CHAPTER 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 Nine Mile Point Nuclear Station - Unit 2 (Unit 2) as compared to the Nuclear Regulatory Commission (NRC) General Design Criteria (GDC) for Nuclear Power Plants, Appendix A to 10CFR50. For each of the 64 criteria, a specific assessment of the plant design is made, and references are listed to identify where detailed information pertinent to each criterion is presented.

Based on the content herein, the Applicant concludes that, with the exception of the inapplicable portions of Criterion 56, Unit 2 fully satisfies and is in compliance with the GDC.

3.1.2 Criterion Conformance

3.1.2.1 Quality Standards and Records (Criterion 1)

Criterion

"Structures, systems, and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety function. A quality assurance program shall be established and implemented in order to provide adequate assurance that these structures, systems, and components will satisfactorily perform their safety functions. Appropriate records of the design, fabrication, erection, and testing of structures, systems, and components important to safety shall be maintained by or under the control of the nuclear power unit licensee throughout the life of the unit."

Design Conformance

Structures, systems, and components important to safety are listed in Table 3.2-1. The total quality assurance (QA) program is described in the Quality Assurance Topical Report (QATR) and is applied to the items indicated in Table 3.2-1. The intent of the QA program is to ensure sound engineering in all phases of design, construction and operation through conformity to regulatory requirements and design bases described in the USAR. In addition, the program ensures adherence to specified standards of workmanship and implementation of recognized codes and standards in fabrication and construction. It also includes the observance of proper preoperational and operational testing and maintenance procedures, as well as documentation by keeping appropriate records. The QA program of Nine Mile Point Nuclear Station, LLC (NMPNS), is responsive to and satisfies the intent of the quality-related requirements of 10CFR50, including Appendix B.

Structures, systems, and components are classified in Chapter 3 (Table 3.2-1) with respect to location, service, and relationship to the safety function to be performed. Recognized codes and standards are applied to the equipment in these classifications as necessary to ensure a quality product in keeping with the required safety function.

Documents are maintained that demonstrate that all 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 utilized; qualified personnel are employed to perform work; and finished parts and components meet applicable specifications for safe and reliable operation. These records are available so that any desired information is retrievable for reference. They are maintained during the life of the operating licenses.

The detailed QA program of NMPNS and its contractors satisfies the requirements of GDC 1.

For further discussion see the following sections:

Principal Design Criteria	1.2.1
Plant Description	1.2.2
Classification of Structures, Components, and Systems	3.2
Quality Assurance	17, Appendix B

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

<u>Criterion</u>

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

Design Conformance

Structures, systems, and components important to safety are designed to withstand effects of the most severe natural phenomena, specific to the site, combined with appropriate normal upset and accident conditions to ensure that there is no loss of capability to perform safety functions. Historical data are utilized with appropriate margin for the specific geographical area in determining effects of natural phenomena.

For further discussion, see the following sections:

Meteorology	2.3
Hydrologic Engineering	2.4
Geology, Seismology, and Geotechnical Engineering	2.5
Classification of Structures, Components, and Systems	3.2
Wind and Tornado Loadings	3.3
Water Level (Flood) Design	3.4
Missile Protection	3.5
Seismic Design	3.7
Design of Category I Structures	3.8
Mechanical Systems and Components	3.9
Qualification of Category I Instrumentation and Electrical Equipment	3.10
Environmental Design of Mechanical and Electrical Equipment	3.11

3.1.2.3 Fire Protection (Criterion 3)

Criterion

"Structures, systems, and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions. Noncombustible and heat-resistant materials shall be used wherever practical throughout the unit, particularly in locations such as the containment and control room. Fire detection and fighting systems of appropriate capacity and capability shall be provided and designed to minimize the adverse effects of fires on structures, systems, and components important to safety. 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."

Design Conformance

The power plant is designed to minimize the occurrence of fire by the use of noncombustible and heat-resistant materials wherever practicable. Plant arrangement allows for isolation of known fire hazards. Noncombustible materials are used to the greatest extent possible to retard the creation and subsequent spread of fire. Automatic and manual fire protection systems are provided throughout the plant. National Fire Protection Association (NFPA) standards are used as guides for the development of all fire protection systems.

For further discussion, see Section 9.5.1, Fire Protection System.

3.1.2.4 Environmental and Missile Design Bases (Criterion 4)

Criterion

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

Design Conformance

Structures, systems, and components important to safety are designed to accommodate the effects of and are compatible with environmental conditions associated with normal operation, maintenance, testing, and postulated accidents (including loss-of-coolant accident [LOCA]) (Section 3.11). These structures, systems, and components are appropriately protected against dynamic effects including missiles, pipe whipping, and discharging fluids that may result from pipe or equipment failures.

Electrical instrumentation and engineered safety feature (ESF) systems located inside the containment are discussed in sections listed below. The design requirements for environmental conditions associated with normal operation, maintenance, testing, and postulated accident events (including LOCA) are given in the sections listed below. Structures, systems, and components design meets the requirements of Criterion 4.

For further discussion, see the following sections:

Meteorology	2.3
Hydrologic Engineering	2.4
Geology, Seismology, and Geotechnical Engineering	2.5
Classification of Structures, Components, and Systems	3.2
Wind and Tornado Loadings	3.3
Water Level (Flood) Design	3.4
Missile Protection	3.5
Protection Against Dynamic Effects Associated with the Postulated Rupture of Piping	3.6
Seismic Design	3.7
Design of Category I Structures	3.8
Mechanical Systems and Components	3.9
Instrumentation and Electrical Equipment Environmental Design of Mechanical and Electrical Equipment	3.11
Integrity of Reactor Coolant Pressure Boundary	5.2
Engineered Safety Features	6
Instruments and Controls	7
Electric Power	8
Design Assessment Report for Hydrodynamic Loads	Appendix 6A

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

<u>Criterion</u>

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

Design Conformance

Unit 2 is a single unit and does not share structures, systems, and components important to safety.

3.1.2.6 (Not Promulgated by NRC)

3.1.2.7 (Not Promulgated by NRC)

3.1.2.8 (Not Promulgated by NRC)

3.1.2.9 (Not Promulgated by NRC)

3.1.2.10 Reactor Design (Criterion 10)

Criterion

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

Design Conformance

The reactor core components consist of fuel assemblies, control rods, in-core ion chambers, neutron sources (for initial startup), 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 occurrence.

The reactor protection system (RPS) is designed to monitor certain reactor parameters, to sense abnormalities, and to initiate a reactor scram, thereby preventing fuel design limits from being exceeded when trip setpoints are exceeded. Scram trip setpoints are selected based on operating experience and the safety design basis. 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 through an uninterruptible power supply (UPS) system. An alternate power source is available for each bus.

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

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

For further discussion, see the following sections:

Principal Design Criteria	1.2.1
Station Description	1.2.2
Fuel System Design	4.2
Nuclear Design	4.3
Thermal-Hydraulic Design	4.4
Reactor Coolant	5.4.1
Reactor Core Isolation Cooling System	5.4.6
Residual Heat Removal System	5.4.7
Accident Analysis	15

3.1.2.11 Reactor Inherent Protection (Criterion 11)

Criterion

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

Design Conformance

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

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

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; it contributes to system stability. Since the Doppler reactivity opposes load changes, it is desirable to maintain a large ratio of moderator void coefficient to Doppler coefficient for optimum load-following capability. The BWR has an inherently large moderator-to-Doppler coefficient ratio.

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

1. Use of coolant flow as opposed to control rods for load-following.
2. Inherent 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 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:

Principal Design Criteria	1.2.1
Nuclear Design	4.3
Thermal-Hydraulic Design	4.4

3.1.2.12 Suppression of Reactor Power Oscillations (Criterion 12)

Criterion

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

Design Conformance

The 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 boiling water reactors (BWR) underdamped, 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-damped 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. For further discussion, see the following sections:

Principal Design Criteria	1.2.1
Nuclear Design	4.3
Thermal-Hydraulic Design	4.4
Systems Required for Safe Shutdown	7.4
Control Systems Not Required for Safety	7.7
Accident Analysis	15

3.1.2.13 Instrumentation and Control (Criterion 13)

Criterion

"Instrumentation and control 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."

Design Conformance

The neutron flux in the reactor core is monitored by the neutron monitoring system (NMS), which has 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 detectors that make up the local power range monitor (LPRM) subsystem. The average power range monitor (APRM) subsystem is composed of core-wide sets of LPRM detectors that are averaged to provide a core average neutron flux. The traversing in-core probe (TIP) subsystem provides a means for calibrating the LPRM system. Both the IRM and APRM subsystems generate scram trips to the reactor trip system. All subsystems, except the TIP subsystem, generate rod block trips. Additional information on the NMS is given in Chapter 7.

The RPS protects the fuel barrier and the nuclear process barrier by monitoring plant parameters and initiating 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 release of radioactive materials from the fuel and RCPB, the containment and reactor vessel isolation control system initiates automatic isolation of appropriate piping when monitored variables exceed preselected operational limits.

Drywell leakage limits are established to ensure that the integrity of the RCPB is maintained. Leakage rates are classified as identified and unidentified corresponding, respectively, to the flow to the drywell equipment drain and floor drain sumps. The permissible total leakage rate limit to these sumps is based upon the capabilities of the various reactor component systems. High sump fillup rate and pumpout rate are alarmed in the main control room. The unidentified leakage rate established in Chapter 5 is less than the value that has been conservatively calculated to be the minimum detectable leakage from a crack large enough to propagate rapidly, but which still allows time for identification and corrective action before integrity of the RCPB is threatened.

As previously noted, adequate instrumentation has been provided to monitor system variables in the reactor core, RCPB, and containment. Appropriate controls have been provided to maintain variables in the operating range and initiate necessary corrective action in the event of abnormal operational occurrence or accident. Instrumentation and controls meet the requirements of Criterion 13.

For further discussion see the following sections:

Principal Design Criteria	1.2.1
Functional Design of Reactivity Control Systems	4.6
Integrity of Reactor Coolant Pressure Boundary	5.2
Main Steam Isolation System	5.4.5
Containment Systems	6.2
Reactor Protection (Trip) System	7.2
Containment and Reactor Vessel Isolation	7.3.1
Control System	7.5.1.3
All Other Instrumentation Systems Required for Safety	7.6

3.1.2.14 Reactor Coolant Pressure Boundary (Criterion 14)

Criterion

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

Design Conformance

The piping and equipment pressure parts within the RCPB extending to and including the outer containment isolation valves 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, 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, fracture toughness properties and the operating temperature of ferritic materials are controlled to ensure adequate toughness. Section 3.1.2.31 describes the methods used to control toughness properties. Materials are impact tested in accordance with applicable portions of the ASME Boiler and Pressure Vessel Code, Section III. Where RCPB piping penetrates the containment, fracture toughness temperature requirements of the RCPB materials apply.

Piping and pressure-retaining portions of components that compose the RCPB are assembled and erected by welding unless applicable codes permit flanged or screwed joints. All welding procedures, welders, and welding machine operators used in producing pressure-retaining welds are qualified in accordance with the requirements of the ASME Boiler and Pressure Vessel Code, Section IX, for the materials to be welded. Qualification records, including the results of procedure and performance qualification tests and identification symbols assigned to each welder, are maintained.

Section 5.2.3 contains the detailed material and examination requirements for the piping and components of the RCPB prior to and after its assembly and erection. Leakage testing and surveillance are accomplished as described in Section 3.1.2.30. The design, fabrication, erection, and testing of the RCPB assure an extremely low probability of failure or abnormal leakage, thus satisfying the requirements of Criterion 14.

For further discussion, see the following sections:

Principal Design Criteria	1.2.1
Design of Structures, Components,Equipment, and Systems	3
In-service Inspection and Testing of Reactor Coolant Pressure Boundary	5.2.4
Reactor Vessel	5.3
Component and Subsystem Design	5.4
Accident Analysis	15
Quality Assurance Program	17

3.1.2.15 Reactor Coolant System Design (Criterion 15)

Criterion

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

Design Conformance

The reactor coolant system (RCS) and associated auxiliary, control, and protection systems consist of the reactor vessel and appurtenances, reactor recirculation system, main steam safety relief valve (SRV) system, reactor core isolation cooling (RCIC) system, nuclear boiler instrumentation system, and portions of the main steam and feedwater systems located inside the outboard containment isolation valves. These systems are designed, fabricated, erected, and tested to stringent quality requirements and appropriate codes and standards that assure high integrity of the RCPB throughout the plant lifetime. As indicated in Section 3.2, these systems are designed and fabricated in accordance with the requirements of ASME Boiler and Pressure Vessel Code, Section III.

The auxiliary, control, and protection systems associated with the RCS act to provide sufficient margin to ensure that design conditions of the RCPB are not exceeded during any condition of normal operation, including anticipated operational occurrences. Appropriate instrumentation is provided to monitor essential variables to ensure that they are within prescribed operating limits (Section 3.1.2.13). If the monitored variables exceed predetermined setpoints, the auxiliary, control, and protection systems automatically respond to maintain the variables and systems within allowable design limits.

An example of the integrated protective action scheme that provides margin to ensure that design conditions of the RCPB are not exceeded is the automatic initiation of the main steam SRV system upon receipt of an overpressure signal. To accomplish overpressure protection, 18 SRVs which discharge to the suppression pool are provided. In addition, seven of these valves provide automatic depressurization of the RCPB in the event of a LOCA in which the vessel is not depressurized by the accident. Automatic depressurization of the vessel in this situation allows operation of the low-pressure ECCS to supply cooling water to cool the core adequately.
Application of appropriate codes and standards, high quality requirements to the RCS, and the design features of associated auxiliary, control, and protection systems assure that the requirements of Criterion 15 are fully satisfied.

For further discussion, see the following sections:

Principal Design Criteria	1.2.1
Design of Structures, Components, Equipment, and Systems	3
Overpressurization Protection	5.2.2
Reactor Vessel	5.3
Component and Subsystem Design	5.4
Reactor Protection (Trip) System	7.2
Accident Analysis	15

3.1.2.16 Containment Design (Criterion 16)

<u>Criterion</u>

"Reactor containment and associated systems shall be provided to establish an essentially leak-tight 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."

<u>Design Conformance</u>

Containment is provided by two major systems: the primary containment system and the secondary containment system. The primary system, which includes the drywell and suppression chamber, provides an essentially leak-tight barrier and is designed and constructed to accommodate, without loss of function, the pressures and temperature resulting from a double-ended rupture or equivalent failure of any coolant pipe within the primary containment. Containment temperature and pressure following an accident are limited by the suppression pool and residual heat removal (RHR) system.

The secondary containment system includes the reactor building structure and the safety-related systems that control the ventilation and cleanup of potentially-contaminated air volumes accumulated within the reactor building structure following a design basis accident (DBA). The two containment systems and their associated safety systems are designed and constructed so that release of radioactivity that could result from postulated DBAs and operating conditions remains below the criteria stated in 10CFR50.67 when calculated by the methods of RG 1.183.

For further discussion, see the following sections:

General Plant Description	1.2
Design of Seismic Category I Structures	3.8
Containment Systems	6.2
Accident Analysis	15

3.1.2.17 Electric Power Systems (Criterion 17)

<u>Criterion</u>

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

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

"Provisions shall be included to minimize the probability of losing electric power from any of the remaining sources 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 sources."

Design Conformance

An onsite electric power system and an offsite electric power system have been provided for operation of the systems and components important to safety. The onsite electric power system consists of three onsite standby diesel generators and three batteries with primary and backup battery chargers. The offsite electric power system consists of two 115-kV transmission circuits from two offsite power sources. Either of the two offsite power sources or any two of the three standby diesel generators has sufficient capacity to operate sufficient safety-related equipment required to shut down the plant in the event of postulated accidents.

Three standby diesel generators are connected to three 4.16-kV safety-related buses forming three independent divisions of the safety-related auxiliary power distribution system. Division I and Division II feed all nuclear safety-related loads (except the high-pressure core spray [HPCS] system loads) divided into redundant load groups; Division III feeds all HPCS loads. The 125-V safety-related dc power system feeds all safety-related dc protection, control, and instrumentation loads and safety-related dc motors. The system is divided into three independent and redundant divisions each consisting of its own battery, primary and backup battery chargers, motor control centers (MCC), and other distribution equipment. Each division feeds the dc loads associated with the corresponding divisions of the safety-related auxiliary power distribution system, thereby maintaining the independence of the three divisions.

Two 115-kV offsite power circuits providing offsite power sources to the plant auxiliary power distribution system are electrically independent and physically separated to minimize the possibility of their simultaneous failure under operating and postulated accident and environmental conditions.

For further discussion, see Chapter 8, Electric Power.

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

Criterion

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

Design Conformance

The station safety-related auxiliary power distribution system is divided into three separate and independent divisions feeding redundant safety-related load groups. This arrangement provides for the scope of periodic inspection and testing of any division while other divisions are feeding their connected loads. The standby diesel generator, associated switchgear assemblies, and batteries in each division are designed and arranged to permit independent periodic inspection and testing.

Each standby diesel generator is testable for its automatic starting and load sequencing capability simulating a loss of bus voltage. Full load testing of each standby diesel generator can be performed periodically by manually starting each standby diesel generator, synchronizing to the offsite power supply at its associated emergency bus, and loading the unit by governor adjustment. The standby diesel generators have an exercise mode for this purpose. These tests prove the operability of the onsite power system under conditions as close to design conditions as practicable.

Transfer of auxiliary power from the main generator source through the normal Station service transformer, to the offsite power via the reserve Station service transformers, will be tested when the reactor power is at a low level. Transfer of auxiliary power from offsite power sources through the reserve Station service transformers, to the onsite power from the standby diesel generators, can be tested during reactor operation or during refueling shutdowns.

For further discussion see Chapter 8, Electric Power.

3.1.2.19 Control Room (Criterion 19)

Criterion

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

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

"Applicants for and holders of construction permits and operating licenses under this part who apply on or after January 10, 1997, applicants for design approvals or certifications under part 52 of this chapter who apply on or after January 10, 1997, applicants for and holders of combined licenses or manufacturing licenses under part 52 of this chapter who do not reference a standard design approval or certification, or holders of operating licenses using an alternative source term under §50.67, shall meet the requirements of this criterion, except that with regard to control room access and occupancy, adequate radiation protection shall be provided to ensure that radiation exposures shall not exceed 0.05 Sv (5 rem) total effective dose equivalent (TEDE) as defined in §50.2 for the duration of the accident."

Design Conformance

A main control room is provided and equipped to operate the unit safely under normal and accident conditions. Main control room shielding and ventilation are designed to permit continuous occupancy of the main control room for the duration of a DBA while limiting the dosage to personnel to not more than 5 Rem TEDE in accordance with 10CFR50.67.

A remote shutdown room complete with equipment, controls, and instrumentation is provided to bring the reactor through shutdown to hot standby or to subsequent cold shutdown in a safe manner. The remote shutdown room and associated controls are located in an area that is physically isolated from the main control room so that any event that could cause the main control room to become uninhabitable or inaccessible has no effect on the accessibility of the remote shutdown room and adjacent controls. Also, equipment, controls, and instrumentation are located throughout the unit to provide capability for a subsequent cold shutdown through the use of suitable procedures. The main control room and the remote shutdown room conform to the requirements of Criterion 19 and 10CFR50.67.

For further discussion, see the following sections:

Seismic Classification	3.2.1
Habitability Systems	6.4
Systems Required for Safe Shutdown	7.4
Air Conditioning, Heating, Cooling, and Ventilating Systems	9.4
Radiation Protection Design Features	12.3

3.1.2.20 Protection System Functions (Criterion 20)

<u>Criterion</u>

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

<u>Design Conformance</u>

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 preestablished 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 not be subject to spurious scrams. The RPS includes the high-inertia, uninterruptible power system, sensors, transmitters, trip units, bypass circuitry, and switches that signal the control rod system to scram and shut down the reactor. The scrams initiated by NMS variables, nuclear system high pressure, turbine stop valve closure, turbine control valve (TCV) fast closure, main steam isolation valve (MSIV) closure, and reactor vessel low water level, prevent fuel damage following abnormal operational transients. Specifically, these process parameters initiate a scram in time to prevent the core from exceeding thermal-hydraulic safety limits during abnormal operational transients. Additional scram trips are initiated by drywell high pressure generator load rejection and scram discharge volume (SDV) high water level. Response by the RPS is prompt and the total scram time is short. Control rod scram motion starts in less than 250 msec after the sensor contacts actuate.

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

Other systems automatically isolate the reactor vessel or containment to prevent the release of significant amounts of radioactive materials from the fuel and the RCPB. Controls and instrumentation for the ECCS and isolation systems are automatically initiated when monitored variables exceed preselected operational limits. The design of the protection system satisfies the functional requirements specified in Criterion 20.

For further discussion, see the following sections:

Principal Design Criteria1	1.2.1
Functional Design of Reactivity Control Systems	4.6
Overpressure Protection	5.2.2
Main Steam Isolation System	5.4.5
Emergency Core Cooling System	6.3
Reactor Protection (Trip) System	7.2
Engineered Safety Feature Systems	7.3
All Other Instrumentation Systems Required for Safety	4.6
Accident Analysis	15

3.1.2.21 Protection System Reliability and Testability (Criterion 21)

Criterion

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

Design Conformance

RPS design provides assurance that, through redundancy, each channel has sufficient reliability to fulfill the single-failure criterion. No single component failure, intentional bypass, maintenance operation, calibration operation, or test to verify operational availability impairs the ability of the system to perform its intended safety function. Additionally, the system design assures that when a scram trip point is exceeded there is a high probability of scram. However, should a scram not occur, other monitored components will scram the reactor if their trip points are exceeded. There is sufficient electrical and physical separation between channels and between logics monitoring the same variable to prevent environmental factors, electrical transients, and physical events from impairing the ability of the system to respond correctly.

The RPS includes design features that permit in-service testing (IST). This ensures functional reliability of the system should the reactor variable exceed the corrective action setpoint.

The RPS initiates an automatic reactor shutdown if the monitored plant variables exceed preestablished 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 occurs when both trip systems have tripped. This logic scheme is called a one-out-of-two taken twice arrangement. The RPS can be tested during reactor operation. Manual scram testing is performed by operating one of the four manual scram controls. Two manual scram controls are associated with each trip system, one in each trip channel. Operating one manual scram control tests one trip channel and one trip system. The total test verifies the ability to de-energize the scram pilot valve solenoids. Indicating lights verify that the actuator contacts have opened. This capability for a thorough testing program significantly increases reliability.

Control rod drive (CRD) operability can be tested during normal reactor operation. Drive position indicators and in-core neutron detectors are used to verify control rod movement. Each control rod can be withdrawn one notch and then reinserted to the original position without significantly perturbing the nuclear system at most power levels. One control rod is tested at a time. CRD mechanism overtravel testing demonstrates rod-to-drive coupling integrity. Hydraulic supply subsystem pressures can be observed on main control room instrumentation. More importantly, the hydraulic control unit (HCU) scram accumulators pressure and the SDV water level are monitored. The SDV is sensed by level switches which automatically scram the reactor when the volume is high enough to verify that the volume is filling up, yet low enough to ensure that the remaining capacity can accommodate a scram.

MSIVs can be tested during reactor operation. For test, the valves move shut a very small distance from the fully-open position, then open fully without affecting reactor operation. If reactor power is sufficiently reduced, the isolation valves can be fully closed. During the refueling operation, valve leakage rates can be determined.

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

Each active component of the ECCS required to operate in a DBA is designed to be operable for test purposes during normal operation of the nuclear system.

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

For further discussion, see the following sections:

Principal Design Criteria	1.2.1
Functional Design of Reactivity Control System	4.6
Main Steam Line Isolation System	5.4.5
Residual Heat Removal System	5.4.7
Containment Systems	6.2
Emergency Core Cooling System	6.3
Reactor Protection (Trip) System	7.2
Engineered Safety Feature Systems	7.3
All Other Instrumentation Systems	
Required for Safety	7.6
Accident Analysis	15

3.1.2.22 Protection System Independence (Criterion 22)

Criterion

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

Design Conformance

The components of the protection system are designed so that environments resulting from any emergency situation in which the components are required to function do not interfere with the operation of that function. Wiring for the RPS outside the main control room is run in rigid or flexible conduit. No other wiring is run in these conduits. The wires from duplicate sensors on a common process tap are run in separate conduits. The system sensors are electrically and physically separated. Only one trip channel actuator logic circuit from each trip system is run in the same conduit.

The RPS is designed to permit maintenance and diagnostic work while the reactor is operating without restricting plant operation or hindering the output of that safety function. Flexibility in design afforded the protection system allows operational system testing by the use of an independent trip channel for each trip logic input. When an essential monitored variable exceeds its scram trip point, it is sensed by at least two independent sensors in each trip system. Maintenance operation, calibration operation, or test, unless manually bypassed, can result in a single channel trip and one trip system trip (half scram). This leaves at least two trip channels per monitored variable of the other trip system capable of initiating a scram. Thus, the arrangement of two trip channels per trip system ensures that a scram occurs as a monitored variable exceeds its scram setting. Only one trip channel in each trip system must trip to initiate a scram.

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

For further discussion, see the following sections:

Principal Design Criteria	1.2.1
Functional Design of Reactivity Control System	4.6

Main Steam Line Isolation System	5.4.5
Residual Heat Removal System	5.4.7
Emergency Core Cooling System	6.3
Reactor Protection (Trip) System	7.2
Engineered Safety Feature Systems	7.3
All Other Instrumentation Systems Required for Safety	7.6
Accident Analysis	15

3.1.2.23 Protection System Failure Modes (Criterion 23)

Criterion

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

Design Conformance

The RPS is designed to fail into a safe state. Use of an independent trip channel for each trip logic allows the system to sustain any trip channel failure without preventing other sensors monitoring the same variable from initiating a scram. A single sensor or trip channel failure causes a channel trip. Only one trip channel in each trip system must be actuated to initiate a scram. Maintenance operation, calibration operation, or test, unless manually bypassed, can result in a single channel trip and one trip system trip (half scram). A failure of any one RPS input or subsystem component produces a trip in one of two channels and therefore in 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.

Environmental conditions in which the instrumentation and equipment of the RPS must operate were considered in establishing component specifications. Instrumentation is designed to function in the worst expected ambient conditions in which the instruments must operate.

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

For further discussion, see the following sections:

Principal Design Criteria	1.2.1
Equipment Qualification	3.11
Emergency Core Cooling System	6.3
Reactor Protection (Trip) System	7.2
Engineered Safety Feature Systems	7.3

3.1.2.24 Separation of Protection and Control Systems (Criterion 24)

<u>Criterion</u>

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

<u>Design Conformance</u>

There is separation between the RPS and the process control systems. 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 RPS. High scram reliability is designed into the RPS and HCU for the CRD. The scram signal and mode of operation override all other signals.

The primary containment isolation control system is designed so that any one failure, maintenance operation, calibration operation, or test to verify operational availability does not impair the functional ability of the isolation control system to respond to essential variables.

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

For further discussion, see the following sections:

Principal Design Criteria	1.2.1

Functional Design of Reactivity Control System	4.6
Emergency Core Cooling System	6.3
Reactor Protection (Trip) System	7.2
Engineered Safety Feature Systems	7.3
All Other Instrumentation Systems Required for Safety	7.6

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

Criterion

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

Design Conformance

The 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 function.

The reactor manual control system (RMCS) is designed so that no single failure negates the effectiveness of a reactor scram. Circuitry for the RMCS is completely independent of the circuitry controlling the scram valves. This separation of the scram and normal rod control functions prevents failures in the RMCS 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. Effectiveness of a reactor scram is not impaired by the malfunctioning of any one control rod.

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

For further discussion, see the following sections:

Principal Design Criteria	1.2.1
Nuclear Design	4.3

3.1.2.26 Reactivity Control System Redundancy and Capability (Criterion 26)

Criterion

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

Design Conformance

Two independent reactivity control systems utilizing different design principles are provided. The normal method of reactivity control employs control rod assemblies containing boron carbide (B_4C) or B_4C and hafnium. Positive insertion of these control rods is provided by means of 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, normal startup, and shutdown) via operator-controlled insertions and withdrawals. Control rods are also capable of maintaining the core within acceptable fuel design limits during anticipated operational occurrences via the automatic scram function. The unlikely occurrence of a limited number of stuck rods during a scram will not adversely affect the capability to maintain the core within fuel design limits.

Circuitry for manual insertion or withdrawal of control rods is completely independent of the circuitry for reactor scram. Separation of the scram and normal rod control functions prevents failures in the reactor manual control circuitry from affecting scram circuitry. Two sources of scram energy (accumulator pressure and reactor vessel pressure) provide needed scram performance over the entire range of reactor pressure, i.e., from operating conditions to cold shutdown. 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. Operating procedures to accomplish such patterns are supplemented by the rod worth minimizer (RWM), which prevents rod withdrawals yielding a rod worth greater than permitted by the preselected rod withdrawal pattern. Because of the carefully planned and regulated rod withdrawal sequence, prompt shutdown of the reactor can be achieved by the insertion of a small number of 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 effect 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 in such a way that the pump runout does 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 fuel with burnable poison (Gd_2O_3) to control high reactivity of fresh fuel. In addition, the standby liquid control system (SLCS) is available to add soluble boron to the core and render it subcritical, as discussed in Sections 3.1.2.27 and 9.3.5.

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

For further discussion, see the following sections:

Principal Design Criteria	1.2.1
Functional Design of Reactivity Control System	4.6
Engineered Safety Feature Systems	7.3
Standby Liquid Control System	7.4.1.3

Control Systems Not Required for Safety 7.7

3.1.2.27 Combined Reactivity Control Systems Capability (Criterion 27)

Criterion

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

Design Conformance

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

In the unlikely event that more than one control rod fails to insert and the core cannot be maintained in a subcritical condition by the control rods alone as the reactor cools down subsequent to initial shutdown, the SLCS is activated manually to inject soluble boron into the reactor core. The SLCS has sufficient capacity to ensure that the reactor can always be maintained subcritical; hence, only decay heat is generated by the core which can be removed by the RHR system, ensuring that the core is always coolable.

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

For further discussion, see the following sections:

Principal Design Criteria	1.2.1
Nuclear Design	4.3
Thermal-Hydraulic Design	4.4
Functional Design of Reactivity Control System	4.6
Reactor Protection (Trip) System	7.2
Control Systems Not Required for Safety	7.7
Accident Analysis	15

3.1.2.28 Reactivity Limits (Criterion 28)

Criterion

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

Design Conformance

The control rod system design incorporates appropriate limits on the potential amount and rate of reactivity increase. Control rod withdrawal sequences and patterns are selected to achieve optimum core performance and low individual rod worths. The RWM prevents withdrawal other than by the preselected rod withdrawal pattern. The RWM 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 ejection velocity to less than 5 fps. Normal rod movement is limited to 6-in increments and the rod withdrawal rate is limited through the hydraulic valve to 3 in/sec.

For Cycle 7, a cycle-specific analysis has been completed for rod withdrawal rates up to 6.0 in per second. For all other cycles, a cycle-generic analysis has been completed for rod withdrawal rates up to 5.0 in per second.

The accident analysis (Chapter 15) evaluates the postulated reactivity accidents, as well as abnormal operational transients. Analyses are included for rod dropout, steam line rupture, changes in reactor coolant temperature and pressure, and cold water addition. The initial conditions, assumptions, calculational models, sequences of events, and anticipated results of each postulated occurrence are covered in detail. 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, or other reactor pressure vessel internals are maintained so that the capability to cool the core is not impaired for any of the postulated reactivity accidents described in the accident analysis.

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

For further discussion, see the following sections:

Principal Design Criteria	1.2.1
Control Rod Drive Systems	3.9B.4.3
Nuclear Design	4.3
Control Rod Drive Housing Supports	4.5.3
Functional Design of Reactivity Control System	4.6
Overpressurization Protection	5.2.2
Reactor Vessel	5.3
Main Steam Line Flow Restrictions	5.4.4
Main Steam Line Isolation System	5.4.5
All Other Instrumentation Systems Required for Safety	7.6
Accident Analysis	15

3.1.2.29 Protection Against Anticipated Operational Occurrences (Criterion 29)

<u>Criterion</u>

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

Design Conformance

High functional reliability of the protection and reactivity control systems is achieved through the combination of logic arrangement, redundancy, physical and electrical independence, functional separation, fail-safe design, and in-service testability. The 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 IST and surveillance. Active components can be tested or removed from service for maintenance during reactor operation without compromising the protection or reactivity control functions even in the event of a subsequent single failure. Components important to safety such as CRDs, MSIVs, and ECCS pumps, are tested during normal reactor operation. Functional testing and calibration schedules are developed using available failure rate data, reliability analyses, and operating experience. These schedules represent an optimization of protection and reactivity control system reliability by considering, on the one hand, the failure probabilities of individual components and, on the other hand, the reliability effects during individual component testing on that portion of the system not undergoing testing. The program for IST 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.

Design and testing features described above ensure the high reliability of the reactor protection and reactivity control systems. However, in the unlikely event these systems fail to respond when required during an anticipated operational occurrence, additional plant capability exists to mitigate such a condition. Recirculation pump trip (RPT), alternate rod insertion (ARI), and manual initiation of SLCS operation provide additional assurance of acceptable plant response to anticipated operational occurrences.

For further discussion, see the following sections:

General Plant Description	1.2
Main Steam Line Isolation System	5.4.5

Residual Heat Removal System	5.4.7
Containment Systems	6.2
Emergency Core Cooling System	6.3
Reactor Protection (Trip) System	7.2
Engineered Safety Feature Systems	7.3
Accident Analysis	15

3.1.2.30 Quality of Reactor Coolant Pressure Boundary (Criterion 30)

Criterion

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

Design Conformance

By utilizing conservative design practices and detailed quality control procedures, the pressure-retaining components of the RCPB are designed and fabricated to retain their integrity during normal and postulated accident conditions. Components that compose the RCPB are designed, fabricated, erected, and tested in accordance with recognized industry codes and standards listed in Sections 5.2.2.6 and 5.3.1.1 and Table 3.2-1. Further, product and process quality planning is provided to ensure conformance with applicable codes and standards, and to retain appropriate documented evidence verifying compliance. Because the subject matter of this criterion deals with aspects of the RCPB, further discussion on this subject is provided in Section 3.1.2.14.

Means are provided for detecting reactor coolant leakage. The leak detection system (LDS) consists of sensors and instruments to detect, annunciate and, in some cases, isolate the RCPB from potentially-hazardous leaks before predetermined limits are exceeded. Small leaks are detected by temperature and pressure changes, by increased frequency of sump pump operation, and by measuring fission product concentration. In addition to these means of detection, large leaks are detected by changes in flow rates in process lines, increases in drywell pressure or temperature, and changes in reactor water level. Allowable leakage rates have been based on the predicted and experimentally determined behavior of cracks in pipes, ability to make up coolant system leakage, normally-expected background leakage due to equipment design, and detection capability of various sensors and instruments. The total leakage rate limit is established so that, in the absence of normal ac power with a loss of feedwater supply, makeup capabilities are provided by the RCIC system. While the LDS 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 LDS are designed to meet the requirements of Criterion 30.

For further discussion, see the following sections:

Principal Design Criteria	1.2.1
Design of Structures, Components, Equipment, and Systems	3
Integrity of Reactor Coolant Pressure Boundary	5.2
Overpressurization Protection	5.2.2
In-service Inspection and Testing of the Reactor Coolant Pressure Boundary	5.2.4
Detection of Leakage Through the Reactor Coolant Pressure Boundary	5.2.5
Reactor Vessel	5.3
Component and Subsystem Design	5.4
Reactor Recirculation Pumps	5.4.1
Emergency Core Cooling System	6.3
Reactor Manual Control System	7.1.1
Leak Detection System - Instrumentation and Control	7.6.1
Quality Assurance	17

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

Criterion

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

Design Conformance

Brittle fracture control of pressure-retaining ferritic materials is provided to ensure protection against nonductile fracture. To minimize the possibility of brittle fracture failure of the reactor pressure vessel (RPV), the RPV is designed to meet the requirements of 10CFR50. The nil ductility transition temperature (NDTT) is defined as the temperature below which ferritic steel breaks in a brittle rather than ductile manner. The NDTT increases as a function of neutron exposure at integrated neutron exposures greater than 1 x 10^{17} nvt with neutrons of energy in excess of 1 Mev.

The balance of the RCPB is designed, maintained, and tested in such a way that adequate assurance is provided that the boundary will behave in a nonbrittle manner throughout the life of the plant. Section 5.2 discusses this in further detail. Therefore, the RCPB is in conformance with Criterion 31.

For further discussion, see the following sections:

Design of Structures, Components, Equipment and Systems	3
Integrity of the Reactor Coolant Pressure Boundary	5.2
Reactor Vessel	5.3
Compliance with 10CFR50, Appendixes G and H	Appendix 5A

3.1.2.32 Inspection of Reactor Coolant Pressure Boundary (Criterion 32)

Criterion

"Components which 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 leak-tight integrity, and (2) an appropriate material surveillance program for the reactor pressure vessel (RPV)."

Design Conformance

The RPV design and engineering effort includes provisions for in-service inspection (ISI). Removable plugs in the biological shield wall (BSW) and/or removable panels in the insulation provide access for examination of the vessel and its appurtenances. Also, removable insulation is provided on the reactor coolant system SRVs, recirculation system, and on the main steam and feedwater systems extending out to and including the first isolation valve outside containment. Inspection of the RCPB is in accordance with the ASME Boiler and Pressure Vessel Code Section XI. Section 5.2 defines the ISI plan, access provisions, and areas of restricted access.

Vessel material surveillance samples are located within the RPV. The program includes specimens of the base metal, weld metal, and heat-affected zone (HAZ) metal.

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

For further discussion, see the following sections:

Mechanical Systems and Components	3.9
Integrity of Reactor Coolant Pressure Boundary	5.2

3.1.2.33 Reactor Coolant Makeup (Criterion 33)

Criterion

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

Design Conformance

Means are provided for detecting loss of reactor coolant through leakage or small breaks. The LDS consists of sensors and instruments to detect, annunciate and, in some cases, isolate the RCPB from potential hazardous loss-of-coolant situations before predetermined limits are exceeded. Small leaks or breaks are detected by temperature and pressure changes, by increased frequency of sump pump operation, and by measuring fission product concentration. In addition to these means of detection, large leaks or breaks are detected by changes in flow rates in process lines, increases in drywell pressure and temperature, and changes in reactor water level. 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 concurrent with a loss of feedwater supply, makeup capabilities are provided by the RCIC system and/or HPCS system.

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

For further discussion, see the following sections:

Detection of Leakage Through the Reactor Coolant Pressure Boundary	5.2.5
Reactor Core Isolation Cooling System	5.4.6
Emergency Core Cooling System	6.3
Instrumentation and Controls	7

3.1.2.34 Residual Heat Removal (RHR) (Criterion 34)

Criterion

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

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

Design Conformance

The RHR system provides the means to:

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

2. Condense reactor steam so decay heat and residual heat can be removed if the normal heat sink is unavailable.

3. Maintain water level in the reactor vessel in conjunction with other ECCSs following a LOCA.

4. Remove energy from the drywell atmosphere in the form of hot water and cool any noncondensable gases in the free volume above the suppression pool following a LOCA.

The major equipment of the RHR system consists of two heat exchangers and three main system pumps. Equipment is connected by associated valves and piping, and controls and instrumentation are provided for proper system operation. The main system's pumps are sized on the basis of the flow required during the LPCI mode of operation. Heat exchangers are sized on the basis of the required duty for the shutdown cooling function which is the mode requiring the maximum heat exchanger area. Three independent loops are located in separate protected areas.

Pumps and heat exchangers are located in separate areas of the auxiliary bays, and the piping associated with each of the three loops is physically separated to provide redundancy and independence.

The RHR system is designed for three major modes of operation:

1. LPCI.

2. Containment cooling:

    a. Containment spray.
    b. Suppression pool cooling.

3. Shutdown cooling.

Both offsite ac power and emergency onsite power systems provide adequate power to operate all auxiliary loads necessary for RHR system operation. Power sources for plant power systems are sufficient in number, and of such electrical and physical independence that no single probable event could interrupt all safety-related power at one time.

Redundant loops of the RHR system are energized from separate and independent ac power buses of the plant power distribution system. Under normal operating conditions these buses are energized by two separate sources of offsite power supply. In the event of loss of offsite power (LOOP), buses providing power to the RHR are energized from two separate and independent onsite standby diesel generators.

The plant layout is designed to provide physical separation of redundant standby diesel generators, switchgear, load centers, MCCs, and other system components.

The RHR system is adequate to remove residual heat from the reactor core to ensure fuel and RCPB design limits are not exceeded. System components are physically and electrically separated and redundant onsite electric power systems are provided. Redundancy of system components as well as physical and electrical separation of system components enable the system to perform its safety function assuming a single failure. The design of the RHR system, including its power supply, meets the requirements of Criterion 34.

For further discussion, see the following sections:

Residual Heat Removal System	5.4.7
Containment Heat Removal System	6.2.2
Emergency Core Cooling System	6.3
Emergency Core Cooling Systems - Instrumentation and Control	7.3
Electric Power	8
Station Service Water System	9.2.1
Accident Analysis	15

3.1.2.35 Emergency Core Cooling System (ECCS) (Criterion 35)

Criterion

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

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

Design Conformance

The ECCS consists of the following:

1. HPCS system.

2. Automatic depressurization system (ADS).

3. Low-pressure core spray (LPCS) system.

4. LPCI - an operating mode of the RHR system.

The ECCS is designed to limit fuel cladding temperature to less than 10CFR50 Appendix K guidelines 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 HPCS system consists of a single motor-driven centrifugal pump, system piping, valves, controls, and instrumentation. The HPCS system ensures 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 RPV. The HPCS continues to operate when reactor vessel pressure is below the pressure at which LPCI or LPCS operation maintains core cooling. A source of water is available from either the condensate storage tank (CST) or the suppression pool.

The ADS functions to reduce the reactor pressure so that flow from LPCI and/or the LPCS enters the RPV in time to cool the core and prevent excessive fuel clad temperature. The ADS uses seven SRVs to relieve high-pressure steam to the suppression pool.

The LPCS system consists of: a centrifugal pump that can be powered by offsite power or standby ac power system; a spray sparger in the RPV above the core (separate from the HPCS sparger); piping and valves to convey water from the suppression pool to the sparger; and associated controls and instrumentation. In case of low water level in the RPV or high pressure in the drywell, the LPCS system automatically sprays water onto the top of the fuel assemblies in sufficient time and at a flow rate to cool the core and prevent excessive fuel temperature.

The LPCI system starts from the same signals that initiate the LPCS system and operates independently to achieve the same objective by flooding the RPV.

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

Results of the performance of the ECCS for the entire spectrum of liquid line breaks are discussed in Section 6.3. Peak cladding temperatures (PCT) are well below the 2,200°F design basis.

Also provided in Section 6.3.3 is an analysis showing that the ECCS conforms to 10CFR50 Appendix K. This analysis shows complete compliance with the final acceptance criteria with the following results:

1. Peak clad temperatures are well below the 2,200°F NRC acceptability limit.

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

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

4. The core temperature is reduced and decay heat removed for an extended period of time.

Redundancy and capability of the onsite electrical power systems for the ECCS are discussed in the evaluation of Criterion 34.

The ECCS is adequate to prevent fuel and clad damage that could interfere with effective core cooling and limits clad metal-water reaction to a negligible amount. Design of the ECCS, including power supplies, meets the requirements of Criterion 35.

For further discussion, see the following sections:

Residual Heat Removal System	5.4.7
Emergency Core Cooling System	6.3
Offsite Power Systems	8.2
Onsite Power Systems	8.3
Water Systems	9.2
Accident Analysis	15

3.1.2.36 Inspection of Emergency Core Cooling System (ECCS) (Criterion 36)

<u>Criterion</u>

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

Design Conformance

The ECCS is designed to permit appropriate periodic inspection in accordance with the intent of ASME Section XI. Spargers within the vessel are accessible for inspection during each refueling outage. Removable insulation is provided on the ECCS piping out to and including the first isolation valve outside the drywell to facilitate inspection. Inspection access doors are provided in the BSW; insulation covering the ECCS RPV nozzles is removable to permit inspection. Components inside the drywell can be inspected when the drywell is open for access. Remaining portions of the ECCS outside the drywell can be inspected at any time. Portions of the ECCS that are part of the RCPB are designed to specifications for ISI to detect defects that might affect the cooling performance. Particular inspection attention will be given to the reactor nozzles, core spray, and feedwater spargers. The design of the reactor vessel and internals for ISI, and the plant testing and inspection program ensures that the requirements of Criterion 36 are met.

For further discussion, see the following sections:

Reactor Pressure Vessel Internals	3.9.5
In-service Inspection and Testing of the Reactor Coolant Pressure Boundary	5.2.4
Reactor Vessel	5.3
Emergency Core Cooling System	6.3
In-service Inspection of ASME Safety Class 2 and Class 3 Components	6.6

3.1.2.37 Testing of Emergency Core Cooling System (ECCS) (Criterion 37)

Criterion

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

Design Conformance

The ECCS consists of the HPCS system, the ADS, the LPCI system, and the LPCS system. Each of these systems has test connections and isolation valves (except ADS) to permit periodic pressure testing to assure the structural and leak-tight integrity of its components. The HPCS, LPCS, LPCI, and ADS systems are designed to permit periodic testing to ensure operability and performance of the active components of each system.

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

The ECCS will be subjected to tests to verify performance of the full operational sequence that brings each system into operation. Operation of the associated cooling water systems is discussed in Criterion 46, Design Conformance. The design of the ECCS meets the requirements of Criterion 37.

For further discussion, see the following sections:

Overpressurization Protection	5.2.2
Reactor Coolant Pressure Boundary In-service Inspection and Testing	5.2.4
Tests and Inspections	6.3.4
In-service Inspection of ASME Safety Class 2 and Class 3 Components	6.6
Engineered Safety Feature Systems	7.3
Electric Power	8

3.1.2.38 Containment Heat Removal (Criterion 38)

Criterion

"A system to remove heat from the reactor containment shall be provided. The system safety function shall be to reduce rapidly, consistent with the functioning of other associated systems, the containment pressure and temperature following any 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 for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure."

Design Conformance

The containment heat removal function is accomplished by the RHR. Following a LOCA, one or both of the following operating modes of the RHR system will 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 activated.

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

For further discussion, see the following sections:

Residual Heat Removal System	5.4.7
Containment Systems	6.2
Standby Ac Power Supply and Distribution	8.3.1
Water Systems	9.2
Accident Analysis	15

3.1.2.39 Inspection of Containment Heat Removal System (Criterion 39)

Criterion

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

Design Conformance

Provisions are made to facilitate periodic inspections of active components and other important equipment of the containment heat removal systems. During plant operation the pumps, valves piping, instrumentation, wiring, and other components outside the drywell can be inspected at any time and will be inspected periodically. Testing frequencies of most components will be correlated with component inspection. The pressure suppression pool is designed to permit appropriate periodic inspection. Access is provided for inspections and maintenance.

The containment heat removal system design meets the requirements of Criterion 39.

For further discussion, see the following sections:

Residual Heat Removal System	5.4.7
Containment Systems	6.2
Emergency Core Cooling Systems	6.3
In-service Inspection of ASME Safety Class 2 and Class 3 Components	6.6
Water Systems	9.2

3.1.2.40 Testing of Containment Heat Removal System (Criterion 40)

Criterion

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

Design Conformance

The containment heat removal function is accomplished by either the containment spray or suppression pool cooling modes of the RHR system. The RHR system has sufficient test connections and isolation valves to permit periodic pressure and flow rate testing. Pumps and valves of the RHR are operated periodically in test modes that simulate actual pumping requirements to verify operability. The containment cooling modes are not automatically initiated, but operation of the components is periodically verified. Operation of associated cooling water systems is discussed in the response to Criterion 46. It is concluded that the requirements of Criterion 40 are met.

3.1.2.41 Containment Atmosphere Cleanup (Criterion 41)

Criterion

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

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

Design Conformance

During normal operation and following a LOCA, all releases from the reactor are confined within the primary containment. In the event of a LOCA, gaseous leakage from the primary containment is processed automatically by the standby gas treatment system (SGTS). For LOCAs with significant fuel damage, operation of the containment sprays removes airborne fission products to reduce radionuclide concentrations prior to leakage from the primary containment.

The hydrogen recombiner system is provided to control the post-LOCA concentration of hydrogen in the primary containment. This system recirculates a portion of the containment atmosphere through a recombiner to maintain hydrogen concentration below

Note: As per NRC revision of 10CFR50.44, which eliminated the design basis LOCA hydrogen release, along with NRC-approved License Amendment 124, which removed the hydrogen recombiner requirements from Technical Specifications, the hydrogen recombiners are no longer required for DBA LOCA hydrogen concentration control. The hydrogen recombiners do, however, remain necessary to ensure adequate atmospheric mixing in the primary containment during and following DBA LOCA. 5-volume percent. The primary containment purge system is available to be used as backup to the hydrogen recombiner system.

The SGTS, containment spray, and the hydrogen recombiner system are designed as redundant systems to ensure that failure of an active component or loss of either offsite power (normal operation) or standby diesel generators (emergency condition) power supplies would not impair the system's ability to perform its safety function. These systems satisfy the requirements of Criterion 41.

For further discussion, see the following sections:

Containment Sprays	6.2.2
Combustible Gas Control in Containment	6.2.5
Engineered Safety Feature Filter Systems	6.5.1
Electric Power	8
Reactor Building HVAC System	9.4.2

3.1.2.42 Inspection of Containment Atmosphere Cleanup Systems (Criterion 42)

Criterion

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

Design Conformance

The SGTS and hydrogen recombiner system are designed to allow appropriate periodic inspection of important system components and satisfy the requirements of Criterion 42.

For further discussion, see the following sections:

Combustible Gas Control in Containment	6.2.5
Fission Product Removal and Control Systems	6.5
Standby Gas Treatment System	6.5.1
Hydrogen Recombiner System	9.4.2

3.1.2.43 Testing of Containment Atmospheric Cleanup Systems (Criterion 43)

Criterion

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

1. The structural and leak-tight 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."

Design Conformance

The SGTS and hydrogen recombiner system are designed to permit appropriate periodic pressure and functional testing and satisfy the requirements of Criterion 43.

For further discussion, see the following sections:

Engineered Safety Feature Filter System	6.5.1
Testability of Offsite/Onsite Power Systems	8.2.1, 8.3.1.1
Reactor Building HVAC System	9.4.2

3.1.2.44 Cooling Water (Criterion 44)

Criterion

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

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

Design Conformance

The service water system (SWP) provides cooling water for removal of heat from all structures, systems, and components important to safety during all normal operating and accident conditions. The safety-related portion of the SWP is designed to ASME Section III Class 3 and seismic Category I design criteria.

Redundant safety-related components that receive cooling water from cooling water systems are supplied through redundant supply headers. Cooling water is returned to the discharge bay through redundant discharge headers. From the discharge bay the cooling water is returned to the UHS via the discharge structure. Electric power for the operation of redundant safety-related components of the SWP is supplied from separate redundant offsite and onsite power sources. No single failure can render the SWP incapable of performing its intended safety function. Thus, the requirements of Criterion 44 are satisfied.

Principal Design Criteria	1.2.1
Classification of Structures, Components, and Systems	3.2
Wind and Tornado Loadings	3.3
Water Level (Flood) Design	3.4
Design of Category I Structures	3.8
Electric Power	8
Service Water System	9.2.1
Ultimate Heat Sink	9.2.5

3.1.2.45 Inspection of Cooling Water System (Criterion 45)

Criterion

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

Design Conformance

The cooling water system is designed to permit periodic inspection of those portions of the SWP important to safety to ensure the integrity and capability of the system. Access is provided for inspection and maintenance. During plant operation, the pumps, valves, piping, instrumentation, wiring, and other components can be inspected at any time and will be inspected periodically. This design satisfies the requirements of Criterion 45.

For further discussion, see the following sections:

Principal Design Criteria	1.2.1
In-service Inspection of ASME Safety Class 2 and Class 3 Components	6.6
Service Water System	9.2.1
Initial Test Program	14

3.1.2.46 Testing of Cooling Water System (Criterion 46)

Criterion

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

Design Conformance

The SWP and closed loop cooling water system are in operation during normal unit operation and shutdown. Thus, component performance is continuously demonstrated.

The safety-related portion of the SWP is designed to permit periodic pressure and functional testing to assure structural and leak-tight integrity. In addition, this portion of the system is designed to permit, to the extent practicable, periodic operability testing with simulation of emergency reactor shutdown or LOCA conditions with or without coincident transfer between normal and emergency power sources. Criterion 46 is satisfied.

For further discussion, see the following sections:

Principal Design Criteria	1.2
In-service Testing of Pumps and Valves	3.9.6
In-service Inspection of ASME Safety Class 2 and Class 3 Components	6.6
Service Water System	9.2.1
Testability of Offsite/Onsite Power Systems	8.2.1, 8.3.1.1

Initial Test Program	14.0
Technical Specifications	16.0

3.1.2.47 (Not Promulgated by NRC)

3.1.2.48 (Not Promulgated by NRC)

3.1.2.49 (Not Promulgated by NRC)

3.1.2.50 Containment Design Basis (Criterion 50)

Criterion

"The reactor containment structure, including access openings, penetrations, and the containment heat removal system, shall be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any loss-of-coolant accident. This margin shall reflect consideration of (1) the effects of potential energy sources which have not been included in the determination of the peak conditions, such as energy in steam generators as required by paragraph 50.44, energy from metal-water and other chemical reactions that may result from degradation, but not total failure, of emergency core cooling functioning, (2) the limited experience and experimental data available for defining accident phenomena and containment responses, and (3) the conservatism of the calculational model and input parameters."

Design Conformance

The containment structure, including access openings and penetrations, is designed to accommodate, without exceeding the design leak rate, the transient peak pressure and temperature associated with a LOCA up to and including a double-ended rupture of the largest reactor coolant pipe.

The containment structure and ESF systems have been evaluated for various combinations of energy release. The analysis accounts for system thermal energy, chemical energy, and nuclear decay heat energy.

Maximum temperature and pressure reached in the primary containment drywell and suppression chamber during the worst-case accident are shown in Chapter 1 to be below the design temperature and pressure of this structure.

The cooling capacity of the containment heat removal systems is adequate to prevent overpressurization of the structure and return the containment to near atmospheric pressure. Therefore, Criterion 50 is met.

For further discussion, see the following sections:

Classification of Structures, Components, and Sytsems	3.2
Missile Protection	3.5
Protection Against Dynamic Effects Associated with the Postulated Rupture of Piping	3.6
Seismic Design	3.7
Design of Category I Structures	3.8
Engineered Safety Features	6

3.1.2.51 Fracture Prevention of Containment Pressure Boundary (Criterion 51)

Criterion

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

Design Conformance

The primary containment boundary is designed to the load combinations shown in Section 3.8 which cover 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 to compare with the allowable limits. The ferritic steel for the primary containment boundary has been qualified by metallurgical characterization, correlation with fracture toughness data, and evaluation to the criteria for Class 2 components identified in the Summer 1977 Addenda of Section III of the ASME Code. This will ensure nonbrittle behavior and minimize the probability of a rapidly propagating fracture under the above established conditions.

The preoperational test program and QA program ensure the integrity of the containment and its ability to meet all normal operating and accident pressures. Therefore, Criterion 51 is met.

For further discussion, see the following sections:

Design of Category I Structures	3.8
Air Conditioning, Heating, Cooling, and Ventilation Systems	9.4
Initial Test Program	14
Quality Assurance	17
Unit 2 Assessment of General Design Criteria 51 to 10CFR50	3D

3.1.2.52 Capability for Containment Leakage Rate Testing (Criterion 52)

Criterion

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

Design Conformance

The design of the reactor containment including all equipment and material subject to leakage rate test conditions is provided with means to facilitate periodic leakage rate tests throughout the plant lifetime. The testing program will be conducted in accordance with the requirements of Appendix J to 10CFR50. Therefore, Criterion 52 is met.

For further discussion, see Section 6.2.6, Containment Leakage Testing.

3.1.2.53 Provisions for Containment Testing and Inspection (Criterion 53)

Criterion

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

Design Conformance

The Unit 2 containment is designed for periodic inspection of all important areas and penetrations. The Unit 2 Technical

Specifications address the surveillance program for the containment and also testing at design pressure for leak-tightness. The periodic testing of the containment and penetrations for leak-tightness is described in detail in Section 6.2.6. Therefore, Criterion 53 is met.

3.1.2.54 Piping Systems Penetrating Containment (Criterion 54)

Criterion

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

Design Conformance

Piping systems penetrating containment are designed to provide the required isolation and testing capabilities. These piping systems have test connections to allow periodic leak detection tests to be performed.

The ESF actuation system test circuitry provides the means for testing isolation valve operability. Therefore, Criterion 54 is met.

For further discussion, see the following sections:

Reactor Coolant Pressure Boundary Penetrating Containment (Criterion 55)	3.1.2.55
Primary Containment Isolation (Criterion 56)	3.1.2.56
Closed-System Isolation Valves (Criterion 57)	3.1.2.57
Engineered Safety Features	6

3.1.2.55 Reactor Coolant Pressure Boundary Penetrating Containment (Criterion 55)

Criterion

"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; or

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

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

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

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

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

<u>Design Conformance</u>

The RCPB, as defined in 10CFR50 Section 50.2(V), consists of the RPV, pressure-retaining appurtenances attached to the vessel, and piping, pumps, and valves that are part of the RCS, or connected to the RCS, up to and including any and all of the following:

1. The outermost containment isolation valve in system piping that penetrates the primary reactor containment.

2. The second of two valves normally closed during normal reactor operation in system piping that does not penetrate primary reactor containment.

3. The RCS safety and relief valves. For a BWR, the RCS extends to and includes the outermost containment isolation valve in the main steam and feedwater piping. RCPB lines that penetrate the containment have isolation valves capable of isolating the containment

to preclude any significant releases of radioactivity. Lines that do not penetrate the containment but form a portion of the RCPB can be isolated from the RCPB.

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

For further discussion, see the following sections:

Integrity of Reactor Coolant Pressure Boundary	5.2
Containment Isolation System	6.2
Engineered Safety Feature Systems	7.3
Accident Analysis	15
Technical Specifications	16

3.1.2.56 Primary Containment Isolation (Criterion 56)

Criterion

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

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

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

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

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

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

Design Conformance

Criterion 56 requires that lines that penetrate the containment and communicate with the containment interior must have two isolation valves; one inside the containment and the other outside. It should be noted that this criterion does not reflect consideration of the BWR suppression pool design. For instance, lines connecting to the suppression pool do not have an isolation valve located inside the containment, as this would necessitate placement of the valve underwater. In effect, this would result in introducing a potentially unreliable valve in a highly-reliable system, thereby compromising design. For this reason, application of Criterion 56 to lines entering the suppression pool is not appropriate. Additionally, one of the Containment Purge & Vent System (CPS) lines, extending into the suppression chamber, is designed with both PCIVs outside of primary containment (see Figure 9.4-8k). While this does not meet Criterion 56 requirements, this configuration is consistant with all of the Mark II containment nuclear plants in the United States and is consistant with that described in the NMP2 Safety Evaluation report, NUREG-1047 and the current Standard Review Plan (SRP) for "Containment Isolation System" (SRP 6.4.2). Locating both PCIVs outside of Primary Containment improves the reliability of the valves, sisnce neither valve is subjected to the more severe environmental conditions within the suppression chamber. With the exception of the aforementioned lines, Criterion 56 is satisfied.

For further discussion, see the following sections:

Containment Isolation System	6.2
Engineered Safety Feature Systems	7.3
Accident Analysis	15
Technical Specifications	16

3.1.2.57 Closed System Isolation Valves (Criterion 57)

Criterion

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

Design Conformance

Each line that penetrates the reactor containment and is neither part of the RCPB nor connected directly to the containment atmosphere has at least one containment isolation valve that is automatic, locked closed, or capable of remote manual operation, located outside the containment as close to the containment as practical. Simple check valves are not used as automatic isolation valves on these lines. Therefore, Criterion 57 is met.

For further discussion, see the following sections:

Containment Isolation System	6.2.4
Engineered Safety Feature Systems	7.3

3.1.2.58 (Not Promulgated by NRC)

3.1.2.59 (Not Promulgated by NRC)

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

Criterion

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

Design Conformance

In all cases, the design for radioactivity control is based on (1) the requirements of 10CFR20, 10CFR50, and applicable regulations for normal operations and any transient situation that may reasonably be anticipated to occur; and (2) 10CFR100 dosage level guidelines for potential accidents of exceedingly low probability of occurrence. All releases are expected to be reported consistent with RG 1.21.

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

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

Solid wastes are prepared for offsite disposal by approved procedures. Shielded and reinforced containers that meet applicable NRC and Department of Transportation requirements are used for the shipment of solid wastes when use of unshielded containers exceeds the NRC mandated dose criteria (Section 11.4). Therefore, Criterion 60 is met.

For further discussion, see the following sections:

General Plant Description	1.2.1
Detection of Leakage through Reactor Coolant Pressure Boundary	5.2.5
Containment Systems	6.2
Liquid Waste Management Systems	11.2
Gaseous Waste Management Systems	11.3
Solid Waste Management System	11.4
Process and Effluent Radiological Monitoring and Sampling Systems	11.5
Accident Analysis	15

3.1.2.61 Fuel Storage and Handling and Radioactivity Control (Criterion 61)

Criterion

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

Design Conformance

New Fuel Storage New fuel is placed in dry storage in the new fuel storage vault located inside the reactor building, which provides adequate shielding for radiation protection. Storage racks preclude accidental criticality (see Design Conformance for Criterion 62). New fuel storage racks do not require inspection or testing for nuclear safety purposes.

Spent Fuel Handling and Storage Irradiated fuel is also stored in the reactor building. Fuel pool water is circulated through the spent fuel pool cooling and cleanup system (SFC) to maintain fuel pool water temperature, purity, and water clarity. A high water level in the spent fuel pool is lowered by runoff of excess water to the skimmer surge tanks. A low water level in the pool is raised by pumping water from the skimmer surge tanks which automatically receive makeup water from the condensate makeup and drawoff system. The SFC is designed with provisions to preclude siphoning or draining the spent fuel pool, as discussed in Section 9.1.4. Storage rack design precludes accidental criticality (see design conformance for Criterion 62). The fuel pool cooling system is designed for the maximum heat load from decaying fuel, including a full core discharge.

No tests are required for nuclear safety purposes. At least one pump and heat exchanger are normally in operation while fuel is stored in the pool. Duplicate units are operated periodically to handle abnormal heat loads or to replace a unit for servicing. The RHR heat exchangers are available as backup to the two independent spent fuel pool cooling loops. Routine visual inspection of the system, component instrumentation, trouble alarms, and IST are adequate to verify system operability.

Radioactive Waste Systems The radioactive waste systems provide all equipment necessary to collect, process, and prepare for disposal of all radioactive liquids, gases, and solid wastes produced as a result of reactor operation. Liquid radwastes are classified, contained, and treated as high- or low-conductivity chemical, detergent, sludge, or concentrated wastes. Processing includes filtration, ion exchange, analysis, and dilution. Liquid wastes are also decanted and sludge is accumulated for disposal as solid radwaste. Wet solid wastes are packaged in shielded steel drums. Dry solid radwastes are packaged in shielded steel drums or other suitable containers. Gaseous radwastes are monitored, processed, recorded, and controlled so that radiation doses to persons outside the controlled area are below those allowed by applicable regulations.

Accessible portions of the reactor and radwaste buildings have sufficient shielding to maintain dose rates within the limits set forth in 10CFR20 and 10CFR50. The radwaste building is designed to preclude accidental release of radioactive materials to the environs. Radwaste systems are used on a routine basis and do not require specific testing to ensure operability. Performance is monitored by radiation monitors during operation.

Fuel storage and handling and radioactive waste systems are designed to ensure adequate safety under normal and postulated accident conditions.

Radiation Shielding All plant system arrangements are reviewed for radiation shielding requirements, and shielding is provided to reduce operations personnel exposure as low as reasonably achievable (ALARA) levels.

Design of these systems meets the requirements of Criterion 61.

For further discussion, see the following sections:

Residual Heat Removal System	5.4.7
Containment Systems	6.2
Fuel Storage and Handling	9.1
Radioactive Waste Management	11
Radiation Protection	12
Accident Analysis	15

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

Criterion

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

Design Conformance

Appropriate plant fuel handling and storage facilities are provided to preclude accidental criticality for new and spent fuel. Criticality in new and spent fuel storage is prevented by the safe configuration of the storage rack. There is sufficient spacing between the assemblies in the new fuel racks to ensure that the array when fully loaded is substantially subcritical. The spent fuel pool uses poison 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 nonclosable drain to prevent accumulation of water. The new fuel storage vault racks (located inside the reactor building) are designed to prevent an accidental critical array, even in the event the vault becomes flooded or subjected to seismic loadings. Center-to-center new fuel assembly spacing limits the effective multiplication factor (k_{eff}) of the array to not more than 0.90 for new dry fuel. If the new fuel is flooded, k_{eff} does not exceed 0.95.

Spent fuel is stored underwater in high-density poison racks in the spent fuel storage pool. Racks in which spent fuel assemblies are placed are designed and arranged to ensure subcriticality in the storage pool. Spent fuel is maintained at a subcritical multiplication factor k_{eff} of less than 0.95 under normal and abnormal conditions. Abnormal conditions may result from earthquake, accidental dropping of equipment, or damage caused by the horizontal movement of fuel handling equipment without first disengaging the fuel from the hoisting equipment.

Refueling interlocks include circuitry that senses conditions of the refueling equipment and 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 malfunction.

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

For further discussion, see the following sections:

New Fuel Storage	9.1.1
Spent Fuel Storage	9.1.2
Fuel Handling System	9.1.4

3.1.2.63 Monitoring Fuel and Waste Storage (Criterion 63)

Criterion

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

Design Conformance

Appropriate systems have been provided to meet the requirements of this criterion. A malfunction of the SFC that could result in loss of RHR capability or excessive radiation levels is alarmed in the control room. Alarmed conditions include high/low water levels in the spent fuel pool and skimmer surge tanks, low pressure at the discharge of the cooling water pumps, and low flow through the cooling water heat exchangers. System temperature is also continuously monitored and alarmed in the control room. Area radiation monitors sense radioactivity in this area and initiate an alarm in the control room on abnormal radiation levels.

Area radiation and tank and sump levels are monitored throughout the plant and alarmed to indicate conditions that may result in excessive radiation levels in radioactive waste or ventilation systems. These systems satisfy the requirements of Criterion 63.

For further discussion, see the following sections:

Fuel Storage and Handling	9.1
Liquid Waste Systems	11.2
Gaseous Waste Systems	11.3
Solid Waste System	11.4
Area Radiation and Airborne Radioactivity Monitoring and Instrumentation	12.3.4

3.1.2.64 Monitoring Radioactivity Releases (Criterion 64)
Criterion

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

Design Conformance

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

1. Liquid discharge to discharge tunnel.

2. Radwaste and reactor building vents.

3. Gaseous releases from the main stack.

During normal plant operation and anticipated operational occurrences, the drywell atmosphere is continuously monitored.

In the event of an accident, the radioactivity level of the containment atmosphere is monitored by radiation elements located inside containment. All potential release points are monitored during the accident condition by individual monitors for each release path and/or by the main stack exhaust monitors. Radioactivity levels in the environs for both normal and accident conditions are monitored by the offsite radiological monitoring program.

Annual reports of operation are submitted to the NRC. These reports include specific information concerning the quantities of principal radionuclides released to the environs, and are submitted within 60 days after each successive 12-month operating period. Therefore, Criterion 64 is met.

For further discussion, see the following sections:

Integrity of Reactor Coolant Pressure Boundary	5.2
Air Conditioning, Heating, Cooling, and Ventilation Systems	9.4
Radioactive Waste Management	11.1-11.5
Radiation Protection Design Features	12.3

3.2 CLASSIFICATION OF STRUCTURES, SYSTEMS, AND COMPONENTS

3.2.1 Seismic Classification

The seismic classification for Unit 2 structures, systems, and components is listed in Table 3.2-1. The classification meets the intent of RG 1.29, except as otherwise noted in the table.

Seismic Category I structures, systems, and components are those necessary to ensure:

1. The integrity of the RCPB.

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

3. The capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the guideline exposures of 10CFR100.

Seismic Category I structures, systems, and components, including their foundations and supports, are designed to withstand the effects of a safe shutdown earthquake (SSE) and remain functional. The term Category I Structures used elsewhere in this section means Seismic Category I Structures as defined herein.

All seismic Category I structures, systems, and components are analyzed for the loading conditions of the SSE and the operating basis earthquake (OBE). Since the two earthquakes have different intensities, the design of seismic Category I structures, components, equipment, and systems to resist each earthquake and other loads is based on levels of material stress or load factors, whichever are applicable, and provides margins of safety appropriate for each earthquake. The margin of safety provided for structures, components, and systems important to safety for the SSE is sufficiently large to assure that their design functions are not jeopardized.

For further details of seismic design criteria, refer to the following sections:

Seismic Design	3.7
Design of Category I Structures	3.8
Mechanical Systems and Components	3.9
Seismic Qualification of Category I Instrumentation and Electrical Equipment	3.10
Design Assessment Report for Hydrodynamic Loads	Appendix 6A

3.2.2 System Quality Group Classifications

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

1. Prevent or mitigate the consequence of accidents and malfunctions originating within the RCPB.

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

3. Contain radioactive material.

A tabulation of quality group classifications for each component so defined is shown in Table 3.2-1 under the heading Quality Group Classification. Corresponding design and fabrication requirements are provided in Table 3.2-2. Figure 3.2-1 depicts the relative locations of these components along with their quality group classifications. For a more detailed guide to quality group boundaries for each safety-related system, refer to the system diagram given in the applicable system section of this Final Safety Analysis Report (FSAR).

Table 3.2-4 identifies the code, code edition, and addenda used in the construction of each Quality Group A (ASME Section III, Class I) component in the RCPB.

3.2.3 Quality Assurance

Structures, systems, and components whose safety functions require conformance to the QA requirement of 10CFR50 Appendix B are summarized in Table 3.2-1 under the heading Quality Assurance Requirement. The QA program is described in Chapter 17.

3.2.4 Correlation of Safety Classes with Industry Codes

The design of plant equipment is commensurate with the safety importance of the equipment. Hence, the various safety classes have a gradation of design requirements. The correlation of safety classes with other design requirements is summarized in Tables 3.2-2 and 3.2-3.

TABLE 3.2-1
(Sheet 1 of 38)

EQUIPMENT AND STRUCTURE CLASSIFICATION

	Scope of Supply	Location	Electrical Classifi-cation	Seismic Category	Quality Group Classifi-cation	QA Requirement (31,32,33,34)	Tornado Protection	Notes
Reactor System								
Reactor vessel	GE	PC	NA	I	A	I	P	
Reactor vessel support skirt	GE	PC	NA	I	A	I	P	
Reactor vessel appurtenances, pressure-retaining portions	GE	PC	NA	I	A	I	P	
CRD housing supports	GE	PC	NA	I	NA	I	P	
Reactor internal structures, engineering safety features	GE	PC	NA	I	NA	I	P	(1)
Reactor internal structures, other	GE	PC	NA	NA	NA	NA	P	(2,34)
Control rods	GE	PC	NA	I	NA	I	P	
Control rod drives	GE	PC	NA	I	NA	I	P	
Core support structure	GE	PC	NA	I	NA	I	P	
Fuel assemblies	GE	PC	NA	I	NA	I	P	
Reactor vessel stabilizer	GE	PC	NA	I	NA	I	P	(30)
Reactor vessel insulation	P	PC	NA	I	NA	NA	P	(34)
Nuclear Boiler System								
Instrumentation condensing chambers	GE	PC	NA	I	A	I	P	(38)
SRV air accumulators	P	PC	NA	I	B	I	P	
Piping, SRV discharge	P	PC	NA	I	C	I	P	
Piping, main steam within outermost isolation valve	P	PC	NA	I	A	I	P	(3)
Pipe supports, main steam within outermost isolation valve	P	PC	NA	I	A	I	P	
Pipe whip restraints, main steam, and feedwater	P	PC,RB	NA	I	NA	I	P	
Piping, feedwater within outermost isolation valve	P	PC	NA	I	A	I	P	
Piping, other RCPB piping within outermost isolation valve	P	PC	NA	I	A	I	P	(3)
Piping, instrumentation beyond outermost isolation valve	P	RB,TB	NA	I or NA	B or D	I or NA	P	(3)
Safety/relief valves	GE	PC	1E	I	A	I	P	(38)
Valves, main steam isolation valves (MSIV)	GE	PC,RB	1E	I	A	I	P	
Valves, feedwater isolation valves	P	PC,RB	1E	I	A	I	P	
Valves, other isolation valves and within outermost isolation valve	P	PC,RB	1E	I	A	I	P	(3)

TABLE 3.2-1
(Sheet 2 of 38)

EQUIPMENT AND STRUCTURE CLASSIFICATION

	Scope of Supply	Location	Electrical Classifi-Cation	Seismic Category	Quality Group Classifi-cation	QA Requirement (31,32,33,34)	Tornado Protection	Notes
Valves, instrumentation beyond outermost isolation valve	P	RB	NA	I or NA	B or D	I or NA	P	(3)
Instrumentation modules (mechanical portion) with safety function	GE	RB	NA	I	B	I	P	
Instrumentation modules (electrical portion) with safety function	GE	RB	1E	I	NA	I	P	
Cable	P	C,RB,M	1E	NA	NA	I	P	
Cable trays and fabricated supports with safety function	P	C,RB,M	NA	I	NA	I	P	
T-quenchers	P	PC	NA	I	C	I	P	
<u>Recirculation System</u>								
Piping, essential	GE,P	PC,RB	NA	I	A,B	I	P	(46)
Pipe suspension, recirculation line	GE	PC	NA	I	A	I	P	
Pipe restraints, recirculation line	GE	PC	NA	I	NA	I	P	(4)
Pumps	GE	PC	NA	I	A	I	P	
Valves, essential, including containment isolation	GE,P	PC,RB	1E	I	A,B,C	I	P	(24)
Piping and valves, other	P	RB,PC	NA or Non-1E	NA	D	NA	P	(25)
Pump motors	GE	PC	Non-1E	NA	NA	NA	P	
Electrical modules with safety function	GE	RB	1E	I	NA	I	P	
Cable	P	C,RB,M	1E	NA	NA	I	P	
Cable trays and fabricated supports with safety function	P	C,RB,M	NA	I	NA	I	P	
LFMG set	GE	N	Non-1E	NA	NA	NA	NR	
Piping, hydraulic lines	P	PC,RB	NA	I	D	NA	P	
<u>CRD Hydraulic System</u>								
Valves, scram discharge volume lines	GE	RB	NA	I	B	I	P	
Valves, insert and withdraw lines	P	RB	NA	I	B	I	P	(5)
Valves, other	GE,P	RB	Non-1E	NA	D	NA	P	
Piping, scram discharge volume lines	P	RB	NA	I	B	I	P	
Piping, insert and withdraw lines	P	PC,RB	NA	I	B	I	P	(5)
Piping, other	P	RB	NA	NA	D	NA	P	
Hydraulic control unit	GE	RB	NA	I	Special	I	P	(6)
CRD pumps, filters and strainers	GE	RB	Non-1E	NA	D	NA	P	
Electric modules with safety function	GE	RB	1E	I	NA	I	P	
Cable	P	C,RB,M	1E	NA	NA	I	P	
Cable trays, and fabricated supports with safety function	P	C,RB,M	NA	I	NA	I	P	
Scram discharge volume header	P	RB	NA	I	B	I	P	

TABLE 3.2-1
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EQUIPMENT AND STRUCTURE CLASSIFICATION

	Scope of Supply	Location	Electrical Classifi-Cation	Seismic Category	Quality Group Classifi-cation	QA Requirement (31,32,33,34)	Tornado Protection	Notes
Standby Liquid Control System								
Standby liquid control storage tank	GE	RB	NA	I	B	I	P	
Pumps	GE	RB	NA	I	B	I	P	
Pump motors	GE	RB	1E	I	NA	I	P	
Valves, explosive	GE	RB	1E	I	B	I	P	
Valves, isolation and within primary containment	P	PC,RB	1E	I	A	I	P	
Valves, beyond isolation valves	P	RB	1E	I	B	I	P	
Piping, downstream of containment valves	P	PC,RB	NA	I	A	I	P	
Piping, upstream of containment valves	P	RB	NA	I	B	I	P	
Electrical modules with safety function	GE	RB	1E	I	NA	I	P	
Cable	P	C,RB,M	1E	NA	NA	I	P	
Cable trays and fabricated supports with safety function	P	C,RB,M	NA	I	NA	I	P	
Test tank	P	RB	NA	NA	D	NA	P	
Piping and valves, other	P	RB	NA	NA	D	NA	P	
Neutron Monitoring System								
Piping, TIP, isolation	P,GE	PC,RB	NA	I	B	I	P	(42,46)
Valves, isolation, TIP subsystem	GE	RB	Non-1E	I	B	I	P	
Electrical modules, IRM, SRM, APRM and OPRM	GE	RB	1E	I	NA	I	P	
Cable, IRM, SRM, and APRM	P	PC,RB	1E	NA	NA	I	P	
Reactor Protection System								
Electrical modules	GE	C,PC,RB,T	1E	I	NA	I	P	
Cable	P	C,PC,RB,T	1E	NA	NA	I	P	
Leak Detection System								
Temperature elements (sensors)	GE	PC,RB,M	1E	I	NA	I	P	
Temperature switches	GE	C	1E	I	NA	I	P	
Pressure transmitters	GE	C	1E	I	NA	I	P	
Pressure switches	GE	C	1E	I	NA	I	P	
Differential temperature switches	GE	C	1E	I	NA	I	P	
Differential pressure switches	GE	C	1E	I	NA	I	P	
Differential pressure transmitters	GE	C	1E	I	NA	I	P	
Flow transmitters	GE	RB	1E	I	NA	I	P	
Differential flow switches	GE	C	1E	I	NA	I	P	

TABLE 3.2-1
(Sheet 4 of 38)

EQUIPMENT AND STRUCTURE CLASSIFICATION

	Scope of Supply	Location	Electrical Classifi-Cation	Seismic Category	Quality Group Classifi-cation	QA Requirement (31,32,33,34)	Tornado Protection	Notes
Primary containment radiation monitors (containment atmosphere monitoring system)	P	RB	1E	I	NA	I	P	
Drywell floor and equipment drain tank level transmitters	P	RB	Non-1E	NA	NA	NA	P	
Reactor building floor drain sump level switches	P	RB	Non-1E	NA	NA	NA	P	
Reactor building equipment drain tank level switches	P	RB	Non-1E	NA	NA	NA	P	
Differential flow summers	GE	M	1E	I	NA	I	P	
Timer switches	GE	M	1E	I	NA	I	P	
Reactor building general area flood level switches	P	RB	1E	I	NA	I	P	
ECCS pump room flood level switches	P	RB	1E	I	NA	I	P	
Power Supplies	GE	M	1E	I	NA	I	P	
<u>Area, Process, and Effluent Radiation Monitors</u>								
Nonsafety plant area monitors	P	RB,M,T,W	Non-1E	NA	NA	NA	P,NR	
Main steam line monitors	GE	M	1E	I	NA	I	P	
Process ventilation monitors for control room and reactor building with isolation signals	P	RB,C	1E	I	NA	I	P	
Process and effluent liquid monitors on service water system	P	M	1E	I	NA	I	P	
High-range containment area monitors (NUREG-0737, Item II.F.1)	P	PC	1E	I	NA	I	P	
Effluent monitors with high-range capabilities (NUREG-0737, Item II.F.1)	P	T,M	Non-1E	NA	NA	NA	P,NR	(27)
Nonsafety process and effluent monitors on liquid and gaseous radwaste, reactor and turbine water, circulating water, spent fuel cooling and cleanup, and standby gas treatment (normal drywell purge) systems	P	RB,M,T,W	Non-1E	NA	NA	NA	P,NR	
Cable	P	C,RB,M	1E	NA	NA	I	P	
Cable trays and fabricated supports with safety function	P	C,RB,M	NA	I	NA	I	P	

TABLE 3.2-1
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EQUIPMENT AND STRUCTURE CLASSIFICATION

	Scope of Supply	Location	Electrical Classification	Seismic Category	Quality Group Classification	QA Requirement (31,32,33,34)	Tornado Protection	Notes
Residual Heat Removal (RHR) System								
Heat exchangers, primary side	GE	RB	NA	I	B	I	P	
Heat exchangers, secondary side	GE	RB	NA	I	C	I	P	
Piping, connected to RCPB within outermost isolation valves	P	PC,RB	NA	I	A	I	P	(3)
Piping, other	P	PC,RB	NA	I	B,C	I	P	
Pumps	GE,P	RB	NA	I	B	I	P	
Pump motors	GE,P	RB	1E	I	NA	I	P	
Pump suction strainers in suppression pool	P	PC	NA	I	B	I	P	(43)
Containment spray nozzles	P	PC	NA	I	B	I	P	
Valves, isolation, RCPB	P	PC,RB	1E	I	A	I	P	
Valves, other	P	PC,RB	1E	I	B,C	I	P	
Electrical modules with safety function	GE	RB	1E	I	NA	I	P	
Cable	P	C,RB,M	1E	NA	NA	I	P	
Cable trays and fabricated supports with safety function	P	C,RB,M	NA	I	NA	I	P	
Pipe whip restraints	P	PC	NA	I	NA	I	P	
Low-Pressure Core Spray (LPCS) System								
Piping, connected to RCPB within outermost isolation valves	P	PC,RB	NA	I	A	I	P	(3)
Piping, other	P	RB,PC	NA	I	B	I	P	(43)
Pumps	GE,P	RB	NA	I	B	I	P	
Pump motors	GE,P	RB	1E	I	NA	I	P	
Valves, isolation, RCPB	P	PC,RB	1E	I	A	I	P	
Valves, other	P	RB	1E	I	B	I	P	
Electrical modules with safety function	P	RB	1E	I	NA	I	P	
Cable	P	C,RB,M	1E	NA	NA	I	P	
Cable trays and fabricated supports with safety function	P	C,RB,M	NA	I	NA	I	P	
Pipe whip restraints	P	PC	NA	I	NA	I	P	
High-Pressure Core Spray (HPCS) System								
Piping, connected to RCPB within outermost isolation valves	P	PC,RB	NA	I	A	I	P	(3)
Piping, other	P	RB,PC	NA	I	B	I	P	(43)
Piping, return test line to condensate storage tank beyond second isolation valve	P	RB,M	NA	NA	D	NA	P,NR	

TABLE 3.2-1
(Sheet 6 of 38)

EQUIPMENT AND STRUCTURE CLASSIFICATION

	Scope of Supply	Location	Electrical Classification	Seismic Category	Quality Group Classification	QA Requirement (31,32,33,34)	Tornado Protection	Notes
Piping and valves, suction line from condensate storage tank to the radwaste building tunnel	P	M	NA	NA	D	NA	NR	
Pumps	GE,P	RB	NA	I	B	I	P	
Pump motors	GE,P	RB	1E	I	NA	I	P	
Valves, isolation, RCPB	P	PC,RB	1E	I	A	I	P	
Valves, other	P	RB	1E	I	B	I	P	
Electrical modules with safety function	GE	RB	1E	I	NA	I	P	
Cable	P	C,RB,M	1E	NA	NA	I	P	
Cable trays and fabricated supports with safety function	P	C,RB,M	NA	I	NA	I	P	
Pipe whip restraints	P	PC	NA	I	NA	I	P	
<u>Reactor Core Isolation Cooling (RCIC) System</u>								
Piping, connected to RCPB within outermost isolation valves	P	PC,RB	NA	I	A	I	P	(3)
Piping, other	P	RB,PC	NA	I	B	I	P	(43)
Piping, return test line to condensate storage tank beyond second isolation valve	P	M,RB	NA	NA	D	NA	NR,P	
Piping and valves, suction line from condensate storage tank to radwaste building tunnel	P	M	NA	NA	D	NA	NR	
Pumps, RCIC and system pressure	GE,P	RB	NA	I	B	I	P	
System pressure pump motor	P	RB	1E	I	NA	I	P	
Turbine	GE	RB	NA	I	NA	I	P	(7)
Valves, valve motors, isolation, RCPB	P	PC,RB	1E	I	A	I	P	
Valves, other	P	RB	1E	I	B	I	P	
Electrical modules with safety function	GE	RB	1E	I	NA	I	P	
Cable	P	C,RB,M	1E	NA	NA	I	P	
Cable trays and fabricated supports with safety function	P	C,RB,M	NA	I	NA	I	P	
Pipe whip restraints	P	PC,RB	NA	I	NA	I	P	
<u>Fuel Service Equipment</u>								
Fuel preparation machine	GE	RB	NA	I	NA	I	P	
General purpose grapple	GE	RB	NA	NA	NA	I	P	(8)

TABLE 3.2-1
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EQUIPMENT AND STRUCTURE CLASSIFICATION

	Scope of Supply	Location	Electrical Classifi-cation	Seismic Category	Quality Group Classifi-cation	QA Requirement (31,32,33,34)	Tornado Protection	Notes
Reactor Vessel Service Equipment								
Steam line plugs	GE	RB	NA	NA	NA	I	P	
Dryer and separator sling and head Strongback	P	RB	NA	NA	NA	I	P	(8)
In-vessel Service Equipment								
Auxiliary service platform	P	RB	NA	NA	NA	NA		
Control rod grapple	GE	RB	NA	NA	NA	I	P	(8)
Refueling Equipment								
Refueling equipment platform assembly	GE	RB	NA	I	NA	I	P	(37)
Refueling bellows	P	RB	NA	NA	NA	NA	P	
Spent fuel pool liner	P	RB	NA	I	NA	I	P	
Storage Equipment								
Fuel storage racks	GE,P	RB	NA	I	NA	I	P	
Fuel storage container	GE	RB	NA	I	NA	I	P	
Radwaste Management Systems								
Liquid Radwaste Systems								
Tanks and vessels	P	RB,W	NA	NA	D	NA	P,NR	(10,11)
Heat exchangers	P	W	NA	NA	D	NA	NR	(11)
Piping	P	RB,W	NA	NA	D	NA	P,NR	(11)
Valves	P	RB,W	Non-1E	NA	D	NA	P,NR	(11)
Pumps	P	RB,W	Non-1E	NA	D	NA	P,NR	(11)
Solid Radwaste System								
Tanks and vessels	P	W	NA	NA	D	NA	NR	(11,12)
Heat exchangers	P	W	NA	NA	D	NA	NR	(11,12)
Piping	P	W	NA	NA	D	NA	NR	(11,12)
Valves	P	W	Non-1E	NA	D	NA	NR	(11,12)
Pumps	P	W	Non-1E	NA	D	NA	NR	(11,12)
Offgas System								
Tanks and vessels	P	T	NA	NA	D	NA	NR	(11)
Heat exchangers	P	T	NA	NA	D	NA	NR	(11)

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EQUIPMENT AND STRUCTURE CLASSIFICATION

	Scope of Supply	Location	Electrical Classification	Seismic Category	Quality Group Classification	QA Requirement (31,22,33,34)	Tornado Protection	Notes
Piping	P	T	NA	NA	D	NA	NR	(11)
Valves	P	T	Non-1E	NA	D	NA	NR	(11)
Pumps	P	T	Non-1E	NA	D	NA	NR	(11)
Mechanical modules	P	T	NA	NA	D	NA	NR	(11)
<u>Reactor Water Cleanup System</u>								
Vessels, filter/demineralizers	GE	RB	NA	NA	C	NA	P	
Heat exchangers, reactor water sides	GE	RB	NA	NA	C	NA	P	
Heat exchanger, cooling water side	GE	RB	NA	NA	C	NA	P	
Piping, within outermost isolation valves	P	PC,RB	NA	I	A	I	P	(3)
Piping, beyond outermost isolation valves	P	RB	NA	I	C	I	P	
Piping, auxiliary	P	M,RB,T,W	NA	NA	D	NA	P,NR	
Pumps	GE	RB	NA	NA	C	NA	P	
Pump motors	GE	RB	Non-1E	NA	NA	NA	P	
Valves, isolation and within outermost isolation valves	P	PC,RB	1E	I	A	I	P	
Valves, beyond outermost isolation valves	a) GE	RB	Non-1E	NA	C	NA	P	
	b) P	RB	Non-1E	NA	C	I	P	(35)
Valves, auxiliary	GE	RB	Non-1E	NA	D	NA	P	
Electrical modules with safety function	P	RB	1E	I	NA	I	P	
Cable	P	C,RB,M	1E	NA	NA	I	P	
Cable trays and fabricated supports with safety function	P	C,RB,M	NA	I	NA	I	P	
Pipe whip restraints	P	PC,RB	NA	I	NA	I	P	
<u>Post-accident Sampling System</u>								
Sample panel piping station and control panel	GE	T	Non-1E	NA	D	NA	NR	(34)
Other piping tubing, valves, and Components	P	RB,T	Non-1E	NA	D	NA	NR	
<u>Containment Atmosphere Monitoring System</u>								
Hydrogen/oxygen analyzers	P	RB	1E	I	NA	I	P	
Piping	P	PC,RB	NA	I	B	I	P	
Valves, automatic - isolation and other	P	PC,RB	1E	I	B	I	P	
Valves, manual	P	PC,RB	NA	I	B	I	P	
Electrical modules with safety functions	P	PC,RB	1E	I	NA	I	P	
Cable	P	PC,RB,C	1E	NA	NA	I	P	

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EQUIPMENT AND STRUCTURE CLASSIFICATION

	Scope of Supply	Location	Electrical Classifi-cation	Seismic Category	Quality Group Classifi-cation	QA Requirement (31,32,33,34)	Tornado Protection	Notes
Cable trays and fabricated supports with safety function	P	C,PC,RB,M	1E	I	NA	I	P	
<u>Containment Leakage Monitoring System</u>								
Valves, isolation	P	PC,RB	1E	I	B	I	P	
Valves, manual	P	PC,RB	NA	NA	D	NA	P	
Piping, essential	P	PC,RB	NA	I	B	I	P	
Piping, nonessential	P	PC,RB	NA	NA	D	NA	P	
<u>Fuel Pool Cooling and Cleanup System</u>								
<u>Fuel Pool Cleanup Subsystem</u>								
Vessels, filter demineralizers	P	RB	NA	NA	D	NA	P	
Piping	P	RB	NA	NA	D	NA	P	
Valves	P	RB	Non-1E	NA	D	NA	P	
Pumps, holding and mixing	P	RB	Non-1E	NA	D	NA	P	
<u>Fuel Pool Cooling Subsystem</u>								
Tanks, skimmer surge	P	RB	NA	I	C	I	P	
Heat exchangers	P	RB	NA	I	C	I	P	
Pumps, cooling	P	RB	NA	I	C	I	P	
Pump motors	P	RB	1E	I	NA	I	P	
Piping, safety related	P	PC,RB	NA	I	B,C	I	P	
Piping, nonsafety related	P	PC,RB	NA	NA	D	NA	P	
Valves, safety related	P	PC,RB	1E	I	B,C	I	P	
Valves, containment isolation	P	PC,RB	NA	I	B	I	P	
<u>Control Room Panels</u>								
Electrical modules with safety function	GE	C	1E	I	NA	I	P	
Cables	P	C,RB,M	1E	NA	NA	I	P	
Cable trays and fabricated supports with safety function	P	C	NA	I	NA	I	P	
<u>Local Control Panels and Racks</u>								
Electrical modules with safety function	GE,F	RB	1E	I	NA	I	P	
Cables	P	C,RB,M	1E	NA	NA	I	P	
Cable trays and fabricated supports with safety function	P	C,RB	1E	I	NA	I	P	

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EQUIPMENT AND STRUCTURE CLASSIFICATION

	Scope of Supply	Location	Electrical Classification	Seismic Category	Quality Group Classification	QA Requirement (31,32,33,34)	Tornado Protection	Notes
Remote shutdown panel	P	C	1E	I	NA	I	P	
Controls/instruments with safety function	GE,P	C,RB	1E	I	NA	I	P	
Controls/instruments with nonsafety Functions	P	C	Non-1E	NA	NA	NA	P	
<u>Instrument Air System</u>								
ADS accumulators	P	PC	NA	I	B	I	P	
ADS piping lines between accumulators and safety-related equipment	P	PC	NA	I	C	I	P	
ADS valves in lines between accumulators and safety-related equipment	P	PC	1E	I	C	I	P	
ADS piping lines for long-term makeup from outside the standby gas treatment building (nitrogen system)	P	RB,O	NA	I	C	I	P	
ADS valves for long-term makeup from outside the standby gas treatment building (nitrogen system)	P	RB,O	NA	I	C	I	P	
ADS piping containment isolation	P	PC,RB	NA	I	B	I	P	
ADS valves containment isolation	P	PC,RB	1E	I	B	I	P	
ADS instrumentation	P	PC,RB	1E	I	C,NA	I,NA	P	
Vessels, accumulators, supporting safety-related equipment	P	PC,RB	NA	I	C	I	P	
Piping in lines between accumulators and safety-related equipment	P	PC,RB	NA	I	C	I	P	
Valves in lines between accumulators and safety-related equipment	P	PC,RB	1E	I	C	I	P	
Piping containment isolation	P	PC,RB	NA	I	B	I	P	
Valves containment isolation	P	PC,RB	1E	I	B	I	P	
Electrical modules with safety function	P	PC	1E	I	NA	I	P	
Cables	P	C,RB,M	1E	NA	NA	I	P	
Cable trays and fabricated supports with safety function	P	C,RB,M	NA	I	NA	I	P	
Piping, nonessential	P	C,PC,RB, M,T,P,S,W	NA	NA	D	NA	P,NR	(50)
Valves, nonessential	P	C,PC,RB, M,T,P,S,W	Non-1E	NA	D	NA	P,NR	(50)
Other equipment	P	RB,T	NA	NA	D	NA	P,NR	

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EQUIPMENT AND STRUCTURE CLASSIFICATION

	Scope of Supply	Location	Electrical Classification	Seismic Category	Quality Group Classification	QA Requirement (31,32,33,34)	Tornado Protection	Notes
Service and Breathing Air Systems								
Piping, containment isolation	P	PC,RB	NA	I	B	I	P	(42)
Valves, containment isolation	P	PC,RB	1E	I	B	I	P	(42)
Electric modules with safety function	P	PC,RB	1E	I	NA	I	P	
Cables	P	C,RB,M	1E	NA	NA	I	P	
Cable trays and fabricated supports with safety function	P	C,RB,M	NA	I	NA	I	P	
Piping, other	P	C,PC,RB, M,T,P,S,W	NA	NA	D	NA	P,NR	
Valves, other	P	C,PC,RB, M,T,P,S,W	Non-1E	NA	D	NA	P,NR	
Other equipment	P	T	NA	NA	D	NA	NR	
Service Water System								
Piping, for essential components	P	C,M,P,RB, S	NA	I	C	I	P	(46)
Piping, for nonessential components	P	M,P,RB,T	NA	NA	D	NA	P,NR	
Valves, for essential components	P	C,M,P,RB, S	1E	I	C	I	P	
Valves, for nonessential components	P	M,P,RB,T	Non-1E	NA	D	NA	P,NR	
Pumps	P	P	NA	I	C	I	P	
Pump motors	P	P	1E	I	C	I	P	
Strainers, self-cleaning	P	P	1E	I	C	I	P	
Electrical modules with safety function	P	C,M,P,RB, S	1E	I	NA	I	P	
Cable	P	C,RB,M	1E	NA	NA	I	P	
Cable trays and fabricated supports with safety function	P	C,RB,M	NA	I	NA	I	P	
Service Water Chemical Treatment System								
Piping, valves, pumps located above el 261 ft in screenwell building	P	P	NA	NA	D	NA	NR	
Piping, valves located below el 261 ft in screenwell building	P	P	NA	I	D	I	P	
Electrical equipment	P	P	Non-1E	NA	NA	NA	NR	
Reactor Building Closed Loop Cooling Water System								
Piping, between containment isolation valves	P	PC,RB	NA	I	B	I	P	
Piping, for essential components	P	PC,RB	NA	I	C	I	P	
Piping, for nonessential components	P	PC,RB,T	NA	NA	D	NA	P,NR	
Valves, isolation	P	PC,RB	1E	I	B	I	P	
Valves, for essential components	P	RB	1E	I	C	I	P	
Valves for nonessential components	P	PC,RB,T	Non-1E	NA	D	NA	P,NR	
Pumps	P	RB	Non-1E	NA	D	NA	P	

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EQUIPMENT AND STRUCTURE CLASSIFICATION

	Scope of Supply	Location	Electrical Classification	Seismic Category	Quality Group Classification	QA Requirement (31,32,33,34)	Tornado Protection	Notes
Reactor Building Closed Loop Cooling Water System (cont'd.)								
Heat exchangers	P	RB	NA	NA	D	NA	P	
Expansion tank and strainers	P	RB	NA	NA	D	NA	P	
Turbine Building Closed Loop Cooling Water System								
Piping	P	T,W	NA	NA	D	NA	NR	
Valves	P	T,W	Non-1E	NA	D	NA	NR	
Heat exchangers	P	T	NA	NA	D	NA	NR	
Pumps	P	T	Non-1E	NA	D	NA	NR	
Power Conversion System								
Main steam piping between outermost isolation valves up to but not including turbine stop valves	P	RB,T,M	NA	I	D	I	P	(13,14)
Main steam branch piping to first valve that leads to the moisture separator reheater, turbine gland seal system, or auxiliary steam header	P	T	NA	I	D	I	NR	(13)
Main turbine bypass piping up to bypass valve	P	T	NA	I	D	I	NR	(13)
First valve that is normally closed or that leads to the moisture separator reheater, turbine gland seal system, or auxiliary steam header in branch piping connected to main steam and turbine bypass piping	P	T	Non-1E	NA	D	NA	NR	(14)
Turbine stop valves, turbine control valves, and turbine bypass valves	P	T	Non-1E	NA	D	NA	NR	(15-17)
Main steam leads from turbine control valves to turbine casing	P	T	NA	NA	D	NA	NR	(15,17)
Feedwater and condensate system beyond long-term isolation valve	P	RB,T	NA	NA	D	NA	NR	(18)
Condensate Storage and Transfer System								
Condensate storage tank	P	M	NA	NA	D	NA	NR	(19)
Piping	P	M,P,RB,T,W	NA	NA	D	NA	NR	
Valves and other components	P	M,P,RB,T,W	Non-1E	NA	D	NA	NR	

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EQUIPMENT AND STRUCTURE CLASSIFICATION

	Scope of Supply	Location	Electrical Classifi-cation	Seismic Category	Quality Group Classifi-cation	QA Requirement (31,32,33,34)	Tornado Protection	Notes
Standby Gas Treatment System								
Filter units, including electrical heating coils	P	M	1E	I	NA	I	P	
Automatic valves	P	M,RB	1E	I	B,C	I	P	(51)
Piping and manual valves, essential	P	M,RB	NA	I	B,C	I	P	(40)(51)
Piping and manual valves, nonessential	P	M,RB	NA	NA	D	NA	P	
All other components, essential	P	M	1E	I	NA	I	P	
All other components, nonessential	P	M,RB	Non-1E	NA	NA	NA	P	
Primary Containment Purge System								
Automatic isolation valves	P	RB	1E	I	B	I	P	
Piping and manual valves, essential	P	RB	NA	I	C	I	P	
Piping and manual valves, nonessential	P	RB	NA	NA	D	NA	P	(50)
All other components, essential	P	RB	Non-1E	I	C	I	P	
All other components, nonessential	P	RB	NA	NA	D	NA	P	
Diesel Generator Systems								
Piping, fuel oil	P	O,S	NA	I	C	I	P	
Valves, fuel oil	P	O,S	1E	I	C	I	P	
Pumps, fuel oil	P	S	NA	I	C	I	P	
Pump motors, fuel oil system	P	S	1E	I	NA	I	P	
Day tanks	P	S	NA	I	C	I	P	
Diesel fuel storage tanks	P	S	NA	I	C	I	P	
Piping, air startup, essential	P	S	NA	I	C	I	P	
Valves, air startup, essential	P	S	1E	I	C	I	P	
Piping, air startup, nonessential	P	S	NA	NA	D	NA	P	
Air dryers	P,GE	S	NA	NA	D	NA	P	
Compressors, air startup	P,GE	S	NA	I	D	NA	P	(47)
Compressor motor	P,GE	S	1E, Non-1E	I(48)	D	NA	P	(47)
Receivers, air startup	P,GE	S	NA	I	C	I	P	(47)
Lube oil cooler	P	S	NA	I	C	I	P	
Piping and valves, cooling water	P	S	NA	I		I	P	(45)
Piping and valves, lube oil	P	S	NA	I	C	I	P	(45)
Pumps, motors	P	S	1E	I	C	I	P	
Standby diesel generators	P	S	1E	I	NA	I	P	
HPCS diesel generator	GE	S	1E	I	NA	I	P	

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EQUIPMENT AND STRUCTURE CLASSIFICATION

	Scope of Supply	Location	Electrical Classifi-cation	Seismic Category	Quality Group Classifi-cation	QA Requirement (31,32,33,34)	Tornado Protection	Notes
<u>HPCS Diesel Generator Cooling Water System</u>								
Heat exchanger								
Piping and valves, engine mounted	GE	S	NA	I	C	I	P	
Piping and valves, other	GE	S	NA	I	(25)	I	P	(26)
	P	S	NA	I	C	I	P	
<u>HPCS Diesel Generator Lube Oil System</u>								
Heat exchanger	GE	S	NA	I	(26)	I	P	(26)
Piping and valves	GE	S	NA	I	(26)	I	P	(26)
Pumps, motors	GE	S	1E	I	(26)	I	P	(26)
<u>HPCS Diesel Generator Combustion Air Intake and Exhaust System</u>								
Intake silencer	GE	S	NA	I	NA	I	P	
Exhaust silencer	GE	S	NA	I	NA	I	NR	(39)
Piping, essential	P	S	NA	I	C	I	P	
Piping, nonessential	P	S	NA	NA	D	NA	NR	(39)
Filter	GE	S	NA	I	(26)	I	P	(26)
<u>Diesel Generator Systems</u>								
Electrical modules with safety function	P	S,M	1E	I	C	I	P	
Cable	P	C,RB,M	1E	NA	NA	I	P	
Cable trays and fabricated supports with safety function	P	C,RB,M	NA	I	NA	I	P	
<u>Floor and Equipment Drainage Systems</u>								
Sumps	P	RB,C,T,W, P,S,M	NA	NA	D	NA	P,NR	
Pumps	P	RB,C,T,W, P,S,M	Non-1E	NA	D	NA	P,NR	
Piping, containment isolation	P	PC,RB	NA	I	B	I	P	(46)
Piping, other	P	RB,C,T,W, P,S,N,M	NA	NA	D	NA	P,NR	
Valves, containment isolation	P	PC,RB	1E	I	B	I	P	
Valves, other	P	RB,C,T,W, P,S,M	Non-1E	NA	D	NA	P,NR	
Tanks	P	RB	NA	NA	D	NA	P	

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EQUIPMENT AND STRUCTURE CLASSIFICATION

	Scope of Supply	Location	Electrical Classification	Seismic Category	Quality Group Classification	QA Requirement (31,32,33,34)	Tornado Protection	Notes
Hydrogen Recombiner System								
Recombiners	P	RB	1E	I	B	I	P	
Piping, essential	P	PC,RB	NA	I	B	I	P	
Valves, essential	P	PC,RB	1E	I	B	I	P	
Piping, containment isolation	P	PC,RB	NA	I	B	I	P	
Valves, containment isolation	P	PC,RB	1E	I	B	I	P	
Strainers, essential	P	RB	NA	I	B	I	P	
Fire Protection Systems								
Water spray deluge systems	P	PC,RB,C, T,W,M,O	Non-1E	NA	D	NA	P,NR	
Sprinkler systems	P	RB,C,T,W, P,S,M	Non-1E	NA	D	NA	P,NR	
Carbon dioxide systems	P	C,T,N,RB	Non-1E	NA	D	NA	P,NR	
Halon systems	P	C,W	Non-1E	NA	D	NA	P,NR	
Portable and wheeled extinguishers	P	RB,C,T,W, P,N,S	NA	NA	D	NA	P,NR	
Foam systems	P	T	Non-1E	NA	D	NA	NR	
Piping and valves, containment isolation	P	PC,RB	1E	I	B	I	P	(42)
HVAC Systems								
Liquid chillers, essential	P	C	1E	I	C	I	P	
Liquid chillers, nonessential	P	M	Non-1E	NA	D	NA	NR	
Chilled water pumps, piping and accessories, essential	P	C	1E	I	C	I	P	
Chilled water pumps, piping and accessories, nonessential	P	M	Non-1E	NA	D	NA	NR	
Air conditioning units, essential	P	C	1E	I	C	I	P	
Air conditioning units, nonessential	P	C,T,M	Non-1E	NA	D	NA	P,NR	
Unit coolers, essential	P	C,RB,P,S	1E	I	C	I	P	
Unit coolers, nonessential	P	PC,T,P	Non-1E	NA	D	NA	P,NR	
Cooling coils	P	RB,T,W	Non-1E	NA	D	NA	P,NR	
Unit heaters, electric	P	RB,C,T,W, P,S,N,M	Non-1E	NA	D	NA	P,NR	
Hot water heating/glycol piping and specialties	P	RB,T,W	Non-1E	NA	D	NA	P,NR	
Heat exchangers, steam to water	P	T	NA	NA	D	NA	NR	
Heating coils, essential	P	C	1E	I	NA	I	P	

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EQUIPMENT AND STRUCTURE CLASSIFICATION

	Scope of Supply	Location	Electrical Classification	Seismic Category	Quality Group Classification	QA Requirement (31,32,33,34)	Tornado Protection	Notes
Heating coils, nonessential	P	C,RB,N,M	Non-1E	NA	D	NA	P,NR	
Air filters, essential	P	C,M	NA	I	C	I	P	
Air filters, nonessential	P	RB,N,T,W, M,P,S	NA	NA	D	NA	P,NR	
Fans and motors, essential	P	C,S,M	1E	I	C	I	P	
Fans and motors, nonessential	P	C,N,PB,T, P,W,M,S	Non-1E	NA	D	NA	P,NR	
Ductwork and accessories, essential	P	C,RB,PC, P,S	1E	I	C	I	P	(40,41)
Ductwork and accessories, nonessential	P	C,RB,PC, P,S,N,T, W,M,O	Non-1E	NA	D	NA	P,NR	
<u>Auxiliary ac Power System</u>								
13.8 kV emergency switchgear	P	RB	1E	I	NA	I	P	
4.16 kV emergency switchgear	P	C	1E	I	NA	I	P	
600 V emergency load centers	P	C	1E	I	NA	I	P	
600 V emergency motor control centers	P	C,RB	1E	I	NA	I	P	
600 V emergency distribution panels	P	M,C,RB	1E	I	NA	I	P	
120 V, 208/120 V and 120/240 V emergency distribution panels	P	M,C,RB	1E	I	NA	I	P	
Emergency distribution transformers	P	M,C,RB	1E	I	NA	I	P	
Containment electrical penetrations	P	RB	1E	I	NA	I	P	
120 V ac emergency uninterruptible power supply systems	P	C	1E	I	NA	I	P	
Emergency cables	P	M,C,RB	1E	NA	NA	I	P	
Cable trays and fabricated supports with safety functions	P	M,C,RB	NA	I	NA	I	P	
Conduit (except when part of an environmental seal) and nonemergency cable tray	P	M,C,RB	NA	I	NA	NA	P	(34)
<u>125-V dc Power Systems</u>								
125-V dc emergency batteries and racks	P	C	1E	I	NA	I	P	
125-V dc emergency battery chargers	P	C	1E	I	NA	I	P	
125-V dc emergency switchgear	P	C	1E	I	NA	I	P	
125-V dc emergency centers	P	C	1E	I	NA	I	P	
125-V dc emergency distribution panels	P	M,C,RB	1E	I	NA	I	P	
Emergency cables	P	M,C,RB	1E	NA	NA	I	P	
Cable tray and fabricated supports with safety functions	P	M,C,RB	NA	I	NA	I	P	

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EQUIPMENT AND STRUCTURE CLASSIFICATION

	Scope of Supply	Location	Electrical Classification	Seismic Category	Quality Group Classification	QA Requirement (31,32,33,34)	Tornado Protection	Notes
Discharge tunnel and diffuser	NA	O	NA	NA	NA	NA	NR	
Main stack	NA	O	NA	I	NA	I	NR	
Offgas room	NA	M	NA	I	NA	NA	NR	
Electrical tunnels, with safety-related cable	NA	M	NA	I	NA	I	T	
Normal switchgear building	NA	N	NA	NA	NA	NA	NR	
Auxiliary boiler building	NA	M	NA	NA	NA	NA	NR	
Standby gas treatment building	NA	M	NA	I	NA	I	T	(21)
Transformer foundations and fire walls	NA	M	NA	NA	NA	NA	NR	
DC containment penetrations	P	RB	1E	I	NA	I	P	
Conduit and nonemergency cable tray	P	M,C,RB	NA	I	NA	NA	P	(34)
<u>Miscellaneous Components</u>								
Reactor building polar crane	P	RB	Non-1E	I	NA	I	(22)	
<u>Civil Structures</u>								(29)
Primary containment	NA	RB	NA	I	NA	I	P	(28,44)
Reactor building, including fuel storage facilities and auxiliary bays	NA	RB	NA	I	NA	I	T	(28,29, 36,49)
Radwaste building	NA	W	NA	I	NA	NA	NR	(20)
Control building	NA	C	NA	I	NA	I	T	
Diesel generator building	NA	S	NA	I	NA	I	T	
Turbine building, including heater bay, except as noted	NA	T	NA	NA	NA	NA	NR	
Main steam tunnel portion of turbine building	NA	T	NA	I	NA	I	T	
Pipe tunnel portion of the turbine building between column lines AK and AM below el 261 ft	NA	T	NA	I	NA	I	T	
Pipe tunnel portion of the turbine building between column lines 10 and 12 below el 248 ft	NA	T	NA	I	NA	I	T	
Turbine building el 250 ft slab over the pipe tunnels between column lines 10 and 12	NA	T	NA	I	NA	NA	T	

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EQUIPMENT AND STRUCTURE CLASSIFICATION

	Scope of Supply	Location	Electrical Classifi-cation	Seismic Category	Quality Group Classifi-cation	QA Requirement (31,32,33,34)	Tornado Protection	Notes
Service building, including foam room	NA	S	NA	NA	NA	NA	NR	
Screenwell service water pumphouse	NA	P	NA	I	NA	I	T	
Screenwell building, superstructure	NA	M	NA	NA	NA	NA	NR	
Intake structures and tunnels	NA	O	NA	I	NA	I	T	
Railroad access lock	NA	M	NA	I	NA	I	T	
Railroad passage to turbine building	NA	M	NA	NA	NA	NA	NR	
Electrical bay	NA	M	NA	NA	NA	NA	NR	
Condensate storage tank building	NA	M	NA	NA	NA	NA	NR	
Access passageway, Unit 2 turbine building to administration building	NA	M	NA	NA	NA	NA	NR	
Cooling tower and flume	NA	O	NA	NA	NA	NA	NR	
Regeneration and condensate demineralizer rooms	NA	M	NA	NA	NA	NA	NR	
Auxiliary service building, substructure	NA	M	NA	I	NA	I	T	
Auxiliary service building, superstructure	NA	M	NA	NA	NA	NA	NR	
Demineralized water storage and waste neutralizing tank building	NA	M	NA	NA	NA	NA	NR	
Shorefront revetment ditch	NA	O	NA	NA(23)	NA	I	NR	
PMP exterior flood protection berms	NA	O	NA	NA	NA	I	NR	
Roof and storm drainage systems	P	RB,S,T,W, C,N,P,M,O	NA	NA	NA	NA	NR	(34)
Spent fuel pool and liner	NA	RB	NA	I	NA	I	T	(29)
Auxiliary service building, superstructure	NA	M	NA	NA	NA	NA	NR	
Demineralized water storage and waste neutralizing tank building	NA	M	NA	NA	NA	NA	NR	
Shorefront revetment ditch	NA	O	NA	NA(23)	NA	I	NR	
PMP exterior flood protection berms	NA	O	NA	NA	NA	I	NR	
Roof and storm drainage systems	P	RB,S,T,W, C,N,P,M,O	NA	NA	NA	NA	NR	(34)
Spent fuel pool and liner	NA	RB	NA	I	NA	I	T	(29)
<u>Miscellaneous Radiation Protection Equipment and Programs</u>								
Portable radioactivity monitoring equipment	P	M	Non-1E	NA	NA	NA	NR	(34a)
Radioactivity sampling equipment	P	M	Non-1E	NA	NA	NA	NR	(34a)
Radioactivity contamination measurement and analysis equipment	P	M	Non-1E	NA	NA	NA	NR	(34a)
Personnel monitoring equipment	P	M	Non-1E	NA	NA	NA	NR	(34a)

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EQUIPMENT AND STRUCTURE CLASSIFICATION

	Scope of Supply	Location	Electrical Classifi-cation	Seismic Category	Quality Group Classifi-cation	QA Requirement (31,32,33,34)	Tornado Protection	Notes
Instrument storage, calibration, and maintenance program	P	M	NA	NA	NA	NA	NR	(34a)
Decontamination facilities	P	TB,W,M	NA	NA	NA	NA	NR	(34a)
Respiratory protection equipment	P	M	NA	NA	NA	NA	NR	(34a)
Contamination control equipment	P	M	Non-1E	NA	NA	NA	NR	(34a)
In-plant I_2 monitoring equipment (NUREG-0737, Item III.D.3.3)	P	M	Non-1E	NA	NA	NA	NR	(34a)
<u>Crack Arrest Verification System</u>								
Piping, Other	GE,P	M	NA	NA	D	NA	P	
Valves, Other	GE,P	M	NA	NA	D	NA	P	
Pressure Vessel	GE	M	NA	NA	D	NA	P	
Controls/Instruments	GE,P	M	Non-1E	NA	NA	NA	P	
<u>Oxygen Feedwater Injection System</u>								
Piping, Other	GE,P	M	NA	NA	D	NA	P,NR	
Valves, Other	GE,P	M	NA	NA	D	NA	P,NR	
Controls/Instruments	GE,P	M	Non-1E	NA	NA	NA	P	
<u>Primary Loop</u>								
Piping	P	RB	NA	NA	D	NA	P	
Valves	P	RB	NA	NA	D	NA	P	
Pumps	P	RB	NA	NA	D	NA	P	
Heat Exchangers	P	RB	NA	NA	D	NA	P	
Pump Motors	P	RB	Non-1E	NA	NA	NA	P	
<u>Secondary Loop</u>								
Secondary Containment Boundary Piping	P	RB	NA	I	C	I	T	
Secondary Containment Boundary Valves	P	RB	NA	I	C	I	T	
Piping (all other)	P	RB,O	NA	NA	D	NA	P,NR	
Valves (all other)	P	RB,O	NA	NA	D	NA	P,NR	
Cooling Towers	P	O	NA	NA	D	NA	NR	
Motor Control Center	P	O	Non-1E	NA	NA	NA	NR	
Pump/Cooling Tower Fan Motors	P	O	Non-1E	NA	NA	NA	NR	

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EQUIPMENT AND STRUCTURE CLASSIFICATION

	Scope of Supply	Location	Electrical Classifi-cation	Seismic Category	Quality Group Classifi-cation	QA Requirement (31,32,33,34,52)	Tornado Protection	Notes
<u>Local Control Panels & Racks</u>								
Cables	P	RB,O	Non-1E	NA	NA	NA	NR	
Controls/Instruments Inside Reactor Building	P	RB	Non-1E	NA	NA	NA	P	
Controls/Instruments in Yard	P	O	Non-1E	NA	NA	NA	NR	
<u>Hydrogen Water Chemistry System</u>								
Piping, Other	GE,P	T	NA	NA	D	NA	NR	
Valves, Other	GE,P	T	NA	NA	D	NA	NR	
Controls/Instruments	GE,P	T	Non-1E	NA	NA	NA	NR	
Electrical Equipment	GE,P	T	Non-1E	NA	NA	NA	NR	
H_2 Transportable Trailers	Gas Vendor	O	NA	NA	NA	NA	NR	(52)
H_2 Transportable Trailer Foundations	P	O	NA	NA	NA	NA	NR	(52)
O_2 Permanent Tank	Gas Vendor	O	NA	NA	NA	NA	NR	(52)
O_2 Tank Foundations	P	O	NA	NA	NA	NA	NR	(52)

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EQUIPMENT AND STRUCTURE CLASSIFICATION

Keys to Abbreviations

KEY TO SCOPE OF SUPPLY:

GE	=	General Electric
P	=	Nine Mile Point Nuclear Station, LLC

KEY TO LOCATION:

PC	=	Primary containment
RB	=	Reactor building
M	=	Any other location
O	=	Outdoors onsite
S	=	Diesel generator building
T	=	Turbine building
W	=	Radwaste building
C	=	Control building
N	=	Normal switchgear building
P	=	Screenwell building

KEY TO ELECTRICAL CLASSIFICATION:

1E	=	Electrical equipment that meets the quality assurance standards of NRC guidelines and IEEE-323-1974.
Non-1E	=	Electrical equipment that is not required to meet 1E requirements.
NA	=	Not applicable because the equipment is not electrical.

KEY TO SEISMIC CATEGORY:

I	=	The equipment and structures are constructed in accordance with the requirements for Category I structures and components (Section 3.7).
NA	=	The seismic requirements for the SSE are not applicable to the equipment. No specific design is made to resist seismic forces. However, each system and component and its supporting elements is reviewed for proper anchorage and load carrying capability under seismic forces and evaluated on

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EQUIPMENT AND STRUCTURE CLASSIFICATION

the basis of sound engineering judgment, to ensure that failure of this class of equipment does not affect the operation of any Category I equipment or cause detrimental damage to Category I structures.

KEY TO QUALITY GROUP CLASSIFICATION:

A,B, C,D	=	NRC quality group classification as defined in RG 1.26. The equipment is constructed in accordance with the codes listed in Table 3.2-2.
N/A	=	Quality group classification is not applicable to this equipment.

KEY TO QA REQUIREMENT:

I	=	Equipment meets the QA requirements of 10CFR50, in accordance with the QA program described in Chapter 17.
NA	=	QA requirements of 10CFR50 Appendix B are not applicable to this equipment.

KEY TO TORNADO PROTECTION:

T	=	Designed for tornado protection.
P	=	Tornado protection provided by virtue of location within a tornado-protected structure.
NR	=	Tornado protection is not provided.

NOTES

(1) Application of Category I design criteria is limited to those reactor vessel internals that are part of engineered safety features, such as the core spray piping, core spray sparger, and hardware.

(2) These reactor vessel internal structures include the steam separators, steam dryers, and miscellaneous hardware items.

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EQUIPMENT AND STRUCTURE CLASSIFICATION

(3) a. Lines equivalent to 1-in or smaller liquid line that are part of the RCPB and not connected to Quality Group A condensing chambers are Quality Group B and Category I.

b. All instrument lines connected to the RCPB and utilized to actuate and monitor safety systems are Quality Group B from the outer isolation valve or process shutoff valve (root valve) to the sensing instrumentation.

c. All instrument lines connected to the RCPB and not utilized to actuate and monitor safety systems are Quality Group D from the outer isolation valve or the process shutoff valve (root valve) to the sensing instrumentation.

d. All other instrument lines:

1) through the root valve are of the same classification as the system to which they are attached.

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

3) beyond the root valve, if not used to actuate a safety system, may be Quality Group D.

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

(4) Recirculation system pipe restraints are not required to function (i.e., restrain a pipe) during an earthquake. These restraints are designed to withstand a SSE without loss of functional capability.

(5) The CRD insert and withdraw lines from the drive flange up to and including the first valve on the HCU are Quality Group B.

(6) The HCU is a GE factory-assembled engineered module of valves, tubing, piping, and stored water which controls a single CRD 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.

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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 Group A, B, C, and D pressure integrity quality levels clearly apply at all levels to the interfaces between the HCU and the connecting conventional piping components (e.g., pipe nipples, fittings, simple hand valves), it is considered that they do not apply to the specialty parts (e.g., solenoid valves, pneumatic components, and instruments).

The design and construction specifications for the HCU do invoke such codes and standards as can be reasonably applied to individual parts in developing required quality levels, but these codes and standards are supplemented with additional requirements for these parts and for the remaining parts and details. For example, (1) all welds are LP inspected, (2) all socket welds are inspected for gap between pipe and socket bottom, (3) all welding is performed by qualified welders, and (4) all work is done in accordance with 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 not explicitly defined within the codes for Quality Groups A, B, or C. This is supplemented by the QC techniques previously described.

(7) The RCIC turbine does not fall within the applicable design codes. To assure that the turbine is fabricated to standards commensurate with safety and performance requirements, GE has established specific design requirements for this component as follows (all references below to the ASME Boiler and Pressure Vessel Code Section III are to the 1968 edition):

a. All pressure-containing castings and fabrications are hydrotested at 1.5 x design pressure.
b. All high-pressure castings are radiographed according to:

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EQUIPMENT AND STRUCTURE CLASSIFICATION

ASTM E-94
E-14 for maximum feasible volume
E-71, 186, or 280 for Severity Level 3

c. As-cast surfaces are magnetic particle or liquid penetrant tested according to ASME Section III, Paragraph N-323.4 or N-323.3.
d. Wheel and shaft forgings are ultrasonically tested according to ASTM A-388.
e. Butt welds are radiographed and magnetic particle or liquid penetrant tested according to ASME Section III, Paragraph N-626 or N-627, respectively.
f. GE is to be notified of major repairs and records maintained thereof.
g. Record system and traceability is according to ASME Boiler and Pressure Vessel Code, Section III, Appendix IX, Paragraph IX-225.
h. Control and identification is according to ASME Section III, Appendix IX, Paragraph IX-226.
i. Procedures conform to ASME Section III, Appendix IX, Paragraph IX-300.
j. Inspection personnel are qualified according to ASME Section III, Appendix IX, Paragraph IX-400.

(8) These items are classified as Seismic NA (except from seismic evaluation) because they suspend from a cable that dampens out the transmission of floor response spectra.

(9) DELETED.

(10) Liquid radwaste system atmospheric storage tanks made of fiberglass are designed, constructed, and tested in accordance with the requirements of ASTM D-3299-74 or NBS PS 15-69 (Section 11.2).

(11) Although RG 1.26 is not applicable, the equivalent quality group classification for radwaste management systems is Quality Group D. The radwaste management systems are designed, constructed, and tested in accordance with the QA provisions of RG 1.143.

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EQUIPMENT AND STRUCTURE CLASSIFICATION

(12) Waste solidification system components are designed, fabricated, inspected, and tested in accordance with Topical Report No. WPC-VRS-001 (Section 11.4). This report was prepared by the Werner and Pfleiderer Corporation, supplier of the waste solidification system for Unit 2, and has been accepted by the NRC for reference in license applications.

(13) The main steam lines between the outermost containment isolation valve up to the turbine stop valve, the main turbine bypass lines up to the turbine bypass valve, and all branch lines connected to these portions of the main steam and turbine bypass lines up to the first valve that leads to the moisture separator and reheater, turbine gland seal system, and auxiliary steam header are Quality Group D. These sections of pipe meet all of the pressure integrity requirements of Quality Group D plus the following additional requirements:

a. All longitudinal and circumferential butt weld joints are radiographed (or ultrasonically tested to equivalent standards). Where size or configuration does not permit effective volumetric examination, magnetic particle or liquid penetrant examination may be substituted. Examination procedures and acceptance standards are at least equivalent to those specified in ANSI B31.1.0.

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

c. The main steam line from its outer isolation valve up to and including the turbine stop valve, and all branch lines 2 1/2 in in diameter and larger up to and including the first valve (including restraints), is designed by the use of an appropriate dynamic

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EQUIPMENT AND STRUCTURE CLASSIFICATION

seismic -system analysis to withstand OBE and DBE design loads in combination with other appropriate loads within the limits specified for Quality Group B pipe in ASME Section III. The mathematical model for the dynamic seismic analyses of the main steam line and branch line piping includes the turbine stop valves and piping beyond the stop valves, including the piping to the turbine casing. The dynamic input loads for design of the main steam line are derived from a time history model analysis (or an equivalent method) of the reactor and applicable portions of the turbine building. The turbine building, housing the main steam lines, may undergo some plastic deformation under the DBE; however, the plastic deformation will be limited to a ductility factor (defined as the ratio between the maximum displacement and the yield displacement) of 2 and an elastic multidegree-of-freedom system analysis will be used to determine the input to the main steam line. The stress allowable and associated deformation limits for piping will be in accordance with Quality Group B requirements for the OBE and DBE loading combinations. The main steam line supporting structures (those portions of the turbine building) are such that the main steam line and its supports can maintain their integrity within the Quality Group B requirements under the Category I seismic loading condition.

d. The high integrity classification of the main steam line from its outer isolation valve up to and including the turbine stop valve, and all branch lines 2 1/2 in and larger up to and including the first valve (as tabulated in this table, Power Conversion System, Items 1 through 6, and Notes 13 and 14), is an acceptable equivalent of the integrity requirements of Quality Group B. The turbine stop valves and the piping beyond the stop valves to the turbine casing and the first valve that leads to the moisture separator and reheater, turbine gland seal system, and auxiliary steam header and the piping

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EQUIPMENT AND STRUCTURE CLASSIFICATION

beyond the first valve are not classified to Category I requirements. However, the turbine stop valves and the first valves, and appropriate portions of the piping beyond these valves (including their restraints) are included in the mathematical model for the dynamic seismic analyses indicated in Note 13c.

e. All inspection records will be maintained for the life of the Station and will include data pertaining to qualification of inspection personnel, examination procedures, and examination results.

(14) The first valve that leads to the moisture separator reheater, turbine gland seal system, and auxiliary steam header in branch lines connected to the main steam lines between the outermost containment isolation valve and turbine stop valve, and in branch lines connected to turbine bypass lineup to the turbine bypass valve, will meet all the pressure integrity requirements of Quality Group D plus the following additional requirements:

a. Pressure-retaining components of all cast parts of valves of a size and configuration for which volumetric examination methods are effective will be radiographed. Ultrasonic examination to equivalent standards may be used as an alternate to radiographic methods. If size or configuration does not permit effective volumetric examination, magnetic particle or liquid penetrant methods may be substituted. Examination procedures and acceptance standards will be at least equivalent to those specified in ANSI B31.1.0.

b. All inspection records will be retained for the life of the Station and will include data pertaining to the qualification of inspection personnel, examination procedures, and examination results.

(15) A number of turbine generator components, including the stop and control valves, turbine bypass valve chest, and high-pressure turbine casing are made of a special GE proprietary alloy (copper-bearing carbon steel) that has

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EQUIPMENT AND STRUCTURE CLASSIFICATION

no assigned ASME or ASTM material number. All welding to this material will be performed to the technical and quality requirements of the GE installation requirements. These requirements match or exceed those given in Note 13 of this table.

(16) A certification will be obtained from the vendors of the turbine stop valves and turbine bypass valves that all cast pressure-retaining parts of a size and configuration for which volumetric examination methods are effective have been examined by radiographic methods by qualified personnel. Ultrasonic examination to equivalent standards may be used as an alternate to radiographic methods. Examination procedures and acceptance standards will be at least equivalent to those specified as supplementary types of examination in ANSI B31.1.0, Paragraph 136.4.3.

(17) The turbine stop and control valves, turbine bypass valves (including the bypass valve chest), and main steam leads between the stop and control valves and the high-pressure turbine casing are fabricated under the requirements of GE's GEZ 4982A, General Electric Large Steam Turbine-Generator Quality Control Program.

The turbine stop and control valves and the main steam leads to the turbine chest will be installed to GE technical and quality requirements equivalent to the fabrication requirements. The erection activity is of a quality level generally equivalent to QA Category I.

(18) In addition to a swing check valve inside containment and a positive-acting check valve outside containment, a third valve with high leak-tight integrity is provided in each line outside containment. The classification of the feedwater lines from the reactor vessel to and including the third isolation valve (2FWS*MOV21A, B) is Quality Group A; beyond the third valve is Quality Group D.

(19) The condensate storage tank is designed, fabricated, and tested in accordance with the requirements of ASTM D-3299-74 or NBS PS 15-69.

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EQUIPMENT AND STRUCTURE CLASSIFICATION

20) The radwaste building is designed and constructed in full compliance with RG 1.143. The radwaste building has been designed to withstand loads associated with the SSE.

(21) The standby gas treatment building is designed and constructed in accordance with seismic and QA Category I requirements up to el 286 ft only.

(22) The RBPC is designed to withstand the spectrum of tornado-generated missiles (Section 3.5.1.4). The metal siding above the refueling floor is designed to withstand the wind loading generated by a tornadic event. This precludes RBPC from exposure to tornadic wind loading.

(23) The revetment ditch system has been analyzed with respect to the factor of safety versus slope failure during a combination of storm and earthquake events, as discussed in Section 2.5.5.2.

(24) Examples of the Quality Group B essential valves in the recirculation system are the following:

a. Valves F001, F002, F009, F013, F014, and F017 for pump seal purge line (inside containment) to recirculation pump.
b. Valves F019, F020, F021, F022, and F059 for sample line from recirculation loops.
c. Vent valves F025, F026, F068, and F069 for remote operated valves.

An example of the Quality Group C essential piping and valves in the recirculation system is the following:

a. Pump seal leak detection piping up to and including valve F086.

(25) Examples of Quality Group D nonessential piping and valves in the recirculation system are the following:

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EQUIPMENT AND STRUCTURE CLASSIFICATION

a. Pump seal purge piping (outside containment) to recirculation pump including valves F008, F016 and F015.
b. Recirculation pump seal staging piping including valves F080, F084, and F088.
c. Pump seal leak detection piping beyond valve F086.

(26) This equipment conforms to ANSI Standard B31.1 and IEEE-344-71 seismic requirements.

To meet the intent of ASME Section III requirements for the Division III (HPCS) diesel generator engine and starting air skids:

a. The pressure test for the equipment has been performed using ASME Section III, Class 3 hydrostatic parameters.
b. The jacket cooling water expansion tank has been hydrostatically tested at 1.5 times its design pressure.
c. Piping over 4 in (6-in lines between the cooling water heat exchanger, expansion tank, and engine block) has been liquid penetrant examined prior to preoperational testing.
d. The use of correct piping and component materials has been verified (material certification) during the manufacturing process (eliminates the need for actual mill test reports for piping).

(27) Effluent monitors meet the environmental qualification and quality assurance requirements of RG 1.97 Revision 3.

(28) Structures installed for biological or post-accident shielding which provide support for safety-related components are of the same QA classification as the structures in which they are located (e.g., biological shield wall). Other components serving only a shielding function are classified QA NA; however, the design considerations of said components ensure no adverse effects on safety-related components.

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EQUIPMENT AND STRUCTURE CLASSIFICATION

(29) The classification of a structure described herein also applies to all major structural components of that structure, unless noted otherwise herein.

(30) Reactor pressure vessel stabilizers are constructed in accordance with ASME III, 1977 Edition through Summer 1978 Addenda, except that for installation, the requirements of Paragraph NF-4600 of ASME III, 1974 Code are applicable.

(31) All safety-related instrumentation and controls (I&C) described in FSAR Sections 7.1 through 7.6 and other safety-related I&C for safety-related systems meet the QA requirements of 10CFR50 Appendix B. These safety-related I&C are listed in FSAR Table 3.2-1, as, for example, "electrical modules with safety function," or "instrument modules with safety function."

In Table 3.2-1, the designation "I" indicates that these safety-related I&C meet the QA requirements of 10CFR50 Appendix B, as described in FSAR Chapter 17.

(32) Those structures, components, and equipment described by RG 1.29, Sections C2 and C3, are described in Section 3.5.1.1.4, 3.7, and 3.8. The pertinent provisions of the operational QA program apply.

(33) All containment isolation valves not listed specifically in the table are seismic and QA Category I. See Table 6.2-56 for additional information concerning these valves.

(34) Pertinent provisions of the operational QA program apply to:

a. Modifications to roof and site drainage systems and grading used for handling the probable maximum precipitation.
b. Reactor vessel steam dryer and steam separator and miscellaneous hardware.

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EQUIPMENT AND STRUCTURE CLASSIFICATION

c. Post-accident sampling system (until the potential secondary containment bypass leakage paths and the leakage paths for highly-radioactive fluids are eliminated).

d. Conduit (except when part of an environmental seal) and nonemergency cable tray whose failure could affect safety-related equipment.

e. Reactor pressure vessel insulation.

(34a) Controlled procedures are provided to ensure proper storage, calibration, and maintenance to:

1. Portable radioactive monitoring equipment used for emergency purposes.
2. Air and liquid sampling equipment for emergency purposes.
3. Portable equipment used to perform radioactivity contamination measurement and analysis.
4. Personnel monitoring and decontamination equipment including TLDs and whole body counter.
5. Instrumentation storage, calibration, and maintenance for instruments used during emergencies.
6. Respiratory protection equipment, including testing.
7. Emergency plans and related equipment/components/structures described in the emergency procedures.
8. In-plant post-accident I_2 monitoring equipment (NUREG-0737, Item III.D.3.3).

(35) The reactor water cleanup system classification meets Standard Review Plan 5.4.8, paragraph II.3.

(36) The stair tower attached to the exterior side of the secondary containment wall in the southeast corner of the reactor building is a non-Category I structure.

(37) The refueling platform is not necessary to:

a. Assure the integrity of the reactor coolant boundary.
b. Assure the capability to shut down the reactor and maintain it in a safe shutdown condition.
c. Assure the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposure.

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EQUIPMENT AND STRUCTURE CLASSIFICATION

Therefore, the installation of the refueling platform has been done in accordance with QA Category II requirements as defined in the SWEC QA Program for Unit 2.

(38) The condensing chambers connected to the steam line flow restrictions are Quality Group B.

(39) The exhaust silencer is not protected by a missile hood; however, an exhaust relief valve is provided to maintain the diesel generator function (see Sections 9.5.8.1 and 9.5.8.2).

(40) The SGTS and a portion of the control room/relay room air intake utilize pipe in lieu of ductwork. Since these pipes are intended to fulfill the function of ductwork, pipes and their components, including supports, are designed, fabricated, and installed in accordance with ASME III, Class 2 or 3 (SGTS is Class 2) requirements, with the following exceptions:

a. Visual inspection of the welds is performed.
b. ASME III Code data reports, N-stamping, and ANSI acceptance are not required.
c. Being part of an engineered safety filtration system operating at low pressure (in W.G.), any leak testing will be in accordance with ANSI N509, as discussed in Section 1.8.

(41) The Category I portion of piping acts as a portion of the ductwork system in the control building ventilation system. Since this is not an ASME piping system, it will not be N-5 code stamped. The ductwork was designed to ASME requirements to ensure a qualified seismically supported duct in excess of the requirements for Category I ductwork and to allow installation of valves, in lieu of dampers, to ensure added leak-tightness.

(42) The Category I portion of this system is designed to ASME Section III and is an extension of the primary containment, which is not code stamped. This system is nonsafety related and provides no safe shutdown or primary coolant pressure boundary function. These sections of pipe are ASME because they are an extension of the primary containment. The Category I pipe spools meet ASME Section III requirements, but will not be N-5 code stamped, since the primary containment is not N stamped.

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EQUIPMENT AND STRUCTURE CLASSIFICATION

(43) The ECCS strainers are fabricated to ASME Section III requirements and meet the ASME Section III requirements, except they are not N-5 code stamped. The strainers are not a pressure boundary, and they are attached to the end of the piping within the suppression pool.

(44) The drywell floor downcomers are fabricated to meet ASME Section III requirements but are not N-5 code stamped. This piping is actually a structural member of the primary containment, which is not code stamped and, therefore, the downcomers need not be N-5 code stamped.

(45) All essential vendor-supplied skid-mounted piping subassemblies and components are designed, fabricated and tested in accordance with ASME Section III, Class 3, requirements with the following exception: ASME Section III system (N-5) certification is not required. The HPCS diesel generator subsystems are designed ANSI B31 and, therefore, could not be N-5 code stamped. The Division I and Division II diesel generators meet all ASME Section III requirements, except for code stamping. (Refer to Table 3.2-1 Note 26 and Sections 9.5-4 through 9.5-8 for further information.)

(46) The following portions of systems are designed, fabricated, and installed in accordance with ASME Section III, Class 2 or Class 3, requirements but are not certified on the N-5 Code Data Report. These systems are nonsafety related and perform no safety function to shut down the reactor and are not part of the primary coolant pressure boundary. These systems are either an extension of the primary containment as noted (see Note 42 above), or these systems are part of the service water intake/discharge structure which is a concrete intake bay.

<u>Extension of the Intake/Discharge Structure</u> is three service water sections of piping. These include:

2SWP-024-01M-3

2SWP-024-01Y-3

2SWP-016-61J-3

These sections have only a small hydrostatic head (less than 10 psi) from the water in the intake/discharge bays, contain isolation valves which are normally closed and perform no safety function except to provide a low pressure boundary for the two bays.

Extension of the Primary Containment - Nitrogen Purge Line for TIP System has two sections of tubing which are nonsafety related:

2GSN-500-151-2

2GSN-500-152-2

These sections are not safety related and perform no safety function, nor are they part of the primary coolant pressure boundary. These pieces of tubing are extensions of the primary containment pressure boundary and, therefore, as described in Note 42 above, need not be N-5 code stamped.

Extension of the Primary Containment - Reactor Building Closed Loop Cooling System

The following reactor building closed loop cooling piping system penetrations are nonsafety related and do not provide any RCPB but are extensions of the primary containment and need not be N-5 stamped in accordance with Note 42 above. These penetrations and pieces of pipe include:

Penetration Z-34B

2CCP-004-677-2
2CCP-004-105-2
2CCP-750-688-2

Penetration Z-33B

2CCP-004-555-2
2CCP-750-689-2

Penetration Z-33A

Penetration Z-34A

Penetration Z-46A

Penetration Z-47

Extension of the Primary Containment - Reactor Recirculation System

Similarly, the reactor recirculation flow control hydraulic system, which performs no safety function and is

TABLE 3.2-1
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EQUIPMENT AND STRUCTURE CLASSIFICATION

not part of the primary coolant pressure boundary but is part of the primary containment pressure boundary as noted in Note 42 above, are also not code stamped. The list of the penetrations follows:

Penetration Z-99A

Penetration Z-99B

2RCS-001-201-2

Penetration Z-99C

Penetration Z-99D

2RCS-001-202-2

Penetration Z-100A

Penetration Z-100B

2RCS-001-203-2

Penetration Z-100C

Penetration Z-100D

2RCS-001-204-2

Penetration Z-319-2

2RCS-750-168-2

Penetration Z-328-3

2RCS-750-151-2

<u>Extension of Primary Containment - Reactor Building Equipment Drain System</u>

(DER) Penetration Z-45

2DER-002-034-2

TABLE 3.2-1
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EQUIPMENT AND STRUCTURE CLASSIFICATION

(47) The equipment and its classification also apply to the air start system associated with the Division III (HPCS) diesel generator.

(48) The GE compressor motor is seismic Category I. The compressor motor starter is seismic Category NA, but it is seismically mounted and evaluated to ensure that if failure occurs it will not cause degradation of safety-related equipment.

(49) The storage pool gate is nonsafety related.

(50) Nonessential portions of these systems within the containment, from the containment penetration up to but not including the safety-related end users, are designed, fabricated and erected to Quality Group C requirements. The safety-related end user components function without or upon loss of nitrogen, or are provided with safety-related accumulators capable of supplying the required quantities of gas.

(51) The essential components of the GTS air supply system are designed and installed in accordance with the requirements of ASME III, Class 3, except ASME III Code data reports and N-stamping are not required since GTS is not a stamped system (See Note 40).

(52) Anchorage for up to two transportable gaseous hydrogen trailers and their foundations are designed to remain in place for design basis earthquake, tornado wind/missile and flood conditions. Anchorage of permanent liquid oxygen tank and its foundation are designed to remain in place for both design basis tornado wind and flood conditions. These meet requirements of EPRI Report NP-5283-SR-A, 1987 revision.

Although the HWC system is not nuclear safety related, the design, procurement, fabrication and construction activities shall conform to the quality assurance provision of codes and standards specified in EPRI Report NP-5283-SR-A, 1987 revision.

(1) Lease II User's Manual, "Slope Stability Analysis," by P. J. Trudeau and J. T. Christian, August 1980, Stone & Webster Engineering Corporation.

TABLE 3.2-2
(Sheet 1 of 1)

CODE GROUP DESIGNATIONS, INDUSTRY CODES, AND STANDARDS FOR MECHANICAL COMPONENTS[1]
ASME Section III Code Applicable Sections

Quality Group Classification	ASME Section III Code Class 8	Pressure Vessels and Heat Exchangers	Pumps, Valves and Piping	Metal Containment Components	Storage Tanks 0-15 (psig)	Storage Tanks Atmospheric
A[2]	1	NA or NCA & NB TEMA C	NA or NCA & NB[2,3]	--	--	--
B	2	NA or NCA & NC TEMA C	NA or NCA & NC[2,3]		NA or NCA & NC[4]	NA or NCA & NC[5]
	MC			NA or NCA & NE		
C	3	NA or NCA & ND TEMA R,C	NA or NCA & ND[2,3]	--	NA or NCA & ND[4]	NA or NCA & ND[5]
D		ASME VIII Div. 1 TEMA C,R	Piping & valves B31.1.0 pumps[6]	--	ASME VIII or equivalent[7]	ASME VIII, NBS-PS15-69 API-650 or equivalent[7]

(1) Components required to be stamped to ASME Boiler and Pressure Vessel Code are stamped with the applicable ASME Code symbol and third-party inspected by a qualified inspector.

(2) Components of the RCPB comply with the requirements of 10CFR50.55a codes and standards. All other components satisfy codes and addenda in effect at the time of the component order.

(3) For pumps classified in A, B, or C, the applicable subsection NB, NC, or ND, respectively, of ASME Section III is used as a guide in calculating the thickness of pressure-retaining portions of the pump and in sizing cover bolting.

(4) 100% volumetric examination of the sidewall and roof weld joints for plates over 3/16-in thick and 100% surface examination of weld joints for plates 3/16-in thick or less of the sidewall-to-bottom and sidewall roof joints. These examination requirements are performed in accordance with the rules of ASME Section III, Code Class 2 and 3.

(5) 100% volumetric examination of the sidewall weld joints for plates over 3/16-in thick and 100% surface examination of the sidewall-to-bottom joints. These examination requirements are performed in accordance with the rules of ASME Section III, Safety Class 2 and 3.

(6) For GE-supplied pumps classified D, the ASME Section VIII, Division I pump design for the intended service is utilized. For other pumps classified D, the manufacturers' standard pump design for the intended service is utilized.

(7) Storage tanks are designed to meet the intent of API, NBS-PS, and/or ASME Section VIII standards as applicable.

(8) In the case that material cannot be purchased to meet the specified ASME III Code, then material that meets subsequent ASME III Code Editions/Addenda up to and including 1980 Edition/Summer 1982 Addenda may be substituted after a review and reconciliation of related requirements of the ASME III Code are performed and documented.

TABLE 3.2-3
(Sheet 1 of 1)

SUMMARY OF SAFETY CLASS DESIGN REQUIREMENTS

Design Requirements	Safety Class 1	Safety Class 2	Safety Class 3	Safety Class Other
Quality group classification[1,2]	A	B	C	D
Quality assurance requirement[3]	I	I	I	NA
Seismic category[4]	I	I	I	NA

[1] Equipment is constructed in accordance with the indicated code group listed in Table 3.2-1 and defined in Table 3.2-2.

[2] As indicated in Table 3.2-1, for QA Category I components, the quality group classification and/or seismic category is not applicable (NA) in certain cases, e.g., reactor internal structures and the shorefront revetment ditch.

[3] I = Equipment meets the QA requirements of 10CFR50 Appendix B.
NA = Conformance with 10CFR50 Appendix B is not required.

[4] I = Equipment is constructed in accordance with the requirements for Category I components (Section 3.7).
NA = The requirement to withstand the SSE is not applicable to this equipment.

TABLE 3.2-4
(Sheet 1 of 2)

REACTOR COOLANT PRESSURE BOUNDARY
CLASS I EQUIPMENT CODE APPLICATION

Equipment	MPL/Mark	Code[1] Edition	Addenda
Reactor pressure vessel	B13-D003	1971	Winter 1972
Main steam system			
Piping		1974	No Addenda
Containment isolation valves	B22-F022A	1977	Summer 1977
	B22-F022B	1977	Summer 1977
	B22-F022C	1977	Summer 1977
	B22-F022D	1977	Summer 1977
	B22-F016	1974	Winter 1975
	B22-F019	1974	Winter 1975
	B22-F028A	1977	Summer 1977
	B22-F028B	1977	Summer 1977
	B22-F028C	1977	Summer 1977
	B22-F028D	1977	Summer 1977
	2MSS*MOV208	1974	Winter 1975
Manual block valve	2MSS*MOV207	1974	Winter 1975
Safety/relief valves	B22-F013	1974	Summer 1976
Recirculation system			
Piping[3]	B35-G001	1977	Summer 1977
Pumps	B35-C001	1971	Summer 1973
Gate valves	B35-F023 B35-F067	1974	Winter 1974
Flow control valves	B35-F060A B35-F060B	1971	Winter 1973

TABLE 3.2-4
(Sheet 2 of 2)

REACTOR COOLANT PRESSURE BOUNDARY
CLASS I EQUIPMENT CODE APPLICATION

Equipment	MPL/Mark	Code[1] Edition	Addenda
High-pressure core spray system			
Isolation valve	E22-F004	1971	Winter 1973
Piping		1974	No Addenda
Standby liquid control system			
Explosive valve	C41-F004	1977	Summer 1977

(1) Code invoked in purchase order. The reference construction permit docket date was June 15, 1972.
(2) Deleted.
(3) See Section 5.4.1.3.

3.3 WIND AND TORNADO LOADINGS

3.3.1 Wind Loadings

3.3.1.1 Design Wind Velocity

Category I structures are designed to withstand the basic wind velocity (fastest mile of wind at 30 ft above the ground) of 90 mph based upon a 100-yr recurrence interval.

3.3.1.1.1 Basis for Design Wind Velocity Selection

ASCE Paper No. 3269 is used as the basis for the wind loading conditions[(1)]. Figure 1(b) of this paper indicates that the fastest mile of wind 30 ft aboveground for a 100-yr period of recurrence for the Nine Mile Point vicinity is 90 mph. A summary of wind records since 1893, as recorded at Rochester, NY; Buffalo, NY; Toronto, Canada; and site area locations, is presented in Section 2.3. An absolute peak wind speed of 73 mph was recorded at Rochester by the U.S. Weather Service Station during this period.

3.3.1.1.2 Vertical Velocity Distribution and Gust Factor

Table 1(b), ASCE Paper No. 3269[(1)] (for coastal areas), is used to obtain the variation of wind velocity with height (Vz) for the corresponding value of the basic wind velocity of 90 mph. The gust factor applied to the velocity Vz is conservatively assumed to be 1.1 over the entire height of a structure.

3.3.1.2 Determination of Applied Forces

The dynamic wind pressure q is computed using the following equation:

$$q = 0.002558\ (Vz \times \text{gust factor})^2 \qquad (3.3\text{-}1)$$

Where:

q = Dynamic wind pressure, psf

Vz = Wind velocity corresponding to height z, mph

These pressures are tabulated for various height zones in Table 3.3-1. The design wind pressure for each building is obtained by multiplying the dynamic wind pressures of Table 3.3-1 by appropriate shape and drag factors for each building[(1)]. The positive pressure on the windward side, the negative pressure on the leeward side, the negative pressure on the sides parallel to the wind direction, and the suction on the roof of the structure are considered to act simultaneously.

The design wind pressures for cylindrical structures are obtained from Table 4(f), ASCE Paper No. 3269[(1)]. The open-framed steel structures are designed to withstand the wind pressures multiplied by the appropriate shape or drag factors in accordance with ASCE Paper No. 3269[1].

3.3.2 Tornado Loadings

Systems and components important to safety and requiring tornado protection are listed in Table 3.2-1. These systems and components are located within the structures that are designed to retain their integrity without loss of function under the tornadic loadings described herein. These structures are also listed in Table 3.2-1.

3.3.2.1 Applicable Design Parameters

The structures referenced above are designed using the following tornado design parameters, as applicable:

1. Maximum rotational velocity of 290 mph.

2. Maximum translational velocity of 70 mph.

3. Minimum translational velocity of 5 mph.

4. Maximum external pressure drop of 3 psi at the vortex. Maximum rate of pressure drop is 2 psi/sec.

5. At maximum rotational speed the radius of influence is 150 ft.

6. Postulated tornado-generated missiles (Section 3.5.1).

3.3.2.2 Determination of Forces on Structures

3.3.2.2.1 Transformation of Tornadic Winds

The maximum resultant tornado wind velocity of 360 mph is obtained by the summation of maximum rotational (tangential) and maximum translational velocities. This resultant velocity is converted into an equivalent static pressure in pounds per square foot using the procedures outlined in ASCE Paper No. 3269[1] for gust, shape, and drag factors.

Although the wind velocity during a tornadic event may vary with the height of the structures, the velocity is conservatively assumed to be constant regardless of the height above grade level. The gust factor is assumed to be 1.0, since the tornadic winds are of short duration.

Tornado wind pressures and differential pressure effects are considered static loading, since the natural period of the components of structures exposed to tornadic loading is short compared to its period of application.

3.3.2.2.2 Venting of Structures

A rapid depressurization of the ambient air can occur if the low pressure within the funnel of a tornado engulfs a structure. This phenomenon would generate a maximum external pressure drop of 3 psi between the inside and outside of the structures. All Category I structures are designed to withstand an internal pressure that varies from 0 to 3 psi at a rate of 2 psi/sec, remaining at 3 psi for 2 sec, and then returning to 0 psi at a rate of 2 psi/sec.

3.3.2.2.3 Missile Impact Loads

The tornado-generated missile loads are considered impactive dynamic loads. The procedures used for converting these impactive dynamic loads from tornado-generated missiles (Section 3.5.1.4) into equivalent static loads on the structures are outlined in Section 3.5.3.

3.3.2.2.4 Tornado Load Combinations

Tornado-generated load combinations for all permanently-enclosed structures that must withstand the design basis tornado are as follows:

1. Wt = Ww

2. Wt = Wp

3. Wt = Wm

4. Wt = Ww + 0.5 Wp

5. Wt = Ww + Wm

6. Wt = Ww + 0.5 Wp + Wm

Where:

Wt = Total tornado load

Ww = Tornado wind load

Wp = Tornado-generated differential pressure load

Wm = Tornado-generated missile load

The most adverse of these combinations is used for designing each component of a structure (as applicable) in combination with other appropriate loads as specified in Sections 3.8.3 through 3.8.5.

The controlling tornado load combinations used in designing various structural elements are described below.

1. The design of the exterior concrete walls and concrete roofs is governed by either loading combination Wt = Ww + Wm or Wt = Wp. The reinforcing requirements on each face of the walls (or roofs) are checked using the above combination.

2. The design of metal siding for the cylindrical portion, above the refueling floor in the reactor building, is governed by loading combination Wt = Ww + 1/2 Wp.

3. The design of open-frame structure (e.g., roof steel framing in the reactor building following blowing off of metal roof decking) is governed by loading combination Wt = Ww.

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

The portions of the structures not designed for tornadic forces when located adjacent to or above the structures housing safety-related systems and equipment are designed so that deflection, translation, or other movement of the structure will not adversely affect the ability of these structures to perform the intended safety functions.

The reactor building superstructure above the crane rail level consists of metal siding, steel framing, and roof decking. The metal siding (Figure 1.2-11) and structural steel framing are designed to withstand the wind loads generated during a tornadic event. The roof decking is designed for normal wind loading. When normal design wind velocity is appreciably exceeded, the roof decking may blow off. The objects on or above the refueling floor are either of insufficient size to become significant missiles or are secured against tornadic forces. If the roof decking blows off the structure, externally-generated tornadic missiles may enter the spent fuel storage pool only if they are raised more than 100 ft above the ground. APED-5696 indicates that the potential for external missile impingement into the spent fuel storage pool is quite low[2]. This topical report also shows that potential depletion of water in the spent fuel storage pool during a tornadic event is of no major concern.

Objects such as steel columns, beams, bracing, and purlins, contained in non-Category I structures located in close proximity to Category I structures, are not considered potential sources of significant missiles described in Section 3.5.1.4. The design basis for tornado-generated missile protection for structures containing safety-related components is the selected external missiles shown in Table 3.5-21 (missile spectrum A of SRP 3.5.1.4), which does not include these type objects. Objects such as metal siding, roofing, roof decks, and parapets may blow off during a tornadic event. These objects are not capable of producing significant missiles (i.e., missiles capable of generating impactive dynamic loads greater than those generated by the postulated tornado-generated missiles described in Section 3.5.1.4). Since the components of the Category I structures are designed to withstand the postulated tornado-generated missile loads and tornadic wind loads simultaneously, the failure of the components that may blow off is not considered to have a detrimental effect on these structures. The tall structures such as the cooling tower and main stack are not designed for tornadic loading. However, the plant arrangement provides sufficient distance from the cooling tower or main stack to Category I structures to preclude the possibility of partial or complete collapse of these structures on Category I structures.

3.3.3 References

1. ASCE Paper No. 3269, Wind Forces on Structures, Final Report of the Task Committee on Wind Forces. Reprinted from transactions of the American Society of Civil Engineers, Vol. 126, Part II, pp 1124-1167, 1961.

2. APED-5696, Miller, D. R. and Williams, W. A. Tornado Protection for the Spent Fuel Storage Pool. General Electric, November 1968.

TABLE 3.3-1
(Sheet 1 of 1)

DYNAMIC WIND PRESSURE FOR CATEGORY I STRUCTURES

Height Above Grade (ft)	Basic Wind Velocity Corresponding to Height (mph)	Dynamic Wind Pressures (psf)
0 to 50	90	26
50 to 150	115	42
150 to 400	145	66
400 to 600	175	95

3.4 WATER LEVEL (FLOOD) DESIGN

3.4.1 Flood Protection

3.4.1.1 Flood Protection Measures for Category I Structures

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

3.4.1.1.1 Identification of Safety-Related Systems and Components

The systems and components necessary for safe shutdown of the plant (Table 3.4-1) are flood protected by physically locating them within flood-protected structures. The postulated flood levels and conditions are described in Section 2.4. Flooding that might result from the failure of systems or components containing liquid is evaluated in the appropriate failure modes and effects analysis (FMEA) sections, such as Chapter 15. The penetrations through the exterior walls of Category I structures below the design basis flood level (DBFL) are listed in Tables 3.4-2 through 3.4-6.

3.4.1.1.2 Description of Structures Housing Safety-Related Equipment

The structures housing safety-related equipment and systems, such as the reactor building, diesel generator building, and control building, are constructed with reinforced concrete walls below grade level. The personnel entrance and equipment access to these buildings are provided at or above el 261 ft LSD (Lake Survey Datum of 1935). All penetrations through the exterior walls below grade level have watertight penetration sleeves. Underground cables are protected from wetting or flooding by being housed in watertight conduits which are enclosed in reinforced concrete encasements to form electrical ductlines. As electrical ductlines enter the structure, the joints are provided with waterstops to prevent in-leakage of the design basis groundwater or floodwater into the structures. The structures housing safety-related equipment, systems, and components are identified in Table 3.4-1. The arrangement and layout drawings for plant structures are furnished in Section 1.2.

3.4.1.1.3 Means of Providing Flood Protection

<u>Exterior Flooding</u>

External flood protection is provided to prevent flood damage from the following combinations of events (Section 2.4.2.2):

1. Probable maximum precipitation (PMP) (Section 2.4.2.3.1) and historical maximum lake water level, 250.19 ft U.S. Land Survey (USLS) (Section 2.4.2.1).

2. Historical maximum precipitation (Section 2.4.1.2) and probable maximum lake stillwater level, 254 ft USLS (Section 2.4.5).

The exterior barriers, maximum flood flows, and maximum water surface elevations of the two combinations of events are shown on Figure 2.4-1. The maximum water surface elevations shown on Figure 2.4-1 are based on the PMP as determined from National Oceanic and Atmospheric Administration (NOAA) Hydrometeorological Report Nos. 51 and 52, and were evaluated with regard to roof drainage design and possible water in-leakage for safety-related buildings. These evaluations are summarized in Sections 2.4.2.3 and 2.4.10. However, plant structures are generally designed for a PMP based on Hydrometeorological Report No. 33, which results in a design basis flood level in the vicinity of the plant of el 260.6 ft.

A revetment ditch system (Section 2.4.5.5) constructed along the shore of Lake Ontario in front of Unit 2 is designed to withstand the effect of probable maximum windstorm (PMWS)[1]. This eliminates the possibility of plant flooding due to lake wave action during PMWS. The top of the revetment is at el 263 ft.

Exterior barriers (Section 2.4.2.3) located on the other three sides of the immediate plant area prevent plant flooding due to rainfall runoff from the watershed encompassing the Unit 2 site. Flood flows from the area south of the plant are routed west of Nine Mile Point Nuclear Station - Unit 1 (Unit 1) by Lake Road and the west berm. The west entrance road prevents flood flows to the east from reaching the plant area.

A storm drainage system is provided in the immediate plant area, inside the exterior barriers, to collect the normal rainfall runoff. However, runoff from the PMP will flow overland to the ditch immediately south of the shorefront revetment and to the ditch south of the plant buildings and to the lake. The storm drainage system is conservatively assumed inoperable during the PMP.

Interior Flooding

The only portions of the large-diameter circulating water system (CWS) pipe (Section 10.4.5) not encased in concrete are in the circulating water pump pits and the condenser pit shown on Figure 3.8-17. There are no safety-related system components located in the area adjacent to either of these pits.

The circulating water pump pits are physically separated by elevation and/or in a different building than the structures that house safety-related equipment. Therefore, the safety-related equipment will not be flooded as a result of a line break (including all water in the CWS and cooling tower) in the CWS in the turbine or screenwell buildings.

To eliminate potential flooding due to a rupture in the circulating water blowdown and/or tempering pipes and components in the screenwell building, the safety-related service water pumps (Section 9.2.1) and all safety-related components in the screenwell area are completely enclosed by Category 1 concrete walls. Access to these service water components is at el 261 ft. All pipe penetrations through these walls below el 261 ft have watertight penetration sleeves. Therefore, flooding of the safety-related equipment in the screenwell building is improbable.

Additionally, all construction joints below el 261 ft in the foundation mats and exterior substructure walls of the Category 1 structures have waterstops or other means to prevent in-leakage through the joints. In an isolated instance when waterstops were inadvertently omitted or damaged, the following alternate method was used to prevent in-leakage through the construction joints: in the reactor building, the sumps were lined using steel liners.

An evaluation of water levels resulting from the failure of nonseismic Category I or nontornado-protected vessels, tanks, and pipes located outside of buildings, indicates there will be no flooding of safety-related systems or components during either earthquake or tornado conditions.

For protection against flooding of ECCS components, the design features in the reactor building include flood troughs, floor drain sumps, safety-related compartment level switches, and watertight compartments for ECCS components. Appendix 3C provides additional analysis to ensure that a safe shutdown can be achieved in the event of flooding from internal sources.

3.4.1.1.4 Procedures Required for Cold Shutdown of Reactor in Flood

Special procedures for cold shutdown in case of flooding are not required. The Unit 2 site is designed against flooding by use of a revetment ditch and berms which are "hardened protection" as defined in RG 1.102.

3.4.1.1.5 Identity of Safety-Related Systems or Components Capable of Functioning in Flooded or Partially-Flooded Conditions

Safety-related systems or components supplied for Unit 2 are not designed to be capable of performing their normal functions while partially or completely flooded, with the exception of electrical cable, cable trays, and electrical penetrations.

3.4.1.2 Permanent Dewatering System

Although no permanent Category 1 dewatering system is used in the Unit 2 design, a nonsafety-related dewatering system is provided for the reactor building and vicinity area to control the groundwater drainage around the reactor building. The dewatering system is a groundwater drainage system that removes groundwater from around and below the reactor building mat (Section 2.5.4.6). Groundwater around the building is collected in embedded porous concrete piping and diverted to sumps located in the reactor building mat.

Sumps constructed of unlined concrete have access shafts and are vented to the outside. Sumps and sump pumps are sized to handle the groundwater drainage. Instrumentation requirements are described in Section 9.3.3.5.

3.4.2 Analytic and Test Procedures

The revetment ditch system along the shore of Lake Ontario in front of Unit 2 precludes generation of hydrodynamic forces from lake wave action upon Category I structures during PMWS.

The postulated flood conditions identified in Section 3.4.1 result in standing water 7.2 in deep over the average yard grade of el 260 ft LSD. The dynamic effect resulting from splashing 7-in high flood waters is considered negligible. Therefore, the hydrodynamic loads due to floods are not considered in designing the Category I structures.

The Category I structures are designed for lateral hydrostatic loads to el 261 ft, as well as the buoyant force of the water due to the probable maximum flood (PMF). For this event, the structural analysis and design of the Station structures is performed using the factor of safety against flotation of 1.1. In addition to this, the Category I structures are designed for the flood loads in combination with other loading (Section 3.8.4). The offshore intake structures are designed for the critical wave forces under different combinations of lake levels and wave heights. For example, the maximum horizontal wave force of 350 kips is calculated for the PMWS water level of el 254 ft and a maximum sustainable nonbreaking wave of 10.1 ft. The horizontal and vertical uplift forces are calculated by using methods developed by Chakradbart and by Durgin and Shiau, respectively[2,3]. In addition, the cap is conservatively designed to sustain a downward force of 1,185 kips which consists of a dynamic force of 321 kips at the PMWS water level of el 254 ft.

The revetment ditch system is designed for the probable maximum surge (PMS) and the associated wave activity on Lake Ontario due to the PMWS. Physical model tests were conducted to assure that the revetment is stable and the entire system is effective under the PMWS conditions. The testing is described in Section 2.4.5.5.

3.4.3 References

1. Design and Analysis Method for Revetment Ditch System, Nine Mile Point Nuclear Station Unit 2, Niagara Mohawk Power Corporation, August 1977.

2. Chakradbart, S. K. Wave Force on Submerged Objects of Symmetry. ASCE Waterways, Harbors, and Coastal Engineering, Vol. 99, No. WW2, pp 147-164, 1973.

3. Durgin, W. W. and Shiau, J. C. Wave Induced Pressure on Submerged Plates. ASCE Waterways, Harbors, and Coastal Engineering, Vol. 101, No. WW1, pp 59-71, 1975.

TABLE 3.4-1
(Sheet 1 of 2)

FLOOD PROTECTION FOR SAFETY-RELATED STRUCTURES AND SYSTEMS

Safety-Related Structures	Safety-Related Systems	Ground Water Level		Average Plant Grade	Elevation of Lowest Exterior Access Openings Below DBFL[5]	Elevation of Penetrations Through Exterior Walls Below DBFL[5]	Elevation of Electrical Duct Bank Penetrations Through Exterior Walls Below DBFL[5]
		Normal	DBFL[5]				
Reactor building including auxiliary bays	Primary containment, RHR, RCIC, LPCS, HPCS, ADS, service water reactor protection and standby liquid control	255	260.6	260	None	See Table 3.4-2	Top of duct el 253'
Control building	Control room with PGCC, emergency switchgear rooms, battery rooms	255	260.6	260	None	See Table 3.4-3	Top of duct el 234'-2"
Diesel generator building	Standby diesel generators and related support systems	255	260.6	260	None	See Table 3.4-6	-
Screenwell building	Service water pumps and related piping	255	260.6	260	None	See Table 3.4-4	Top of duct el 257' Two duct lines at same elevation
Main stack	Standby gas treatment system	255	260.6	260	None	See Table 3.4-6	-
Standby gas treatment building	Standby gas treatment system	255	260.6	260	None	None	-
Turbine building (main steam tunnel area)	Main steam and feedwater isolation valves and safety-related instruments	255	260.6	260	None	See Table 3.4-3	-
Piping and electrical tunnels	Service water system and electrical systems necessary for reactor control	255	260.6	260	None	See Table 3.4-5	-

TABLE 3.4-1
(Sheet 2 of 2)

FLOOD PROTECTION FOR SAFETY-RELATED STRUCTURES AND SYSTEMS

NOTES:

1. All dimensions are lake survey datum (LSD) elevations in feet and inches.

2. The tunnels housing Category I systems and components are accessible only from the adjoining buildings.

3. Pipe penetrations through exterior walls of Category I structures have watertight seals designed to withstand the flood loads.

4. Where electrical ducts penetrate Category I structures, waterstops are provided to prevent any adverse effect from flooding.

5. The DBFL of 260.6 is based on the PMP as determined from NOAA Hydrometeorological Report No. 33 and is the original basis for the plant design. See Sections 2.4.10 and 3.4 for further discussion concerning DBFL.

TABLE 3.4-2
(Sheet 1 of 3)

PENETRATIONS THROUGH EXTERIOR WALLS OF REACTOR BUILDING BELOW DBFL*

Sleeve No.	Elevation	System
W3888	210'-8"	RHR
W31670	227'	CRD, Electrical
W3227R	235'-1"	HVR
W3226R	225'-11"	HVR
W3225R	235'-7"	HVR
W3455C	235'-3"	SWP
W4288C	224'-6 1/2" HWP	Spare
W4025C	236'-2"	Instrument
W3224R	230'-3"	HVR
W4241C	256'	CMS
W3596C	247'-3"	FPL
W3177C	251'-4"	SWP
W4374C	251'-4"	SWP
W3190C	247'	RCIC
W3820C	248'-3"	FPW
W3546C	256'-3 3/8"	FPW
W4364C	254' HWP	6" Drain
W3943C	254'	DFR
W3944C	254'	DFR
W3945C	254'	DFR
W3946C	254'	DFR
W3947C	254'	DFR
W3545C	246'-3"	FPW
W3102C	208'-8" HWP	RHR
W3890C	210'-8"	RHR
W3625C	203'-4"	Instrument
W3082C	208'	HPCS
W3083C	208'	HPCS
W3084C	210'-4"	RHR
W3045C	208'	SWP
W3046C	208'	SWP
W3042C	206'	SWP
W3821C	208'-11" HWP	DFR
W3041C	206'	SWP
W3079C	203'	SWP
W3108C	208'-9 9/16" HWP	RHR
W3550C	253'-11 1/2"	DFR
W3815C	254'-6"	FPW
W3544C	256'	FPW
W4242C	256'-6"	CMS
W4243C	256'-6"	CMS
W3598C	255'	FPL
W3482O	256'-9"	SWP
W3582C	249'-11"	SWP

TABLE 3.4-2
(Sheet 2 of 3)

PENETRATIONS THROUGH EXTERIOR WALLS OF REACTOR
BUILDING BELOW DBFL*

Sleeve No.	Elevation	System
W3583C	248'-5"	SWP
W3584C	252'-11"	SWP
W3585C	251'-5"	SWP
W3570S	248'-8 1/4"	MWS
W3185C	246'-6"	CNS
W4240C	256'	CMS
W3008R	194'-3"	RHR
W3065C	182'-6"	RBCLCW
W3066C	183'-6"	RBCLCW
W4121S	177'-1 7/16"	O_2 Mapp argon
W4122S	178'-5 1/16"	O_2 Mapp argon
W4123S	179'-7 5/16"	O_2 Mapp argon
W4120S	178'-5 1/16"	Electrical
W4118S	179'-7 5/16"	Compressed air
W4119S	182'-4"	Construction water
W4112S	179'-7 5/16"	Electrical
W4113S	177'-1 7/16"	Electrical
W4110S	182'-4"	Spare
W4111S	182'-4"	Spare
W3062C	192'-3"	RCIC
W3074S	193'-2"	SWP
W3075S	193'-2"	SWP
W3018C	189'-9"	RHR
W3019C	186'-9"	RHR
W3615C	193'-2"	Instrument
W4150S	181'-4 9/16"	Spare
W4152S	178'-7 1/16"	O_2 Mapp argon
W4153S	177'-2 1/16"	Compressed air
W4151S	179'-7 9/16"	Construction water
W4154S	178'-7 1/16"	O_2 Mapp argon
W4155S	179'-7 9/16"	O_2 Mapp argon
W4156S	177'-2 1/16"	Sump discharge
W4157S	179'-7 9/16"	Electrical
W4158S	179'-7 9/16"	Electrical
W4159S	177'-2 1/16"	Electrical
W3616C	190'	Instrument
W3622C	189'	Spare
W4160S	178'-7 1/16"	Electrical
W4161S	179'-7 9/16"	Electrical
W4162S	178'-7 1/16"	Electrical
W3023R	194'	AAS, IAS, SAS, RHR
W3040C	182'-6"	SWP
W3068C	176'-6"	RBCLCW
W3067C	176'-3"	RBCLCW

TABLE 3.4-2
(Sheet 3 of 3)

PENETRATIONS THROUGH EXTERIOR WALLS OF REACTOR
BUILDING BELOW DBFL*

Sleeve No.	Elevation	System
W3022R	194'	RHR
W3617C	190'	Instrument
W3787C	185'	DFR
W3037C	190'-3"	SWP
W3038C	185'-9"	SWP
W3568C	184'-10"	MWS
W3051C	192'-6"	RBCLCW
W3048O	192'-9"	SWP
W3606C	187'-9"	Spare
W3605C	189'	MWS
W3623C	188'-2"	Spare
W3052C	186'	RBCLCW
W3565S	183'	Instrument

KEY:
- AAS = Breathing air
- CMS = Containment atmosphere monitoring
- CNS = Condensate makeup and drawoff
- CRD = Control rod drive hydraulic
- DFR = Reactor building floor drains
- FPL = Fire protection - low-pressure CO_2
- FPW = Fir protection - water
- HPCS = High-pressure core spray
- IAS = Instrument air
- HVR = Reactor building ventilation
- HWP = High work point
- MWS = Makeup water
- RBCLCW = Reactor building closed loop cooling water
- RCIC = Reactor core isolation cooling
- RHR = Residual heat removal
- SAS = Service air
- SWP = Service water

* The table is based on a DBFL of 260.6 ft, the original basis for plant design. See Sections 2.4.10 and 3.4 for further discussion concerning DBFL.

TABLE 3.4-3
(Sheet 1 of 1)

PENETRATIONS THROUGH EXTERIOR WALLS OF CONTROL AND TURBINE BUILDINGS BELOW DBFL[1]

Sleeve No.	Elevation	System
Turbine Building[2]		
W5335C	253'-9"	DWS
W0504C	255'-11 1/16"	15" Roof drain
W0505C	252'-9"	FPW
W5045C	254'	4" Drain
W0508C	254'-6"	FPW
W0509C	256'	FPW
W0024C	257'	FWS
W5275C	257'	FWS
W0023C	257'	FWS
W5274C	257'	FWS
W0507C	254'	FPW
W0506C	254'	FPW
Control Building		
W6185R	230'-7"	Vent duct
W6186R	230'-7"	Vent duct

KEY: DWS = Domestic water
FPW = Fire protection - water
FWS = Feedwater

(1) The table is based on a DBFL of 260.6 ft, the original basis for plant design. See Sections 2.4.10 and 3.4 for further discussion concerning DBFL.

(2) The turbine building is Category I in the following areas only:

a. Electrical bay area between column lines AM and AK, up to el 261'.
b. Main steam tunnel area and area underneath main steam tunnel between column lines 10 and 12, when providing support to main steam tunnel.
c. Pipe tunnels between column lines 10 and 12 up to el 248'.

TABLE 3.4-4
(Sheet 1 of 1)

PENETRATIONS THROUGH EXTERIOR WALLS OF RADWASTE BUILDING AND SCREENWELL BUILDING BELOW DBFL[1]

Sleeve No.	Elevation	System
Radwaste Building[2]		
W8001C	248'-9"	TBCLCW
W8002C	250'-9"	TBCLCW
Screenwell Building		
W2250C	253'-7 3/8"	DFM
W2247C	252'-1 1/4"	10" Roof drain

[1] The table is based on a DBFL of 260.6 ft, the original basis for plant design. See Sections 2.4.10 and 3.4 for further discussion concerning DBFL.

[2] The radwaste building is designed as a Category I structure. However, Category I classification is not used in construction of the radwaste building.

KEY: TBCLCW = Turbine building closed loop cooling water
DFM = Miscellaneous floor and equipment drains

TABLE 3.4-5
(Sheet 1 of 3)

PENETRATIONS THROUGH EXTERIOR WALLS OF PIPE TUNNELS BELOW DBFL*

Sleeve No.	Elevation	System
W1251S	256'-1 3/4"	LWS
W1252S	256'-1 3/4"	MWS
W1253S	255'-1 3/4"	LWS
W1254S	256'-1 3/4"	LWS
W1255S	255'-1 3/4"	LWS
W1256S	256'-2 1/4"	LWS
W1257S	255'-1 3/4"	LWS
W1258S	249'-5"	Drain
W1132R	251'-10 1/4"	Spare
W1134R	250'-6"	Spare
W1133C	256'-2 1/4"	OFG
W1215C	255'-10 7/8"	4" Drain
W1020C	257'-0 3/4"	4" Drain
W1285C	247'-4"	CCS
W1208C	254'	DFM
W1077C	254'	4" Sump discharge
W1143C	246'-11 3/8"	SWP
W1145C	245'-11 7/16"	SWP
W1147C	244'-11 7/16"	SWP
W1207C	254'	DFM
W1006C	252'-7 3/16"	AAS
W1141C	247'-10 3/4"	SWP
W1078C	253'-8 5/8"	4" Sump discharge
W1137C	250'-0 5/8"	SWP
W1139C	248'-11 5/16"	SWP
W1286C	251'-0 1/8"	LWS
W1140C	248'-11 9/16"	SWP
W1144C	246'-11 7/16"	SWP
W1148C	244'-11 1/4"	SWP
W1138C	249'-11 15/16"	SWP
W1142C	247'-11 1/16"	SWP
W1146C	245'-11 3/8"	SWP
W1081C	255'-3 1/2"	Spare
W1259C	251'	LWS
W1276C	249'	DFT
W1243C	256'-6"	Abandoned
W1244C	251'-8 1/4"	Abandoned
W1109S	252'-5"	8" Drain line
W1219S	251'-3"	CCS
W1220S	248'-9"	CCS
W1069C	251'-4"	SWP
W1283C	251'-4"	SWP
W1070C	247'	RCIC
W1057C	254'-6"	SWP
W1058C	254'-6"	CWS

TABLE 3.4-5
(Sheet 2 of 3)

PENETRATIONS THROUGH EXTERIOR WALLS OF PIPE TUNNELS BELOW DBFL*

Sleeve No.	Elevation	System
W1056C	247'-7"	SWP
W1097C	249'-5"	Spare
W1099C	249'-5"	FPW
W1100C	249'-5"	FPW
W1101C	249'-5"	FPW
W1102C	249'-5"	FPW
W1103C	249'-5"	FPW
W1104C	249'-5"	FPW
W1105C	249'-5"	FPW
W1106C	249'-5"	FPW
W1107C	249'-5"	FPW
W1108C	249'-5"	FPW
W1186C	245'-3 15/16"	MSS
W1187C	245'-3 15/16"	MSS
W1188C	245'-3 15/16"	MSS
W1238C	245'-3 15/16"	MSS
W1239C	245'-3 15/16"	MSS
W1240C	245'-3 15/16"	MSS
W1241C	245'-3 15/16"	MSS
W1242C	245'-6 1/16"	MSS
W1189C	245'-3 15/16"	MSS
W1190C	245'-6 1/16"	MSS
W1249C	247'	IAS
W1152C	252'-7"	SWP
W1153C	249'-10"	SWP
W1154C	247'-9 3/8"	SWP
W1155C	246'-0 3/8"	SWP
W1195C	252'-7"	WTS
W1196C	252'-7"	WTS
W1216C252'-4"FPW		

KEY:
- AAS = Breathing air
- CWS = Circulating water
- DFT = Turbine building floor drains
- DM = Miscellaneous floor and equipment drains
- FPW = Fire protection - water
- IAS = Instrument air
- LWS = Radioactive liquid waste
- MSS = Main steam
- MWS = Makeup water
- OFG = Offgas
- RCIC = Reactor core isolation cooling
- SWP = Service water
- CCS = Turbine building closed loop cooling water
- WTS = Water treating

TABLE 3.4-5
(Sheet 3 of 3)

PENETRATIONS THROUGH EXTERIOR WALLS OF PIPE TUNNELS BELOW DBFL*

* The table is based on a DBFL of 260.6 ft, the original basis for plant design. See Sections 2.4.10 and 3.4 for further discussion concerning DBFL.

TABLE 3.4-6
(Sheet 1 of 1)

PENETRATIONS THROUGH EXTERIOR WALLS OF DIESEL GENERATOR BUILDING AND MAIN STACK BELOW DBFL*

Sleeve No.	Elevation	System
Diesel Generator Building		
W1294C	258'-3"	6" drainage
W1295C	256'-6"	12" drainage
W1328C	255'-10"	6" drainage
W1329C	255'-10"	6" drainage
Main Stack		
W2275C	256'-6"	DFM
W2276C	254'-0 1/8"	ARC

KEY: ARC = Condenser air removal
DFM = Miscellaneous floor and equipment drain

* The table is based on a DBFL of 260.6 ft, the original basis for plant design. See Sections 2.4.10 and 3.4 for further discussion concerning DBFL.

TABLE 3.4-7
(Sheet 1 of 1)

PERFORMANCE OF WATER STOP MATERIAL IN EXPECTED ENVIRONMENT

Material	Expected Environment				Expected Performance of Material[2][3]			
	Temperature Range[1]	Chemicals	Radiation Level	Aging	Temperature Range	Chemicals	Radiation Level	Aging
Styrene-Butadiene synthetic rubber waterstops	-20°F to +325°F	Unit 2 site has average pH -8.0-8.4. No acidic environment expected within the walls below grade area.	Below 1.4×10^7 rads	40 yr at normal operating temperature (109°F)	-35°F to +176°F	Unaffected by acidic or alkaline soils or soil bacteria.	2×10^6 rads before threshold damage. 1×10^7 rads before 25% damage. 6.0×10^7 rads before 50% damage.	40 yr at 109°F

(1) Temperature range varies from -26°F minimum outside at Site, 109°F normal operating inside secondary containment, to 325°F maximum accident inside secondary containment. The worst-case design conditions during which the water stop must function do not exceed the expected performance temperature of the material.

(2) Safety-related water stops are insulated and sealed from ambient environmental conditions to establish 40-yr qualified life.

(3) Water stop systems are required to contain long-term flooding from cracks in low temperature (<175°F) systems which have a large inventory of water, e.g., systems connected to the suppression pool. Cracks in these systems could remain undetected for long periods of time assuming failure of nonredundant leak detection systems. Under these conditions, watertight cubicles, which employ several water stops, prevent the spread of flooding to redundant safe shutdown equipment. This is discussed in Appendix 3C.5.

Water stops are not required to contain flooding from high temperature (>175°F) systems, e.g., RCIC, RWCU, or RHR (shutdown cooling) systems. Loss of water from a HELB or moderate-energy line crack in these systems is quickly detected and isolated either by the respective system instrumentation or by redundant leak detection system. Under these conditions, only a limited quantity of water is released, and flooding of redundant safe shutdown equipment is not a concern. Thus, the water stop function is not required.

Required service conditions have been evaluated in establishing the 40-yr qualified life of the water stops. In addition, the water stops are protected with layers of insulation and caulking material for added assurance that the system remains functional under all conditions.

Expected environmental radiation level of 1.4×10^7 rads is consistent with the manufacturer's design data.

3.5 MISSILE PROTECTION

3.5.1 Missile Selection and Description

The following criteria have been adopted to assess the plant's integrated design to afford protection from generated missiles of the type postulated in this section:

1. No loss of RCPB and containment function.

2. No direct loss of reactor coolant.

3. No offsite dosage exceeding the limits of 10CFR100.

4. No loss of integrity of the spent fuel pool.

5. No loss of function for systems required to shut down the reactor and maintain it in a cold shutdown condition (considering that offsite power is not available for shutdown of the plant), or mitigate the consequences of the missile damage so that:

   a. No equipment is allowed to be damaged in one safety-related division, e.g., Division I, from internally-generated missiles originated from another safety-related division, e.g., Division II.

   b. Missiles generated from nonsafety-related equipment do not damage any safe shutdown equipment.

   The systems required to be protected are:

   a. Emergency core cooling systems (ECCS).

   b. Service water system (SWP).

   c. Reactor core isolation cooling (RCIC) and residual heat removal (RHR) systems.

   d. Diesel generator and associated support systems.

   e. Control rod drive (CRD) hydraulic (scram section) system.

   f. Spent fuel pool cooling (SFC) system.

   g. Remote shutdown system.

   h. Reactor protection system (RPS).

   i. All containment isolation valves.

j. Heating, ventilating, and air conditioning (HVAC) systems required during operation of the essential components.

6. When a potential missile-generating component is considered an initiating event, the single-failure criteria is used in the missile protection assessment.

Although Class 1E sensors to the RPS are located on the TCV, high-pressure turbine stop valve, and main condenser, failure of these sensors does not prevent the reactor from being safely shut down since other RPS sensors (high pressure scram or high flux scram) located in safety-related buildings provide sufficient backup. Therefore, these sensors are not analyzed for missile hazards inside the turbine building.

Essential structures, systems, and components are protected from the effects of internal missiles by one or more of the following practices:

1. Locating the system or component in an individual missile-proof structure.

2. Physically separating redundant systems or components of the system from the missile trajectory path.

3. Providing localized protective shields or barriers for systems and components.

4. Designing the particular structure or local protective shield/barrier to withstand the impact of the most damaging missile.

5. Providing design features on the potential missile source to minimize the probability of missile generation.

6. Orienting the potential missile source in such a manner as to avoid detrimental missile impact.

3.5.1.1 Internally-Generated Missiles

3.5.1.1.1 Locations of Structures, Systems, and Components

All structures, systems, and components important to safety and their locations are listed in Table 3.2-1. Items are protected against damage for internally-generated missiles as described in Sections 3.5.1.1.4 and 3.5.1.1.5. For locations of these structures, systems, and components see general arrangement drawings, Figures 1.2-1 through 1.2-40.

3.5.1.1.2 Applicable Seismic Category and Quality Group Classifications

For designations of seismic category and quality group classification, see Table 3.2-1.

3.5.1.1.3 Piping and Instrumentation Drawings

Piping and instrumentation drawings (P&IDs) of safety-related structures, systems, and components are identified in the applicable section, as shown in Table 3.5-1.

3.5.1.1.4 Identification of Missiles, Their Sources, and Basis of Selection

Missile protection is provided within the plant structures that are important to safety inside and outside the containment for three general sources of postulated internal missiles:

Rotating Component Failure

Potential missiles from rotating equipment include:

1. Pumps.

2. Fans.

3. Compressors.

4. Turbines.

5. Electric motors.

Pressurized Component Failure

Potential missiles from pressurized equipment include:

1. Valve bonnets.

2. Valve stems.

3. Thermowells.

4. Vessel head bolts.

5. Carbon dioxide (CO_2), Halon, and dry chemical - CO_2 activated fire extinguishers.

6. Self-contained breathing apparatus.

7. Cutting, burning, and welding pressurized components.

8. Hydrogen, oxygen, and nitrogen compressed gas cylinders.

Pipe whip and jet impingement effects due to postulated pipe failures are discussed in Sections 3.6.1 and 3.6.2.

With the exception of compressed gas bottles labeled "DOT-3AA (Pressure)," inspected and handled in accordance with Compressed Gas Association, Inc., Pamphlets C-6 and P-1 (Standard for Visual Inspection of Compressed Gas Cylinders and Standard for Safe Handling of Compressed Gas in Containers, respectively), protection is provided against the potential failure of high-pressure gas bottles and non-ASME III accumulators by their physical separation from locations with safety-related equipment. Fire extinguishers are UL labeled and have passed tests qualifying them as not being potential missiles.

Pressurized equipment noted in items 5, 6, 7, and 8 above will be removed and transported to a designated area for periodic inspection and maintenance.

Gravitational Missiles

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

Nonseismic items and systems in locations where their failure could adversely affect safety-related equipment are classified as follows:

1. General

   All suspended nonsafety-related piping, conduit, instrument tubing, structures, architectural features, and HVAC ducting are supported to prevent collapse during SSE.

2. Cable Trays

   All cable trays for both Class 1E and non-Class 1E circuits are seismically supported when located in seismic Category I locations.

3. Equipment for Maintenance

   All other equipment, such as hoists, that is required during maintenance either will be removed during operation, is designed to prevent it from becoming a missile during a SSE, or is so located such that it cannot become a gravitational missile.

4. Fire Extinguishers

   Mounted on commercially available brackets, units are not supported seismically because the latching will interfere with firefighting activities. A walkdown will be performed to ensure sufficient separation exists, such that if support failure occurs and the extinguisher falls, safety-related components in the vicinity will not be affected.

5. Self-Contained Breathing Apparatus

   Stored on commercially available racks in normally locked cabinets. Locations are selected for convenience to firefighting activities that may be in the general vicinity of safety-related equipment.

6. Cutting, Burning, and Welding Pressurized Components

   Units are transported, stored, and used in accordance with a plant administrative procedure to ensure a low probability of damage to safety-related components.

3.5.1.1.5 Missile Protection Provided

Rotating Component Failure Missiles

Catastrophic failure of rotating equipment (i.e., pumps, turbines, motors, and compressors) leading to the generation of missiles is not considered credible. Massive and rapid failure of these components is not considered credible because of the material characteristics; inspections; quality control during fabrication, erection, and operation; conservative design; and prudent operation as applied to the particular component.

The most substantial piece of nuclear steam supply system (NSSS) rotating equipment is the recirculation pump and motor. The potential for missiles from this source has been evaluated and it has been concluded that destructive pump overspeed cannot result in the generation of missiles[1].

Motors will not become missiles because the rotation speed is limited to within design speed. The pump shaft failure decouples the rotor for the overspeed driving blowdown force. Only those cases with peak torques less than those required to fail the pump shaft (five times rated) will have the capability to drive the motor to overspeed. When missile generation probabilities are considered along with a discussion of the actual load-bearing capabilities of the system, it is evident that these considerations support the conclusion that it is unrealistic that the motors would become missiles.

The pump impeller or fan blades will not become missiles during coupling failures because braking forces applied by the process fluids will limit the rotational speed to less than the normal operating speed.

Although missile generation is considered incredible, all safety-related and nonsafety-related rotating components, including pumps, fans, and turbines, are analyzed to demonstrate that no missiles are generated that could strike equipment whose failure could impact plant shutdown. Any one of the following methods to achieve this demonstration may be applied:

1. Casings and guards are analyzed to show that potential missiles are retained.

2. Missile generation is shown to be conclusively incredible due to components testing, quality control, modes of operation, and designing with a very large safety margin which prevents failure of those parts designated as potential missiles.

3. The trajectory of potential missiles is analyzed to show that it does not strike equipment whose failure could impact plant shutdown.

Pump and fan casings are analyzed for penetration of potential missiles using the Ballistic Research Lab Formula (BRL) by Gwaltney[7].

When the potential for missile penetration of casings is shown to be possible, the following missile characteristics are used for evaluation:

1. Missile flight path will be within ±25 percent of the plane of rotation.

2. Ricocheting is not assumed if missile strike direction is within 45 degrees to the local normal of the target surface.

3. Secondary missiles from concrete barriers are considered if the primary missile has more than 4,000 ft/lb of kinetic energy.

4. Nonsafety-related targets are evaluated to ensure that secondary missiles could not strike targets whose failure could impact safe plant shutdown.

Redundant overspeed tripping devices ensure that the turbine does not reach runaway speed where possible component failure could take place.

Pressurized Component Failure Missiles

The internally-generated missiles in this category are not considered credible for the following reasons:

1. Thermometers installed on piping or in wells are evaluated in Table 3.5-2. The analysis of the thermowell shows that thermowell ejection is very improbable because of its highly conservative design. Consequently, it is not considered a probable missile source.

2. Valves of ANSI 900-psig rating and above, constructed in accordance with ASME Section III, are pressure seal

bonnet-type valves. For pressure seal bonnet valves, valve bonnets are prevented from becoming missiles by the retaining ring, which would have to fail by the yoke, capturing the bonnet or reducing bonnet energy. Because of the highly conservative design of the retaining ring of these valves, bonnet ejection is highly improbable. Hence, bonnets are not considered credible missiles.

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

4. 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. No credible valve stem missiles were identified at Unit 2.

5. Pressurized compressed gas cylinders are manufactured to Department of Transportation standards and stamped with a "DOT-3AA (Pressure)" designation. These cylinders have stringent manufacturing controls which meet the requirements of 10CFR50 when used in missile-sensitive areas. Controls such as receipt inspection and safe handling requirements in accordance with Compressed Gas Association, Inc., Pamphlets C-6 and P-1 are imposed by the requirements of the procurement specification. Under these conditions, the causes of failures are virtually eliminated, and such cylinders need not be considered as credible missiles.

6. Nuts, bolts, nut and bolt combinations, and nut and stud combinations have only a small amount of stored energy and thus are of minimal concern as potential missiles.

3.5.1.2 Internally-Generated Missiles (Inside Containment)

Location of Structures, Systems, or Components

All safety-related systems and components inside containment are listed in Table 3.2-1. They are protected against damage from internally-generated missiles as described in Sections 3.5.1.1.4 and 3.5.1.1.5.

3.5.1.3 Turbine Missiles

3.5.1.3.1 Turbine Placement and Orientation

Turbine placement and orientation for the three units affecting the turbine missile evaluation for Unit 2 are shown on Figure 3.5-1. They are the turbines of the Nine Mile Point Nuclear Station - Unit 1 (Unit l), Unit 2, and James A. FitzPatrick Power Station. The spin axes of the Unit 1 and Unit 2 turbine generators are oriented in an east-west direction. The James A. FitzPatrick Power Station turbine generator has its spin axis oriented with a 15-deg clockwise rotation from a north-south direction. Figure 3.5-1 also indicates the ±25-deg missile ejection zone for low-trajectory turbine missiles resulting from low-pressure turbine discs[2].

A plan view of plant regions located at Unit 2 is shown on Figure 3.5-1 along with the turbine generators of Unit 1, Unit 2 and the James A. FitzPatrick plant. Note that applicable low-trajectory targets are those within the ±25-deg missile ejection zones. For high-trajectory missiles, target areas are all aboveground, Category I structures.

Tables 3.5-3 through 3.5-16 provide the probabilities of turbine missile strikes in various events. Due to very low probability of turbine missile strikes, as demonstrated by the above-referenced data, it is not necessary to design the safety-related structures for turbine missiles.

3.5.1.3.2 Missile Identification, Characteristics, and Target Description

<u>Missile Identification and Characteristics</u>

The turbine generators located at Unit 1, Unit 2 and James A. FitzPatrick Power Station are manufactured by General Electric Company (GE). The turbine type for Unit 1 and Unit 2 is a 38-in last-stage bucket, while the turbine type for the James A. FitzPatrick plant is a 43-in last-stage bucket.

At Unit 2, the original built-up type rotor design has been replaced with a monoblock type rotor design which reduces the susceptibility to stress corrosion cracking. Due to this replacement, the probability of missile generation from the Unit 2 turbine is statistically insignificant.

For turbine missile evaluation at Unit 1 and the James A. FitzPatrick plant, a hypothetical missile is considered generated in the disc plane. As it penetrates the stationary turbine parts, the missile is deflected from the vertical plane. It has been determined that the deflection angle is a maximum of 5 deg on each side of the plane of the disc for inner-stage buckets. For last-stage buckets, the deflection angles may be up to 25 deg on each side of the plane of the disc. The missile characteristics used for this turbine missile strike probability evaluation are provided in Tables 3.5-17 and 3.5-18[3,4].

Target Description

Systems, equipment, and components required for safe shutdown and to maintain cold shutdown of the reactor, or to prevent the release of radiation to within allowable limits, are housed in the following structures:

1. Reactor building.
2. Control building.
3. Diesel generator building.
4. Screenwell building service water pump room.
5. Standby gas treatment building and railroad access lock.
6. Radwaste building.
7. Auxiliary service building and north and south auxiliary bays.
8. Intake structure, pipe, and shaft.
9. Main steam tunnel.

These targets are considered to be the safety-related regions for turbine missile evaluation.

3.5.1.3.3 Probability Analysis

For Unit 2, the probability of generation and ejection (P1) value from a Unit 2 postulated missile is insignificant (~0), due to the replacement of original built-up type rotor with monoblock type rotor. Therefore, the overall probability (P_4) for a Unit 2 postulated missile is insignificant (~0).

The evaluation of a postulated turbine missile from Unit 1 and the James A. FitzPatrick plant is based on the probability of missile generation and on the effects attributed to it. The overall probability of unacceptable damage to the critical plant regions, P_4, is the product of three contributing factors:

$$P_4 = P_1 \times P_2 \times P_3 \qquad (3.5\text{-}1)$$

Where:

P_1 = The probability of generation and ejection of a high-energy missile

P_2 = The probability that the missile will strike a safety-related region

P_3 = The probability that the missile strike will damage its target in a manner leading to unacceptable consequences

P_4 = The overall probability

Probability of Generation and Ejection (P_1)

A turbine missile can be caused by brittle fracture of a rotating turbine part at or near turbine operating speed or by ductile fracture upon runaway after extensive, highly-improbable control system failures[(2)]. The operating experience of GE turbines clearly demonstrates that the structural integrity record of discs and rotor has been excellent. This excellent operational record can be attributed to:

1. Careful control of alloy chemistry and forging heat-treating cycles.

2. Improved steel mill practices in vacuum pouring and alloy addition resulting in more uniform and defect-free forgings.

3. Improved ultrasonic and magnetic particle testing techniques that ensure sound discs, which equal or exceed the specified design standard.

4. Redundancy in the control systems. As a minimum, the GE turbine is equipped with two separate and redundant overspeed protection systems.

These factors minimize the probability of missile generation.

For a postulated turbine failure, two speed failures were considered: the design overspeed failure (120 percent rated speed) and the destructive overspeed failure (180 percent rated speed). The GE estimate for the turbine failure rate is 8.67×10^{-9}/yr/turbine for design overspeed (120 percent rated speed) for 43-in last-stage buckets[(3)], whereas turbine failure rate is statistically insignificant for 38-in last-stage buckets[(4)] for design overspeed (120 percent rated speed). For destructive overspeed (180 percent rated speed), failure rate has been estimated to be 5.0×10^{-9}/yr/turbine for both 43-in last-stage buckets[(3)] and 38-in last-stage buckets[(4)]. Bush has obtained a failure rate of 3.3×10^{-5} to 3.1×10^{-4}/turbine yr for a turbine population corrected to be relevant to nuclear reactors[(5)].

RG 1.115, which is based on Bush's results, recommends a failure rate of 1.0 x 10^{-4} for design overspeed and for destructive overspeed turbine failures. However, Standard Review Plan (SRP) 3.5.1.3 allows an annual turbine failure rate of 10^{-4} (subdivided as 6 x 10^{-5}/turbine yr for design speed failures and 4 x 10^{-5}/turbine yr for destructive overspeed failures). Hence, in addition to using the turbine manufacturer's estimated failure rate, the value of 1 x 10^{-4} was used for the overall turbine failure rate (subdivided as 6 x 10^{-5}/turbine yr for design speed failure and 4 x 10^{-5}/turbine yr for destructive overspeed failure).

Probability of Missile Strike (P_2)

The probability of a strike on a safety-related region, P_2, is a function of the number of missiles, the ejection velocity, the envelope of possible ejection directions, and the plant layout. A solid-angle approach has been used to calculate the strike probability. The analytical approach, based on the assumptions described in Table 3.5-19, is as follows.

The turbine spins about the z-axis of the reference system (Figure 3.5-2). The variable angles required to describe the missile motion are also displayed. Based on data provided by the turbine manufacturer (GE), a postulated missile is thrown from the turbine with initial velocity V_0[3,4].

The angle θ (Figures 3.5-2 and 3.5-3) is limited to the range:

$$\frac{\pi}{2} - \delta_1 \leq \theta \leq \frac{\pi}{2} + \delta_2 \tag{3.5-2}$$

The 25-deg deflection angle recommended by the NRC was interpreted as applying only to the end discs in each hood in accordance with the GE report. A 5-deg deflection angle was assumed for interior discs.

The probability that a single disc fragment strikes a critical area, A_0, is defined as:

$$P(A_o) = \int_{\Omega_o} f(\Omega)\, d\Omega \tag{3.5-3}$$

Where:

Ω_o = Solid angle that must be subtended by the initial velocity vector for a missile to strike A_0

$d\Omega$ = Differential solid angle

$f(\Omega) =$ Probability density function

From Figure 3.5-2:

$$d\Omega = \cos\phi \; d\phi \; d\Psi \qquad (3.5\text{-}4)$$

Given V_0, the elevation angle, ϕ, necessary to hit any point on A_0 (described by r, y, and Ψ on Figure 3.5-2) is determined from classical trajectory theory as:

$$\phi = \tan^{-1}\left[\frac{1 \pm \left[1-\left(\frac{rg}{v_o^2}\right)^2 - 2\left(\frac{yg}{v_o^2}\right)\right] 1/2}{\left(\frac{rg}{v_o^2}\right)}\right]$$

(3.5-5)

In Equation 3.5-5, air resistance is neglected and the ± refers to high- and low-trajectory missiles, respectively.

The probability density function, $f(\Omega)$, is determined by assuming:

$$f(\Omega) = \text{constant} = f_0 \qquad (3.5\text{-}6)$$

for:

$$0 \leq \beta \leq 2\pi$$

$$\frac{\pi}{2} - \delta_1 \leq \theta \leq \frac{\pi}{2} + \delta_2$$

(3.5-7)

and $f(\Omega) = 0$, for all other θ.

From probability theory, it is required that:

$$\int_{all\,\Omega} f(\Omega)\, d\Omega = 1 \qquad (3.5\text{-}8)$$

Therefore,

$$f_o = \frac{1}{2\pi\left(\sin\delta_1 + \sin\delta_2\right)} \qquad (3.5\text{-}9)$$

If n disc fragments are generated, the strike probability is:

$$P_2\ (A_o) = \frac{n}{2\pi\left(\sin\delta_1 + \sin\delta_2\right)}\int_{\Omega o}^{d\Omega} \tag{3.5-10}$$

The computer program MISSILE (Appendix 3A) has been developed to calculate the strike probability using Equation 3.5-10. The analysis considers both high- and low-trajectory missiles. The discrimination between high and low trajectory is based on the elevation angle at which the missiles are ejected, the missiles' initial velocity, and the distance from the missiles' origins to the critical plant region. Figures 3.5-4 and 3.5-5 represent the top and side views of an idealized target. The strike probability of the target is found by numerically integrating Equation 3.5-10, to give:

$$P_2 = \frac{n}{2\pi\left(\sin\delta_1 + \sin\delta_2\right)}\sum_{i=1}^{n\Psi}\left(\cos\phi_i\right)\left(\Delta\phi_i\right)\Delta\Psi \tag{3.5-11}$$

for:

$$\pi/2 - \delta_1 \leq \phi_i \leq \pi/2 + \delta_2 \tag{3.5-12}$$

and $P_2 = 0$ for:

$$o \leq \theta_i < \frac{\pi}{2} - \delta_1,\ \frac{\pi}{2} + \delta_2 < \theta_i \leq \pi \tag{3.5-13}$$

Where:

θ_i = $\cos^{-1}(\cos\phi_i \cos\Psi_i)$

n_Ψ = Number of ground angle increments taken through the target

From Figures 3.5-4 and 3.5-5:

$$\Delta\Psi = \frac{\Psi_{max} - \Psi_{min}}{n_\Psi}$$

$$\Psi i = \Psi_{min} + (i - 1/2)\Delta\Psi$$

$$\Delta\phi i = \left|\phi_2^i - \phi_1^i\right|$$

(3.5-14)

$$\phi i = 1/2\left(\phi_1^i + \phi_2^i\right)$$

Equation 3.5-5 is used to determine $\phi^i_{1,2}$.

Probability of Penetrating Concrete Structures (P_3)

Considering the random missile orientation, the concept of penetration probability, P_3, recommended by the NRC (SRP 3.5.1.3), was introduced by assuming that the variation in the concrete structural thickness, T, is uniformly distributed between the maximum and minimum value for a particular fragment. For each type of missile fragment, the probability of penetrating a concrete structure, P_3, is:

$$P_3 = \frac{T_{max} - T}{T_{max} - T_{min}}$$

(3.5-15)

Where:

T = Thickness of the concrete structure

T_{min}, T_{max} = Concrete thickness required to prevent penetration, defined by the extreme values of the missile. T_{min} and T_{max} are obtained from the modified National Defense Research Council (NDRC) formula[6].

If multiple barriers are considered, the protection is deemed adequate if the last barrier stops the missile without generating secondary missiles that could damage the essential systems. For calculating residual velocities after the missile has perforated a barrier, the following relationship taken from RG 1.115 is conservative:

$$V_r = (V_i^2 - V_p^2)^{½} \qquad (3.5\text{-}16)$$

Where:

V_r = Residual missile velocity after perforation

V_I = Incident missile velocity

V_p = Incident missile velocity required to just perforate the barrier, calculated by conservative use of penetration data

The probability of penetrating a concrete structure, P_3, can be calculated based on this residual velocity, taking into account the perforation of missile barriers.

Probability Calculation and Acceptance Criteria

In RG 1.115, the NRC considers the value of 10^{-7}/yr an acceptable risk rate for the loss of an essential system from a single event due to low-trajectory turbine missiles. Also, SRP 2.2.3 indicates that an "expected rate of occurrence of potential exposures in excess of the 10CFR100 guidelines of approximately 10^{-6}/yr is acceptable if, when combined with reasonable qualitative arguments, the realistic probability can be shown to be lower."

The probability calculation was based on estimates of the individual buildings considering both low and high trajectories and both design and destructive overspeed failures from the turbine generators of Unit 1, Unit 2 and the James A. FitzPatrick plant. These were then combined to determine the overall probability of damage (P_4) for Unit 2. The overall probability of damage (P_4) has been estimated using the turbine failure probabilities (P_1) suggested by both the turbine manufacturer (GE) and the NRC.

The turbine manufacturer's missile data (missile ejection velocity, missile weight, characteristic dimensions, etc.) have been used to calculate the strike probability (P_2) on the critical plant regions for both low- and high-trajectory turbine missiles. These data have also been used to evaluate the penetration probability (P_3) defined in Section 3.5.1.3.3.

Tables 3.5-3 through 3.5-8 show the calculated probabilities of damage on the Unit 2 safety-related regions due to low-trajectory turbine missiles for design overspeed and destructive overspeed turbine failures. Tables 3.5-3 through 3.5-5 show the calculated results using the turbine manufacturer's failure rates, while Tables 3.5-6 through 3.5-8 show the calculated results using the turbine failure rates suggested by the NRC. It should be noted that the information in Tables 3.5-3 through 3.5-8 was determined by considering the entire front surface areas and roof areas of all buildings containing the essential systems, as shown on Figure 3.5-1. This is conservative since it is much greater than the actual projected areas of the essential systems.

Similarly, Tables 3.5-9 through 3.5-14 present the calculated probabilities of damage to the Unit 2 safety-related regions due to high-trajectory turbine missiles for design overspeed and destructive overspeed turbine failures, respectively. The entire front surface areas and roof areas of the buildings containing essential systems were used to determine the strike probabilities rather than the areas of the essential systems. These probabilities were obtained from the sum of the probabilities due to various missile ejection velocities and due to all fragments for turbine generators of Unit 1, Unit 2 and the James A. FitzPatrick plant.

The probability of damage for all Unit 2 buildings due to turbine missiles generated from Unit 1, Unit 2 and the James A. FitzPatrick plant for the two failure modes is presented in Tables 3.5-15 and 3.5-16. It can be observed that the overall probability of damage by turbine missiles is 0.962×10^{-7}/yr for Unit 2 if the probability of turbine failure rate of 1.0×10^{-4}/yr recommended by the NRC is used for Unit 1 and the James A. FitzPatrick plant. These results are within the acceptance value of 10^{-6}/yr as outlined in SRP 2.2.3, and the acceptance value of 10^{-7}/yr as specified in RG 1.115. These calculated figures are conservative. The overall probability for damage by turbine missiles for Unit 2, when estimated on a more realistic basis with manufacturer's probability, is much lower.

3.5.1.3.4 Turbine Overspeed Protection

The turbine is equipped with a redundant, testable overspeed trip system to minimize the possibility of a turbine overspeed event. The system and its test program are described in Section 10.2.2.2.

3.5.1.3.5 Turbine Valve Testing

Turbine valve testing and test frequency are described in Section 10.2.

3.5.1.3.6 Turbine Characteristics

Turbine characteristics, design, and operation are described in Section 10.2.

3.5.1.4 Missiles Generated by Natural Phenomena

It is assumed that a tornado could generate missiles as listed in Table 3.5-21. These design basis missiles are in accordance with the spectrum of missiles of SRP 3.5.1.4.

The structures and/or barriers used to provide missile protection are listed in Table 3.5-22 along with the list of the safety-related systems protected by these barriers. The minimum thickness of reinforced concrete barriers that are designed to provide protection against missiles generated by natural phenomena is 24 in. The strength of concrete used in the construction of these barriers is 3,000 psi at 28 days as a minimum. The corresponding curing time conforms to ACI 301, Chapter 12, as supplemented in Section 3.8.4.6.

3.5.1.5 Missiles Generated by Events Near the Site

No missiles of any significance are expected to be generated by events near the site, due to the distances from nearby transportation routes. The nearest transportation route is about 11.3 km (7 mi) for waterbound traffic, approximately 3.5 km (2.1 mi) for rail traffic, and about 6.2 km (3.9 mi) for road traffic. Any explosion on one of these routes would not generate significant missiles at the plant site (Section 2.2).

3.5.1.6 Aircraft Hazards

It has been determined that there is no significant aircraft hazard at Nine Mile Point site, as discussed in Section 2.2.3.1.7.

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

The systems and components required for a safe shutdown of the reactor and maintenance of a safe shutdown condition are identified in Section 3.5.1. It should be noted that there is no buried safety-related piping. The missiles considered in this section are turbine missiles and tornado-generated missiles. All other equipment-generated missiles have been evaluated and are considered noncredible (Section 3.5.1).

All safety-related components are protected from tornado-generated missiles by virtue of being located in structures designed to withstand tornadic events, including tornado-generated missiles. Tables 3.2-1 and 3.5-22 provide the location and listing of all safety-related structures, systems, and equipment (components), with details of protection from tornado-generated missiles.

HVAC systems intake and exhaust air openings that are required in the above-referenced structures are tornado- and missile-protected. Locations of missile-protected openings are shown on the general arrangement drawings contained in Section 1.2.

The reactor building penetrations associated with the nonsafety-related alternate decay heat (ADH) system are not missile protected. As stated in Section 9.1.6, analysis has determined that the probability of a missile strike to these penetrations is less than the acceptable limits of RG 1.117. The probability analysis was performed in accordance with the methodology developed by L. A. Twisdale in EPRI Report No. NP-2005.

<u>Missile Barriers</u>

The protective structures and barriers designed to withstand the effects of turbine-generated missiles are listed in Table 3.5-22 and are shown on Figure 3.5-1. The exterior walls and roof of the Category I structures are designed to withstand the effects of tornado-generated missiles, except the reactor building steel superstructure. These structures are listed in Table 3.5-22 and are shown on Figure 1.2-2.

Category I Electrical Ductlines and Manholes

Category I electrical ductlines are protected from tornado-generated missiles either by being buried under at least 8 ft-0 in of earth cover or by being located directly underneath plant structures which provide missile protection. Category I electrical manholes are provided with a minimum of 12-in earth cover and 2 ft-0 in thick concrete roof which prevents perforation by tornado-generated missiles. Additionally, a 2 ft-0 in thick concrete slab block is provided at the top of each Category I manhole cover to prevent impingement and perforation of the manhole cover by tornado-generated missiles.

3.5.3 Barrier Design Procedures

Missile barriers are designed to defeat the missiles described in Section 3.5.1. Defeat of the missile is achieved if the missile is stopped with no generation of secondary missiles and structural collapse of the barrier is precluded.

Local response of steel barriers is evaluated by using the Ballistic Research Laboratory Formula in Gwaltney[7]. The thickness of steel barriers to prevent perforation is obtained by multiplying 1.25 by the thickness for threshold perforation (P) as determined by the Ballistic Research Laboratory Formula.

The procedure used to evaluate the local response of concrete barriers to missile impact with no scabbing is based on Appendix B of SWECO-7703[8]. The minimum thickness of concrete barriers is 24 in, which conforms to the minimum acceptable barrier thickness requirements of Table 1 of the SRP, Section 3.5.3, except that 20-in thickness with 4,000 psi concrete is used for missile protection enclosure of valves 2SWP*MOV77A and B in the screenwell building, which is also in compliance with Table 1 of SRP Section 3.5.3. There are no openings in the missile barriers which would allow a tornado-generated missile to pass through the barrier into the building.

Unless otherwise stated in this section, the missile spectrum A of SRP 3.5.1.4 was chosen for Unit 2 design, since the values of missile impact loads derived from spectrum A are more conservative than the same from spectrum II (i.e., the missile spectrum of Table 2 of NUREG-0800, SRP 3.5.3), except for the automobile missile. In case of the automobile missile, the only difference between the two spectra is in the velocity of missile strike; i.e., the horizontal impact velocity listed in spectrum A is lower than that listed in spectrum II. Spectrum II missiles are considered in designing the missile protection shield structures from motor-operated valves (MOVs) in the screenwell at el 261'-0" and the tornado missile analysis for the diesel generator building exhaust line penetrations.

Unit 2 design is based on the methods and procedures outlined in Appendixes B and C of SWECO-7703[8]. This topical report was submitted to the NRC on September 23, 1977. This report indicates that 24-in thick concrete barriers are capable of withstanding the automobile missile of spectrum II (i.e., with higher velocity) without loss of function. (See Tables C.3-1 through C.3-6, Appendix C of SWECO-7703.) Therefore, since the minimum concrete barrier thickness used in the Unit 2 design is 24 in, the structural barriers are capable of withstanding the missiles from either spectrum A or spectrum II.

The overall structural response of concrete barriers to missile impact is evaluated using the methods presented in Appendix C of SWECO-7703. Using these methods, the structural design of the barrier is controlled by the ductility factor as described herein.

If the barrier is required to carry loads during and after missile impact, the maximum allowable ductility is limited to a factor of 10. In particular:

1. For beam-column members where the compressive load is equal to or less than one-third of that which would produce balanced conditions (i.e., P_b or 0.1 $Fc'A_g$, whichever is smaller, the allowable ductility is 10.

   Where:

   P_b = Axial load capacity at simultaneous assumed ultimate strain of concrete and yielding of tension steel

   Fc' = Specified compressive strength of concrete, psi

   A_g = Gross area of section, sq in

2. For beam-column members where the design is controlled by compression, the allowable ductility is 1.3.

3. For members that are between the cases of Items 1 and 2, the ductility ratio is taken as decreasing linearly from 10 to 1.3.

4. Where shear controls the design, the permissible ductility ratios are as follows:

   a. When shear is carried by concrete alone, the allowable ductility is ≤1.0.

b. When shear is carried by a combination of concrete and stirrups (or bent bars), the allowable ductility is ≤1.3.

The overall structural response of the steel barriers to missile impact is evaluated in accordance with the following:

1. When flexural compression or shear governs, the allowable ductility is ≤10.

2. For columns with slenderness ratio (l/γ):

   a. Equal to or less than 20, the allowable ductility is ≤1.3.

   b. Greater than 20, the allowable ductility is ≤1.0.

   Where:

   l = Effective length of the member

   γ = Least radius of gyration

3. When the members are subjected to tension, the ductility ratio (u) is given by:

$$u = 0.5 \frac{\varepsilon u}{\varepsilon y} \tag{3.5-17}$$

   Where:

   εu = Ultimate strain

   εy = Yield strain

If a concrete barrier is not required to carry other loads during and after impact, the maximum allowable ductility is limited to correspond to a rebar elongation of 5 percent. Similarly, for steel barriers not required to carry other loads, the maximum allowable ductility is also limited to correspond to an elongation of 5 percent.

3.5.4 References

1. General Electric Report, Analysis of Recirculation Pump under Accident Conditions. Submitted to the NRC (Attention: Mr. D. R. Vassallo, Assistant Director for Light Water Reactors) by GE via GE Letter No. MFN-104-79 dated March 30, 1979.

2. General Electric Memo Report. Hypothetical Turbine Missile - Probability of Occurrence. Stone & Webster Engineering Corporation, Document No. BN8104140013 F2.20, March 14, 1973.

3. General Electric Memo Report. Hypothetical Turbine Missile Data, 43-inch Last Stage Bucket Units. Stone & Webster Engineering Corporation, Document No. BN8104140013 F2.20, March 15, 1973.

4. General Electric Memo Report. Hypothetical Turbine Missile Data, 38-inch Last Stage Bucket Units. Stone and Webster Engineering Corporation, March 16, 1973.

5. Bush, S. H. A Reassessment of Turbine-Generator Failure Probability. Nuclear Safety, Vol. 19, No. 6, November-December 1978.

6. Bush, S. H. Probability of Damage to Nuclear Components Due to Turbine Failure. Nuclear Safety, Vol. 14, No. 3, May-June 1973.

7. Gwaltney, R. C. Missile Generation and Protection in Light Water-Cooled Power Reactor Plants. ORNL-NISC-22, September 1968.

8. SWECO-7703, Missile-Barrier Interaction. A Topical Report Prepared By Stone & Webster Engineering Corporation, September 1977.

TABLE 3.5-1
(Sheet 1 of 2)

SAFETY-RELATED STRUCTURES, SYSTEMS, AND COMPONENTS

System/Structure	Section
Recirculation	5.4.1
Reactor core isolation cooling	5.4.6
High-pressure core spray	6.3
Low-pressure core spray	6.3
Automatic depressurization	6.3
Residual heat removal	5.4.7, 6.3
Control rod drive	3.9B.4
Spent fuel cooling	9.1.3
Standby liquid control	9.3.5
Service water	9.2.1
Main steam isolation	5.4.5
Standby gas treatment	6.5.1
Diesel generator systems	9.5.4 thru 9.5.8
Reactor protection	7.2
Remote shutdown	7.4.1.4
Safety-related control room indications	7.5
Neutron monitoring	7.6.1.4
Radiation monitoring (main steam lines - main steam tunnel)	7.6.1.1
Reactor building (including auxiliary bays)	3.8, 6.2.3
Control building	6.4.2.1, 9.4.1

TABLE 3.5-1
(Sheet 2 of 2)

SAFETY-RELATED STRUCTURES, SYSTEMS, AND COMPONENTS

System/Structure	Section
Diesel generator building	3.8.4, 9.4.6
Screenwell pumphouse	3.8.4, 9.2.5
Intake structures	3.8.4, 9.2.5
Electrical tunnels (safety-related)	3.8.4
Standby gas treatment building	3.8.4
Railroad access lock	3.8.4
Primary containment	3.8, 6.2.1, 6.2.2, 6.2.4

TABLE 3.5-2
(Sheet 1 of 1)

SUMMARY OF FORCES AND STRESSES ACTING ON THERMOWELL WELDS IN THE VARIOUS HIGH-ENERGY SYSTEMS

System	Pipe Size (in)	System Data (operating)		Pipe Wall Thickness (in)	Thermowell Type	Acting Forces		Moment (in-lb)	Unit Shear Stress (lb/in^2)	Stress (lb/in^2)	Combined Stress (lb/in^2)	Factor of Safety
		Press (psia)	Temp (F°)			Horiz (lbs)	Vertical (lbs)					
MSS	28	964	540	1.339	Weld-in	565.6	1,743.4	2,615.9	308.32	485.51	575.13	37
MSS	26	964	540	1.266	Weld-in	565.6	1,743.4	2,615.9	318.95	516.0	606.6	35
MSS	6	964	540	0.432	Socket	514.7	2,450.3	1,608.7	2,658.2	2,885.3	3,923.1	5
MSS	2	964	540	0.344	Socket	514.7	2,450.3	1,608.7	2,658.2	2,885.3	3,923.1	5
IWS	24	1,124	425	1.219	Weld-in	565.6	2,026.2	2,615.9	430.0	675.1	800.4	27
IWS	18	1,069	425	0.937	Weld-in	565.6	1,929.1	2,615.9	409.9	676.6	791.1	27
IWS	12	1,069	425	0.688	Weld-in	565.6	1,929.1	2,615.9	468.2	863.45	982.4	22
ICS	10	1,155	560	0.594 0.718	Weld-in	565.6	2,081.0	2,615.9	533.79	956.14	1,095.0	19
ICS	8	1,293	284	0.500	Socket	514.7	3,162.3	2,380.8	3,430.7	4,002.2	5,271.3	3
CSL	12	142.7	170	0.688	Weld-in	565.6	292.1	2,615.9	70.89	863.45	866.3	24
CSL	10	142.7	170	0.594	Weld-in	565.6	292.1	2,615.9	74.92	956.14	959.0	22
CSH	12	1,050	550	0.844	Weld-in	565.5	1,895.5	2,615.9	422.4	738.15	850.4	25
CSH	10	1,050	550	0.594	Weld-in	565.5	1,895.5	2,615.9	486.2	956.14	1,072.6	19
RHR	20	1,043	533	1.031	Weld-in	565.5	1,883.1	2,615.9	399.6	675.2	784.5	27
RHR	12	1,300	536	0.844	Weld-in	565.5	2,337.3	2,615.9	520.8	738.15	903.4	23
WCS	10	1,190	545	0.844	Weld-in	565.5	2,142.9	2,615.9	477.5	738.15	879.1	24
WCS	8	1,190	545	0.906	Weld-in	565.5	2,142.9	2,615.9	462.4	696.18	835.8	25
WCS	6	1,190	545	0.562	Weld-in	565.5	2,142.9	2,615.9	560.5	991.3	1,138.7	18
WCS	4	1,150	140	0.438	Socket	514.7	2,852.9	1,608.7	3,095.0	2,885.3	4,231.3	4
WCS	3	1,170	140	0.300	Socket	514.7	2,852.9	1,608.7	3,095.0	2,885.3	4,231.3	4
WCS	2	1,150	140	0.218	Socket	514.7	2,852.9	1,608.7	3,095.0	2,885.3	4,231.3	4

TABLE 3.5-3
(Sheet 1 of 1)

DAMAGE PROBABILITY DUE TO LOW-TRAJECTORY TURBINE MISSILES FROM UNIT 2 STRIKING PLANT REGIONS AT UNIT 2

Manufacturer's Probability

Safety-Related Regions	Design Overspeed Failure			Destructive Overspeed Failure		
	p_1	p_2xp_3	$p_1xp_2xp_3$	p_1	p_2xp_3	$p_1xp_2xp_3$
Reactor building	Statistically Insignificant (Reference 4)			Statistically Insignificant		
Control building	Statistically Insignificant (Reference 4)			Statistically Insignificant		
Diesel generator building	Statistically Insignificant (Reference 4)			Statistically Insignificant		
Screenwell building service water pump room	Statistically Insignificant (Reference 4)			Statistically Insignificant		
Standby gas treatment and RR access lock	Statistically Insignificant (Reference 4)			Statistically Insignificant		
Radwaste area	Statistically Insignificant (Reference 4)			Statistically Insignificant		
Auxiliary service building and north and south auxiliary bays	Statistically Insignificant (Reference 4)			Statistically Insignificant		
Intake and discharge shaft area	Statistically Insignificant (Reference 4)			Statistically Insignificant		
Main steam tunnel	Statistically Insignificant (Reference 4)			Statistically Insignificant		

TABLE 3.5-4
(Sheet 1 of 1)

DAMAGE PROBABILITY DUE TO LOW-TRAJECTORY TURBINE MISSILES FROM UNIT 1 STRIKING PLANT REGIONS AT UNIT 2

Manufacturer's Probability

Safety-Related Regions	Design Overspeed Failure			Destructive Overspeed Failure		
	p_1	p_2xp_3	$p_1xp_2xp_3$	p_1	p_2xp_3	$p_1xp_2xp_3$
Reactor building	Statistically Insignificant (Reference 4)			1.37×10^{-8}	0	0
Control building	Statistically Insignificant (Reference 4)			1.37×10^{-8}	0	0
Diesel generator building	Statistically Insignificant (Reference 4)			1.37×10^{-8}	0	0
Screenwell building - service water pump room	Statistically Insignificant (Reference 4)			1.37×10^{-8}	0	0
Standby gas treatment and RR access lock	Statistically Insignificant (Reference 4)			1.37×10^{-8}	0	0
Radwaste area	Statistically Insignificant (Reference 4)			1.37×10^{-8}	0	0
Auxiliary service building and north and south auxiliary bays	Statistically Insignificant (Reference 4)			1.37×10^{-8}	0	0
Intake and discharge shaft area	Statistically Insignificant (Reference 4)			1.37×10^{-8}	0	0
Main steam tunnel	Statistically Insignificant (Reference 4)			1.37×10^{-8}	0	0

NOTE: Manufacturer's probability: $p_1 = 1.37 \times 10^{-8}$

TABLE 3.5-5
(Sheet 1 of 1)

DAMAGE PROBABILITY DUE TO LOW-TRAJECTORY TURBINE MISSILES FROM JAMES A. FITZPATRICK POWER STATION STRIKING PLANT REGIONS AT UNIT 2

Manufacturer's Probability

Safety-Related Region	Design Overspeed Failure			Destructive Overspeed Failure		
	p_1	p_2xp_3	$p_1xp_2xp_3$	p_1	p_2xp_3	$p_1xp_2xp_3$
Reactor building	1.37×10^{-8}	5.807×10^{-4}	7.956×10^{-12}	1.37×10^{-8}	5.446×10^{-5}	7.461×10^{-13}
Control building	1.37×10^{-8}	1.712×10^{-4}	2.345×10^{-12}	1.37×10^{-8}	1.508×10^{-5}	2.066×10^{-13}
Diesel generator building	1.37×10^{-8}	5.453×10^{-5}	7.471×10^{-13}	1.37×10^{-8}	5.099×10^{-6}	6.986×10^{-14}
Screenwell building - service water pump room	1.37×10^{-8}	3.140×10^{-6}	0	1.37×10^{-8}	0	0
Standby gas treatment and RR access lock	1.37×10^{-8}	1.553×10^{-4}	2.128×10^{-12}	1.37×10^{-8}	1.457×10^{-5}	1.996×10^{-13}
Radwaste building	1.37×10^{-8}	2.397×10^{-4}	3.288×10^{-12}	1.37×10^{-8}	2.243×10^{-5}	3.073×10^{-13}
Auxiliary service building and north and south auxiliary bays	1.37×10^{-8}	4.616×10^{-5}	6.324×10^{-13}	1.37×10^{-8}	4.319×10^{-6}	5.917×10^{-14}
Intake and discharge shaft area	1.37×10^{-8}	7.182×10^{-5}	9.839×10^{-13}	1.37×10^{-8}	6.688×10^{-6}	9.163×10^{-14}
Main steam tunnel	1.37×10^{-8}	3.830×10^{-5}	5.247×10^{-13}	1.37×10^{-8}	3.582×10^{-6}	4.907×10^{-14}

NOTE: Manufacturer's probability: $p_1 = 1.37 \times 10^{-8}$

TABLE 3.5-6
(Sheet 1 of 1)

DAMAGE PROBABILITY DUE TO LOW-TRAJECTORY TURBINE MISSILES FROM UNIT 2 STRIKING PLANT REGIONS AT UNIT 2

NRC Probability

Safety-Related Regions	Design Overspeed Failure			Destructive Overspeed Failure		
	p_1	p_2xp_3	$p_1xp_2xp_3$	p_1	p_2xp_3	$p_1xp_2xp_3$
Reactor building	Statistically Insignificant (Reference 4)			Statistically Insignificant		
Control building	Statistically Insignificant (Reference 4)			Statistically Insignificant		
Diesel generator building	Statistically Insignificant (Reference 4)			Statistically Insignificant		
Screenwell building - service	Statistically Insignificant (Reference 4)			Statistically Insignificant		
Standby gas treatment and RR access lock	Statistically Insignificant (Reference 4)			Statistically Insignificant		
Radwaste building	Statistically Insignificant (Reference 4)			Statistically Insignificant		
Auxiliary service building and north and south auxiliary bays	Statistically Insignificant (Reference 4)			Statistically Insignificant		
Intake and discharge shaft area	Statistically Insignificant (Reference 4)			Statistically Insignificant		
Main steam tunnel	Statistically Insignificant (Reference 4)			Statistically Insignificant		

TABLE 3.5-7
(Sheet 1 of 1)

DAMAGE PROBABILITY DUE TO LOW-TRAJECTORY TURBINE MISSILES FROM UNIT 1 STRIKING PLANT REGIONS AT UNIT 2

NRC Probability

Safety-Related Regions	Design Overspeed Failure			Destructive Overspeed Failure		
	p_1	$p_2 x p_3$	$p_1 x p_2 x p_3$	p_1	$p_2 x p_3$	$p_1 x p_2 x p_3$
Reactor building	Statistically Insignificant (Reference 4)			1×10^{-4}	0	0
Control building	Statistically Insignificant (Reference 4)			1×10^{-4}	0	0
Diesel generator building	Statistically Insignificant (Reference 4)			1×10^{-4}	0	0
Screenwell building - service water pump room	Statistically Insignificant (Reference 4)			1×10^{-4}	0	0
Standby gas treatment and RR access lock	Statistically Insignificant (Reference 4)			1×10^{-4}	0	0
Radwaste building	Statistically Insignificant (Reference 4)			1×10^{-4}	0	0
Auxiliary service building and north and south auxiliary bays	Statistically Insignificant (Reference 4)			1×10^{-4}	0	0
Intake and discharge shaft area	Statistically Insignificant (Reference 4)			1×10^{-4}	0	0
Main steam tunnel	Statistically Insignificant (Reference 4)			1×10^{-4}	0	0

NOTE: NRC probability: $p_1 = 1 \times 10^{-4}$

TABLE 3.5-8
(Sheet 1 of 1)

DAMAGE PROBABILITY DUE TO LOW-TRAJECTORY TURBINE MISSILES FROM JAMES A. FITZPATRICK POWER STATION STRIKING PLANT REGIONS AT UNIT 2

NRC Probability

	Design Overspeed Failure			Destructive Overspeed Failure		
Safety-Related Regions	p_1	p_2xp_3	$p_1xp_2xp_3$	p_1	p_2xp_3	$p_1xp_2xp_3$
Reactor building	1×10^{-4}	5.807×10^{-4}	5.807×10^{-8}	1×10^{-4}	5.446×10^{-5}	5.446×10^{-9}
Control building	1×10^{-4}	1.712×10^{-4}	1.712×10^{-8}	1×10^{-4}	1.508×10^{-5}	1.508×10^{-9}
Diesel generator building	1×10^{-4}	5.453×10^{-5}	5.453×10^{-9}	1×10^{-4}	5.099×10^{-6}	5.099×10^{-10}
Screenwell building - service water pump room	1×10^{-4}				1×10^{-4}	
Standby gas treatment and RR access lock	1×10^{-4}	1.553×10^{-4}	1.553×10^{-8}	1×10^{-4}	1.457×10^{-5}	1.457×10^{-9}
Radwaste building	1×10^{-4}	2.397×10^{-4}	2.397×10^{-8}	1×10^{-4}	2.243×10^{-5}	2.243×10^{-10}
Auxiliary service building and north and south auxiliary bays	1×10^{-4}	4.616×10^{-5}	4.616×10^{-9}	1×10^{-4}	4.319×10^{-6}	4.319×10^{-10}
Intake and discharge shaft area	1×10^{-4}	7.182×10^{-5}	7.182×10^{-9}	1×10^{-4}	6.688×10^{-6}	6.688×10^{-10}
Main steam tunnel	1×10^{-4}	3.830×10^{-5}	3.830×10^{-9}	1×10^{-4}	3.582×10^{-6}	3.582×10^{-10}

NOTE: NRC probability: $p_1 = 1 \times 10^{-4}$

TABLE 3.5-9
(Sheet 1 of 1)

DAMAGE PROBABILITY DUE TO HIGH-TRAJECTORY TURBINE MISSILES FROM UNIT 2 STRIKING PLANT REGIONS AT UNIT 2

Manufacturer's Probability

Safety-Related Regions	Design Overspeed Failure			Destructive Overspeed Failure		
	p_1	p_2xp_3	$p_1xp_2xp_3$	p_1	p_2xp_3	$p_1xp_2xp_3$
Reactor building	Statistically Insignificant (Reference 4)			Statistically Insignificant		
Control building	Statistically Insignificant (Reference 4)			Statistically Insignificant		
Diesel generator building	Statistically Insignificant (Reference 4)			Statistically Insignificant		
Screenwell building - service water pump room	Statistically Insignificant (Reference 4)			Statistically Insignificant		
Standby gas treatment and RR access lock	Statistically Insignificant (Reference 4)			Statistically Insignificant		
Radwaste area	Statistically Insignificant (Reference 4)			Statistically Insignificant		
Auxiliary service building and north and south auxiliary bays	Statistically Insignificant (Reference 4)			Statistically Insignificant		
Intake and discharge shaft area	Statistically Insignificant (Reference 4)			Statistically Insignificant		
Main steam tunnel	Statistically Insignificant (Reference 4)			Statistically Insignificant		

TABLE 3.5-10
(Sheet 1 of 1)

DAMAGE PROBABILITY DUE TO HIGH-TRAJECTORY TURBINE MISSILES FROM UNIT 1 STRIKING PLANT REGIONS AT UNIT 2

Manufacturer's Probability

Safety-Related Regions	Design Overspeed Failure			Destructive Overspeed Failure		
	p_1	p_2xp_3	$p_1xp_2xp_3$	p_1	p_2xp_3	$p_1xp_2xp_3$
Reactor building	Statistically Insignificant (Reference 4)			1.37×10^{-8}	1.871×10^{-6}	2.563×10^{-14}
Control building	Statistically Insignificant (Reference 4)			1.37×10^{-8}	8.321×10^{-6}	1.140×10^{-13}
Diesel generator building	Statistically Insignificant (Reference 4)			1.37×10^{-8}	3.202×10^{-6}	4.387×10^{-14}
Screenwell building - service water pump room	Statistically Insignificant (Reference 4)			1.37×10^{-8}	8.822×10^{-7}	1.209×10^{-14}
Standby gas treatment and RR access lock	Statistically Insignificant (Reference 4)			1.37×10^{-8}	2.161×10^{-6}	2.961×10^{-14}
Radwaste building	Statistically Insignificant (Reference 4)			1.37×10^{-8}	9.232×10^{-6}	1.265×10^{-13}
Auxiliary service building and north and south auxiliary bays	Statistically Insignificant (Reference 4)			1.37×10^{-8}	7.955×10^{-6}	1.090×10^{-13}
Intake and discharge shaft area	Statistically Insignificant (Reference 4)			1.37×10^{-8}	4.373×10^{-6}	5.991×10^{-14}
Main steam tunnel	Statistically Insignificant (Reference 4)			1.37×10^{-8}	6.257×10^{-7}	8.572×10^{-15}

NOTE: Manufacturer's probability: $p_1 = 1.37 \times 10^{-8}$

TABLE 3.5-11
(Sheet 1 of 1)

DAMAGE PROBABILITY DUE TO HIGH-TRAJECTORY TURBINE MISSILES FROM JAMES A. FITZPATRICK POWER STATION STRIKING PLANT REGIONS AT UNIT 2

Manufacturer's Probability

Safety-Related Region	Design Overspeed Failure			Destructive Overspeed Failure		
	p_1	$p_2 x p_3$	$p_1 x p_2 x p_3$	p_1	$p_2 x p_3$	$p_1 x p_2 x p_3$
Reactor building	1.37×10^{-8}	1.161×10^{-5}	2.206×10^{-13}	1.37×10^{-8}	4.839×10^{-6}	6.629×10^{-14}
Control building	1.37×10^{-8}	8.779×10^{-6}	1.203×10^{-13}	1.37×10^{-8}	2.641×10^{-5}	3.618×10^{-13}
Diesel generator building	1.37×10^{-8}	4.800×10^{-6}	6.576×10^{-14}	1.37×10^{-8}	2.695×10^{-5}	3.692×10^{-13}
Screenwell building - service water pump room	1.37×10^{-8}	3.760×10^{-6}	5.151×10^{-14}	1.37×10^{-8}	4.449×10^{-7}	6.095×10^{-15}
Standby gas treatment and RR access lock	1.37×10^{-8}	4.272×10^{-5}	5.853×10^{-13}	1.37×10^{-8}	9.526×10^{-6}	1.305×10^{-13}
Radwaste building	1.37×10^{-8}	9.584×10^{-6}	1.308×10^{-13}	1.37×10^{-8}	8.962×10^{-6}	1.228×10^{-13}
Auxiliary service building and north and south auxiliary bays	1.37×10^{-8}	1.449×10^{-4}	1.985×10^{-12}	1.37×10^{-8}	2.570×10^{-5}	3.523×10^{-13}
Intake and discharge shaft area	1.37×10^{-8}	3.010×10^{-6}	4.124×10^{-14}	1.37×10^{-8}	2.400×10^{-6}	3.288×10^{-14}
Main steam tunnel	1.37×10^{-8}	6.524×10^{-7}	8.938×10^{-15}	1.37×10^{-8}	1.267×10^{-6}	1.736×10^{-14}

NOTE: Manufacturer's probability: $p_1 = 1.37 \times 10^{-8}$

TABLE 3.5-12
(Sheet 1 of 1)

DAMAGE PROBABILITY DUE TO HIGH-TRAJECTORY TURBINE MISSILES FROM UNIT 2 STRIKING PLANT REGIONS AT UNIT 2

NRC Probability

Safety-Related Regions	Design Overspeed Failure			Destructive Overspeed Failure		
	p_1	$p_2 x p_3$	$p_1 x p_2 x p_3$	p_1	$p_2 x p_3$	$p_1 x p_2 x p_3$
Reactor building	Statistically Insignificant (Reference 4)			Statistically Insignificant		
Control building	Statistically Insignificant (Reference 4)			Statistically Insignificant		
Diesel generator building	Statistically Insignificant (Reference 4)			Statistically Insignificant		
Screenwell building	Statistically Insignificant (Reference 4)			Statistically Insignificant		
Standby gas treatment and RR access lock	Statistically Insignificant (Reference 4)			Statistically Insignificant		
Radwaste building	Statistically Insignificant (Reference 4)			Statistically Insignificant		
Auxiliary service building and north and south auxiliary bays	Statistically Insignificant (Reference 4)			Statistically Insignificant		
Intake and discharge shaft area	Statistically Insignificant (Reference 4)			Statistically Insignificant		
Main steam tunnel	Statistically Insignificant (Reference 4)			Statistically Insignificant		

TABLE 3.5-13
(Sheet 1 of 1)

DAMAGE PROBABILITY DUE TO HIGH-TRAJECTORY TURBINE MISSILES FROM UNIT 1 STRIKING PLANT REGIONS AT UNIT 2

NRC Probability

Safety-Related Regions	Design Overspeed Failure			Destructive Overspeed Failure		
	p_1	$p_2 x p_3$	$p_1 x p_2 x p_3$	p_1	$p_2 x p_3$	$p_1 x p_2 x p_3$
Reactor building	Statistically Insignificant (Reference 4)			1×10^{-4}	1.871×10^{-6}	1.871×10^{-10}
Control building	Statistically Insignificant (Reference 4)			1×10^{-4}	8.321×10^{-6}	8.321×10^{-10}
Diesel generator building	Statistically Insignificant (Reference 4)			1×10^{-4}	3.202×10^{-6}	3.202×10^{-10}
Screenwell building - service water pump room	Statistically Insignificant (Reference 4)			1×10^{-4}	8.822×10^{-7}	8.822×10^{-11}
Standby gas treatment and RR access lock	Statistically Insignificant (Reference 4)			1×10^{-4}	2.161×10^{-6}	2.161×10^{-10}
Radwaste building	Statistically Insignificant (Reference 4)			1×10^{-4}	9.232×10^{-6}	9.232×10^{-10}
Auxiliary service building and north and south auxiliary bays	Statistically Insignificant (Reference 4)			1×10^{-4}	7.955×10^{-6}	7.955×10^{-10}
Intake and discharge shaft area	Statistically Insignificant (Reference 4)			1×10^{-4}	4.373×10^{-6}	4.373×10^{-10}
Main steam tunnel	Statistically Insignificant (Reference 4)			1×10^{-4}	6.257×10^{-7}	6.257×10^{-11}

NOTE: NRC probability: $p_1 = 1 \times 10^{-4}$

TABLE 3.5-14
(Sheet 1 of 1)

DAMAGE PROBABILITY DUE TO HIGH-TRAJECTORY TURBINE MISSILES FROM JAMES A. FITZPATRICK POWER STATION STRIKING PLANT REGIONS AT UNIT 2

NRC Probability

Safety-Related Regions	Design Overspeed Failure			Destructive Overspeed Failure		
	p_1	$p_2 x p_3$	$p_1 x p_2 x p_3$	p_1	$p_2 x p_3$	$p_1 x p_2 x p_3$
Reactor building	1×10^{-4}	1.161×10^{-5}	1.161×10^{-9}	1×10^{-4}	4.839×10^{-6}	4.839×10^{-10}
Control building	1×10^{-4}	8.779×10^{-6}	8.779×10^{-10}	1×10^{-4}	2.641×10^{-5}	2.641×10^{-9}
Diesel generator building	1×10^{-4}	4.800×10^{-6}	4.800×10^{-10}	1×10^{-4}	2.695×10^{-5}	2.695×10^{-9}
Screenwell building	1×10^{-4}	3.760×10^{-6}	3.760×10^{-10}	1×10^{-4}	4.449×10^{-7}	4.449×10^{-11}
Standby gas treatment and RR access lock	1×10^{-4}	4.272×10^{-5}	4.272×10^{-9}	1×10^{-4}	9.526×10^{-6}	9.526×10^{-10}
Radwaste building	1×10^{-4}	9.584×10^{-6}	9.584×10^{-10}	1×10^{-4}	8.962×10^{-6}	8.962×10^{-10}
Auxiliary service building and north and south auxiliary bays	1×10^{-4}	1.449×10^{-4}	1.449×10^{-8}	1×10^{-4}	2.570×10^{-5}	2.570×10^{-9}
Intake and discharge shaft area	1×10^{-4}	3.010×10^{-6}	3.010×10^{-10}	1×10^{-4}	2.400×10^{-6}	2.400×10^{-10}
Main steam tunnel	1×10^{-4}	6.524×10^{-7}	6.524×10^{-11}	1×10^{-4}	1.267×10^{-6}	1.267×10^{-10}

NOTE: NRC probability: $p_1 = 1 \times 10^{-4}$

TABLE 3.5-15
(Sheet 1 of 1)

SUM OF DAMAGE PROBABILITY DUE TO LOW- AND HIGH-TRAJECTORY TURBINE MISSILES GENERATED FROM TURBINES AT UNITS 1 AND 2 AND JAMES A. FITZPATRICK TO PLANT REGIONS OF UNIT 2

Manufacturer's Probability

Trajectory and Turbine	Design Overspeed Failure			Destructive Overspeed Failure		
	p_1	$p_2 x p_3$	$p_1 x p_2 x p_3$	p_1	$p_2 x p_3$	$p_1 x p_2 x p_3$
Low trajectory from Unit 2	Statistically Insignificant (Reference 4)			Statistically Insignificant		
Low trajectory from Unit 1	Statistically Insignificant (Reference 4)			1.37×10^{-8}	0	0
Low trajectory from FitzPatrick	1.37×10^{-8}	5.373×10^{-4}	7.361×10^{-12}	1.37×10^{-8}	4.930×10^{-5}	6.754×10^{-13}
High trajectory from Unit 2	Statistically Insignificant (Reference 4)			Statistically Insignificant		
High trajectory from Unit 1	Statistically Insignificant (Reference 4)			1.37×10^{-8}	3.862×10^{-5}	5.291×10^{-13}
High trajectory from FitzPatrick	1.37×10^{-8}	2.299×10^{-4}	3.150×10^{-12}	1.37×10^{-8}	1.064×10^{-4}	1.459×10^{-12}
Total	1.37×10^{-8}	7.672×10^{-4}	1.051×10^{-11}	1.37×10^{-8}	1.943×10^{-4}	2.66×10^{-12}

The total p_2p_3 for design and destructive overspeed failure is 0.962×10^{-3}.
The total $p_1p_2p_3$ for design and destructive overspeed failure is 1.078×10^{-11}.

NOTE: Manufacturer's probability for Unit 1 and Fitzpatrick: $p_1 = 1.37 \times 10^{-8}$

TABLE 3.5-16
(Sheet 1 of 1)

SUM OF DAMAGE PROBABILITY DUE TO LOW- AND HIGH-TRAJECTORY TURBINE MISSILES GENERATED FROM TURBINES AT UNITS 1 AND 2 AND JAMES A. FITZPATRICK TO PLANT REGIONS OF UNIT 2

NRC Probability

Trajectory and Turbine	Design Overspeed Failure			Destructive Overspeed Failure		
	p_1	p_2xp_3	$p_1xp_2xp_3$	p_1	p_2xp_3	$p_1xp_2xp_3$
Low trajectory from Unit 2	Statistically Insignificant (Reference 4)			Statistically Insignificant		
Low trajectory from Unit 1	Statistically Insignificant (Reference 4)			1×10^{-4}	0	0
Low trajectory from FitzPatrick	1×10^{-4}	5.373×10^{-4}	5.373×10^{-8}	1×10^{-4}	4.930×10^{-5}	4.930×10^{-9}
High trajectory from Unit 2	Statistically Insignificant (Reference 4)			Statistically Insignificant		
High trajectory from Unit 1	Statistically Insignificant (Reference 4)			1×10^{-4}	3.862×10^{-5}	3.862×10^{-9}
High trajectory from FitzPatrick	1×10^{-4}	2.299×10^{-4}	2.299×10^{-8}	1×10^{-4}	1.064×10^{-4}	1.064×10^{-8}
Total	1×10^{-4}	7.672×10^{-4}	7.672×10^{-8}	1×10^{-4}	1.943×10^{-4}	1.943×10^{-8}

The total p_2p_3 for design and destructive overspeed failure is 0.962×10^{-3}.
The total $p_1p_2p_3$ for design and destructive overspeed failure is 0.962×10^{-7}.

NOTE: NRC probability (required for Unit 1 and FitzPatrick only): $p_1 = 1 \times 10^{-4}$.

TABLE 3.5-17
(Sheet 1 of 3)

TURBINE MISSILE INFORMATION

43-Inch Last Stage Bucket Unit

Stage Group	I(1)				II(1)				III(2)			
Stage numbers in group:	1 - 3				4 - 6				7 (Last)			
Number of representative stage	2				5				7			
Missile Dimensions	a	b	c	d	a	b	c	d	a	b	c	d
Fragment group												
Number of fragments in group	2	1	3	10	2	1	3	10	2	1	3	10
Sector angle, deg	120	60			120	60			120	60		
Fragment weight, lb	2,000	1,000	300	100	4,000	2,000	600	150	8,200	4,100	1,400	200
Radius, in*												
R_1 Bore	20	20	NA	NA	18	18	NA	NA	17	17	NA	NA
R_2 Hub	27	27			27	27			28	28		
R_3 Vane root	48	48			47	47			45	45		
Thickness, in*												
T_1 Hub	9	9			12	12			27	27		
T_2 Web	3	3			5	5			12	12		
Approximate rectangular dimensions, in*			19x19x3	11x11x3			20x20x5	10x10x5			20x20x14	8x8x12

* See Figure 3.5-6.

TABLE 3.5-17
(Sheet 2 of 3)

HYPOTHETICAL TURBINE MISSILE INFORMATION

43-In Last Stage Bucket, 1,800 rpm Low-Pressure Turbine

Low Speed Burst
Postulated speed: 2,160 rpm (120%)
Lifetime probability: 2.6 x 10^{-7}

	Stage Group I	Stage Group II	Stage Group III(3)				
Conditional probability of occurrence in stage group	Not statistically significant	Not statistically significant	1				
Probability of occurrence in stage group			2.6 x 10^{-7}				
Fragment group	Not statistically significant	Not statistically significant	a			b	
			Energy	Velocity		Energy	Velocity
Minimum			10	280	0	0	
Maximum			22	420	18	530	
Midpoint			16	350	9	380	
Fragment group	Not statistically significant	Not statistically significant	c			d	
			Energy	Velocity		Energy	Velocity
Minimum			0	0	0	0	
Maximum			8	610	2	800	
Midpoint			4	430	1	560	

TABLE 3.5-17
(Sheet 3 of 3)

43-Inch Last Stage Bucket

High Speed Burst
Postulated speed: 3,240 rpm (180%)
Lifetime probability: 1.5×10^{-7}

	Stage Group I[3]				Stage Group II[3]				Stage Group III[4]			
Conditional probability of occurrence in stage group	3/7				3/7				1/7			
Probability of occurrence in stage group	6.4×10^{-8}				6.4×10^{-8}				2.1×10^{-8}			
Fragment group[5]	a		b		a		b		a		b	
	Energy	Velocity	Energy	Velocity	Energy	Velocity	Energy	Velocity	Energy	Velocity	Energy	Velocity
Minimum	0	0	0	0	0	0	0	0	26	450	0	0
Maximum	8	510	8	720	17	520	16	720	53	650	38	770
Midpoint	4	360	4	510	8.5	370	8	510	39.5	560	19	550
Fragment group[5]	c		d		c		d		c		d	
	Energy	Velocity	Energy	Velocity	Energy	Velocity	Energy	Velocity	Energy	Velocity	Energy	Velocity
Minimum	0	0	0	0	0	0	0	0	0	0	0	0
Maximum	5	1,040	2	1,130	8	930	2	930	16	860	3	980
Midpoint	2.5	730	1	800	4	660	1	660	8	610	1.5	690

(1) For interior disc, δ_1 and δ_2 are 5 deg, respectively.
(2) For end disc, δ_1 and δ_2 are 25 deg and 0 deg or 0 deg and 25 deg, respectively.
(3) The deflection angles δ_1 and δ_2 are 5 deg for inner stage buckets.
(4) For last stage buckets, the deflection angles are 25 deg and 0 deg or 0 deg and 25 deg, respectively.
(5) Missiles in four size classes, a, b, c, and d, are postulated to occur per burst.

NOTES:

a. Energy of ejected missiles is given in million ft lb; velocity in fps.

b. Energies are postulated to be uniformly distributed over stated ranges.

SOURCE: General Electric Memo Report. Hypothetical Turbine Missile Data, 43-inch Last Stage Bucket Units. Stone & Webster Engineering Corporation, Document No. CD7912100015, March 15, 1973.

TABLE 3.5-18
(Sheet 1 of 3)

TURBINE MISSILE INFORMATION

38-Inch Last Stage Bucket Units

Stage Group	I[1]				II[1]				III[2]			
Stage numbers in group:	1 - 3				4 - 6				7 (Last)			
Number of representative stage	2				5				7			
Missile Dimensions	a	b	c	d	a	b	c	d	a	b	c	d
Fragment group												
Number of fragments in group	2	1	3	10	2	1	3	10	2	1	3	10
Sector angle, deg	120	60			120	60			120	60		
Fragment weight, lb	2,000	1,000	300	100	3,000	1,500	500	150	6,500	3,200	1,000	200
Radius, in*												
R_1 Bore	18	18			17	17			16	16		
R_2 Hub	24	24			25	25			25	25		
R_3 Vane root	45	45			45	45			45	45		
Thickness, in*												
T_1 Hub	10	10			12	12			21	21		
T_2 Web	3	3			5	5			10	10		
Approximate rectangular dimensions, in*	NA	NA	19x19x3	11x11x3	NA	NA	17x19x5	10x10x5	NA	NA	19x19x10	8x8x10

* See Figure 3.5-6.

TABLE 3.5-18
(Sheet 2 of 3)

HYPOTHETICAL TURBINE MISSILE INFORMATION

38-Inch Last Stage Buckets, 1,800 rpm Low Pressure Turbine

Low Speed Burst
Postulated speed: 2,160 rpm (120%)
Lifetime probability: not statistically significant

	Stage Group I	Stage Group II	Stage Group III[1]
Conditional probability of occurrence in stage group	Not statistically significant	Not statistically significant	Not statistically significant
Probability of occurrence in stage group			
Fragment group Minimum Maximum Midpoint	Not statistically significant	Not statistically significant	Not statistically significant
Fragment group Minimum Maximum Midpoint	Not statistically significant	Not statistically significant	Not statistically significant

TABLE 3.5-18
(Sheet 3 of 3)

38-Inch Last Stage Buckets, 1,800 rpm Low Pressure Turbine

High Speed Burst
Postulated speed: 3,240 rpm (180%)
Lifetime probability: 1.5×10^{-7}

	Stage Group I[3]				Stage Group II[3]				Stage Group III[4]			
Conditional probability of occurrence in stage group	3/7				3/7				1/7			
Probability of occurrence in stage group	6.4×10^{-8}				6.4×10^{-8}				2.1×10^{-8}			
Fragment group[5]	a		b		a		b		a		b	
	Energy	Velocity	Energy	Velocity	Energy	Velocity	Energy	Velocity	Energy	Velocity	Energy	Velocity
Minimum	0	0	0	0	0	0		0	16	400	0	0
Maximum	7	470	6	620	14	550	13	750	38	610	30	780
Midpoint	3.5	340	3	440	7	390	6.5	530	27	520	15	550
Fragment group[5]	c		d		c		d		c		d	
	Energy	Velocity	Energy	Velocity	Energy	Velocity	Energy	Velocity	Energy	Velocity	Energy	Velocity
Minimum	0	0	0	0	0	0	0	0	0	0	0	0
Maximum	4	930	2	1,130	6	880	2	930	13	910	3	980
Midpoint	2	660	1	800	3	620	1	660	6.5	650	1.5	690

(1) For interior disc, δ_1 and δ_2 are 5 deg, respectively.
(2) For end disc, δ_1, and δ_2 are 25 deg and 0 deg or 0 deg and 25 deg, respectively.
(3) The deflection angles δ_1 and δ_2 are 5 deg for inner stage buckets.
(4) For last stage buckets, the deflection angles are 25 deg and 0 deg or 0 deg and 25 deg, respectively.
(5) Missiles in four size classes are postulated to occur per burst.

NOTES:

a. Energy of ejected missiles is given in million ft lb; velocity in fps.

b. Energies are postulated to be uniformly distributed over stated ranges.

SOURCE: General Electric Memo Report. Hypothetical Turbine Missile Data, 38-inch Last Stage Bucket Units. Stone & Webster Engineering Corporation, March 16, 1973.

TABLE 3.5-19
(Sheet 1 of 1)

BASIC ASSUMPTIONS FOR PROBABILITY ANALYSIS OF TURBINE MISSILE STRIKE

1.	Air resistance is neglected.
2.	The probability density function is constant over the range: $$o \leq \rho \leq 2\pi \text{ also } \frac{\pi}{2} - \delta_1 \leq \Theta \leq \frac{\pi}{2} - \delta_2$$ The probability density function is zero for all other values of θ.
3.	Only one disc fractures during an incident.
4.	A disc break generates 16 fragments.

TABLE 3.5-20

THIS TABLE HAS BEEN DELETED

TABLE 3.5-21
(Sheet 1 of 1)

SELECTED EXTERNAL MISSILES[1]

Missile[2]	Weight (lb)	Horizontal Impact Velocity (mph)
Wood plank, 4" x 12" x 12'[3]	200	288
Steel pipe, 3-in diameter, Schedule 40, 10 ft long[3]	78	144
Steel rod, 1-in diameter x 3 ft long[3]	8	216
Steel pipe, 6-in diameter, Schedule 40, 15 ft long[3]	285	144
Steel pipe, 12-in diameter, Schedule 40, 15 ft long[3]	743	144
Utility pole, 13 1/2-in diameter, 35 ft long[4]	1,490	144
Automobile, frontal area 20 sq ft[4]	4,000	72

(1) This table is extracted from the missile spectrum A of SRP 3.5.1.4 Revision 2 - July 1981; alternatively, as otherwise noted in Section 3.5.3, Spectrum II missiles may be considered for analysis.

(2) All missiles are considered to be capable of striking in all directions, with vertical velocities equal to 80% of the horizontal impact velocities.

(3) These missiles are to be considered at all elevations.

(4) These missiles are to be considered at elevations up to 30 ft above all grade levels within 1/2 mi of the facility structures.

TABLE 3.5-22
(Sheet 1 of 2)

MISSILE BARRIERS FOR NATURAL PHENOMENA AND
TURBINE-GENERATED MISSILES

Protected Components	Missile Barrier
RCPB, ECCS, CRD and other safety-related equipment inside containment	Exterior reactor building wall, primary containment structure, internal structures
Main control room and related electrical, instrumentation, control, and ventilation equipment in control building	Control building
Spent fuel pool	Reactor building wall below el 353' and 353' slab
Emergency diesel generators	Diesel generator building*
Diesel generator support System	Diesel generator building*
Service water pumps and piping	Screenwell service water pump room
Service water pump bay unit coolers	Service water pump bay - screenwell building
Standby gas treatment system	Standby gas treatment building
ECCS, MCCs, and other safety-related equipment	North and south auxiliary bay roof
HVAC, SWP valves, and related equipment	Auxiliary service building slab at el 261'

* The nonsafety-related exhaust line penetrations through the roof of the diesel generator building have been determined by a probability analysis as not requiring tornado missile protection. The analysis indicated that the evaluated damage probability to the generators is less than the criteria of RG 1.117 for no protection.

TABLE 3.5-22
(Sheet 2 of 2)

MISSILE BARRIERS FOR NATURAL PHENOMENA AND
TURBINE-GENERATED MISSILES

NOTES:

1. Wherever any structure is referenced as a barrier, the exterior walls and roof of that structure are designed to act as missile barriers. Each ventilation opening through missile barrier exterior walls or roofs is provided with a missile barrier hood.

2. Entrances to safety-related areas either are designed to withstand tornado-generated missiles or are provided with a labyrinth to prevent missile impingement on safety-related systems and components. The labyrinth is designed to withstand tornado-generated missiles.