutdown conditions, based on the most critical transient or steady state condition

P_o Pressure loads during startup, normal operating, or shutdown conditions

P_e Design external pressure

T_e Thermal loads under thermal conditions during event causing external pressure

R_e Pipe reactions under thermal conditions during event causing external pressure

E Loads generated by the operating basis earthquake, including sloshing effects, if applicable

E' Loads generated by the safe shutdown earthquake, including sloshing effects, if applicable

P_a Pressure equivalent static load generated by the postulated design basis accident, including P_o, pool swell, and subsequent hydrodynamic loads

T_a Thermal loads under thermal conditions generated by the postulated design basis accident, including T_o, pool swell, and subsequent hydrodynamic reaction loads

R_a Pipe reactions under thermal conditions generated by the postulated design basis accident, including R , pool swell, and subsequent hydrodynamic reaction loads

P_s All pressure loads which are caused by the actuation of safety/relief valve discharge

T_s All thermal loads which are generated by the actuation of safety/relief valve discharge

R_s All pipe reaction loads which are generated by the actuation of safety/relief valve discharge

Y_r Equivalent static load on the structure generated by the reaction on the broken pipe during the design basis accident

Y_j Jet impingement equivalent static load on the structure generated by the broken pipe during the design basis accident

Y_m Missile impact equivalent static load on the structure generated by or during the design basis accident, such as pipe whipping

F_1 Loads generated by the post-LOCA flooding of the primary containment, if any

σ_1, σ_2 Calculated stress

TABLE 3.8-4

LOAD COMBINATIONS FOR REINFORCED CONCRETE SEISMIC CATEGORY I STRUCTURES IN THE PRIMARY CONTAINMENT

a. For normal and severe environmental conditions, the following load combinations and allowables[(1)] are considered:

$U = 1.4D + 1.7L$

$U = 1.4D + 1.7L_o + 1.9E_o$

When thermal stresses due to T_o and R_o are present, the following combinations are considered:

$U = (0.75)\ (1.4D + 1.7L + 1.7T_o + 1.7R_o)$

$U = (0.75)\ (1.4D + 1.7L_o + 1.9E_o + 1.7T_o + 1.7R_o)$

b. For extreme environmental, abnormal, abnormal/severe environmental, and abnormal/extreme environmental conditions, the following load combinations and allowables[(1)] are considered:

$U = D + L_o + T_o + R_o + E'$

$U = D + L + T_a + R_a + 1.5P$

$U = 1.4D + 1.7L + F$

$U = D + L_o + T_a + R_a + 1.25P_a + R_r + 1.25E_o$

$U = D + L_o + E_o + F_1$

$U = D + L_o + T_a + R_a + P_a + R_r + E'$

---

(1) Symbols used in load combinations:

D	Dead loads
E_0	Operating basis earthquake loads
E'	Safe shutdown earthquake loads
F	Post-accident containment flooding loads
L	Live loads

L_o Operating live loads (combined with seismic loads) - The live load expected to be present when the plant is operating, established in accordance with the layout and mechanical requirements. These loads may vary, depending upon the function of a specific area

P_a Abnormal pressure loads

T_a Abnormal temperature loads

T_o Operating thermal loads

R_a Abnormal pipe reaction loads

R_o Operating pipe reaction loads

R_r Abnormal local effects, including reactions from pipe rupture, jet impingement, pipe whip, or any nonmechanistic - break inside the bioshield

U The section strength, based on the strength design method of ACI-318

(2) Both cases of L (or L_o), having its full value or being completely absent, are checked.

TABLE 3.8-5

LOAD COMBINATIONS FOR DRYWELL STRUCTURAL STEEL
SEISMIC CATEGORY I STRUCTURES

1. For normal and severe environmental conditions, the following load combinations and allowables[1] are considered:

$S = D + L$

$S = D + L_o + E_o$

When thermal stresses due to T_o and R_o are present, the following combinations are considered:

$1.5S = D + L + T_o + R_o$

$1.5S = D + L_o + T_o + R_o + E_o$

2. For extreme environmental, abnormal, abnormal/severe environmental, and abnormal/extreme environmental conditions, the following load combinations and allowables[1] are considered:

$1.6S = D + L_o + T_o + R_o + E'$

$1.6S = D + L + T_a + R_a + P_a$

$S = D + L + F_L$

$1.6S = D + L_o + T_a + R_a + P_a + R_r + E_o$

$1.6S = D + L_o + E_o + F_L$

$1.7S = D + L_o + T_a + R_a + P_a + R_r + E'$

---

(1) Symbols used in load combinations:

D Dead loads

E_o Operating basis earthquake loads

E' Safe shutdown earthquake loads

F_L Post-accident containment flooding loads

L Live loads

L_o Operating live loads (combined with seismic loads) - The live load expected to be present when the plant is operating, established in accordance with the layout and mechanical requirements. These loads may vary, depending upon the function of a specific area.

P Abnormal pressure loads

T Abnormal temperature loads

T Operating thermal loads

R_a Abnormal pipe reaction loads

R_o Operating pipe reaction loads

R Abnormal local effects, including reactions from pipe rupture, jet impingement, and pipe whip. In load combinations involving R , the corresponding structural acceptance criteria (1.6S or 1.7S) is first satisfied without considering the effect of R . When considering the effects of R , local section strength capacities may be exceeded provided that there will be no loss of function of any safety related system. The local effects are evaluated in accordance with Bechtel Topical Report BN-TOP-2 (Reference 3.8-4).

S The section strength, determined as the allowable set forth in the AISC Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings, Part I.

(2) Both cases of L (or L_o), having its full value or being completely absent, are checked.

TABLE 3.8-6

ALLOWABLE STRESSES FOR SUPPRESSION CHAMBER INTERNAL STRUCTURES OUTSIDE THE SCOPE OF THE ASME CODE[1]

Combination[2]	Allowables
$D + L + P_t + T_t$	Stress limits in accordance with the AISC Code
$D + L + F_1$	
$D + L + T_o + R_o$	
$D + L + T_o + R_o + E$	
$D + L + T_a + R_a + P_a + E$	
$D + L + T_e + R_e + P_e + E$	
$D + L + T_s + R_s + P_s + E$	
$D + L + T_a + R_a + P_a + E'$	1.5 times the stress limits in accordance with the AISC Code.
$D + L + T_e + R_e + Pe + E'$	
$D + L + F_1 + E$	
$D + L + T_s + R_s + P_s + E'$	

(1) The allowable stresses for the structures within the scope of the ASME B&PV Code are given in Table 3.8-2.

(2) For the definition of symbols used in the load combinations, see Table 3.8-2.

TABLE 3.8-7

LIST OF APPLICABLE CODES, STANDARDS, RECOMMENDATIONS, AND SPECIFICATIONS

Designation	Title	Edition[1]
ACI	American Concrete Institute:	
318	Building Code Requirements for Reinforced Concrete	1971, with the 1974 supplement, including 1973 and 1974 revisions
531	Building Code Requirements for Concrete Masonry Structures	1979
AISI	American Iron and Steel Institute:	
	Specification for the Design of Light Gauge Cold-Formed Steel Structural Members	1968
AISC	American Institute of Steel Construction: Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings	February 12, 1969 with: Supplement 1 of Nov 1, 1970 Supplement 2 of Dec 8, 1971 and Supplement 3 of June 12, 1974
ANSI	American National Standards Institute:	
A58.1	Building Code Requirements for Minimum Design Loads in Buildings and Other Structures	1972
AREA	American Railway Engineering Association:	
	Manual for Railway Engineering (for possible future railway)	March 1972
ASME	American Society of Mechanical Engineers:	
	Boiler and Pressure Vessel Code	
Section II	Materials Specifications	1974 with Winter 1974 Addenda
Section III	Nuclear Power Plant Components	1974 with Winter 1974 Addenda
Section V	Nondestructive Examination	1974 with Winter 1974 Addenda
Section IX	Welding and Brazing Qualifications	1974 with Winter 1974 Addenda
Section XI	Inservice Inspection	1974 with Winter 1974 Addenda
AWS	American Welding Society:	
D 1.1	Structural Welding Code (See NCIG-01)	1975
NML	Nuclear Mutual Limited: Property Loss Prevention Standards	

TABLE 3.8-7 (Cont)

Designation	Title	Edition[1]
NCIG-01	Nuclear Construction Issue Group Visual Weld Acceptance Criteria	5/7/1985, Rev 2
PCA	Portland Cement Association:	
	Design of Multistory Reinforced Concrete Buildings for Earthquake Motions J.A. Blume, N.M. Newmark, and L.H. Corning	1961
UBC	International Conference of Building Officials:	
	Uniform Building Code	1973

(1) Principle editions used are listed; later editions may be applied in specific cases.

TABLE 3.8-8

LOAD COMBINATIONS FOR REINFORCED CONCRETE SEISMIC CATEGORY I STRUCTURES OTHER THAN THE PRIMARY CONTAINMENT AND ITS INTERNALS

1. For normal and severe environmental conditions, the following load combinations and allowables are considered:

   $U = 1.4D + 1.7L$

   $U = 1.4D + 1.7L_o + 1.9E_o$

   $U = 1.4D + 1.7L + 1.7W$

   If the thermal stresses due to T_o and R_o are present, the following combinations are considered:

   $U = (0.75)\ (1.4D + 1.7L + 1.7T_o + 1.7R_o)$

   $U = (0.75)\ (1.4D + 1.7L_o + 1.9E_o + 1.7T_o + 1.7R_o)$

   $U = (0.75)\ (1.4D + 1.7L + 1.7W + 1.7T_o + 1.7R_o)$

   In addition, the following combinations are considered:

   $U = 1.2D + 1.9E_o$

   $U = 1.2D + 1.7W$

2. For extreme environmental, abnormal, abnormal/severe environmental, and abnormal/extreme environmental conditions, the following load combinations and allowables are considered:

   $U = D + L_o + T_o + R_o + E_s$

   $U = D + L + T_o + R_o + W_t$

   $U = D + L + T_o + R_o + W_e$

   $U = D + L + T_a + R_a + 1.5P_a$

   $U = D + L_o + T_a + R_a + 1.25P_a + R_r + 1.25E_o$

   $U = D + L_o + T_a + R_a + P_a + R_r + E_s$

TABLE 3.8-9

LOAD COMBINATIONS FOR REINFORCED CONCRETE
NON-SEISMIC CATEGORY I STRUCTURES [1]

1. $U = 1.4D + 1.7\ L$

2. $U = 0.75\ (1.4\ D + 1.7\ L_o + 1.87\ E_u) + 1.0\ R_o^{(2)}$

3. $U = 0.75\ (1.4\ D + 1.7\ L + 1.7\ W) + 1.0\ R_o^{(2)}$

4. $U = 0.9\ D + 1.3\ W + 1.0\ R_o^{(2)}$

5. $U = 0.9\ D + 1.43\ E_u + 1.0\ R_o^{(2)}$

In addition to the requirements above, the structures are checked to verify that they do not collapse on, or interact with, adjacent Seismic Category I structures for the following load combinations under abnormal and/or extreme environmental conditions:

6. $U = D + L_o + E_s + T_o + R_o^{(2)}$

7. $U = D + L + W_t + T_o + R_o^{(2)}$

8. $U = D + L + W_e + T_o + R_o^{(2)}$

9. $U = D + L + R_a + 1.5\ P_a + T_a$

? 10. $U = D + L_0 + R_a + R_r + P_a + E_s + T_a$

---

(1) The Turbine Building and the administration facility.

(2) R_o, which produces the most critical combination loading, is used.

TABLE 3.8-10

LOAD COMBINATIONS FOR STRUCTURAL STEEL SEISMIC CATEGORY I STRUCTURES OTHER THAN THE PRIMARY CONTAINMENT AND ITS INTERNALS[1]

1. For normal and severe environmental conditions, the following load combinations and allowables are considered:

   $S = D + L$

   $S = D + L_o + E_o$

   $S = D + L + W$

   If thermal stresses due to T_o and R_o are present, the following combinations are also considered:

   $1.5S = D + L + T_o + R_o$

   $1.5S = D + L_o + T_o + R_o + E_o$

   $1.5S = D + L + T_o + R_o + W$

2. For extreme environmental, abnormal, abnormal/severe environmental, and abnormal/extreme environmental, the following load combinations and allowables apply:

   $1.6S = D + L_o + T_o + R_o + E_s$

   $1.6S = D + L + T_o + R_o + W_t$

   $1.6S = D + L + T_o + R_o + W_e$

   $1.6S = D + L + T_a + R_a + P_a$

   $1.6S = D + L_o + T_a + R_a + P_a + R_r + E_o$

   $1.7S = D + L_o + T_a + R_a + P_a + R_r + E_s$

---

(1) Also applicable to composite (structural steel and concrete) construction.

TABLE 3.8-11

LOAD COMBINATIONS FOR STRUCTURAL STEEL AND CONCRETE MASONRY
NON-SEISMIC CATEGORY I STRUCTURES (1)

$$S = D + L$$
$$1.33S = D + L_o + E_u + R_o^{(2)}$$
$$1.33S = D + L + W + R_o^{(2)}$$
$$S = D + L_o + E_u + R_o^{(2)(3)}$$

In addition to the requirements above, the structures are checked to verify that they do not collapse on, or interact with, adjacent Seismic Category I structures for the following load combinations for structural steel structures under abnormal and/or extreme environmental conditions:

$$1.5S = D + L_o + T_o + R_o^{(2)}$$
$$1.6S = D + L_o + E_s + T_o + R_o^{(2)}$$
$$1.6S = D + L + W_t + T_o + R_o^{(2)}$$
$$1.6S = D + L + W_e + T_o + R_o^{(2)}$$
$$1.6S = D + L + R_a + P_a + T_a$$
$$1.7S = D + L_0 + R_a + R_r + P_a + E_s + T_a$$

---

(1) The Turbine Building and the administration facility.

(2) R_o, which produces the most critical combination of loading, is used.

(3) This load case is used only for structural elements carrying mainly earthquake forces, e.g., struts and bracings.

TABLE 3.8-12

SYMBOLS USED IN LOAD COMBINATIONS

Symbol		Description
D	———	Dead loads
E_o	———	Operating basis earthquake loads
E_s	———	Safe shutdown earthquake loads
E_u	———	UBC seismic zone 1 loads
$H^{(1)}$	———	Lateral earth pressure
L	———	Live loads
L_o	———	Operating live loads (used with seismic loads)
P_a	———	Abnormal pressure loads
R_a	———	Abnormal pipe reaction loads
R_0	———	Operating pipe reaction loads
R_r	———	Abnormal local effects of pipe rupture
R_{rr}	———	See Section 3.8.4.3.1.4.4(1)
R_{rj}	———	See Section 3.8.4.3.1.4.4(2)
R_{rm}	———	See Section 8.3.4.3.1.4.4(3)

Symbol		Description
S	———	Section strength for structural steel determined as the allowable set forth in the AISC specifications, as listed in Table 3.8-7.
T_a	———	Abnormal temperature loads
T_o	———	Operating thermal loads
U	———	Section strength for reinforced concrete based on the strength design method of ACI 318 as listed in Table 3.8-7, and a reduced effective depth (d), resulting from specified rebar placing tolerances in excess of those specified in ACI 318.
W	———	Severe wind loads
W_e	———	Extreme wind and flood loads
W_t	———	Tornado loads
W_{tq}	———	See Section 3.8.4.3.1.3.2
W_{tp}	———	See Section 3.8.4.3.1.3.2
W_{tm}	———	See Section 3.8.4.3.1.3.2

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(1) Combinations including this load are described in Section 3.8.5

TABLE 3.8-13

SUMMARY OF STRUCTURAL FOUNDATIONS

Structure	Elev, Bottom of Base Mat[1] PSE&G Datum	Elev, Bottom of Base Mat[1] CGS Datum	Elev, Top of Vincentown Formation[2] PSE&G Datum	Type of Foundation
Reactor Building, Mat 3	+40.0	-49.0	+36.0	Base mat bearing on engineered backfill and/or concrete fill on Vincentown Formation
Auxiliary Building, Mat 5, and parts of Mats 3 and 4	+40.0	-49.0	+36.0	Base mat bearing on engineered backfill and/or concrete fill on Vincentown Formation
Turbine Building, Mat 1	+40.0	-49.0	+36.0	Base mat bearing on engineered backfill and/or concrete fill on Vincentown Formation
SSWS intake structure, Mat 6	+65.5	-22.0	+25.0	Direct bearing of slab on the Vincentown Formation by placing a tremie concrete plug between the top of the Vincentown Formation and the bottom of the base slab.

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(1) Grade level at +101.5 PSE&G datum or +12.5 CGS datum

(2) Elevations shown are average.

TABLE 3.8-14

GROUNDWATER TABLE RESTRICTIONS FOR POWER BLOCK COMPLEX

	Area (See Key Plan - Figure 3.8-37)[1]			
Maximum Allowable Water Table Elevation	Turbine Building	Reactor Building and Southern Section of Radwaste/Service Area of Auxiliary Building	Plant Cancelled Area and Northern Section of Radwaste/Service Area	Auxiliary Building Control/Diesel Generator Area and Central Section of the Radwaste/Service Area
	Mat 1 (Same for Administration Facility - Mat 2)	Mat 3	Mat 4	Mat 5
Natural water table El 96 ft (dewatering discontinued)	all concrete walls and floor slabs to El 137; Turbine pedestal concrete complete	All floors to El 132; floors and cylinder walls to El 145; drywell shield to El 98; exterior walls to El 132; selected interior walls to El 102	Selected floors to El 102; Cylinder walls to El 132; Drywell shield to El 98; Exterior walls to El 132; selected interior walls to El 102	All to El 155
77 ft (dewatering)	All walls to El 77; floor slab to El 77; turbine pedestal complete except for center portion of top deck	Cylinder wll to El 77; drywell shield to El 71 ft 8 in.; exterior walls to El 102; interior walls to El 77	Cylinder wall to El 77; drywell shield to El 71 ft 8 in.; exterior walls to El 102; interior walls to El 77	All to El 102
52 ft (dewatering)	Entire base slab	Entire base slab	Entire base slab	Entire base slab

(1)There is no dewatering for Mat 6, SSWS intake structure, until the tremie concrete plug is placed and cured.

TABLE 3.8-15

LIST OF APPLICABLE CODES, STANDARDS, RECOMMENDATIONS, AND SPECIFICATIONS

Designation	Title	Edition
American Concrete Institute		
ACI 211.1	Recommended Practice for Selecting Proportions for Normal and Heavyweight Concrete	1974
ACI 214	Recommended Practice for Evaluation of Compression Test Results of Field Concrete	1965
ACI 301	Specifications for Structural Concrete for Buildings	1972, 1973
ACI 304	Recommended Practice for Measuring, Mixing, Transporting, and Placing Concrete	1973
ACI 305	Recommended Practice for Hot Weather Concreting	1972
ACI 306	Recommended Practice for Cold Weather Concreting	1966
ACI 309	Recommended Practice for Consolidation of Concrete	1974
ACI 315	Manual of Standard Practice for Detailing Reinforcing Concrete Structures	1974
ACI 318	Building Code Requirements for Reinforced Concrete	1971
American Welding Society		
AWS D1.1	Structural Welding Code (See NCIG-01)	1975
AWS D12.1	Recommended Practice for Welding Reinforcing Steel and Connections in Reinforced Concrete Construction	1975
U.S. Nuclear Regulatory Commission		
RG 1.10	Mechanical (Cadweld) Splices in Reinforcing Bars of Category I Concrete Structures	Revision 1 Jan 1973
RG 1.15	Testing of Reinforcing Bars for Category I Concrete Structures	Revision 1 Dec 1972
RG 1.18	Structural Acceptance Test for Concrete Primary Reactor Containments	Revision 1 Dec 1972

TABLE 3.8-15 (Cont)

Designation	Title	Edition
RG 1.19	Nondestructive Examination of Primary Containment Liner Welds	Revision 1 Aug 1972
RG 1.54	Quality Assurance Requirements for Protective Coatings Applied to Water Cooled Power Plants	June 1973
RG 1.55	Concrete Placement in Category I Structures	June 1973
RG 1.57	Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components	June 1973
RG 1.58	Qualification of Nuclear Power Plant Inspection, Examination, and Testing Personnel	Aug 1973
RG 1.69	Concrete Radiation Shields for Nuclear Power Plants	Dec 1973
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	Revision 1 Apr 1976
American Society for Testing and Materials		
ASTM A 36	Structural Steel	1970, 1974, 1975, 1977, 1981
ASTM A 307	Carbon Steel Externally and Internally Threaded Standard Fasteners	1968, 1974, 1976, 1978
ASTM A 325	High Strength Bolts for Structural Steel Joints, Including Suitable Nuts and Plain Hardened Washers	1971, 1974, 1976, 1978
ASTM A 441	High Strength, Low Alloy Structural Manganese Vanadium Steel	1970
ASTM A 490	Quenched and Tempered Alloy Steel Bolts for Structural Steel Joints	1971, 1974, 1976
ASTM A 500	Cold Formed Welded and Seamless Carbon Steel Structural Tubing in Rounds and Shapes	1973, 1977
ASTM A 615	Deformed and Plain Billet Steel Bars for Concrete Reinforcement	1975
ASTM C 31	Making and Curing Concrete Test Specimens in the Field	1969, 1975
ASTM C 33	Concrete Aggregates	1977

Designation	Title	Edition
ASTM C 39	Compressive Strength of Cylindrical Concrete Specimens	1972
ASTM C 40	Organic Impurities in Sands for Concrete	1973
ASTM C 78	Flexural Strength of Concrete	1975
ASTM C 87	Effect of Organic Impurities in Fine Aggregate on Strength of Mortar	1969
ASTM C 88	Soundness of Aggregates by Use of Sodium Sulfate or Magnesium Sulfate	1973
ASTM C 94	Ready Mixed Concrete	1973, 1974
ASTM C 109	Compressive Strength of Hydraulic Cement Mortars	1975
ASTM C 117	Materials Finer than No. 200 Sieve in Mineral Aggregates by Washing	1969
ASTM C 127	Specific Gravity and Absorption of Coarse Aggregate	1973
ASTM C 128	Specific Gravity and Absorption of Fine Aggregate	1968
ASTM C 131	Resistance to Abrasion of Small Size Coarse Aggregate by Use of the Los Angeles Machine	1969
ASTM C 136	Sieve or Screen Analysis of Fine and Coarse Aggregates	1971
ASTM C 138	Unit Weight, Yield, and Air Content of Concrete	1973, 1975
ASTM C 142	Clay Lumps and Friable Particles in Aggregates	1971
ASTM C 143	Slump of Portland Cement Concrete	1974
ASTM C 150	Portland Cement	1973, 1974
ASTM C 156	Water Retention by Concrete Curing Materials	1971
ASTM C 171	Sheet Materials for Curing Concrete	1969
ASTM C 172	Sampling Fresh Concrete	1971
ASTM C 186	Heat of Hydration of Hydraulic Cement	1973
ASTM C 192	Making and Curing Concrete Test Specimens in the Laboratory	1969

TABLE 3.8-15 (Cont)

Designation	Title	Edition
ASTM C 231	Air Content of Freshly Mixed Concrete by the Pressure Method	1975
ASTM C 260	Air Entraining Admixtures for Concrete	1973
ASTM C 266	Time of Setting of Hydraulic Cement by Gillmore Needles	1971
ASTM C 289	Potential Reactivity of Aggregates	1971
ASTM C 295	Petrographic Examination of Aggregates for Concrete	1973
ASTM C 311	Sampling and Testing Fly Ash or Natural Pozzolans for Use as a Mineral Admixture in Portland Cement Concrete	1977
ASTM C 469	Static Modulus of Elasticity and Poisson's Ratio for Concrete in Compression	1965
ASTM C 494	Chemical Admixtures for Concrete .	1971
ASTM C 618	Fly Ash and Raw or Calcined Natural Pozzolans for Use as a Mineral Admixture in Portland Cement Concrete	1978
ASTM C 637	Aggregates for Radiation Shielding Concrete	1973
ASTM C 845	Expansive Hydraulic Cement	1980
ASTM D 75	Sampling Aggregates	1971
ASTM D 422	Standard Method for Particle Size Analysis of Soils	1972
ASTM D 1140	Standard Method of Test in Soils Finer Than the No. 200 Sieve	1971
ASTM D 1556	Standard Method of Test for Density of Soil in Place by the Sand Cone Method	1974
ASTM D 1557	Standard Methods of Test for Moisture Density Relations of Soils Using 10-lb. Rammer and 18-in. Drop (Method D)	1970
ASTM D 2922	Standard Test for Density of Soil and Soil Aggregate in Place by Nuclear Methods (Shallow Depth)	1971

TABLE 3.8-15 (Cont)

Designation	Title	Edition
American National Standards Institute		
ANSI N45.2.2	Packaging, Shipping, Receiving, Storing, and Handling of Items for Nuclear Power Plants.	1972
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	1974
ANSI N45.2.6	Qualifications of Inspection, Examination, and Testing Personnel for the Construction Phase of Nuclear Power Plants	1973
ANSI B31.1	Power Piping	1973
American Institute of Steel Construction		
AISC	Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings and Supplement Nos. 1, 2 and 3	1969
AISC	Code of Standard Practice for Steel Buildings and Bridges	1972
AISC	Specification for Structural Joints Using ASTM A 325 or A 490 Bolts	1966, 1972, and 1976
American Society of Mechanical Engineers		
ASME	ASME Boiler and Pressure Vessel Code, Sections II, III, V, VIII, and IX	1971 with Addenda through Summer 1972
International Conference of Building Officials		
UBC	Uniform Building Code	1973, 1976
Building Officials and Code Administrators		
BOCA	Basic Plumbing Code	1975

TABLE 3.8-15 (Cont)

Designation	Title	Edition
Nuclear Construction Issue Group		
NCIG-01	Nuclear Construction Issue Group Visual Weld Acceptance Criteria	5/7/1985 (Rev. 2)

TABLE 3.8-16

MINIMUM TESTING FREQUENCIES FOR CONCRETE MATERIALS AND CONCRETE

Material	Requirement	Test	Frequency
Cement	Standard physical and chemical properties	ASTM C 150	Each 6000 bbl of cement delivered or each silo of cement certified at the mill
Pozzolan	Chemical and physical properties per ASTM C 618	ASTM C 311	Each shipment of Pozzolan of 400 tons or less
Aggregate	Organic impurities in sands	ASTM C 40	Once daily for each day of production
	Soundness of aggregates	ASTM C 88	Once every 6 months
	L.A. abrasion	ASTM C 131	Once every 6 months
	Gradation	ASTM C 136	
	Coarse aggregate		Once daily for each 500 cubic yards of production
	Fine aggregate		Once daily for each 500 cubic yards of production
	Moisture		
	Coarse aggregate	-	Once daily for each 500 cubic yards of production
	Fine aggregate	-	Once daily for each 500 cubic yards of production
	Flat and elongated particles	-	Twice per month
Water and Ice	Quality of water to be used in concrete (to meet the requirements herein)	-	Once each 3 months
Admixtures	Air entraining agent	ASTM C 260	Uniformity testing upon occasion at jobsite or supplier facility
	Water reducing and retarding agent(s)	ASTM C 494	Uniformity testing upon occasion at jobsite or supplier facility
Concrete	Mixer uniformity	NRMCA	Initially and as required to maintain NRMCA certification
	Slump, air content, and temperature	ASTM C 143 ASTM C 231	First batch and every 50 cubic yards per class of concrete produced each day, except for grout and starter mix every 50 cubic yards produced
	Unit weight	ASTM C 138	Daily per class of concrete produced

TABLE 3.8-16 (Cont)

Material	Requirement	Test	Frequency
	Compressive strength	ASTM C 31/C 39	One set of two cylinders for testing at 7,28, and 90 days for design strength for each 100 cubic yards of production

TABLE 3.8-17

CONCRETE COVER OF REINFORCING STEEL[1]

Type of Placement	Member	Minimum Clear Cover (inches)	Nominal Cover (inches)	Reduction in Nominal Cover (inches)	Increase in Nominal Cover (inches)
Formed exposed to earth, weather, or water	Members more than 24 in. thick				
	No. 14 and 18 bars	2	3	1	1-1/2
	All other bars	1-1/2	2-1/2	1	1-1/2
	Members 12 to 24 in. thick	1-1/2	2-1/2	1	1
Formed not exposed to earth, weather, or water	Members more than 24 in. thick				
	No. 14 and 18 bars	1-1/2	2-1/2	1	1-1/2
	All other bars	1	2	1	1-1/2
	Members 12 to 24 in. thick	1	2	1	1
	Members less than 12 in. thick	5/8	1	3/8	3/8

(1) Nominal cover for reinforcing steel is the sum of minimum clear cover plus the allowable reduction in nominal cover.

TABLE 3.8-18

ALLOWABLE DUCTILITY RATIOS (μ) FOR REINFORCED CONCRETE
SUBJECTED TO IMPACTIVE AND IMPULSIVE LOADINGS

Member Type and Load Condition	HCGS - Structural Design Criteria		ACI-349 as Modified by Reg. Guide 1.142
	IMPACTIVE LOADINGS	IMPULSIVE LOADINGS	
Flexure:			
Beams	$\frac{0.10}{P-P} \le 10$	3.0	$\frac{0.05}{P-P'} \le 10$
Slabs	$\frac{0.10}{P-P} \le 30$	3.0	$\frac{0.05}{P-P'} \le 10$
Beams and slabs shall also satisfy the following requirements:	$\frac{f'_c}{60f_y} \le \frac{A_s}{bd}$	Ditto	
	$\frac{A_s - A'_s}{bd} \le .25 \frac{f'_c}{f_y}$	Ditto	
Axial compression: Wall and columns	1.3	1.0	1.3
Shear, concrete beams and slabs in region controlled by shear:			
Shear carried by concrete only	1.0	1.0	1.0
Shear carried by concrete and stirrups	1.3	1.0	1.3
Shear carried completely by stirrups	3.0	1.5	3.0

TABLE 3.8-18 (Cont)

Notation:

f'_c = Concrete compressive strength (psi)

f_y = Reinforcing yield strength (psi)

A_s = Area of tensile reinforcement (in^2)

A'_s = Area of compressive reinforcement (in^2)

b = Effective width of the member (in)

d = Distance from the extreme compression (in) fiber to the centroid of the tension reinforcement

TABLE 3.8-19

ALLOWABLE DUCTILITY RATIOS FOR STRUCTURAL STEEL
SUBJECTED TO IMPACTIVE AND IMPULSIVE LOADING

	HCGS STRUCTURAL DESIGN CRITERIA		NUREG 0800
Member Type And Load Combination	Impactive Loading	Impulsive Loading	Appendix A
BEAMS IN FLEXURE			10 (Tension Due To Flexure)
1. Tornado missiles	10		
2. Other loads	20[1] (Proportioned to preclude buckling)	3	
COLUMNS	1.3 (Proportioned to preclude buckling)	1	$\frac{1}{r} \leq 20$, $\mu \leq 1.3$
			$\frac{1}{r} > 20$, $\mu \leq 1.0$
AXIAL TENSION	$0.5 \frac{\Sigma u}{y}$	3	$0.5 \frac{\Sigma u}{y}$

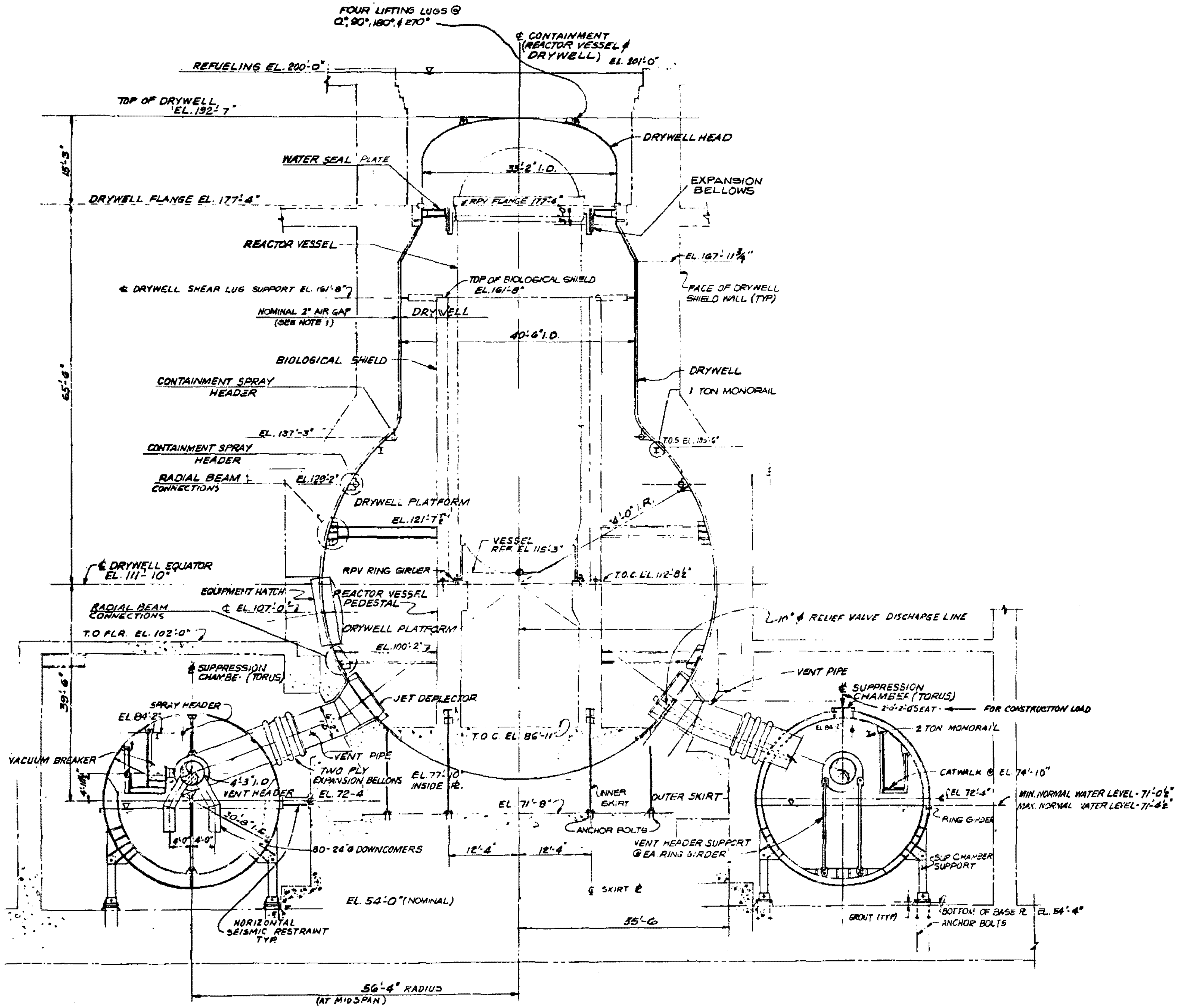
(1) Actual ductility ratios are less than or equal to 10.

TABLE 3.8-20

TYPE A TRIPLE FLUED HEAD CONTAINMENT PENETRATIONS

Penetration Assembly Number	Nom. Elev.	Service	Process Pipe Size	Scope of Supply
P-1A	107'-0"	Main Steam	26"	NSSS
P-1B	107'-0"	Main Steam	26"	NSSS
P-1C	107'-0"	Main Steam	26"	NSSS
P-1D	107'-0"	Main Steam	26"	NSSS
P-2A	113'-2"	Feed Water	24	Non-NSSS
P-2B	113'-2"	Feed Water	24"	Non-NSSS
P-3	106'-0"	RHR Shutdown Cooling From RPV	20"	Non-NSSS
P-4A	106'-0"	RHR Shutdown Cooling Return	12"	Non-NSSS
P-4B	106'-0"	RHR Shutdown Cooling Return	12"	Non-NSSS
P-5A	108'-9"	Core Spray to Reactor	12"	Non-NSSS
P-5B	108'-9"	Core Spray to Reactor	12"	Non-NSSS
P-6A	106'-0"	LPCI	12"	Non-NSSS
P-6B	106'-0"	LPCI	12"	Non-NSSS
P-6C	106'-0"	LPCI	12"	Non-NSSS
P-6D	106'-0"	LPCI	12"	Non-NSSS
P-7	106'-0"	HPCI Turbine Steam Supply	10"	Non-NSSS
P-8A	114'-0"	Chilled Water From Drywell Coolers	8"	Non-NSSS
P-8B	114'-0"	Chilled Water to Drywell Coolers	8"	Non-NSSS
P-9	150'-6"	RWCU Supply	6"	Non-NSSS
P-10	148'-0"	SPARED*	6"	Non-NSSS
P-11	106'-0"	RCIC Turbine Steam Supply	4"	Non-NSSS
P-12	103'-9"	Main Steam Drains3"	Non-NSSS	
P-38A	110'-0"	Chilled Water to Drywell Coolers	8"	Non-NSSS
P-38B	114'-0"	Chilled Water from Drywell Coolers	8"	Non-NSSS

- Process piping capped inside and outside containment. Flued head remains in place.



1. NOTES:

A CLEAR AIR GAP IS PROVIDED BETWEEN THE DRYWELL SHELL & SHIELD WALL FROM EL. 86'-11" TO EL. 174'-0". BELOW EL. 100'-0" A FEW VERY LOCALIZED AREAS EXIST WITH CLEAR AREA GAP REDUCED TO AS NARROW AS ½".

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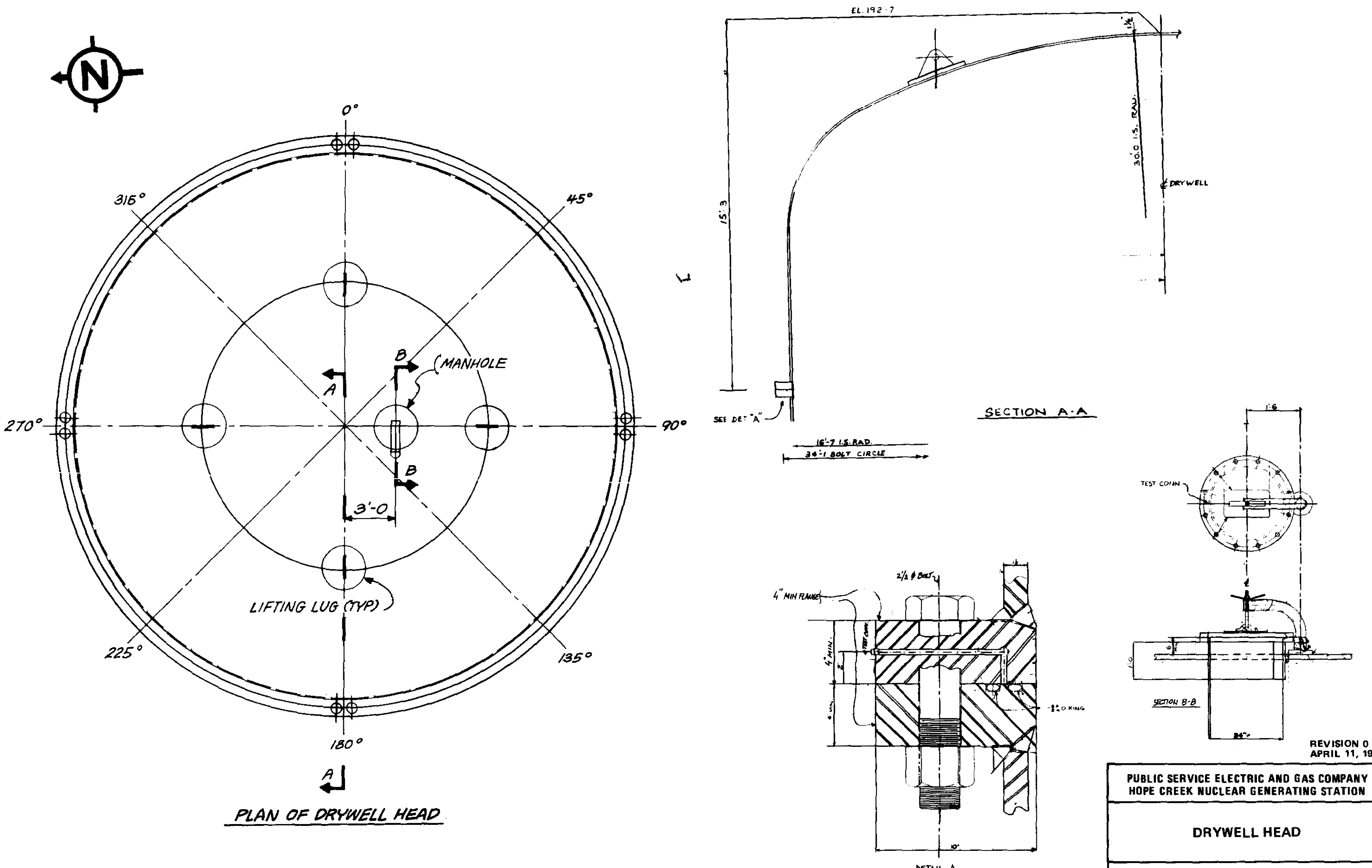
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

PRIMARY CONTAINMENT ELEVATION

UPDATED FSAR

FIGURE 3.8-1



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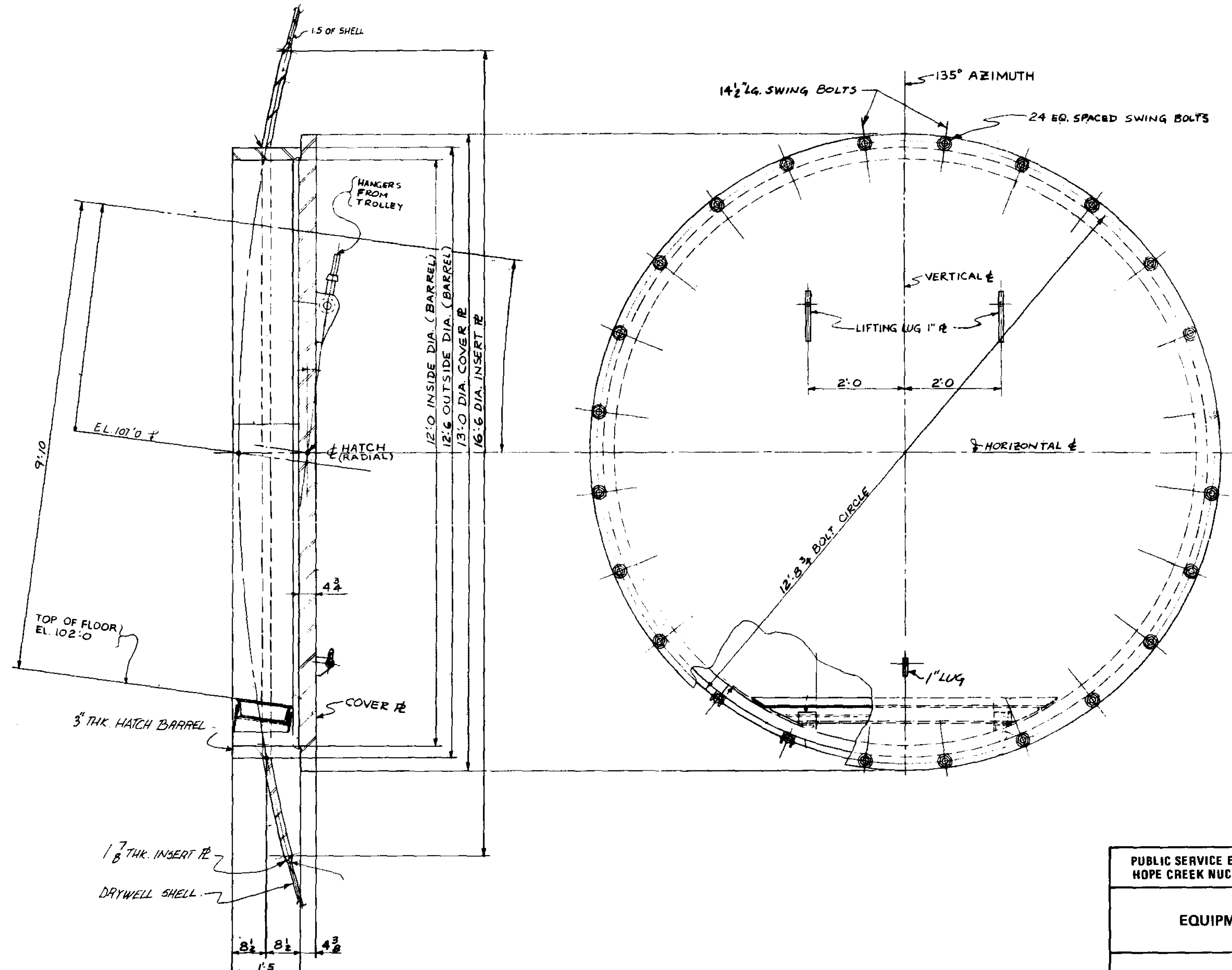
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

DRYWELL HEAD

UPDATED FSAR

FIGURE 3.8-2



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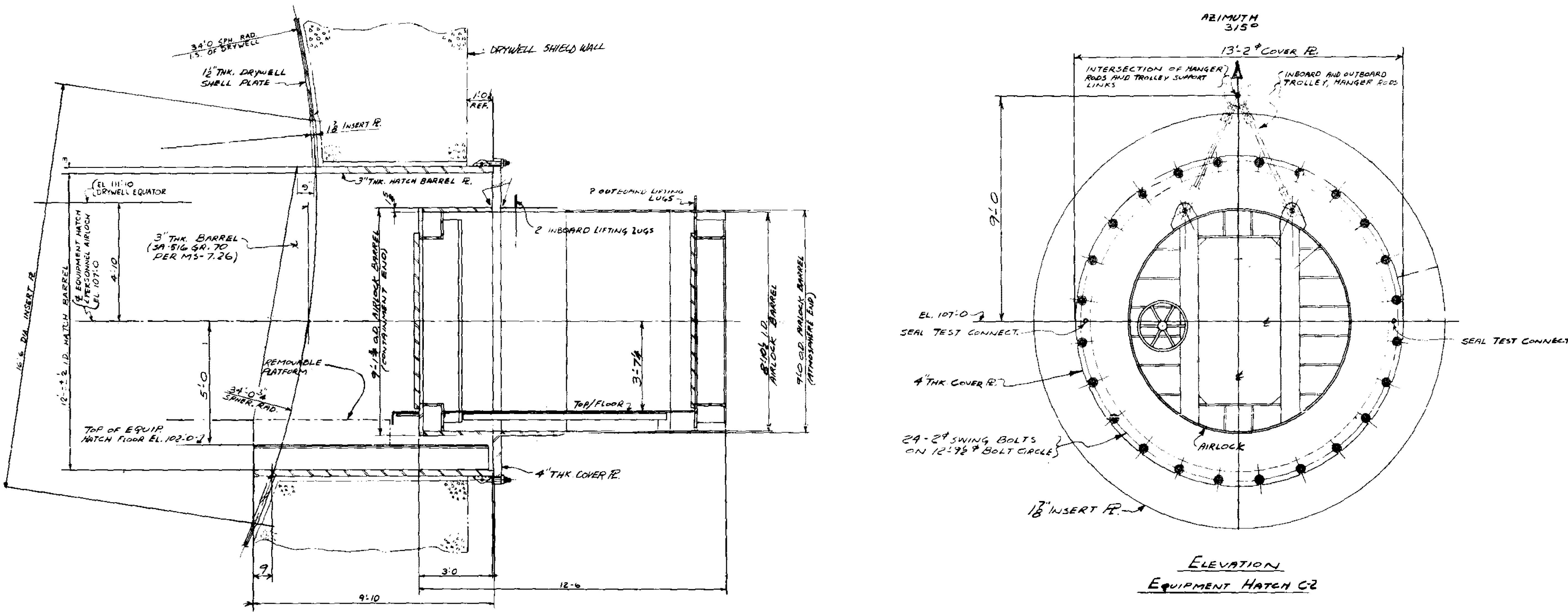
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

EQUIPMENT HATCH C-1

UPDATED FSAR

FIGURE 3.8-3



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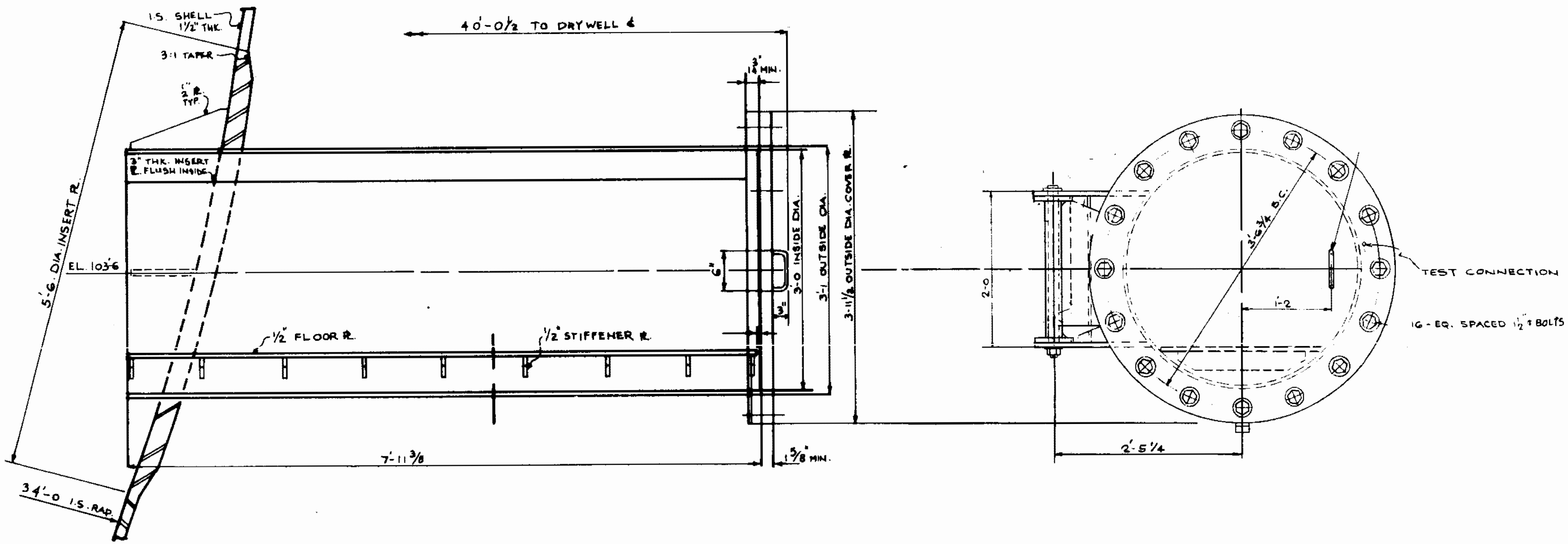
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

EQUIPMENT HATCH C-2 AND PERSONNEL AIR LOCK

UPDATED FSAR

FIGURE 3.8-4



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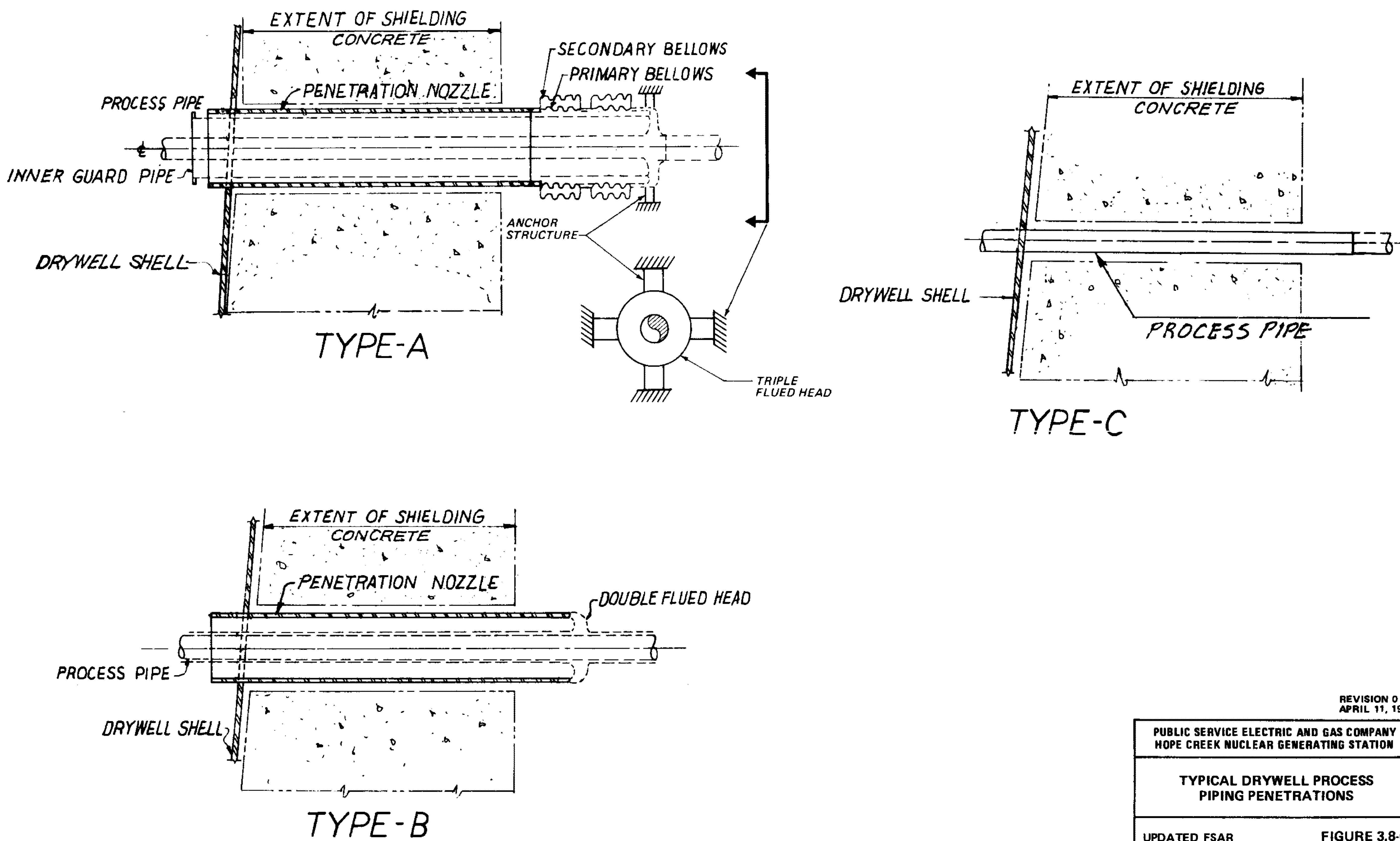
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

C.R.D. REMOVAL
HATCH C-3

UPDATED FSAR

FIGURE 3.8-5



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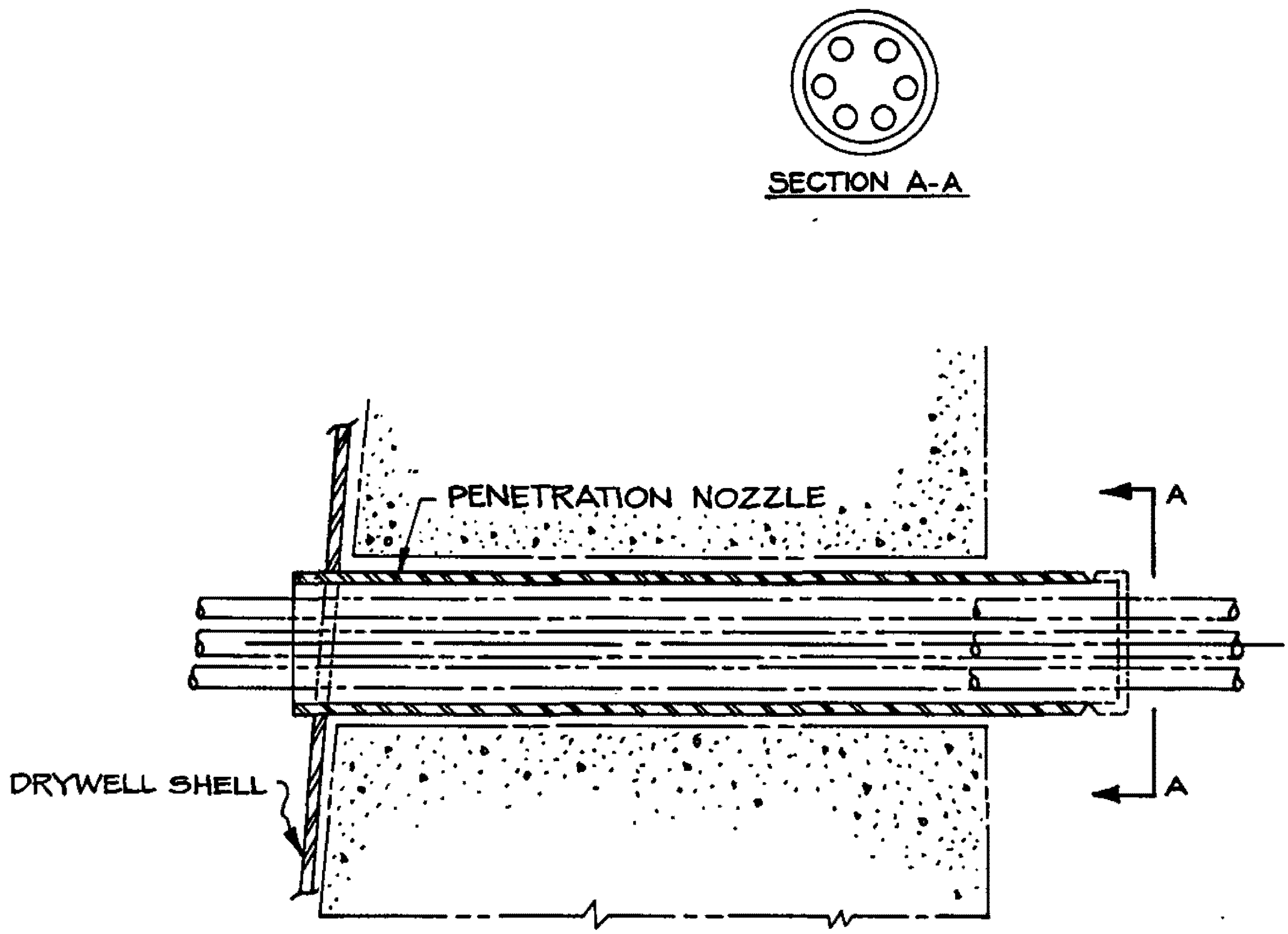
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

TYPICAL DRYWELL PROCESS PIPING PENETRATIONS

UPDATED FSAR

FIGURE 3.8-6



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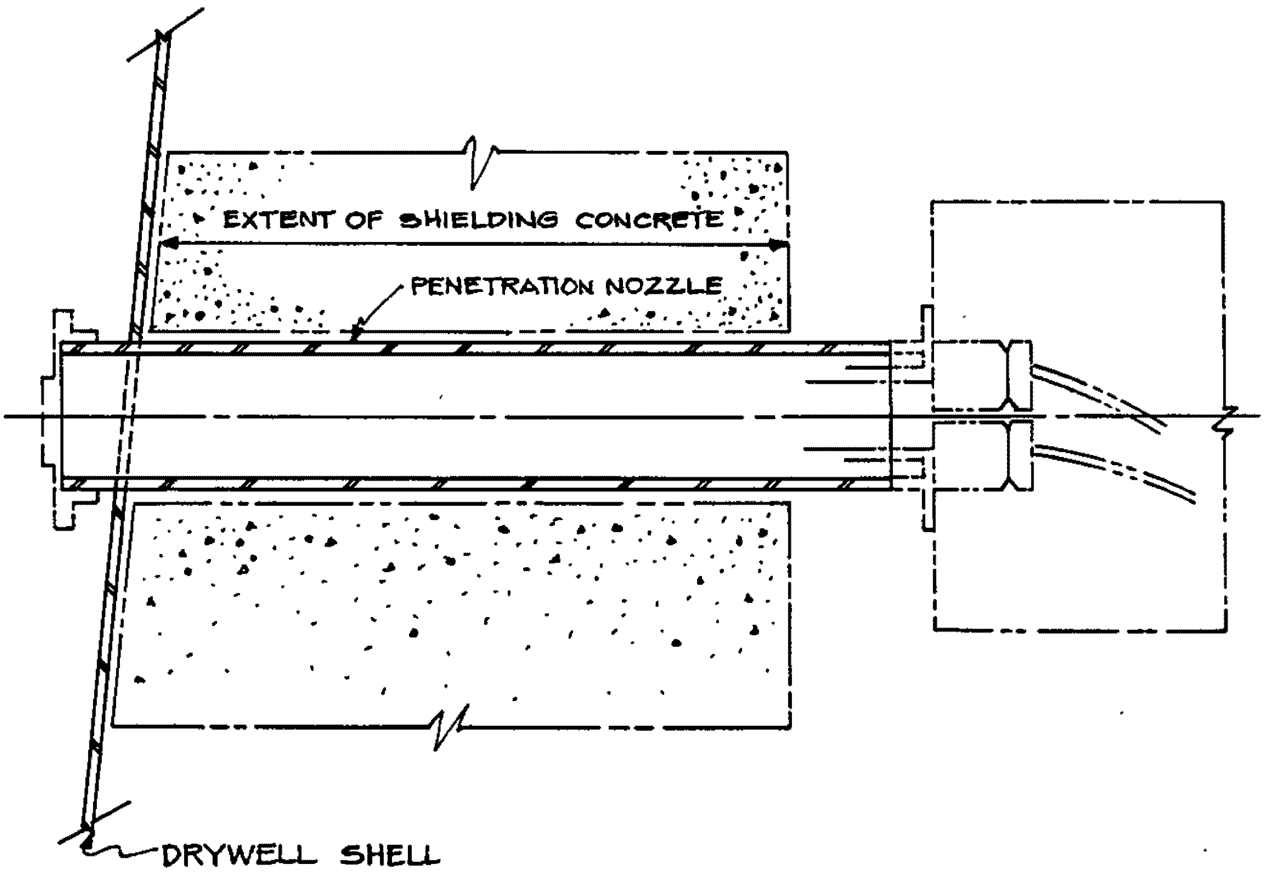
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

TYPICAL DRYWELL
INSTRUMENTATION PENETRATION

UPDATED FSAR

FIGURE 3.8-7



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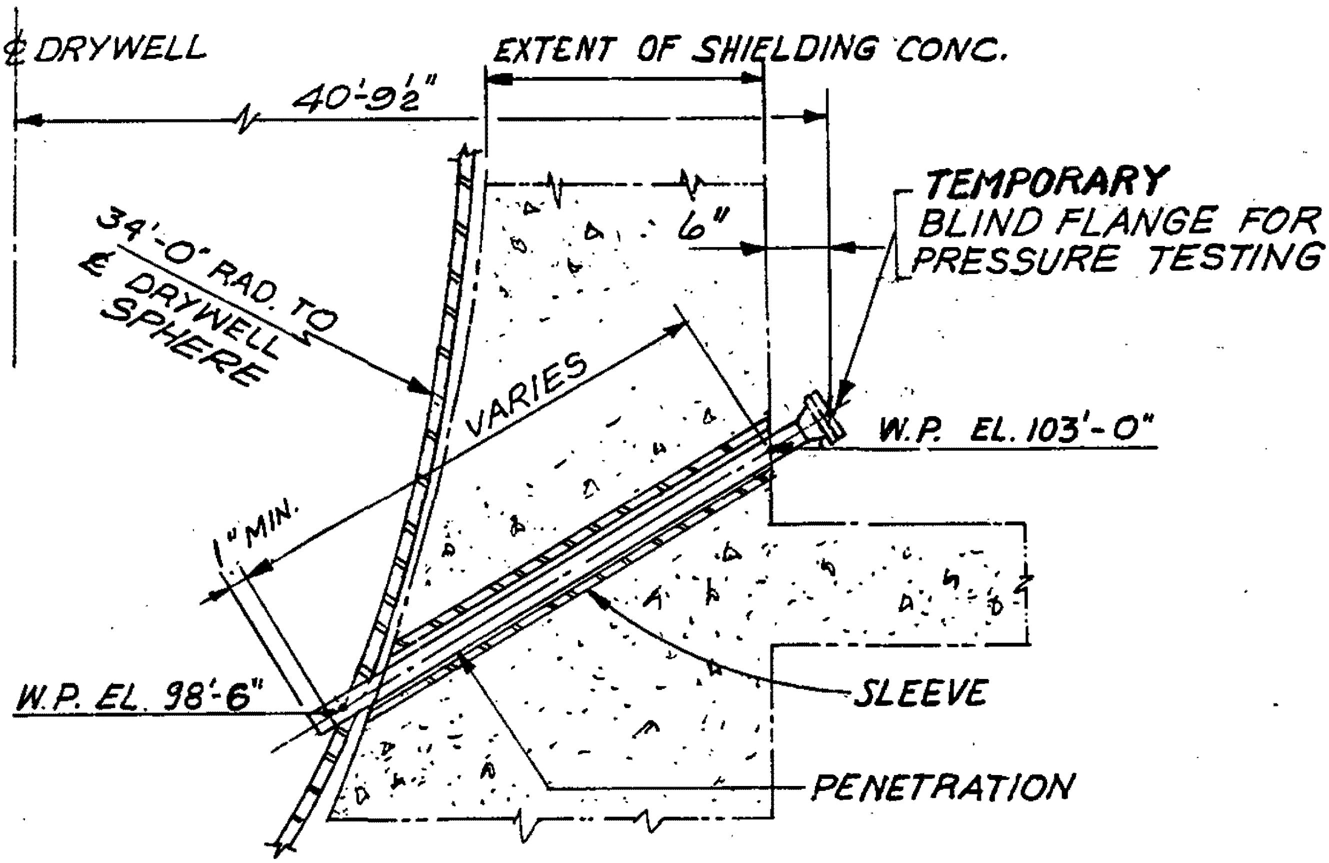
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

TYPICAL DRYWELL
ELECTRICAL PENETRATION

UPDATED FSAR

FIGURE 3.8-8



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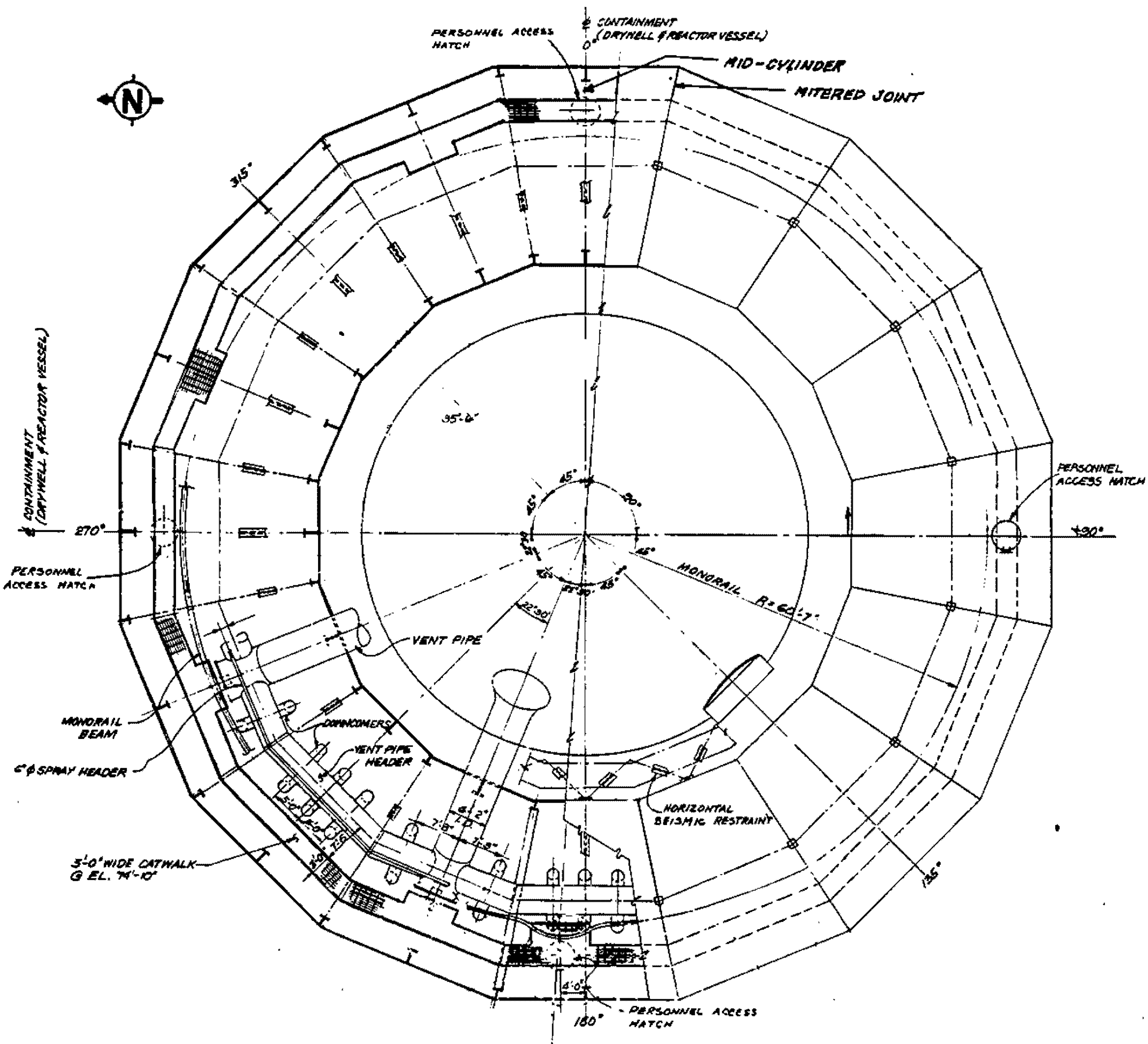
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

TYPICAL DRYWELL
T.I.P. SYSTEM PENETRATION

UPDATED FSAR

FIGURE 3.8-9



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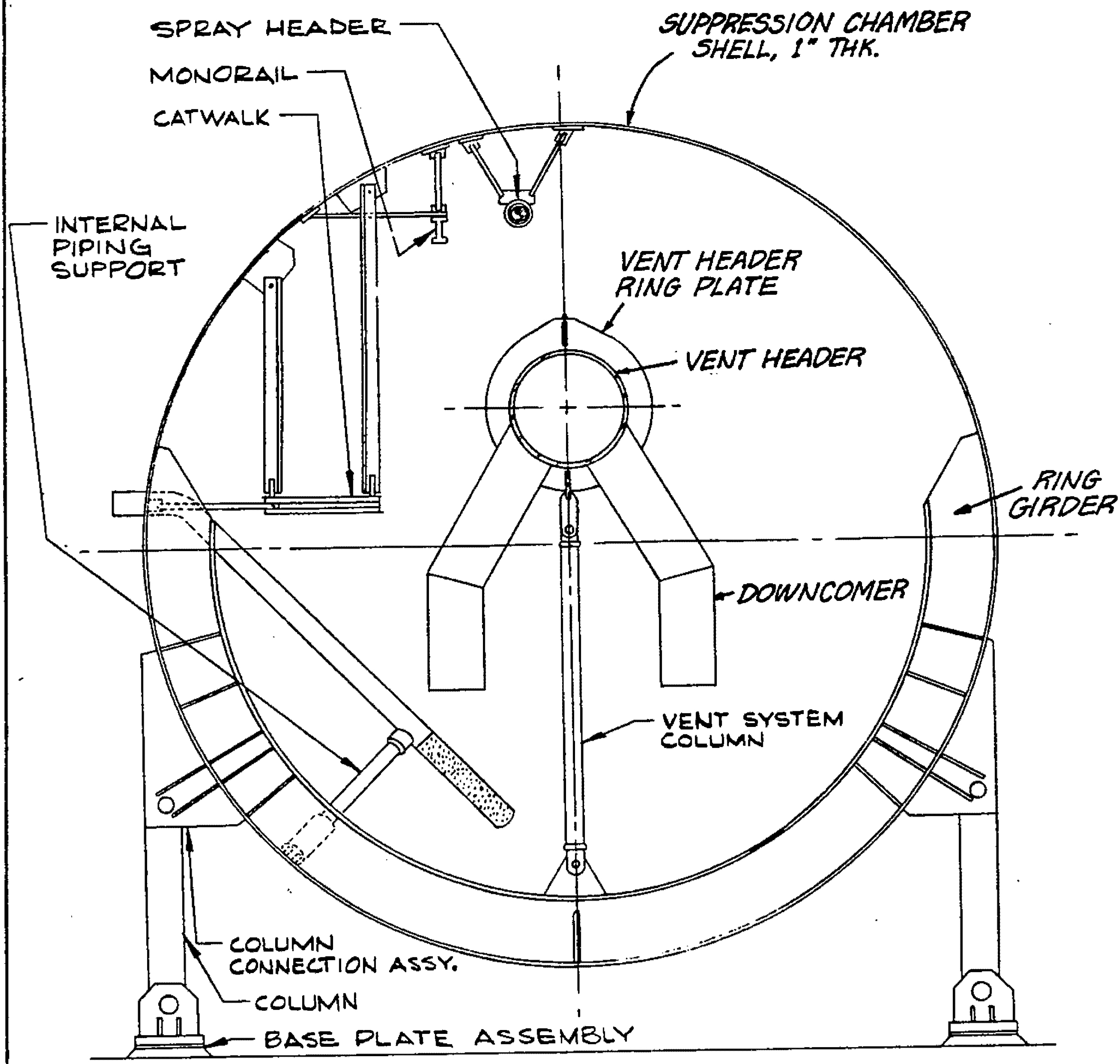
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

PLAN – SUPPRESSION CHAMBER

UPDATED FSAR

FIGURE 3.8-10



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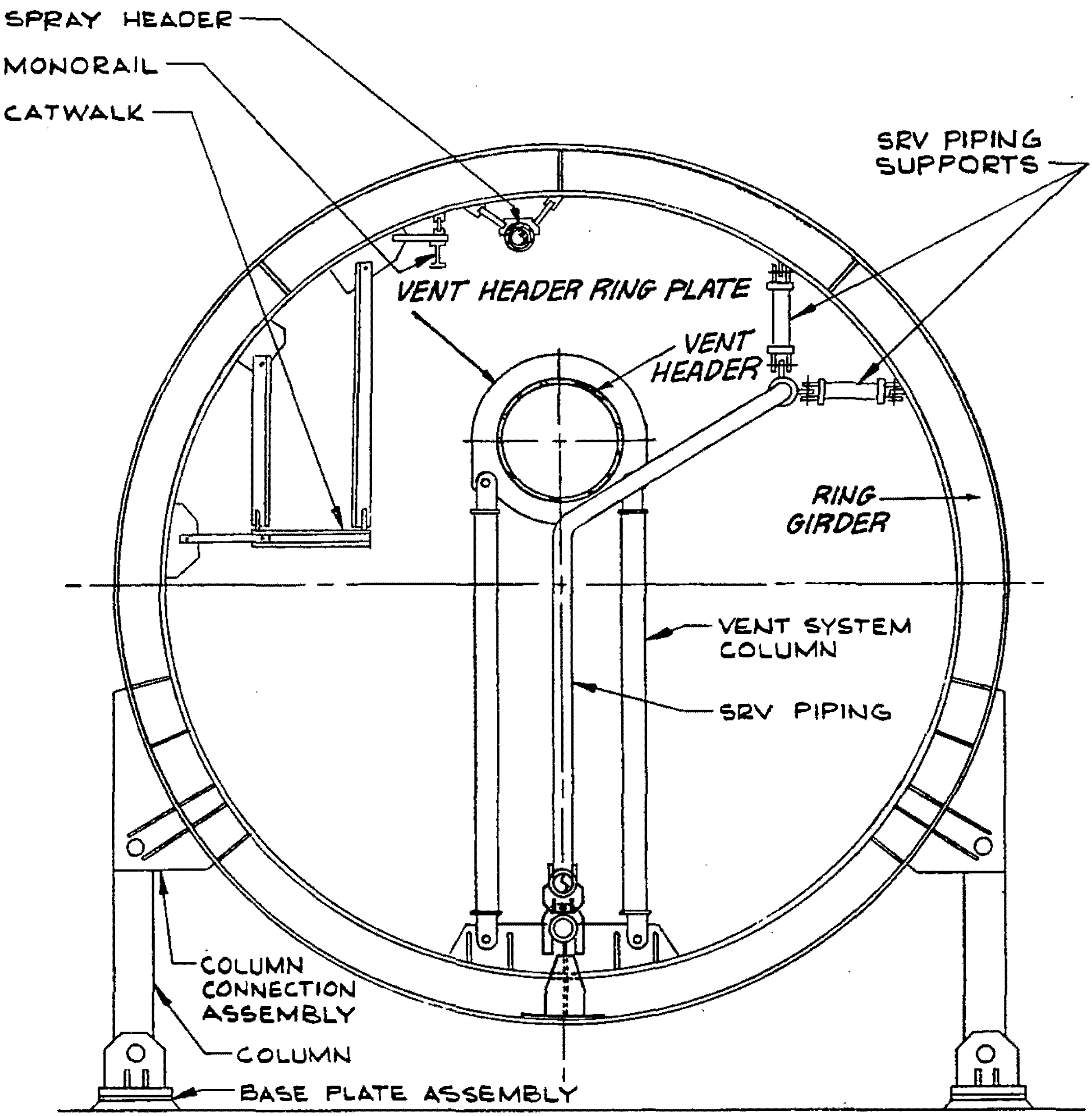
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

SUPPRESSION CHAMBER
SECTION AT MIDCYLINDER –
NON-VENT BAY

UPDATED FSAR

FIGURE 3.8-11



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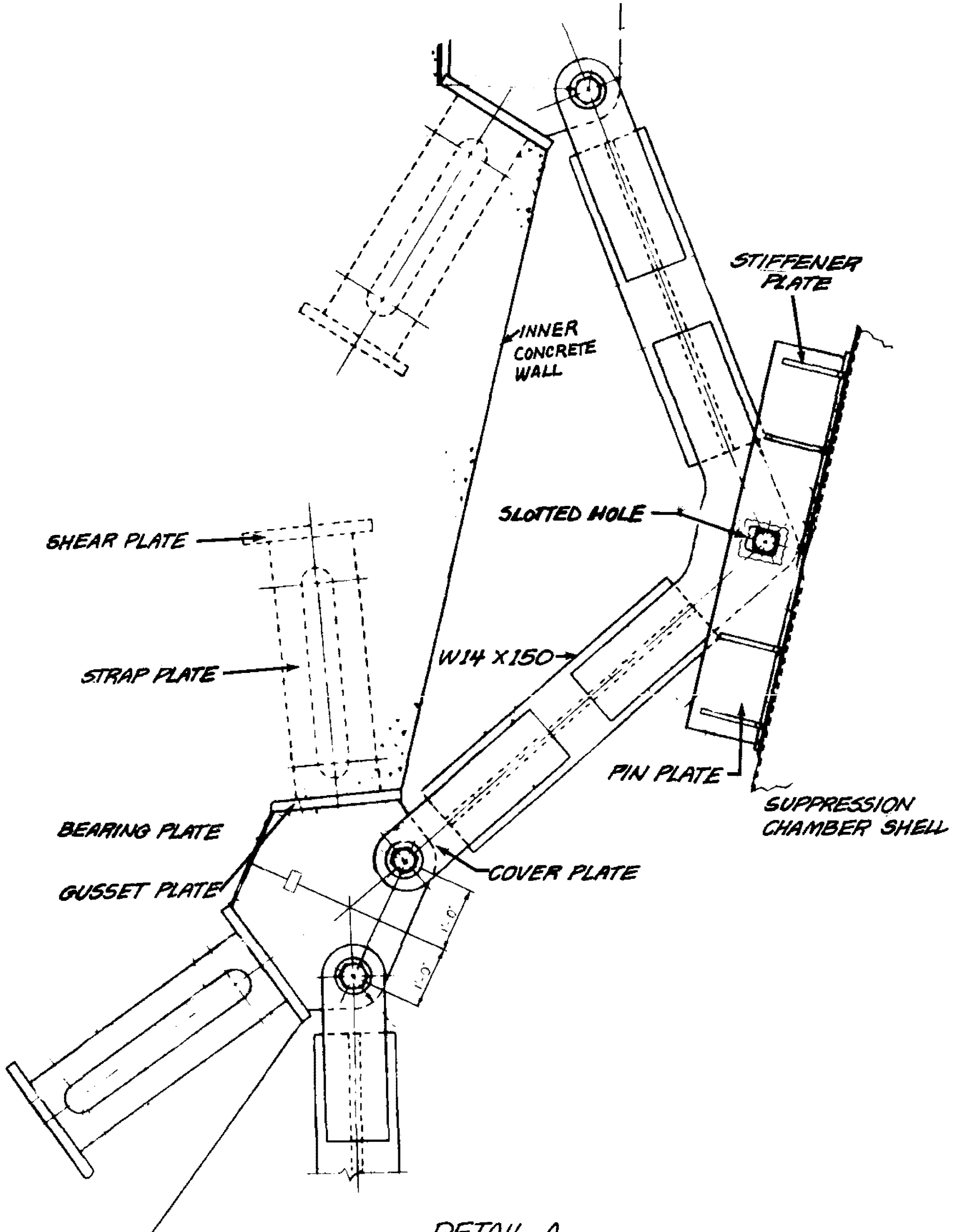
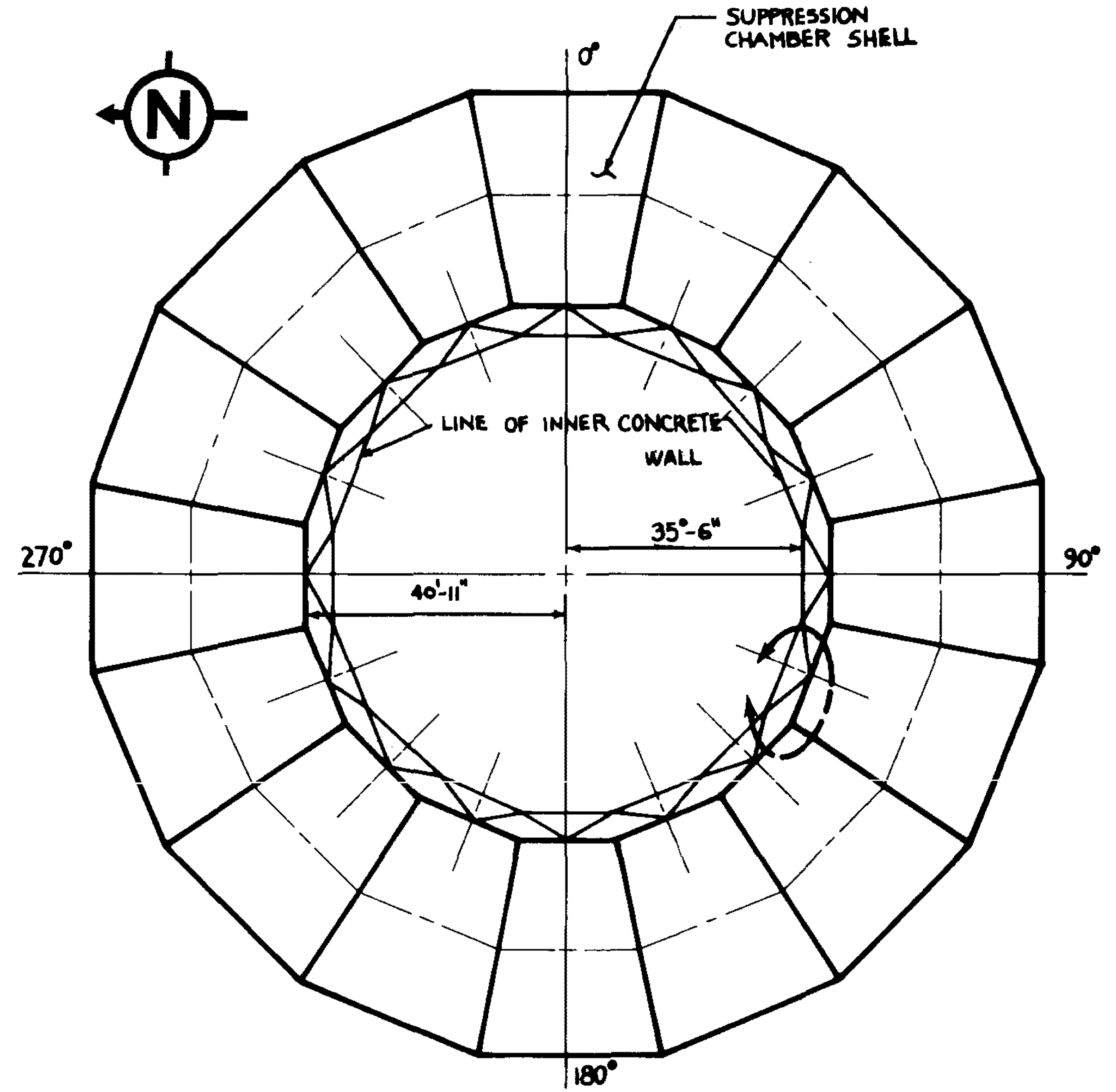
HOPE CREEK NUCLEAR GENERATING STATION

SUPPRESSION CHAMBER
SECTION AT MITERED JOINT



DETAIL A



SUPPRESSION CHAMBER - PLAN VIEW

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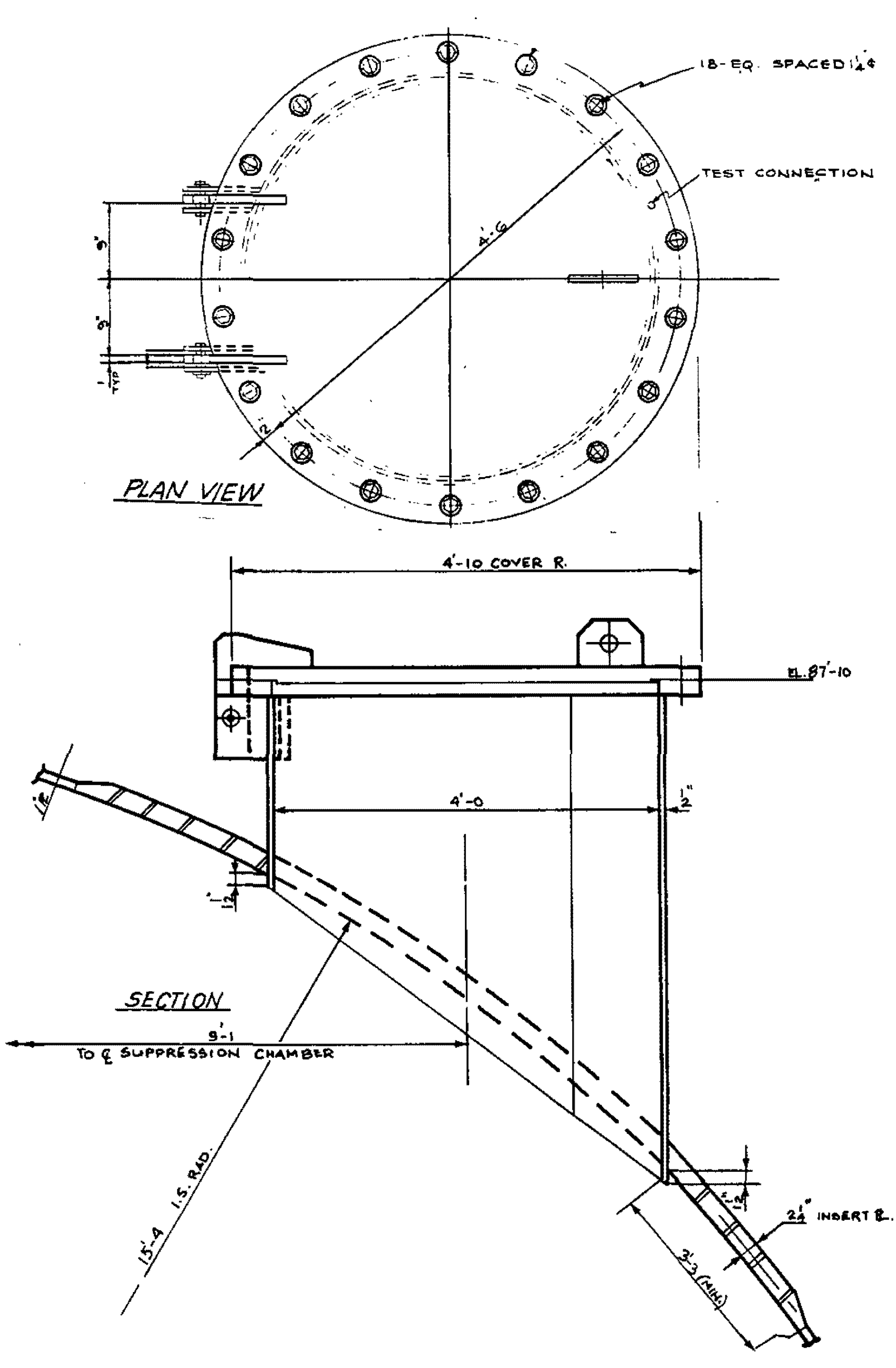
HOPE CREEK NUCLEAR GENERATING STATION

SUPPRESSION CHAMBER
HORIZONTAL SEISMIC RESTRAINT

UPDATED FSAR

FIGURE 3.8-13



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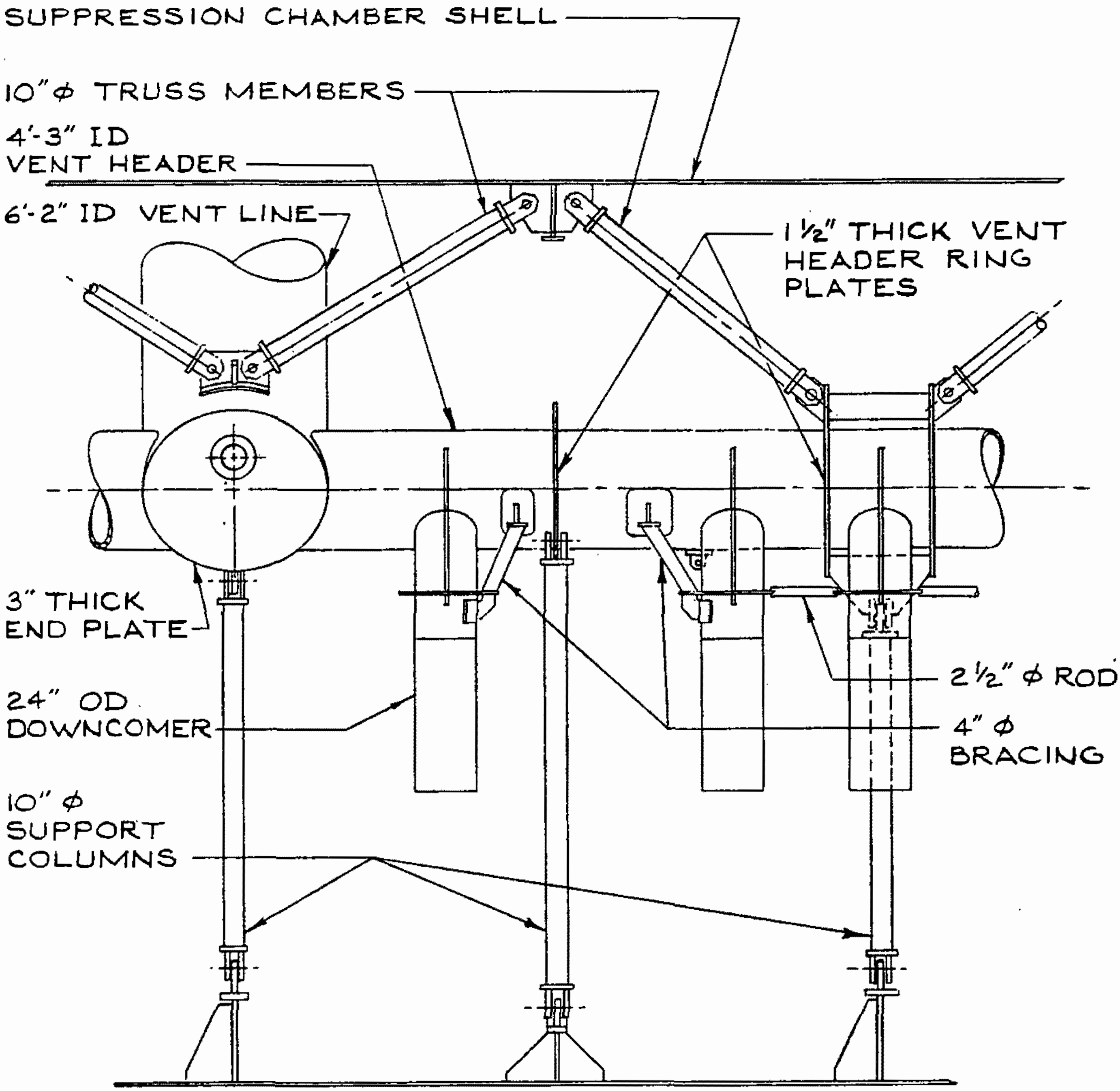
HOPE CREEK NUCLEAR GENERATING STATION

SUPPRESSION CHAMBER
ACCESS HATCH

UPDATED FSAR

FIGURE 3.8-14



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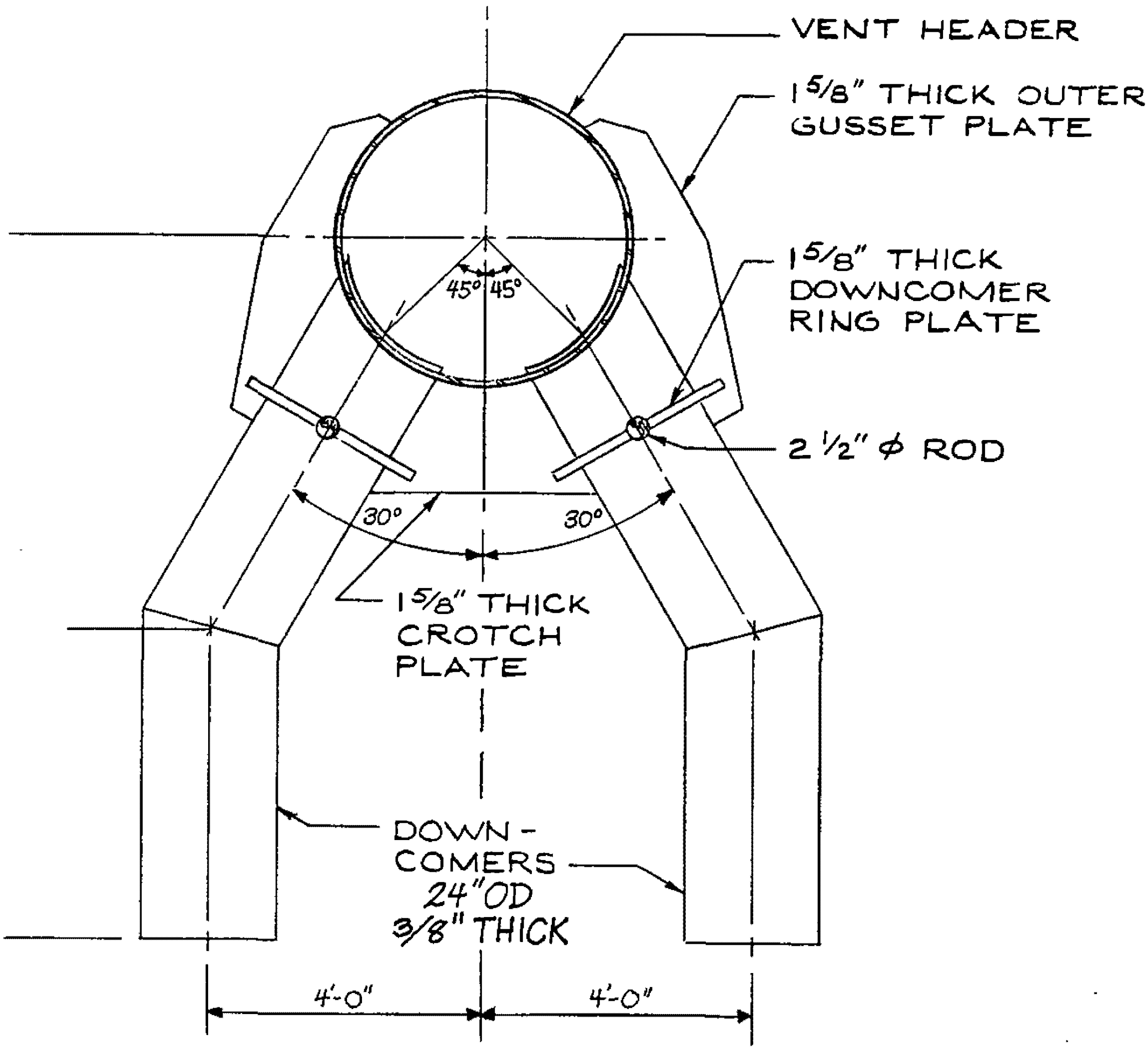
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

ELEVATION OF VENT SYSTEM

UPDATED FSAR

FIGURE 3.8-15



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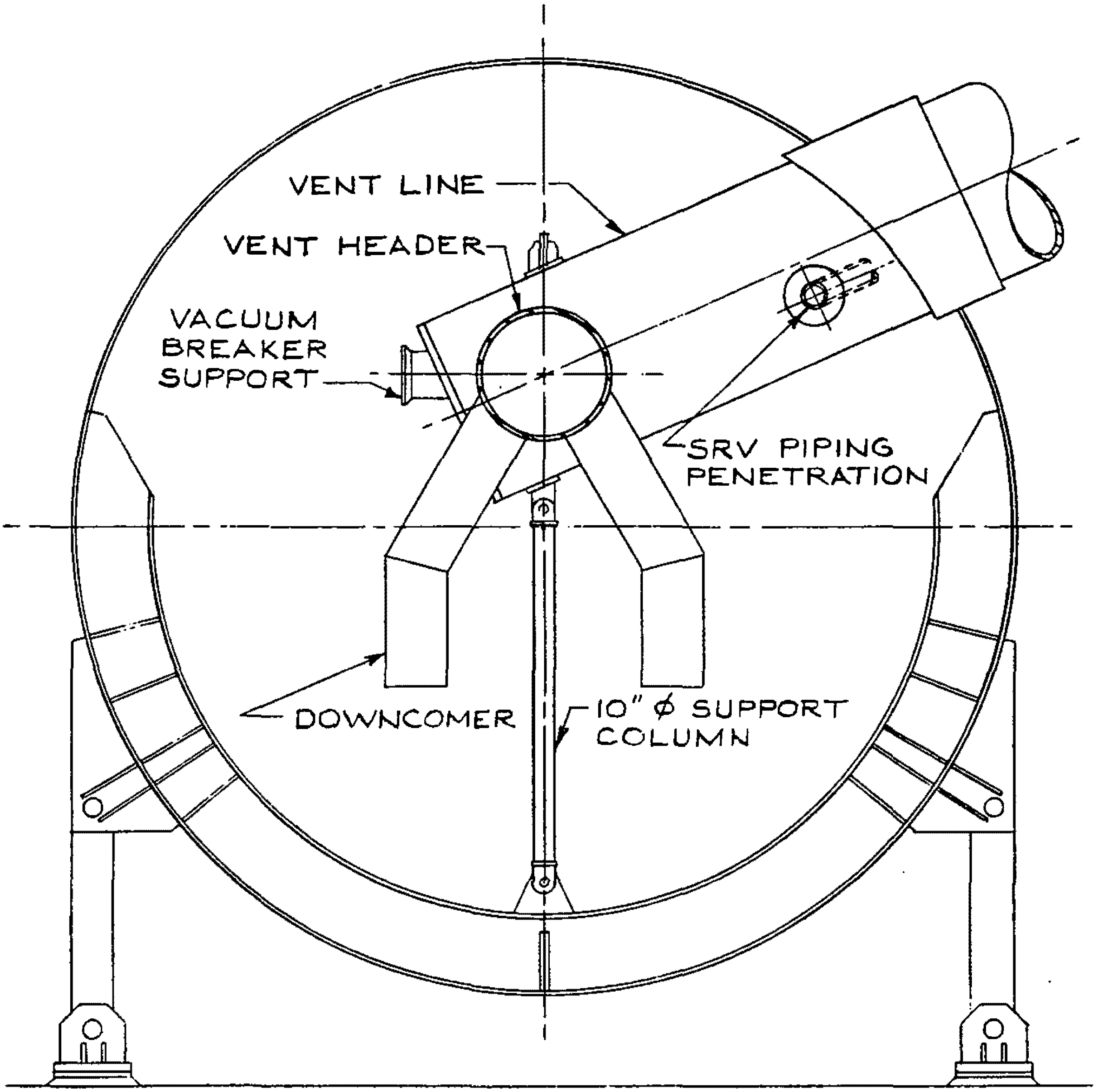
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

DETAIL OF VENT HEADER – DOWNCOMER INTERSECTION

UPDATED FSAR

FIGURE 3.8-16



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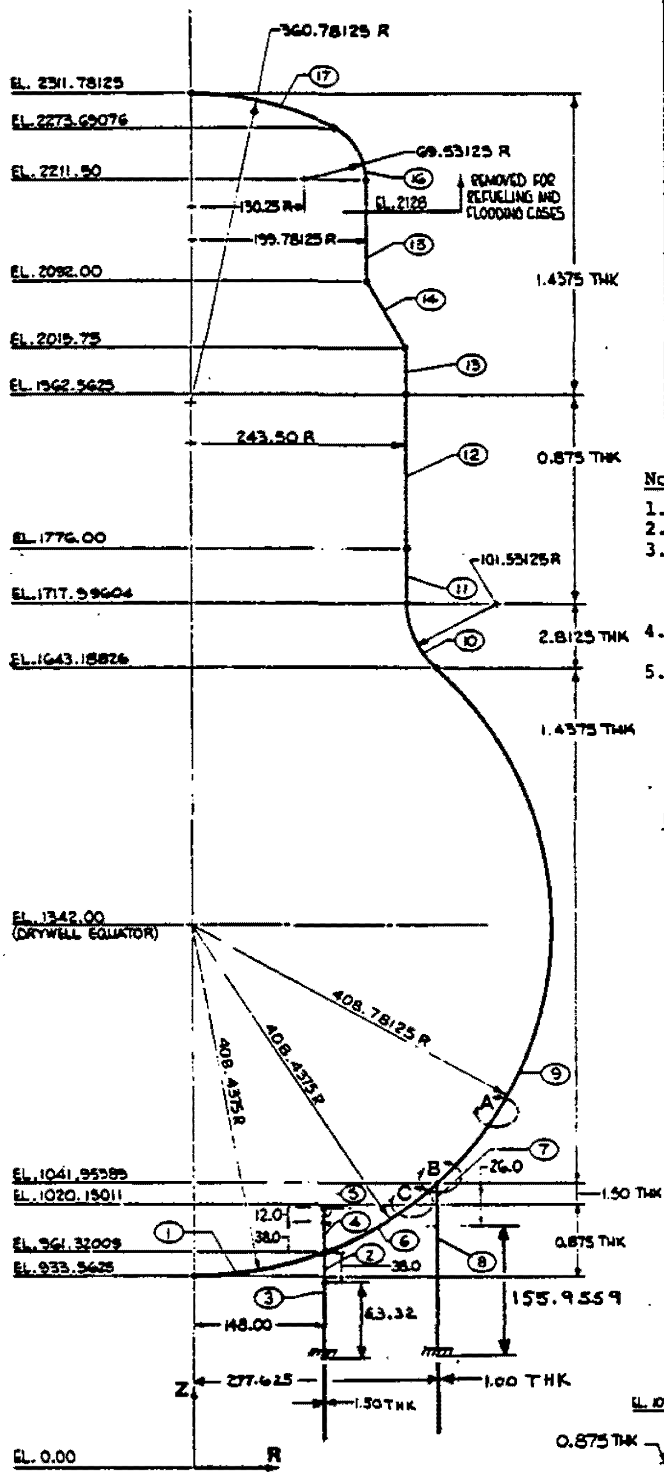
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

SUPPRESSION CHAMBER
SECTION AT MIDCYLINDER –
VENT LINE BAY

UPDATED FSAR

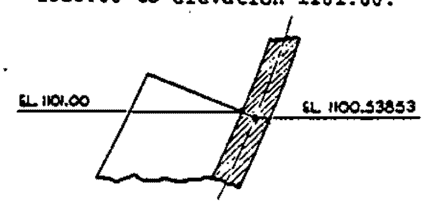
FIGURE 3.8-17



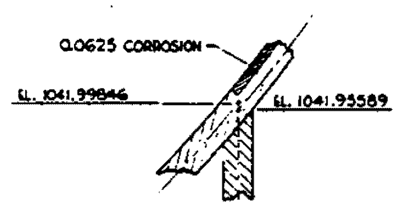
Segment	Arc Length	Mesh Pts.
1	151.4466	28
2	38.00	10
3	63.32	25
4	38.00	10
5	12.00	5
6	153.8143	29
7	26.00	10
8	155.9559	30
9	675.5139	80
10	84.1002	17
11	58.00	14
12	186.525	43
13	53.1875	14
14	88.0345	14
15	119.50	25
16	76.9817	10
17	167.2740	10
Total		374

Notes:

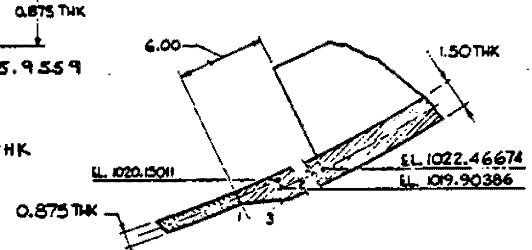
1. All dimensions are in inches.
2. Segments are shown as (x) .
3. Thicknesses examined are given in the corroded condition. (Note that no corrosion is specified for the bottom 7/8" & 1 1/2" plate since they are buried in concrete.)
4. The number of mesh points used in BOSOR4 model is specified above.
5. The drywell sphere is vertically stiffened with 64-1"x8" bars from approximately elevation 1023.00 to elevation 1101.00.



Detail A



Detail B



Detail C

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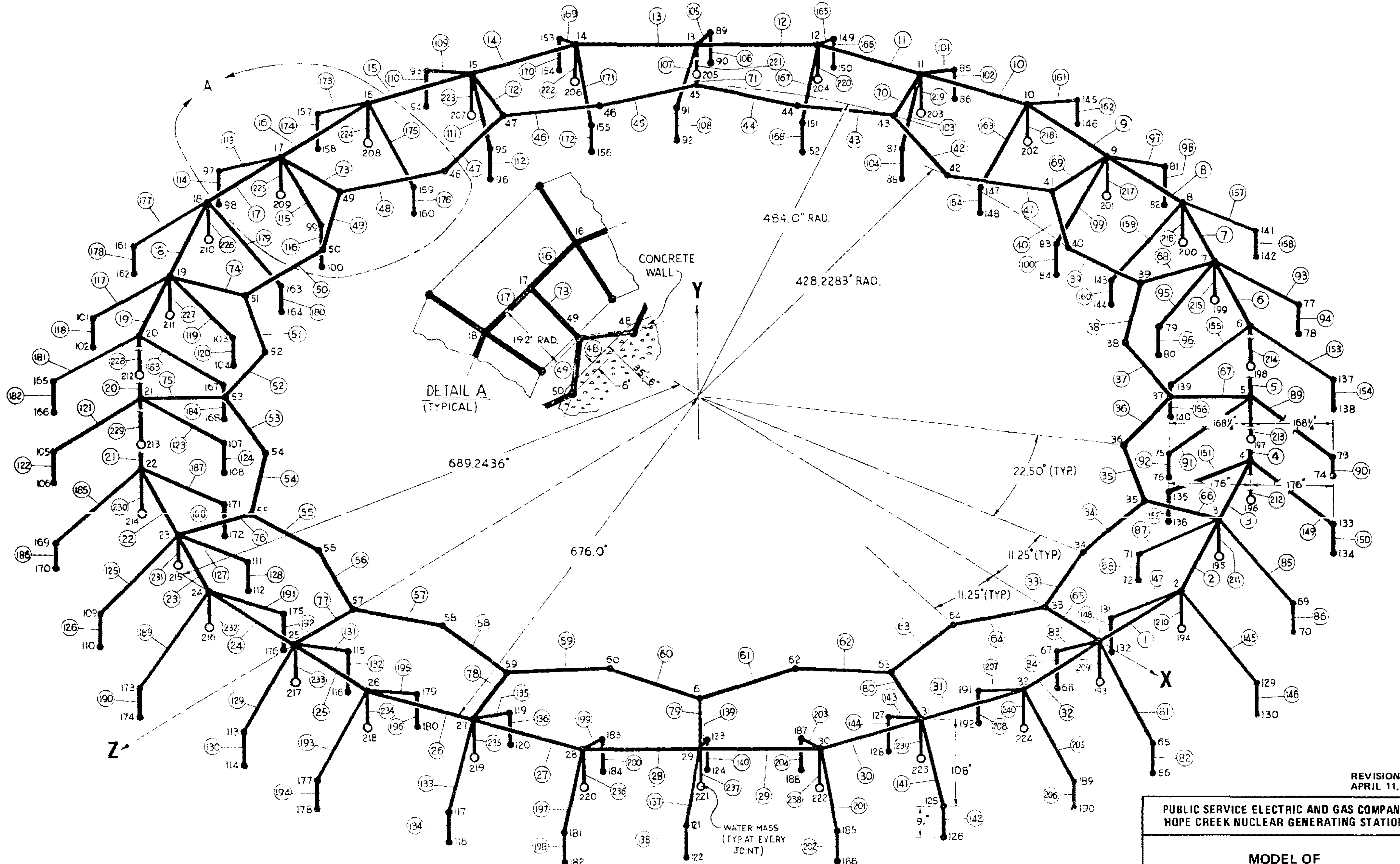
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

BOSOR4
MODEL OF DRYWELL SHELL

UPDATED FSAR

FIGURE 3.8-18



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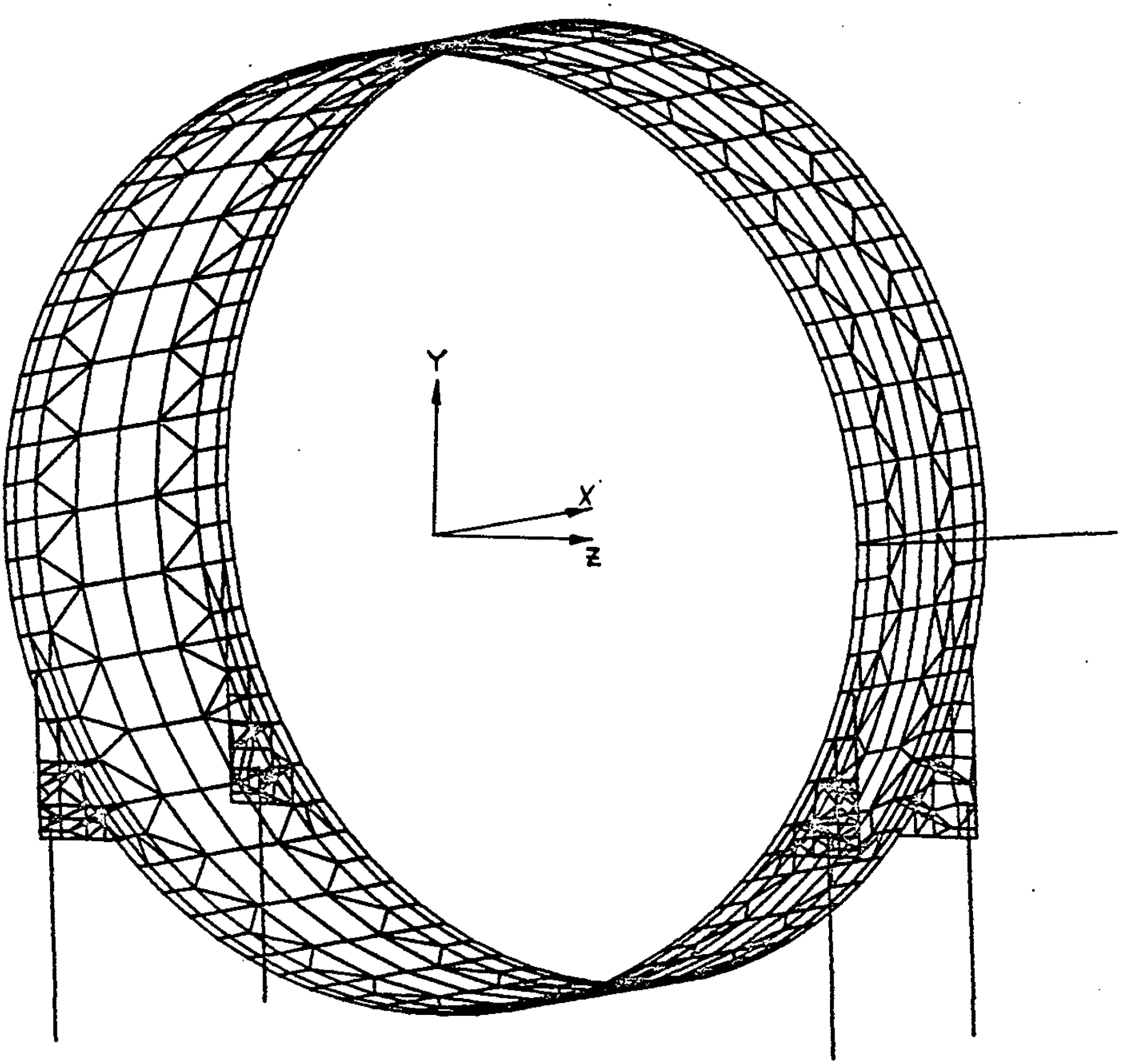
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

MODEL OF
SUPPRESSION CHAMBER

UPDATED FSAR

FIGURE 3.8-19



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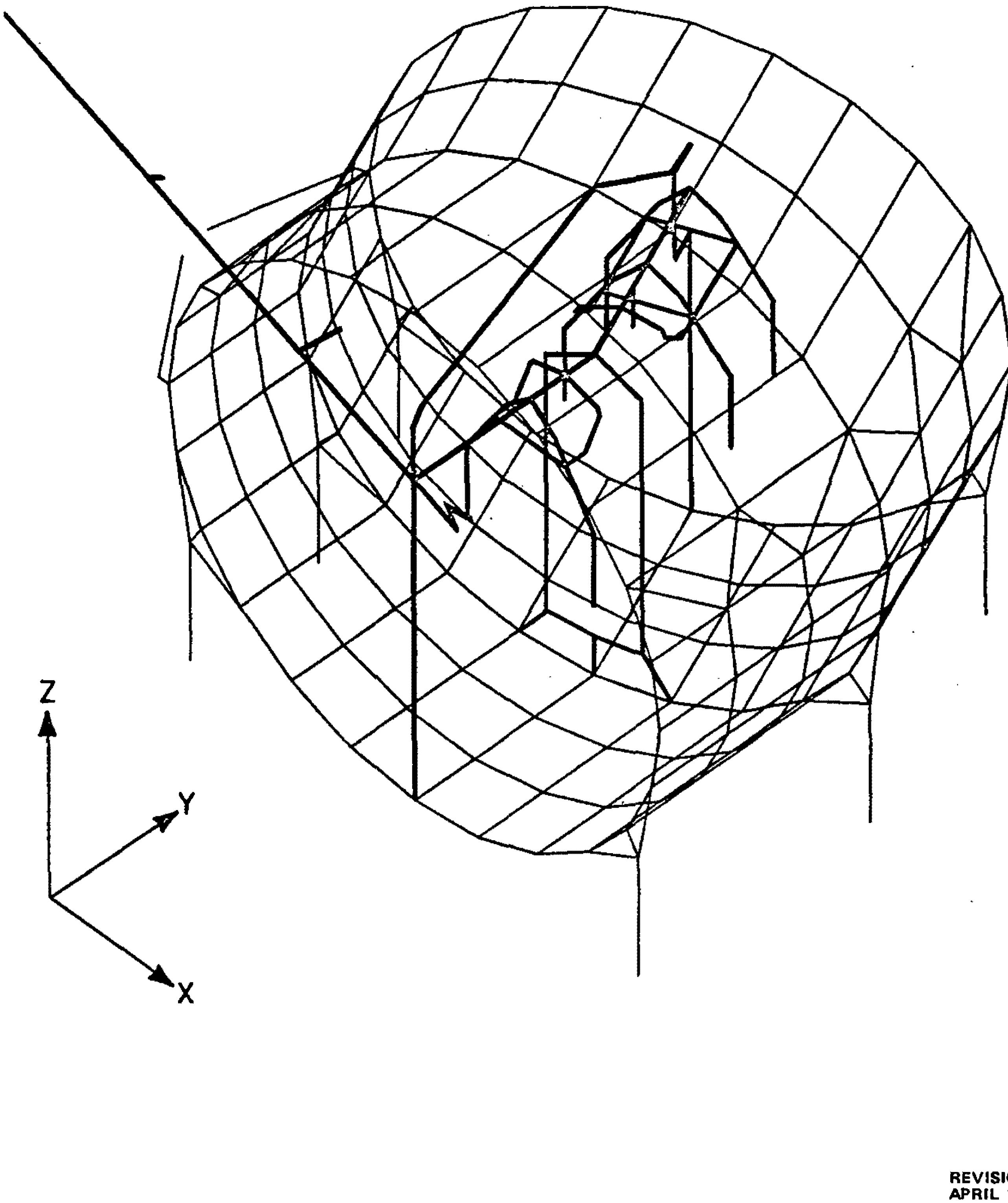
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

MODEL OF
SUPPRESSION CHAMBER
1/32 SEGMENT – ISOMETRIC VIEW

UPDATED FSAR

FIGURE 3.8-20



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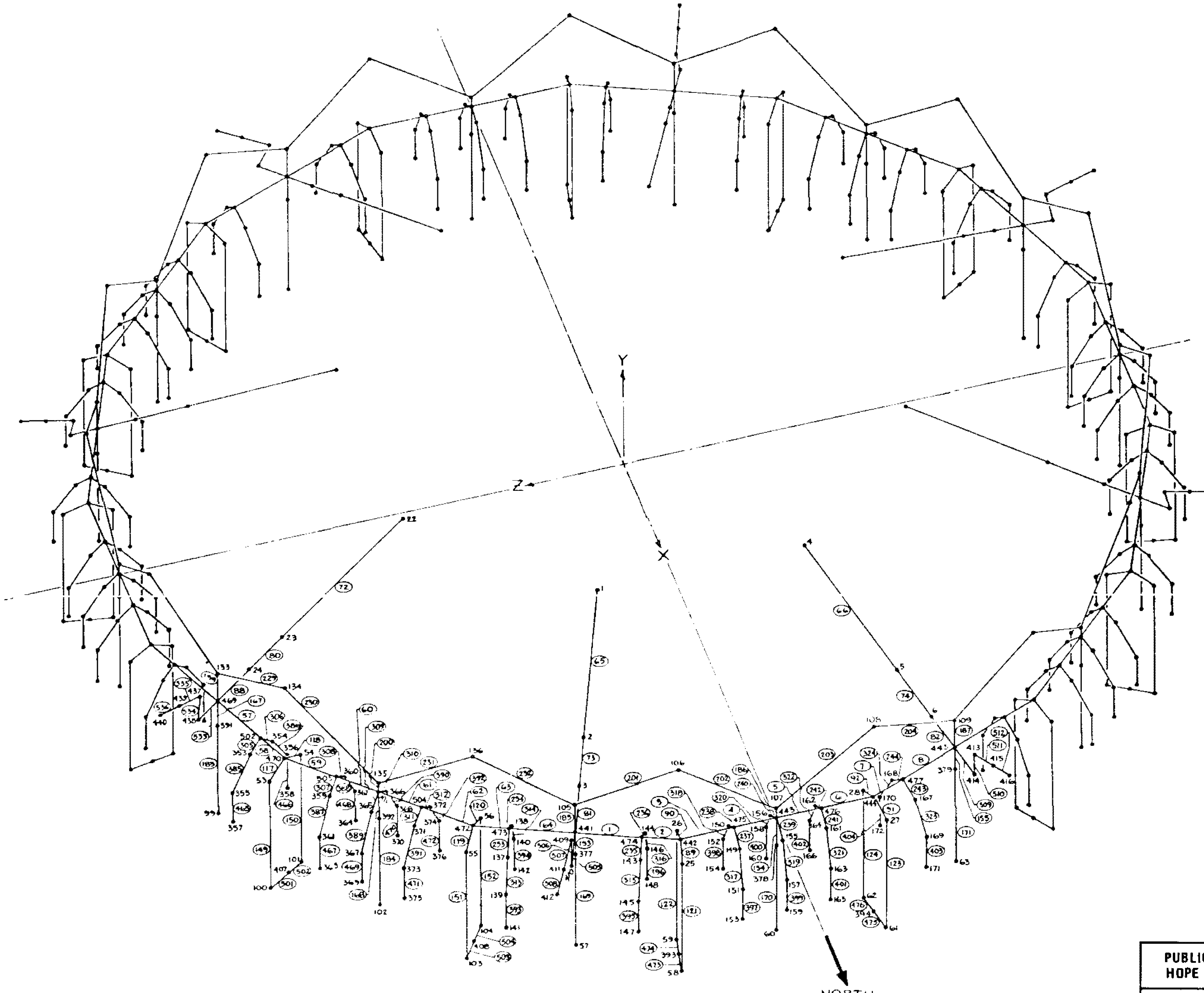
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

MODEL OF VENT SYSTEM AND SUPPRESSION CHAMBER 1/16 SEGMENT – ISOMETRIC VIEW

UPDATED FSAR

FIGURE 3.8-21



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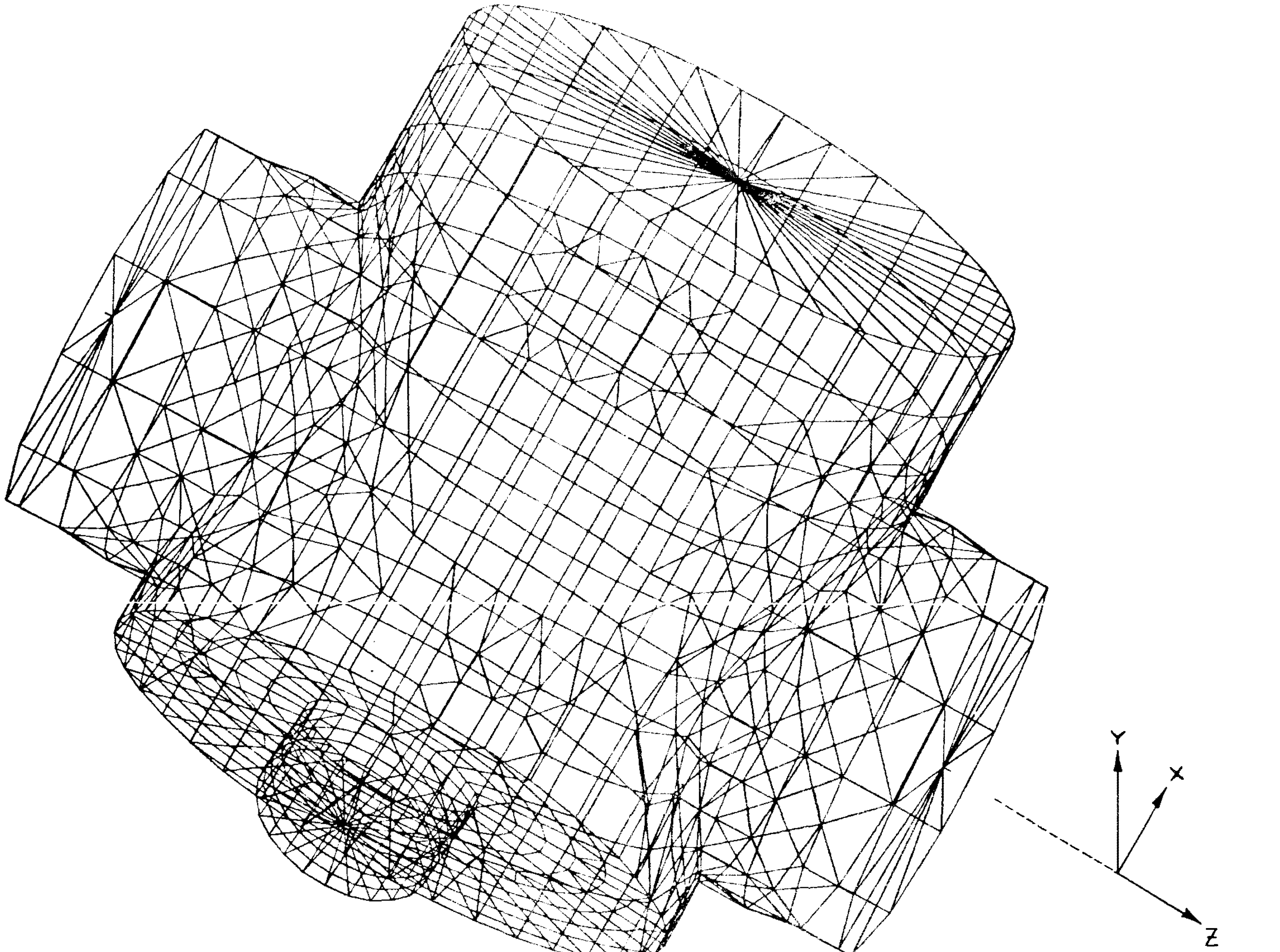
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

MODEL OF VENT SYSTEM

UPDATED FSAR

FIGURE 3.8-22



ORIGIN IS LOCATED ALONG THE ℄ OF THE VENT LINE AT THE ℄ OF THE VENT HEADER.

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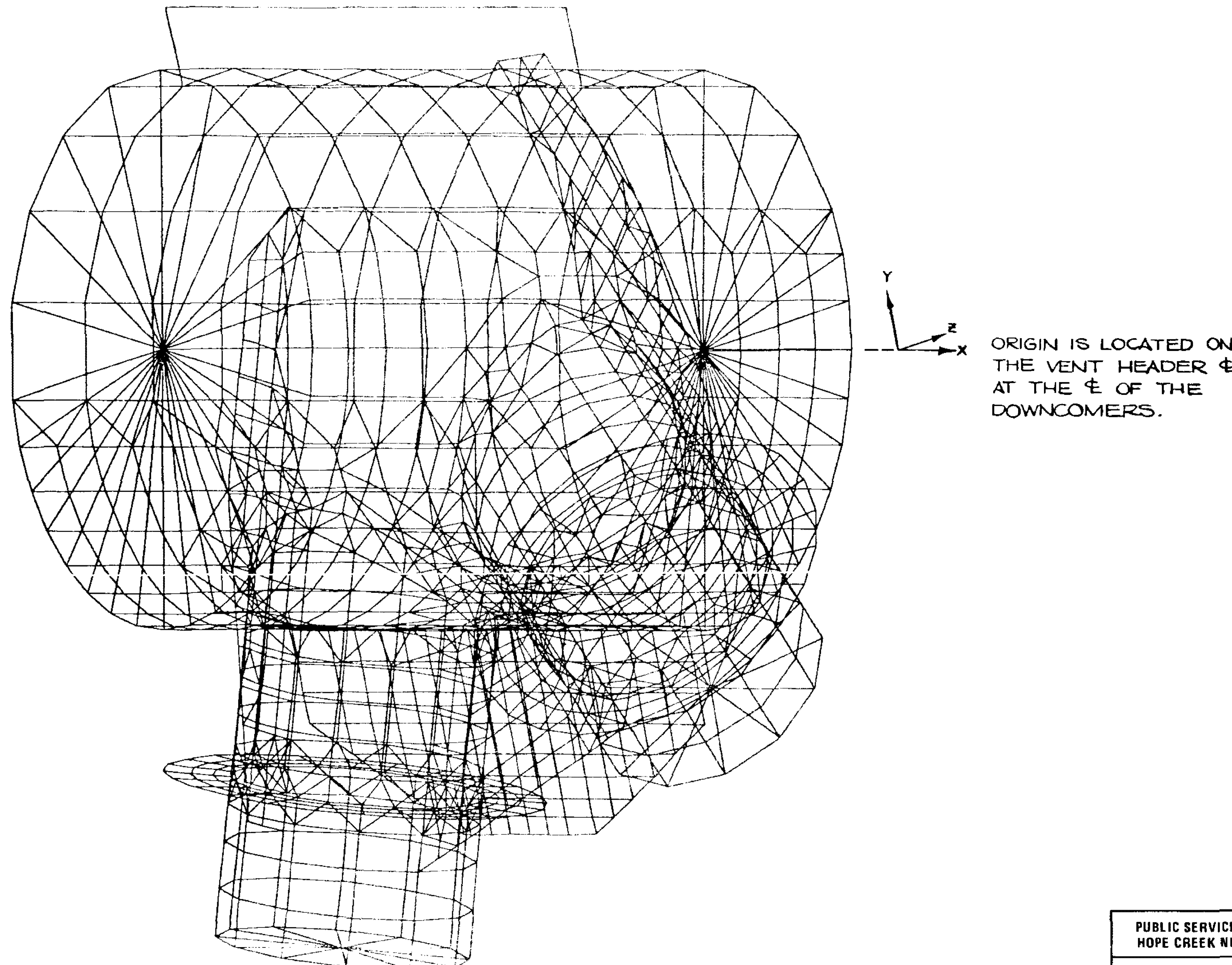
HOPE CREEK NUCLEAR GENERATING STATION

MODEL OF VENT LINE – VENT HEADER INTERSECTION – ISOMETRIC VIEW

UPDATED FSAR

FIGURE 3.8-23



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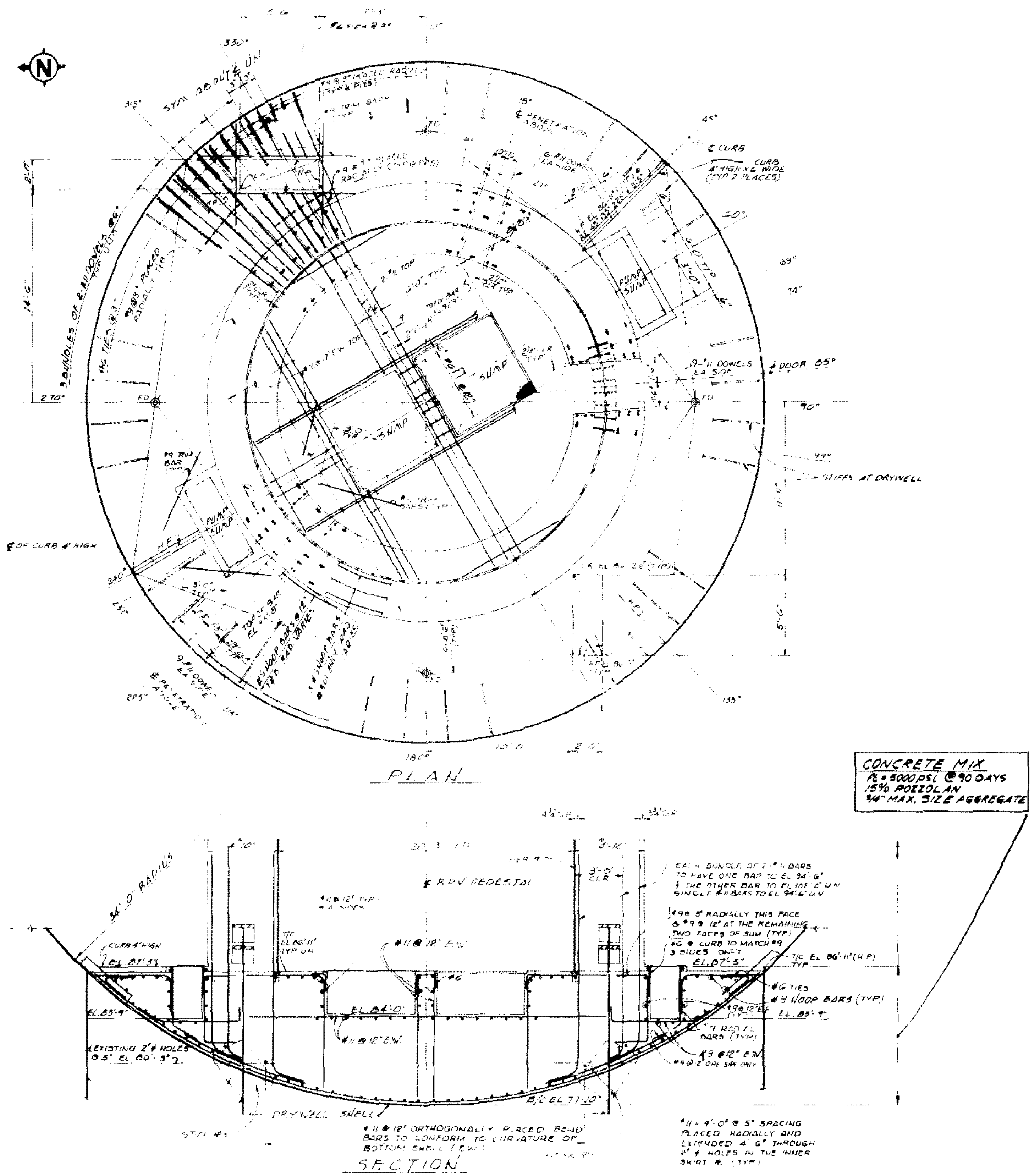
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

MODEL OF VENT HEADER – DOWNCOMER INTERSECTION – ISOMETRIC VIEW

UPDATED FSAR

FIGURE 3.8-24



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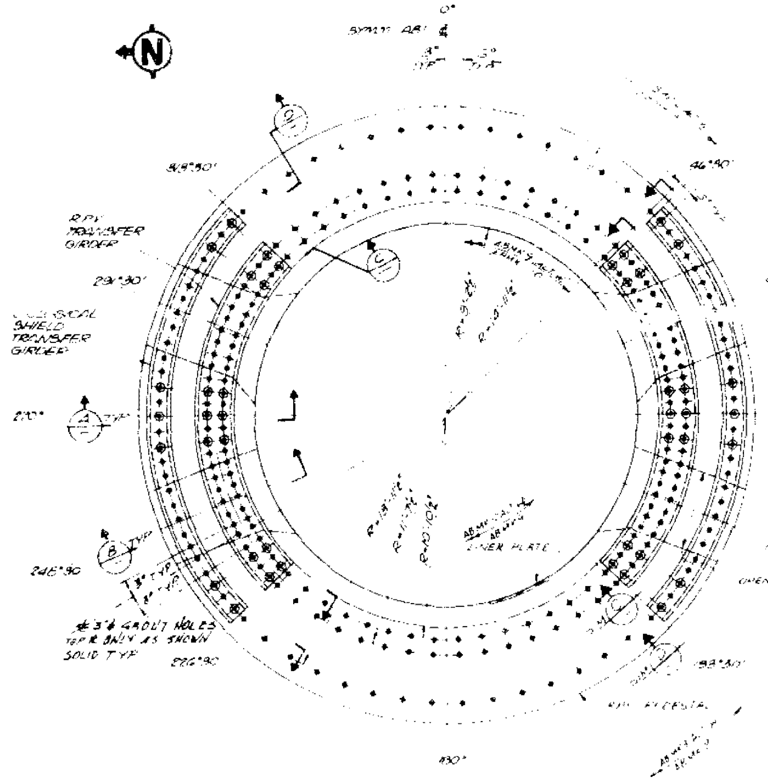
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

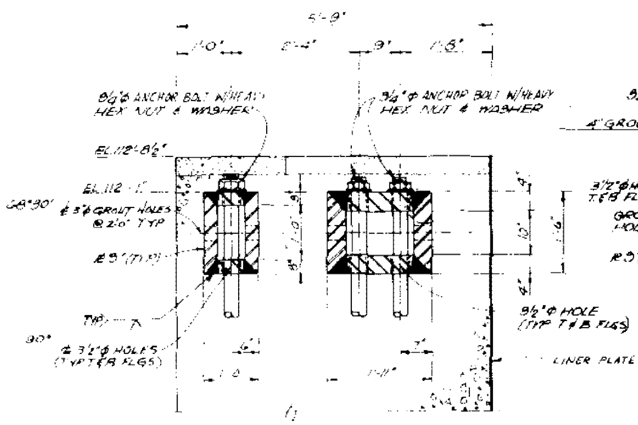
RPV PEDESTAL
FLOOR REINFORCING

UPDATED FSAR

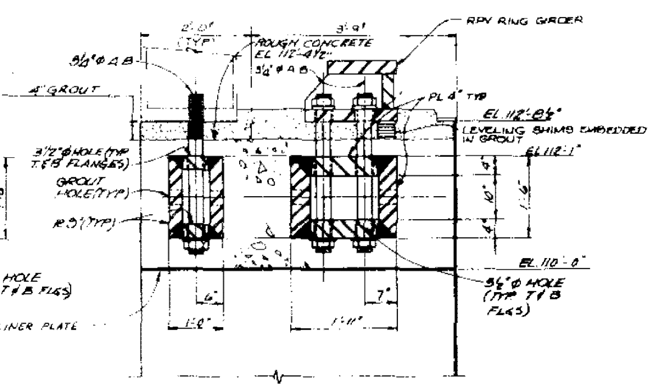
FIGURE 3.8-25



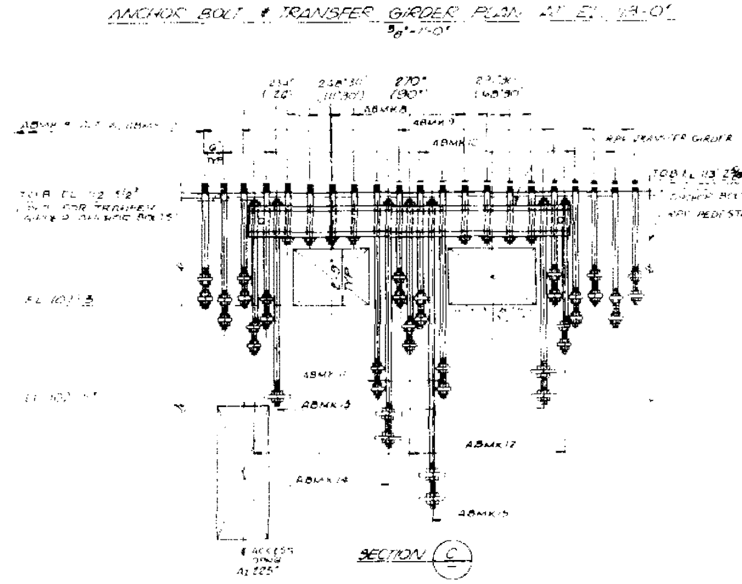
ANCHOR BOLT & TRANSFER GIRDER PLAN AT EL. 118'-0"



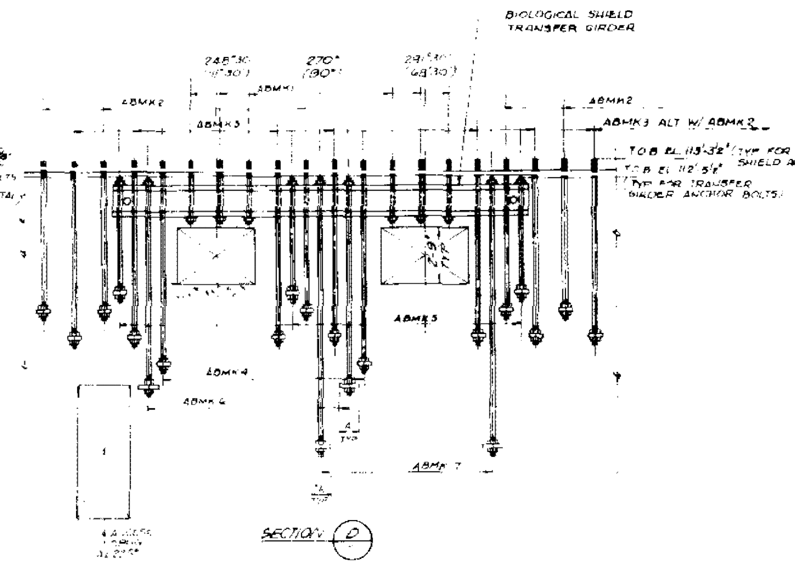
SECTION A



SECTION B

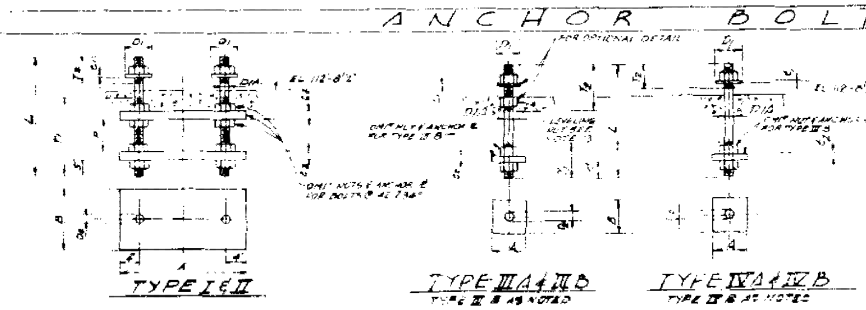


SECTION C



SECTION D

ANCHOR BOLT SCHEDULE (PER UNIT)



TYPE I & II

TYPE III A & III B

TYPE IV A & IV B

MARK	NO. OF BOLTS	NO. OF BRG R'S	TYPE	BOLT DIA.	BOLT L	BOLT P	BOLT T_1	BOLT T_2	ANCHOR PLATE A	ANCHOR PLATE B	ANCHOR PLATE t_2	ANCHOR PLATE D_2	WASHERS D_1	WASHERS D_2	WASHERS E_1	REMARKS
1	12	—	III B	3¼"	3'-1"	—	6"	12"	—	—	—	—	6"	3⅝"	¼"	BIOLOGICAL SHIELD ANCHOR BOLTS
2	20	20	III A	3¼"	7'-9"	—	12"	12"	8"	8"	2½"	3⅝"	6"	3⅝"	¼"	" " " "
3	24	24	III A	3¼"	9'-0"	—	12"	12"	8"	8"	2½"	3⅝"	6"	3⅝"	¼"	" " " "
4	4	4	III A	3¼"	10'-3"	—	12"	12"	8"	8"	2½"	3⅝"	6"	3⅝"	¼"	" " " "
5	6	6	IV A	3¼"	6'-0"	—	12"	6"	8"	8"	2½"	3⅝"	6"	3⅝"	¼"	BIO SHIELD TRANSFER GIRDER A-BOLTS
6	4	4	IV A	3¼"	10'-6"	—	13"	6"	12"	11"	3¼"	3⅝"	6"	3⅝"	¼"	" " " " " "
7	4	4	IV A	3¼"	15'-6"	—	13"	6"	12"	11"	3¼"	3⅝"	6"	3⅝"	¼"	" " " " " "
8	72	—	IV B	3¼"	3'-1"	—	6"	6"	—	—	—	—	6"	3⅝"	⅜"	RPV ANCHOR BOLTS SEE NOTE 6
9	40	20 SETS	I	3¼"	6'-0"	8"	2'-0"	6"	17"	8"	2½"	3⅝"	6"	3⅝"	⅜"	" " " " " "
10	40	20 SETS	I	3¼"	7'-3"	8"	2'-0"	6"	17"	8"	2½"	3⅝"	6"	3⅝"	⅜"	" " " " " "
11	8	4 SETS	I	3¼"	10'-6"	8"	2'-0"	6"	17"	8"	2½"	3⅝"	6"	3⅝"	⅜"	" " " " " "
12	12	6 SETS	I	3¼"	7'-6"	8"	2'-0"	6"	17"	8"	2½"	3⅝"	6"	3⅝"	⅜"	RPV TRANSFER GIRDER ANCHOR BOLTS SEE NOTE 6
13	8	4 SETS	II	3¼"	10'-0"	12"	2'-4"	6"	17"	15"	3¼"	3⅝"	6"	3⅝"	⅜"	" " " " " " " "
14	4	2 SETS	II	3¼"	12'-0"	18"	2'-4"	6"	17"	15"	3¼"	3⅝"	6"	3⅝"	⅜"	" " " " " " " "
15	4	2 SETS	II	3¼"	15'-0"	12"	2'-4"	6"	17"	15"	3¼"	3⅝"	6"	3⅝"	⅜"	" " " " " " " "

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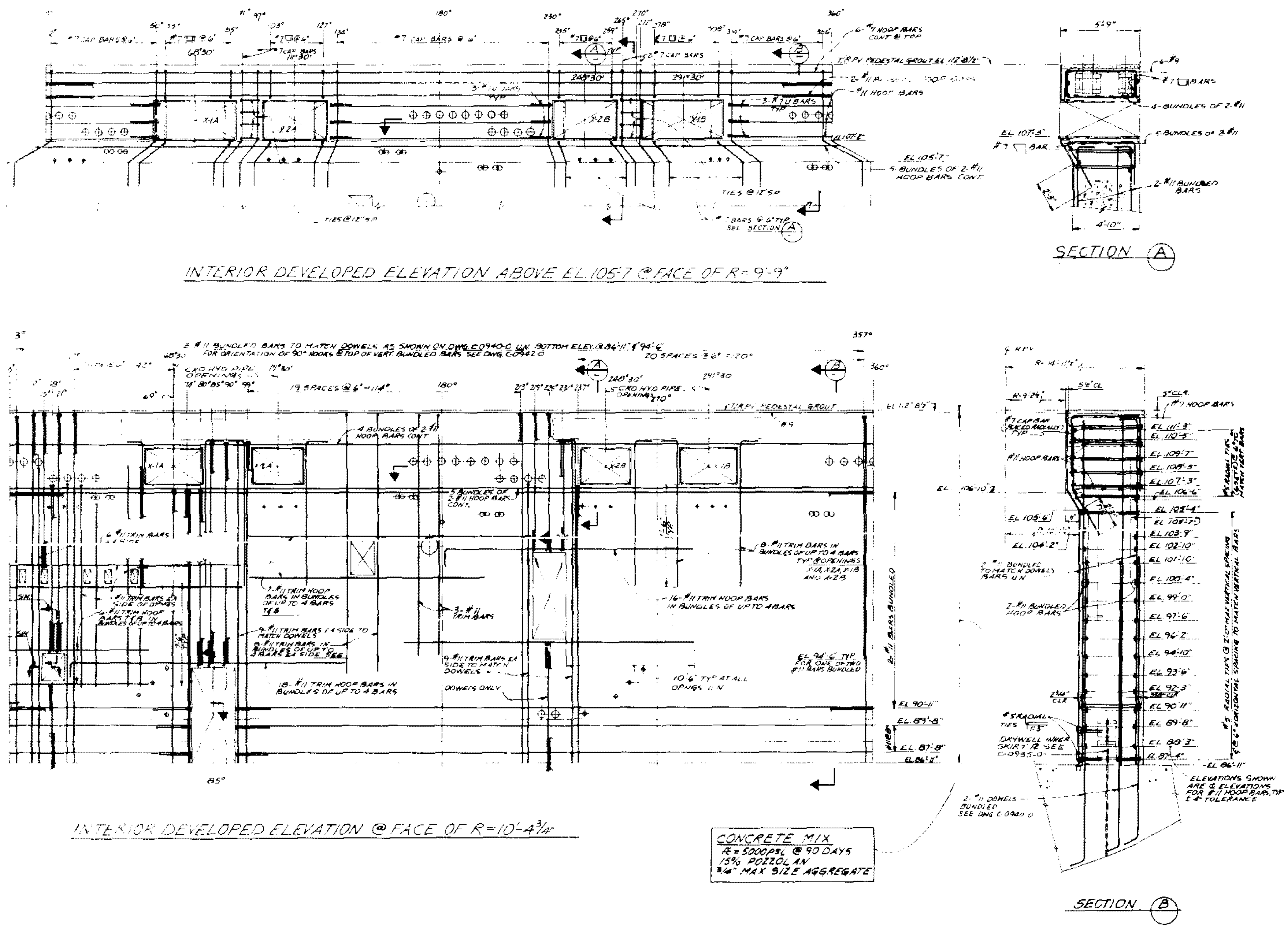
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

REACTOR PRESSURE VESSEL AND BIOLOGICAL SHIELD WALL BASE CONNECTIONS

UPDATED FSAR

FIGURE 3.8-26



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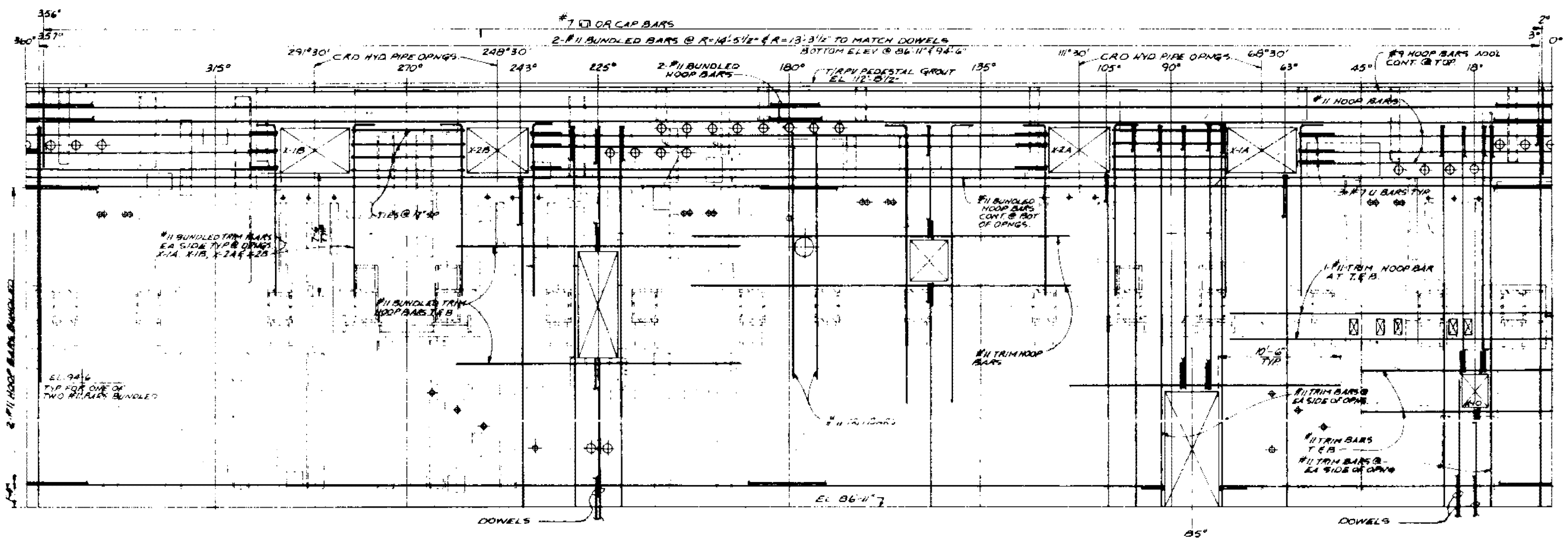
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

RPV PEDESTAL
REINFORCING

UPDATED FSAR

FIGURE 3.8-27



EXTERIOR DEVELOPED ELEVATION @ FACE OF R=14'-5½"

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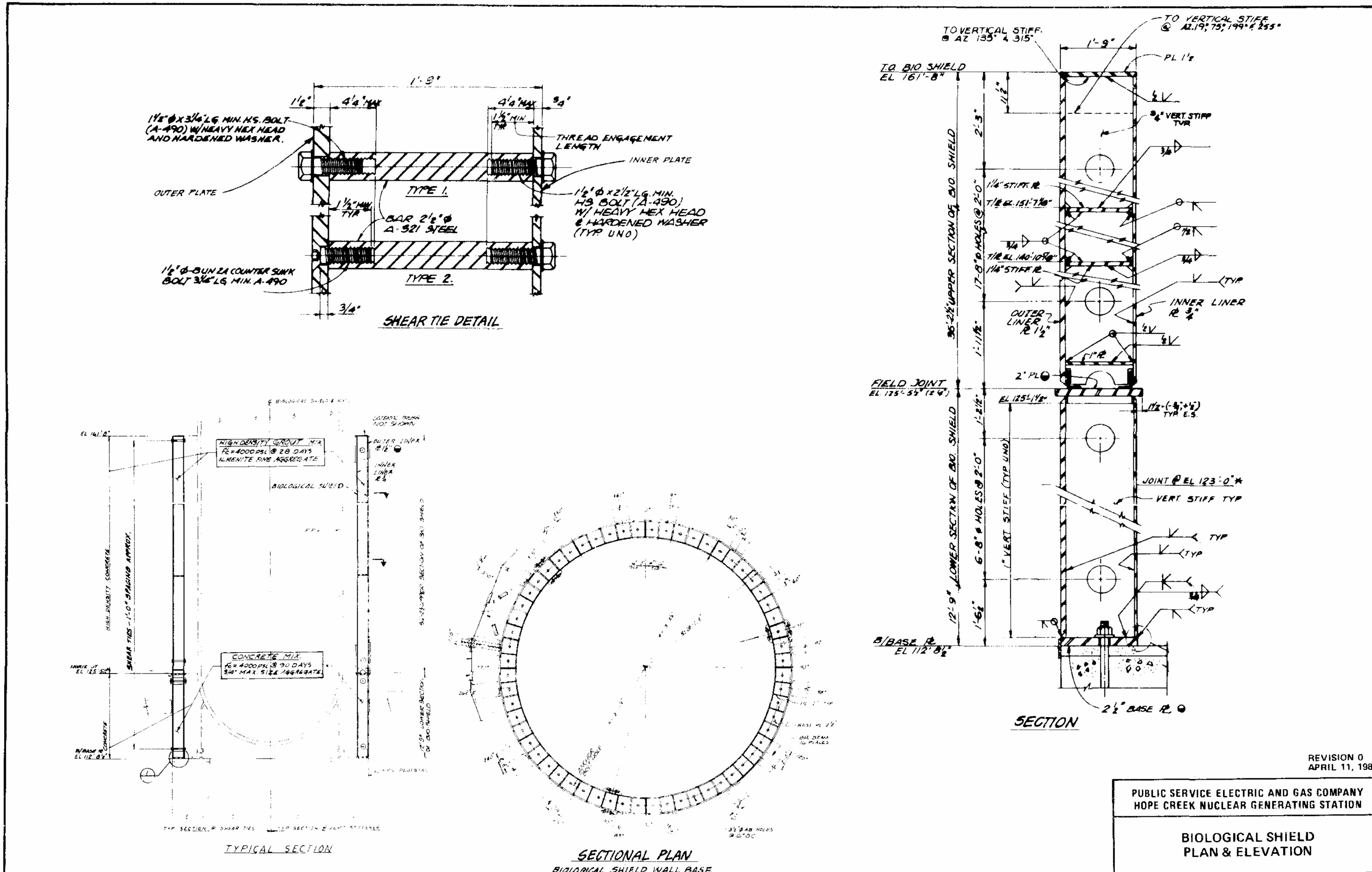
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

RPV PEDESTAL
REINFORCING

UPDATED FSAR

FIGURE 3.8-28



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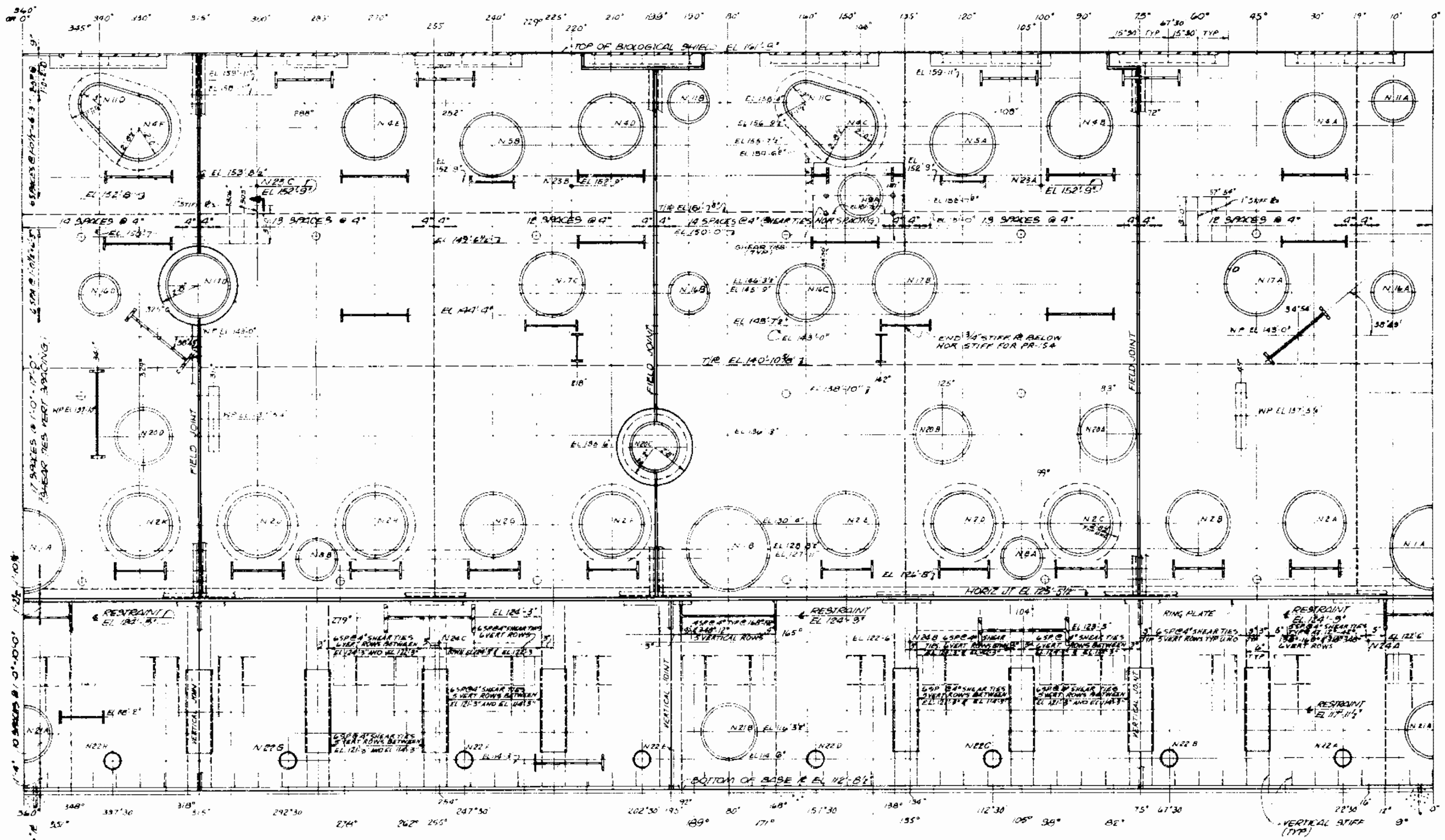
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

BIOLOGICAL SHIELD
PLAN & ELEVATION

UPDATED FSAR

FIGURE 3.8-29



EXTERIOR DEVELOPED ELEVATION
LOOKING FROM OUTSIDE

PENETRATION SCHEDULE

PENETRATION NUMBER	℄ ELEVATION	SLEEVE REQD	SLEEVE TYPE *	SLEEVE ID (IN)	SLEEVE THICKNESS	FUNCTION
N1A & N1B	128'-8½"	2	I	3'-4"	2½"	RECIRCULATION OUTLET
N2A THRU N2H, N2J & N2K	130'-4"	10	I	4'-0"	2½"	RECIRCULATION INLET
N4A, N4B, N4D & N4E	156'-9½"	4	I	4'-3"	2"	FEEDWATER
N4C & N11C	AS SHOWN	1	II	AS SHOWN	2½"	FEEDWATER & INSTRUMENTATION
N4F & N11D	AS SHOWN	1	II	AS SHOWN	2½"	FEEDWATER & INSTRUMENTATION
N5A & N5B	155'-7½"	2	I	4'-0"	2"	CORE SPRAY
N8A & N8B	127'-11"	2	I	2'-6"	2"	JET PUMP INSTRUMENTATION
N20C	135'-6"	1	I	3'-0"	2"	CORE BELT INSPECTION
N11A & N11B	158'-4"	2	I	2'-6"	2"	INSTRUMENTATION
N16A, N16B & N16D	145'-9"	3		2'-6"		INSTRUMENTATION
N16C	145'-9"	1		3'-6"		INSTRUMENTATION
N17A THRU N17D	146'-3½"	4		4'-0"		LPCI INLET
N20A, N20B & N20D	146'-5"	3		3'-6"		CORE BELT INSPECTION
N21A & N21B	118'-3½"	2	I	3'-6"	2"	SKIRT INSPECTION
N22A THRU N22H	114'-6"	8	III	14" OD	SCH 80	VENTILATION DUCT
N23A THRU N23C	152'-9"	3	IV	5" NOM DIA	SCH 80	ELECT THERMOCOUPLE CABLES
N24A THRU N24C	122'-6"	3	IV	5" NOM DIA	SCH 80	ELECT THERMOCOUPLE CABLES
N8A	152'-7½"	1	I	2'-6"	2"	CRD NOZZLE INSPECTION

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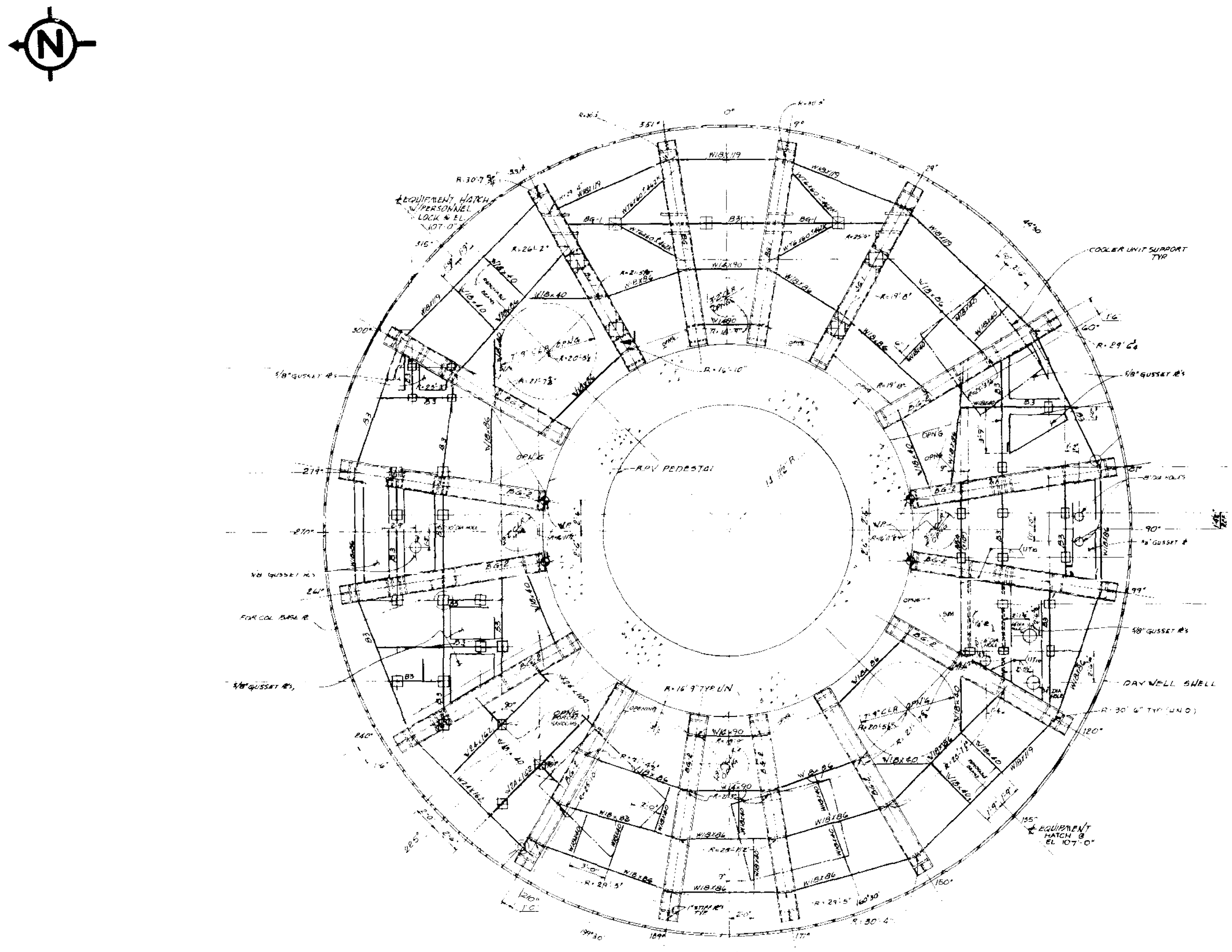
HOPE CREEK NUCLEAR GENERATING STATION

BIOLOGICAL SHIELD WALL

UPDATED FSAR

FIGURE 3.8-30



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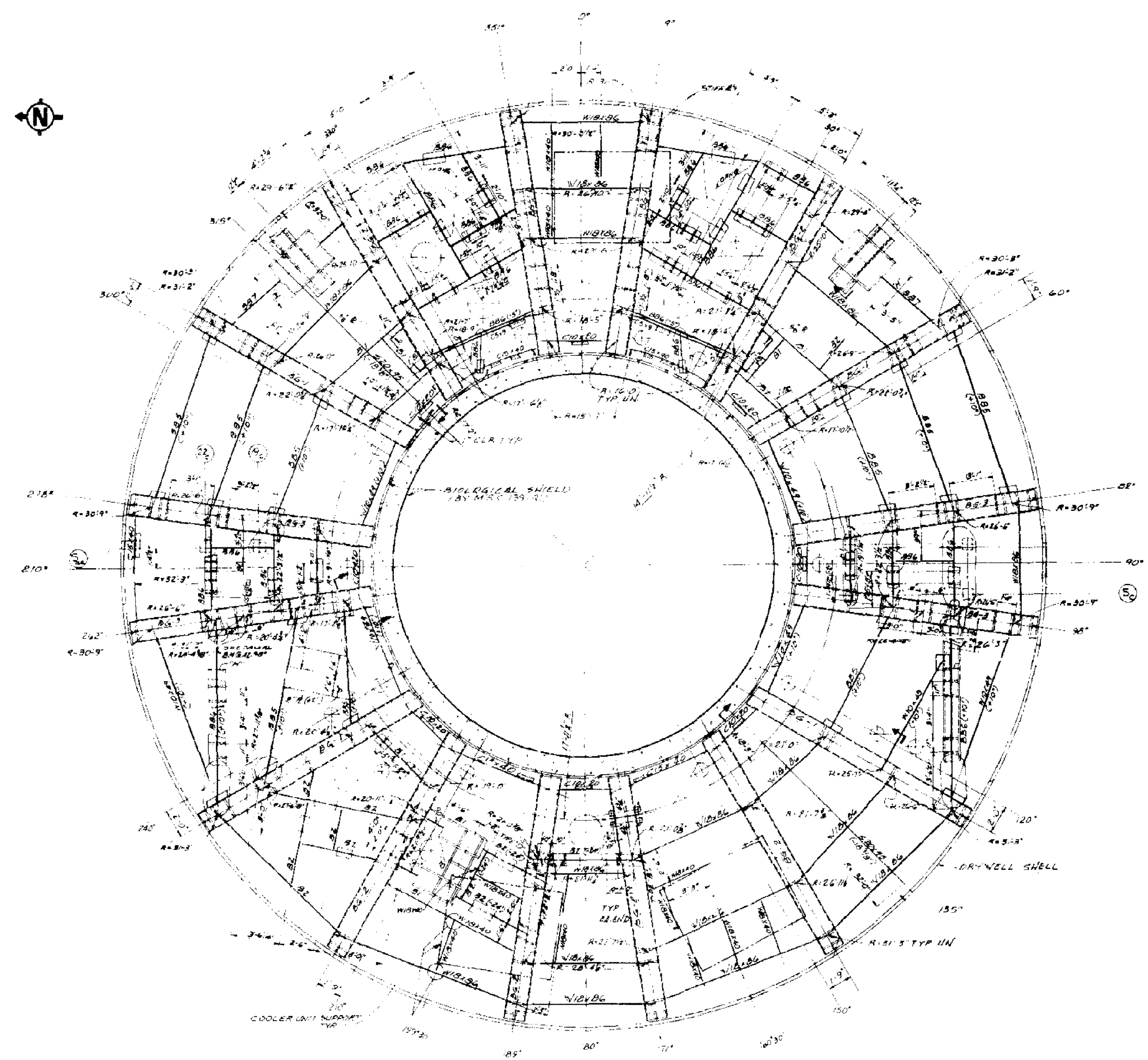
DRYWELL PLATFORM
AT EL. 100'-2"

UPDATED FSAR

FIGURE 3.8-31



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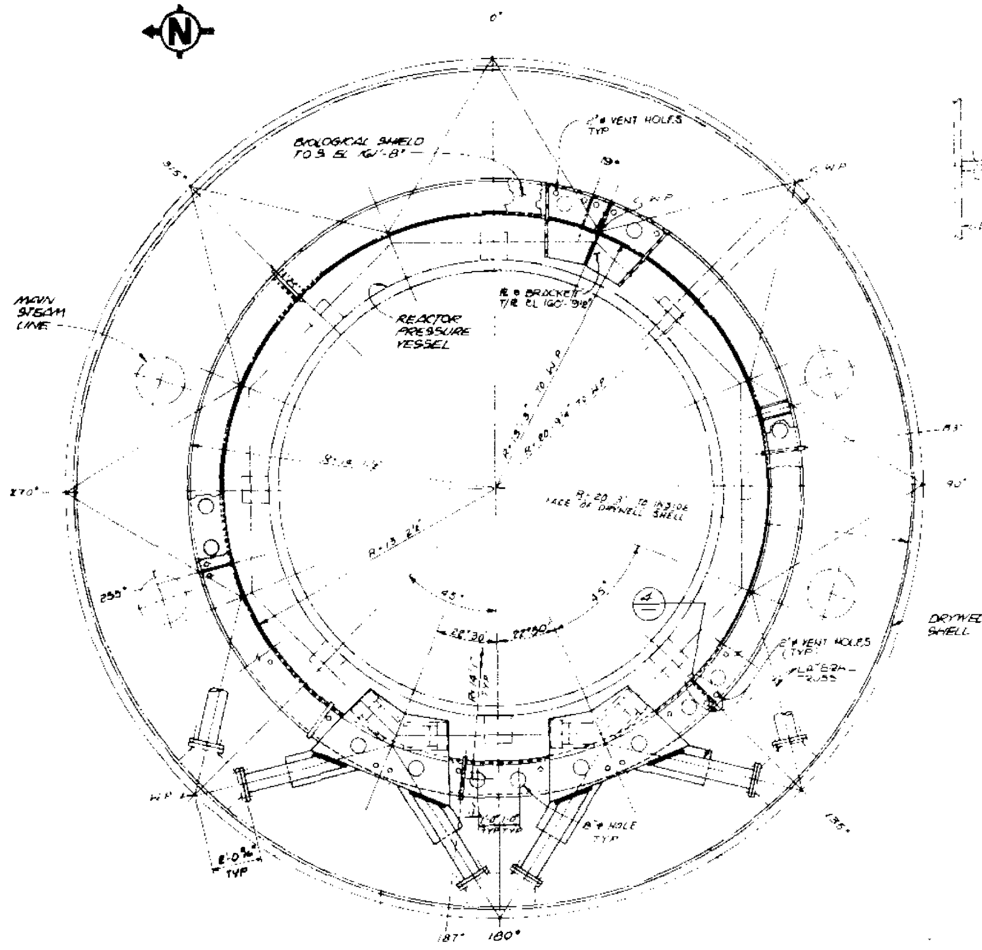
HOPE CREEK NUCLEAR GENERATING STATION

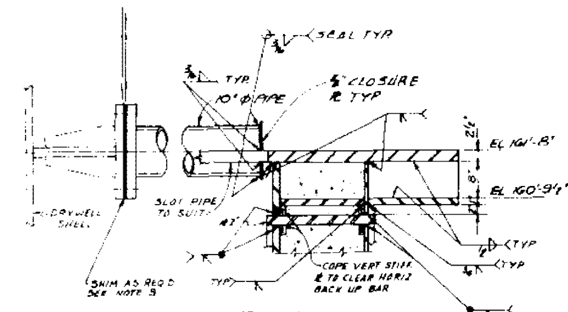
DRYWELL PLATFORM
AT EL. 121'-7½"

UPDATED FSAR

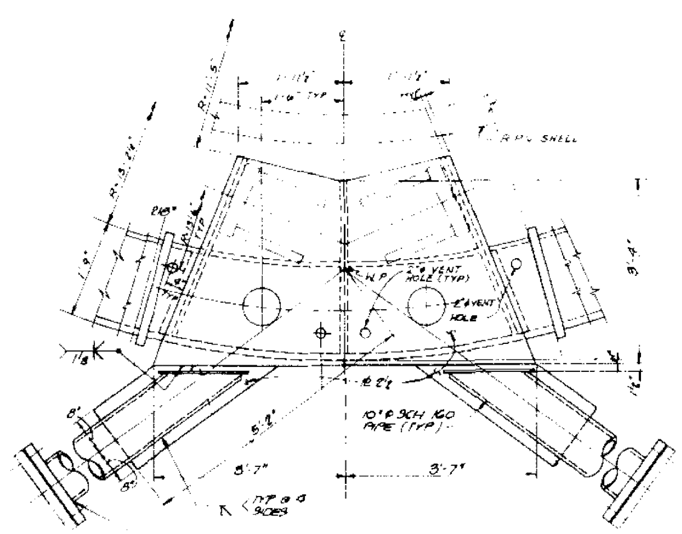
FIGURE 3.8-32



PLAN AT EL 161'-8"



SECTION



DETAIL

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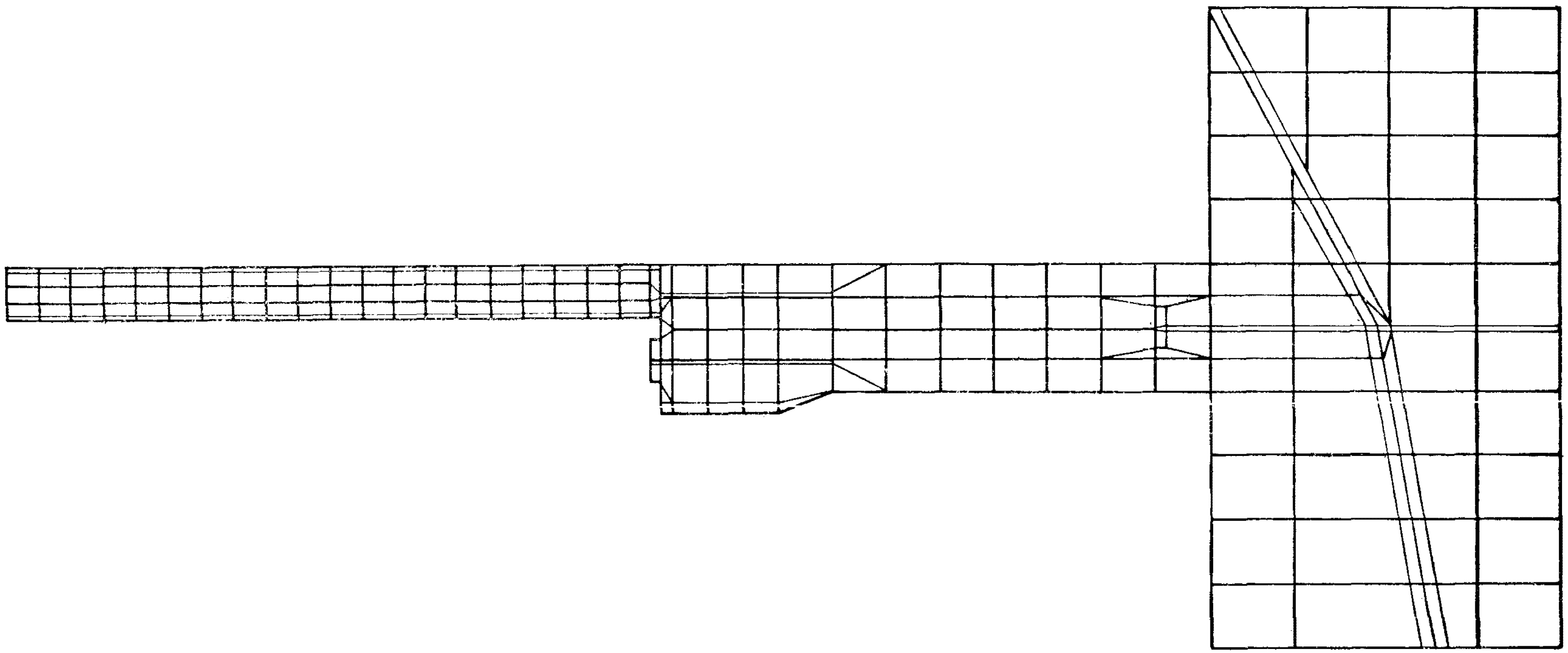
HOPE CREEK NUCLEAR GENERATING STATION

LATERAL TRUSS

UPDATED FSAR

FIGURE 3.8-33



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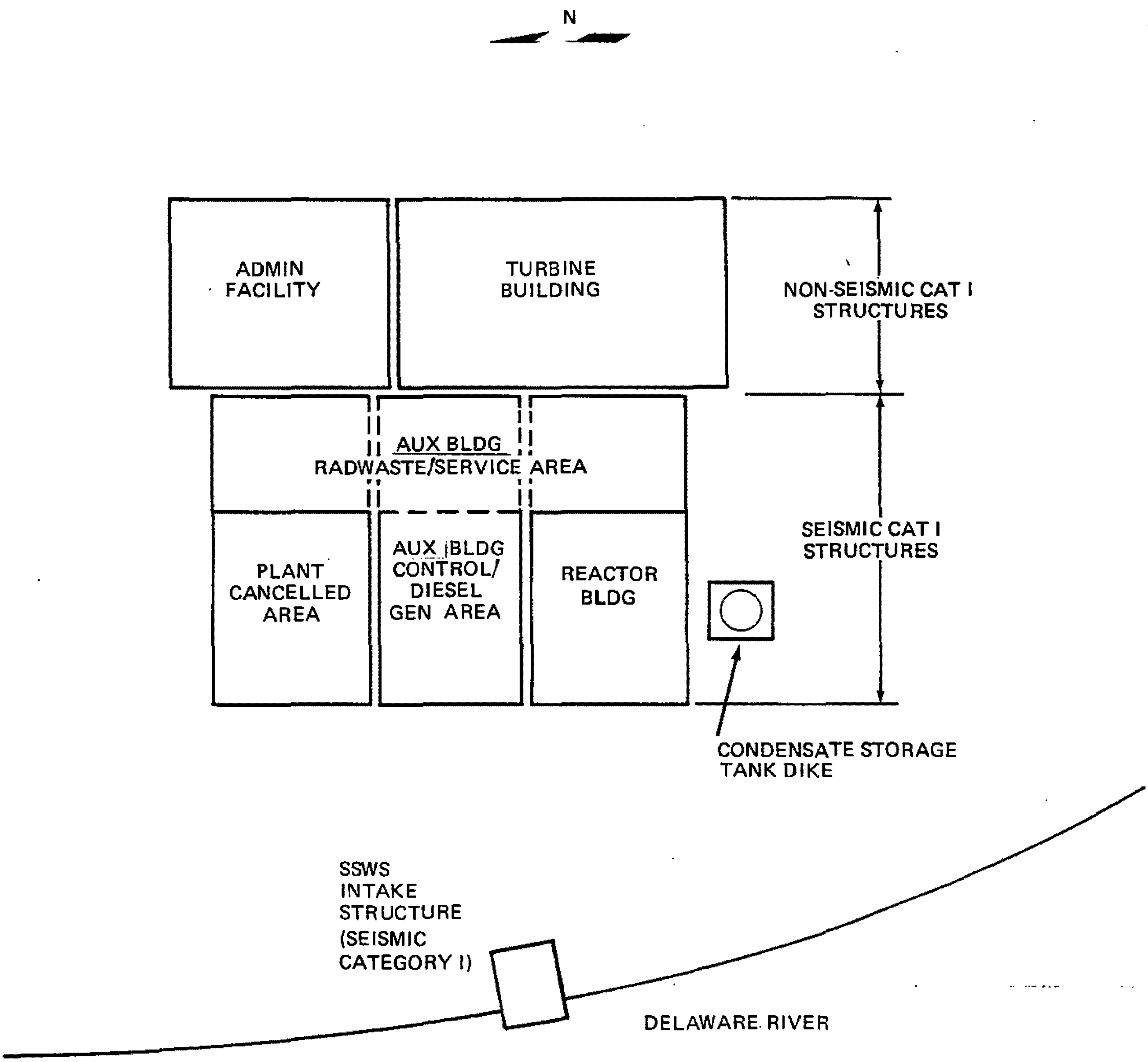
HOPE CREEK NUCLEAR GENERATING STATION

FINITE-ELEMENT MODEL
OF RPV PEDESTAL AND
BIOLOGICAL SHIELD

UPDATED FSAR

FIGURE 3.8-34



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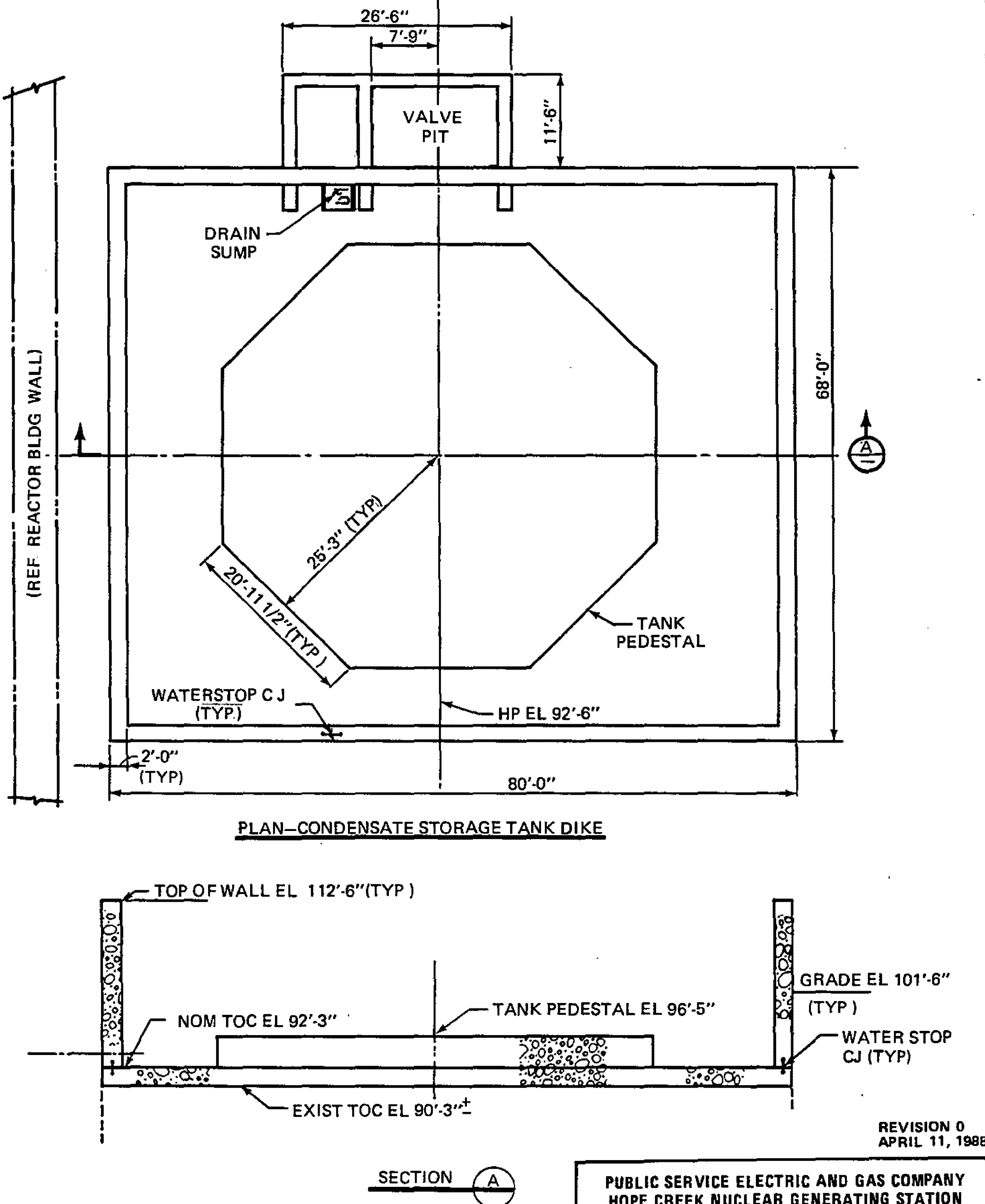
OTHER SEISMIC CATEGORY I STRUCTURES AND RELATED NON-SEISMIC CATEGORY I STRUCTURES

UPDATED FSAR

FIGURE 3.8-35



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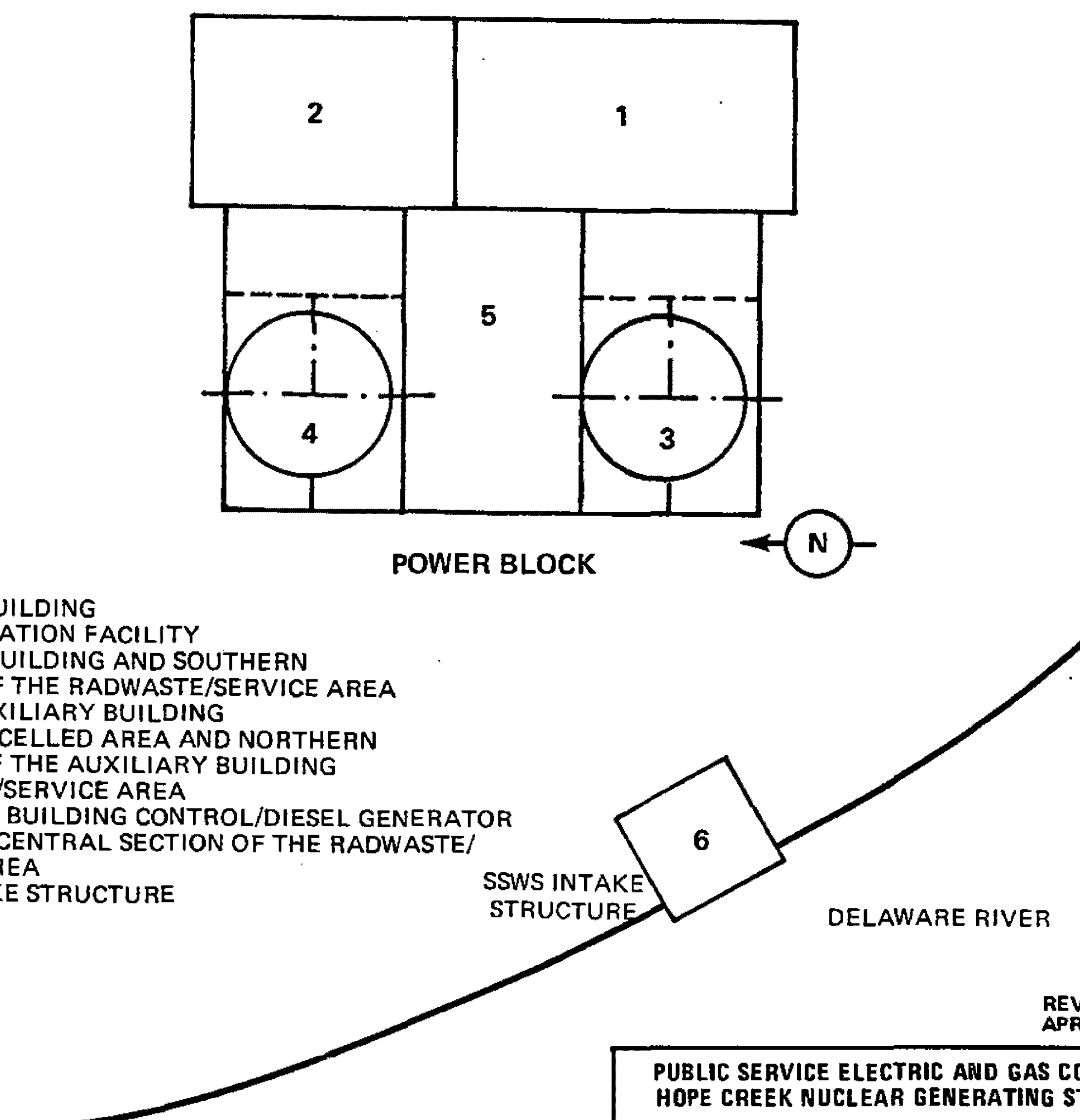
HOPE CREEK NUCLEAR GENERATING STATION

CONDENSATE STORAGE TANK DIKE FOUNDATION DETAILS AND SECTIONS

UPDATED FSAR

FIGURE 3.8-36



1. TURBINE BUILDING
2. ADMINISTRATION FACILITY
3. REACTOR BUILDING AND SOUTHERN SECTION OF THE RADWASTE/SERVICE AREA OF THE AUXILIARY BUILDING
4. PLANT CANCELLED AREA AND NORTHERN SECTION OF THE AUXILIARY BUILDING RADWASTE/SERVICE AREA
5. AUXILIARY BUILDING CONTROL/DIESEL GENERATOR AREA AND CENTRAL SECTION OF THE RADWASTE/ SERVICE AREA
6. SSWS INTAKE STRUCTURE

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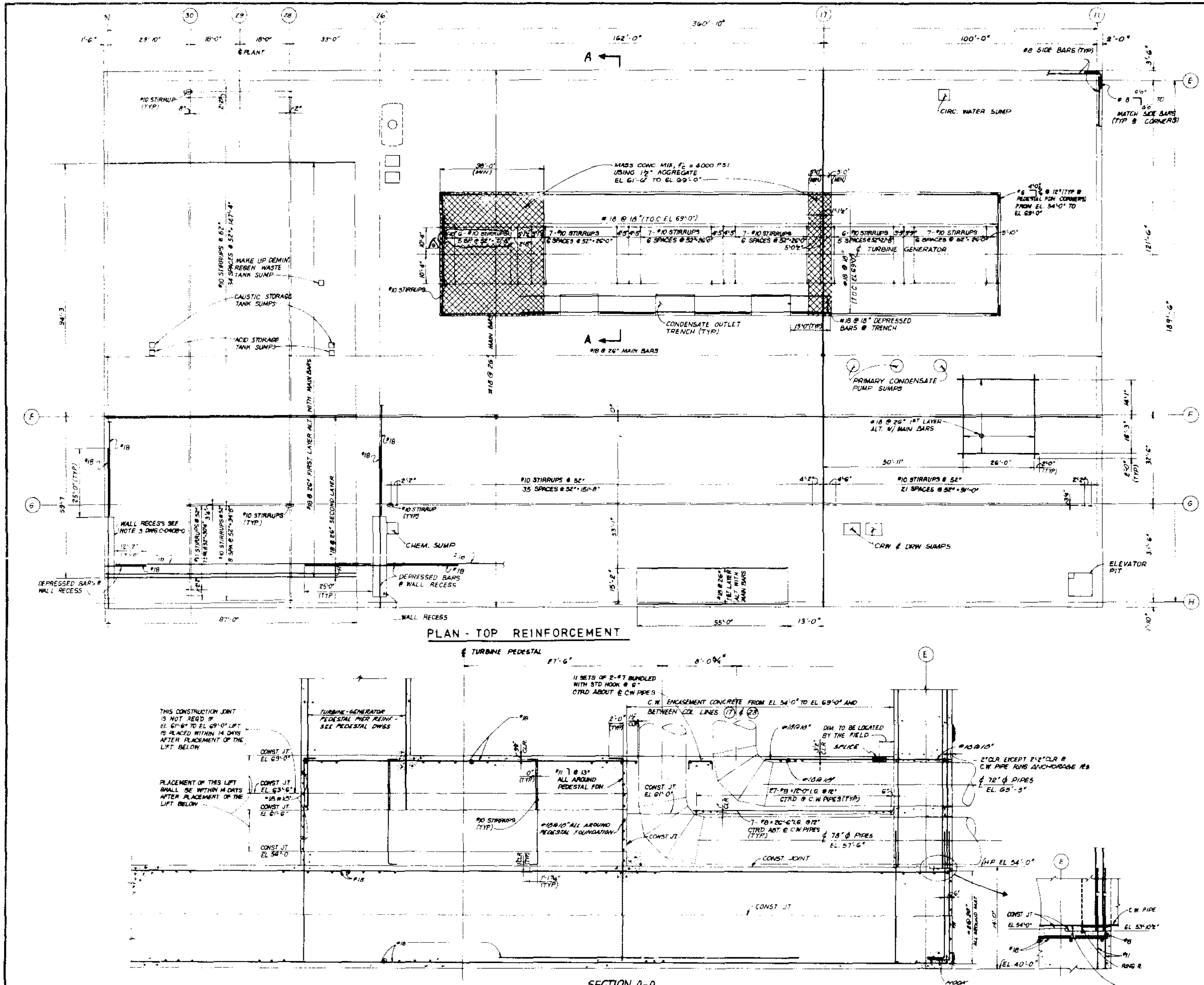
KEY PLAN – FOUNDATION MATS

UPDATED FSAR

FIGURE 3.8-37



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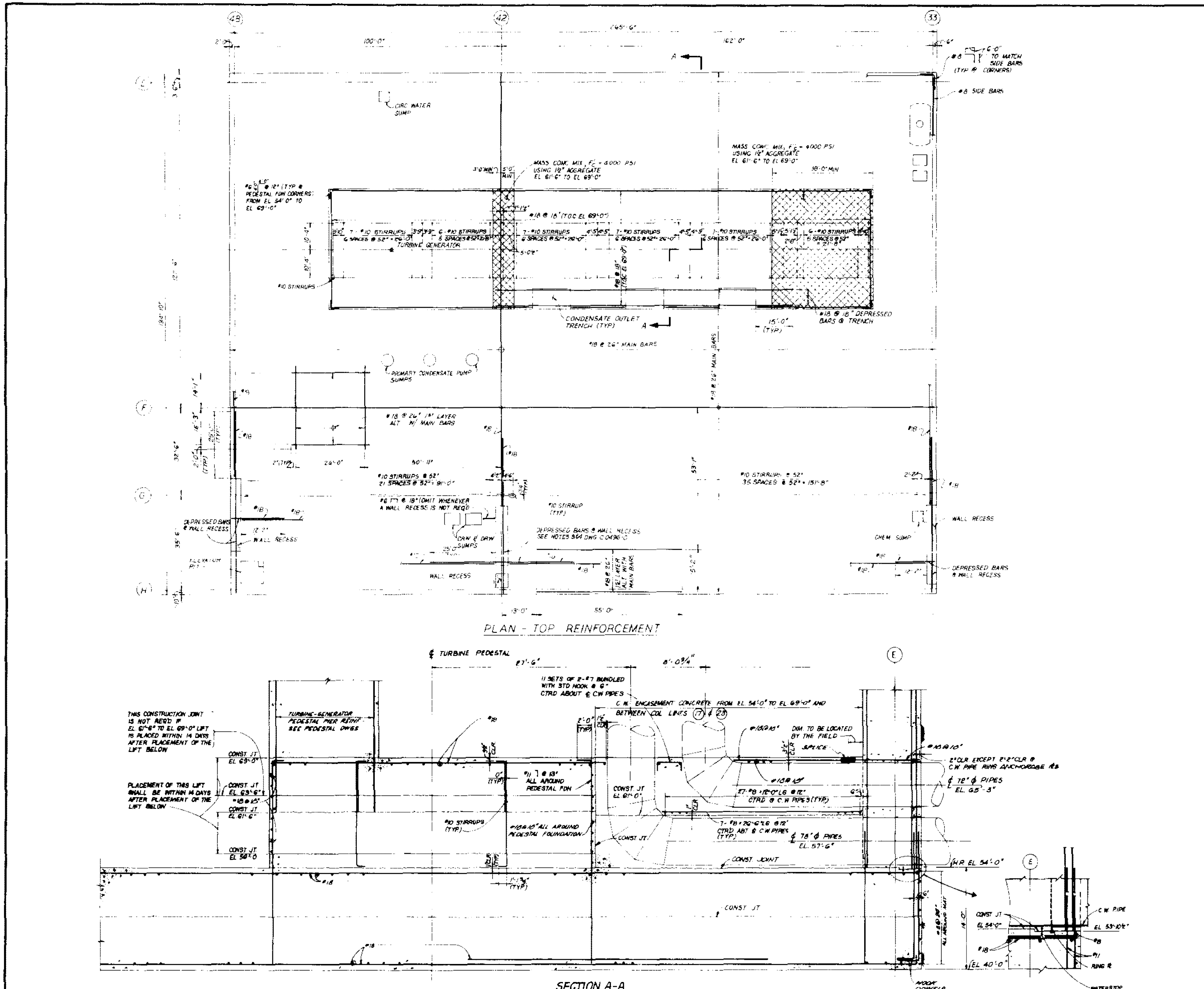
HOPE CREEK NUCLEAR GENERATING STATION

FOUNDATIONS
MAT 1

UPDATED FSAR

FIGURE 3.8-38



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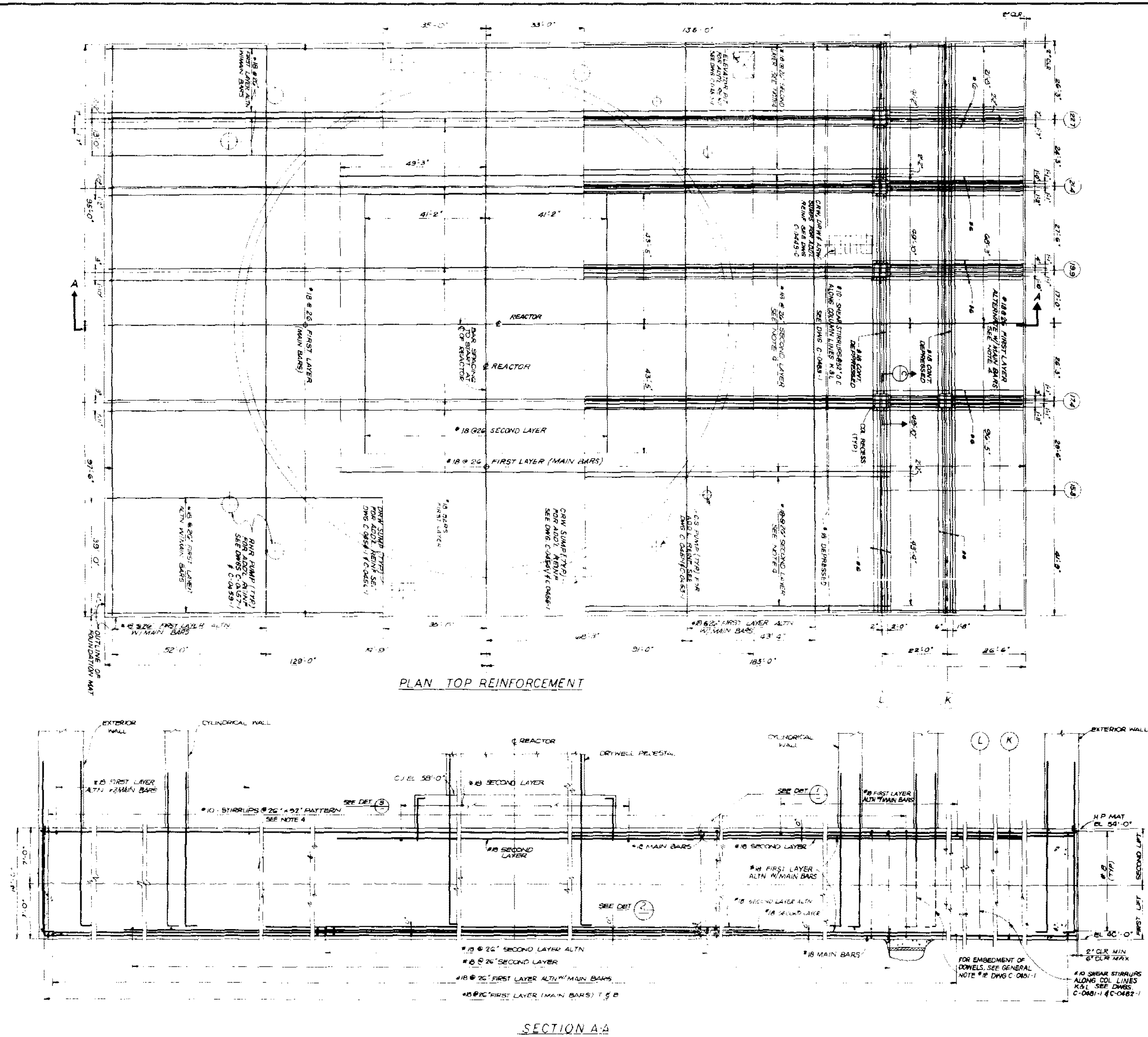
HOPE CREEK NUCLEAR GENERATING STATION

FOUNDATIONS
MAT 2

UPDATED FSAR

FIGURE 3.8-39



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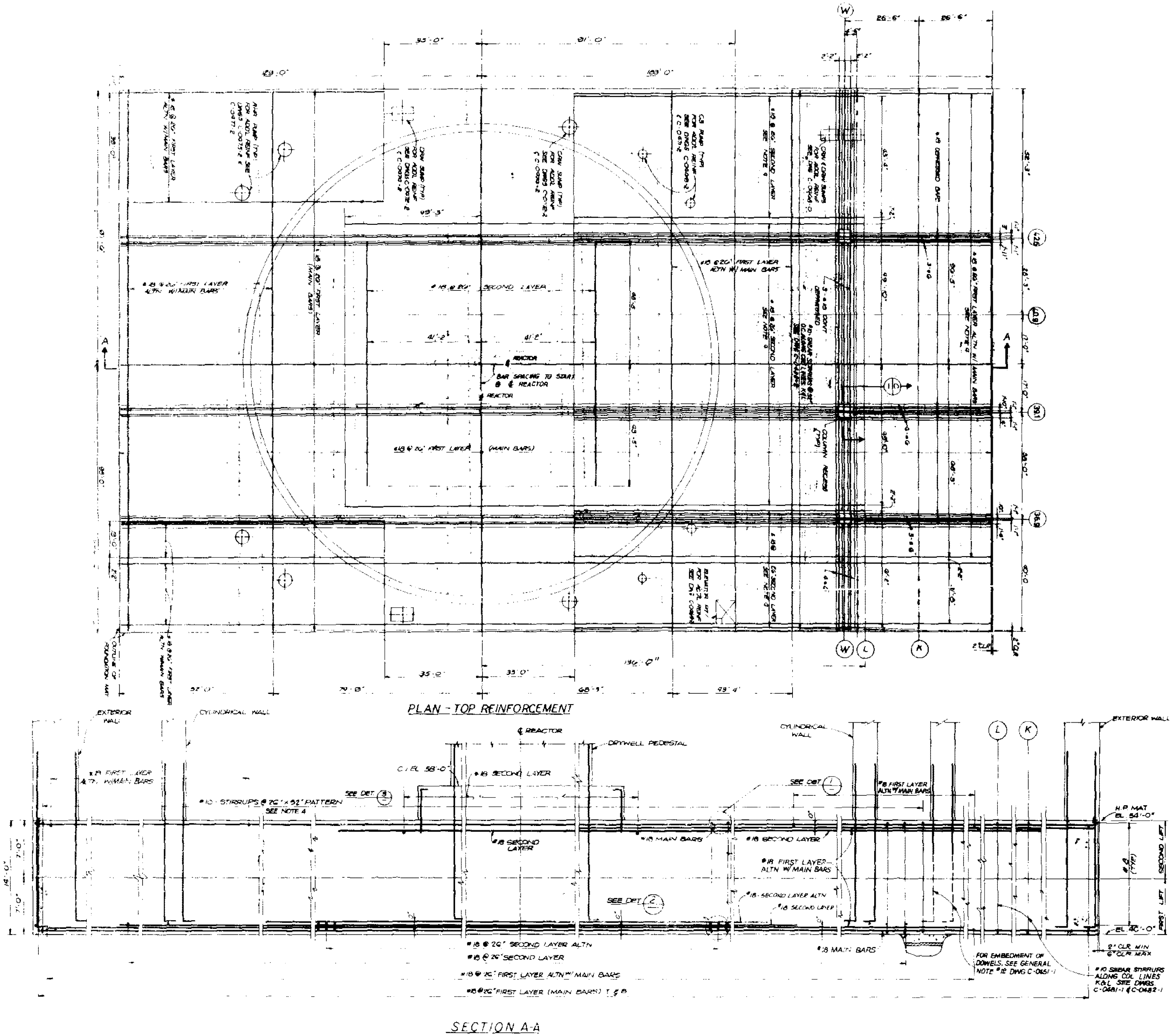
HOPE CREEK NUCLEAR GENERATING STATION

FOUNDATIONS
MAT 3

UPDATED FSAR

FIGURE 3.8-40



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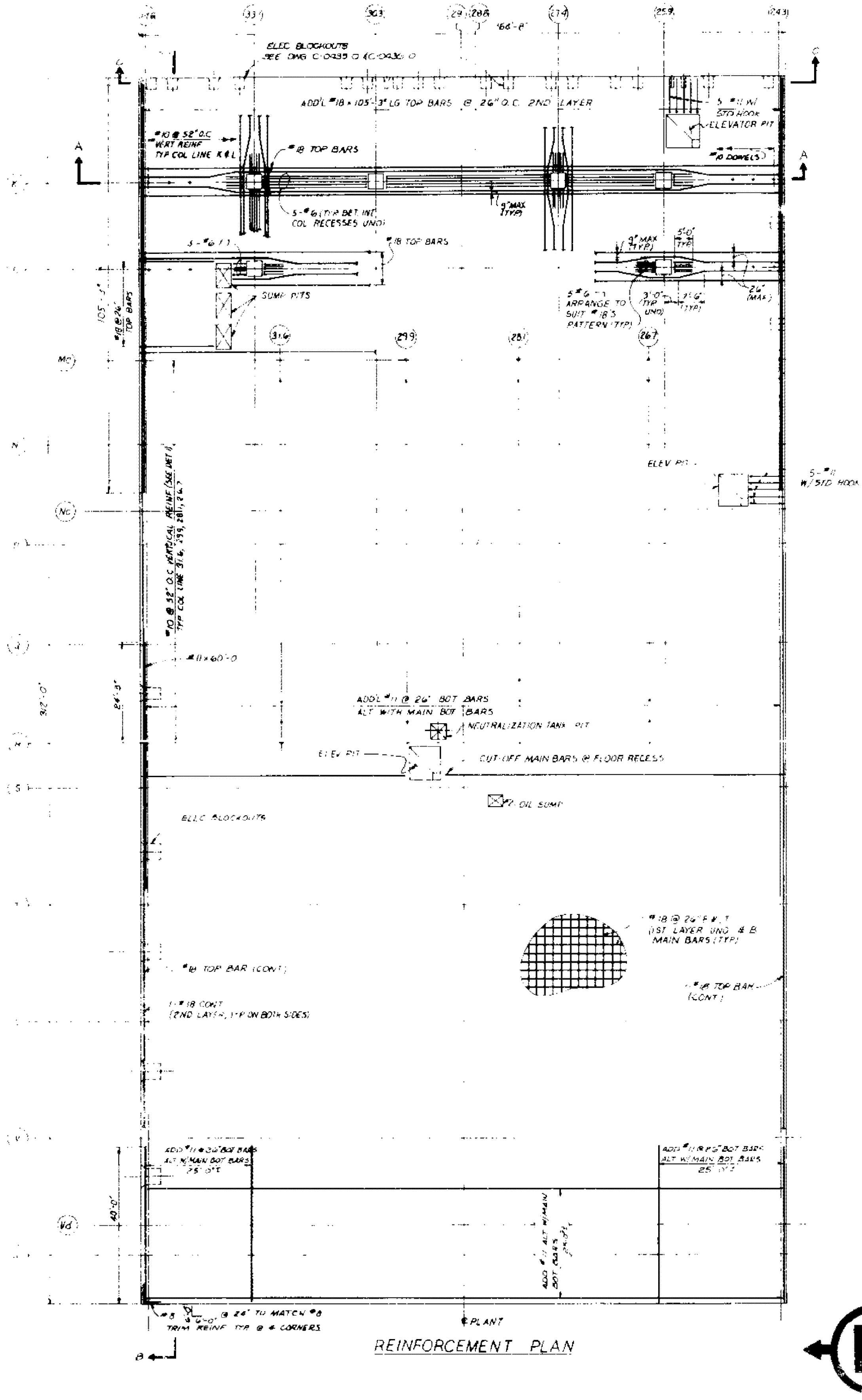
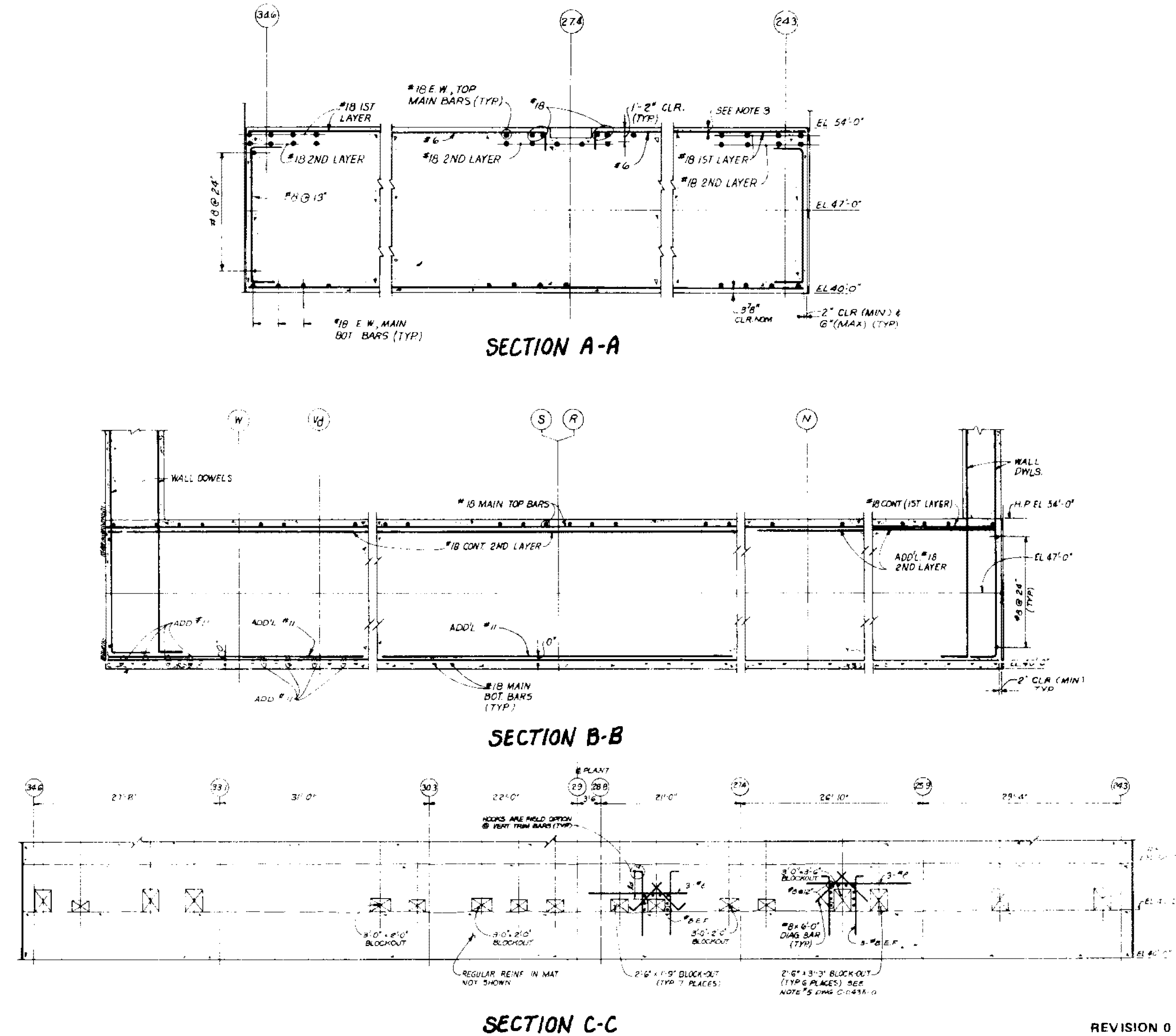
HOPE CREEK NUCLEAR GENERATING STATION

FOUNDATIONS
MAT 4

UPDATED FSAR

FIGURE 3.8-41





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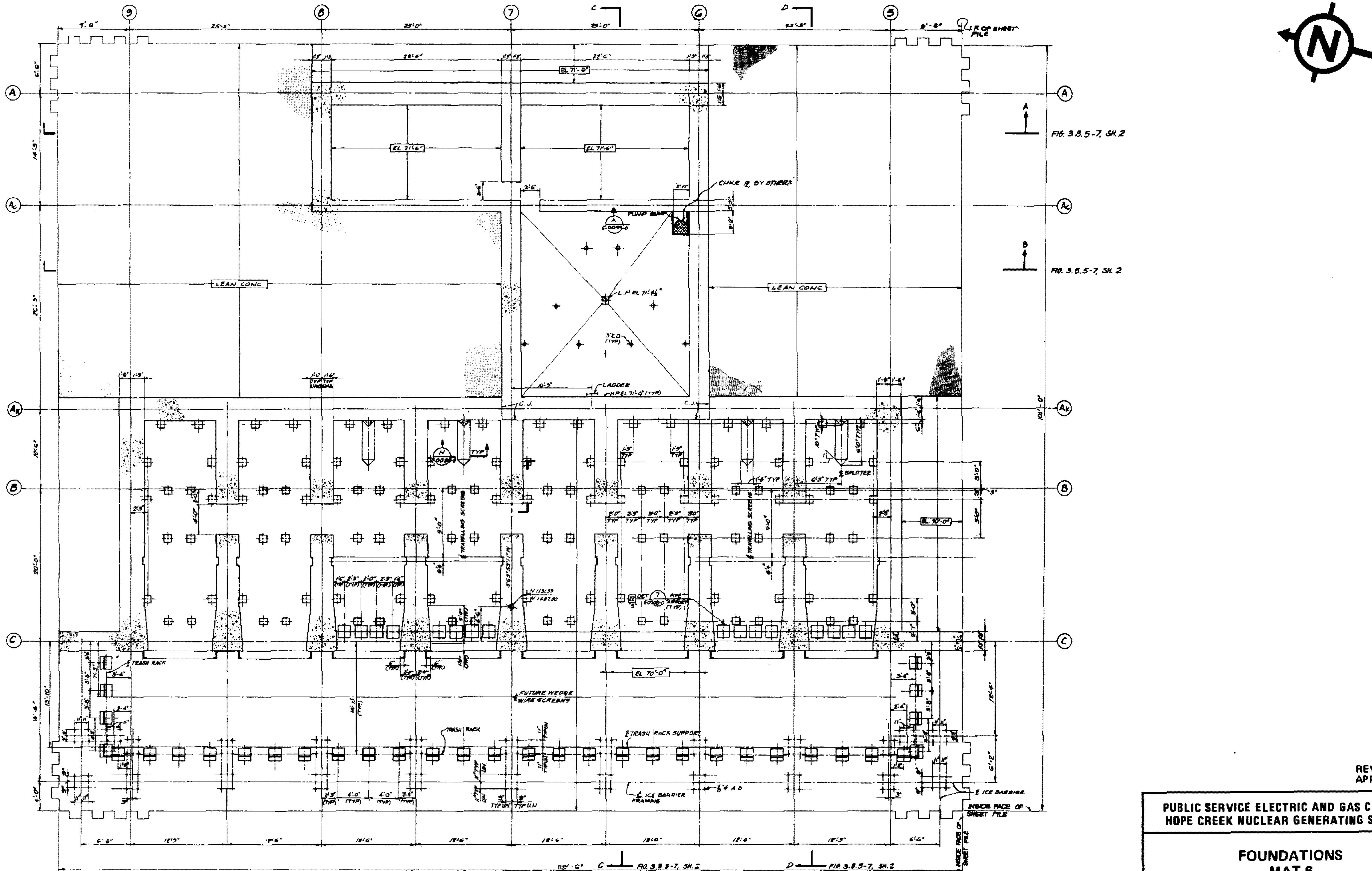
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

FOUNDATIONS
MAT 5

UPDATED FSAR

FIGURE 3.8-42



FOUNDATION MAT PLAN
AT ELEVATION 70'-0" U.N.O.

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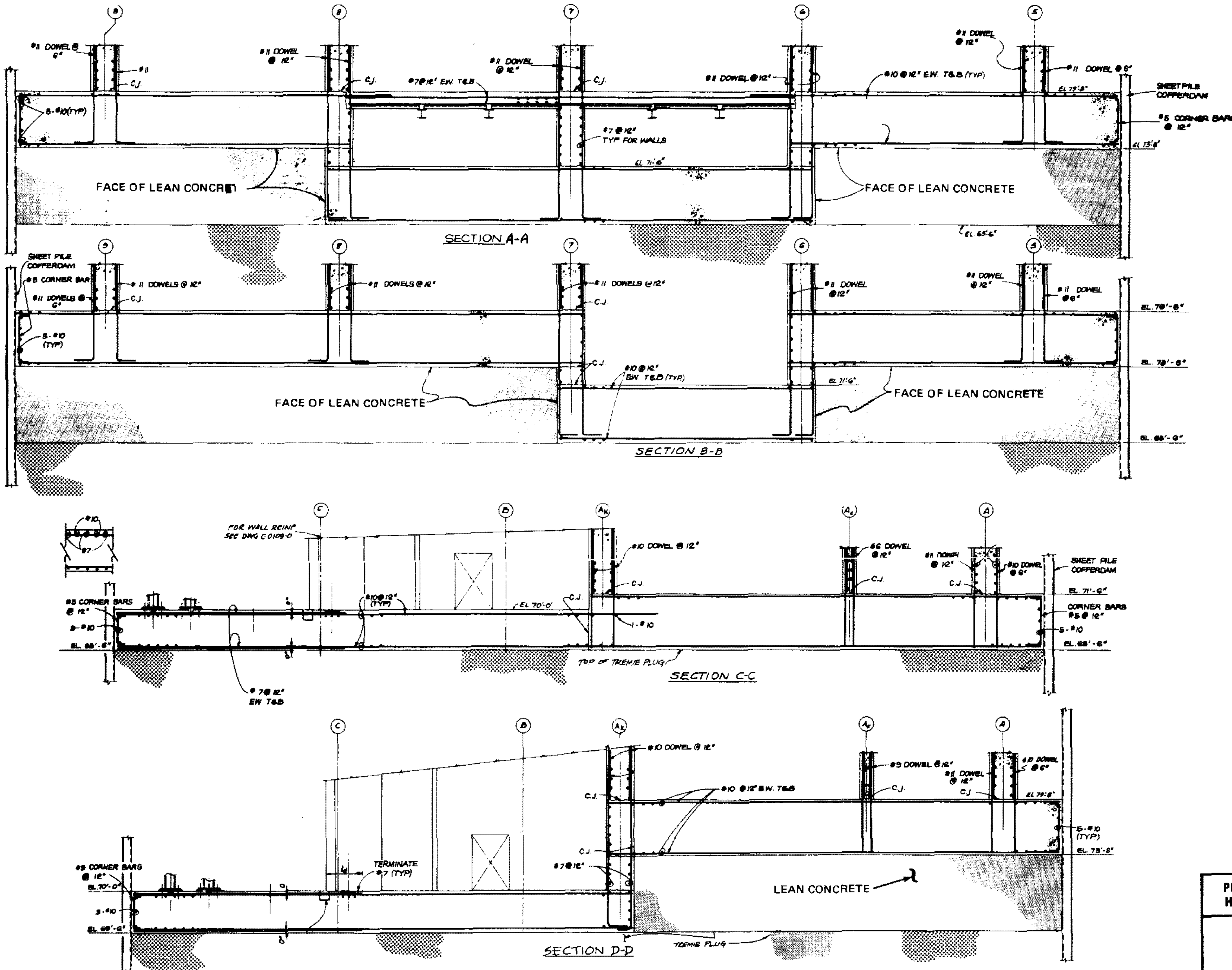
HOPE CREEK NUCLEAR GENERATING STATION

FOUNDATIONS
MAT 6

UPDATED FSAR

Sheet 1 of 2
FIGURE 3.8-43



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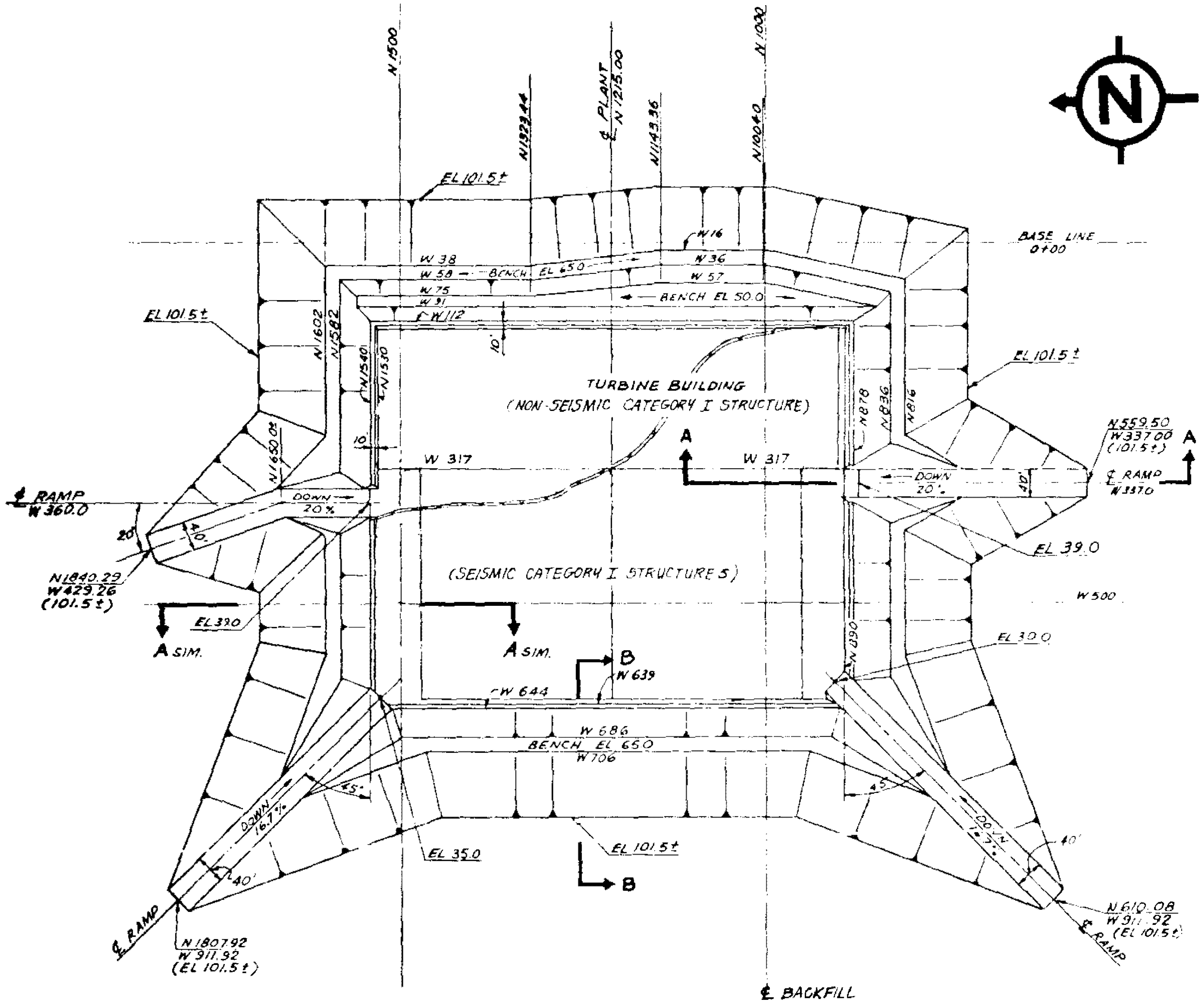
HOPE CREEK NUCLEAR GENERATING STATION

FOUNDATIONS
MAT 6

UPDATED FSAR

Sheet 2 of 2
FIGURE 3.8-43



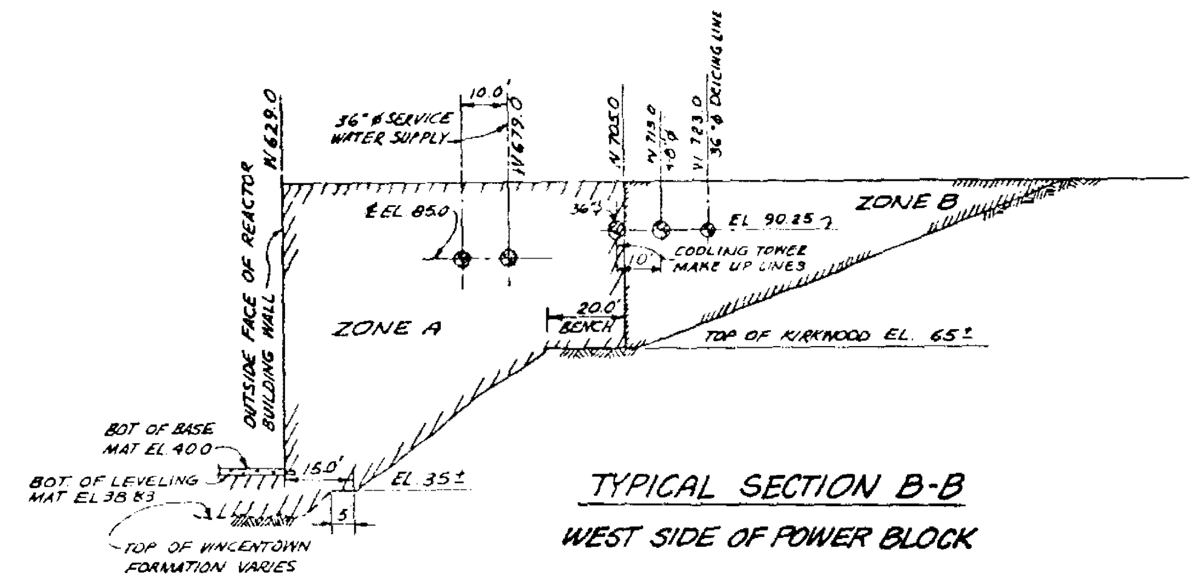
EXCAVATION AND BACKFILL PLAN

MATERIAL AND COMPACTION SCHEDULE		
ZONE	DEGREE * OF COMPACTION	MATERIAL DESCRIPTION
A	98 % **	BACKFILL MATERIAL IS FROM APPROVED BORROW SOURCE
B	95 % ***	
C, 1	90 % ***	
2	85 % ***	BACKFILL IS FILL MATERIAL OBTAINED FROM EXCAVATION AT PLANT SITE

* DEGREE OF COMPACTION INDICATED IS EXPRESSED AS PERCENTAGE OF THE MAXIMUM DRY DENSITY DETERMINED BY ASTM D-1557.

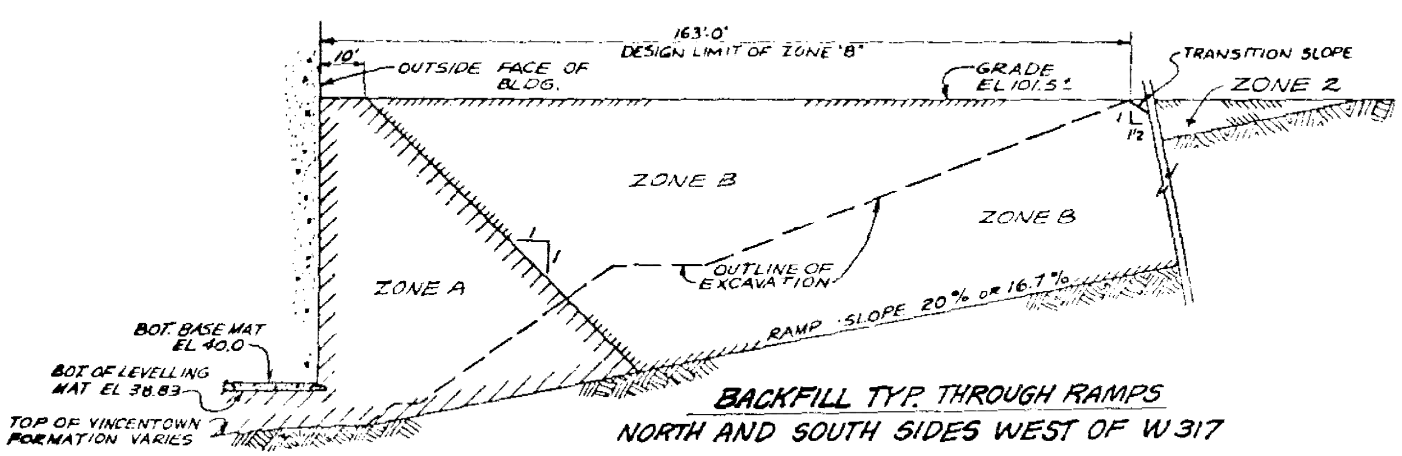
** DEGREE OF COMPACTION IS EXPRESSED AS **AN AVERAGE WITH A MINIMUM OF 95%.**

*** DEGREE OF COMPACTION IS EXPRESSED **A MINIMUM.**



TYPICAL SECTION B-B

WEST SIDE OF POWER BLOCK



BACKFILL TYP. THROUGH RAMPS

NORTH AND SOUTH SIDES WEST OF W 317

SECTION A-A

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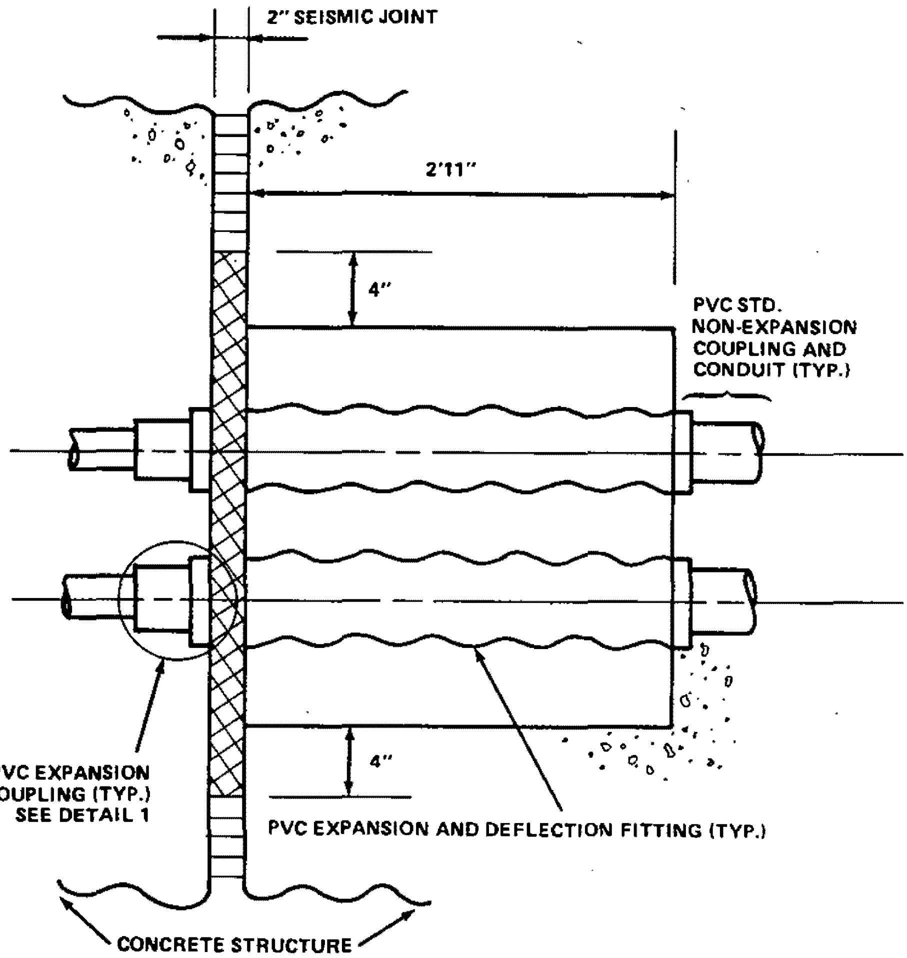
HOPE CREEK NUCLEAR GENERATING STATION

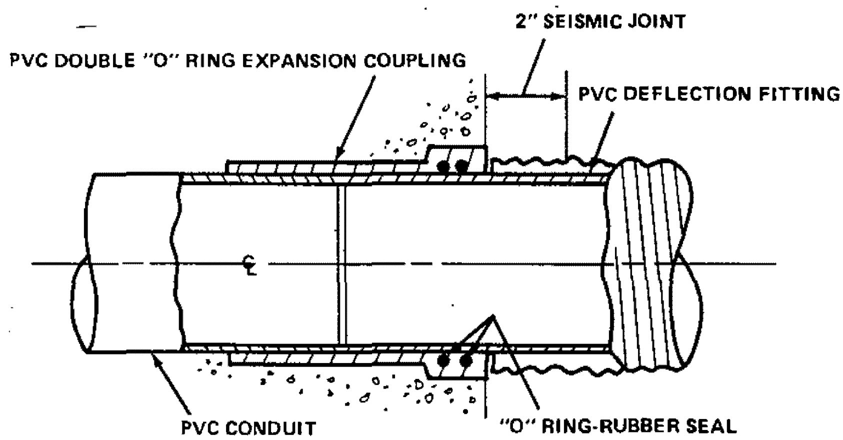
EXCAVATION AND BACKFILL AREAS

UPDATED FSAR

FIGURE 3.8-44



TYPICAL SECTION



DETAIL I

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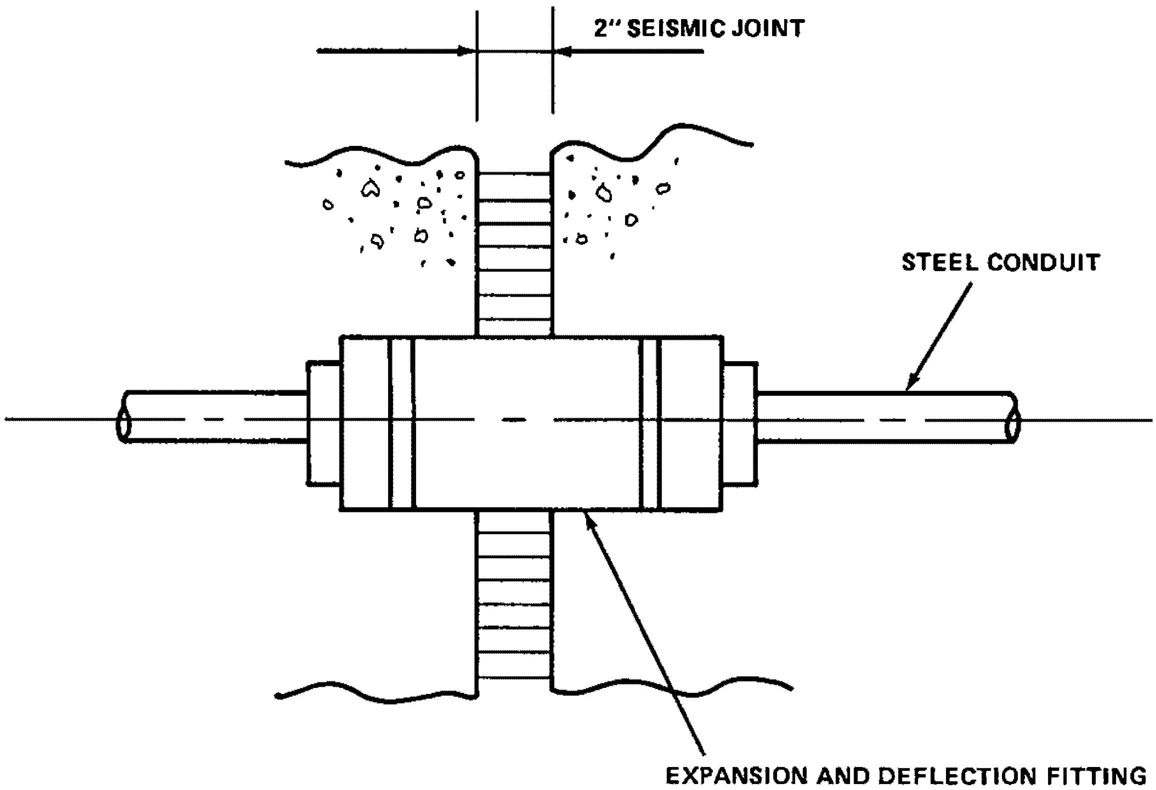
HOPE CREEK NUCLEAR GENERATING STATION

TYPICAL DETAIL FOR EMBEDDED
PVC CONDUIT CROSSING SEISMIC JOINT
FOUNDATION MAT ONLY)

UPDATED FSAR

FIGURE 3.8-45



TYPICAL SECTION

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PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

TYPICAL DETAIL
FOR EMBEDDED STEEL CONDUIT
CROSSING SEISMIC JOINT

UPDATED FSAR

FIGURE 3.8-46

3.9 MECHANICAL SYSTEMS AND COMPONENTS

3.9.1 Special Topics for Mechanical Components

3.9.1.1 Design Transients

This section describes the transients that are used in the design of ASME Boiler and Pressure Vessel (B&PV) Code, Section III Class 1 components, the core support structures, the reactor internals, and the hydraulic control units. The number of cycles or events for each transient is included. These transients are included in the design specifications and/or stress reports for the components. Transients selected for fatigue evaluation include conservative estimates of magnitude and frequency of temperature and pressure conditions resulting from the design transients. 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 B&PV Code, as applicable.

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 drives (CRD) are as follows:

Transient	Category	Cycles
Reactor startup/shutdown	Normal/Upset	120
Vessel pressure tests	Normal/Upset	130
Vessel overpressure	Normal/Upset	10
Scram test plus startup scrams	Normal/Upset	300
Operational scrams	Normal/Upset	300
Jog cycles	Normal/Upset	30,000
Shim/drive cycles	Normal/Upset	1000

In addition to the above cycles, the following cycles were considered in the design of the CRDs:

Transient	Category	Cycles
Scram with inoperative buffer	Normal/Upset	10
Scram with stuck control blade	Normal/Upset	1
Operating basis earthquake	Upset	10
Safe shutdown earthquake	Faulted	1

The frequency of occurrence of the operating basis earthquake (OBE) indicates the emergency category. However, for conservatism, the OBE was analyzed as an upset condition.

All ASME B&PV, Section III, Class 1 components of the CRD have been analyzed according to Section III of the ASME B&PV Code. The capability of the CRDs to withstand other emergency and faulted conditions is verified by test rather than by analysis.

3.9.1.1.2 Control Rod Drive Housing and In-core Instrument Housing Transients

The number of transients, their cycles, and classifications, as considered in the design and fatigue analysis of the CRD housing and in-core instrument housing, are as follows:

Transient	Category	Cycles
Normal startup/shutdown	Normal/upset	120
Vessel pressure/overpressure tests	Normal/upset	130
Interruption of feedwater flow	Normal/upset	80
Scram	Normal/upset	200
OBE(1)	Normal/upset	10
Safe shutdown earthquake (SSE)(2)	Emergency	1
CRD Housing Only		
Stuck rod scram	Normal/upset	1
Scram without buffer	Normal/upset	10

(1) The frequency of an OBE indicates an emergency category. However, for conservatism, this OBE condition is analyzed as an upset.

(2) SSE is a faulted condition. However, in the stress analysis report, it is treated as emergency with lower stress limits.

3.9.1.1.3 Hydraulic Control Unit Transients

The transients used in the design and analysis of the hydraulic control unit (HCU) and its components are as follows:

Transient	Category	Cycles
Reactor startup and shutdown	Normal/upset	120
Scram tests	Normal/upset	300
Operational scrams	Normal/upset	300
Jog cycles	Normal/upset	30,000
Scram with stuck scram discharge valve	emergency	1
OBE	Normal/upset	10
SSE	Faulted	1

3.9.1.1.4 Core Support Structures and Reactor Internals Transients

The cycles listed in Table 3.9-1 are considered in the design and fatigue analysis for the core support structures and the reactor internals.

3.9.1.1.5 Main Steam System Transients

The following transients are considered in the stress analyses of the main steam piping:

Transient	Category	Cycles
Startup	normal	120

Transient	Category	Cycles
Loss of feedwater pumps, isolation valves closed	Upset	10
Scram	Upset	180
Shutdown	normal	111
Reactor overpressure delayed scram	Emergency	1
Single safety/relief valve (SRV) blowdown	Upset	8
Automatic blowdown	Emergency	1
Hydrotest	Test	130
OBE	Upset	50
Turbine main stop valve (MSV) closure	Upset	40
Safety/relief valve lift	Upset	5,000

3.9.1.1.6 Recirculation System Transients

The following transients are considered in the stress analyses of the recirculation piping:

Transient	Category	Cycles
Startup	Normal	120
Turbine roll and increase to power	Normal	120
Loss of feedwater heater	Upset	10
Partial feedwater heater bypass	Upset	70
Scrams	Upset	180
Shutdown	Normal	111
Loss of feedwater pumps, isolation valves closed	Upset	10
Reactor overpressure with delayed scram	Emergency	1
Single SRV blowdown	Upset	8
Automatic blowdown	Emergency	1
Hydrotest	Test	130
OBE	Normal/upset	50

3.9.1.1.7 Reactor Assembly Transients

The reactor assembly includes the reactor pressure vessel (RPV), support skirt, and shroud support, including legs, cylinder and plate. The cycles listed in Table 3.9-1 are specified in the reactor assembly design and fatigue analysis except for feedwater nozzles. The cycles listed in Table 3.9-1a are specified in the feedwater nozzle stress and fatigue re-analysis.

3.9.1.1.8 Main Steam Isolation Valve Transients

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

Transient	Category	Cycles
Preoperational at 100°F/h	Normal/upset	150
Startup (heating at 100°F/h), Pressure 0 to 1000 psig	Normal/upset	120
Shutdown:		
Cooling cycles at 100°F/h, 540 to 375°F, Pressure 1000 psig to 0 psig	Normal/upset	120
Cooling cycles at 270°F/h, 375 to 330°F, Pressure 1000 psig to 0 psig	Normal/upset	120
Cooling cycles at 100°F/h, 330 to 100°F, Pressure 1000 psig to 0 psig	Normal/upset	120
Scram cooling cycles at 100°F/h, Pressure 1000 psig increasing to 1125 psig, then decreasing to 240 then increasing to 1000 psig	Normal/upset	180
Emergency and faulted transients:		

Transient	Category	Cycles
1000 psig and 546°F to 35 psig and 281°F in 15 s	Emergency/faulted	1
546°F to 375°F in 3.3 min, 375°F to 281°F at 300°F/h, pressure 1000 to 35 psig	Emergency/faulted	1
546°F to 375°F in 10 min, 375°F to 281°F at 100°F/h, pressure 1000 to 35 psig	Emergency/faulted	8
546°F to 583°F in 2 s, 583°F to 538°F in 30 s, 538°F to 400°F then return to 546°F at 100°F/h, pressure 1000 to 1350 then to 240 then 240 then to 1000 psig	Emergency/faulted	1
561°F to 500°F in 7 min, 500°F to 400°F then to 546°F at 100°F/h, pressure 1000 to 1180 then to 240 then to 1000 psig	Emergency/faulted	10

3.9.1.1.9 Safety/Relief Valve Transients

The transients used in the analysis of the SRVs are as follows:

Transient	Category	Cycles
Preoperational and inservice testing (100°F/h)	Normal/upset	150
Startup (100°F/h) and pressure increase (0 to 1000 psig)	Normal/upset	120

Transient	Category	Cycles
Shutdown 540 to 375°F and 330 to 100°F at 100°F/h, (270°F/h between 375 and 330°F) pressure decrease to 0 psig	Normal/upset	120
Scram. Temperature change greater than 30°F will be at 100°F/h. Pressure increasing from 1000 to 1125 psig than decreasing to 240 psig then returning to normal operating pressure of 1000 psig.	Normal/upset	180
System pressure and temperature decay from 1000 psig, 546°F to 35 psig and 281°F within 15 s	Emergency/ faulted	1
System temperature change from 546 to 375°F within 3.3 min, and from 375° to 281°F at a rate of 300°F/h; pressure change from 1000 to 35 psig	Emergency/ faulted	1
System temperature change from 546 to 375°F within 10 min, and from 375 to 281°F at a rate of 100°F/h; pressure change from 1000 to 35 psig	Emergency/ faulted	8
System temperature change from 546 to 583°F within 2 s, from 583 to 538°F within 30 s, and from 538° to 400°F and return to 546°F at a rate of 100°F/h; pressure change from 1000 to 1350 psig, then to 240 psig and return to 1000 psig	Emergency/ faulted	1

System temperature changes greater than 30°F, from 561 to 500°F within 7 min, and from 500 to 400°F and return to normal operating temperature of 546°F at a rate of 100°F/h; pressure change from 1000 to 1180 to 240 psig, and return to normal operating pressure of 1000 psig	Emergency/ faulted	10

Paragraph NB-3552 of the ASME B&PV Code, Section III, excludes various transients and provides means for combining those that are not excluded. Review and approval of the equipment supplier's certified calculation ensures proper accounting of the specified transients.

3.9.1.1.10 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 pump case shall be 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
Heatup and cooldown at 100°F/h	Normal/upset	300
~29°F temperature changes	Normal/upset	600
~50°F temperature changes	Normal/upset	200

Transient	Category	Cycles
RPV pressure transients to 100 percent design pressure	Normal/upset	1
SRV blowdowns, 546 to 375°F in 10 min	Normal/upset	30
Improper pump startup, 130 to 546°F in 15 s	Emergency	1
Cooling transient, 546 to 281°F in 15 s	Emergency	2
Hydrotest to 1300 psig	Testing	130
Hydrotest to 1670 psig	Testing	3

3.9.1.1.11 Recirculation Gate Valve Transients

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

Transient	Cycles
50 to 575 to 50°F at 100°F/h	300
~29°F between limits of 50 to 575°F, instantaneous	600
~50F between limits of 50° and 546°F, instantaneous	200
546 to 375°F, in 10 minutes	30
546 to 281°F, in 15 seconds	2
130 to 546°F, in 15 seconds	1

Transient	Cycles
110 percent design pressure at 575°F	1
1300 psi at 100°F installed hydrostatic test	130
1670 psi at 100°F installed hydrostatic test	3

3.9.1.2 Computer Programs Used in Analyses

The following sections discuss computer programs used in the analyses of specific NSSS components. Computer programs were not used in the analyses of all components; thus, not all components are listed.

Sections 3.9.1.2.1 through 3.9.1.2.4 and 3.9.1.2.6 reference computer programs used by GE, its vendors and Structural Integrity Associates for analyzing NSSS components. These NSSS programs can be divided into three categories:

1. GE computer programs - Verification of the following GE computer programs has been performed in accordance with the requirements of 10CFR50, Appendix B. Evidence of the verification of the input, output, and methodology of the programs is documented in GE design record files:

    a. PISYS

    b. ANSI 7

    c. RVFOR

    d. TSFOR

    e. PDA

    f. SAP-4

g. ED-6

h. ED-8

i. EZPYP

j. DYSEA

k. SPECA04

l. GEAPL01

m. PIPST01

2. Hitachi computer programs - Verification of the following Hitachi computer programs is assured by the contractual requirements, such that the quality assurance procedure governing these proprietary programs used in the design of N-stamped equipment is in full compliance with the requirements of 10CFR50, Appendix B:

   a. TERESS

   b. FEMR

   c. FEASEL

   d. TSCHOCK

   e. ASSAL

   f. S-3, S-5, and S-7

   g. FLAHDERS and JESPO

   h. TEDS

3. Structural Integrity Associates computer program - The following Structural Integrity computer program was developed and verified in accordance with the Structural Integrity QA Program, which is in compliance with the requirements of 10CFR50, Appendix B and ANSI/ASME NQA-1-1989, and meets the intent of applicable portions of ANSI N45.2. The Structural Integrity implementation of the QA Program has been audited and accepted by many nuclear utilities. Structural Integrity's Quality Assurance Program is controlled by Structural Integrity's Quality Assurance Manual.

   a. VESLFAT

3.9.1.2.1 Reactor Vessel Assembly

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

3.9.1.2.1.1 TERESS

This program is used for the analysis of stresses and deflections in shell elements of axial symmetry, i.e., shells of revolution, due to thermal loadings. Input to this program consists of dimensions of shell elements, material properties, and temperature distribution. Output consists of stresses, edge rotations, and deflections of each shell. These edge rotations and deflections are used, in turn, for other computer programs, such as FLAHDERS for flange stress analysis.

3.9.1.2.1.2 FEMR

This program, based on an ordinary finite element method, is used for calculating stresses and deformations of axisymmetrical and plane stress fields. It is also capable of dealing with thermal stress and body force.

3.9.1.2.1.3 FEASEL

This program is a finite element method program including plane stress and axisymmetrical fields. The program can deal with elements that have no circumferential stress and have arbitrary thickness in the axisymmetrical problem.

3.9.1.2.1.4 TSCHOCK

This program is used for the analysis of peak stress in a cylindrical shell, due to the loading produced by a radial temperature distribution caused by a change of surface temperature owing to sudden contact with a fluid of different temperature. The time when the maximum peak stress occurs is also obtained.

3.9.1.2.1.5 ASSAL

This program is for calculating the stresses and deformation of a rotationally symmetrical shell subject to symmetrical and nonsymmetrical loads. The program is based on the theory of A. Kalnins, which reduces fundamental equations of shell to first order differential equations, including the linear theory of shell bending. The results yield a solution using the direct integral (boundary condition problem) and finite difference methods.

3.9.1.2.1.6 S-3, S-5, and S-7

These programs are general finite element analysis programs using the well known methods of quadratic shape displacement functions. They calculate elastic stresses and deformation of two and three-dimensional problems. The S-3 program is used for a plane stress field, and S-5 can be used for an axisymmetrical field. For three-dimensional problems, the S-7 program is applicable.

3.9.1.2.1.7 FLAHDERS and JESPO

These programs are used to compute discontinuity stresses in discontinuity regions using a method described in the ASME B&PV Code, Section III. FLAHDERS is used for stress analysis of flange stresses due to bolt tightening, internal pressure, and temperature distribution loads. The JESPO program is used for the stress analysis of the shroud support, considering stresses due to internal pressure, vertical force, and temperature distribution loads. Structural Integrity Associates computer program PIPX-TS2 was used to calculate the flange thermal transient and stress response for the alternate flood-up procedure.

3.9.1.2.1.8 TEDS

This program is used to calculate steady and variable transient temperature distributions in any odd shaped body in two-dimensional and three-dimensional fields, respectively. TEDS accommodates variations of boundary temperature and internal heat generation in both position and time. The program is useful in obtaining a temperature distribution for the evaluation of thermal stresses in any part of the reactor vessel. This program uses the shapes of internal nodes of triangle, square, and trapezoid in two-dimensional fields; therefore, the internal node input data agree with the element data of the FEMR computer program for the finite element stress analysis.

3.9.1.2.1.9 Reactor Internals

3.9.1.2.1.9.1 Core-Plate Beam Buckling - PIPST01

PIPST01 is a computer program which calculates approximate potential for core plate beam buckling. It uses the Rayleigh-Ritz energy method to determine the applied moment needed to initiate yielding and then finally to buckle a given tee beam. The tee beam model covers a segment of a BWR/2-5 core plate with a stiffener beam. The pressure differential across the plate that would have created this moment is calculated for a given length of beam or size of core plate.

Generic dimensions and material properties are all input by the user.

3.9.1.2.1.9.2 Structural Analysis Program - SAP 4

SAP 4 is a general structural analysis program for static and dynamic analyses of linear, elastic complex structures. The finite element displacement method is used to solve the displacements and to compute the stresses of each element of the structure. The structure can be composed of unlimited numbers of three dimensional truss, beam, plate, shell, solid, plate strain plane stress, brick, thick shell, spring, or axisymmetric elements. The program can treat thermal and various forms of mechanical loading, as well as internal element loading. The dynamic analysis includes mode superposition, time history, and response spectrum analyses. Earthquake loading, as well as time varying pressure, can be treated. The program is very versatile and efficient in solving large and complex structural systems. The output contains displacements of each nodal point, as well as stresses at the surface of each element.

3.9.1.2.1.9.3 Other Programs

Other computer codes used for the analysis of the internal components are described in detail in Section 4.1.

3.9.1.2.1.10 Feedwater Nozzle Re-analysis

Structural Integrity's re-analysis of the feedwater nozzles utilized the ANSYS finite element computer program. Thermal cycles for pre-extended power uprate (Pre-EPU) and extended power uprate (EPU) conditions were analyzed. Each of the thermal cycles was applied to the Hope Creek feedwater nozzle finite element model (FEM) to perform a thermal stress analysis using the ANSYS finite element software. The thermal stresses are added to pressure stresses and attached piping load stresses to obtain stress output. The ANSYS stress output was used in a subsequent ASME Code stress and fatigue analysis using Structural Integrity's VESLFAT computer program.

Structural Integrity's VESLFAT program performs fatigue usage calculations in accordance with ASME code, Section III, Sub article NB-3222.4(e) for Service Levels A and B conditions. The VESLFAT program computes the primary-plus-secondary and total stress ranges for all events and performs a correction for elastic-plastic analysis (K_e), if necessary.

The program evaluates the stress ranges for primary-plus-secondary and primary-plus-secondary-plus-peak stresses based on all six components of stress (3 normal and 3 shear stresses). The input maximum temperature for both states of a load set pair is used to establish the S_m value used in the fatigue calculations from the user-defined input values.

When more than one stress set is defined for either of the event pair loadings, the stress differences are determined for all of the potential stress pairs, saving the maximum for the event pair, based on the pair producing the largest alternating total stress intensity (S_{alt}), including any effects of K_e.

The stress intensities for the event pairs are reordered in decreasing order of S_{alt}, including a correction for the ratio of modulus of elasticity (E) from the fatigue curve divided by E from the material evaluated at the maximum event temperature. This allows a fatigue table to be created to eliminate the number of cycles available for each of the transient events. This fatigue table is based on a worst-case progressive pairing of events in order of the most severe alternating stress to the least severe, allowing determination of a bounding fatigue usage per NB-3222.4(e). For each load set pair in the fatigue table, the allowable number of cycles is determined based on S_{alt} and a cumulative usage factor (CUF) is calculated.

3.9.1.2.2 Piping

3.9.1.2.2.1 Piping Analysis Program - PISYS

PISYS is a computer code specialized for piping load calculations. It uses 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 that 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 Reference 3.9-1.

3.9.1.2.2.2 Component Analysis - ANSI 7

The ANSI 7 computer program determines stress and accumulative usage factors in accordance with NB-3600 of the ASME B&PV Code, Section III. The program was written to perform stress analysis in accordance with the ASME sample problem, and has been verified by reproducing the results of the sample problem analysis.

3.9.1.2.2.3 Safety/Relief Valve Discharge Pipe Forces-RVFOR

The safety/relief valve (SRV) discharge pipe connects the SRV 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.

See Section 3.9.1.2.5.4 for descriptions of the non-NSSS computer programs used to analyze relief valve discharge pipe forces.

3.9.1.2.2.4 Turbine Main Stop Valve Closure - TSFOR

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

3.9.1.2.2.5 Piping Dynamic Analysis - PDA

The pipe whip analyses use the PDA computer program. PDA is used to determine the response of a pipe subjected to the thrust force occurring after a pipe break. The program treats the situation in terms of a generic pipe break configuration, which involves a straight, uniform pipe fixed at one end and subjected to a time dependent thrust force at the other end. A typical restraint used to reduce the resulting deformation is also included at a location between the two ends. Nonlinear and time dependent stress strain relations are used to model the pipe and the

restraint. Similar to the popular elastic hinge concept, bending of the pipe is assumed to occur only at the fixed end and at the location supported by the restraint.

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

3.9.1.2.2.6 Piping Analysis Program - EZPYP

EZPYP links the ANSI-7 and SAP programs together. The EZPYP program can be used to run several SAP cases by making user specified changes to the basic SAP pipe model. By controlling files and SAP runs, the EZPYP program makes it possible to perform a complete piping analysis in one computer run.

3.9.1.2.3 Pumps and Motors

SAP is a general structural analysis program for static and dynamic analysis of linear elastic complex structures. The finite-element displacement method is used to solve the displacements and compute the stresses of each element of the structure. The structure can be composed of unlimited numbers of three dimensional truss, beam, plate, shell, solid, plane strain plane stress, brick, thick shell, spring, and axisymmetric elements. The program can treat thermal and various forms of mechanical loading as well as internal element loading. The dynamic analysis includes mode superposition, time history, and response spectrum analyses. Earthquake types of loading, as well as time varying pressure, can be treated. The program is very versatile and efficient in solving large and complex structural systems. The output contains displacements of each nodal point, as well as stresses at the surface of each element.

3.9.1.2.4 Residual Heat Removal Heat Exchangers

The following are the computer programs used in dynamic and static analysis to determine structural and functional integrity of the residual heat removal (RHR) heat exchangers:

3.9.1.2.4.1 Support Load Seismic Analysis - ED-6

This computer program computes the total loads at the upper and lower supports of the RHR heat exchanger. It takes into account the heat exchanger flooded weight, seismic loads (either OBE or SSE), and the allowable nozzle loads, and sets up the worst combination of these loads. By maximizing seismic loads together with nozzle loads, maximum conservative moments and forces at the upper and lower supports are calculated.

3.9.1.2.4.2 Stress Analysis of Supports - ED-8

This program performs a full stress analysis of the upper and lower supports of the RHR heat exchanger. The stresses in the supports, both upper and lower, caused by loads resulting from seismic and nozzle loads, are computed in support load program ED-6 and are used as input values for this program. This program computes the membrane stresses on the shell of the heat exchanger by the use of the Bijlaard's analysis, as well as the net section stresses (shear, tensile, bearing) on the lower support plate and upper lugs. It also computes the stresses on the welds that hold the supports to the shell of the heat exchanger.

3.9.1.2.5 Non-NSSS Seismic Category I System Components

The following sections discuss computer programs developed and/or owned by Bechtel, and programs that are recognized and widely used in industry. These computer programs are used in the analysis of non-NSSS Seismic Category I system components.

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 Bechtel programs is provided in the following paragraphs.

3.9.1.2.5.1 ME101-Linear Elastic Analysis

ME101 is a finite element computer program that performs linear elastic analysis of piping systems using standard beam theory techniques. The input data format is specifically designed for pipe stress engineering. ME101 performs a thorough check of the input prior to analysis. In addition, the program automatically modifies the geometry to improve the finite element model.

The output may be used directly for piping design, for conformation to Code, and for other regulatory requirements. Two piping codes, ASME B&PV Code, 1974, and ANSI B31.1, Summer 1973 Addenda, are incorporated in ME101 to the extent of computing flexibility factors, stress intensification factors, and stresses.

ME101 performs static and dynamic load analysis of piping systems, effective weight calculations, and ASME B&PV Code, Section III Class 2 and 3 and ANSI B31.1 Code stress checks.

Static analysis considers one or more of the following: thermal expansion, dead weight, uniformly distributed loads, and externally applied forces, moments, imposed displacements and rotations, individual force loads, static seismic (uniform directional acceleration) loads, or seismic anchor movement analysis.

Dynamic analysis is based on the standard normal superposition techniques. The input excitation may be in the form of seismic response spectra or time-dependent loading functions. In the single or multiple response spectrum analysis, the user may request modal synthesis by square root of the sum of the squares (SRSS) method or by NRC Regulatory Guide 1.92 closely spaced mode 10 percent

(Equation 4) method. ME101 can consider further differential damping for large and small pipe according to NRC Regulatory Guide 1.61. Various methods of eigenvalue solution are available. Determinant search or subspace iteration considers all data points as mass points. In the time history analysis, the excitation may be in the form of arbitrary nodal forces, support displacements, rotations, or support accelerations that are not necessarily in phase.

ME101 checks stresses from design loads versus allowable stresses according to ASME/ANSI Code equations. The user may request design load checks for sustained loads, occasional loads, multimode thermal expansion and pipe break, except for time history load cases.

The ME101 restraint load summary report prints the support load results from several load cases together in the same report, except for time history load cases.

The general loading combinations capability for ME101 can combine the results of several load cases together, according to certain algebraic rules, to form a new load case. The new load case resulting from this may be used in stress comparisons or restraint load summaries, except for time history load cases. ME101 has the capability of saving load case results on a tape and using these results in late runs for stress checks, restraint load summary reports, and general loading combinations, except for time history load cases.

For piping configurations with optional node numbering, ME101 generates isometric plots. The user may obtain plots on ZETA or CALCOMP plotters, on a Tektronix 4014 graphics terminal, or on an RMS-600 printer/plotter.

ME101 uses out-of-core techniques for both static and response spectra analysis and has no practical limitations to the number of equations or band width. However, the use of very large systems may become prohibitive due to cost of computation. The maximum number

of mode shapes allowable for response spectra analysis is currently 125.

This program considers the zero period acceleration effect in seismic response analysis. It accepts coordinate and keyword data in English or metric units.

The various versions of ME101 used in piping analysis for HCGS are C2, E1, G1, G3, H1, H3, I1, I2, J1, J2, J3, J4, J5, K1, K2 and K3.

Bechtel began this piping program's development in July 1975, and has continuously supported ME101. The program has been used by various Bechtel projects.

The ASME Benchmark Problem 1 demonstrates the solution for natural frequencies of a three dimensional structure, as described in Reference 3.9-2.

Natural frequencies, in hertz, from ME101 and Reference 3.9-2, are as follows:

Mode	Reference 3.9- 2	ME101
1	110	112
2	117	116
3	134	138

A total of 26 test problems were used for the verification of the ME101 results. These verification problems have been compared against one of the following:

1. ME632, Computer Program, "Seismic Analysis of Piping Systems," VERB MOD8, 1976, Bechtel International Corporation, San Francisco, California

2. "Pressure Vessel and Piping 1972 Computer Programs Verification," The American Society of Mechanical Engineers

3. Hand calculations

4. EDS Superpipe, EDS Nuclear, San Francisco, California

5. NUPIPE-IIM, Nuclear Services Corporation Piping Analysis Program, Campbell, California

6. TPIPE, A Computer Program for Analysis of Piping Systems, PMB Systems Engineering, San Francisco, California

7. ADINA, A Computer Program, Massachusetts Institute of Technology, Boston, Massachusetts

8. MSC/NASTRAN Program, McNeal Schwendler Corporation, Los Angeles, California

9. EASE2 Program, Engineering/Analysis Corporation, San Francisco, California

10. ANSYS, Swanson Analysis Systems, Inc, 1975, Elizabeth, Pennsylvania.

The J1 version of ME101 also includes seven NRC benchmarked problems, as referenced in NUREG/CR-1677, dated August 1980.

3.9.1.2.5.2 ME912-Thermal Stress

In ME912, a finite difference representation of the heat diffusion equation is used for the pipe or component wall section in contact with fluid of a specified temperature and flow rate time histories. The program is quasi two dimensional, accounting for the reduction of severity of a given transient with distance from inlet.

Thermal properties of water, stainless steel, and carbon steel are built in the program. Film transfer coefficients for water 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 the inlet to the length downstream. Average temperature differences, T_a-T_b, are thus calculated for structural discontinuity.

The ME912 program has been used by Bechtel on various projects. A Univac 1100 computer is used to run the program.

The ME912 program was developed from References 3.9-3, 3.9-4, and 3.9-5 by Bechtel. It has been used extensively since 1975 for Class 1 component design on the Fast Flux Test Facility (FFTF) project.

For local gradients, the program has been compared with analytical flat plate data of Reference 3.9-4 and numerical results by in-house program ME643. The results are acceptable. Table 3.9-2 shows the comparison of ME912 with ME643 and analytical results from Reference 3.9-4 for axial variations of fluid and wall temperatures; the program agrees closely with the analytical solution of Reference 3.9-5.

The ME643 program was developed from References 3.9-6 and 3.9-7 by Bechtel.

The results of ME643 transient temperature responses on both inside and outside surfaces of a sample pipe are compared with the chart of Reference 3.9-8 and plotted on Figure 3.9-1.

3.9.1.2.5.3 ME913-Nuclear Class 1 Piping Stress Analysis

ME913 can determine stress intensity levels for Class 1 nuclear power piping components (see equations 9 through 14 of subarticle NB-3650, Analysis of Piping Components, ASME B&PV Code, Section III).

Prior to using this program, the following information external to the program is required:

1. Piping configuration

2. Piping and piping component properties

3. Moment reactions due to:

   a. Thermal expansion loads

   b. Weight loads

   c. Earthquake loads and other dynamic loads

4. Thermal response of the piping system due to the specified transients is as follows:

   DT_1, DT_2 and the $(T_a T_b)$ values for the key points during system life.

ME913, Revision 4, is used by Bechtel. A Univac 1100 computer is used to run the program.

ME913 is the revised and expanded version of the LOTEMP program, originally developed by Bechtel and made available for use through the CDC 6600 computer. The LOTEMP program has been used extensively by the Bechtel FFTF systems analysis group since 1972 in the preliminary design of FFTF Class 1 piping. The ME913 program has

been used to analyze Class 1 piping for Bechtel nuclear power plant projects.

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 discussed in Reference 3.9-9. The results were compared with the computer output for equations 9 through 14 in ME913.

Table 3.9-3 shows the comparison between the ASME sample problem, Reference 3.9-9, and ME913 results.

3.9.1.2.5.4 Relief Valve Discharge Line Dynamic Forces Computer Program (NE805)

Following an actuation of a relief valve, the incoming steam pressurizes the discharge line, forcing the water leg (which could be at the normal water level, as in a first actuation of the relief valve, or an elevated level which is possible for subsequent actuation) out of the discharge line. Following water clearing, steam is discharged into the suppression pool. The water clearing and steam blowdown cause dynamic forces on the relief valve discharge line. Bechtel computer code RVCL (NE805) is used for analysis of discharge transients following a relief valve opening. NE805 is capable of modeling a discharge line of changing cross-sectional area. The code predicts the time dependent forces on the various segments of the relief valve discharge line. It models the steam flow through the relief valve and the steam air flows in the line. It also models the water flow in the discharge line during water clearing. The options for the exit device are a straight pipe, a ramshead, or a quencher model in a reservoir. The quencher model considers sequential uncovering the quencher holes during air/water clearing. NE805 uses the method of characteristics and allows for heat transfer through the pipe wall. It calculates

flow parameters, pressure, velocity, and density as functions of time and the distance along the discharge line. Using these calculated values, the code computes the dynamic forcing functions induced on various pipe segments of the relief valve discharge line.

The force output can be used directly for piping stress analysis in codes such as ME101, as described in Section 3.9.1.2.5.1. NE805 generates plots of flow parameter histories and/or force time histories, an option specified by the user. The plots can be obtained on both CALCOMP 1036 and the Tektronix plotters.

Development of the RVCL (NE805) program began in 1975 and is being continuously supported by Bechtel Power Corporation. It has been used by various Bechtel projects. The NE805 program has been verified against Monticello Mark I T-quencher test, Karlstein Mark II T-quencher test, and Caorso X-quencher test. Comparison with test data was found to be reasonable.

The current NE805 version is being used by Bechtel Power Corporation. A UNIVAC 1100 computer is used for executing the NE805 program.

3.9.1.2.5.5 ME210 - Local Stress in Cylindrical Shells due to External Loads

This standard presents a method of analyzing and determining local stresses in cylindrical shells due to external moments and forces acting on rigid attachments of circular or rectangular shape. This program is based on a paper 'Local Stresses in Spherical and Cylindrical Shells due to External Loadings' by Wichman, Hopper & Mershon, published in Welding Research Council Bulletin No. 107, August 1965 and March 1979 Revision. Values from Bijlaard curves are obtained by interpolation procedures.

This program also calculates piping stress intensity due to internal pressures and moments in accordance with the pressure and moment

stress calculations specified in EQ. 9 and EQ. 10 of ASME Section III NB-3650. The local stress intensity and piping stress intensity are summed and printed out if the required information for piping stress calculation is specified in the input. If no information for piping stress calculation is given, only the local stresses including primary plus secondary stress intensity and primary membrane stress intensities are printed out.

ME210 is executed on the UNIVAC 1100 Mainframe Computer (System B). This program has been utilized by Bechtel on various projects.

3.9.1.2.5.6 ME602 - Spectra Merging and Simplified Seismic Analysis

ME602 performs the seismic analysis of small diameter piping systems (2 inch and under) using the modified response spectrum method described in BP-TOP-1, Rev. 3. The program generates a set of tables of seismic spans, support reactions and stresses for various pipe sizes.

This program performs response spectrum curve merging along with the calculation of the seismic span. The program can be also used independently for the sole purpose of merging spectrum curves and storing the combined spectrum data for ME101 analysis. A neutral plot file of the "RAW" or "COMBINED" spectrum curves can be generated for plotting on RMS, TEKTRONIX, CALCOMP or any neutral file compatible plotter.

ME602 is executed on the UNIVAC 1100 Mainframe Computer (System B). This program has been utilized by Bechtel on various projects.

3.9.1.2.5.7 MA099 - ME101 Interactive Program

The ME101 Interactive Program operates on three types of data. The first type is an ME101 image deck. In this case the deck can be transferred from the Univac to the Chromatics (typically CADME (ME099) deck from CAD file) then converted by the Chromatics into an

ME101 Interactive Random I/O access data base. The second type is created directly on the ME101 workstation using the Data Manager and Input/Modify program modules.

Any ME101 Interactive data (Random I/O access) file can be converted back into ME101 image file by going through "FORMAT" in "RUN" module. ME101 results from the Univac are the third type, and must be converted into ME101 Interactive data base in order to display results in tables or graphic plots. These three types of data file can be saved on the Chromatics internal hard disk or stored on 8-inch single double density floppy disk. The data stored on floppy disk, which provides inexpensive storage, can be used on any other ME101 workstation.

The ME101 Interactive Program uses a Chromatics CGC7900 color graphics microcomputer.

3.9.1.2.5.8 CE798 - ANSYS Program

The ANSYS program is a self-contained general purpose finite element program developed and maintained by Swanson Analysis Systems, Inc. The program contains many routines, all interrelated, and all for the main purpose of achieving a solution to an engineering problem by the finite element method.

The ANSYS program is designed to be user oriented. It does not require special knowledge of system operations or computer programming in order to be used. The basic input is straightforward and easily learned. In addition, the program flexibility allows various methods of arriving at the same solution. The flexibility, capabilities, and options have been developed over many years, at the request of a worldwide user community, such that the ANSYS program can be applied to a wide variety of engineering applications.

CE798 is executed on the UNIVAC 1100 Mainframe Computer (System B).

3.9.1.2.5.9 ME351 - Pipe Rupture Analysis Program

This program performs nonlinear elastic plastic analysis of three dimensional piping systems subjected to concentrated static or dynamic time history forcing functions. These forces may result from fluid jet thrust at the location of a postulated rupture of high energy piping. PIPERUP is an adaptation of the finite element method to the specific requirements of pipe rupture analysis. Straight and curved beam (elbow) elements are used to mathematically represent the piping and axial and rotational springs are used to represent restraints. The stiffness characteristics of piping and restraints can reflect elastic/linear strain hardening material properties, and gaps between piping and restraints can be modeled.

ME351 is executed on the UNIVAC 1100 Mainframe Computer (System B). Various Bechtel projects utilize this program.

3.9.1.2.5.10 ME632 - Piping System Analysis

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.

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.

ME632 is executed on the UNIVAC 1100 Mainframe Computer (System B).

3.9.1.2.5.12 ME909 - Spectra Curves Merging

This program merges individual response spectra curves, makes a neutral plot file of these curves, and produces data cards for ME101 seismic analysis.

The test data to validate this conversion program is available upon request from the Program Manual Library in San Francisco or the Library in Gaithersburg.

ME909 is executed on UNIVAC 1100 Mainframe Computer (System B) and has been utilized by Bechtel on various projects.

3.9.1.2.5.12 BASEPLATE II (CE-035)

BASEPLATE II consist of a pre- and a post-processor for the STARDYNE structural analysis system, with the specific purpose of analyzing flexible baseplates on a geometrically non-linear foundation.

The pre-processor program automatically creates a finite element model of the prescribed structure and generates the appropriate STARDYNE input data and control cards. The capabilities include multiple attachments specification, standard library attachments and non-library attachments, integral framed structure, mixed support conditions and several load cases. The post-processor program sorts the STARDYNE output and tabulates the relevant results.

CE-035 utilizes the CDC computer.

3.9.1.2.5.13 BASEPLATE (ME-035)

The finite element method is used to analyze the baseplate response. The EAL (Engineering Analysis Language) is employed for solution of the problem and for post-processing of results, special pre-processor was written for model generation. The model generation is controlled through input parameters by the user. The plate elements are based on mixed formulation which allows computation of stresses in nodal points, the contact between the baseplate and the concrete or deactivated depending on the negative or positive displacement. Bolts are considered as combination of general stiffness matrix and one dimensional elements. Iterative solutions are used to find the final configuration starting with the assumption that all bolt springs are active.

ME-035 utilizes the U1100 computer, system B.

3.9.1.2.5.14 BOLT Program (CE-050)

CE-050, "BOLTS", is a FORTRAN computer program which will determine concrete expansion and grouted-in anchor loads and interaction values for baseplates anchored with symmetrical 4, 6, or 8 bolt patterns which incorporate the effects of the base plate flexibility and bolt stiffness. The program resides on the Bechtel Univac System under the EXEC 8 operating system. BOLTS is intended to be executed interactively from a remote terminal thus providing immediate results for design/analysis problems.

The program is well suited to the evaluation of base plates commonly used in pipe, conduit, HVAC, cable tray, and other small equipment supports. Empirically generated relationships are employed to analyse centrally loaded, symmetrical, unstiffened plates.

CE-050 utilizes the U1100 computer, system B.

3.9.1.2.5.15 WELD Program (ME-120)

This program presents a method of determining fillet weld size for the connecting structural member based on the approach described in "Design of Welded Structures" by O.W. Blodgett, and "Solutions to Design of Weldments" by O.W. Blodgett.

The main objective of this program is to minimize the total weld length.

ME-120 utilizes the U1100 computer, system B.

3.9.1.2.5.16 FAPPS (ME-150)

"FAPPS" (Frame Analysis Program for Pipe Support) is an interactive computer program specifically developed for the analysis and design of 18 standard frames ("easy input") as well as any non-standard

frame for pipe support. It optimizes member sizes, welds, base plates and embedments based upon various user specified design limitations.

ME-150 utilizes the U1100 computer, system B.

3.9.1.2.5.17 Anchor Plate (ME-225)

This program presents a method of analyzing plate type piping anchors. The program determines the following:

1. Thickness of anchor plate

2. Thickness of guide plate

3. Welds joining the plate and process pipe

4. Welds joining the supporting structure

5. Take-out dimensions of the anchor plate and the guide plate

ME-225 utilizes the U1100 computer, system B.

3.9.1.2.5.18 Pipe Clamp (ME-226)

This theory is developed for a program which computes the minimum required thickness and the stresses at two critical sections, for six special cases of pipe support clamps. The forces and resultant stress in the clamp bolts are also computed. The program will also compute the forces and stress in the stanchion, welds, and the base plate if applicable. Finally, the program will compute certain clamp dimensions and the total weight of the clamp and its associated hardware.

This theory uses linear elastic analysis.

ME-226 utilizes the U1100 computer, system B.

3.9.1.2.5.19 ICES STRUDL II (CE-901)

STRUDL is a broad, extensive and general program for solving problems in Structural Engineering.

CE-901 utilizes the U1100 computer, system B.

3.9.1.2.5.20 SI Weld (ME-799)

This program presents a method of determining fillet weld size for the connecting structural member based on the approach described in "Design of Welded Structures" by O.W. Blodgett, and "Solutions to Design of Weldments" by O.W. Blodgett.

ME-799 utilizes the U1100 computer, system B.

3.9.1.2.5.21 EZPLOT (UE-170)

EZPLOT is an interactive program installed on service bureaus which have CDC NOS 175 and NOS 176 computers. It plots structural analysis data written for the following programs:

STARDYNE	SACS	(By special request:
EASE 2	STRAN	FLUSH, STRUDL, BSAP)
NASTRAN		

In the case of STARDYNE, EZPLOT can create post processor plots of deflected shapes resulting from static or dynamic analyses.

EZPLOT will read data for these programs and plot it directly on the screen of a plotting terminal, or it can create a neutral plot file which can be directed to any type of supported plotting device including pen plotters, VERSATEC, etc. Plotting terminals include TEKTRONIX (e.g., 4014) or TEKTRONIX compatible terminals.

UE-170 utilizes the CDC computer.

3.9.1.2.5.22 UNIPLOT (UE-188)

UNIPLOT (UE-188) was developed at Bechtel to provide the users with a device independent computer graphics library. The neutral plot file, generated by UE-188 subroutines, can be directed to any one of Bechtel's graphics devices (e.g., Calcomp, Tektronix, RMS, Versatec, etc.), located at both the home and area offices.

UE-188 operates on UNIVAC, IBM, and VAX, and can be used with programs written in FORTRAN (either the FOR or FTN compilers).

All subroutines in the UE-188 library are Calcomp compatible. Therefore, any program written for the Calcomp plotters can be mapped with UE-188 to take advantage of its device independence. This includes programs that use the DISSPLA (UE-188) subroutines.

UE-188 utilizes the U1100 computer, system B.

3.9.1.2.5.23 STARDYNE (CE-991)

The STARDYNE Analysis System consists of a series of compatible digital computer programs designed to analyze linear elastic structural models. The system encompasses the full range of static and dynamic analyses. These programs provide the analyst with a sophisticated, cost effective, structural dynamical analysis system.

CE-991 utilizes the CDC computer.

3.9.1.2.5.24 Hydraulic System Transient Analysis Computer Program (NE820)

Bechtel's computer code HSTA (Hydraulic System Transient Analysis, NE820) is a code that can be used to analyze flow transients in

liquid systems. These transients are commonly referred to as water hammer and are a result of a rapid interruption of flow, filling of an empty line or collapsing of a void, etc.

While the code can be considered as a general water hammer computer code, the specific needs that arise in the analysis of various nuclear power plant systems have received considerable attention. The code primarily calculates the pressure and velocity changes with time for various locations in the piping network. These variables can then be used to compute the dynamic forces for various pipe segments of the system. These forces represent the dynamic forcing function to be utilized in the structural analysis of the piping system and can be used directly in codes such as ME101.

The code enables analysis of transients resulting from valve closure, valve opening, pump failure, pump startup, rapid depressurization, etc. It can model the liquid (water) column separation phenomenon (encountered in transients where pressures drop to the vapor pressure of the liquid). The code can also be used in the analysis of startup transients in cases where the piping system is initially not full of liquid. The code can be applied to analyze steam or gaseous flow systems under situations having no significant changes in fluid density during the transient.

To solve one dimensional unsteady flow problems, the mass and momentum conservation equations (along the pipe axis) are utilized to obtain pressure and velocity as functions of time and distance. The solution is based on the finite difference solution of the method of characteristics (MOC). The numerical scheme uses a constant time step constant distance grid. The solution starts from steady initial conditions and progresses to the desired time level. It allows for compressibility of the fluid and takes into account the elasticity of the pipe walls. Pipe frictional losses are also considered. Hydraulic devices are treated as boundary conditions.

Several options of the HSTA code have been verified. In this verification effort, emphasis was placed on comparison with

experimental or test data; however, in some instances comparisons were made to independent numerically predicted results. The code verification incorporates comparison with test data or independently calculated data for valve closure, branching, pump failure, open surge vessel (with one way check valve), air tank (for suppression of pressure surge), positive displacement pump, liquid (water) column separation in a horizontal line and in a siphon system and finally for line filling case.

The current NE820 version is being extensively used by Bechtel Power Corporation. A UNIVAC 1100 Computer is used for executing the NE820 program.

3.9.1.2.5.26 GAFT - Program (NE810)

GAFT (Gaseous Fluid Transients NE810), is a computer code that can be used to analyze flow transients in gaseous systems. When the fluid is steam these transients are commonly referred to as steam hammer. GAFTPLOT, designated as program number NE811, is a post processor plot program that allows the plotting of flow variables and force time histories that are calculated by GAFT.

This code has been developed by Bechtel Power Corporation over the last several years. While it can be considered as a general computer code for gaseous fluid transients, the specific needs that arise in the analysis of nuclear power plant systems have received considerable attention. The code primarily calculates the pressure, velocity, density and sonic speed changes with time for various locations in the piping network. These variables can then be used to compute the dynamic forces for various pipe segments of the system. These forces represent the dynamic forcing functions to be utilized in the structural analysis of the piping system and can be used directly in codes such as ME101.

The code enables analysis of transients resulting from valve closure, valve opening, pipe break, rapid depressurization, etc. As an example of the code application, the main steam piping system of

a nuclear power plant can be modeled on GAFT. Events such as the turbine stop valve closure or a break in the main steam line, can then be analyzed to calculate the dynamic forcing functions for pipe segments in the main steam piping. An assumption in such an analysis would be that steam behaves as an ideal gas.

To solve one dimensional unsteady flow problems, the mass, energy, momentum and conservation equations (along the pipe axis) are utilized together with an equation of state to obtain pressure, velocity, density and sonic speed as functions of time and distance. The solution is based on the finite difference solution of the method of characteristics (MOC). The numerical scheme uses a constant time step constant distance grid. The solution starts from steady initial conditions and progresses to the desired time level. It allows for compressibility of the fluid but assumes inelastic pipe walls. Pipe frictional losses are also considered. Hydraulic devices such as valves, restrictions, branching, etc. are treated as boundary conditions.

Several options of the GAFT code have been verified. In the GAFT verifications endeavors, emphasis was placed on comparison with experimental or test data; however, in some instances comparisons were made to independent numerically predicted results. The present version incorporates comparisons between the code results and either measured or independently calculated data for valve closure, branching, pipe break and dead end hydraulic boundary devices.

The current NE810/NE811 versions are being used extensively by Bechtel Power Corporation. A UNIVAC 1100 Computer is used for executing the NE810/NE811 Programs.

3.9.1.2.5.26 REPIPE - Program (NE565)

Program REPIPE (NE565) computes the loading time history on a piping network from the Program RELAP hydrodynamic analysis of the contained fluid.

Complex piping networks experience a variety of forces generated by the fluid flow within them. These forces can be simply classified as steady state and transient. Steady state forces are those which are present during the normal operation of the system. Initial design efforts recognize these forces and compensate for them with appropriate support. However, in abnormal situations, such as inadvertent valve operation, pipeline rupture, or a pump malfunction within the network, the fluid forces change rapidly. These momentary or transient forces can be quite large, particularly in a power plant where the fluid is initially at high pressure and temperature.

Determining the distortion of a piping network during a transient usually becomes a three step process. First, the time history behavior of the fluid within the pipe is calculated (the hydraulics programs). Second, the force time histories are calculated from the fluid behavior. Third, the piping network stresses over time are determined (the stress program).

The transient hydraulic analysis employs either RELAP4 or RELAP5. The RELAP series of computer programs compute time varying pressure, momentum flux and energy states throughout a fluid system containing water, steam, and/or a two phase mixture. The programs utilize a one dimensional fluid flow solution, and are particularly useful and efficient for transient flow in piping networks.

REPIPE is the intermediate processor which operates on the output of the hydraulic program to produce force time histories for input to the stress program.

3.9.1.2.5.27 Thermal Hydraulic Transient Analysis Program (NE458)

Program RELAP5/MOD1 (NE458) is an advanced thermal hydraulics program intended for the analysis of complex transients in nuclear reactors and piping networks. Equations of conservation of mass, energy, and momentum are solved in one dimension for steam and/or

water flow. The equations assume a non-homogenous mixture of steam and liquid, and non-equilibrium between phases can be modeled. The effects of noncondensible gas on steam/liquid flow are considered in the equations. Models are available to simulate pump, valve, and heat exchanger components, as well as complex control systems. RELAP5 is expressly written for the analysis of both small and large break reactor loss-of-coolant accidents, but can be used to analyze many power plant operational transients. The program is frequently coupled with the post processor REPIPE to generate hydrodynamic loads on power plant piping.

The current NE458 version is on the CDC computer system.

3.9.1.2.6 Dynamic Loads Analyses

3.9.1.2.6.1 Program for Dynamic/Seismic Responses of Three Dimensional Members - DYSEA

The DYSEA program embodies the spatial finite element method together with temporal model superposition and response spectrum analysis features. The timewise solutions of the decoupled modal equations were obtained by using the Newmark-[Sb] integration scheme. The program permit one to predict the dynamic and/or seismic responses of structural systems that may be composed of three dimensional truss, beam, and spring members. The material properties are restricted to the linear, elastic stress strain behavior range. The program can handle dynamic systems having mass coupling such as through the hydrodynamic mass matrix, which may arise from the hydrodynamic interaction effect of the structure submerged in a potential fluid field. The structural system may be subjected to externally applied time dependent mechanical forces, earthquake excitations, and/or multiple support excitations.

3.9.1.2.6.2 Acceleration Response Spectrum Program - SPECA04

The SPECA04 computer program generates acceleration response spectrum, consistent with the requirements of Regulatory Guide 1.122

for an arbitrary input of the time history of piecewise linear accelerations, i.e., to compute the 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/enveloping spectra in conformance with Regulatory Guide 1.122 when the spectral points are generated equally spaced on a logarithmic axis of period/frequency. This program is also used in seismic and safety/relief transient analyses.

3.9.1.2.6.3 Forces and Moment Time Histories Program - GEAPL01

The GEAPL01 computer program converts distributed asymmetric pressure time histories over a given area into equivalent time varying nodal forces and moments for use as input to perform dynamic analysis of a system. The overall resultant forces and moment time histories at specified points of resolution can also be obtained from GEAPL01.

3.9.1.2.6.4 Dry Storage Cask Dynamic Loading - Visual Nastran

The freestanding cask configurations are modeled using kinematic computer program Visual Nastran as an assemblage of rigid bodies (of approximate size and mass) interconnected by spring and damping elements. The acceleration time histories generated in HI-2043145 are applied to the model as the driving input.

3.9.1.2.6.5 Generation of 3-D Acceleration Time Histories for Dry Storage Casks

The time histories are developed in accordance with the applicable USNRC requirements (i.e., SRP 3.7.1) using the Holtec computer code GENEQ, which is a derivative of the commercially available code SIMQKE. This method of generation has been used by Holtec and approved by the NRC on numerous dockets in spent fuel rack licensing.

3.9.1.2.6.6 Reactor Building Structural Analysis for Dry Storage Casks

The forces and moments in concrete slab and the underlying steel beams are solved using the finite element code ANSYS. The model is comprised of 4-noded shell elements representing the concrete slab and 2-noded linear beam elements representing the steel I-beams. A response spectrum analysis is performed within ANSYS to obtain the force and moment distribution in the slab due to its self-weight excitation under the OBE and SSE loading. The live loads and cask induced loads are solved by static linear analysis. The factored load combinations are assembled in ANSYS and the results are compared with the allowable load and stress limits in accordance with HCGS Design Criteria Document 10855-D2.1.

3.9.1.3 Experimental Stress Analyses

No experimental stress analyses were utilized.

3.9.1.4 Considerations for the Evaluation of Faulted Conditions

Each Seismic Category I component is evaluated for the faulted loading conditions. In all cases, calculated stresses are within the allowable limits. The following paragraphs show examples of the treatment of faulted conditions for the major components on a component-by-component basis. Elastic-plastic analyses have not been used in evaluating the HCGS Seismic Category I systems and components for compliance with service level D limits. The stress levels of these components are below the stresses allowed by the ASME B&PV Code. Additional discussion of the faulted event and analyses can be found in Sections 3.9.3 and 3.9.5, and Tables 3.9-1 and 3.9-4.

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Seismically designed piping is separated from non-Seismic Category I piping by seismic boundary anchors. These anchors are designed for the combined load generated from both sides of the boundary anchor. The loads from the Seismic Category I side are actual calculated loads, and the loads from the non-Seismic Category I side are determined by one of the following:

a. Loads determined by the plastic capability of the piping,

b. If the loads calculated by item a. are excessively high, then seismic loads for the non-seismic Category side are calculated using dynamic seismic analysis or equivalent static seismic analysis,

c. Actual calculated loads if the non-seismic side piping is isolated by a combination of anchor and restraints. This combination is designed to a simplified seismic design criteria (e.g., by simplified span method such as those used for the design of small piping).

See Section 3.7.3.13.

Sections 3.9.2.3 and 3.7 discuss the treatment of dynamic loads resulting from the postulated SSE. Section 3.9.2.7 discusses the dynamic analysis of loads on NSSS equipment resulting from blowdown. Deformations under faulted conditions were evaluated in critical areas, and no cases are identified where design limits, such as clearance limits, are exceeded.

3.9.1.4.1 Control Rod Drive System Components

3.9.1.4.1.1 Control Rod Drives

The major CRD components that were analyzed for the faulted conditions are the ring flange, and the indicator tube. The maximum stresses for these components, and for various plant operating conditions, including faulted conditions, are given in Table 3.9-4w.

The ASME B&PV Code, Section III components of the CRD have been analyzed for scram with an inoperative buffer and for a scram with a stuck control blade conditions shown in Section 3.9.1.1.1. The loads and stresses are within the elastic limits of the material.

No analysis was made for the non-Code components of the CRD for the abnormal condition. The design adequacy of non-Code components of the CRD has been verified by 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 cycles listed in Section 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 3.9.2.3.2.4 discusses the dynamic qualification of the HCU.

3.9.1.4.2 Standard Reactor Internal Components

3.9.1.4.2.1 Control Rod Guide Tube

The maximum calculated stress on the control rod guide tube occurs in the base during the faulted condition. The faulted limit is the lesser of 2.4 S or 0.7 S at the design temperature, per ASME B&PV Code, Section III, Table F1322-1. Per ASME B&PV Code,

Section III, Table I-1.2, S equals 57,500 psi and S_m equals 16,000 psi at 575°F, and the results are summarized in Table 3.9-4aa.

3.9.1.4.2.2 In-core Instrument Housing

The maximum calculated stress on the in-core instrument housing occurs at the outer surface of the vessel penetration during the faulted condition. The calculated and allowable stresses are shown in Table 3.9-4bb.

3.9.1.4.2.3 Jet Pump

The elastic analysis for the jet pump faulted conditions shows that the maximum stress occurs at the jet pump riser. The stress analysis results are summarized in Table 3.9-4y.

3.9.1.4.2.4 Low Pressure Coolant Injection Coupling

The stresses on the low pressure coolant injection (LPCI) coupling are very low during a faulted event, due to its flexibility. The maximum stress due, to faulted loads, and the allowable stress are given in Table 3.9-4z.

3.9.1.4.2.5 Orificed Fuel Support

See Section 3.9.1.3.2.

3.9.1.4.2.6 CRD Housing

The SSE is classified as a faulted condition. The maximum stress on the CRD housing during an SSE is 22,030 psi. The maximum design stress limit for this event is 2.4 S_m which equals 40,000 psi, and the ultimate strength of the material is 57,000 psi.

The CRD housing was analyzed for a faulted condition, including an SSE. Table 3.9-4x shows the loading conditions, load combinations, analytical methods, and allowable and calculated stress values for the highly stressed areas of the CRD housing.

3.9.1.4.3 Reactor Pressure Vessel Assembly

For the faulted conditions, the RPV and shroud support were evaluated using elastic analysis methods. Ultimate strength allowable values were not used since the emergency allowable stress limits of ASME B&PV Code, Section III, were used for the faulted stress cases. Table 3.9-4b lists the 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 Section 3.9.5. The calculated and allowable stresses are summarized in Tables 3.9-4b and 3.9-4c.

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

Tables 3.9-4i, 3.9-4j, and 3.9-4l provide a summary of the analyses of the safety/relief, main steam isolation, and recirculation gate valves, respectively.

Standard design rules, as defined in applied codes, are used in the analysis of pressure boundary components of Class 1 active valves. Conventional elastic stress analyses are used to evaluate components not defined in the ASME B&PV Code. The allowable Code stresses are applied to determine acceptability of the structure under applicable loading conditions, including the 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 the ASME B&PV Code, Section III, Appendix F, Rules for Evaluation of Faulted Conditions, and these are above elastic limits. Additional information on the main steam and recirculation piping is presented in Tables 3.9-4e through 3.9-4h.

3.9.1.4.7 Nuclear Steam Supply System Pumps, Heat Exchangers, and Turbine

The recirculation, Emergency Core Cooling System (ECCS), reactor core isolation cooling (RCIC), and standby liquid control (SLC) pumps; the residual heat removal (RHR) heat exchangers; and the RCIC turbine were analyzed for the faulted loading conditions identified in Section 3.9.3.1. In all cases, stresses are within the elastic limits. The analytical methods, stress limits, and allowable stresses are discussed in Sections 3.9.2.3 and 3.9.3.1.

3.9.1.4.8 Control Rod Drive Housing Supports

Examples of the calculated stresses, and the allowable stress limits for the faulted condition for the CRD housing supports, are shown in Table 3.9-4cc.

3.9.1.4.9 Fuel Storage Racks

Examples of the calculated stresses and stress limits for the faulted conditions for the new fuel storage racks are shown in Table 3.9-4u.

3.9.1.4.10 Fuel Assembly (Including Channel)

GE boiling water reactor (BWR) fuel assembly (including channel) design bases, analytical methods, and evaluation results, including

those applicable to the faulted conditions, are contained in References 3.9-10 and 3.9-11. The acceleration profiles are summarized in Table 3.9-4ee.

Specific analysis of ABB fuel is performed to demonstrate compliance with applicable design criteria. The ABB fuel Assembly (including channel) design bases and analytical methods are described in Reference 3.9-24 and 3.9-25.

3.9.1.4.11 Refueling Equipment

Refueling and servicing equipment important to safety are classified as essential components, per the requirements of 10CFR50, Appendix A. This equipment, and other equipment whose failure would degrade an essential component, are defined in Section 9.1 and are classified as Seismic Category I. These components are subjected to an elastic, dynamic, finite element analysis to generate loadings. This analysis uses appropriate seismic floor response spectra and combines loads at frequencies up to 33 hertz 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, or AISC allowables. The calculated and allowable stresses are summarized in Table 3.9-4u.

3.9.1.4.12 Non-NSSS Seismic Category I System Components

The stress allowables of Appendix F of the ASME B&PV Code, Section III, in effect at the award of each purchase order, 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 of the ASME B&PV Code, Section III 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 standard 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.

3.9.1.4.13 Piping Branch Connection Design

All branch design branch connections for ASME Section III Class 1, 2, and 3 piping allow only the use of integrally reinforced fittings. Design drawings supplied to the piping fabrication stipulate the type of branch connection to be used based on run pipe size, branch pipe size and piping class, temperature and pressure. The piping fabricator is required to perform reinforcement calculations for all branch connections except where the ASME Section III excludes such calculations. For Class 1 piping, reinforcement calculations are included in the stress report.

At the branch connection, appropriate stress intensification or stress indices are used to determine the maximum stresses due to mechanical loads. Moments from three legs are considered in evaluation of branch connection.

NSSS:

All branch connections are designed to meet the requirements of paragraph NB-3643.3 of Section III of the ASME B&PV Code for the reinforcement of openings. For branch connections, the stress indices of NB-3681 are used to compute the stresses resulting from internal pressure and mechanical loadings. Compliance with the B&PV Code stress criteria is documented in the piping design reports. Branches with nominal diameters less than one-fourth of the diameter of the pipe run are decoupled from the large pipe models and

analyzed separately. The small branch pipes are not in the General Electric scope of supply; however, to ensure they have a negligible effect on the large piping runs, the mechanical loadings from the small pipe are provided by the organization responsible for the small pipe (Bechtel Power Corporation).

3.9.2 Dynamic Testing and Analysis

Dynamic testing and analysis are performed for the purpose of verifying the capacity of mechanical systems and components to satisfy dynamic design requirements under plant operating conditions. Depending on the components, the operating event, and the feasibility of either analysis or testing, the design for dynamic performance normally rests generally upon analytical calculations. In this context, testing generally serves to verify the dynamic calculations of the system dynamic performance. In some cases, due to either the unpredictability of loads or the complexity of structural response, primary design verification results from testing. An example of this latter case occurs in the steady state vibration evaluation of piping systems.

In general, dynamic testing and analysis are organized in two parts:

1. Nuclear Steam Supply Systems (NSSS) Systems

2. Non-NSSS, Balance Of Plant (BOP) Systems.

3.9.2.1 Thermal Expansion, Piping Vibration, and Dynamic Effects in NSSS Piping

The test program is divided into three phases: thermal expansion, piping vibration, and transient dynamic effects.

3.9.2.1.1 Thermal Expansion Testing of Main Steam and Recirculation Piping

The thermal expansion preoperational and startup testing program is performed through the use of potentiometer sensors and verifies that normal thermal movement occurs in the piping systems. The main purpose of this program is to ensure the following:

1. The piping system during system heatup and cooldown is free to expand, contract, and move without unplanned obstruction or restraint in the x, y, and z directions.

2. The piping system is working in a manner consistent with the assumptions of the stress analyses.

3. There is adequate agreement between calculated and measured values of displacement.

4. There is consistency and repeatability in thermal displacements during heatup and cooldown of the systems.

Thermal expansion displacement limits are established before the start of testing. The measured displacements are compared with these limits to determine the acceptability of the measured motion. If the measured displacement is shown to be within the acceptance limits, the piping system is moving in a manner consistent with the stress analyses. On that basis, the pipe thermal expansion behavior is acceptable. Two levels of displacement limits criteria, level 1 and level 2, are described in Section 3.9.2.1.4.

3.9.2.1.2 Piping Vibration

3.9.2.1.2.1 Preoperational and Startup Vibration Testing of Recirculation Piping

The purpose of the vibration test phase is to verify that the operating vibration level in the recirculation piping is within

acceptable limits. This phase of the test uses visual observation and, if necessary, hand held instrumentation to supplement remote measurements. If, during steady state operation, visual observation indicates that vibration is significant, measurements are made with a hand held vibrograph. Testing is done during the following steady-state conditions:

1. Minimum flow

2. 50 percent of rated flow

3. 75 percent of rated flow

4. 100 percent of rated flow

5. With RHR in the shutdown cooling mode at rated RHR shutdown cooling loop flow.

3.9.2.1.2.2 Preoperational Vibration Testing of Small Attached Piping

During visual observation of each of the above test conditions, 1. through 5., attention is given to small attached piping and instrument connections to ensure that they are not in resonance with the recirculation pump motors or flow induced vibrations. If the operating vibration acceptance criteria are not met, corrective action such as modification of supports is taken.

3.9.2.1.2.3 Operating Transient Loads on Main Steam and Recirculation Piping

The purpose of the operating transient test phase is to verify that pipe stresses are within ASME B&PV Code limits. The amplitude of displacements and number of cycles per transient of the main steam and recirculation piping are measured and displacements compared with acceptance criteria. The deflections are correlated with stresses to verify that the pipe stresses remain within ASME B&PV

Code limits. Remote vibration and deflection measurements are taken during the following transients:

1. Recirculation pump start

2. Recirculation pump trip at 100 percent of rated flow

3. Turbine main stop valve closure at 100 percent power

4. Manual discharge of each safety/relief valve (SRV) at 1000 psig and at planned transient tests that result in SRV discharge.

3.9.2.1.3 Dynamic Effects Testing of Main Steam and Recirculation Piping Systems

To verify that snubbers are adequately performing their intended function during the plant operation, a program for dynamic testing as a part of the initial startup operation testing is conducted. The main purpose of this program is to ensure the following:

1. The vibration levels from the various dynamic loadings during transient and steady state conditions are below the predetermined acceptable limits.

2. Long term fatigue failure does not occur due to underestimating the dynamic effects caused by cyclic loading during plant transient operations.

The purpose of dynamic testing is to account for the acoustic wave due to the SRV lift (RV1), SRV loads resulting from air clearing (RV2), and turbine main stop valve closure loads (TSVC). The maximum stress developed in the piping from the RV1, RV2, and TSVC transients is used as a basis for establishing criteria that ensure proper functioning of the snubbers. If field measurements are within criteria limits, snubbers are assumed to be functioning properly. Snubbers are tested to allow free piping movements at low

velocity. During plant startup, the snubbers are checked for proper settings and for any evidence of hydraulic fluid leakage.

The above testing was performed with the original Target Rock 2-Stage SRVs installed at original plant startup. Subsequently, Target Rock 3-Stage SRVs have been evaluated and approved for installation at Hope Creek. Results from the above testing remain valid because the 3-Stage SRVs have the same set pressures, capacities, and response times as the 2-Stage SRVs.

3.9.2.1.4 Test Evaluation and Acceptance Criteria for Main Steam and Recirculation Piping Systems

The piping response to test conditions is 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 limits of the ASME B&PV Code, Section III, NB-3600. Acceptable deflection and acceleration limits are determined after the completion of piping systems stress analyses and are provided in the startup test specifications. To ensure test data integrity and test safety, criteria are established to facilitate assessment of the test while it is in progress. These criteria, designated level 1 and level 2, are described in the following paragraphs.

The criteria for vibration displacements are based on an assumed linear relationship between displacements, snubber loads, and magnitude of applied loads for any function and response of the system. Thus the magnitude of the limits of displacements, snubber loads, and nozzle loads, are all proportional. Maximum displacements, level 1 limits, are established to prevent the maximum stress in the piping systems from exceeding the normal and upset primary stress limits and/or the maximum snubber load from exceeding the maximum load to which the snubber has been tested.

Based on the above criteria, level 1 displacement limits are established for all instrumented points in the piping system. These limits are compared with the field measured piping displacements. The methods of acceptance are explained in the following paragraphs.

3.9.2.1.4.1 Level 1 Criteria

Level 1 criteria establish the maximum limits for the level of pipe

motion, which, if exceeded, would make 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 will be repeated to verify that the requirements of the level 1 limits are satisfied.

3.9.2.1.4.2 Level 2 Criteria

If the level 2 criteria are satisfied for both steady state and operating transient vibrations, there will be no fatigue damage to the piping system due to steady state vibration; and all operating transient vibration will be bounded by the values in the stress report.

If the pipe motions specified by level 2 criteria are exceeded, the responsible piping design engineer will be advised. Plant operational and startup testing plans will not necessarily be altered. Investigations of the measurements, criteria, and calculations used to generate the pipe motion limits will be initiated. An acceptable resolution must be reached by all appropriate and involved parties, including the responsible piping design engineer. Detailed evaluation will be needed to develop corrective action or to show that the measurements are acceptable. Depending upon the nature of the resolution, the applicable tests may or may not be repeated.

3.9.2.1.4.3 Acceptance Limits

For steady state vibration, the pipe break stress due to vibration only (neglecting pressure) will not exceed 10,000 psi for level 1 criteria and 5,000 psi for level 2 criteria. These limits are below the piping material's fatigue endurance limits as defined by the design fatigue curves for 10^6 cycles in Appendix I of the ASME B&PV Code.

For operating transient vibration, the piping bending stress (zero to peak) due to an operating transient only will not exceed 1.2S or pipe support loads will not exceed the service level D ratings for level 1 criteria. The 1.2S limits ensures that the total primary stress, including pressure and dead weight, will not exceed $1.8S_m$, the new service level B limit. Level 2 criteria are based on pipe stresses and support loads not exceeding design basis predictions. Design basis criteria require that operating transient stresses and loads not exceed any of the service level B limits including the primary stress limits, the fatigue usage factor limits, and the allowed loads on snubbers.

3.9.2.1.5 Corrective Actions for Main Steam and Recirculation Piping Systems

During the course of the tests, the remote measurements are regularly checked to determine compliance with level 1 criteria. If trends indicate that level 1 criteria may be exceeded, the measurements are monitored at more frequent intervals. The tests are placed on hold or terminated as soon as level 1 criteria are exceeded. As soon as possible after the test hold or termination, the following corrective actions are taken:

1. Installation inspection - A walkdown of the piping and suspension is made to identify any obstruction or improperly operating suspension components. Snubbers are installed to about the midpoint of their total travel range at operating temperature. Hangers are in their operating range between the hot and cold settings. If vibration exceeds criteria, the source of the excitation is identified to determine if it is related to equipment failure. Action is taken to correct any discrepancies before repeating the test.

2. Instrumentation inspection - The instrumentation installation and calibration are checked and any

discrepancies corrected. Additional instrumentation is added, if necessary.

3. Repeat test - If actions 1. and 2. above identify discrepancies that could account for failure to meet level 1 criteria, the test is repeated after correction of the discrepancies.

4. Resolution of findings - If the level 1 criteria are violated on the repeat test or no relevant discrepancies are identified in actions 1. and 2. above, the organization responsible for the stress report reviews the test results and criteria to determine if the test can be safely continued.

If the test measurements indicate failure to meet level 2 criteria, the following corrective actions are taken after completion of the test:

1. Installation inspection - A walkdown of the piping and suspension is made to identify any obstruction or improperly operating suspension components. If vibration exceeds limits, the source of vibration is identified. Action, such as suspension adjustment, is taken to correct any discrepancies.

2. Instrumentation inspection - The instrumentation installation and calibration are checked and any discrepancies corrected.

3. Repeat test - If actions 1. and 2. above identify a malfunction or discrepancy that could account for failure to comply with level 2 criteria, and appropriate corrective action has been taken, the test may be repeated.

4. Documentation of discrepancies - If the test is not repeated, the discrepancies found under actions 1. and 2. above are documented in the test evaluation report and correlated with the test condition. The tests are not considered complete until the test results are reconciled with the acceptance criteria.

3.9.2.1.6 Measurement Locations for Main Steam and Recirculation Piping

Remote shock and vibration measurements are made in the three orthogonal directions near the first downstream SRV 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 recirculation piping is made, and any visible vibration is measured with a hand held instrument.

For each of the selected remote measurement locations, level 1 and 2 deflection limits are prescribed in the startup test specification. The exact location of measuring devices and visual inspection points will be supplied in the test program. Measurements taken at these points will show if the stress and fatigue limits are within acceptable levels.

3.9.2.2 Preoperational and Startup Testing of Non-NSSS Piping

3.9.2.2.1 Scope

The test program is organized on the basis of three of the main types of non-NSSS pipe loads: thermal expansion, steady state pipe vibration, and dynamic transient pipe vibration. Testing under earthquake or postulated accident conditions is not included. The scope of piping to be considered in testing includes:

1. All ASME B&PV Code Class 1, 2, and 3 piping systems, other than NSSS piping

2. Other high energy piping within Seismic Category I structures and selected piping systems in the turbine building

3. High energy portions of any system whose failure could reduce the functioning of any Seismic Category I system to an unacceptable safety level

4. Seismic Category I portions of moderate energy piping systems located outside primary containment.

A list of all non-NSSS piping systems within this scope is provided in Table 3.9-5. This table identifies the preoperational or power ascension test category applicable for each system. Where appropriate, remarks indicate the reasons for including or excluding testing.

A test specification covers test activities. It provides, in detail, the test scope, the purpose, the exact location of measuring devices and visual inspections points, and the instrumentation requirements. Criteria for acceptance of test measurement data are also provided in the specification. Cognizant design personnel familiar with the systems to be tested develop the test plans, witness the test, and evaluate the test results. The data acquired from the tests are compared with the expected results to determine the acceptability of the system response.

3.9.2.2.2 Thermal Expansion Testing of Non-NSSS Piping

Piping thermal expansion tests are performed for safety related piping systems with a normal operating temperature that exceeds 300°F. Operating experience has shown that piping systems with an operating temperature of less than 300°F do not warrant thermal expansion tests. Engineering review of all Seismic Category I piping systems is performed after completion of construction and prior to fuel load. Supports, restraints, and snubbers are reviewed to ensure that normal thermal movement is not restrained as a result

of interferences or obstructions. In the tests, pipe thermal movements are monitored to confirm that restraint does not occur as a result of interference or obstruction at locations other than at the designed restraint locations. Free thermal expansion movement at snubbers is verified in the tests by data evaluation for remotely monitored piping, and by monitoring compression or extension travel of snubbers for visually checked piping.

Acceptance criteria for thermal expansion tests are based on two requirements:

1. Restraint of thermal expansion movement must be compatible with design restraint conditions

2. Thermal expansion movements must agree with calculated movements within the tolerance band prescribed in the test specification.

Corrective actions to be taken in instances where thermal expansion acceptance criteria are not satisfied are:

1. Identification of the location and cause of the abnormal restraint of thermal movement

2. Initiation of administrative procedures to eliminate the abnormal restraint

3. Repetition of the test for demonstration of compliance with acceptance criteria.

Non-NSSS piping systems included in preoperational and startup testing for thermal expansion movement are listed in Table 3.9-5.

3.9.2.2.3 Steady State Vibration Testing of Non-NSSS Piping Systems

Pipe vibration is the result of periodic application of forces induced by fluid flow oscillations or by equipment vibration. Such

vibration, if prolonged and at a high level, contributes to reduction of the fatigue life of the pipe system.

Steady state vibration level is categorized according to whether it is negligible or questionable. Negligible vibration amplitude is small enough that the judgement of the qualified test engineer is sufficient to determine that it is clearly acceptable. The acceptability of a questionable vibration is determined on the basis of one of the three acceptance procedures noted below.

The acceptance criterion for steady state vibration is that the maximum vibratory stress in the pipe does not exceed one-half the endurance limit for 10^6 cycles of vibration. Within that limit, vibration does not contribute to reduction of piping fatigue life. Compliance with the acceptance criterion is judged on the basis of criteria obtained from one of the following sources:

1. Prior operating experience with identical systems

2. General acceptance data based on operating experience with similar systems and/or generalized stress analysis calculations

3. Specific and detailed stress calculations based on computer model representation of the system operation with the vibration modes identified from the test data.

Vibration test evaluations are made under three different conditions:

1. When piping is inaccessible during power ascension, but operating conditions are duplicated under preoperational test conditions

2. When piping is accessible during power ascension testing

3. When piping is inaccessible during power ascension testing and the operating conditions are not duplicated under preoperational test conditions.

3.9.2.2.3.1 Preoperational Testing of Inaccessible Non-NSSS Piping for Steady State Vibration

The purpose of this preoperational vibration testing is to verify acceptable vibration of piping that is inaccessible during startup testing but that can be tested, before fuel load, under the same operating conditions that occur after fuel load. Amplitude and frequency measurements, if required, are obtained by hand held or rigidly attached instrumentation capable of the required accuracy over the expected range of frequency. The measurement system incorporates filtering devices capable of resolution of multiple mode vibrations.

The piping is visually examined by a qualified test engineer familiar with the structural design and operation of the piping. The test engineer makes a judgement as to whether the vibration observed during the visual examination is negligible. If it is not negligible, it is considered questionable and requires further measurement and interpretation. Further testing consists of such vibration measurements as the test engineer determines necessary to ensure that the vibration amplitude is within acceptable limits.

3.9.2.2.3.2 Power Ascension Testing of Accessible Non-NSSS Piping for Steady State Vibrations

This piping is accessible to the test personnel during plant operating conditions. The purpose of startup testing of accessible piping is to verify that the steady state vibration during any operating mode does not exceed the acceptable limits.

The piping is visually examined by a qualified test engineer familiar with the structural design and operation of the piping. Vibration that, based on this visual examination, the test engineer

does not judge negligible, is further examined. Further testing consists of such vibration measurements as the test engineer determines necessary to ensure that vibration amplitude is within acceptable limits.

Amplitude and frequency measurements, if required, are obtained by hand held or rigidly attached instrumentation capable of the required accuracy over the expected range of frequency. The measurement system incorporates filtering that enables the resolution of multiple mode vibrations.

3.9.2.2.3.3 Power Ascension Testing of Inaccessible Non-NSSS Piping for Steady State Vibrations

Piping is not always accessible to personnel during plant operation due to prohibitive radiation conditions. The purpose of startup testing of inaccessible piping is to verify that the steady state vibration during any operating mode does not exceed acceptable limits.

Because this piping is not always accessible to test engineers during startup testing, only limited, if any, walkdown visual survey is made. Remote readout instrumentation is rigidly installed at selected locations on the piping to provide vibration measurements during all startup test conditions and within the frequency and amplitude range of any expected vibration mode.

Qualified design and stress analysis personnel select the instrument locations needed to monitor vibration. They determine acceptable limits of maximum measured vibration for each system, and later evaluate the acceptability of questionable instrument vibration test measurements.

Qualified instrumentation specialists design the instrumentation installation hardware, to ensure that spurious vibration is not generated in operation. They select signal conditioning, calibration, and recording equipment capable of producing vibration

test records of the accuracy and range prescribed in the test specification.

Qualified test engineers perform the test measurements in conformance with the test procedures, making determinations of vibration as either negligible or questionable. Design engineers familiar with the piping system dynamics and with the acceptance criteria make determinations of questionable vibration as either acceptable or unacceptable, recommending specific corrective actions to reduce or eliminate excessive vibration.

3.9.2.2.4 Startup Dynamic Transient Tests of Non-NSSS Piping

During plant startup, dynamic transient tests are performed for the following piping systems and the indicated modes of operation:

1. Main steam piping outside the primary containment for main steam turbine trip at 25 percent, 75 percent, and 100 percent power

2. Main steam bypass piping for main stop valve closure

3. Main steam SRV discharge piping for main steam SRV opening

4. High pressure coolant injection (HPCI) turbine steam supply piping for HPCI turbine trip

5. Feedwater piping for reactor feed pump trip/coastdown.

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

Dynamic transient analysis of the subject lines is performed to determine the response of the system to 100 percent of the design condition loads. Pipe stress and deflection as well as the restraint design maximum load are determined.

During the test, a time history record of the load at selected restraints is obtained. Pipe dynamic pressure is measured at selected locations, and valve opening or closure time history is also measured. Pipe acceleration is measured at selected locations.

Acceptance criterion for dynamic transient testing is that the restraint maximum load measured in the test does not exceed the design calculated maximum load. Valve opening/closing stroke time is for information only, to be correlated with the piping design calculation. Pipe acceleration is not to exceed the design calculated maximum acceleration.

A test specification describes in detail the pipe system scope and objectives of the test. Instrumentation requirements and the acceptance limits for restraints and pipe acceleration are provided.

3.9.2.2.4.1 Corrective Action for Unacceptable Steady or Transient Vibration

Non-NSSS piping systems having vibration in excess of the acceptance limits are modified in accordance with plant procedures adopted to:

1. Modify the dynamic response of the piping system by addition, modification, or deletion of vibration restraints

2. Suppress the vibration by means of damping devices

3. Eliminate or reduce the source of the vibration.

Following adoption of any corrective measure, the piping system is again tested under the same conditions and evaluated for compliance with the acceptance criteria.

3.9.2.3 Seismic Qualification of Safety-Related NSSS Mechanical Equipment

This section describes the criteria for seismic qualification of safety-related mechanical equipment and the qualification testing and/or analyses applicable to this plant for all the major components on a component by component basis. In some cases, a module or assembly of mechanical and electrical equipment is qualified as a unit, e.g., the Emergency Core Cooling System (ECCS) pumps. These modules are generally discussed in this section. Seismic qualification testing for active pumps and valves is also discussed in Section 3.9.3.2. Electrical supporting equipment, such as control consoles, cabinets, and panels, that are part of the NSSS, are discussed in Section 3.10. The seismic test and/or evaluation results for safety-related mechanical equipment are maintained in a permanent file by GE and are readily auditable in all cases.

3.9.2.3.1 Tests and Analysis Criteria and Methods

The ability of equipment to perform its safety-related function during and after an earthquake is demonstrated by tests and/or analyses. Selection of testing, analysis, or a combination of the two is determined by the type, size, shape, and complexity of the equipment being considered. When practical, the safety-related operations are performed simultaneously with vibratory testing. Where this is not practical, operability is demonstrated by mathematical analysis.

The NSSS seismic qualification program for HCGS utilizes seismic data generated over a number of years. Since it was not a licensing requirement at the time, most of these data were developed in earlier years without pre-aging or sequential testing of the equipment. However, NSSS equipment located in harsh environments that has been qualified in recent years has generally been pre-aged and sequentially tested in accordance with the guidelines of IEEE 323-1974.

NSSS equipment on HCGS is being seismically evaluated using pre-aged and sequential testing data where it is available. Otherwise, the earlier data without pre-aging and sequential testing are being used.

The aging requirement is described in Section 3.11.2.7.2. Maintenance and surveillance program requirements given in Section 3.11.2.7.6 incorporate the results of testing, as applicable.

Equipment that is large, simple, and/or consumes large amounts of power is usually qualified by analysis or static test to show that the loads, stresses, and deflections are less than the allowable maximums. Analysis and/or testing are also used to show there are no natural frequencies below 33 hertz. If a natural frequency lower than 33 hertz is discovered, dynamic tests may be conducted and, in conjunction with mathematical analysis, used to verify operability and structural integrity at the required seismic input conditions.

When the equipment is qualified by dynamic test, the response spectrum or the time history of the attachment point is used in determining input motion.

Natural frequency may be determined by running a continuous sweep frequency search using a sinusoidal steady state input of low magnitude. Seismic conditions are simulated by testing using random vibration input or single frequency input within equipment capability at frequencies up to 33 hertz. Whichever method is used, the input motion during testing envelops the actual input motion expected during earthquake conditions.

The equipment being dynamically tested is mounted on a fixture that 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 seismic safe shutdown

earthquake (SSE) loads at the center of gravity of the extended structure. In cases where the equipment structural complexity makes mathematical analysis impractical, a static bend test is used to determine spring constant and operational capability at maximum equivalent seismic load conditions.

RPV and attached piping and pipe-mounted equipment are analyzed for annulus pressurization loads in the range of 60 to 100 Hz frequency depending on the dynamic characteristics of the equipment and its installation. The effect of hydrodynamic loads is limited to the torus and torus attached piping in accordance with the Mark I Containment Long-Term Program (NUREG 0661). The qualification test frequencies, in general, range up to 50 Hz, which is the upperbound hydrodynamic loading frequency.

Non-ASME B&PV code components are qualified by tests that address the "strong motion" phase of seismic (and, if applicable, SRV) dynamic motion sufficient to generate the maximum equipment response. This testing generally consists of five OBE tests and one SSE test of 30 seconds each. Non-ASME B&PV code components are also qualified by analyses that have not considered vibration fatigue-cycle effects.

Some equipment is shown to be qualified by single axis and/or single frequency testing. However, all essential equipment is reevaluated for seismic qualification according to the requirements or recommendations of IEEE 344-1975, Regulatory Guides 1.92 and 1.100, and Standard Review Plans 3.9.2, 3.10, and HCGS specific requirements.

In most instances, use of single axis test data is restricted to equipment with a response that shows a predominant single mode of vibration in each direction with minimal cross coupling. In some cases, if the response shows a single mode of vibration in each direction but also has cross coupling, the existing single axis test data are still used if the test response spectra (TRS) can be shown

to exceed the required response spectra (RRS) by a factor of 1.4 over all frequencies.

In most instances, use of single frequency test data is restricted to cases where the required input motion is dominated by one frequency, where response of the equipment is adequately represented by one mode, or where the input motion has sufficient intensity and duration to produce sufficiently high levels of stress to assure structural integrity where structural integrity is the determinant requirement. In some cases, if the input motion is sufficiently high so as to excite secondary modes, such that modal responses can be shown to occur out of phase and at high enough levels, existing single frequency test data are also used to demonstrate operability.

The determination of which dynamic loads to address in a qualification program is made on the basis of both load evaluations made on similar designed facilities and on plant specific assessments. From this basis, those loads which are considered to be significant are then selected and used in the qualification demonstration program. As described in the NRC approved NEDE-24326-1-P operational aging, vibration aging for pipe mounted equipment, applicable dynamic event aging, etc, are all considered. Specific loads, such as those generated for the sudden closure of valves, have been considered when they are determined to be critical (i.e., loads from the closing of the SRVs and turbine stop valve are considered, but loads from the closure of a MSIV are not because of the relatively slow closure time of the MSIV).

Vibration fatigue cycle effects for NSSS equipment designed to ASME B&PV Code requirements are evaluated in a manner found satisfactory to NRC consultants. The approach taken encompasses OBE, SRV where applicable, thermal, and pressure cycles (see References 3.9-18, 3.9-19 and 3.9-20).

Table 3.9-25 (SQRT devices) provides a listing of typical NSSS equipment showing the methods used for their qualification.

3.9.2.3.1.1 Random Vibration Input

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

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

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

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

3.9.2.3.1.2 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 is such that a purely rectilinear resultant input is avoided.

3.9.2.3.1.3 Fixture Design

The fixture design simulates the actual service mounting and causes no dynamic coupling to the equipment.

3.9.2.3.1.4 Prototype Testing

Testing is conducted on prototypes of the equipment installed in this plant.

3.9.2.3.2 Seismic Qualification of Specific NSSS Mechanical Components

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

3.9.2.3.2.1 Jet Pumps

A static analysis of the jet pumps is performed assuming 3.0g horizontal acceleration and 1.5g vertical acceleration. The stresses resulting from the analysis are below the design allowables. Static analysis with an appropriate amplification factor is used in lieu of dynamic analysis since the jet pump is a simple component with a natural frequency of slightly less than 33 hertz.

3.9.2.3.2.2 Control Rod Drive and Control Rod Drive Housing

The seismic qualification of the control rod drive (CRD) housing, with the CRD enclosed, for an operating basis earthquake (OBE) and a safe shutdown earthquake (SSE), is done analytically, and the stress results of the analyses establish the structural integrity of these components. Preliminary tests were conducted to verify the operability of the CRD during a seismic event. A simulated test, imposing a static bow in the fuel channels, was performed to show the CRD function satisfactorily.

3.9.2.3.2.3 Core Support - Fuel Support and Control Rod Guide Tube

No dynamic testing of the control rod guide tube is conducted. However, a detailed analysis imposing dynamic effects due to seismic

events shows that the maximum stresses developed during these events are much lower than the maximum allowed for the component material.

3.9.2.3.2.4 Hydraulic Control Unit

The seismic loads adequacy of the hydraulic control unit (HCU), for the faulted condition, is demonstrated by test and analysis. With the HCUs mounted on a seismic support structure, the dynamic loads are 1.8g vertical at the natural frequency of 7 to 30 hz, and 1.75g horizontal at 2 to 6 hz and 4g horizontal at 10 hz. At these frequencies, the maximum HCU capability (by test) for dynamic loads is 20g vertical at 7 to 30 hz, and greater than 4g horizontal at 2 to 6 hz and 8g horizontal at 10 hz.

3.9.2.3.2.5 Fuel Assembly (Including Channel)

Refer to Section 3.9.1.4.10.

3.9.2.3.2.6 Recirculation Pump and Motor Assembly

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

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

3.9.2.3.2.7 Emergency Core Cooling System Pump and Motor Assembly

The qualification of ECCS pump and motor assemblies as a unit, while operating under SSE conditions, is provided in the form of a static

earthquake acceleration analysis. The maximum specified vertical and horizontal accelerations are constantly applied simultaneously, and the worst case combination in the results of the analysis indicate the pump is capable of sustaining the above loading without overstressing the pump components.

A similar motor design is seismically qualified via a combination of static analysis and dynamic testing. The complete motor assembly is seismically qualified via dynamic testing, in accordance with IEEE 344-1975. The qualification test program includes demonstration of startup and shutdown capabilities, as well as no-load operability during seismic loading conditions.

For static analysis on a similar motor design, the seismic forces of each component or assembly are obtained by concentrating its mass at the center of gravity of component or assembly, and multiplying by the seismic acceleration, the earthquake coefficient. The magnitude of the earthquake coefficients are 0.14g vertical and 1.5g horizontal for the SSE condition.

3.9.2.3.2.8 Reactor Core Isolation Cooling Pump Assembly

The reactor core isolation cooling (RCIC) pump construction is of a barrel type on a large cross section pedestal. Qualification by analysis is performed. The seismic design analysis is based on 1.5g horizontal and 0.14g vertical accelerations. Results are obtained by using acceleration forces acting simultaneously in two directions: one vertical and one horizontal. The pump mass, support system, and accessory piping are shown, by analysis, to have a natural frequency greater than 33 hertz.

The RCIC pump assembly is analytically qualified by static analysis for seismic loading as well as the design operating loads of pressure, temperature, and external piping loads. The results of this analysis confirm that the stresses are substantially less than 90 percent of the allowable.

The RCIC pump has been analytically qualified by static analysis. The following operability statement appears as Note 2 on Table 3.9-4t. Static analysis on a similar type pump, for the emergency or faulted conditions, shows that the maximum shaft deflection is 0.002 in. with 0.006 in. allowable; shaft stresses are 3080 psi with 25,000 psi allowable; and, bearing loads for the drive end are 98 lb, with 7670 lb allowable. Bearing loads for thrust end are approximately 765 lb, with 17,600 lb allowable.

The RCIC pumps were installed in the pump manufacturer's closed test loop and subjected to an hydraulic performance test. The pump was driven with an electric motor in the test speed range of 3585 to 3590 rpm. All test setups, test procedures, and instrumentation were in accordance with the standards of the Hydraulic Institute and the ASME Power Test Code 8.2. Several points of data were taken to accurately determine the performance of the pump and to satisfy all the requirements listed in the HCGS pump data sheets.

3.9.2.3.2.9 Reactor Core Isolation Cooling Turbine Assembly

The RCIC turbine is seismically qualified by static analysis. The turbine assembly and its components are considered to be supported as designed, and horizontal/vertical accelerations are applied to the mass centers of gravity. The magnitude of the acceleration coefficient is 1.5g horizontal and 0.48g vertical. The results of the analysis indicate the turbine assembly is capable of sustaining the above loadings without overstressing any components.

3.9.2.3.2.10 Standby Liquid Control Pump and Motor Assembly

The standby liquid control (SLC) positive displacement pump and motor assembly is mounted on a common base plate and is qualified by static analysis.

The seismic design analysis is based on 1.5g horizontal and 0.14g vertical accelerations. Results are obtained by using acceleration forces acting simultaneously in two directions: one vertical and one horizontal. The pump/motor/base assembly is shown by static analysis to have a natural frequency greater than 33 hertz. The SLC pump and motor assembly is analytically qualified by static analysis for seismic loading as well as the design operating loads of pressure, temperature, and external piping loads. The results of this analysis confirm that the stresses are substantially less than 90 percent of the allowable.

The SLC pump has been qualified by static analysis. Coefficients used included 1.5g horizontal and 0.14 g vertical for safe shutdown earthquake (SSE). A subsequent seismic analysis for the SLC pump and motor has been accomplished using coefficients of 1.75g for both horizontal and vertical loads. Although these analyses are not specifically applied to the HCGS SLC pump and motor, the results are applicable due to equipment similarity, which is based on physical configuration, materials of construction, weight of equipment, base plate size, model number, allowable nozzle loads, and operating functions. A comparison evaluation showed the equipment to be alike in all respects.

An SLC motor, dimensionally identical to the HCGS SLC motors, has been subjected to dynamic testing in the no-load condition. The test-motor insulation materials differ from those used in the HCGS motors, but this difference was not sufficient to change the loads evaluation. With the test motor bolted to a vibration excitation table, a resonance search was performed in a frequency range of 10-80-10 Hz with a sweep of 10 Hz per minute. Input acceleration for the resonance search was 0.2g peak. There was no detectable resonant frequency within the frequency range tested.

The test motor was also subjected to an operational basis earthquake (OBE) test with the test motor in the operational no-load mode. The OBE test consisted of a minimum of 20 seconds of applied vibratory

motion at a frequency of 33 Hz with an acceleration level of 1.4g peak for periods of 20 seconds each at zero-degree and 180-degree phases. Testing was completed with no adverse effects noted:

A simulated SSE test was performed at an applied frequency of 33 Hz and an acceleration level of 2.0g peak. Tests were performed with the vertical and horizontal axes in phase and then 180 degrees out of phase at each of the following conditions:

a. a 15-second run was performed in each axis configuration with the test specimen in the nonoperating mode.

b. The seismic vibration was started for two seconds, and then the test specimen was energized for a period of 15 seconds in each axis configuration.

c. At the same time, the seismic vibration was started and the specimen was energized, and the test was continued for 15 seconds in each axis configuration.

d. The test specimen was energized and after three seconds, the seismic vibration was started and run for 15 seconds.

An operational test was performed for 125 minutes with the specimen loaded to 40 horsepower using a dynamotor. Operational data was recorded every 10 minutes. The 125-minute operating time complies with the maximum period required by the system design specification for the HCGS.

The SLC pump was installed in the pump manufacturer's closed test loop and subjected to a hydraulic performance test. Testing was in accordance with the reciprocating pump section of the Hydraulic Institute standards. With the pump speed relatively constant, data were accurately taken at six points to determine the performance of the pump and to satisfy all the requirements listed in the HCGS pump data sheet.

3.9.2.3.2.11 Residual Heat Removal Heat Exchangers

A dynamic analysis is performed to verify that the RHR heat exchanger withstands seismic loadings in accordance with its seismic classification. Seismic testing is an impractical method to verify the seismic adequacy of equipment when predictable seismic loads can be determined by dynamic and static analysis.

The heat exchanger, including its appurtenances and supports, is designed to withstand the effects of pressure, dead weight, nozzle loads due to attached piping, and seismic accelerations. The heat exchanger is analyzed for these loads using finite element techniques. The seismic accelerations used in the analyses are in the form of HCGS unique response spectra. The spectra are used as direct input to the dynamic finite element analyses.

3.9.2.3.2.12 Standby Liquid Control Tank

The SLC storage tank is a cylindrical tank 9 feet in diameter and 12-feet high, bolted to the concrete floor. Stresses are calculated readily by conventional methods. The magnitude of the earthquake coefficients for the SSE are 1.5g horizontal and 0.14g vertical. The SLC tank has been qualified by analysis for:

1. Stresses in the tank bearing plate

2. Bolt stresses

3. Sloshing loads imposed by an earthquake sloshing natural frequency of 0.58 hertz

4. Minimum wall thickness

5. Buckling.

3.9.2.3.2.13 Main Steam Isolation Valves

The main steam isolation valves (MSIVs) are qualified for operability by analysis and tests for seismic loading.

3.9.2.3.2.14 Main Steam Safety/Relief Valves

Due to the complexity of this structure and the performance requirements of the valve, the total assembly of the 2-Stage main steam SRV (including electrical and pneumatic devices) was dynamically tested at seismic accelerations equal to or greater than the SSE levels determined for this plant. Satisfactory operation of the valves is demonstrated during and after the test. Tests and analysis satisfy operability criteria, as defined in Section 3.9.2.3.

Seismic qualification of the 3-Stage SRVs was performed by analysis, by the manufacturer. The model used for analyses was benchmarked against actual testing performed for the Limerick 3-Stage SRV. The analysis was independently verified by a third party.

3.9.2.3.2.15 High Pressure Coolant Injection Turbine Assembly

The HPCI turbine is seismically qualified by static analysis. The turbine assembly and its components are considered to be supported as designed, and horizontal/vertical accelerations are applied to the mass centers of gravity. The magnitude of the acceleration coefficients is 1.5g horizontal and 0.48g vertical. The results of the analysis indicate the turbine assembly is capable of sustaining the above loadings without overstressing any components.

3.9.2.3.2.16 High Pressure Coolant Injection Pump Assembly

The HPCI pump assembly consists of a main pump, a gear reducer, and a booster pump. Both pumps are split body type, mounted on a common base plate. The assembly is seismically qualified by dynamic analysis using the response spectrum modal analysis technique. The structure's response at each of the lowest 60 modes is determined due to SSE seismic input in each of the three global directions. The total SSE seismic response (loads and deformations) is then determined by superposition, using the method of Regulatory

Guide 1.92, Revision 1. The pump mass support system and accessory piping have been shown by analysis to have a natural frequency greater than 60 hertz.

The HPCI pump assembly has been seismically qualified by a dynamic analysis performed by Byron Jackson Pump Company. A three dimensional, finite element model was developed and dynamically analyzed using the response spectrum method. Static nozzle loads, pump thrust loads and dead weight were considered. Critical location stresses were evaluated and compared with the allowable stresses based on Section III of the ASME Code.

The HPCI and RCIC pumps were installed in the pump manufacturer's closed test loop and subjected to a hydraulic performance test. The pump was driven with an electric motor in the test speed range of 3585 to 3590 rpm. All test setups, test procedures, and instrumentation were in accordance with the standards of the Hydraulic Institute and the ASME Power Test Code 8.2. Several points of data were taken to accurately determine the performance of the pump and to satisfy all the requirements listed in the HCGS pump data sheets.

3.9.2.4 Seismic Qualification Testing of Safety-Related Non-NSSS Mechanical Equipment

3.9.2.4.1 Seismic Qualification Criteria

All non-NSSS Seismic Category I mechanical equipment is designed to withstand the simultaneous horizontal and vertical accelerations caused by the OBE and the SSE in conjunction with other normal operating loads.

Seismic qualification criteria used for the Seismic Category I mechanical equipment, with the exception of pumps and active valves, are in compliance with Regulatory Guide 1.100 and IEEE 344-1975. The seismic qualification of pumps and active valves is discussed more fully in Section 3.9.3.2.

Where applicable, all equipment is pre-aged prior to seismic testing as part of the test sequence. The aging requirement is described in Section 3.11.2.7.2. Maintenance and Surveillance program requirements given in Section 3.11.2.7.6 incorporate the results of testing, as applicable.

The criteria for selecting a qualification method, by analysis and/or by test, is based on the practicality of the method for the function, type, size, shape, and complexity of the equipment.

Table 3.9-7 list all non-NSSS Seismic Category I mechanical equipment, equipment locations and qualification methods.

3.9.2.4.2 Methods and Procedures for Qualifying Non-NSSS Mechanical Equipment

Seismic Category I equipment is shown to be capable of withstanding the horizontal and vertical accelerations of five OBEs and one SSE by dynamic analysis, dynamic testing, or a combination of dynamic analysis and testing.

The seismic qualification methods and procedures are in compliance with the requirements of IEEE 344-1975 and Regulatory Guide 1.100.

Pipe mounted equipment is qualified by analysis and/or testing to the acceleration levels allowed for piping systems. These levels include gravity and operation loading, as well as loading that is due to seismic or any other accident related excitation, if applicable.

The plant operating vibration loads are insignificant compared to seismic loads considered for equipment qualification. However, applicable transient loads caused by sudden valve actuation (e.g., main steam turbine trips, HPCI turbine stop valve closure, MSSRV discharge, etc.) are considered in the design loading of non-NSSS ASME components as specified in Table 3.9-8. Force time histories of transient loads are developed using one of the computer codes referenced in Section 3.9.1.2. These forcing functions are then input to the finite element piping analysis along with the applicable seismic response spectra. The combined seismic and transient piping responses are evaluated against the equipment allowables specified for the appropriate service level. Selected systems are subsequently subjected to inplant dynamic transient testing to confirm the acceptability of the analysis.

All pipemounted valve operators and accessories are qualified by using a single axis, single frequency testing (required input motion (RIM) test). This is justified on the ground that the seismic floor motion is filtered through the piping system, which generally has one predominant structural mode. Thus the resulting motion that reaches the linemounted equipment is predominantly a single frequency and singleaxis motion. The test is performed by using RIM in each of the three axes, independently.

In accordance with the Mark I Containment Long-Term Program (NUREG-0661), non-NSSS equipment attached to the torus has been evaluated for appropriate hydrodynamic loads, including fatigue effects.

Wetwell to drywell vacuum breakers inside the torus are also qualified for hydrodynamic loads for frequencies up to 50 hz.

3.9.2.4.2.1 Dynamic Analysis

Dynamic analysis without testing is used if structural integrity alone ensures the intended design function. Included is mechanical

equipment such as tanks and vessels, ductwork, heat exchangers, filters, and inactive valves.

The methods and procedures for the seismic analysis of Seismic Category I equipment are discussed in Section 3.7.3. For equipment such as pumps, rotational analysis is used to qualify heavy rotating machinery items, where it must be verified that deformations due to seismic loading will not cause binding of the rotating element to the extent that the component cannot perform its intended design function.

3.9.2.4.2.2 Dynamic Testing

Dynamic testing is used for equipment that requires confirmation of operability during and after seismic events. Loadings include the OBE, the SSE, and all static and dynamic loads. Included is mechanical equipment such as fans, pumps, and valve actuators.

Dynamic testing is performed by subjecting equipment to vibratory motions that conservatively simulate the required response spectrum or the required input motion at the equipment-mounting location.

Equipment is tested in the operational condition. Operability is verified during and/or after the testing, as applicable to the equipment being tested. The requirements of testing procedures and methods are in accordance with Section 6 of IEEE 344-1975. The test results have demonstrated that the test response spectrum closely resembles and envelops the required response spectrum over the critical frequency range.

3.9.2.4.2.3 Combined Analysis and Testing

Equipment that cannot be qualified practically by analysis or testing because of its size and/or complexity is qualified by combined analysis and testing. Combined analysis and testing methods are in accordance with Section 7 of IEEE 344-1975.

3.9.2.4.3 Methods and Procedures of Analysis or Testing of Supports of Mechanical Equipment

Analysis or testing of supports of Class 1E equipment and instrumentation is discussed in Section 3.10.3.2.

The design of supports of mechanical equipment is confirmed by analysis or test to ensure their structural capability to withstand dynamic excitation.

In general, equipment is tested or analyzed with equipment supports and base connections simulating the actual installation. Therefore, the seismic qualification methods and procedures for supports are similar to those for mechanical equipment, as discussed in Section 3.9.2.4.2. When the equipment and support are seismically qualified separately, the required response spectrum curves at the location of the equipment are produced to account for the possible seismic amplification between support and equipment.

3.9.2.4.4 Operating License Review

Qualification documents containing results of qualification tests and analyses for non-NSSS mechanical equipment are maintained by PSE&G in a centrally located, readily auditable, permanent file.

3.9.2.5 <u>Dynamic Response of Reactor Internals Under Operational Flow Transients and Steady State Conditions</u>

The major reactor internal components within the vessel are subjected to extensive testing coupled with dynamic system analysis to describe properly the resulting flow induced vibration phenomena incurred from normal reactor operation and from anticipated operational transients.

In general, the vibration forcing functions for operational flow transients and steady state conditions are not predetermined by detailed analyses. Special analyses of the response signals

measured for the reactor internals of many similar designs are performed to predict amplitude and modal contributions. Parametric studies, useful for extrapolating the results from tests of internals and components of similar designs, are also performed. This vibration prediction method is appropriate where standard hydrodynamic theory cannot be applied due to the complexity of the structure and flow conditions. Elements of this vibration prediction method are as follows:

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

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

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

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

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 from paragraph 1. above.

The dynamic modal analyses also form the basis for interpretation of the prototype plant preoperational and initial startup test results, as discussed in Section 3.9.2.6. Modal stresses are calculated and relationships are obtained between sensor response amplitudes and peak component stresses. The allowable amplitude is that which produces a peak stress of ~10,000 psi.

3.9.2.6 Preoperational Flow Induced Vibration Testing of Reactor Internals

Hope Creek Generating Station reactor internals are tested in accordance with the provisions of Regulatory Guide 1.20, Revision 2, for nonprototype Category I plants. The test procedure requires operation of the recirculation system at rated flow with internals important to safety installed, followed by inspection for evidence of vibration, wear, or loose parts. Blade guides, in-core instruments, neutron sources, steam dryer, and fuel are not installed. Control rods are either not installed or are fully withdrawn and prevented from being inserted. The test duration is sufficient to subject critical components to at least 10^6 cycles of vibration during two loop and single loop operation of the recirculation system. Upon completion of the flow test, the vessel head and shroud head are removed, the vessel drained, and major components inspected on a selected basis. The inspection covers all components that were examined on the prototype design, including the shroud, shroud head, core support structures, the jet pumps, and the peripheral CRD and in-core guide tubes. Access is provided to the reactor lower plenum.

Reactor internals design configurations for the Hope Creek Generating Station are substantially the same as those that have been tested in prototype BWR/4 plants. Results of the prototype tests are presented in a licensing topical report, Reference 3.9-12. This report also contains additional information on the confirmatory inspection program.

3.9.2.6.1 Compliance With Regulatory Guide 1.20

GE supplied NSSS analyses, design, and equipment used in this facility are in compliance with the intent of Regulatory Guide 1.20, through the incorporation of the alternate approach cited below.

Regulatory Guide 1.20 describes a comprehensive vibration assessment program for reactor internals during preoperational and initial startup testing. This regulatory guide is applicable to the core support structure and other reactor internals. The vibration assessment program meets the requirements of Criterion 1, Quality Standards and Records, of Appendix A to 10CFR50 and Section 50.34, Contents of Applications; Technical Information, of 10CFR50.

Vibration testing of reactor internals is performed on all GE BWR plants. At the time of the original issue of Regulatory Guide 1.20, test programs for compliance were instituted. The first BWR/4 plant of each size was considered a prototype and was instrumented and subjected to preoperational and startup flow testing to demonstrate that flow induced vibrations similar to those expected during operation do not cause damage. Subsequent plants that have reactor internals similar to those of the prototypes have also been tested, in compliance with the requirements of Regulatory Guide 1.20.

GE is committed to confirm satisfactory vibration performance of reactor internals through preoperational flow testing, followed by inspection for evidence of excessive vibration. Extensive vibration measurements in prototype plants, together with satisfactory operating experience in all BWR/4 plants, have established the adequacy of BWR/4 reactor internals designs.

3.9.2.7 Dynamic System Analysis of Reactor Internals Under Faulted Conditions

To ensure that no significant dynamic amplification of load occurs as a result of the oscillatory nature of the blowdown forces, a comparison is 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 are determined from a comprehensive dynamic model of the reactor pressure vessel (RPV) and its internals with 12 degrees of freedom. Only motion in the vertical direction is considered here; hence, each structural member (between two mass points) can only have an axial load. Besides the real masses of the RPV and core support structures, account is taken of the water inside the RPV.

The accident analysis method is described in Sections 3.9.5.2 and 3.9.5.3. The time varying pressures are applied to the dynamic model of the reactor internals described above. Except for the nature and locations of the forcing functions and the dynamic model, the dynamic analysis method is identical to that described for seismic analysis and is detailed in Section 3.7.3.1. The dynamic components of forces from these loads are combined with dynamic force components from other dynamic loads (including seismic), all acting in the same direction, by the square root of the sum of the squares (SRSS) method. This resultant force is then combined with other steady state and static loads on an absolute sum basis to determine the design load in a given direction.

The loads and load combinations acting upon the jet pumps and low pressure coolant injection (LPCI) coupling are listed on Tables 3.9-4y and 3.9-4z, respectively. Reactor asymmetric loads analysis is described in Appendix 3C.

3.9.2.8 Correlations of Reactor Internals Vibration Tests With the Analytical Results

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

Such comparisons provide the analysts with added insight into the dynamic behavior of the reactor internals. The additional knowledge gained is used in the generation of the dynamic models for seismic and loss-of-coolant accident (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, Browns Ferry-1.

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

3.9.3 ASME B&PV 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 events for the design of safety-related ASME B&PV Code components, except primary containment components discussed in Section 3.8.

This section also lists the major ASME B&PV Code 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 are covered in Section 3.9.1.1. Design transients for ASME B&PV Code Class 2 equipment are not addressed in this section. Seismic related loads are discussed in Sections 3.9.2.3 and 3.7.

Table 3.9-4 presents the loading combination analytical methods (by reference or example), and the calculated stress or other design values for the most critical areas of the NSSS components, supports, and core support structures. These values are also compared to applicable allowable values in the ASME B&PV Code.

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 B&PV Code, Section III.

3.9.3.1.1.1 Normal Conditions

Normal conditions are any conditions in the course of system startup, operation in the design power range, normal hot standby (with the main condenser available), and system shutdown other than upset, emergency, faulted, or testing conditions.

3.9.3.1.1.2 Upset Conditions

Upset conditions are any deviations from normal conditions anticipated to occur often enough so that design should include the capability to withstand the conditions without operational impairment. The upset conditions include those transients that 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. Vibratory motions due to an operating basis earthquake (OBE) are conservatively treated as an upset condition. Hot standby with the main condenser isolated is an upset condition.

3.9.3.1.1.3 Emergency Conditions

Emergency conditions are those deviations from normal conditions that require shutdown for correction of the conditions or repair of damage in the reactor coolant pressure boundary (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:

1. A multiple safety/relief valve blowdown of the reactor vessel

2. 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 primary containment and reactor shutdown

3. Improper assembly of the core during refueling

4. Vibratory motions of an OBE in combination with associated system transients.

3.9.3.1.1.4 Faulted Conditions

Faulted conditions are 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 the following:

1. A control rod drop accident

2. A fuel handling accident

3. A main steam line break

4. A recirculation loop break

5. The combination of any pipe break plus the seismic motion associated with a safe shutdown earthquake (SSE) plus a loss of offsite power (LOP), and

6. An SSE.

3.9.3.1.1.5 Correlation of Plant Conditions with Event Probability

The range of probabilities of events occurring associated with the plant conditions are listed below. These correlations can be used to identify the appropriate plant condition for any hypothesized event or sequence of events.

Plant Conditions	Event Encounter Probability Per Reactor Year
Normal (planned)	1
Upset (moderate probability)	$1 > P > 10^{-2}$
Emergency (low probability)	$10^{-2} > P > 10^{-4}$
Faulted (extremely low probability)	$10^{-4} > P > 10^{-6}$

3.9.3.1.2 Reactor Pressure Vessel Assembly, Core Support Structures, and Reactor Internals

The reactor pressure vessel (RPV) assembly consists of the RPV, support skirt, and shroud support.

The RPV assembly is constructed in accordance with Section III of the ASME B&PV Code. The shroud support consists of the shroud support plate and the shroud support cylinder and its legs. The RPV is an ASME B&PV Code Class 1 component. Complete stress reports on these components have been prepared in accordance with ASME B&PV Code requirements. Table 3.9-4b summarizes stress criteria, loading combinations, and calculated and allowable stresses for each category of plant conditions. The stress analyses performed for the reactor vessel assembly, including the faulted condition, were completed using elastic methods. The stress load combinations and stress analyses for other core support structures and reactor internals are discussed in Section 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 RPV to the outboard main steam isolation valve (MSIV). This piping is designed in accordance with the ASME B&PV Code, Section III, Subsection NB-3600. The load combinations and stress criteria are shown in Table 3.9-4e.

3.9.3.1.4 Recirculation Loop Piping

The recirculation system piping that is bounded by the RPV nozzles is designed in accordance with the ASME B&PV Code, Section III, Subsection NB-3600. The load combinations and allowables are shown in Table 3.9-4g.

3.9.3.1.5 Recirculation System Valves

The recirculation system suction and discharge gate valves are designed in accordance with the ASME B&PV Code, Section III, Class 1, Subsection NB-3500. Loading combinations and other stress analysis information are presented in Table 3.9-41.

3.9.3.1.6 Recirculation Pumps

In the design of the recirculation pumps, the ASME B&PV Code, Section VIII, Division 1, with edition and addenda as specified in Table 3.2-2, is used as a guide in calculations made for determining the thickness of pressure retaining parts and in sizing the pressure retaining bolting.

The pump supplier's calculations for the design of the pressure retaining components include the determination of minimum wall thickness, allowable stress and pressures, and calculations for pressurized bolted flange covers and for sizing of bolting. Pertinent examples of these calculations are shown in Table 3.9-4k.

3.9.3.1.7 Standby Liquid Control Storage Tank

The loads considered in the design of the standby liquid control (SLC) storage tank and the categorization of these loads are as follows:

1. Pressure (atmospheric) - Normal/upset

2. Temperature (200°F) - Normal/upset

3. OBE (1/2 SSE) - Upset

4. Piping nozzle loads - Upset/faulted

5. SSE - Faulted.

The stress limits allowed by the ASME B&PV Code for the normal and upset conditions are 1.0 S for general membrane and 1.5 S for bending plus local membrane.

The stress limits allowed by the ASME B&PV Code for the faulted conditions are 1.2 S for general membrane and 1.8 S for bending plus local membrane.

A summary of the design calculations and methods used is given in Table 3.9-4o.

3.9.3.1.8 Residual Heat Removal Heat Exchangers

The loading combinations and other stress analysis information for the residual heat removal (RHR) heat exchangers is presented in Table 3.9-4q.

The design of the RHR heat exchanger is discussed in Section 3.9.2.3.2.11.

3.9.3.1.9 Reactor Core Isolation Cooling Turbine

Although not under the jurisdiction of the ASME B&PV Code, the Reactor Core Isolation Cooling (RCIC) System turbine is designed and fabricated following the basic guidelines for an ASME B&PV Code, Section III, Class 2 component.

1. The operating conditions for the RCIC turbine include:

    a. Surveillance testing - Monthly operation with reactor pressure at 1000 psia (nominal) and saturated temperature; turbine exhaust pressure at 25 psia (peak) and saturated temperature

    b. Auto-startup - 30 cycles per year with reactor pressure at 1150 psia (nominal) and saturated

temperature; turbine exhaust pressure at 25 psia (peak) and saturated temperature.

2. The design conditions for the RCIC turbine include:

   a. Turbine inlet - 1250 psig at saturated temperature

   b. Turbine exhaust - 165 psig at saturated temperature.

   c. The upset conditions that control the turbine design include:

      (1) Design pressure

      (2) Design temperature

      (3) OBE

      (4) Inlet and exhaust piping nozzle loads.

      The stress limits for the pressure boundary are the ASME B&PV Code allowable stresses, 1.0 S for general membrane and 1.5 S for bending plus local membrane.

   d. The faulted or emergency conditions include:

      (1) Design pressure

      (2) Design temperature

      (3) SSE

      (4) Inlet and exhaust piping nozzle loads.

      The stress limits for the pressure boundary are 120 percent of the ASME B&PV Code allowable stresses,

1.2 S for general membrane, and 1.8 S for bending plus local membrane.

Table 3.9-4s contains a summary of the calculated and allowable loads for the RCIC turbine components.

3.9.3.1.10 Reactor Core Isolation Cooling Pump

The RCIC pump is designed and fabricated to the requirements for an ASME B&PV Code, Section III, Class 2 component.

1. The operating conditions for the RCIC pump are surveillance tested in conjunction with the RCIC turbine. A monthly operational test is performed, during which the RCIC pump takes condensate from the condensate storage tank and, at design flow, discharges condensate back to the condensate storage tank via a closed test loop.

2. The design conditions for the RCIC pump include:

    a. Required net positive suction head (NPSH) - 20.3 feet

    b. Total head - High speed: 3052 feet

       - Low speed: 525 feet

    c. Constant flow rate - 625 gpm

    d. Ambient room conditions can be found in the Hope Creek Environmental Design Criteria (EDC), Document no. D7.5.

    e. The normal plus upset conditions that control the pump design include:

(1)	Design pressure	1500 psig
(2)	Design temperature	40 to 140°F
(3)	OBE	2/3 of SSE

(4)	Suction nozzle loads	F_o = 1940 pounds,
		M_o = 2460 foot-pounds
(5)	Discharge nozzle loads	F_o = 3715 pounds,
		M_o = 4330 foot-pounds

where:

F_o and M_o are as defined in Table 3.9-4t.

The stress limits for the pressure boundary are the ASME B&PV Code allowable stresses, 1.0 S for general membrane and 1.5 S for bending plus local membrane.

f. The faulted or emergency conditions include:

(1)	Design pressure	1500 psig
(2)	Design temperature	40 to 140°F
(3)	SSE	Horizontal - 1.5 g
		Vertical - 0.14 g
(4)	Suction nozzle loads	F_o = 2325 pounds,
		M_o = 2950 foot-pounds
(5)	Discharge nozzle loads	F_o = 4450 pounds,
		M_o = 5200 foot-pounds

The stress limits for the pressure boundary are 120 percent of the ASME B&PV Code allowable stresses, 1.2 S for general membrane and 1.8 S for bending plus local membrane.

Table 3.9-4t contains a summary of the design calculations and nozzle loads for the RCIC pump components.

3.9.3.1.11 Emergency Core Cooling System Pumps

This section discusses the RHR and core spray pumps. The High Pressure Coolant Injection (HPCI) System pump is discussed in a later section.

The design conditions for the 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	360 °F	148°F

1. Normal/upset condition:

   The design pressures are as tabulated above. The OBE seismic accelerations are 0.75g horizontal and 0.07g vertical. The stress limits for the pressure boundary are the ASME B&PV Code allowable stresses, 1.0 S for general membrane, and 1.5 S for bending plus local membrane.

2. Faulted or emergency condition:

   Design pressures are as tabulated above. The SSE seismic accelerations are 1.5g horizontal and 0.14g vertical. The stress limits for the pressure boundary are 120 percent of the ASME B&PV Code allowable stresses, 1.2 S for general membrane, and 1.8 S for bending plus local membrane.

Table 3.9-5p summarizes the design calculation for the RHR and core spray pumps.

3.9.3.1.12 Standby Liquid Control Pumps

The SLC pumps are designed and fabricated following the requirements for ASME B&PV Code, Section III, Class 2 components.

1. The operating conditions for each SLC pump and motor are functionally tested by pumping demineralized water through a closed test loop. Each SLC pump is capable of

injecting the net contents of the SLC tank into the reactor in not less than 50 minutes and not more than 125 minutes. Each pump is capable of injecting flow into the reactor against a pressure of zero psig up to the initial setpoint pressure of the reactor safety/relief valves (SRVs).

2. The design conditions for each SLC pump include:

   a. Flow rate: 43 gpm

   b. Available NPSH, maximum: 12.9 psi @ 110°F

   c. Maximum operating discharge pressure: 1255 psig

   d. Ambient room conditions can be found in the Hope Creek Environmental Design Criteria (EDC), Document no. D7.5.

   e. The normal plus upset conditions which control the pump design include:

(1)	Design pressure	1400 psig
(2)	Design temperature	150°F
(3)	OBE	0.75 g (Horiz.)
		0.07 g (Vert.)

   The stress limit for the pressure boundary is the ASME B&PV Code allowable stress, 1.0 S for general membrane.

   f. The faulted or emergency conditions include:

(1)	Design pressure	1400 psig
(2)	Design temperature	150°F

(3) SSE Horizontal - 1.5 g
Vertical - 0.14 g

The stress limits for the pressure boundary are 120 percent of ASME B&PV Code allowable stresses, 1.2 S for general membrane, and 1.8 S for bending plus local membrane.

A summary of the design calculations and nozzle loads for the SLC pump components is contained in Table 3.9-4n.

3.9.3.1.13 Main Steam Isolation and Safety/Relief Valves

Load combination analytical methods, calculated stresses, and allowable limits are shown for the SRVs and the MSIVs in Tables 3.9-4i and 3.9-4j, respectively.

3.9.3.1.14 Safety/Relief Valve Discharge Piping

See Section 3.9.3.1.20.

3.9.3.1.15 High Pressure Coolant Injection Turbine

Although not under the jurisdiction of the ASME B&PV Code, Section III, the HPCI turbine is designed and fabricated following the basic guidelines for an ASME B&PV Code, Section III, Class 2 component.

1. The operating conditions for the HPCI turbine include:

   a. Surveillance testing - Monthly operation with reactor pressure at 1000 psia (nominal) and saturated temperature; turbine exhaust pressure at 65 psia (peak) and saturated temperature

   b. Auto-startup - 30 cycles per year with reactor pressure at 1150 psia (nominal) and saturated

temperature; turbine exhaust pressure at 65 psia (peak) and saturated temperature.

2. The design conditions for the HPCI turbine include:

   a. Turbine inlet - 1250 psig at saturated temperature

   b. Turbine exhaust - 185 psig at saturated temperature

   c. The upset conditions that control the turbine design include:

      (1) Design pressure

      (2) Design temperature

      (3) OBE

      (4) Inlet and exhaust piping nozzle loads.

      The stress limits for the pressure boundary are the ASME B&PV Code allowable stresses, 1.0 S for general membrane, and 1.5 S for bending plus local membrane.

   d. The faulted or emergency conditions include:

      (1) Design pressure

      (2) Design temperature

      (3) SSE

      (4) Inlet and exhaust piping nozzle loads.

      The stress limits for the pressure boundary are 120 percent of the ASME B&PV Code allowable stresses,

1.2 S for general membrane, and 1.8 S for bending plus local membrane.

A summary of the design calculations for the HPCI turbine components is shown in Table 3.9-4dd.

3.9.3.1.16 High Pressure Coolant Injection Pump

The HPCI pump is designed and fabricated following the requirements for an ASME B&PV Code, Section III, Class 2 component.

1. The operating conditions for the HPCI pump are surveillance tested in conjunction with the HPCI turbine. A monthly operational test is performed, during which the HPCI pump takes condensate from the condensate storage tank (CST) and, at design flow, discharges condensate back to the CST via a closed test loop.

2. The Design conditions for the HPCI pump include:

   a. The required NPSH of the HPCI Booster Pump at speed 2093 rpm and flow 5920 gpm is 19.7 feet. [Reference Calculation BJ-0002, Rev. 6, Section 7.3]

   b. Total head - High speed: 3162 feet

      - Low speed: 1038 feet

   c. Constant flow rate - 5600 gpm

   d. Ambient room conditions can be found in the Hope Creek Environmental Design Criteria (EDC), Document no. D7.5.

   e. The normal plus upset conditions that control the pump design include:

(1)	Design pressure	1500 psig
(2)	Design temperature	40 to 140°F
(3)	OBE	2/3 of SSE

(4)	Suction nozzle loads	F_o = 5570 pounds,
		M_o = 15,370 foot-pounds
(5)	Discharge nozzle loads	F_o = 7850 pounds,
		M_o = 15,385 foot-pounds

where:

F_o and M_o are as defined in Table 3.9-4v.

The stress limits for the pressure boundary are the ASME B&PV Code allowable stresses, 1.0 S for general membrane, and 1.5 S for bending plus local membrane.

f. The faulted or emergency conditions include:

(1)	Design pressure	1500 psig
(2)	Design temperature	40 to 140°F
(3)	SSE	Horizontal - 1.50 g
		Vertical - 0.14 g
(4)	Suction nozzle loads	F_o = 6680 pounds,
		M_o = 18,450 foot-pounds
(5)	Discharge nozzle loads	F_o = 9420 pounds,
		M_o = 18,465 foot-pounds

where:

F_o and M_o are as defined in Table 3.9-4v.

The stress limits for the pressure boundary are 120 percent of the ASME B&PV Code allowable stresses, 1.2 S for general membrane, and 1.8 S for bending plus local membrane.

The calculated stress values are compared with allowable stresses for critical components in Table 3.9-4v.

3.9.3.1.17 Reactor Water Cleanup System Pumps

The Reactor Water Cleanup (RWCU) System pumps are not part of a safety system and are not required to meet Seismic Category I requirements.

The static analysis considers static equilibrium forces on the equipment including the effect of OBE loads. This analysis considers piping loads as well as torsional moment produced by the rotating assembly.

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

1. Normal plus upset loads include the simultaneous effect of normal operating loads, design pressure, temperature, nozzle loads, dead weight loads including seismic due to OBE loads, plus torsional load due to rotation of the component assembly.

2. The pump and its supports are designed to withstand the OBE loads applied at the mass center, assuming that the pump is flooded.

3. Stresses in the supports and the anchor bolts due to OBE 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 stresses set forth in the applicable codes.

4. The ASME B&PV Code, Section III, is used as a guide in calculating the thickness of the pressure retaining parts and for sizing the pressurized cover bolting.

5. Transient analysis is not required for Class 3 components operating in the 70 to 545°F temperature range.

Table 3.9-5r shows the calculated stress values and allowable stress limits for the pumps.

3.9.3.1.18 Fuel Pool Cooling and Cleanup System Heat Exchangers

See Section 3.9.3.1.20.

3.9.3.1.19 Reactor Water Cleanup System Heat Exchangers

The RWCU regenerative and nonregenerative heat exchangers are not part of a safety system and are not required to meet Seismic Category I requirements. However, a static seismic analysis is done on these heat exchangers. Static seismic forces of 0.2 g horizontal and 0 g vertical are used in this analysis.

The loadings considered in the design of the heat exchangers include:

1. Normal plus upset loads include the simultaneous effect of normal operating loads, design pressure, temperature, nozzle loads, and dead weight loads.

2. The heat exchangers and their supports are designed to withstand the static seismic forces applied.

3. 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 stresses set forth in the applicable codes.

4. The allowable shear on anchor bolts set in concrete are in accordance with Table Number 26-1 of the Uniform Building Code.

Table 3.9-4d shows the calculated stress values and allowable stress limits for the heat exchangers.

3.9.3.1.20 Non-NSSS ASME B&PV Code Constructed Items

The design loading combinations categorized with respect to plant operating conditions identified as normal, upset, emergency, and faulted for the non-NSSS ASME B&PV Code constructed items are presented in Table 3.9-7.

The design criteria and stress limits associated with each of the plant operating conditions for each type of ASME B&PV Code constructed item are presented in Tables 3.9-8 through 3.9-14.

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

3.9.3.2 NSSS Pump and Valve Operability Assurance

The NSSS active pumps are listed in Table 3.9-15 and the NSSS active valves are listed in Table 3.9-16. Table 3.9-26 lists examples of PVORT NSSS equipment qualification methodology.

Active mechanical equipment classified as Seismic Category I is designed to perform its function during the life of the plant under postulated plant conditions. Equipment with faulted condition functional requirements include active pumps and valves in fluid systems such as the RHR system and the core spray system. Active equipment must perform a mechanical motion during the course of accomplishing a safety function.

Periodic inspection and operational testing is performed as per the requirements in Section 16. See Section 3.9.6 for operational testing outline.

The only NSSS active valves subjected to hydrodynamic loads are the safety/relief valves (B21-F013) and the main steam isolation valves (B21-F022). Both of these valve types are being dynamically qualified by test up to 100 hz.

The load and conditions considered in the qualification of safety-related pumps and valves are given in Tables 3.9-4 and 3.9-4(a).

Deflections due to piping loads and dynamic loads are addressed for active essential pumps and valves by several methods depending on the situation. Methods used include static deflection analysis, dynamic deflection analysis, and dynamic seismic testing.

Operability is ensured by satisfying the requirements of the following programs. Safety-related active valves are qualified by prototype testing and analysis, and safety-related active pumps by analysis with suitable stress limits and nozzle loads. The content of these programs is detailed below.

3.9.3.2.1 Emergency Core Cooling System Pumps

All active ECCS pumps are qualified for operability by first being subjected to rigid tests, both prior to and after installation in the plant. The in-shop tests include:

1. Hydrostatic tests of pressure retaining parts to 125 percent of the design pressure (times the ratio of material allowable stress at room temperature to the allowable stress value at the design temperature)

2. Seal leakage tests

3. Performance tests, while the pump is operated with flow, determines total developed head, minimum and maximum head, and NPSH requirements.

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 inspection and operational testing per the requirements in Chapter 16. 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 an SSE condition by ensuring that the pump is not damaged during the seismic event, and that the pump will continue operating despite the SSE loads.

3.9.3.2.1.1 Analysis of Loading, Stress, and Acceleration Conditions

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 Section 3.9.3.1 and Table 3.9-5. The average membrane stress (0[16]fm) for the faulted condition loads is limited to 1.20[16]fm. The maximum stress in local fibers (0[16]fm + bending stress (0[16]fb)) is limited to 1.8 S.

The qualification of the pump and motor as an integral unit while operating under OBE and SSE conditions is provided in the form of a static earthquake-acceleration analysis. Under this criterion, the unit is considered to be supported as designed, and the maximum specified vertical and horizontal accelerations are constantly and simultaneously applied in the worst case combination. The maximum seismic nozzle loads from the attached piping system are also considered in an analysis of the pump support to ensure that there is no geometrical/dimensional deformation of the pump components.

3.9.3.2.1.2 Pump Operation During and Following Safe Shutdown Earthquake Loading

Active pump/motor rotor combinations are designed to rotate at a constant speed under all conditions. Motors are designed to withstand short periods of severe overload. The high rotary inertia in the operating pump rotor, and the nature of the random, short duration loading characteristics of the seismic event, prevent the rotor from becoming seized. In actuality, the seismic loadings cause only a slight increase, if any, in the torque, i.e., motor current, necessary to drive the pump at the constant design speed. Therefore, the pump does not shut down during the SSE and operates at the design speed, despite the SSE loads.

The functional ability of the active pumps after a faulted condition is ensured since only normal operating loads and steady state nozzle loads exist. For the active pumps, the faulted condition is greater than the normal condition due to seismic SSE loads on the equipment itself. 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 are not damaged during the faulted condition, the post-faulted condition operating loads will be no worse than the normal plant operating limits. This is ensured by requiring that the imposed nozzle loads (steady state loads) for normal conditions and post-faulted conditions are limited by the magnitudes of the normal condition nozzle loads. The ability of the pumps to function under these applied loads for post-faulted conditions is proven during the normal operating plant conditions for active pumps.

3.9.3.2.2 Standby Liquid Control Pump and Motor Assemblies and Reactor Core Isolation Cooling Pump Assembly

These equipment assemblies are small, compact, rigid assemblies with natural frequencies well above 33 hertz. With this fact verified, each equipment assembly is seismically qualified via static analysis only. This static qualification verifies

operability under seismic conditions and ensures that structural loading stresses are within ASME B&PV Code limitations.

3.9.3.2.3 Emergency Core Cooling System Motors

Qualification of the Class 1E motors used for the ECCS motors is in compliance with IEEE 323-1974. The qualification of all motor sizes is based on completion of a type test, followed with review and comparison of design and material details and seismic analysis of production units, ranging from 500 to 3500 Bhp, with the motor used in the type test. All manufacturing, inspection, and routine tests by motor manufacturer on production units are performed on the test motor.

The type test was performed on a 1250 hp vertical motor, in accordance with IEEE 323-1974. First normal operation during the design life was simulated; then the motor was subjected to a number of seismic events; and then subjected to the abnormal environmental conditions possible during and after a LOCA. The type test plan was as follows:

1. Thermal aging of the motor electrical insulation system (which is a part of the stator only) was based on extrapolation, in accordance with the temperature life of the characteristic curve from IEEE 275-1966 for the insulation type used on the ECCS motors. The amount of aging equals the total estimated operation days of maximum insulation surface temperature.

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

3. The normal induced current vibration effect on the insulation system was simulated by a 1.5 g horizontal vibration acceleration for one hour at current frequency.

4. Motor bearings were selected and their operating life established on the basis of bearing manufacturer's tests and operating data using the loads calculated to act on the bearings.

5. The dynamic load deflection analysis on the rotor shaft, performed to ensure adequate rotation clearance, is verified by static loading deflection of the rotor for the type test motor.

6. Dynamic loading aging and testing were 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 SSE loads with motor starts and operation combination, as may possibly occur during the life of the plant.

7. An environmental test simulating a LOCA condition was performed for 100 days with the test motor fully loaded, simulating pump operation. The test consists of startup and six hours of operation at 212°F ambient temperature, and a 100 percent steam environment. After one hour standstill in the same environment, another startup and operation of the test motor was followed by sufficient operation at high humidity and temperature. This was based on extrapolation in accordance with the temperature life characteristic curve from IEEE 275-1966 for the insulation type used on the ECCS motors.

3.9.3.2.4 High Pressure Coolant Injection Pump Assembly

Operability of the HPCI pump assembly is demonstrated by a combination of analytical stress calculations, pump manufacturer's operating experience, and testing. The stress definitions and the allowable stress criteria are based on the ASME B&PV Code, Section III. The Code is directly applicable to the stamped pressure boundary components of the pump.

The witnessed hydrostatic and performance tests, as performed at the pump manufacturer's plant, demonstrate that the pump, as designed, meets the ASME B&PV Code, Section III, requirements and the parameters of the design specification.

A three dimensional, finite element model was developed and dynamically analyzed using the response spectrum analysis method. The model was analyzed using static nozzle loads, pump thrust loads, and dead weight. Critical location stresses were evaluated and compared with the allowable stress based on the ASME B&PV Code, Section III. Shaft deflections and accelerations were analyzed to ensure that the rotating parts have no contact with the stationary parts, except at engineered wear points based on the pump manufacturer's operating experience. The above considerations provide adequate assurance that the pump will remain operable during the SSE load condition.

3.9.3.2.5 NSSS ASME B&PV Code Class 1 Active Valves

Each of the Class 1 valves is designed to perform its mechanical motion in conjunction with a design basis accident (DBA). Seismic qualification for operability is unique for each valve type. Each method of qualification is detailed individually below.

3.9.3.2.5.1 Main Steam Isolation Valves

The MSIVs are evaluated for operability during a seismic event by analysis and testing as follows:

1. First, the design of the valve body is evaluated in accordance with the applicable code that limits deformations in the operating area of the valve body to be within the elastic limit of the material, by limiting pressure and pipe reaction input loads, including seismic, thereby ensuring no interference with valve operability.

2. An analysis is completed on the actuator structure to determine component stresses and actuator deflection at loads under faulted conditions, including seismic acceleration loads. Component stresses of the actuator structure are limited to be within the material's elastic limit.

3. A dynamic test is conducted to qualify the control components and the safety-related limit switches for seismic condition loadings.

To ensure 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's actual input loads, amplified accelerations, and resonance frequencies are determined based on site excitation input to the system as a part of the overall steam line analysis. Pipe anchors and restraints are applied as required to limit pipe system resonance frequencies and amplified acceleration to be within acceptable limits for the MSIVs.

MSIV operability during seismic acceleration is addressed in Sections 3.9.2.3.1 and 3.9.2.3.2.13.

MSIV operability during LOCA conditions has been demonstrated, as defined in Reference 3.9-13. The test specimen was a 20-inch valve of a design representative of the MSIVs.

Qualification testing of sensitive electrical/pneumatic equipment to meet performance requirements is completed.

3.9.3.2.5.2 Safety/Relief Valves

The SRVs are qualified by test for operability during a seismic event. Structural integrity of the configuration during a seismic event is demonstrated by both analysis and test. The test includes the following steps:

1. Each valve is designed for the maximum moments that may be imposed when installed in service with inlet and outlet conditions of 400,000 inch-pounds and 300,000 inch-pounds, respectively. These moments are resultants of dead weight plus seismic loading (3g horizontal and 1g vertical) of both valve and connecting pipe, thermal expansion of the connecting pipe, and reaction forces from valve discharge.

2. A production 2-Stage SRV demonstrated its operability during a dynamic qualification (shake table) test when moment and seismic loads were applied that were greater than the required design limit loads and conditions for this equipment.

A mathematical model of this valve is included in the main steam piping system analysis along with one for the MSIVs. This analysis ensures that the equipment design limits are not exceeded.

Seismic tests were conducted on the SRVs, and the natural frequencies were determined to be greater than or equal to 33 hertz. The tests also determined that the equipment remains functional during application of the specified seismic loads.

In addition to the testing described above, and in Section 3.9.2.3, the sensitive electrical/pneumatic equipment of the SRVs were qualified to perform during and after emergency environmental conditions. The SRV analytical qualifications results are shown in Table 3.9-4i.

The original SRVs at Hope Creek were the 2-Stage Target Rock SRVs. Target Rock 3-Stage SRVs have been evaluated and approved for installation at Hope Creek. The 3-Stage SRVs were qualified as follows:

The original NSSS vendor evaluated that the use of the 2-Stage Required Response Spectrum for seismic testing (in the original product qualification specification) remained appropriate for the 3-Stage SRV. Target Rock document TR 9384, “Seismic Qual. Report" (VTD 432427) documents the seismic qualification of the 3-Stage SRV by analysis that was benchmarked against actual testing for the Limerick 3-Stage SRV. An independent third party verified that the seismic qualification performed by Target Rock met the requirement of original 2-Stage SRV. The natural frequencies of the 3-Stage SRVs were also demonstrated to be greater than 33 Hz. The 3-Stage SRVs were analyzed in accordance with the ASME Code (ASME Boiler and Pressure Vessel Code, Section III, Division 1, Class A, 1968 Edition with Addenda thru Winter 1970) in VTD 432428, to the same values of moment (300,000 in-lb and 400,000 in-lb, and greater values of acceleration (two horizontal accelerations of 4.5g, and one vertical acceleration of 4.5g).

3.9.3.2.5.3 Check Valves for the Residual Heat Removal and Core Spray Systems

GE scope of supply includes eight air operable and testable check valves: six for the RHR system, and two for the Core Spray System. Operability of these check valves is ensured by design calculations and sufficient structural margins, so that movement of the disk/hinge pin is not impaired under any loading conditions.

3.9.3.2.6 NSSS ASME B&PV Code Class 2 and 3 Active Valves

3.9.3.2.6.1 Control Rod Drive Valves

GE scope of supply for the Control Rod Drive (CRD) system includes four ASME B&PV Code Class 2 active valves, but no Class 3 valves.

The four Class 2 active valves in the CRD scram discharge volume vent and drain lines are air operated. However, the valves are designed to be fail-safe; the safety operation of the valve closure does not depend upon the plant air supply or on electrical operation of the controlling solenoid valves. In the event that the solenoid valves that control the globe valves are deenergized, or the plant air supply is interrupted for any reason, the yoke springs held in tension are capable of closing the valve. The valves are analyzed and type tested per IEEE 344-1975 to ensure operability during and after the dynamic loadings due to an earthquake.

3.9.3.2.6.2 Standby Liquid Control Valve (Explosive Valve)

The standby liquid control explosive valves are qualified to IEEE 344-1975. The qualification test included a demonstration of the absence of natural frequencies below 33 hertz, and the ability to remain operable under a horizontal seismic coefficient of 6.5g and a vertical seismic coefficient of 4.5g at 33 hertz.

3.9.3.2.7 Non-NSSS Pump and Valve Operability Assurance

3.9.3.2.7.1 Non-NSSS Active Pumps

The non-NSSS active pumps are tabulated in Table 3.9-17. Non-NSSS active pumps 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 design specifications. Table 3.9-27 provides examples of Non-NSSS active pumps, indicating their qualification method and the industry standards met.

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

After the pumps have been assembled, they are hydrostatically and performance tested in the manufacturer's shop in accordance with Hydraulic Institute standards. After the pumps are installed, they undergo functional tests. Provisions are made for inspection and operational testing per the requirements in Section 16. See Section 3.9.6 for operational testing outline. All of these tests demonstrate that the pumps are reliable and will function as specified.

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 OBE and SSE events.

Information on loading combinations, system operating transients, and stress limits for pumps is given in the response to Question 210.52.

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

Deflection due to piping loads and dynamic loads is addressed for active essential pumps by several methods depending on the situation. Methods used include static deflection analysis, dynamic deflection analysis, static bend testing, and dynamic seismic testing. These methods account for pump deflection due to the application of nozzle allowable loadings and demonstrate component operability.

Each pump/motor combination is designed to rotate at a constant speed under all conditions, unless the rotor becomes completely seized, i.e., fails to rotate at all. Motors are designed to withstand short periods of severe overload and, typically, the rotor can be seized a short period of time before a circuit breaker shuts down the pump. 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 perform their intended functions. These proposed requirements take into account the complex characteristics of the pump, and they 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.

3.9.3.2.7.2 Non-NSSS Active Valves

Non-NSSS active valves are tabulated in Table 3.9-18. See Sections 3.9.3.2.5 and 3.9.3.2.6 for a discussion of operability assurance of active valves supplied by the NSSS vendor. Table 3.9-27 provides examples of non-NSSS active valves, indicating their qualification method and the industry standards met.

Safety-related non-NSSS active valves are subjected to a series of stringent tests prior to service and during the plant life. Before installation, the following tests are performed: the shell hydrostatic test, in accordance with ASME B&PV Section III requirements; backseat and main seat leakage tests; the disc hydrostatic test; functional tests which verify that the valve opens and closes within the specified time limits; and the operability qualification of motor, air, and hydraulic operators for environmental conditions over the installed life, i.e., aging, radiation, accident environment simulation, etc, in accordance with IEEE 382-1972. It is the intent of PSE&G to review the qualification reports of all safety-related non-NSSS active valve operators (electric, air, and hydraulic) using the requirements of IEEE 382-1980. The intent of IEEE 382-1980 has been met to the maximum degree possible, without embarking on a completely new qualification program. This has been done by reviewing the existing qualification reports using the IEEE 382-1980 requirements and supplementing these original tests with additional test and/or analyses, as applicable, to demonstrate the upgraded qualification. After installation, cold hydrostatic tests, functional tests (in accordance with the requirements of Section 14), and periodic inservice operation (in accordance with the requirements of Section 16) are performed to verify and ensure the functional ability of the valve. See Section 3.9.6 for operational testing outline.

The method of qualification for soft parts of safety-related valves is addressed in Section 3.11.2.6. In addition, maintenance and surveillance program requirements are given in Section 3.11.2.7.6.

The valves are designed using either stress analyses or pressure containing minimum wall thickness requirements. For all active valves with extended topworks, an analysis is also performed for static equivalent SSE loads applied at the extended structure's center of gravity. The maximum stress limits allowed in the analyses demonstrate structural integrity and are equal to the limits recommended by ASME for the particular ASME class of valve analyzed. The loads and conditions considered in the qualification of Class 1 valves are given in Table 3.9-9. The loads and conditions considered for Class 2 and 3 valves are given in Table 3.9-14.

In addition to the foregoing, a representative valve of each type is factory tested to verify operability during a simulated seismic event. The factory qualification testing procedures are as described below.

Deflection due to piping loads and dynamic loads are addressed for active essential valves by several methods depending on the situation. Methods used include static deflection analysis, dynamic deflection analysis, static bend testing, and dynamic seismic testing.

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

1. All active valves with topworks must have a first natural frequency greater than 33 hz. This is proven by analyses. For valves mounted on lines connected directly or

indirectly to the RPV or the biological shield, resonant frequencies up to 100 hertz are determined. Such frequencies are used as input to the dynamic analysis of the piping systems for annulus pressurization effects. Because of the unique and heavier loads imposed by hydrodynamic forces on piping attached to the suppression chamber, active valves installed in such piping subjected to hydrodynamic loads are additionally analyzed to determine all resonant frequencies between 0 and 100 hertz. These valves are listed in Table 3.9-30. Such frequencies are used as input to the dynamic analysis of these piping systems.

2. While in the shop and installed in a suitable test rig, the extended topworks of the valve are subjected to a statically applied equivalent seismic load. The load, specified as 4.5g times the weight of the topworks, 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 is simultaneously applied to the valve during the static load tests.

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

4. Valve operators (motor, air, and hydraulic) are independently qualified as operable during the SSE prior to their installation on the valve.

The equivalent seismic and hydrodynamic static load, which is used for the static valve qualification, is the maximum load which the valve is designed to withstand. The piping designer maintains the valve operator accelerations within these levels.

The valve is leaktested 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 factory testing program applies only to valves with overhanging structures, e.g., the motor operator, air, or hydraulic actuator assembly. The testing is conducted on a representative number of valves, a representative valve being selected as described below.

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 by tests, and the results are used to qualify all valves within the intermediate range of sizes, as shown in Table 3.9-19. 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 it relates 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.

Additionally, the stress calculations for each valve assembly are reviewed, and 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 nontested valves by tested valves.

Because of their compact configuration, check valves are not adversely affected by seismic acceleration. They have no extended structures to distort the bodies and cause malfunctions. Their discs are designed to allow sufficient clearance within the body to prevent binding or interference due to distortions from nozzle or other imposed loads. They are qualified by a combination of the following factory tests and analysis:

1. Stress analysis of critical areas and parts for SSE loads, in accordance with the allowables specified in Tables 3.9-9 and 3.9-14.

2. In-shop hydrostatic test

3. In-shop seat leakage test

4. Periodic valve exercise and inspection to ensure the functional ability of the valve, in accordance with the requirements of Section 16.

Seismic operability testing is not performed for vacuum relief valves. Due to the particularly simple characteristics of these valves, and the lack of extended structures, they are qualified by a combination of the following tests and analysis:

1. Stress analysis, including seismic loads where applicable

2. In-shop hydrostatic test

3. In-shop seat leakage test

4. Performance tests

5. Periodic in situ valve inspection, as applicable, and periodic valve removal, refurbishment, performance testing, and reinstallation

The above testing and analysis is sufficient to ensure the functional capability of the valve.

The following applies to 2-Stage SRVs:

SRVs that have an extended structure go through a similar qualification procedure as the vacuum relief valves, with the addition of a seismic operability qualification test, as described below.

The SRVs are type tested. A random valve is selected from a lot of valves of similar design and size. The valve is tested at operating pressure and at ambient temperature conditions. The test is a four-part procedure that consists of:

1. Verifying the operability of the valve before the simulated seismic event

2. Applying a static coefficient seismic load to the valve superstructure and verifying its operability during the event

3. Removing the load and verifying its operability after the event

4. Subsequent inspection after the test.

The seismic test demonstrates that SRVs can open within a specific pressure band to protect vessels and equipment from abnormal pressure, and that they are able to reseat, preventing a further flow of fluid after normal pressure conditions have been restored.

The original SRVs at Hope Creek were the 2-Stage Target Rock SRVs. Target Rock 3-Stage SRVs have been evaluated and approved for installation at Hope Creek. The 3-Stage SRVs installed at Hope Creek were qualified as follows:

The original NSSS vendor evaluated that the use of the 2-Stage Required Response Spectrum for seismic testing (in the original product qualification specification) remained appropriate for the 3-Stage SRV. Target Rock document TR 9384, "Seismic Qual. Report" documents the seismic qualification of the 3-Stage SRV by analysis that was benchmarked against actual testing for the Limerick 3-Stage SRV. An independent third party verified that the seismic qualification performed by Target Rock met the requirement of original 2-Stage SRV. The natural frequencies of the 3-Stage SRVs were also demonstrated to be greater than 33 Hz.

During a seismic event, it is anticipated that the seismic acceleration imposed upon the valve may cause it to open momentarily and discharge under system conditions that otherwise would not result in valve opening, but this is considered to be of no real safety or other consequence.

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

3.9.3.2.7.3 Extent of Pump and Valve Qualification/Operability per Draft Standards ANSI/ASME QP-1 (N551.1), QP-2 (N551.2), QP-3 (N551.3), QP-4 (N551.4), and Issued ANSI Standards N41.6 and B16.41-1983

The subject standards are not included in the Acceptance Criteria of NRC Standard Review Plan (SRP) 3.10. However, these standards are identified in the review procedure. The extent to which these draft standards are used in the qualification of pumps and valves is given below:

Valves: ANSI Std. N 41.6 is the same as IEEE Std. 382-1972. This standard is complied fully in the qualification of all the valves.

The extent of qualification of one 18" motor operated valve has been reviewed against the requirements of ANSI Std. B16.41-1983. Essentially the requirements of this standard are met in the qualification of 18" motor operated valves.

Pumps and Motors: The qualifications of Safety Auxiliaries Cooling System pumps and valves has been reviewed per the requirements of Draft Standards QP-1, QP-2, QP-3, QP-4. Some additional information was obtained from the manufacturer recently in this connection.

Essentially, the requirements of these standards are met, except for that of QP-3 (Shaft Seal Assemblies).

The detailed response was provided by PSE&G Letter to NRC, R. L. Mittl to W. Butler, dated September 18, 1985.

3.9.3.3 Design and Installation of Pressure Relief Devices

3.9.3.3.1 NSSS Safety/Relief Valves

The original SRVs at Hope Creek were the 2-Stage Target Rock SRVs. Target Rock 3-Stage SRVs have been evaluated and approved for installation at Hope Creek. Because the 2-Stage and 3-Stage SRVs have the same set pressures, capacities, and response times, the following discussion is applicable to both 2-Stage and 3-Stage SRVs.

An SRV lift results in a transient that produces momentary unbalanced forces acting on the discharge piping system, for the period from opening of the SRV until a steady discharge flow from the RPV 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 SRV, cause the SRV 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 sequential time history solution of the fluid flow equation to generate a time history of the fluid properties at numerous locations along the pipe. The fluid transient properties are calculated based on the maximum set pressure specified in the steam system specification and the value of the ASME B&PV Code flow rating increased by a factor to account for the conservative method of establishing the rating. Simultaneous discharge of all valves is assumed in the analysis, because simultaneous discharge is considered to induce maximum stress in the piping. 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.

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 SRV, the main steam line, and the discharge piping are combined with loads due to other effects, as specified in Section 3.9.3.1. The ASME B&PV Code stress limits corresponding to load combinations classified as normal, upset, emergency, and faulted are applied to the main steam lines and to the SRV discharge piping.

The drywell SRV piping system for HCGS consists of 14 individual Schedule 40, SA-106, Grade B piping lines. The nominal pipe size of the piping is 10" Schedule 40 at the outlet flange of the SRV, changing to 10" Schedule 80 immediately before the vent pipe jet deflector and 10" Schedule 160 at the vent pipe penetration (VPP). Figure 3.9-7 shows the routing, support locations, and support types, for a representative SRV line in the drywell.

The 14 SRV lines initiate at the 4 main steam lines and are grouped in sets of three and four, as shown schematically in Figure 3.9-8. The lines are routed from the drywell area through the vent lines and into the suppression chamber.

The 14 SRV lines are attached to the 4 main steam lines in the drywell at the safety-relief valves, as shown in Figure 3.9-9. Each SRV line passes through a vent pipe jet deflector and is supported at an intermediate location in the vent pipe. Beyond this support, the SRV line turns 90° and exits the vent pipe at the VPP. This arrangement is shown in Figure 3.9-10.

The wetwell SRV piping system for HCGS consists of fourteen 10" diameter, Schedule 80, SA-106 Grade B piping lines. Figure 3.9-11 shows a typical wetwell SRV line and support locations.

The support system for the wetwell SRV piping consists of a stiffened penetration support at the VPP, vertical and horizontal struts attached to the ring girder, and a lateral strut attached to the vent header. Details of the strut supports attached to the ring girder and vent header are shown in Figure 3.9-12.

At the lower end of each SRV line is a 12" diameter T-quencher device.

The T-quencher is supported by a 14" diameter pipe beam located directly below the T-quencher arms. The T-quencher arms are connected to the support beam by plate-type supports as shown in Figure 3.9-15. The T-quencher ramshead support assembly consists of the ramshead saddle plate and two attached pin plates with stiffeners. The assembly pin plates are connected to pin plates on the mitered joint ring girder by a 2-1/2 in. diameter pin as shown in Figures 3.9-13 through 3.9-15.

The imbalanced thrust load on the last vertical SRVDL segment in the wetwell is shown on Figure 3.9-16. The reaction loads at the ramshead support are contained in Table 3.9-29.

3.9.3.3.2 Design and Installation Details for Mounting of Pressure Relief Devices in ASME B&PV Code Class 1 and 2 Systems (Non-NSSS)

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

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

2. 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 loading combinations for a given transient are shown in Table 3.9-7, and the design criteria and stress limits are shown in Tables 3.9-8 and 3.9-12.

3.9.3.4 Component Supports

For GE and Bechtel designed pipe supports, the reactions produced by primary and secondary pipe loads are categorized as primary. The primary and secondary loads are summed and compared to the load rating to ensure that the rating is not exceeded. Since no distinction is made between primary and secondary loads, and load rated components are designed to primary limits or they are qualified by testing; the supports meet primary stress criteria for the combination of the primary and secondary loads.

3.9.3.4.1 Piping (NSSS)

Piping supports are designed in accordance with Subsection NF of the ASME B&PV Code, Section III. Supports are either designed by load rating, per Subsection NF-3260, or to the stress limits for linear supports, per Subection NF-3231. To avoid buckling in the component supports, Appendix F of the ASME B&PV Code requires that

the allowable loads be limited to two-thirds of the critical buckling loads. The critical buckling loads for Class 1 component supports in the NSSS scope subjected to faulted loads that are more severe than normal, upset, and emergency loads, are determined by the supplier, using the methods discussed in Appendix F of the ASME B&PV Code. In general, the load combinations for the conditions correspond to those used to design the supported pipe. Design transient cyclic data are not applicable to piping supports, since no fatigue evaluation is necessary to meet the ASME B&PV Code requirements.

The design criteria and dynamic testing requirements for component supports are given below:

1. Component supports - All component supports are designed, fabricated, and assembled so that they cannot become disengaged by the movement of the supported pipe or equipment after they are installed. All component supports are designed in accordance with the rules of Subsection NF of the ASME B&PV Code (Table 3.2-3). For the NSSS scope of supply, all valve operators that are mounted on Class 1 piping are not used as component supports.

2. 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.

3. Snubbers

   a. Required load capacity and snubber location - The entire piping system, including valves and the suspension system between anchor points, is mathematically modeled for complete structural

analysis. In the mathematical model, the snubbers are modeled as a spring with a given spring stiffness depending on the snubber size. The analysis determines the forces and moments acting on each component and the forces acting on the snubbers due to all dynamic loading conditions defined in the piping design specification. The design load on the snubbers includes those loads caused by seismic forces (OBE and SSE), system anchor movements, and reaction forces caused by SRV discharge, main stop valve closure, etc.

The snubber location and loading direction are first decided by estimation so that the stresses in the piping system have acceptable values. The snubber locations and direction are refined by performing the computer analysis on the piping system, as described above.

The spring constant required by the suspension design specification for a given load capacity snubber is compared against the spring constant used in the piping system model. If the spring constants are not in agreement, they are brought into agreement, and the system analysis is redone to confirm the snubber loads. If the stiffness of the backup structure for the snubber is not large compared to that of the snubber, the reduced effective snubber stiffness (spring constant) is used in the analysis to account for the backup structure flexibility.

b. Design specification requirements - To ensure that the required structural and mechanical performance characteristics and product quality are achieved, the following requirements for design and testing are imposed:

(1) The snubbers are required by the suspension design specification to be designed in accordance with all of the rules and regulations of the ASME B&PV Code, Section III, Subsection NF (Table 3.2-3). This design requirement includes calculation of the stresses in the snubber component parts under normal, upset, emergency, and faulted loads. These calculated stresses are then compared against the allowable stresses of the material, as given in the ASME B&PV Code, Section III, to make sure that they are below the allowable limits.

(2) The snubbers are tested to ensure that they can perform as required during OBE and SSE events, and under anticipated operational transient loads or other mechanical loads associated with the design requirements for the plant. The test requirements include:

(a) Snubbers are subjected to loading of force or displacement versus time loading at frequencies within the range of significant modes of the piping system.

(b) Displacements are measured to determine the performance characteristics specified.

(c) Tests are conducted at various temperatures to ensure operability over the specified range.

(d) Peak test loads in both tension and compression are verified to be equal to or higher than the rated load requirements.

(e) The snubbers are also tested for various abnormal environmental conditions. Upon completion of the abnormal environmental transient test, the snubber is tested dynamically at a frequency within a specified frequency range. The snubber must operate normally during the dynamic test.

c. Snubber installation requirements - An installation instruction manual is required by the suspension design specification. This manual must contain instructions for storage, handling, erection, and adjustments (if necessary) of snubbers. Each snubber has an installation location drawing that contains the installation, location of the snubber on the pipe and structure, the hot and cold settings, and additional information needed to install the particular snubber.

The suspension design specification requires that snubbers be provided with position indicators to identify the rod position. This indicator facilitates the checking of hot and cold settings of the snubber, as specified in the installation manual, during plant preoperational and startup testing.

d. Inspection, testing, repair, and/or replacement of snubbers - The suspension design specification requires that the snubber supplier prepare an installation instruction manual. This manual must contain complete instructions for the testing, maintenance, and repair of the snubber. It must also contain inspection points and the period for inspection.

   The suspension design specification requires that hydraulic snubbers be equipped with a fluid level indicator so that the level of fluid in the snubber can be ascertained easily.

e. Struts - The design load on struts includes those loads caused by dead weight, thermal expansion, primary seismic forces (OBE and SSE), system anchor displacements, and reaction forces caused SRV discharge, main stop valve closure, etc.

   Struts are designed in accordance with NF-3000 of the ASME B&PV Code to be capable of carrying the design load for all conditions (Table 3.2-3).

f. 1. <u>Equipment Anchorage</u>

   Equipment anchorage is not in the NSSS scope.

   2. <u>Component Support Bolting</u>

   The following bolting design limits are typical of components mounted directly on base plates.

- RWCU Pump

  The support bolting of this pump, which is not safety-related, is designed for the effects of pipe load and SSE loads to the requirements of the ASME B&PV Code, Section III, Appendix XVII. The stress limits of 0.41 Sy for tension and 0.15 Sy for shear are used.

- RCIC/SLC Pumps and RCIC Turbine

  The equipment-to-base-plate bolting satisfies the following design criteria: For normal and upset conditions, 1.0 S is used for primary membrane (or tension), and 1.5 S for primary membrane plus bending (if applicable), where S is the allowed stress limits from the ASME B&PV Code, Section III, Appendix I, Table I-7.3. For emergency and faulted conditions, stresses shall be less than 1.2 times the allowed limits for normal and upset conditions.

The allowed stress limits used for bolting in pipe supports and pipe mounted equipment supports are as per ASME B&PV Code, Section III, Subsection NF.

For service level A and B, the bolts meet the criteria of Paragraph NF-3280. For service level C and D, Article 2460 of Appendix XVII, with the factors indicated in Article 2110 of Appendix XVII, is the applicable design

requirements for bolting. The stresses calculated under these criteria do not exceed the specified minimum yield stresses at temperature.

3. Flanged Connections

Flanged connections are not in the NSSS scope.

g. Expansion Anchor

Expansion anchors are not in the NSSS scope.

3.9.3.4.2 Emergency Core Cooling System Pumps

The core spray and RHR pumps are tested in the shop and tested as defined in Section 3.9.3.2. These tests prove the adequacy of the support structure for the pump assembly under operating conditions. Furthermore, the stress calculation summary provided in Section 3.9.3.1 defines the stress levels in the critical support areas; namely, the pressure boundary parts and nonpressure boundary parts. The stress level margins prove the adequacy of the equipment.

3.9.3.4.3 Reactor Core Isolation Cooling and High Pressure Coolant Injection Turbines

The RCIC and HPCI turbine assemblies are analyzed as defined in Section 3.9.3.1. The analyses verify the adequacy of the supports under various operating conditions. In all cases, the calculated stresses in the critical support areas are within the stresses allowed by the ASME B&PV Code. Tables 3.9-4s and 3.9-4dd give the summary of the design calculations for RCIC and HPCI turbine components, respectively.

3.9.3.4.4 Reactor Water Cleanup System Pump (NSSS)

The pump pedestal bolts are analyzed as discussed in Section 3.9.3.1. Loads from seismic, dead weight, connecting pipes, and temperature are considered.

3.9.3.4.5 Reactor Pressure Vessel Support Skirt and Stabilizer

The RPV support skirt is designed as an ASME B&PV Code Class 1 plate and shell type component support, per the requirements of the ASME B&PV Code, Section III. The loading conditions, stress criteria, calculated stresses, and the allowable stresses in the critical support areas for various plant operating conditions are summarized in Table 3.9-4b. The stress level margins prove the adequacy of the RPV support skirt.

The RPV stabilizer is designed as a Class 1, linear type component support, per the requirements of the ASME B&PV Code, Section III, Subsection NF. The stabilizer provides a reaction point near the upper end of the RPV to resist horizontal loads due to effects such as earthquake and pipe rupture. The design loading conditions, stress criteria, calculated stresses, and the allowable stresses in the critical areas are summarized in Table 3.9-4bb.

A generic BWR 4/5 study was conducted using the design of the Limerick 1 and 2 cylindrical support skirts, which have the smallest ratio of thickness to radius. The study examined the skirt buckling under axial compression, hoop stress, and transverse sheer (see Reference 3.9-21) and showed that under each of these types of loads the critical buckling stress is much greater than the yield stress.

Since this study showed that inelastic stability limits the skirt's integrity, the permitted critical buckling stress should be less than 90 percent of the stress from a compressive load that would produce a yield stress, and the permitted buckling stress divided by 1.125 to provide margin for the variations of fabrication or

eccentricity. Analyses have shown that the loads on the HCGS support skirt will not produce stresses that exceed two-thirds of the at-temperature critical buckling stress, prescribed by paragraph F-1370(e) of Section III of the ASME B&PV Code.

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3.9.3.4.6 Non-NSSS Component Supports

Piping supports are designed in accordance with Article 3000 of Subsection NF of the ASME B&PV Code, Section III. In general, the loading combinations for supports for ASME B&PV 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 3.9-20. The stress limits are those given in Article 3000 of Subsection NF of the ASME B&PV Code, Section III, 1974 Edition, including Addenda through Winter of 1975. In addition, the design of such supports also conforms to Subsection NF-3280, NF-3290, NF-3380, and NF-3390 of Section III of the ASME B&PV Code, 1980 Edition. The nondestructive examination of welds for NF support elements shall be performed using the requirements noted in Subsection NF-5200 of Section III of the ASME B&PV Code, 1977 Edition, including Addenda through Winter of 1978. For component supports of essential systems, emergency stress limits are used in lieu of the faulted condition limits. This is to ensure that, under all loading conditions, their stresses are contained within the yield stress.

The allowable stress limits used for bolting in equipment anchorage and in pipe support components is 0.5 Su but shall not exceed 0.9 Sy under all service levels.

For flanged connections, the bolt allowable stress used in the piping analysis is ASME Subsection III, 1979 Summer Addenda, Subsections NB, NC, and ND for Classes 1, 2 and 3, respectively.

The capacities of concrete expansion anchors are based on actual testing of anchors to failure. The failure loads are divided by the factor of safety (typically 4 in accordance with NRC Bulletin 79-02) to establish the allowable design loads. Baseplate flexibility is considered in the design of concrete expansion anchor bolts in accordance with IE Bulletin 79-02.

3.9.3.4.6.1 Snubbers

Snubbers are used in Seismic Category I & II systems. For both inside and outside primary containment, snubbers are of the mechanical and hydraulic type. The mechanical snubbers are purchased from Pacific Scientific Corporation and hydraulic snubbers from Lisega with loading ratings appropriate for the design conditions and load combinations. The snubbers are designed in accordance with ASME B&PV Code, Section III, Subsection NF (see Table 3.2.-3).

The effective stiffness of the snubber is considered in evaluating the piping system response.

A summary of the types and sizes of snubbers at HCGS is provided in the ISI Snubber Tracking Program.

Functional tests for hydraulic and mechanical snubbers are required:

The snubber functional test shall verify that:

(1) Activation (restraining action) is achieved within the specified range in both tension and compression;

(2) Snubber in bleed, or release rate where required, is present in both tension and compression, within the specified range (hydraulic snubbers only);

(3) For mechanical snubbers, the force required to initiate or maintain motion of the snubber is within the specified range in both directions of travel; and

(4) For snubbers specifically required not to displace under continuous load, the ability of the snubber to withstand load without displacement.

Technical Specification LCO 3.0.8 allows snubbers to be inoperable without declaring the affected supported LCO(s) not met for 72 hours if the inoperable snubber affects one subsystem or 12 hours if it affects multiple subsystems. The following Tier 2 restrictions from Reference 3.9-29 apply to the use of LCO 3.0.8 and have been incorporated into the TS Bases:

1. For BWR plants, one of the following two means of heat removal must be available when LCO 3.0.8a is used:

   - At least one high pressure makeup path (e.g., using high pressure coolant injection (HPCI) or reactor core isolation cooling (RCIC) or equivalent) and heat removal capability (e.g., suppression pool cooling), including a minimum set of supporting equipment required for success, not associated with the inoperable snubber(s), or
   - At least one low pressure makeup path (e.g., low pressure coolant injection (LPCI) or core spray (CS)) and heat removal capability(e.g., suppression pool cooling or shutdown cooling), including a minimum set of supporting equipment required for success, not associated with the inoperable snubber(s).

2. When LCO 3.0.8b is used at BWR plants, it must be verified that at least one success path exists, using equipment not associated with the inoperable snubber(s), to provide makeup and core cooling needed to mitigate LOOP accident sequences.

3. Every time the provisions of LCO 3.0.8 are used licensees will be required to confirm that at least one train (or subsystem) of systems supported by the inoperable snubbers would remain capable of performing their required safety or support functions for postulated design loads other than seismic loads. LCO 3.0.8 does not apply to non-seismic snubbers. In addition, a record of the design function of the inoperable snubber (i.e., seismic vs. non-seismic), implementation of any applicable Tier 2 restrictions, and the associated plant configuration shall be available on a recoverable basis for staff inspection.

   LCO 3.0.8 only applies to snubber support functions that are seismic related. In OPERATIONAL CONDITIONS 4 and 5, snubbers only perform seismic support functions. In OPERATIONAL CONDITIONS 1, 2, and 3, some snubbers also perform non-seismic support functions (e.g., hydrodynamic loads, turbine trip loads, etc.). When LCO 3.0.8 is used, confirm that at least one train (or subsystem) of systems supported by the inoperable snubbers would remain capable of performing their required safety or support functions for postulated design loads other than seismic loads.

3.9.3.5 SRP Rule Review

Acceptance Criterion II.1, of SRP Section 3.9.3, requires that the combination of design and service loadings applicable to the design of Class 1, 2 and 3 components be in agreement with positions stated in Appendix A of SRP Section 3.9.3.

At HCGS, Appendix A was not used as a design basis requirement. However, Bechtel used representative industry practice at the time of design, procurement, and manufacturing, and is in agreement with the general intent of this appendix. The actual loading combinations are specified in the component purchase specifications.

3.9.4 Control Rod Drive System

This plant is equipped with a hydraulic Control Rod Drive (CRD) System. The discussion in this section includes the CRDs, the hydraulic control units (HCUs), 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 the Control Rod Drive

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

3.9.4.2 Applicable Control Rod Drive System Design Specifications

The CRD system is designed to meet the functional design criteria outlined in Section 4.6 and consists of the following:

1. Locking piston CRDs

2. HCUs

3. Hydraulic power supply (pumps)

4. Interconnecting piping

5. Flow and pressure control valves

6. Instrumentation and electrical controls.

Those components of the CRD forming part of the reactor coolant pressure boundary are designed in accordance with the ASME B&PV Code, Section III.

The quality group classification of the CRD hydraulic system is outlined in Table 3.2-1. The components are designed in accordance with the codes and standards governing the individual quality groups. Quality group classification is not applicable to the nonpressurized parts of the CRDs.

Pertinent aspects of the design and qualification of the CRD components are discussed in the following sections: transients in Section 3.9.1.1, faulted conditions in Section 3.9.1.4, seismic testing in Section 3.9.2.3, and loading combinations and stress limits in Table 3.9-4w.

3.9.4.3 Design Loads, Stress Limits, and Allowable Deformations

The ASME B&PV Code, Section III components of the CRDs have been evaluated analytically. The design load combinations and stress limits are listed in Table 3.9-4w. For the non-Code components, experimental testing is used to determine the CRD performance under all possible conditions, as described in Section 3.9.4.4.

Deformation has been compared with the allowable limits, and is not a controlling factor, based upon numerous tests performed on CRDs.

3.9.4.4 Control Rod Drive System Performance Assurance Program

The CRD system test program consists of the following tests:

1. Development tests
2. Factory quality control tests
3. Five-year maintenance life tests
4. 1.5X design life tests
5. Operational tests
6. Acceptance tests
7. Surveillance tests.

All of the above tests, except 3. and 4., are discussed in Section 4.6.3. Tests 3. and 4. are discussed below:

Test 3. - Five-Year Maintenance Life Tests

Four CRDs are normally picked at random from the production stock each year and subjected to various tests under simulated reactor conditions and approximately one-sixth of the cycles specified in Section 3.9.1.1.

Upon completion of the test program, the CRDs must meet, or surpass, the minimum specified performance requirements.

Test 4. - 1.5X Design Life Tests

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

Two CRDs underwent such testing in 1976. Upon completion of the test program, these CRDs met or surpassed the minimum specified performance requirements.

3.9.5 Reactor Pressure Vessel Internals

This section identifies and discusses the structural and functional integrity of the major reactor pressure vessel (RPV) internals.

3.9.5.1 Design Arrangements

The core support structures and RPV internals (exclusive of fuel, control rods, control rod drives (CRDs), and in-core nuclear instrumentation) are identified below:

1a. Core support structures

- a. Shroud
- b. Shroud support (cylinder, plate, and legs)
- c. Core plate and holddown bolts
- d. Top guide
- e. CRD housings
- f. Fuel supports
- g. Control rod guide tubes.

2. Reactor internals

- a. Feedwater spargers
- b. Initial startup neutron sources

c. Surveillance sample holders

d. In-core instrument housings

e. Steam dryer assembly

f. Shroud head and steam separator assembly

g. Guide rods

h. Jet pump assemblies and instrumentation

i. Deleted

j. Core plate differential pressure sensing lines

k. In-core flux monitor guide tubes

l. Core spray lines and spargers

m. Low pressure coolant injection (LPCI) lines

n. CRD thermal sleeves.

A general assembly drawing of the important reactor components is shown on Figure 3.9-2.

The floodable inner volume of the RPV is shown on Figure 3.9-3. This 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 tubes, is such that one end is unrestricted and thus free to expand.

The LPCI lines include couplings with slip joint sleeves to allow free thermal expansion.

3.9.5.1.1 Core Support Structures

The core support structures consist of those items listed in Section 3.9.5.1.a. These structures form partitions within the reactor vessel, to sustain pressure differentials across the partitions, to direct the flow of coolant, and to laterally locate and support the fuel assemblies. Figure 3.9-3 shows the reactor vessel internal flow paths.

3.9.5.1.1.1 Shroud

The stainless steel shroud with shroud support makes up a 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 grid below. The central portion of the shroud surrounds the active fuel and forms the longest section of the assembly. This section is bounded at the top by the top guide grid and at the bottom by the core plate. The lower portion, surrounding part of the lower plenum, is welded to the RPV shroud support.

3.9.5.1.1.2 Shroud Support

The shroud support is furnished as a completed assembly by the RPV manufacturer and is designed, analyzed, and built as an integral part of the RPV. The shroud support is designed to support the shroud and to support and locate the jet pumps. The shroud support provides an annular baffle between the RPV and the shroud. The jet

pump discharge diffusers penetrate the shroud support to introduce the coolant to the inlet plenum below the core. The loading conditions, stress criteria, and calculated and allowable stresses are summarized in Table 3.9-4b.

3.9.5.1.1.3 Core Plate

The core plate is a circular stainless steel plate with bored holes, which is 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 plate.

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

3.9.5.1.1.4 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, for less than four fuel assemblies. Sockets are provided in the bottom of the beam intersections to anchor the in-core flux monitors and the 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 that 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. The loading conditions, stress criteria, and calculated and allowable stresses are summarized in Table 3.9-4c.

3.9.5.1.1.5 Fuel Supports

The fuel supports, shown on Figure 3.9-4, are of two basic types; namely, peripheral fuel supports, and four-lobed orificed fuel supports. The peripheral fuel supports are located at the outer edge of the active core and are not adjacent to control rods. Each peripheral fuel support holds one fuel assembly and contains a single orifice assembly designed to ensure proper coolant flow to the peripheral fuel assembly. Each four lobed orificed fuel support holds four fuel assemblies and is provided with four orifice plates to ensure proper coolant flow distribution to each rod controlled fuel assembly. The four lobed orificed fuel supports rest in the top of the control rod guide tubes, which are supported laterally by the core plate. The control rods pass through slots in the center of the four-lobed orificed fuel support. A control rod and the four adjacent fuel assemblies represent a core cell. See Section 4.2.

3.9.5.1.1.6 Control Rod Guide Tubes

The control rod guide tubes, located inside the vessel, extend from the top of the CRD housings up through holes in the core plate. Each tube is designed as the guide for a control rod, as well as being the vertical support for a four lobed orificed fuel support and the four fuel assemblies surrounding the control rod. The bottom of the guide tube is supported by the CRD 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 CRD 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 CRD housing to hold the thermal sleeve in position.

3.9.5.1.2 Reactor Internals

The reactor internals consist of those items listed in Section 3.9.5.1.b. Those that involve coolant flow paths are described in the following paragraphs.

3.9.5.1.2.1 Jet Pump Assemblies

The jet pump assemblies are located in two semicircular groups in the downcomer annulus between the core shroud and the RPV wall. The design and performance of the jet pumps are discussed in References 3.9-14 and 3.9-15. Each stainless steel jet pump consists of a driving nozzle, a suction inlet, a throat or mixing section, and a diffuser, as shown on Figure 3.9-5. 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 RPV wall.

The nozzle entry section is connected to the riser by a metal to metal, spherical to conical seal joint. Firm contact is maintained by a holddown clamp. The throat section is supported laterally by a bracket attached to the riser. There is a slip fit joint between the throat and diffuser. The diffuser is a gradual conical section changing to a straight cylindrical section at the lower end.

The preload on the hold down beams will be reduced from 30 to 25 kips in accordance with General Electric recommendations. This will increase the expected life of the beams to 19-40 years. The need for inservice inspection will be based on a lead plant experience and GE testing, and will be conducted such that any initiation will be detected prior to beam failure.

Repairs were made to Jet Pumps 8, 9 and 15 where cracks have been found in the welds that attach the jet pump flow instrument pressure sensing lines to brackets on the jet pump diffusers. The repair involves installation of a semi-circular clamp on each jet pump diffuser to hold the sensing line against the jet pump. Consequently, the clamp in effect augments or replaces the function of the weld attachment.

A repair was made to Jet Pump 16 where excessive gap was found at the vessel side setscrew. The repair involved installation of auxiliary spring wedge assembly against the jet pump. This assembly functionally replaces the restrainer bracket setscrew and re-establishes the lateral three-point support. In addition, on Jet Pump 9, the shroud side setscrew tack welds were found cracked. This setscrew was staked to prevent back out of the setscrew and an auxiliary spring wedge assembly was installed in case a gap developed at the setscrew location.

3.9.5.1.2.2 Steam Dryer Assembly

The steam dryer removes moisture from the wet steam leaving the steam separators. The extracted moisture flows down the dryer vanes to the collecting troughs, and 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 dryer 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 RPV wall. Upward movement of the dryer assembly, which may occur under accident conditions, is restricted by steam dryer holddown brackets attached to the RPV top head.

3.9.5.1.2.3 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. To conform with NUREG-0619 the design of the feedwater nozzles and spargers follow the issue resolution presented in Reference 3.9-17. 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, mixing the cooler feedwater with the downcomer flow from the steam separators and steam dryer before it contacts the vessel wall. The feedwater also serves to condense the steam in the region above the downcomer annulus and to subcool the water flowing to the jet pumps and recirculation pumps.

3.9.5.1.2.4 Core Spray Lines and Spargers

The core spray lines and spargers distribute coolant to the reactor core during accident conditions. Two core spray lines enter the reactor vessel through the two core spray nozzles. 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, passing 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.

3.9.5.1.2.5 Deleted

3.9.5.1.2.6 Core Plate Differential Pressure Sensing Lines

These lines are used to sense the differential pressure across the core plate. The lines enter the reactor vessel at a point below the

core shroud as two concentric lines. In the lower plenum, the two lines separate. The inner line terminates near the lower shroud, below the core plate. It is used to sense the pressure below the core plate during normal operation. The outer line terminates immediately above the core plate and senses the pressure in the region outside the fuel assemblies.

3.9.5.1.2.7 In-core Flux Monitor Guide Tubes

These guide tubes provide a means of positioning fixed detectors in the core, as well as providing a path for calibration monitors of the Traversing In-core Probe (TIP) System.

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

3.9.5.1.2.8 Surveillance Sample Holders

The surveillance sample holders are welded baskets containing impact and tensile specimen capsules. The baskets hang from the brackets that are attached to the inside wall of the RPV 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 RPV itself, while avoiding jet pump removal interference or damage.

3.9.5.1.2.9 Shroud Head and Steam Separator Assembly

The shroud head and steam separator assembly is bolted to the top of the shroud, forming 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 that establishes a vortex, separating the water from the steam. The separated water flows from the lower portion of the steam separator into the downcomer annulus.

3.9.5.1.2.10 Low Pressure Coolant Injection Lines

The LPCI lines penetrate the core shroud through separate LPCI nozzles. Coolant is discharged inside the core shroud.

3.9.5.2 Design Loading Conditions

3.9.5.2.1 Events to be Evaluated

Examination of the spectrum of conditions for which the safety design basis must be satisfied, by core support structures and other engineered safety feature (ESF) reactor internals, reveals three significant faulted events:

1. Recirculation line break accident - A break in a recirculation line between the RPV and the recirculation pump suction

2. Steam line break accident - A break in one main steam line between the RPV and the flow restrictor, resulting in significant pressure differentials across some of the structures within the reactor

3. Earthquake - Subjects the core support structures and reactor internals to significant forces as a result of ground motion.

Analysis of other conditions existing during normal operation, abnormal operational transients, and accidents shows that the loads affecting the core support structures and other ESF reactor internals are less severe than these three postulated events. The faulted conditions for the RPV internals are discussed in Section 3.9.1.4. Loading combinations and analyses for the RPV internals are discussed in Section 3.9.3.1 and Tables 3.9-1 and 3.9-4.

3.9.5.2.2 Pressure Differential During Rapid Depressurization

A digital computer code is used to analyze the transient conditions within the RPV following the recirculation line break accident and the steam line break accident. The analytical model of the vessel consists of nine nodes, 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, giving the depressurization rates and pressures in the various regions of the reactor.

Figure 3.9-6 shows the nine reactor nodes. The computer code used is the GE Short-Term Thermal-Hydraulic Model described in Reference 3.9-16. This model has been approved for use in Emergency Core Cooling System (ECCS) conformance evaluation under 10CFR50, Appendix K. 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 3.9-16. These additional features are discussed below:

1. The liquid level in the steam separator region and in the annulus between the steam dryer skirt and the vessel wall

is tracked to more accurately determine the flow and mixture quality in the steam dryer and in the steam line.

2. 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 in the bypass region for a steam line 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.

3. The enthalpies in the guide tubes and in the bypass region are calculated separately since the fuel assembly pressure differential 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 line break) and a main steam line break (the largest steam line break) are considered in determining the design basis accident (DBA) for the ESF reactor internals. The recirculation line break is the same as the design basis loss-of-coolant accident (LOCA) 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 RPV and the main steam line flow 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 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 that influence the reactor's internal pressure differentials. The core flow essentially affects only the steady state part. For a fixed power, as the core flow increases, so do 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 RPV and the depressurization rate and the transient part of the maximum pressure load is thus increased. As a result, the total loads on some components are higher at low power.

To ensure that the calculated pressure differentials bound those that 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 a high power, rated recirculation flow condition. The power chosen for analysis is the minimum value permitted by the recirculation system controls at rated recirculation drive flow, i.e., the drive flow necessary to achieve rated core flow at rated power.

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

3.9.5.2.4 Earthquake

The seismic loads acting on the structures within the RPV are based on a dynamic analysis, as described in Section 3.7. The seismic analysis is performed by coupling the lumped mass model of the RPV and internals, as described in Section 3.7, with the building model, to determine the system natural frequencies and mode shapes. The acceleration and load response are then determined by either the time history method or the response spectrum method.

In the time history method, the dynamic response is determined for each mode of interest and added algebraically for each instant of time. Resulting response time histories are then examined, and the maximum values of acceleration, shear, and moment are used for design calculations.

In the response spectrum method, the accelerations, shears, and moments are determined for each mode of interest. The square roots of the sum of the squares (SRSS) of these individual responses are then used for design calculations.

The detailed descriptions of the earthquake analysis are given in Section 3.7. The detailed description of the dynamic response analysis to these forcing functions is given in Section 3.9.2.5.

3.9.5.3 <u>Design Bases</u>

3.9.5.3.1 Safety Design Bases

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

1. They are 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 RPV

2. Deformation is limited to ensure that the control rods and the ECCS can perform their safety functions

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

3.9.5.3.2 Power Generation Design Bases

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

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

2. They are arranged to facilitate refueling operations

3. They are designed to facilitate inspection.

3.9.5.3.3 Design Loading Categories

Loading combinations for the core support structures are shown in Table 3.9-5. The basis for determining faulted loads on the reactor internals is shown for seismic loads in Section 3.7, and for pipe rupture loads in Sections 3.9.5.2.3 and 3.9.5.3.4. Table 3.9-4c shows loading combinations, analytical methods, and allowable and calculated stress values for highly stressed areas of selected reactor internal components. Table 3.9-4aa provides this same type of information for the control rod guide tubes.

The stress limits for core support structures and other ESF reactor internals are consistent with the ASME B&PV Code, Section III, Subsection NA-2140, and associated stress limits contained in ASME Addenda dated through Summer 1976. Level A, B, C, and D service limits defined in the ASME Winter 1976 Addenda (which replace normal, upset, emergency, and faulted condition limits) are not reflected in design documents for core support structures and other ESF reactor internals for this reactor. However, for these components, level A, B, C, and D service limits are judged to be equivalent to the normal, upset, emergency, and faulted loading condition limits. Therefore, for clarity, both sets of nomenclature are retained herein.

Stress intensity and other design limits are discussed in Sections 3.9.5.3.5 and 3.9.5.3.6. The core support structures that are fabricated as part of the RPV assembly are discussed in Section 3.9.3.1.2.

Subsection NG of Section III of the ASME B&PV Code was not issued at the time the core support structures for this plant were designed. The criteria presented in Section 3.9.5.3.6 were used in lieu of Subsection NG.

The design requirements for equipment classified as "other i.e., non-ESF, internals" e.g., steam dryer and shroud head, are specified by the designer with appropriate consideration of the intended service of the equipment and the expected plant and environmental conditions under which it operates. 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. Section 3.9.5.3.5 presents criteria for the ESF reactor internals.

3.9.5.3.4 Response of Internals Due to Steam Line Break Accident

The maximum pressure loads acting on the reactor internal components result from a steam line break upstream of the flow restrictor. On some components, the loads are maximum when operating at the minimum power associated with the maximum core flow. This has been substantiated by the analytical comparison of liquid line versus steam line 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 minimum power and maximum core flow. More realistically, the reactor would be at or near a full power condition, resulting in a lessening of the maximum pressure loads acting on the internal components.

3.9.5.3.5 Stress, Deformation, and Fatigue Limits for Engineered Safety Feature Reactor Internals (Except Core Support Structure)

The stress, deformation, and fatigue criteria listed in Tables 3.9-21, 3.9-22 3.9-23, and 3.9-24 are used as design limits for the ESF reactor internals. Other criteria established by applicable codes and standards for similar equipment, by manufacturers' standards, or by empirical methods based on field

experience and testing may also be used. For the quantity SF_{min} (minimum safety factor) appearing in the tables, the following values are used:

Service Level	Design Condition	SF_{min}
A	Normal	2.25
B	Upset	2.25
C	Emergency	1.5
D	Faulted	1.125

Components inside the RPV, such as control rods, that must move during an accident condition, have been examined for adequate clearances during emergency and faulted conditions. No mechanical clearance problems have been identified. The forcing functions applicable to the reactor internals are discussed in Section 3.9.2.5.

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

Stress, deformation, and fatigue criteria discussed in Section 3.9.5.3.3 and presented in Table 3.9-4 are used as design limits for the core support structures. These criteria are supplemented, where applicable, by the criteria for the reactor internals in Section 3.9.5.3.5. However, in no case are the criteria presented in Table 3.9-4 exceeded in the design of the core support structures.

3.9.5.4 SRP Rule Review

3.9.5.4.1 Acceptance Criterion II.b

Acceptance Criterion II.b of SRP Section 3.9.5 provides that the design and construction of the core support structures should conform to the requirements of the ASME B&PV Code, Section III, Subsection NG.

The HCGS core support structures were designed and purchased in 1971, prior to the 1974 issue of Subsection NG of the ASME B&PV Code. However, during the design of the HCGS core supports, an earlier draft of the ASME B&PV Code, Section III, Subsection NB was used as a guide in developing the design criteria. These criteria are presented in Section 3.9.5.3. Subsequent to the issuance of Subsection NG of the Code, comparisons were made to assure that the pre-NG design meets the equivalent level of safety as presented by Subsection NG.

3.9.5.4.2 Acceptance Criterion II.c

Acceptance Criterion II.c of SRP Section 3.9.5 provides that the design basis for reactor internals should meet guidelines of the ASME B&PV Code, Section III, Subsection NG-3000, and should not adversely affect the integrity of core support structures, as described in Subsection NG-1122 of the ASME B&PV Code, Section III.

Guidelines similar to Subsection NG-3000 of the ASME B&PV Code, Section III were used in the design of the reactor internals because Subsection NG-1122 was not written at the time the HCGS reactor internals were procured and designed.

As part of a generic study, assurance was obtained to demonstrate that failure of the nonsafety-related or nonsafety-graded reactor internals will not impair the function of other safety-related reactor internals.

3.9.5.5 Steam Dryer Structural Integrity

A plan of steam dryer monitoring, evaluation and inspection is established as part of the implementation of License Amendment No. 174. The three main elements of the plan related to steam dryer structural integrity are: (1) a slow and deliberate power ascension with defined hold points and durations; (2) a detailed power ascension monitoring and analysis program to trend steam dryer performance; and (3) a long term inspection/monitoring program to verify steam dryer performance at EPU conditions.

3.9.5.5.1 Power Ascension

Power above 3,339 MWt is initially increased at a rate of about 1 percent of 3,339 MWt per hour. Main steam line (MSL) strain gage and accelerometer vibration data are collected hourly during power ascension. At every 2.5 percent of 3,339 MWt step, MSL strain gage and accelerometer data, reactor pressure vessel water level instrumentation, and moisture carryover data, are evaluated against acceptance criteria. At every 5 percent CLTP plateau, the data is evaluated against the acceptance criteria; plant walkdowns are conducted; and information is forwarded to the NRC. The durations of certain hold points are defined in the Facility Operating License. For all other hold points, the duration is determined by the time required to obtain the specified data, complete the evaluation, and obtain the required level of approval to proceed.

3.9.5.5.2 Power Ascension Monitoring and Analysis

Level 1 and Level 2 acceptance criteria is established for MSL strain gage and accelerometer data and for moisture carryover data, where Level 1 requires that power be reduced to a previous acceptable level and Level 2 requires that power be held at that level with a re-evaluation of the data.

The Level 1 limit curves for MSL strain gages are based on not exceeding the ASME allowable alternating stress value on the dryer's limiting component. The Level 2 limit curves are based on not exceeding 80 percent of the allowable alternating stress value on the dryer. MSL strain gage limit curves are developed as follows:

- Collect in-plant MSL strain gage data.
- Calculate the stress ratio at the limiting steam dryer locations (loads are based on Acoustic Circuit Model (ACM) Rev. 4, and stresses are determined using the HCGS harmonic domain finite element model methodology).
- Generate revised limit curves based on the lowest calculated alternating stress ratio.

Reference 3.9-28 provides greater detail on the methodology for developing the strain gage limit curves including the NRC-accepted biases and uncertainties that must be applied.

3.9.5.5.3 Long Term Inspection/Monitoring

Station operating procedures are used to monitor operating moisture carryover conditions. Results are reviewed / evaluated on a defined basis to monitor moisture carryover conditions.

Steam dryer inspections are tracked, planned, and scheduled via normal commitment and planning processes already in place at the station. Steam dryer inspections and monitoring of plant parameters potentially indicative of steam dryer failure are conducted as recommended in Electric Power Research Institute (EPRI) Technical Report 1011463, "BWR Vessel and Internals Project, Steam Dryer Inspection and Flaw Evaluation Guidelines (BWRVIP-139)."

3.9.6 Inservice Testing of Pumps and Valves

Inservice testing of certain safety-related pumps and valves is accomplished in accordance with the requirements of 10CFR50, Appendix A, GDC 37, 40, 43, 46, 54, and 10CFR50.55a(g), using the date of commercial operation for determining the test interval.

The design includes access provisions for pre-service operational readiness testing of valves that comply, at the minimum, with the 1974 ASME B&PV Code, Section XI, with Addenda through Summer 1975.

The pumps and valves that require testing and the tests performed are determined in accordance with the 1983 ASME B&PV Code, Section XI, with Addenda through Summer 1983 (hereafter referred to as ASME XI, 83S83), Subsections IWP and IWV, to the extent practical. Tests conducted during each 120-month inspection interval comply with the requirements of the latest edition and addenda of the ASME B&PV Code incorporated or modified by reference in 10CFR50.55a, 12 months prior to the start of the 120-month inspection interval. The testing program verifies that the pumps and valves required for safety remain in a state of operational readiness to perform their safety-related functions throughout the lifetime of the plant.

Pumps required for safety, and provided with an emergency power source, are inservice tested, where applicable, in agreement with the requirements of the ASME B&PV Code, Section XI, Article IWP-1000. The reference values and periodic testing schedule for the inservice testing program are derived in accordance with the inservice testing procedures of IWP-3000. The surveillance requirements are defined by the Inservice Testing Program, as described in Section 6.0 of the Technical Specifications. Methods of measuring reference values and inservice values for pump parameters are in agreement with IWP-4000.

The pump test schedule, test method, and procedures for testing are presented separately from the FSAR in the HCGS inservice pump testing program. See Reference 3.9-22 for the Hope Creek IST program.

3.9.6.2 Inservice Testing of Valves

The inservice testing programs for valves whose function is required for safety are in the valve testing list, as required by IWV-1100. The program does not include those valves exempted by IWV-1200. The valves required to be tested in accordance with the rules of Subsection IWV are listed by type, identification number, ASME B&PV Code Class, and Section XI, Article IWV-2000 valve category. The initial periodic inservice valve testing complies with the provisions of IWV-3000 of ASME XI, 83S83.

Included in the valve testing list are valves in the normal or alternate flow path of the main process piping of systems required for safety. These valves are not required to change position to perform their safety-related function and are included in the list for administrative control to verify valve position quarterly or each time the valve is cycled.

The valve testing list and relief requests for valve testing are presented separately from the FSAR in the HCGS inservice valve testing program. See Reference 3.9-22 for the Hope Creek IST program. See Reference 3.9-23 regarding leak rate testing reactor coolant boundary valves at Hope Creek.

3.9.6.3 Relief Request

Where pump or valve testing proves to be impractical to meet the requirements of the inservice testing program of the ASME B&PV Code, Section XI, relief requests are submitted on a case by case basis to the NRC staff for review and approval.

3.9.7 References

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TABLE 3.9-1

PLANT EVENTS

Normal, Upset, and Testing Conditions		Number of Cycles
1.	Bolt-up(1)	123
2.	Design hydrostatic test	130
3.	Startup (100°F/h heatup rate)(2)	117
4.	Daily reduction to 75% power(1)	10,000
5.	Weekly reduction 50% power(1)	2000
6.	Control rod pattern changes(1)	400
7.	Loss of feedwater heaters	80
8.	OBE	10/50(3)
9.	Scram:	
	a. Turbine generator trip, feedwater on, isolation valves stay open	40
	b. Other scrams	140
10.	Reduction to 0% power, hot standby, shutdown (100°F/h cooldown rate)(2)	111
10a.	Alternate Flood-Up Event	53 (60-year plant life) 28 (40-year plant life)
11.	Unbolt	123
12.	Preoperational blowdown	10

	Number of Cycles
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[4]
b. Automatic blowdown	1[4]
c. Loss of feedwater pumps, isolation valves closed	5
d. Single safety/relief valve blowdown	8
16. Improper start of cold recirculation loop	1[4]
17. Sudden start of pump in cold recirculation loop	1[4]
18. Improper startup with reactor drain shut off	1[4]
Faulted Condition	
19. Pipe rupture and blowdown, including annulus pressurization	1[4]
20. SSE at rated operating conditions	1[4]

---

(1) Applies to RPV only.

(2) Bulk average vessel coolant temperature change in any consecutive one-hour period.

(3) 50 peak OBE cycles for NSSS piping; 10 peak OBE cycles for other NSSS equipment and components.

(4) The annual encounter probability of the one cycle events is $<10^{-2}$ for emergency and $<10^{-4}$ for faulted events.

TABLE 3.9-1a

PLANT EVENTS - FEEDWATER NOZZLES

	TRANSIENT	CATEGORY	NUMBER OF CYCLES
1	Boltup	Normal/upset	44
2	Design Hydrostatic Test	Testing	44
3	Startup	Normal/upset	117
4	Turbine Roll & Increase to Rated Power	Normal/upset	117
5	Daily Reduction to 75% Power	Normal/upset	6,667
6	Weekly Reduction to 50% Power	Normal/upset	1,233
7	Loss of Feedwater Heaters, Turbine Trip with 100% Steam Bypass	Normal/upset	3
8	Loss of Feedwater Heaters, Partial Feedwater Heater Bypass	Normal/upset	20
9	SCRAM, Turbine Generator Trip, Feedwater On, Isolation Valves Open, all other SCRAMs	Normal/upset	136
10	Reduction to 0% Power	Normal/upset	111
11	Hot Standby Normal/upset	Normal/upset	666
12	Initial Shutdown, Vessel Flooding, Final Shutdown	Normal/upset	111
13	Hydrostatic Test	Testing	1
14	Unbolt	Normal/upset	44
15	Pre-Op Blowdown	Normal/upset	10
16	Loss of AC Power, Natural Recirculation Re-Start	Normal/upset	5

TABLE 3.9-2

COMPARISON OF ME912 WITH ME643 AND ANALYTICAL RESULTS

Case	Program	ΔT_1	ΔT_2	$T_a - T_b$ (1)
		Temperature Gradients (1)		
450 to 553°F step	ME643	79.0	38.0	24.0
3-inch Sch 160, stainless	ME912	79.7	40.6	24.3
Thicknesses 1.50:1	Ref 3.9-4	82.0	41.0	-
408 to 100°F step	ME643	136.2	40.1	83.0
12-inch Sch 80 carbon	ME912	134.4	41.9	81.6
steel thicknesses	Ref 3.9-4	139.0	43.0	-
1.69:1				

(1) Defined in the ASME B&PV Code, Section III, Subsection NB-3650.

TABLE 3.9-3

COMPARISON BETWEEN ASME SAMPLE PROBLEM AND COMPUTER PROGRAM ME913 RESULTS[(1)]

	ME 913	Sample Problem[(2)]
Equation 9	20,810 psi	20,825 psi
Equation 10	65,567 psi	65,596 psi
Equation 11	128,950 psi	128,920 psi
Equation 12	39,536 psi	39,564 psi
Equation 13	23,152 psi	23,155 psi
Total usage factor	0.3439	0.3699

---

(1) Comparison made for butt welding tee, location 10.

(2) See Reference 3.9-9.

TABLE 3.9-4

INDEX

o. Standby Liquid Control Tank

p. ECCS Pumps

q. RHR Heat Exchanger

r. RWCU Pump

s. RCIC Turbine

t. RCIC Pump

u. Fuel Storage Racks

v. High Pressure Coolant Injection Pump

w. Control Rod Drive

x. Control Rod Drive Housing

y. Jet Pumps

z. LPCI Coupling

aa. Control Rod Guide Tube

bb. In-core Instrument Housing

cc. Reactor Vessel Support Equipment CRD Housing Support

dd. HPCI Turbine

ee. Fuel Assembly (Including Channel)

TABLE 3.9-4a

LOADING COMBINATIONS AND CRITERIA FOR
NSSS ASME B&PV CODE CLASS 1, 2, and 3
PIPING AND COMPONENTS

Load Case	Events(1)	ASME B&PV Code Service Limit
1	N	A
2	N+OBE+SOT(2)	B
3	N+SBA(3)	C
4	N+LOCA	D
5	N+LOCA+SSE	D

---

(1) Key to load definitions:

N - Normal load consisting of pressure, dead weight, and thermal loads.

OBE - Operating basis earthquake.

SOT - Systems operating transients.

SBA - Small break accident

LOCA - Loads associated with the design basis loss-of-coolant accident (LOCA). The LOCA analysis considers the effects of a break in the main stream, recirculation, or feedwater line.

SSE - Safe shutdown earthquake.

(2) The effects of pool dynamic loads due to LOCA and SRV have been assessed by the Mark I Owner's Group to be negligible on the NSSS vessel, internals, components, piping, and floor mounted and pipe mounted equipment.

(3) Other than the effect on long term containment temperature, SBA has negligible effects on vessel, supports, internals, piping, and floor mounted and pipe mounted equipment.

TABLE 3.9-4aa

CONTROL ROD GUIDE TUBE

Criteria	Loading	Primary Stress Type	Allowable Stress, psi	HISTORICAL INFORMATION Calculated Stress, psi
Control rod guide tube Primary Stress Limit - The allowable primary membrane stress, plus bending stress, is based on the ASME B&PV Code, Section III, for type 304 stainless steel tubing and SA351 Type CF8 casting (base) S_m = 16,000 psi @ 575°F				
For normal and upset condition: $S_{limit} = 1.5\ S_m = 24{,}000$ psi	1. External pressure 2. Vertical seismic and weight 3. Horizontal seismic 4. Lateral flow impingement 5. Vibration	Applying vertical seismic loads plus dead weight under normal and upset conditions, the maximum stress occurs at the guide tube base	24,000	16,340
For faulted condition: $S_{limit} = 2.4\ S_m = 38{,}400$ psi	1. External pressure 2. Vertical seismic and weight 3. Horizontal seismic 4. Lateral flow impingement 5. Vibration	Applying vertical seismic loads plus dead weight under faulted conditions, the maximum stress occurs at the guide tube base	38,400	21,763

TABLE 3.9-4b

REACTOR PRESSURE VESSEL AND SHROUD SUPPORT ASSEMBLY[1]

ASME B&PV Code Section III Primary Stress Limit Criteria	Load Case Number	Primary Stress Type	Allowable Stress, psi	Maximum Calculated Stress, psi
Vessel Support Skirt[3] (Material: SA533 Gr B CL1)				
Normal & upset condition:				
$P_m \le S_m$	Normal & upset condition loads:	Primary membrane	26,700	15,110
Sm = 26,700 @ 575°F	1. Normal loads 2. Upset pressure 3. OBE 4. SRV			
$P_L + P_b \le 1.5$ Sm		Primary membrane plus	40,050	28,700
S_m = 26,700 @ 575°F		bending		
Emergency condition[2]:				
$P_m \le S_y$	Emergency condition loads:	Primary membrane	42,600	18,460
S_y = 42,600 @ 547°F	1. Normal loads 2. Upset pressure 3. OBE 4. SRV			
$P_L + P_b \le 1.5\ S_y$		Primary membrane plus	63,900	34,820
S_y = 42,600 @ 547°F		bending		
Faulted condition[2]:				
$P_m \le S_y$	Faulted condition loads:	Primary membrane	42,600	16,680
S_y = 42,600 @ 547°F	1. Normal loads 2. Accident pressure 3. Jet reaction 4. Annulus pressurization 5. Safe shutdown earthquake			
$P_L + P_b \le 1.5\ S_y$		Primary membrane plus	63,900	44,754
S_y = 42,600 @ 547°F		bending		

Maximum cumulative usage factor: 0.2087 @ skirt knuckle

ASME B&PV Code Section III Primary Stress Limit Criteria	Load Case Number	Primary Stress Type	Allowable Stress, psi	Maximum Calculated Stress, psi
Shroud Support[4] (Material SB-168)				
Normal & upset condition:				
$P_m \le S_m$	Normal & upset condition loads:	Primary membrane	23,300	21,958
S_m = 23,300 @ 575°F	1. Normal loads 2. Upset pressure 3. OBE			
$P_L + P_b \le 1.5\ S_m$		Primary membrane plus	34,950	22,663
S_m = 23,300 @ 575°F		bending		
Emergency condition[2]:				
$P_m \le S_y$	Emergency condition loads:	Primary membrane	28,400	23,026
S_y = 28,400 @ 547°F	1. Normal loads 2. Upset pressure 3. OBE 4. Chugging			
$P_L + P_b \le 1.5\ S_y$		Primary membrane plus	42,600	25,056
S_y = 28,400 @ 547°F		bending		
Faulted condition[2]:				
$P_m \le 2S_y$	Faulted condition loads:	Primary membrane	46,600	39,485
$S_m\ S_y$ = 23,300 @ 547°F	1. Normal loads 2. Accident pressure 3. Safe shutdown earthquake 4. Acoustic pressure			
$P_L + P_b \le 1.5\ S_y\ 3\ S_m$		Primary membrane plus	69,900	41,617
$S_m\ S_y$ = 23,300 @ 547°F		bending		
Maximum cumulative usage factor is 0.672		At shroud cylinder		

ASME B&PV Code Section III Primary Stress Limit Criteria	Load Case Number	Primary Stress Type	Allowable Stress, psi	Maximum Calculated Stress, psi
RPV Feedwater Nozzle[3] (Material: SA508 CL safe-end)				
Normal & upset condition:				
$P_m \leq S_m$	Normal & upset condition loads:	Primary membrane	17,700	16,220
S_m - 17,700 @ 575°F	1. Normal loads 2. Upset pressure 3. Operating basis earthquake 4. SRV			
$P_L + P_b \leq 1.5\ S_m$		Primary membrane plus	26,550	22,930
S_m = 17,700 @ 575°F		bending		
Emergency condition[2]:				
$P_m \leq S_y$	Emergency condition loads:	Primary membrane	25,900	21,420
S_y = 25,900 @ 594°F	1. Normal loads 2. Upset pressure 3. Chugging 4. SRV-ADS			
$P_L + P_b \leq 1.5\ S_y$		Primary membrane plus	38,900	22,400
S_y = 25,900 @ 594°F		bending		
Faulted condition[2]:				
$P_m \leq 2.4\ S_m$	Faulted condition loads:	Primary membrane	42,480	28,300
S_m = 17,700 @ 575°F	1. Normal loads 2. Accident pressure 3. Chugging 4. SRV-ADS 5. Safe shutdown earthquake			
$P_L + P_b \leq 3.6\ S_m$		Primary membrane plus	63,720	33,740
S_m = 17,700 @ 594°F		bending		

Maximum cumulative usage factor: 0.12149 @ safe end

TABLE 3.9-4b (Cont)

ASME B&PV Code Section III Primary Stress Limit Criteria	Load Case Number	Primary Stress Type	Allowable Stress, psi	Maximum Calculated Stress, psi
CRD Penetration[3] (Material: SA-312, 304SS)				
Normal & upset condition:				
$P_m \leq S_m$ S_m = 14,150 @ 575°F	Normal & upset condition loads: 1. Normal loads 2. Upset pressure 3. OBE	Primary membrane	14,150	8,840
Emergency condition[2]:				
$P_m \leq 1.2\ S_m$ S_m = 14,150 @ 575°F	Emergency condition loads: 1. Normal loads 2. Upset pressure 3. Safe shutdown earthquake	Primary membrane	16,980	13,090
Faulted condition[2]				
$P_m \leq 2.4\ S_m$ S_m = 14,150 @ 575°F	Faulted condition loads: 1. Normal loads 2. Upset pressure 3. Safe shutdown earthquake	Primary membrane	33,960	13,090

Maximum cumulative usage factor: 0.021 @ CRD housing

(1) The vessel, support skirt, and shroud support, including legs, cylinder, and plates are furnished as a completed assembly by the vessel manufacturer.

(2) Value of S_m or S_y is shown depending upon the controlling criteria (e.g., 1.8 S_m or 1.5 S_y for emergency or faulted condition).

(3) In early 1984, a revised seismic input was generated to accommodate the effects of Unit 2 cancellation. The new seismic building basemat time histories were compared to the inputs used in the original calculation. The comparisons indicated no need to recalculate the maximum stresses, since the impact of the revised input was judged to be insignificant.

(4) Maximum calculated stresses as shown were generated based on the revised seismic building basemat time histories.

TABLE 3.9-4bb

IN-CORE INSTRUMENT HOUSING [1]

Criteria	Loading	Primary Stress Type	Allowable Stress, psi	HISTORICAL INFORMATION Calculated Stress, psi
Primary Stress Limit - The allowable primary membrane stress is based on ASME B&PV Code, Section III for Class I vessels, for type 304 austenitic stainless steel S_m = 16,660 psi @ 575°F				
For normal and upset conditions: $S_{limit} = S_m$ = 16,660 psi	1. Design pressure 2. OBE	Maximum membrane stress occurs at the outer surface of the vessel penetration	16,660	16,660
For faulted condition: $S_{limit} = 1.2\ S_m$ = 19,920 psi	1. Design pressure 2. SSE 3. Static weights	Maximum membrane stress occurs at the outer surface of the vessel penetration	19,920	19,920

(1) Analyzed to emergency condition limits.

TABLE 3.9-4c

REACTOR INTERNALS AND ASSOCIATED EQUIPMENT

ASME B&PV Code Section III Primary Stress Limit Criteria	Load Case Number	Primary Stress Type	Allowable Stress, psi	HISTORICAL INFORMATION Maximum Calculated Stress, psi
Top Guide - Highest Stressed Beam (Material: 304 SS)				
Normal & upset condition:[3]				
	Normal & upset condition loads: 1. Normal loads 2. Normal pressure 3. Operating basis earthquake 4. SRV	Primary membrane plus bending	25,350	24,916
$P_L + P_b \leq 1.5\ S_m$ S_m = 16,900 @ 550°F				
Emergency condition[1]:				
	Emergency loading condition: 1. Normal loads 2. Upset pressure 3. Chugging 4. SRV	Primary membrane plus bending	38,025	24,338
$P_L + P_b \leq 2.25\ S_m$ S_m= 16,900 @ 550°F				
Faulted condition[1]:				
	Faulted condition loads: 1. Normal loads 2. Faulted pressure 3. Safe shutdown earthquake 4. SRV 5. Chugging	Primary membrane plus bending	50,700	27,842
$P_L + P_b \leq 3\ S_m$ S_m = 16,900 @ 550°F				

HISTORICAL INFORMATION

Maximum cumulative usage factor: 0.435 at beam slot

ASME B&PV Code Section III Primary Stress Limit Criteria	Load Case Number	Primary Stress Type	Allowable Stress, psi	HISTORICAL INFORMATION Maximum Calculated Stress, psi
Core Plate (Ligament in Top Plate)[2] (Material: 304 SS)				
Normal & upset condition:				
	Normal & upset condition loads: 1. Normal loads 2. Upset pressure 3. Operating basis earthquake 4. SRV	Primary membrane plus bending	25,350	10,190
$P_L + P_b \leq 1.5\ S_m$				
$S_m = 16.9$ @ 550°F				
Emergency condition[1]				
	Emergency condition loads: 1. Normal loads 2. Upset pressure 3. Operating basis earthquake 4. SRV	Primary membrane plus bending	38,025	10,190
$P_L + P_b \leq 2.25\ S_m$				
$S_m = 16.9$ @ 550°F				
Faulted condition[1]:				
	Faulted condition loads: 1. Normal loads 2. Accident pressure 3. Safe shutdown earthquake 4. Chugging 5. SRV	Primary membrane plus bending	50,700	12,170
$P_L + P_b \leq 3\ S_m$				
$S_m = 16.9$ @ 550°F				

HISTORICAL INFORMATION

Maximum cumulative usage factor: 0.111 @ core plate stud

ASME B&PV Code Section III Primary Stress Limit Criteria	Load Case Number	Primary Stress Type	Allowable Stress, psi	HISTORICAL INFORMATION Maximum Calculated Stress, psi
Differential Pressure Sensing Lines[2] (Material: 304 SS)				
Normal & upset condition:				
$P_L + P_b \leq 3.0\ S_m$	Normal & upset condition loads:	Primary membrane plus bending	41,850	29,830
S_m = 13,950 @ 550°F	1. Normal loads 2. Normal pressure 3. OBE			
Emergency condition[1]:				
$P_L + P_b \leq 2.25\ S_m$	Emergency condition loads:	Primary membrane plus bending	31,390	21,440
S_m = 13,950 @ 550°F	1. Normal loads 2. Faulted pressure 3. Jet reaction 4. Annulus pressurization 5. Safe shutdown earthquake			
Faulted condition[1]				
$P_L + P_b \leq 3\ S_m$	Faulted condition loads:	Primary membrane plus bending	41,850	21,440
S_m = 13,950 @ 550°F	1. Normal loads 2. Faulted pressure 3. Jet reaction 4. Annulus pressurization 5. Safe shutdown earthquake			

HISTORICAL INFORMATION

Maximum cumulative usage factor: <.05 at elbow

---

(1) Value of S_m or S_y is shown depending upon the controlling criteria, e.g., 1.8 S_m or 1.5 S_y for emergency or faulted condition.
(2) In early 1984, a revised seismic input was generated to accommodate the effects of Unit 2 cancellation. The new seismic building basemat time histories were compared to the inputs used in the original calculation. The comparisons indicated no need to recalculate the maximum stresses, since the impact of the revised input was judged to be insignificant.
(3) Maximum calculated stresses as shown were generated based on the revised seismic building basemat time histories.

TABLE 3.9-4cc

REACTOR PRESSURE VESSEL SUPPORT EQUIPMENT

Criteria	Loading	Location	Allowable Stress, psi	Calculated Stress, psi
RPV Support (ring girder)				
Primary stress limit:				
AISC specification for the design, fabrication, and erection of structural steel for buildings				
For normal & upset conditions: AISC allowable stresses, but without the usual increase for earthquake loads	Normal and upset condition:	Top flange	22,000	f_b = 10,000
	1. Dead loads	Bottom flange	22,000	f_b = 10,000
	2. Operating basis earthquake	Vessel to girder bolts	54,000	f_t = 35,200
	3. Loads due to scram		14,000	f_v = 4450
For emergency conditions: 1.5 x AISC allowable stresses	Emergency condition:	Top flange	33,000	f_b = 22,000
		Bottom flange	33,000	
	1. Dead loads	Vessel to girder bolts	81,000	f_b = 20,000
	2. Safe shutdown earthquake		21,000	
	3. Loads due to scram			f_t = 70,400
				f_v = 8,900
For faulted conditions: 1.67 x AISC allowable stresses for structural steel members	Faulted condition:	Top flange	36,800	f_b = 28,000
	1. Dead loads	Bottom flange	36,800	f_b = 23,400
	2. Safe shutdown earthquake			
	3. Jet reaction load	Vessel to girder bolts	125,000	f_t = 94,000
			72,000	f_v = 9,650
RPV Stabilizer				
Primary stress limit:				
Section III, Subsection NF of the ASME B&PV Code - linear type support				
For emergency conditions: 1.3 x NF allowable stresses (1)	Emergency condition:	Bracket	30,200	f_b = 18,800
	1. Spring preload	Bracket	18,300	f_v = 5,500
	2. Safe shutdown earthquake	Rod (stress index)	1.0	
				0.97
	Fouled condition:	Bracket		
	1. Annulus Pressurization Loads		12,800	P_m = 75,505
	2. Safe shutdown earthquake		63,900	$P_m + P_b$ = 38,761

Criteria	Loading	Location	Allowable Stress, psi	Calculated Stress, psi
CRD Housing[2] Supports - Beams				
Based on AISC specification for the design, fabrication, and erection of structural steel for buildings				
f_y @ 150°F = 36,000 psi				
For normal and upset condition:	Normal and upset loads: [3]			
f_a = 0.60 f (tension)	(Negligible)			
f_b = 0.66 f (bending)				
f_v = 0.40 f (shear)				
For emergency condition:	Emergency loads: [3] (negligible)			
For faulted condition:	Faulted loads:			
$f_{a(limit)}$ = 1.5 x 0.60 x f (tension)	1. Weight of structure	Top chord	33,000	f_a = 12,200
$f_{b(limit)}$ = 1.5 x 0.60 x f (bending)	2. Impact force from failure of CRD housing	Top chord	33,000	f_b = 16,500
$f_{v(limit)}$ = 1.5 x 0.40 x f (shear)				
CRD Housing Supports - Grid Structure				
Based on AISC specification for the design, fabrication, and erection of structural steel for buildings				
f_y @ 150°F = 46,000 psi				
For normal and upset condition:	Normal and upset loads: [3]			
f_a = 0.60 f (tension)	(negligible)			
f_b = 0.66 f (bending)				
f_v = 0.40 f (shear)				
For emergency condition:	Emergency loads: [3] (negligible)			

TABLE 3.9-4cc (Cont)

Criteria	Loading	Location	Allowable Stress, psi	Calculated Stress, psi
For faulted condition:	Faulted loads:			
$f_{a(limit)} = 1.5 \times 0.60 \times f$ (tension)	1. Weight of structure		46,000	$f_b = 40,700$
$f_{b(limit)} = 1.5 \times 0.66 \times f$ (bending)	2. Impact force from failing of CRD housing		27,600	$f_v = 11,100$
$f_{v(limit)} = 1.50 \times 0.40 \times f$ (shear)				

(1) The emergency condition provides the least margin and, therefore, is the only case analyzed.
(2) f_y = Material yield strength.
(3) Dead weights and earthquake loads are very small compared to impact force.

TABLE 3.9-4d

REACTOR WATER CLEANUP HEAT EXCHANGERS

Criteria	Loading	Component	Minimum Thickness Required, in.	Actual Thickness, in.
	Regenerative Heat Exchangers			
Closure bolting				
Bolting requirements are calculated per rules of ASME B&PV Code, Section III Primary stress limit for SA-193-B7, S = 20,000 psi	Design basis loads consisting of: 1. Design pressure 2. Design temperature 3. Design gasket load	Bolting - channel to shell flange	NA	1.25 dia
Wall thickness				
Wall thickness requirements are calculated per rules of ASME B&PV, Section III, Class 3, and TEMA Class C	1. Design pressure 2. Design temperature	Shell	0.858	0.938
		Shell head	0.704	1.00
		Channel shell	0.858	1.00
		Channel cover	3.53	3.75
		Tube sheet	2.87	2.875
Primary stress limit for: Carbon steel, S = 17,500 psi Austenitic stainless steel, S = 15,900 psi		Tubes	0.084	0.095
Nozzle loads				
The maximum forces and moments due to pipe reactions shall not exceed the allowable limits	Design basis loads consisting of: 1. Design pressure 2. Design temperature 3. Dead weight 4. Thermal expansion 5. Seismic (Class II basis)	Nozzle N1 (tube inlet)	F_o = 3760 lb M_o = 15,100 in.-lb	F = 1474 lb M = 40,920 in.-lb
		Nozzle N2 (tube outlet)	F_o = 3760 lb M_o = 15,100 in.-lb	F = 198 lb F = 10,498 in.-lb
		Nozzle N3 (shell inlet)	F_o = 3760 lb M_o = 15,100 in.-lb	F = 56 lb M = 1808 in.-lb
		Nozzle N4 (shell outlet)	F_o = 3760 lb M_o = 15,100 in.-lb	F = 3453 lb M = 44,268 in.-lb

TABLE 3.9-4d (Cont)

Criteria	Loading	Component	Minimum Thickness Required, in.	Actual Thickness, in.
		Nonregenerative Heat Exchangers		
Closure bolting				
Bolting requirements are calculated per rules of ASME B&PV Code, Section III	Design basis loads consisting of: 1. Design pressure 2. Design temperature 3. Design gasket load	Bolting - channel to shell flange	NA	1.25 dia
Primary stress limit for SA-193-B7, S = 20,000 psi				
Wall thickness				
Wall thickness requirements are calculated per rules of ASME B&PV, Section III, Class 3, and TEMA Class C	1. Design pressure 2. Design temperature	Shell	0.118	0.375
		Shell head	0.104	0.375
		Channel shell	0.917	1.00
		Channel cover	3.53	3.75
		Tube sheet	2.87	2.875
Primary stress limit for:		Tubes	0.056	0.065
Carbon steel, S = 17,500 psi				
Austenitic stainless steel, S = 15,900 psi				
Nozzle loads				
The maximum forces and moments due to pipe reactions shall not exceed the allowable limits	1. Design pressure 2. Design temperature 3. Dead weight 4. Thermal expansion 5. Seismic (Class II basis)	Nozzle N1 (tube inlet)	F_o = 654 lb M_o = 2620 in.-lb	F = 126 lb M = 2206 in.-lb
		Nozzle N2 (tube outlet)	F_o = 654 lb M_o = 2620 in.-lb	F = 268 lb M = 4276 in.-lb
		Nozzle N3 (shell inlet)	F_o = 654 lb M_o = 3920 in.-lb	F = 396 lb M = 4281 in.-lb
		Nozzle N4 (shell outlet)	F_o = 654 lb M_o = 3920 in.-lb	F = 126 lb M = 2206 in.-lb

TABLE 3.9-4d (Cont)

---

(1) F_o = Allowable resultant nozzle force, lb.

F_x, F_y, F_z = Actual orthogonal nozzle forces in x, y, and z directions.

$\sqrt{F_x^2 + F_y^2 + F_z^2}$ must be equal to or less than F_0

M_0 = Allowable resultant nozzle moment, in.-lb.

M_x, M_y, M_z = Actual orthogonal nozzle moments in x, y, and z directions.

$\sqrt{M_x^2 + M_y^2 + M_z^2}$ must be equal to or less than M_o.

(2) Calculated loads were evaluated and accepted by General Electric per letter GB-86-27 dated 1/31/86.

TABLE 3.9-4dd

HPCI TURBINE DESIGN CALCULATIONS

Turbine Part	Calculated	Allowable
Pressure boundary castings		
Stop valve	8,975 psi	14,000 psi
Turbine inlet (high press)	6,550 psi	14,000 psi
Turbine wheel case (low press)	6,000 psi	14,000 psi
Pressure boundary bolting		
Stop valve	17,600 psi	20,000 psi
Turbine flange	18,290 psi	20,000 psi
Nonpressure boundary components		
Turbine shaft	5,000 psi	50,000 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

(Historical Information)

TABLE 3.9-4e

ASME CODE CLASS 1 MAIN STEAM PIPING AND PIPE-MOUNTED EQUIPMENT - HIGHEST STRESS LOCATION

Acceptance Criteria	Limiting Stress Type	Calculated Stress(1) or Usage Factor	Allowable Limits	Ratio Actual/ Allowable	Loading	Identification of Locations of Highest Stress Points - Node Point Numbers
ASME B&PV Code Section III, NB-3600						
Design condition: $Eq.\ 9 \geq 1.5\ S_m$	Primary	24,064 psi	26,500 psi	0.91	1. Pressure 2. Weight 3. OBE	Loop B (009) Lug, Riser
Service levels A & B (normal & upset) condition: $Eq.\ 12 \geq 3.0\ S_m$	Secondary	51,854 psi	53,100 psi	0.98	1. Pressure 2. Weight 3. Thermal expansion 4. SRV (Pedestal, Accln:)	Line D (039 F) Elbow - end
Service levels A & B (normal & upset) condition: $Eq.\ 13 \geq 3.0\ S_m$	Primary plus secondary (except thermal expansion)	28,772 psi	53,100 psi	0.54	1. Pressure 2. Weight	Line C (051) Elbow - end
Service levels A & B (normal and upset) condition: Cumulative usage factor	NA	0.05	1.0	0.05		Line D (039 F) Elbow - end
ASME B&PV Code Section III, NB-3600						
Service level B (upset) condition: $Eq.\ 9 \geq 1.8\ S_m\ \&\ 1.5\ S_y$	Primary	25,778 psi	31,860 psi	0.81	1. Pressure 2. Weight 3. OBE 4. Turbine MSV closure (TSVC)	Line D Lug Riser
Service level C (emergency) condition: $Eq.\ 9 \geq 2.25\ S_m\ \&\ 1.8\ S_y$	Primary	25,464 psi	39,825 psi	0.64	1. Pressure 2. Weight 3. SRV (acoustic wave)	Line B (009) Lug Riser

THIS TABLE CONTAINS HISTORICAL DATA ONLY AND IS NO LONGER UPDATED.
FOR CURRENT INFORMATION, SEE THE LATEST APPLICABLE STRESS CALCULATIONS.

(Historical Information)

TABLE 3.9-4e (Cont)

Acceptance Criteria	Limiting Stress Type	Calculated Stress(1) or Usage Factor	Allowable Limits	Ratio Actual/ Allowable	Loading	Identification of Locations of Highest Stress Points - Node Point Numbers
Service level D (faulted) condition:					1. Pressure 2. Weight 3. SSE	Line B (009) Lug Riser
Eq. 9 ≥ 3.0 S_m	Primary	26,030 psi	53,100 psi	0.49	4. LOCA (AP)	

(1) Appropriate loading combinations of Table 3.9-5a were considered, and the calculated stresses or loads are reported for the governing load combinations.

THIS TABLE CONTAINS HISTORICAL DATA ONLY AND IS NO LONGER UPDATED.
FOR CURRENT INFORMATION, SEE THE LATEST APPLICABLE STRESS CALCULATIONS.

TABLE 3.9-4ee

FUEL ASSEMBLY (INCLUDING CHANNEL)

Acceptance Criteria	Loading	Primary Load Type	Calculated Peak[2,3] Acceleration	Evaluation Basis Acceleration
Acceleration Envelope	Horizontal Direction: 1. Peak Pressure 2. Safe Shutdown Earthquake 3. Annulus Pressurization	Horizontal Acceleration Profile	2.5G	(1,3)
	Vertical Direction: 1. Peak Pressure 2. Safe Shutdown Earthquake	Vertical Accelerations	0.3G	(1,3)

(1) Evaluation Basis Accelerations and Evaluations are contained in NEDE-21175-P and NEDE-21175-3-P.

(2) For the most limiting load combination, the fuel assembly gap opening for Hope Creek is expected to be negligible. This is based on an assessment comparing the positive net hold down forces to those of other plants for which the calculated fuel assembly gap opening is found to be negligible.

(3) Calculated peak acceleration values apply to GE fuel. Evaluation basis and acceptance limits for ABB fuel are contained in Reference 3.9-25.

(Historical Information)

TABLE 3.9-4f

ASME CODE CLASS 1 MAIN STEAM PIPING AND PIPE MOUNTED EQUIPMENT - HIGHEST STRESSED EQUIPMENT

Component/Load Type	Highest Calculated Load(1)	Allowable Load	Ratio Calculated/ Allowable	Loading	Identification of Equipment With Highest Loads
Service level B	21,682 lb	50,000 lb	0.434	1. OBE 2. Turbine MSV closure (TSVC)	Main steam line C snubber SSC8
Service level D	23,157 lb	75,000 lb	0.309	1. SSE 2. LOCA (AP)	Main steam line C snubber SSC8
Safety/Relief Valve - horizontal acceleration Service level D	2.92 g	8.0 g	0.37	1. SRV (acoustic wave 2. SSE	Main steam line D SRV Inlet (306)
Safety/Relief Valve - vertical acceleration Service level D	2.23 g	6.0 g	0.37	1. SRV (acoustic wave 2. SSE	Main steam line B SRV Inlet (303)
Main Steam Isolation Valve - inlet/outlet - Axial Service level A	7,802 psi	13,280 psi	0.59	1. Normal loads 2. Turbine MSV closure (TSVC) 3. SSE	Main steam line D Outlet (063)
Main Steam isolation Valve - Inlet/outlet-Bending Service level D	9,550 psi	13,280 psi	0.72	1. Normal loads 2. Turbine MSV closure (TSIV) 3. SSE	Main steam line B Inlet (070)
Main Steam Isolation Valve - bonnet moment Service level D	610,529 in.-lb	1,618,000 in.-lb	0.38	1. Normal loads 2. LOCA (AP) 3. SSE	Main steam line A MSIV Bonnet (073)

(1) Appropriate loading combinations of Table 3.9-5a were considered, and the calculated stresses or loads are prepared for the governing loading combinations.

THIS TABLE CONTAINS HISTORICAL DATA ONLY AND IS NO LONGER UPDATED.
FOR CURRENT INFORMATION, SEE THE LATEST APPLICABLE STRESS CALCULATIONS.

TABLE 3.9-4g

ASME CODE CLASS 1 RECIRCULATION PIPING AND PIPE MOUNTED EQUIPMENT - HIGHEST STRESS LOCATION

Acceptance Criteria	Limiting Stress Type	Stress(1) or Usage Factor	Allowable Limits	Calculated Actual/ Allowable	Loading	Identification of Locations at Highest Stress Points - Node Point Numbers
ASME B&PV Code Section III, NB-3600						
Design condition: Eq.9 ≥ 1.5 S_m	Primary	15,650 psi	25,013 psi	0.62	1.Pressure 2.Weight 3.OBE	Lug suction - Loop B (006)
Service levels A & B (normal & upset) condition: Eq. 12 ≥ 3.0 S_m	Secondary	30,417 psi	50,025 psi	0.61	1.Pressure 2.Weight 3.Thermal expansion 4.OBE	RHR tee - Loop B (601)
Service levels A & B (normal & upset) condition: Eq. 13 ≥ 3.0 S_m	Primary plus secondary (except thermal expansion)	30,296 psi	50,025 psi	0.61	1.Pressure 2.Weight 3.OBE	Sweepolet discharge header - Loop A (108)
Service levels A & B (normal and upset) condition:						
Cumulative usage factor	NA	0.02	1.0	0.02		RHR tee - Loop B (601)
Service level B (upset) condition: Eq. 9 ≥ 1.8 S_m& 1.5 S_y	Primary	19,863 psi	28,596 psi	0.69	1.Pressure 2.Weight 3.OBE	RHR tee - Loop A (601)
Service level C (emergency) condition: Eq. 9 ≥ 2.25 S_m& 1.8 S_m	Primary	17,527 psi	34,315 psi	0.51	1.Pressure 2.Weight	Lug suction - Loop B (006)
Service level D (faulted) condition: Eq. 9 ≥ 3.0 S_m	Primary	20,316 psi	38,128 psi	0.53	1.Pressure 2.Weight 3.SSE 4.LOCA (AP)	RHR tee - Loop A (601)

(1) Appropriate loading combinations of Table 3.9-5a were considered and the calculated stresses are reported for the governing loads.

NOTE: THIS TABLE CONTAINS HISTORICAL DATA ONLY AND IS NO LONGER UPDATED.
FOR CURRENT INFORMATION, SEE THE LATEST APPLICABLE STRESS CALCULATION.

TABLE 3.9-4h

ASME CODE CLASS 1 RECIRCULATION PIPING AND PIPING MOUNTED EQUIPMENT - HIGHEST STRESSED EQUIPMENT

Component/Load Type	Highest Calculated Load(1)	Allowable Load	Ratio Calculated/ Allowable	Loading	Identification of Equipment With Highest Loads
Service level B	33,244 lb	100,000 lb	0.332	1. OBE	Snubber SA11 - loop A
Service level D	44,310 lb	150,000 lb	0.295	1. LOCA (AP) 2. SSE	Snubber SA11 - loop A
Discharge valve:					
Flange Moment Service Level B	101,543 in-lb	2,067,952 in-lb	0.05	1. Normal loads 2. OBE	Loop A
Flange Moment Service Level D	268,036 in-lb	2,067,952 in-lb	0.13	1. Normal loads 2. LOCA (AP) 3. SSE	Loop A
Suction valve:					
Flange Moment Service Level B	97,366 in-lb	2,067,952 in-lb	0.05	1. Normal Loads 2. OBE	Loop A
Flange Moment Service Level D	263,858 in-lb	2,067,952 in-lb	0.13	1. Normal loads 2. LOCA (AP) 3. SSE	Loop B
Recirculation pump/ motor:					
Horizontal acceleration - Level D	0.86 g	2.7 g	0.32	1. Normal loads 2. SSE 3. AP	Loop A pump motor C.G.
Vertical acceleration - Level D	0.42 g	2.7 g	0.16		Loop B pump C.G.

(1) Appropriate loading combinations of Table 3.9-5a were considered and the calculated stresses are for the governing loading combinations.

NOTE: THIS TABLE CONTAINS HISTORICAL DATA ONLY AND IS NO LONGER UPDATED.
FOR CURRENT INFORMATION, SEE THE LATEST APPLICABLE STRESS CALCULATION.

(Historical Information)

TABLE 3.9-4i

MAIN STEAM SAFETY/RELIEF VALVES (PILOT-OPERATED)

ASME Code, Section III, 1968, Including Addenda through Summer 1970

Topic	Method of Analysis	Target Rock 7567F Analysis	Allowable Value	Calculated
1. Body inlet and outlet flange stresses	$S_H = \frac{fMo}{Lg_1^2B} + \frac{PB}{4go} < 1.5\,S_m$	P_D (target rock) = P (codes)	$1.5\,S_m$ = 29,100 psi	Inlet: $S_H = 1.2\,S_m$ = 0.77 (allowable)
Note for Topics 1 and 2:	$S_R = \frac{(4te\,/\,3 + 1)Mo}{Lt^2\,B} < 1.5\,S_m$			
Design pressures:	$S_T = \frac{TMo}{t^2\,B} - Z\,S_R < 1.5\,S_m$			$S_R = 0.52\,S_m$ = 0.35 (allowable)
P_a = 1375 psig (inlet) P_b = 625 psig (outlet)				S_T 1.2 S_m = 0.76 (allowable)
These are the equivalent max anticipated pressures under all operating conditions. Analyses include applied moments of: M = 400,000 in.-lb (inlet) and M = 300,000 in.-lb (outlet) Actual tested capability (including accelerations and moments) is as described in Topic No. 11 The analyses also include consideration of seismic, operational, and flow reaction forces. Allowable vs. tested capabilities are provided in Topic No. 12 No. 12	where: S_H = Longitudinal hub wall stress, psi S_R = Radial "flange" stress, psi S_T = Tangential "flange" stress, psi	Body material: A105 Gr. II S_m = 19,400 psi (500°F, equivalent inlet and outlet temperature)		Outlet: S_H= 0.36 S_m = 0.24 (allowable) S_R = 0.5 S_m = 0.33 (allowable) S_T = 1.36 S_m = 0.91 (allowable)

THIS TABLE CONTAINS HISTORICAL DATA ONLY AND IS NO LONGER UPDATED.
FOR CURRENT INFORMATION, SEE THE LATEST APPLICABLE STRESS CALCULATIONS.

TABLE 3.9-4i (Cont)

Topic	Method of Analysis	Target Rock 7567F Analysis	Allowable Value	Calculated
2. Inlet and outlet stud area requirements See above note.	Total cross-sectional area shall exceed the greater of: $Am_1 = \frac{Wm_1}{Sb}$, or $Am_1 = \frac{Wm_1}{Sb}$ $Am_2 = \frac{Wm_2}{Sa}$ $\quad Am_2 = \frac{Wm_2}{Sn}$ \} *		Inlet: Am_1 ($>Am_2$) $=Am$ $= 8.02\ in^2$ Outlet: $Am = 4.73\ in^2$	Inlet: A_b (actual area) $= 1.72$ Am (required min) Outlet: $A_b = 2.04$ Am

THIS TABLE CONTAINS HISTORICAL DATA ONLY AND IS NO LONGER UPDATED.
FOR CURRENT INFORMATION, SEE THE LATEST APPLICABLE STRESS CALCULATIONS.

TABLE 3.9-4i (Cont)

Topic	Method of Analysis	Target Rock 7567F Analysis	Allowable Value	Calculated
	where:	Bolting material: SA193 Gr B7		
	Am_1 = total required bolt (stud) area for operating condition	*Where Am (required minimum) is the greater of Am_1 and Am_2 and A_b (actual bolt area) must exceed Am.		
	Am_2 = total required bolt (stud) area for gasket seating			
3. Body wall thickness	1. Valve wall thickness criterion:	Section at inlet:		t_{ACT}= 1.67 (t_{RQD})
	$t_{min} < t_A$	$t_{RQD} < t_{ACT}$		
	where:	Section at middle of body	t = 0.67 in.	t_{ACTo} = 1.28 (t_{RQD})
	t_{min} = lated thickness requirement, including corrosion allowance	minimum calcu-	t_{RQD} < greater than t_m at the section under consideration.	Actual thickness
	t_A = Actual wall thickness			
	(Note: This t_{min} is t_m per notation of the codes.)			
	2. Cyclic rating:			
	Thermal			
	$I_t = \sum \frac{Nri}{Ni}$	$I_t = \sum \frac{Nri}{Ni}$ (i=1,2,&3)	I_t (max) = 1.0	I_t = 0.33 = 0.33 (allowable)
	Fatigue			
	Na calculation as based on Sa, where Sa is defined as the larger of:	Na calculation as based on $S_a = S_{a2}$ ($>S_{pi}$), where S_a (Target Rock) = S_a (codes)	Na $\geq$ 2000 cycles	Na (based on S_{a2}) = 1.8 x 10 cycles: .. satisfies criterion
	$S_{p1} = (2/3)Qp + \frac{Peb}{2} + Q_{T2} + 1.3\,Q_{T1}$ (Uses same notation as codes)			

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FOR CURRENT INFORMATION, SEE THE LATEST APPLICABLE STRESS CALCULATIONS.

Topic	Method of Analysis or	Target Rock 7567F Analysis	Allowable Value	Calculated
	$S_{p2}=0.4\ Qp+\frac{K}{2}\ (Peb+2Q_T3)$ where: S_{p1} = Fatigue stress intensity at inside surface of crotch, psi. S_{p2} = Fatigue stress intensity at outside surface of crotch, psi			
4. Bonnet flange stresses (body side)	$S_{H1} = \frac{PB_1}{\pi g_1} - \frac{6M\ H_2}{+ \pi R_1 R_2}$ (longitude hub stress adjacent to flange)	$S_H < 1.5$ Sm; $S_R < 1.5$ Sm; $S_T < 1.5$ Sm	1.5 Sm = 29,100 psi	S_H = 0.82 Sm = 0.55 (allowable); S_R = 0.5 S_m = 0.33 (allowable)

$$S_{H2} = \left[\frac{Q}{\pi B_1 t} + P\left[Z + V \pm\right.\right.$$

(Target Rock) = P (codes)

$$\frac{E + B}{B_1} + \frac{0.075\ PB_1}{g_1} + \frac{1.8\ M\ H}{\pi B_1\ g\ 2_1}$$

S_T = 0.27 Sm
= 0.18 (allowable)

(circumferential stress in hub adjacent to flange)

Material: A105 Gr II.

Sm = 19,400 psi
(@ 500°F)

(@ bolt circle)

$$S_R = \left[\frac{Q}{\pi B_1 t} + P\right] \pm \frac{6M}{\pi B_1 t} 7$$

(adjacent to hub)

$$S_T = \left[\frac{Q}{\pi B_1 t}\quad P\right] Z \pm \frac{Et\theta}{\pi B_1} R + \frac{1.8\ M_S}{\pi B_1 t^2}$$

THIS TABLE CONTAINS HISTORICAL DATA ONLY AND IS NO LONGER UPDATED.
FOR CURRENT INFORMATION, SEE THE LATEST APPLICABLE STRESS CALCULATIONS.

Topic	Method of Analysis	Target Rock 7567F Analysis	Allowable Value	Calculated
5. Bonnet flange stresses (bonnet side)	Using Roark's formulas for stress and strain, Table X, 4th. Edition, superposition of cases 2 and 3:	$S_R < 1.5\ S_m$ $S_T < 1.5\ S_m$	1.5 Sm = 29,100 psi	$S_R = S\ \ = 1.27\ S_m$ = 0.85 (allowable)

$$S_R = S_T = \frac{-3W}{2\ wmt^2}\left[m + (m+1)\log\frac{a}{r_\sigma} - (m-1)\ \frac{r_o^2}{4\ n^2}\right]$$

$$S_R = S_T = \frac{-3W}{2\ wmt^2}\left[1/2\ \ (m-1) + (m+1)\log\frac{a}{r_\sigma} - (m-1)\frac{ro^2}{2\ a^2}\right]$$

Material: A105 Gr II

Sm = 19,400 psi (@ 500°F)

Topic	Method of Analysis	Target Rock 7567F Analysis	Allowable Value	Calculated
6. Bonnet stud area requirements	Total cross-sectional area shall exceed: $Am_1 = \frac{Wm_1}{Sb}$ where: Am_1 = Total required bolt (stud) area	$Am_1 = \frac{Wm_1}{Sb}$ Bolting material: SA 193 Gr B7	$Am_1 = 9.839\ in.^2 = Am$	Ab (actual) = 1.044 Am (required minimum)
7. Bonnet wall thickness	Using Roark's formulas for stress and strain, Table XIII, case 35, considering the circumferential stress, S_2 (the governing stress), and setting equal to Sm: where: P = design pressure a = inside diameter b = outside diameter	$t_m < t_a$	tm = 0.119 in.	ta = 3.75 t

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FOR CURRENT INFORMATION, SEE THE LATEST APPLICABLE STRESS CALCULATIONS.

TABLE 3.9-4i (Cont)

Topic	Method of Analysis	Target Rock 7567F Analysis	Allowable Value	Calculated
8. Pilot valve housing flange	$S_H = \frac{FM_o}{Lg^2B}$ $S_R = \frac{(4t\ \theta\ 3+1)\ M}{Lt^2B}$ $S_T = \frac{TMo}{t2B} - Z\ S_R$ where: S_H = Longitudinal "hub" wall stress, psi S_R = Radial "flange" stress, psi S_T = Tangential "flange" stress, psi	$S_H < 1.5$ Sm $S_R < 1.5$ Sm $S_T < 1.5$ Sm Material A105 Gr II Sm = 19,400 psi (@ 500°F)	1.5 Sm = 29,100 psi	S_H = 0.54 Sm = 0.36 (allowable) S_R = 0.36 Sm = 0.24 (allowable) S_T = 0.30 Sm = 0.20 (allowable)
9. Pilot valve body flange stress	Using Roark's formulas for stress and strain, 4th edition, Table X, Case 2,	$S_R = S_T < S_H$ Material: A105 Gr II Sm = 19,400 psi (@500°F)	Sm = 19,400	$S_R = S_T$ = 0.34 Sm = 0.34 (allowable)

$$S_R = S_T = \frac{-3W}{2^{\pi}\ m +^{2}} [m + (m+1) \log \frac{a}{r_o} - (m-1) \frac{r_o^2}{4_a{}^2}]$$

where:

W = applied load
m = reciprocal of Poisson's ratio
a = radius of flange
r_o= radius of applied load

Topic	Method of Analysis	Target Rock 7567F Analysis	Allowable Value	Calculated
10. Main disc stress	Using Roark's formulas for stress and strain, 4th edition, page 250, $S_{max} = \frac{BWa^2}{+_o 2}$	$S_{max} < S_m$ Material: SA182 S_m= 13,600 psi (@ 500°F)	S_m = 13,600 psi	Smax = 0.68

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FOR CURRENT INFORMATION, SEE THE LATEST APPLICABLE STRESS CALCULATIONS.

TABLE 3.9-4i (Cont)

Topic	Method of Analysis	Target Rock 7567F Analysis	Allowable Value	Calculated
	where: β = 1.63 W = applied load a = radius of disc t_0 = thickness at center			
11. Seismic Capability:	Stresss analysis uses $F_{vertical}$ = (mass of valve) x (2.0g) and $F_{horizontal}$ = (mass of valve) x (3.0g), with concurrent 400,000 in.-lb and 300,000 in.-lb applied at the inlet and outlet, respectively. Valve operability has been verified by test, with applied moments of 800,000 in.-lb and 600,000 in.-lb at the inlet and outlet, respectively, and at actual acceleration levels of vertical = 6g and horizontal = 8g. Tests are per IEEE 344 (1975).			
12. Dynamic Loads Capability:	A table is included in Table 3.9-5f that provides a comparison between the calculated valve valve loadings (acceleration levels and moments), as based on the piping stress analysis, and the allowable values.			

THIS TABLE CONTAINS HISTORICAL DATA ONLY AND IS NO LONGER UPDATED.

FOR CURRENT INFORMATION, SEE THE LATEST APPLICABLE STRESS CALCULATIONS.

(Historical Information)

TABLE 3.9-4j

MAIN STEAM ISOLATION VALVE

Criteria	Method of Analysis[(1)]	Allowable Stress or Minimum Thickness, in.	Calculated Stress or Actual Thickness, in.
Design of pressure retaining parts			
Body minimum wall thickness	Reference paragraph NB 3543, nonstandard pressure-rated valve, Tables NB 451.3, 451.4, and 452.1 For design condition of 1250 psig and 575°F. The primary service rating = 655 based on a core diameter of 23 in. t_m = 1.925 in. (including a corrosion allowance of 0.12 in.)	1.925	1.937
Body shape rule	Reference articles 452.2 and 452.2a(1), body shape rules		
Radius of crotch	Reference Article 452.3, radius of crotch criterion $r_2 \geq 0.3\ t_m$ as $r_2 = 1$ in., $t_m = 1.8125 + 0.12 \geq 0.3 \times 1.935 = 0.58$ criterion satisfied		
Longitudinal curvature	Reference article 452.2f, longitudinal curvature criterion $\frac{1}{r_{long}} + \frac{1}{r_{lat}} \geq \frac{4}{3d_m}$ is met		

THIS TABLE CONTAINS HISTORICAL DATA ONLY AND IS NO LONGER UPDATED.
FOR CURRENT INFORMATION, SEE THE LATEST APPLICABLE STRESS CALCULATIONS.

TABLE 3.9-4j (Cont)

Criteria	Method of Analysis[1]	Allowable Stress or Minimum Thickness, in.	Calculated Stress or Actual Thickness, in.
Flat wall limitation	Reference article 452.2g, flat wall limitation. Since no flat sections were built into the valve body design, the requirements of this article are satisfied.		
Primary crotch			
Stress due to internal pressure	Reference article 452.3 criterion $P_m = \left[\frac{A_f}{A_m} + 0.5\right] P_s < S_m$, where A_f 504 in^2, A_m = 58 in^2, P_s = 1375 psig, P_m = 12,650 psi, S_m = 19,400 psi, since $S_m > P_m$, criterion satisfied	19,400	12,650
Valve body secondary stress	Reference article 452.4		
Primary plus secondary stress due to internal pressure	Reference article 452.4a $Q_P = C_P\left[\frac{r_i}{t\theta} + 0.5\right] P_S C_a$, where C_p = 3, r_i = 11.625 in, P_e = 1375 psi, t_e = 2.75 for wye-type valve C_a = 1.33 + Q_p = 25,965 psi		

THIS TABLE CONTAINS HISTORICAL DATA ONLY AND IS NO LONGER UPDATED.
FOR CURRENT INFORMATION, SEE THE LATEST APPLICABLE STRESS CALCULATIONS.

(Historical Information)

TABLE 3.9-4j (Cont)

Criteria	Method of Analysis[1]	Allowable Stress or Minimum Thickness, in.	Calculated Stress or Actual Thickness, in.
Secondary stress due to pipe reaction	Reference article 452.4b, Figure 452.4b(3)		
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	19,400	5040
	P_{ed} = 5040 psi		
Bending load effect	$P_{eb} = C_b \frac{F_b S}{G_b}$, $C_b F_b S$, where S = 30,750 F_b = 340 in^3,		
	i.d. = 23.25 in, r_i = 11.625 t_e = 2.75 o.d = 27.8125		
	as $\frac{t_e}{r}$ = 0.197 > 0.19 + C_b = 1	19,400	9940
	$G_b = \frac{I}{r_i + t_e}$, where I = 15,121 in^4, r_i = 11.625 in,		
	t_e = 2.75 in + G_b 10.52 in^3		
	$P_{eb} = \frac{1(340)(30,750)}{1052}$ = 9940 psi		

THIS TABLE CONTAINS HISTORICAL DATA ONLY AND IS NO LONGER UPDATED.
FOR CURRENT INFORMATION, SEE THE LATEST APPLICABLE STRESS CALCULATIONS.

TABLE 3.9-4j (Cont)

Criteria	Method of Analysis[1]	Allowable Stress or Minimum Thickness, in.	Calculated Stress or Actual Thickness, in.
Torsion load effect	Reference article 452.4b		
	$P_{et} = \dfrac{2F_b S}{G_t}$, where $F_b = 340\ in^3$, $S = 30,750$ psi		
	$G_t = 2162\ in^3$	19,400	9670
	$P_{et} = 9670$ psi		
Thermal secondary stress at crotch region	Reference article 452.4c and Figures 452.4c(4) and 452.4c(5)		
	$Q_T = Q_{T1} + Q_{T_2}$,		
	where $T_{e1} = 3$ in, $Q_{T_1} = 1100$,		
	$Q_{T2} = C_6 C_2 \Delta T_2$, where $C_2 = 0.21$, $C_6 = 220$, and $\Delta T_2 = 5.6$		
	$Q_{T2} = 260$ psi, $Q_T = 1360$ psi		
	criterion $S_N = Q_p + P_e \leq 3\ S_m$,		
	where $Q_p = 25,965$, $P_e = 9940$, $Q_{T2} = 1360$,	58,200	38,625

THIS TABLE CONTAINS HISTORICAL DATA ONLY AND IS NO LONGER UPDATED.
FOR CURRENT INFORMATION, SEE THE LATEST APPLICABLE STRESS CALCULATIONS.

(Historical Information)

TABLE 3.9-4j (Cont)

Criteria	Method of Analysis[1]	Allowable Stress or Minimum Thickness, in.	Calculated Stress or Actual Thickness, in.
	as 38,625 ≤ 58,200, criterion satisfied		
Normal duty valve fatigue requirements	Reference article 452.5 criterion $N_a \geq 2000$ cycles		
	$$S_{p1} = \frac{2}{3} Q'_p + \frac{P_{eb}}{2} + Q_{T3} + 1.3\, Q_{T1}'$$		
	$$S_{p2} = 0.4\, Q'_p + \frac{K}{2}\left(P_{eb} + 2\, Q_{T2}\right)$$		
	where Q'_p = 25,965, P_{eb} = 9942 K-2, Q_{T2} = 260, Q_{T3} = 1100 psi		
	S_{p1} = 23,970, S_{p2} = 20,845 S_a equal to the larger of S_{p1} and S_{p2}		
	S_a = 23,970		
	N_a = 55,000 ≥ 2000, criterion satisfied		
Disk design calculation	Reference I-1120, Section III of ASME B&PV Code; Roark, 4th Ed., Pages 198, 200, 201		

THIS TABLE CONTAINS HISTORICAL DATA ONLY AND IS NO LONGER UPDATED.
FOR CURRENT INFORMATION, SEE THE LATEST APPLICABLE STRESS CALCULATIONS.

TABLE 3.9-4j (Cont)

Criteria	Method of Analysis[1]	Allowable Stress or Minimum Thickness, in.	Calculated Stress or Actual Thickness, in.
	Disk design conditions, P_s = 1756 psi at 500°F,		
	Case No. 13, $S_t = \frac{3W}{4mt^2(a^2 - b^2)}\left[a^4(3m + 1) + b^4(m - 1) - 4_m a^2 b^2 - 4(m+1)a^2 b^2(\ln(a/b))\right]$,		
	where W = 1250 psi, $m = \frac{10}{3}$, t = 5.875 in, a = 10.75 in,		
	b = 1.75 in, S_t = 9489 psi		
	Case No. 14, $S_t = \frac{3W}{2\pi m t^2}\left[\frac{2a^2(m+1)}{a^2 - b^2}\ln(a / b) + (m - 1)\right]$		
	where W = 59,044 lb_f, t = 5.875 in, $m = \frac{10}{3}$,		
	a = 10.75 in, b = 1.75 in., S_t = 4531 psi		
		17,800	14,020($S_{t1}+S_{t2}$)

THIS TABLE CONTAINS HISTORICAL DATA ONLY AND IS NO LONGER UPDATED.
FOR CURRENT INFORMATION, SEE THE LATEST APPLICABLE STRESS CALCULATIONS.

TABLE 3.9-4j (Cont)

Criteria	Method of Analysis[(1)]	Allowable Stress or Minimum Thickness, in.	Calculated Stress or Actual Thickness, in.
	Case No. 21, $S_r = \frac{3W}{4t^2}\left[\frac{4a^4(m+1)\ln(a/b)a^4(m+3)+b^4(m-1)+4a^2b^2}{a^2(m+1)+b^2(m-1)}\right]$		
	where W = 1,375, m = 10/3, t = 3.188 in, a = 10.75 in, b = 7.25 in.		
	S_r = 6090 psi		
	Case No. 22, $S = \frac{3W}{2\pi t^2}\left[\frac{2a^2(m+1)\ln(a/b)+a^2(m-1)-b_2(m-1)}{a^2(m+1)+b^2(m-1)}\right]$		
	where W = 252.755, m = 10/3, t = 3.125, a = 10.75, b = 7.25		
	S_p = 10,740 psi		

Note:

(1) All references are made to ASME B&PV Code for Pumps and Valves for Nuclear Power, dated November 1968. Reference the same code for explanation of the symbols used.

THIS TABLE CONTAINS HISTORICAL DATA ONLY AND IS NO LONGER UPDATED.
FOR CURRENT INFORMATION, SEE THE LATEST APPLICABLE STRESS CALCULATIONS.

TABLE 3.9-4k

RECIRCULATION PUMP

Criteria	Normal and Upset Condition	Analytical Results	Allowable Stress or Actual Thickness
1. Casing minimum wall thickness loads: $t = \frac{PR}{SE - 0.6P} + C$ 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. Primary membrane stress limit: Allowable working stress per ASME Section III, Class C	Design pressure & temperature	t = 2.69 in.	S_{allow} = 15,075 psi t_{act} = 3.00 in.
2. Casing cover minimum thickness loads: $S_s = \frac{F}{A}$ F = force A = area at shear point	Design pressure & temperature	S_s = 3380 psi	S_{allow} = 8750 psi t_{act} = 3.5 in.
$S_b = \frac{Kga^2}{h^2}$ q = pressure load a = radius of o.d. b = radius of i.d. h = plate thickness Primary bending & shear stress limit: 1.5 S_m per ASME Code for pumps and valves for nuclear power Class I		S_b = 5950 psi	S_{allow} = 1.5 x 15,075 psi t_{act} = 7 in.

Criteria	Normal and Upset Condition	Analytical Results	Allowable Stress or Actual Thickness
3. Cover and seal flange bolt areas loads:	Design pressure & temperature	For cover flange bolts:	
Bolting loads, areas, and stresses shall be calculated in accordance with "Rules for Bolted Flange Connections" - ASME Section VIII, Paragraph UA-49.	Design gasket load	S_{act} = 19,400 psi	S_{allow} = 20,000 psi
		A_m = 90.2 sq in.	A_{act} = 101 sq in.
		For seal flange bolts:	
Bolting stress limit: Allowable working stress per ASME Section III, Class C		S_{act} = 18,000 psi	S_{allow} = 20,000 psi
		A_m = 9.85 sq in.	A_{act} = 11.1 sq in.
4. Cover clamp flange thickness loads:	Design pressure & temperature	For flange thickness stress:	
Flange thickness and stress shall be calculated in accordance with "Rules for Bolted Flange Connections" - ASME Section VIII, Para UA-51	Design gasket load Design bolting load	t = 8.9 in.	t_{act} = 9.25 in.
Tangential flange stress limit: Allowable working stress per ASME Section III, Class C.			S_{allow} = 17,500 psi
5. Seal cover loads:	Design pressure & temperature	S_b = 2870 psi	S_{allow} = 15,075 psi
$S_b = \frac{KP}{+2}$		t = 1.10 in.	t_{act} = 2.56 in.
P = bolt load due to pressure t = thickness, in. K = constant or shape factor			
6. Seal chamber minimum wall thickness loads:	Design pressure & temperature	t = 0.741 in.	S_{allow} = 1.5 x 17,075 psi
$t = \frac{PR}{SE - 0.6P} + C$	Piping reactions during normal operation		t_{act} = 1.375 in.

TABLE 3.9-4k (Cont)

Criteria	Normal and Upset Condition	Analytical Results	Allowable Stress or Actual Thickness
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.			
Combined stress limit: 1.5 S_m per ASME Code for pumps and valves for nuclear power Class I			
7. Mounting bracket combined stress loads:	Flooded weight DBE horizontal seismic force = 1.07 g DBE vertical seismic force = 0.67 g		1.5 S_m = 25,013 psi
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).		Lug No. 1, S_c = 20,857 psi Lug No. 2, S_c = 11,393 psi	
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 No. 3, S_c = 7,380 psi	
Combined stress limit: 1.5 S_m per ASME Code for pumps and valves for nuclear power Class I			

Criteria	Normal and Upset Condition	Analytical Results	Allowable Stress or Actual Thickness
8. Stresses due to seismic loads: The flooded pump-motor assembly shall be analyzed as a free body supported by constant support hangers from the pump brackets. Horizontal and vertical seismic forces shall be applied at mass center of assembly and equilibrium reactions shall be determined for the motor and pump brackets. Loads, shear, and moment diagrams shall be constructed brackets. Loads, shear, and moment 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 determine from loading diagram plus tensile stress from operating pressure Combined stress limit: Yield stress	Operation pressure and temperature DBE horizontal seismic force = 2.05g DBE vertical seismic force	For motor bolt tensile stress: S_{act} = 21,300 psi For pump cover bolt tensile stress: = 1.61g S_{act} = 20,000 psi For motor support barrel combined stress: S_{act} = 2830 psi	 S_{allow} = 30,800 psi S_{allow} = 32,000 psi S_{allow} = 22,400 psi

TABLE 3.9-41

REACTOR RECIRCULATION SYSTEM GATE VALVES
STRUCTURAL AND MECHANICAL LOADING CRITERIA

Discharge Valves

	Component/ Loads/ Design	Design Procedure	Required Design Value	Actual Design Value
1.	Body and Bonnet			
1.1	Loads: design pressure, design temperature	Vendor's design calculations	1525 psi 575°F	1525 psi 575°F
1.2	Pressure rating, psi	Used Draft ASME Code, Section 452.1(3)	P_r = 800 psi	P_r = 800 psi
1.3	Minimum wall thickness, in.	Used Draft ASME Code, Section 452.1(3)	$t_m \geq 2.1164$ in.	t_m = 2.5 min.
1.4	Primary membrane stress, psi	Used Draft ASME Code, Section 452.3(3)	$P_m \leq S_m$(500°F) = 19,600 psi	P_m = 11,068 psi
1.5	Secondary stress due to pipe reaction	Used Draft ASME Code, Section 452.4(3)	P_e = greatest value of P_{ed} P_{eb} and $P_{et} \leq 1.5\ S_m$ (500°F) 1.5 (16,800) = 25,200 psi	P_{ed} = 5,580 psi P_{eb} = 12,702 psi P_{et} = 12,277 psi $P_e = P_{eb}$ = 12,702 psi
1.6	Primary plus secondary stress due to internal pressure	Used Draft ASME Code, Section 452.4(3)	See 1.8 below	Q_p = 24,284 psi
1.7	Thermal secondary stress	Used Draft ASME Code, Section 452.4(3)	See 1.8 below	Q_T = 5409 psi
1.8	Sum of primary plus secondary stress	Used Draft ASME Code, Section 452.4(3)	$S_n \leq 3S$ (500°F) = 58,800 psi	$S_n = Q_p + P_e + 2Q_T$ S_n = 47,804 psi

Discharge Valves

	Component/ Loads/ Design	Design Procedure	Required Design Value	Actual Design Value
1.9	Fatigue requirements	Used Draft ASME Code, Section 452.5[(3)]	$N_a \geq 2000$ cycles	$N_a > 10^5$ cycles
1.10	Cyclic rating	Used Draft ASME Code, Section 454.2[(3)]	$I_t \leq 1$	$I_t = 0.111$
2.	Body to Bonnet Bolting			
2.1	Loads: design pressure & temperature, gasket loads, stem operational load, seismic load (design basis earthquake)	Used ASME B&PV Code, Section VIII[(1)] Paragraph UA-47 thru UA-51		
2.2	Bolt area	Used ASME B&PV Code, Section VIII[(1)] Paragraph UA-47 thru UA-51	$A_b \geq 44.41$ in.^{2} $S_b \leq 27,975$ psi (575°F)	$A_b = 55.86$ in.^{2} $S_b = 22,834$ psi
2.3	Body flange stresses	Used ASME B&PV Code, Section VIII[(1)], Paragraph UA-47 thru UA-51		
2.3.1	Operating condition	Used ASME B&PV Code, Section VIII[(1)], Paragraph UA-47 thru UA-51	$S_H \leq 1.5$ S (575°F) = 28,837 psi $S_R \leq 1.5$ S (575°F) = 14,225 psi $S_T \leq 1.5$ S (575°F) = 19,285 psi	$S_H = 27,260$ psi $S_R = 8004$ psi $S_T = 9031$ psi

Discharge Valves

	Component/ Loads/ Design	Design Procedure	Required Design Value	Actual Design Value
2.3.2	Gasket seating condition	Used ASME B&PV Code, Section VIII(1), Paragraph UA-47 thru UA-51	$S_H \leq 1.5\ S\ (100°F) = 30{,}000$ psi $S_R \leq 1.5\ S\ (100°F) = 30{,}000$ psi $S_T \leq 1.5\ S\ (100°F) = 30{,}000$ psi	$S_H = 29{,}981$ psi $S_R = 11{,}671$ psi $S_T = 12{,}972$ psi
3.	<u>Stresses in Stem</u>			
3.1	Loads: operator thrust and torque			
3.2	Stem thrust stress	Calculate stress due to operator thrust in critical cross section	$S_T \leq S_m = 43{,}675$ psi	$S_T = 6344$ psi
3.3	Stem torque stress	Calculate shear stress due to operator torque in critical cross section	$S_s \leq 0.6\ S_T = 26{,}205$ psi	$S_S = 7732$ psi
3.4	Buckling on stem	Calculate slenderness ratio. If greater than 30, calculate allowable load from Rankine's formula using safety factor of 9.	max. allowable = 38,800 lb	Slenderness ratio = 60 Actual load on stem = 31,842 lb Therefore, no buckling
4.	<u>Disc Analysis</u>			
4.1	Loads: maximum differential pressure(2)			
4.2	Maximum stress in the disc	Calculate maximum stress according to Table 10 of Roark's "Formula for Stress and Strain"	$S_{max} \leq 1.5\ S_m (500°F) = 28{,}500$ psi	Max stress = 22,885 psi

Discharge Valves

	Component/ Loads/ Design	Design Procedure	Required Design Value	Actual Design Value
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	Stress in yoke legs bolts		$S_{max} \leq S_m$ = 28,800 psi (500°F)	Max. stress = 5,134 psi
5.3	Stress of yoke legs		$S_{max} \leq 1.5\ S_m$ = 19,480 psi (500°F)	S_{max} = 7,213 psi
5.4	Stress of yoke - ?????? connection		$S_{max} \leq S_m$ = 19,225 psi (575°F)	S_{max} = 8,190 psi

Suction Valves

Component/ Loads/ Design		Design Procedure	Required Design Value	Actual Design Value
1.	Body and Bonnet			
1.1	Loads: Design pressure, design temperature	Vendor's design calculation	1275 psi 575°F	1275 psi 575°F
1.2	Pressure rating, psi	Used Draft ASME Code, Section 452.1[(5)], Figure NB-3545.1-2	P_r = 668 psi	P_r = 668 psi
1.3	Minimum wall thickness, in.	Used Draft ASME Code, Section 452.1[(5)], Paragraph NB-1542	$t_m \geq$ 1.7724 in.	t_m = 2.5 minimum
1.4	Primary membrane stress, psi	Used Draft ASME Code, Section 452.3[(5)], Paragraph NB-3545.1	$P_m \leq S_m$ (500°F) = 19,600 psi	P_m = 9,275 psi
1.5	Secondary stress due to pipe reaction	Used Draft ASME Code, Section 452.4[(5)]	Pe = greatest value of Ped Peb and Pet $\leq 1.5\ S_m$ 1.5 (16,800) = 25,200 psi	P_{ed} = 5318 psi P_{eb} = 11,980 psi P_{et} = 11,575 psi P_e = Peb = 11,980 psi
1.6	Primary plus secondary stress due to internal pressure	Used Draft ASME Code, Section 452.4[(5)]	See 1.8 below	Q_p = 20,580 psi
1.7	Thermal secondary stress	Used Draft ASME Code, Section 452.4[(5)]	See 1.8 below	Q_t = 5484 psi
1.8	Sum of primary plus secondary stress	Used Draft ASME Code, Section 452.4[(5)]	Sn $\leq 3\ S_m$ (500°F) = 58,800 psi	S_n = Op + Pe + 2Qt S_r = 43,538 psi
1.9	Fatigue requirements	Used Draft ASME Code, Section 452.5[(3)]	Na $\geq$ 2000 cycles	$N_a = 10^6$ cycles
1.10	Cyclic rating	Used Draft ASME Code, Section 454.2[(5)]	$I_t \leq 1$	I_t = 0.111

Suction Valves

Component/ Loads/ Design		Design Procedure	Required Design Value	Actual Design Value
2.	Body to Bonnet Bolting			
2.1	pressure & temperature, Gasket loads, Stem operational load (design basis earthquake)	Loads: design Paragraph UA-47 thru UA-51	Used ASME Section VIII[1]	
2.2	Bolt area	Used ASME Section VIII[1] Paragraph UA-47 thru UA-51	$A_b \geq 37.53$ in.^{2}	$A_b = 55.86$ in.^{2}
			$S_b \leq 27,975$ psi (575°F)	$S_b = 19,470$ psi
2.3	Body flange stresses	Used ASME Section VIII[1] Paragraph UA-47 thru UA-51		
2.3.1	Operating condition	Used ASME Section VIII[1] Paragraph UA-47 thru UA-51	$S_H \leq 1.5\ S_m$(575°F) = 28,837 psi	$S_H = 24,456$ psi
			$S_R \leq 1.5\ S_m$(575°F) = 19,225 psi	$S_R = 6539$ psi
			$S_T \leq 1.5\ S_m$(575°F) = 19,225 psi	$S_T = 8718$ psi
2.3.2	Gasket seating condition	Used ASME Section VIII[1] Paragraph UA-47 thru UA-51	$S_H \leq 1.5\ S_m$(100°F) = 30,000 psi	$S_H = 28,945$ psi
			$S_R \leq 1.5\ S_m$(100°F) = 30,000 psi	$S_R = 10,253$ psi
			$S_T \leq 1.5\ S_m$(100°F) = 30,000 psi	$S_T = 13,619$ psi
3.	Stress in stem			
3.1	Loads: 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 = 37,750 lb	Slenderness ratio = 61 Actual load on stem = 31,842 lb Therefore, no buckling.

Suction Valves

Component/ Loads/ Design		Design Procedure	Required Design Value	Actual Design Value
3.3	Stem thrust stress	Calculate stress due to operator thrust in critical cross section	$S_T \le S_T$ = 43,675 psi	S_T = 6461 psi
3.4	Stem torque stress	Calculate shear stress due to operator torque in critical cross section	$S_s \le 0.6\ S_m$ = 26,205 psi	S_S = 7947 psi
4.	Disc Analysis			
4.1	Loads: maximum differential pressure(4)			
4.2	Maximum stress in the disc	Calculate maximum stress according to table 10 of Roark's Formula for Stress and Strain	$S_{max} \le 1.5\ S_m$ (500°F) = 28,500 psi	Max stress = 19,418 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	Stress in yoke legs bolts		$S_{max} \le S_m$ = 28,800 psi (500°F)	Max stress = 5053 psi
5.3	Stress at yoke legs		$S_{max} \le S_m$ = 19,400 psi (500°F)	S_{max} = 7092 psi
5.4	Stress at yoke-bonnet conn.		$S_{max} \le S_m$ = 19,225 psi (575°F)	S_{max} = 8052 psi

(1) ASME B&PV Code Section VIII, 1968 Edition only.
(2) Valve differential pressure = 200 psid.
(3) Draft, ASME Code for Pumps and Valves For Nuclear Power.
(4) Valve differential pressure = 50 psid.

TABLE 3.9-4m

ASME B&PV CODE CLASS 3 SAFETY/RELIEF VALVE DISCHARGE PIPING

See Tables 3.9-8 and 3.9-13

TABLE 3.9-4n

STANDBY LIQUID CONTROL PUMP

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 & 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 193-B7	S_y = 105,000 psi
6.	Dowel pins(2) alignment, SAE 4140	S_a = 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

Criteria/Loading [1]	Component	Limiting Stress Type	Allowable Stress, psi	Calculated Stress, psi
Normal and upset condition loads:				
1. Design pressure	1. Fluid cylinder	General membrane	17,800	See note[4]
2. Design temperature	2. Discharge valve stop	General membrane	17,800	See note[4]
3. Operating basis earthquake	3. Cylinder head extension	General membrane	17,800	See note[4]
4. Nozzle loads[3]	4. Discharge valve cover	General membrane	17,800	See note[4]
5. Dead weight	5. Cylinder head	General membrane	17,800	See note[4]
6. Thermal expansion	6. Stuffing box flange plate	General membrane	17,800	See note[4]
7. SRV discharge	7. Stuffing box gland	General membrane	35,000	See note[4]
	8. Cylinder head studs	Tensile	25,000	See note[4]
	9. Stuffing box studs	Tensile	25,000	See note[4]
Emergency condition loads:				
1. Design pressure	1. Fluid cylinder	General membrane	21,360	4450
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	8150
5. Nozzle loads[3]	5. Cylinder head	General membrane	21,360	8150
	6. Stuffing box flange plate	General membrane	21,360	10,390
	7. Stuffing box gland	General membrane	42,000	11,420
	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	8685
	11. Pump holddown bolts	Shear	12,000	9415
	12. Pump holddown bolts	Tensile	15,000	12,675
	13. Power frame-foot area	Shear	15,000	1850
	14. Power frame-foot area	Tensile	15,000	11,390
	15. Motor holddown bolts	Shear	12,000	3020
	16. Motor holddown bolts	Tensile	15,000	5290
	17. Motor frame-foot area	Shear	15,000	2070
	18. Motor frame-foot area	Tensile	15,000	4125

Nozzle load: Allowable nozzle loads are given in the form of the following equation and must be satisfied to prevent excessive shear stress in the pump holddown bolts.

Units: Forces - lb
Moments - in.-lb

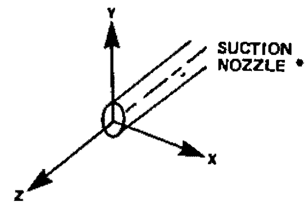
$$0.78F_{sx} + 1.32F_{sz} + 1.32F_{dz} + 0.146F_{dx} + 0.0412M_{sy} + 0.0412M_{dx} \leq 2315$$

where:

F_{sx} = Force on suction nozzle flange in x direction.

F_{sz} = Force on suction nozzle flange in z direction.

Criteria/Loading [1]	Component	Limiting Stress Type	Allowable Stress, psi	Calculated Stress, psi
	F_{dz} = Force on discharge nozzle flange in z direction.			
	F_{dx} = Force on discharge flange in x direction.			
	M_{sy} = Moment on suction flange about y axis.			
	M_{dy} = Moment on discharge flange about y axis.			



* Same axis orientation applies to discharge nozzle.

(1) Based on ASME B&PV Code, Section III.
(2) Dowel pins take all shear.
(3) Nozzle loads produce shear loads only.
(4) Calculated stresses for emergency or faulted condition are less than the allowable stresses for the normal and upset condition stresses; therefore, the normal and upset condition is not evaluated.
(5) 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 3.9-4o

STANDBY LIQUID CONTROL TANK

Criteria	Method of Analysis	Allowable Stress of Min. Thickness Req'd or Load		Actual Stress or Thickness Req'd or Load	
1. Shell thickness					
Loads: normal & upset design pressure and temperature	Brownell & 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		1602 psi	
2. Nozzle loads					
Loads: normal & upset design pressure and temperature	The maximum moments due to pipe reaction and maximum forces shall not exceed the allowable limits.	Fo (lb)	Mo (ft-lb)	F(lb)	M(ft-lb)
Overflow nozzle		440	300	78	169
Discharge nozzle		440	300	Noz 1:151 Noz 2:171	142 136
Loads: faulted dead weight, thermal expansion, and SSE	The maximum moments due to pipe reaction and maximum forces shall not exceed the allowable limits	Fo (lb)	Mo (ft-lb)	F(lb)	M(ft-lb)
Overflow nozzle		530	360	78	205
Discharge nozzle		530	360	Noz 1:234 Noz 2:253	200 180
3. Anchor bolts	ASME Section III	18,750 psi		9617 psi	

TABLE 3.9-4p

ECCS PUMPS

Location	Loading Condition	Criteria	Actual Thickness, in.	Minimum Thickness, in.
Residual heat removal pump				
Discharge head shell	Faulted condition Design pressure Nozzle loads SSE loads	ASME B&PV Code, Section VIII, Division 1, Paragraph UG-32	0.750	0.526
Discharge nozzle	Faulted condition Design pressure Nozzle loads	ASME B&PV Code, Section VIII, Division 1, Paragraph UG-27	0.562	0.328
Shell	Faulted condition Design pressure Nozzle loads SSE loads	ASME B&PV Code, Section VIII, Division 1, Paragraph UG-27	0.750	0.376
Core spray pump				
Discharge head shell	Faulted condition Design pressure Nozzle loads SSE loads	ASME B&PV Code, Section VIII, Division 1, Paragraph UG-32	0.500	0.268
Discharge nozzle	Faulted condition Design pressure Nozzle loads	ASME B&PV Code, Section VIII, Division 1, Paragraph UG-27	0.365	0.228
Shell	Faulted condition Design pressure Nozzle loads SSE loads	ASME B&PV Code, Section VIII, Division 1, Paragraph UG-27	0.500	0.185

(1) Operability demonstrated by analysis.

TABLE 3.9-4q

RHR HEAT EXCHANGER

Loading/Component	Criteria/Location	Allowable Stress or Min. Thickness Req'd.	Calculated Stress or Thickness
1. Closure bolting Loads: normal	Bolting loads and stresses calculated per "Rules for Bolted Flange Connections" ASME Section VIII, App II		
Design pressure and temperature, design gasket load	a. Shell to tube sheet bolts b. Channel to tube sheet bolts c. Channel to corner bolts	25,000 psi 25,000 psi 25,000 psi	24,230 psi 24,230 psi 24,230 psi
2. Wall thickness Loads: normal	Shell side, ASME Section III, Class C, and TEMA, Class C		
Design pressure and temperature	Tube side, ASME Section VIII, Div. 1, and TEMA, Class C		
	a. Shell b. Shell cover c. Channel ring d. Tubes e. Channel cover f. Tube sheet	0.736 in. 0.728 in. 0.736 in. 0.047 in. 6.2069 in. 6.2859 in.	0.750 in. 1.0 in. min 1.0625 in. 0.049 in. 6.25 in. 6.3125 in.

Loading	Criteria	Allowable Nozzle Forces and Moments	Actual Nozzle Forces and Moments[1]
3. Nozzle Loads: faulted Design pressure and temperature, dead weight, thermal expansion, safe shutdown earthquake	The maximum moments due to pipe reaction and the maximum forces shall not exceed the allowable limits Primary stress smaller of 0.75 S_u or 2.4 S_m (ASME Section III allowable)	See below *(a) *(b)	*(c)

Loading/Component	Criteria/Location	Allowable Stress or Min. Thickness Req'd.	Calculated Stress or Thickness

*(a) Allowable limits (design bases)

	N1	N2	N3	N4
F_x =	9041 lb	9041 lb	40,738 lb	20,325 lb
F_y =	20,325 lb	20,325 lb	18,122 lb	20,325 lb
F_z =	20,325 lb	20,325 lb	40,738 lb	9041 lb
M_x =	627,621 in.-lb	627,621 in.-lb	246,774 in.-lb	121,927 in.-lb
M_y =	121,927 in.-lb	121,927 in.-lb	1,230,600 in.-lb	121,927 in.-lb
M_z =	121,927 in.-lb	121,927 in.-lb	246,774 in.-lb	627,621 in.-lb

*(b) Forces and moments are given in global coordinate system defined on heat exchanger

*(c) Calculated Nozzle Forces and Moments (design bases)

	N1	N2	N3	N4
F_x =	4547 lb	3297 lb	6604 lb	2779 lb
F_y =	4233 lb	3671 lb	5912 lb	3524 lb
F_z =	1922 lb	4697 lb	4432 lb	2777 lb
M_x =	58,716 in.-lb	94,032 in.-lb	241,752 in.-lb	145,080 in.-lb
M_y =	116,376 in.-lb	120,120 in.-lb	306,967 in.-lb	105,636 in.-lb
M_z =	226,524 in.-lb	114,792 in.-lb	236,574 in.-lb	117,036 in.-lb

Component/Loading	Criteria/Location	Allowable Stress, psi	Actual Stress, psi
4. Support brackets & attachment welds	Stress allowables as per ASME B&PV Code, Section III, Subsection NT (upset condition).		
Loads: faulted			
	Lower bracket welds		
Design pressure and temperature, dead weight, nozzle loads, SSE	- Bending stress	14,437.5	7,153.9
	- Shear stress	14,437.5	13,998

Loading/Component	Criteria/Location	Allowable Stress or Min. Thickness Req'd.	Calculated Stress or Thickness
5. Anchor bolts	Stress allowable as per ASME B&PV Code, Section III, Appendix XVII		
Loads: faulted	Lower support bolting		
Design pressure and temperature, dead weight, nozzle loads, SSE	- Tension stress	52,500	42,494
	- Shear stress	21,700	10,358
6. Shell adjacent to support brackets	Shell stress allowables as per ASME Section III, Subsection NC (upset condition)		
Loads: faulted			
Design pressure and temperature, dead weight, nozzle loads, SSE	1. Maximum principal stress adjacent to upper support	28,875	14,667
	2. Maximum principal stress adjacent to lower support	28,875	19,550
7. Shell away from discontinuities	Stress allowable as per ASME Section III, Subsection NC (upset condition)		
Loads: faulted	Principal stress	19,250	19,620
Design pressure and temperature, dead weight, nozzle loads, SSE			

Note:
(1) Calculated nozzle forces and moments are enveloped values of RHR Ht. Exchanger AE205 & BE205. Calculated loads were evaluated and accepted by General Electric per letter GB-86-27 dated 1/31/86. outline drawings GE VPF #3239-97-4.

Table 3.9-4r

Reactor Water Cleanup Pump

Component Design Margin	Safety Factor (SF)[1] 1AP-221	Safety Factor (SF)[1] 1BP-221
Motor Case Flange to Pump Case Bolting	1.105	1.103
Suction Nozzle Max Loading		
Faulted	1.762	1.762
Upset	1.138	1.138
Normal	1.099	1.099
Seismic	4.173	4.210
Discharge Nozzle Max Loading		
Faulted	3.379	3.540
Upset	2.204	2.331
Normal	2.242	2.994
Seismic	15.447	15.300
Support Skirt Bolt Seismic Loading		
Axial	7.954	7.683
Shear	32.755	89.314

[1] Designed to ASME Section III, Class 3, 2004 Edition

TABLE 3.9-4s

RCIC TURBINE

Criteria Loading	Component	Limiting Stress Type	Allowable Stress, psi	Calculated Stress, psi
The highest stressed sections of the various components of the RCIC turbine assembly are identified. Allowable stresses are based on ASME B&PV Code, Section III, for:				
Normal and upset condition:				
Pressure boundary castings SA216-NCB:				
S_a (general membrane) = 0.8S, S = 17,500 psi				
S_a (bending) = 1.5 x 0.8 x S, S = 17,500 psi				
Pressure boundary boltings, SA193-B7 @ 500°F				
S_a = 1.0S S = 25,000 psi				
Alignment dowel pins: AISI4037, PC28-35				
T_a = 61,100 psi				
S_a = 106,000 psi				
Normal and upset condition loads:	Castings: 1) Stop valve	General membrane	14,000	
	2) Governor valve	General membrane	14,000	
1. Design pressure	3) Turbine inlet	Local bending	21,000	(1)
2. Design temperature	4) Turbine case	Local bending	21,000	
3. Operating basis earthquake	Pressure containing bolts	Tensile	25,000	
4. Inlet nozzle loads	Structure alignment pins	Shear	61,100	
5. Exhaust nozzle loads				
Faulted condition loads:	Castings: 1) Stop valve	General membrane	16,800	9800
	2) Governor valve	General membrane	16,800	13,200
1. Design pressure	3) Turbine inlet	Local bending	25,200	15,300
2. Design temperature	4) Turbine case	Local bending	25,200	18,000
3. Safe shutdown earthquake	Pressure containing bolts	Tensile	25,000	20,100
4. Inlet nozzle loads	Structure alignments pins	shear	61,100	46,880
5. Exhaust nozzle loads				
6. Safety/relief valve discharge				
7. LOCA				

Criteria Loading	Component	Limiting Stress Type	Allowable Load Criteria	Calculated Loads
Nozzle load definition:				
Turbine vendor has defined allowable nozzle loads for the turbine assembly. The above calculated stresses assume these allowable nozzle loads have been satisfied			Inlet: $F = \frac{(2620-M)}{3}$	F = 390 M = 1419
Normal condition loads:				
1. Design pressure 2. Design temperature 3. Weight of structure 4. Thermal expansion			Exhaust: $F = \frac{(6000-M)}{3}$ where: F = resultant force (lb) M = resultant moment (ft-lb)	F = 219 M = 2398
Upset, emergency, or faulted condition loads:				
1. Design pressure 2. Design temperature 3. Weight of structure 4. Thermal expansion 5. Safe shutdown earthquake/operating basis earthquake			Inlet: $F = \frac{(7000-M)}{4.7}$ Exhaust: $F = \frac{(8500-M)}{0.34}$ but less than 7000 F = resultant force (lb) M = resultant moment (ft-lb)	Inlet: Upset: F=1050; M=1743 Emerg: F=1066; M=1806 Faulted: F=585; M=1651 Exhaust: Upset: F=1860; M=3487 Emerg: F=1718; M=4006 Faulted: F=3076; M=4999

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(1) Calculated stresses for the faulted condition are lower than the allowable stresses for the normal plus upset condition; therefore, the normal, upset, and emergency conditions are not evaluated.

(2) Operability: Analysis indicates that shaft deflection with faulted loads is 0.006 inch (this is fully acceptable) and maximum bearing load with faulted condition is 80 percent of allowable. Furthermore, as indicated in Paragraph 3.9.2.3.2.9, the turbine assembly has been seismically qualified via dynamic testing. This qualification included demonstration of startup and shutdown capabilities, as well as no load operability during seismic loading conditions.

TABLE 3.9-4t

RCIC PUMP

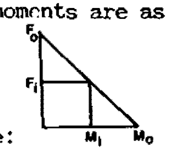
Criteria/Loading	Component	Limiting Stress Type	Allowable Stress, psi	Calculated Stress, psi
Pressure boundary stress limits of the various components for the RCIC pump assembly are based on the ASME B&PV Code, Section III, for pressure boundary part @ 140°F				
1. Forged barrel, SA105 GR. II				
S_y = 36,000 psi				
2. End cover plates, SA105 GR. II				
S_y = 36,000 psi				
3. Nozzle connections, SA105 GR. II				
S_y = 36,000 psi				
4. Aligning pin, SA105 GR. II				
S_y = 36,000 psi				
5. Closure bolting, SA193-B7				
S_y = 125,000 psi				
6. Pump holddown bolting, SA 325				
S_y = 81,000 psi				
7. Taper pins, SA108 GR B1112,				
S_y = 70,000 psi				
<u>Normal and upset condition loads:</u>				
1. Design pressure	1. Forged barrel	General membrane	See note(1)	See note(1)
2. Design temperature	2. End cover (suction)	General membrane	See note(1)	See note(1)
3. Operating basis earthquake	3. End cover (discharge)	General membrane	See note(1)	See note(1)
4. Suction nozzle loads	4. Nozzle reinforcement	Tensile, shear	See note(1)	See note(1)
5. Discharge nozzle loads	5. Alignment pin	Tensile	See note(1)	See note(1)
	6. Closure bolting		See note(1)	See note(1)

Criteria/Loading	Component	Limiting Stress Type	Allowable Stress, psi	Calculated Stress, psi
	7. Taper pins		See note(1)	See note(1)
	8. Pump holddown bolts		See note(1)	See note(1)
Emergency or faulted condition loads:				
1. Design pressure	1. Forged barrel	General membrane	21,000	8100
2. Design temperature	2. End cover (suction)	General membrane	21,000	10,660
3. Safe shutdown earthquake	3. End cover (discharge)	General membrane	21,000	17,115
4. Suction nozzle loads	4. Nozzle reinforcement at barrel	General membrane	21,000	5340
5. Discharge nozzle loads	5. Alignment pin	Shear	25,200	15,000
	6. Closure bolting	Tensile	30,000	20,740
	7. Taper pins (bearing housing)	Shear	25,200	1220
	8. Pump holddown bolts	Tension	38,880	12,450

Nozzle load definition:

Units: Forces = lb
Moments = ft-lb

The allowable combinations of forces and moments are as follows:



$$\frac{Fi}{Fo} + \frac{Mi}{Mo} \leq 1$$

where:

Fi = Largest absolute value of the three actual external orthogonal forces (Fx, Fy, Fz) that may be imposed by the interface pipe

Mi = Largest absolute value of the three actual external orthogonal moments (Mx, My, Mz) permitted from the interface pipe when they are combined simultaneously for a specific condition

Criteria/Loading	Component	Limiting Stress Type	Allowable Loads	Calculated Load(3)
Normal and upset condition loads:		Fo = Allowable value of Fi when all moments are zero (lbs)	Suction:	
1. Design pressure 2. Design temperature 3. Weight of structure 4. Thermal expansion 5. Operating basis earthquake		Mo = Allowable value of Mi when all forces are zero (ft.-lbs)	Fo = 1940 Mo = 2460	F2 = 1745 M2 = 1436
		F2 = Maximum of the three orthogonal forces (lbs)	Discharge: Fo = 3715 Mo = 4330	F2 = 816 M2 = 880
Emergency or faulted condition loads:			Suction:	
1. Design pressure 2. Design temperature 3. Weight of structure		M2 = Maximum of the three orthogonal moments (ft.-lbs)	Fo = 2325 Mo = 2950	F2 = 2556 M2 = 1659
4. Thermal expansion 5. Safe shutdown earthquake			Discharge: Fo = 4450 Mo = 5200	F2 = 949 M2 = 896

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(1) Calculated stresses for emergency or faulted condition are less than the allowable for normal plus upset condition; therefore, the normal and upset condition is not evaluated.

(2) Operability: Static analyses for emergency or faulted condition show that the maximum shaft deflection is 0.002 in. with 0.006 in. allowable; shaft stresses are 3080 psi with 25,000 psi allowable; and bearing loads for drive end are 98 lb, with 7670 lb allowable. Bearing loads for thrust end are 765 lb, with 17,600 lb allowable.

(3) Calculated loads were evaluated and accepted by General Electric per letter GB-86-27 dated 1/31/86.

TABLE 3.9-4u

FUEL STORAGE RACKS

Acceptance Criteria	Loading	Primary Stress Type	Allowable Stress, psi	Calculated Stress, psi[3]
The allowable primary bending stress is based on ASME Section for type ASTM B221 6061-T6 aluminum alloy.				
F_u = 38,000 psi				
F_y = 35,000 psi[1]				
For normal condition:	For normal condition:			
$S_{limit} = 0.66\ F_y$	1. Normal pressure 2. Weight	Bending	23,100	15,230
For emergency condition:[2]	For emergency condition:			
$S_{limit} = 0.88\ F$	1. Normal loads 2. OBE 3. SRV 4. LOCA	Bending	30,800	30,800
For faulted condition:	For faulted condition:			
$S_{limit} = 0.88\ F_y$	1. Normal loads 2. Safe shutdown earthquake 3. SRV 4. LOCA	Bending	30,800	30,800

(1) Operability assurance is demonstrated by analysis.

(2) Normal and upset condition allowable is used to evaluate the emergency condition.

(3) Above values are taken from the generic analysis using La Salle's equipment/fuel storage racks which envelope all other storage racks. Hope Creek's storage fuel pool is smaller than LaSalle's; however, the rack system for both are comprised of identical components. Calculated stresses for Hope Creek will be lower than shown above because of the size and lesser loadings.

TABLE 3.9-4v

HIGH PRESSURE COOLANT INJECTION PUMP

Criterion	Loading	Component and Controlling Stress	Allowable Stress, psi	Calculated Stress, psi
Pressure boundary stress limits of the various components for the HPCI pump assembly are based on the ASME B&PV Code Sections III and VIII. Design temperature 140°F, design pressure 1500 psig.				
Pressure Boundary Parts				
Pump case - A216 Gr.WCB - S_y = 30,000 psi (main & booster pumps)	(SEE THE FOLLOWING PAGES FOR DETAILED INFORMATION)			
Case bolting - A193 Gr.B7 - S_y = 25,000 psi (main & booster pumps)				
Component Parts				
Holddown bolts (main pump) A325, (Yield Stress is based on A307 allowable values which is conservative)				
Holddown bolts (booster pump) A325, (Yield Stress is based on A307 allowable values which is conservative)				
Pump pins (main) A193Gr.B7				
Pump pins (booster) A193Gr.B7				
For the normal plus upset condition(1):	Normal plus upset loads include:			

Criterion	Loading	Component and Controlling Stress	Allowable Stress, psi	Calculated Stress, psi	
Pump casings: S_A = 17,500 x .8 = 14,000 psi	Design pressure and temperature, dead weight and thermal expansion, operating basis earthquake (OBE), and suction and discharge nozzle loads	Pump casings, gen. memb.	14,000		
Case bolting (main) S_A = 20,000 x 1.2 = 24,000 psi		Case bolting (main) tensile	24,000		
Case bolting (booster) S_A = 20,000 x 1.2 = 24,000 psi		Case bolting (booster) tensile	24,000		
Holddown bolts (main) S_A = 25,000 x 1.2 = 30,000 psi		Holddown bolt (main) tensile	30,000	See Note 1	
Holddown bolts (booster) S_A = 25,000 x 1.2 = 30,000 psi		Holddown bolt (booster) tensile	30,000		
Pump pins (main) S_A = 25,000 x 1.2 = 30,000 psi		Pump pins (main) shear	30,000		
Pump pins (booster) S_A = 25,000 x 1.2 = 30,000 psi		Pump pins (booster) shear	30,000		
<u>For the Emergency or Faulted Conditions</u>:	Emergency or faulted condition loads include:				
Pump case, S_A (general membrane) = 1.2 x 14,000 = 16,800 psi	Design pressure and temperature, dead weight and thermal expansion, LOCA load and design basis earthquake (SSE), and emergency and faulted nozzle	Pump case (main) gen. memb.	14,000	12,050	
S_A ($\bar{\ }_M$ or $\sqcap_L$ + $\sqcap_F$ = 1.8 14,000 x 14,000 = 25,200 psi	3,630 loads.	Pump case (booster) gen. memb.			
Case bolting (main) S_A = 20,000 x 1.2 = 24,000 psi		Case bolting (main) tensile	24,000	10,100	
Case bolting (booster) S_A = 20,000 x 1.2 = 24,000 psi		Case bolting (booster) tensile	24,000	17,400	
Holddown bolts (main) S_A = 25,000 x 1.2 = 30,000 psi		Holddown bolts (main) tensile		25,000	20,240

Criterion	Loading	Component and Controlling Stress	Allowable Stress, psi	Calculated Stress, psi
Holddown bolts (booster) S_A = 25,000 x 1.2 = 30,000 psi		Holddown bolts (booster) tensile	25,000	14,870
Pump pins (main) S_A = 25,000 x 1.8 = 45,000 psi		Pump pins (main) shear	30,000	23,180
Pump pins (booster) S_A = 25,000 x 1.8 = 45,000 psi		Pump pins (booster) shear	30,000	19,760
Nozzle Load Definition:				
Forces are in (lb) and moments are in (ft-lb) the allowable combination of forces and moments are as follows: $\frac{F_i}{F_0} + \frac{M_1}{M_0} \leq 1$ SEE HARD COPY FOR DIAGRAM	The normal plus upset condition loads include: Design pressure and temperature, dead weight and thermal expansion, and operating basis earthquake.	F_0 = Allowable value of F when all moments are zero (lbs). M_0 = Allowable value of M when all forces are zero. (ft.-lbs)	Suction nozzle: F_0 = 5,570 M_0 = 15,370 Discharge nozzle: F_0 = 7,850 M_0 = 15,385	F2=2998 M2=4899 F2=4700 M2=13191
	The emergency or faulted condition loads include: Design pressure and temperature, dead weight and thermal expansion, and safe shutdown earthquake.		Suction nozzle: F_0 = 6,680 M_0 = 18,450 Discharge nozzle: F_0 = 9,420 M_0 = 18,465	F2=4829 M2=4019 F2=4813 M2=13792

Where F_i (lb) is the maximum of the three orthogonal forces F_x, F_y, F_z and M_i (ft.-lbs) is the maximum of any of the three orthogonal moments M_x, M_y, M_z) for the same reference coordinates. F_0 and M_0 for upset and faulted conditions are base values given above.

Note:

(1) The calculated stresses for the emergency or faulted conditions are less than their corresponding allowable stresses for the normal plus upset condition; therefore, normal plus upset condition is not evaluated.
(2) Calculated loads were evaluated and accepted by General Electric per letter GB-86-27 dated 1/31/86.

TABLE 3.9-4w

CONTROL ROD DRIVE

Criteria	Loading[1]	Primary Stress Type	Allowable Stress, psi	Calculated Stress, psi
Ring Flange				
Allowable primary membrane stress plus bending stress is based on ASME B&PV Code, Section III, for type 304 stainless steel @ 250°F: S_m = 20,000 psi				
For normal and upset condition: $S_{allow} = 1.5 \times S_m$	For normal & upset condition: 1. Normal loads[1] 2. Scram with OBE	General membrane & bending	30,000	8285
For emergency condition: $S_{allow} = 1.8 \times S_m$	For emergency condition: 1. Normal loads[1] 2. Scram with accumulator at overpressure	General membrane & bending	36,000	1370
For faulted condition: $S_{allow} = 0.80\ S_u$ for gen. memb. $= 2.16\ S_y$ for gen. memb. & bending	For faulted condition: 1. Normal loads[1] 2. Scram with SSE 3. Scram with stuck rod	General membrane & bending	71,925	3563
Indicator Tube				
Allowable primary membrane stress plus bending stress is based on ASME B&PV Code, Section III for type 316 stainless steel @ 250°F: S_m = 20,000 psi				
For normal and upset condition: $S_{allow} = 1.5 \times S_m$	For normal & upset condition: 1. Normal loads[1] 2. Scram with OBE	General membrane & bending	30,000	15,242

TABLE 3.9-4w (Cont)

Criteria	Loading[1]	Primary Stress Type	Allowable Stress, psi	Calculated Stress, psi
For emergency condition: $S_{allow} = 1.8 \times S_m$	For emergency condition: 1. Normal loads[1] 2. Failure of pressure regulating system 3. Scram with accumulator at overpressure	General membrane & bending	31,028	20,795
For faulted condition: $S_{allow} = 3.6 \times S_m$	For faulted condition: 1. Normal loads[1] 2. Scram with SSE 3. Scram with stuck rod	General membrane & bending	72,000	25,700

(1) Normal loads include pressure, temperature, weight, and mechanical loads.

TABLE 3.9-4x

CONTROL ROD DRIVE HOUSING

Criteria	Loading	Primary Stress Type	Allowable Stress, psi	HISTORICAL INFORMATION Calculated Stress, psi
Primary Stress Limit - The allowable primary membrane stress is based on the ASME B&PV 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. OBE with housing lateral support installed	Maximum membrane stress intensity occurs at the tube to tube weld near the center of the housing for normal, upset, and emergency conditions	16,660	14,480
For normal and upset condition: $S_{limit} = 1.0\ S_m$ $S_m = 16,660$ psi @ 575°F				
For faulted condition:[1] $S_{limit} = 2.4\ S_m$	Faulted condition loads: 1. Design pressure 2. Stuck rod scram loads 3. SSE with housing lateral support installed		40,000	22,030

(1) Analyzed to emergency conditions limits.

TABLE 3.9-4y

JET PUMPS

Criteria	Loading Combinations	Stress Type	Allowable Stress, psi	HISTORICAL INFORMATION Calculated Stress, psi
Primary membrane plus bending stress based on ASME B&PV Code, Section III.				
For service levels A & B (normal and upset) condition: For type 304SS: $S_m = 16,900$ psi @ 550°F $S_{limit} = 3.0\ S_m$	1. Dead weight 2. Pressure 3. SRV 4. OBE	Primary membrane plus bending plus secondary membrane	50,700	19,346
For service level C (emergency) condition: For type 304SS: $S_m = 16,900$ psi @ 550°F $S_{limit} = 2.25\ S_m$	1. Dead weight 2. Pressure 3. SRV 4. OBE	Primary membrane plus bending	30,420	19,346
For service level D (faulted) condition: For type 304SS: $S_m = 16,900$ psi @ 550°F $S_{limit} = 3.6\ S_m$	1. Dead weight 2. Pressure 3. Chugging 4. SRV 5. SSE	Primary membrane plus bending	60,840	34,417

(1) Maximum stress occurs at the jet pump riser brace.

TABLE 3.9-4z

LPCI COUPLING[(1)]

Criteria	Loading Combinations	Stress Type	Allowable Stress, psi	HISTORICAL INFORMATION Calculated Stress, psi
Primary membrane plus bending stress based on ASME B&PV Code Section III for type CF3				
For service levels A & B (normal & upset) condition:	NL + ΔP + OBE + SRV	Primary membrane & bending	25,350	3839
S_{limit} = 25,350 psi				
For service level C (emergency) condition:	NL + ΔP + Chugging + SRV	Primary membrane & bending	38,025	Negligible
S_{limit} = 38,025 psi				
For service level D (faulted) condition:	NL + ΔP + AP + SSE	Primary membrane & bending	60,840	15,174
S_{limit} = 60, 840 psi				

(1) Highest stressed region is attachment ring.

TABLE 3.9-5

NON-NSSS PIPING SYSTEMS STARTUP TESTING

Piping System	Code(s)/ S.C./H.E. M.E.(1)	Temp >300°F	Thermal Expansion(2)	Dynamic Transient(3)	Steady State Vibration(4)	Remarks
Main steam (Bechtel-supplied)	ASME III - 1,2,3; B31.1/ SC 1 & Non-SC 1/ HE	Yes	Yes	Yes	Yes	Main stop valve closure and SRV opening transient
Extraction steam	B31.1/ Non-SC 1/ HE	Yes	N/R(5)	N/R(5)	N/R(5)	
Condensate transfer and storage	ASME III - 2, 3; B31.1/ SC 1 & Non-SC 1/ HE & ME	No	N/R(5)	N/R	N/R	
Feedwater	ASME III-1,2; B31.1/ SC 1 & Non-SC 1/ HE	Yes	Yes(8)	Yes	Yes	Power ascension test for safety-related piping portion only
Liquid radwaste	ASME III-2; B31.1/SC 1 & Non-SC 1/ ME	No	N/R	N/R	N/R	Piping between containment isolation valves
Condenser air removal	B31.1/ Non-SC 1/ HE	Yes	N/R(5)	N/R	N/R	
Service water	ASME III-3; B31.1; AWWA/ SC 1 & Non-SC 1/ ME	No	N/R	N/R	N/R	
Lube oil and diesel fuel oil storage and transfer	ASME III-3; B31.1/ SC 1 & Non-SC 1/ ME	No	N/R	N/R	N/R	
Auxiliary steam	B31.1/ Non-SC 1/ HE & ME	Yes	Yes(8)	N/R	N/R	

Piping System	Code(s)/ S.C./H.E. M.E.(1)	Temp >300°F	Thermal Expansion(2)	Dynamic Transient(3)	Steady State Vibration(4)	Remarks
Fire protection	ASME III-3; B31.1; NFPA/ SC 1 & Non-SC 1/ ME	No	N/R	N/R	N/R	
Flow instrument lines	ASME III-1/ SC 1/ HE	Yes	N/R	N/R	Yes	Portion of system from steam line to containment penetration
Instrument compressed air	B31.1; ASME III-3/ SC 1 & Non-SC 1/ ME	Yes	N/R(5)	N/R	N/R	
Primary containment instrument gas	ASME III-2, 3; B31.1/SC 1 & Non-SC 1/ ME	No	N/R	N/R	N/R	
Reactor feed pump turbine steam	B31.1/ Non-SC 1/ HE	Yes	Yes(6)	N/R	N/R	
Breathing air	ASME III-3; B31.1/ SC 1 & Non-SC 1/ ME	No	N/R	N/R	N/R	
Diesel engine auxiliaries	ASME III-3; B31.1/ SC 1 & Non-SC 1/ HE	Yes	Yes(8)	N/R	Yes	Steady state vibration for diesel starting air only.
Safety and turbine auxiliaries cooling	ASME III-3; B31.1/ SC 1 & Non-SC 1/ ME	No	N/R	N/R	N/R	
Torus water cleanup	ASME III-2,3; B31.1/ SC 1 & Non-SC 1/ ME	No	N/R	N/R	N/R	

TABLE 3.9-5 (Cont)

Piping System	Code(s)/ S.C./H.E. M.E.(1)	Temp >300°F	Thermal Expansion(2)	Dynamic Transient(3)	Steady State Vibration(4)	Remarks
High pressure coolant injection (HPCI)	ASME III-1,2; B31.1/ SC 1 & Non-SC 1/ HE & ME	Yes	Yes(8)	Yes	Yes	HPCI turbine stop valve closure (TSVC) transient. Steady state vibration for steam supply, turbine exhaust, and HPCI pump suction and discharge lines
Reactor core isolation cooling (RCIC)	ASME III-1,2; B31.1/ SC 1 & Non-SC 1/ HE & ME	Yes	Yes(8)	N/R	Yes	Steady state vibration for RCIC steam supply, turbine exhaust, and RCIC pump suction and discharge piping
Reactor water cleanup (RWCU)	ASME III-1,3; B31.1/ SC 1 & Non-SC 1/ HE & ME	Yes	Yes(8)	N/R	Yes(6)	Steady state vibration for RWCU from recirculating loops A and B and RPV to CE 207, and from CE 207 to the feedwater tie-in.
Residual heat removal (RHR)	ASME III-1,2, 3; B31.1/ SC 1 & Non-SC 1/ HE & ME	Yes	Yes(8)	N/R	Yes(6)	
Control rod drive (CRD)	ASME III-3; B31.1/ SC 1 & Non-SC 1/ ME & HE	No	N/R	N/R	N/R	
Standby liquid control (SLC)	ASME III-1,2; B31.1/ SC 1 & Non-SC 1/ HE & ME	No	N/R	N/R	N/R	
Core spray	ASME III-1,2; B31.1/ SC 1 & Non-SC 1/ HE & ME	Yes	Yes(7)(8)	N/R	Yes(6)	Steady state vibration for core spray pump suction and discharge
Plant steam leak detection	ASME III-2; B31.1/ SC 1 & Non-SC 1/ ME	No	N/R	N/R	N/R	

TABLE 3.9-5 (Cont)

Piping System	Code(s)/ S.C./H.E. M.E.(1)	Temp >300°F	Thermal Expansion(2)	Dynamic Transient(3)	Steady State Vibration(4)	Remarks
Fuel pool cooling, cleanup, and demineralizer	ASME III-2,3; B31.1/ SC 1 & Non-SC 1/ ME	No	N/R	N/R	N/R	
Containment atmosphere control	ASME III-2; B31.1/ SC 1 & Non-SC 1/ ME & HE	Yes	N/R(5)	N/R	N/R	
Offgas recombiner	B31.1/ Non-SC 1/ HE	Yes	Yes(8)	N/R	N/R	
Chilled water	ASME III-2,3; B31.1/ SC 1 & Non-SC 1/ ME	No	N/R	N/R	N/R	

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(1) Codes: ASME III; B&PV Code, Section 1, 2, or 3 denotes nuclear Class 1, 2, or 3 piping.

SC 1 or Non-SC 1 denotes Seismic Category I or II.

B31.1 denotes ANSI B31.1.

AWWA denotes American Water Works Association

NFPA denotes National Fire Protection Association

HE denotes high energy piping system, i.e., pressure $\geq$275 psi or temperature $\geq$200°F during normal plant operation.

ME denotes moderate energy piping system.

(2) Thermal expansion tests for the indicated systems correspond to the test boundaries and requirements stated in the vendor test specification.

(3) Dynamic transient tests for the indicated systems correspond to the test boundaries and requirements stated in the vendor test specification.

(4) Steady-state vibration tests for the indicated systems correspond to the test boundaries and requirements stated in the vendor test specification.

(5) N/R denotes not required by the criteria of SRP 3.9.2 and Regulatory Guide 1.70, and means the test is not performed for the reasons listed below. Bechtel exercises judgement to include other piping systems in the testing program on a prudent engineering basis.

a) For thermal expansion tests: the system is not safety-related, or the normal operating temperature is less than 300°F.
b) For dynamic transient test: the system is not safety-related, or does not experience any significant transients.
c) For steady state vibration tests: the system is not safety-related, or no significant vibration is expected, based on previous experience with similar systems.

(6) Test to be done during preoperational test program.

(7) For the effect of RPV expansion only. No flow in the core spray line.

(8) Thermal expansion testing is required only for those portions of the system that have an operating temperature of 300°F or higher per the line index.

TABLE 3.9-6

SEISMIC ANALYSIS FOR NON-NSSS MECHANICAL EQUIPMENT

Equipment Identification			Location				
Description	Equipment Number	Bldg	Elevation	Vendor	PO	Qualification Method(1)	Standard(2)
Containment hydrogen recombiner control panel	1A, 1B-C633	Aux	137' 0"	Rockwell	M047A	DT	A,I
RHR blowout panel to torus compartment	1A, 1B-S284	Reac	54' 0"	W.J. Woolley	M177	SA	A,I
RCIC blowout panel to torus compartment	1C-S284	Reac	54' 0"	W.J. Woolley	M177	SA	A,I
HPCI blowout panel to torus compartment	1D-S284	Reac	54' 0"	W.J. Woolley	M177	SA	A,I
Diesel generator	1A, 1B-G400, 1C, 1D-G400	Aux	102' 0"	Colt/FMED	M018	DA	A,F,I
Service water traveling screen control panel	1A, 1B-C515, 1C, 1D-C515	Intake struct	107' 0"	Royce	M020	DT	A,I
Service water strainer	1A, 1B-F509, 1C, 1D-F509	Intake struct	93' 0"	Zurn Ind Inc	M076	SA	A,I
Station service water pump	1A, 1B-P502, 1C, 1D-P502	Intake struct	93' 0"	Hayward Tyler	M080	SA	A,F,I
Spray water booster pump	1A, 1B-P507, 1C, 1D-P507	Intake struct	79' 0"	Hayward Tyler	M082	SA	A,F,I
Service water traveling screen	1A, 1B-S501, 1C, 1D-S501	Intake struct	114' 0"	Royce	M020	DA	A,I
Service water pump lubrication water tank	10-T543 10-T544	Intake struct	122' 0" 122' 0"	CVI CVI	M707 M707	SA SA	A,F,I
SACS heat exchanger	1A1E, 1A2E 201, 1B1E, 1B2E 201	Reac	102' 0"	Graham	M069	DA	A,I

TABLE 3.9-6 (Cont)

Equipment Identification			Location				
Description	Equipment Number	Bldg	Elevation	Vendor	PO	Qualification Method[1]	Standard[2]
SACS pump	1A, 1B-P210, 1C, 1D-P210	Reac	102' 0"	Ingersoll Rand	M070	SA	A,F,I
SACS expansion tank	1A, 1B-T205	Reac	201' 0"	CVI	M707	DA	A,F,I
Diesel generator lube oil heat exchanger	1A, 1B-E404, 1C, 1D-E404	Aux	102' 0"	Colt/FMED	M018	DA	F
Diesel generator jacket water heat exchanger	1A, 1B-E405, 1C, 1D-E405	Aux	102' 0"	Colt/FMED	M018	DA	F
Diesel generator exciter panel	1A, 1B-C420, 1C, 1D-C420	Aux	102' 0"	Colt/FMED	M018	DT	A,I
Diesel generator local engine control panel	1A, 1B-C421, 1C, 1D-C421	Aux	102' 0"	Colt/FMED	M018	DT	A,I
Diesel generator remote central generator panel	1A, 1B-C422, 1C, 1D-C422	Aux	130' 0"	Colt/FMED	M018	DT	A,I
Diesel generator remote engine control panel	1A, 1B-C423, 1C, 1D-C423	Aux	130' 0"	Colt/FMED	M018	DT	A,I
Diesel generator load sequencer panel	1A, 1B-C428 1C, 1D-C428	Aux	130' 0"	Consolidated Control	J810	DT	A,I
Diesel fuel oil filter	1A, 1B-F405, 1C, 1D-F405	Aux	102' 0"	Colt/FMED	M018	DA	F
Diesel fuel oil strainer	1A, 1B-F406, 1C, 1D-F406	Aux	102' 0"	Colt/FMED	M018	DA	F
Diesel fuel oil transfer pump	1A, 1B-P401, 1C, 1D-P401, 1E, 1F-P401, 1G, 1H-P401	Aux	54' 0"	Crane-Chempump	M092	SA	A,F,I

TABLE 3.9-6 (Cont)

Equipment Identification		Location					
Description	Equipment Number	Bldg	Elevation	Vendor	PO	Qualification Method(1)	Standard(2)
Motor driven fuel oil pump	1A, 1B-P402, 1C, 1D-P402	Aux	102' 0"	Colt/FMED	M018	SA	F
Engine driven fuel oil pump	1A, 1B-P404, 1C, 1D-P404	Aux	102' 0"	Colt/FMED	M018	SA	F
Diesel fuel oil storage tank	1A, 1B-T403, 1C, 1D-T403, 1E, 1F-T403, 1G, 1H-T403	Aux	54' 0"	Buffalo Tank	M105	SA	A,F,I
Diesel fuel oil day tank	1A, 1B-T404 1C, 1D-T404	Aux	102' 0"	Colt/FMED	M018	SA	F
Jacket water keep warm heater	1A, 1B-E407, 1C, 1D-E407	Aux	102' 0"	Colt/FMED	M018	SA	F
Combustion air intercooler	1A, 1B-E408, 1C, 1D-E408	Aux	102' 0"	Colt/FMED	M018	SA	F
Combustion air intake filter	1A, 1B-F413, 1C, 1D-F413	Aux	130' 0"	Colt/FMED	M018	DA	F
Intake silencer	1A, 1B-F414, 1C, 1D-F414	Aux	102' 0"	Colt/FMED	M018	DA	F
Diesel generator exhaust silencer	1A, 1B-F415, 1C, 1D-F415	Aux	102' 0"	Colt/FMED	M018	DA	F
Diesel engine jacket water pump	1A, 1B-P408, 1C, 1D-P408	Aux	102' 0"	Colt/FMED	M018	DA	F
Jacket water keep-warm pump	1A, 1B-P410, 1C, 1D-P410,	Aux	102' 0"	Colt/FMED	M018		F
Lube oil keep warm heater	1A, 1B-E406, 1C, 1D-E406	Aux	102' 0"	Colt/FMED	M018	SA	F
Rocker arm lube oil filter	1A, 1B-F403, 1C, 1D-F403	Aux	102' 0"	Colt/FMED	M018	DA	F

TABLE 3.9-6 (Cont)

Equipment Identification		Location					
Description	Equipment Number	Bldg	Elevation	Vendor	PO	Qualification Method(1)	Standard(2)
Lube oil filter	1A, 1B-F404, 1C, 1D-F404	Aux	102' 0"	Colt/FMED	M018	DA	F
Lube oil strainer	1A, 1B-F407, 1C, 1D-F407	Aux	102' 0"	Colt/FMED	M018	DA	F
Rocker arm lube oil pump	1A, 1B-P403, 1C, 1D-P403	Aux	102' 0"	Colt/FMED	M018	SA	F
Engine driven lube oil pump	1A, 1B-P405, 1C, 1D-P405	Aux	102' 0"	Colt/FMED	M018	SA	F
Rocker arm motor driven prelube pump	1A, 1B-P406, 1C, 1D-P406	Aux	102' 0"	Colt/FMED	M018	SA	F
Lube oil keep warm pump	1A, 1B-P407, 1C, 1D-P407	Aux	102' 0"	Colt/FMED	M018	SA	F
Lube oil makeup tank	1A, 1B-T406, 1C, 1D-T406	Aux	102' 0"	Colt/FMED	M018	SA	F
Start air receiver tank	1A, 1B-T408, 1C, 1D-T408, 1E, 1F-T408, 1G, 1H-T408	Aux	102' 0"	Colt/FMED	M018	DA	F
SRV control air supply accumulator	1A, 1B-T210, 1C, 1D-T210, 1E, 1F-T210, 1G, 1H-T210, 1J, 1K-T210, 1L, 1M-T210, 1P, 1R-T210	Reac	102' 0"	CVI	M707	SA	A,F,I
MSIV control air supply accumulator	1A, 1B-T211, 1C, 1D-T211,	Reac	102' 0"	CVI	M707	DA	A,F,I
	1A, 1B-T212, 1C, 1D-T212	Reac	102' 0"	CVI	M707	DA	A,F,I
ECCS jockey pump	1A, 1B-P228, 1C, 1D-P228	Reac	54' 0"	Hayward Tyler	M082	SA	A,F,I

TABLE 3.9-6 (Cont)

Equipment Identification			Location				
Description	Equipment Number	Bldg	Elevation	Vendor	PO	Qualification Method(1)	Standard(2)
Fuel pool heat exchanger	1A, 1B-E202	Reac	162' 0"	Alfa-Laval	M071	DA	A,I
Fuel pool cooling pump	1A, 1B-P211	Reac	162' 0"	Hayward Tyler	M082	SA	A,F,I
High density spent fuel storage rack	10-S287	Reac	168' 0"	GCA	M178	DA	J
Air accumulator for torus isolation vacuum relief valve	1A, 1B-T277	Reac	77' 0"	CVI	M707	SA	A,F,I
Instrument gas compressor skid	1A, 1B-S934	Reac	132' 0"	CVI	M048	DT	A,I
Instrument gas receiver	1A, 1B-T201	Reac	132' 0"	CVI	M048	SA	A,I
Control room return air fan	1A, 1B-V415	Aux	155' 0"	Buffalo	M719	SA	A,I
Technical support center emergency filter fan	00-V314	Aux	153' 0"	Buffalo	M713	DA	A,I
Technical support center emergency filter unit	00-VH313	Aux	153' 0"	AAF	M786	DA	A,I
Technical support center supply unit	00-VH314	Aux	153' 0"	AAF	M711	DA	A,I
Service area air handling unit	00-VH316	Aux	137' 0"	AAF	M711	DA	A,I
Traveling screen fans	0A, 0B-V558	Intake struct	114' 0"	Joy	M719A	SA	A,I
Reactor Building FRVS recirculation system fan	1A, 1C-V213 1B, 1F-V213 1D, 1E-V213	Reac Reac Reac	132' 0" 178' 0" 162' 0"	Buffalo Buffalo Buffalo	M713 M713 M713	DT DT DT	A,I A,I A,I

Equipment Identification		Location					
Description	Equipment Number	Bldg	Elevation	Vendor	PO	Qualification Method(1)	Standard(2)
RCIC pump room unit cooler	1A, 1B-VH208	Reac	54' 0"	AAF	M711	DA	A,I
HPCI pump room unit cooler	1A, 1B-VH209	Reac	54' 0"	AAF	M711	DA	A,I
RHR pump room unit cooler	1A, 1B-VH210,	Reac	54' 0"	AAF	M711	DA	A,I
	1C, 1D-VH210,	Reac	54'-0"			DA	A,I
	1E, 1F-VH210,	Reac	77'-0"			DA	A,I
	1G, 1H-VH210	Reac	54'-0"			DA	A,I
Core spray pump room unit cooler	1A, 1B-VH211,	Reac	54' 0"	AAF	M711	DA	A,I
	1C, 1D-VH211,						A,I
	1E, 1F-VH211,						A,I
	1G, 1H-VH211						A,I
Reactor Building FRVS recirculation filter system	1A-VH213	Reac	132' 0"	AAF	M786	DA	A,I
	1B-VH213	Reac	178' 0"	AAF	M786	DA	A,I
	1C-VH213	Reac	132' 0"	AAF	M786	DA	A,I
	1D, 1E-VH213	Reac	162' 0"	AAF	M786	DA	A,I
	1F-VH213	Reac	178' 0"	AAF	M786	DA	A,I
SACS pump room unit cooler	1A, 1B-VH214, 1C, 1D-VH214	Reac	102' 0"	AAF	M711	DA	A,I
Emergency area cooling system cooler control panel	1A, 1B-C281	Reac	102' 0"	Comsip	M780A	DA	A,I
	1C, 10-C281	Reac	77'-0"	Comsip	M78	DA	A,I
Reactor Building FRVS control panel	1A, 1C-C285,	Aux	178' 0"	Comsip	M780A	DA	A,I
	1B, 1D-C285	Aux	124'-0"				
Reactor Building FRVS vent filter	1A, 1B-VH206	Reac	145' 0"	AAF	M786	DA	A,I
RCIC pump room duct heater	10-VE-259	Reac	54' 0"	AAF	M786	DA	A,I
HPCI pump room duct heater	10-VE260	Reac	54' 0"	AAF	M786	DA	A,I

TABLE 3.9-6 (Cont)

Equipment Identification		Location					
Description	Equipment Number	Bldg	Elevation	Vendor	PO	Qualification Method(1)	Standard(2)
Standby liquid control room duct heater	1A, 1B-VE261	Reac	162' 0"	AAF	M786	DA	A,I
Diesel generator area HVAC	1A, 1B-C483, 1C, 1D-C483	Aux	178' 0"	Comsip	M780A	DA	A,I
Diesel area battery room exhaust fan	1A, 1B-V406, 1C, 1D-V406	Aux	163' 6"	Buffalo	M713	DT	A,I
Diesel generator room recirculation fan	1A, 1B-V412, 1C, 1D-V412, 1E, 1F-V412, 1G, 1H-V412	Aux	77' 0"	Buffalo	M719	DA	A,I
Diesel generator room cooling coil	1A, 1B-VE412, 1C, 1D-VE412, 1E, 1F-VE412, 1G, 1H-VE412	Aux	77' 0"	Trane	M731	DA	A,I
Switchgear room unit cooler	1A, 1B-VH401, 1C, 1D-VH401	Aux	163' 6"	AAF	M711	DT	A,I
Class 1E panel room supply air unit	1A, 1B-VH408	Aux	163' 6"	AAF	M711	DA	A,I
Battery room duct heater	1A, 1B-VE420, 1C, 1D-VE420	Aux	146' 0"	AAF	M786	SA	A,I
Control room emergency air supply fan	1A, 1B-V400	Aux	155' 3"	Buffalo	M713	DT	A,I
Control area battery room heat exchanger fan	1A, 1B-V410	Aux	178' 0"	Buffalo	M713	DT	A,I
Control room emergency supply unit	1A, 1B-VH400	Aux	155' 0"	AAF	M786	DA	A,I
Control room supply unit	1A, 1B-VH403	Aux	155' 0"	AAF	M711	DA	A,I

TABLE 3.9-6 (Cont)

Equipment Identification			Location				
Description	Equipment Number	Bldg	Elevation	Vendor	PO	Qualification Method(1)	Standard(2)
Control equipment room supply unit	1A, 1B-VH407	Aux	178' 0"	AAF	M711	DA	A,I
Control room water chiller	1A, 1B-K400	Aux	155' 0"	Carrier	M723	DA	A,I
Control room chilled water system head tank	1A, 1B-T410	Aux	178' 0"	CVI	M707	DA	A,I,F
Control room chilled water circulation pump	1A, 1B-P400	Aux	155' 0"	Hayward Tyler	M082	SA	A,I,F
Control room water chiller pumpout unit							
Control room chilled water head tank	1A, 1B-T413	Aux	178' 0"	CVI	M707	DA	A,I,F
Intake structure supply fan	1A, 1B-V503, 1C, 1D-V503	Intake struct	122' 0"	Joy	M719A	SA	A,I
Intake structure exhaust fan	1A, 1B-V504, 1C, 1D-V504	Intake struct	122' 0"	Joy	M719A	SA	A,I
Class 1E channel diesel generator grounding transformer	1A, 1B-G403, 1C, 1D-G403	Aux	102' 0"	Colt/FMED	M018	DT	F
Chilled water circulation pump	1A, 1B-P414	Aux	178' 0"	Hayward Tyler	M082	DA	A,I,F

(1) DA - dynamic analysis.
DT - dynamic testing.
SA - static analysis

(2) A - IEEE 344-1975
F - ASME Code, Section III
I - NRC R.G. 1.100, Rev. 1
J - Appendix D of SRP 3.8.4

TABLE 3.9-7

DESIGN LOADING COMBINATIONS FOR ASME B&PV CIDE
CLASS 1, 2 AND 3 NON-NSSS COMPONENTS

Condition	Design Loading Combinations(1)(2)(3)(4)
Design	(a) PD
Normal	(a) PD + DW
Upset	(a) $PO + DW + (OBE^2 + RVC^2)^{1/2}$
	(b) PO + DW + OBE + RVO
	(c) PO + DW + FV
Emergency	(a) $PO + DW + (OBE^2 + FV^2)^{1/2}$
Faulted	(a) PO + DW + SSE + RVO
	(b) $PO + DW + (SSE^2 + RVC^2)^{1/2}$
	(c) $PO + DW + (SSE^2 + DBA^2)^{1/2}$

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(1) As required by the appropriate subsection, i.e., NB, NC, or ND, of the ASME B&PV Code, Section III, Division I, other secondary loads, such as thermal expansion, thermal transient, thermal gradients, and anchor point displacement portion of the OBE, may require consideration in addition to the primary stress producing loads listed.

(2) For torus attached piping, the loading combinations used in the piping analysis are those given in the Plant Unique Analysis Application Guide (PUAAG) (NEDO-24583-1, October 1978, Table 5-2).

(3) Definition of symbols used:

PD	-	design pressure
PO	-	operating pressure
DW	-	dead weight
OBE	-	operating basis earthquake (inertia portion)
SSE	-	safe shutdown earthquake (inertia portion)
FV	-	transient response of the piping system associated with fast valve closure time less than 5 seconds
RVC	-	transient response of the piping system associated with relief valve opening in a closed system
RVO	-	transient response of the piping system associated with relief valve opening in an open system
DBA	-	design basis accident.

(4) For components other than Bechtel supplied piping, the pressure load for the normal condition may be either PD or PO depending upon the requirements of the ASME Code in effect for that component or whichever is more conservative.

TABLE 3.9-8

DESIGN CRITERIA FOR ASME B&PV CODE CLASS 1 NON-NSSS PIPING

Condition	Applicable Code Paragraph(1)(2)	Primary Stress Limits
Design	NB-3221 and NB-3652	1.5 S_m
Normal	NB-3222 and NB-3653	1.5 S_m
Upset	NB-3223 and NB-3654	1.8 Sm but not greater than 1.5 S_y
Emergency	NB-3224 and NB-3655	2.25 Sm but not greater than 1.8 S_y
Faulted	NB-3225 and NB-3656	3.0 S_m

(1) As specified by the ASME B&PV Code, Section III, 1977 through Summer 1979 Addenda.

(2) Functional capability of essential piping is ensured per NEDO-21985, September 1978.

TABLE 3.9-9

DESIGN CRITERIA FOR ASME B&PV CODE CLASS 1 NON-NSSS VALVES

Plant Condition	Design Loading Combinations (4)	Stress Limits (1)
Design	PD	The valve shall conform to the requirement of Paragraph NB-3500 (Standard Design Rules)
Normal	POn + Bn	
Upset (3)	POu + OBE + Bu	NB-3525
Emergency (2)	POe + Be	NB-3526
Faulted (2)	POf + SSE + Bf	NB-3527

(1) As specified by the ASME B&PV Code, Section III, 1974 through Winter 1974 Addenda.

(2) Where valve function must be ensured (active valves) during emergency or faulted conditions, the specified emergency or faulted conditions for the plant is considered the normal condition for the valve.

(3) As required by subsection NB of ASME Section III, other loads such as thermal transient and thermal gradients may require additional consideration in addition to those primary stress producing loads listed.

(4) Definition of symbols used:

PD	-	Design pressure
PO	-	Operating pressure at noted plant condition
OBE	-	Operating Basis Earthquake loads (inertia portion) excluding loads from attached piping
SSE	-	Safe Shutdown Earthquake loads (inertia portion) excluding loads from attached piping
B	-	Piping end loads at noted plant condition
n	-	Normal
f	-	Faulted
e	-	Emergency
u	-	Upset

TABLE 3.9-10

DESIGN CRITERIA FOR ASME B&PV CODE CLASS 2 AND 3 NON-NSS VESSELS DESIGNED TONC-3300 AND ND-3300

Condition	Stress Limits(1)
Design and normal	The vessel conforms to the requirements of NC-3300 and ND-3300
Upset, emergency, and faulted	The vessel conforms to the requirements of ASME Code Case 1607-1

(1) As specified by the ASME B&PV Code, Section III, 1974 through Winter 1974 Addenda.

TABLE 3.9-11

DESIGN CRITERIA FOR ASME B&PV CODE CLASS 2 NON-NSSS VESSELS DESIGNED TO ALTERNATE RULES OF NC-3200

Condition	Stress Limits[(1)(2)]
Design and normal	The vessel conforms 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)]	$P_m \leq$ greater of 1.5 S_m or 1.2 S_y but not to exceed 0.7 S_u $(P_m \text{ or } P_L) + P_b \leq 2.4\ S_m$

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(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. It excludes discontinuities and concentrations. It is produced only by pressure and other mechanical loads.

P_L - local primary membrane stress intensity. It is the same as P_m, except that discontinuities are considered.

P_b - primary bending stress intensity. A component of primary stress intensity proportional to distance from the centroid of the solid section. It excludes discontinuities and concentrations. It is 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. It 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 of the Code.

S_y - yield strength value, Appendix I, Table I-2.0 of the Code.

S_u - Ultimate tensile strength.

(2) These limits do not take into account either local or general buckling that might occur in thin wall vessels. Such buckling must be considered for upset conditions, but need not be considered for emergency or faulted conditions unless required by the design specification.

(3) Fatigue analysis requirements of NC-3219 and Appendix XIV of the Code must also be considered.

(4) As an alternative to satisfying these limits, the faulted condition stress limits of Appendix F of the Code may be applied, provided that a complete analysis in accordance with NC-3211.1(c) is performed.

TABLE 3.9-12

DESIGN CRITERIA FOR ASME B&PV CODE CLASS 2 AND 3 NON-NSSS PIPING

Condition	Applicable Code Paragraph (1)(2)	Primary Stress Limits
Design:		
Sustained Loads	NC, ND-3652.1	$1.0S_h$
Occasional Loads	NC, ND-3652.2	$1.2S_h$
Normal and Upset	NC, ND-3652.2 & 3611	$1.2S_h$
Emergency	NC, ND-3611	$1.8S_h$
Faulted	Code Case 1606-1	$2.4S_h$

(1) As specified by the ASME B&PV Code, Section III, 1974 through Winter 1974 Addenda, except for Class 2 and 3 flanges, which are designed to 1979 Summer Addenda, Paragraph NC and ND-3658, and CRD piping which is designed to ASME B&PV Code Section III, 1980 Edition through Winter 1981 addenda, Section 3650.

(2) Functional capability of essential piping is ensured per NEDO-21985, September 1978.

TABLE 3.9-13

DESIGN CRITERIA FOR ASME B&PV CODE CLASS 2 AND 3 NON-NSSS PUMPS

Condition	Stress Limits (1)
Design and normal	The pump conforms to the requirements of Section III, Paragraphs NC-3400 and ND-3400
Upset, emergency, and faulted (2)	The pump conforms to the requirements of ASME Code Case 1636-1

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(1) As specified by the ASME B&PV Code, Section III, 1974 through Winter 1974 Addenda.

(2) Where pump function must be ensured (active pumps) during emergency or faulted conditions, the pumps nozzle loads due to the specified emergency or faulted plant conditions are considered in satisfying the normal condition stress limits for the pump.

TABLE 3.9-14

DESIGN CRITERIA FOR ASME B&PV CODE CLASS 2 AND 3 NON-NSSS VALVES

Plant Condition	Design Loading Combination (1)(4)	Stress Limits (1)(2)(3)
Design	PD	The valve shall conform to the requirements of Section III, 1974 Paragraphs NC-3500 or ND-3500, as applicable
Normal	POn + Bn	$S_m \leq 1.0S$ $(S_m \text{ or } S_L) + S_b \leq 1.50S$
Upset	POu + OBE + Bu	$Sm \leq 1.1S$ $(S_m \text{ or } S_L) + S_b \leq 1.65S$
Emergency	POe + Be	$Sm < 1.5 S$ $(S_m \text{ or } S_L) + S_b \leq 1.8S$
Faulted	POf + SSE + Bf	$Sm \leq 2.0S$ $(S_m \text{ or } S_L) + S_b \leq 2.4S$

(1) Definition of symbols:

S_m - General membrane stress
S_L - Local membrane stress
S_b - Bending stress
S - Allowable stress
PD - Design pressure
PO - Operating pressure at noted plant condition
n - Normal
u - Upset
e - Emergency
f - Faulted

OBE - Operating basis earthquake loads (inertia portion) excluding loads from attached piping

SSE - Safe shutdown earthquake loads (inertia portion) excluding loads from attached piping

B - Piping end loads at noted plant condition

(2) As specified by the ASME B&PV Code, Section III, 1974 through Winter 1974 Addenda.

(3) Where valve function must be ensured (active valves) during emergency or faulted conditions, the specified emergency or faulted plant conditions are considered as the normal condition for the valve.

(4) As required by subsection NC, ND of ASME Section III, other loads such as thermal transient and thermal gradients may require additional consideration in addition to those primary stress producing loads listed.

TABLE 3.9-15

NSSS ACTIVE PUMPS

Component Name	Identification as Shown on Applicable Figures
RHR pump	1A-P202 1B-P202 1C-P202 1D-P202
RCIC pump	10-P203
HPCI main pump	10-P204
Core spray pump	1A-P206 1B-P206 1C-P206 1D-P206
Standby liquid control pump	1A-P208 1B-P208
HPCI auxiliary oil pump	10-P213 (10-S211)[(1)]
HPCI booster pump	10-P217
RCIC turbine main oil pump	10-P271 (10-S212)
HPCI turbine main oil pump	10-P272 (10-S211)

(1) Pump is located on skid noted in parenthesis

TABLE 3.9-16

NSSS ACTIVE VALVES

System	Valve Type(1)	Quantity	Size, in.	Actuation(1)	Valve Identification Number
Main steam	GB	8	26	AO	F022A-D & F028A-D
Main steam	PSV	14	8	Self	F013A-H, J-M, P&R
Core spray	TCK	2	12	AO	F006A, B
Control rod drive hydraulic supply	GB	4	1	AO	F010, F011, F180, F181
	SV	2	1/2	Sol (Dual)	F009A & B, F182A & B
	SV	2	1	Sol	F160A & B
	GT	6	1	Sol	F163A & B, F162A-D
CRD - HCU	GB	2	0.5	AO	126, 127
	GB	1	0.5	AO/Sol (Dual)	139
Standby liquid control	GT	2	1-1/2	(2)	F004A, B
RCIC - turbine steam	GB	1	3	MO	HV: 4282
RCIC - turbine steam	GB	1	3	HO	FV: 4283
Neutron monitoring	BL	5	.375	Sol	J004A-1 thru -5
Neutron monitoring	GT	5	.375	(2)	J004B-1 thru -5
Residual heat removal	TCK	6	12	AO	F041A-D & F050A, B
HPCI - turbine steam	GT	1	10	MO	FV: 4880
HPCI - turbine steam	GT	1	10	HO	FV: 4879

(1) Definition of symbols:

BL - ball valve
BF - butterfly valve
GB - globe valve
GT - gate valve
TCK - testable check valve
PSV - pressure relief or safety valve
AO - air operator
MO - motor operator
HO - hydraulic operator
Sol - solenoid operator
DIA - diaphragm valve
E/H - electro hydraulic
SV - solenoid valve - 3-way

(2) Explosive valve.

TABLE 3.9-17

NON-NSSS ACTIVE PUMPS

Component Name	Identification as Shown on Applicable Figures
SACS pump	1A-P210
	1B-P210
	1C-P210
	1D-P210
Fuel pool cooling pump	1A-P211
	1B-P211
ECCS jockey pump	1A-P228
	1B-P228
	1C-P228
	1D-P228
Control room chilled water circulation pump	1A-P400
	1B-P400
Diesel fuel oil transfer pump	1A-P401
	1B-P401
	1C-P401
	1D-P401
	1E-P401
	1F-P401
	1G-P401
	1H-P401
Motor driven diesel fuel oil pump	1A-P402 (1A-G400)[(1)]
	1B-P402 (1B-G400)
	1C-P402 (1C-G400)
	1D-P402 (1D-G400)

Component Name	Identification as Shown on Applicable Figures
Rocker arm motor driven prelube pump	1A-P406 (1A-G400) 1B-P406 (1B-G400) 1C-P406 (1C-G400) 1D-P406 (1D-G400)
Lube oil keepwarm pump	1A-P407 (1A-G400) 1B-P407 (1B-G400) 1C-P407 (1C-G400) 1D-P407 (1D-G400)
Jacket water keepwarm pump	1A-P410 (1A-G400) 1B-P410 (1B-G400) 1C-P410 (1C-G400) 1D-P410 (1D-G400)
Water chiller oil pump	1A-P412 (1A-K400) 1B-P412 (1B-K400)
Water chiller pumpout pump	1A-P413 (1A-K400) 1B-P413 (1B-K400)
1E panel room chilled water pump	1A-P414 1B-P414
Water chiller oil pump	1A-P416 (1A-K403) 1B-P416 (1B-K403)
Water chiller pumpout pump	1A-P417 (1A-K403) 1B-P417 (1B-K403)

Component Name	Identification as Shown on Applicable Figures
Station service water pump	1A-P502 1B-P502 1C-P502 1D-P502
Spray water booster pump	1A-P507 1B-P507 1C-P507 1D-P507
Rocker arm lube oil engine driven pump	1A-P403 (1A-G400) 1B-P403 (1B-G400) 1C-P403 (1C-G400) 1D-P403 (1D-G400)
Engine driven diesel fuel oil pump	1A-P404 (1A-G400) 1B-P404 (1B-G400) 1C-P404 (1C-G400) 1D-P404 (1D-G400)
Engine driven fuel oil pump	1A-P405 (1A-G400) 1B-P405 (1B-G400) 1C-P405 (1C-G400) 1D-P405 (1D-G400)
Engine driven jacket water pump	1A-P408 (1A-G400) 1B-P408 (1B-G400) 1C-P408 (1C-G400) 1D-P408 (1D-G400)

Component Name	Identification as Shown on Applicable Figures
Engine driven cooling water pump	1A-P411 (1A-G400) 1B-P411 (1B-G400) 1C-P411 (1C-G400) 1D-P411 (1D-G400)

(1) Pump is located on skid noted in parenthesis.

TABLE 3.9-18

NON-NSSS ACTIVE VALVES

System	Valve Type(1)	Quantity	Size, in.	Actuation(1)	Valve Identification Number
Main steam	GB	1	2	MO	F071
Main steam	GT	2	3	MO	F016, F019
Main steam	CK	22	1	Self	V043 thru V058; V109 thru V114
Main steam	PSV	28	6	Self	F037A-H, J-M, P&R; 4500A-H, J-M, P&R
Feedwater	CK	1	4	MO	F039
Feedwater	CK	2	24	MO	F032 A, B
Feedwater	CK	2	24	AO	F074 A, B
Feedwater	CK	2	24	Self	V003, V007
Feedwater	TCK	2	4	Self	V127, V128
Condensate transfer and storage	GT	1	10	MO	F011
Condensate transfer and storage	TCK	8	3	Self	V036, V037, V039, V040, V050, V051, V060, V061
Condensate transfer and storage	TCK	9	4	Self	V005, V042, V043, V045, V046, V054, V055, V057, V058
Condensate transfer and storage	TCK	1	10	Self	V003
Reactor recirculation	CK	2	.75	Self	V043, V047
RHR	BF	2	18	MO	F048A, B
RHR	CK	10	1	Self	V089, V090, V194, V195, V206, V207, V208, V210, V211, V260
RHR	CK	5	2	Self	V308, V309, V312, V313, V423
RHR	CK	4	18	Self	V002, V008, V099, V105
RHR	GB/SV	6	2	AO	F122A, B; F146A-D
RHR	GB	1	4	MO	F040
RHR	GB	2	6	MO	F027A, B
RHR	GB	2	12	MO	F015A, B
RHR	GB	6	18	MO	F003A, B; F010A, B; F024A,B
RHR	GT	6	4	MO	F007A-D; F049, 4439
RHR	GT	1	6	MO	F075
RHR	GT	4	12	MO	F017A-D
RHR	GT	4	16	MO	F016A, B; F021A, B
RHR	GT	4	18	MO	F006A, B; F047A, B
RHR	GT	2	20	MO	F008, F009
RHR	GT	4	24	MO	F004A-D
RHR	TCK	4	4	self	V030, V033, V127, V130
RHR	TCK	1	6	Self	V038

System	Valve Type(1)	Quantity	Size, in.	Actuation(1)	Valve Identification Number
RCIC	CK	2	1	Self	V028, V029
RCIC	CK	2	2	Self	V006, V023
RCIC	CK	1	6	Self	V010
RCIC	GB	2	2	MO	F019, F046
RCIC	GB	1	4	MO	F022
RCIC	GT	4	6	MO	F010, F012, F013, F031
RCIC	TCK	2	6	Self	V002, V004
Core spray	CK	4	12	Self	V013, V014, V015, V016
Core spray	GB/SV	2	2	AO	F039A, B
Core spray	GB	2	4	MO	F031A, B
Core spray	GB	2	10	MO	F015A, B
Core spray	GT	4	12	MO	F004A, B, F005A, B
Core spray	GT	4	16	MO	F001 A-D
Core spray	TCK	4	3	Self	V028, V030, V032, V034
Control rod drive hydraulic supply	GB	2	2	MO	3800 A, B
Reactor water cleanup	GT	2	6	MO	F001, F004
Reactor water cleanup	CK	1	4	MO	F039
Standby liquid control	CK	3	1 1/2	Self	V004, V005, V029
Standby liquid control	GCK	2	2	MO	F006A, B
HPCI	CK	2	1	Self	V014, V023
HPCI	CK	1	2	Self	V027
HPCI	CK	1	14	Self	V003
HPCI	GB	5	2	MO	F059, 4803, 4804, 4865, 4866
HPCI	GB	1	4	MO	F012
HPCI	GB	1	10	MO	F008
HPCI	GT	1	8	MO	8278
HPCI	GT	2	14	MO	F006, F007
HPCI	GT	2	16	MO	F004, F042
HPCI	TCK	1	4	Self	V015
HPCI	TCK	2	16	Self	V006, V008
Station service water	BF	2	3	MO	2234, 2236
Station service water	BF	6	6	MO	F073, 2197A-D, 2238
Station service water	BF	2	20	MO	2356 A, B
Station service water	BF	5	24	MO	2355A, B; 2371A, B; 2207
Station service water	BF	4	28	MO	2198 A-D
Station service water	BF	2	30	MO	2203, 2204
Station service water	CK	3	2	Self	V543, V556, V557
Station service water	CK	4	2	Self	V544 thru V547

System	Valve Type(1)	Quantity	Size, in.	Actuation(1)	Valve Identification Number
Station service water	CK	4	28	Self	V359, V361, V363, V365
Fuel pool cooling and cleanup	CK	2	6	Self	V007, V040
Fuel pool cooling and cleanup	CK	1	8	Self	V015
Fuel pool cooling and cleanup	GB	2	2	MO	4647, 4648
Fuel pool cooling and cleanup	GB	2	6	MO	4689A, B
Fuel pool cooling and cleanup	GT/SV	3	8	AO	4676A, B; 4678
RACS	GT	4	4	MO	2553, 2554, 2555, 2556
Torus water cleanup	GT	4	6	MO	4679, 4652, 4680, 4681
SACS	BF	8	8	MO	2314A, B; 7921A, B; 2317A, B; 7922A, B
SACS	BF/SV	4	8	AO	2395A-D
SACS	BF	4	20	MO	2512A, B; 2457A, B
SACS	BF	8	30	MO	2491A, B; 2494A, B; 2496A-D
SACS	BF/SV	4	30	HO	2522A-D
SACS	CK	4	2	Self	V704, V705, V706, V707
SACS	CK	4	20	Self	V010, V013, V016, V019
SACS	CK	2	30	Self	V029, V031
SACS	GB/SV	6	2	AO	2293A, B; 2520A-D
SACS	BL/SV	18	3	AO	2325A-H, 2290A-H, 2292A,B
SACS	BL/SV	6	4	AO	2302A-F
SACS	BL/SV	8	6	AO	2398A-H
SACS	GB	10	2	MO	2452A,B; 2320A, B; 2453A,B; 2321A,B 2446; 2447
Station service water screen wash	BF	4	6	MO	2225A-D
Station service water screen wash	CK	4	6	Self	V003, V010, V016, V023
RCIC turbine steam	CK	1	2	Self	V010
RCIC turbine steam	GB	2	2	MO	F060, F076
RCIC turbine steam	GB/SV	3	2	AO	F004; F025; F026
RCIC turbine steam	GB	1	4	MO	F045
RCIC turbine steam	GT	2	3	MO	F062; F084
RCIC turbine steam	GT	2	4	MO	F007; F008
RCIC turbine steam	GT	1	10	MO	F059
RCIC turbine steam	TCK	1	10	Self	V003
RCIC turbine steam	SV	1	1	SOL	F054
HPCI turbine steam	CK	2	2	Self	V032, V038
HPCI turbine steam	GB/SV	3	2	AO	F026, F028, F029
HPCI turbine steam	GB	1	2	MO	F100, 4922
HPCI turbine steam	GT	2	3	MO	F075, F079
HPCI turbine steam	TCK	1	20	Self	V004

System	Valve Type(1)	Quantity	Size, in.	Actuation(1)	Valve Identification Number
Chilled water	GT	8	8	MO	9531A-1 thru -4; 9531B-1 thru -4
Containment atmosphere control	BF/SV	1	6	AO	4978
Containment atmosphere control	BF/SV	6	24	AO	4958, 4962, 4964, 4980, 5029, 5031
Containment atmosphere control	BF/SV	4	26	AO	4950, 4952, 4956, 4979
Containment atmosphere control	CK	6	1	Self	V054, V055, V081, V093, V138, V139
Containment atmosphere control	GB	25	2	MO	4951, 4955A, B; 4983A, B; 5019A, B; 4984A, B; 4959A, B; 4965A, B; 4966A, B; 5022A, B; 4974; 4963; 5055A, B; 5057A, B.
Containment atmosphere control	GT	4	4	MO	5050A, B; 5052A, B
Containment atmosphere control	GT	4	6	MO	5053A, B, 5054A, B
Containment atmosphere control	CK	8	24	Self	4946 A-H
FRVS	CK	2	1	Self	V032
Liquid R/W	GT	4	3	MO	F003; F004; F019; F020
Liquid R/W	GT	2	4	MO	5262, 5275
Solid radwaste	GT	1	3	MO	5551
Diesel fuel oil transfer & storage	CK	16	2	Self	V001 thru V008; V095 thru V098; V121 thru V124
Primary containment instrument gas	CK	2	1	Self	V005, V006
Primary containment	CK	4	2	Self	V023, V024, V217, V219
instrument gas	GB	13	2	MO	5147; 5148, 5160A, B; 5126A, B; 5152A, B; 5162; 5124A, B; 5172A, B;
Primary containment	GT	7	2	AO	5156A,B; 5154; 5155;
instrument gas	CK	2	1/2	Self	V223, V224
Neutron monitoring	GB/SV	1	2	AO	5161
Neutron monitoring	CK	1	.375	Self	V006
Steam leak detection	GB	4	2	MO	5018, 4953, 4957, 4981
Reactor recirculation	GB	2	.75	Sol	SV-4310, 4311

System	Valve Type(1)	Quantity	Size, in.	Actuation(1)	Valve Identification Number
RHR	GB	1	1	Sol	SV-F074
RHR	GB	4	.75	Sol	SV-F079A, B; F080A, B
RHR	BF	2	18	MO	F048A, B
Station service water	GB	7	1	Sol	SV-2235, 2237, 2239, 2247A-D,
Station service water	GB	4	2	Sol	SV-2367A-D
Safety and turbine auxiliaries cooling	BF/SV	2	20	AO	TV-2517A, B
Safety and turbine auxiliaries cooling	GB	4	1	Sol	2281-1, -2; 2288-1, -2
RCIC turbine steam	GB/SV	1	1	AO	LV-F005
RCIC turbine steam	GB	1	2	Self	PCV-F015
HPCI turbine steam	GT	3	10	MO	F001, F002, F003
HPCI turbine steam	GT	1	20	MO	F071
HPCI turbine steam	GB/SV	1	1	AO	LV-F025
HPCI turbine steam	SV	1	1	Sol	F054
HPCI turbine steam	GB	1	2	Self	PCV-F035
Primary Cont-Instrument Gas	GT	2	2	Sol	5157A,B
Aux. Building Chilled Water	SV	4	4	E/H	TV-9637A,B; TV-9667A,B
Aux. Building Chilled Water	SV	2	6	E/H	TV-9634A,B
Aux. Building Chilled Water	GT	2	1 1/2	E/H	TV-9768A,B
Aux. Building HVAC	SV	8	1	Sol	9588AA, AB, BA, BB; 9589A,B; 9598A,B
FRVS	GT	1	8	MO	9451
FRVS	SV	10	1	Sol	9370A,B; 9372A,C; 9395A,B; 9414A,B; 9450A,B
Standby Diesel Engines	GB	20	1	Sol	6615A-D; 7534A-D; 7535A-D; 7536A-D; 7537A-D

TABLE 3.9-18 (Cont)

(1) Definition of symbols used:

BF - butterfly valve
CK - check valve
GB - globe valve
GCK - globe (stop check) valve
GT - gate valve
TCK - testable check valve
AO - air operator
HO - hydraulic operator
MO - motor operator
SOL - solenoid operator.
SV - solenoid valve - 3-way
E/H - electro-hydraulic
BL - ball valve

TABLE 3.9-19

EXTRAPOLATION OF TESTED NON-NSSS VALVE TO OTHER SIZES

Size of Qualified Valve V	0.5	1	1.5	2	3	4	6	8	10	12	14	16	18	20	22	24	26	28	30	36
0.5	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	

Qualification Extends To: (columns 0.5 through 36)

Size of Qualified Valve — Qualification Extends To:

V	0.5	1	1.5	2	3	4	6	8	10	12	14	16	18	20	22	24	26	28	30	36	
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 3.9-20

DESIGN LOADING COMBINATIONS FOR SUPPORTS
FOR ASME B&PV CODE CLASS 1, 2 AND 3 NON-NSSS COMPONENTS

Condition	Design Loading Combinations (1)(2)	Allowable Stress
Hydrostatic test	(a) HTDW	0.8 S_y
Normal and upset	(a) $DW+TH+(OBE^2+RVC^2)^{1/2}$	ASME Section III, Appendix XVII
	(b) DW+TH+OBE+RVO	
	(c) DW+TH+FV	
Emergency	(a) $DW+TH+(OBE^2+FV^2)^{1/2}$	ASME Section III, Appendix XVII
Faulted	(a) DW+TH+SSE+RVO	ASME Section III, Appendix F(4)
	(b) $DW+TH+(SSE^2+RVC^2)^{1/2}$	
	(c) $DW+TH+(SSE^2+DBA^2)^{1/2}$	

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(1) Loads due to OBE, SSE, and DBA include both inertia portion and anchor movement portion when spectra method is used. The loads from the inertia portion and anchor movement portion are combined by the SRSS method.

(2) For torus attached piping, the loading combinations used in evaluating the pipe support loads are those given in the Plant

Unique Analysis Application Guide (PUAAG) (NEDO-24583-1, October 1979 (Table 5-2)).

(3) Definition of symbols used:

HTDW - piping dead weight due to hydrostatic test

TH - reaction at the support due to thermal expansion of the pipe

DW - dead weight

OBE - operating basis earthquake[1]

RVC - transient response of the piping system associated with relief valve opening in a closed system

RVO - transient response of the piping system associated with relief valve opening in an open system

FV - transient response of the piping system associated with fast valve closure time less than 5 seconds

SSE - safe shutdown earthquake[1]

DBA - design basis accident[1]

(4) For essential safety-related (ESR) systems allowable stress not to exceed S_y.

TABLE 3.9-21

DEFORMATION LIMIT
(FOR ESF REACTOR INTERNALS ONLY)

Either One of (Not Both)	General Limit
1. $\left[\dfrac{\text{Permissible deformation, DP}}{\text{Analyzed deformation causing loss of function, DL}}\right]$	$\dfrac{0.9}{SF_{min}}$
2. $\left[\dfrac{\text{Permissible deformation, DP}}{\text{Experimental deformation causing loss of function, DE}}\right]^{(1)}$	$\dfrac{1.0}{SF_{min}}$

where:

DP - permissible deformation under stated conditions of service levels A, B, C, or D (normal, upset, emergency, or faulted)

DL - analyzed deformation that could cause a system loss of function(2)

DE - experimentally determined deformation that could cause a system loss of function

SF_{min} - minimum safety factor (see Section 3.9.5.3.5)

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(1) Equation b. is not used unless supporting data are provided to the NRC by General Electric.

(2) "Loss of function" can only be defined quite generally, until attention is focused on the component of interest. In cases of interest, where deformation limits can affect the function of equipment and components, they are specifically delineated.

From a practical viewpoint, it is convenient to interchange some deformation condition at which function is assured with the loss of function condition if the required safety margins from the functioning conditions can be achieved. Therefore, it is often unnecessary to determine the actual loss of function condition, because this interchange procedure produces conservative and safe designs. Examples where deformation limits apply are: control rod drive alignment and clearances for proper insertion; core support deformation causing fuel disarrangement, or excess leakage of any component.

TABLE 3.9-22

PRIMARY STRESS LIMIT

(FOR ESF REACTOR INTERNALS ONLY)

Any One Of (No More Than One Required)	General Limit
1. $\left[\dfrac{\text{Elastic evaluated primary stresses, PE}}{\text{Permissible primary stresses, PN}}\right]$	$\le \dfrac{2.25}{SF_{min}}$
2. $\left[\dfrac{\text{Permissible load, LP}}{\text{Largest lower bound limit load, CL}}\right]$	$\le \dfrac{1.5}{SF_{min}}$
3. $\left[\dfrac{\text{Elastic evaluated primary stresses, PE}}{\text{Conventional ultimate strength at temperature, US}}\right]$	$\le \dfrac{0.75}{SF_{min}}$
4. $\left[\dfrac{\text{Elastic plastic evaluated nominal primary stress, EP}}{\text{Conventional ultimate strength at temperature, US}}\right]$	$\le \dfrac{0.9}{SF_{min}}$
5. $\left[\dfrac{\text{Permissible load, LP}}{\text{Plastic instability load, PL}}\right]^{(1)}$	$\le \dfrac{0.9}{SF_{min}}$
6. $\left[\dfrac{\text{Permissible load, LP}}{\text{Ultimate load from fracture analysis, UF}}\right]^{(1)}$	$\le \dfrac{0.9}{SF_{min}}$
7. $\left[\dfrac{\text{Permissible load, LP}}{\text{Ultimate load or loss of function load from test, LE}}\right]^{(1)}$	$\le \dfrac{1.0}{SF_{min}}$

where:

PE - primary stresses evaluated on an elastic basis. The effective membrane stresses are averaged through the load carrying section of interest. The simplest average bending, shear, or torsion stress distribution that support the external loading are added to the membrane stresses at the section of interest.

PN - permissible primary stress levels under service level A or B (normal or upset) conditions under ASME B&PV Code, Section III

LP - permissible load under stated conditions of service levels A, B, C, or D (normal, upset, emergency, or faulted)

CL - lower bound limit load with yield point equal to $1.5S_m$, where S_m is the tabulated value of allowable stress at temperature as contained in ASME B&PV Code, Section III. The "lower bound limit load" is defined as that produced from the analysis of an ideally plastic (nonstrain hardening) material, where deformations increase with no further increase in applied load. The lower bound load is one in which the material everywhere satisfies equilibrium, and nowhere exceeds the defined material yield strength, using either a shear theory or a strain energy of distortion theory to relate multiaxial yielding to the uniaxial case.

US - conventional ultimate strength at temperature or loading, whichever is more limiting, causing a system malfunction

EP - elastic plastic evaluated nominal primary stress. Strain hardening of the material may be used for the actual monotonic stress strain curve at the temperature of loading. An approximation to the actual stress strain curve that everywhere

has a lower stress for the same strain as the actual monotonic curve may also be used. Either the shear or strain energy of distortion flow rule is used.

PL = plastic instability load, defined here as the load at which any load bearing section begins to diminish its cross-sectional area at a faster rate than the strain hardening can accommodate the loss in area. This type of analysis requires a true stress true strain curve, or a close approximation based on monotonic loading at the temperature of loading.

UF = ultimate load from fracture analyses. For components that involve sharp discontinuities (local theoretical stress concentration <3), the use of a "fracture mechanics" analysis, where applicable, using measurements of plane strain fracture toughness, may be applied to compute fracture loads. Correction for finite plastic zones and thickness effects, as well as gross yielding, may be necessary. The methods of linear elastic stress analysis may be used in the fracture analysis, where its use is clearly conservative or supported by experimental evidence. Examples of where fracture mechanics may be applied are: for fillet welds; for end of fatigue life crack propagation.

LE = ultimate load or loss of function load as determined from experiment. In using this method, account is taken of the dimensional tolerances that may exist between the actual part and the tested part, or parts, as well as differences that may exist in the ultimate tensile strength of the actual part and the tested parts. In each of these areas, the experimentally determined load uses adjusted values to account for material property and dimensional variations, each of which has no greater probability than 0.1 of being exceeded in the actual part.

SF_{min} = minimum safety factor (see Section 3.9.5.3.5)

---

(1) Equations 5., 6., and 7. are not used unless supporting data are provided to the NRC by General Electric.

TABLE 3.9-23

BUCKLING STABILITY LIMIT
(FOR ESF REACTOR INTERNALS ONLY)

Any One of (No More Than One Required)	General Limit
1. $\left[\dfrac{\text{Permissible load, LP}}{\text{Service level A (normal) permissible load, PN}}\right]$	$\le\dfrac{2.25}{SF_{min}}$
2. $\left[\dfrac{\text{Permissible load, LP}}{\text{Stability analysis load, SL}}\right]$	$\le\dfrac{0.9}{SF_{min}}$
1	
3. $\left[\dfrac{\text{Permissible load, LP}}{\text{Ultimate buckling collapse load from test, SE}}\right]$	$\le\dfrac{1.0}{SF_{min}}$

where:

LP = permissible load under stated conditions of service levels A, B, C, or D (normal, upset, emergency, or faulted)

PN = applicable service level A (normal) permissible load

SL = stability analysis load. The ideal buckling analysis is often sensitive to otherwise minor deviations from ideal geometry and boundary conditions. These effects are accounted for in the analysis of the buckling stability load. Examples of this are: ovality in externally pressurized shells; eccentricity in column members.

SE = ultimate buckling collapse load as determined from experiment. In using this method, account is taken of the dimensional tolerances that may exist between the actual part and the tested part. In each of these areas, the experimentally

determined load is adjusted to account for material property and dimensional variations, each of which has no greater probability than 0.1 of being exceeded in the actual part.

SF_{min} = minimum safety factor (see Section 3.9.5.3.5)

---

(1) Equation 3. is not used unless supporting data are provided to the NRC by General Electric.

TABLE 3.9-24

FATIGUE LIMIT
(FOR ESF REACTOR INTERNALS ONLY)

Summation of fatigue damage usage with design and operation loads following Miner hypotheses[(1)]

Cumulative Damage in Fatigue	Limit for Service Level A and B (Normal and Upset) Design Conditions
Design fatigue cycle usage from analysis using the method of ASME Code	≤1.0

(1) M.A. Miner, "Cumulative Damage in Fatigue," Journal of Applied Mechanics, Vol. 12, ASME, Vol. 67, pp A159-A164, September 1945.

TABLE 3.9-25

TYPICAL NSSS SQRT EQUIPMENT QUALIFICATION METHODOLOGY

MPL Number	Equipment	Qualification Methodology
E11-B001	RHR heat exchanger	Response spectrum dynamic analysis
E11-C002	RHR pump motor	Response spectrum dynamic analysis
E21-C001	Reactor core spray pump	Response spectrum dynamic analysis
C41-C001	Standby liquid control pump	Static analysis/Dynamic test - IEEE 344-1975
C41-A001	SLC tank	Static analysis to 1.75 g
E41-C002	HPCI turbine	Dynamic test
E51-C002	RCIC turbine	Dynamic test
145C3103	Thermometer	Static analysis
145C3224	Temperature element	Multi frequency, multi-axis test
159C4361	Level switch	Single axis, single frequency test
163C1303	Limit switch	Single frequency - multi axis test

TABLE 3.9-26

EXAMPLES OF NSSS PVORT EQUIPMENT QUALIFICATION METHODOLOGY

MPL Number	Equipment	Qualification Methodology
E91-C001	HPCI pump	Static analysis for dynamic analysis
C51-J004	Guide tube valve	Single frequency - static analysis
B21-F013	SRV	Static analysis, comparison to other
B21-F022/ F028	MSIV	Single axis/multiaxis test
C11-F009/ F182	CRD solenoid valve	Single axis/multiaxis test

TABLE 3.9-27

EXAMPLES OF NON-NSSS PVORT EQUIPMENT QUALIFICATION METHODOLOGY

Equipment Number	Description	Qualification Method	Standards[1]
1A, 1B-P210 1C, 1D-P210	SACS Pump	Static Analysis	A, F, I
1A, 1B-P211	Fuel Pool Cooling Pump	Static Analysis	A, F, I
1A, 1B-P402 1C, 1D-P402	Motor Driven Diesel Fuel Oil Pump	Static Analysis	F
1A, 1B-P228 1C, 1D-P228	ECCS Jockey Pump	Static Analysis	A, F, I
1A, 1B-P414	Chilled Water Circulation Pump	Dynamic Analysis	A, F, I
1A, 1B-P408 1C, 1D-P408	Engine Driven Jacket Water Pump	Dynamic Analysis	F
1A, 1B-P507 1C, 1D-P507	Spray Water Booster Pump	Static Analysis	A, F, I
1-BC-HV-F047A	18"-GBB-GT-MO	Static Analysis & Dynamic Testing & Pull Test	A, E, F, I
1-BE-HV-F031A	4"-GBB-GB-MO	Static Analysis & Dynamic Testing & Pull Test	A, E, F, I

Equipment Number	Description	Qualification Method	Standards[1]
1-FC-HV-F059	10"-HBB-GT-MO	Static Analysis & Dynamic Testing & Pull Test	A, E, F, I
1-EE-HV-4655	6"-HBC-GT-AO	Static Analysis & Dynamic Testing & Pull Test	A, E, F, I
1-AB-HV-F019	3"-DBA-GT-MO	Static Analysis & Dynamic Testing & Pull Test	A, E, F, I
1-AE-HV-F074A	24"-DLA-CK-AO	Static Analysis & Dynamic Testing & Pull Test	A, E, F, I
1-BD-HV-F046	2"-CBA-GB-MO	Static Analysis & Dynamic Testing & Pull Test	A, E, F, I
1-EG-HV-2522A	30"-HBC-BF-HYDRAU	Static Analysis & Dynamic Testing & Pull Test	A, E, F, I
1-EG-HV-2395A	8"-HBC-BF-AO	Static Analysis & Dynamic Testing & Pull Test	A, E, F, I
1-EA-HV-2356A	20"-HEC-BF-MO	Static Analysis & Dynamic Testing & Pull Test	A, E, F, I

---

(1) Standards

A - IEEE-344-1975

F - ASME B&PV Code, Section III

I - NRC Regulatory Guide 1.100, Rev. 1

E - IEEE 382, 1972

TABLE 3.9-28

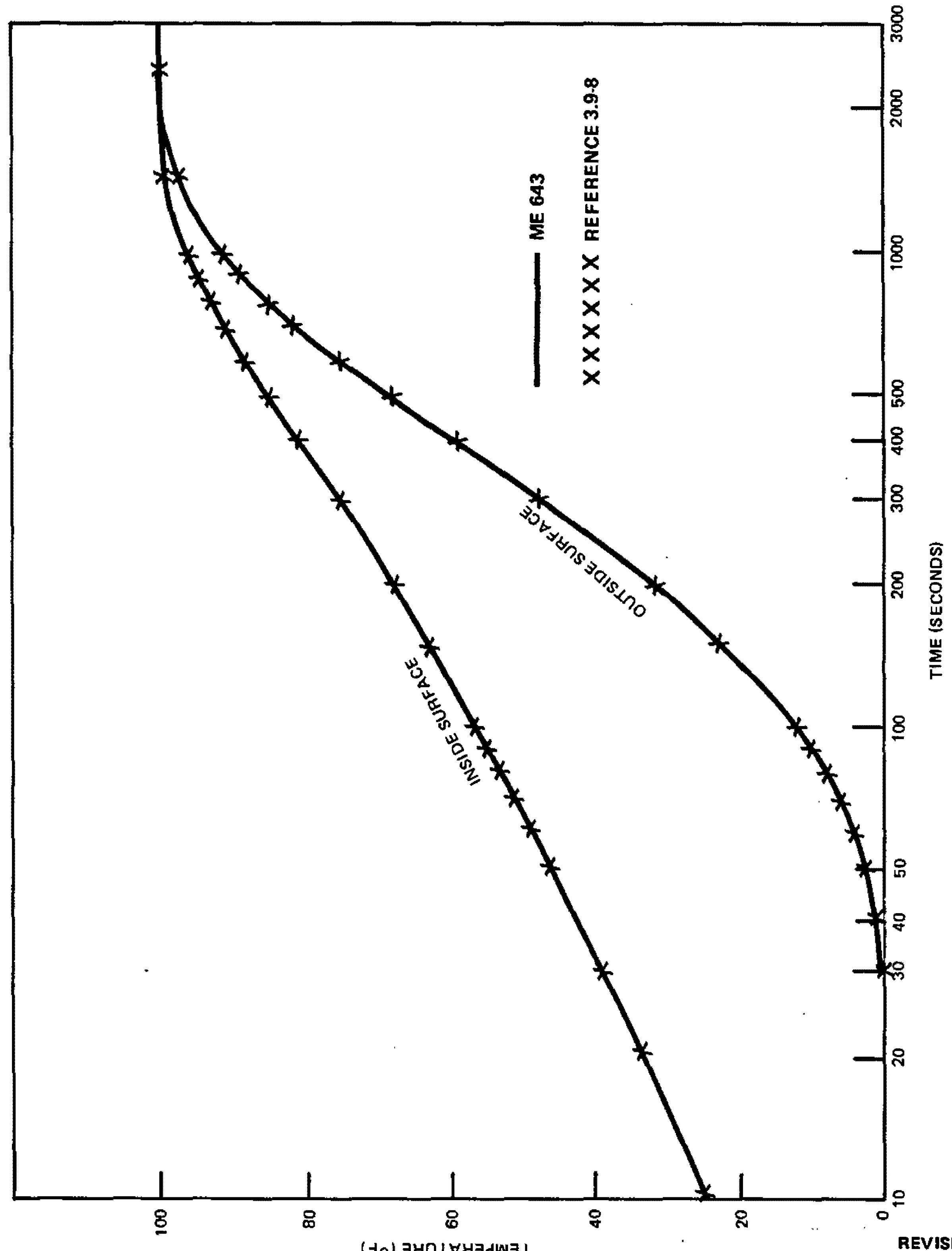
NON-NSSS VALVES SUBJECTED TO HYDRODYNAMIC LOADS

1-BE-HV-F001A to D
1-FC-HV-F059
1-BJ-HV-F042
1-FD-HV-F071
1-EE-HV-4680
1-EE-HV-4681
1-BD-HV-F031
1-BC-HV-F004A to D
1-EE-HV-4652
1-EE-HV-4679
1-GS-HV-4958
1-BE-HV-F015A and B
1-BC-HV-F024A and B
1-BC-HV-4421
1-BC-HV-4420A and B
1-BE-HV-F031A
1-BJ-HV-F012
1-FD-HV-F079
1-FD-HV-F075
1-BC-HV-F007C
1-AB-PSV-F037A to H
1-AB-PSV-F037J to M
1-AB-PSV-F037P, Q, R
1-AB-PSV-4500A to H
1-AB-PSV-4500J to M
1-AB-PSV-4500P, Q, R
1-BD-SV-F019
1-GS-PSV-4946A to H
1-BJ-HV-4865
1-BJ-HV-4866
1-BJ-HV-4804
1-FC-HV-F060

TABLE 3.9-29

SRV WATER CLEARING REACTION LOADS ON RAMSHEAD SUPPORT

Reaction Loads	Value (kips)
Horizontal load perpendicular to the plane of the ring girder	10.4
Horizontal load parallel to the plane of the ring girder	18.59
Vertical load	83.12



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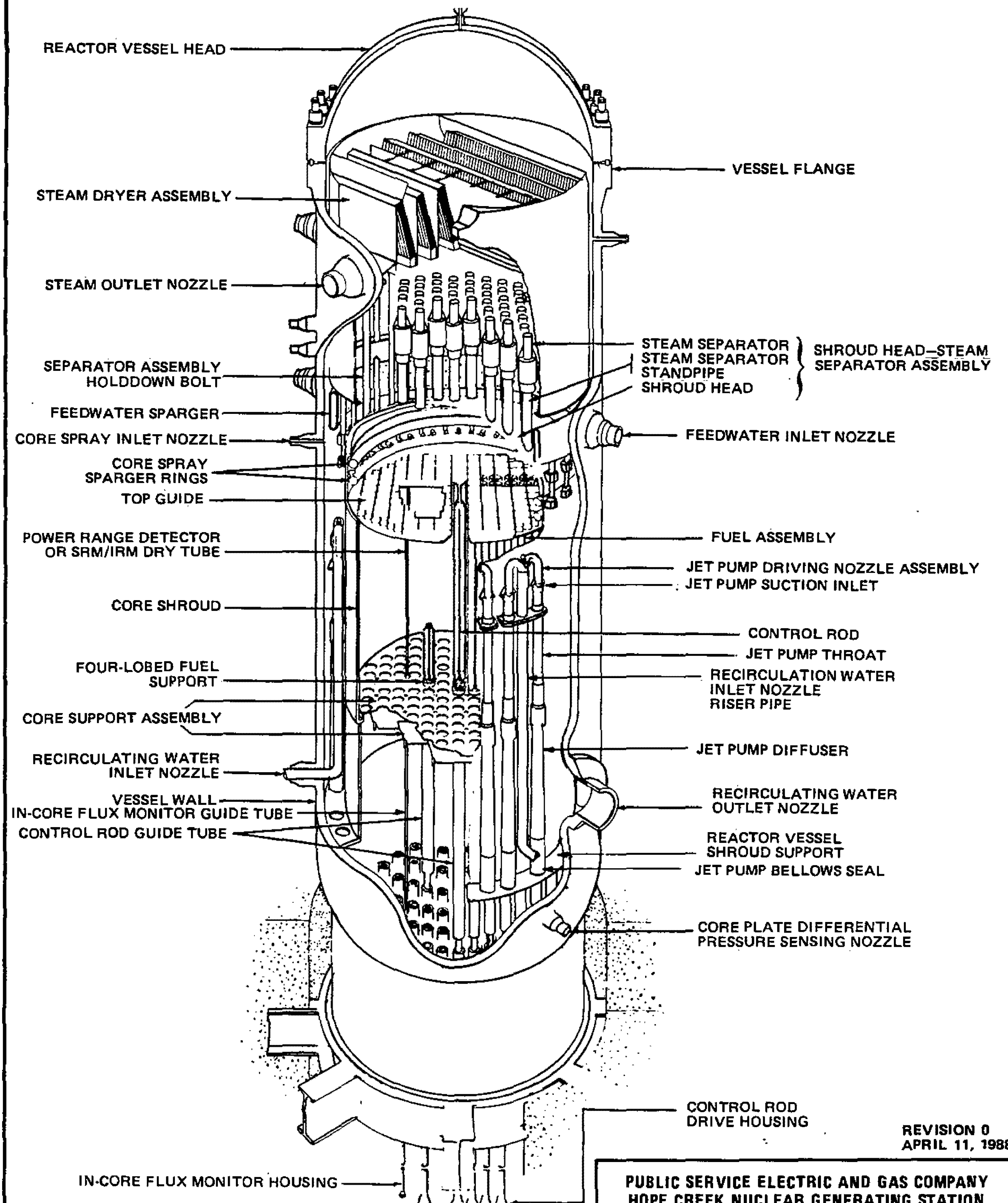
HOPE CREEK NUCLEAR GENERATING STATION

TRANSIENT TEMPERATURE RESPONSE

UPDATED FSAR

FIGURE 3.9-1



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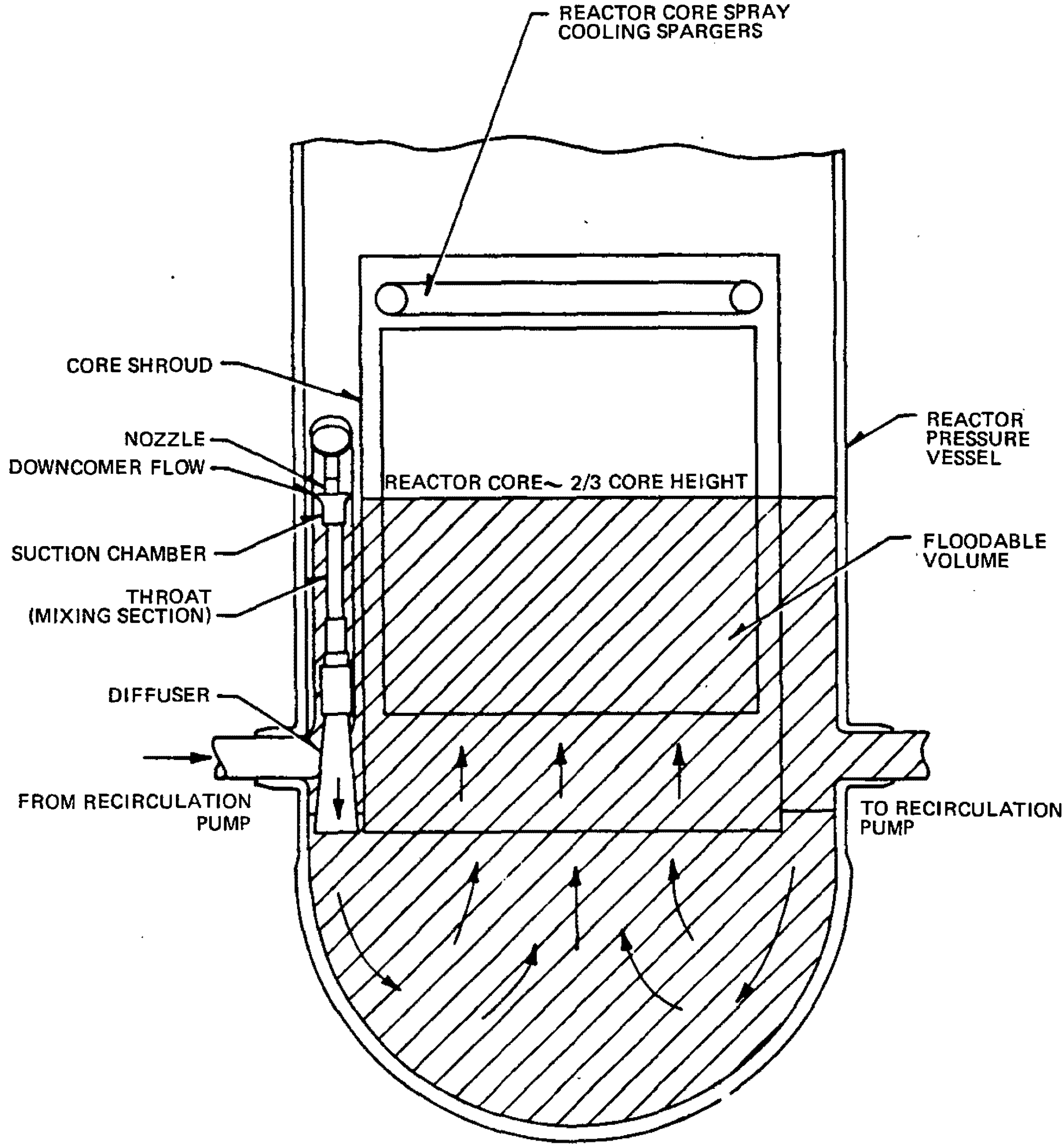
REACTOR VESSEL CUTAWAY

UPDATED FSAR

FIGURE 3.9-2



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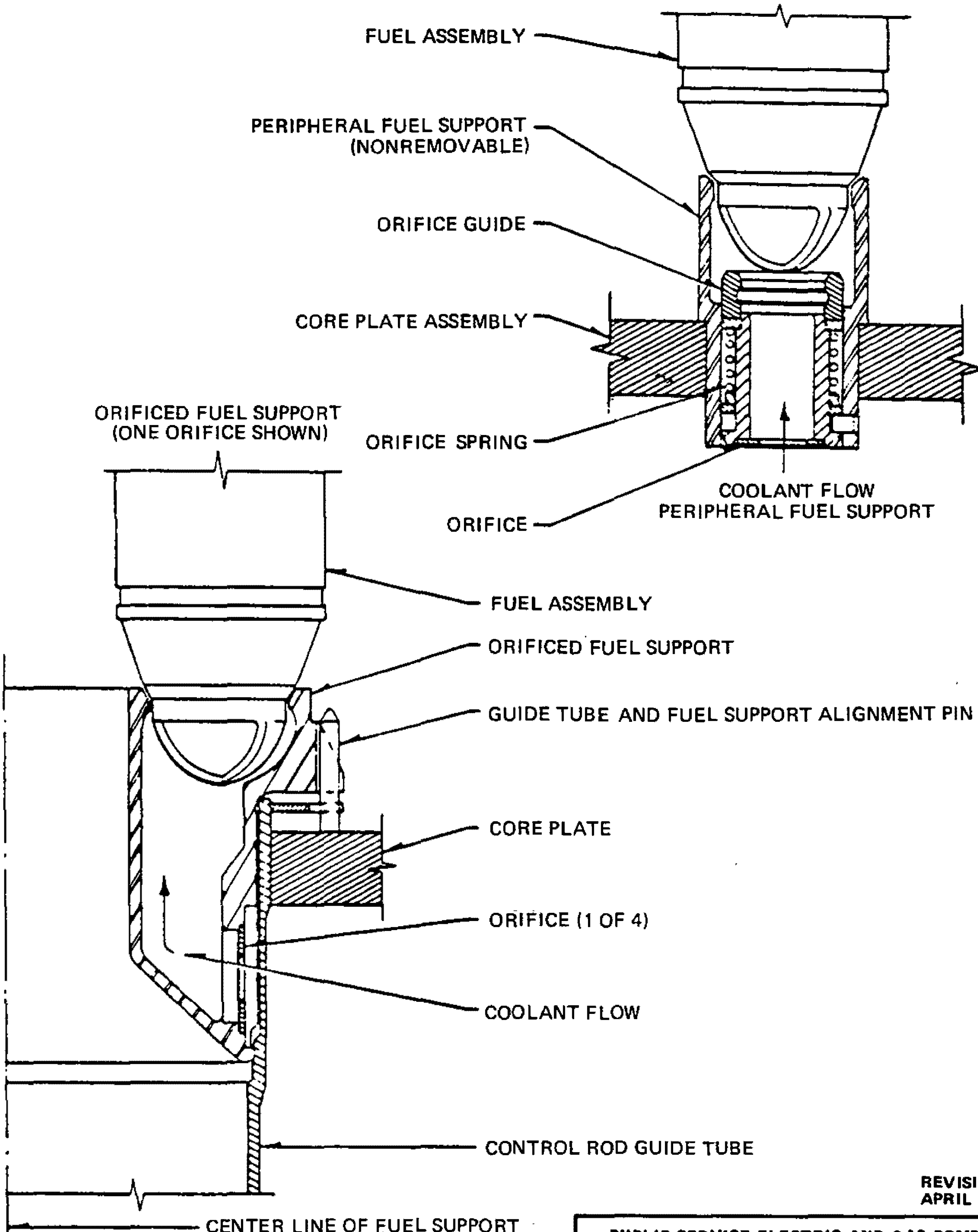
HOPE CREEK NUCLEAR GENERATING STATION

REACTOR INTERNALS
FLOW PATHS

UPDATED FSAR

FIGURE 3.9-3



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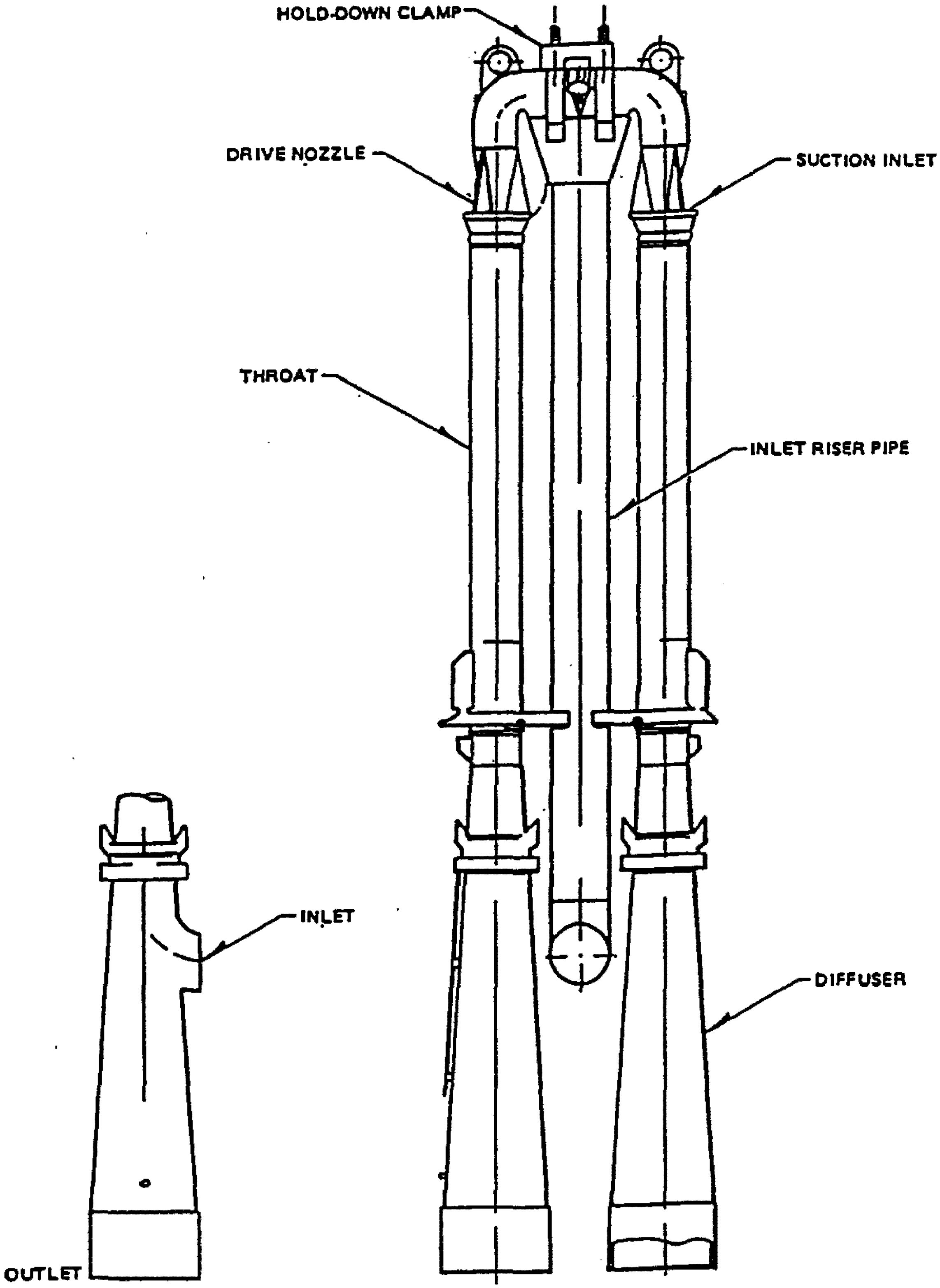
HOPE CREEK NUCLEAR GENERATING STATION

FUEL SUPPORTS

UPDATED FSAR

FIGURE 3.9-4



Note: Jet Pumps 8, 9, and 15 have clamps installed around the diffuser to support the instrument sensing line lower bracket.

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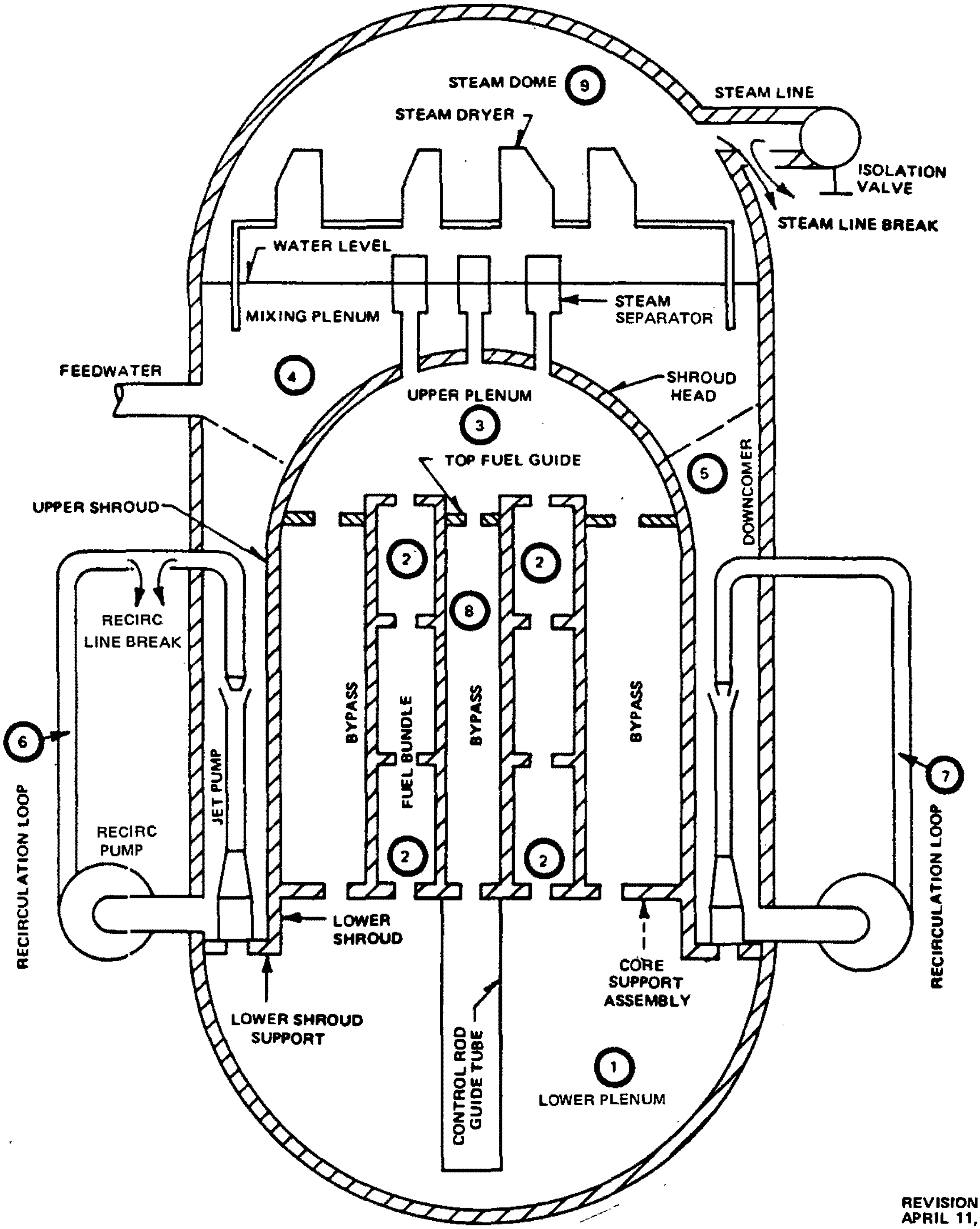
JET PUMP

Updated FSAR
Revision 9, June 13, 1998

Sheet 1 of 1
Figure 3.9-5



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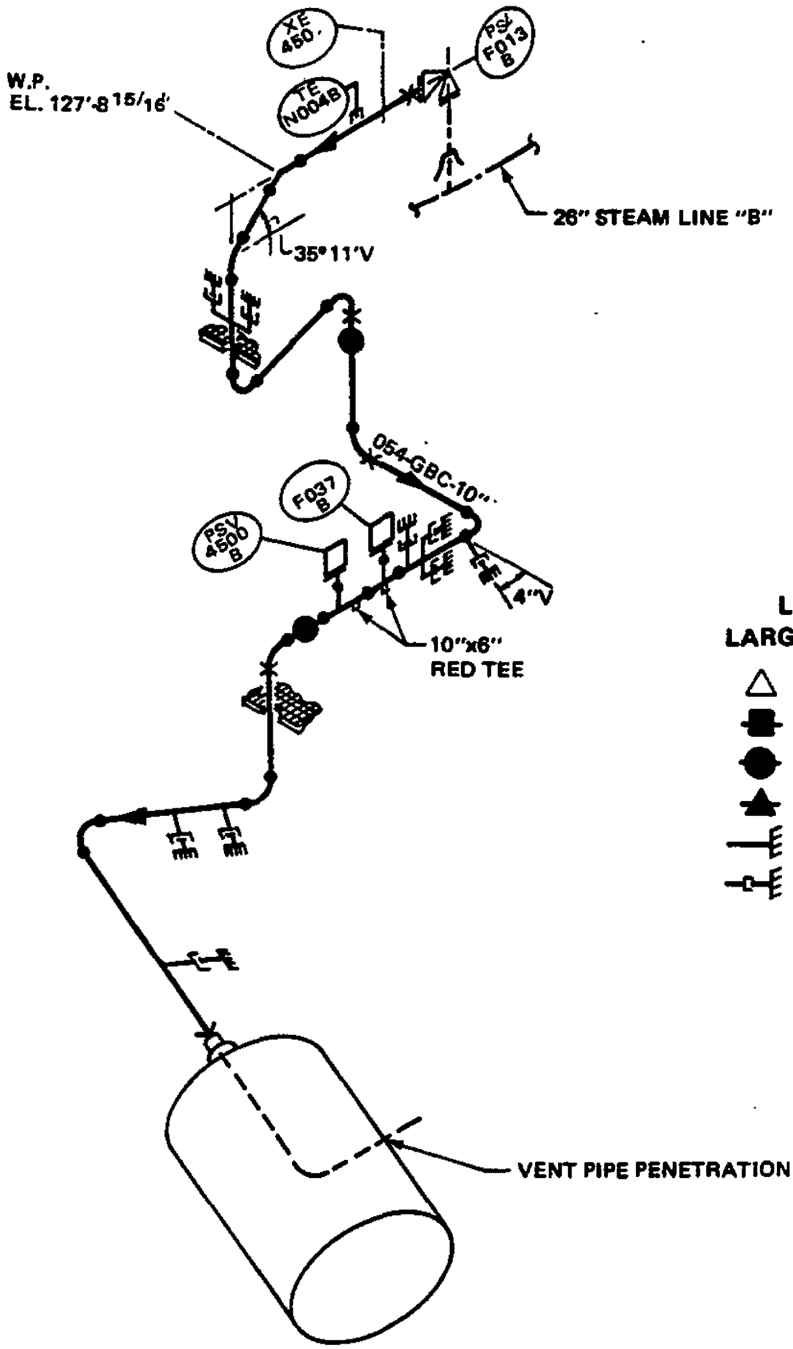
HOPE CREEK NUCLEAR GENERATING STATION

PRESSURE NODES USED FOR DEPRESSURIZATION ANALYSIS

UPDATED FSAR

FIGURE 3.9-6



LEGEND FOR LARGE PIPE SUPPORTS

- HANGER NO.
- RIGID HANGER
- SPRING HANGER
- ANCHOR
- RESTRAINT
- SNUBBER

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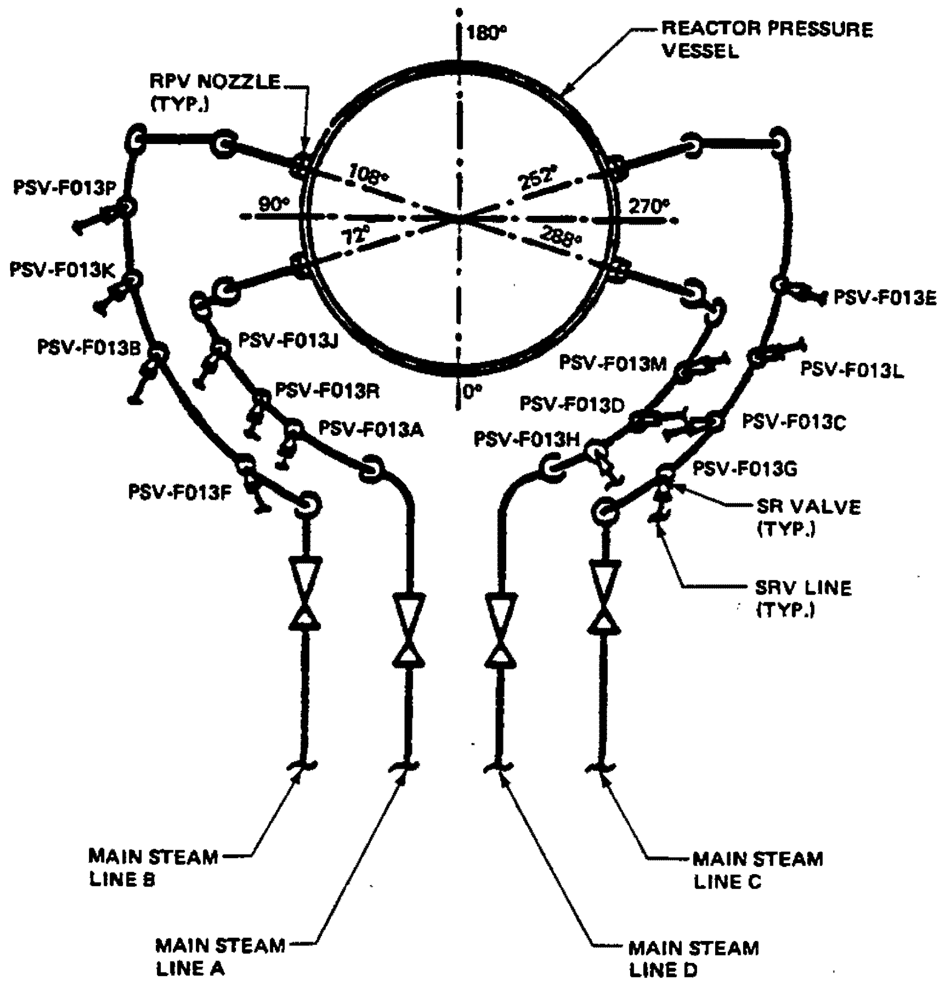
HOPE CREEK NUCLEAR GENERATING STATION

REPRESENTATIVE SRV PIPING
SYSTEM IN THE DRYWELL

UPDATED FSAR

FIGURE 3.9-7



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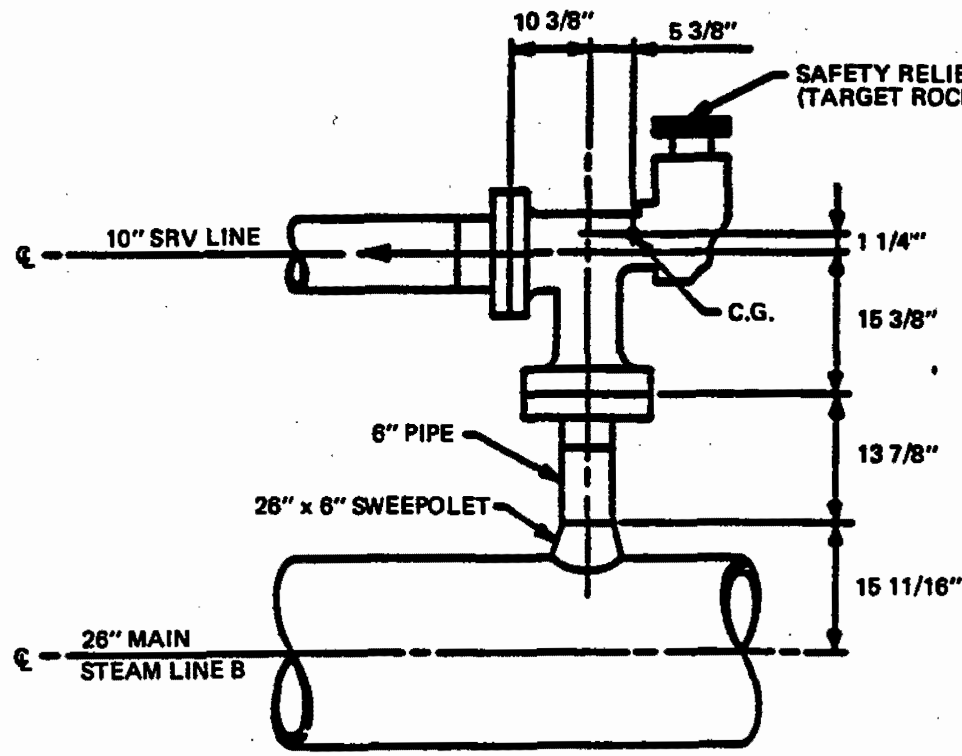
HOPE CREEK NUCLEAR GENERATING STATION

SRV DISCHARGE LINE AND
MAIN STEAM LINE SCHEMATIC

UPDATED FSAR

FIGURE 3.9-8



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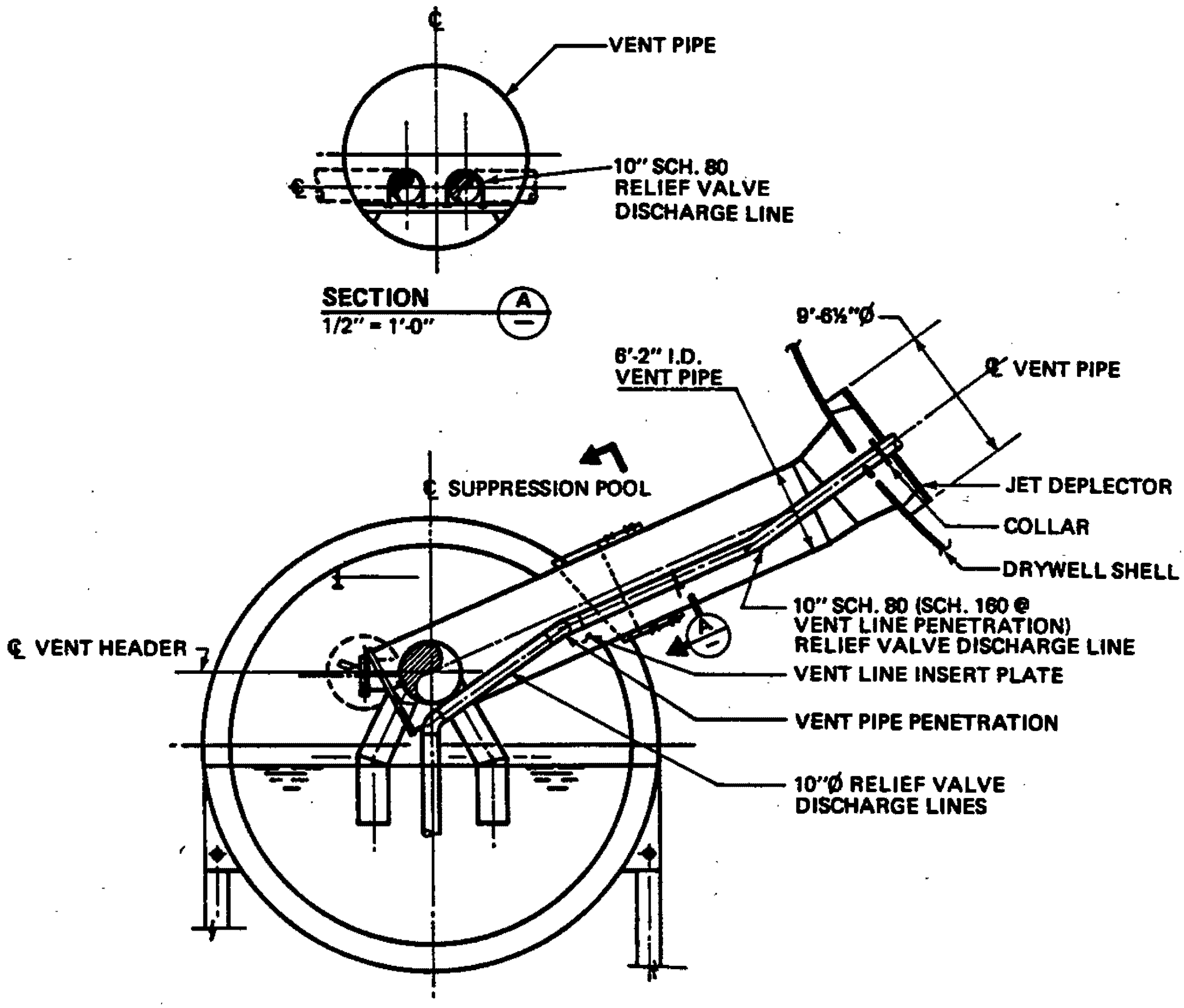
HOPE CREEK NUCLEAR GENERATING STATION

SRV DISCHARGE LINE
CONNECTION TO THE MAIN STEAM
SAFETY RELIEF VALVE

UPDATED FSAR

FIGURE 3.9-9



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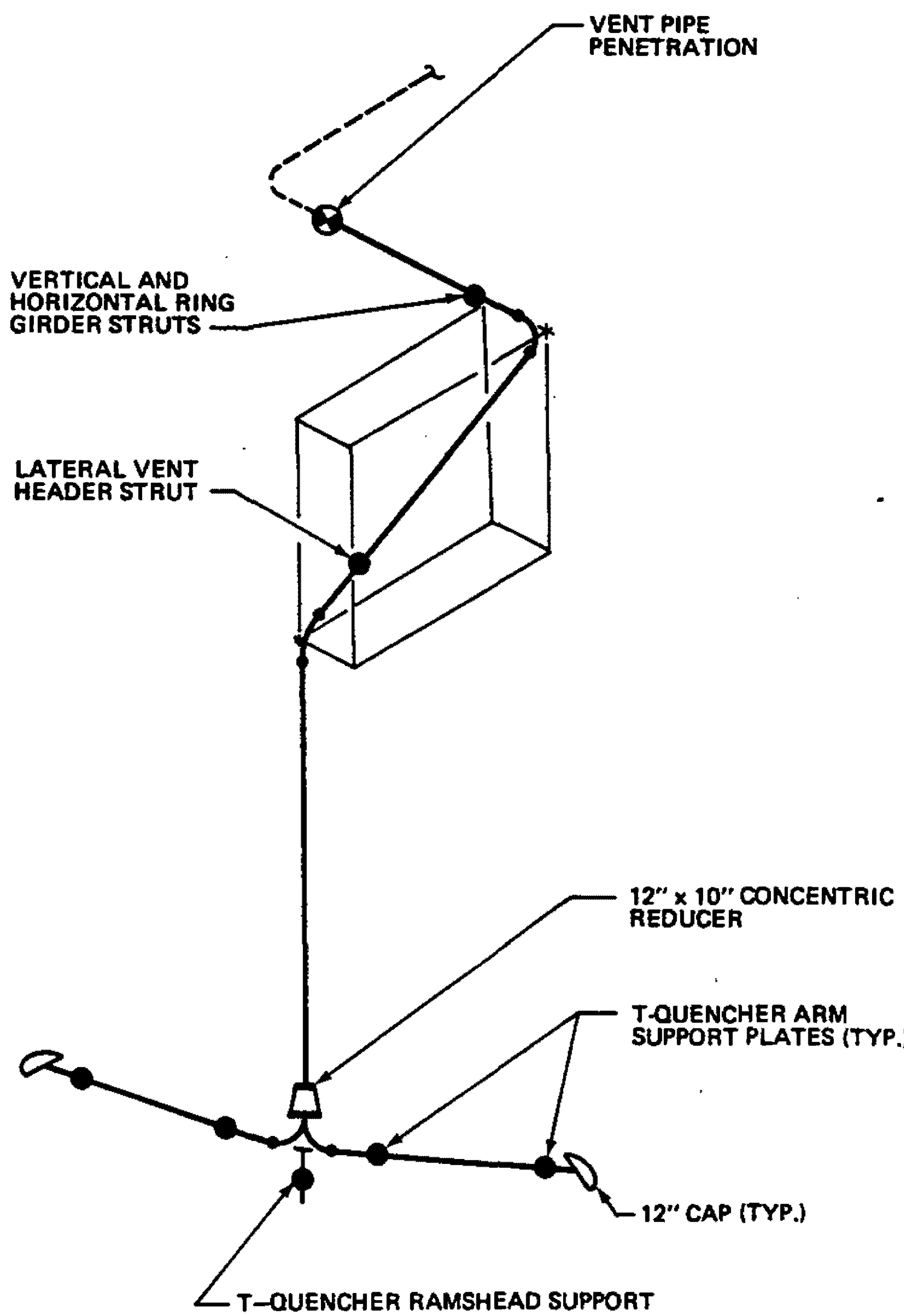
HOPE CREEK NUCLEAR GENERATING STATION

TYPICAL SRV PIPING SYSTEM
IN THE VENT PIPE

UPDATED FSAR

FIGURE 3.9-10



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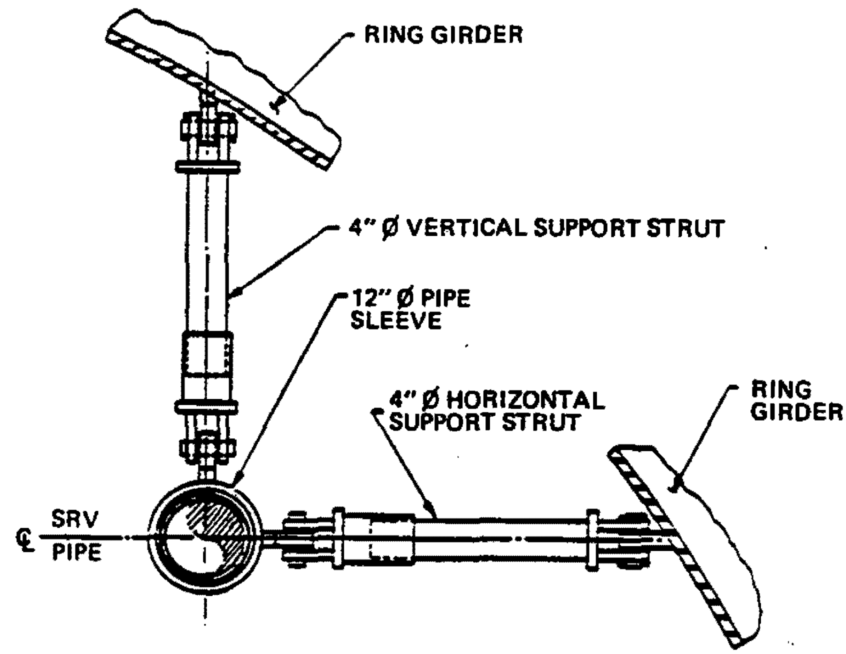
HOPE CREEK NUCLEAR GENERATING STATION

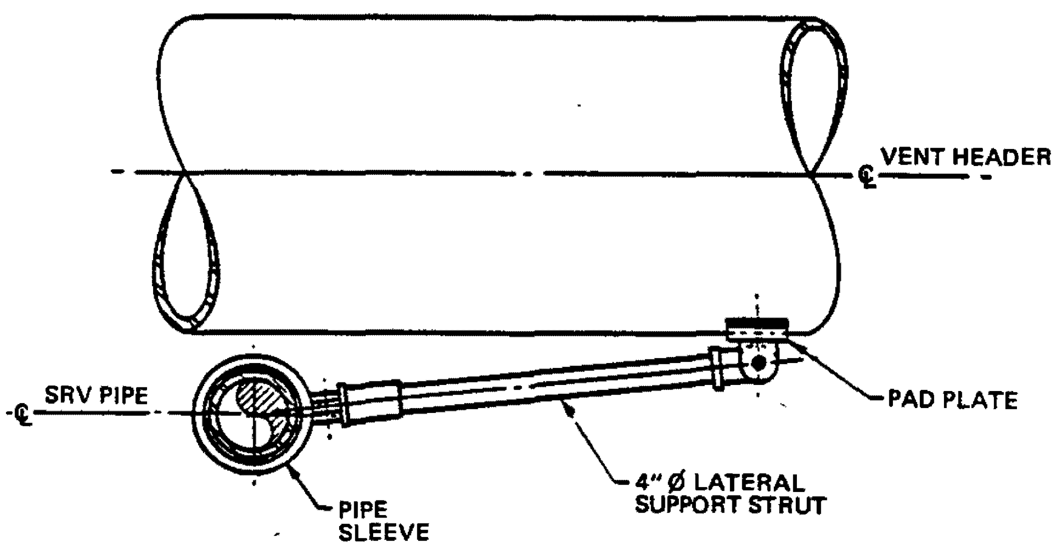
TYPICAL WETWELL SRV PIPING SYSTEM AND QUENCHER – ISOMETRIC VIEW

UPDATED FSAR

FIGURE 3.9-11



SUPPORT STRUTS AT RING GIRDER



SUPPORT STRUT AT VENT HEADER

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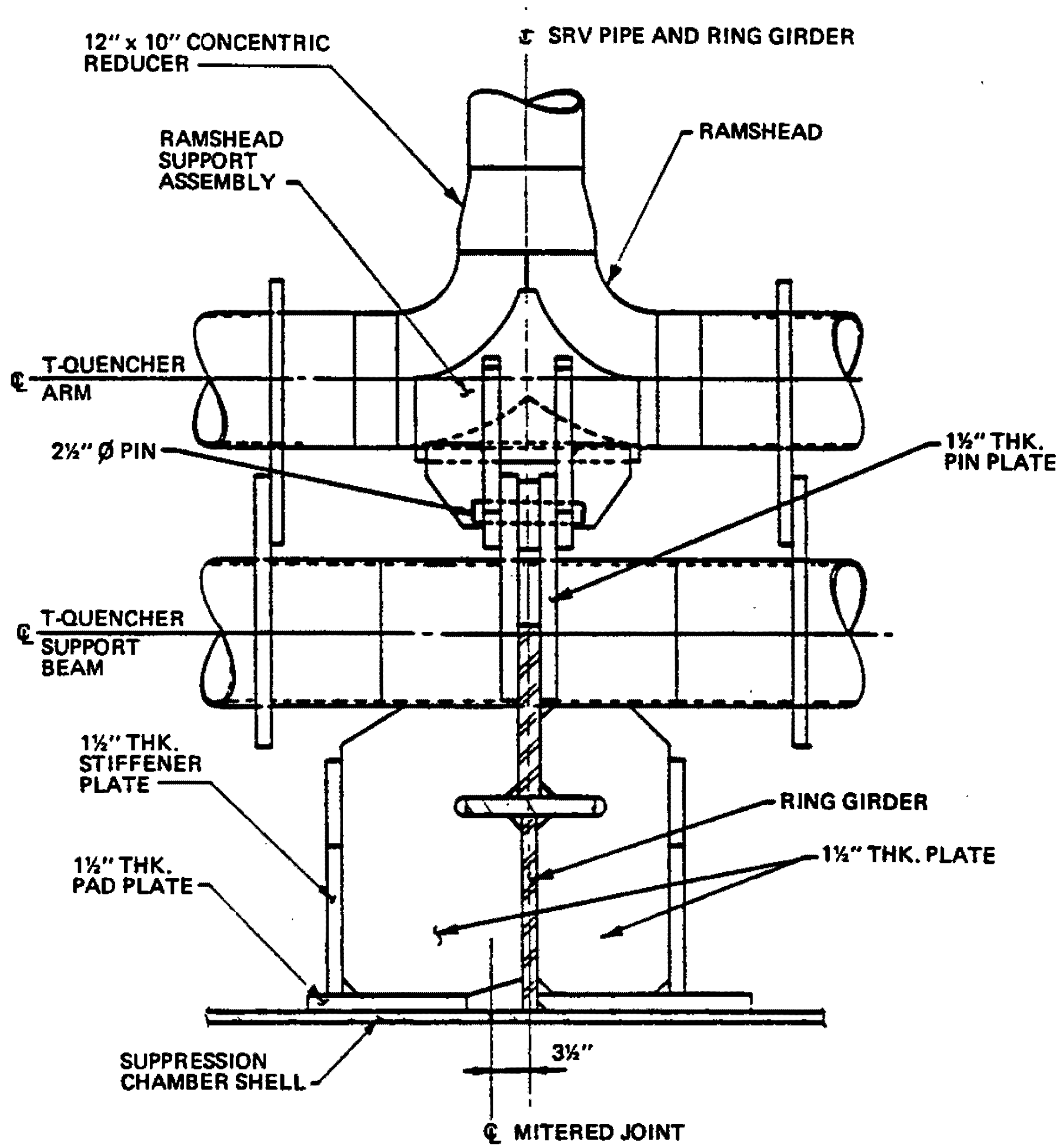
PUBLIC SERVICE ELECTRIC AND GAS COMPANY HOPE CREEK NUCLEAR GENERATING STATION
WETWELL SRV PIPING SUPPORT DETAILS
UPDATED FSAR FIGURE 3.9-12



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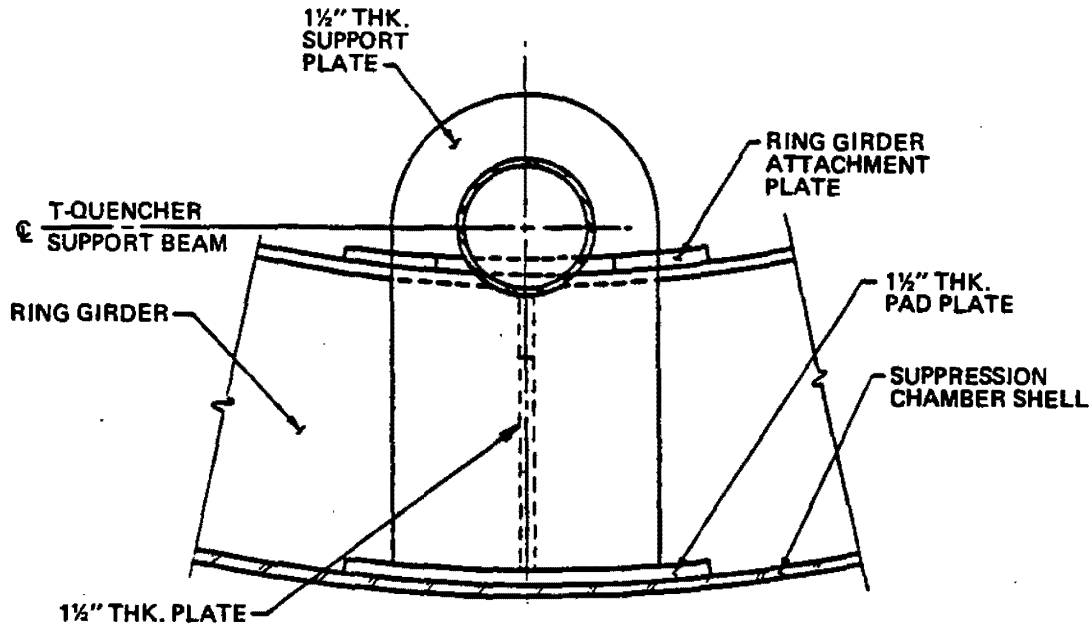
HOPE CREEK NUCLEAR GENERATING STATION

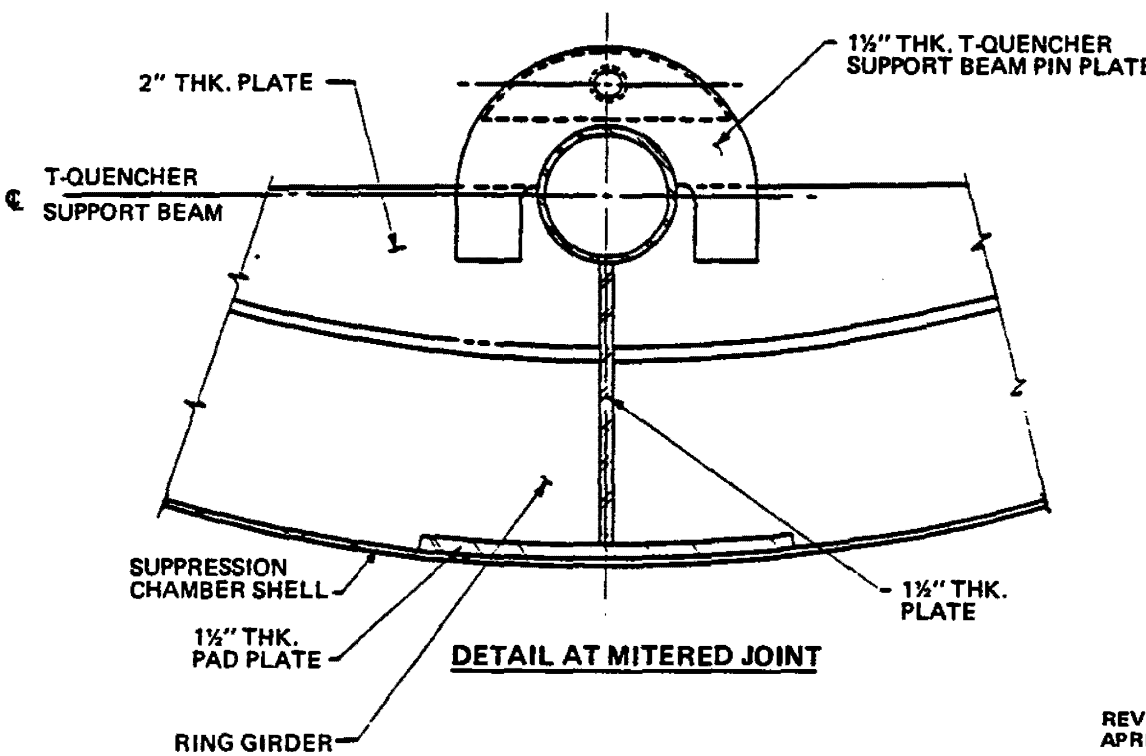
T-QUENCHER RAMSHEAD
AND SUPPORT DETAILS

UPDATED FSAR

FIGURE 3.9-13



DETAIL AT MIDCYLINDER



DETAIL AT MITERED JOINT

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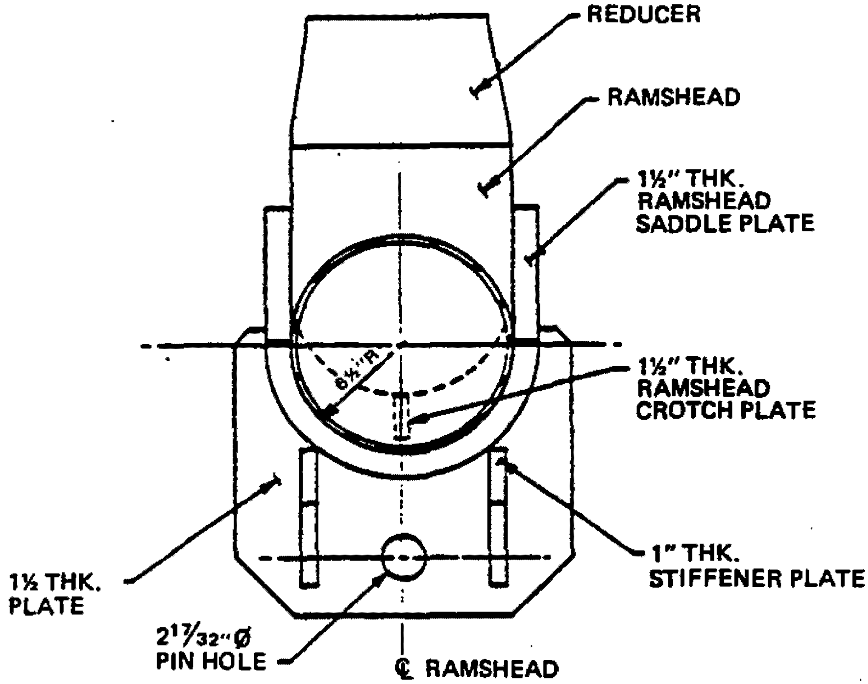
HOPE CREEK NUCLEAR GENERATING STATION

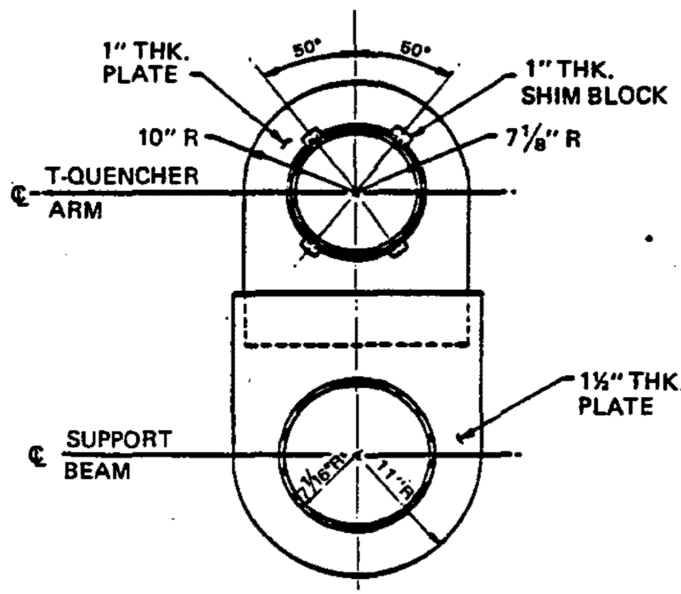
MIDCYLINDER AND MITERED
JOINT RING GIRDER T-QUENCHER
SUPPORT DETAILS

UPDATED FSAR

FIGURE 3.9-14



RAMSHEAD SUPPORT ASSEMBLY



T-QUENCHER ARM SUPPORT PLATE

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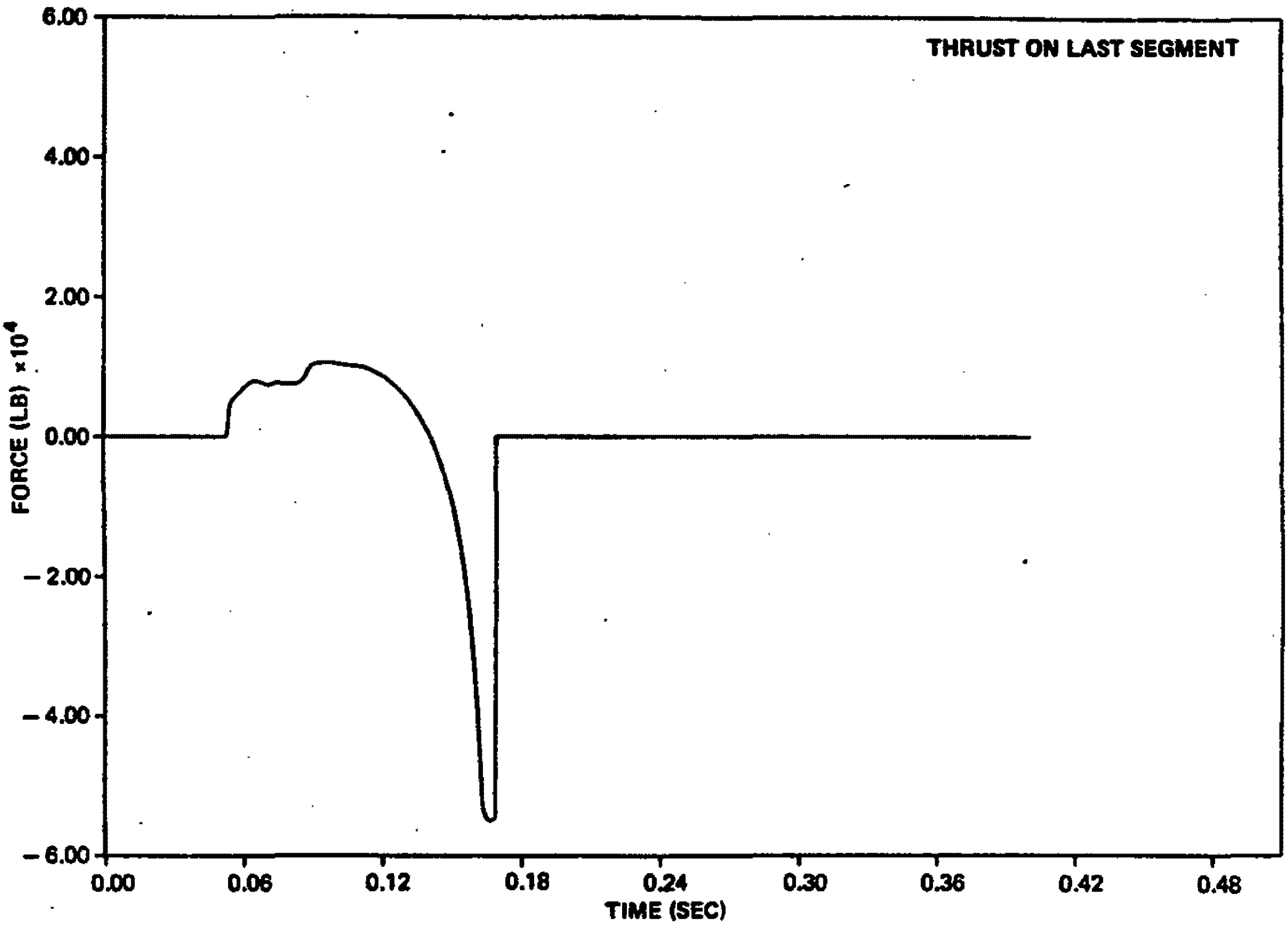
HOPE CREEK NUCLEAR GENERATING STATION

RAMSHEAD AND T-QUENCHER ARM SUPPORT DETAILS

UPDATED FSAR

FIGURE 3.9-15



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HOPE CREEK NUCLEAR GENERATING STATION

S/RV LINE P232R,
CASE A1.1A3.1/B3.1

UPDATED FSAR

FIGURE 3.9-16

3.10 SEISMIC QUALIFICATION OF SEISMIC CATEGORY I INSTRUMENTATION AND ELECTRICAL EQUIPMENT

The seismic qualification of Seismic Category I instrumentation, electrical equipment, and their supports is described in this section. Sections 3.9.2.3, 3.9.2.4, 3.9.3.2, and 3.9.3.4 address similar topics on the seismic qualification, testing, and analysis of the Seismic Category I mechanical components, equipment, and their supports, including the integral or associated electrical components such as valve mounted components and pump motors.

Dynamic testing methods and the results of the qualification of active pumps with motors and supports and pipe mounted valves listed in Table 3.9-5 are addressed in Sections 3.9.3.2 and 3.9.3.4. All safety-related equipment will be investigated further to demonstrate compliance with the requirements for the seismic qualification review team (SQRT) of the NRC. Reports will be submitted to the NRC following the completion of these programs.

3.10.1 Seismic Qualification Criteria

3.10.1.1 Seismic Category I Equipment Identification

3.10.1.1.1 Seismic Category I NSSS Equipment Identification, Excluding Motors and Valve-Mounted Equipment

Seismic Category I Nuclear Steam Supply System (NSSS) instrumentation and electrical equipment, as well as other equipment, is identified in Table 3.2-1.

All the plant Seismic Category I instrumentation and electrical equipment is qualified to resist and withstand the effects of the postulated earthquakes.

The Class 1E electrical and instrumentation equipment including the condensing chambers which are part of the Pressure Transmitter Measuring System, excluding motors and valve mounted equipment,

supplied by GE and requiring seismic qualification are identified in Table 3.10-3. The supporting structures for this equipment are identified in Table 3.10-4.

3.10.1.1.2 Seismic Category I Non-NSSS Equipment Identification

Non-NSSS safety-related instrumentation and Class 1E electrical equipment that requires seismic qualification are identified in Tables 3.10-1 and 3.10-2, respectively.

3.10.1.2 Seismic Design Criteria

3.10.1.2.1 Seismic Design Criteria (NSSS)

The NSSS seismic qualification program for HCGS utilizes seismic data generated over a number of years. Since it was not a licensing requirement at the time, most of these data were developed in earlier years without pre-aging or sequential testing of the equipment However, NSSS equipment located in harsh environments that has been qualified in recent years has generally been pre-aged and sequentially tested in accordance with the guidelines of IEEE 323-1974.

NSSS equipment on HCGS is being seismically evaluated using pre-aged and sequential testing data where it is available. Otherwise,the earlier data without pre-aging and sequential testing are being used. The aging requirement is described in Section 3.11.2.7.2.

Seismic Category I instrumentation and electrical equipment are designed to withstand the safe shutdown earthquake (SSE) defined in Section 3.7.1.

The seismic criteria used in the design and subsequent qualification of all Class 1E instrumentation and electrical equipment supplied by GE are as described below.

The Class 1E equipment is capable of performing all safety-related functions during normal plant operation, anticipated transients, design basis accidents (DBAs), and post accident operation, while being subjected to, and after the cessation of, the accelerations resulting from the SSE at the point of attachment of the equipment to the building or supporting structure.

The criteria for each of the devices used in the Class 1E systems depend on the use in a given system. For example, a relay in one system may, as its safety function, have to deenergize and open its contacts within a certain time, while in another system, it must energize and close its contacts. Since General Electric (GE) supplies many devices for many applications, the approach taken was to test the device in the worst case configuration. In this way, the capability of protective action initiation and the proper operation of safety failure circuits are ensured.

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 for 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 loadings. The magnitude and frequency of the loadings that each component will experience are determined by its specific location within the plant.

3.10.1.2.2 Seismic Design Criteria (Non-NSSS)

Seismic qualification criteria used for the Seismic Category I instrumentation and electrical equipment are in compliance with IEEE 344-1975 and Regulatory Guide 1.100.

All plant Seismic Category I instrumentation and electrical equipment are seismically qualified by one of the following methods:

1. Dynamic analysis

2. Testing under simulated seismic conditions

3. Combination of testing and analysis.

The decision criteria for selecting a qualification method are based on the practicality of the method for the function, type, size, shape, and complexity of the equipment.

Seismic input motion for the plant is defined in the project specification. Discussion of development of response spectra curves from the basic input ground motion is provided in Section 3.7.2.

3.10.2 Methods and Procedures for Qualifying Electrical Equipment and Instrumentation, Excluding Motors and Valve Mounted Equipment

3.10.2.1 NSSS Equipment Compliance with IEEE 344-1971, Regulatory Guide 1.100 and IEEE 344-1975

The following compliance statements are with respect to IEEE 344-1971, which is the plant commitment, and as such do not demonstrate compliance with Regulatory Guide 1.100. However, the seismic qualification requirements used for this plant ensure an adequate degree of equipment performance and thereby represent an acceptable basis for qualifying the equipment.

Some equipment is shown to be qualified by single axis and/or single frequency testing. However, all essential equipment is reevaluated and upgraded using SQRT program methodology for seismic qualification according to the requirements or recommendations of IEEE 344-1975, Regulatory Guides 1.92 and

1.100, and Standard Review Plans 3.9.2, 3.10, and HCGS specific requirements as described in Table 3.10-3.

In most instances, use of single axis test data is restricted to equipment with response that shows a predominant single mode of vibration in each direction with a minimal cross coupling. In some cases, if the response shows a single mode of vibration in each direction but also has cross coupling, the existing single axis test data are still used if the test response spectra (TRS) can be shown to exceed the required response spectra (RRS) by a factor of 1.4 over all frequencies.

In most instances, use of single frequency test data is restricted to cases where the required input motion is dominated by one frequency, where response of the equipment is adequately represented by one mode, or where the input motion has sufficient intensity and duration to produce sufficiently high levels of stress to assure structural integrity where structural integrity is the determinant requirement. In some cases, if the input motion is sufficiently high so as to excite secondary modes, such that modal responses can be shown to occur out of phase and at high enough levels, existing single frequency test data are also used to demonstrate operability.

3.10.2.1.1 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 effects of the safe shutdown earthquake (SSE). Both analysis and testing are used, but most equipment is tested. Analyses are used, utilizing SQRT program methodology, to demonstrate compliance with IEEE 344-1975. Also, analysis is used to determine the adequacy of mechanical strength, e.g., mounting bolts, etc., after operating capability is established by testing as follows:

1. Analysis - GE-supplied Class 1E equipment performing primarily a mechanical safety function, e.g., pressure boundary devices, etc, is analyzed since the passive nature of their critical safety role usually makes testing impractical. Analytical methods sanctioned by IEEE 344-1971 are used in such cases. Also Class 1E equipment is analyzed to meet IEEE 344-1975 utilizing SQRT program methodology. See Table 3.10-3 for indication of which items were qualified by analysis.

2. Testing - GE-supplied Class 1E equipment having primarily an active electrical safety function is tested in compliance with IEEE 344-1971, Section 3.2.

3.10.2.1.2 Documentation

Available documentation verifies that the seismic qualification of GE-supplied Class 1E equipment is in accordance with the requirements of IEEE 344-1971 and IEEE 344-1975 Section 4.

3.10.2.2 Testing Procedures for Qualifying NSSS Electrical Equipment and Instrumentation, Excluding Motors and Valve Mounted Equipment

The test procedures require that the device be mounted on the table of the vibration machine in a manner similar to the actual mounting condition. The device is tested in the operating states as if it were performing its Class 1E functions. These states are 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 are tested if the relay is used in those configurations in its Class 1E functions.

The seismic excitation is a single frequency test in which the applied vibration is a sinusoidal table motion at a fixed peak acceleration and a discrete frequency at any given time. Each

frequency and acceleration combination is maintained for about 30 seconds, except when the resonance search is made. See IEEE 344-1971, Paragraph 3.2. The vibratory excitation is applied in three orthogonal axes individually with the axes chosen as those coincident with the most probable mounting configuration.

The first step is to search for resonances in each device. This is done since resonances cause amplification of the input vibration and are the most likely cause of malfunction. The resonance search is usually run at low acceleration levels (0.2 g) to avoid damaging the test sample in case a severe resonance is encountered. The resonance search is run from 1 to 33 hertz, in accordance with IEEE 344, in no less than 7 minutes. If the device is large enough, the vibrations are monitored by accelerometers placed at critical locations. Resonances are determined by comparing the acceleration level with that at the table of the vibration machine. Sometimes, the devices either are too small for an accelerometer, have their critical parts in an inaccessible location, or have critical parts that would be adversely affected by the mounting of an accelerometer. In these cases, the resonances were detected by visual (strobe light) or audible observation or by performance.

Following the frequency scan and resonance determination, the devices are tested to determine their malfunction limit. This test is a necessary adjunct to the assembly test as will be shown later. The malfunction limit test is run at each resonant frequency as determined by the frequency scan. In this test, the acceleration level is gradually increased until either the device malfunctions or the limit of the device is reached. If no resonances are detected, as is usually the case, the device is considered to be rigid (all parts move in unison), and the malfunction limit is therefore independent of frequency. To achieve maximum acceleration from the vibration machine, rigid devices are malfunction tested at the upper test frequency, 33 hertz, since that allows 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 3.10-3 includes the qualification limit for each device tested.

The above procedures are required of purchased devices as well as those made by GE. Vendor test results are reviewed, and if unacceptable, the tests are repeated either by GE or the vendor. If the vendor tests were adequate, the device is considered qualified to the limits of the test.

3.10.2.3 Methods and Procedures for Qualifying Non-NSSS Instrumentation and Electrical Equipment

Where applicable, all equipment is pre-aged prior to seismic testing as part of the test sequence. The aging requirement is described in Section 3.11.2.7.2. Maintenance and surveillance program requirements given in Section 3.11.2.7.6 incorporate the results of testing, as applicable.

The analysis and testing for the seismic qualification of non-NSSS Class 1E instrumentation and electrical equipment are in compliance with the appropriate project seismic specifications that meet the requirements of IEEE 344-1975 and Regulatory Guide 1.100.

Pipe mounted instrumentation is qualified by analysis and/or testing to the acceleration levels allowed for piping systems. These levels include gravity and operation loading, as well as loading that is due to seismic or any other accident related excitation, if applicable.

All pipe mounted control valve operators and accessories are qualified by using a single axis, single frequency test (required input motion (RIM) test). This is justified on the grounds that the seismic floor motion is filtered through the piping system, which generally has one predominant structural mode. Thus the resulting motion that reaches the line mounted equipment is

predominantly a single frequency and a single axis motion. The test is performed by using RIM in each of the three axes, independently.

Seismic Category I equipment is shown to be capable of withstanding the horizontal and vertical accelerations of five OBEs and one SSE by dynamic analysis, dynamic testing, or a combination of dynamic analysis and testing.

In accordance with the Mark I Containment Long Term Program (NUREG-0661), non-NSSS equipment attached to the torus has been evaluated for appropriate hydrodynamic loads, including fatigue effects.

Thermowells provided on the torus shell for temperature monitoring are also qualified for hydrodynamic loads for frequencies up to 50 hz, including fatigue effects.

3.10.2.3.1 Dynamic Analysis

Dynamic analysis without testing is performed as a basis for qualification only if the necessary functional operability of the equipment is ensured by its structural integrity alone.

For this analysis, equipment is idealized using a mathematical model in which frequencies and mode shapes are determined for vibration in the vertical direction and two orthogonal horizontal directions. For each direction of vibration, the spectral accelerations per mode are obtained from the appropriate response spectrum curve corresponding to the location and damping of the equipment.

Seismic loading in terms of inertia forces, moments, and shears for each direction is determined by combining the response of the individual modes. The criteria used in combining modal responses and spatial components are in accordance with Regulatory Guide 1.92. If the orientation of the equipment is not

designated, the horizontal seismic loading is taken as the maximum loading obtained from either horizontal direction.

If the frequencies of all equipment modes, determined by either analysis or testing, are greater than the frequency of the appropriate spectrum response curve at which the acceleration is constant in the rigid or high frequency range, the seismic loading consists of the static equivalent loading corresponding to that acceleration level. The damping values used are in accordance with Regulatory Guide 1.61 and IEEE 344-1975 for electrical equipment and instrumentation, respectively.

In lieu of determining the vibrational frequencies and corresponding loads for equipment which can be represented by a simple model, the equivalent static load is adequately represented by a 1.5 factor applied to the peak acceleration of the applicable response spectrum. A factor of less than 1.5 is used if adequate justification is provided.

The dynamic analysis is done in accordance with Section 5 of IEEE 344-1975.

3.10.2.3.2 Dynamic Tests

In lieu of performing a dynamic analysis, seismic adequacy is established by providing seismic tests. Dynamic tests are performed by subjecting equipment to vibratory motions that conservatively simulate the required response spectrum or the required input motion at the equipment mounting location. Equipment is tested in the operational condition. Operability is verified during and/or after the testing, as applicable to the equipment being tested.

The test results have demonstrated that the test response spectrum closely resemble and envelope the required response spectrum over

the critical frequency range. For the pipe mounted instrumentation, the required input motion is enveloped by the test input motion.

The requirements of testing procedures and methods are in accordance with the project seismic specification and with IEEE 344-1975, Section 6. The tests are performed using a combination or one of the following techniques:

1. Proof testing
2. Fragility testing
3. Device testing
4. Assembly testing
5. Generic testing.

3.10.2.3.3 Combined Analysis and Testing

Equipment that cannot be qualified practically by analysis or testing because of its size and/or complexity is qualified by combined analysis and testing. Combined analysis and testing methods are in accordance with IEEE 344-1975, Section 7.

3.10.3 Methods and Procedure of Analysis or Testing of Supports of Electrical Equipment and Instrumentation

3.10.3.1 NSSS-Seismic Analysis Testing Procedures and Restraint Measures

The Class 1E equipment supplied by GE is used in many systems on many different plants under widely varying seismic requirements.

The HCGS control room panels and local instrument panels are qualified for seismic adequacy by comparison to tested equivalent panels and devices by SQRT program methodology as described in Table 3.10-4.

Some GE supplied Class 1E devices are qualified by analysis only, as shown in Tables 3.10-3 and 3.10-4. Analysis is used for active devices in accordance with SQRT program methodology and for passive mechanical devices and is sometimes used in combination with testing for larger assemblies containing Class 1E devices. For instance, a test might run to determine if there are natural frequencies in the equipment within the critical seismic frequency range discussed in IEEE 344-1971, Paragraph 3.2.2.3.1. If the equipment has a frequency greater than the critical seismic frequency range, it is assumed to be rigid, and a static analysis is performed. If it has natural frequencies in the critical frequency range, then calculations of transmissibility and responses to varying input accelerations are determined to see if Class 1E devices mounted in the assembly would operate without malfunctioning. The testing of Class 1E equipment is accomplished using the following procedure.

Assemblies, i.e., control panels, containing devices that have had seismic malfunction limits established are tested by mounting the assembly on the table of a vibration machine in the manner it is to be mounted when in use. It is vibration tested by running a low level resonance search. As with the devices, the assemblies are tested in the three major orthogonal axes. The resonance search is run in the same manner as described for devices. If resonances are present, the transmissibility between the input and the location of each Class 1E device is determined by measuring the accelerations at each device location and calculating the magnification between it and the input. Once known, the transmissibilities can be used analytically to determine the response at any Class 1E device location for any given input. It is conservatively assumed that the transmissibilities are linear as a function of acceleration, even though they actually decrease

as acceleration is increased. As long as the device input accelerations are determined to be below their malfunction limits, then the assembly is considered a rigid body with a transmissibility equal to 1, so that a device mounted on it would be limited directly by the assembly input acceleration.

Since control panels and racks constitute the majority of Class 1E assemblies supplied by GE, seismic qualification testing of these is discussed in more detail. There are basically four generic types. One or more of each type is tested using the above procedures.

Figures 3.10-1 through 3.10-4 illustrate the four basic types referenced above and show typical accelerometer locations. The status of the seismic tests on the Class 1E panels supplied by GE for this plant is summarized in Table 3.10-4.

The full acceleration level tests described above disclose that most of the panel types had more than adequate mechanical strength. A given panel design acceptability is just a function of its amplification factor and the malfunction levels of the devices mounted in it. Subsequent panels are, therefore, tested at lower acceleration levels, and the transmissibilities are measured to the various devices, as described above. By dividing the device malfunction levels by the panel transmissibility between the device and the panel input, the panel seismic qualification level can be determined. Several high level tests are run on selected generic panel designs to ensure conservatism in using the transmissibility analysis described.

3.10.3.2 Non-NSSS Equipment Supports

Analyses or tests are performed for all supports of Class 1E electrical equipment and instrumentation, i.e., switchgears, battery racks, instrument racks, control consoles, cabinets, panels, and electrical raceways, to ensure their structural capability to withstand seismic excitation.

Required response spectra for devices that are mounted on flexible supports or panels are modified for the flexibility of the supports.

Supports are tested with equipment installed or with a dummy simulating the equivalent equipment inertial mass effects and dynamic coupling to the support. If the equipment is installed in a nonoperational mode for the support test, the response in the test at the equipment mounting location is monitored and is characterized in the form of response spectrum. In such a case, equipment is tested separately for operability using this response spectrum.

The analytical method and much of the design criteria pertaining to electrical raceway systems is verified by the Cable Tray and Conduit Raceway Test Program, as shown in Reference 3.10-1. During that program, approximately 2000 dynamic tests were performed on several hundred varied cable tray and conduit support systems. As a result of this extensive test program, a conservative design criteria for Class 1E cable tray and conduit support systems has been developed. The design criteria also consider the low cycle fatigue phenomena for connections and include justification on use of higher damping values for cable tray and conduit support systems.

The following bases are used in the seismic design and analysis of Class 1E electrical raceway supports and instrument tubing supports:

1. All Class 1E electrical raceway supports and instruments tubing supports are qualified by analysis, using the response spectrum method described in Section 3.10.2.3.1.

2. Analysis and design of seismic restraint measures for electrical raceway supports and instrument tubing supports are based on combined limiting values for static load, span length, and computed seismic response.

3. Maximum stress is limited to 90 percent of minimum yield stress.

4. The Seismic Category I instrument tubing systems are supported so that the allowable stresses permitted by Section III of the ASME B&PV Code are not exceeded when the tubing is subjected to the loads specified in Section 3.9 for Class 2 and 3 piping.

For field mounted instruments, the following are applicable:

1. The mounting structures for Seismic Category I instruments have a fundamental frequency equal to or greater than the appropriate ZPA frequency or the frequency as required by the supplier of the equipment being mounted whichever is greater and thus are in the rigid range for seismic acceleration.

2. The stress level in the mounting structure does not exceed the material allowable stress when the mounting structure is subjected to the maximum acceleration level for its location in combination with other design loads.

3.10.4 Operating License Review

3.10.4.1 NSSS Control and Electrical Equipment Other Than Motors and Valve Mounted Equipment

The seismic test results for safety-related panels and control equipment within the NSSS scope are maintained in a permanent file by General Electric (GE), and can be readily audited in all cases. The equipment used in Class 1E applications passes the prescribed tests. Where equipment fails to pass the tests, it is rejected. In some cases, equipment that fails one test is modified or repaired to meet the performance requirements and retested. If the retested equipment passes, it may be used in a Class 1E application.

Table 3.10-3 lists the NSSS control devices by master parts list (MPL), item number, and vendor. A summary of the test conditions for the devices used in Class 1E applications is also given in Table 3.10-4. The acceleration level shown in the right columns of Table 3.10-3 is the acceleration at which either the device malfunctions or the limit of the vibration machine is reached.

3.10.4.2 NSSS Motors

The dynamic qualification results for the Emergency Core Cooling System (ECCS) motors and standby liquid control (SLC) are discussed in Section 3.9.2.3 in conjunction with their respective pump motor assemblies.

3.10.4.3 Non-NSSS Equipment

Qualification documents containing results of qualification tests and analysis are listed in Tables 3.10-1 and 3.10-2 for non-NSSS instruments and electrical equipment. The qualification documents are maintained by PSE&G in a centrally located, readily auditable, permanent file.

3.10.5 References

3.10-1 ANCO Engineers, Inc, "Cable Tray and Conduit Raceway Seismic Test Program - Release-4," Report 1053-21.1-4, December 15, 1978.

TABLE 3.10-1

SEISMIC QUALIFICATION TEST SUMMARY
NON-NSSS INSTRUMENTS

Item Number	Description	Area/Elevation	Supplier	Qualification Method	Standards (1)
J-200Q	Main control panels	Main Control rm/137ft, Control eqpt rm/102ft	Bailey	Test & analysis	A, I
J-201Q	Remote control panels	Various	Comsip/ Customline	Test & analysis	A, I
J-301Q	Electronic field transmitters	Various	Tobar	Test	A, I
J-305Q	Panel-mounted instruments	Various	Westinghouse	Test	A, I
J-359Q	H_2, O_2 analyzer	React bldg/162 ft	Comsip Delphi	Test & analysis	A, I
J-371Q	Radiation monitoring (FRVS-Duct Section)	React bldg/186 ft	TEC	Analysis	A, I
J-371Q	Radiation Monitoring SAC-Sample Shield	React bldg/102 ft	TEC	Analysis	A, I
J-373Q	Radiation Monitoring System	Various	General Atomic TEC	Test & Analysis	A, I
J-483Q	Flood alarms	Various	Fluid C (FCI)	Test	A, I
J-525Q	Pressure indicators	Various	Dresser	Test	A, I
J-556Q	Temp Wells, RDTs	Various	Thermo Electric	Test & analysis	A, I
J-601Q	Control valves	Various	Masoneilan	Test & analysis	A, F, I
J-603Q	Solenoid valves	Various	Valcor	Test & analysis	D, F
J-605Q	Control butterfly valves	Various	Fisher	Test & analysis	F
J-610Q	Pressure regulators	Various	Marotta	Test	A, I
J-703Q	Excess flow check valves	Various	Dragon	Test	F
J-705Q	Bypass manifolds	Various	Dragon	Analysis	F
J-730Q	Flexible metal hose	Various	Metal Bellows	Test	A, I

Item Number	Description	Area/Elevation	Supplier	Qualification Method	Standards (1)
J-810Q	Emergency load sequence	Aux 130'0"	Consolidated Controls	Test	A, I
F37917Q	Terminal and fuse blocks	Various	Amerace Corp.	Test	A
F51216Q	Electric cond. seal assembly	Various	Conax Corp.	Test	A
F61553Q	Fittings	React bldg	PG Grand	Test	A

(1) Standards
A - IEEE 344-1975
D - IEEE 382-1980
F - ASME Code-Section III
I - NRC REG. GUIDE 1.100, Rev. 1

TABLE 3.10-2

SEISMIC QUALIFICATION TEST SUMMARY

NON-NSSS ELECTRICAL EQUIPMENT

P.O. NUMBER	EQUIPMENT TAG NUMBER	DESCRIPTION	MANUFACTURER	BLDG AND ELEV (1)	QUALIFICATION	STANDARDS (2)
E-109	1AN205	1E 5kV SWGR	BROWN BOVERI	RB 102	TEST	A, I
E-109	1BN205	1E 5kV SWGR	BROWN BOVERI	RB 102	TEST	A, I
E-109	1CN205	1E 5kV SWGR	BROWN BOVERI	RB 102	TEST	A, I
E-109	1DN205	1E 5kV SWGR	BROWN BOVERI	RB 102	TEST	A, I
E-109	10A401	1E 4.16kV SWGR	BROWN BOVERI	DG 130	TEST	A, I
E-109	10A402	1E 4.16kV SWGR	BROWN BOVERI	DG 130	TEST	A, I
E-109	10A403	1E 4.16kV SWGR	BROWN BOVERI	DG 130	TEST	A, I
E-109	10A404	1E 4.16kV SWGR	BROWN BOVERI	DG 130	TEST	A, I
E-112A	1AP210	SACS MOTORS	WESTINGHOUSE	RB 102	ANALYSIS	A, I
E-112A	1BP210	SACS MOTORS	WESTINGHOUSE	RB 102	ANALYSIS	A, I
E-112A	1CP210	SACS MOTORS	WESTINGHOUSE	RB 102	ANALYSIS	A, I
E-112A	1DP210	SACS MOTORS	WESTINGHOUSE	RB 102	ANALYSIS	A, I
E-112B	1AP502	SVC WTR PP MTR	GE	1S093	ANALYSIS	A, I
E-112B	1BP502	SVC WTR PP MTR	GE	1S093	ANALYSIS	A, I
E-112B	1CP502	SVC WTR PP MTR	GE	1S093	ANALYSIS	A, I
E-112B	1DP502	SVC WTR PP MTR	GE	1S093	ANALYSIS	A, I
E-117	10B410	480V UNIT SUB	GE	DG130	TEST & ANALYSIS	A, I
E-117	10B420	480V UNIT SUB	GE	DG130	TEST & ANALYSIS	A, I
E-117	10B430	480V UNIT SUB	GE	DG130	TEST & ANALYSIS	A, I
E-117	10B440	480V UNIT SUB	GE	DG130	TEST & ANALYSIS	A, I
E-117	10B450	480V UNIT SUB	GE	DG130	TEST & ANALYSIS	A, I
E-117	10B460	480V UNIT SUB	GE	DG130	TEST & ANALYSIS	A, I
E-117	10B470	480V UNIT SUB	GE	DG130	TEST & ANALYSIS	A, I
E-117	10B480	480V UNIT SUB	GE	DG130	TEST & ANALYSIS	A, I
E-118	10B242	480V MCC	CUTLER HAMMER UNITROL	RB077	TEST & ANALYSIS	A, I
E-118	10B212	480V MCC	CUTLER HAMMER UNITROL	RB102	TEST & ANALYSIS	A, I
E-118	10B222	480V MCC	CUTLER HAMMER UNITROL	RB102	TEST & ANALYSIS	A, I
E-118	10B232	480V MCC	CUTLER HAMMER UNITROL	RB102	TEST & ANALYSIS	A, I
E-118	10B411	480V MCC	CUTLER HAMMER UNITROL	DG130	TEST & ANALYSIS	A, I
E-118	10B421	480V MCC	CUTLER HAMMER UNITROL	DG130	TEST & ANALYSIS	A, I
E-118	10B431	480V MCC	CUTLER HAMMER UNITROL	DG130	TEST & ANALYSIS	A, I
E-118	10B441	480V MCC	CUTLER HAMMER UNITROL	DG130	TEST & ANALYSIS	A, I
E-118	10B451	480V MCC	CUTLER HAMMER UNITROL	DG130	TEST & ANALYSIS	A, I
E-118	10B461	480V MCC	CUTLER HAMMER UNITROL	DG130	TEST & ANALYSIS	A, I
E-118	10B471	480V MCC	CUTLER HAMMER UNITROL	DG130	TEST & ANALYSIS	A, I
E-118	10B481	480V MCC	CUTLER HAMMER UNITROL	DG130	TEST & ANALYSIS	A, I

P.O. NUMBER	EQUIPMENT TAG NUMBER	DESCRIPTION	MANUFACTURER	BLDG AND ELEV (1)	QUALIFICATION	STANDARDS (2)
E-118	10B553	480V MCC	CUTLER HAMMER UNITROL	IS093	TEST & ANALYSIS	A, I
E-118	10B563	480V MCC	CUTLER HAMMER UNITROL	IS093	TEST & ANALYSIS	A, I
E-118	10B573	480V MCC	CUTLER HAMMER UNITROL	IS093	TEST & ANALYSIS	A, I
E-118	10B583	480V MCC	CUTLER HAMMER UNITROL	IS093	TEST & ANALYSIS	A, I
E-120	1AD417	125VDC DIST PNL	BROWN BOVERI	DG130	TEST	A, I
E-120	1BD417	125VDC DIST PNL	BROWN BOVERI	DG130	TEST	A, I
E-120	1CD417	125VDC DIST PNL	BROWN BOVERI	DG130	TEST	A, I
E-120	1DD417	125VDC DIST PNL	BROWN BOVERI	DG130	TEST	A, I
E-121	10D410	125VDC SWGR	GE	DG130	TEST & ANALYSIS	A, I
E-121	10D420	125VDC SWGR	GE	DG130	TEST & ANALYSIS	A, I
E-121	10D430	125VDC SWGR	GE	DG130	TEST & ANALYSIS	A, I
E-121	10D436	125VDC SWGR	GE	DG163	TEST & ANALYSIS	A, I
E-121	10D440	125VDC SWGR	GE	DG130	TEST & ANALYSIS	A, I
E-121	10D446	125VDC SWGR	GE	DG163	TEST & ANALYSIS	A, I
E-121	10D470	125VDC SWGR	GE	CR054	TEST & ANALYSIS	A, I
E-121	10D480	125VDC SWGR	GE	CR054	TEST & ANALYSIS	A, I
E-121	10D476	125VDC SWGR	GE	DG163	TEST & ANALYSIS	A, I
E-121	10D486	125VDC SWGR	GE	DG163	TEST & ANALYSIS	A, I
E-121	10D450	250VDC SWGR	GE	CR054	TEST & ANALYSIS	A, I
E-121	10D460	250VDC SWGR	GE	CR054	TEST & ANALYSIS	A, I
E-121	10D251	250VDC SWGR	GE	RB054	TEST & ANALYSIS	A, I
E-121	10D261	250VDC SWGR	GE	RB054	TEST & ANALYSIS	A, I
E-122	1AC680A/B	AUX CAB (SKIN)	COMSIP	DG163	TEST & ANALYSIS	A, I
E-122	1BC680A/B	AUX CAB (SKIN)	COMSIP	DG163	TEST & ANALYSIS	A, I
E-122	1CC680A/B	AUX CAB (SKIN)	COMSIP	DG163	TEST & ANALYSIS	A, I
E-122	1DC680A/B	AUX CAB (SKIN)	COMSIP	DG163	TEST & ANALYSIS	A, I
E-122	1AC680C/D	AUX CAB (SKIN)	COMSIP	DG163	TEST & ANALYSIS	A, I
E-122	1BC680C/D	AUX CAB (SKIN)	COMSIP	DG163	TEST & ANALYSIS	A, I
E-122	1CC680C/D	AUX CAB (SKIN)	COMSIP	DG163	TEST & ANALYSIS	A, I
E-122	1DC680C/D	AUX CAB (SKIN)	COMSIP	DG163	TEST & ANALYSIS	A, I
E-122	1(A-D)C680	AUX CAB (SKIN)	COMSIP	DG163	TEST & ANALYSIS	A, I
E-135	1AW201	M.V. PEN.	WESTINGHOUSE	RB114	TEST & ANALYSIS	A, I
E-135	1BW201	M.V. PEN.	WESTINGHOUSE	RB114	TEST & ANALYSIS	A, I
E-135	1CW201	M.V. PEN.	WESTINGHOUSE	RB114	TEST & ANALYSIS	A, I
E-135	1DW201	M.V. PEN.	WESTINGHOUSE	RB114	TEST & ANALYSIS	A, I
E-135	1EW201	M.V. PEN.	WESTINGHOUSE	RB114	TEST & ANALYSIS	A, I

P.O. NUMBER	EQUIPMENT TAG NUMBER	DESCRIPTION	MANUFACTURER	BLDG AND ELEV (1)	QUALIFICATION	STANDARDS (2)
E-135	1FW201	M.V. PEN.	WESTINGHOUSE	RB114	TEST & ANALYSIS	A, I
E-135	1AW200	L.V. PEN.	WESTINGHOUSE	RB114	TEST & ANALYSIS	A, I
E-135	1BW200	L.V. PEN.	WESTINGHOUSE	RB114	TEST & ANALYSIS	A, I
E-135	1CW200	L.V. PEN.	WESTINGHOUSE	RB117	TEST & ANALYSIS	A, I
E-135	1DW200	L.V. PEN.	WESTINGHOUSE	RB108	TEST & ANALYSIS	A, I
E-135	1AW202	L.V. PEN.	WESTINGHOUSE	RB111	TEST & ANALYSIS	A, I
E-135	1BW202	L.V. PEN.	WESTINGHOUSE	RB117	TEST & ANALYSIS	A, I
E-135	1BW205	L.V. PEN.	WESTINGHOUSE	RB117	TEST & ANALYSIS	A, I
E-135	1JW204	L.V. PEN.	WESTINGHOUSE	RB111	TEST & ANALYSIS	A, I
E-135	1AW204	L.V. PEN.	WESTINGHOUSE	RB111	TEST & ANALYSIS	A, I
E-135	1BW204	L.V. PEN.	WESTINGHOUSE	RB108	TEST & ANALYSIS	A, I
E-135	1CW204	L.V. PEN.	WESTINGHOUSE	RB111	TEST & ANALYSIS	A, I
E-135	1DW204	L.V. PEN.	WESTINGHOUSE	RB108	TEST & ANALYSIS	A, I
E-135	1EW204	L.V. PEN.	WESTINGHOUSE	RB111	TEST & ANALYSIS	A, I
E-135	1FW204	L.V. PEN	WESTINGHOUSE	RB115	TEST & ANALYSIS	A, I
E-135	1GW204	L.V. PEN	WESTINGHOUSE	RB111	TEST & ANALYSIS	A, I
E-135	1HW204	L.V. PEN	WESTINGHOUSE	RB115	TEST & ANALYSIS	A, I
E-135	1KW204	L.V. PEN	WESTINGHOUSE	RB111	TEST & ANALYSIS	A, I
E-135	1AW203	L.V. PEN	WESTINGHOUSE	RB108	TEST & ANALYSIS	A, I
E-135	1DW202	L.V. PEN.	WESTINGHOUSE	RB111	TEST & ANALYSIS	A, I
E-135	1BW203	L.V. PEN.	WESTINGHOUSE	RB115	TEST & ANALYSIS	A, I
E-135	1CW202	L.V. PEN.	WESTINGHOUSE	RB111	TEST & ANALYSIS	A, I
E-135	1CW205	L.V. PEN.	WESTINGHOUSE	RB117	TEST & ANALYSIS	A, I
E-135	1AW205	L.V. PEN.	WESTINGHOUSE	RB115	TEST & ANALYSIS	A, I
E-135	1DW205	L.V. PEN.	WESTINGHOUSE	RB117	TEST & ANALYSIS	A, I
E-135	1EW205	L.V. PEN.	WESTINGHOUSE	RB115	TEST & ANALYSIS	A, I
E-135	1FW205	L.V. PEN.	WESTINGHOUSE	RB115	TEST & ANALYSIS	A, I
E-135	1GW205	L.V. PEN.	WESTINGHOUSE	RB117	TEST & ANALYSIS	A, I
E-135	1HW205	L.V. PEN.	WESTINGHOUSE	RB117	TEST & ANALYSIS	A, I
E-135	1AW206	L.V. PEN.	WESTINGHOUSE	RB115	TEST & ANALYSIS	A, I
E-135	1CW206	L.V. PEN.	WESTINGHOUSE	RB115	TEST & ANALYSIS	A, I
E-135	1AW207	L.V. PEN.	WESTINGHOUSE	RB079	TEST & ANALYSIS	A, I
E-135	1BW207	L.V. PEN.	WESTINGHOUSE	RB078	TEST & ANALYSIS	A, I
E-135	1CW207	L.V. PEN.	WESTINGHOUSE	RB078	TEST & ANALYSIS	A, I
E-135	1DW207	L.V. PEN.	WESTINGHOUSE	RB079	TEST & ANALYSIS	A, I
E-135	1AW208	L.V. PEN.	WESTINGHOUSE	RB078	TEST & ANALYSIS	A, I
E-135	1AW209	L.V. PEN.	WESTINGHOUSE	RB075	TEST & ANALYSIS	A, I

P.O. NUMBER	EQUIPMENT TAG NUMBER	DESCRIPTION	MANUFACTURER	BLDG AND ELEV (1)	QUALIFICATION	STANDARDS (2)
E-139	10X201	15kVa XFMR	SQUARE D	RB102	TEST	A, I
E-139	10X202	15kVa XFMR	SQUARE D	RB102	TEST	A, I
E-139	10X203	15kVa XFMR	SQUARE D	RB102	TEST	A, I
E-139	10X204	15kVa XFMR	SQUARE D	RB102	TEST	A, I
E-139	10X411	30kVa XFMR	SQUARE D	DG130	TEST	A, I
E-139	10X412	30kVa XFMR	SQUARE D	DG130	TEST	A, I
E-139	10X413	30kVa XFMR	SQUARE D	DG130	TEST	A, I
E-139	10X414	30kVa XFMR	SQUARE D	DG130	TEST	A, I
E-139	10X421	15kVa XFMR	SQUARE D	DG130	TEST	A, I
E-139	10X422	15kVa XFMR	SQUARE D	DG130	TEST	A, I
E-139	10X423	15kVa XFMR	SQUARE D	DG130	TEST	A, I
E-139	10X424	15kVa XFMR	SQUARE D	DG130	TEST	A, I
E-139	10X501	15kVa XFMR	SQUARE D	IS093	TEST	A, I
E-139	10X502	15kVa XFMR	SQUARE D	IS093	TEST	A, I
E-139	10X503	15kVa XFMR	SQUARE D	IS093	TEST	A, I
E-139	10X504	15kVa XFMR	SQUARE D	IS093	TEST	A, I
E-139	10Y201	120VAC DIST PNL	BROWN BOVERI	RB102	TEST	A, I
E-139	10Y202	120VAC DIST PNL	BROWN BOVERI	RB102	TEST	A, I
E-139	10Y203	120VAC DIST PNL	BROWN BOVERI	RB102	TEST	A, I
E-139	10Y204	120VAC DIST PNL	BROWN BOVERI	RB077	TEST	A, I
E-139	10Y401	120VAC DIST PNL	BROWN BOVERI	RB130	TEST	A, I
E-139	10Y402	120VAC DIST PNL	BROWN BOVERI	RB130	TEST	A, I
E-139	10Y403	120VAC DIST PNL	BROWN BOVERI	RB130	TEST	A, I
E-139	10Y404	120VAC DIST PNL	BROWN BOVERI	RB130	TEST	A, I
E-139	10Y411	120VAC DIST PNL	BROWN BOVERI	RB130	TEST	A, I
E-139	10Y412	120VAC DIST PNL	BROWN BOVERI	RB130	TEST	A, I
E-139	10Y413	120VAC DIST PNL	BROWN BOVERI	RB130	TEST	A, I
E-139	10Y414	120VAC DIST PNL	BROWN BOVERI	RB130	TEST	A, I
E-139	10Y501	120VAC DIST PNL	BROWN BOVERI	IS093	TEST	A, I
E-139	10Y502	120VAC DIST PNL	BROWN BOVERI	IS093	TEST	A, I
E-139	10Y503	120VAC DIST PNL	BROWN BOVERI	IS093	TEST	A, I
E-139	10Y504	120VAC DIST PNL	BROWN BOVERI	IS093	TEST	A, I
E-150	1AD411	125V BATTERY	C&D	DG146	TEST	A, I
E-150	1BD411	125V BATTERY	C&D	DG146	TEST	A, I
E-150	1CD411	125V BATTERY	C&D	DG146	TEST	A, I
E-150	1DD411	125V BATTERY	C&D	DG146	TEST	A, I
E-150	10D411	125V BATTERY	C&D	CR054	TEST	A, I
E-150	10D431	125V BATTERY	C&D	CR054	TEST	A, I
E-150	1CD447	125V BATTERY	C&D	DG163	TEST	A, I

P.O. NUMBER	EQUIPMENT TAG NUMBER	DESCRIPTION	MANUFACTURER	BLDG AND ELEV (1)	QUALIFICATION	STANDARDS (2)
E-150	1DD447	125V BATTERY	C&D	DG163	TEST	A, I
E-150	1DD447	125V BATTERY	C&D	DG163	TEST	A, I
E-150	1AD412	125V SW	PRINGLE	DG146	TEST	A, I
E-150	1BD412	125V SW	PRINGLE	DG146	TEST	A, I
E-150	1CD412	125V SW	PRINGLE	DG146	TEST	A, I
E-150	1DD412	125V SW	PRINGLE	DG146	TEST	A, I
E-150	10D422	125V SW	PRINGLE	CR054	TEST	A, I
E-150	10D432	125V SW	PRINGLE	CR054	TEST	A, I
E-150	1CD448	125V SW	PRINGLE	DG163	TEST	A, I
E-150	1DD448	125V SW	PRINGLE	DG163	TEST	A, I
E-151	1AD413	125V BATT CHGR	C&D	DG146	TEST	A, I
E-151	1BD413	125V BATT CHGR	C&D	DG146	TEST	A, I
E-151	1CD413	125V BATT CHGR	C&D	DG146	TEST	A, I
E-151	1DD413	125V BATT CHGR	C&D	DG146	TEST	A, I
E-151	1AD414	125V BATT CHGR	C&D	DG146	TEST	A, I
E-151	1BD414	125V BATT CHGR	C&D	DG146	TEST	A, I
E-151	1CD414	125V BATT CHGR	C&D	DG146	TEST	A, I
E-151	1DD414	125V BATT CHGR	C&D	DG146	TEST	A, I
E-151	1ED414	125V BATT CHGR	C&D	DG146	TEST	A, I
E-151	1FD414	125V BATT CHGR	C&D	DG146	TEST	A, I
E-151	1CD444	125V BATT CHGR	C&D	DG163	TEST	A, I
E-151	1DD444	125V BATT CHGR	C&D	DG163	TEST	A, I
E-151	10D423	250V BATT CHGR	C&D	CR054	TEST	A, I
E-151	10D433	250V BATT CHGR	C&D	CR054	TEST	A, I
E-151	10D424	250V BATT CHGR	C&D	CR054	TEST	A, I
E-153A	1AJ481	INSTR PWR PNL	SIMMONDS	CR137	TEST & ANALYSIS	A, I
E-153A	1BJ481	INSTR PWR PNL	SIMMONDS	CR124	TEST & ANALYSIS	A, I
E-153A	1CJ481	INSTR PWR PNL	SIMMONDS	CR137	TEST & ANALYSIS	A, I
E-153A	1DJ481	INSTR PWR PNL	SIMMONDS	CR124	TEST & ANALYSIS	A, I
E-153B	1AJ482	1E 225A PNL BD	ECS	DG163	TEST	A, I
E-153B	1BJ482	1E 225A PNL BD	ECS	DG163	TEST	A, I
E-153B	1CJ482	1E 225A PNL BD	ECS	DG163	TEST	A, I
E-153B	1DJ482	1E 225A PNL BD	ECS	DG163	TEST	A, I
E-153B	1YF401	1E 100A FUSE PNL	ECS	CR102	TEST	A, I
E-153B	1YF402	1E 100A FUSE PNL	ECS	CR102	TEST	A, I
E-153B	1YF403	1E 100A FUSE PNL	ECS	CR102	TEST	A, I
E-153B	1YF404	1E 100A FUSE PNL	ECS	CR102	TEST	A, I

TABLE 3.10-2 (Cont)

P.O. NUMBER	EQUIPMENT TAG NUMBER	DESCRIPTION	MANUFACTURER	BLDG AND ELEV (1)	QUALIFICATION	STANDARDS (2)
E-154	1AD481	20kVa UPS	CYBEREX	CR137	TEST	A, I
E-154	1BD481	20kVa UPS	CYBEREX	CR124	TEST	A, I
E-154	1CD481	20kVa UPS	CYBEREX	CR137	TEST	A, I
E-154	1DD481	20kVa UPS	CYBEREX	CR124	TEST	A, I
E-154	10D485	20kVa UPS	CYBEREX	DG163	TEST	A, I
E-154	10D496	20kVa UPS	CYBEREX	CR137	TEST	A, I
E-154	0AD495	20kVa UPS	CYBEREX	CR137	TEST	A, I
E-154	1AD482	20kVa UPS	CYBEREX	DG163	TEST	A, I
E-154	1BD482	20kVa UPS	CYBEREX	DG163	TEST	A, I
E-154	1CD482	20kVa UPS	CYBEREX	DG163	TEST	A, I
E-154	1DD482	20kVa UPS	CYBEREX	DG163	TEST	A, I
E-154	1AD492	20kVa UPS	CYBEREX	DG163	TEST	A, I
E-154	1BD492	20kVa UPS	CYBEREX	DG163	TEST	A, I
E-154	1CD492	20kVa UPS	CYBEREX	DG163	TEST	A, I

(1) LOCATION

DG = DIESEL GENERATOR AREA
CR = CONTROL AREA
IS = INTAKE STRUCTURE
RB = REACTOR BUILDING

(2) STANDARDS

A - IEEE 344-1975
I - NRC R.G. 1.100, Rev. 1

TABLE 3.10-3

NSSS SEISMIC CATEGORY 1 ELECTRICAL AND INSTRUMENTATION EQUIPMENT QUALIFICATION RESULTS

Equipment	Method	Results
Temperature Elements	The temperature elements are qualified by both dynamic testing and analysis. The applicable standard is IEEE 344-1975.	The temperature elements designated as having an active safety function have been dynamically tested demonstrating qualification. Mounted similar to field conditions, they have been subjected to SRV vibration aging, chugging, seismic, and hydrodynamic loads. Biaxially testing, over the frequency range of 1 to 100 Hz, was accomplished in three mutually perpendicular axes with Test Response Spectra (TRS) enveloping the Required Response Spectra (RRS). The temperature elements maintained their functional and structural integrity during testing. Those elements having a passive safety function were analyzed to show structural integrity when subjected to process pressures and loads in excess of the requirement for their location.
Temperature Switch	The temperature switch is shown to be qualified by an analysis of its structural capability thereby meeting the guidelines of IEEE 344-1975.	The safety fucntion of the temperature switch is passive. Analysis shows that it exceeds its structural requirements when subjected to required seismic and hydrodynamic loads. Calculations indicate a high natural frequency making it a rigid body in the range of interest and its capability far exceeds its stress requirements.
Pressure Transmitters; Differential, Absolute, and Gauge	The transmitters are qualified by dynamic testing meeting the guidelines of IEEE 344-1975.	The transmitters can be subjected to both seismic and hydrodynamic loads during their installed life. Testing in an as-installed condition included random frequency excitation to meet SRV aging, upset and faulted seismic, and chugging requirements. Tests were performed in three mutually perpendicular axes. During testing the transmitters maintained structural integrity and met functional requirements.
Level Transmitters	Level transmitters are shown to be qualified for their application by both analysis and testing. Testing was performed to meet the guidelines of IEEE 344-1975.	The level transmitters have both an active or passive safety function depending on their application. Those transmitters with a passive safety function have been shown to meet structural requirement by analysis. They have natural frequencies higher than the range of interest and have been shown to have structural integrity to withstand the required seismic and dynamic conditions.

TABLE 3.10-3 (Cont)

Equipment	Method	Results
		Those transmitters whose safety function is active were tested in their safety-related operating mode and were continuously monitored. They maintained their structural integrity and met accuracy requirements during testing. Five OBE and one SSE tests were performed in three mutually perpendicular axes. Excitation was applied biaxially over a frequency range of 1 to 100 Hz.
Level Switch	The switches are shown to be qualified for their installed location by testing performed to meet the guidelines of IEEE 344-1975.	The level switch has an active safety function and can be subjected to seismic and hydrodynamic loads during its plant life. Vibration aging, SRV, OBE, SSE, and sine beat testing was performed in three mutually perpendicular areas to levels greater than required for their installed location. During testing the switches met structural and functional requirements.
Pressure Switch	The pressure switch is qualified by dynamic testing to meet the applicable standards of IEEE 344-1975.	This switch has an active safety function and can be subjected to seismic loads during its plant life. Five OBE and one SSE multifrequency, biaxial seismic tests were performed on the switch at levels exceeding the requirements. Excitation was applied in three mutually perpendicular axes. The switch met its functional structural requirements.
Stop Valve Switch	This switch is qualified based on dynamic testing at levels greater than its requirement.	This device has an active safety function and has demonstrated structural and functional integrity when subjected to seismic conditions in excess of the HCGS requirement. It demonstrated no natural frequencies below the Zero Period Acceleration (ZPA) point. Discrete frequency dwells were applied, biaxially, to a maximum level of 6.5 g from 1 to 35 Hz in three mutually perpendicular axes.
Pressure Indicators	The pressure indicators have been qualified by dynamic testing to meet the guidelines of IEEE 334-1975.	Indicators can have an active or passive safety function. The indicators mounted in an as-installed condition were subjected to biaxial random testing over a frequency range of 1 to 250 Hz. Five OBE and one SSE tests were applied in three mutually perpendicular axes. TRS that included both seismic and hydrodynamic loads enveloped the RRS. The indicator maintained structural integrity throughout testing.

Equipment	Method	Results
Insulated Detectors	The detectors have been qualified by dynamic testing to meet the guidelines of IEEE 344-1975.	Detectors have an active safety function and met structural and functional requirements when subjected to seismic testing at amplitudes greater than required. Five OBE and one SSE biaxial random tests were performed in three mutually perpendicular axes over a frequency range of 1 to 100 Hz. Functional performance was demonstrated before, during, and after seismic excitation.
IRM Detector	A combination of test and analysis demonstrates qualification of the detectors for their installed location to meet the guidelines of IEEE 344-1975.	The IRM detector movement during a seismic event is controlled by the fuel bundle and maximum excitation occurs at the natural frequency of the bundle. The detector was tested at discrete frequencies in the horizontal axes and analyzed for vertical loads. Capabilities, both tested and analyzed, exceed the requirements, demonstrating qualification.
Conductivity Element	The conductivity cell was analyzed to withstand seismic loads significantly greater than required to meet the guidelines of IEEE 344-1975.	The safety function of the cell is passive, however, it must maintain its structural integrity. Analysis indicates no resonances in any axis below 100 Hz and the ability to withstand loads more than 15 times greater than required.
Condensing Chamber	This equipment is qualified by analysis to meet the HCGS seismic requirement applying the ASME Boiler and Pressure Vessel Code Section 111.	Stress analysis indicates that the condensing chamber meets the requirements of the ASME code and that the lowest calculated allowable moment reaction exceeds the maximum moment of any condensing chamber installation.

Equipment/Manufacturer Cross-Reference

Equipment	Purchased Part Drawing	Equipment Code(1)	Manufacturer
Temperature Element	133D9679	P	Pyco
Temperature Element	145C3224	A	Pyco
Temperature Element	159C4520	P	Rosemount
Temperature Element	117C3485	P	California Alloy
Temperature Indicator	145C3103	P	Weed, Inc.

Equipment	Purchased Part Drawing	Equipment Code[1]	Manufacturer
Temperature Indicator	169C8974	P	Weed, Inc.
Temperature Switch	157C4629	P	Weed, Inc.
Pressure Transmitter	189C7360	A,P	Rosemount
Pressure Transmitter	163C1563	A	Rosemount
Pressure Transmitter	163C1564	P	Rosemount
Pressure Transmitter	188C7360	A,P	Rosemount
Pressure Indicator	163C1184	A,P	Robert Shaw
Diff Press Indicator	163C1181	P	Robert Shaw
Level Transmitter	188C7360	A	Rosemount
Level Transmitter	184C4775	A	Gould
Level Transmitter	169C8392	A	Rosemount
Level Transmitter	163C1973	P	Rosemount
Level Transmitter	163C1560	P	Rosemount
Level Switch	184C4776	A	Magnetrol
Level Switch	159C4361	A,P	Magnetrol
Pressure Switch	184C4770	A	Barksdale
TSV Limit Switch	163C1303	A	Namco
Detector	237X731	A	GE
Detector	112C3144	A	GE
Detector	163C1154	A	GE
Conductivity Element	163C1544	P	Balsbaugh
Condensing Chamber	204B7269	P	GE

TABLE 3.10-3 (Cont)

Equipment	Purchased Part Drawing	Equipment Code(1)	Manufacturer
Flow Transmitter	188C7360	A	Rosemount
Flow Transmitter	163C1560	P	Rosemount
Flow Transmitter	169C8392	P	Rosemount
Flow Element	169C8733	P	Vickery Simms
Power Supply	184C4571	A	GE
Power Supply	164C5660	A	GE
Diff Press Transmitter	163C1560	P	GE
Diff Press Transmitter	188C7360	A	Rosemount
Master Trip Unit	164C5150	A	Rosemount
Slave Trip Unit	164C5150	A	Rosemount
Voltage Preamp - IRM	163C1262	A	GE
Intermediate Range Monitor	368X102AA	A	GE
Power Range Instr.	328X105AC	A	GE
Log Rad Monitor	304A3700	A	GE
Sq. Root Converter	159C4486	M	Bailer Meter
Summer	159C4659	A	GE
Alarm Unit	159C4660	A	Bailey Meter
Local Panels		A	GE
Control Room Panels		A	GE

(1) P = Passive Class 1E
A = Active Class 1E
M = Manual Class 1E

TABLE 3.10-4

NSSS SEISMIC QUALIFICATION TEST SUMMARY
CLASS 1E CONTROL PANELS AND LOCAL PANELS AND RACKS

Local Panels	Panel qualification is by similarity to tested equivalent panels and devices by SQRT program methodology. The applicable standard is IEEE 344-1975.	The local instrument panels are qualified for structural integrity based upon comparisons of similar panel Test Response Spectra (TRS) to an envelope of Required Response Spectra (RRS) for the Hope Creek Reactor Building. The panels were installed in an equivalent manner to those tested. The panels were tested to a spectrum of loads more severe than the Design Basis Event (DBE) and the inputs consist of seismic (SSE) loads. Multifrequency, biaxial testing was performed by applying five OBE and two SSE level tests in each of three mutually-perpendicular axes. Functional performance and structural integrity were monitored through the test series. In the instance where instruments were not tested on the panel, the response at the device location was determined by multiplying the RRS and Zero Period Acceleration (ZPA) by the amplification factor for that device location on the panel and comparing amplification factor for that device location on the panel and comparing the result with individual instrument test data. Qualification of panels is ensured since the TRS enveloped the RRS, and functional and structural requirements were met.
Control Room Panels	The control room panels are qualified by similarity to tested equivalent panels and devices by SQRT program methodology. The applicable standard is IEEE 344-1975.	Control room panels and essential devices are seismically qualified to the IEEE 344-1975 criteria by comparing these panels to similar panels that have been qualified by test. The panels were tested to a spectrum of loads in the frequency range of 1-45 Hz. The design inputs are seismic (SSE) loads. For individual Class 1E devices in the panels, the malfunction limits obtained by single axis, single frequency testing in the 1-33 Hz range. The design of the panels is representative of a generic GE control room panel design. These control room panels are vertical boards consisting of 2 or more bays. Each bay is 24 or 30 inches wide by 36 inches deep by 90 inches high. Each vertical board is a squared corner enclosure (cabinet) manufactured from 0.18 inch carbon steel with angle iron bracket attached top and bottom. Vertical boards contain steel channels as stiffeners, located generally in the

vicinity of the instrumentation. In some cases, the channel is used as an instrument mounting structure.

Most Hope Creek vertical boards are structurally identical to vertical boards that have been tested to IEEE 344-1975. Three tested vertical boards were used for comparison purposes.

The APRM instrument panels are designed to IEEE 344-2004.

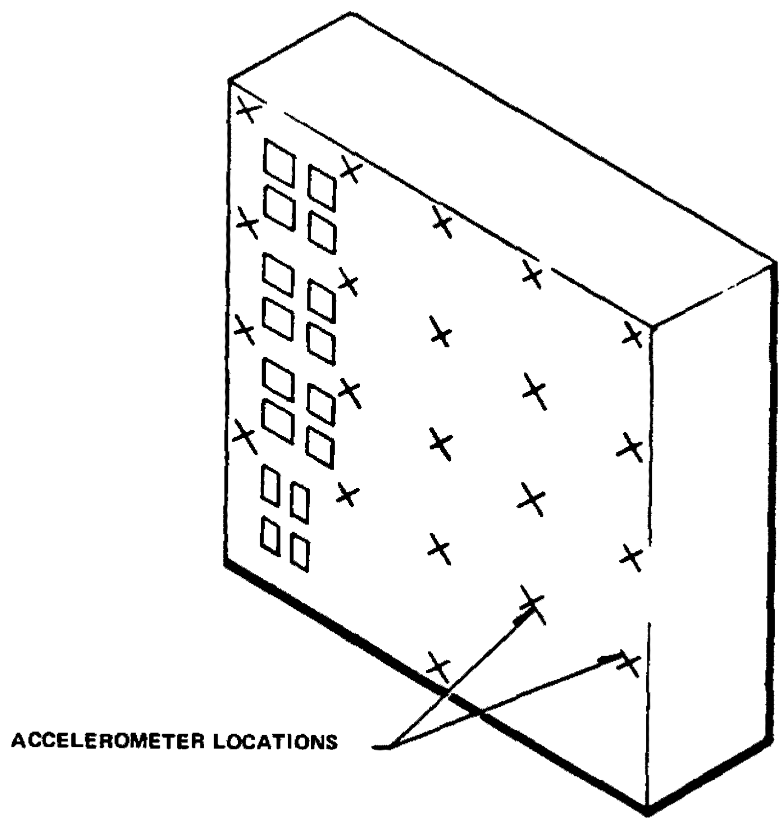
The mechanical characteristics that affect the structural response such as damping, section modulus, stiffness, and mass distribution were considered as a basis for similarity. Theoretical relationships demonstrate that the smaller width structures, heights and depth being constant, have higher natural frequencies and maximum peak accelerations. Higher resonant frequencies indicate a more rigid structure. In a few cases, test data for a given structure deviates from the theoretical. In these cases, examination reveals additional stiffening, more massive equipment or existence of a local mode.

Redundant reactivity control panels C22-P001 and P002 were qualified based upon the satisfactory functional operation of the equipment during testing.

Panel	Description	Type	Class 1E Equipment Description
H11-P608	Power range neutron monitor	Instrument Panel	APRM, QLVPS, 2/4 Logic Module APRM Cal Monitor Panel, Analog Isolator, RRCS Power Supply
H11-P609	Reactor protection system Division 1 & 3 logic	Vertical Board	HFA & HMA relays, CR 105 contractor
H11-P611	Reactor protection system Division 2 & 4 logic	Vertical Board	HFA & HMA relays, CR 105 contractor
H11-P617	Division I RHR relay	Vertical Board	HFA & HMA relays
H11-P618	Division II RHR relay	Vertical Board	HFA & HMA relays
P11-P620	HPCI relay	Vertical Board	HFA & HMA relays
H11-P621	RCIC relays	Vertical Board	HFA & HMA relays

Panel	Description	Type	Equipment Description
H11-P622	Reactor core inboard isolation valve relays	Vertical Board	HFA & HMA relays
H11-P623	Reactor core outboard isolation valve relays	Vertical Board	HFA & HMA relays
H11-P628	ADS Division II relay	Vertical Board	HFA & HMA relays
H11-P631	ADS Division IV relay	Vertical Board	HFA & HMA relays
H11-P635	Division A radiation mon. instrument panel	Instrument Panel	Neutron monitoring electronics
H11-P636	Division B radiation mon. instrument panel	Instrument Panel	Neutron monitoring electronics
H11-P640	Division 4 RHR relay	Vertical Board	HFA & HMA relays
H11-P641	Division 3 RHR relay	Vertical Board	HFA & HMA relays
H21-P001	Core spray A & ADS D Panel	Local Panel	Pressure switches
H21-P002	Reactor water cleanup	Local Panel	Pressure transmitters
H21-P004	Reactor vessel level & press A	Local Panel	Pressure switches, level indicator/transmitters
H21-P005	Reactor vessel level & press C	Local Panel	Pressure switches, level indicator/transmitter
H21-P006	Recirculation pump A	Local Panel	Pressure transmitters
P21-P009	Jet pump A	Local Panel	Pressure transmitters
H21-P010	Jet pump B	Local Panel	Pressure transmitters
H21-P011	Standby liquid control	Local Panel	CR2940 switches
H21-P014	HPCI Panel A	Local Panel	Pressure transmitters/ switches
H21-P015	Main steam flow A/B	Local Panel	Pressure switch
H21-P016	HPCI Panel C	Local Panel	Pressure transmitters/ switches

Panel	Description	Type	Equipment Description
H21-P017	RCIC division panel B	Local Panel	Pressure transmitter/ switches
H21-P018	RHR A panel	Local Panel	Pressure transmitter/ switch
H21-P019	Core spray B & ADS B	Local Panel	Pressure transmitter/ switch
H21-P021	RHR B Panel	Local Panel	Pressure transmitter/ switch
H21-P022	Recirculation pump B	Local Panel	Pressure transmitters
H21-P025	Main steam flow C/D	Local Panel	Pressure switches
H21-P026	Reactor vessel level & Press D	Local Panel	Pressure switches/ transmitters
H21-P027	Reactor vessel level & Press B	Local Panel	Pressure switches/ transmitters
H21-P030	SRM & IRM preamp A-D	NEMA 12 enclosures	SRM-IRM preamplifiers
H21-P031	SRM & IRM preamp A-D	NEMA 12 enclosures	SRM-IRM preamplifiers
H21-P032	SRM & IRM preamp A-D	NEMA 12 enclosures	SRM-IRM preamplifiers
H21-P033	SRM & IRM preamp A-D	NEMA 12 enclosures	SRM-IRM preamplifiers
H21-P037	RCIC Panel D	Local Panel	Pressure Transmitter/Switches
H21-P041	Main steam flow A/B	Local Panel	Transmitters
H21-P042	Main steam flow C/D	Local Panel	Transmitters
H21-P055	RHR C panel	Local Panel	Pressure transmitter/switch
H21-P069	RHR D panel	Local Panel	Pressure transmitter/switch



(BENCHBOARD WOULD BE THE SAME WITH A BENCH SECTION PROTRUDING OUT ABOUT HALF-WAY DOWN)

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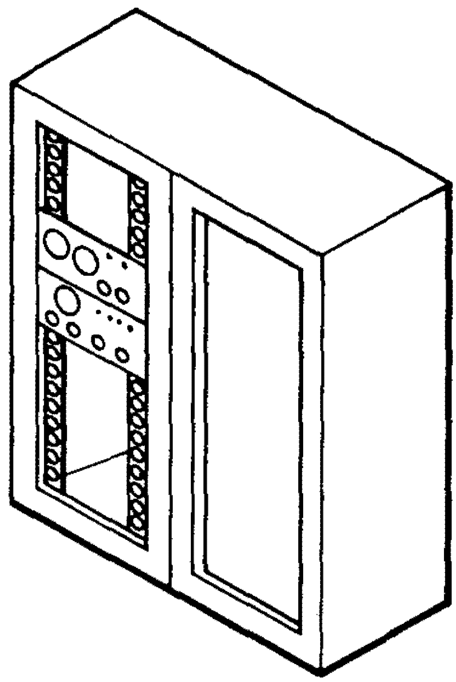
APRIL 11, 1988

PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

TYPICAL VERTICAL BOARD

UPDATED FSAR

FIGURE 3.10-1



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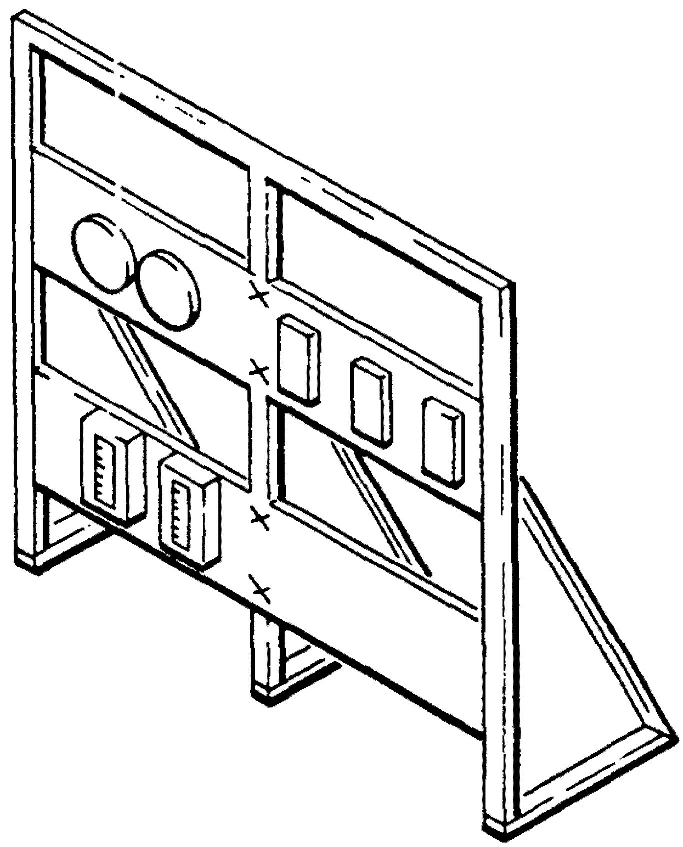
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

INSTRUMENT RACK
(CABINET CONTAINS PAGES OR OTHER SPECIAL INSTRUMENTS INSTEAD OF ONLY DRAWER INSTRUMENTS)

UPDATED FSAR

FIGURE 3.10-2



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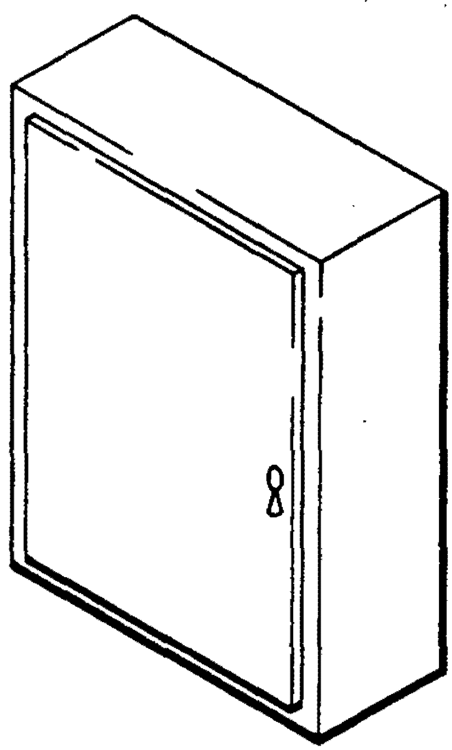
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

TYPICAL LOCAL RACK
(PIPING AND OTHER EXTERNAL CONNECTIONS NOT SHOWN)

UPDATED FSAR

FIGURE 3.10-3



(INSTRUMENTS MOUNTED INSIDE ON INTERNAL MEMBRANE
MOUNTED ON STANDOFFS ATTACHED TO BACK)

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PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

NEMA TYPE-12 ENCLOSURE

UPDATED FSAR

FIGURE 3.10-4

3.11 ENVIRONMENTAL DESIGN OF MECHANICAL AND ELECTRICAL EQUIPMENT

Engineered Safety Feature (ESF) Systems, including the Reactor Protection System (RPS), are provided with safety-related equipment, which is installed in accordance with mechanical and electrical separation requirements, and designed to function properly in the following service environments:

1. For all normal and abnormal environmental design conditions, including the maximum and minimum limits for temperature, pressure, relative humidity, and radiation (gamma, beta, and neutron).

   Normal environmental conditions are defined as those conditions existing during routine plant operations, including startup, shutdown, refueling, and maintenance operations.

   Abnormal conditions are defined as those that can be experienced at different times during the operating lifetime of the plant. These abnormal conditions are anticipated operational occurrences, such as loss of offsite power (LOP), and should not be interpreted to be accident conditions.

2. In addition to the normal and abnormal operation environmental design bases stated above, the safety-related ESF equipment is designed to perform its safety function during exposure to the applicable design basis event (DBE) environment present in its operational area and remain in a safe mode after its safety function is performed.

   The DBE conditions are those that would exist as a result of an accident.

Environmental design criteria for the design of mechanical and electrical components of the ESF systems conform to 10CFR50, Appendix A, GDC 1, Quality Standards and Records; GDC 2, Design Bases for Protection Against Natural Phenomena; GDC 4, Environmental and Missile Design Bases; GDC 23, Protection Systems Failure Modes; GDC 50, Containment Design Basis; 10CFR50, Appendix B, Section XI, XVII; and 10CFR50.49.

3.11.1 Equipment Identification and Environmental Conditions

3.11.1.1 <u>Nuclear Steam Supply System (NSSS) Class 1E Equipment Identification</u>

3.11.1.1.1 NSSS Class 1E Equipment

NSSS safety-related systems and equipment requiring qualification is qualified to perform its design safety functions in normal, abnormal, accident, and post-accident environments, as applicable. SAP database identifies NSSS Class IE equipment requiring qualification.

3.11.1.1.2 NSSS Class 1E Equipment Environments

The Hope Creek Environmental Design Criteria (EDC), Document no. D7.5 (Ref. 3.11-5) provides the environmental conditions by plant zones during normal, abnormal, accident, and post-accident conditions. Information concerning the temperature, pressure, relative humidity, and radiation is provided. Tables 1,2,3,5 and 6 cover all locations which contain Class 1E NSSS equipment.

3.11.1.2 <u>Non-NSSS Class 1E Equipment</u>

Qualification of the ESF components to the applicable environmental conditions is provided to fulfill the following design criteria:

1. For normal and abnormal plant operations, ESF components required to prevent, limit, or mitigate the consequences of a DBE and affect a safe shutdown will be qualified to remain functional during exposure to the combined following environmental conditions:

   Enveloping values of the radiation, temperatures, pressures, and relative humidities at the component location during normal and abnormal operation are given in Tables 1,2,3,5 and 6 of EDC D7.5.

2. In addition to the above, ESF components required to mitigate the consequences of a DBE and affect a safe shutdown are designed to accomplish their design safety function during exposure to the applicable accident environmental conditions. The enveloping environmental conditions are those anticipated to follow a DBE and are listed below:

   a. The enveloping conditions that components inside primary containment may be exposed to following a DBE are provided in the tables and figures contained in EDC D7.5. The DBE-TID inside primary containment is calculated using AST as described in RG 1.183, Appendix I. The TID is calculated for a period of 100 days, at which time the dose has essentially saturated. The service conditions are established in accordance with NUREG-0588.

   b. The enveloping conditions that components may be exposed to outside primary containment, but inside the Reactor Building, following a DBE are provided in the tables and figures contained in EDC D7.5. The DBE-TID outside primary containment is calculated for contributions from two sources. The first assumes that 50 percent of the

core halogen inventory and 1 percent of the core solid fission product inventory are contained in the Emergency Core Cooling System (ECCS) water after an accident. The second source is from the airborne cloud that has leaked from the primary containment. The TID is calculated for a period of 180 days, at which time it has essentially saturated. The service conditions are established in accordance with NUREG-0588. These values are considered bounding and are not analyzed for AST.

c. The enveloping conditions that components outside the Reactor Building may be exposed to are shown in the table and figures contained in EDC D7.5. Environmental conditions will not vary significantly from those experienced during normal or abnormal plant operations.

   Radiation exposures to components will be minor and will be due to two sources: radiation shine, and immersion in airborne radioactivity released in a controlled manner from the Reactor Building. The TID from both sources will be less than 100 rads for 180 days.

SAP database identifies non-NSSS Class IE equipment requiring qualification.

3.11.1.3 Excluded Systems - NSSS and Non-NSSS

Systems containing active or passive mechanical equipment, non-1E systems, or systems located in mild environment that are designated as Seismic Category I in Table 3.2-1 are excluded from the HCGS Environmental Qualification Program.

3.11.1.4 Environmental Conditions - NSSS and Non-NSSS

The environmental conditions shown in the tables and the associated figures of EDC D7.5 may change from time to time because of evaluations that may be performed on a case by case basis for specific plant operating conditions. The maximum outdoor design ambient temperature at which these compartment temperatures can be maintained is 94°FDB/78°FWB. This outside air temperature is the same for each HVAC system.

The environmental conditions that components may be exposed to in areas outside primary containment, but inside the Reactor Building, are not capable of becoming harsh until after initial criticality is achieved.

3.11.2 Qualification Tests and Analyses

3.11.2.1 NSSS Safety-Related Class 1E Electrical Equipment Harsh Environment Qualification

Components of the Nuclear Steam Supply System (NSSS) Class 1E electrical equipment are qualified in accordance with the environmental qualification criteria and guidelines specified as Category II in NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment," dated December 1979 (for comment), and IEEE-323-1971, "IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations." However, the HCGS environmental qualification program has upgraded qualification of equipment to the requirements of NUREG-0588, Category I and IEEE-323-1974.

Components of the NSSS Class 1E electrical equipment in a harsh environment are qualified by test, or a combination of testing and analysis. Those components used in more than one system, which can be or are located in different plant areas, are qualified for the worst environmental conditions in which they are required to

function. Design limits for critical parameters for different applications are considered in the test procedures. Licensing Topical Report (NEDE-24326-1-P) presents information on a portion of the upgrade effort.

The identification, location, and conditions to which NSSS supplied Class 1E equipment is required to function are shown in a separate environmental qualification report.

See References 3.11-2 and 3.11-3 for additional information regarding equipment qualification compliance.

3.11.2.2 Regulatory Guides and General Design Criteria - NSSS Equipment

All components of safety-related NSSS equipment are tested or analyzed to meet the requirements of 10CFR50, Appendix A, GDC 1, 2, 4, 23, and 50.

3.11.2.2.1 Criterion 1 - Quality Standards and Records

Satisfaction of criterion 1 is achieved by reviews to ensure that tests or analyses conform to the design, procurement, fabrication, quality assurance, and environmental qualification documentation. Refer to Chapter 17 for further definition of the quality assurance program.

3.11.2.2.2 Criterion 4 - Environmental and Missile Design Bases

The environmental requirements of criterion 4 are addressed in this section, and considerations relating to missiles are addressed in Section 3.5.

3.11.2.2.3 Criterion 23 - Protection System Failure Modes

Compliance with criterion 23 is addressed in Section 7.

3.11.2.2.4 Criterion 50 - Containment Design Bases

Compliance with criterion 50 is discussed in Section 6.

3.11.2.2.5 Regulatory Guide 1.30, Quality Assurance Requirements for Instrumentation and Electrical Equipment

Although Regulatory Guide 1.30 is not part of the NSSS design basis, the extent of conformance is provided in Section 7.

3.11.2.2.6 Regulatory Guide 1.40, Qualification Tests of Continuous Duty Motors Installed Inside the Containment of Water Cooled Nuclear Power Plants

Regulatory Guide 1.40 is not applicable to the HCGS because there are no NSSS supplied continuous duty, safety-related motors inside the primary containment.

3.11.2.2.7 Regulatory Guide 1.73 - Qualification Tests of Electrical Valve Operators Installed Inside the Containment of Nuclear Power Plants

Regulatory Guide 1.73 is not applicable to HCGS because there are no NSSS supplied electric motor operated valves inside the primary containment.

3.11.2.2.8 Regulatory Guide 1.89 - Qualification of Class 1E Equipment for Nuclear Power Plants

Hope Creek Generating Station will attempt to comply on a case by case basis.

3.11.2.3 Non-NSSS Class 1E Electrical Equipment Harsh Environment Qualification

Harsh environment qualification of Class 1E equipment is accomplished by test or analysis (where analysis is supported by test data or otherwise justifiable) for the applicable environmental conditions postulated to exist at the equipment location. These components are qualified to the requirements of IEEE-323-1974 and NUREG-0588, Category I.

The identification, location, and conditions to which non-NSSS supplied 1E equipment is required to function are shown in a separate environmental qualification report.

3.11.2.4 Regulatory Guides and General Design Criteria - Non-NSSS Equipment

The requirements of GDC 1, 2, 4, 23, and 50 of Appendix A to 10CFR50 are met, as discussed in Section 3.1. The enveloping environmental conditions that the equipment may be exposed to are discussed in Section 3.11.1. Conformance to the criteria of Appendix B to 10CFR50 is discussed in Section 17.

Section 1.8 discusses the extent of compliance with Regulatory Guides 1.30, 1.40, 1.63, 1.73, 1.89, and 1.131.

3.11.2.5 NSSS and Non-NSSS Class 1E Equipment Located in a Mild Environment

Mild environment equipment is that equipment which, during or after a design basis event (DBE), does not experience an environment that is significantly more severe than that existing during normal and abnormal events. Additionally, equipment that experiences the environment of a DBE can be treated as if it were

in a mild environment if the device falls into either of the following categories:

1. The equipment accomplishes its safety function prior to experiencing the environment of the DBE, and the equipment will not fail in a manner detrimental to plant safety.

2. The equipment is not needed to mitigate the DBE and the equipment will not fail in a manner detrimental to plant safety.

The vendors of mild environment equipment have been requested to submit a certificate of compliance certifying that the equipment will perform its required safety function in its applicable environment.

In addition, a surveillance and maintenance program will be developed to ensure equipment operability during its design life.

3.11.2.6 Deleted

3.11.2.7 Qualification Methods for NSSS and Non-NSSS Safety-Related Electrical Equipment

3.11.2.7.1 Margin

IEEE-323 and NUREG-0588, Rev. 1, Paragraph 3.(4) are used as a basis for determining margin. The equipment technical specification for safety-related electrical equipment required to be environmentally qualified include conservative environmental conditions which were derived using environmental parameters which contain conservatisms applied during the derivation of local environmental conditions. The equipment vendor determines what margin must be added to allow for variations in production processes, for inaccuracies in the test equipment and for errors associated with defining satisfactory performance.

The qualification documentation for safety-related equipment will include documented provision that adequate margin has been applied. Methods of allowing for margin will be included, such as the following:

1. Increasing the levels of testing

2. Increasing the number of test cycles

3. Increasing test duration

4. Increasing the number of stress reversals in analysis

5. Increasing or decreasing (e.g., line voltage) service conditions to provide more severe conditions.

3.11.2.7.2 Aging

IEEE-323 is used as a basis for establishing an aging program. Safety-related electrical equipment that must meet aging requirements is subjected to an aging program designed to place the equipment in its end of qualified life condition before performing the functional tests under the DBE and post-DBE environments. The significant aging mechanisms expected to be present during actual service are identified. Age-related degradation resulting from exposure to elevated temperature, radiation and cyclical mechanical and electrical stresses under specified normal, abnormal (excluding DBE) and test conditions anticipated during the installed life of the equipment, are considered. If it is demonstrated that no age related failure mechanisms exist that can impair the product's ability to perform its safety function throughout its installed life, the equipment to be tested can be considered exempt from age conditioning.

The Arrhenius methodology is considered an acceptable method of addressing accelerated thermal aging. Other aging may be considered appropriate if justified for the application.

The need for aging of a particular piece of equipment is determined based on an evaluation of the specific design and application.

3.11.2.7.3 Dose Rate

NUREG-0588 and AST are used as a basis for establishing radiation exposure. Safety-related electrical equipment that may be exposed to radiation is irradiated to simulate the effects of this exposure. The subject equipment is exposed to radiation levels anticipated during the design life of the equipment, including the DBE condition and radiation from recirculating fluids. The radiation dose inside the Reactor Building is not reanalyzed using AST since these values are conservative.

The exposure rate and source(s) is selected to produce degradation representative of the anticipated service. The radiation specified for normal, abnormal, DBE and post-DBE exposure may be combined in one test. The Tables in EDC D7.5 list the dose rates for normal, abnormal and DBE conditions for equipment and areas.

3.11.2.7.4 Synergistic Effects

NUREG-0588, Rev. 1, Paragraph 4(3), is used as a basis for determining synergistic effects. The aging process accounts for the occurrences of known synergistic effects that may occur. Any identified synergistic effects are accounted for in the qualification program. The results of the evaluation and determination will be justified and documented.

3.11.2.7.5 Use of Analysis for Qualification

IEEE-323 is used as a guide for qualification of safety-related electrical equipment by analysis. Analysis may be used to supplement type testing. Such analysis performed for full environmental qualification or to supplement the testing is justified in qualification documentation. When possible, partial

test data is used to support the analytical assumptions and conclusions. The analytical techniques that may be used are similarity, extrapolation, and mathematical modeling.

3.11.2.7.6 Maintenance/Surveillance Program

A maintenance/surveillance program in compliance with 10CFR50.49 and Regulatory Guide 1.33 will be established for the HCGS.

The maintenance/surveillance program for environmentally qualified electrical equipment will encompass equipment located in both harsh and mild environments. To provide the necessary controls the following approach will be used:

1. Master Equipment List

   A master equipment list for safety-related electrical equipment will be used as the basis. Equipment located in a harsh environment will be identified and designated on this list. This will result in the safety-related equipment list being divided into two categories:

   a. Safety-related electrical equipment located in a harsh environment.

   b. Safety-related electrical equipment located in a mild environment.

2. Components in a Harsh Environment

   These components have been identified and fall into two categories:

   a. Components with a qualified 40 year life which require no replacement parts or planned maintenance for their stated life.

b. Components with a qualified life of 40 years subject to replacement of noted parts, or components requiring replacement in total in less than 40 years. The EQ program will provide the information, on a real time basis, to notify the station maintenance/surveillance personnel as to both the required action and the date by which the action must be complete to meet the 40 year life criteria or total replacement criteria.

3. Components in a Mild Environment

These components will be compiled from a master equipment list for safety-related electrical equipment. Component operability will be demonstrated via normal operating procedures, Technical Specification requirements, or specified maintenance/surveillance requirements, which result from the equipment qualification program.

3.11.3 Qualification Test Results

Environmental qualification documentation for safety-related electrical equipment has been available for NRC audit.

3.11.4 Loss of Ventilation

The maximum indoor temperatures used for the sizing of air conditioning systems serving safety-related systems are determined by additive analysis of the following factors:

1. Outdoor design temperatures for the geographical location of the plant (both wet and dry bulb values).

2. Maximum piping thermal loads, if applicable, using maximum operating temperatures for the pipe contents and maximum footage of active pipe for each mode of operation.

3. Maximum electrical loads, assuming full lighting, and using, if applicable, the maximum panel and equipment resistance losses for each mode of operation.

4. Maximum heat release from operating equipment, if applicable (for example, the diesel generator, electric motors, heat exchangers, etc).

5. Maximum heat and humidity transfer from the surface of open pools and tanks, if applicable, using the maximum operating temperature of the contents for each mode of operation.

6. Maximum heat transfer through the space envelope, including walls, floor, and ceiling or roof (individual values may be negative).

7. Maximum heat and humidity releases resulting from assumed values for fluid leakage rates, if applicable.

Seismic Category I air conditioning and ventilation systems described in Section 9.4 are powered from the Class 1E electrical power supplies. The environmental conditions to which safety-related equipment are qualified are described in Section 3.11.2.

These Seismic Category I air conditioning and ventilation systems are so designed that the single failure of an active mechanical component, or an active or passive electrical component, after a design basis event (DBE), cannot impair the ability of other safety-related systems to fulfill their safety functions. If a train in a Seismic Category I air conditioning system becomes inoperative redundant equipment is available to fulfill the safety function and mitigate the consequences of a DBE.

Engineered safety feature (ESF) and Reactor Protection System (RPS) instrumentation that is not served by a Seismic Category I ventilation system is designed for continued operation in the event of failure of the normal ventilation system.

Seismic Category I air conditioning trains provide cooling to areas containing safety-related equipment. Nonsafety-related equipment rooms and corridors through which power and instrumentation cables are routed are not cooled by Seismic Category I air conditioning trains since, upon loss of ventilation, temperatures there will be lower than cable design requirements.

Power cable insulation is designed for a conductor temperature of 194°F (90°C) based on an EQ service life of 40 years. The allowable current carrying capacity of the cable is based on not exceeding this conductor design temperature while the surrounding air during normal plant conditions is at a maximum

temperature of 150°F (65.5°C) inside the primary containment and the Reactor Building and 104°F (40°C) to 122°F (50°C) for the rest of the plant.

Instrumentation cable is designed for a conductor temperature of 194°F (90°C) based on an EQ service life of 40 years. Operating currents of these cables are low and will not cause this temperature to be exceeded at maximum design ambient air temperature.

3.11.5 Estimated Radiation Environment

For each location, environmental conditions (including radiation levels) are provided in the environmental qualification summary report. The applicant will correlate the data with Tables 1,2,5,6 and 8 of EDC D7.5 and describe how these environments were derived.

3.11.6 SRP Rule Review

3.11.6.1 <u>Acceptance Criteria Deviation 2</u>

In SRP Section 3.11, Subsection II, it states that complete and auditable records will be available at the time of OL application. The files will not be available at the time of OL application, but should be available in time for the environmental qualification audit prior to fuel load.

The records will be complete and available for audit prior to fuel load.

3.11.6.2 <u>Acceptance Criteria Deviation 3</u>

In SRP Section 3.11, Subsection II, it states that, at the time of the CP and OL application, complete and auditable records which describe the environmental qualification method used for all mechanical and electrical equipment in sufficient detail to document the degree of compliance with the requirements of the SRP, must be available and maintained at a central location.

3.11.6.3 Acceptance Criteria Deviation 4

SRP Section 3.11, Subsection II, lists IEEE-323 (augmented by Regulatory Guide 1.89) as acceptance criteria. Where the HCGS environmental qualification program is upgraded to NUREG-0588, Category I, the guidance contained in Regulatory Guide 1.89 is followed.

3.11.7 References

3.11-1 Deleted

3.11-2 R.L. Mittl, PSE&G, to W. Butler, NRC, "Equipment Qualification", dated July 29, 31, and August 15, 1985.

3.11-3 C. McNeill, PSE&G, to E. Adensam, NRC, "Equipment Qualification," dated December 4, 1985, January 7, February 25, and March 26 and 31, 1986.

3.11-4 Deleted

3.11-5 Hope Creek Generating Station Environmental Design Criteria, Document No. D7.5.

Tables 3.11-1a through 3.11-1h, Table 3.11-2, and Figures 3.11-1 through 3.11-5 have been deleted and information on the Environmental Qualification parameters for Class 1E equipment have been provided separately in the "Environmental Design Criteria", Document No. D7.5.

Tables 3.11-4, 3.11-5, and 3.11-6 have been deleted and information on the environmental qualification of Class 1E equipment is identified in SAP.

TABLE 3-11.1

TABLE 3.11-1A THROUGH TABLE 3.11-1H
HAVE BEEN INTENTIONALLY DELETED

TABLE 3-11.2

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TABLE 3.11-3

SUMMARY OF HCGS COMPLIANCE WITH 10CFR50.49

This table represents a summary of Hope Creek Generating Station compliance with 10CFR50.49.

Paragraph(a)	-	Requirement incorporated. A program has been established for qualification of electric equipment in a harsh environment that is safety-related. The present program will be discussed in detail by an EQ summary report as referenced in Section 3.11.
Paragraph(b) (1)	-	Requirements incorporated. Safety-related electrical equipment, needed to mitigate design basis events, has been identified, designed and has been qualified to function properly in the environmental conditions during normal, abnormal and design basis events.
Paragraph(b)	-	Requirement incorporated. To date no nonsafety-related equipment in this category has been identified.
Paragraph(b) (3)	-	Requirement incorporated. The parameters required to be measured by Regulatory Guide 1.97 are included to the extent noted in Section 1.8.1.97. Equipment required by Regulatory Guide 1.97 to be environmentally qualified has been included in the equipment qualification program.

Paragraph(c)	-	No requirement. This section details items (mild environment, seismic qualification, etc.) that are not included within the scope of this rule.
Paragraph(d)	-	Requirement incorporated. SAP database identifies safety-related electric equipment located in a harsh environment.
Paragraph(d) (1)	-	Requirement incorporated. The equipment evaluation summary (EES) sheets in the EQ Binders provide this information.
Paragraph(d) (2)	-	Requirement incorporated. Equipment test reports provide this information.
Paragraph(d) (3)	-	Requirement incorporated. The EES sheets in the EQ Binders provide this information.
Paragraph(e) (1)	-	Requirement incorporated. Section 3.11 discusses the design basis including temperature and pressure. A plant specific profile for temperature and pressure vs. time for equipment qualification has been included in the Environmental Design Criteria DITS D7.5. Temperature and pressure limits are included on the EES's and in the tables included in D7.5.

TABLE 3.11-3 (Cont)

Paragraph(e) (2) -	Requirement incorporated. Humidity has been considered where it is applicable and is included on the EES's and in the tables included in D7.5.
Paragraph(e) (3) -	Requirement incorporated. Chemical effects are not applicable since demineralized water is used. Effects on demineralized spray are encompassed by testing at 100 percent relative humidity. Equipment subjected to direct spray impingement has been evaluated to determine if testing under spray conditions in addition to 100 percent relative humidity conditions is required.
Paragraph(e) (4) -	Requirement incorporated. Radiation effects on safety-related electrical equipment have been taken into account, where applicable, including radiation resulting from recirculating fluids. Radiation levels are included on the EES's and in the tables included in D7.5.
Paragraph(e) (5) -	Requirement incorporated. Aging is included as part of equipment qualification except where equipment is not considered to be age sensitive. Qualified life is included in the EESs.
Paragraph(e) (6) -	Requirement incorporated. Equipment that could be submerged has been identified and demonstrated to be qualified by test for the duration required.

Paragraph(e) (7) -	Requirement incorporated. Synergistic effects have been considered in the accelerated aging programs. An engineering evaluation will be performed to identify known synergistic effects for materials that are included in the equipment qualified. Any identified synergistic effects are accounted for in the qualification programs. Section 3.11.2.7.4 discusses the design basis for synergistic effects.
Paragraph(e) (8) -	Requirement incorporated. The equipment technical specification includes the margin in the environmental conditions of the plant and the margin to be applied to service conditions. Section 3.11.2.7.1 discusses margins as part of the design basis.
Paragraph(f) (1-4) -	Requirement incorporated. Section 3.11.2.1 and 3.11.2.3 discusses performance of environmental qualification.
Paragraph(g) -	N/A. Pertains to plants receiving operating licenses prior to February 22, 1983.
Paragraph(h) -	N/A. Pertains to plants receiving operating licenses prior to February 22, 1983.
Paragraph(i) -	N/A. Pertains to plants receiving operating licenses prior to November 30, 1985.
Paragraph(j) -	Requirement incorporated. Section 3.11.3 states that environmental qualification documentation for safety-related electrical equipment will be available for NRC audit.

Paragraph(k) - No requirement. This section permits applicants for, and holders of, operating licenses exemption from this rule if the Commission previously required qualification of equipment in accordance with DOR guidelines or NUREG-0588.

Paragraph(l) - Requirement incorporated. Replacement equipment will be qualified in accordance with 10CFR50.49.

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PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK GENERATING STATION

PRIMARY CONTAINMENT ZONES

Updated FSAR
Revision 5, May 11, 1993

Sheet 1 of 1
Figure 3.11-1

"THIS FIGURE HAS BEEN DELETED"

PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK GENERATING STATION

PRIMARY CONTAINMENT ENVELOPING TEMPERATURE RESPONSE- LOSS OF OFFSITE POWER

Updated FSAR	Sheet 1 of 1
Revision 5, May 11, 1993	Figure 3.11-2

"THIS FIGURE HAS BEEN DELETED"

PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK GENERATING STATION

PRIMARY CONTAINMENT ENVELOPING PRESSURE RESPONSE- LOSS OF OFFSITE POWER

Updated FSAR
Revision 5, May 11, 1993

Sheet 1 of 1
Figure 3.11-3

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PUBLIC SERVICE ELECTRIC AND GAS COMPANY HOPE CREEK GENERATING STATION	
ENVELOPING PRIMARY CONTAINMENT TEMPERATURE RESPONSE-DBE	
Updated FSAR Revision 5, May 11, 1993	Sheet 1 of 1 Figure 3.11-4

"THIS FIGURE HAS BEEN DELETED"

PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK GENERATING STATION

ENVELOPING PRIMARY CONTAINMENT PRESSURE RESPONSE-DBE

Updated FSAR
Revision 5, May 11, 1993

Sheet 1 of 1
Figure 3.11-5