CHAPTER 3 - DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

3.1 CONFORMANCE WITH NRC GENERAL DESIGN CRITERIA

This section discusses the extent to which the design criteria for the plant structures, systems, and components important to safety meet the General Design Criteria for Nuclear Power Plants specified in 10CFR50, Appendix A. For each criterion, a summary is provided to show how the principal design features meet the criterion. The discussion of each criterion also gives the section of the UFSAR where more detailed information is presented to demonstrate compliance with the criterion.

GDC 1 - Quality Standards and Records

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

Design Evaluation

Structures, systems, and components important to safety are designed, fabricated, erected, tested, and operated under a QA program that satisfies the requirements of 10CFR50, Appendix B. Chapter 17.2 of the UFSAR discusses the QA program during operation, which is designed and implemented to ensure that LGS is tested and operated in conformance with the regulatory requirements and design bases outlined in the license application.

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

Structures, systems, and components are classified with regard to location, service, and relationship to the safety function to be performed. Recognized codes and standards are applied to the equipment in keeping with the appropriate classification. Where codes are not available or where the existing code must be modified, justification is provided in the UFSAR.

Documents and records are available to show objective evidence that the requirements of the QA program have been satisfied. The documentation shows that the required codes, standards, and specifications were observed; specified materials were used; correct procedures were used; qualified personnel performed the work; and inspections and tests verified that finished parts and components meet the applicable specifications. Appropriate records are maintained during the operational life of the plant.

The QA program developed by the licensee and it's contractors satisfies the requirements of GDC 1.

GDC 2 - Design Bases for Protection Against Natural Phenomena

> Structures, systems, and components important to safety shall be designed to withstand the effect 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 Evaluation

The design basis for protection against natural phenomena is in accordance with GDC 2. Structures, systems, and components important to safety are designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, and floods without loss of the capability to perform required safety functions, with appropriate margin to account for uncertainties in the historical data. The natural phenomena postulated in the design are presented in Sections 2.3, 2.4, and 2.5. The design criteria for the structures, systems, and components affected by each natural phenomenon are presented in Sections 3.2, 3.3, 3.4, 3.5, 3.7, and 3.8. Those combinations of natural phenomena and plant originated accidents that are considered in the design are identified in Sections 3.8, 3.9, 3.10, and 3.11.

GDC 3 - Fire Protection

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

Design Evaluation

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

Appropriate equipment and facilities for fire protection, including the detection, alarm, and extinguishing of fires, are provided to protect plant equipment and personnel from fire, explosions, and the resultant release of toxic vapors. Automatic and manual types of fire protection equipment are provided.

The fire protection system provides an adequate supply of water to the deluge systems, sprinkler systems, and hose stations located throughout the plant. Two separate Halon extinguishing systems are provided for the raised flooring in the auxiliary equipment room. Portable fire extinguishers are provided throughout the plant. A detailed description of the fire protection system and its design bases is provided in Section 9.5.1.

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

The fire protection system is designed, fabricated, and installed in accordance with the requirements of the NFPA, ANI, OSHA, and applicable local codes and regulations as listed in Section 9.5.1.

The fire protection system was inspected and functionally tested prior to plant operation in order to ensure its proper operation. The fire suppression systems are provided with test valves and facilities for periodic testing. All equipment is accessible for periodic inspection.

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

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

<u>GDC 4 - Environmental and Missile Design Bases</u>

> 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.

<u>Design Evaluation</u>

Structures, systems, and components important to safety are designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including the design basis LOCA. These structures, systems, and components are appropriately protected against dynamic effects and discharging fluids that may result from equipment failures. Normal and postulated accident effects and load combinations are given in Sections 3.6, 3.8, 3.9, 3.10, and 3.11.

Special attention has been directed to the effects of pipe movement, jet forces, and missiles within the primary containment. Pipe whip restraints have been provided to the extent practicable (Section 3.6). Primary containment integrity protection is discussed in Section 6.2.1. The structures, systems, and components important to safety are protected from dynamic effects by separating redundant counterparts so that no single event can prevent a required safety action, and by routing and locating these components, to the extent practicable, to avoid potentially hazardous areas. The means used to preserve the independence of redundant counterparts of safety-related systems are discussed in Chapter 6.

Dynamic effects external to the plant, induced by natural phenomena (e.g., tornado-produced missiles), are discussed in Section 3.5. Section 3.11 contains a discussion of design environmental conditions.

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

<u>GDC 5 - Sharing of Structures, Systems, and Components</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.

<u>Design Evaluation</u>

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

The following safety-related structures are shared between both units:

a. Control enclosure and support subsystems

b. Spray pond pumphouse and support subsystems

c. Spray pond

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

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

The below listed safety systems and subsystems are shared by both units. Refer to the section listed by each system or subsystem for discussion of design criteria for instrumentation. The instrumentation for these systems is available on common panels in the control room and therefore is available to the operators of both units.

SHARED SAFETY SYSTEMS	SECTION
ESW system	7.3.2.11
RHRSW system	7.3.2.12
RHRSW-RMS	7.6.1.1
Control structure support systems	
Habitability, control room isolation subsystem	7.3.2.10
Emergency switchgear and battery room cooling subsystem	7.3.2.15
CECWS	7.3.2.13
AERVS	7.3.2.15
CRV-RMS	7.6.1.1
CREFA-RMS	7.6.1.1
SGTS	7.3.2.7
SGTS-UC	7.3.2.15
RAVE-RMS and REVE-RMS	7.6.1.1
North stack radiation monitoring system	7.6.1.1
Spray pond pumphouse support system	7.3.2.15

The shared systems that are important to safety are discussed below. A more detailed discussion may be found in these referenced sections.

Emergency Service Water System

The ESW system is designed to supply cooling water to safety- related components, including the diesel generators, room coolers and chillers, and the RHR pumps during LOOP and accident conditions. Certain nonessential components can be cooled by the ESW system also, at the operator's option.

The ESW pumps are located in the spray pond pumphouse with the RHRSW pumps. The spray pond pumphouse is designed as seismic Category I. The ESW system consists of two redundant loops, each capable of simultaneously providing 100% of the cooling water required by both Units 1 and 2. The system is designed so that no single active or passive electrical or control component failure or active mechanical component failure can prevent it from achieving its safety-related objective.

For additional discussion, see Section 9.2.2.

RHR Service Water System

The RHRSW system is designed to supply cooling water to the RHR heat exchangers during normal shutdown cooling operations as well as during LOOP and accident conditions.

The RHRSW pumps are located in the spray pond pumphouse with the ESW pumps. The spray pond pumphouse is designed as seismic Category I. The RHRSW system consists of two redundant loops, each supplying one RHR heat exchanger in each unit and capable of simultaneously providing 100% of the cooling water required by both Units 1 and 2. The system is designed so that no single active or passive component failure can prevent it from achieving its safety-related objective.

For additional discussion, see Section 9.2.3.

Ultimate Heat Sink (Spray Pond)

The spray pond provides the water for both the ESW and the RHRSW systems. It is the UHS for both Units 1 and 2. The return lines from the ESW and the RHRSW system are combined, and the total quantity of water from both these systems is discharged through spray networks, which dissipate the heat. There are two redundant return loops. Either one is capable of handling the full flow from the ESW and RHRSW systems when shutting down two units simultaneously.

Each return loop supplies two spray networks. Two of the four networks provide sufficient cooling for the design basis conditions.

The spray pond contains sufficient water to meet the requirements for shutting down one unit if there is an accident and to permit the safe shutdown of the second unit for a period of 30 days without makeup.

For additional discussion, see Section 9.2.6.

Standby Gas Treatment System

The SGTS is designed to maintain both reactor enclosures and refueling area at the required negative pressure when any of these areas are isolated.

The SGTS filter train and fans are located in the control enclosure. The control enclosure is a seismic Category I structure. The SGTS consists of two 100% capacity redundant filter trains and two 100% capacity fans. The system is designed so that no single failure can prevent it from achieving its safety-related objective.

Additional discussion is given in Section 6.5.1.1.

Control Room and Control Structure HVAC Systems

The control room and control structure HVAC systems are designed to maintain the space temperature and pressure in the various areas within the common control structure at their design conditions.

Each system consists of two 100% capacity redundant HVAC units and the system is designed so that a single failure will not prevent the system from achieving its safety-related objective.

Additional discussion is given in Section 9.4.1.

Control Structure Chilled Water System

The CSCWS provides chilled water to maintain stipulated ambient air temperature in various areas inside the common control structure.

The system consists of two 100% capacity redundant chillers and pumps and is designed so that a single failure will not prevent the system from achieving its safety-related objective.

Additional discussion is given in Section 9.2.10.2.

GDC 10 - Reactor Design

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

Design Evaluation

The reactor core components consist of fuel assemblies, control rods, incore ion chambers, neutron sources, and related items. The mechanical design is based on a conservative application of stress limits, operating experience, and experimental test results.

The core has sufficient heat transfer area and coolant flow to ensure that there is no fuel damage under normal conditions or anticipated operational occurrences. The RPS is designed to monitor certain reactor parameters, sense abnormalities, and shut down the reactor, thereby preventing fuel damage when trip setpoints are exceeded. Trip setpoints are selected according to operating experience and the design bases. 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 provided by dc-ac static inverters. Alternate electrical power is available to the RPS buses. The RPS is fail-safe, i.e., scram is initiated on loss of power.

An analysis and evaluation have been made of the effects on core fuel following adverse plant operating conditions. The results of abnormal operational transients are presented in Chapter 15 and show that the MCPR does not fall below the specified limit, thereby satisfying the transient design basis. The conditions assumed in the analysis and the control systems used to accommodate these transients are identified in Chapter 15.

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 GDC 10.

Referenced sections are as follows:

a.	Fuel system design	Section 4.2
b.	Nuclear design	Section 4.3
c.	Thermal and hydraulic design	Section 4.4
d.	Component and subsystem design	Section 5.4
e.	RPS	Section 7.2
f.	All other instrumentation systems required for safety	Section 7.6
g.	Control systems not required for safety	Section 7.7
h.	Accident analyses	Chapter 15

GDC 11 - Reactor Inherent Protection

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

Design Evaluation

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

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

a. Fuel temperature or Doppler coefficient

b. Moderator void coefficient

c. Moderator temperature coefficient

The combined effect of these coefficients in the power range is termed the power coefficient.

Doppler reactivity feedback occurs simultaneously with a change in fuel temperature and opposes the power change that caused it. Thus, 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 that permits the use of coolant flow rate for load-following.

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

a. Use of coolant flow as opposed to control rods for load-following

b. Inherent self-flattening of the radial power distribution

c. Ease of control

d. 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.

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 GDC 11. Referenced sections are as follows:

a. Nuclear design Section 4.3

b. Thermal and hydraulic design Section 4.4

GDC 12 - Suppression of Reactor Power Oscillations

> The reactor core and associated coolant, control, and protection systems shall be designed to ensure 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 Evaluation

The reactor core is designed to ensure that no power oscillation can cause 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. Operating experience has shown large BWRs to be inherently stable against xenon-induced power instability.

The RPS design and the recirculation pump trip system provide protection from excessive fuel cladding temperatures and protect the nuclear system process barrier 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 and recirculation pump trip. High reliability of the RPS 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 the fuel design limits. These systems ensure that GDC 12 is met. Referenced sections are as follows:

a.	Fuel system design	Section 4.2
b.	Nuclear design	Section 4.3
c.	Thermal and hydraulic design	Section 4.4
d.	Integrity of RCPB	Section 5.2
e.	RPS	Section 7.2
f.	All other instrumentation systems required for safety	Section 7.6
g.	Accident analyses	Chapter 15

GDC 13 - Instrumentation and Control

Instrumentation shall be provided to monitor variables and systems over their anticipated ranges for normal operations, 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 Evaluation

Instrumentation is provided to monitor variables and systems over their anticipated ranges for normal operations, for anticipated operational occurrences, and for accident conditions as appropriate to assure adequate safety. Appropriate controls are provided to maintain these variables and systems within prescribed operating ranges. A summary description for each instrumentation and control system is provided in Section 7.1. The instrumentation and controls provided meet the requirements of GDC 13. Referenced sections are as follows:

a.	Reactor Trip System (Reactor Protection System) – Instrumentation And Controls	Section 7.2
b.	ESF Systems	Section 7.3
c.	Systems Required For Safe Shutdown	Section 7.4
d.	Information Systems Important To Safety	Section 7.5
e.	All Other Instrumentation Systems Important To Safety	Section 7.6
f.	Control Systems Not Required For Safety	Section 7.7

GDC 14 - Reactor Coolant Pressure Boundary

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

Design Evaluation

The piping and equipment pressure parts within the RCPB through the outer isolation valves are designed, fabricated, erected, and tested to provide a high degree of integrity throughout the plant lifetime. Section 3.2 classifies the systems and components within the RCPB. The design requirements and codes and standards applied to the quality group ensure a quality product in keeping with the safety functions to be performed.

To minimize the possibility of brittle fracture within the RCPB, the fracture or notch properties and the operating temperature of ferritic materials are controlled to ensure adequate toughness when the system is pressurized to more than 20% of the design pressure. Section 5.2 describes the methods used to control toughness properties. Materials to be impact-tested are tested by the Charpy

V-notch method in accordance with ASME B&PV Code, Section III. The service temperature of these materials is maintained above the NDTT. The fracture toughness temperature requirements of the RCPB materials also apply for the RCPB piping that penetrates the containment.

Piping and equipment pressure parts of the RCPB are assembled and erected by welding unless applicable codes permit flanged or screwed joints. Assembly is according to ANSI B31.7 and ASME Section III. Erection is according to ASME Section III. Welding procedures are employed that produce welds of complete penetration, complete fusion, and freedom from unacceptable defects. All welding procedures, welders, and welding machine operators are qualified in accordance with the requirements of ASME B&PV Code, Section IX for the materials to be welded.

Section 5.2 contains the detailed material and examination requirements for the piping and equipment of the RCPB before and after its assembly and erection. Leakage testing and surveillance is accomplished as described in the evaluation against GDC 30.

The design, fabrication, erection, and testing of the RCPB ensure an extremely low probability of failure or abnormal leakage, thus satisfying the requirements of GDC 14. Referenced sections are as follows:

a.	Design of structures, components, equipment, and systems	Chapter 3
b.	Integrity of RCPB	Section 5.2
c.	Reactor vessel and appurtenances	Section 5.3
d.	Component and subsystem design	Section 5.4
e.	Accident analyses	Chapter 15
f.	QA	Chapter 17

GDC 15 - Reactor Coolant System Design

> The RCS 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 Evaluation

The RCS consists of the reactor vessel and appurtenances, the reactor recirculation system, the pressure relief system, and the main steam and feedwater lines. This system is designed, fabricated, erected, and tested to stringent requirements and appropriate codes and standards that ensure high integrity of the RCPB throughout the plant lifetime. The RCS is designed and fabricated to meet the requirements of the ASME B&PV Code, Section III.

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

An example of the integrated protective action scheme that provides sufficient margin to ensure that the design conditions of the RCPB are not exceeded is the automatic initiation of the pressure relief system on receipt of an overpressure signal. To accomplish overpressure protection, a number of pressure-operated relief valves are provided that can discharge steam from the nuclear system to the pressure-suppression pool. The pressure relief system also provides for automatic depressurization of the nuclear system if there is a LOCA in which the vessel is not depressurized by the accident. The depressurization of the nuclear system in this situation allows operation of the low pressure ECCS to supply enough cooling water to adequately cool the core. In a similar manner, other auxiliary, control, and protection systems provide assurance that the design conditions of the RCPB are not exceeded during any conditions of normal operation, including anticipated operational occurrences.

The application of appropriate codes, standards, and high quality requirements to the RCS and the design features of its associated auxiliary, control, and protection systems ensure that the requirements of GDC 15 are satisfied.
Referenced sections are as follows:

a.	Integrity of RCPB	Section 5.2
b.	Reactor vessel and appurtenances	Section 5.3
c.	Component and subsystem design	Section 5.4
d.	ECCS	Section 6.3
e.	ESF systems	Section 7.3
f.	Control systems not required for safety	Section 7.7
g.	Accident analyses	Chapter 15

GDC 16 - Containment Design

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.

Design Evaluation

The primary containment system, which includes the drywell and suppression chamber, is designed, fabricated, and erected to accommodate, without failure, the pressures and temperatures resulting from the double-ended rupture or equivalent failure of any coolant pipe within the primary containment. The reactor enclosure encompassing the primary containment provides secondary containment. To provide protection against the consequences of accidents involving the release of radioactive materials from the fuel and nuclear system process barrier, the PCRVICS initiates automatic isolation of appropriate pipelines that penetrate the primary containment whenever monitored variables exceed preselected operational limits. The two containment systems and their associated safety systems are designed and maintained so that offsite doses, which could result from postulated design basis accidents, remain below the values stated in 10CFR50.67 when calculated by the methods of Regulatory Guide 1.183. The referenced sections provide detailed information that demonstrates compliance with GDC 16. Referenced sections are as follows:

a.	Containment systems	Section 6.2
b.	MSIV Leakage Alternate Drain Pathway	Section 6.7
c.	Primary containment ventilation system	Section 9.4.5
d.	Accident analysis	Chapter 15

GDC 17 - Electric Power Systems

> An onsite electric power system and an offsite electric power system shall be provided to permit functioning of structures, systems, and components important to safety. The safety function for each system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to assure that 1) specified acceptable fuel design limits and design conditions of the RCPB are not exceeded as a result of anticipated operational occurrences and 2) the core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents.
>
> The onsite electric power supplies (including the batteries) and the onsite electric distribution system shall have sufficient independence, redundancy, and testability to perform their safety functions assuming a single failure.
>
> Electric power from the transmission network to the onsite electric distribution system shall be supplied by two physically independent circuits (not necessarily on separate rights of way) designed and located so as to minimize to the extent practical the likelihood of their simultaneous failure under operating and postulated accident and environmental conditions. A switchyard common to both circuits is acceptable. Each of these circuits shall be designed to be available in sufficient time, following a loss of all onsite alternating current power supplies and the other offsite electric power circuit, to assure that specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded. One of these circuits shall be designed to be available within a few seconds following a LOCA to assure that core cooling, containment integrity, and other vital safety functions are maintained.
>
> Provisions shall be included to minimize the probability of losing electric power from any of the remaining supplies as a result of, or coincident with, the loss of power generated by the nuclear power unit, the loss of power from the transmission network, or the loss of power from the onsite electric power supplies.

Design Evaluation

Two separate offsite power systems that are common to both units and four physically independent, onsite, standby diesel generators per unit, with associated battery systems, are provided to supply adequate power to all the functions important to safety. Either of the two offsite power systems or any three of the four onsite standby diesel generator systems in each unit have sufficient capability to operate safety-related equipment so that specified acceptable fuel design limits and design conditions of the RCPB are not exceeded as a result of anticipated operational occurrences and to cool the reactor core and maintain primary containment integrity and other vital functions if there are postulated accidents. For a further and more detailed discussion on safety-related load distribution on diesel generators, including safety-related equipment common to both units, see Chapter 8.

The two independent offsite power systems supply electric power to the onsite power distribution system via the 220 kV and 500 kV switchyards. The two switchyards are approximately 1.7 miles apart. These two power systems are physically independent and are designed to minimize the possibility of their simultaneous failure under operating and postulated accident and environmental conditions. Each offsite source is capable of supplying all safety-related loads during a LOCA in one unit with a simultaneous safe shutdown of the other unit while maintaining proper voltage regulation.

Each offsite power source can supply all engineered safeguard buses through its associated transformer. Power is available to the safeguard buses from their preferred offsite power source during normal operation and from the alternate offsite power source if the preferred power is unavailable. Each of the offsite power sources is available immediately following a loss of all onsite alternating current power supplies and the other offsite electric power circuit to ensure that fuel design limits and design conditions of the RCPB are not exceeded. The loss of both offsite power sources to a safeguard bus results in the automatic starting and connection of the associated diesel generator. Loads are progressively and sequentially added to prevent generator instabilities. The sequential loading is such that core cooling, containment integrity, and other vital safety functions are maintained.

In addition to the two offsite sources described above, a third offsite source is available from the 33 kV distribution system to supply power to the engineered safeguard loads. This source can be connected to the safeguard buses within 72 hours if there is a loss of one of the two offsite sources or of one of the safeguard transformers.

The onsite safeguard power supplies, including the safeguard batteries and onsite safeguard electric distribution systems, are independent, redundant, and testable, thus ensuring their operability and ability to perform their safety functions, assuming that there is a single failure. For a further and detailed discussion of electric power systems, see Chapter 8.

Onsite and offsite electric power systems are provided in accordance with GDC 17. Provisions are included in the design of the electric power system to minimize the probability of losing electric power from any of the remaining supplies as a result of, or coincident with, the loss of power generated by the nuclear power unit, the loss of power from the transmission network, or the loss of power from the onsite electric power supplies.

GDC 18 - Inspection and Testing of Electric Power Systems

> Electric power systems important to safety shall be designed to permit appropriate periodic inspection and testing of important areas and features, such as wiring, insulation, connections, and switchboards, to assess the continuity of the systems and the condition 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 operational sequence that brings the systems into operation, including operation of applicable portions of the protection system, and the transfer of power among the nuclear power unit, the offsite power system, and the onsite power system.

Design Evaluation

Transmission line protective relaying will be tested on a routine basis. This can be accomplished without removing the transmission lines from service.

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

Full load testing of each standby diesel generator can be performed while the plant is at power by manually starting each standby generator and by manual synchronization to the normal power supply. These tests prove the operability of the electric power systems under conditions as close to design as practicable to assess the continuity of these systems and the condition of the components. The inspection and testing of electric power systems, described in Chapters 8 and 16, conform to GDC 18.

<u>GDC 19 - Control Room</u>

> A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including 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 its equivalent to any part of the body, for the duration of the accident.
>
> Equipment at appropriate locations outside the control room shall be provided 1) with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown and 2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures.
>
> Note: The application of Alternative Source Terms in accordance with 10CFR50.67 modified the GDC 19 dose limit from 5 Rem whole body to 5 Rem TEDE.

<u>Design Evaluation</u>

A control room is provided in which appropriate controls and instrumentation are located to permit personnel to operate the unit safely under normal conditions or maintain it in a safe condition under accident conditions. The control room and associated postaccident ventilation systems are designed in accordance with seismic Category I requirements.

The design of the control room permits access and occupancy during a LOCA. Sufficient shielding and ventilation are provided to permit occupancy of the control room for a period of 30 days following the LOCA, without receiving more than a 5 Rem total effective dose equivalent (TEDE). An analysis of exposures within the control room for each of the postulated accidents is presented in Chapter 15.

The capability for prompt hot shutdown of the reactor and the capability for subsequent cold shutdown through suitable procedures from locations outside the control room is provided by the remote shutdown system, if the control room becomes inaccessible. The remote shutdown system has the 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 subsequent cold shutdown of the reactor. The remote shutdown system panel contains controls for the following equipment:

a. RHR system - The controls for one loop of the RHR system and associated RHRSW system are provided on the remote shutdown panel. The suppression pool cooling and shutdown cooling modes of RHR system operation can be controlled from the remote shutdown panel.

b. RCIC system - All basic RCIC equipment can be controlled from the remote shutdown panel.

c. Reactor recirculation system - The suction valve of one recirculation pump can be controlled from the remote shutdown panel.

d. MSRVs - Three MSRVs can be operated from the remote shutdown panel.

e. Ac power supplies for the above systems can be controlled from the remote shutdown panel.

See Section 7.4 for a further detailed discussion of remote shutdown capabilities. The control room and the remote shutdown panels conform to GDC 19.

GDC 20 - Protection System Functions

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

Design Evaluation

The RPS is designed to provide timely protection against the onset and consequences of conditions that threaten the integrity of the fuel barrier and nuclear system process barrier. Fuel damage is prevented by initiation of an automatic reactor shutdown if monitored nuclear system variables exceed pre-established limits of anticipated operational occurrences. Trip settings are selected and verified to be far enough above or below operating levels to provide proper protection but not be subject to spurious actuation. The RPS includes the sensors, relays, bypass circuitry, and switches that signal the control rod system to shut down the reactor. The shutdowns initiated by neutron monitoring system variables, nuclear system high pressure, turbine stop valve closure, TCV fast closure, and reactor vessel low water level prevent fuel damage following abnormal operational transients. Specifically, these process parameters initiate a shutdown in time to prevent the core from exceeding thermal-hydraulic safety limits during abnormal operational transients. Response by the RPS is prompt, and the total shutdown time is short.

A fully withdrawn control rod (withdrawn to 144 inches) traverses 90% of its full stroke at Technical Specification insertion rates which is sufficient to ensure that acceptable fuel design limits are not exceeded.

In addition to the RPS, which provides for automatic shutdown of the reactor to prevent fuel damage, other portions of the protection systems are provided to sense accident conditions and automatically initiate the operation of other systems and components important to safety. The ECCS and the primary containment and reactor containment isolation control system are initiated automatically following a LOCA to limit the extent of fuel damage and prevent the release of significant amounts of radioactive materials from the fuel and the nuclear system process barrier. The control and instrumentation for the ECCS and the isolation systems are initiated automatically when monitored variables exceed preselected operational limits.

The design of the protection system satisfies the functional requirements as specified in GDC 20. Referenced sections are as follows:

a.	Fuel System Design	Section 4.2
b.	ECCS	Section 6.3
c.	Reactor Trip System (Reactor Protection System) – Instrumentation And Controls	Section 7.2
d.	ESF systems	Section 7.3
e.	All Other Instrumentation Systems Important To Safety	Section 7.6
f.	Accident analyses	Chapter 15

GDC 21 - Protection System Reliability and Testability

> The protection system shall be designed for high functional reliability and inservice testability commensurate with the safety functions to be performed. Redundancy and independence designed into the protection system shall be sufficient to assure that 1) no single failure results in loss of the protection function 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 capability to test channels independently to determine failures and losses of redundancy that may have occurred.

Design Evaluation

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

The protection system includes design features that permit inservice testing. This ensures the functional reliability of the system if the monitored variables exceed the corrective action setpoint.

The RPS initiates an automatic reactor shutdown if the monitored plant variables exceed pre-established limits. The RPS consists of two independent trip systems. Each trip system has two trip logics arranged in a one-out-of-two-twice logic, to produce an automatic trip signal.

Both the manual and automatic portions of each trip logic of the RPS can be tested independently during reactor operation. The 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 determines if failures or losses of redundancy have occurred. CRD operability can be tested during normal reactor operation. CRD position indicators and the incore neutron detectors are used to verify control rod movement. Each control rod can be withdrawn one notch and then reinserted to the original position without significantly disturbing the reactor system. One control rod is tested at a time. The control rod mechanism overdrive demonstrates rod-to-drive coupling integrity. Hydraulic supply subsystem pressures can be observed on control room instrumentation. The HCU scram accumulator pressure is monitored and the scram discharge volume level is continuously monitored.

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

a.	Component and subsystem design	Section 5.4
b.	Containment systems	Section 6.2
c.	ECCS	Section 6.3
d.	RPS	Section 7.2
e.	ESF systems	Section 7.3
f.	All other instrumentation systems required for safety	Section 7.6
g.	Accident analyses	Chapter 15

GDC 22 - Protection System Independence

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

Design Evaluation

The components of protection systems are designed so that the mechanical and thermal environment resulting from any potential accident condition in which the components are required to function does not interfere with that function. The wiring for the protection system outside the control room is run in rigid metallic conduit or enclosed raceways segregated from all other wiring.

Only one trip actuator logic circuit from each trip system may be run in the same wireway.
The system sensors are electrically and physically separated. In general, redundant sensors have separate process taps. Where common process taps are used, analysis shows that failure of the common process tap will not interfere with the protection function. The wires from duplicate sensors on a common process tap are run in separate wireways.

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

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

The protection system meets the design requirements for functional and physical independence as specified in GDC 22. Referenced sections are as follows:

a.	Component and subsystem design	Section 5.4
b.	ECCS	Section 6.3
c.	RPS	Section 7.2
d.	ESF system	Section 7.3
e.	All other instrumentation systems required for safety	Section 7.6
f.	Accident analyses	Chapter 15

GDC 23 - Protection System Failure Modes

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

Design Evaluation

The RPS and the normally energized portion of the PCRVICS are designed to fail in a safe state on disconnection or loss of energy supply. The rest of the ESFs will not initiate a protection function on disconnection or loss of energy supply. This is acceptable since these systems are designed with more than one independent trip logic, each with its own independent power supply. Disconnection of one of the trip logics or loss of one of the power supplies will not prevent accomplishing a protection function when required. The environmental conditions in which the instrumentation and equipment of the protection system must operate were considered in establishing the component specifications. Instrumentation specifications are based on the worst expected ambient conditions in which the instruments must operate.

The failure modes of the protection system are such that it fails in an acceptable state as required by GDC 23. Referenced sections are as follows:

a.	Environmental design of mechanical and electrical equipment	Section 3.11
b.	RPS	Section 7.2
c.	ESF systems	Section 7.3

d. All other instrumentation systems required for safety — Section 7.6

GDC 24 - Separation of Protection and Control Systems

The protection system shall be separated from control systems to the extent that failure of any single control system component or channel or failure or removal from service of any single protection system component or channel 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.

Design Evaluation

There is separation between the protection system and the control systems. Outputs from the control system components and channels are not used as protection system inputs. The sensors, trip channels, and trip logics of the protection system are not used for automatic control of process systems. Therefore, failure in the controls and instrumentation of process systems cannot induce failure of any portion of the protection system.

For additional details of evaluation of the reactivity control system, see the evaluation for GDC 25.

The protection system is separated from control systems as required in GDC 24. Referenced sections are as follows:

a. RPS — Section 7.2

b. ESF systems — Section 7.3

c. All other instrumentation systems required for safety — Section 7.6

GDC 25 - Protection System Requirements for Reactivity Control Malfunctions

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

Design Evaluation

The RPS provides protection against the onset and consequences of conditions that threaten the integrity of the fuel barrier and the nuclear system process barrier. Any monitored variable that exceeds the scram setpoint initiates an automatic scram and does not prevent the remaining variables from being monitored, and if one channel fails, the remaining portions of the RPS can function.

The reactor manual control system is designed so that no single failure can negate the effectiveness of a reactor scram. The circuitry for the reactor manual control system is completely independent of the circuitry controlling the scram valves. This separation of the scram and normal rod control functions prevents failures in the reactor manual control circuitry from affecting the scram circuitry. Because each control rod is controlled as an individual unit, a failure that results in energizing any of the insert or withdraw solenoid valves can affect only one control rod. The effectiveness of a reactor scram is not impaired by the malfunctioning of any one control rod.

The most serious rod withdrawal errors occur when an out-of-sequence rod is continuously withdrawn while the reactor is just subcritical. The RWM normally prevents the withdrawal of out-of-sequence control rods. If such a continuous rod withdrawal were to occur, the increase in fuel temperature subsequent to scram would not be sufficient to exceed acceptable fuel design limits.

The design of the protection system ensures that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control systems as specified in GDC 25. Referenced sections are as follows:

a.	Fuel mechanical design	Section 4.2
b.	Nuclear design	Section 4.3
c.	Thermal and hydraulic design	Section 4.4
d.	RPS	Section 7.2
e.	All other instrument systems required for safety	Section 7.6
f.	Control systems not required for safety	Section 7.7
g.	Accident analyses	Chapter 15

GDC 26 - Reactivity Control System Redundancy and Capability

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

Design Evaluation

Two independent reactivity control systems using different design principles are provided. The normal method of reactivity control employs control rod assemblies that contain a neutron absorbing material as described in Section 4.2. Control of reactivity is operationally provided by a combination of these movable control rods, burnable poisons, and reactor coolant recirculation system flow. These systems accommodate fuel burnup, load changes, and long-term reactivity changes.

Reactor shutdown by the CRD system is sufficiently rapid to prevent the exceeding of acceptable fuel design limits for normal operation and all abnormal operational transients. The circuitry for manual insertion or withdrawal of control rods is completely independent of the circuitry for reactor scram. Two sources of scram energy (accumulator pressure and reactor vessel pressure) provide the needed scram performance over the entire range of reactor pressure (i.e., from operating conditions to cold shutdown).

The design of the control rod system includes appropriate margin for malfunctions, such as stuck rods, in the highly unlikely event that they do occur. Control rod withdrawal sequences and patterns are selected before operation to achieve optimum core performance and, simultaneously, low individual rod worths. The operating procedures for accomplishing such patterns are supplemented by the RWM, which prevents rod withdrawals yielding a rod worth greater than permitted by the preselected rod withdrawal pattern. An additional design basis of the control rod system requires that the core in its maximum reactivity condition be subcritical with the control rod of the highest worth fully withdrawn and all other rods fully inserted. Because of the carefully planned and regulated rod withdrawal sequence, prompt shutdown of the reactor can be achieved with the insertion of a small number of the many independent control rods. If a reactor scram is necessary, the unlikely occurrence of a limited number of stuck rods does not impair the capability of the control rod system to render the core subcritical.

A SLCS containing a neutron-absorbing sodium pentaborate solution is the independent backup system. This system has the capability to shut the reactor down from full power and maintain it in a subcritical condition at any time during the core life. The reactivity control provided to reduce reactor power from rated power to a shutdown condition with the control rods withdrawn in the power pattern accounts for the reactivity effects of xenon decay, elimination of steam voids, change in water density due to the reduction in water temperature, Doppler effect in uranium, change in neutron leakage from boiling to cold, and change in rod worth as boron affects the neutron migration length.

The redundancy and capabilities of the reactivity control systems satisfy the requirements of GDC 26. Referenced sections are as follows:

a.	Fuel mechanical design	Section 4.2
b.	Reactor trip system	Section 7.2
c.	Systems required for safe shutdown	Section 7.4
d.	Control systems not required for safety	Section 7.7

GDC 27 - Combined Reactivity Control Systems Capability

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

Design Evaluation

There is no credible event applicable to the BWR that requires combined capability of the control rod system and poison additions by SLCS. The primary reactivity control system for the BWR during postulated accident conditions is the control rod system. Abnormalities are sensed, and, if protection system limits are reached, corrective action is initiated through an automatic scram. High integrity of the protection system is achieved through the combination of logic arrangement, trip channel redundancy, power supply redundancy, and physical separation. High reliability of reactor scram is further achieved by separation of scram and manual control circuitry, individual control units for each control rod, and fail-safe design features built into the CRD system. Response by the RPS is prompt, and the total scram time is short.

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

The reactor core design assists in maintaining the stability of the core under accident conditions as well as during power operation. Reactivity coefficients in the power range that contribute to system stability are: the fuel temperature or Doppler coefficient, moderator void coefficient, and moderator temperature coefficient. The overall power reactivity coefficient is negative and provides a strong negative reactivity feedback under severe power transient conditions.

The design of the reactivity control system ensures reliable control of reactivity under postulated accident conditions with appropriate margin for stuck rods. The capability to cool the core is maintained under all postulated accident conditions. Thus, GDC 27 is satisfied. Referenced sections are as follows:

a.	Fuel mechanical design	Section 4.2
b.	Nuclear design	Section 4.3
c.	Thermal and hydraulic design	Section 4.4
d.	RPS	Section 7.2
e.	All other instrumentation systems required for safety	Section 7.6
f.	Control systems not required for safety	Section 7.7
g.	Accident analyses	Chapter 15

GDC 28 - Reactivity Limits

The reactivity control systems shall be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effect 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 Evaluation

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. This function provides the operator with an effective backup control rod monitoring routine that enforces adherence to established startup, shutdown, and low power level operations control rod procedures.

The control rod mechanical design incorporates a hydraulic velocity limiter in the control rod that prevents rapid rod ejection. This engineered safeguard protects against a high reactivity insertion rate by limiting the control rod velocity to less than 5 ft/sec. Normal rod movement is limited to 6 inch increments, and the rod withdrawal rate is limited through the hydraulic valve to 3 in/sec.

The accident analyses in Chapter 15 evaluate in detail postulated reactivity accidents as well as the 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. The results of these analyses indicate that none of the postulated reactivity transients or accidents results in damage to the RCPB. In addition, the integrity of the core, its support structures, or other RPV internals is maintained so that the capability to cool the core is not impaired for any of the postulated reactivity accidents described in Chapter 15.

The design features of the reactivity control system, which limit the potential amount and rate of reactivity increase, ensure that GDC 28 is satisfied for all postulated reactivity accidents.
Referenced sections are as follows:

a.	Design of structures, components, equipment, and systems	Chapter 3
b.	Fuel mechanical design	Section 4.2
c.	Nuclear design	Section 4.3
d.	Nuclear design	Section 4.5
e.	Integrity of RCPB	Section 5.2
f.	Reactor vessel and appurtenances	Section 5.3
g.	Component and subsystem design	Section 5.4
h.	Reactor trip system	Section 7.2
i.	All other instrumentation systems	Section 7.6

required for safety

j.	Accident analyses	Chapter 15

GDC 29 - Protection Against Anticipated Operational Occurrences

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

Design Evaluation

The high functional reliability of the protection and reactivity control systems is achieved through the combination of logic arrangement, redundancy, physical and electrical independence, functional separation, fail-safe design, and inservice testability. These design features are discussed in GDC 21, 22, 23, 24, 25, and 26.

An extremely high probability of correct protection and reactivity control systems response to anticipated operational occurrences is maintained by a thorough program of inservice testing and surveillance. Active components can be tested or removed from service for maintenance during reactor operation without compromising the protection or reactivity control functions. Components important to safety such as CRDs, MSIVs, RHR pumps, etc, are tested during normal reactor operation. Functional testing and calibration schedules are developed using available failure rate data, reliability analyses, and operating experience. These schedules represent an optimization of protection and reactivity control system reliability by considering the reliability effects during individual component testing on the portion of the system not undergoing testing. The capability for inservice testing ensures the high functional reliability of protection and reactivity control systems if a reactor variable exceeds the corrective action setpoint.

The capabilities of the protection and reactivity control systems to perform their safety functions if there are anticipated operational occurrences satisfy the requirements of GDC 29. Referenced sections are as follows:

a.	Fuel mechanical design	Section 4.2
b.	Component and subsystem design	Section 5.4
c.	Containment systems	Section 6.2
d.	ECCS	Section 6.3
e.	RPS	Section 7.2
f.	ESF system	Section 7.3
g.	All other instrumentation systems required for safety	Section 7.6
h.	Accident analyses	Chapter 15

GDC 30 - Quality of Reactor Coolant Pressure Boundary

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

Design Evaluation

By using conservative design practices and detailed quality control procedures, the pressure-retaining components of the RCPB are designed and fabricated to retain their integrity during normal and postulated accident conditions. Accordingly, components that comprise the RCPB are designed, fabricated, erected, and tested in accordance with recognized industry codes and standards listed in Chapter 5. Further product and process quality planning is provided as discussed in the evaluation of GDC 1 to ensure conformance with the applicable codes and standards and to retain appropriate documented evidence verifying compliance. Because the subject matter of this criterion deals with the aspects of the RCPB, further discussion of this subject appears in the response to GDC 14.

Means are provided for detecting reactor coolant leakage. The leak detection system consists of sensors and instruments to detect, annunciate, and, in some cases, isolate the RCPB from potential hazardous leaks before predetermined limits are exceeded. As described in Section 5.2.5, small leaks are detected by temperature and pressure changes, increased frequency of sump pump operation, and measurement of airborne radioactivity in the primary containment atmosphere. In addition to these means of detection, large leaks are detected by flow rates in process lines and changes in reactor water level. The allowable leakage rates are based on the predicted and experimentally determined behavior of cracks in pipes, the ability to make up coolant system leakage, the normally expected background leakage due to equipment design, and the detection capability of the various sensors and instruments. The total leakage rate limit is established so that, in the absence of normal ac power concurrent with a loss of feedwater supply, makeup capabilities are provided by the RCIC system. While the leak detection system provides protection from small leaks, the ECCS network provides protection for the complete range of discharges from ruptured pipes. Thus, protection is provided for the full spectrum of possible discharges. The RCPB and the leak detection system are designed to meet the requirements of GDC 30. Referenced sections are as follows:

a.	Design of structures, components, equipment, and systems	Chapter 3
b.	Integrity of RCPB	Section 5.2
c.	Reactor vessel and appurtenances	Section 5.3
d.	Component and subsystem design	Section 5.4
e.	All other instrumentation systems required for safety	Section 7.6
f.	Accident analyses	Chapter 15

GDC 31 - Fracture Prevention of Reactor Coolant Pressure Boundary

> The RCPB shall be designed with sufficient margin to assure that, when stressed under operating, maintenance, testing, and postulated accident conditions, 1) the boundary behaves in a nonbrittle manner 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 Evaluation

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

The 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 about $1x10^{17}$ nvt with neutrons of energies in excess of 1 MeV. Since the material NDTT dictates the minimum operating temperature at which the reactor vessel can be pressurized, it is desirable to keep the NDTT as low as possible.

The reactor assembly design provides an annular space from the outermost fuel assemblies to the inner surface of the reactor vessel that serves to attenuate the fast neutron flux incident on the reactor vessel wall. This annular volume contains the core shroud, jet pump assemblies, and reactor coolant. Assuming that plant operation is at rated power and the plant has an availability of 100% for the plant lifetime, the neutron fluence at the inner surface of the vessel is not sufficient to appreciably shift the transition temperature. Expected shifts in transition temperature during design life as a result of environmental conditions, such as neutron flux, are considered in the design. Operational limitations assume that NDTT shifts are accounted for in the reactor operation.

The RCPB is designed, maintained, and tested so that adequate assurance is provided that the boundary will behave in a nonbrittle manner throughout the life of the plant. Therefore, the RCPB is in conformance with GDC 31. Referenced sections are as follows:

a.	Design of structures, components, equipment, and systems	Chapter 3
b.	Integrity of RCPB	Section 5.2
c.	Reactor vessel and appurtenances	Section 5.3

GDC 32 - Inspection of Reactor Coolant Pressure Boundary

> 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 RPV.

Design Evaluation

The LGS design conforms to GDC 32. The RCPB design meets the requirements of the ASME B&PV Code, Section XI, including Summer 1971 Addenda, which requires access for all mandatory inspections. The design also permits the conduct of a material surveillance program for the RPV. Additional details of these features are in Section 5.2.4.

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

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

The plant testing and inspection programs ensure that the requirements of GDC 32 are met. Referenced sections are as follows:

a.	Design of structures, components, equipment, and systems	Chapter 3
b.	RCPB	Section 5.2
c.	Reactor vessel and appurtenances	Section 5.3
d.	Component and subsystem design	Section 5.4

GDC 33 - Reactor Coolant Makeup

> A system to supply reactor coolant makeup for protection against small breaks in the RCPB shall be provided. The system safety function shall be to assure that specified acceptable fuel design limits are not exceeded as a result of reactor coolant loss due to leakage from the RCPB and rupture of small piping or other small components 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 Evaluation

Means are provided for detecting reactor coolant leakage. The leak detection system consists of sensors and instruments to detect, annunciate, and in some cases isolate the RCPB from potential hazardous leaks before predetermined limits are exceeded. As described in Section 5.2.5, small leaks are detected by temperature and pressure changes, increased frequency of sump pump operation, and the measurement of airborne radioactivity. The allowable leakage rates are based on predicted and experimentally determined behavior of cracks in pipes, the ability to make up coolant system leakage, the normal 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 systems. Thus, protection is provided to ensure that fuel clad temperature limits are not exceeded.

The plant is designed to provide ample reactor coolant makeup for protection against small leaks in the RCPB. The design of these systems meets the requirements of GDC 33. Referenced sections are as follows:

a.	Integrity of RCPB	Section 5.2
b.	ECCS	Section 6.3
c.	ESF Systems	Section 7.3

GDC 34 - Residual Heat Removal

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 RCPB 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 Evaluation

The RHR system provides the means to:

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

b. Deleted

The major equipment of the RHR system consists of two heat exchangers, four main system pumps, and two service water pumps. The equipment is connected by associated valves and piping, and the controls and instrumentation are provided for proper system operation.

One loop consists of a heat exchanger, two main system pumps, and associated piping, and is located in one area of the reactor enclosure. A redundant loop is located in another area of the reactor enclosure to minimize the possibility of a single physical event causing the loss of the entire system.

Each heat exhanger is alignable to one of two RHR pumps. During cold shutdown and refueling operations, this results in availability of four subsystems of shutdown cooling.

As described in the evaluation for GDC 44, the RHRSW system is used to remove heat from the RHR heat exchanger.

Both RHR loops take suction from a common line coming from the reactor. A failure of either containment isolation valve in this common suction line would therefore prevent use of this flow path. In the event of such a failure, a flow path can be established through the ADS valves (Section 15.2.9).

During cold shutdown and refueling operation conditions when two subsystems of shutdown cooling may be aligned to a common passive heat exchanger and discharge piping, a failure of the associated shutdown cooling discharge valve or check valve will require manual actions to repair the valve to restore flow, or if unsuccessful, may require cooling water flow to be returned through the LPCI injection line.

The systems used for residual heat removal are powered from the safeguard buses. The design of the safeguard buses, as described in the evaluation for GDC 17, assures that residual heat can be removed, assuming a single failure, when onsite electric power is available (assuming offsite power is not available) and when offsite electric power is available (assuming onsite power is not available).

The RHR system is adequate to remove residual heat from the reactor core to ensure that fuel and RCPB design limits are not exceeded. Redundant offsite and onsite electric power systems are provided. The design of the systems used to remove residual heat, including their power supplies, meets the requirements of GDC 34. Referenced sections are as follows:

a.	Component and subsystem design	Section 5.4
b.	ECCS	Section 6.3
c.	ESF systems	Section 7.3
d.	Systems required for safe shutdown	Section 7.4
e.	Onsite power systems	Section 8.3
f.	Water systems	Section 9.2
g.	Accident analyses	Chapter 15

GDC 35 - Emergency Core Cooling

> A system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate such that 1) fuel and clad damage that could interfere with continued effective core cooling is prevented 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 Evaluation

The ECCS consists of the following: HPCI system; ADS; CS system; and LPCI system (an operating mode of the RHR system). The ECCS is designed to limit fuel cladding temperature over the complete spectrum of possible break sizes in the nuclear system process barrier, including a complete and sudden circumferential rupture of the largest pipe connected to the reactor vessel.

The HPCI system consists of a steam turbine, a constant flow pump, system piping, valves, controls, and instrumentation. The HPCI system is provided to ensure that the reactor core is adequately cooled to prevent excessive fuel clad temperatures for breaks in the nuclear system that do not result in rapid depressurization of the reactor vessel. The HPCI system continues to operate until the reactor vessel pressure is below the pressure at which LPCI operation or CS system operation maintains core cooling. Two sources of water are available, the CST and the suppression pool.

In case the capability of the feedwater pumps, CRD water pumps, and RCIC and HPCI systems is not sufficient to maintain the reactor water level, the ADS functions to reduce the reactor pressure so that flow from the LPCI and the CS systems enters the reactor vessel in time to cool the core and prevent excessive fuel clad temperature. The ADS uses several of the nuclear system pressure relief valves to relieve the high pressure steam to the suppression pool.

Two independent loops are provided as a part of the CS system. Each loop consists of two centrifugal water pumps driven by electric motors, a spray sparger in the reactor vessel above the core, piping and valves to convey water from the suppression pool to the sparger, and the associated controls and instrumentation. In case of low water level in the reactor vessel or high pressure in the drywell, the CS system automatically sprays water onto the top of the fuel assemblies in time and at a sufficient flow rate to cool the core and prevent excessive fuel temperature. The LPCI system starts from the same signals that initiate the CS system and operates independently to achieve the same objective by flooding the reactor vessel.

LPCI and CS operation provide protection to the core if there is a large break in the nuclear system when the feedwater pumps, CRD, RCIC, and the HPCI systems are unable to maintain reactor vessel water level. Protection provided by LPCI and CS also extends to a small break for which the ADS has operated to lower the reactor vessel pressure so that these systems start to provide core cooling.

The results of ECCS performance for the entire spectrum of liquid line breaks are discussed in Section 6.3.

The RHR and CS systems are powered from the safeguard buses. The design of the safeguard buses, as described in the evaluation for GDC 17, assures that emergency core cooling can be provided, assuming a single failure, when onsite electric power is available (assuming offsite power is not available) and when offsite electric power is available (assuming onsite power is not available).

The ECCS provided is adequate to prevent fuel and clad damage that could interfere with effective core cooling and to limit clad metal-water reaction to a negligible amount. Redundant offsite and onsite electric power systems are provided. The design of the ECCS, including its power supplies, meets the requirements of GDC 35. Referenced sections are as follows:

a.	Component and subsystem design	Section 5.4
b.	ECCS	Section 6.3
c.	ESF systems	Section 7.3
d.	Onsite power systems	Section 8.3
e.	Water systems	Section 9.2

f. Accident analyses Chapter 15

GDC 36 - Inspection of Emergency Core Cooling System

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

Design Evaluation

The ECCS discussion in GDC 35 includes inservice inspection considerations.

The CS spargers within the reactor vessel are accessible for remote visual inspection during refueling outages. Removable plugs in the sacrificial shield and/or panels in the insulation provide access for the examination of nozzles from the outside of the vessel. Removable insulation is provided on the ECCS piping out to and including the first isolation valve outside containment. Inspection of the ECCS is in accordance with Section XI of the ASME B&PV Code, or 10 CFR 50.69 Alternative Treatment, when applicable. Refer to Section 6.6.1 for further information.

During plant operations, the pumps, valves, piping, instrumentation, wiring, and other components outside the primary containment can be visually inspected at any time. Components inside the primary containment can be inspected when the drywell is open for access. When the reactor vessel is open for refueling or other purposes, the spargers and other internals can be inspected. Portions of the ECCS that are part of the RCPB are designed to specifications for inservice inspection to detect defects that might affect the cooling performance. The design of the reactor vessel and internals for inservice inspection and the plant testing and inspection program ensures that the requirements of GDC 36 are met. See Section 5.2.4 for further discussion of ECCS inservice inspection.

GDC 37 - Testing of Emergency Core Cooling System

> The ECCS shall be designed to permit appropriate periodic pressure and functional testing to assure 1) the structural and 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 the operation of the associated cooling water system.

Design Evaluation

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

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

The pumps and valves of these systems are capable of being tested periodically to verify operability. Flow rate tests can be conducted on the CS, LPCI, and HPCI systems.

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

a.	Overpressurization protection	Section 5.2
b.	ECCS inspection and testing	Section 6.3
c.	ECCS instrumentation and controls	Section 7.3
d.	Standby ac power system	Section 8.3

GDC 38 - Containment Heat Removal

> 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 to 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 Evaluation

If there is a LOCA, the pressure-suppression system rapidly condenses the steam to prevent containment overpressure. The containment feature of pressure-suppression employs two separate compartmented sections of the primary containment: the drywell that houses the nuclear system and the suppression chamber containing a large volume of water. Any increase in pressure in the drywell from a leak in the nuclear system is relieved below the surface of the suppression chamber water pool by connecting vent lines. The pressure buildup in the suppression chamber is equalized with the drywell by a vent line and vacuum breaker arrangement. Cooling systems remove heat from the reactor core, the drywell, and from the water in the suppression chamber during an accident condition; thus, continuous cooling of the primary containment is provided.

Sufficient water is provided in the suppression pool to accommodate the initial energy that can be transiently released into the drywell from a postulated pipe failure.

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

The containment heat removal function is accomplished by the containment cooling mode of the RHR system. This mode consists of the suppression pool cooling subsystem and the containment spray subsystem.

Either or both RHR heat exchangers can be manually activated to remove energy from the containment. The redundancy and capability of the offsite and onsite electrical power systems for the RHR system, discussed in the evaluation against GDC 34, ensure that the system safety function can be accomplished, assuming there is a single failure, for onsite electric power system operation (assuming that offsite power is not available) and for offsite electric power system operation (assuming that onsite power is not available).

The pressure-suppression system is capable of rapid containment pressure and temperature reduction following a LOCA to ensure that the design limits are not exceeded. Redundant offsite and onsite electrical power systems are provided. The design of the containment heat removal system meets the requirements of GDC 38. Referenced sections are as follows:

a.	Component and subsystem design	Section 5.4
b.	Containment systems	Section 6.2
c.	ECCS	Section 6.3
d.	ESF systems	Section 7.3
e.	Electric power systems	Chapter 8
f.	Auxiliary systems	Chapter 9
g.	Accident analyses	Chapter 15

GDC 39 - Inspection of Containment Heat Removal System

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

Design Evaluation

Provisions are made to facilitate periodic inspections of active components and other important equipment of the containment pressure-reducing systems. During plant operations, the pumps, valves, piping, instrumentation, wiring, and other components outside the primary containment can be periodically visually inspected. Components inside the primary containment can be inspected when the drywell is open for access. The testing frequencies of most components are correlated with the component inspection.

The pressure-suppression chamber is designed to permit appropriate periodic inspection. Space is provided inside the chamber for inspection and maintenance. There are two hatches that permit access to the suppression chamber for inspection.

The containment heat removal system is designed to permit periodic inspection of major components both outside and within the primary containment. This design meets the requirements of GDC 39. Referenced sections are as follows:

a.	Component and subsystem design	Section 5.4

b.	Containment systems	Section 6.2
c.	ECCS	Section 6.3
d.	Water systems	Section 9.2

GDC 40 - Testing of Containment Heat Removal System

> The containment heat removal system shall be designed to permit appropriate periodic pressure and functional testing to assure 1) the structural and 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 Evaluation

The containment heat removal function is accomplished by the containment cooling mode of the RHR system. This mode consists of the suppression pool cooling subsystem and containment spray subsystem.

The RHR system is provided with sufficient test connections and isolation valves to permit periodic pressure and flow rate testing.

The pumps and valves of the RHR system can be operated periodically to verify operability. The containment cooling mode is not automatically initiated, but operation of the components can be periodically verified. The operation of associated cooling water systems is discussed in the conformance to GDC 46. It is concluded that the requirements of GDC 40 are met.

GDC 41 - Containment Atmosphere Cleanup

> 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 quantity of fission products released to the environment following postulated accidents and to control the concentration of hydrogen or oxygen and other substances in the containment atmosphere following postulated accidents to assure that containment integrity is maintained. Each system shall have suitable redundancy in components and features and suitable interconnections, leak detection, isolation, and containment capabilities to assure that, for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available), its safety function can be accomplished, assuming a single failure.

Design Evaluation

Fission products released into the secondary containment following postulated accidents are automatically processed by the RERS and/or the SGTS. Initiation of the SGTS and RERS follows high radiation signals from monitors located in the reactor enclosure exhaust ducts, a high drywell pressure signal, or a low reactor level signal. Initiation of the SGTS follows a high radiation signal from monitors in the refueling area exhaust ducts.

The ability of these systems to remove radioactivity from the process stream is discussed in Section 6.5. The SGTS and RERS are each composed of two trains that are separated physically and electrically so that a single failure does not prevent their function, except as described in Table 6.5-2. The redundancy of these systems is also discussed in Section 6.5.3.

A combustible gas control system consisting of redundant hydrogen recombiners maintains hydrogen and oxygen concentrations below flammable limits following a postulated LOCA. The system continuously processes the primary containment atmosphere following manual initiation. As a backup to the combustible gas control system, the containment atmosphere can be purged through the SGTS filters to the turbine enclosure north vent stack. A detailed description is provided in Section 6.2.5.

The SGTS, RERS, and the combustible gas control system meet the requirements of GDC 41 except as described above.

The containment atmospheric cleanup systems are connected to the safeguard buses. The design of the safeguard buses, as described in the evaluation for GDC 17, ensures that the containment atmospheric cleanup function can be provided, assuming a single failure, when onsite electric power is available (assuming that offsite power is not available) and when offsite electric power is available (assuming that onsite power is not available).

GDC 42 - Inspection of Containment Atmosphere Cleanup Systems

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

Design Evaluation

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

The charcoal beds are provided with removable canisters for taking charcoal samples.

For a further discussion of the SGTS and RERS inspection features, see Section 6.5.

All active components of the combustible gas control system are located externally to the primary containment and are accessible for inspection during normal operation of the plant. For a discussion of inspection, see Section 6.5.

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

GDC 43 - Testing of Containment Atmosphere Cleanup Systems

> The containment atmosphere cleanup systems shall be designed to permit appropriate periodic pressure and functional testing to assure 1) the structural and leak-tight integrity of its components, 2) the operability and performance of the active components of the

systems such as fans, filters, dampers, and valves, and 3) the operability of the systems as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system 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 Evaluation

Each unit of the SGTS and the RERS is operated periodically to ascertain the operability and performance of the major active components, such as fans, filters, motors, pumps, and valves, as well as the structural integrity of the unit. This test verifies the operability of the system as a whole and the operability of all associated subsystems. See Section 8.3.1 for a discussion of the testing of the auxiliary power system.

The leak-tightness of the HEPA filters is measured by the DOP test. A halogenated hydrocarbon refrigerant system is provided to test the activated charcoal filters. For a further discussion of testing, see Section 6.5. Each loop of the combustible gas control system is designed for periodic pressure and operability testing. For a further discussion of testing, see Section 6.5.

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

GDC 44 - Cooling Water

A system to transfer heat from structures, systems, and components important to safety to an 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 system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

Design Evaluation

The ESW system and the RHRSW system provide cooling water for the removal of excess heat from all structures, systems, and components that are necessary to maintain safety during all abnormal and accident conditions. These include the standby diesel generators, the RHR motor oil coolers and pump compartment unit coolers, the control room chillers, the CS pump compartment unit coolers, the RCIC pump compartment unit coolers, the HPCI pump compartment unit coolers, and the RHR heat exchangers.

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

The ESW system and RHRSW system meet the requirements of GDC 44. Referenced sections are as follows:

a. AC power systems Section 8.3

b. Service water system Section 9.2.1

c. ESW system Section 9.2.2

d. RHRSW system Section 9.2.3

e. UHS Section 9.2.6

GDC 45 - Inspection of Cooling Water System

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

Design Evaluation

The ESW system and RHRSW system are designed to permit appropriate periodic inspection to ensure the integrity of system components, thus meeting the requirements of GDC 45. Referenced sections are as follows:

a. ESW system Section 9.2.2

b. RHRSW system Section 9.2.3

GDC 46 - Testing of Cooling Water System

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

Design Evaluation

The RHRSW system described in GDC 44 is in operation during all plant shutdowns. The ESW system is periodically tested when the diesel generators are tested. This testing includes transfer between the normal offsite power supply and the emergency onsite power system. These systems are designed to the extent practicable to permit demonstration of operability of the systems as required for operation during a LOCA or a LOOP. Thus, the ESW and RHRSW systems meet the requirements of GDC 46. Referenced sections are as follows:

a. ESW system Section 9.2.2

b. RHRSW system Section 9.2.3

GDC 50 - Containment Design Basis

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

<u>Design Evaluation</u>

The primary containment structure, access openings, penetrations, heat removal system, and internal compartments are designed with sufficient margin to meet the requirements of GDC 50. Containment design is described in Section 6.2.

<u>GDC 51 - Fracture Prevention of Containment Pressure Boundary</u>

> 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.

<u>Design Evaluation</u>

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

The ferritic steel used for the primary containment boundary is specified so that the toughness of the material meets the above established conditions.

The weld procedure qualification ensures that the toughness of the weld metal and heat-affected zones follow the same criteria as for the base metal.

Since the primary containment is located within the reactor enclosure, the possibility of brittle fracture of ferritic material under low temperature is considerably reduced.

The lowest design service temperature is conservatively taken as 30ºF. The actual service temperature is calculated to be approximately 135ºF. Thus sufficient margin is inherent in the design to account for the various uncertainties involved in design and fabrication. The design of the reactor containment boundary meets the requirements of GDC 51. Actual service conditions above 135°F do not adversely affect this conclusion.

Records for the materials of the flued head fittings, main steam piping, and MSIVs are available for inspection. Additional information is provided in Section 5.2.3.3.

GDC 52 - Capability for Containment Leakage Rate Testing

> 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 Evaluation

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

GDC 53 - Provisions for Containment Testing and Inspection

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

Design Evaluation

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

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

Expansion bellows are not used on containment penetrations.

The reactor containment meets the requirements of GDC 53.

GDC 54 - Piping Systems Penetrating Containment

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

Design Evaluation

Piping systems that penetrate the primary containment have been accorded special design considerations to reflect their importance in accomplishing safety-related functions and in achieving isolation, if required. The penetrations are discussed in Section 6.2.4. Both the isolation valves and the system that initiates isolation use components whose quality maximizes reliability.

Sufficient independence and redundancy is provided to ensure effective isolation. Containment isolation is discussed in Section 6.2.4.

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

Piping systems penetrating primary reactor containment meet GDC 54.

<u>GDC 55 - Reactor Coolant Pressure Boundary Penetrating Containment</u>

Each line that is part of the RCPB and that penetrates the 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:

a. One locked closed isolation valve inside and one locked closed isolation valve outside containment, or

b. One automatic isolation valve inside and one locked closed isolation valve outside containment, or

c. 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

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

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

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

<u>Design Evaluation</u>

The piping systems that are part of the RCPB and penetrate primary containment conform to the requirements of GDC 55 as described in Section 6.2.4. Similarly, for lines that do not penetrate the primary containment but form a portion of the RCPB, the design ensures that isolation from the RCPB can be achieved.

<u>GDC 56 - Primary Containment Isolation</u>

Each line that connects directly to the containment atmosphere and penetrates the 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:

a. One locked closed isolation valve inside and one locked closed isolation valve outside containment, or

b. One automatic isolation valve inside and one locked closed isolation valve outside containment, or

c. 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

d. 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 Evaluation

The piping systems penetrating primary containment conform to the requirements of GDC 56 as described in Section 6.2.4.

GDC 57 - Closed System Isolation Valves

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

Design Evaluation

The piping systems penetrating primary containment conform to the requirements of GDC 57 as described in Section 6.2.4.

GDC 60 - Control of Releases of Radioactive Materials to the Environment

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 Evaluation

In all cases, the design for radioactivity control is on the basis of the requirements of 10CFR20, 10CFR50, and applicable regulations for normal operations and for any transient situation that might reasonably be anticipated to occur, and is on the basis of 10CFR50.67 dose limits for potential accidents of exceedingly low probability of occurrence. The activity level of waste gas effluents is substantially reduced by holdup of noble gases from the offgas system in charcoal decay beds and subsequent release at the plant exhaust duct.

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

Solid wastes are shipped offsite for disposal in shielded and reinforced containers that meet applicable NRC and DOT requirements (Section 11.4).

The liquid, gaseous, and solid waste systems meet the requirements of GDC 60. Referenced sections are as follows:

a.	Liquid waste system	Section 11.2
b.	Gaseous waste system	Section 11.3
c.	Solid waste system	Section 11.4
d.	Process and effluent radiological monitoring system	Section 11.5
e.	Accident analyses	Chapter 15
f.	Technical Specifications	Chapter 16

GDC 61 - Fuel Storage and Handling and Radioactivity Control

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

Design Evaluation

a. New fuel storage

   New fuel is stored in the spent fuel storage pool.

b. Spent fuel handling and storage

Irradiated fuel is stored submerged in the spent fuel storage pool located in the reactor enclosure. Fuel pool water is circulated through the FPCC system to maintain fuel pool water temperature, purity, water clarity, and water level. Storage racks preclude accidental criticality (GDC 62 design evaluation).

Reliable decay heat removal is provided by the FPCC system. The pool water is circulated through the system with suction taken from the pool and is discharged through diffusers at the bottom of the fuel pool. Pool water temperature will not exceed 143ºF during removal of the maximum normal heat load from the pool with the service water temperature at its maximum. The RHR system with its substantially larger heat removal capacity can be used as a backup for fuel pool cooling.

If there is a complete loss of capability to remove heat from the spent fuel pool using heat exchangers, heat can be removed by allowing the pool to boil and adding makeup water from the UHS (spray pond), by either of two seismic Category I flow paths, to maintain the pool water level.

High and low level switches indicate pool water level changes in the control room. Fission product concentration in the pool water is minimized by the fuel pool filter/demineralizer. This minimizes the radioactivity release from the pool to the reactor enclosure.

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

No special tests are required, because at least one pump and one heat exchanger are continuously in operation while fuel is stored in the pool. Other cooling trains are operated periodically to handle high heat loads or to replace a unit for servicing. Routine visual inspection of the system components, instrumentation, and trouble alarms is adequate to verify system operability.

c. Radioactive waste systems

The radioactive waste systems provide all equipment necessary to collect, process, and prepare for disposal all radioactive liquids, gases, and solid waste produced as a result of reactor operation.

Liquid radwastes are classified, contained, and treated as high or low conductivity, chemical, detergent or sludge wastes. Processing includes filtration, ion exchange, analysis, and dilution. Wet solid wastes are dewatered and packaged in steel containers. Dry solid wastes are compressed and packaged in steel drums. Gaseous radwastes are monitored, processed, recorded, and released so that radiation doses to persons outside the controlled area are below those allowed by applicable regulations.

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

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

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

Referenced sections are as follows:

a.	RHR system	Section 5.4
b.	Containment systems	Section 6.2
c.	New fuel storage	Section 9.1
d.	Spent fuel storage	Section 9.1
e.	FPCC system	Section 9.1
f.	HVAC systems	Section 9.4
g.	Radioactive waste management	Chapter 11
h.	Radiation protection	Chapter 12

GDC 62 - Prevention of Criticality in Fuel Storage and Handling

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

Design Evaluation

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 geometrically safe configuration of the storage rack. There is sufficient spacing between the assemblies to ensure that the array, when fully loaded, is substantially subcritical. Fuel elements are limited by rack design to only top loading and fuel assembly positions. The new and spent fuel storage racks are designed to seismic Category I requirements.

New fuel storage is discussed in Section 9.1.1. New fuel is stored in the spent fuel pool.

Spent fuel is stored underwater in the spent fuel pool. The racks in which spent fuel assemblies are placed are designed and arranged to ensure subcriticality in the storage pool. Spent fuel storage is discussed in Section 9.1.2.

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 misoperation.

The use of geometrically safe configurations for new and spent fuel storage and the design of fuel handling systems precludes accidental criticality in accordance with GDC 62. Referenced section is as follows:

a. Fuel storage and handling Section 9.1

GDC 63 - Monitoring Fuel and Waste Storage

> 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 Evaluation

Appropriate systems are provided to meet the requirements of GDC 63. A malfunction of the FPCC system that could result in a loss of residual heat removal capability and excessive radiation levels is alarmed in the control room. Alarmed conditions include low fuel pool cooling water pump discharge pressure, high and low levels in the fuel storage pool and skimmer surge tanks, and flow in the drain lines between the fuel pool gates between fuel pool and reactor well. System temperature is also continuously monitored and alarmed in the control room. Spent fuel storage is discussed in Section 9.1.2, and fuel pool cooling and cleanup are discussed in Section 9.1.3.

The reactor enclosure and refueling floor ventilation radiation monitoring systems detect abnormal amounts of radioactivity and initiate appropriate action to control the release of radioactive material to the environment. These systems are discussed in Sections 9.4 and 11.5.

Area radiation and tank and sump levels are monitored and alarmed to indicate conditions that may result in excessive radiation levels in radioactive waste system areas. Area radiation monitoring is discussed in Section 12.3.

GDC 64 - Monitoring Radioactivity Releases

> Means shall be provided for monitoring the reactor containment atmosphere, spaces containing components for recirculation of LOCA 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 Evaluation

Appropriate means are provided for monitoring radioactivity releases to meet the requirements of GDC 64.

A fission products monitoring system is provided to sample the containment (both drywell and suppression pool) atmosphere for radioactive particulates, noble gases, and iodine during normal operation. A hydrogen-oxygen analyzer system is provided to monitor the hydrogen-oxygen concentration in the containment during normal operation and following an accident.

Means are provided to monitor radioactive effluent discharge paths and the site environment for radioactivity releases. Referenced sections are as follows:

a.	RCPB leakage detection system	Section 5.2
b.	ESF systems	Section 7.3
c.	All other systems required for safety	Section 7.6
d.	Control systems not required for safety	Section 7.7
e.	Radioactive waste management	Chapter 11
f.	Airborne radioactivity monitoring	Section 12.3

3.2 CLASSIFICATION OF STRUCTURES, COMPONENTS, AND SYSTEMS

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

3.2.1 SEISMIC CLASSIFICATION

GDC 2 and 10CFR100, Appendix A require that nuclear power plant structures, systems, and components important to safety be designed to withstand the effect of earthquakes. Regulatory Guide 1.29 (Rev 3) provides additional guidance, and defines seismic Category I structures, components, and systems and those necessary to ensure:

a. The integrity of the RCPB,

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

c. The capability to prevent, or mitigate the consequences of accidents which could result in potential offsite exposures in excess of the dose limits of 10CFR50.67.

Plant structures, systems, and components, including their foundations and supports, that are designed to remain functional if there is a SSE are designated as seismic Category I, as indicated in Table 3.2-1. Class 1E electric equipment is seismic Category I equipment. Seismic classification of systems instrumentation is discussed in Chapter 7.

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

Seismic Category I structures are sufficiently isolated, or protected, from other structures to ensure that their integrity is maintained at all times.

Components and their supporting structures that are not seismic Category I, but are located in the vicinity of seismic Category I items, are listed as seismic Category IIA in Table 3.2-1. Those components listed as seismic Category IIA are either designed to seismic Category I criteria or are reviewed to identify those whose failure could result in loss of required function of seismic Category I structures, equipment, or systems required after an SSE. Components identified by this review are considered safety-impact items and are either analytically checked to confirm their integrity against collapse when subjected to seismic loading from the SSE or are separated from seismic Category I equipment by a barrier.

Structures, systems, and components that are not required to maintain their structural integrity or function during or after, the SSE are classified nonseismic (seismic Category II). The boundaries of each seismic category portion of the systems are shown on the piping and instrument diagrams in appropriate sections of this document. A cross reference of system to former UFSAR figure number is provided in Section 1.7.

Clarifications and alternate approaches to the guidelines of Regulatory Guide 1.29, as applied to LGS, are the following:

a. Regarding the seismic design of the reactor core and internals of paragraph C.1.b, reactor internals that are not required for safe shutdown of the plant are designed to maintain their relative positions following an SSE, but may not remain functional.

   These internals are listed below:

   1. Feedwater spargers

   2. Initial startup neutron sources

   3. Surveillance sample holders

   4. Incore instrument housings

   5. Steam dryer

   6. Shroud head and separator assembly

   7. Guide rods

   Generic evaluation has demonstrated that the failure of these structures will not jeopardize the safety function of other safety-related internals during a seismic event.

b. The spent fuel pool cooling system, and the associated service water supply, discussed in paragraph C.1.d and C.1.g of the guide, are seismic Category IIA rather than seismic Category I systems. However, two alternate seismic Category I cooling sources are provided via connections from the RHR system and the ESW system, as discussed in Section 9.1.3. The portions of the spent fuel pool cooling system necessary for use of these alternate cooling sources are seismic Category I. This arrangement meets the intent of the guide.

c. The seismic Category I boundary of the main steam system and connected piping outside primary containment may not always be at a valve that is normally closed, or capable of automatic isolation, as discussed in paragraph C.1.e of the guide. In these cases, the boundary is at a valve that is capable of remote manual closure to avoid unnecessary complication in lines that normally would not be provided with automatically closing valves. The remote manual valves are the following (shown in drawing M-01):

   1. Main steam to air ejectors isolation valve, HV-150
   and HV-250

   2. Main steam to steam seal evaporator isolation valve, HV-111 and HV-211

3. Main steam to reactor feed pump-turbine high pressure steam supply valve, HV-108 and HV-208.

The use of remotely operated manual valves in lieu of normally closed or automatic valves is justifiable for the following reasons:

Those portions of the steam system extending from the outermost containment isolation valve up to but not including the turbine stop valves, and connected piping of 2½ inches nominal pipe size or larger up to and including the remotely operated manual valves are classified seismic Category I. In addition, these valves are Class 1E powered, and the controls are installed on seismic Category I panels located in the control room for ready operator access to remotely close the valves when required.

During normal plant operation, in case of a pipe break downstream of any one of the remotely operated manual valves, radiation monitors in the turbine enclosure exhaust will detect radiation and alert the operator in the control room. Temperature elements will also show an increase in temperature.

Each of the three remotely operated manual valves in question is downstream of the MSIVs which automatically close in the event of a large pipe break in the main steam line.

Following a Design Basis Large Break LOCA, the operator can manually align the MSIV Leakage Alternate Drain Pathway, as described in section 6.7. This alignment includes the shutting of each three remotely operated manual valves in question.

Even assuming the unlikely event of a pipe break downstream of any one of these remotely operated manual valves coincident with a LOCA, during the time period before the MSIV Leakage Alternate Drain Pathway is aligned, the radiation doses are well below the values of 10CFR50.67. The activity levels in the residual steam would be comparable to normal operation activity levels. Core activity would not be transported past the already closed MSIVs due to the transport delay time of the residual steam and water.

Consistent with Regulatory Guide 1.26, the turbine bypass valve chest is designed to Quality Group D. In accordance with the Regulatory Guide 1.29 for the turbine stop valve, the turbine bypass valve chest is not designed to seismic Category I requirements.

d. Paragraph C.3 of Regulatory Guide 1.29 recommends seismic Category I design requirements be extended "to the first seismic restraint beyond the defined boundaries." Since seismic analysis of a piping system required division of the system into discrete segments terminated by fixed points, this means that the seismic design cannot be terminated at a seismic restraint, but is extended to the first point in the system which can be treated as an anchor to the plant structure. In addition, paragraph C.4 of Regulatory Guide 1.29 states that the pertinent quality assurance requirement of 10CFR50, Appendix B, should be applied to the safety requirements of such items. Both these guidelines are considered to be met adequately by applying the following practices to such items:

1. Design and design control for such items are carried out in the same manner as that for items directly important to safety. This includes the performance of appropriate design reviews.

2. Field audits are performed by representatives of the originating design group to ensure that the final installation of such items is in accordance with documents that formed the basis for the seismic analysis of the items.

3. Such items are not included in the Q-List.

3.2.2 SYSTEM QUALITY GROUP CLASSIFICATIONS

GDC 1 of 10CFR50, Appendix A, requires that structures, systems and components important to safety be designed, fabricated, erected, and tested to quality standards commensurate with their importance to safety. Components of the RCPB meet the requirements of Class 1 components of the ASME B&PV Code, Section III, or equivalent quality standards, as required by 10CFR50.55a. Regulatory Guide 1.26 (Rev 3), describes a quality classification system that may be used to determine applicable standards for other components in nuclear power plants. Quality group classifications are assigned to systems and components in accordance with the reliance placed on these systems to:

a. Prevent, or mitigate the consequences of, accidents and malfunctions originating within the RCPB

b. Permit shutdown or the reactor, and maintain it in the safe shutdown condition

c. Contain radioactive material

A tabulation of quality group classification for each component so defined is shown in Table 3.2-1 under the heading "Quality Group Classification." The applicable codes and standards of each quality group, as described by Regulatory Guide 1.26, are given in Tables 3.2-2 and 3.2-3. The locations of these components, and the quality group classification of the piping, valves, and interfaces between components of different classifications, are indicated on the system piping and instrumentation diagrams in the pertinent section of the UFSAR. A cross reference of system to former UFSAR figure number is provided in Section 1.7.

System quality group classifications, and design and fabrication requirements as indicated in Table 3.2-1, meet the guidelines of Regulatory Guide 1.26, except as noted below.

The LGS design is based on quality group commitments made before Regulatory Guide 1.26 was issued, as shown in Table 3.2-3, and in some cases alternate approaches to the guide have been used, as follows:

a. Regarding systems important for reactor shutdown, as discussed in paragraph C.1.b (Quality Group B) of the guide, the CRD system HCUs are classified as "special equipment" by GE because the codes and standard of a quality group are not strictly applicable to the HCUs. A detailed discussion is given in the notes of Table 3.2-1.

b. The Quality Group B classification may terminate on some steam system connected piping at the first valve capable of remote manual closure, rather than at a normally, or automatically, closed valve. Additionally, Quality Group B is applied only to piping 2½ inches in diameter and larger, similar to the guidelines of Regulatory Guide 1.29 for application of seismic Category I classification.

c. Certain components in the normal spent fuel pool cooling system were designed , fabricated, procured, installed, and tested to the requirements of ASME Section III, Class 3, prior to May 1978. After May 1978, system design, fabrication, materials, procurement, installation, and testing are, at a minimum, in accordance with Quality Group D and the intent of Regulatory Guide 1.143 (Rev 1). However, as discussed in Section 3.2.1, backup cooling and makeup sources are provided. These sources are at least Quality Group C, as are the connecting portions of the normal cooling system.

d. The standby diesel generator piping is designed as shown in drawing M-20. Supplementary material certification, design, and examination requirements have been applied to the Seismic Category I off-skid portions of the emergency diesel auxiliary systems to ensure that their quality is essentially equivalent to ASME Section III, Class 3. The technical differences between ANSI B31.1 and ASME Section III, Class 3 are few. The major differences were addressed by supplemental requirements and are listed in Table 3.2-4.

The on-skid piping and components in the diesel generator auxiliary systems were provided in accordance with ASME Section III, Class 3 or manufacturer's standards, as shown in the vendor manual and also drawing M-20. Auxiliary system components were supplied to ASME Section III, Class 3 to the greatest extend practicable at the time of procurement. The referenced drawing indicate that this encompasses most of the equipment within the main process loop of each skid-mounted auxiliary system.

All skid-mounted components, regardless of design code, have been designed to withstand seismic accelerations (Seismic Category I) as well as normal diesel operating loads. Portions of the Air Start System, which do not have a safety related function, have been downgraded to Seismic Category IIA as shown on drawing M-20. Motors Associated with stand-by components such as the Oil Pre-Lube Pump, The Oil Circulating Pump and the Jacket Water Keep Warm Pump, have been reclassified as non-safety related. The associated pumps are considered safety related (passive) since they are required for pressure boundary integrity only. These non-safety related components will have mountings that are calssified as seismic Category IIA. Each assembled diesel generator skid was subjected to a series of operating tests including load acceptance and rejection, air start capacity, variable load, overspeed, 300 hr rating, normal operating, and contract acceptance tests. In addition, to the operating tests, the following specific component tests were conducted:

1. Fuel Oil Injectors - These were functionally tested and calibrated to deliver a metered amount of fuel to the combustion chamber. Each injector has a

unique serial number which permits traceability to its shop test and calibration records.

2. Flexible Hose Assemblies and Pipe Coupling Connectors - These components (drawing M-20) are manufacturer unique designs used to ensure flexibility in the piping systems and are not supplied to ASME Section III, Class 3. However, these components were fabricated in accordance with engineering specifications, and each component was hydrotested to in excess of 1.25 times design pressure.

3. ASME Section III, Class 3 and non-ASME Section III, Class 3 Piping - All piping systems were hydrostatically tested to 1.5 times the design pressure regardless of design code, except for the on-skid starting air system piping which was pneumatically tested to 1.25 times the design pressure. These tests are documented in the shop test records.

All diesel components were supplied or manufactured in accordance with the following vendor's quality control standards:

1. Subsupplied Components - Design and procurement controls were used in the procurement of all subsupplied components. These components were purchased to detailed engineering specifications and drawings. Upon receipt, all components (or a representative sample) were inspected against the specification, drawings, and purchase order requirements, and affixed with appropriate tags to be removed at the point of use in the manufacturing cycle. Periodic reviews of subsupplier performance and audits or vendor records were conducted to ensure that the quality of the items provided remained acceptable.

   Typical examples of subsupplied items procured under this program are as follows:

   - Air start solenoid valves, filters, strainers, and compressors
   - Inlet and exhaust expansion joints and the exhaust silencer
   - Motor-driven fuel oil, jacket water, and lube oil pumps
   - Combustion air coolers, lube oil and jacket water standby heaters
   - Fuel and lube oil strainers
   - Jacket water, air cooler coolant, and lube oil thermostatic bypass valves
   - Electrical and pneumatic instrumentation and controls.

2. Supplier Manufactured Components - Equipment and components designed and manufactured by the diesel generator vendor were designed in accordance with written procedures for design control which required appropriate reviews and approvals of all detail drawings, specifications,

procedures, and instructions. During the manufacturing process, visual inspection, dimensional checks, final inspections, and customer hold and witness points were used. Also, procedures were developed and implemented for rejection of components, recall of materials, and internal audits. Typical examples of supplier manufactured components incorporating the above standards include the following:

- Air admission check valves, pilot valves, and air start distributors
- Engine-mounted intake and exhaust air piping, scavenger air receivers, evacuation system components
- Fuel injectors, dirty fuel drip tank, fuel and oil drip pan
- Governor control linkage.

The above standards are fully documented in the diesel generator suppliers' Quality Assurance Plan, which addresses the 18 criteria contained with 10CFR50, Appendix B. It is invoked upon all diesel generators supplied by this manufacturer for use in nuclear plants and has received the approval of the appropriate operating utilities and the NRC.

The above described controls were invoked on all piping and components supplied by the manufacturer. In addition to these requirements, the applicant invoked supplemental quality assurance requirements on the following selected diesel generator and auxiliary system components:

1. Subsupplier Items:
    - Generators, generator controls, and static exciters
    - Starting air receivers and inlet valves
    - Jacket water, lube oil, and air cooler coolant heat exchangers
    - Lube oil strainers and filters
    - Governors
    - Inlet air filters
    - Turbochargers
    - Fuel oil day tank
    - Lube oil storage tank
    - Jacket water expansion tank.
2. Diesel Generator Supplier Manufactured Items:

- Engine blocks
- Subbase oil pan
- Cylinder liners
- Exhaust belts
- Pistons and piston inserts
- Crankshafts and connecting rods
- Vertical drive assembly
- Scavenging air blower gears, housings, impellers
- Engine driven lube oil, jacket water, and air cooler coolant pumps
- Jacket liners
- Skid-mounted piping and valves provided to ASME Section III, Class 3.

The additional quality assurance requirements invoked by the applicant include: (1) periodic documented subsupplier audits (including plant visits), (2) review and approval of subsupplier QA programs and manuals, (3) test and inspection audits, (4) calibration of test gauges before and after use, and (5) control of calibration records and acceptance devices.
With the imposition of the above design, manufacturing, and testing controls, the on-skid and off-skid piping and components have been made to be equivalent to Quality Group C.

e. The chilled water piping system for the control structure chilled water system is designed to ANSI B31.1 with supplementary material certification and design requirements to ensure that the quality is essentially equivalent to ASME Section III, Class 3. The technical differences between ANSI B31.1 and ASME Section III, Class 3 are few. The major differences were addressed by supplemental requirements and are listed in Table 3.2-5.

f. Instrument tubing downstream of the containment isolation valve of instrument lines connected to the RCPB is Quality Group D for instruments that are "passive": (i.e., do not actuate safety systems), rather than Quality Group B or C as discussed in paragraphs 1.e and 2.c of the regulatory guide. This is based on considerations given in Regulatory Guide 1.11 for instrument lines penetrating containment and having two restriction devices.

g. The piping between the two containment isolation valves, and the outboard isolation valves, on the drywell chilled water system are the equivalent of Quality Group B although they were originally designed and constructed as ANSI B31.1. Equivalency has been assured through the imposition of supplemental design, fabrication, and testing requirements:

- Piping material properties are essentially equivalent to the requirements of ASME Section III, Class 2

- Seismic and hydrodynamic accelerations were considered in design and analysis

- Design temperatures and pressures are greater than design values for the primary containment

- Documented quality control inspections were performed (by trained and qualified inspectors) on piping installation, welds, valves, and hangers

- The installation is of fully welded construction except for the outboard isolation valves, which are flanged

- A hydrostatic test was performed at a pressure 3 times greater than containment design pressure

These provisions meet or exceed the commitments made in PSAR Appendix A and PSAR Figure A.2.1.

h. The water chillers in the control room HVAC system are designed and fabricated in accordance with ASME Section VIII, Division I requirements, with the exception of the ASME Section III, Class 3 condenser. The condenser is connected to the ESW system. Supplementary material certification and design requirements have been applied to the ASME Section VIII portions of the chillers to ensure that their quality is essentially equivalent to ASME Section III, Class 3 at the time of purchase. The chillers were designed to seismic Category I requirements and fabricated under an approved quality assurance program. The major similarities and differences are listed in Table 3.2-6 along with the supplementary requirements.

i. The control structure chilled water pumps are designed and fabricated in accordance with the manufacturer's standards with supplementary requirements applied. The pumps are designed to seismic Category I requirements and fabricated under an approved quality assurance program used for ASME Section III, Class 3 pumps. The pressure-retaining materials used are approved for use in ASME Section III, Class 3 pumps and were supplied with CMTRs. The pumps were hydrotested at a pressure greater than that required by the code. The electrical components were environmentally qualified. Therefore, the quality of the pumps is essentially equivalent to ASME Section III, Class 3.

j. The seismic Category I control structure chilled water cooling coils in safety-related air handling units are fabricated from ASME Section III, Class 3 approved materials, with CMTRs, for the pressure-retaining parts. The coils are fabricated in accordance with the same quality assurance program using equivalent processes, are tested to the same procedures, and are of the same design as the ASME Section III, Class 3 ECCS unit cooler cooling coils, but are not N-stamped.

k. The seismic Category I SLCS tank was designed to API-650 standards and supplementary ASME Section III, Class C testing and examination requirements. The tank meets key later requirements of NC-3800, ASME Section III, Class 2, and is judged to be of equivalent quality.

l. All three of the Unit 1 and one of the Unit 2 SLCS pump discharge accumulator vessels were designed to ASME Section VIII, Division 1 requirements. These weldless vessels, except for stamping, meet the code requirements of ASME Section III, Class C cited in Table 3.2-3.

m. The HPCI turbine exhaust line globe stop-check valves meet the material, design, fabrication, inspection, and testing requirements of the 1968 Draft ASME Code for Pumps and Valves indicated in Table 3.2-3 and the nondestructive inspection requirements of ASME Section III. They are, therefore, equivalent to Quality Group B. These valves were manufactured, however, to ANSI B31.1 (1967), ANSI B16.5 and MSS-SP-66 requirements.

n. The CRD system piping between the HCUs and the containment isolation boundary valves was originally designed and constructed according to ANSI B31.1. This piping has been reanalyzed and reevaluated as part of the modification which provided new isolation valves. This piping is now considered to be equivalent to ASME Section III, Class 2. The Upgrade Program included the following:

   - The design requirements of the piping and the applicable stress calculations have been reviewed and found adequate to meet the requirements of ASME III, Class 2 and Seismic Category I.

   - The installation rules and processes have been reviewed and found adequate to meet the intent of ASME III, Class 2.

   - The inspection and testing procedures and acceptance criteria have been reviewed and found adequate to meet the requirements of ASME III, Class 2.

   - The documentation required by the specifications and programs satisfies the intent of ASME III, Class 2 to provide written evidence of an acceptable level of quality. Engineering review of the documents has been found to be satisfactory.

o. The Unit 1 Corrosion Monitoring Piping and Components for the RHR Heat Exchanger IBE205 (RHR HX "B") connected to the RHRSW system was installed using ANSI B31.1 materials. This piping and components are considered to be equivalent to ASME Section III, Class 3. This consideration is based on the following:

   - The design requirements of the piping and the applicable stress calculations have been performed in accordance with the requirements of ASME III, Class 3 and Seismic Category I.

   - This equipment and piping was installed in accordance with the requirements of ASME III, Class 3.

- The inspection and testing procedures and acceptance criteria have been done in accordance with the requirements of ASME III, Class 3.

- The documentation required by the specifications and programs satisfies the intent of ASME III, Class 3 to provide written evidence of an acceptable level of quality. Engineering review of the documents has been found to be satisfactory.

3.2.3 QUALITY ASSURANCE

Those structures, components, and systems necessary to ensure:

a. the integrity of the RCPB,

b. the capability to shut down the reactor and maintain it in a safe shutdown condition,

c. the capability to prevent, or mitigate the consequences of, accidents that could result in potential offsite exposures in excess of the values of 10CFR50.67.

are classified as Q-listed, require conformance to the applicable quality assurance requirements of 10CFR50, Appendix B and are summarized in Table 3.2-1 under the heading, "Q-List". Table 3.2-1 (LGS Design Criteria Summary) is intended, in part, to provide identification of safety-related structures, systems, and components. The LGS Q-List is not part of the UFSAR; it is a controlled QA program document that serves to identify structures, systems and components requiring compliance with 10CFR50, Appendix B.

Quality assurance during construction is discussed in the document "Limerick Generating Station Units 1 and 2; Summary Description of the Quality Assurance Program for Design and Construction," referenced in Section 17.1. The Quality Assurance Program during the Operational Phase is described in Section 17.2.

The Q-listed boundaries for piping systems terminate at the outermost containment or system isolation valve. The piping downstream of this boundary is not required to ensure items a, b, or c above and is therefore not required to be Q-listed. However, in order that failure of the non Q-listed piping not affect the Q-listed piping or the isolation valves, the non Q-listed piping is designed to seismic Category I requirements up to and including the first point in the system that can be treated as an anchor to the plant structure except as indicated in part c of the discussion on Regulatory Guide 1.29 in Section 3.2.1. Stress analysis, support design, and design control for this non Q-listed piping, classified as seismic Category IIA, is carried out in the same manner as it is for Q-listed, seismic Category I items.

The pertinent quality assurance requirements of 10CFR50, Appendix B, are considered to be adequately met for the seismic Category IIA piping as indicated in Section 3.2.1, part d.

The design control and construction control practices that were used during the design and construction phase are also applied during the operations phase to ensure that the same stringent requirements are maintained for any changes to seismic Category IIA piping. In addition, seismic Category IIA piping and supports which are included in seismic Category I stress calculations will be included as part of the ISI program, to the extent that repairs and replacements to the IIA supports are performed in accordance with the ASME Section XI Repair and Replacement

Program. The ISI program is included in the operations phase of the Nuclear Quality Assurance Plan.

Table 3.2-1

LGS DESIGN CRITERIA SUMMARY

SYSTEM/COMPONENT[40]		UFSAR SECTION	SOURCE OF SUPPLY[1]	LOCA-TION[2]	QUALITY GROUP CLASSI-FICATION[3]	PRINCIPAL CODES AND STANDARDS[4]	SEISMIC CATEGORY[5]	Q-LIST[6]	COMMENTS
I	NSSS								
A.	Reactor System	4,5							
	1. Reactor vessel		GE	C	A	III-1	I	Y	(7)
	2. Reactor vessel support skirt		GE	C	-	III-1	I	Y	(49)
	3. Reactor vessel appurtenances, pressure-retaining portions		GE	C	A	III-1	I	Y	
	4. CRD housing supports		GE	C	-	MF STD	I	Y	(50)
	5. Reactor internals, ESFs		GE	C	-	MF STD	I	Y	
	6. Reactor internals, other,		GE	C	-	MF STD	II	Y/N	(8)(62)
	7. Control rods		GE	C	-	MF STD	I	Y	
	8. Core support structure		GE	C	-	MF STD	I	Y	(72)
	9. Power range detector hardware		GE	C	A	III-1	I	Y	
	10. Fuel assembles		GE	C	-	MF STD	I	Y	
B.	Nuclear Boiler System	4,5							(87)
	1. Vessels, level instrumentation condensing chambers		GE	C	A	III-1	I	Y	(9)
	2. Vessels, air accumulators (except in item 8 below)		P	C	C	III-3	I	Y	
	3. Piping, relief valve discharge		P	C	C	III-3	I	Y	(48)
	4. Piping and valves, RCPB		GE/P	C,R	A	III-1	I	Y	(7)(9)(48)
	5. Mechanical components, instrumentation with safety function		GE	C	B	MF STD	I	Y	(11)
	6. Electrical modules, with safety function		GE	C,R,CS	-	IEEE 323, 344	I	Y	(11),(12)
	7. Quenchers and quencher supports		P	C	C	III-3/III-NF	I	Y	
	8. Air accumulators on non-ADS SRVs		P	C	D	B31.1	IIA	N	
C.	Control Rod Drive Hydraulic System	4.6.1							Class 1
	1. CRDs		GE	C	-	III-1	I	Y	Appurtenance
	2. HCU including scram accumulators		GE	R	-	MF STD	I	Y	(14)
	3. Piping and valves, insert and withdraw lines, and scram discharge volume lines		P/GE	R,C	B	III-2	I	Y	(15)
	4. Piping and valves, other		P	R	D	B31.1	I/II/IIA	Y/N	(67)
	5. Pumps		GE	T	D	VIII-1	II	N	
	6. Electrical modules, with safety function		GE	C,R,CS	-	IEEE 323, 344	I	Y	(11)(12)

Table 3.2-1 (Cont'd)

SYSTEM/COMPONENT[40]	UFSAR SECTION	SOURCE OF SUPPLY[1]	LOCATION[2]	QUALITY GROUP CLASSIFICATION[3]	PRINCIPAL CODES AND STANDARDS[4]	SEISMIC CATEGORY[5]	Q-LIST[6]	COMMENTS
D. Recirculation System	5							
1. Piping		GE	C	A	III-1	I	Y	(7)(9)(48)
2. Valves		GE	C	A	III-1	I	Y	(7)(9)
3. Pumps		GE	C	A	III-1	I	Y	(7)
4. Motors, pump		GE	C	-	NEMA MG-1	I	N	
5. Electrical modules, with safety function		P/GE	C,R,CS	-	IEEE 323, 344	I	Y	(11)(12)
6. ASD Piping & Valves		P	T	D	ANSI B31.1	II	N	
7. ASD Heat Exchangers		P	T	D	VIII-1	II	N	
E. Reactor Water Cleanup System	5.4.8							(86)
1. Filter/demineralizer vessels		GE	R	C	III-3	II	N	
2. Heat exchangers		GE	R	C	III-C/ TEMA R	IIA	N	(71)
3. Piping and valves, RCPB		P	R	A	III-1	I	Y	(7)(9)(16)(48)
4. Piping and valves, connections to RCIC, and feedwater		P	R	B	III-2	I	Y	(48)(16)
4A. Piping and valves, connection to pump suction outboard containment isolation valve		P	R	B	III-2	IIA	N	(16)
5. Piping and valves, other		P	R	C	III-3	II/IIA	N	(16)
6. Pumps								
A pump		P	R	C	III-3	IIA	N	
B & C pumps		GE	R	C	III-3	II	N	
7. Mechanical components		P	R	-	MF STD	II	N	(11)
F. Traversing Incore Probe System	7.7.1							
1. Drive mechanism, chamber shield		GE	R	-	MF STD	II	N	
2. Indexing mechanism		GE	C	-	MF STD	II	N	
3. Tubing, TIP drive		GE	C,R	B	MF STD	IIA	N	
4. Valves and tubing, TIP drive isolation		GE	R	B	MF STD	I	Y	
5. Purge equipment		GE	R	-	MF STD	II	N	
6. Piping, TIP purge		P	C,R	D	B31.1	IIA	N	
7. Valves and piping, TIP purge isolation		P	C,R	B	III-2	I	Y	

Table 3.2-1 (Cont'd)

SYSTEM/COMPONENT[40]		UFSAR SECTION	SOURCE OF SUPPLY[1]	LOCATION[2]	QUALITY GROUP CLASSIFICATION[3]	PRINCIPAL CODES AND STANDARDS[4]	SEISMIC CATEGORY[5]	Q-LIST[6]	COMMENTS
II	ENGINEERED SAFETY FEATURES								(61)
A.	Reactor Core Isolation Cooling System	5.4.6							(86)
1.	RCIC turbine		GE	R	-	MF STD	I	Y	(17)
2.	RCIC barometric condenser		GE	R	D	B31.1	II	N	
3.	Piping and valves, RCPB		P	C	A	III-1	I	Y	(7)(9)(48)
4.	Piping within outermost containment isolation valves, discharges to suppression pool		P	C	D	B31.1	I	Y	(48)
5.	Piping and valves, other safety-related		P	R	B	III-2	I	Y	(9)(48)
6.	Portion of piping for RCIC turbine drains		P	R	D	B31.1	I	Y	(55)
7.	Pumps, RCIC condensate and condenser vacuum		GE	R	D	MF STD	II	N	
8.	Pump, RCIC		GE	R	B	III-2	I	Y	
9.	Electrical modules, with safety function		GE	R,CS	-	IEEE 323, 344, 279	I	Y	(11)(12)
10.	Out of service piping connected to RHR		P	R	-	B31.1	IIA	N	(84)
B.	Residual Heat Removal System	5.4.7							(86)
1.	Heat exchangers, primary (process) side		GE	R	B	III-2/TEMA C	I	Y	
2.	Heat exchangers, secondary (service water) side		GE	R	C	VIII-1/ TEMA C	I	Y	
3.	Piping, RCPB		P	C	A	III-1	I	Y	(7)(9)(48)
4.	Piping, containment spray line (inside containment)		P	C	B	III-3	I	Y	(48)(58)
5.	Piping and valves, other safety-related		P	R	B	III-2	I	Y	(48)
6.	Valves, isolation		GE/P	C,R	A,B	III-1,2	I	Y	(7)
7.	Pumps		GE	R	B	III-2	I	Y	
8.	Motors, pump		GE	R	-	NEMA MG-1	I	Y	
9.	Mechanical components		GE	R	B	MF STD	I	Y	(11)
10.	Electrical modules, with safety function		P/GE	R,CS	-	IEEE 323, 344, 279	I	Y	(11)(12)
11.	Containment spray nozzles		P	C	-	MF STD	I	Y	(59)
12.	Out of service piping connected to RCIC		P	R	-	B31.1	IIA	N	(84)

Table 3.2-1 (Cont'd)

SYSTEM/COMPONENT[40]	UFSAR SECTION	SOURCE OF SUPPLY[1]	LOCA-TION[2]	QUALITY GROUP CLASSI-FICATION[3]	PRINCIPAL CODES AND STANDARDS[4]	SEISMIC CATEGORY[5]	Q-LIST[6]	COMMENTS
C. Core Spray System	6.3							(86)
1. Piping and valves, RCPB		P/GE	C	A	III-1	I	Y	(7)(9)(48)
2. Piping and valves, other safety-related		P	R	B	III-2	I	Y	(9)(48)
3. Pumps		GE	R	B	III-2	I	Y	
4. Motors, pump		GE	R	-	NEMA MG-1	I	Y	
5. Electrical modules, with safety function	GE	R,CS	-	IEEE 323	I	Y 344, 279	(11)(12)	
D. High Pressure Coolant Injection System	6.3							(86)
1. HPCI turbine		GE	R	-	MF STD	I	Y	(17)
2. Piping and valves, RCPB		P	C	A	III-1	I	Y	(7)(9)(48)
3. Piping and valves, other safety-related		P/GE	R	B	III-2	I	Y	(9)(48)(70)
4. Piping, return test line to CST beyond second isolation valve		P	R,AB, RW,O	B	III-2	IIA/II	N	
5. Piping, remainder		P	AB, RW,O	D	B31.1	II	N	
6. Pumps, HPCI and booster		GE	R	B	III-2	I	Y	
7. Electrical modules, with safety function		GE	R,CS	-	IEEE 323, 344, 279	I	Y	(11)(12)
8. HPCI barometric condenser		GE	R	D	MF STD	II	N	
9. Pumps, HPCI condensate and condenser vacuum		GE	R	D	MF STD	II	N	
10. Piping within outermost containment isolation valves, discharges to suppression pool		P	C	D	B31.1	I	Y	(48)
11. Portion of piping for HPCI turbine drains		P	R	D	B31.1	I	Y	(55)
E. Standby Liquid Control System	9.3.5							(86)
1. SLCS tank		GE	R	B	API 650	I	Y	(68)
2. Test tank		P	R	D	API 650	IIA	N	
3. Piping and valves, RCPB		P	C	A	III-1	I	Y	(7)(48)
4. Piping and valves, other safety-related		P	R	B	III-2	I	Y	(9)(48)(69)
5. Piping, service and drain		P	R	D	B31.1	I/IIA	N	
6. Pumps		GE,P	R	B	III-2	I	Y	
7. Motors, pump		GE,P	R	-	NEMA MG-1	I	Y	
8. Electrical modules, with safety function		GE	C,R,CS	-	IEEE 323, 344, 279	I	Y	(11)(12)

Table 3.2-1 (Cont'd)

SYSTEM/COMPONENT(40)		UFSAR SECTION	SOURCE OF SUPPLY(1)	LOCA-TION(2)	QUALITY GROUP CLASSI-FICATION(3)	PRINCIPAL CODES AND STANDARDS(4)	SEISMIC CATEGORY(5)	Q-LIST(6)	COMMENTS
F.	Section Deleted								
III	FUEL STORAGE AND HANDLING; REACTOR VESSEL SERVICING								
A.	Storage Equipment	9.1.1, 9.1.2							
	1. Spent fuel storage racks (also used for new fuel)		P	R	-	MF STD	I	Y	
	2. Channel storage racks		GE	R	-	MF STD	II	N	
	3. In-vessel racks		GE	R	-	MF STD	I	Y	
	4. Defective fuel storage containers		GE	R	-	MF STD	I	Y	
B.	Fuel Pool Cooling and Cleanup System	9.1.3							
	1. Heat exchangers		P	R	C	III-3/TEMA C	IIA	N	(18)
	2. Skimmer surge tanks		P	R	C	AWWA D-100	I	Y	
	3. Filter/demineralizer vessels		P	RW	C	III-3	II	N	(18)
	4. Resin and precoat tanks		P	RW	D	MF STD	II	N	(18)
	5. Piping and valves, cooling loop		P	R	C	B31.1	IIA	N	(8)(18)(19)(52)
	6. Piping and valves, RHR intertie		P	R	B	III-2	I	Y	(48)
	7. Piping and valves, ESW makeup		P	R,CS	C	III-3	I	Y	(48)
	8. Piping and valves, other		P	R	D	B31.1	II	N	(18)
	9. Pumps		P	R	C	III-3	IIA	N	(18)
C.	Fuel Servicing Equipment	9.1.4							
	1. Fuel preparation machine		GE	R	-	MF STD	I	Y	
	2. New fuel inspection stand		GE	R	-	MF STD	II	N	
	3. General purpose grapple		GE	R	-	MF STD	I	Y	
	4. Jib cranes		GE	R	-	B30.11/ B30.16	I	Y	
D.	Refueling Equipment	9.1.4							
	1. Refueling platforms		GE	R	-	MF STD	I	Y	(20)
	2. Fuel grapples		GE	R	-	MF STD	I	Y	(20)
	3. Fuel grapples (Main Hoist)		PaR	R	-	MF STD	IIA	N	(20)
	4. Refueling Mast		GE	R	-	MF STD	IIA	N	(20)

Table 3.2-1 (Cont'd)

SYSTEM/COMPONENT[40]		UFSAR SECTION	SOURCE OF SUPPLY[1]	LOCA-TION[2]	QUALITY GROUP CLASSI-FICATION[3]	PRINCIPAL CODES AND STANDARDS[4]	SEISMIC CATEGORY[5]	Q-LIST[6]	COMMENTS
E.	Under Reactor Vessel Service Equipment	9.1.4							
	1. Equipment handling platform		GE	C	-	MF STD	II	N	
	2. CRD handling equipment		GE	C	-	MF STD	II	N	
F.	Reactor Vessel Servicing Equipment	9.1.4, 9.1.5							
	1. Steam line plugs		GE	R	-	MF STD	I	Y	
	2. Dryer and separator sling		GE	R	-	MF STD	I	Y	
	3. RPV head strong-back		GE	R	-	MF STD	I	Y	(85)
	4. Deleted								
	5. Control rod grapple		GE	R	-	MF STD	I	Y	
	6. Reactor enclosure crane		P	R	-	CMAA 70	I	Y	(20)
	7. Combined Grapple, CRB/FSP		P	R	-	MF STD	I	Y	
	8. Service Pole Caddy		GE	R		MF STD	IIA	N	
	9. Fuel Floor Auxiliary Platform		P	R	-	CMAA 70	IIA	N	
	10. Reactor Cavity Work Platform Assembly		P	R	-	AISC	IIA	N	
IV	RADIOACTIVE WASTE MANAGEMENT	11							
A.	Liquid Waste Management Systems	11.2							(86)
	1. Atmospheric tanks		P	RW	D	API 650	II	N	(18)
	2. Filter vessels		P	RW	D	III-3	II	N	(18)
	3. Demineralizer vessels		P	RW	D	III-3	II	N	(18)
	4. Evaporator, complete system (abandoned)		P	RW	D	III-3	II	N	(18)
	5. Laundry drain filter		P	RW	D	III-3	II	N	(18)
	6. Piping and valves		P	RW	D	III-3/ B31.1	II	N	(18)
	7. Pumps, centrifugal		P	RW	D	III-3	II	N	(18)
B.	Gaseous Waste Management System	11.3							
	1. Heat exchanger		P	RW	D	VIII-1	II	N	(18)
	2. Pressure vessels		P	RW	D	VIII-1	II	N	(18)
	3. Atmospheric tanks		P	RW	D	API 650	II	N	(18)
	4. 0-15 psig tanks		P	RW	D	API 620	II	N	(18)
	5. Recombiner		P	CS	D	III-3	II	N	(18)
	6. Preheater		P	CS	D	III-3	II	N	(18)
	7. Aftercondenser		P	CS	D	VIII-1	II	N	(18)
	8. Refrigeration equipment, piping		P	RW	D	B9.1/B31.5	II	N	(18)
	9. Refrigeration equipment, other		P	RW	D	VIII-1/ TEMA C	II	N	(18)
	10. Piping and valves		P	RW	D	III-3/ B31.1	II	N	(18)
	11. Pumps/compressors		P	RW	D	MF STD	II	N	(18)

Table 3.2-1 (Cont'd)

SYSTEM/COMPONENT(40)	UFSAR SECTION	SOURCE OF SUPPLY(1)	LOCA-TION(2)	QUALITY GROUP CLASSI-FICATION(3)	PRINCIPAL CODES AND STANDARDS(4)	SEISMIC CATEGORY(5)	Q-LIST(6)	COMMENTS
C. Solid Waste Management System	11.4							
1. Tanks, atmospheric		P	RW,T	D	API 650	II	N	(18)
2. Phase separators		P	RW	D	API 650	II	N	(18)
3. Waste containers		P	RW	-	MF STD	II	N	(18)
4. Centrifuges		P	RW	-	MF STD	II	N	(18)
5. Piping and valves		P	RW	D	III-3/ B31.1	II	N	(18)
6. Pumps		P	RW	D	III-3	II	N	(18)
V WATER SYSTEMS								
A. Service Water System	9.2.1							(87)
1. Heat exchangers		P	R,T	D	VIII-1/TEMA C	II	N	
2. Piping and valves		P	R,CW,T	D	B31.1	II/IIA	N	
3. Pumps		P	R,CW	D	MF STD	II	N	
B. Emergency Service Water System	9.2.2							(86)
1. Piping and valves		P	R,T, O,S	C/D	III-3/ B31.1	I/IIA	Y/N	(48)-seismic Category I only
2. Pumps		P	S	C	III-3	I	Y	
3. Motors, pump		P	S	-	IEEE 323, 344	I	Y	(12)
4. Electrical modules, with safety function		P	O,S, CS,R	-	IEEE 323, 344, 279	I	Y	(11)(12)
C. RHR Service Water System	9.2.3							(86)
1. Piping and valves		P	R,S,O	C/D	III-3/ B31.1	I/IIA/II	Y/N	(48)-seismic Category I only
2. Pumps		P	S	C	III-3	I	Y	
3. Motors, pump		P	S	-	IEEE 323, 344	I	Y	(12)
4. Electrical modules, with safety function		P	O,S, CS,R	-	IEEE 323, 344, 279	I	Y	(11)(12)

Table 3.2-1 (Cont'd)

SYSTEM/COMPONENT(40)	UFSAR SECTION	SOURCE OF SUPPLY(1)	LOCA-TION(2)	QUALITY GROUP CLASSI-FICATION(3)	PRINCIPAL CODES AND STANDARDS(4)	SEISMIC CATEGORY(5)	Q-LIST(6)	COMMENTS
D. Reactor Enclosure Cooling Water System	9.2.8							(87)
1. Tanks, pressure		P	R	D	VIII-1	IIA	N	
2. Tanks, atmospheric		P	R	D	API 650	IIA	N	
3. Cooler, reactor enclosure equipment drain sump		P	R	D	B31.1	IIA	N	
4. Heat exchangers, other		P	R	D	VIII-1/ TEMA C	II	N	
5. Piping and valves forming part of containment boundary		P	R,C	B	III-2	I	Y	(48)
6. Piping to recirculation pumps (Unit 1)		P	R	C	III-3	IIA	N	
(Unit 2)		P	R	D	B31.1	IIA	N	
7. Piping and valves, other		P	R	D	B31.1	IIA	N	
8. Pumps		P	R	D	MF STD	IIA	N	
E. Turbine Enclosure Cooling Water System	9.2.9							
1. Heat exchangers		P	T	D	VIII-1/ TEMA C	II	N	
2. Tanks, atmospheric		P	T	D	API 650	II	N	
3. Piping and valves		P	T	D	B31.1	II	N	
4. Pumps		P	T	D	MF STD	II	N	
F. Circulating Water System	10.4.5							
1. Condenser		P	T	D	HEI	II	N	
2. Cooling tower		P	O	-	MF STD	II	N	
3. Piping		P	O,CW,T, SP, PP	D	B31.1	II	N	
4. Valves		P	CW,SP, PP,T	D	B31.1	II	N	
5. Pumps		P	CW,SP, PP	D	MF STD/ HYDI	II	N	

Table 3.2-1 (Cont'd)

SYSTEM/COMPONENT(40)			UFSAR SECTION	SOURCE OF SUPPLY(1)	LOCA-TION(2)	QUALITY GROUP CLASSI-FICATION(3)	PRINCIPAL CODES AND STANDARDS(4)	SEISMIC CATEGORY(5)	Q-LIST(6)	COMMENTS
VI	DIESEL GENERATOR SYSTEM		9.5.4,9.5.5, 9.5.6, 9.5.7							(86)
	1.	Diesel generators		P	G	C	IEEE 387	I	Y	(47)
	2.	Fuel oil storage and transfer		P	G,O	C	III-3 B31.1, MF STDS	I	Y	(47)(22)
	3.	Lubrication system		P	G	C	III-3 IV, VIII B31.1, MF STD	I	Y	(47)(83)
	4.	Air start system		P	G	C	III-3 VIII B31.1 MF STD	I	Y	(47)(81)
	5.	Cooling water systems		P	G	C	III-3 B31.1 MF STD	I	Y	(47)(83)
	6.	Air intake and exhaust system		P	G	C	III-3 B31.1 MF STD	I	Y	(47)(44)(83)
	7.	Electrical modules, with safety function		P	G,CS	C	IEEE 323, 344, 279	I	Y	(11)(12)
VII	HEATING, VENTILATING, AND AIR CONDITIONING SYSTEMS									
A.	Control Structure									
	1.	Control Room HVAC System	9.4.1.1							(74)(86)
		a. Water chillers (except condenser)		P	CS	C	VIII-1/ IEEE 323, 344	I	Y	(54)
		b. Water chiller condensers		P	CS	C	III-3	I	Y	
		c. Chilled water pumps		P	CS	C	MF STD IEEE 323, 344	I	Y	(56)
		d. Chilled water piping and valves		P	CS	C	B31.1/IEEE 323	I	Y	(19)(48)(53)
		e. Fans		P	CS	-	AMCA	I	Y	
		f. Motors, fan		P	CS	-	NEMA MG-1/ IEEE 323, 344	I	Y	
		g. Coils, cooling		P	CS	C	ARI	I	Y	(57)
		h. Coils, electric heating		P	CS	-	NEC	IIA	N	
		i. Duct-work and registers		P	CS	-	AISI/AWS	I	Y	(23)
		j. Dampers, isolation and control		P	CS	-	AMCA	I	Y	(25)

Table 3.2-1 (Cont'd)

SYSTEM/COMPONENT[40]	UFSAR SECTION	SOURCE OF SUPPLY[1]	LOCA-TION[2]	QUALITY GROUP CLASSI-FICATION[3]	PRINCIPAL CODES AND STANDARDS[4]	SEISMIC CATEGORY[5]	Q-LIST[6]	COMMENTS
2. Auxiliary Equipment Room HVAC System	9.4.1.2							(74)(86)
a. Chilled water system		P	CS	C	See Item VII .A.1.a/d	I	Y	(19)(53)
b. Fans		P	CS	-	AMCA	I	Y	
c. Motors, fans		P	CS	-	NEMA MG-1/ IEEE 323, 344	I	Y	
d. Coils, cooling		P	CS	C	ARI	I	Y	(57)
e. Coils, electric heating		P	CS	-	NEC	IIA	N	
f. Duct-work and registers		P	CS	-	AISI/AWS	I	Y	(23)
g. Dampers, isolation and control		P	CS	-	AMCA	I	Y	(25)
3. Emergency Fresh Air Supply System	9.4.1.3							(74)(86)
a. Fans		P	CS	-	AMCA	I	Y	
b. Motors, fans		P	CS	-	NEMA MG-1/ IEEE 323, 344	I	Y	
c. Coils, electric heating		P	CS	-	NEC	I	Y	
d. Duct-work		P	CS	-	AISI/AWS	I	Y	(23)
e. Dampers, isolation and control		P	CS	-	AMCA	I	Y	(25)
f. Prefilters		P	CS	-	ANSI N509	I	Y	(24)
g. HEPA filters		P	CS	-	ANSI N509	I	Y	(24)
h. Charcoal adsorbers		P	CS	-	ANSI N509	I	Y	(24)
4. Cable Spreading/Auxiliary Switchgear Room HVAC	9.4.1.4							
a. Chilled water and steam heating systems		P	T	D	MF STD/ B31.1	II	N	
b. Fans		P	T	-	AMCA	II	N	
c. Coils, cooling		P	T	-	ARI	II	N	
d. Coils, steam heating		P	T	-	MF STD	II	N	
e. Duct-work and registers		P	CS	-	AISI/AWS	IIA	N	(23)
f. Dampers, fire		P	CS	-	UL	IIA	N	(25)

Table 3.2-1 (Cont'd)

SYSTEM/COMPONENT[40]	UFSAR SECTION	SOURCE OF SUPPLY[1]	LOCA-TION[2]	QUALITY GROUP CLASSI-FICATION[3]	PRINCIPAL CODES AND STANDARDS[4]	SEISMIC CATEGORY[5]	Q-LIST[6]	COMMENTS
5. SGTS Equipment Compartment HVAC System	9.4.1.5							(74)(86)
a. Chilled water system		P	CS	C/D	See Item VII .A.1.a/d	I	Y	(19)
b. Air supply system		P	CS	-	See Item VII .A.2.b/f	I	Y	
c. Unit coolers		P	CS	-	AMCA/ IEEE 323, 344	I	Y	
d. Unit heaters, electric		P	CS	-	NEC	IIA	N	
e. Exhaust duct-work and registers		P	CS	-	AISI/AWS	IIA/II	N	(23)
f. Exhaust fans		P	CS	-	AMCA	II	N	
g. Prefilters, exhaust		P	CS	-	ANSI N509	II	N	
h. HEPA filters, exhaust		P	CS	-	ANSI N509	II	N	
i. Charcoal adsorbers, exhaust		P	CS	-	ANSI N509	II	N	
6. Emergency Switchgear, Battery and Inverter Rooms HVAC System	9.4.1.6							(74)(86)
a. Chilled water system		P	CS	C/D	See Item VII .A.1.a/d	I	Y	(19)
b. Recirculation fans		P	CS	-	AMCA/IEEE 323, 344	I	Y	
c. Coils, cooling		P	CS	-	ARI	I	Y	
d. Recirculating duct-work and registers		P	CS	-	AISI/AWS	I	Y	
e. Dampers, isolation		P	CS	-	AMCA	I	Y	(25)
f. Battery rooms normal exhaust fans		P	CS	-	AMCA	II	N	
g. Battery rooms normal exhaust duct-work		P	CS	-	AISI/AWS	IIA/II	N	(23)
B. Reactor Enclosure and Refueling Area								(86)
1. Reactor Enclosure HVAC System (Normal Operation)	9.4.2.1							
a. Chilled water and steam heat equipment		P	R,AB	D	MF STD	II	N	
b. Piping		P	R	D	B31.1	IIA/II	N	
c. Valves, isolation, chilled water to primary containment		P	R	B	III-2	I	Y	
d. Valves, remainder		P	R	D	B31.1	II	N	
e. Fans		P	R	-	AMCA	II	N	
f. Coils, cooling		P	R	-	ARI	II	N	
g. Coils, heating		P	R	-	MF STD	II	N	

Table 3.2-1 (Cont'd)

SYSTEM/COMPONENT(40)	UFSAR SECTION	SOURCE OF SUPPLY(1)	LOCA-TION(2)	QUALITY GROUP CLASSI-FICATION(3)	PRINCIPAL CODES AND STANDARDS(4)	SEISMIC CATEGORY(5)	Q-LIST(6)	COMMENTS
h. Duct-work and registers		P	R	-	AISI/AWS	I/II	Y/N	(23)(26)
i. Dampers, isolation and control		P	R	-	AMCA	I/II	Y/N	(25)(26)
j. Prefilters, exhaust		P	R	-	ANSI N509	II	N	
k. HEPA Filters, exhaust		P	R	-	ANSI N509	II	N	
l. Charcoal adsorbers, exhaust		P	R	-	ANSI N509	II	N	
2. Refueling Floor HVAC System (Normal Operation)	9.4.2.1							
a. Chilled water and steam heating equipment		P	R	D	MF STD	II	N	
b. Piping and valves		P	R	D	B31.1	II	N	
c. Fans		P	R	-	AMCA	II	N	
d. Coils, cooling		P	R	-	ARI	II	N	
e. Coils, heating		P	R	-	MF STD	II	N	
f. Duct-work and registers		P	R	-	AISI/AWS	I/II	Y/N	(23)(26)
g. Dampers, isolation and control		P	R	-	AMCA	I/II	Y/N	(25)(26)
3. Reactor Enclosure Air Recirculation System	6.5.1							(74)
a. Fans		P	R	-	AMCA	I	Y	
b. Motors, fans		P	R	-	NEMA MG-1/ IEEE 323, 344	I	Y	
c. Duct-work		P	R	-	AISI/AWS	I	Y	(23)
d. Dampers, isolation and control		P	R	-	AMCA/IEEE 323, 344	I	Y	(25)
e. Prefilters		P	R	-	ANSI N509	I	Y	(24)
f. HEPA filters		P	R	-	ANSI N509	I	Y	(24)
g. Charcoal adsorbers		P	R	-	ANSI N509	I	Y	(24)
4. Standby Gas Treatment System	6.5.1							(74)
a. Exhaust fans		P	CS	-	AMCA	I	Y	
b. Motors		P	CS	-	NEMA MG-1/ IEEE 323, 344	I	Y	
c. Coils, electric heating		P	CS	-	NEC	I	Y	
d. Duct-work		P	R/CS	-	AISI/AWS	I	Y	(23)
e. Dampers, isolation and control		P	R/CS	-	AMCA	I	Y	(25)
f. HEPA filters		P	CS	-	ANSI N509	I	Y	(24)
g. Charcoal adsorbers		P	CS	-	ANSI N509	I	Y	(24)
h. Prefilters		P	R	-	ANSI N509	I	Y	(65)(24)

Table 3.2-1 (Cont'd)

SYSTEM/COMPONENT(40)			UFSAR SECTION	SOURCE OF SUPPLY(1)	LOCA-TION(2)	QUALITY GROUP CLASSI-FICATION(3)	PRINCIPAL CODES AND STANDARDS(4)	SEISMIC CATEGORY(5)	Q-LIST(6)	COMMENTS
	5.	RHR, HPCI, RCIC and CS Rooms HVAC	9.4.2.2							(74)
		a. Piping and valves		P	R	C	III-3	I	Y	(48)
		b. Fans, unit coolers		P	R	-	AMCA	I	Y	
		c. Motors, fans		P	R	-	NEMA MG-1/ IEEE 323, 344	I	Y	
		d. Coils, cooling		P	R	C	III-3	I	Y	
		e. Duct-work and registers		P	R	-	AISI/AWS	I	Y	(23)
C.	Primary Containment									
	1.	Drywell Cooling System	9.2.10, 9.4.5							(86)
		a. Piping and valves		P	T,R	D	B31.1	II,IIA	N	
		b. Motors, fan		P	C	-	NEMA MG-1/ IEEE 323, 334, 344	I	Y	
		c. Fans (operate after LOCA mixing)		P	C	-	AMCA	I	Y	
		d. Fans (other)		P	C	-	AMCA	IIA	N	
		e. Coils, cooling		P	C	-	ARI	IIA	N	
		f. Duct-work (operate after LOCA mixing)		P	C	-	AISI/AWS	I	Y	
		g. Duct-work (other)		P	C	-	AISI/AWS	IIA	N	
		h. Dampers (operate after LOCA mixing)		P	C	-	AMCA	I	Y	
		i. Dampers (other)		P	C	-	AMCA	IIA	N	
		j. Chilled water equipment		P	R	D	MF STD	II	N	
		k. Chilled water isolation valves at primary containment		P	R	B	III-2	I	Y	
		l. Piping associated with isolation valves at primary containment penetration		P	C	B	B31.1	I	Y	(21)(48)
	2.	Purge System								
		a. Piping and valves		P	R	B	III-2	I	Y	(48)
		b. Piping and valves, beyond outermost containment isolation valves (smaller than 18 inch nominal diameter)		P	R	D	B31.1	IIA	N	

Table 3.2-1 (Cont'd)

SYSTEM/COMPONENT(40)	UFSAR SECTION	SOURCE OF SUPPLY(1)	LOCA-TION(2)	QUALITY GROUP CLASSI-FICATION(3)	PRINCIPAL CODES AND STANDARDS(4)	SEISMIC CATEGORY(5)	Q-LIST(6)	COMMENTS
3. Hydrogen recombiner								(86)
a. Piping and valves		P	R	B	III-2	I	Y	(48)(27)
b. Reaction chamber		P	R	B	III-2	I	Y	
c. Blower		P	R	B	III-2	I	Y	
4. Vacuum relief system								
a. Valves		P	C	B	III-2	I	Y	
b. Tees supporting valves		P	C	B	ASTM/B31.1	I	Y	(66)
D. Radwaste and Offgas Enclosure	9.4.3							
1. Fans		P	RW	-	AMCA	II	N	
2. Coils, cooling		P	RW	-	ARI	II	N	
3. Heating coil, steam		P	RW	-	MF STD	II	N	
4. Duct-work		P	RW,T,CS	-	SMACNA	II	N	
5. Dampers		P	RW	-	AMCA	II	N	
6. Prefilters, exhaust		P	RW	-	ANSI N509	II	N	
7. HEPA filters, exhaust		P	RW	-	ANSI N509	II	N	
8. Chilled water, direct expansion and steam equipment		P	R,RW AB	D	MF STD	II	N	
9. Charcoal adsorbers, exhaust		P	RW	-	ANSI N509	II	N	
10. Piping, chilled water		P	RW	D	B31.1	II	N	
E. Turbine Enclosure	9.4.4							(86)
1. Piping, chilled water		P	T	D	B31.1	II	N	
2. Fans		P	T	-	AMCA	II	N	
3. Coils, cooling		P	T	-	ARI	II	N	
4. Coils, heating, steam		P	T	-	MF STD	II	N	
5. Duct-work		P	T,CS	-	SMACNA	II	N	
6. Dampers		P	T	-	AMCA	II	N	
7. Prefilters, exhaust		P	T	-	ANSI N509	II	N	
8. HEPA filters, exhaust		P	T	-	ANSI N509	II	N	
9. Charcoal adsorbers, exhaust		P	T	-	ANSI N509	II	N	
10. Chilled water and steam heat equipment		P	R,AB	D	MF STD	II	N	

Table 3.2-1 (Cont'd)

SYSTEM/COMPONENT(40)	UFSAR SECTION	SOURCE OF SUPPLY(1)	LOCA-TION(2)	QUALITY GROUP CLASSI-FICATION(3)	PRINCIPAL CODES AND STANDARDS(4)	SEISMIC CATEGORY(5)	Q-LIST(6)	COMMENTS
F. Diesel Generator Enclosure	9.4.6							(86)
1. Motors, fan		P	G	-	IEEE 323, 344/ NEMA, MG-1	I	Y	
2. Fans		P	G	-	AMCA	I	Y	
3. Duct-work		P	G	-	MF STD	I	Y	(23)
4. Dampers		P	G	-	AMCA/IEEE 323	I	Y	(25)
5. Unit heaters, steam		P	G	-	MF STD	IIA	N	
G. Spray Pond Pump Structure	9.4.7							(86)
1. Motors, fan		P	S	-	IEEE 323, 344/NEMA MG-1	I	Y	
2. Fans		P	S	-	AMCA	I	Y	
3. Dampers		P	S	-	AMCA	I	Y	(25)
4. Coils, electric heating		P	S	-	NEC	IIA	N	
H. Miscellaneous Pump Structures Schuylkill, Perkiomen, Circulating Water)	9.4.9							
1. Fans		P	SP,PP, CW	-	AMCA	II	N	
2. Dampers		P	SP,PP, CW	-	AMCA	II	N	
I. Miscellaneous Enclosures (Auxiliary Boiler, Fuel Oil Transfer, Water Treatment, Sewage Treatment)	9.4.9							
1. Fans		P	AB,F, W,ST	-	AMCA	II	N	
2. Dampers		P	AB,F, W,ST	-	AMCA	II	N	
J. Administration Building	9.4.9							
1. Fans		P	A	-	AMCA	II	N	
2. Coils, cooling		P	A	-	ARI	II	N	
3. Coils, heating		P	A	-	MF STD	II	N	
4. Duct-work		P	A	-	SMACNA	II	N	
5. Dampers		P	A	-	AMCA	II	N	
6. Chilled hot water, and direct expansion systems		P	A	-	MF STD	II	N	

Table 3.2-1 (Cont'd)

SYSTEM/COMPONENT(40)	UFSAR SECTION	SOURCE OF SUPPLY(1)	LOCA-TION(2)	QUALITY GROUP CLASSI-FICATION(3)	PRINCIPAL CODES AND STANDARDS(4)	SEISMIC CATEGORY(5)	Q-LIST(6)	COMMENTS
K. Hot Maintenance Shop	9.4.8							
1. Direct expansion and hot water equipment		P	HS,A	-	MF STD	II	N	
2. Fans		P	HS	-	AMCA	II	N	
3. Coils, cool and heat		P	HS	-	ARI/MF STD	II	N	
4. Duct-work		P	HS	-	SMACNA	II	N	
5. Dampers		P	HS	-	AMCA	II	N	
6. Dust collectors		P	HS	-	MF STD	II	N	
7. HEPA Filters, exhaust		P	HS	-	ANSI N509	II	N	
8. Prefilters, exhaust		P	HS	-	ANSI N509	II	N	
VIII MAIN STEAM AND POWER CONVERSION SYSTEM								
A. Condensate Storage and Transfer System	9.2.7							
1. Tanks		P	O	D	AWWA D100	II	N	
2. Piping and valves		P	T,O	D	B31.1	II	N	
3. Pumps		P	T	D	HYD I/MF STD	II	N	
4. Dikes		P	O	-	-	IIA	N	
B. Main Steam System	10							(86)
1. Piping, main steam to turbine stop valves and branch line		P	R,T	B	III-2	I	Y	(8)(19)(28) (48)
2. Piping, main steam from, and including, the turbine stop valve to turbine casing and branch line up to, and including, first valve		P	T	D	B31.1	II	N	(29)(30)(31)
3. Piping and valves, other, steam		P	T	D	B31.1	II	N	
C. Main Condenser Evacuation System	10.4.2							
1. Condensers		P	T	D	VIII-1	II	N	
2. Air ejectors		P	T	D	B31.1	II	N	
3. Pumps, mechanical vacuum, and accessories		P	T	D	MF STD	II	N	
4. Piping and valves		P	T,RW	D	B31.1	II	N	
D. Turbine Gland Sealing System	10.4.3							
1. Steam seal evaporator		P	T	D	VIII-1	II	N	
2. Gland steam condenser		P	T	D	VIII-1	II	N	
3. Piping and valves		P	T	D	B31.1	II	N	

Table 3.2-1 (Cont'd)

SYSTEM/COMPONENT(40)	UFSAR SECTION	SOURCE OF SUPPLY(1)	LOCA-TION(2)	QUALITY GROUP CLASSI-FICATION(3)	PRINCIPAL CODES AND STANDARDS(4)	SEISMIC CATEGORY(5)	Q-LIST(6)	COMMENTS
E. Condensate Cleanup System	10.4.6							
1. Pressure vessels (filter and deep bed/demineralizers)		P	T	D	VIII-1	II	N	
2. Piping and valves		P	T	D	B31.1	II	N	
3. Atmospheric tanks (spent resin)		P	T	D	API 650	II	N	(18)
4. Piping and valves (spent resin)		P	T	D	B31.1	II	N	(18)
F. Condensate and Feedwater System	10.4.7							
1. Heat exchangers		P	T	D	VIII-1	II	N	
2. Piping and valves, reactor feedwater, RPV to outermost isolation valve		P	C,R	A	III-1	I	Y	(32)(48)
3. Piping and valves, reactor feedwater, other		P	R,T	B,D	III-2/ B31.1	I,II	Y,N	(48)-seismic Category I only
4. Piping, steam, to feedwater pump-turbine		P	T	D	B31.1	II	N	
5. Piping, crossover (low pressure)		P	T	D	B31.1	II	N	
6. Piping, bypass (high pressure) downstream of first isolation valve		P	T	B	III-2	I	Y	(48)
7. Piping and valves, condensate		P	T	D	B31.1	II	N	
8. Pumps, feedwater and condensate		P	T	D	B31.1/ HYD I	II	N	
G. Auxiliary Steam System	10.4.10							
1. Auxiliary boilers		P	AB	-	VIII-1	II	N	
2. Piping and valves		P	AB,T	-	B31.1	II	N	
H. Main Chlorination System	9.2.4							
1. Piping and valves		P	CW	-	B31.1	II	N	
I. Lube Oil System								
1. Lube oil storage tank		P	T	-	API 650	II	N	
2. Reservoirs		P	T	-	MF STD	II	N	
3. Centrifuges		P	T	-	MF STD	II	N	
4. Heat exchangers		P	T	-	VIII-1/ TEMA C	II	N	
5. Piping and valves		P	T	-	B31.1	II	N	
6. Pumps		P	T	-	B31.1/ HYD I	II	N	
IX INSTRUMENTATION AND CONTROL SYSTEMS								(10)
A. Reactor Protection (Trip) System	7.2	GE	C,R	-	IEEE 279	I	Y	(45)(87)

Table 3.2-1 (Cont'd)

SYSTEM/COMPONENT[40]	UFSAR SECTION	SOURCE OF SUPPLY[1]	LOCA-TION[2]	QUALITY GROUP CLASSI-FICATION[3]	PRINCIPAL CODES AND STANDARDS[4]	SEISMIC CATEGORY[5]	Q-LIST[6]	COMMENTS
B. Engineered Safety Features Systems	7.3							
1. ECCS: HPCI, ADS, CS, LPCI (RHR)		GE	C,R	-	IEEE 279	I	Y	(45)(87)
2. Primary containment and reactor vessel isolation control system		GE	C,R	-	IEEE 279	I	Y	(45)
3. Class 1E power system		P	G	-	IEEE 279	I	Y	(45)
4. RHR containment spray mode		GE	C,R	-	IEEE 279	I	Y	(45)(87)
5. Service water systems: RHRSW, ESW		P	C,R	-	IEEE 279	I	Y	(45)(87)
6. Containment atmospheric control systems:								(86)
CGCS		P	C	-	IEEE 279	I	Y	(45)
Combustible Gas monitoring system		P	C	-	IEEE 279	I	Y	(45)(27)
Primary containment vacuum relief system		P	C	-	MF STD	II	N	
7. Deleted								
8. RHR suppression pool cooling system		GE	C,R	-	IEEE 279	I	Y	(45)(87)
9. RERS		P	R	-	IEEE 279	I	Y	(45)(87)
10. Reactor enclosure isolation system		P	R	-	IEEE 279	I	Y	(45)(87)
11. Habitability and control room isolation		P	CS	-	IEEE 279	I	Y	(45)
12. SGTS filter room and access area unit coolers		P	CS	-	IEEE 279	I	Y	(45)
13. Diesel generator enclosure ventilation system		P	G	-	IEEE 279	I	Y	(45)(87)
14. Spray pond pump structure ventilation system		P	S	-	IEEE 279	I	Y	(45)
15. ESF switchgear and battery rooms cooling system		P	CS	-	IEEE 279	I	Y	(45)
16. ECCS pump compartment unit coolers		P	R	-	IEEE 279	I	Y	(45)
17. Drywell unit coolers		P	R	-	IEEE 279	I	Y	(45)(86)
18. CECWS		P	CS	-	IEEE 279	I	Y	(45)
19. Auxiliary equipment room ventilation system		P	CS	-	IEEE 279	I	Y	(45)
20. SGTS		P	CS	-	IEEE 279	I	Y	(45)
C. Safety-Related Display Instrumentation	7.5	P/GE	CS	-	IEEE 279	I	Y	(45)(27)
D. Systems Required for Safe Shutdown	7.4							
1. RCIC system		GE	C,R	-	IEEE 279	I	Y	(45)
2. SLCS		GE	C,R	-	MF STD	I	Y	(45)
3. Reactor shutdown cooling mode of the RHR System		GE	R	-	IEEE 279	I	Y	(45)(87)
4. Remote shutdown system		GE	CS	-	MF STD	I	Y	(45)(87)

Table 3.2-1 (cont'd)

SYSTEM/COMPONENT(40)	UFSAR SECTION	SOURCE OF SUPPLY(1)	LOCA-TION(2)	QUALITY GROUP CLASSI-FICATION(3)	PRINCIPAL CODES AND STANDARDS(4)	SEISMIC CATEGORY(5)	Q-LIST(6)	COMMENTS
E. All Other Instrumentation Systems Required for Safety	7.6							
1. Process radiation monitoring system		P/GE	C,R,CS	-	MF STD	I	Y	(45)(86)
2. NMS-IRM, APRM, LPRM		GE	R	-	IEEE 279	I	Y	(45)
3. SRV position indication		P	C,CS	-	-	IIA	N	(51)(27)
4. Leak detection systems Main steam line leak detection RCIC system leak detection RWCU system leak detection HPCI system leak detection		P	C,R	-	-	I	Y	(45)(86)
5. CIGS-ADS control		P	R	-	IEEE 279	I	Y	(45)
6. High pressure/low pressure systems interlocks		GE	R	-	IEEE 279	I	Y	(45)
7. Safeguard piping fill system		P	R	-	IEEE 279	I	Y	(45)
F. Control Systems Not Required for Safety	7.7							
1. RPV instrumentation		GE	C	-	MF STD	I	Y	(45)
2. RMCS		GE	C,R	-	MF STD	II	N	
3. Recirculation flow control system		GE	C,R,T	-	MF STD	I,IIA,II	Y/N	-
3A. Recirc Motor Adjustable Speed Drive Set		P	T	-	NEMA MG-1	II	N	
4. Feedwater control system		P/GE/W	T	-	MF STD	I/II	Y/N	(45)
5. Pressure regulator and turbine generator system		P	T	-	MF STD	II	N	
6. NMS-TIP, RBM, SRM		GE	R	-	MF STD	II	N	
7. RWCU system		GE	R	-	MF STD	I/IIA	Y/N	(45)
8. FPCC		P	R,RW	-	MF STD	II	N	(45)
9. Radwaste system - Gaseous radwaste system Liquid radwaste system Solid radwaste system		P	RW	-	MF STD	II/IIA	N	
10. ARMS		P/GE	C,R,T, CS,RW,HS	-	MF STD	II	N	(87)
11. Plant Monitoring System (PMS)		RTP	CS	-	MF STD	II	N	
12. CIGS		GE	C,R	-	MF STD	I/II	Y/N	(45)
13. Refueling interlocks		GE	C	-	MF STD	I	Y	
14. Leak detection system		P	C,R	-	MF STD	II	N	
15. Fire protection and suppression system		GE	CS	-	MF STD	II	N	
16. Nonsafety-related equipment area cooling ventilation systems		P	R,T,RW	-	MF STD	II	N	
17. Process radiation monitoring systems		P/GE	R,T,RW, HS	-	MF STD	II	N	(86)

Table 3.2-1 (Cont'd)

SYSTEM/COMPONENT[40]	UFSAR SECTION	SOURCE OF SUPPLY[1]	LOCA-TION[2]	QUALITY GROUP CLASSI-FICATION[3]	PRINCIPAL CODES AND STANDARDS[4]	SEISMIC CATEGORY[5]	Q-LIST[6]	COMMENTS
X ELECTRIC SYSTEMS								
A. Engineered Safety Features AC Equipment	8.3							
1. 4 kV switchgear, including safeguard bus feeder breakers, protective relays, control panels		P	G	- 323, 344	IEEE 308,	I	Y	(43)(45)
2. 440 V load centers, including 4160/440 V transformers, protective relays, control panels		P	R	-	IEEE 308, 323, 344	I	Y	(43)(45)
3. 440 V motor control centers		P	R	-	IEEE 308, 323, 344	I	Y	(43)(45)
B. Engineered Safety Features DC Equipment	8.3							
1. 125 V and 125/250 V station batteries and racks, battery chargers		P	CS	-	IEEE 308, 323, 344	I	Y	(45)
2. MCC and distribution panels, including protective relays		P	CS,R, G	-	IEEE 308, 323, 344	I	Y	(43)(45)
C. 120 V Vital AC System Equipment	8.3							
1. 120 V distribution panels		P	CS	-	IEEE 308,323,344	I	Y	(43)(45)
D. Electric Cables for Safety-related Equipment	8.3							
1. 5 kV power cables		P	ALL	-	IEEE 279 308, 323, 383	-	Y	(12)(45)
2. 600 V power cables, including all dc power cables		P	ALL	-	IEEE 279, 308, 323, 383	-	Y	(12)(45)
3. Control and instrumentation cables		P/GE	ALL	-	IEEE 279, 308, 323, 383	-	Y	(12)(45)
E. Miscellaneous Electrical	8.3, 9.5							
1. Primary containment enclosure electrical penetration assemblies		P	C	-	IEEE 317, 344, 383/III-MC	I	Y	(45)
2. Raceway systems, safety-related		P	ALL	-	IEEE 344	I	Y	(41)(45)
3. Emergency lighting batteries		P	ALL	-	MF STD	II	N	
4. Emergency lighting systems		P	ALL	-	MF STD	II	N	(63)
5. Emergency communications systems		P	ALL	-	MF STD	II	N	
6. Motors, nonsafety-related		P	ALL	-	NEMA MG-I	II	N	
7. Inverters		P	ALL	-	MF STD	II	N	(42)
8. Valve operators		-	-	-	-	-	-	(40)
9. Raceway systems, nonsafety-related		P	ALL	-	NEC	II/IIA	N	

Table 3.2-1 (Cont'd)

SYSTEM/COMPONENT(40)	UFSAR SECTION	SOURCE OF SUPPLY(1)	LOCA-TION(2)	QUALITY GROUP CLASSI-FICATION(3)	PRINCIPAL CODES AND STANDARDS(4)	SEISMIC CATEGORY(5)	Q-LIST(6)	COMMENTS
F. Offsite Power Systems	8.2	P	-	-	MF STD	II	N	
XI AUXILIARY SYSTEMS								
A. Safeguard Piping Fill System, Including Feedwater Fill System	6.3							
1. Piping and valves, from and including isolation valves, to feedwater lines		P	R	A	III-1	I	Y	(48)
2. Piping and valves, other		P	R	B	III-2	I	Y	(48)
3. Pumps		P	R	B	III-2	I	Y	
B. Suppression Pool Cleanup System	P&ID M-52							
1. Piping and valves, to second isolation valve		P	R	B	III-2	I	Y	(48)
2. Piping and valves, after second isolation valve		P	R	D	B31.1	IIA	N	
3. Pumps		P	R	D	MF STD	IIA	N	
C. Demineralized Water Makeup System	9.2.5							
1. Tanks		P	W	D	API 650/ AWWA D100	II	N	
2. Piping and valves		P	ALL	D	B31.1	II	N	
3. Pumps		P	W	D	B31.1/HYD I	II	N	
4. Filter vessels		P	W	D	VIII-1	II	N	
5. Demineralizer vessels		P	W	D	VIII-1	II	N	
D. Drywell Chilled Water System	9.2.10							(86)
1. Chillers		P	T	D	VIII-1	II	N	
2. Cooling coils		P	T	-	ARI	II,IIA	N	
3. Piping and valves, other		P	T,R	D	B31.1	II,IIA	N	
4. Valves, isolation to primary containment		P	R	B	III-2	I	Y	
5. Pumps		P	T	D	HYD I/B31.1	II	N	
6. Piping associated with isolation valves at primary containment penetration		P	C	B	B31.1	I	Y	(21)(48)

Table 3.2-1 (Cont'd)

SYSTEM/COMPONENT(40)			UFSAR SECTION	SOURCE OF SUPPLY(1)	LOCATION(2)	QUALITY GROUP CLASSIFICATION(3)	PRINCIPAL CODES AND STANDARDS(4)	SEISMIC CATEGORY(5)	Q-LIST(6)	COMMENTS
E.	Control Structure Chilled Water System		9.2.10							(86)
	1.	Piping		P	CS	C	B31.1	I	Y	(48)(53)
	2.	Valves		P	CS	C	B31.1	I	Y	(53)
	3.	Pumps		P	CS	C	III-3	I	Y	(56)
	4.	Motors, pump		P	CS	-	IEEE 323, 344	I	Y	
	5.	Chillers (except condensers)		P	CS	C	VIII-1/IEEE 323	I	Y	(54)
	6.	Chiller condensers		P	CS	C	III-3	I	Y	
F.	Compressed Air and Instrument Gas System		9.3.1							(86)
	1.	Compressors		P	T	D	MF STD	II	N	
	2.	Instrument gas bottles (ADS – Long Term)		P	T	-	MF STD	I	Y	(13)
	3.	Air and gas receivers		P	T	D	VIII-1	II	N	
	4.	Piping and valves forming part of containment boundary		P	C,R	B	III-2	I	Y	(48)
	5.	Piping and valves, safety-related (except as in item 7 below)		P	C,R, CS,G	C	III-3	I	Y	(48)
	6.	Piping and valves, other		P	ALL	D	B31.1	II/IIA	N	(82)
	7.	Piping and components to Unit I & 2 inflatable seals 1,2,3,4,7 & 10		P	R	D	B31.1	I	Y	(80)
G.	Sampling System		9.3.2							
	1.	Sample coolers		P	C,R, T,RW	D	MF STD	II	N	(73)
	2.	Piping and valves on III-1 system		P	C	A	III-1	I	Y	(9)(48)
	3.	Piping and valves on III-2 system (includes containment penetration isolation)		P	R	B	III-2	I	Y	(9)(48)
	4.	Piping and valves on III-3 system		P	R	B	III-2	I	Y	(9)(48)
	5.	Piping and valves, other systems		P	R,T,RW	D	B31.1	II	N	(9)
H.	Equipment and Floor Drains		9.3.3							(87)
	1.	Piping, radioactive		P	C,R, T,RW	D	B31.1	II/IIA	N	
	2.	Piping, nonradioactive		P	ALL	D	B31.1	II	N	
	3.	Piping and valves primary containment isolation boundary		P	C	B	III-2	I	Y	(48)
I.	Fire Protection System		9.5.1							
	1.	Pumps, piping and water system components		P	ALL	-	NFPA/ANI	II/IIA	N	
	2.	Gas system components (CO_2 and Halon 1301)		P	CS	-	NFPA/ANI	II/IIA	N	
	3.	Fire and smoke detection and alarm system		P	ALL	-	NFPA/ANI	II/IIA	N	

Table 3.2-1 (Cont'd)

SYSTEM/COMPONENT(40)		UFSAR SECTION	SOURCE OF SUPPLY(1)	LOCA-TION(2)	QUALITY GROUP CLASSI-FICATION(3)	PRINCIPAL CODES AND STANDARDS(4)	SEISMIC CATEGORY(5)	Q-LIST(6)	COMMENTS
J.	Nitrogen System and Generator External Hydrogen System								
	1. Vessels (nitrogen only)		P	H,O	D	VIII-1	II	N	
	2. Piping		P	H,T,R, O,RW	D	B31.1	II	N	
	3. Valves		P R,O	H,T,	D	B31.1	II	N	
K.	Postaccident Sampling System	11.5.5							(79)(27)
	1. Sample coolers		GE	R	D	VIII-1	IIA	N	
	2. Sample line root valves		P	R	B	III-2	I	Y	(46)
	3. System piping		P,GE	R,CS	D	B31.1	IIA/II	N	
	4. System tubing and other valves		P,GE	R,CS	-	MF STD	IIA/II	N	
L.	Zinc Injection System								
	1. Tanks		GE	T	D	B31.1	II	N	
	2. Pipes		P,GE	T	D	B31.1	II	N	
	3. Valves		P,GE	T	D	B31.1	II	N	
XII	ENCLOSURES								(33)
A.	Reactor Enclosure and Refueling Area	3.8.4	P	R	B	ACI/AISC	I	Y	(75)
	1. Roof scuppers and parapet openings		P	R	-	UBC	II	N	
	2. Pressure resisting doors		P	R	-	MF STD	I	Y	
	3. Missile barriers for safety-related equipment		P	R	-	ACI/AISC	I	Y	
	4. Spent fuel pool liner		P	R	-	AISC	I	Y	
	5. Safety-related masonry walls		P	R	-	ACI/UBC	I	Y	(75)
	6. Fabricated supports for safety-related systems and components	3.7.3/ 3.10.3	P	R	-	AISC/AISI	I	Y	
	7. Inflatable Seals (Unit 1 & 2, seals 1,2,3,4,7 & 10 only)	9.3.1	P	R	-	MF STD	I	Y	
B.	Primary Containment	3.8.1	P	C	B	ACI/AISC/III	I	Y	(64)
	1. Access hatches/locks/doors		P	C	B	III-2	I	Y	(64)
	2. Liner plate ?¼"		P	C	B	VIII-1	I	Y	(64)
	3. Penetration assemblies and liner plate >¼"		P	C	B	III-2	I	Y	(39)(64)(27)
	4. Vacuum relief valves		P	C	B	III-2	I	Y	(66)
	5. Downcomers		P	C	B	III-2	I	Y	(64)(66)
	6. Downcomer bracing		P	C	-	AISC	I	Y	
	7. Biological (primary) shield		P	C	-	ACI/AISC	I	Y	
	8. Fabricated supports for safety-related systems and components	3.7.3/ 3.10.3	P	C	-	AISC/AISI	I	Y	

Table 3.2-1 (Cont'd)

SYSTEM/COMPONENT[40]	UFSAR SECTION	SOURCE OF SUPPLY[1]	LOCATION[2]	QUALITY GROUP CLASSIFICATION[3]	PRINCIPAL CODES AND STANDARDS[4]	SEISMIC CATEGORY[5]	Q-LIST[6]	COMMENTS
C. Turbine Enclosure	3.8.4.1	P	T	-	ACI/AISC	II	N	
D. Control Structure	3.8.4.1	P	CS	-	ACI/AISC	I	Y	(75)
1. Roof scuppers and parapet openings		P	CS	-	UBC	II	N	
2. Pressure resisting doors		P	CS	-	MF STD	I	Y	
3. Missile barriers for safety-related equipment		P	CS	-	ACI/AISC	I	Y	
4. Safety-related masonry walls		P	CS	-	ACI/UBC	I	Y	
5. Fabricated supports for safety-related systems and components	3.7.3/ 3.10.3	P	CS	-	AISC/AISI	I	Y	(75)
E. Radwaste and Offgas Enclosure	3.8.4.1	P	RW	-	ACI/AISC	IIA	N	(18)
F. Diesel Generator Enclosure	3.8.4.1	P	G	-	ACI/AISC	I	Y	
1. Roof scuppers and parapet openings		P	G	-	UBC	II	N	
2. Missile barriers for safety-related equipment		P	G	-	ACI/AISC	I	Y	
3. Safety-related masonry walls		P	G	-	ACI/UBC	I	Y	
4. Fabricated supports for safety-related systems and components	3.7.3/ 3.10.3	P	G	-	AISC/AISI	I	Y	
G. Spray Pond Pump Structure	3.8.4.1	P	S	-	ACI/AISC	I	Y	
1. Roof scuppers and parapet openings		P	S	-	UBC	II	N	
2. Fabricated supports for safety-related systems and components	3.7.3/ 3.10.3	P	S	-	AISC/AISI	I	Y	
H. Schuylkill Pump Structure		P	SP	-	ACI/AISC	II	N	
I. Perkiomen Pump Structure		P	PP	-	ACI/AISC	II	N	
J. Circulating Water Pump Structure		P	CW	-	ACI/AISC	II	N	
K. Auxiliary Boiler Enclosure		P	AB	-	ACI/AISC	II	N	
L. Fuel Oil Pump Structure		P	F	-	ACI/AISC	II	N	
M. Water Treatment Enclosure		P	W	-	ACI/AISC	II	N	
N. Sewage Treatment Enclosure		P	ST	-	ACI/AISC	II	N	
O. Administration Building		P	A	-	ACI/AISC	II	N	
P. 5-Line Outage Support Facility		P	T	-	ACI/AISC UBC	II	N	
Q. Chemistry Lab Expanion		p	CL	-	B OCA/UBS	II	N	

Table 3.2-1 (Cont'd)

SYSTEM/COMPONENT[40]		UFSAR SECTION	SOURCE OF SUPPLY[1]	LOCA-TION[2]	QUALITY GROUP CLASSI-FICATION[3]	PRINCIPAL CODES AND STANDARDS[4]	SEISMIC CATEGORY[5]	Q-LIST[6]	COMMENTS
XIII	SPRAY POND								
A.	Pond	3.8	P	S	-	-	I	Y	(34)(35)
B.	Support Columns	3.8	P	S	-	ACI/AISC	I	Y	(36)
C.	Overflow Structure	3.8	P	S	-	ACI/AISC	I	Y	
D.	Spray-Network Piping	9.2	P	S	C	III-3	I	Y	(38)(60)
E.	Soil-Bentonite Lining and Soil Cover	2.5	P	S	-	-	-	N	
F.	Unreinforced Concrete (excluding foundations) Concrete Backfill, Exploration, Trench Backfill, and Soil Cover		P	S	-	-	-	N	
G.	Riprap and Riprap Bedding	2.5	P	S	-	-	-	N	
H.	Shotcrete 2.5		P	S	-	-	-	N	
I.	Rock Bolts	2.5	P	S	-	-	-	N	
J.	Roadwork-		P	S	-	-	-	N	
K.	Emergency Spillway	3.8	P	S	-	-	II	Y	(37)

Table 3.2-1 (Cont'd)

NOTES ON
LGS DESIGN CRITERIA SUMMARY

(1)	GE	-	General Electric
	W	-	Westinghouse Nuclear Automation
	P	-	PECO Energy
(2)	Location		
	R	-	Reactor Enclosure
	C	-	Containment
	T	-	Turbine Enclosure
	CS	-	Control Structure
	RW	-	Radwaste and Offgas Enclosure
	G	-	Diesel Generator Enclosure
	AB	-	Auxiliary Boiler Enclosure
	F	-	Fuel Oil Pump Structure
	W	-	Water Treatment Enclosure
	ST	-	Sewage Treatment Enclosure
	A	-	Administration Building
	S	-	Spray Pond Pump Structure
	SP	-	Schuylkill Pump Structure
	PP	-	Perkiomen Pump Structure
	CW	-	Circulating Water Pump Structure
	O	-	Outdoors, Onsite
	HS	-	Hot Maintenance Shop
	CL	-	Chemistry Laboratory Expansion

(3) A,B,C,D - Quality group classification as defined in

Regulatory Guide 1.26, see also Tables 3.2-2 and 3.2-3.
\- = Not applicable to quality group classification

Table 3.2-1 (Cont'd)

(4) Notations for principal construction codes:

AA	-	Aluminum Association
ACI	-	American Concrete Institute
AFBMA	-	Anti Friction Bearing Manufacturers Association
AISC	-	American Institute of Steel Construction
AISI	-	American Iron and Steel Institute
AMCA	-	Air Moving and Conditioning Association
AMCA 210	-	"Test Codes for Air Moving Devices"
AMCA 211A	-	"AMCA Certified Ratings Program for Air Performance"
ANI	-	American Nuclear Institute
ANSI	-	American National Standards Institute
ANSI B9.1	-	"Safety Code for Mechanical Refrigeration"
ANSI B30.11	-	"Monorail Systems and Underhung Cranes"
ANSI B30.16	-	"Overhead Hoists"
ANSI B31.1	-	"Code for Pressure Piping"
ANSI B31.5	-	"Refrigeration Piping"
ANSI N509	-	"Nuclear Power Plant Air Cleaning Units and Components" (1976, 1980)
API	-	American Petroleum Institute
API 620	-	"Recommended Rules for Design and Construction of Large, Welded, Low Pressure Storage Tanks"
API 650	-	"Welded Steel Tanks for Oil Storage"
ARI	-	Air Conditioning and Refrigeration Institute
AWS	-	American Welding Society
BOCA	-	National Building Code

Table 3.2-1 (Cont'd)

CMAA 70	-	Crane Manufacturers Association of America, Specification Number 70, "Electric Overhead Traveling Cranes"
HEI	-	Heat Exchange Institute
HYD I	-	Hydraulic Institute
IEEE	-	Institute of Electrical and Electronics Engineers
IEEE 279	-	"Criteria for Protection Systems for Nuclear Power Generating Stations" (1971)
IEEE 308	-	"Standard Criteria for Class 1E Electric Systems for Nuclear Power Generating Stations" (1971)
IEEE 317	-	"Standard for Electrical Penetration Assemblies in Containment Structures for Nuclear Fueled Power Generating Stations" (April 1971)
IEEE 323	-	"General Guide for Qualifying Class 1E Electric Equipment for Nuclear Power Generating Stations" (1971, 1974)
IEEE 334	-	"Trial Use Guide for Type Test of Continuous-Duty Class 1 Motors Installed Inside the Containment of Nuclear Power Generating Stations (ANSI N41.9)" (1971)
IEEE 344	-	"Guide for Seismic Qualification of Class 1E Electric Equipment for Nuclear Power Generating Stations" (1971, 1975)
IEEE 383	-	"Type Test of Class 1E Electrical Cables, Field Splices, and Connections for Nuclear Power Generating Stations" (1974)
IEEE 387	-	"Criteria for Diesel Generator Units Applied as Standby Power Supplies for Nuclear Power Generating Stations" (1972)
MF STD	-	Manufacturer's Standard
NEC	-	National Electric Code
NEMA MG-1	-	National Electrical Manufacturers' Association, "Motors and Generators" (1971)
NFPA	-	National Fire Protection Association
RDT-M-16-1T	-	Reactor Research and Development Gas Phase Adsorbents for Trapping Radioactive Iodine and Iodine Compounds
SMACNA	-	Sheet Metal and Air Conditioning Contractors National Association Inc.
TEMA C	-	Tubular Exchanger Manufacturers Association, Class C

Table 3.2-1 (Cont'd)

	UBC	-	Uniform Building Code
	UL	-	Underwriters Laboratories Standard
	III	-	"1, 2, 3, MC" - ASME B&PV Code, Section III, Class 1,2,3, MC (Certain NSSS mechanical components ordered prior to July 1, 1971, were built to the older code categories. Refer to Table 3.2-3 for the appropriate code)
	VIII-1	-	ASME B&PV Code, Section VIII, Division 1
(5)	I	-	The equipment is constructed in accordance with the seismic requirements for the SSE.
	IIA	-	The equipment of this category is designed so that it cannot adversely affect plant safety features during and after SSE.
	II	-	The equipment of this category is not designed for the SSE.
(6)	Y	-	Requires compliance with the requirements of 10CFR50, Appendix B
	N	-	Not within the scope of 10CFR50, Appendix B

(7) PECo's request to use alternative codes to those required in 10CFR50.55a for primary pressure boundary components was sent to the NRC on July 15, 1975. The applicable codes, code dates, and addenda are listed in the request letter for the following components which are still included in the LGS design:

A. RPV

B. Main steam SRVs

C. Main steam piping (26") from RPV to 2nd isolation valve

D. Main steam line suspension

E. Recirculation pump

F. Recirculation gate valves (motor-operated):
28" suction
28" discharge

G. Recirculation loop piping (28")

H. Recirculation loop suspension

Table 3.2-1 (Cont'd)

I. RHR valves
20" gate valve

J. Core spray valves
12" check valve (air-operated)

K. Nuclear Class I piping (all except main steam and recirculation)

NRC approval of the request was given on November 18, 1975 in a letter from R.C. De Young (NRC) to E.G. Bauer (PECo).

The applicable code for design, fabrication, and testing of the MSIVs is the ASME Standard Code for Pumps and Valves for Nuclear Power - 1968 Draft including March 1970 Addenda.

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

(9) Instrument and sampling lines Quality Group, seismic category, and quality assurance requirements are as follows:

A. Instrument Lines

1. From the process boundary through the process root valve (including restriction orifice adapter), or, for lines penetrating primary containment, through the containment isolation valve or excess flow check valve, whichever applies: same Quality Group, seismic category, and QA requirements as the process line

2. Downstream of the boundary defined in 1, above:

   a. Lines penetrating primary containment and whose associated instruments are required to function to perform a safety function (Q-active) are Quality Group B, seismic Category I, and are Q-listed.

   b. Lines penetrating primary containment and whose associated instruments are only required to maintain their pressure boundary integrity (Q-passive) are Quality Group D, seismic Category I, and are Q-listed.

   c. Lines not penetrating primary containment whose associated instruments are Q-active are Quality Group B, seismic Category I, and Q-listed.

   d. Lines not penetrating primary containment whose associated instruments are Q-passive are Quality Group D, seismic Category I, and Q-listed.

   e. Other lines are Quality Group D, nonseismic Category I, and not Q-listed.

   f. Certain adapters in nonsafety-related, non Q-listed instrument lines are not manufactured per ANSI B31.1.

Table 3.2-1 (Cont'd)

B. Sampling Lines

1. From process line to root valve: Same quality group, seismic category, and quality assurance as process line.

2. From root valve through sample rack isolation valve:

   a. Sampling lines from Q-listed Quality Group A, B, and C process lines are Quality Group B, seismic Category I, and Q-listed.

   b. Sampling lines from non Q-listed Quality Group C process lines are Quality Group B or D, nonseismic Category I, and not Q-listed

   c. Sampling lines from Quality Group D process lines are Quality Group D, nonseismic Category I, and non Q-listed.

3. Downstream of sample rack isolation valve: Quality Group D, nonseismic Category I, and non Q-listed

(10) Accident monitoring instrumentation was designed using the guidance provided in Regulatory Guide 1.97 Revision 2. Regulatory Guide 1.97 Category I and most of Category II instrumentation is Q-listed. For additional design criteria information, see Section 7.5.

(11) Components include any assembly of interconnected parts that constitutes an identifiable device or piece of equipment. For example, electrical components include sensors, power supplies, and signal processors; and mechanical components include turbines, strainers, and orifices.

(12) Refer to Section 7.1 for descriptions of conformance with IEEE 279, IEEE 308, IEEE 323, and IEEE 344.

(13) These bottles are not available as seismic Category I items. However, the piping from the bottles is seismic Category I and Q-listed. DOT specifications were used for the bottles.

(14) The HCU is a GE factory assembled, engineered module of valves, tubing, piping, and stored water that controls a single CRD by the application of precisely times sequences of pressures and flows to accomplish slow insertion, or withdrawal, of the control rods for power control, while providing rapid insertion for reactor scram.

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

Table 3.2-1 (Cont'd)

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 by additional requirements for these parts, and for the remaining parts and details. For example, all welds are liquid penetrant inspected; all socket welds are inspected for gap between pipe and socket bottom; all welding is performed by qualified welders; and all work is done by written procedures.

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

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

B. The scram accumulator is joined to the HCU by a split flange joint chosen for its compact design to facilitate both assembly and maintenance. Both the design and construction conform to ANSI B31.1, Power Piping Code, This joint, which requires a design pressure of 1750 psig, has been proof tested to 10,000 psi.

C. The accumulator nitrogen shutoff valve is a 6000 psi cartridge valve whose copper alloy material is listed by ASME Section VIII. The valve was chosen for this service partly because it is qualified by the U.S. Navy for submarine service.

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

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

(15) Quality Group B on the CRD insert and withdraw lines and scram discharge lines extends from the drive flange up to, and including, the first valve on the HCU.

(16) The RWCU pump suction piping from its tap off the recirculation loop up to and including the first valve outside primary containment (HV-44-F004) is classified Group A. The piping between valve HV-44-F004 up to and including the second valve (HV-44-F040) is classified Group B. The remainder of the pump suction piping is classified Group C.

Table 3.2-1 (Cont'd)

The RWCU system discharges into the feedwater lines and the RCIC line. There are three RWCU discharge paths: a tap into the RCIC line outside containment, a tap to feedwater loop A outside containment, and a tap to feedwater loop A inside containment. The RWCU piping upstream of the feedwater outboard containment isolation valve (HV-44-F039) is classified Group C. Valve HV-44-F039 and piping up to the feedwater and RCIC line taps (outside primary containment) is classified Group B. The RWCU tap into the feedwater line inside primary containment is classified Group B up to the outboard containment isolation valve (41-1016). Valve 41-1016 and piping up to the tap into the feedwater line is classified Group A.

(17) The HPCI and RCIC turbine do not fall within the applicable design codes. To ensure that the turbine is fabricated to the standards commensurate with their safety and performance requirements, GE has established specific design requirements for this component in their specification.

(18) Certain major liquid, solid, and gaseous radwaste system components were designed, fabricated, procured, installed, and tested to the requirements of ASME Section III, Class 3, prior to May 1978. After May 1978, radwaste and deep bed spent resin system design, fabrication, materials, procurement, installation, and testing are at a minimum, in accordance with quality group D and the intent of Regulatory Guide 1.143 (Rev 1), subject to the following clarifications and exceptions:

A. Certain atmospheric tanks are welded to API/AWS standards in lieu of ASME Section IX.

B. Curbs or elevated thresholds are not provided for indoor tanks because of the watertight integrity of the surrounding structure.

C. Hydrotest pressure is held for 10 minutes, in accordance with ASME Section III, rather than 30 minutes. When in-place pressure testing is not practical, the tie-in welds will be examined via NDE per ANSI B31.1.

D. The radwaste enclosure is designed in accordance with seismic Category I criteria (Section 3.8.4). LGS does not use Regulatory Guide 1.60, as stated in Section 1.8. Alternate methods are discussed in Sections 3.7 and 3.8.

E. LGS's quality program during construction did not require audits of activities associated with radwaste systems, and items of nonconformance and their regulation are not always documented. Beginning February 14, 1983, auditing or monitoring is required for activities associated with radwaste systems, and items of nonconformance and their regulation shall be documented.

F. Cleaning and welding of piping is conducted in accordance with the specified piping quality group.

Table 3.2-1 (Cont'd)

G. Inspection of all Q-listed instrument installation is performed by Bechtel QC personnel. Inspection of BOP instrumentation installations is performed by Bechtel field engineering personnel. In addition, final inspection of instrumentation up to the first root valve is performed by PECo QC personnel. Calibration and functional testing of all instrumentation is performed by PECo personnel.

Decontamination equipment and facilities are associated with the liquid radwaste system and are not safety-related.

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

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

(21) The basis for classification of non-ASME Section III equipment as Quality Group B is given in Section 3.2.2.g.

(22) Diesel fuel oil storage tanks and transfer pumps were designed to ASME Section III, Class 3 but were not stamped.

(23) The structural design of seismic Category I and IIA HVAC ducts was verified by testing duct specimens as permitted by the AISI Code, to substantiate the duct width to duct sheet thickness ratio (w/t) and cut height to duct sheet thickness ratio (h/t) of up to 1500. Seismic Category II ducts were designed and constructed in accordance with SMACNA.

(24) Regulatory Guide 1.52 (July 1976) suggests various industry standards and codes for this equipment. These references were used for system design, with exceptions as noted in Section 6.5.

(25) Dampers with electro-hydraulic operators were designed to IEEE 323. Fire dampers are labeled by UL.

(26) Portions of ducts and dampers in the reactor enclosure and refueling floor HVAC system are seismic Category II non Q-listed, and the remainder are seismic Category I, Q-listed.

(27) For discussion of design criteria related to TMI Action Plan requirements, see Section 1.13.2.

(28) The main steam system from its outer isolation valve up to, but not including, the turbine stop valve and bypass valve chest, and all branch lines 2½ inches in diameter and larger up to, and including, the first valve (including their restraints), will be designed by the use of an appropriate dynamic seismic-system analysis to withstand the OBE and SSE design loads in combination with other appropriate loads, within the limits specified for Class 2 pipe in the ASME Section III Code. The mathematical model for the dynamic seismic analyses of the main steam system and branch line piping includes the turbine stop valves and the piping from the stop valves to the turbine casing. The dynamic input loads

Table 3.2-1 (Cont'd)

for design of the main steam system are derived from a history model analysis (or an equivalent method) of the reactor and applicable portions of the turbine building. An elastic multi-degree-of-freedom system analysis is used to determine the input to the main steam system. The allowable stress and associated deformation limits for piping are in accordance with the ASME Section III Class 2 requirements for the OBE loading combinations. The main steam system supporting structures (those portions of the turbine enclosure) are such that the main steam system and its supports can maintain their integrity.

(29) The following qualification has been met with respect to the certification requirements:

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

B. The manufacturer of these valves and steam leads has certified that the quality control program so defined has been accomplished.

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

(31) The main steam loads from the turbine control valve to the turbine casing meet all of the requirements of Group D, plus the addition of the following 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 appropriate sections of ANSI B31.1.

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 appropriate sections of ANSI B31.1.

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

(32) The classification of the feedwater line from the reactor vessel through the second isolation valve is Group A. The classification of the feedwater line from the second isolation valve through the third valve is Group B. Beyond the third valve the classification is Group D.

(33) The listed design criteria apply to the external and internal structure of each enclosure. Refer to Sections 3.7 and 3.8 for discussion of seismic design and Category I structure design, respectively.

Table 3.2-1 (Cont'd)

(34) The following construction activities are conducted under the quality assurance program to ensure that the pond will perform its safety function:

A. Inspection and treatment of the rock surface at the pond bottom and sides to ensure that permeability requirements are met.

B. Final survey and measurement of the as-built pond to ensure that the water volume and side slopes satisfy design requirements.

C. Performance of a seepage test to ensure that the design basis seepage rate assumptions are not exceeded.

In addition, material used for treatment of fracture zones and capping observation wells is Q-listed.

(35) The pond is built completely by excavation. Ability to meet seismic Category I criteria is verified by measurement of as-built side slope areas (see Note 34.b) to ensure that design requirements are met.

(36) Includes unreinforced concrete used for support column foundations.

(37) The final survey and measurement of the as-built emergency spillway are conducted under the applicable portions of the QA program to ensure that the geometry satisfies design requirements.

(38) A complete description of the codes and standards, seismic category, and Q-list status of piping and instrumentation within the spray pond is shown on drawing M-12.

(39) Containment isolation barriers listed in Table 6.2-17 are Q-listed. Containment isolation valves are also included in the entries for each applicable system in this table.

(40) Specific components that comprise parts of major components with the same design criteria are generally not listed. For example, transformers are a part of load centers or switchgear, and valves operators are a part of motor-operated valves. Class 1E valve operators are in compliance with IEEE 323 and IEEE 344.

(41) Raceway systems include conduit, cable trays, and their supports. Raceway fire stops and seals are not Q-listed. However, quality control provisions commensurate with BTP CMEB 9.5-1 are applied to the raceway fire stops and seals.

(42) Inverters do not supply power to safety-related loads. The Class 1E battery loads are discussed in Section 8.3.2.1.1.4.

(43) Primary, backup and fault current protection devices are subcomponents of switchgear, load centers, MCC and distribution panels, which are Q-listed as shown in items X.A, X.B and X.C.

(44) Exhaust piping beyond the roof penetration is not Q-listed.

Table 3.2-1 (Cont'd)

(45) Equipment is qualified in accordance with the conformance statements made in Section 7.2, 7.3, 7.4, 7.5, and 7.6 in reference to IEEE 279 paragraph 4.4 and IEEE 323.

(46) Primary containment gas sample lines from the sample taps to and including the outboard containment isolation valves are seismic Category I. Liquid sample lines from the RHR system are seismic Category I up to and including the second system isolation valves. The sample line from the jet pump instrument system is seismic Category I to the PASS isolation valves.

(47) Delineation of applicable codes or standards and seismic category is shown in applicable piping specification for line class as indicated in drawing M-20 and in vendor manual for on-skid piping. The basis for classification of non-ASME Section III equipment as Quality Group C is given in Section 3.2.2.d.

(48) Supports associated with this piping are constructed in accordance with quality assurance and seismic Category I requirements.

The parts of the ESW system that are seismic Category IIA are indicated on drawing M-11. Nonseismic Category I drain and vent lines and capped ends extending from seismic Category I piping are seismic Category IIA downstream of the last isolation valve.

The operator may also elect to provide ESW to the following nonseismic Category I equipment:

A. RECW heat exchanger

B. TECW heat exchanger

C. Reactor recirculation pump seal and motor oil coolers.

ESW flow to and from these components is controlled by redundant key-locked remote manual valves for RECW and TECW and locked closed manual valves for the recirculation pump.

(49) The reactor vessel support skirt is designed to ASME Section III, Class I, subsection NF criteria.

(50) CRD housing supports are designed in accordance with the AISC code.

(51) Equipment is qualified in accordance with the conformance statements made in Section 7.6.

(52) The original design was to ASME Section III. The actual requirements are that of ANSI B31.1. The cleaning and welding of piping is conducted in accordance with the specified piping quality group.

Table 3.2-1 (Cont'd)

(53) The basis of classification of non-ASME Section III equipment as Quality Group C is given in Section 3.2.2.e.

(54) The basis for classification of non-ASME Section III equipment as Quality Group C is given in Section 3.2.2.h.

(55) Short welded sections of ANSI B31.1 piping in the turbine stop valve seat drains, stop valve leak-offs, governing valve leak-offs, casing drains, ring drains, chest drains, and turbine shaft seal leak-offs that cannot be hydrotested will be inservice tested to ANSI B31.1 requirements and the welds will be surface examined.

(56) The basis for classification of non-ASME Section III equipment as Quality Group C is provided in Section 3.2.2.i.

(57) The basis for classification of non-ASME Section III equipment as Quality Group C is provided in Section 3.2.2.j.

(58) This piping was purchased and constructed to Quality Group C requirements and was subsequently upgraded to Quality Group B by volumetrically examining all circumferential welds over two inches using radiography. Visual examination will be performed inservice in accordance with the Inservice Inspection Program.

(59) The containment spray nozzles are fabricated to manufacturer's standards. Inservice inspection requirements will be consistent with Quality Group B requirements.

(60) Spray pond nozzles and junction boxes were designed and built to ASME III, Class 3 requirements, except that they were not N-stamped by the manufacturer. No manufacturer had an N-stamp at that time.

(61) ESF also include those systems in Chapter 6 and Table 6.1-1. Although RCIC is not an ESF, it is listed in this subsection because, although not required to mitigate the consequences of an accident, it may be used.

(62) The reactor internal structures, other, include the steam dryer, shroud head and steam separator assembly, incore guide tubes and incore guide tube stabilizers, differential pressure and liquid control lines inside the RPV, fuel orifices, and feedwater spargers. These structures are neither required for safe shutdown of the plant nor will their failure jeopardize the safety function of other safety-related reactor internals. Differential pressure and liquid control lines inside the RPV that form part of the RCPB are Q-listed. With the exception of these lines, the aforementioned components are not safety-related, are not Q-listed, and are not under 10CFR50, Appendix B. However, they are inspected as part of the inservice inspection program which is included in the operations phase of the QA Plan.

(63) Control room emergency lighting meets seismic Category I requirements as described in Section 9.5.3.2.2.

(64) The metal components for the primary containment (including hatches, liner plate, penetration sleeves, and downcomers) are designed and fabricated in accordance with the

Table 3.2-1 (Cont'd)

principal codes and standards noted in this table and Sections 3.8.1 and 3.8.2, but are not N-stamped except the personnel airlock.

(65) Prefilters for the refueling area to SGTS alignment are installed in the refueling area duct-work. Prefiltering for the reactor enclosure to SGTS alignment is provided by the RERS.

(66) The tees supporting the vacuum relief valves are fabricated from ASTM material and welded into the downcomers. All welds have been radiographed.

(67) Exceptions: Valves HV-046-127, 128, 227, and 228 are seismic Class I and Q-listed. Isolation signals must close these block valves to prevent bypass leakage. Valves HV-046-125, 126, 225, and 226 are also seismic Class I and Q-listed. The piping and valves from the HCUs to and including the Containment Isolation Valves 46-1101, 46-1102, 46-1108, 46-1109, 46-1115, 46-1116, 46-1122, 46-1123, 46-2101, 46-2102, 46-2108, 46-2109, 46-2115, 46-2116, 46-2122, and 46-2123 are equivalent to ASME Section III, Class 2, seismic Class I and are Q-listed.

(68) The SLCS storage tanks were purchased before Article NC-3800 on atmospheric storage tanks was included in the ASME Section III, Class 2 code, and were hence designed and fabricated to API 650 and supplemental requirements at the time of purchase as discussed in Section 3.2.2(k).

(69) All three of the Unit 1 and one of the Unit 2 SLCS pump discharge accumulator vessels were purchased to ASME Section VIII, Division 1. These weldless vessels, except for the code stamping, also meet the ASME Section III, Class C requirements. Per Table 3.2-3, vessels purchased prior to July 1, 1971, are purchased to ASME Section III, Class C rather than Class 2.

(70) The HPCI turbine exhaust line globe stop-check valves were manufactured to ANSI B31.1 (1967), ANSI B16.5, and MSS-SP-66. They have been shown to meet the requirements of 1968 Draft ASME Code for Pumps and Valves and the nondestructive inspection requirements of ASME Section III.

(71) The shell side of the nonregenerative heat exchangers is constructed to ASME Section VIII, Division 1 and TEMA R.

(72) The core support structure was designed and procured prior to the issuance of subsection NG of ASME Section III. However, an earlier draft of the ASME Code was used as a guide in developing the design criteria in lieu of subsection NG.

Subsequent to the issuance of subsection NG, NG-3000 has been used, for evaluation purposes, in the core support structure evaluation. A detailed comparison between the original design basis and Appendix F (referenced by NG-3000) shows that the two sets of limits have no significant differences.

(73) The sample coolers were manufactured to the equivalent of ASME VIII, Division 1, but not code stamped. It was not customary at the time of manufacture to code stamp these coolers.

Table 3.2-1 (Cont'd)

(74) The safety-related portions of the HVAC systems are designed, fabricated, erected, and tested to quality standards commensurate with the safety function to be performed. Fans, filters, plenums, dampers, and duct-work in these systems are not classified as piping or pressure vessels and, as such, do not fall under the jurisdiction of ASME Section III. All of the equipment in these systems performing safety-related functions is Q-listed and seismic Category I, and all electrical components performing safety-related functions are environmentally qualified in accordance with Regulatory Guide 1.89. In addition, these systems provide substantial conformance with Regulatory Guide 1.52 as discussed in Sections 1.8 and 6.5.1 and Table 6.5-2.

(75) Includes shield walls identified by the plant shielding study (Section 1.13.2, Item II.B.2).

(76) Deleted

(77) Deleted

(78) Deleted

(79) The PASS does not function to mitigate the consequences of an accident. Therefore, with the exception of its interfaces with Q-listed systems, the PASS is not Q-listed. The PASS design is consistent with the guidance of Regulatory Guide 1.97 for the monitoring of Type E variables. While not a mitigating system, PASS is an integral part of the plant's capability to obtain and analyze samples and is therefore administratively controlled based on the guidance in Generic Letter 83-36, enclosure I.

(80) Select valves per QAD-M-15, to Unit 1 & 2 inflatable seals 1, 2, 3, 4, 7 and 10 are leak tested under the ASME section XI IST program but are not required to be installed to meet ASME class codes, but rather to ANSI B31.1 codes. These valves have been upgraded to meet seismic category I requirements and are Q-listed.

(81) Portions of the Air Start System have been downgraded to Seismic Category IIA as shown on drawing M-20. These portions were designed and installed to Seismic Category I standards, however, they do not perform a safety related function.

(82) Portion of external pneumatic connection beyond 59-*137 is seismic category IIA. The remainder of Item F.6 is seismic category II.

(83) The Emergency Diesel Generator Air Cooler Coolant, Jacket Water, and Lube Oil Cooler Heat Exchangers, sub-assembly parts purchased after 11/2000 have been procured in accordance with US NRC Generic Letter 89-09 and ASME Section III, Class 3 equivalent requirements.

(84) A portion of the piping connecting the Unit 2 RCIC Pump suction to the RHR heat exchangers has been downgraded to ANSI/ASME B31.1 Power Piping and Seismic Category IIA because it no longer serves a safety-related function.

Table 3.2-1 (Cont'd)

(85) The original RPV strongback was replaced in 1994 with a RPV strongback carousel by GE that was designed and fabricated to ANSI N14.6-1978. GE later identified that the hook pins, clevis pins, and clevis rods did not meet the material testing as required by paragraph 3.2.6 of ANSI N14.6-1978. Because the new RPV strongback carousel was proof tested and is periodically inspected and load tested, its design function to provide a single-failure proof device to lift the RPV head during refueling is not impacted although it is not completely compliant to ANSI N14.6-1978.

(86) This system has been categorized under 10 CFR 50.69. See Section 13.5.5 for further information.

(87) Components within this system have been categorized under the 10 CFR 50.69 process. This is based on the 10 CFR 50.69 system boundaries as defined in the associated System Categorization Document. See Section 13.5.5 for further information.

Table 3.2-2

CLASSIFICATION AND CODE COMPLIANCE REQUIREMENTS FOR NON-NSSS MECHANICAL COMPONENTS

	CODE CLASSIFICATIONS			
COMPONENT	GROUP A	GROUP B	GROUP C	GROUP D[1]
Pressure Vessels	ASME B&PV Code, Section III, Nuclear Power Plant Components-CLASS 1	ASME B&PV Code, Section III, Nuclear Power Plant Components-CLASS 2	ASME B&PV Code, Section III, Nuclear Power Plant Components-CLASS 3[5]	ASME B&PV Code, Section VIII, Division 1
Piping Systems [4] (including pipe supports)	As above [3]	As above [3]	As above [3]	ANSI B 31.1 Power Piping
Pumps	As above	As above	As above	Manufacturer's Standards
Valves	As above	As above	As above	ANSI B 31.1
0-15 psig Storage Tanks	-	As above	As above	API 620, or ASME B&PV Code Section VIII, Division 1
Atmospheric Storage Tanks	-	As above	As above [2]	API 650, AWWA D 100, ANSI B 96.1, or ASME B&PV Code Section VIII, Div. 1

(1) Certain portions of the radwaste systems meet the additional requirements of Quality Group D (Augmented), as defined in BTP ETSB 11-1, Parts B.4 and B.5.

(2) Atmospheric storage tanks fabricated to Group C requirements may be used in a Group D or Group D (Augmented) system.

(3) ASME Section III piping systems (including pipe supports) are constructed in accordance with ASME Section III, 1971 Edition with Addenda through Winter 1971, except:

a. Piping material conforms to ASME Section III, 1971 Edition with Addenda through Winter 1971, or to later editions or addenda approved by the AE.

b. Field fabrication, installation, examination, and testing are in accordance with ASME Section III, 1974 Edition with Addenda through Winter 1974.

c. Paragraphs NC-4436 and ND-4436 of ASME Section III, 1980 Edition with Addenda through Winter 1981, is used for installation of attachments to Class 2 and 3 piping systems after testing.

Table 3.2-2 (Cont'd)

d. Pipe supports are constructed and installed in accordance with the 1969 Edition of ANSI B31.7 with Addenda approved March 10, 1971. Snubbers supplied through the architect-engineer are manufactured in accordance with ASME Section III, 1977 Edition with Addenda through Winter 1977, and installed in accordance with the 1969 Edition of ANSI B31.7 with Addenda approved March 10, 1971.

e. Stress analysis is in accordance with ASME Section III, 1971 Edition with Addenda through Winter 1972, except:

1. Class 1 piping systems stress analysis is in accordance with the 1977 Edition with Addenda through Summer 1979.

2. Class 2 and 3 flange stress analysis is in accordance with the 1977 Edition with Addenda through Summer 1979.

3. Unit 2 only. The SIF values for nuclear class 2 and 3 weldolets, sockolets and half coupling branch connections are in accordance with the 1974 Edition of the code.

f. Containment penetration flued heads are manufactured in accordance with ASME Section III with Addenda through Summer 1974, and installed in accordance with item b above. Diaphragm penetration flued heads are manufactured in accordance with ASME Section III, 1980 Edition with Addenda through Winter 1981, and installed in accordance with item b above.

g. Group B and C instrument sensing lines that are attached to Group A instrument sensing lines are hydrotested in accordance with ASME Section III, 1974 Edition with Addenda through Winter 1975.

h. Orifice plates (which are clamped between flanges and used in flow measuring service) that do not exceed ½ inch nominal thickness are not considered to be an ASME piping subassembly, part, appurtenance, component, or material in accordance with Paragraph NCA-1273 of ASME Section III, 1980 Edition with Addenda through Summer 1980.

i. For installation of instrument lines supplied through the architect-engineer, the minimum fillet weld size is in accordance with figure NB/NC/ND-4427-1 of ASME Section III 1980 Edition with Addenda through Summer 1980.

j. The NSSS piping systems supplied by GE and installed by the architect-engineer are installed in accordance with ASME Section III, 1974 Edition with Addenda through Winter 1974. Supports are manufactured and installed and snubbers are installed for the GE-supplied systems in accordance with ANSI B31.7, 1969 Edition with Addenda approved March 10, 1971. Snubbers are manufactured in accordance with ASME Section III, 1977 Edition with Addenda through Winter 1977.

k. Paragraph NB-4436 of ASME Section III, 1977 Edition, is used for installation of attachments to Class 1 piping systems after testing.

l. Subparagraph N(X)-4453.1 of ASME Section III, 1983 Edition with Addenda through Summer 1983, is used for making repairs to welds in Class 1, 2, and 3 piping systems.

m. Unit 2 only. Nameplate removal/replacement of N-stamped items shall be in accordance with the 1983 Edition through the Winter of 1984 Addenda, Paragraph NCA-8240.

n. Unit 2 only. Venting prior to hydrostatic test shall be in accordance with the 1980 Edition through the Summer 1981 Addenda, Paragraph NB/NC/ND-6211.

(4) CRD and TIP piping systems are constructed in accordance with ASME Section III, 1974 Edition with Addenda through Summer 1976. For evaluation of CRD insert/withdrawal piping interference with the CRD housing, Paragraph NC-3600 of the 1974 Edition with Addenda through Winter 1976 is used.

(5) The Emergency Diesel Generator Air Cooler Coolant, Jacket Water, and Lube Oil Cooler Heat Exchangers, sub-assembly parts purchased after 11/2000 have been procured in accordance with US NRC Generic Letter 89-09 and ASME Section III, Class 3 equivalent requirements.

Table 3.2-3

CLASSIFICATION AND CODE COMPLIANCE REQUIREMENTS FORNSSS MECHANICAL COMPONENTS

Group Classification	ASME III Code 1968 Ed.	Classes 1971 Ed.	Components Ordered on or after January 1, 1970 to July 1, 1971	Components Ordered on or after July 1, 1971
A	A	1	ASME I ASME III, A ASME IX ANSI B16.5 ANSI B16.11 ANSI B31.1 ANSI B31.7, I NP&VC, I TEMA C (2)	ASME I ASME III, 1 ASME IX ANSI B16.5 ANSI B31.7 NA&NB Subsections TEMA C (2) (5)
B	B[(1)],C	2, MC[(1)]	ASME III, B[(1)],C[(7)] ANSI B31.7, II NP&VC, II[(8)] TEMA C TANKS	ASME III,2 & MC[(1)] NA&NC Subsections NA&NE Subsections TEMA C TANKS[(6)] (5)
C	-	3	ASME VIII, Div. 1 ANSI B31.7, III NP&VC, III TEMA C TANKS (5)	ASME III, 3 NA&ND Subsections TEMA C TANKS (5)
D	-	-	ASME VIII, Div. 1 ANSI B31.1.0 TEMA C TANKS[(3)] (4)	ASME VIII, Division 1 ANSI B31.1 TEMA C TANKS[(3)] (4)

Table 3.2-3 (Cont'd)

(1) Metal containment vessel (as applicable) and extensions of containment only.

(2) PECo's request to use alternative codes to those required in 10CFR50.55a for primary pressure boundary components was sent to the NRC on July 15, 1975. The applicable codes, code dates, and addenda are listed in the request letter for the following components that are still included in the LGS design:

a. PRV
b. Main steam SRVs
c. Main steam piping (26 inches) from RPV to second isolation valve
d. Main steam line suspension
e. Recirculation pump
f. Recirculation gate valves (motor-operated):
   28 inch suction
   28 inch discharge
g. Recirculation loop piping (28 inches)
h. Recirculation loop suspension
i. RHR valves
   20 inch gate valve
j. Core spray valves
   12 inch check valve (air operated)
k. Nuclear Class 1 piping (all except main steam and recirculation)

NRC approval of the request was given on November 18, 1975 in a letter from R.C. De Young (NRC) to E.G. Bauer (PECo).

The applicable code for design, fabrication, and testing of the MSIVs is the ASME Standard Code for Pumps and Valves for Nuclear Power - 1968 Draft including March 1970 Addenda.

Paragraphs NB-4433 and NB-3123.2 of the 1980 Edition of the ASME Section III are applicable for the design of hanger lugs welded to the main steam piping pressure boundary.

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

(4) For pumps classified Group D and operating above 150 psi or 212°F, ASME Section VIII, Division 1 shall be used as a guide in calculating the wall thickness for pressure- retaining parts and in sizing the cover bolting. For pumps operating below 150 psi and 212°F, manufacturer's standard pump for service intended may be used.

Table 3.2-3 (Cont'd)

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

(6) The SLCS storage tanks were designed and constructed to API 650; ASME Section III, Class 2 did not contain Article NC-3800 on atmospheric storage tanks at the time these tanks were purchased.

(7) All three of the Unit 1 and one of the Unit 2 SLCS accumulator vessels are ASME Section VIII, Division 1. The ASME Section III, Class C code requires only that these weldless vessels meet the ASME Section VIII, Division 1 standards, and hence, except for the code stamping, they meet all ASME Section III, Class C requirements.

(8) The HPCI turbine exhaust line globe stop-check valves meet the requirements of the cited code but were manufactured to ANSI B31.1 (1967), ANSI B16.5, and MSS-SP-66 requirements.

Table 3.2-4

OFF-SKID PORTIONS OF THE EMERGENCY DIESEL GENERATOR AUXILIARY SYSTEMS

ASME Section III Class 3	ANSI B31.1	LGS Supplementary Requirements
Requires ASME materials and CMTRs for all piping larger than ¾ inch nominal pipe size. Certificates of compliance may be substituted for CMTRs for piping less than ¾ inch.	Requires materials that conform to either ASME or ASTM specification.	ASTM and ASME materials were procured and CMTRs were supplied for all piping larger than ¾ inch nominal pipe size. Certificates of compliance may be substituted for CMTRs for piping less than ¾ inch.
Requires seismic design in addition to the B31.1 requirements.	Requires design for pressure, temperature, and normal operating loads.	Piping is designed to seismic Category I with minimum wall thicknesses in conformance with ASME Section III, Class 3.
Requires liquid penetrant, magnetic particle, or radiographic examination for circumferential welds greater than 2 inches nominal pipe size.	Requires only visual inspection of welds at the design pressure and temperature of the auxiliary systems.	Same as ANSI B31.1.
Requires pneumatic testing at 1.2x design pressure.	Requires pneumatic test at 1.2x design pressure with initial service leak test as an alternative.	All piping is pneumatically tested to 1.2x design pressure except for piping with design pressure at or below 0 psig.

Table 3.2-5

CHILLED WATER PIPING SYSTEM FOR THE CONTROL STRUCTURE CHILLED WATER SYSTEM

ASME Section III Class 3	ANSI B31.1	LGS Supplementary Requirements
Requires ASME materials and CMTRs for all piping larger than ¾ inch nominal pipe size. Certificates of compliance may be substituted for CMTRs for piping less than ¾ inch.	Requires materials that conform to either ASME or ASTM specification.	ASTM and ASME materials were procured and CMTRs were supplied for all piping larger than ¾ inch nominal pipe size. Certificates of compliance may be substituted for CMTRs for piping less than ¾ inch.
Requires seismic design in addition to the ANSI B31.1 requirements.	Requires design for pressure, temperature, and normal operating loads.	Piping is designed to seismic Category I with minimum wall thicknesses in conformance with ASME Section III, Class 3.
Requires liquid penetrant, magnetic particle, or radiographic examination for circumferential welds greater than 2 inches nominal pipe size.	Requires only visual inspection of welds at the design pressure and temperature of the chilled water system.	Same as ANSI B31.1.
Requires hydrostatic testing at 1.25x design pressure.	Requires hydrostatic test at 1.5x design pressure.	Same as ANSI B31.1.

Table 3.2-6

CONTROL ROOM HVAC CHILLERS

ASME Section III Class 3	ASME Section VIII Division 1	LGS Supplementary Requirements
Requires use of ASME materials that are listed in the stress tables or in ASME Section VIII for nonferrous materials.	Requires use of ASME materials that are listed in the stress tables.	Materials used in the vessels are permitted by ASME Section III, with the exception of SA306, Grade 60 bar used for the vessel water box flange. Use of this material is permitted by ASME Section VIII.
Requires CMTRs.	CMTRs or certificates of compliance not required.	CMTRS were provided for the pressure retaining material, with the exception of some vessel nozzles.
Requires examination of materials in accordance with the ASME material specification for the product forms involved.	Requires examination of materials in accordance with the ASME material specification.	
Requires the vessel design to be in accordance with ASME Section VIII, Division 1.	Provides rules for vessel design.	Requires vessel to be designed to seismic Category I requirements.
Requires the vessel fabrication to be in accordance with ASME Section VIII, Division 1.	Provides rules for vessel fabrication.	
Requires the vessel weld examination to be in accordance with ASME Section VIII, Division 1.	Provides rules for vessel weld examination.	
Requires hydrostatic testing at 1.5x design pressure.	Requires hydrostatic testing at 1.5x design pressure.	

Table 3.2-6 (Cont'd)

ASME Section III Class 3	ASME Section VIII Division 1	LGS Supplementary Requirements
Requires the manufacturer to implement a quality control system.	Requires the manufacturer to implement a quality control system.	
Requires authorized inspector and Code Data Report.	Requires authorized inspector and Code Data Report.	
Requires the material manufacturer to document and maintain a QA program.	No requirement.	Requires the material manufacturer to document and maintain a QA program.

Table 3.2-7

RHR Heat Exchanger "B"
Corrosion Monitoring Components
Connected to RHRSW System Pressure Boundary

ASME Section III, Class 3	ANSI B31.1	LGS Supplementary Requirements
Requirements ASME materials and CMTRs for all piping larger than 3/4 inch nominal pipe size. Certificates of compliance may be substituted for CMTRs for piping less than 3/4 inch.	Requires materials that conform to either ASME or ASTM specification.	ASTM and ASME materials were procured and CMTRs were supplied for all ASME piping larger than 3/4 inch nominal pipe size. Certificates of compliance may be substituted for CMTRs for ASME piping less than 3/4 inch.
Requires seismic design in addition to the ANSI B31.1 requirements	Requires design for pressure, temperature, and normal operating loads.	Piping is designed to seismic Category I with minimum wall thicknesses in conformance with ASME Section III, Class 3.
Requires liquid penetrant, magnetic particle, or radiographic examination for circumferential welds greater than 2 inches nominal pipe size.	Requires only visual inspection of welds at the design pressure and temperature of the chilled water system.	Same as ASME
Requires hydrostatic testing at 1.25X design pressure.	Requires hydrostatic test at 1.5X design pressure.	Same as ANSI B31.1, for HBD components. Same as ASME for GBC and GBD components.
Requires use of ASME materials that are listed in the stress tables or in ASME Section VIII for nonferrous materials.	Requires use of ASME materials that are listed in the stress tables.	Materials used in the vessels are permitted by ASME Section III.
Requires CMTRs.	CMTRs or certificates of compliance not required.	CMTRs were provided for the pressure retaining material.

Table 3.2-7 (Cont'd)

RHR Heat Exchanger "B"
Corrosion Monitoring Components
Connected to RHRSW System Pressure Boundary

ASME Section III, Class 3	ANSI B31.1	LGS Supplementary Requirements
Requires examination of materials in accordance with the ASME material specification for the produce forms involved.	Requires examination of materials in accordance with the ASME material specification.	
Requires the vessel design to be in accordance with ASME Section VIII, Division 1.	Provides rules for vessel design.	Requires vessel to be designed to seismic Category I requirements.
Requires vessel fabrication to be in accordance with ASME Section VIII, Division 1.	Provides rules for vessel fabrication.	
Requires the vessel weld examination to be in accordance with ASME Section VIII, Division 1	Provides rules for vessel weld examination.	
Requires hydrostatic testing at 1.5X design pressure.	Requires hydrostatic testing at 1.5X design pressure.	
Requires the manufacturer to implement a quality control system.	Requires the manufacturer to implement a quality control system.	
Requires authorized inspector and Code Data Report.	Requires authorized inspector and Code Data Report.	
Requires the material manufacturer to document and maintain a QA program.	No requirement.	Requires the material manufacturer to document and maintain a QA program.

3.3 WIND AND TORNADO LOADINGS

3.3.1 WIND LOADINGS

Design wind loads for all exposed structures are based on Reference 3.3-1 and Reference 3.3-2.

3.3.1.1 Design Wind Velocity

Exposed structures are designed to withstand a basic wind velocity of 90 mph at 30 feet above ground. The recurrence interval of this wind velocity is estimated to be at least 100 years (Reference 3.3-1). A gust factor of 1.1 is used in conjunction with this basic wind velocity. The variation of wind velocity with height is given in Table 3.3-1.

3.3.1.2 Determination of Applied Forces

The dynamic pressures on the exposed structural surfaces due to the design wind are computed in accordance with the requirements of Reference 3.3-1, which are summarized as follows:

$$q = 0.002558V^2$$

where:

q = dynamic pressure (psf)

V = design wind velocity, mph

$$V = G\ (v)$$

where:

G = gust factor

v = basic wind velocity

The total design pressure for a structure is the product of dynamic pressure and shape coefficient. A shape coefficient of 1.3 is applied with all wind loads. This total design pressure is distributed among different exposed surfaces of the structure, on the basis of location with respect to the wind direction. Pressures and suction developed by using the above procedure and used in the design of plant structures are given in Table 3.3-1.

Table 3.3-4 provides a comparison between the maximum expected external pressure and the external design pressures. This table shows that an adequate margin of safety for wind loadings exists for the secondary containment structure.

3.3.2 TORNADO LOADINGS

Structures that directly affect the ultimate safe shutdown of the plant are designed to resist applicable design basis tornado forces. Table 3.3-2 lists the systems and components that are protected against tornados and the enclosures that provide this protection. The radwaste enclosure is tornado-resistant only to the extent of protecting the gaseous radwaste treatment system and retaining approximately 500,000 gallons of solid and liquid radwaste within the confines of the enclosure.

3.3.2.1 Applicable Design Parameters

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

a. Dynamic wind loadings - These are the external pressure or suction forces on a structure due to the passage of a tornado funnel. The design basis tornado has a rotational speed of 300 mph and a translational speed of 60 mph. Conservatively, this is taken as a 300 mph wind applied uniformly over an entire structure.

b. Differential pressures - When the low pressure within a tornado funnel engulfs a structure, a rapid depressurization occurs and produces differential pressures between the inside and outside of the structure and between the compartments inside the structure depending on the available vent paths. The pressure transient caused by the design basis tornado is a 3 psi pressure drop at the rate of 1 psi/sec, followed by a 2 second calm and then a repressurization to the original pressure at a rate of 1 psi/sec.

c. Tornado missiles - The types of missiles postulated to be generated by a tornado are discussed in Section 3.5.1.4.

Regulatory Guide 1.76, "Design Basis Tornado for Nuclear Power Plants", calls for a pressure drop of 3 psi at the rate of 2 psi/sec, however, the increased depressurization rate of 2 psi/sec in Regulatory Guide 1.76 would have no significant effect on the external structural elements of tornado-resistant structures. These structures are designed for the maximum pressure differential due to a tornado of 3 psi regardless of the rate of depressurization.

The increased depressurization rate would increase the maximum differential pressure for internal structural elements. These elements were checked for the maximum differential pressure caused by the following design basis tornado pressurization profile: a 1 psi/sec pressure decrease for 3 seconds; a 2 second calm; a 1 psi/sec pressure increase for 3 seconds. An analysis to determine the effects of the increased maximum differential pressures on the internal structural elements has not been performed because the 1 psi/sec depressurization rate is considered to be conservative. The design basis tornado pressurization profile used was committed to in the LGS PSAR, Appendix C, Section C.2.4, prior to the issuance of Regulatory Guide 1.76. This pressurization profile was based on Reference 3.3-4. Previous documents used as the design basis for tornado effects were References 3.3-5 and 3.3-6. Actual tornado parameters were developed by studies of the tornado damage, eyewitness accounts of the maximum tornado depressurization on barometric instruments, and analysis of films of actual tornadoes. The depressurization effects defined in References 3.3-4, 3.3-5, and 3.3-6 substantiate the conservatism of the 1 psi/sec depressurization rate because they exceed the observed effects of actual tornadoes.

To further demonstrate the conservatism of the 1 psi/sec depressurization rate, site specific tornado parameters have been determined for LGS using the methodology of Reference 3.3-7. This reference is the basis document for Regulatory Guide 1.76.

In accordance with Regulatory Guide 1.76, LGS is in zone I, a region which encompasses a wide variation of tornado risks, from high tornado risk areas such as the Midwest to low risk areas such as New England. For the following analysis, a site specific tornado risk data base is used to determine the design basis tornado parameters.

Tornado data was obtained from the National Severe Storms Forecast Center for the years 1950 through 1981 for an area 125 nautical miles in radius centered on Pottstown, Pennsylvania. During this period, these were 322 tornadoes, or 10.1 tornadoes per year.

The probability that a tornado will strike a particular area is given by Reference 3.3-7 as:

$$P_s = \bar{n}\,(a/A)$$

where:

P_s = tornado strike probability

$\bar{n}$ = average number of tornadoes per year

a = average individual tornado area

A = land area within 125 nm of Pottstown (1 square nautical mile = 1.32 square miles)

From the tornado data, tornado areas were calculated for 307 tornadoes (data was not available for 15 tornadoes). The average area was 0.24 square miles. The land area within 125 nm of Pottstown is approximately 53,500 square miles (the total area is 64,900 square miles), of which 82% is land area. Therefore, the average probability of a tornado strike is:

$$P_s = \frac{10.1(0.24)}{53{,}500} = 4.6\text{x}10^{-5} \text{ per year}$$

In accordance with Reference 3.3-7, the probability of occurrence of a tornado that exceeds the design basis tornado should be on the order of 10^{-7} per year to adequately protect public health and safety. Therefore:

$$P_s \cdot P_i \leq 10^{-7}$$

where:

P_i = acceptable intensity probability

Thus,

$$P_i = 10^{-7}/4.6\text{x}10^{-5} = 2.2\text{x}10^{-3} \text{ per year} = 0.22\%$$

Each tornado in the tornado data base has been classified according to a wind speed scale (the Fujita parameters). The distribution of tornadoes with respect to wind speed is given in Table 3.3-3. The cumulative distribution from Table 3.3-3 is plotted in Figure 3.3-1. Using Figure 3.3-1, the maximum wind speed corresponding to a probability P_i of 0.22% is 280 mph.

To determine the rotational and translational components of the maximum wind speed, the values obtained from table 4 of Reference 3.3-7 are used for interpolation. The values thus obtained are translational wind speed of 56.5 mph and rotational wind speed of 223.5 mph.

The depressurization rate is calculated by Reference 3.3-7 as follows:

$$\frac{dp}{dt} = \frac{T}{r_m} \rho V_m^2$$

where:

p = pressure

t = time

T = translational wind speed

V_m = rotational wind speed

ρ = density of air

r_m = radius of maximum rotational wind speed

From table 4 and table 5 of Reference 3.3-7, the parameters for a Region I design basis tornado are:

T = 70 mph

V_m = 290 mph

$\frac{dp}{dt}$ = 2 psi/sec

Therefore, by ratio

$$\frac{dp_2}{dt} = \frac{dp_1}{dt} \frac{T_2 V_{m2}^2}{T_1 V_{m1}^2} = \frac{2.0 \cdot 56.5 \cdot (223.5)^2}{70 \cdot (290)^2}$$

$$= 0.96 \text{ psi/sec}$$

These site specific tornado parameters are less limiting than the values used for the design basis tornado in Section 2.3.1.

3.3.2.2 Determination of Forces on Structures

The procedures used to transform the tornado loadings into effective loads on structures are described below under separate headings for each parameter described in Section 3.3.2.1.

a. Dynamic wind loadings - The procedure used to transform the tornado wind velocity into an effective pressure is the same as described in Section 3.3.1.2, with the following exceptions: velocity and velocity pressure are not assumed to vary with height, and a gust factor of 1.0 is used. The resulting pressure loadings are as follows:

1. Pressure on windward side (0.8q) 185 psf

2. Suction on leeward side (0.5q) 115 psf

3. Total pressure on structure (1.3q) 300 psf

4. Uplift on roof (0.6q) 140 psf

b. Differential pressures - The maximum differential pressure between the inside and outside of fully enclosed areas is taken as 3.0 psi. Blowout panels are provided where necessary to lower the design differential pressures. For vented compartments, a flow analysis of all interconnecting air volumes is performed and the maximum transient pressure differentials across walls, floors, and roofs are calculated. All structural components in the vented and nonvented compartments are then checked to confirm that they can withstand the maximum calculated transient differential pressure.

c. Tornado missiles - The procedures and methods outlined in Reference 3.3-3 are used to transform the dynamic loads into effective loads and to determine the structural response of the elements subjected to missile impingement.

All of the above design basis tornado loadings are considered as loadings that act simultaneously.

Analytical techniques were used for estimating the values of tornado parameters for purposes of design with an adequate level of conservatism. Although the tornado parameters used in the LGS external design pressure analyses differ slightly from those defined in Regulatory Guide 1.76, the LGS analyses are either equivalent or conservative compared to analyses using the Regulatory Guide 1.76 parameters (Section 2.3.1.2.4). The use of these parameters for calculating the external design pressure ensures an adequate margin above the maximum expected external pressure.

3.3.2.3 <u>Effect Of Failure Of Structures Or Components Not Designed For Tornado Loadings</u>

Structures such as the turbine enclosure and outdoor storage tanks, which are not designed for tornado loads, are investigated and checked to ensure that they cannot produce missiles that have more severe effects than those listed in Section 3.5.1.4. The modes of failure of these structures are analyzed to verify that their failure due to tornado loading cannot prevent structures or components needed for safe shutdown from performing their intended functions.

3.3.3 REFERENCES

3.3-1 "Wind Forces on Structures", Transactions, Volume 126, Part II, ASCE paper No. 3269, (1961).

3.3-2 H.C. Thom, "New Distributions of Extreme Winds in the United States", Journal of the Structural Division, ASCE, (1969).

3.3-3 "Design of Structures For Missile Impact", BC-TOP-9-A, Rev. 2, Bechtel Power Corporation, (September 1974).

3.3-4 J.A. Dunlop and K. Wiedner, "Nuclear Power Plant Tornado Design Considerations", Journal of the Power Division, Proceedings of the ASCE, (March 1971).

3.3-5 "Design Criteria for Nuclear Power Plants Against Tornadoes", Bechtel Power Corporation, B-TOP-3, (March 12, 1970).

3.3-6 "Tornado Criteria for Nuclear Plants", Bechtel Power Corporation, (July 1969).

3.3-7 "Technical Basis for Interim Regional Tornado Criteria", AEC, WASH-1300, (May 1974).

Table 3.3-1

DESIGN WIND LOADS ON STRUCTURES

			WALL LOAD[2]			
HEIGHT ZONE (ft)	BASIC WIND VELOCITY v (mph)	DYNAMIC PRESSURE, q[1] (psf)	Total Design Pressure, 1.3 q (psf)	Windward Pressure, 0.8 q (psf)	Leeward Suction, 0.5 q (psf)	Roof Load Suction, 0.6 q (psf)
0-50	90	25	33	20	13	15
50-150	105	34	44	27	17	20
150-400	125	48	63	39	24	29
>400	135	56	73	45	28	34

(1) Includes gust factor of 1.1

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

Square or rectangular structures - 1.00

Round or elliptical structures - 0.60

Table 3.3-2

TORNADO PROTECTED SYSTEMS AND TORNADO-RESISTANT ENCLOSURES

PROTECTED SYSTEM OR COMPONENT	TORNADO-RESISTANT ENCLOSURE
RCPB	Reactor enclosure
ECCS	Reactor enclosure
RHR system	Reactor enclosure
RHRSW system	Reactor enclosure and spray pond pumphouse
ESW system	Reactor enclosure, diesel generator enclosure, control structure, and spray pond pumphouse
RECW system	Reactor enclosure
Fuel pool cooling system	Reactor enclosure
Fuel pool	Reactor enclosure
CRD hydraulic system	Reactor enclosure
SLCS	Reactor enclosure
Standby diesel generators	Diesel generator enclosure
Gaseous radwaste system	Radwaste enclosure
Control room	Control structure
CSCWS	Control structure
Various electrical, instrumentation, and control equipment required for safe shutdown	Reactor enclosure, diesel generator enclosure, control structure, and spray pond pumphouse
SGTS *	Reactor enclosure
RERS *	Reactor enclosure

Table 3.3-2 (Cont'd)

NOTE *	The SGTS and the RERS are provided to reduce airborne radiation concentrations in the secondary containment prior to release to the environment for low probability events such as a LOCA. They are not required to achieve safe shutdown. In accordance with Regulatory Guide 1.117, such systems are not required to be protected against tornadoes. However, they are totally enclosed in the reactor enclosure and control structure, which are tornado-resistant enclosures.

Table 3.3-3

WIND SPEED AND CUMULATIVE WIND SPEED DISTRIBUTION FOR TORNADOES WITHIN 125 nm OF POTTSTOWN, PA

Wind Speed Classification	No. of Tornadoes	Percent of Total	Cumulative Percentage
F5 (261 - 308 mph)	0	0.0	0.0
F4 (207 - 260 mph)	0	0.0	0.0
F3 (158 - 206 mph)	15	4.9	4.9
F2 (113 - 157 mph)	93	30.2	35.2
F1 (73 - 112 mph)	154	50.2	85.3
F0 (40 - 72 mph)	43	14.0	99.3
F-1 (<40 mph)	2	0.7	100.0

Table 3.3-4

CALCULATION OF MARGIN BETWEEN DESIGN
AND MAXIMUM EXPECTED WIND LOADINGS

Height Above Ground (Feet)	Fastest Mile[1] (mph)	Max. Expected Wind Pressure on Wall[2] (psf)	Basic Wind Velocity[3] (mph)	Total Design Pressure on Wall[3] (psf)	Margin[4]
30	82	27	90	33	1.22
100	97	38	105	44	1.16
200 (Reactor Enclosure Roof)	108	47	125	63	1.34
300	114	52	125	63	1.21

(1) The fastest mile values of wind are taken from Table 2.3.1-8.

(2) The maximum expected wind pressure on the wall included a gust factor of 1.1 and a shape factor of 1.3 corresponding to its respective fastest mile velocity. The approach described in Section 3.3.1.2 can be used to calculate the maximum expected wind pressure, i.e., 0.002558 (fastest mile $V \cdot 1.1)^2 \cdot 1.3$ or $0.004024V^2$.

(3) Data are taken from Table 3.3-1.

(4) The horizontal and vertical components of the wind pressure are based on the height above ground. Therefore, the margin associated with roof pressure (suction) will be the same as that due to the horizontal pressure on the walls (i.e. 34%).