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 <u>Group IV - Fluid Systems</u>

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 Group VI - Fuel and Radioactivity Control

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

The nuclear power unit design shall include means to 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	

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)

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
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	
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	(48)
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	
4. HPCI turbine		GE	C	NA	VIII-1	I	Y	(11)(59)
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)

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
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. Reactor Core Isolation Cooling (RCIC) System:	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. Fuel Handling and Storage								
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)

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	

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
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)

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	

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
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	

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. Heating, Ventilating, and Air Conditioning Systems								
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	(16)
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)(48)
j) Valves		P	C	C	III-3	I	Y	
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. Main Steam and Power Conversion Systems								
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	

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. <u>Instrumentation and Control Systems</u>								
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	

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
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	I	Y	(15)
d) Cable, IRM and APRM, with safety function		P	A,C	NA	IEEE-279/323	NA	Y	
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								(51)
9. Area Radiation Monitoring Systems		P	All	NA	None	NA	N	
10. Deleted								
11. Reactor Water Cleanup System		GE	C	NA	None	NA	N	
12. Radwaste systems		P	R	NA	None	NA	N	(51)
13. Fuel Pool Cooling System		P	C	NA	None	NA	Y	
14. Fuel Pool Cleanup System		P	C	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
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. <u>Electric System</u>								
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	

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)

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
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	

(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-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 exchangers	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

		WALL LOAD(2)		
Height Zone (ft)	Dynamic Pressure, q(1) (psf)	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

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.

<u>Failure of Cooling Tower Basin Wall</u> (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 <u>Means of Protecting Safety-Related Components From Internal Flood</u>

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 <u>Permanent Dewatering Systems</u>

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:

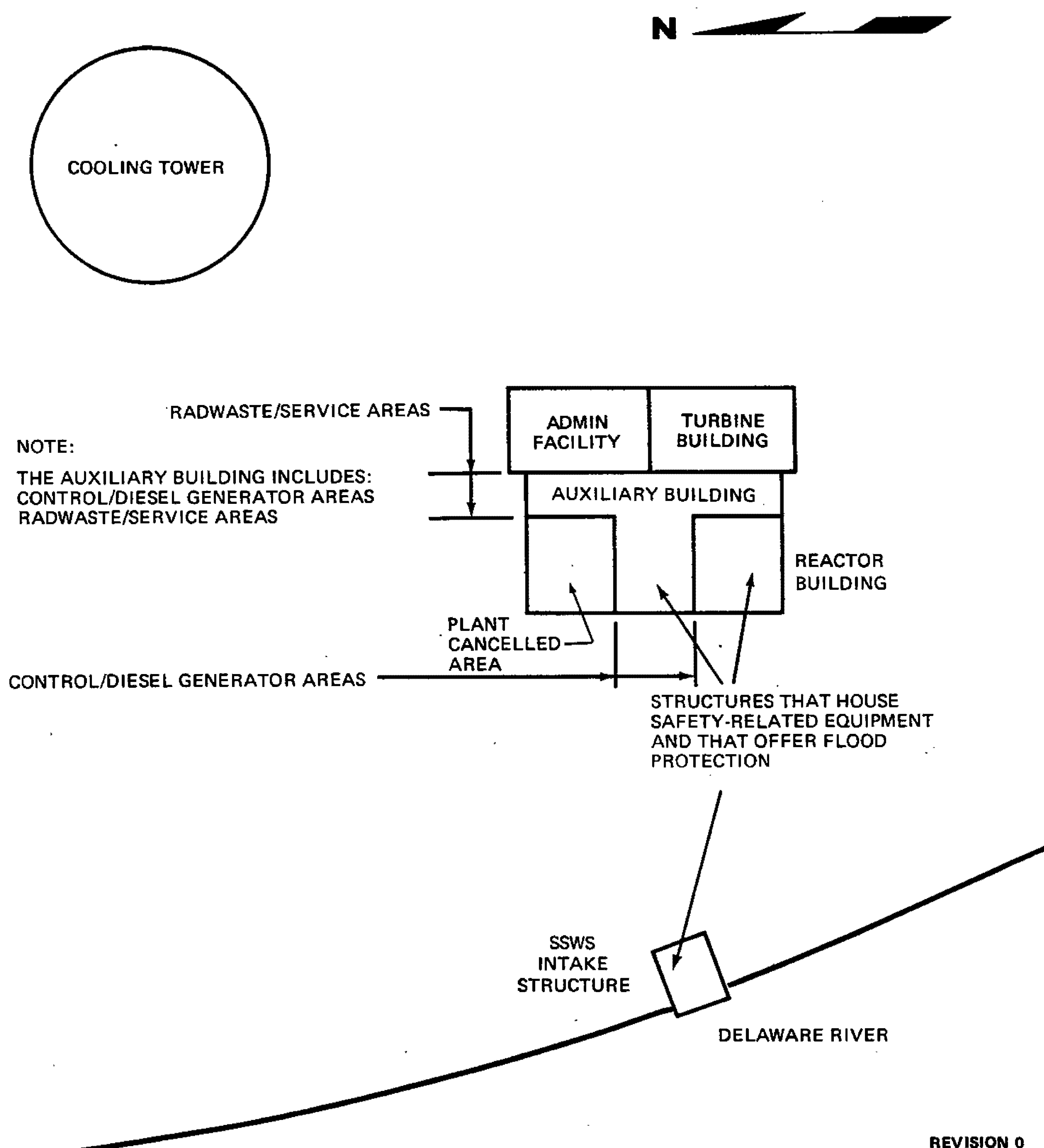
(1) The invert elevation of openings and the crest elevation of slots and parapet openings are 6 inches above the roof drain elevation.

(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



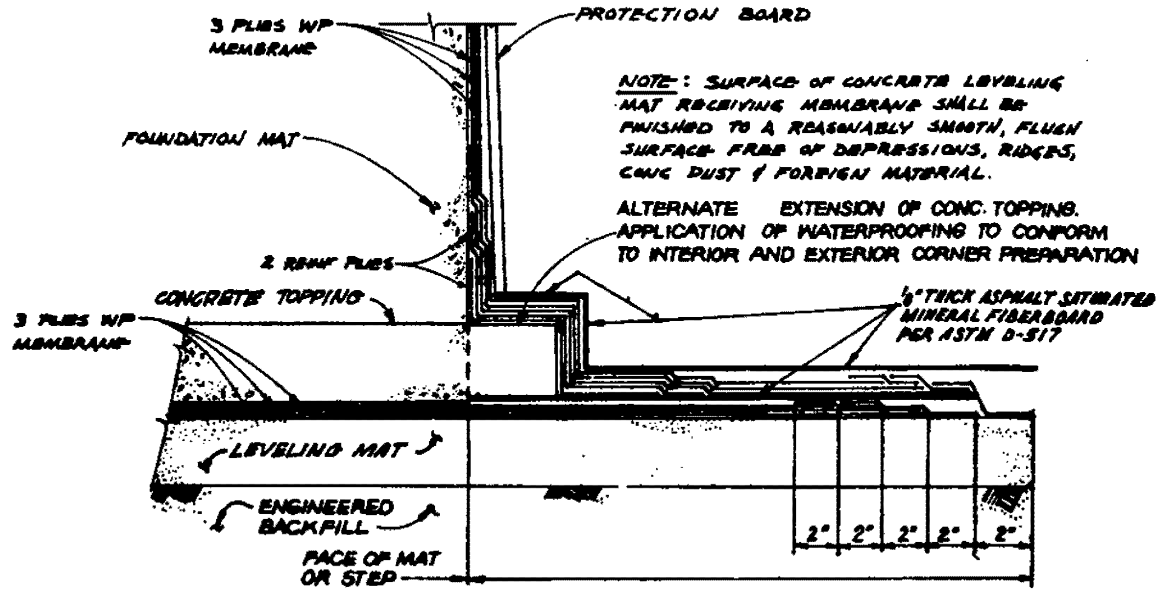
REVISION 0
APRIL 11, 1988

PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

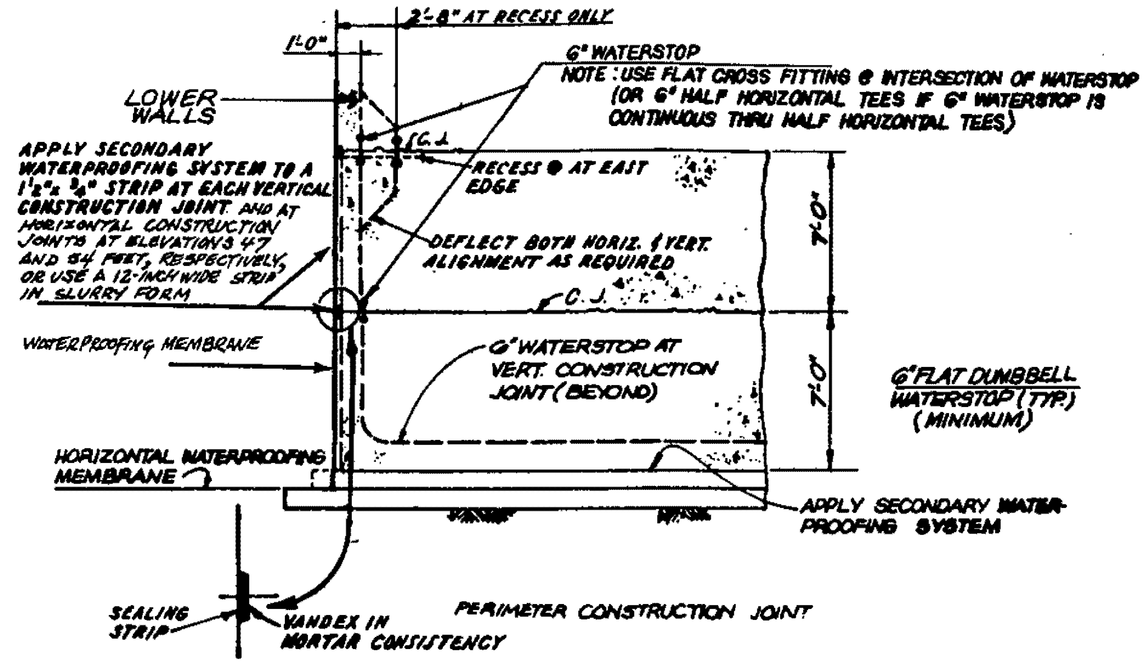
STRUCTURES THAT PROVIDE FLOOD PROTECTION

UPDATED FSAR

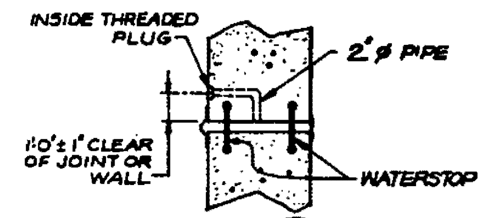
FIGURE 3.4-1



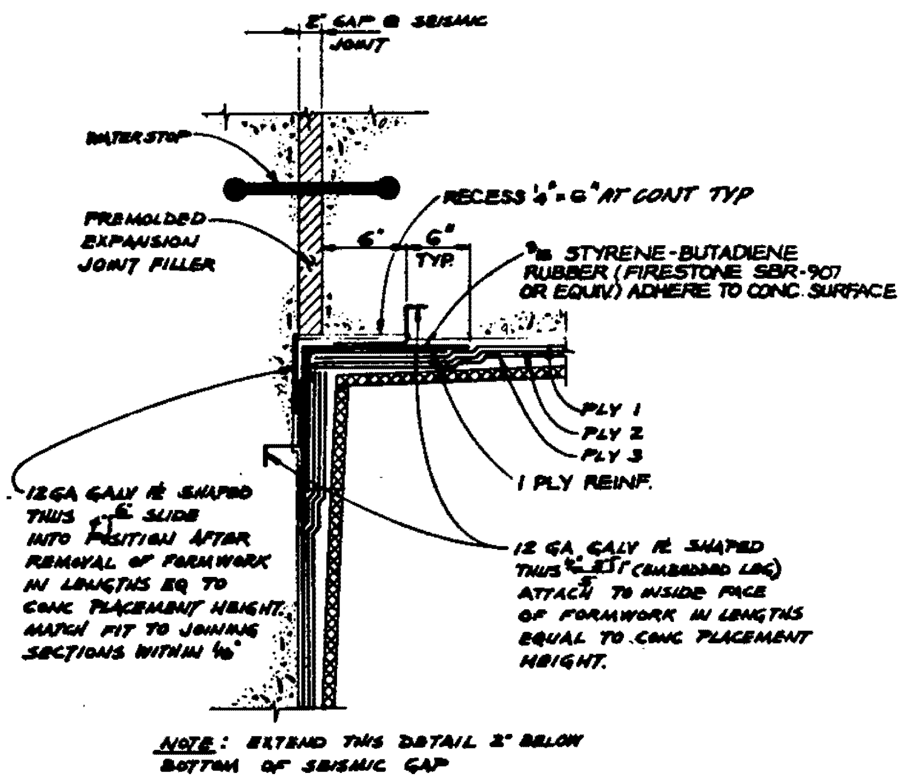
TYPICAL HORIZONTAL WATERPROOFING APPLICATION & EXTERIOR TERMINATION



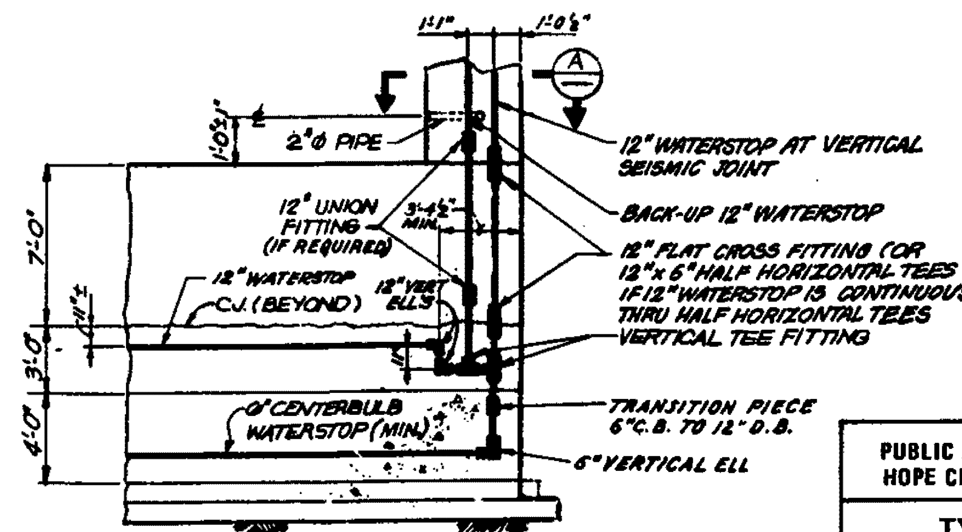
TYPICAL CONSTRUCTION JOINT DETAILS



SECTION A
N.T.S.



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



TYPICAL SEISMIC JOINT DETAILS

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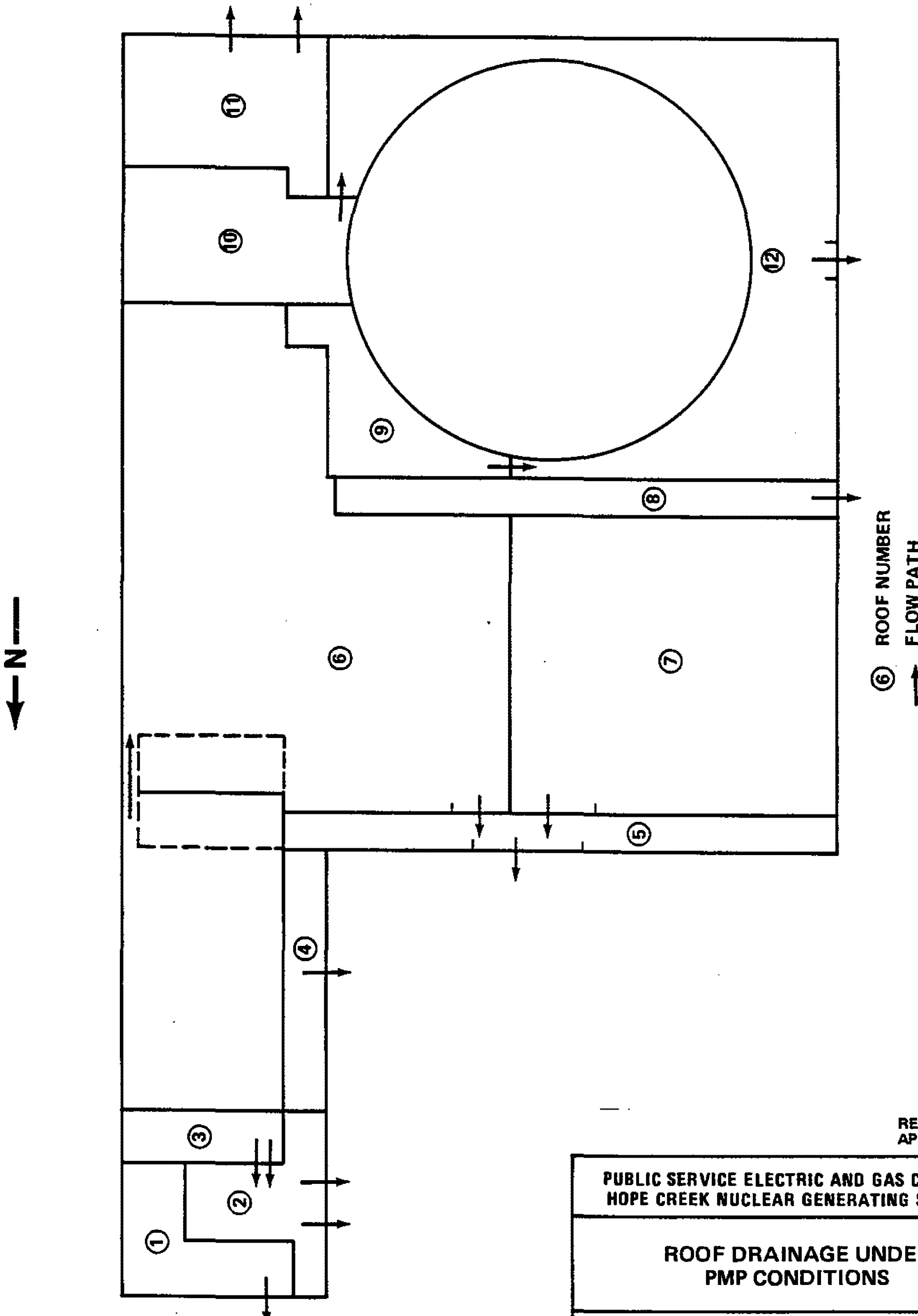
APRIL 11, 1988

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:

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

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

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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 $1.02x10^7$ takeoffs or landings for 1979 ($2.56x10^6$ flight hours x 4 operations/hour). This produces a fatal crash rate of $4.88x10^{-7}$ per takeoff or landing for 1979 (five divided by $1.02x10^7$).

[redacted]

The impact anywhere on a safety-related structure by an air carrier or military aircraft, such as a Boeing 727 or Lockheed C-130 (greater than 12,500 lbs), is conservatively assumed to cause unacceptable damage. Therefore, the auxiliary building, the reactor building, and the SSWS intake structure, are considered vulnerable to damage from impact by this category of aircraft. The effective impact areas for all categories of aircraft considered are shown in Table 3.5-9.

3.5.1.6.5 Aircraft Crash Density

For an aircraft flying directly over a plant, the crash density can be shown to be as indicated in Reference 3.5-18:

$$\rho = \frac{D}{\pi D^2/4} \qquad (3.5\text{-}75)$$

where:

D = 2 x altitude x glide ratio

This approach has been used for those airways passing within 2 miles of the HCGS plant site.

For the general aviation small fixed-wing VFR traffic near the HCGS, the following equation has been used because it is difficult to establish the exact location (i.e. distance from the plant and altitude) of the aircraft when the trouble leading to the crash originated; see Reference 3.5-19:

$$\rho = 1/2\ \gamma e^{-\gamma \chi} \qquad (3.5\ \text{-}76)$$

where:

x = lateral distance

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

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[redacted]

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:

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

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.

3.5-17 National Transportation Safety Board, Annual Review of Aircraft Accident Data, U.S. General Aviation, Calendar Year 1979, Report No. NTSB-ARG-81-1, November 5, 1981.

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."

3.5-22 C. McNeill, PSE&G, to E. Adensam, NRC, "Turbine System Maintenance Program", dated July 7, 1986.

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

THIS TABLE INTENTIONALLY DELETED

TABLE 3.5-4

THIS TABLE INTENTIONALLY DELETED

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	$6.82x10^{-8}$	$4.82x10^{-4}$
GA-SFW Multi-Engine	$4.66x10^{-8}$	$8.97x10^{-4}$
Commuter Air Carrier/On Demand Air Taxi (<12,500 lbs)	$1.75x10^{-8}$	$8.97x10^{-4}$
Air Carrier (>12,500 lbs)	$5.27x10^{-10}$	0.0212
Military (>12,500 lbs)	$2.64x10^{-9}$	0.0227
Helicopter (Crashes per takeoff and landing operation)	$4.88x10^{-7}$	$6.3x10^{-4}$

(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	2.5×10^{-3}
VFR 3-5 miles	4.3×10^{-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^{-1} (Mi^{-1})	P	P (Totals)
GA-SFW[1] Single Engine						
VFR						
w/in 1 mile	6.82×10^{-8}	150	4.82×10^{-4}	.1353	6.67×10^{-10}	
2 mile		430		.0183	2.59×10^{-10}	9.39×10^{-10}
3 mile		150		.0025	1.23×10^{-11}	
5 mile		525		4.54×10^{-5}	7.89×10^{-13}	
IFR						
TRACON-V123-312, J150 @ 7000 ft	6.82×10^{-8}	616	4.82×10^{-4}	.0480	9.72×10^{-10}	
TRACON-V29-157 @ 9000		4448		.0373	5.45×10^{-9}	9.43×10^{-9}
ARTCC-V123-312, J150 @ 7000		1551		.0480	$2.45-10^{-9}$	
ARTCC-V29-157 @ 9000		456		.0373	5.59×10^{-10}	
GA-SFW Multi Engine						
VFR						
w/in 1 mile	4.66×10^{-8}	50	8.97×10^{-4}	.1353	2.83×10^{-10}	
2 mile		145		.0183	1.11×10^{-10}	
3 mile		50		.0025	5.22×10^{-12}	4.0×10^{-10}
5 mile		175		4.54×10^{-5}	3.32×10^{-13}	
IFR						
TRACON-V123-312, J150 @ 7000 ft	4.66×10^{-8}	205	8.97×10^{-4}	.048	4.11×10^{-10}	
TRACON-V29-157 @ 9000		1483		.0373	2.31×10^{-9}	1.42×10^{-8}
ARTCC-V123-312, J150 @ 7000		4654		.048	9.34×10^{-9}	
ARTCC-V29-157 @ 9000		1369		.0373	2.13×10^{-9}	
IFR Commuter Air Carrier and On-Demand Air Taxi (<12,500 lbs)						
TRACON-V123-312, J150 @ 7000 ft	1.75×10^{-8}	1916	8.97×10^{-4}	.048	1.44×10^{-9}	
TRACON-V29-157 @ 9000		7026		.0373	4.11×10^{-9}	
ARTCC-V123-312, J150 @ 16000		1825		.021	6.02×10^{-10}	6.75×10^{-9}
ARTCC-V29-157 @ 16000		1825		.021	6.02×10^{-10}	
IFR Air Carrier (>12,500 lbs)						
TRACON-V123-312, J150 @ 7000 ft	5.27×10^{-10}	182	.0212	.048	9.76×10^{-11}	
TRACON-V29-157 @ 16000		1642		.021	3.85×10^{-10}	
ARTCC-V123-312, J150 @ 20000		116070		.017	2.2×10^{-8}	2.28×10^{-8}
ARTCC-V29-157 @ 16000		1460		.021	3.42×10^{-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.

MISSILE IDENTI- FICATION	SOURCE OF MISSILE						

Security Related Information
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TABLE 3.5-13 (Cont)

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

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
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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
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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
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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

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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 3.5-13

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 3.5-14

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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

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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 3.5-16

SECURITY - RELATED
INFORMATION WITHHELD
UNDER 10 CFR 2.390

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REV
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APRIL 11, 1988

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

HELICOPTER GLIDE PATHS

Updated FSAR Fig. 3.5-17

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INFORMATION WITHHELD
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

SECURITY - RELATED
INFORMATION WITHHELD
UNDER 10 CFR 2.390

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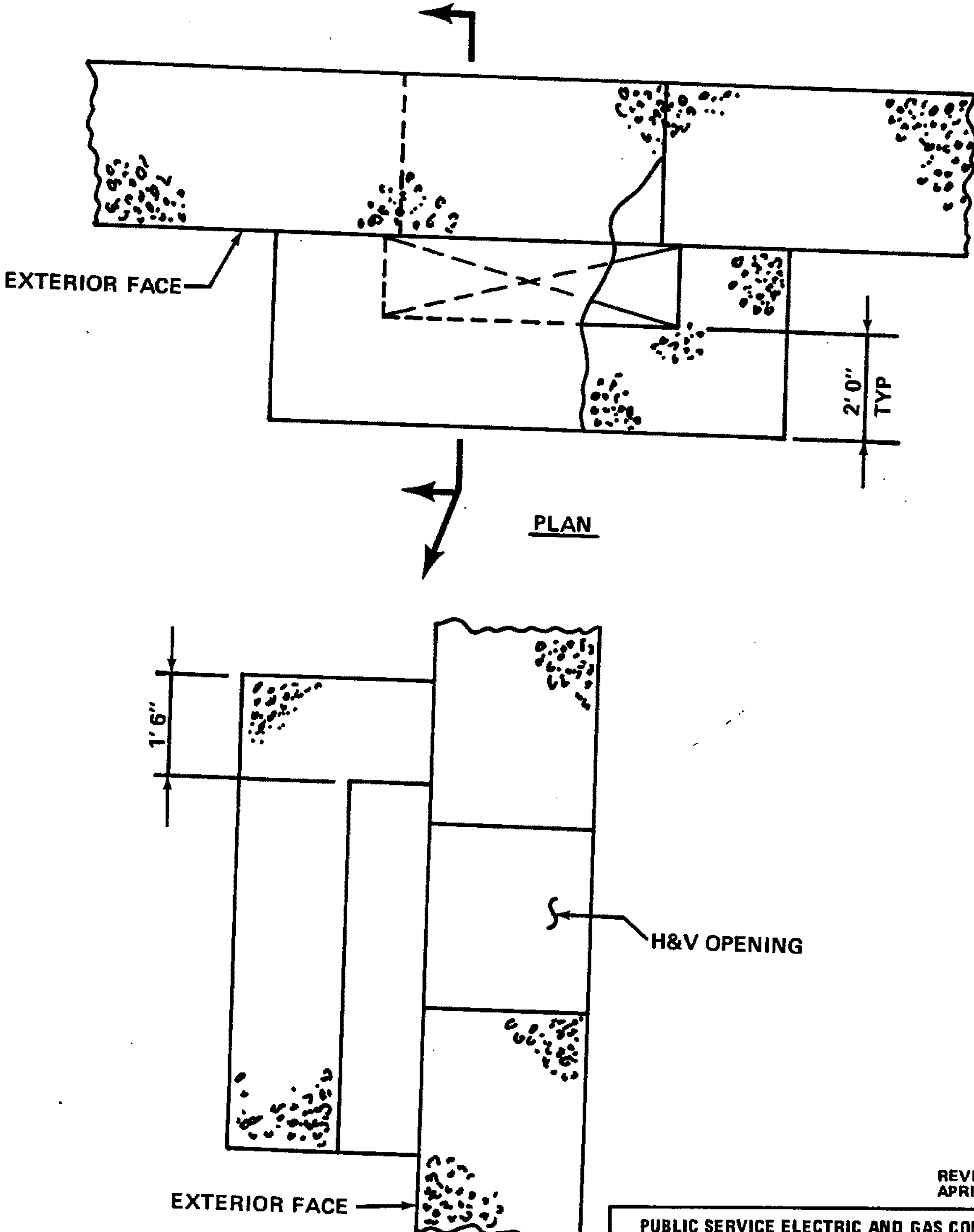
APRIL 11, 1988

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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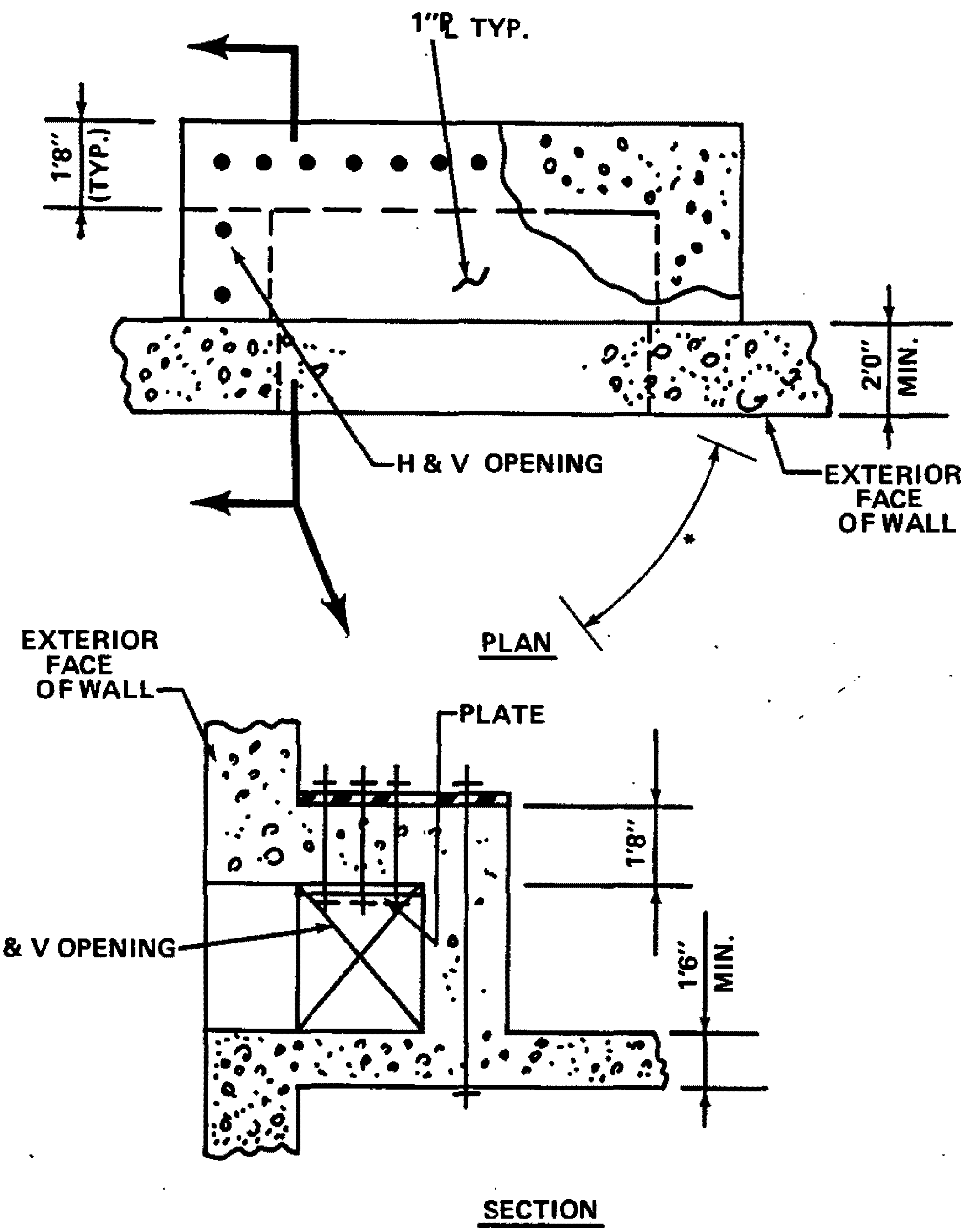
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

WALL OPENING
MISSILE BARRIER
TYPE – W1

UPDATED FSAR

FIGURE 3.5-20



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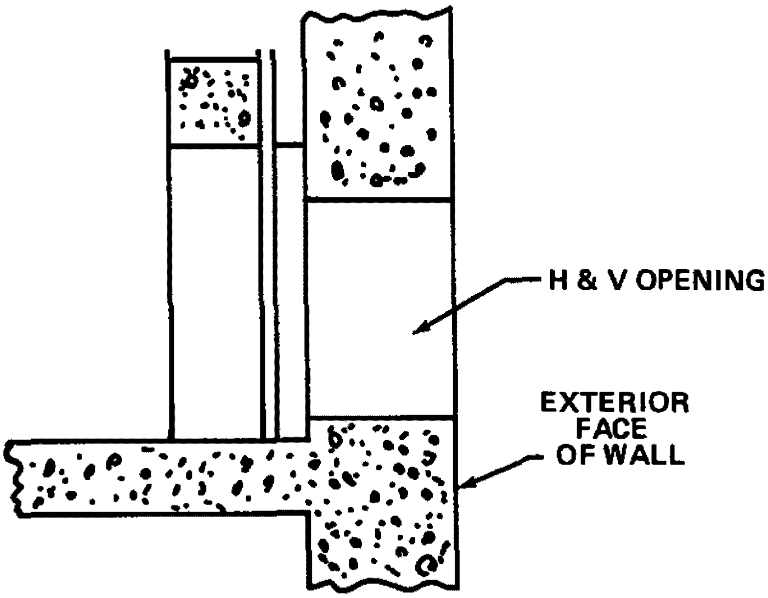
* POTENTIAL MISSILES IN THIS ZONE ARE SHIELDED BY ADJACENT STRUCTURES

PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

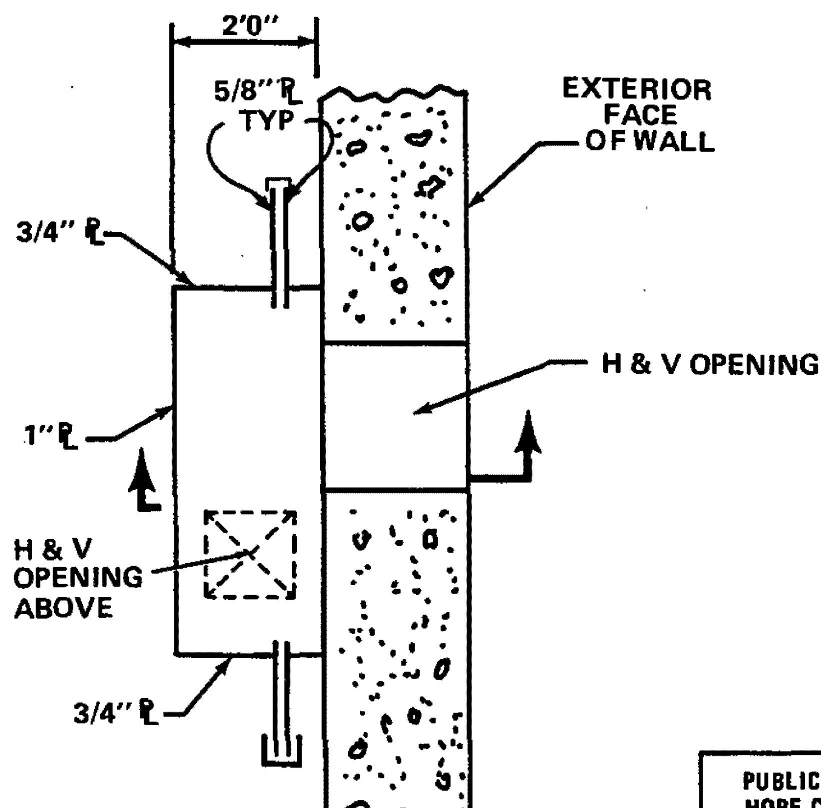
WALL OPENING
MISSILE BARRIER
TYPE – W2

UPDATED FSAR

FIGURE 3.5-21



ELEVATION



PLAN

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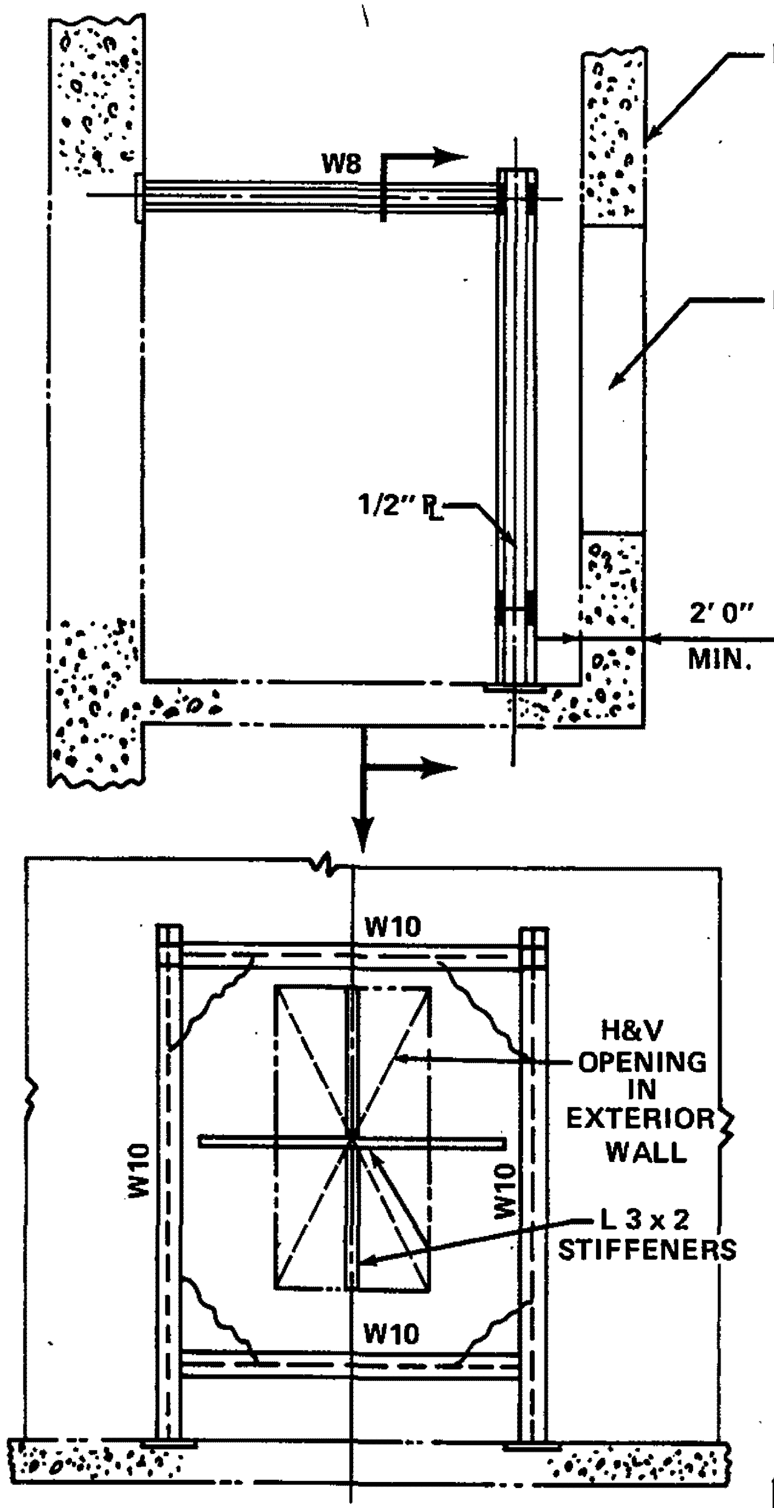
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

WALL OPENING
MISSILE BARRIER
TYPE – W3

UPDATED FSAR

FIGURE 3.5-22



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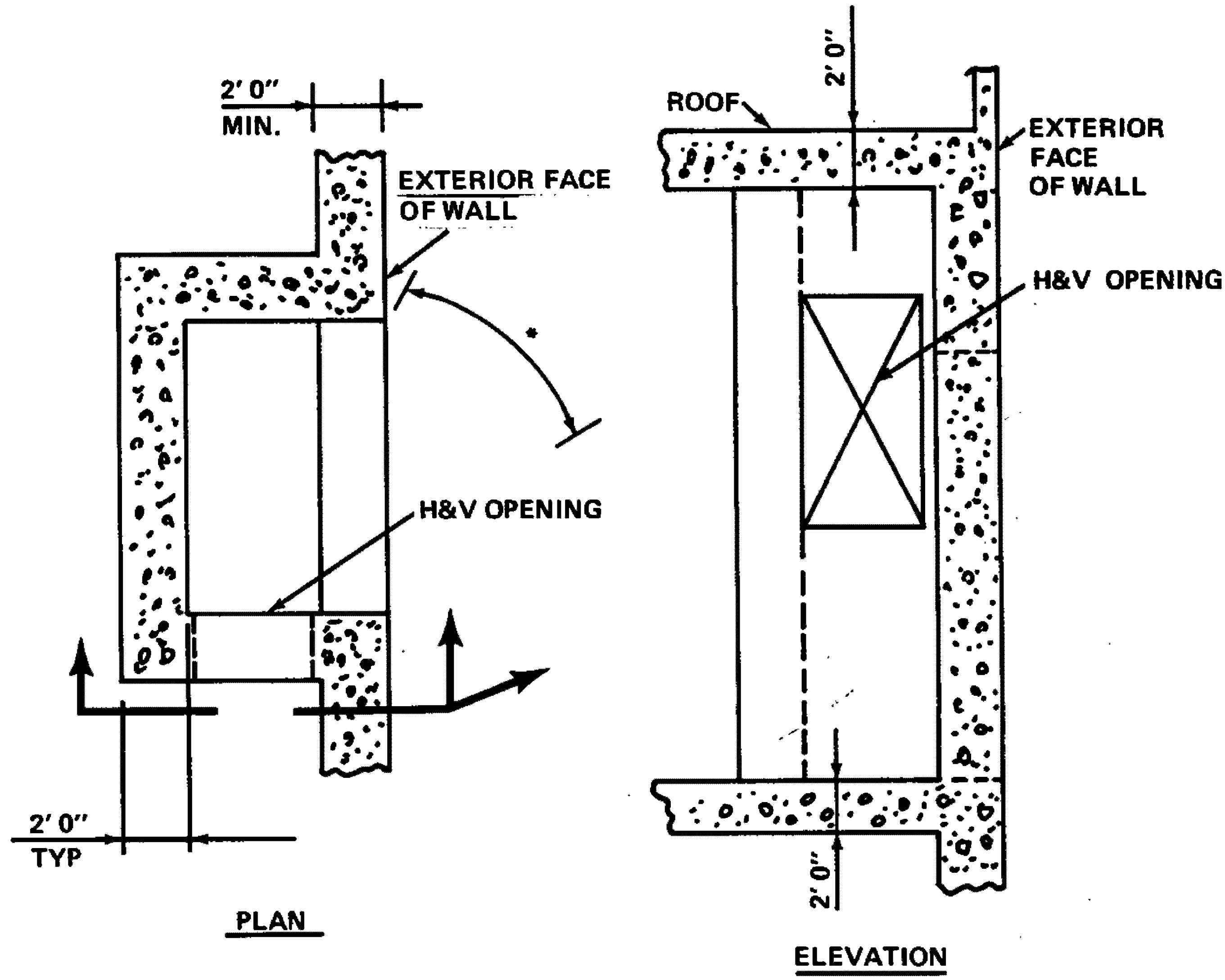
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

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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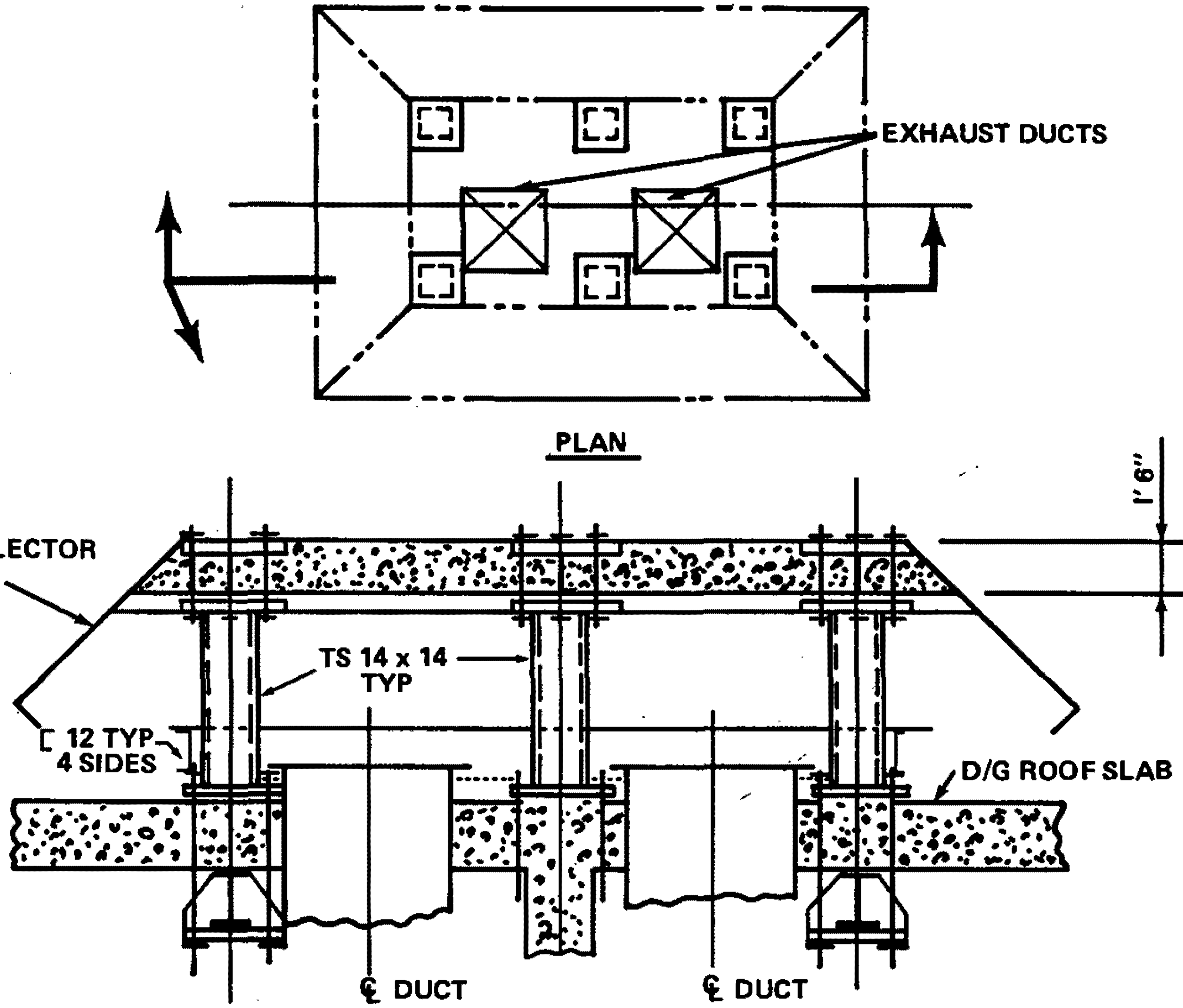
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

WALL OPENING
MISSILE BARRIER
TYPE – W5

UPDATED FSAR

FIGURE 3.5-24



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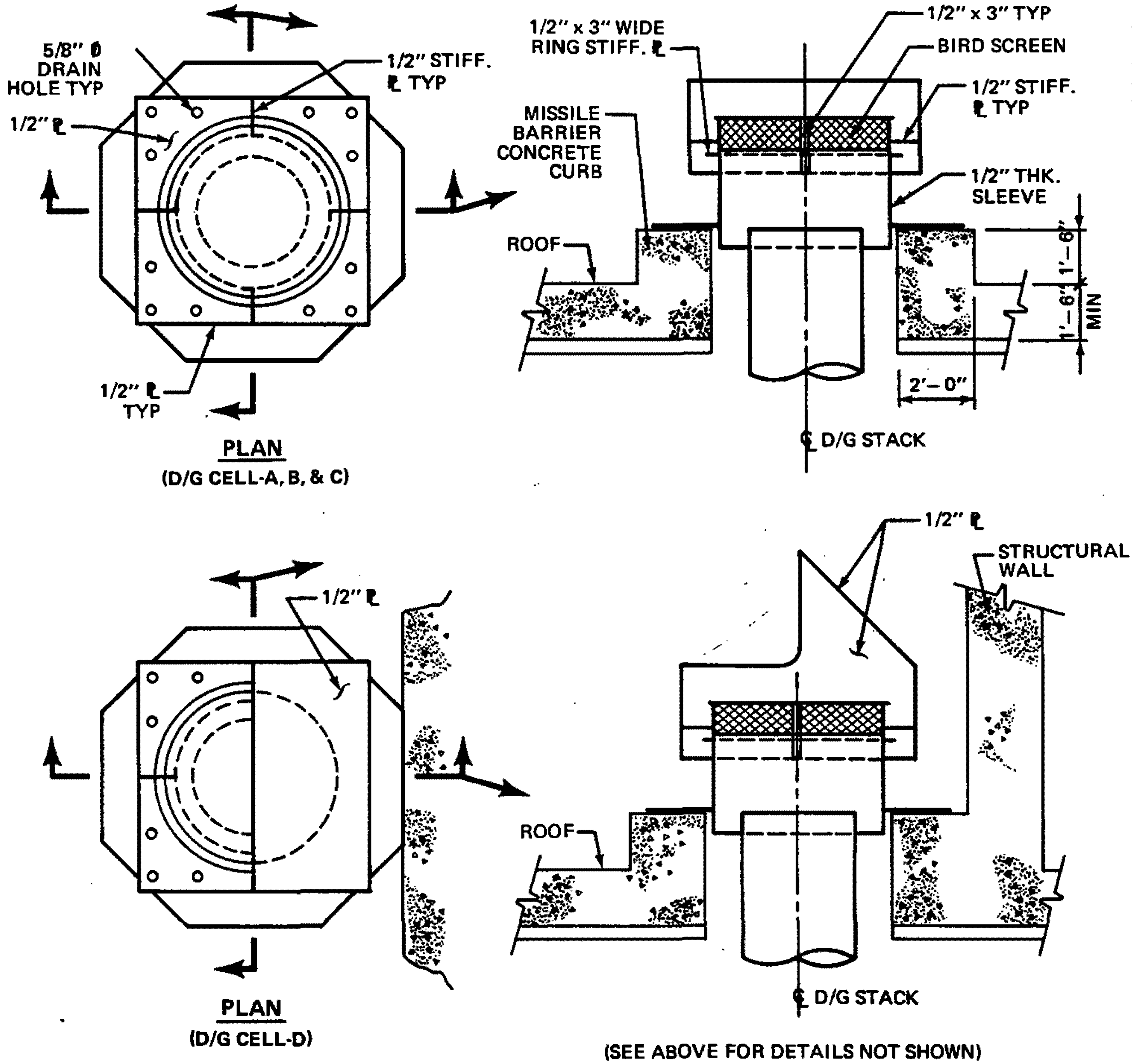
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

ROOF OPENING
MISSILE BARRIER
TYPE – R1

UPDATED FSAR

FIGURE 3.5-25



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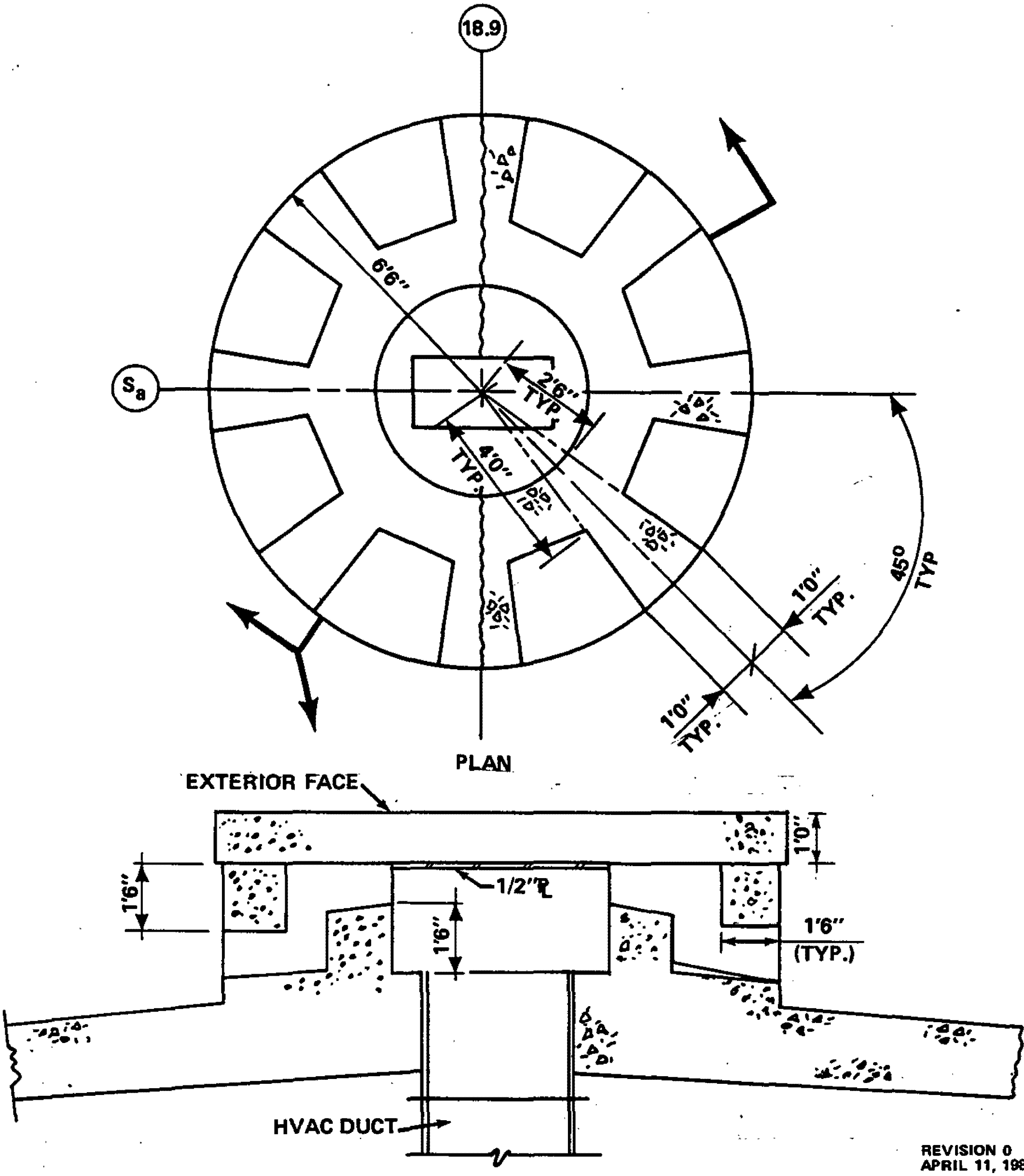
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

ROOF OPENING
MISSILE BARRIER
TYPE – R2

UPDATED FSAR

FIGURE 3.5-26



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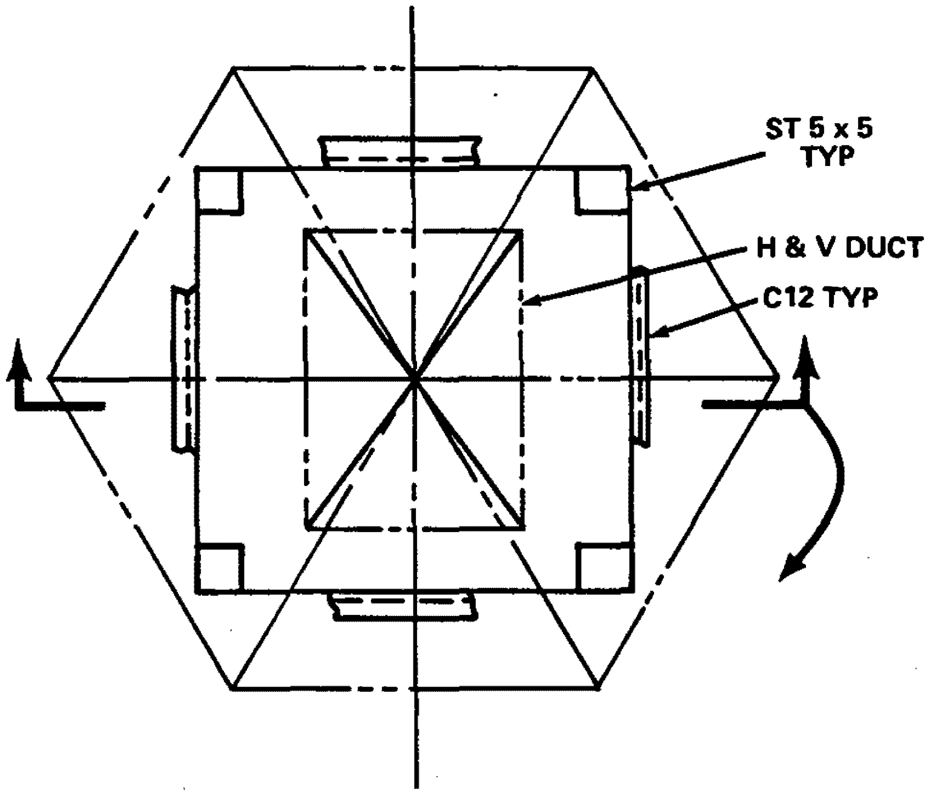
APRIL 11, 1988

PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

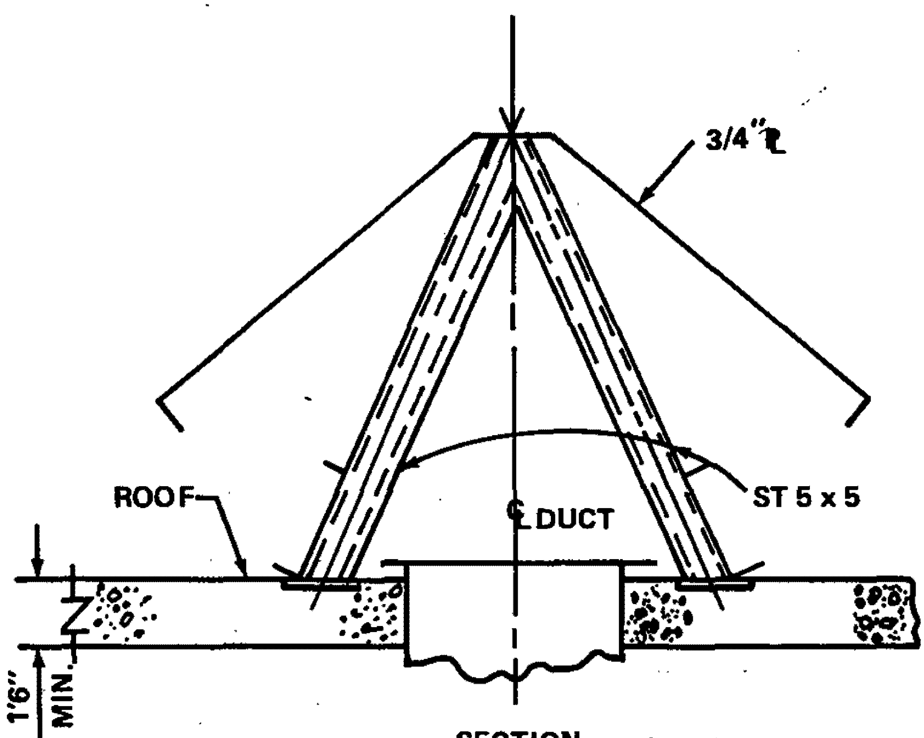
ROOF OPENING
MISSILE BARRIER
TYPE – R3

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