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

TABLE OF CONTENTS

TABLE OF CONTENTS (Cont'd)

TABLE OF CONTENTS (Cont'd)

TABLE OF CONTENTS (Cont'd)

TABLE OF CONTENTS (Cont'd)

TABLE OF CONTENTS (Cont'd)

TABLE OF CONTENTS (Cont'd)

TABLE OF CONTENTS (Cont'd)

TABLE OF CONTENTS (Cont'd)

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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. A carbon dioxide system provides protection for the cable spreading room. 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.

GDC 4 - Environmental and Missile Design Bases

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

Design Evaluation

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.

GDC 5 - Sharing of Structures, Systems, and Components

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

Design Evaluation

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.

GDC 19 - Control Room

> 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.

Design Evaluation

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 required for safety Section 7.6

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.

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.

Design Evaluation

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.

GDC 51 - Fracture Prevention of Containment Pressure Boundary

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

Design Evaluation

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.

GDC 55 - Reactor Coolant Pressure Boundary Penetrating Containment

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.

Design Evaluation

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.

GDC 56 - Primary Containment Isolation

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 143°F 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							
	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							
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]	LOCA-TION[2]	QUALITY GROUP CLASSI-FICATION[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							
	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							
	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							
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							
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							
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							
	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							
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							
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							
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							
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							
	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)
		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)
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)
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)	LOCATION(2)	QUALITY GROUP CLASSIFICATION(3)	PRINCIPAL CODES AND STANDARDS(4)	SEISMIC CATEGORY(5)	Q-LIST(6)	COMMENTS
5. SGTS Equipment Compartment HVAC System	9.4.1.5							(74)
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)
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								
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							
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								
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							
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							
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							
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							
	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)

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
B. Engineered Safety Features Systems	7.3							
1. ECCS: HPCI, ADS, CS, LPCI (RHR)		GE	C,R	-	IEEE 279	I	Y	(45)
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)
5. Service water systems: RHRSW, ESW		P	C,R	-	IEEE 279	I	Y	(45)
6. Containment atmospheric control systems:								
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)
9. RERS		P	R	-	IEEE 279	I	Y	(45)
10. Reactor enclosure isolation system		P	R	-	IEEE 279	I	Y	(45)
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)
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)
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)
4. Remote shutdown system		GE	CS	-	MF STD	I	Y	(45)

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)
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)
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	
11. Plant Monitoring System (PMS) Unit 1 Only		GE	CS	-	MF STD	II	N	
Plant Monitoring System (PMS) Unit 2 Only		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	

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]	LOCATION[2]	QUALITY GROUP CLASSIFICATION[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							
	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]	LOCA-TION[2]	QUALITY GROUP CLASSI-FICATION[3]	PRINCIPAL CODES AND STANDARDS[4]	SEISMIC CATEGORY[5]	Q-LIST[6]	COMMENTS
E. Control Structure Chilled Water System	9.2.10							
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							
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							
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]	LOCATION[2]	QUALITY GROUP CLASSIFICATION[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.

Table 3.2-1 (Cont'd)

	TEMA C	-	Tubular Exchanger Manufacturers Association, Class C
	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

Table 3.2-1 (Cont'd)

stop valves and the piping from the stop valves to the turbine casing. The dynamic input loads 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.

Table 3.2-1 (Cont'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.

(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.

Table 3.2-1 (Cont'd)

(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.

(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.

Table 3.2-1 (Cont'd)

(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.

(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

Table 3.2-1 (Cont'd)

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 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.

Table 3.2-1 (Cont'd)

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.

(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.

Table 3.2-1 (Cont'd)

(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.

(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.

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.	ASME III 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^{o}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^{o}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:

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

where:

P_s = tornado strike probability

$\overline{n}$ = 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 Effect Of Failure Of Structures Or Components Not Designed For Tornado Loadings

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

HEIGHT ZONE (ft)	BASIC WIND VELOCITY v (mph)	DYNAMIC PRESSURE, q[1] (psf)	WALL LOAD[2] 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%).

3.4 WATER LEVEL (FLOOD) DESIGN

3.4.1 FLOOD PROTECTION

3.4.1.1 Flood Protection Measures for Seismic Category I Structures

Seismic Category I structures and the safety-related systems and components housed within them are listed in Table 3.4-1. As discussed in Section 2.4.2.2, the design basis flood level of the Schuylkill River at the site is 207 feet, including wave activity. The shortest horizontal distance from the contour at el 207' to the nearest safety-related structure is approximately 200 feet. Grade level is no lower than el 215' at any of the safety-related structures, and none of the safety-related structures has exterior openings below el 217'. Therefore, the safety-related structures are secure from Schuylkill River flooding and no special provisions for flood protection are necessary. Flooding from internal events is discussed in Section 3.6.1. The impact of local intense precipitation is discussed in Section 2.4.2.3.

Table 2.4-1 lists the access openings in safety-related structures. Two of these doors are below the design basis flood (el 207' MSL) but do not represent a flooding concern. The doors between the reactor enclosure and water pipe tunnel room 202 at el 201' are watertight. The doors between the control structure and the turbine enclosure at el 200' are watertight because the turbine enclosure is not watertight against the PMP as discussed in Section 2.4.2.3.

The failure of nonseismic Category I and nontornado protected tanks, vessels, and major pipes located outside buildings (Table 3.4-2) has also been evaluated and determined to not adversely affect safety-related structures, systems, and components as discussed below.

Tank Failure

The location of tanks in the yard area is shown in Figure 3.8-58. Failure of the tanks on the west and south sides of the power plant complex (Table 3.4-2, items 1 through 5) will not cause potential flooding of safety-related structures, systems, and components. Any flooding due to a failure of these tanks will be contained within seismic Category IIA earth dikes, which will remain stable under both static and dynamic conditions. The design of the earth dikes is discussed in Sections 2.4.12 and 2.5.5.5.

The tanks on the north side of the power plant complex (Table 3.4-2, items 6 through 9) do not have seismically designed containments around them. Failure of these tanks could cause local flooding. This flooding would not adversely affect safety-related facilities for the following reasons:

a. Surface drainage in this area will drain water towards the Schuylkill River and Possum Hollow Run before it can reach the power plant complex.

b. Seismic Category I electrical cable and duct banks located in the vicinity of these tanks are adequate, as discussed below.

Even if the above dikes were to fail, there would be no impact on other safety-related structures, systems, or components due to site drainage.

Failure of Cooling Tower Basin Wall

The failure of the cooling tower basin wall (Table 3.4-2, items 10 & 11) would not adversely affect safety-related structures, systems, and components as discussed below.

The runoff pattern of water from the cooling tower basin wall failure would be similar to that caused by intense storm precipitation as discussed in Section 2.4.2.3 and shown in Figures 2.4-4, 2.4-5 and 2.4-6. Most of the flood water from the cooling tower basin would run away from the power plant complex. The worst case flood conditions for the power plant complex would be created by a failure of the south side of the Unit 1 cooling tower basin wall. For this case, a portion of the cooling tower basin water would flow towards the turbine enclosure. Although some limited turbine enclosure flooding may occur, there would be no impact on safety-related components.

The seismic Category I electrical cable and duct banks and valve pits located in the flow path of the water from the failed cooling tower basin are adequately protected as discussed below.

Failure of Circulating Water Conduit

Failure of the conduit within the yard area between the cooling tower basin and the turbine enclosure (Table 3.4-2, item 12) will cause flooding of this area. Water from the damaged conduit will erode the soil cover and flood the yard.

The runoff pattern will be similar to that shown in Figure 2.4-4. The seismic Category I electrical cable and duct banks and valve pits, located in this area are adequate, as discussed below.

In the most severe case, all the water from the cooling tower basin could drain through the damaged conduit into the yard area between the cooling water pumphouse and turbine enclosure and cause flooding of the condenser pit. However, safety-related systems and components would not be damaged, as discussed in Section 10.4.1.3.3.

Failure of Major Yard Piping

Failure of any of the pipes identified in Table 3.4-2, items 13 through 17, may cause local flooding. However, the intensity and volume of water discharge from any of the pipes is less than that of the cooling water conduit failure discussed above and would not cause damage to any safety-related facilities. Soil erosion caused by failure of these pipes is discussed in Section 3.8.4.1.6.

Safety-related structures, systems, and components, including underground cables, will not be adversely affected by flooding or wetting caused by (1) design basis flood, (2) design basis precipitation or (3) failure of an outdoor tank or tank truck.

The safety-related structures, systems, and components located within the yard area are shown in Figure 3.8-58. Excavation for seismic Category I structures, pipelines, electrical duct banks, manholes, and underground diesel oil storage tanks are shown in Figure 2.5-37.

The design basis flood is not applicable to structures and yard facilities. The design basis flood level with respect to the Schuylkill River is 207 ft (Section 2.4.2.2), which is 10 feet lower than the lowest grade level entrance to any of the safety-related facilities.

The protection of safety-related structures, systems, and components with respect to flooding caused by the design basis precipitation is discussed in Section 2.4.2.3. The runoff patterns for flood water caused by precipitation are shown in Figures 2.4-4 and 2.4-5. Below-grade parts of structures are protected from water intrusion as stated in Table 3.4-1.

The following seismic Category I yard facilities may be susceptible to flooding:

a. 5 Valve pits for RHRSW and ESW piping

b. 8 Valve pits for diesel generator fuel oil storage tanks

The valve pits may be temporarily covered with water during an intense storm or major failure of an outdoor tank. However, the following protective features have been provided to resist resultant flooding of these facilities:

a. They are built as reinforced concrete boxes and are equipped with solid steel manhole covers with gaskets.

b. The tops of slabs of these facilities are elevated 3 to 12 inches above grade.

c. The RHRSW/ESW valve pits are equipped with drain pipes leading seepage water into normal waste drainage system.

d. The diesel generator fuel oil storage tank valve pits' drain lines have been sealed off from the normal waste drainage system to minimize the potential for back flow from the downstream drainage pipe. Administrative controls shall assure that:

    - These pits are periodically inspected (at least quarterly) at a frequency sufficient to assure that seepage water does not accumulate to a level at which it would enter the fuel oil tank or jeopardize safety related components in the pits.

    - Accumulated seepage water is pumped out of the pits and out of the collection sumps of the concrete base surrounding the fuel oil storage tanks, as necessary.

    - The pits are periodically inspected for material conditions adverse to pit flood resistance.

All electrical cables are designed to operate under water. Water absorption characteristics for all cables have been reviewed to confirm that even under flooded conditions, electrical cabling in manholes and duct banks will continue to operate properly. In addition, all electrical conduits that travel to electrical manholes outside the structures are sealed to prevent water from entering the structures through the electrical duct banks (Table 3.4-1).

The design features described above will also protect seismic Category I structures and yard facilities against internal flooding in case of failure of an outdoor tank or tank truck.

3.4.1.2 Permanent Dewatering System

All safety-related structures founded below the design basis groundwater levels can withstand the resulting hydrostatic loading. Safety-related systems and components not located within structures are founded above the maximum groundwater levels. Furthermore, the groundwater levels by safety-related structures, systems, and components are located below the top of rock. Therefore, a permanent dewatering system to protect safety-related structures, systems, and components is not required.

Groundwater does not affect the yard facilities because the design groundwater elevation (Section 2.4.13.5) is generally below yard facilities.

Rubberized flat dumbbell-type water stops have been provided at construction joints below the maximum expected groundwater level listed in Section 2.4.13.5 for all safety-related enclosures.

3.4.2 ANALYTICAL AND TEST PROCEDURES

The safety-related structures listed in Table 3.4-1 are built of reinforced concrete with below-grade exterior walls that are a minimum of 1 foot thick for yard structures and 2 feet thick for the remaining listed structures. The roofs and parapets of all safety-related structures have been analyzed to verify their ability to withstand the static loading resulting from any water confined on the top of the structures due to local intense precipitation (discussed in Section 2.4.2.3).

The safety-related structures founded below the design basis groundwater levels (see Section 2.4.13.5) have been analyzed to verify their ability to withstand the resulting hydrostatic loading.

The spray pond pump structure and other safety-related structures at the spray pond have been designed to withstand the maximum water level in the spray pond.

Table 3.4-1

FLOOD LEVELS AT SAFETY-RELATED STRUCTURES

STRUCTURE		SAFETY-RELATED SYSTEM OR COMPONENTS HOUSED IN STRUCTURE	ELEVATION OF LOWEST EXTERIOR ACCESS OPENING[(1)]	DESIGN FLOOD ELEVATION
1.	Reactor Enclosure	Primary containment, ECCS, miscellaneous safety-related systems and components	217'-0"	207.0'
2.	Diesel Generator Enclosure	Diesel generator	217'-0"	207.0'
3.	Spray Pond and Spray Pond Pump Structure	ESW system and RHRSW system	268'-0"	254.9'[(2)]
4.	Miscellaneous Yard Structures	See Figure 3.8-58	See Figure 3.8-58	207.0'

(1) Penetrations below the lowest exterior access openings typically include underground electrical and piping penetrations. These penetrations are sealed water-tight.

(2) The design basis flood level for the spray pond pumphouse in based on the design basis flood level of the spray pond (Section 2.4.8.2.1).

Table 3.4-2

YARD TANKS AND MAJOR PIPING (NONSEISMIC)

ITEM NO.	TANK OR PIPE DESCRIPTION	CAPACITY OR FLOWS	LOCATION	TYPE OF CONTAINMENT	TORNADO PROTECTION
1	Unit 1 CST tank	200,000 gal.	West of power plant complex	Earth dikes	None
2	Refueling Water	550,000 gal.	West of power plant complex	Earth dikes	None
3	Unit 2 CST tank	200,000 gal.	South of power plant complex	Earth dikes	None
4	Fuel oil storage tank	200,000 gal.	South of power plant complex	Earth dikes	None
5	Deleted				
6	Deleted				
7	Deleted				
8	Clarified water tank	200,000 gal.	Water treatment plant north of power plant complex	None	None
9	Demineralized water tank	50,000 gal.	Water treatment plant north of power plant complex	None	None
10	Cooling tower Basin 1	$7x10^6$ gal.	North of power plant complex	Reinforced concrete walls	None
11	Cooling tower Basin 2	$7x10^6$ gal.	North of power plant complex	Reinforced concrete walls	None
12	Cooling water pipes 8.0 ft diam. 4 pressure pipe each unit	450,000 gpm per unit	Between cooling tower and turbine enclosure	Underground	Soil cover
13	36" ϕ makeup water pressure pipe	30,000 gpm	From Schuylkill River to cooling tower	Underground	Soil cover
14	36" ϕ blowdown gravity pipe	18,000 gpm	From cooling tower to Schuylkill River	Underground	Soil cover

Table 3.4-2 (Cont'd)

15	36" ϕ makeup water pressure pipe	30,000 gpm	From Perkiomen Creek to cooling tower	Underground	Soil cover
16	36" ϕ service water pressure pipe	35,000 gpm	From cooling tower to turbine enclosure	Underground	Soil cover
17	12" ϕ fire loop pressure pipe	2,500 gpm	Around plant complex	Underground	Soil cover

3.5 MISSILE PROTECTION

Where possible, the seismic Category I and safety-related structures, equipment, and systems are protected from missiles through basic station component arrangement so that the missile does not cause the failure of these structures, equipment, or systems. Where it is impossible to provide protection through plant layout, suitable physical barriers are provided to isolate the missile or to shield the critical system or component. Also, redundant seismic Category I components are suitably protected so that a single missile cannot simultaneously damage a critical system component and its backup system. Missile protection provides for either safe operation or shutdown during all operating conditions, operational transients, and postulated accident conditions. Table 3.2-1 provides a tabulation of safety-related structures, systems, and components, along with their applicable seismic category and quality group classification.

3.5.1 MISSILE SELECTION AND DESCRIPTION

3.5.1.1 Internally Generated Missiles (Outside Primary Containment)

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

a. Rotating component failure missiles

b. Pressurized component failure missiles

c. Gravitationally generated missiles

3.5.1.1.1 Rotating Component Failure Missiles

The systems located outside the primary containment have been examined to identify and classify potential missiles. Redundant equipment is normally located in different areas of the plant or separated by walls so that a single missile from a rotating mass does not damage both redundant systems.

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

Additionally, the bases for considering it unlikely for rotating components, other than those identified in this section, to break through their casings and adversely impact safety-related equipment are the following:

A review of event reports on file at the Nuclear Safety Information Center, Oak Ridge National Laboratory, concerning failures of fans and missile generation indicated that a few fan failures have resulted in generation of missiles in safety-related areas of a nuclear facility. Small pump failures resulting in generation of missiles are considered more improbable than fan failures resulting in generation of missiles because pump casings are generally thicker than fan casings and pump speeds are generally lower than fan speeds. Even in the unlikely event that a rotating component does break through its casing, much of the component's kinetic energy would be dissipated in moving through the casing, thereby decreasing the probability of the component adversely damaging a safety-related component. Therefore, generation of secondary missiles from the internally generated missiles described above is not considered credible. It is an even lower probability that a rotating component would adversely affect redundant safety-related systems because redundant equipment is generally located in different areas or separated by barriers.

Large pumps, such as the RHR and CS pumps, are installed so that the impeller section is surrounded by concrete; additionally, these pumps are located in separate compartments. Therefore, they are not considered a missile hazard to other systems.

Missiles from HPCI or RCIC turbines would be contained by the walls of their compartments that form suitable barriers so that no other systems can be affected by a HPCI or RCIC turbine failure.

Other rotating equipment is not considered to constitute missile hazards because of its small size and/or the unlikelihood that it could move its rotating components through its housing. A review of the analyses of internally generated missiles performed for Palo Verde and San Onofre verified that postulated missiles from pumps and fans (e.g., a pump impeller or fan blade) typically do not have sufficient energy to penetrate the component casings. Because LGS uses pumps and fans that are generally designed and constructed in accordance with the same recognized industry codes and standards as those installed at Palo Verde and San Onofre, the results of the rigorous structural analyses conducted for those plants are indicative of the integrity of LGS equipment.

It was further verified that no internally generated missiles will cause loss of function of any system required for safe shutdown. The plant arrangement was reviewed to ensure that all essential systems or components are either remote from or separated by adequate barriers from potential missile sources.

In any case where a direct path exists between a potential missile source and equipment required for safe shutdown, potential missiles cannot impact more than a single component; therefore redundant equipment will be available to effect safe shutdown.

All HVAC fans located in safety-related enclosures are listed in Table 3.5-6, along with the reason why they are not credible missiles that could adversely affect redundant equipment needed for safe shutdown. For many fans, there are barriers that separate the potential fan blade missiles from essential systems. These barriers consist of walls, floors, and ceilings that totally enclose the potential missile and are sufficiently thick to prevent spalling. All redundant fans needed for safe shutdown are separated by adequate barriers. Class I seismic duct-work of a thickness substantially greater than normal duct-work is utilized in safety-related areas that have duct-work, which should preclude escape of a potential missile from connecting duct-work.

A review was also made of the fan installation drawings to ensure no inlet conditions to fans existed that could result in operating conditions that would impose cyclical fatigue on fan blades resulting in blade cracks. No such conditions exist.

The only fan blades that have the potential for damaging redundant components needed for safe shutdown, if they escape through their casing, are the drywell area unit cooler fans. A study was performed for these fans which demonstrated that the fan blades do not have sufficient kinetic energy to penetrate their casings. In addition, a complete physical inspection of all drywell unit cooler fan blades for blade angle, verification of proper torque on blade retaining bolts, and lock wiring of the bolts by the manufacturer's representative was conducted to ensure that no blades would loosen during operation. This inspection is documented in the quality records.

3.5.1.1.2 Pressurized Component Failure Missiles

The following are potential internal missiles from pressurized equipment:

a. Valve bonnets

b. Valve stems

c. Temperature detectors

d. Nuts and bolts

Pressurized components in systems where service pressure exceeds 275 psig are evaluated as to their potential for becoming missiles.

Most valves of ANSI 900 psig rating and above and most valves of 600 psig rating, constructed in accordance with ASME Section III, are pressure seal bonnet type valves. For pressure seal bonnet valves, valve bonnets are prevented from becoming missiles by the retaining ring, which would have to fail in shear, and by the yoke, which would capture the bonnet or reduce bonnet energy. Because of the highly conservative design of the retaining ring of these valves, bonnet ejection is highly improbable and hence the bonnets are not considered credible missiles.

The remaining valves of ANSI rating 900 psig and below are valves with bolted bonnets. Valve bonnets are prevented from becoming missiles by limiting stresses in the bonnet-to-body bolting material by requirements set forth in the ASME Section III and by designing flanges in accordance with applicable code requirements. Even if bolt failure were to occur, the likelihood of all bolts experiencing simultaneous complete severance failure is remote. The widespread use of valves with bolted bonnets and the low historical incidence of complete severance failure of bonnets confirm that bolted valve bonnets need not be considered as credible missiles.

Valve stems are not considered potential missiles if at least one feature in addition to the stem threads is included in their design to prevent ejection. Valves with back-seats are prevented from becoming missiles by this feature. In addition, air or motor-operated valve stems are effectively restrained by the valve operators.

Temperature or other detectors installed on piping or in wells are considered highly improbable missiles, since a complete and sudden failure of a circumferential weld is needed for a detector to become a missile.

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

In addition, high pressure gas cylinders and accumulators are not considered credible internally generated missiles impacting on redundant safety-related equipment for the following reasons:

a. All accumulators located in areas that contain safety-related equipment are listed in Table 3.5-3, along with the maximum pressure and temperature at which they operate. These accumulators have low stresses and operate in the "moderate energy" range. Therefore, any failures would not be of concern for missile generation.

b. Safety-related gas cylinders and accumulators are seismically supported, which ensures that they will not become gravity-generated missiles during design basis conditions. Nonsafety-related cylinders and accumulators are seismically restrained if they are located in the vicinity of safety-related components.

c. High pressure gas (H_2, O_2, N_2, Kr) cylinders meet the manufacturing and test requirements of the DOT standards, 49CFR178.37, Specification 3AA and ICC standards. Vessel wall thickness requirements of both standards are comparable to ASME Section III, Class 3 requirements. Because of this highly conservative design, any failure of these cylinders that would cause them to become missiles is highly improbable.

d. Halon gas cylinders used in the fire protection system are located in the turbine enclosure and are isolated from safety-related equipment.

3.5.1.1.3 Gravitationally Generated Missiles

Installed equipment and components in safety-related plant areas outside containment are designed and installed so that they would not present gravitational missile hazards to safety-related structures, systems, or components during or after an SSE. This is achieved for safety-related equipment by seismic Category I design and installation (in accordance with Regulatory Guide 1.29 criteria) and for nonsafety-related equipment by seismic Category IIA design and installation (Section 3.2). Nonpermanently installed equipment is either removed from the safety-related areas, adequately separated from safety-related components, or secured in place before reactor operation to ensure that it would not present a missile hazard.

The information requested in the generic letter dated December 22, 1980 regarding conformance to the criteria contained in NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants" is provided in Reference 9.1-1.

3.5.1.2 Internally Generated Missiles (Inside Containment)

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

a. Rotating component failure missiles

b. Pressurized component failure missiles

c. Gravitationally generated missiles

3.5.1.2.1 Rotating Component Failure Missiles

The most substantial piece of NSSS rotating equipment is the recirculation pump and motor. This potential missile source is covered in detail in Reference 3.5-1.

It is concluded in Reference 3.5-1 that destructive pump overspeed can result in certain types of missiles. A careful examination of shaft and coupling failures shows that the fragments do not result in damage to the containment or to vital equipment.

a. Low energy missiles (kinetic energy less than 1,000 ft-lbs):

   Low energy level missiles may be created at motor speeds of 300% of rated through failure of the end structure of the rotor. The structure consists of the retaining ring, the end ring, and the fans. Missiles potentially generated in this manner would strike the overhanging ends of the stator coils, the stator coil bracing, support structures, and two walls of ½ inch thick steel plate. Due to the ability of these structures to absorb energy, it is concluded that missiles would not escape this structure. It is at this point that frictional forces would tend to bring the overspeed sequence to a stop.

b. Medium energy missiles (kinetic energy less than 20,000 ft-lbs):

   In the postulated event that the body of the rotor bursts, medium energy missiles could be created. The likelihood that these missiles would escape the motor is considered less than the likelihood of escape for the low energy missiles described above, due to the additional amount of material constraining missile escape, such as the stator coils and stator frame directly adjacent to the rotor.

c. The motor as a potential missile:

   Since bolting is capable of carrying greater torque loads than the pump shaft, pump bolt failure is precluded. Since pump shaft failure decouples the rotor for the overspeed driving blowdown force, only those cases with peak torques less than that required to fail the pump shaft (five times rated) have the capability to drive the motor to overspeed. When missile generation probabilities are considered along with a discussion of the actual load-bearing capabilities of the system, it is evident that these considerations support the conclusion that it is unrealistic that the motor would become a missile.

It is concluded that the other rotating components inside the containment, such as fans, do not have sufficient energy to move the masses of their rotating parts through the housings in which they are contained and therefore are not considered missile hazards. Additional evaluation and discussion of the potential for rotating equipment generated missiles to impact safety-related equipment is provided in Section 3.5.1.1.1 above.

3.5.1.2.2 Pressurized Component Failure Missiles

It is concluded that potential internal missiles inside the containment from pressurized components are not considered credible for the same reasons listed in Section 3.5.1.1.2.

3.5.1.2.3 Gravitationally Generated Missiles

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

3.5.1.3 Turbine Missiles

The turbine inspection program which is being implemented in Section 10.2.3.6 uses a probabilistic approach for scheduling the inspection and replacement of the low pressure turbine rotors with shrunk-on discs. This turbine inspection program is based upon the missile probability analysis methodology in Reference 3.5-15 and the probabilistic approach established in NUREG 1048 (Reference 3.5-14). The approach in Reference 3.5-14 was found acceptable by the NRC for use in establishing maintenance and inspection schedules for specific turbine systems including the original Main Turbines installed at Limerick Generating Station. As a result of the replacement of the original Main Turbines, the missile probability analysis methodology in Reference 3.5-15 has been adopted to maintain the probabilistic approach to determine the inspection program for the installed turbine system. The overspeed protection system test interval was determined by Reference 3.5-16 in accordance with the missile probability analysis methodology in Reference 3.5-15.

The intent of the program is to ensure that the probability of a turbine generating a missile is maintained less than 1×10^{-5} per year. The program takes into account specific turbine wheel operating conditions, material properties, results of periodic in service inspections, and other factors. The program's determination of missile probability is based on the probabilities of individual parameters which may lead to the generation of the turbine missile. As a result, the program can facilitate evaluations of the effects of changes in any parameter. Table 3.5-9, Turbine System Reliability Criteria, has been extracted from table U.1 of Reference 3.5-14, for use with an unfavorably oriented turbine. The probability of unacceptable damage from turbine missile will be maintained at less than or equal to 1×10^{-7} per year.

Schedules for future inspection and replacement of low pressure turbine rotors with shrunk-on discs will be based on the probabilistic approach.

3.5.1.4 Missiles Generated by Natural Phenomena

Only tornado-generated missiles have been considered. Missiles used in the design and assessment of structures and openings are listed in Table 3.5-4. All safety-related structures, systems, and components listed in Table 3.2-1 were reviewed for adequacy against tornado-generated missiles listed in Table 3.5-4. These safety-related structures, systems, and components are designed either to resist tornado missiles in accordance with Reference 3.5-6 or are protected by these tornado-resistant enclosures. The structures designed for these tornado missiles and the systems protected are listed in Table 3.3-2. Table 3.5-8 provides information on the characteristics of these barriers. Additionally, ESW and RHRSW systems yard piping is protected by burial and separation of redundant loops.

The exterior walls and roof thicknesses have been evaluated for the tornado-resistant enclosures listed in Table 3.3-2 and are capable of withstanding all of the missiles listed in Table 3.5-4. The 4000 psi strength concrete walls and roofs have minimum thicknesses of 24 inches and 18 inches, respectively (Table 3.5-8). This exceeds the minimum acceptable missile barrier thickness requirements specified in table 1 of SRP (NUREG-0800, July 1981) section 3.5.3.

Where necessary to protect safety-related components, the doors in these enclosures have been designed to withstand the 1 inch steel rod, the utility pole, the 6 inch steel pipe, and the 12 inch pipe in addition to the original three design basis missiles. This design is in accordance with the SRP (NUREG-0800, July 1981).

The probability of any of the above tornado missiles penetrating any of the openings and structures (not specifically designed to be resistant to the above missiles) and adversely impacting safety-related components, such that safe shutdown is prevented, is considered low for the following reasons:

a. The total area of the nontornado-resistant features that have safety-related components located behind them is extremely small compared to the total area of the tornado-resistant portions of the enclosures.

b. To penetrate a nontornado-resistant feature and travel a sufficient distance to impact a safety-related component, a missile would need to strike the feature at a perpendicular angle.

c. Much of the missile's kinetic energy would be dissipated in breaking through the nontornado-resistant feature, which reduces the possibility of the missile adversely damaging a safety-related component even if it strikes one.

d. Redundant safety-related components are normally located in different areas of the plant or yard or are separated by walls so that a single tornado missile would not damage both redundant systems.

Evaluations have been carried out considering all seven tornado missiles listed in Table 3.5-4 and their penetration/impact on the openings and nontornado designed structures of the plant including manhole covers and valve pit roofs. Even though some singular safety-related components may be affected, damage would not occur to all redundant systems, and safe shutdown could be achieved even when assuming an additional single active failure.

LGS is in conformance with Regulatory Guide 1.117 regarding systems to be protected from tornado missiles except as discussed below where unacceptable damage to unprotected spray networks is not considered credible.

As described in Section 9.2.6, the ultimate heat sink at LGS is an excavated spray pond with a surface area of 9.9 acres. Four spray networks, each having 50% capacity for shutdown of two units, are provided.

Details of the spray pond excavation and finished grading are shown in Figures 3.8-55, 3.8-56, and 3.8-57. The general arrangement of the spray pond, spray networks, and spray pond pump structure is shown in Figure 9.2-6. The layout of the spray networks is shown in M-384.

As discussed in Section 3.5.1.4, all essential structures, systems, and components related to the ESW system, RHRSW system, and the UHS are protected from the effects of tornadoes and tornado missiles. Protection of the spray networks from tornado missiles is provided by location of the network piping and sprays below the surrounding grade and by physical separation of the networks:

a. In all but the spillway area, the surrounding grade is in excess of el 260' while the top of the sprays are at el 258' and the spray network piping is between el 253'-5" and el 256'-8".

b. The closest branches of adjacent spray networks are separated by 65 ft.

c. The supply piping to adjacent networks is separated by 215 ft.

d. The networks are located at a minimum distance of 72 ft from the edge of the pond.

The use of elevational differences and physical separation to provide protection of the spray pond networks from tornado missiles is justified by the following considerations:

a. Only two spray networks are required for the safe shutdown of both units.

b. The only active failure that can compromise the operability of a spray network is failure of its supply valve (HV-012-032A, B, C or D). These valves may be manually operated to isolate damaged networks or to initiate the use of undamaged networks if their controls or motors are inoperable.

c. The physical arrangement of the spray networks precludes the possibility that large missiles can damage more than one spray network due to trajectory considerations. Multiple missiles of sufficient energy and distribution to substantially damage multiple networks are unlikely. Network piping varies in size from the 30 inch diameter supply headers to the 2 inch diameter piping at the extreme ends of the distribution branches. Network piping wall thickness varies from 0.337-0.500 inch.

d. The loss of some sprays in a network does not result in substantial loss of heat removal capability for the entire network (each network contains 240 spray nozzles).

e. The design thermal performance of the spray pond is based on conservative design values of initial pond temperature and meteorology as described in Section 9.2.6.4. For all expected conditions, the margin in thermal performance would be considerably greater than the 10% margin demonstrated under design conditions. In fact, for average meteorological conditions, a single spray network is sufficient for the removal of the heat rejected from both units for at least a 24 hour period.

f. Interconnections are provided that allow the use of the cooling towers as a heat sink for ESW and RHRSW systems. Such operation may be initiated from the control room or locally by manual operation.

   It is unlikely that tornado winds would compromise the heat removal capability of the spray pond networks, or the cooling towers, to the extent that safe shutdown of the units would be affected. The spray networks have been designed to withstand design basis tornado winds. While not specifically designed to withstand design basis tornado winds, the cooling tower shell and supporting structure have been designed to withstand the following wind loading when either operating or dry:

Elevation Above Grade (ft)	Wind Velocity (mph)
30	90
150	113
200	118
300	125
400	130
500	135

The cooling towers are expected to provide sufficient heat removal capability for the safe shutdown of the units even in the event that the tower fill is extensively damaged.

g. The loss of more than two spray networks and the coincident loss of the cooling towers due to tornado missiles is unlikely due to physical separation of the cooling towers and the spray pond. The cooling towers are located approximately 600 feet from the nearest portion of a spray network.

   The likelihood of tornado winds and/or missiles affecting the safe shutdown capability of the cooling towers and spray networks at the same time is quite remote when the above described design factors are considered together with the variation in tornado intensity along its path length and width (Reference 3.5-2).

h. Tornado missiles are an insignificant contributor to plant risk because of the low frequency of occurrence of tornadoes in this region (EROL section 2.3.1.2.2) and the low likelihood of damaging missiles if one were to occur.

   Even if the safe shutdown capability of the cooling towers and spray networks were compromised by tornado effects, use of the cooling tower basins and/or UHS in a "cooling pond type" mode would allow substantial time for spray network repair. A plant procedure governs such repair activities. This procedure will contain, at a minimum, the following elements:

   1. Repair work on damaged spray networks will begin immediately, using materials, equipment, and personnel that have been verified to be available. Procedure verification will be made each year.

   2. On current loss of all UHS spray headers and all cooling tower cooling capacity, the spray pond will be operated as a closed cycle cooling pond until the temperature of the water reaches the design limit of 95°F. In this mode, water will be returned to the pond via the winter bypass line to promote thermal mixing and minimize the likelihood or recirculation.

      Under design basis conditions of initial pond temperature and meteorology, it would take approximately 6 hours for the pond to reach the 95°F limit.

Under average conditions, it would take approximately 10 hours to reach this limit. Both numbers are for two-unit, full power operation. For single-unit operation, these times would be approximately 12 hours and 20 hours, respectively. The heat rejection rate can be further reduced by depressurizing the reactor at a slower rate than 100°F/hr assumed in the design basis analysis.

3. When the pond reaches the design temperature limit, the sluice gates between the spray pond pumphouse wet wells and the spray pond will be closed. Water will be released from the cooling tower basins into the wet wells and pumped through the plant to service the required heat loads. The water will be returned to the spray pond and allowed to discharge over the blowdown weir and storm spillway.

   The two cooling tower basins contain a total of 14 million gallons. If it is conservatively assumed that only one-half of this volume of water is available, there is sufficient water to provide makeup for the ESW and RHRSW pumps, operating in a once-through mode, for an additional 4 hours. In the unlikely event that the cooling tower basin walls have failed due to tornado missiles, the additional time of 4 hours would not be available. However, the spray pond PRA demonstrates that it is extremely improbable that the four spray pond networks would not be available.

4. Sufficient makeup water can be supplied to the cooling tower basins to sustain continuous operation in this mode for a number of sources as described in Item i.

i. The Schuylkill River makeup pumphouse is located approximately 1500 ft from the nearest cooling tower, making it unlikely that the pumphouse would be damaged by a tornado that would also compromise the spray pond networks and the cooling towers. The pumphouse is powered from the 2300 V plant services switchgear. The switchgear can be fed using offsite power from either of the two plant substation via underground lines. The two substations are approximately 2000 ft apart, making it highly unlikely that both substation would be disabled by a tornado that would also compromise the spray pond networks and the cooling towers.

   While an additional source of water is available from the pump station to provide the Perkiomen makeup supply located approximately 8 miles from the plant site, no reliance is being placed on this intake for the purpose of safety analysis nor the safety licensing basis for the facility.

   If existing sources of makeup cannot be made available in a timely manner, makeup will be provided using available portable pumps of required size and capacity to pump water from the Schuylkill River to the spray pond pumphouse wet wells. The water would be pumped via a tie-in to the existing underground water pipeline from the Schuylkill River intake pumphouse to the cooling tower basins. It would flow via gravity to the pump pits. If a tie-in to the existing pipeline is not possible, the water would be pumped directly to the wetwell through temporary lines. The portable pumps that would be used are either licensee owned or rental pumps. The required pumps will be verified to be available on an annual basis.

j. Plant EOPs address the various contingency actions available to the operators to deal with degraded UHS conditions. As indicated in the above discussions, substantial time is available for corrective operator actions. If UHS capability should be lost for such a period of time that conditions degraded considerably, the EOPs would direct the use of equipment that would achieve a safe stable state regardless of UHS capability.

3.5.1.5 Missiles Generated by Events Near the Site

The safety-related structures, systems, and components were reviewed for adequacy against missiles externally generated by railroad explosions. The safety-related facilities are either designed to resist the externally generated missiles in accordance with Reference 3.5-6 or are protected by these missile-resistant barriers. The barriers designed to resist externally generated missiles and the corresponding systems and components protected by these missile barriers are listed in Table 3.5-7.

The nearest possible train explosion accident and its consequent missiles are considered to be the most severe missile-generating event that could occur near the site. The postulated missiles resulting from such an accident considered in the design of structures protecting safety-related systems are listed in Table 3.5-5. Missiles resulting from truck, industrial, and pipeline explosions would be less severe and therefore are not considered. As demonstrated in Section 2.2, there is no potential for missiles from ship or barge explosions or military installations. Descriptions of the railroad, its location relative to the plant, the railroad explosion, and explosions from other sources are given in Section 2.2.

3.5.1.6 Aircraft Hazards

The safety-related structures, systems, and components were reviewed for adequacy against missiles externally generated by aircraft accidents. The safety-related facilities are either designed to resist the externally generated missiles in accordance with Reference 3.5-6 or are protected by these missile-resistant barriers. The barriers designed to resist externally generated missiles and the corresponding systems and components protected by these missile barriers are listed in Table 3.5-7.

Airports and aircraft activity in the vicinity of the site are discussed in Section 2.2.2. Based on the airport and aircraft information provided in Section 2.2.2, an analysis has been performed using the methodology of SRP section 3.5.1.6 to demonstrate that the probability of an aircraft accident causing damage to safety-related equipment or structures sufficient to result in radiological consequences that are a significant fraction of 10CFR50.67 exposure guidelines is lower than the acceptance criteria of SRP section 3.5.1.6.

3.5.1.6.1 Design of Safety-Related Structures

Aircraft operating or capable of operating out of nearby airports are listed in Table 3.5-1. Based on the estimated maximum kinetic energy at impact and corresponding local damage to concrete elements, the Learjet is determined as the design aircraft for the design of safety-related structures.

The characteristics of the Learjet are presented in Table 3.5-2, and the associated load curve acting on structures due to the impact is shown in Figure 3.5-1. The design aircraft is assumed to impact perpendicularly to the vertical surface of the buildings or to impact at a glide angle of 15° from horizontal on the roof of buildings. The reactor enclosure, control structure, and spray pond pump structure are designed to withstand the impact of the design aircraft without loss of structural integrity. Safety-related equipment and systems located therein are designed to remain functional. The diesel generator enclosures are not designed to withstand aircraft impact.

3.5.1.6.2 Analysis Method

The probability of damage to structures containing safety-related equipment (P) is calculated from the formula:

$$P = \sum_{i=1}^{L} \sum_{j=1}^{M} N_{ij} C_{ij} A_{ij} \qquad \text{(EQ. 3.5-1)}$$

where:

L	=	number of different flight paths that affect the site
M	=	number of different types of aircraft that may constitute a hazard
N_{ij}	=	number of movements per year by aircraft j operating in flight path i
C_{ij}	=	probability of a crash per square mile per aircraft movement for aircraft j operating in flight path i
A_{ij}	=	effective plant area (square miles) for aircraft j operating in flight path i

Values for the above parameters and related assumptions are discussed in the following sections.

3.5.1.6.2.1 Aircraft Movements (N_{ij})

In addition to the movements, type of aircraft, and other particulars of aircraft operation described in Section 2.2.2, the following considerations for each of the sources of aircraft movement apply.

Pottstown-Limerick Airport - The analysis assumes that 90% of the takeoffs (45% of the movements) are towards the west, in the direction of the site, and that 10% of the landings (5% of the movements) are from the west, in the direction of the site. This is based on the VFR and IFR information provided in Section 2.2.2. The flight restrictions discussed in Section 2.2.2 are accounted for in the analysis.

Based on the approach pattern for Pottstown-Limerick Airport, the holding area is at least 3.8 miles northeast of the runway, which would locate the holding area more than 5 miles northeast from the site. Due to the distance and direction, no additional movements involved with the holding pattern were considered in calculating the aircraft crash probability.

Pottstown Municipal Airport - The proportions of single-engine and twin-engine aircraft using the airport are assumed to be 70% and 30%, respectively. Based on the information provided in Section 2.2.2, it is assumed that 50% of the takeoffs involve eastbound passes over the site, with the remainder to be north and west away from the site. Similarly, it is assumed that 50% of the landings involve northbound flight patterns over the site.
New Hanover Airport, Sunset Landing Strip and Perkiomen Valley Airport - The projected annual number of operations for these facilities located between 5 and 10 miles from the site is less than 500 D^2. In accordance with SRP section 3.5.1.6, their contributions to the total probability are extremely small and are not considered in the analysis.

Federal Airways - The traffic data presented in Section 2.2.2 are employed in the analysis. It is conservatively assumed that the Boeing 727 is representative of traffic in these airways.

LGS Plant Site Heliport - The analysis is based on 156 landings and 156 takeoffs per year. The helicopter parameters used are given in Table 3.5-1. These parameters are bounded by those of the Learjet, which was the design basis aircraft for impact.

3.5.1.6.2.2 Aircraft Crash Probability (C_{ij})

Pottstown-Limerick and Pottstown Municipal Airports

Aircraft crash probabilities from the SRP section 3.5.1.6, are used in this analysis. The SRP specifies the crash probability per square mile in an annular sector 60° on each side of the extended runway centerline as a function of the distance from the end of the runway.

Accordingly, the crash probability for U.S. general aviation at a distance of 1-2 miles of $1.5x10^{-7}$ per square mile per aircraft movement is used for Pottstown-Limerick Airport. This probability is also applied to the rotary-wing operations at that airport. For Pottstown Municipal Airport, the U.S. general aviation probability for 4-5 miles, $1.2x10^{-8}$ per square mile per aircraft movement, is used. The use of this value for Pottstown Municipal Airport is conservative because LGS lies outside the 60° sector of the extended runway centerline.

LGS Plant Site Heliport

The crash probability for the helicopter traffic using the plant site heliport is based on NTSB and FAA nationwide statistics on the estimated number of helicopter flight hours for 1979 and the number of fatal helicopter crashes during takeoff and landing during that same year, and is equal to $4.88x10^{-7}$ per takeoff or landing.

Federal Airways

For commercial aircraft in the Federal airways, a model for the probability that an aircraft crash will result at a distance (x) normal to its flight path is employed, which was developed by Solomon (References 3.5-7 and 3.5-8) using a negative exponential distribution. Solomon's model requires estimation of the deviation, which was determined by a 20° angle from the flight path. A deviation is estimated here by referencing Solomon's angle at 14,000 feet, and adding one mile for deviation in the airway. This gives the probability:

$$f(x) = \frac{1}{3.94} e^{-x/1.97} \qquad \text{(EQ. 3.5-2)}$$

Where:

x = distance in miles

The probability that an enroute crash will occur is about 0.45×10^{-9} per flight path mile, based upon U.S. commercial aviation performance in the period 1970-75 (Reference 3.5-9). Thus, the crash probability at the site per passage in each of the Federal airways can be estimated by the product of these two probabilities.

The resulting probabilities are as follows:

Airway	x (miles)	Probability (per square mile)
V29/147	1.3	5.9×10^{-11}
Pottstown VOR 320° radial	1.3	5.9×10^{-11}
V210	8.0	2.2×10^{-12}
V276	10.0	7.1×10^{-13}

3.5.1.6.2.3 Critical Target Areas (A_{ij})

The aircraft crash probability estimates the chance that a given aircraft maneuver (passage, takeoff, or landing) would result in striking the ground within a specific unit area, without regard to the obstruction of surrounding objects. If the assumed flight ray of the aircraft in crashing should pass through an obstructing object, then the probability of crash into that obstructing object is the same as for the associated unit ground area.

The determination of target areas are based on the following considerations:

a. The skid area, associated with a ground strike ahead of a wall, followed by sliding into the wall. The controlling parameters for the target area of this mode are the width of the wall normal to the skid path (augmented by the wingspan of the striking aircraft), and the distance the aircraft will slide.

b. The roof or plan area (augmented by the wingspan of the striking aircraft), which may be projected to an equal ground area regardless of the angle of flight path slope.

c. The wall shadow area, (Bh) cot (A), which is the ground area projected by a wall of vertical dimension (h), width (B) normal to the flight path (and augmented by the wingspan of the striking aircraft), with aircraft glide path making an angle (A) with the ground.

d. Shadowing of target structures by other structures.

Skid lengths are calculated from the equation provided in Reference 3.5-5:

$$L_s = \frac{6.3x10^{-6} V^2}{k} \qquad \text{(EQ. 3.5-3)}$$

where:

L_s = skid length (miles)

k = roughness factor (set equal to 2.5)

V = velocity (mph)

The glide angle is assumed to be 15° for fixed-wing aircraft. For rotary-wing aircraft, it is assumed to be 0° because of the poor aerodynamic characteristics of such aircraft. The skid length for rotary-wing aircraft is therefore considered to be zero feet.

The impact velocity for single-engine and twin-engine general aviation aircraft is estimated to be 110 mph. Therefore, from Equation 3.5-3, the skid length is 0.0305 miles or 161 feet. The impact velocity of a Boeing 727, assumed to be the average aircraft in the Federal airways, as discussed in Section 3.5.1.6.2.1, is taken as 290 mph, yielding a skid length of 0.212 miles or 1120 feet.

The analysis assumes that single-engine aircraft have a wingspan of 32 feet, twin-engine aircraft have a wingspan of 50 feet, and the wingspan of a Boeing 727 is 110 feet.

Pottstown-Limerick and Pottstown Municipal Airports

As discussed in Section 3.5.1.6.1, the reactor enclosure, control structure, and spray pond pump structure are designed to withstand the impact of the largest aircraft capable of operating out of the nearby airports, the design basis Learjet, and are therefore not considered in the calculation of critical target areas.

The spray pond itself is not susceptible to aircraft damage because it is constructed completely in excavation, as discussed in Section 2.5. Aircraft damage to the spray pond spray network is not considered because the design provides for separation and redundancy so that the loss of a network would not affect the systems safety function.

The diesel generator enclosures for Units 1 and 2 each have an east-west dimension of 108 feet, a north-south dimension of 84.5 feet, and a height of 29 feet. The two structures are separated by 54 feet in the east-west direction.

Flight paths of concern for the diesel generator enclosures are those in the east, west, and northbound directions. There are no intervening structures for northbound aircraft. The critical target area for single-engine aircraft is 3.51×10^{-3} square miles and 4.01×10^{-3} square miles for twin-engine aircraft.

For westbound traffic, the reactor enclosures and the administration building provide shadowing to the diesel generator enclosures. The reactor enclosures, located north of the diesel generator enclosures, reduce the effective target area for westbound aircraft to 84.5 feet plus one-half the aircraft's wingspan. The warehouse and shop area, located to the east of the diesel generator enclosures, effectively eliminates the skid area. The resultant critical target areas for single-engine aircraft is 1.36×10^{-3} square miles, and 1.49×10^{-3} square miles for twin-engine aircraft.

For eastbound traffic, shadowing to the diesel generator enclosures is provided by the reactor enclosures and the radwaste enclosures. The resultant critical target areas are 1.68×10^{-3} square miles for single-engine aircraft and 1.86×10^{-3} square miles for twin-engine aircraft.

For southbound traffic, the critical target area for diesel generators is zero due to the shadowing afforded by the reactor enclosure.

For rotary-wing aircraft, the critical target area of the diesel generator enclosure is 6.55×10^{-4} square miles.

LGS Plant Site Heliport

The probability of a crashing helicopter impacting the diesel generator building is determined based on the known flight path of the helicopter traffic using the heliport, the closest distance this flight path comes to the diesel generator building, and the deviation from the flight path.

The possible deviation from the bounding traffic pattern for landing or takeoff path is represented by an exponential density function of the form:

$$f(x;\lambda) = \lambda e^{-\lambda x} \qquad \text{(EQ. 3.5-4)}$$

where:

$f(x;\lambda)dx$ = the probability that the deviation is in the range $(x, x+dx)$;

x = the perpendicular distance (deviation) from the flight path;

λ = the exponential distribution parameter.

The exponential distribution parameter, λ, is the inverse of the expected value of x. The expected value of x, x has been chosen as the mean distance which the helicopter can travel from an altitude (h). Therefore, x is given by:

$$x = h \tan(\phi)$$

The angle ϕ is conservatively assumed to be the helicopter glide angle due to loss of a rotor. The expected altitude (h) of the helicopter on an approach or departure from the helipad is given by:

$$h = a \tan(\theta)$$

The angle θ, the approach angle, is conservatively assumed to be 15°, and (a) is the distance from the helipad.

Using the above information, λ is given by:

$$\lambda = \frac{1}{a \tan(\phi) \tan(\theta)}$$

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

$$P(>d) = 0.5\, e^{-\lambda|d|}$$

The probability of impact on the diesel generator enclosure is greatest when the helicopter is on the approach path 675 feet from the helipad and 1,100 feet laterally from the diesel generator enclosure. The average conditional probability of impacting the diesel generator enclosure from a crash for all three types of helicopters given in Table 3.5-1 is 7.66×10^{-4}.

Federal Airways

The reactor enclosure, control structure, and diesel generator enclosures are considered in the calculation of critical target areas. The spray pond pump structure represents only about 5% of the total target areas of safety-related structures, and its contribution to the overall probability is considered negligible.

The spray pond is not susceptible to aircraft damage because it is constructed completely in excavation, as discussed in Section 2.5. Aircraft damage to the spray pond spray network is not considered because the design provides for separation and redundancy so that the loss of a network would not affect the systems safety function.

Because the majority of airway traffic is northbound, the analysis considers the south faces of the diesel generator enclosures and the reactor enclosures as the target area. The calculated critical target area is 3.33×10^{-2} square miles.

3.5.1.6.3 Crash Probabilities

Based on the given data and assumptions regarding the number of aircraft movements (N_{ij}), aircraft crash probability (C_{ij}), and critical target areas for the plant (A_{ij}), the probability of an aircraft striking to safety-related structures has been calculated.

For operations out of the Pottstown Municipal Airport, the probability of an aircraft striking the diesel generator enclosures is given by:

$$P = 1.2 \times 10^{-8} \bullet [1.73N_e + 1.40N_w + 3.66N_n] \times 10^{-3}\ yr^{-1} \qquad \text{(EQ. 3.5-5)}$$

where:

N_e = number of eastbound movements over the site

N_w = number of westbound movements over the site

N_n = number of northbound movements over the site

and the target areas have been weighted to reflect the proportion of single and twin-engine traffic.

For

$N_e = N_n = 4000; N_w = 0,$

this yields a crash probability of 2.59×10^{-7} per year.

Similarly, for operations out of the Pottstown-Limerick Airport, the probability is given by:

$$P = 1.5 \times 10^{-7} \cdot [1.73N_e + 1.40N_w + 3.66N_n + 0.655N_r] \times 10^{-3}\ yr^{-1} \quad \text{(EQ. 3.5-6)}$$

where:

N_r = number of rotary-wing operations over the site from the Pottstown-Limerick Airport.

For

$N_e = 1500, N_w = 13500, N_n = 0; N_r = 0,$

this yields a crash probability of 3.2×10^{-6} per year.

For operations out of the LGS plant site heliport, the probability of a helicopter striking the diesel generator enclosure is the product of the number of takeoffs and landings per year, the probability of the helicopter crashing during takeoff or landing, and the probability that the crashing helicopter will impact the diesel generator building:

$$P = 312 \cdot 4.88 \times 10^{-7} \cdot 7.66 \times 10^{-4}$$

$$= 1.17 \times 10^{-7} \text{ per year}$$

The probability of an aircraft in one of the airways near the site striking the diesel generator enclosures, or the reactor enclosures is given by:

$$P = [(8395 + 20240)(590) + (46355)(22) + (4745)(7.1)] \times 10^{-13} \cdot 0.0333$$

$$= 5.98 \times 10^{-8}\ yr^{-1} \quad \text{(EQ. 3.5-7)}$$

3.5.1.6.4 Probability of Aircraft Strike Resulting in Unacceptable Consequences

Given the aircraft impact resistant design of all safety-related structures except the diesel generator enclosures, an aircraft strike could jeopardize safe shutdown capability or have potential offsite dose consequences only under the following scenarios:

(1) The impacting aircraft is larger in size than the design basis aircraft described in Section 3.5.1.6.1, and the impact causes structural failure and/or damage to systems required to achieve or maintain safe shutdown, resulting in the possibility of offsite consequences in excess of 10CFR50.67 guidelines.

(2) The impacting aircraft strikes a diesel generator enclosure, damaging more than 2 diesel generators, thereby leaving that unit dependent on offsite power for achieving and maintaining shutdown. During the period required to return at least 2 diesel generators to service, a complete loss of offsite power occurs, which leaves the affected unit dependent on its Class 1E battery system to achieve and/or maintain safe shutdown. If offsite power is not restored before the batteries are exhausted, inadequate core cooling and offsite radiological consequences in excess of 10CFR50.67 guidelines could result.

The probabilities of scenarios (1) and (2) occurring are calculated as follows:

Scenario 1: Impact By Aircraft Larger Than Design Basis Aircraft

This scenario is credible only for aircraft in Federal airways. It is conservatively assumed for this analysis that the impact of any aircraft using the Federal airways causes damage such that offsite radiological consequences in excess of 10CFR50.67 occur. The probability of such an impact causing unacceptable radiological consequences is therefore equal to the crash probability of an aircraft in one of the airways near the site, shown in Section 3.5.1.6.3 to be 5.98×10^{-8} per year.

Scenario 2: Diesel Generator Damage Caused By Impact Of Design Basis or Smaller Aircraft

The probability that aircraft operating out of the Pottstown- Limerick Airport, the Pottstown Municipal Airport, and the LGS plant site heliport will impact the diesel generator enclosures is shown in Section 3.5.1.6.3 to be 3.2×10^{-6} per year, 2.59×10^{-7} per year, and 1.17×10^{-7} per year respectively, or a total probability,

$$P_{ac} = 3.58 \times 10^{-6} \text{ per year.}$$

Given that each unit's diesel generator enclosure is 108 ft wide, and that each of the 4 diesels per unit is located in a separate compartment inside that unit's diesel generator enclosure, it is highly unlikely that the impact of a typical twin-engine plane with a wingspan of 50 ft could disable more than 2 diesel generators.

In addition, only 2 of a unit's 4 diesel generators are required to achieve and maintain safe shutdown, assuming no additional failures. Despite the above, it is conservatively assumed for this analysis that any aircraft impacting a diesel generator enclosure disables all 4 diesel generators therein, leaving that unit dependent on offsite power for achieving and maintaining safe shutdown. It is further conservatively assumed that immediate action would be taken to repair the diesel generators, and that at least 2 of the 4 would be restored to service 1 year following the aircraft impact.

Using the methodology of appendix III of Reference 3.5-10, the probability, P_{LOOP}, that a LOOP occurs (during the 1 year in which the affected unit would have no onsite ac backup power) is conservatively determined to be 0.2 per year, thus leaving that unit dependent on its Class 1E dc battery system to achieve or maintain shutdown.

It has been shown (Reference 3.5-11) that under loss of all ac power (station blackout) conditions, in a BWR with electrical and mechanical system designs similar to LGS, the batteries could power all systems required to achieve or maintain safe shutdown for as long as 7 hours before exhaustion. For this analysis, it is conservatively assumed that battery exhaustion occurs 4 hours after the LOOP occurs, after which inadequate core cooling and offsite radiological consequences in excess of 10CFR50.67 could occur.

Again using the methodology of appendix III of Reference 3.5-10, the probability, P_4, that at least one offsite source of power is not returned to service within 4 hours following the LOOP is conservatively estimated to be 0.05 per LOOP event.

Based on the preceding information, the probability per year of an aircraft crash that disables one unit's diesel generators, leading to offsite radiological consequences in excess of 10CFR50.67 dose limits, can be calculated as follows:

$$P = P_{AC} \cdot P_{LOOP} \cdot P_4$$

$$= (3.58\text{x}10^{-6}\ \text{yr}^{-1})(0.2\ \text{yr}^{-1})(0.05) \qquad \text{(EQ. 3.5-8)}$$

$$= 3.58\text{x}10^{-8}\ \text{per year.}$$

Conclusion

The total probability per year of an aircraft impact resulting in offsite radiological consequences in excess of 10CFR50.67 dose limits is the sum of the probabilities associated with Scenarios 1 and 2, or $9.56\text{x}10^{-8}$ per year. Realistically, this probability would be lower.

It is concluded that the acceptance criteria of SRP section 3.5.1.6 are met.

3.5.2 SYSTEMS TO BE PROTECTED

Safety-related systems and structures are reviewed for missile protection and are listed in Table 3.2-1. Structures and barriers designed to provide for protection from external missiles are discussed in Section 3.5.1 and listed in Table 3.5-7, and their characteristics are listed in Table 3.5-8.

As discussed in Section 3.5.1, the only postulated internally generated missiles are those arising from failure of the HPCI and RCIC pump turbines. In this case, the pump-room walls are designed as barriers to isolate such missiles from other safety-related components.

The locations of safety-related equipment and the arrangement of the various compartments are shown in drawings M-110, M-111, M-112, M-113, M-114, M-115, M-116, M-117, M-118, M-119, M-120, M-121,M-122, M-123, M-124, M-125, M-126, M-127, M-128, M-129, M-130, M-131, M-132, M-133, M-134, M-135, M-136, M-137, M-138, M-140, M-141, M-142, M-143, M-144, M-145, M-146, M-388, M-389, and M-390.

3.5.3 BARRIER DESIGN PROCEDURES

Safety-related components necessary for safe shutdown and housed in the Category I structures are adequately protected against tornado missiles. The characteristics of structural wall barriers for Category I structures are shown in Table 3.5-8. Figure 3.5-2 shows typical reinforcement provided in the southernmost reactor enclosure concrete wall barrier.

The structures and barriers are designed to resist missile hazards in accordance with the procedures detailed in Reference 3.5-6.

The procedures include:

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

b. Estimation of the barrier thickness required to prevent perforation

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

Tornado missiles, as described in Table 3.5-4, were considered as deformable upon impact, and the structures were considered as rigid targets. Eighteen inch thick concrete, with a compressive strength of 4000 psi, was used in concrete walls and roofs subject to tornado missile impact. This satisfies the penetration formulae for elements subject to missiles as specified in Reference 3.5-6.

The ESW and RHRSW piping located within the yard area are installed underground with adequate cover for missile protection. A detailed assessment of soil cover for Category I yard piping shows a minimum depth of 4 feet, with most soil coverings exceeding 6 feet. The 4 foot depth was found to be adequate for tornado missiles (Table 3.5-4) in accordance with the criteria set forth in Reference 3.5-3.

The physical routing of the piping and typical profiles showing soil cover and installation details are shown in Figure 2.5-37. The cementitious backfill for pipe bedding is shown in detail 3 of Figure 2.5-37. This backfill was used predominantly for Category I piping to provide additional missile protection. The properties of cementitious backfill are defined in Section 2.5.4.5.4. Figure 2.5-37 shows the 4 foot minimum depth dimension of soil cover for Category I yard piping.

Doors are designed for tornado missiles unless internal secondary barriers are provided to protect safety-related components. Louvers are not designed for tornado missiles; however, internal secondary barriers are provided to protect safety-related components which are located behind the louvers. Figure 3.5-3 shows a typical secondary barrier for the south wall of the diesel generator enclosure (concrete labyrinth behind the door) used for tornado missile protection.

3.5.4 REFERENCES

3.5-1 Letter, E.A. Hughes (GE) to R.C. DeYoung (NRC), "GE Recirculation Pump Potential Overspeed", (January 18, 1977).

3.5-2 NUREG/CR-2944, "Tornado Damage Risk Assessment", Reinhold & Ellingwood, Brookhaven National Lab, (September 1982).

3.5-3 "Depth Prediction for Earth-Penetrating Projectiles", Soil Mechanics and Foundation Division, ASCE, p. 6558, (May 1969).

3.5-4 "Probability of Missile Generation in General Electric Nuclear Turbines", GE Proprietary Report.

3.5-5 D.C. Gonyea, "An Analysis of the Energy of Hypothetical Wheel Missiles Escaping from Turbine Casings", GE Technical Information Series No. DF735L12, (February 1973).

3.5-6 "Design of Structures for Missile Impact", BC-TOP-9A, Revision 2, Bechtel Power Corporation, San Francisco, California, (September 1974).

3.5-7 Solomon, K.A., "Hazards Associated with Aircraft and Missiles", presented at American and Canadian Nuclear Society Meeting, Toronto, Canada, (June 1976).

3.5-8 Solomon, K.A., "Estimate of probability that an Aircraft will impact the PVNGS", NUS-1416, NUS Corp., (June 1975).

3.5-9 National Air Transportation Safety Board, "Annual Review of Aircraft Accident Data", (Published 1972 and annually thereafter).

3.5-10 ASH-1400, "Reactor Safety Study: An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants", NRC, (October 1975).

3.5-11 NUREG/CR-2182, Vol. I, "Station Blackout at Browns Ferry Unit One - Accident Sequence Analysis, Oak Ridge National Laboratory", (November 1981).

3.5-12 Barber, R.B., "Steel Road/Concrete Slab Impact Test (Experimental Simulation)", Bechtel Power Corporation, (October 1973).

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

3.5-14 "Safety Evaluation Report Relates to the Operation of Hope Creek Generating Station", NUREG-1048, Supplement 6, (July 1986).

3.5-15 “Missile Analysis Report, Limerick Units 1 and 2," Siemens Energy, Inc., CT-27544, March 4, 2016 (including, as Appendix A: "Missile Probability Analysis Methodology for Limerick Generating Station, Units 1 and 2, with Siemens Retrofit Turbines” Siemens Power Corporation Engineering Report – ER-9605, Revision 2, June 18, 1997 - SIEMENS PROPRIETARY)

3.5-16 “Impact of Increasing Test Interval for Turbine Overspeed Protection System on Turbine Missile Probability,” MPR Associates Inc., MPR-2892 Revision 0, April 4, 2006

Table 3.5-1

AIRCRAFT IMPACT DESIGN PARAMETERS

AIRCRAFT DESCRIPTION	GROSS WEIGHT (lb)	APPROACH VELOCITY (knots)	APPROACH VELOCITY (ft/sec.)	KINETIC ENERGY (K-ft)
Single reciprocating engine	<5,000	110	186	2,686
Cessna (twin recip engine)	7,250	111	188	3,979
Cessna jet (twin gas turbine)	10,500	115	194	6,136
Learjet[(1)] (twin gas turbine)	13,500	139	235	11,577
Navajo Chieftrain	7,000	100	169	3,104
Beech King Air-200	12,500	103	174	5,877
Cheyenne Turbo-Prop	11,200	115	194	6,545
Bell Long Ranger (helicopter)	4,150	50	85	466
Aerospatiale Twin Star (helicopter)	5,291	50	85	594
Bell 222 Model A (helicopter)	7,850	70 (takeoff)	118	1,697

(1) Largest aircraft capable of operating out of nearby airports; none are presently known to exist.

Table 3.5-2

CHARACTERISTICS OF LEARJET

Aircraft Components[1]	Weight (lbs)	Mass (Kip-sec^2/ft)	K.E. (Kip-ft)	Resultant Load on Walls[2] Momentum (Kip-sec)	Assumed Impact Duration (sec)	Peak Force (Kips)
Fuselage	7,178	0.2229	6,155	52.3	0.10	846
Tip Tanks	1,358	0.0422	1,165	9.9	0.05	396
Wing	3,503	0.1088	3,005	25.6	0.10	512
Engines	1,158	0.0360	995	8.5	0.05	340
Empennage	305	0.0095	260	2.2	0.05	88
Total	13,500	0.4194	11,580	98.5	-	(2)

(1) All components are assumed traveling at 235 fps (139 knots) at impact.
(2) Resultant load curves on structure due to the different components and total are shown in Figure 3.5-1.

Table 3.5-3

ACCUMULATORS LOCATED IN SAFETY-RELATED AREAS

Description	Maximum Pressure (psig)	Maximum Temperature (°F)
MSRV Accumulators	125	150
MSIV Accumulators	125	150
Diesel Air Start Reservoirs	250	120
Instrument Gas Receivers	125	135

Table 3.5-4

TORNADO-GENERATED MISSILE PARAMETERS

MISSILE	WEIGHT (lb)	IMPACT AREA (ft^2)	HORIZONTAL[4] VELOCITY (mph)/(ft/sec)	KINETIC ENERGY (ft-lb)
1. Wood plank (4"x12"x12')	200	0.333	300/440	$6.01x10^5$
2. Steel pipe (3" dia x 10', schedule 40)[3]	78	0.067	144/211	$5.39x10^4$
3. Automobile[2][3]	4000	20	72/106	$6.98x10^5$
4. Steel rod (1" dia x 3')	8	0.007	216/317	$1.25x10^4$
5. Utility pole (13½" dia x 35', not more than 30' above all grade elevations within ½ mile of the plant)	1490	1.266	144/211	$1.03x10^6$
6. Steel pipe (6" dia x 15', schedule 40)[1]	285	0.239	144/211	$1.97x10^5$
7. Steel pipe (12" dia x 15', schedule 40)[1]	743	0.886	144/211	$5.14x10^5$

(1) The design basis for LGS included only missiles 1, 2, 3, 4 and 5. All safety-related structures and openings in structures have been assessed for the effects of missiles 6 and 7.

(2) LGS was originally designed for a postulated automobile missile not more than 25 ft above grade for all safety-related structures. All safety-related structures have been reassessed for the effect of the automobile at elevations up to 30 ft above all grade levels within ½ mile of the plant.

(3) LGS was originally designed for postulated missile velocities equal to 100 mph for the 3 in diameter steel pipe and 50 mph for the automobile. All safety-related structures have been reassessed for the revised velocities shown on the table.

(4) These missiles are considered to be capable of striking in all directions with vertical velocities equal to 80% of the horizontal velocities.

Table 3.5-5

RAILROAD ACCIDENT GENERATED MISSILE PARAMETERS

MISSILE	WEIGHT (lb)	INITIAL VELOCITY (mph)
Part of tank car tank (112" dia x 50')[(1)]	62,000	200 (during ignition, traveling end on)
Steel pipe (3" dia x 10' schedule 40)	78	200 (tumbling)
Rail (3' long, 130 lb/yd)	130	200 (tumbling)
Type E coupler (7 in^2 shank)	200	200 (tumbling)
Unfused bomb	750	100

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(1) These have been observed being propelled by burning gases at a very low trajectory not exceeding 30-35 feet above the ground. The elevation difference (77 feet) between the railroad and safety-related structures prevents impingement of this missile, and therefore it was not considered in the design.

Table 3.5-6

POTENTIAL HVAC FAN MISSILES

Fan No.	Description	Area	Elevation	Notes
OAV-163, OBV-109	SGTS exhaust fans	8	350	(2)
OAV-127, OBV-127	Control room emergency fresh air supply fans	8	304	(1)
OOV-126	Toilet room exhaust air fan	8	332	(2)
1AV-206, 1BV-206, 2AV-206, 2BV-206	Reactor enclosure equipment compartment air exhaust fans	15,16, 17,18	313	(2)
OAV-132, OBV-132	Drywell purge exhaust fans	8	350	(2)
OAV-131, OBV-131	SGTS room exhaust fans	8	350	(2)
1AV-213, 1BV-213, 2AV-213, 2BV-213	Reactor enclosure air recirculation fans	15,18	313	(2)
OAV-120, OBV-120	Auxiliary equipment room air return/exhaust fans	8	304	(2)
OAV-121, OBV-121	Control room ac return/exhaust fans	8	304	(2)
1AV-512 thru 1HV-512 and 2AV-512 thru 2HV-512	Diesel generator enclosure ventilation air exhaust fans	DG Bldg		(3)

Table 3.5-6 (Cont'd)

Fan No.	Description	Area	Elevation	Notes
1AV-201 thru 1CV-201 and 2AV-201 thru 2CV-201	Refueling floor air supply fans	15,18	313	(2)
1AV-202 thru 1CV-202 and 2AV-202 thru 2CV-202	Reactor enclosure air supply fans	16,17	313	(2)
1AV-204 thru 1CV-204 and 2AV-204 thru 2CV-204	Refueling floor air exhaust fans	16,17	313	(2)
1AV-205 thru 1CV-205 and 2AV-205 thru 2CV-205	Reactor enclosure air exhaust fans	16,17	331	(2)
OAV-124, OBV-124	Battery room air exhaust fans	8	304	(2)
OAV-114, OBV-114	Auxiliary equipment room supply ac units	8	304	(2)
OAV-116, OBV-116	Control room supply ac units	8	304	(2)
OAV-118, OBV-118	Emergency switchgear and battery room ac units	8	217	(2)

Table 3.5-6 (Cont'd)

Fan No.	Description	Area	Elevation	Notes
1AV-208, 1BV-208, 2AV-208, 2BV-208	RCIC pump-room unit coolers	15,18	177	(3)
1AV-209, 1BV-209, 2AV-209, 2BV-209	HPCI pump-room unit coolers	15,18	177	(3)
1AV-210 thru 1HV-210 and 2AV-210 thru 2HV-210	RHR pump-room unit coolers	15,16, 17,18	177	(3)
1AV-211 thru 1HV-211 and 2AV-211 thru 2HV-211	CS pump-room unit coolers	11,12, 13,14	177	(3)
1AV-212 thru 1HV-212 and 2AV-212 thru 2HV-212	Drywell area unit coolers	Drywell	217	(4)
OAV-140, OBV-140	SGTS room unit coolers	8	332	(3)
OAV-141, OBV-141	SGTS room access area unit coolers	8	332	(3)
OAV-543 Thru ODV-543	Spray pond pumphouse air supply units	Spray pond		(2)
00V900	CRD Decontamination Room Unit Cooler	15	253	(3)

Table 3.5-6 (Cont'd)

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(1) These potential missiles are installed in equipment that is only used during accident situations. They are not considered credible potential missiles because internal missile generation is not postulated to occur concurrently with other accidents.

(2) These potential missiles are either remote from or separated by adequate barriers from all essential systems. Therefore, essential systems are protected from these potential missiles.

(3) The potential missiles cannot impact on more than a single component. Therefore, redundant equipment will be available to effect safe shutdown.

(4) The casing thickness is sufficient to prevent penetration by a loose blade. (During the September 24, 1982 Palo Verde event where the description indicated that a portion of a fan blade cut the liner plate, investigation revealed that the portion of blade escaped through a neoprene-coated fiberglass flexible connection between the fan discharge connection and the downstream duct. Pieces of blade struck the fan casing and the downstream duct. The duct was dented but no penetration of the fan casing or duct by the blade occurred. In the case of the LGS drywell unit coolers, the fan discharge connections are connected to a Class I Seismic Y duct fitting directly. Downstream of the fitting, stainless steel bellows with sliding metal flow liners are utilized rather than flexible fabric connections, thus precluding blade penetration.)

Table 3.5-7

MISSILE BARRIERS AND PROTECTED COMPONENTS FOR EXTERNALLY GENERATED MISSILES

PROTECTED COMPONENTS	MISSILE BARRIER
Reactor coolant and other safety-related equipment inside the containment	Primary containment structure, reactor enclosure walls, refueling cavity walls, internal structures and beams
Control room, cable spreading room, switchgear, batteries, SGTS	Control structure
ECCS and other safety-related equipment outside the primary containment	Reactor enclosure and internal walls
Standby diesel generators	Diesel generator enclosure[1]
Diesel fuel oil tank	Earth; the tank is buried underground
Spent fuel pool	Reactor enclosure and fuel pool walls
ESW pumps and RHRSW pumps	Spray pond pump structure
ESW and RHRSW yard piping	Earth; piping is buried

(1) The diesel generator enclosure is not designed for design aircraft accident.

Table 3.5-8

CHARACTERISTICS OF EXTERNALLY GENERATED MISSILE BARRIERS

BARRIER	MINIMUM THICKNESS (in)	MINIMUM STRENGTH (psi)	CURING TIME (days)
Primary containment	74	4000	28
Reactor enclosure			
Walls	24	4000	28
Roofs	18	4000	28
Refueling cavity walls	48	4000	28
Spent fuel pool walls	54	4000	28
Control structure			
Walls	24	4000	28
Roof	18	4000	28
Diesel generator enclosure			
Walls	24	4000	28
Roof	24	4000	28
Spray pond pump structure			
Walls	24	4000	28
Roof	24	4000	28

(1) On which strength is based

Table 3.5-9

TURBINE SYSTEM RELIABILITY CRITERIA

CRITERION	PROBABILITY, YR^{-1} UNFAVORABLY ORIENTED TURBINE	REQUIRED ACTION
(A)	$P_1 < 10^{-5}$	This is the general, minimum reliability requirement for loading the turbine and bringing the system on-line.
(B)	$10^{-5} < P_1 < 10^{-4}$	If this condition is reached during operation, the turbine may be kept in service until the next scheduled outage, at which time the licensee is to take action to reduce P_1 to meet the appropriate A criterion (above) before returning the turbine to service.
(C)	$10^{-4} < P_1 < 10^{-3}$	If this condition is reached during operation, the turbine is to be isolated from the steam supply within 60 days, at which time the licensee is to take action to reduce P_1 to meet the appropriate (A) criterion (above) before returning the turbine to service.
(D)	$10^{-3} < P_1$	If this condition is reached at any time during operation, the turbine is to be isolated from the steam supply within 6 days, at which time the licensee is to take action to reduce P_1 to meet the appropriate (A) criterion (above) before returning the turbine to service.

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

This section describes the treatment of postulated ruptures in high energy and moderate energy piping located both inside and outside of the primary containment. The methods used to determine pipe rupture locations and to analyze the results of the ruptures, including jet thrust forces, jet impingement forces, piping dynamic responses, and compartment pressure-temperature transients, are described. Description is also provided for the design measures that have been implemented to ensure that the effects of any postulated rupture do not result in the loss of a required function that is necessary to mitigate the consequences of that pipe rupture.

The definitions of certain terms used within this section are provided in Section 3.6.3. The use of the term "low reactor level" in this section is generic rather than referring to a specific isolation signal set point.

NOTE: The computer program "COPDA", as described in Bechtel Topical Report BN-TOP-4 (Reference 3.6-1), was originally used to determine compartment pressure and temperature resulting from line breaks in Unit 1. The computer program "FLUD" (Reference 3.6.3), which replaced COPDA, was typically used for the same analysis in Unit 2. The mainframe program "FLUD" was later replaced in 1985 with a personal computer version "PCFLUD" (Reference 3.6-12). The "PCFLUD" program had additional features added in 1993 (hot pipes, multiple HVAC fans, and "CONCOIL" room air cooler routine) and was renamed "CONCOIL-FLUD" or "CFLUD" (Reference 3.6-13). The supporting calculations for UFSAR Table 3.6-7 must be consulted for information regarding which specific program was used.

3.6.1 POSTULATED PIPING FAILURES IN FLUID SYSTEMS

The failure of piping containing high energy or moderate energy has the potential of causing damage to surrounding structures, systems, and components. Depending on the fluid system involved and the rupture location, postulated piping failures can result in one or more of the effects described below. Essential systems and components are protected from these effects unless it can be demonstrated that their function is not impaired.

Pipe Whip

Pipe whip is the unrestrained movement of a pipe due to the reaction force imposed on the pipe by fluid discharging from a rupture. Protection against pipe whip can be provided by intervening structural members between high energy piping and the essential systems and components, by providing pipe whip restraints on the high energy piping, or by locating essential systems and components sufficiently distant from high energy piping. Examples of typical pipe whip restraints and bumpers are shown in Figure 3.6-1.

Jet Impingement

The blowdown of fluid from a rupture of a high energy pipe can exert forces on nearby equipment that could be high enough to cause damage to the equipment. Protection against jet impingement can be provided by installing jet impingement barriers to deflect the blowdown jet, or by locating essential systems and components a sufficient distance from high energy piping.

Environmental Effects

Pipe ruptures in high energy and moderate energy lines release fluid that can increase temperature, pressure, and humidity in the vicinity of the pipe rupture and also in remote areas that communicate with the local atmosphere. Essential systems and components may be exposed to abnormal conditions that have the potential of degrading the capability of the equipment to perform its function.

Piping systems whose rupture might generate hazardous environmental conditions are generally located in compartments that are capable of being isolated from other compartments containing essential systems and components. Isolation of compartments that enclose high energy lines is provided by maintaining normally closed access ways, automatically isolating ventilation duct-work, and sealing penetrations through walls and slabs. Compartments are designed to withstand the maximum internal pressurization developed as a result of a pipe rupture. Essential systems and components are either located in areas not affected by pipe ruptures or are qualified for operation under the maximum environmental conditions that they may be subjected to as a result of pipe ruptures.

Figures 3.6-43 through 3.6-45 are provided to show additional details of a typical pipe break analysis performed in accordance with SRP 3.6.1 and BTP ASB 3-1. The pipe break chosen is a break of the HPCI steam supply line in the HPCI compartment.

Water Spray

Water released from pipe ruptures is a hazard to certain equipment, particularly electrical components. In most cases, spatial separation is adequate to prevent water spray from reaching essential systems and components. Essential systems and components that can potentially be subjected to water spray are either designed to operate when wetted or will be provided with barriers to deflect the spray.

Flooding

Significant ruptures of fluid system piping may result in flooding in the vicinity of the rupture and in compartments through which the released fluid drains. The flooding rate and the total fluid volume released are computed based on the rupture configuration, the service of the fluid system, and the time required to isolate the rupture. The plant's drainage system handles minor releases of fluid with no adverse effects on essential systems and components.

Compartments that contain fluid systems with the potential for major releases of fluid are designed to contain the flooding and prevent significant leakage to adjoining compartments containing essential systems and components. Essential systems and components are either located in areas not subject to significant flooding or are designed to operate in a flooded environment.

3.6.1.1 Design Bases

Pipe breaks are postulated to occur in high energy fluid system piping in accordance with the criteria stated in Section 3.6.2.1.1. Pipe cracks are postulated to occur in moderate energy fluid system piping in accordance with the criteria stated in Section 3.6.2.1.2. The effects of these piping ruptures require special consideration to ensure the following:

a. The ability to shut the reactor down safely is maintained.

b. Primary containment integrity is maintained.

c. Resultant offsite doses are below the values of 10CFR50.67.

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

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

b. Offsite power is assumed to be unavailable if a trip of the turbine-generator or the RPS is a direct consequence of the postulated piping rupture.

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

d. Where the postulated piping rupture is assumed to occur in one of the two redundant loops of either the RHR system or the RHRSW system, single active failures of components in the other loop of that system need not be assumed. These two systems are dual-purpose moderate energy systems powered from both onsite and offsite sources and are designed, constructed, and inspected to standards appropriate for nuclear safety systems.

e. All available systems, including those actuated by operator actions, may be employed to mitigate the consequences of the postulated piping rupture. In judging the availability of systems, the postulated piping rupture and its direct consequences, and the assumed single active component failure and its direct consequences are considered. The feasibility of carrying out operator actions is judged on the availability of ample time and adequate access to equipment for the required actions.

f. An unrestrained whipping pipe is considered capable of causing breaks in impacted piping of smaller nominal pipe size and developing through-wall leakage cracks in impacted piping of equal or larger nominal pipe size with thinner wall thickness, except where experimental or analytical data demonstrates the capability to withstand the impact without failure. If such experimental or analytical data are used, additional documentation will be submitted to the NRC for review.

g. The effect of the jets resulting from each postulated pipe break in high energy fluid systems is reviewed and is evaluated only when such jets may have an effect on specific safety-related components that are required to effect a safe shutdown, maintain containment integrity, provide post-LOCA monitoring, or limit radioactive releases to site-allowable limits, as a result of a specific postulated pipe break event.

The effects of jet impingement on the piping required to effect a safe shutdown, maintain containment integrity, provide post-LOCA monitoring, or limit radioactive releases to site-allowable limits, regardless of size or wall thickness, are also reviewed. Bounding cases are evaluated to ensure that acceptable stress levels in such piping are not exceeded.

3.6.1.2 Description

A listing of high energy fluid system piping is provided in Table 3.6-1. All other piping in the plant that is pressurized above atmospheric pressure is considered to be moderate energy piping. The routing of piping within the reactor enclosure and the primary containment is shown in Figures 1.2-40 through 1.2-72.

For each pipe rupture location determined in accordance with the criteria of Section 3.6.2.1, an analysis is performed using the assumptions of Section 3.6.1.1 to verify that the consequences of the pipe rupture are acceptable. These analyses are summarized below for high energy and moderate energy fluid systems.

3.6.1.2.1 High Energy Fluid Systems

3.6.1.2.1.1 Reactor Recirculation System

The two reactor recirculation loops are located entirely within the primary containment and are arranged on opposite sides of the reactor pedestal and reactor shield wall. Pipe whip restraints anchored in the reactor pedestal and reactor shield wall are provided for the recirculation loops and are arranged as shown in Figure 3.6-2. This system of restraints prevents unrestrained pipe whip resulting from a postulated rupture at any of the identified break locations. The restraints consist of two basic components: the frame attached to a support member and straps attached to the frame (two straps per frame). Either carbon steel cables or stainless steel bars are used as straps. The restraints on the 28-inch recirculation loop piping utilize stainless steel bars. The restraints on the 22-inch discharge header and the 12-inch risers utilize carbon steel cables. A schematic detail of a restraint is shown in Figure 3.6-3.

A review of the potential consequences of jet impingement resulting from recirculation loop ruptures has shown that breaks at the lower end of two of the 12-inch risers in each recirculation loop could result in impingement on the CRD withdraw piping. An evaluation of the consequences of damage to CRD withdraw piping resulting from jet impingement has shown that such damage would not prevent the associated control rods from being inserted into the reactor as necessary to affect a reactor shutdown. Electrical cabling associated with essential systems and components is routed to avoid jet impingement from postulated ruptures of recirculation loop piping.

Pipe Break Locations

The postulated pipe break locations for the recirculation loop piping are shown in Figure 3.6-4. The calculated stress levels and usage factors, and the postulated break types, are listed in Table 3.6-2. Blowdown thrust time histories for each break location are provided in Table 3.6-3. The recirculation piping design basis has been evaluated for the effects of power rerate and demonstrated to be adequate for the increases in pressure, temperature and flow due to power rerate. Details of the evaluation performed are documented in Reference 3.6-11.

Compartment Pressure-Temperature Transient

The pressure-temperature transient in the primary containment resulting from a complete circumferential rupture of one recirculation loop is discussed in Section 6.2.1.

Verification of Reactor Shutdown Capability

The sequence of events that would occur automatically to shut the reactor down and cool the core in the event of a recirculation loop rupture is discussed in Section 6.3.3. A combination of pipe whip restraints, jet impingement barriers, and separation by distance is used to ensure the availability of sufficient equipment to accomplish these functions.

3.6.1.2.1.2 Main Steam System

The four 26-inch main steam lines are routed as shown in Figure 5.1-7 for the portion inside the primary containment, and in Figure 3.6-5 for the portion outside the primary containment. The A and B steam lines are connected to the east side of the reactor vessel and the C and D steam lines are connected to the west side of the vessel. All four steam lines penetrate the north side of the primary containment. The portion of the reactor enclosure through which the main steam lines are routed (between the primary containment and the turbine enclosure) is referred to as the main steam tunnel and is separated from other areas of the reactor enclosure by concrete walls and slabs. Only piping, valves, and associated instrumentation are located in the main steam tunnel. Figure 3.6-6 shows an elevation view of the main steam tunnel.

The following features are incorporated into the design of the main steam line and nearby structures to mitigate the consequences of a main steam line break or to minimize the probability of its occurrence:

a. A venturi-type flow restrictor is located in each main steam line inside the primary containment. The flow restrictor reduces the rate of loss of reactor coolant from a main steam line break for break locations downstream of the restrictor. (The flow restrictors are described in Section 5.4.)

b. Each main steam line is provided with either three or four MSRVs that reduce the probability of breaks by protecting the steam line against overpressurization. (The MSRVs are described in Section 5.2.)

c. Each main steam line is provided with two fast-acting fail-safe MSIVs, one upstream and one downstream of the primary containment penetration. These valves close automatically upon receipt of signals indicating high steam flow or high temperature in the vicinity of the piping outside the primary containment (as well as upon receipt of other initiating signals), in order to terminate blowdown through breaks outside the primary containment. (The MSIVs are described in Section 5.4.)

d. Moment-limiting pipe whip restraints are located upstream of the inboard MSIVs and downstream of the outboard MSIVs in order to ensure the operability of these valves in the event of a main steam line break in the general vicinity of the valves.

The main steam lines are provided with pipe whip restraints inside the primary containment and in the main steam tunnel. Typical restraints inside the primary containment are shown in Figure

3.6-1. Figure 3.6-6 shows the locations of the restraints in the main steam tunnel. As shown in Figure 3.6-7, the first two restraints downstream of the outboard MSIVs span between the east and west walls of the tunnel and restrain all four steam lines. Corbels extending out from the north wall of the main steam tunnel limit the possible upward movement of the upper elbow of each steam line in the tunnel. Additionally, the vertical portion of the steam line run in the tunnel is restrained against the north wall of the tunnel at two separate locations.

After entering the Unit 1 turbine enclosure from the main steam tunnel, the main steam lines are routed along the west side of the control structure before turning westward to run the length of the turbine enclosure. This arrangement is shown in Figure 3.6-5. In order to prevent a main steam line break in this portion of piping from causing pipe whip impact on the control structure wall, bumpers are provided between the control structure and the elbows of the two steam lines closest to the control structure (26" EBB-103 and 26" EBB-104). The Unit 2 arrangement is similar, but opposite hand.

In reviewing the potential consequences of jet impingement resulting from main steam line breaks, it was determined that breaks at the elbow at the lower end of the near-vertical piping run inside the primary containment could result in impingement on the CRD withdraw piping. An evaluation of the consequences of damage to CRD withdraw piping resulting from jet impingement has shown that such damage would not prevent the associated control rods from being inserted into the reactor core as necessary to affect a reactor shutdown. Electrical cabling associated with essential systems and components is routed to avoid jet impingement from postulated breaks of main steam piping.

Pipe Break Locations

The postulated pipe break locations for the main steam piping, and also the pipe whip restraint locations, are shown in Figures 3.6-8 and 3.6-9. The calculated stress levels and usage factors, and the postulated break types, for the main steam piping inside the primary containment are listed in Table 3.6-4. Breaks for the main steam piping outside containment are postulated to occur at each location of potential high stress, such as pipe fittings, valves, and welded attachments. The main steam piping basis has been evaluated for the effects of power rerate and demonstrated to be adequate for the increases in pressure, temperature and flow due to power rerate. Details of the evaluation performed are documented in Reference 3.6-11.

Compartment Pressure-Temperature Transients

The pressure-temperature transient in the primary containment resulting from a complete circumferential break of one main steam line is discussed in Section 6.2.1.

Protection against overpressurization of the main steam tunnel in the event of a main steam line break in the tunnel is provided by two sets of blowout panels. One set of blowout panels is located in the north wall of the tunnel and vents to the turbine enclosure. The second set of blowout panels is located in the vent stack that leads upward from the tunnel and discharges to the atmosphere above the top of the reactor enclosure. This vent stack can be seen in drawings M-123 and M-138.

A pressure-temperature transient analysis for the case of a main steam line break in the tunnel was performed using the analytical technique described in BN-TOP-4 (Reference 3.6-1) and the blowdown data provided in Table 3.6-6 and Figure 3.6-10. The flow model is shown in Figure 3.6-11 and the results of the analysis are listed in Table 3.6-7. The main steam tunnel is designed to withstand the maximum pressure developed, and the MSIVs are qualified to operate under environmental conditions more severe than those calculated to occur.

Protection against overpressurization of the Unit 1 turbine enclosure as a result of a main steam line break within its boundaries is provided by blowout panels along the north exterior wall of the turbine enclosure. As shown in drawings M-111 and 112, these blowout panels are located between column lines 12 and 18 and vent the main condenser area of the turbine enclosure directly to the atmosphere. The main steam lines are routed in areas that communicate directly with the main condenser area for the entire length of their run within the turbine enclosure. The Unit 2 arrangement is similar.

A pressure-temperature transient analysis for the case of a main steam line break in the turbine enclosure was performed using the analytical technique described in Reference 3.6-1 and with the blowdown data provided in Table 3.6-6. The flow model is shown in Figure 3.6-12 and the results of the analysis are listed in Table 3.6-7. The turbine enclosure is designed to withstand the maximum pressure developed.

The secondary containment structure is capable of withstanding the effects of a high energy pipe rupture occurring inside the secondary containment without loss of integrity. For all pipe breaks except those in the RWCU system, the time-dependent mass and energy release rates are calculated using the Moody model including friction effects. Blowdown for the main steam line breaks is determined in accordance with the assumptions in Reference 3.6-14. A double-ended guillotine break is assumed, and both forward and reverse flows are included for all cases except for HPCI and RCIC line breaks in the HPCI and RCIC compartments, respectively. The postulated pipe breaks in the HPCI and RCIC compartments are at the upstream side of a normally closed isolation valve, therefore only forward flow is considered.

Mass and energy release rates for pipe breaks in the RWCU system are calculated with RELAP4/MOD5 using the Henry-Fauske model option for subcooled conditions and the Moody critical flow model option for saturated conditions.

Verification of Reactor Shutdown Capability

Breakage of a main steam line inside the primary containment would result in an unisolable blowdown of the reactor vessel. The sequence of events that would occur automatically to shut the reactor down and cool the core is discussed in Section 6.3.3.

For a main steam line break outside the primary containment, the MSIVs will be closed automatically because of high steam flow, low reactor water level, or high temperature in the vicinity of the main steam lines. Closure of the MSIVs will cut off steam flow to the feedwater pump turbines, causing the pumps to coast down and stop. The reactor will be scrammed by low reactor water level or closure of the MSIVs. After isolation of the reactor vessel, the RCS pressure will increase until the setpoint of the MSRVs is reached. Steam will then be automatically discharged to the suppression pool to limit the pressure rise.

Low reactor water level will initiate operation of the HPCI and RCIC systems to maintain reactor water level. If both the HPCI and RCIC systems are unavailable, the ADS will be automatically initiated to depressurize the reactor vessel so that the LPCI and core spray systems can inject water into the vessel. These latter two systems are initiated automatically by low reactor water level, and will provide sufficient flow to restore reactor water level and cool the core.

After reactor water level has been restored, and the RCS pressure and temperature have decreased sufficiently, the shutdown cooling mode of the RHR system can be initiated to bring the reactor to cold shutdown.

A combination of pipe whip restraints, jet impingement barriers, and separation by distance or intervening structure is used to ensure the availability of essential systems and components in the event of a main steam line break in either the primary containment or the main steam tunnel. Essential systems and components located in these areas are qualified to operate under the environmental conditions resulting from a break. Since no essential systems and components are located in the turbine enclosure, no special provisions are necessary to provide protection for equipment in this area from the effects of pipe breaks. Blowout panels are provided to prevent overpressurization of the turbine enclosure, and bumpers are provided to prevent a whipping main steam line from impacting the control structure.

3.6.1.2.1.3 Feedwater System

The discharge lines from the three feedwater pumps are routed into a common mixing header in the turbine enclosure. From this header, two parallel 24-inch feedwater lines enter the main steam tunnel and then penetrate the primary containment. Inside the primary containment, the two lines diverge to form symmetrical headers on opposite sides of the reactor vessel. Each header splits into three 12-inch risers that attach to the reactor vessel nozzles. The routing of the feedwater lines in the main steam tunnel and primary containment is shown in drawings M-234, M-305 and Figure 3.6-6.

Each feedwater containment penetration is provided with three check valves as containment isolation valves, one inside the drywell and two in the main steam tunnel. In the event of a feedwater line break outside the primary containment, these check valves will, as described below, close to prevent backflow from the reactor vessel. Thus, flow from the break would be from the feedwater pump side only.

The operability of the first outboard containment isolation valve (i.e. closest to the drywell penetration) is ensured in the event of a feedwater line break by providing moment-limiting whip restraints designed for this purpose. The operability of the check valve inside containment and the second outboard check valve need not be protected from overstress due to pipe whip, since only two isolation valves are necessary to provide adequate redundancy to ensure the containment isolation function, and it is not possible for one pipe break to affect both the inboard and second outboard check valves. More detailed discussion of feedwater check valve operability is provided in Section 3.9.3.2b.2.

The feedwater lines are provided with pipe whip restraints inside the primary containment, in the main steam tunnel, and in the turbine enclosure immediately outside the main steam tunnel. Typical restraints inside the primary containment are shown in Figure 3.6-1. As shown in Figure 3.6-13, the restraint in the tunnel spans between the east and west walls of the tunnel and restrains both feedwater lines. Corbels extending out from the north wall of the main steam tunnel limit the possible upward movement of the upper elbows of the feedwater startup recirculation lines that connect to the feedwater lines in the main steam tunnel. Feedwater line restraints located in the turbine enclosure and which are attached to the control structure and reactor enclosure walls are shown in Figure 3.6-14. Bumpers on the control structure walls protect the control structure from a whipping feedwater line.

Electrical cabling associated with other essential systems and components is routed to avoid jet impingement from postulated breaks of feedwater piping.

Pipe Break Locations

The postulated pipe break locations for the feedwater piping, and also the pipe whip restraint locations, are shown in Figures 3.6-15 and 3.6-16. The calculated stress levels and usage factors, and the postulated break types, for the feedwater piping inside the primary containment are listed in Table 3.6-8. Breaks for the feedwater piping outside containment are postulated to occur at each location of potential high stress, such as pipe fittings, valves, and welded attachments.

Compartment Pressure-Temperature Transients

The pressure-temperature transient in the primary containment resulting from a break of any of the sizes of feedwater lines in the drywell is exceeded in severity by the transients resulting from recirculation loop and main steam line breaks, which are discussed in Section 6.2.1. The pressure-temperature transient resulting from a feedwater line break in the main steam tunnel or turbine enclosure is exceeded in severity by the transient resulting from a main steam line break, which is discussed in Section 3.6.1.2.1.2.

Verification of Reactor Shutdown Capability

Breakage of a feedwater line inside the primary containment would result in an unisolable blowdown of the reactor vessel. The sequence of events that would occur automatically to shut the reactor down and cool the core is discussed in Section 6.3.3.

For a feedwater line break outside the primary containment, differential pressure across the containment isolation check valves in the reverse direction will cause these valves to close rapidly, isolating the reactor vessel from the break. The loss of feedwater flow will cause the reactor water level to drop, initiating a reactor scram. Water level will continue to drop because of steam generation from decay heat, causing closure of the MSIVs as well as initiation of the RCIC and HPCI systems. Once the reactor has been scrammed and the RCS isolated, the sequence of events is similar to that for a main steam line break outside the primary containment.

A combination of pipe whip restraints, jet impingement barriers, and separation by distance or intervening structure is used to ensure the availability of essential systems and components in the event of a feedwater line break in either the drywell or the main steam tunnel. Since no essential systems and components are located in the turbine enclosure, no special provisions are necessary to provide protection for equipment in this area from the effects of pipe breaks. Bumpers are provided in the turbine enclosure to protect the control structure from the impact of a whipping feedwater line.

3.6.1.2.1.4 Condensate System

The condensate system is located entirely within the turbine enclosure. No pipe whip restraints are provided for the condensate piping.

Pipe Break Locations

Since the condensate system consists of non-nuclear class piping, breaks are postulated to occur at each location of potential high stress, such as pipe fittings, valves, and welded attachments.

Compartment Pressure-Temperature Transients

Since the normal fluid temperature in the condensate system is less than 135°F, no significant pressure-temperature transient would result from condensate line breaks.

Verification of Reactor Shutdown Capability

In the event of a condensate line break, the feedwater pumps would trip because of low suction pressure. The resulting loss of feedwater flow would result in closure of the containment isolation check valves, preventing reactor blowdown through the break. The sequence of events that would occur from this point on is the same as for breakage of a feedwater line outside the primary containment.

Since no essential systems and components are located in the turbine enclosure, no special provisions are necessary to provide protection for equipment in this area from the effects of pipe breaks. The flooding effects of a condensate line break are exceeded by the effects of a circulating water line expansion joint rupture in the turbine enclosure, a discussion of which is contained in Sections 10.4.1 and 10.4.5.

3.6.1.2.1.5 Reactor Water Cleanup System

The RWCU system takes suction from reactor recirculation loop B via the RHR shutdown cooling suction line inside primary containment. The 6-inch RWCU suction line is routed up to el 297'-3", at which point it penetrates the primary containment. The RWCU piping is then routed through the various RWCU equipment compartments at el 283' and el 313' in the reactor enclosure, including the containment penetration compartment, three RWCU pump compartments, the regenerative heat exchanger compartment, two nonregenerative heat exchanger compartments, two RWCU holding pump compartments, and two RWCU filter/demineralizer compartments. The RWCU return piping is routed from the containment penetration compartment directly into the main steam tunnel, where the piping splits into two 4-inch lines. One line connects to the A feedwater line (DBB-103) and the other line connects to the RCIC injection line immediately upstream of its connection to the B feedwater line (DBB-104). Both of these connections to the feedwater lines are located between the two outboard containment isolation valves in the feedwater lines. Another RWCU line, the 4-inch RWCU alternate return line, has a branch connection to the RWCU return line at a point just upstream of the location where the return line splits into the two lines leading to the feedwater lines. From this branch connection, the RWCU alternate return line is routed downward to a containment penetration at elevation 261 feet, which is also in the main steam tunnel. The routing of this RWCU piping is shown in drawings M-210, M-211, M-225, M-231, M-232, M-235, M-296, M-302, M-303, M-306, M-322, and M-323.

The RWCU suction line is provided with two fast-acting isolation valves, one upstream and one downstream of the primary containment penetration. These valves close automatically upon receipt of signals indicating high differential flow (RWCU suction vs. return) or high temperature in the vicinity of the piping outside the drywell (as well as upon receipt of other initiating signals), in order to terminate blowdown through breaks outside the drywell. In order to ensure the operability of these valves in the event of an RWCU suction line break, moment-limiting pipe whip restraints are located upstream of the inboard containment isolation valve and downstream of the outboard containment isolation valve. Whip restraints are also located on the portion of the RWCU suction line within the drywell and on the portion of the return line within the main steam tunnel.

In reviewing the potential consequences of jet impingement resulting from RWCU line breaks, it was determined that breaks near the connection of the RWCU return lines to the RCIC injection and feedwater line in the main steam tunnel could result in impingement

on the outboard MSIV operators. A steel plate barrier is provided to protect the MSIV operators from this potential source of jet impingement. It was also determined that RWCU line breaks in the RWCU containment penetration compartment could result in impingement on the RWCU outboard containment isolation valve. A steel plate barrier is provided to protect the isolation valve from this potential source of jet impingement.

Pipe Break Locations

The postulated pipe break locations for the RWCU pump suction piping inside the primary containment, and also the pipe whip restraint locations, are shown in Figure 3.6-17. The calculated stress levels and usage factors, and the postulated break types, for this portion of the RWCU piping are listed in Table 3.6-10. Piping isometrics for the RWCU pump suction piping in the RWCU isolation valve compartment and for the RWCU return piping in the main steam tunnel are shown in Figures 3.6-39 and 3.6-18, respectively. Breaks in the latter portions of the RWCU piping are postulated to occur at each location of potential high stress, such as pipe fittings, valves, and welded attachments.

Compartment Pressure-Temperature Transients

The pressure-temperature transient in the primary containment resulting from a break in the portion of the RWCU suction line in the drywell is exceeded in severity by the transients resulting from recirculation loop breaks, main steam line breaks, and small steam leaks, which are discussed in Section 6.2.1.

Protection against overpressurization of the RWCU equipment compartments in the reactor enclosure as a result of RWCU line breaks in these areas is provided by interconnecting steam venting paths between the various compartments and by blowout panels leading to the outside atmosphere. The two nonregenerative heat exchanger compartments are vented into the regenerative heat exchanger compartment via a short steam venting plenum. The regenerative heat exchanger compartment is vented in turn to the adjacent containment penetration compartment. Each of the three pump rooms is vented individually to the containment penetration compartment. The containment penetration compartment is vented, via a blowout panel, to an exhaust stack located on the west side of the reactor enclosure for Unit 1 and on the east side of the reactor enclosure for Unit 2. The stack is open to the atmosphere at its upper and lower ends. The venting pathway from the containment penetration compartment to the exhaust stack is shown in drawings M-215 and M-316.

Pressure-temperature transient analyses for the cases involving RWCU line breaks in the RWCU equipment compartments were performed using the analytical technique described in Reference 3.6-1 and with the blowdown data provided in Table 3.6-6. These blowdown data were developed using Reference 3.6-2. The flow model for breaks in the RWCU equipment compartments is shown in Figure 3.6-19, and the results of the analyses are listed in Table 3.6-7. No analyses are performed for the RWCU holding pump compartments and filter/demineralizer compartments since the piping in these areas contains fluid at a temperature of 150°F or less, so that no significant pressure-temperature transient results from a pipe break.

The RWCU equipment compartments are designed to withstand the maximum pressure developed due to a pipe break. The outboard containment isolation valves for the RWCU system and other systems located in the containment penetration compartment are qualified to operate under environmental conditions more severe than those calculated to occur due to pipe break.

Verification of Reactor Shutdown Capability

Breakage of the RWCU suction line inside the drywell would result in an unisolable blowdown of the reactor vessel. The sequence of events that would occur automatically to shut the reactor down and cool the core is discussed in Section 6.3.3.

For an RWCU line break outside the drywell, the RWCU containment isolation valves will be closed automatically due to high flow in the RWCU suction line or high temperature in the RWCU equipment compartments. Backflow from the feedwater lines into the RWCU return line will be prevented by closure of the check valve in the return line. If the break occurs in the RWCU piping within the main steam tunnel, the MSIVs will close automatically due to high temperature in the tunnel. Once MSIV isolation has occurred, the sequence of events is similar to that for a main steam line break outside the primary containment. If the break occurs in the RWCU equipment compartments, no reactor scram or MSIV closure will occur, due to the isolation of the RWCU system and rapid termination of the blowdown. After an RWCU isolation, there is sufficient capability to shutdown the reactor. The operator will investigate the cause of the RWCU isolation and take appropriate actions, in accordance with the plant Technical Specifications and approved operating procedures. The Technical Specifications and/or the operating procedures, which include off normal and EOPs, may, based on plant conditions, require the operator to initiate a normal shutdown of the reactor.

A combination of pipe whip restraints, jet impingement barriers, and separation by distance or intervening structure is used to ensure the availability of essential systems and components in the event of an RWCU line break occurring in the drywell, main steam tunnel, or RWCU equipment compartments. Essential systems and components located in these areas are qualified to operate under the environmental conditions resulting from the break. Among the RWCU equipment compartments, only the containment penetration compartment contains safety-related equipment: the primary containment purge line, two of the four LPCI injection lines, and the RWCU outboard containment isolation valve. The containment isolation valves in the purge line and the LPCI injection lines are normally closed during reactor operation and are not required to operate after a RWCU line break outside primary containment; therefore they require no protection. The RWCU outboard containment isolation valve is protected from jet impingement by a steel plate barrier, and the cabling to the valve is routed so as to avoid jet impingement.

3.6.1.2.1.6 Reactor Vessel Drain

The reactor vessel drain consists of 2-inch piping originating at a nozzle on the reactor vessel bottom head. From this location inside the reactor vessel pedestal, the piping increases to 2½ inches in diameter, penetrates the pedestal into the general drywell area, increases to 4 inches in diameter, and is routed upward to its point of connection to the 6-inch RWCU suction line inside the drywell.

Pipe Break Locations

The postulated pipe break locations for the reactor vessel drain line, and also the pipe whip restraint locations, are shown in Figure 3.6-20. The calculated stress levels and usage factors, and the postulated break types, are listed in Table 3.6-12.

Compartment Pressure-Temperature Transients

Since the reactor vessel drain line is located entirely within the drywell, breakage of this line would have no effect on plant areas outside the primary containment. The pressure-temperature transient in the primary containment resulting from a reactor vessel drain line break is exceeded in severity by the transients resulting from recirculation loop breaks and main steam line breaks, which are discussed in Section 6.2.1.

Verification of Reactor Shutdown Capability

Breakage of the reactor vessel drain line would result in an unisolable blowdown of the reactor vessel. The sequence of events that would occur automatically to shut the reactor down and cool the core is discussed in Section 6.3.3. A combination of pipe whip restraints and separation by distance or intervening structure is used to ensure the availability of essential systems and components in the event of a reactor vessel drain line break.

3.6.1.2.1.7 HPCI Steam Supply Line

The HPCI steam supply piping has a nominal diameter of 10 inches for the portion inside the drywell and 12 inches for the portion outside the drywell. The supply line connects to main steam line C inside the drywell. From its connection to the main steam line, the HPCI steam supply line is routed downward and then horizontally along the drywell wall. The line then penetrates the drywell at el 244'-8", entering the isolation valve compartment located at floor el 217' in the reactor enclosure. The steam supply line penetrates the floor of the isolation valve compartment and enters the HPCI pump compartment located at el 177'. The routing of this line is shown in drawings M-213, M-225, M-226, M-227, M-228, M-295, M-297, M-296, M-318, M-319, M-229, and M-320. During normal reactor operation, the line is pressurized from main steam line C up to the HPCI turbine steam supply valve (HV-55-F001).

The HPCI steam supply line is provided with two fast-acting isolation valves, one upstream and one downstream of the primary containment penetration. These valves close automatically upon receipt of signals indicating high steam flow or high temperature in the vicinity of the piping outside the drywell (as well as upon receipt of other initiating signals), in order to terminate blowdown through breaks outside the drywell.

Moment-limiting pipe whip restraints are located upstream of the inboard containment isolation valve and downstream of the outboard containment isolation valve in order to ensure the operability of these valves in the event of a break in the HPCI steam supply line near the valves.

Whip restraints are also located on the HPCI steam supply line and in the isolation valve compartment at el 217' in the reactor enclosure. A typical restraint inside the drywell is shown in Figure 3.6-1.

Pipe Break Locations

The postulated pipe break locations for the HPCI steam supply line, and also the pipe whip restraint locations, are shown in Figure 3.6-21 for the portion of the line inside the drywell and in Figure 3.6-22 for the portion of the line outside the drywell. The calculated stress levels and usage factors, and the postulated break types, are listed in Tables 3.6-13 and 3.6-14.

Compartment Pressure-Temperature Transients

The pressure-temperature transient in the primary containment resulting from a break in the portion of the HPCI steam supply line in the drywell is exceeded in severity by the transients resulting from recirculation loop breaks and main steam line breaks, which are discussed in Section 6.2.1.

Protection against overpressurization of the HPCI pump compartment and the isolation valve compartment at el 217' in the reactor enclosure as a result of HPCI steam supply line breaks in these areas is provided by steam venting paths and blowout panels leading to the outside atmosphere. The isolation valve compartment is vented to the atmosphere via blowout panels located on the south side of the reactor enclosure, as shown in drawings M-118, M-123 and M-138. The HPCI pump compartment is vented to the isolation valve compartment via hinged, metal plate blowout panels located in the floor at el 217'. These latter panels lift to relieve pressurization in the HPCI pump compartment but do not allow pressurization in the isolation valve compartment to result in steam flow in the reverse direction, i.e., down into the HPCI pump compartment.

Pressure-temperature transient analyses for the cases involving HPCI steam supply line breaks in the HPCI pump compartment and the HPCI piping area were performed using the analytical technique described in Reference 3.6-1 and the blowdown data provided in Table 3.6-6. The flow model for breaks in these two compartments is shown in Figure 3.6-23, and the results of the analyses are listed in Table 3.6-7. Figures 3.6-43 through 3.6-45 provide additional details of a typical pipe break analysis. The case of an HPCI steam supply line break in the isolation valve compartment was analyzed using the CFLUD computer program (Reference 3.6-13) and the blowdown data provided in Table 3.6-6. The flow model for this case is shown in Figure 3.6-24, and the results of the analysis are listed in Table 3.6-7.

The HPCI pump compartment and the isolation valve compartment are designed to withstand the maximum pressures developed due to pipe break. All equipment located in the isolation valve compartment that is required to operate following breakage of the HPCI steam supply line is qualified to operate under environmental conditions more severe than those calculated to occur due to pipe break. No equipment located in the HPCI pump compartment is required to operate following breakage of the HPCI steam supply line.

Verification of Reactor Shutdown Capability

Breakage of the HPCI steam supply line inside the drywell would result in an unisolable blowdown of the reactor vessel. The sequence of events that would occur automatically to shut the reactor down and cool the core is discussed in Section 6.3.3.

For an HPCI steam supply line break outside the drywell, the steam supply line containment isolation valves will be closed automatically, terminating reactor vessel blowdown. No reactor scram will occur, due to the isolation of the HPCI steam line and rapid termination of the blowdown. After a HPCI isolation, there is sufficient capability to shutdown the reactor. The operator will investigate the cause of the HPCI isolation and take the appropriate actions, in accordance with the plant Technical Specifications and approved operating procedures, to assure that the plant is in safe condition. The Technical Specifications and/or the operating procedures, which include off normal and EOPs, may, based on plant conditions, require the operator to initiate a normal shutdown of the reactor.

A combination of pipe whip restraints and separation by distance or intervening structure is used to ensure the availability of essential systems and components in the event of an HPCI steam supply line break occurring in the drywell, the isolation valve compartment, or the HPCI pump compartment. Essential systems and components located in these areas are qualified to operate under the environmental conditions resulting from the break. Electrical cabling associated with essential systems and components is routed so as to avoid jet impingement from the postulated break.

No pipe whip restraints are provided for the portion of the HPCI steam supply line located within the HPCI pump compartment. Since no essential systems and components (other than the HPCI system itself) are located in the HPCI pump compartment, such protection is not necessary.

3.6.1.2.1.8 RCIC Steam Supply Line

The RCIC steam supply line has a nominal diameter of 4 inches for the portion inside the drywell and 6 inches for the portion outside the drywell. The supply line connects to main steam line B inside the drywell. From its connection to the main steam line, the RCIC steam supply line is routed generally downward and then horizontally along the drywell wall. The line then penetrates the drywell at el 243'-6", entering the isolation valve compartment located at floor el 217' in the reactor enclosure. The steam supply line penetrates the floor of the isolation valve compartment and enters the RCIC pump compartment located at el 177'. The routing of this line is shown in drawings M-213, M-215, M-225, M-226, M-227, M-229, M-239, M-295, M-296, M-297, M-310, M-316, M-318, and M-320. During normal reactor operation, the line is pressurized from main steam line B up to the RCIC turbine steam supply valve (HV-50-F045).

The RCIC steam supply line is provided with two fast-acting isolation valves, one upstream and one downstream of the primary containment penetration. These valves close automatically upon receipt of signals indicating high steam flow or high temperature in the vicinity of the piping outside the drywell (as well as upon receipt of other initiating signals), in order to terminate blowdown through breaks outside the drywell.

Moment-limiting pipe whip restraints are located upstream of the inboard containment isolation valve and downstream of the outboard containment isolation valve in order to ensure the operability of these valves in the event of a break in the RCIC steam supply line near the valves. Whip restraints are also located on the RCIC steam supply line inside the drywell and in the isolation valve compartment at el 217' in the reactor enclosure. A typical restraint inside the drywell is shown in Figure 3.6-1.

Pipe Break Locations

The postulated pipe break locations for the RCIC steam supply line, and also the pipe whip restraint locations, are shown in Figure 3.6-25 for the portion of the line inside the drywell and in Figure 3.6-26 for the portion of the line outside the drywell. The calculated stress levels and usage factors, and the postulated break types, are listed in Tables 3.6-15 and 3.6-16.

Compartment Pressure-Temperature Transients

The pressure-temperature transient in the primary containment resulting from a break in the portion of the RCIC steam supply line in the drywell is exceeded in severity by the transients resulting from recirculation loop breaks and main steam line breaks, which are discussed in Section 6.2.1.

Protection against overpressurization of the RCIC pump compartment and the isolation valve compartment at el 217' in the reactor enclosure as a result of RCIC steam supply line breaks in these areas is provided by steam venting paths and blowout panels leading to the outside atmosphere. The isolation valve compartment is vented to the atmosphere via blowout panels located on the south side of the reactor enclosure, as shown in drawings M-118, M-123 and M-138. The RCIC pump compartment is vented to the isolation valve compartment via hinged, metal plate blowout panels located in the floor at el 217'. These latter panels lift to relieve pressurization in the RCIC pump compartment but do not allow pressurization in the isolation valve compartment to result in steam flow in the reverse direction, i.e., down into the RCIC pump compartment.

Pressure-temperature transient analyses for the cases involving RCIC steam supply line breaks in the RCIC pump compartment and the RCIC upper pipe tunnel were performed using the analytical technique described in Reference 3.6-1 and the blowdown data provided in Table 3.6-6. The flow model for breaks in these two compartments is shown in Figure 3.6-27, and the results of the analyses are listed in Table 3.6-7.

The RCIC pump compartment is designed to withstand the maximum pressure developed due to pipe break. No equipment located in the RCIC pump compartment is required to operate following breakage of the RCIC steam supply line. The pressure-temperature transient in the isolation valve compartment at el 217' resulting from a break in the portion of the RCIC steam supply line within that compartment is exceeded in severity by the transient resulting from HPCI steam supply line breakage in the same compartment. All equipment located in the isolation valve compartment that is required to operate following breakage of the RCIC steam supply line is qualified to operate under environmental conditions more severe than those calculated to occur due to breakage of the HPCI steam supply line.

Verification of Reactor Shutdown Capability

Breakage of the RCIC steam supply line inside the drywell would result in an unisolable blowdown of the reactor vessel. The sequence of events that would occur automatically to shut the reactor down and cool the core is discussed in Section 6.3.3.

For an RCIC steam supply line break outside the drywell, the steam supply line containment isolation valves will be closed automatically, terminating reactor vessel blowdown. No reactor scram will occur, due to the isolation of the RCIC steam line and rapid termination of the blowdown. After a RCIC isolation, there is sufficient capability to shutdown the reactor. The operator will investigate the cause of the RCIC isolation and take appropriate actions, in accordance with the plant Technical Specifications and approved operating procedures, to assure that the plant is in a safe condition. The Technical Specifications and/or the operating procedures, which include off normal and EOPs, may, based on plant conditions, require the operator to initiate a normal shutdown of the reactor.

A combination of pipe whip restraints and separation by distance or intervening structure is used to ensure the availability of essential systems and components in the event of an RCIC steam supply line break occurring in the drywell, the isolation valve compartment, or the RCIC pump compartment. Essential systems and components located in these areas are qualified to operate under the environmental conditions resulting from the break. Electrical cabling associated with essential systems and components is routed so as to avoid jet impingement from the postulated break.

No pipe whip restraints are provided for the portion of the RCIC steam supply line located within the RCIC pump compartment. Since no essential systems and components are located within the RCIC pump compartment, such protection is not necessary.

3.6.1.2.1.9 Main Steam Drain Lines

The main steam drainage piping connects to the four main steam lines both inside and outside the drywell. Inside the drywell, 2-inch drain lines that connect to each of the main steam lines are headered together into a single 3-inch line, which then penetrates the drywell wall. This 3-inch drain header is provided with two containment isolation valves, one upstream and one downstream of the containment penetration, both of which are normally closed. Outside the drywell (in the main steam tunnel), 2-inch drain lines that connect to each of the main steam lines are headered together into a single 3-inch line, which then connects to the drain header from the drain lines inside the drywell. Downstream of the connection of the two drain headers, the common line splits into two 3-inch lines again, one routed to the main condenser and one routed to the liquid waste management system. These latter two lines are both provided with normally closed valves located within the main steam tunnel. The routing of the main steam drain lines is shown in drawings M-217, M-226, M-297, and M-326.

Pipe Break Locations

The postulated pipe break locations for the main steam drain lines inside the drywell are shown in Figure 3.6-28. The calculated stress levels and usage factors, and the postulated break types, are listed in Table 3.6-17. Breaks in the main steam drain lines within the main steam tunnel are postulated to occur at each location of potential high stress, such as pipe fittings, valves, and welded attachments.

Compartment Pressure-Temperature Transients

The pressure transient in the primary containment resulting from a break in a main steam drain line within the drywell is exceeded in severity by transients resulting from recirculation loop breaks and main steam line breaks. The temperature transient in the primary containment resulting from a main steam drain line break is exceeded in severity by the transient resulting from a small steam leak. These design basis transients are discussed in Section 6.2.1.

The pressure-temperature transient in the main steam tunnel resulting from a main steam drain line break within the tunnel is exceeded in severity by the transient resulting from a main steam line break.

Verification of Reactor Shutdown Capability

Breakage of a main steam drain line inside the drywell would result in an unisolable blowdown of steam from the reactor vessel through the broken line. The sequence of events that would occur automatically to shut the reactor down and cool the core is discussed in Section 6.3.3.

For the case of a main steam drain line break inside the main steam tunnel, the resultant temperature rise in the tunnel would cause the MSIVs and the MSL drain isolation valves, if open to close automatically, thereby terminating steam blowdown through the break. Once the MSIVs and MSL drain isolation valves have been closed, the sequence of events is similar to that for a main steam line break outside the drywell.

Separation by distance and intervening structure is used to ensure the availability of essential systems and components in the event of a main steam drain line break in either the drywell or the main steam tunnel.

3.6.1.2.1.10 RPV Head Vent Line

The RPV head vent line is located entirely within the drywell. From its connection point to a 4-inch flanged nozzle on the RPV head, the line reduces to 2 inches in diameter and then is routed generally downward to a penetration through the containment seal plate. From this point, the line continues on downward to its connection with 26-inch main steam line C.

Pipe Break Locations

The postulated pipe break locations for the RPV head vent line, and also the pipe whip restraint locations, are shown in Figure 3.6-30. The calculated stress levels and usage factors, and the postulated break types, are listed in Table 3.6-19.

Compartment Pressure-Temperature Transients

Since the RPV head vent line is located entirely within the drywell, breakage of this line has no effect on plant areas outside the primary containment. The pressure transient in the primary containment resulting from a break in the RPV head vent line is exceeded in severity by transients resulting from recirculation loop breaks and main steam line breaks. The temperature transient in the primary containment resulting from a break in the RPV head vent line is exceeded in severity by the transient resulting from a small steam leak. These design basis transients are discussed in Section 6.2.1.

Verification of Reactor Shutdown Capability

Breakage of the RPV head vent line would result in an unisolable blowdown of steam into the drywell. The sequence of events that would occur automatically to shut the reactor down and cool the core is discussed in Section 6.3.3.

Separation by distance and intervening structure is used to ensure the availability of essential systems and components in the event of an RPV head vent line break.

3.6.1.2.1.11 Standby Liquid Control System Injection Line

The discharge lines from the SLCS injection pumps penetrate the drywell through two separate penetrations and are headered together inside the drywell to form a single 2-inch line. From the vicinity of the drywell penetrations (between el 265' and el 271'), this line is routed horizontally and upward to its connection to core spray injection line B (12" DCA-319). In addition to the containment isolation valves, the SLCS injection line is provided with a check valve near the connection to the core spray line. Only that portion of the SLCS injection line between the core spray line and the inboard check valve is considered high energy during periods when the reactor is pressurized.

Pipe Break Locations

The postulated pipe break locations for the SLCS injection line are shown in Figure 3.6-31. The calculated stress levels and usage factors, and the postulated break types, are listed in Table 3.6-20.

Compartment Pressure-Temperature Transients

Since the high energy portion of the SLCS injection line is located entirely within the drywell, breakage of this line would have no effect on plant areas outside the primary containment. The pressure-temperature transient in the primary containment resulting from a break in the SLCS injection line is exceeded in severity by the transients resulting from recirculation loop breaks and main steam line breaks, which are discussed in Section 6.2.1.

Verification of Reactor Shutdown Capability

Breakage of the SLCS injection line between the RPV and the first check valve in that line would result in an unisolable blowdown from the reactor vessel into the drywell. The sequence of events that would occur automatically to shut the reactor down and cool the core is discussed in Section 6.3.3.

Separation by distance and intervening structure is used to ensure the availability of essential systems and components in the event of an SLCS injection line break.

3.6.1.2.1.12 RHR Shutdown Cooling Suction Line

The RHR shutdown cooling suction line is a 20-inch line connected to recirculation loop B in the drywell. The line is routed downward and then horizontally to its containment penetration at el 244'-8". The line is provided with two containment isolation valves (one inboard and one outboard of the penetration), both of which are normally closed. Thus, only that portion of the line between the recirculation loop and the inboard containment isolation valve is considered high energy. The routing of the line is shown in drawings M-213, M-217, M-225, M-295, M-296, and M-326.

The RHR shutdown cooling suction line is provided with pipe whip restraints on the portion of the line inside the drywell. Examples of these restraints are shown in Figure 3.6-1.

Pipe Break Locations

The postulated pipe break locations for the RHR shutdown cooling suction line, and also the pipe whip restraint locations, are shown in Figure 3.6-32. The calculated stress levels and usage factors, and the postulated break types, are listed in Table 3.6-21.

Compartment Pressure-Temperature Transients

Since the high energy portion of the RHR shutdown cooling suction line is located entirely within the drywell, breakage of the line would have no effect on plant areas outside the primary containment. The pressure-temperature transient in the primary containment resulting from a break in the RHR shutdown cooling suction line is exceeded in severity by the transients resulting from recirculation loop breaks and main steam line breaks, which are discussed in Section 6.2.1.

Verification of Reactor Shutdown Capability

Breakage of the RHR shutdown cooling suction line would result in an unisolable blowdown of the reactor vessel. The sequence of events that would occur automatically to shut the reactor down and cool the core is discussed in Section 6.3.3.

A combination of pipe whip restraints and separation by distance and intervening structure is used to ensure the availability of essential systems and components in the event of a break in the RHR shutdown cooling suction line.

3.6.1.2.1.13 RHR Shutdown Cooling Return Line

One 12-inch RHR shutdown cooling return line is associated with RHR loop A and a second return line is associated with RHR loop B. Since the two return lines are routed symmetrically on opposite sides of the drywell, the following discussion applies to both lines. The RHR shutdown cooling return line, from the discharge of the associated RHR pump and heat exchanger, penetrates the south side of the drywell at el 244'-8". The line is then routed horizontally following the curvature of the drywell wall until a point due west (for loop A; due east for loop B) of the reactor vessel centerline is reached. From this point, the line is routed upward to el 265'-4" and then horizontally to its connection with the discharge riser of the reactor recirculation loop. The routing of this piping is shown in drawings M-213, M-217, M-225, M-295, M-296, and M-326.

The RHR shutdown cooling return line is provided with two containment isolation valves: a normally closed globe valve outside the drywell and a check valve inside the drywell. Only that portion of the line between the reactor recirculation line and the inboard check valve is considered high energy during periods when the reactor is pressurized.

The RHR shutdown cooling return line is provided with one pipe whip restraint, located at the upper elbow of the vertical portion of the line. A detail of this restraint is shown in Figure 3.6-1.

Pipe Break Locations

The postulated pipe break locations for the RHR shutdown cooling return line, and also the pipe whip restraint locations, are shown in Figure 3.6-32. The calculated stress levels and usage factors, and the postulated break types, are listed in Table 3.6-21.

Compartment Pressure-Temperature Transients

Since the high energy portion of the RHR shutdown cooling return line is located entirely within the drywell, breakage of the line would have no effect on plant areas outside the primary containment. The pressure-temperature transient in the primary containment resulting from a break in the RHR shutdown cooling return line is exceeded in severity by the transients resulting from recirculation loop breaks and main steam line breaks, which are discussed in Section 6.2.1.

Verification of Reactor Shutdown Capability

Breakage of the RHR shutdown cooling return line would result in an unisolable blowdown of the reactor vessel. The sequence of events that would occur automatically to shut the reactor down and cool the core is discussed in Section 6.3.3.

A combination of pipe whip restraints and separation by distance and intervening structure is used to ensure the availability of essential systems and components in the event of a break in the RHR shutdown cooling return line.

3.6.1.2.1.14 LPCI Injection Line

There are four 12-inch LPCI injection lines, one associated with each of the four RHR pumps. The four lines are routed symmetrically inside the drywell, with the A and C injection lines entering the west side of the drywell and the B and D lines entering the east side of the drywell. The following discussion applies to all four lines. The LPCI injection line penetrates the drywell at el 285'-2" and is routed up to el 297'-3" inches where it connects to the reactor vessel nozzle. The routing of this piping is shown in drawings M-215, M-217, M-234, M-235, M-305, M-306, M-316, and M-326.

The LPCI injection line is provided with two containment isolation valves: a normally closed gate valve outside the drywell and a check valve inside the drywell. Only that portion of the line between the reactor vessel nozzle and the inboard check valve is considered high energy during periods when the reactor is pressurized.

The LPCI injection line is restrained to prevent pipe whip inside the drywell. A typical restraint is shown in Figure 3.6-1.

Pipe Break Locations

The postulated pipe break locations for the LPCI injection line, and also the pipe whip restraint locations, are shown in Figure 3.6-33. The calculated stress levels and usage factors, and the postulated break types, are listed in Table 3.6-22

Compartment Pressure-Temperature Transients

Since the high energy portion of the LPCI injection line is located entirely within the drywell, breakage of the line would have no effect on plant areas outside the primary containment. The pressure-temperature transient in the primary containment resulting from a break in the LPCI injection line is exceeded in severity by the transients resulting from recirculation loop breaks and main steam line breaks, which are discussed in Section 6.2.1.

Verification of Reactor Shutdown Capability

Breakage of the LPCI injection line would result in an unisolable blowdown of the reactor vessel. The sequence of events that would occur automatically to shut the reactor down and cool the core is discussed in Section 6.3.3.

A combination of pipe whip restraints and separation by distance and intervening structure is used to ensure the availability of essential systems and components in the event of a break in the LPCI injection line.

3.6.1.2.1.15 Core Spray Injection Line

There are two core spray injection lines, one associated with core spray pumps A and C and one associated with core spray pumps B and D. Since the two lines are routed symmetrically within the drywell, the following discussion applies to both lines. The core spray injection line penetrates the north side of the drywell at el 297'-3" and is routed up to el 306'-7" before connecting to the RPV nozzle. The routing of the piping is shown in drawings M-217, M-235, M-306, and M-326.

The core spray injection line is provided with containment isolation valves both inside and outside the drywell, the inboard valve being a check valve. Only that portion of the line between the reactor vessel nozzle and the inboard check valve is considered high energy during periods when the reactor is pressurized.

The core spray injection line is restrained to prevent pipe whip inside the drywell. Typical restraints are shown in Figure 3.6-1.

Pipe Break Locations

The postulated pipe break locations for the core spray injection line, and also the pipe whip restraint locations, are shown in Figure 3.6-34. The calculated stress levels and usage factors, and the postulated break types, are listed in Table 3.6-23.

Compartment Pressure-Temperature Transients

Since the high energy portion of the core spray injection line is located entirely within the drywell, breakage of the line would have no effect on plant areas outside the primary containment. The pressure-temperature transient in the primary containment resulting from a break in the core spray injection line is exceeded in severity by the transients resulting from recirculation loop breaks and main steam line breaks, which are discussed in Section 6.2.1.

Verification of Reactor Shutdown Capability

Breakage of the core spray injection line would result in an unisolable blowdown of the reactor vessel. The sequence of events that would occur automatically to shut the reactor down and cool the core is discussed in Section 6.3.3.

A combination of pipe whip restraints and separation by distance and intervening structure is used to ensure the availability of essential systems and components in the event of a break in the core spray injection line.

3.6.1.2.1.16 Control Rod Drive Hydraulic

The 2 CRD water pumps are located in the turbine enclosure. The high energy discharge pipes from the 2 pumps are headered together, and a single 2-inch pipe is routed from the turbine enclosure into the reactor enclosure at el 201'. This 2-inch line is then routed upward to the CRD hydraulic system master control station located at el 253' of the reactor enclosure. From the master control station, a 2-inch cooling water header and a 2-inch charging water header are routed to the groups of HCUs on the west side and east side of the drywell. Containment isolation valves are provided in each header between the master control station and the HCUs. The piping between the isolation valves has been upgraded to be equivalent to ASME III, Class 2.

Pipe Break Locations

Since the CRD pump discharge line, the cooling water header, and the charging water header originally consisted entirely of non-nuclear class piping, breaks are postulated to occur at each location of potential high stress, such as pipe fittings, valves, and welded attachments. The original analysis bounds the effects of the addition of containment isolation valves in the headers.

Compartment Pressure-Temperature Transients

Since the normal fluid temperature in the CRD hydraulic system is less than 120°F, no significant pressure-temperature transient would result from postulated breaks.

Verification of Reactor Shutdown Capability

Loss of water pressure due to a break in the CRD pump discharge line, cooling water header, or charging water header will not prevent the control rods from being inserted into the reactor core. At reactor pressures of 450 psig or higher, reactor pressure alone is sufficient to fully insert the control rods. At lower reactor pressures, the scram accumulators assist in supplying the energy necessary to insert the control rods.

3.6.1.2.1.17 Auxiliary Steam Line

From the auxiliary boiler, auxiliary steam is distributed via an 8-inch header to the various steam-consuming components in the turbine enclosure and the radwaste enclosure. This auxiliary steam header passes through the offgas pipe tunnel north of the control structure at el 187'. The auxiliary steam system also provides steam to the RCIC and HPCI systems for testing purposes.

Pipe Break Locations

Since the auxiliary steam line consists of non-nuclear class piping, breaks are postulated to occur at each location of potential high stress, such as pipe fittings, valves, and welded attachments.

Compartment Pressure-Temperature Transients

A pressure-temperature transient analysis for the case of an auxiliary steam line break in a safety-related area was not performed because the piping does not pass through any safety related areas with the exception of the test lines to HPCI and RCIC. However, a pressure-temperature transient analysis was performed for the auxiliary steam line break in the offgas pipe tunnel to ensure control structure integrity. The HPCI and RCIC test lines are normally isolated from the reactor enclosure by closed valves. The lines are only energized in the reactor enclosure for short periods (i.e., less than 1% of the time) and are therefore considered as moderate energy lines.

Verification of Reactor Shutdown Capability

Breakage of the auxiliary steam line would have no effect on operation of the reactor. Except for the test lines for HPCI and RCIC, auxiliary steam lines are not located in areas containing safety related equipment. In the case of the HPCI and RCIC test lines, they are considered to be moderate energy lines based on the short periods in which they are energized. As moderate energy lines, they are acceptable since all rooms containing safety related equipment are qualified for a moderate energy line break.

3.6.1.2.1.18 Plant Heating Steam Piping

From the auxiliary boiler, plant heating steam is distributed to various steam-consuming components in the turbine enclosure and the diesel generator compartments. Additionally, heating steam is provided to the refueling floor and the railroad air-lock in the reactor enclosure and in the diesel corridor outside of the reactor enclosure, however, none of these areas contain safety related equipment necessary to shut down the reactor or maintain primary containment integrity.

Pipe Break Locations

Because the plant heating steam piping consists of non-nuclear class piping, breaks are postulated to occur at each location of potential high stress, such as pipe fittings, valves, and welded attachments.

Verification of Reactor Shutdown Capability

Breakage of the plant heating steam piping would have no effect on operation of the reactor because reactor systems are not located in areas through which the heating steam piping is routed. The only areas of the plant through which heating steam piping is routed that contain safety-related systems are the diesel generator compartments, the refuel floor, the diesel corridor, and the railroad air lock. In addition to the diesel generators, the safeguard MCCs for the following valves are located within diesel generator enclosures:

a. ESW to RHRSW valves (HV-11-11A, 11B, 15A, 15B; drawing M-11).

b. ESW to TECW heat exchanger valves (HV-11-105, 107, 205, 207; Figure 9.2-2).

c. TECW heat exchanger to RHRSW valves (HV-12-110, 210; drawing M-12).

d. CST to ECCS (HV-55-124, 224; drawing M-55).

The MCCs for these valves are distributed throughout the diesel generator cells such that the loss of the MCCs in any one cell would not compromise the safety function of the systems. Thus, an HELB in one cell would not affect the safe shutdown capability of the plant.

A rupture of the plant steam heating piping in one cell would not affect the diesel generators located in other cells. Thus, a failure of this system would result in loss of only the one diesel. The safe shutdown capability of the plant would therefore not be compromised because only three of the four diesels are required in the event of a LOOP.

Because the loss of components in any diesel generator compartment from a high or moderate energy line break would not precipitate a trip of the turbine-generator or a trip of the RPS, offsite power is assumed to remain available in the analysis of the effects of postulated piping failures in these systems (BTP ASB 3.1, paragraph 3.b.1).

There are no essential systems or components as defined in Branch Technical Position SPLB 3-1 (Protection against postulated piping failures in fluid systems outside containment) on the refueling floor, the diesel corridor or the railroad airlock.

3.6.1.2.2 Moderate Energy Fluid Systems

Each room, compartment, or area containing components essential for safe shutdown has been evaluated for the effects of postulated ruptures in moderate energy piping.

Safe shutdown components were evaluated for operability in the postulated water spray and/or flooding environment. Safe shutdown components in the reactor enclosure, the control structure, the diesel generator enclosure, the spray pond pumphouse, the main steam tunnel, and the RHRSW pipe tunnel were considered.

Those components not designed for operability in a water spray and/or a flooding environment were assumed to fail. Electrical components that are needed for safe shutdown, but that would fail under the water spray/flooding conditions, were protected from water spray and flooding conditions.

The synergistic effects of an independent single active failure, in addition to the effects of the pipe break, were evaluated and were found acceptable subject to any limitations discussed below. For some systems (such as RHR), a single active failure in the redundant loop of the same system has been excluded in accordance with Section 3.6.1.1.

The crack sizes postulated, and the nominal pipe sizes in which moderate energy pipe cracks are postulated to occur, are discussed in Section 3.6.2.1.3. Additional criteria used in the moderate energy fluid systems analysis are discussed in Section 3.6.1.1.

3.6.1.2.2.1 Primary Containment

All equipment within the primary containment that must operate during or after a LOCA is qualified for the appropriate environmental conditions, as described in Section 3.11. Wetting associated with the postulated failure of any moderate energy pipe is within the bounds of that qualification. Consequently, no item-by-item discussion of this less severe event is required to verify that capability of safely shutting down the plant.

3.6.1.2.2.2 Reactor Enclosure

Compartments or areas on each elevation that contain safe shutdown electrical equipment were evaluated. Piping and mechanical drawings for each elevation of the reactor enclosure are provided in Section 1.2. All safety-related equipment whose operability in a water spray or flooding environment must be ensured were either protected from the consequences of the pipe break or are designed to remain operable in that environment.

3.6.1.2.2.3 Control Structure

Compartments or areas that contain safety-related equipment on each elevation were evaluated beginning with el 200', the lowest elevation of the control structure. All safety-related equipment whose operability in a water spray or flooding environment must be ensured were either protected from the consequences of the pipe break or were designed to operate in that environment. Piping and mechanical drawings for each elevation of the control structure are provided in Section 1.2.

3.6.1.2.2.4 Diesel Generator Compartments

Moderate energy piping systems that are routed through the diesel generator compartments include:

a. ESW to the diesel generator heat exchangers

b. diesel fuel oil supply lines

c. diesel generator starting air

d. demineralized water from the jacket water expansion tank to the diesel generator

e. lube oil from the storage tank to the diesel generator

f. fire protection system (normally dry)

Each of the moderate energy piping systems is integral to the operation or protection of the diesel generator located in the same compartment as the piping. Thus, a failure of one or more of these systems would result in loss of only the one diesel. The safe shutdown capability of the plant would therefore not be compromised because only three of the four diesels are required in the event of a LOOP.

In addition to the diesel generators, a safeguard MCC is located within each diesel generator enclosure. The impact of a pipe break in the diesel generator enclosure on these MCCs is discussed in Section 3.6.1.2.1.18.

3.6.1.2.2.5 Spray Pond Pumphouse

The spray pond pumphouse contains two separate ESW and RHRSW pump areas, two RHRSW and ESW pipe-ways, two RHRSW valve compartments, and several other compartments.

Each pump-room contains two RHRSW pump motors and two ESW pump motors. The pumps are common to both Unit 1 and Unit 2.

During normal plant operation, the ESW and RHRSW piping is depressurized except during surveillance testing of ESW, RHRSW, and the diesel generators; and during operation for reactor shutdown cooling, suppression pool cooling, and spray pond cooling/chemistry control. The frequency of the surveillance tests is in accordance with the Technical Specifications, and the frequency of operation for other purposes varies with plant conditions.

A moderate energy line break in these lines during normal plant operation is unlikely. However, if a line break should occur, water spray could impinge on the pump motors in the room. In this event, up to two ESW and two RHRSW pumps could be disabled. The loss of the two ESW pumps is of no concern because shutdown cooling loads would be provided by the service water system. Because nonsafeguard heat removal systems are available, the RHRSW system is not critical for safe shutdown of the reactors.

The RHRSW valve compartments contain valves that are essential for the safe operation of the ESW system. However, as discussed above, shutdown cooling loads would be provided by the service water system because offsite power would be assumed to be available.

A pipe break in the wet pit, the ESW and RHRSW pipe-way, or the access hatch area would not impact safe shutdown capability because these compartments do not contain components that are essential for safe shutdown.

Spray pond pumphouse flooding is limited to the compartments adjoining the pump area. The pump area itself cannot flood because water would flow through open grating into the spray pond.

Flooding in the adjacent compartments could damage at most a single mechanical division. Because it is not necessary to postulate a single active failure in the redundant RHRSW system, and because flooding can damage only one mechanical division of RHRSW, safe shutdown capability will not be impaired.

3.6.1.2.2.6 Main Steam Tunnel

Safe shutdown components in the main steam tunnel include the outboard MSIVs.

The MSIVs are qualified to operate in environments more severe than that caused by moderate energy water spray. The valves are located well above the maximum water flood level. Therefore, the occurrence of a moderate energy pipe crack in this compartment would have no effect on safe shutdown capability.

3.6.1.2.2.7 RHRSW Pipe Tunnel

A moderate energy pipe break in the RHRSW pipe tunnel would not impact safe shutdown capability.

The only safe shutdown components on this elevation are two RHRSW valves. Each of these valves are qualified to be submersible and would withstand postulated water spray and flooding conditions.

3.6.1.3 Safety Evaluation

The analyses of postulated pipe ruptures summarized in Section 3.6.1.2 verify that the consequences of any single rupture of fluid system piping in the plant will not prevent safe shutdown of the reactor from being achieved.

The offsite radiological consequences of piping ruptures are enveloped by a reactor recirculation system break for all breaks inside primary containment, and by main steam system and feedwater system breaks for all breaks outside primary containment. The radiological consequences of these breaks are presented in Sections 15.6.5, 15.6.4, and 15.6.6, respectively.

Special consideration has been given to protecting the control room and other areas of the control structure containing essential systems and components from the effects of postulated pipe ruptures. Exterior walls of the control structure above the slab at el 217' are designed as steam-tight in those areas where the walls could be subject to steam pressurization resulting from rupture of high energy fluid system piping outside the control structure. HVAC ducts penetrating these portions of the control structure walls are equipped with back pressure dampers, and other types of penetrations through the walls are designed as steam-tight.

As described in Sections 3.6.1.2.1.2 and 3.6.1.2.1.3, those portions of the main steam and feedwater lines routed near the control structure are provided with pipe whip restraints and bumpers to prevent a postulated rupture of these lines from causing unacceptable damage to the control structure walls.

3.6.2 DETERMINATION OF PIPE FAILURE LOCATIONS AND DYNAMIC EFFECTS ASSOCIATED WITH POSTULATED PIPING FAILURES

Information concerning break and crack location criteria and methods of analysis is presented in this section. The location criteria and methods of analysis are needed to evaluate the dynamic effects associated with postulated ruptures of high energy and moderate energy piping inside and outside the primary containment.

3.6.2.1 Criteria Used to Determine Pipe Break and Crack Locations and Their Configurations

3.6.2.1.1 Break Locations in High Energy Fluid System Piping

3.6.2.1.1.1 Piping in Containment Penetration Areas

High energy pipes penetrating the primary containment are provided with moment-limiting restraints that are located reasonably close to the containment isolation valves and are designed to withstand the loadings resulting from a pipe break either inboard of the inboard isolation valve or outboard of the outboard isolation valve so that neither isolation valve operability nor leak-tight integrity of the containment penetration would be impaired as a result of such pipe breaks. Terminal ends of high energy piping that penetrate the containment are considered to originate beyond the containment isolation valve and its moment-limiting restraint, both inboard and outboard.

Breaks are not postulated in these portions of high energy piping in containment penetration areas provided that the following design stress and fatigue limits are satisfied:

For ASME Section III, Class 1 Piping

a. The stress intensity range S_n, calculated for normal and upset conditions by equation (10) of paragraph NB-3653, does not exceed 2.4 S_m, and the cumulative usage factor associated with normal, upset, and testing conditions is less than 0.1, or

b. The stress intensity range S_n, calculated for normal and upset conditions by equation (10) of paragraph NB-3653 exceeds 2.4 S_m but does not exceed 3.0 S_m and the cumulative usage factor associated with normal, upset, and testing conditions is less than 0.1, or

c. The stress intensity range S_n, calculated for normal and upset conditions by equation (10), exceeds 3.0 S_m, but the stress intensity ranges computed by equations (12) and (13) of paragraph NB-3653 are less than 2.4 S_m and the cumulative usage factor associated with normal, upset, and testing conditions is less than 0.1

d. Breaks are always postulated whenever the usage factor exceeds 0.1 regardless of stress.

e. The loading resulting from a postulated pipe break beyond these portions of the piping does not cause the stress as calculated by equation (9) in paragraph NB-3652 to exceed 2.25 S_m, except for the portion of piping between the isolation valve and the adjacent restraints protecting the operability of the valve. For this latter portion of piping, higher stresses are permitted provided that a plastic-hinge is not formed and the operability of the isolation valve is ensured.

After the as-built analysis is completed, the licensee will conduct a comparison against the SRP criteria (NUREG-0800) (BTP MEB 3-1) and demonstrate no additional breaks for the following situations:

a. The stress calculated by equation (10) is between 2.4 S_m and 3.0 S_m and

b. The ASME Code version of 1979 Summer Addenda or later is used for analysis.

For ASME Section III, Class 2 and 3 Piping

a. The maximum stress ranges as calculated by the sum of equations (9) and (10) in paragraph NC-3652, considering normal and upset plant conditions, does not exceed $0.8(1.2\ S_h + S_A)$.

b. The maximum stress, as calculated by equation (9) in paragraph NC-3652, under the loadings resulting from a postulated rupture of fluid system piping beyond these portions of piping does not exceed $1.8\ S_h$, except for the portion of the piping between the isolation valve and the adjacent restraints protecting the operability of the valve. For this latter portion of the piping, higher stress is permitted, provided that a plastic-hinge is not formed and the operability of the isolation valve is assured.

In addition to these stress and fatigue criteria, high energy piping in containment penetration areas must meet the following requirements:

a. Welded pipe support attachments are avoided to eliminate stress concentrations.

b. The number of circumferential and longitudinal pipe welds and branch connections is minimized.

c. The length of the piping run is minimized, consistent with requirements to keep stress levels low and provide access for inservice inspection.

d. The design at points of pipe fixity (such as pipe anchors or welded connections at containment penetrations) does not require welding directly to the outer surface of the piping (flued, integrally forged pipe fittings are acceptable), except where such welds are 100% volumetrically examinable in service and a detailed stress analysis is performed to demonstrate compliance with the limits of the stress and fatigue criteria stated above.

e. In accordance with the inservice inspection plan, the inservice examination completed during each inspection interval will provide either a 100% volumetric examination of circumferential pipe welds within these portions of piping or those examinations required by the Risk Informed Inservice Inspection (RISI) Program as applied to these portions of piping also known as the Break Exclusion Region (BER). Inservice inspection of the RCPB is discussed in Section 5.2.4.

f. For piping constructed in accordance with ANSI B31.1, all welds will be fully radiographed.

After the as-built analysis is completed, the licensee will conduct a comparison against the SRP criteria (NUREG-0800) (BTP MEB 3-1) and demonstrate no additional breaks for the following situations:

a. The stress calculated by equation (10) is between $2.4\ S_m$ and $3.0\ S_m$ and

b. The ASME Code version of 1979 Summer Addenda or later is used for analysis.

3.6.2.1.1.2 Recirculation System Piping

Pipe breaks in the Unit 1 recirculation system are postulated to occur at the following locations:

a. Terminal ends of a piping run or branch run.

b. At intermediate locations between terminal ends where the maximum stress range between any two load sets (including the zero load set), as calculated according to ASME Section III, subarticle NB-3600, for upset plant conditions and an independent OBE event, meets the following requirements:

  1. The stress range, as calculated using equation (12) or (13), exceeds 2.4 S_m and the cumulative usage factor associated with normal, upset, and testing conditions is less than 0.1.

  2. The stress range calculated using equation (10) exceeds 2.4 S_m but is less than 3.0 S_m, and the cumulative usage factor exceeds 0.1.

  3. The stress range calculated using equation (10) exceeds 3.0 S_m, and the cumulative usage factor exceeds 0.1.

  4. Breaks are always postulated whenever the usage factor exceeds 0.1 regardless of stress.

  5. If two or more intermediate break locations cannot be determined by stress or usage factor limits, two arbitrary intermediate break locations are selected on a reasonable basis. This basis includes consideration of fitting locations and/or highest stress or usage factor locations. Where more than two such intermediate locations are possible using the application of the above reasonable basis, those two locations possessing the greatest damage potential are used. A break at each end of a fitting can be classified as two discrete break locations when the stress analysis is sufficiently detailed to differentiate stresses at each postulated break.

After the as-built analysis is completed, the licensee will conduct a comparison against the SRP criteria (NUREG-0800) (BTP MEB 3-1) and demonstrate no additional breaks for the following situations:

a. The stress calculated by equation (10) is between 2.4 S_m and 3.0 S_m, and

b. The ASME Code version of 1979 Summer Addenda or later is used for analysis.

For Unit 2, break locations are similarly postulated except that the two arbitrary intermediate break locations of b.4 above need not be postulated provided the following criteria are met:

a. The piping system is adequately resistant to IGSCC and is not susceptible to unanticipated water hammer thermal transient events.

b. The piping system is included in the startup testing program for piping steady-state vibrations.

c. The elimination of the break location does not change equipment environmental qualification or structural design criteria.

3.6.2.1.1.3 Class 1 Piping (Other Than Recirculation System Piping and Piping in Containment Penetration Areas)

Breaks in Class 1 piping (ASME Section III) for Unit 1 are postulated to occur at the following locations:

a. At terminal ends of piping runs or branch runs.

b. At intermediate locations between terminal ends, as determined by one of the two following criteria:

1. At each location of potential high stress such as pipe fittings (elbows, tees, reducers, etc) valves, and welded attachments.

2. At each location where, for normal and upset load conditions, none of the following stress and fatigue limits are met:

(a) The stress intensity range S_n, calculated by equation (10) of paragraph NB-3653, does not exceed 2.4 S_m and the cumulative usage factor associated with normal, upset, and testing conditions is less than 0.1.

(b) The stress intensity range S_n, as calculated by equation (10) of paragraph NB-3653, exceeds 2.4 S_m but is less than 3.0 S_m, and the cumulative usage factor is less than 0.1.

(c) The stress intensity range S_n exceeds 3.0 S_m, but the stresses computed by equations (12) and (13) of paragraph NB-3653 are less than 2.4 S_m, and the cumulative usage factor is less than 0.1.

(d) Breaks are always postulated whenever the usage factor exceeds 0.1 regardless of stress.

3. When the above stress and fatigue criteria result in less than two intermediate break locations, a minimum of two locations are chosen based on highest stress, as calculated by equation (10) of paragraph NB-3653. The two locations are separated by a change of direction of the force resulting from pipe break. If the piping run has no more than one change of direction, a minimum of one intermediate break location is chosen. Any given fitting is considered as a single break location regardless of the number of breaks postulated at different locations on that fitting.

Intermediate pipe break locations are initially based on committed design piping stress calculations in accordance with the above criteria. As a result of piping reanalysis, the highest stress locations may be shifted. An initially determined pipe break location will not be changed as a consequence, however, unless one of the following conditions exists:

(a) Reanalysis shows that maximum stress range or cumulative usage factor at another location not only exceeds that for the initial pipe break location but also exceeds the above pipe break criteria. In addition, the break at the new location results in more serious consequences to safety-related systems than the initial break location.

(b) Significant changes are made in the routing, size, or wall thickness of the pipe after the initial pipe break determination.

4. After the as-built analysis is completed, the licensee will conduct a comparison against the SRP criteria (NUREG-0800) (BTP MEB 3-1) and demonstrate no additional breaks for the following situations:

(a) The stress calculated by equation (10) is between 2.4 S_m and 3.0 S_m and

(b) The ASME Code version of 1979 Summer Addenda or later is used for analysis.

For Unit 2, break locations are similarly postulated except that arbitrary intermediate breaks from b.3 above need not be postulated provided the following criteria are met:

a. The piping system is adequately resistant to IGSCC and is not susceptible to unanticipated water hammer thermal transient events.

b. The piping system is included in the startup testing program for piping steady-state vibrations.

c. The elimination of the break location does not change equipment environmental qualification or structural design criteria.

The following Unit 2 class 1 piping systems meet all of the above criteria:

- Main Steam Inside Containment
- Feedwater Inside Containment
- RWCU Inside Containment
- Reactor Vessel Drain
- HPCI Steam Supply Inside Containment

- RCIC Steam Supply Inside Containment
- SLCS
- RHR Shutdown Cooling Suction
- RHR Shutdown Cooling Return
- LPCI Injection
- Core Spray Injection

3.6.2.1.1.4 Class 2 and 3 Piping (Other Than Recirculation System Piping and Piping in Containment Penetration Areas)

Breaks in Class 2 and 3 piping (ASME Section III) for Unit 1 are postulated to occur at the following locations:

a. At terminal ends of piping runs or branch runs.

b. At intermediate locations between terminal ends, as determined by one of the three following criteria:

   1. At each location of potential high stress, such as pipe fittings (elbows, tees, reducers, etc), valves, and welded attachments.

   2. At each location where the maximum stress range, as calculated by the sum of equations (9) and (10) of paragraph NC-3652, considering normal and upset plant conditions, exceeds $0.8(1.2\ S_h + S_A)$.

   3. When the above stress and fatigue criteria result in less than two intermediate break locations, a minimum of two locations are chosen based on highest stress, as calculated by the sum of equations (9) and (10) of paragraph NC-3652. The two locations are separated by a change of direction of the force resulting from pipe break. If the piping run has no more than one change of direction, a minimum of one intermediate break location is chosen. Any given fitting is considered as a single break location regardless of the number of breaks postulated at different locations on that fitting.

      Intermediate pipe break locations are initially based on committed design piping stress calculations in accordance with the above criteria. As a result of piping reanalysis, the highest stress locations may be shifted. An initially determined pipe break location will not be changed as a consequence, however, unless one of the following conditions exists:

      (a) Reanalysis shows that maximum stress range or cumulative usage factor at another location not only exceeds that for the initial pipe break location but also exceeds the above pipe break criteria. In addition, the break at the new location results in more serious consequences to safety-related systems than the initial break location.

(b) Significant changes are made in the routing, size, or wall thickness of the pipe after the initial pipe break determination.

4. At each extreme of the piping run adjacent to basic protective structures, when the piping system contains no fitting, valve, or welded attachment.

For Unit 2, break locations are similarly postulated except that the arbitrary intermediate breaks from b.3 above need not be postulated provided the following criteria are met:

a. The piping system is adequately resistant to IGSCC and is not susceptible to unanticipated water hammer thermal transient events.

b. The piping system is included in the startup testing program for piping steady-state vibrations.

c. The elimination of the break location does not change equipment environmental qualification or structural design criteria.

The following Unit 2 Class 2, 3 piping systems meet all of the above criteria:

- RWCU Outside Containment
- HPCI Steam Supply Outside Containment
- RCIC Steam Supply Outside Containment
- Main Steam Outside Containment

3.6.2.1.1.5 Non-Nuclear Class Piping

Breaks in non-nuclear class piping are postulated to occur at the following locations:

a. At terminal ends of piping runs or branch runs.

b. At each intermediate location of potential high stress, such as pipe fittings (elbows, tees, reducers, etc), valves, and welded attachments.

Alternatively, the break locations for non-nuclear class piping can be selected according to the same criteria used for Class 2 and 3 piping, provided that all necessary analyses are made.

These breaks in nonseismic Category I piping have been postulated at those locations that would result in the maximum amount of damage and the safety-related systems and components have adequate protection from these piping breaks as discussed below.

The only high energy seismic Category II piping in the control structure is the portion of the steam supply line to the offgas recombiner preheater that is located within the recombiner compartments. The only high energy seismic Category II piping in the reactor enclosure is the RWCU piping inside the RWCU filter/demineralizer compartments and the RWCU holding pump compartments. In both cases, the walls of the compartments are capable of withstanding the pipe whip and jet impingement forces and the compartment pressurization that could result from breaks at the most adverse locations. Because there are no safety-related components located in these compartments, safety-related components will not be affected by high energy pipe breaks within these compartments.

The diesel generator enclosures have high energy seismic Category IIA piping in the form of heating steam supply lines (1½" JBD-330/430). A rupture of such a line could create maximum temperatures of less than 340°F and pressures less than 15 psia (0.3 psig) due to the large "bird grill" vent openings. As only one diesel generator enclosure is affected at any one time, the assumed loss of one diesel generator is acceptable.

Other safety-related structures, such as the spray pond pumphouse, do not contain high energy seismic Category II piping.

Safety-related components are protected from the effects of high energy pipe breaks in nonsafety-related structures by the walls that separate the safety-related structures from the nonsafety-related structures. To the extent necessary to prevent unacceptable damage to safety-related components, these walls are designed to withstand the pipe whip and jet impingement forces and compartment pressurization that could result from breaks at the most adverse locations in high energy seismic Category II piping within the nonsafety-related structures.

3.6.2.1.2 Crack Locations in Moderate Energy Fluid System Piping

Through-wall leakage cracks are postulated to occur in moderate energy piping located in areas containing essential systems and components. Cracks are postulated to occur at terminal ends of piping runs or branch runs, and at intermediate locations selected in accordance with either of the two following criteria:

a. At each location of potential high stress, such as pipe fittings (elbows, tees, reducers, etc), valves, and welded attachments

b. For Class 1 piping (ASME Section III), at locations where the maximum range of stress intensity as calculated by equation (10) of paragraph NB-3653 exceeds 1.2 S_m, and for Class 2 or 3 piping (ASME Section III) or non-nuclear piping, at locations where the maximum stress range as calculated by the sum of equations (9) and (10) of paragraph NC-3652 exceeds $0.4(1.2\ S_h + S_A)$.

The above criteria notwithstanding, cracks are not postulated in those portions of moderate energy piping located in the following areas:

a. Areas in which high energy pipe breaks are postulated, provided that moderate energy piping cracks would not result in more severe environmental conditions than the high energy pipe breaks.

b. Between containment isolation valves, provided that:

1. The piping meets the requirements of subarticle NE- 1120 of the ASME B&PV Code

2. The maximum range of stress intensity for Class 1 piping (ASME Section III) as calculated by equation (10) of paragraph NB-3653 does not exceed 1.2 S_m, and the maximum stress range for Class 2 and 3 (ASME Section III) or non-nuclear piping as calculated by the sum of equations (9) and (10) of paragraph NC-3652 does not exceed 0.4(1.2 S_h + S_A).

3.6.2.1.3 Types of Breaks and Cracks in Fluid System Piping

Circumferential Breaks

A circumferential break is assumed to result in (a) severance of a high energy pipe on a plane perpendicular to the pipe axis, and (b) separation amounting to at least a one diameter lateral displacement of the ruptured piping ends unless physically limited by piping restraints, structural members, or piping stiffness. Pipe whipping is assumed to occur in the plane defined by the piping geometry and configuration, and to cause pipe movement in the direction of the jet reaction.

All analyses to determine the blowdown (reaction) force on the segment of piping that contains a circumferential break are based on unobstructed discharge from 100% of the cross-sectional area of the pipe. This is consistent with the assumption of a one diameter lateral displacement of the ruptured piping ends. The only pipe break analyses that have involved lateral displacements of less than one pipe diameter are analyses concerning jet impingement forces. In certain cases where pipe whip restraints are located on both sides of a postulated circumferential break, and the design of the restraints prevents the two ends of the break from achieving a one diameter displacement, credit is taken for one end of the broken pipe causing partial blockage of the fluid being discharged from the opposite side of the break. Similarly, in certain cases where one side of the break is an RPV nozzle safe-end and the other side of the break is restrained from achieving a one diameter displacement relative to the nozzle, credit is taken for partial blockage of the fluid discharging from the restraint pipe end. This methodology can result in a reduction of the jet impingement force on potential impingement targets.

Circumferential breaks are postulated in high energy fluid system piping of nominal pipe size greater than 1-inch, at the locations determined by the criteria listed in Section 3.6.2.1.1, except where it can be shown that the maximum stress is in the circumferential direction and is at least 1.5 times the longitudinal stress, in which case only a longitudinal break is postulated.

Longitudinal Breaks

A longitudinal break is assumed to result in an axial split parallel to the pipe axis, without causing pipe severance. The break opening area is assumed to be equal to the effective cross- sectional flow area of the pipe at the break location. The split is assumed to be oriented (but not concurrently) at two diametrically opposed points on the piping circumference so that the jet reaction force causes out-of-plane bending of the piping configuration. Piping movement is assumed to occur in the direction of the jet reaction unless limited by piping restraints, structural members, or piping stiffness.

Longitudinal breaks are postulated in high energy fluid system piping of nominal pipe sizes of 4 inches and larger, at the locations determined by the criteria listed in Section 3.6.2.1.1, with the following exceptions. Longitudinal breaks are not postulated:

a. At terminal ends

b. At intermediate break locations chosen to satisfy the criterion for a minimum number of break locations

c. At locations where it can be shown that the maximum stress is in the longitudinal direction and is at least 1.5 times the circumferential stress, in which case only circumferential breaks need to be postulated.

Through-Wall Leakage Cracks

Through-wall leakage cracks are postulated to occur in moderate energy fluid system piping exceeding a nominal pipe size of 1 inch, at the locations determined by the criteria listed in Section 3.6.2.1.2. A crack is assumed to occur at any orientation about the circumference of a pipe. Fluid flow from a crack is based on a circular opening with an area equal to that of a rectangle one-half pipe diameter in length and one-half pipe wall thickness in width.

3.6.2.2 Analytical Models to Define Forcing Functions and Response Models (Recirculation System Only)

3.6.2.2.1 Analytical Methods to Define Blowdown Forcing Functions

The rupture of a pressurized pipe causes the flow characteristics of the system to change, creating reaction forces that can dynamically excite the piping system. The reaction forces are a function of time and space and depend upon the fluid state within the pipe prior to rupture, break flow area, frictional losses, plant system characteristics, piping system, and other factors. The methods used to calculate the reaction forces for recirculation system piping are presented below.

The criteria that are used for calculation of fluid blowdown forcing functions include:

a. The dynamic force of the jet discharge at the break location is based on the effective cross-sectional flow area of the pipe and on a calculated fluid pressure as modified by an analytically or experimentally determined thrust coefficient. Limited pipe displacement at the break location, line restrictions, flow limiters, positive pump-controlled flow, and the absence of energy reservoirs may be taken into account, as applicable, in the reduction of jet discharge.

b. All breaks are assumed to attain full pipe break area instantaneously, i.e., a rise time not exceeding one millisecond is used for the initial pulse.

Blowdown forcing functions are determined by either of two methods as described below.

Moody Model

The predicated blowdown forces on pipes fed by a pressure vessel can be described by transient and steady-state forcing functions. The forcing functions used are based on methods described in Reference 3.6-4. These are simply described as follows:

a. The transient forcing functions at points along the pipe result from the propagation of waves (wave thrust) along the pipe, and from the reaction force due to the momentum of the fluid leaving the end of the pipe (blowdown thrust).

b. The waves cause various sections of the pipe to be loaded with time-dependent forces. It is assumed that the pipe is one-dimensional, in that there is no attenuation or reflection of the pressure waves at bends, elbows, and the like. Following the rupture, a decompression wave is assumed to travel from the break at a speed equal to the local speed of sound within the fluid. Wave reflections occur at the break end, changes in direction of piping, and the pressure vessel until a steady flow condition is established. Vessel and free space conditions are used as boundary conditions. The blowdown thrust causes a reaction force perpendicular to the pipe break.

c. The initial blowdown force on the pipe is taken as the sum of the wave and blowdown thrusts and is equal to the vessel pressure (P_o) times the break area (A). After the initial decompression period (i.e., the time it takes for a wave to reach the first change in direction), the force is assumed to drop off to the value of the blowdown thrust (i.e., 0.7 P_oA).

d. Time histories of transient pressure, flow rate, and other thermodynamic properties of the fluid can be used to calculate the blowdown force on the pipe using the following equation:

$$F = \left[(P-P_a) + \frac{\rho u^2}{g_c}\right] A \qquad \text{(EQ. 3.6-1)}$$

where:

F	=	blowdown force
P_a	=	pressure at exit plane
P	=	ambient pressure
u	=	velocity at exit plant
ρ	=	density at exit plane
A	=	area of break
g_c	=	Newton's constant

e. Following the transient period, a steady-state period is assumed to exist. Steady-state blowdown forces are calculated including frictional effects. For saturated steam, these effects reduce the blowdown forces from the theoretical maximum of 1.26 P_oA. The method accounting for these effects is presented in Reference 3.6-4. For subcooled water, a reduction from the theoretical maximum of 2.0 P_oA is found through the use of Bernoulli's equation and standard equations such as Darcy's equation, which account for friction.

RELAP3

The computer code RELAP3 (Reference 3.6-5) is used to obtain exit plane thermodynamic states for postulated ruptures. Specifically, RELAP3 supplies exit pressure, specific volume, and mass rate. From these data the blowdown reaction load is calculated using the following relation:

$$\frac{T}{A_E} = P_E - P_{oo} + \frac{G_E{}^2\overline{V}_E}{G_c} \qquad \text{(EQ. 3.6-2)}$$

$$R = -\frac{T}{A_E} \cdot A_E \qquad \text{(EQ. 3.6-3)}$$

where:

$\frac{T}{A_E}$	=	thrust per unit break area (lb_f/ft^2)
P_E	=	exit pressure (lb_f/ft^2)
P_{oo}	=	receiver pressure (lb_f/ft^2)
G_E	=	exit mass flux ($lb_m/sec\text{-}ft^2$)
$\overline{V}_E$	=	exit specific volume (ft^3/lb_m)
g_c	=	Newton's constant (32.174 $ft\text{-}lb_m/lb_f\text{-}sec^2$)
R	=	reaction force on the pipe (lb_f)

3.6.2.2.2 Pipe Whip Dynamic Response Analyses

The prediction of time-dependent and steady-thrust reaction loads caused by blowdown of subcooled, saturated, and two-phase fluid from a ruptured pipe is used in design and evaluation of dynamic effects of pipe breaks. A detailed discussion of the analytical methods employed to compute these blowdown loads for recirculation system piping is given in Section 3.6.2.2.1. Analytical methods used to account for this loading are discussed below.

The criteria used for performing the pipe whip dynamic response analyses for recirculation system piping include:

a. A pipe whip analysis is performed for each postulated pipe break. However, a given analysis can be used for more than one postulated break location if the blowdown forcing function, piping and restraint system geometry, and piping and restraint system properties are conservative for other break locations.

b. The analysis includes the dynamic response of the pipe in question and the pipe whip restraints that transmit loading to the structures.

c. The analytical model adequately represents the mass/inertia and stiffness properties of the system.

d. Pipe whipping is assumed to occur in the plane defined by the piping geometry and configuration, and is assumed to cause pipe movement in the direction of the jet reaction.

e. Piping within the broken loop is no longer considered part of the RCPB. Plastic deformation in the pipe is considered as a potential energy absorber. Limits of strain are imposed that are similar to strain levels allowed in restraint plastic members. Piping systems are designed so that plastic instability does not occur in the pipe at the design dynamic and static loads, unless damage studies are performed to show that the consequences do not result in direct damage to any essential system or component.

f. Components such as vessel safe ends and valves that are attached to the broken piping system and do not serve a safety function, or whose failure would not further escalate the consequences of the accident, are not designed to meet limits imposed by the ASME B&PV Code for essential components under faulted loading.

The pipe whip analysis was performed using the PDA computer program (Reference 3.6-6). PDA is a computer program used to determine the response of a pipe subjected to the thrust-force occurring after a pipe break. The program treats the situation in terms of generic pipe break configuration, which involves a straight, uniform pipe fixed at one end and subjected to a time-dependent thrust-force at the other end. A typical restraint used to reduce the resulting deformation is also included at a location between the two ends. Nonlinear and time-independent stress-strain relations are used for the pipe and the restraint. A static nonlinear cantilever beam analysis is used for these locations to obtain the relationship between the pipe bending moment and the deflection (or rotation). Similar to the plastic-hinge concept, bending of the pipe is assumed to occur only at the fixed end and at the location supported by the restraint.

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

Considerable testing and analyses have demonstrated that potential rebound does not cause unacceptable increases in restraint deformation following the first quarter cycle loading for the GE restraint design and piping system experiencing blowdown thrust forces. Generic tests were performed on a 12-inch pipe size restraint with two primary loading configurations that represent the typical conditions during the postulated pipe rupture. Any other loading condition results in a combination of these two extremes. These loading configurations are:

a. Load applied perpendicular to the restraint frame base against the cable

b. Load applied parallel to the base against one side of the frame.

The mass/inertia and stiffness properties of the recirculation system in the pipe dynamic analysis (PDA model) are represented as described below. A generic representation of the pipe in any given analysis is shown in Figure 3.6-41. If the stiffness of the piping segment located between A and B is such that:

a. The slope of BD at B = 0, then in the analysis, the pipe is treated as built-in at B.

b. The slope of BD at B ≠ 0 (considerably different), then in the analysis, the pipe is considered to have a fixed, simple support (pinned end) at B.

To analyze the pipe with both ends supported (Figure 3.6-42) with the above computer model, two simplifications were made in the PDA program. First, an equivalent point mass is assumed at D instead of pipe length DE. The inertia characteristics of this mass rotating around point B are calculated to be identical to those of pipe length DE rotating around point E. Secondly, an equivalent resisting force is calculated for any deflection for the case of a built-in end from the bending moment-angular deflection relationship for pipe length DE. This equivalent force is subtracted from the applied thrust-force when calculating the net energy. The new model resulting from these simplifications is shown in Figure 3.6-42. The PDA computer program is further described in Section 3.9.1.2.2.6.

A comprehensive verification has been performed to demonstrate the conservatisms inherent in the PDA pipe whip computer program and the analytical methods utilized. This is described in Reference 3.6-7. Part of this verification program included an independent analysis of the recirculation system piping for the 1969 Standard Plant Design by Nuclear Services Corporation, under contract to GE. The recirculation system piping was chosen for study due to its complex piping arrangement and assorted pipe sizes. The analysis included elastic-plastic pipe properties, elastic-plastic restraint properties, and gaps between the restraint and pipe as documented in Reference 3.6-7. The piping/restraint system geometry and properties and fluid blowdown forces were the same in both analyses. However, a linear approximation was made by Nuclear Services Corporation for the restraint load-deflection curve supplied by GE. This approximation is demonstrated in Figure 3.6-36. The effect of this approximation is to give lower energy absorption of a given restraint deflection. Typically, this yields higher restraint deflections and lower restraint-to-structure loads than the GE analysis. The deflection limit used by Nuclear Services Corporation is the design deflection at one-half of the ultimate uniform strain for the GE restraint design. The restraint properties used for both analyses are provided in Table 3.6-24.

A comparison of the Nuclear Services Corporation analysis with the PDA analysis, as presented in Table 3.6-25 and Figure 3.6-37, shows that PDA predicts higher loads in 15 of the 18 restraints analyzed. This is due to the Nuclear Services Corporation model including energy-absorbing effects in secondary pipe elements and structural members. However, PDA predicts higher restraint deflections in 50% of the restraints. The higher deflections predicted by Nuclear Services Corporation for the lower loads are caused by the linear approximation used for the force-deflection curve rather than by differences in computer techniques. This comparison demonstrates that the simplified modeling system used in PDA is adequate for pipe rupture loading, restraint performance, and pipe movement predictions within the meaningful design requirements for these low probability postulated accidents.

3.6.2.3 Analytical Models to Define Forcing Functions and Response Models (Systems Other Than Recirculation System)

Analyses to determine the jet impingement effects and the piping and restraint displacements resulting from a pipe break are performed in accordance with BN-TOP-2 (Reference 3.6-8). Analysis of jet thrust forces are described in section 2.2 of BN-TOP-2. Fluid jet impingement forces are discussed in section 2.3 of BN-TOP-2. Impulsive loading and impact combined with impulsive loading are described in sections 3.2 and 3.3, respectively, of BN-TOP-2.

Alternatively, nonlinear time history dynamic analyses are performed. The forcing function used in piping dynamic analysis is obtained using either Reference 3.6-2 or 3.6-9. A typical forcing function and the piping system model used for the dynamic response analysis is provided in Figure 3.6-38. Pipe restraint rebound effects are also considered in this analysis.

A specific case for which time history dynamic analyses are used is the analysis of pipe breaks near containment isolation valves whose operability following the break must be ensured. For each break postulated to occur near a containment isolation valve in a high energy line, a dynamic analysis is performed using the appropriate pipe break forcing function in order to determine the stresses in the pipe and the loads on the moment-limiting restraints near the valve. The dynamic analysis verifies that the stress in the pipe at the isolation valve is maintained below the yield strength of material, in order to ensure valve operability. Since the section modulus of the valve is much greater than that of the pipe, the stress in the valve body is kept below the yield strength of the valve. Therefore, deformation in the valve body as a result of nearby pipe breaks is severely limited and remains in the elastic range so that binding of the valve internals cannot occur.

The criteria for dynamic analyses when used in verifying valve operability are as follows:

a. An analysis of the piping system is performed for the most severe of the postulated longitudinal and circumferential breaks at the break locations determined in accordance with the criteria of Section 3.6.2.1.

b. The loading condition of a piping system prior to a postulated rupture in terms of internal pressure, temperature, and stress state is that condition associated with reactor operation at 100% power.

   The basis of selecting 100% power as the loading condition of a piping system prior to rupture is as follows:

   1. Pipe rupture analysis state-of-the-art involves several conservative steps and assumptions in all phases of break design (e.g., probability of break, the postulated speed of break propagation, the structural material properties, and the structural stability characteristics of pipe break restraint structures).

   2. For those portions of piping systems that are normally pressurized during normal plant operation at power mode, the thermodynamic states in the piping systems are those of full (100%) thermal power.

3. There is a much higher probability for scheduled plant operation at 100% power (or less) than at higher ratings.

The combined effects of these considerations result in designs sufficiently capable of sustaining breaks at higher power levels.

c. For a circumferential break, pipe break dynamic transient force analyses are performed only for that end (or ends) of the pipe or branch that is connected to a contained fluid energy reservoir having sufficient capacity to develop a jet stream.

d. Dynamic analytic methods used for calculating the piping and piping/restraint system response to the pipe break forces adequately account for the effects of the following:

1. Translational masses (and rotational masses for major components) and stiffness properties of the piping system, restraint system, major components, and support walls

2. Transient forcing function(s) acting on the piping system

3. Elastic and inelastic deformation of piping and/or restraint

4. The design clearance between the pipe and the restraint

e. A 10% increase in minimum specified design yield strength (S_y) is used to account for strain rate effects in dynamic analyses.

The criteria for dynamic analyses when used in verifying the design of pipe whip restraints other than valve operability restraints are as follows:

a. A design analysis of the pipe whip restraint system is performed for each postulated longitudinal and circumferential break at the break locations determined in accordance with Section 3.6.2.1.

b. The loading condition of a piping system prior to a postulated break in terms of internal pressure, temperature, and stress state is that condition associated with reactor operation at 100% power.

c. For a circumferential break, pipe break dynamic transient forcing function calculations are performed only for that end (or ends) of the pipe or branch that is connected to a contained fluid energy reservoir having sufficient capacity to develop a jet stream.

d. An energy balance is established between the work done by the pipe break force on the pipe during the first quarter cycle of movement and the sum of the strain energy temporarily stored as elastic strain energy and that energy dissipated as plastic strain energy in members of the whip restraint.

e. In the event that the energy balance design analysis criteria are not satisfied, a dynamic analysis of the piping and piping restraint system is performed. This analysis is described by the same criteria described for valve operability dynamic analysis.

f. Dynamic analytical methods used for designing the pipe whip restraints adequately account for the effects of the following:

   1. Transient forcing functions acting on the piping system.

   2. Design clearance between the pipe and the restraint in the direction of pipe whip motion 3. Elastic stiffness and yield load capacity of the restraint under load applied in the direction of the pipe motion

   4. Impact energy-absorbing capacity of crushable material used in the restraint is equivalent to 100% of the available capacity

   5. Ductility capacity of the restraint loaded under impact beyond its yield load corresponds to not more than 50% of the ultimate uniform strain of the material.

3.6.2.4 Dynamic Analysis Methods to Verify Integrity and Operability (Recirculation System Only)

Pipe whip restraints, as differentiated from piping supports, are designed to function and carry load for an extremely low probability gross failure in a piping system carrying high energy fluid. The piping integrity does not usually depend on the piping whip restraints for any loading combination. When the piping integrity is lost because of a postulated break, the pipe whip restraint acts to limit the movement of the broken pipe to an acceptable distance. The pipe whip restraints (i.e., those devices that serve only to control the movement of a ruptured pipe following gross failure) will be subjected to once in a lifetime loading. For the purpose of design, the pipe break event is considered to be a faulted plant condition and the pipe, its restraints, and the structure to which the restraint is attached, are analyzed and designed accordingly.

As described in Section 3.6.1.2.1.1, the pipe whip restraints used for the recirculation system consist of straps (either carbon steel wire ropes or stainless steel bars) attached to a steel frame. The analytical methods used in the design of these restraints are similar to those used for the Fermi Unit 2 and Duane Arnold plants. They have, however, been improved by incorporation of the latest force-deflection data available for wire rope and by using the GE PDA code for the dynamic analysis. Load capacities for the restraint frames were developed by using the SAP code (a finite-element structural analysis program), and were confirmed by a test series using slowly applied loading methods to determine restraint load-deflection data in the tangential direction (parallel to the restraint base). The results of this test program are presented in Reference 3.6-10. The specific design objectives for the restraints are:

a. The restraints shall in no way increase the RCPB stresses by their presence during any normal mode of reactor operation or condition

b. The restraint system shall function to stop the movement of pipe failure (gross loss of piping integrity) without allowing damage to critical components or missile development

c. The restraints shall provide minimum hinderance to inservice inspection of the process piping.

For the purposes of design, the pipe whip restraints are designed for the following dynamic loads:

a. Blowdown thrust of the pipe section that impacts the restraint

b. Dynamic inertia loads of the moving pipe section that is accelerated by the blowdown thrust and subsequent impact on the restraint

c. Design characteristics of the pipe whip restraints are included and verified by the pipe whip dynamic analysis described in Section 3.6.2.2.2

d. Since the pipe whip restraints are not contacted during normal plant operation, the postulated pipe rupture event is the only design loading condition.

As previously described, the recirculation loop pipe whip restraints are composed of two parts, the straps and the restraint frame. Both parts of the restraining device function as load-carrying members, and will deflect under load. The load configurations for a restraint are shown in Figure 3.6-3. The components of the restraints are categorized as Type I and II, as described below:

a. Type I - radial load-carrying members: These members, consisting of cables or bars, will absorb energy when loaded in the direction perpendicular to the restraint base by elastic and plastic deformations (Figure 3.6-3).

b. Type II - tangential load-carry members: These members, consisting of restraint frames, will absorb energy when loaded in the direction parallel to the base by plastic deformation (Figure 3.6-3).

Each of these components is constructed of a different material in order to fulfill different design objectives. The design requirements and design limits for each component are therefore different. They are specified as below:

a. Type I - Straps

  1. For carbon steel wire ropes, the maximum acceptable load is 90% of the load-carrying capacity of the cable in the restraint configuration. This limit takes into consideration the efficiency reduction experienced when a cable is wrapped around a pipe. This means that the design load is limited to about 75% of the minimum certified load-carrying capacity of the cable in tension.

  2. For stainless steel bars, the design limit base was 50% of the minimum uniform ultimate tensile elongation.

b. Type 2 - Restraint Frames

Design limits for the ASTM A36 restraint frames are as follows:

1. Design Load

The load-bearing member is primarily a cantilever beam with an extra support (the diagonal plate) at approximately mid-span. At loads approaching the plastic moment capability of the beam, the plastic-hinge forms at the section determined by an elastic structural analysis. The maximum design load and the ultimate load are calculated based on plastic moment capability of this section, with the diagonal plate stressed uniformly at the minimum ultimate stress of 58,000 psi as specified for ASTM A36 material.

2. Design Deflection

The design and ultimate deflection are calculated assuming the beam remains straight and rotates about a point on the upper surface of the beam. The maximum design deflection at the load point is calculated assuming the diagonal plate undergoes 10% elongation. This corresponds to 50% of the minimum ultimate elongation of 20%, as specified for ASTM A36 material. The ultimate deflection of the beam is based on 20% ultimate elongation of the diagonal plate.

3.6.2.5 Dynamic Analysis Methods to Verify Integrity and Operability (Systems Other Than Recirculation System

The pipe whip restraints provided for protection from high energy pipe breaks are of two basic types: independent restraints and operability restraints. Independent restraints are provided solely to protect nearby structures and equipment from damage due to whipping pipes, and are designed so that a gap is maintained between the pipe and the restraint during normal plant conditions. Operability restraints are provided near primary containment isolation valves whose operability is required following a break of the pipe in which they are installed. These operability restraints are designed to limit the stress in the piping near the valve to below the yield strength of the material in order to ensure operability of the valve. To accomplish this function, it is necessary to minimize the gap between the pipe and the restraint so that contact will occur during normal plant conditions.

The following high energy piping systems in the drywell are not provided with pipe whip restraints:

a. Reactor vessel drain line (4" DCA-101)

b. Main steam drain lines (2" and 3" DBA-105)

c. RPV head vent line (2" DBA-108)

d. SLCS injection line (2" DCA-112)

Each of these lines has been evaluated, using the guidance of Regulatory Guide 1.46, to verify that in the event of a pipe break, damage to structures, systems, or components needed for safe shutdown would not occur. Therefore, the ability to shut the reactor down safely is maintained if a break should occur in any of these lines.

3.6.2.5.1 Design Loading Combinations

The design loading combinations applied in the design of pipe whip restraints are categorized with respect to the plant operating conditions which are identified as normal, upset, emergency, and faulted as described in Section 3.9.3.1.1. Pipe break is considered as a faulted plant condition.

3.6.2.5.2 Design Stress Limits

Operability Restraints

When restraints for piping are designed so that contact between pipe and restraint will occur during normal plant conditions, the design loading combinations for normal, upset, emergency, and faulted conditions are applicable. In evaluating the supports and restraints for ASME Section III Class 1, 2, and 3, the design stress limits applied in evaluating loading combinations for normal, upset, emergency, and faulted (except for pipe rupture) conditions are those given in Tables 3.9-12 and 3.9-16. After rupture of the supported pipe occurs, the piping system is no longer within the jurisdiction of ASME Section III because the pressure boundary has been breached. The restraints are evaluated for pipe rupture loads as described in Section 3.6.2.3.

Pipe restraints are included on main steam, feedwater, RWCU, HPCI, and RCIC system to ensure the operability of the containment isolation valves during a postulated break event that touch the pipe during normal operation. The restraints are modeled in the thermal and dynamic analysis as active (1/16 inch or less gaps) during all loading conditions.

The operability of the isolation valves protected by operability restraints is assured by limiting the pipe break dynamic stress in the adjacent pipe. Stresses at the junction of this component with the pipe are limited to the dynamic yield strength of the pipe material (1.1 S_y). Between the containment penetration inboard/outboard isolation valves, pipe dynamic stress is limited to be less than 2.25 S_m.

Independent Restraints

When restraints are designed solely to control movement following a postulated pipe rupture and to function independently of the normal support system, only the design pipe rupture loads are applicable.

To ensure that restraints function independently of the normal support system, the motions of the intact pipe due to all normal and upset plant conditions and the vibratory motion of the SSE are calculated and used to specify a minimum clearance between the pipe and the restraint. Wherever possible, gaps between pipes and restraints are maximized to avoid possible contact during plant operation. Where a particular location requires minimizing a gap, special features are provided to permit adjustment of the gap size during hot functional testing.

Independent restraints are evaluated for the pipe rupture loads as described in Section 3.6.2.3.

3.6.2.6 Guard Pipe Assembly Design Criteria

Guard pipe assemblies are not used in this plant.

3.6.3 DEFINITIONS

Certain terms used in Sections 3.6.1 and 3.6.2 have specified meanings as described below:

Essential Systems and Components - Systems and components required to shut down the reactor and mitigate the consequences of a postulated piping failure, without offsite power.

High Energy Fluid Systems - Fluid systems that, during normal plant conditions, are either in operation or maintained pressurized under conditions where either or both of the following are met:

a. Maximum operating temperature exceeds 200°F

b. Maximum operating pressure exceeds 275 psig

Moderate Energy Fluid Systems - Fluid systems that, during normal plant conditions, are either in operation or maintained pressurized (above atmospheric pressure) under conditions where both of the following are met:

a. Maximum operating temperature is 200°F or less

b. Maximum operating pressure is 275 psig or less

A system that qualifies as a high energy fluid system for only short periods and qualifies as moderate energy fluid system for the majority of the time is classified as a moderate energy fluid system provided that the total time the system operates within high energy pressure/temperature conditions is less than either of the following:

a. 2% of the time that the system operates as a moderate energy fluid system

b. 1% of the normal operating life span of the plant

Normal Plant Conditions - Plant operating conditions during reactor startup, operation at power, hot standby, or reactor cooldown to cold shutdown condition.

Upset Plant Conditions - Plant operating conditions during system transients that may occur with moderate frequency during plant service life and are anticipated operational occurrences, but not during system testing.

S_h and S_A - Allowable stresses at maximum (hot) temperature and allowable stress range for thermal expansion, respectively, as defined in ASME Section III, Article NC-3600.

S_m - Design stress intensity as defined in ASME Section III, Article NB-3600.

S_n - Primary plus secondary stress intensity range for normal and upset conditions as defined in ASME Section III, paragraph NB-3653.

Single Active Component Failure - Malfunction or loss of function of a component of electrical or fluid systems. The failure of an active component of a fluid system is considered to be a loss of component function as a result of mechanical, hydraulic, pneumatic, or electrical malfunction, but not the loss of component structural integrity. The direct consequences of a single active component failure are considered to be part of the single failure.

Terminal Ends - Extremities of piping runs that connect to structures, components (e.g., vessels, pumps, valves), or pipe anchors that act as rigid constraints to piping thermal expansion. A branch connection to a main piping run is a terminal end of the branch run, except when all three of the following conditions are in effect:

a. The nominal size of the branch run is at least half that of the main run

b. The intersection is not rigidly constrained to the building structure

c. The branch run and main run are included together in the same piping stress analysis model

For piping in containment penetration areas, terminal ends are selected at points located immediately beyond the required moment-limiting restraints inside and outside containment. In piping runs which are maintained pressurized during normal plant conditions for only a portion of the run (i.e., up to the first normally closed valve), a terminal end of such runs is the piping connection to this closed valve.

3.6.4 REFERENCES

3.6-1 "Subcompartment Pressure Analyses", BN-TOP-4, Rev. 0, Bechtel Power Corporation, San Francisco, California, (July 1976).

3.6-2 RELAP4/MOD5, "Computer Program for Transient Thermal-Hydraulic Analysis of Nuclear Reactors and Related Systems," ANCR-NUREG-1335, (September 1976).

3.6-3 FLUD, "Thermofluid Dynamics for a System of Interconnected Compartments," NE017, version 3, (July 18, 1980).

3.6-4 "System Criteria and Applications for Protection Against the Dynamic Effects of Pipe Break", GE Specification No. 22A2625.

3.6-5 RELAP3, "A Computer Program for Reactor Blowdown Analysis", IN-1321, Reactor Technology TID-4500, (June 1970).

3.6-6 "PDA - Pipe Dynamic Analysis Program for Pipe Rupture Movement" (proprietary filing), GE Report NEDE-10813.

3.6-7 "Final Report Pipe Rupture Analysis of Recirculation System for 1969 Standard Plant Design", Nuclear Services Corporation Report No. GEN-02-02.

3.6-8 "Design for Pipe Break Effects", BN-TOP-2, Rev. 2, Bechtel Power Corporation, San Francisco, California, (May 1974).

3.6-9 F.J. Moody, "Fluid Reaction and Impingement Loads", ASCE Specialty Conference on Structural Design of Nuclear Plant Facilities, Vol 1, pp. 219-262, (December 1973).

3.6-10 "Recirculation System Pipe Whip Restraint for the BWR 4, 218 and 251, Mark I and Mark II Product Line Plant", GE Design.

3.6-11 "Limerick Atomic Power Stations Units 1 & 2 Power Rerate Evaluation of Main Steam & Recirculation Piping System, "GE Report GE-NE-123-E014-0193, Rev. 0 (July 1993) (part of N-00E-177-00003).

3.6-12 PCFLUD, "Thermofluid Dynamics for a System of Interconnected Compartments," MAP-120, Version 4.0 (July 15, 1992).

3.6-13 CONCOIL-FLUD (CFLUD), "Thermofluid Dynamics for a System of Interconnected Compartments," Version 1.0 (December 1, 1993).

3.6-14 GEH 0000-0158-9651-NP, Revision 0, October 2013, LGS Main Steam Isolation Valve Response Time Testing Analysis (SDOC G-080-VC-00489).

Table 3.6-1

HIGH ENERGY FLUID SYSTEM PIPING

FLUID SYSTEM	EXTENT OF HIGH ENERGY PIPING
Reactor recirculation	From reactor vessel suction nozzle to recirculation pump to reactor vessel discharge nozzles (Drawing M-43)
Main steam	From reactor vessel nozzles to turbine stop valves (Drawings M-01 and M-41)
Feedwater	From condensate filter/demineralizers through feedwater heaters and feedwater pumps to reactor vessel nozzles (Drawings M-06 and M-41)
Condensate	From condensate pump discharge through steam jet air ejector condenser, steam packing exhauster, and condensate filter/ demineralizers (Drawings M-05 and M-16)
RWCU	From shutdown cooling suction line through RWCU pumps, regenerative and non-regenerative heat exchangers, and cleanup filter/demineralizers to feedwater lines (Drawings M-43, M-44 and M-45)
Reactor vessel drain	From reactor bottom head nozzle to RWCU line inside primary containment (Drawings M-43 and M-44)
HPCI steam supply	From main steam line C to HPCI turbine steam supply valve (Drawing M-55)
RCIC steam supply	From main steam line B to RCIC turbine steam supply valve (Drawings M-49 and M-50)
Main steam drain line	From main steam lines inside drywell to inboard containment isolation valve; from main steam lines outside drywell to outboard containment isolation valve and drain line isolation valves (Drawing M-41)

Table 3.6-1 (Cont'd)

RPV head vent line	From reactor vessel head nozzle to main steam line C (Drawing M-41)
SLCS	From reactor vessel nozzle to first upstream check valve (Drawing M-42 and M-48)
RHR shutdown cooling suction	From reactor recirculation loop to inboard containment isolation valve (Drawing M-51)
RHR shutdown cooling return	From reactor recirculation loop to first upstream check valve (Drawing M-51)
LPCI injection	From reactor vessel nozzle to first upstream check valve (Drawing M-51)
Core spray injection	From reactor vessel nozzle to first upstream check valve (Drawing M-52)
CRD hydraulic	From drive water pump to master control station to HCUs (Drawing M-46)
Auxiliary steam	From auxiliary boiler to various steam-consuming components (Drawing M-21)
Plant heating steam	From auxiliary boiler to various steam-consuming components

Table 3.6-2

RECIRCULATION PIPING SYSTEM
STRESS LEVELS AND PIPE BREAK DATA (5)(6)

(UNIT 1)

Node Point(1)	STRESS RATIOS PER ASME CODE EQUATIONS Eq.(10) $3\ S_m$ (2)	Eq.(12) $3\ S_m$ (2)	Eq.(13) $3\ S_m$ (2)	Usage Factor	Break Type(3)	Basis for Break Selection(4)
A. RECIRCULATION LOOP A						
001	0.630	0.079	0.606	0.00	C	TE
500	1.225	0.447	0.847	0.06	L	MBL
220	1.209	0.136	0.957	0.05	C & L	MBL
236	0.734	0.088	0.646	0.00	C	TE
200	1.39	0.098	0.993	0.15	C & L	MBL
216	0.821	0.199	0.677	0.00	C	TE
165	1.849	0.198	0.831	0.12	C & L	MBL
296	0.786	0.099	0.698	0.00	C	TE
240	1.532	0.312	0.948	0.29	C & L	MBL
256	0.745	0.152	0.635	0.00	C	TE
260	1.222	0.265	0.784	0.05	C & L	MBL
276	0.727	0.218	0.560	0.00	C	TE
B. RECIRCULATION LOOP B						
001	0.575	0.069	0.512	0.00	C	TE
16	1.590	0.281	0.741	0.16	L	MBL
800	1.179	0.401	0.826	0.04	L	MBL
220	1.175	0.112	0.945	0.04	C & L	MBL
236	0.708	0.074	0.642	0.00	C	TE
200	1.355	0.077	1.006	0.13	C & L	MBL
216	0.792	0.209	0.642	0.00	C	TE
165	1.838	0.203	0.832	0.11	C & L	MBL
296	0.817	0.093	0.739	0.00	C	TE
240	1.499	0.326	0.894	0.25	C & L	MBL
256	0.771	0.176	0.648	0.00	C	TE
260	1.235	0.274	0.799	0.06	C & L	MBL
276	0.793	0.223	0.585	0.00	C	TE

Table 3.6-2 (Cont'd)

RECIRCULATION PIPING SYSTEM
STRESS LEVELS AND PIPE BREAK DATA
(UNIT 2)

Node Point[1]	STRESS RATIOS PER ASME CODE EQUATIONS Eq.(10) 3 S_m [2]	Eq.(12) 3 S_m [2]	Eq.(13) 3 S_m [2]	Usage Factor	Break Type[3]	Basis for Break Selection[4]
A. RECIRCULATION LOOP A						
001	0.40	0.08	0.37	0.00	C	TE
236	0.79	0.40	0.40	0.00	C	TE
216	0.79	0.37	0.44	0.00	C	TE
296	0.94	0.51	0.45	0.00	C	TE
256	1.18	0.79	0.43	0.02	C	TE
276	0.73	0.34	0.40	0.00	C	TE
B. RECIRCULATION LOOP B						
001	0.43	0.06	0.41	0.00	C	TE
236	0.57	0.07	0.52	0.00	C	TE
216	0.72	0.19	0.54	0.00	C	TE
296	0.63	0.09	0.57	0.00	C	TE
256	0.70	0.16	0.54	0.00	C	TE
276	0.70	0.20	0.53	0.00	C	TE

(1) Locations of the nodes are shown in Figure 3.6-4
(2) S_m: Design stress intensity as defined in ASME Section III, Article NB-3600 (Section 3.6.3)
(3) Break types are indicated as follows:
C - Circumferential
L - Longitudinal
(4) Symbols used to denote the basis for break selection are as follows:
TE - Terminal end
MBL - Intermediate break locations selected to satisfy the requirement for a minimum number of break locations.
(5) The recirculation piping design basis has been evaluated for the effects of power rerate and demonstrated to be adequate for the increases in pressure, temperature and flow due to power rerate. Detail of the evaluation performed are documented in Reference 3.6-11.
(6) The information posted in this table was used for the original System piping analysis. Refer to System ASME III piping analysis for current information.

Table 3.6-3

REACTOR RECIRCULATION PIPING SYSTEM
FLUID BLOWDOWN THRUST TIME HISTORIES

(UNIT 1)

(Refer to Diagram A below)

Node Point of Break Location[1]	Break Type	Side of Break	Initial Force F_o (kips)	Intermediate Force F_{int} (kips)	Steady-State Force F_{ss} (kips)	Time Duration of $F_o(T_1)$ sec)	Time to Reach Steady-State (T_2) (sec)
001	CRCMF		540.00	476.80	179.80	0.00186	0.08006
16 (loop B only)	LONG		540.00	564.30	621.54	0.00064	0.0032
220	CRCMF	PUMP	373.48	291.98	330.53	0.00145	0.0210
236	CRCMF	PUMP	134.42	101.48	118.96	0.00143	0.02098
200	CRCMF	PUMP	373.48	281.78	330.53	0.00145	0.021
216	CRCMF	PUMP	134.42	101.48	118.96	0.00143	0.02098
296	CRCMF	PUMP	134.42	101.48	118.96	0.00143	0.02098
240	CRCMF	PUMP	373.48	281.98	330.53	0.00145	0.021
256	CRCMF	PUMP	134.42	101.48	118.96	0.00143	0.02098
260	CRCMF	PUMP	373.48	281.98	330.53	0.00145	0.021
276	CRCMF	PUMP	134.42	101.48	118.96	0.00143	0.02098
220	CRCMF	VESSEL	373.48	373.48	373.48	0.0038	0.0368
200	CRCMF	VESSEL	373.48	373.48	373.48	0.0038	0.0368
165	CRCMF	VESSEL	373.48	373.48	373.48	0.0038	0.0368
240	CRCMF	VESSEL	373.48	373.48	373.48	0.0038	0.0368
260	CRCMF	VESSEL	373.48	373.48	373.48	0.0038	0.0368

Table 3.6-3 (Cont'd)

REACTOR RECIRCULATION PIPING SYSTEM
FLUID BLOWDOWN THRUST TIME HISTORIES

(UNIT 2)

(Refer to Diagram A below)

Node Point of Break Location(1)	Break Type	Side of Break	Initial Force F_o (kips)	Intermediate Force F_{int} (kips)	Steady-State Force F_{ss} (kips)	Time Duration sec) of $F_o(T_1)$	Time to Reach (sec) Steady-State (T_2)
001	C		540.00	476.80	179.80	0.00186	0.08006
236	C	PUMP	134.42	101.48	118.96	0.00143	0.02098
216	C	PUMP	134.42	101.48	118.96	0.00143	0.02098
296	C	PUMP	134.42	101.48	118.96	0.00143	0.02098
256	C	PUMP	134.42	101.48	118.96	0.00143	0.02098
276	C	PUMP	134.42	101.48	118.96	0.00143	0.02098

(1) SEE FIGURE 3.6-4 FOR IDENTIFICATION OF POSTULATED BREAK LOCATIONS.

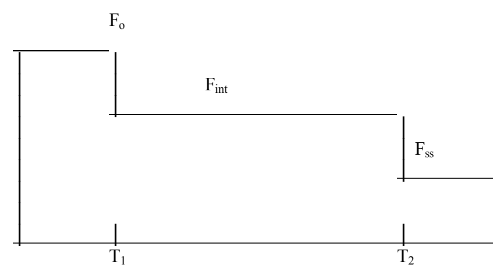

Diagram A

Table 3.6-4

MAIN STEAM PIPING STRESS LEVELS AND PIPE BREAK DATA (7)(8)

(UNIT 1)

Node Point(1)	Node Type(2)	Stress By Eq. 10 Usage (ksi)(6)	Cumulative Stress Limit Factor	Pipe Break Break 2.4 S_m(ksi)	Type(3)	Basis for Break Selection(4)
LINE A						
1	BW	25.94	0.0	42.21	C	TE
3	CURB	51.45	0.14	42.21	C&L	SFL
3	CURE	50.18	0.20	42.21	C&L	SFL
5	SWE	(5)	(5)	42.21	C	TE (TYP for 3 PSVs)
6	FLA	(5)	(5)	42.21	C	TE (TYP for 3 PSVs)
24	CUR	39.03	0.01	42.21	C	MBL
40	BW	34.72	0.01	42.21	C	TE
LINE B						
1	BW	24.81	0.0	42.21	C	TE
3	CURB	48.56	0.11	42.21	C&L	SFL
3	CURE	45.87	0.19	42.21	C&L	SFL
5	SWE	(5)	(5)	42.21	C	TE (TYP for 4 PSVs)
6	FLA	(5)	(5)	42.21	C	TE (TYP for 4 PSVs)
23	CUR	36.19	0.01	42.21	C	MBL
46	BW	33.09	0.01	42.21	C	TE
LINE C						
2	BW	24.03	0.0	42.21	C	TE
3	CURB	46.48	0.10	42.21	C&L	SFL
4	CURE	44.56	0.17	42.21	C&L	SFL
5	SWE	(5)	(5)	42.21	C	TE (TYP for 3 PSVs)
6	FLA	(5)	(5)	42.21	C	TE (TYP for 3 PSVs)
26	CUR	36.73	0.01	42.21	C	MBL
48	BW	34.07	0.01	42.21	C	TE

Table 3.6-4 (Cont'd)

(UNIT 1)

Node Point[1]	Node Type[2]	Stress By Eq. 10 (ksi)[6]	Cumulative Usage Factor	Pipe Break Stress Limit 2.4 S_m(ksi)	Break Type[3]	Basis for Break Selection[4]
LINE D						
1	BW	33.21	0.01	42.21	C	TE
3	CURB	50.28	0.11	42.21	C&L	SFL
3	CURE	48.04	0.13	42.21	C&L	SFL
5	SWE	[5]	[5]	42.21	C	TE (TYP for 4 PSVs)
6	FLA	[5]	[5]	42.21	C	TE (TYP for 4 PSVs)
31	CUR	39.09	0.01	42.21	C	MBL
47	BW	34.13	0.01	42.21	C	TE

Table 3.6-4 (Cont'd)

MAIN STEAM PIPE STRESS LEVELS AND PIPE BREAK DATA

(UNIT 2)

Node Point[1]	Node Type[2]	Stress By Eq. 10 (ksi)[6]	Cumulative Usage Factor	Pipe Break Stress Limit 2.4 S_m(ksi)	Break Type[3]	Basis for Break Selection[4]
LINE A						
1	BW	32.44	0.007	42.48	C	TE
5	SWE	41.77	0.251	42.48	C	TE (TYP for 3 PSVs)
6	FLA	[5]	[5]	42.48	C	TE (TYP for 3 PSVs)
40	BW	32.30	0.008	42.48	C	TE
LINE B						
1	BW	32.00	0.007	42.48	C	TE
5	SWE	37.58	0.218	42.48	C	TE (TYP for 4 PSVs)
6	FLA	[5]	[5]	42.48	C	TE (TYPE for 4 PSVs)
46	BW	32.60	0.009	42.48	C	TE
LINE C						
2	BW	29.77	0.007	42.48	C	TE
5	SWE	41.65	0.294	42.48	C	TE (TYP for 3 PSVs)
6	FLA	[5]	[5]	42.48	C	TE (TYP for 3 PSVs)
48	BW	33.78	0.010	42.48	C	TE
LINE D						
1	BW	29.95	0.006	42.48	C	TE
5	SWE	40.32	0.126	42.48	C	TE (TYP for 4 PSVs)
6	FLA	[5]	[5]	42.48	C	TE (TYP for 4 PSVs)
47	BW	32.18	0.007	42.48	C	TE

Table 3.6-4 (Cont'd)

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(1) Locations of nodes are shown in Figure 3.6-8.

(2) Node types are designated as follows:

BW - Butt-welding tee
CUR - Elbow
CURB - Elbow beginning
CURE - Elbow ending
SWE - Sweepolet
FLA - Flange

(3) Break types are indicated as follows:

C - Circumferential
L - Longitudinal

(4) Symbols used to denote the basis for break selection are as follows:

TE - Terminal end
SFL - The stress and fatigue limits established in Section 3.6.2.1.1.3 are not met.
MBL - Intermediate break location selected to satisfy the criteria for a minimum number of break locations.

(5) Stress values not calculated; terminal end break assumed.

(6) Refer to Section 3.6.2.1.1.3.

(7) The main steam piping design basis has been evaluated for the effects of power rerate and demonstrated to be adequate for the increase in pressure, temperature and flow due to power rerate. Details of the evaluation performed are documented in Reference 3.6-11.

(8) The information posted in this table was used for original System piping analysis. Refer to System ASME III piping analysis for current information.

Table 3.6-5

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Table 3.6-6

BLOWDOWN DATA FOR HIGH ENERGY PIPE BREAKS OUTSIDE PRIMARY CONTAINMENT

HIGH ENERGY LINE	BLOWDOWN (2) TIME AFTER BREAK (sec)	MASS FLOW RATE (lb/sec)	ENTHALPY (Btu/lb)(3)	ISOLATION VALVES	ISOLATION VALVE CLOSURE VALVE CLOSING TIME (sec)	SIGNAL DELAY TIME (sec)	TOTAL INTERVAL (sec)
Main steam line (26 inch EBB-101, EBB-102, EBB-103, or EBB-104)	0.00	13,356	1192.4	HV41-F022A,B,C&D	5.0	1.0	6.0
	0.076	8,543	1192.4	HV41-F028A,B,C&D	5.0	1.0	6.0
	0.16	9,177	1192.4				
	1.0	15,972	1192.4				
	1.001	19,398	595.0				
	5.0	19,398	595.0				
	6.0	0	595.0				
RWCU suction line (6 inch DCC-103)	0.0	2,820	525.3	HV44-F001	10.0 max.	2.0	12.0 max.
	0.1	2,319	524.7	HV44-F004	10.0 max.	2.0	12.0 max.
	0.2	1,818	513.0	HV44-F039(1)	-	-	-
	0.3	1,486	509.8				
	0.4	1,200	506.3				
	0.5	1,027	503.7				
	0.6	905	501.3				
	0.7	791	497.4				
	0.8	723	495.0				
	0.9	701	491.0				
	1.0	638	485.1				
	1.26	563	485.0				
	6.0	563	485.0				
	9.0	0	485.0				
RWCU pump discharge line * (4 inch DCC-101)	0.00	1410 *	526.4	HV44-F001	10.0 max.	2.0	12.0 max.
	0.02	912 *	526.4	HV44-F004	10.0 max.	2.0	12.0 max.
	0.03	623 *	526.4	HV44-F039(1)	-	-	-
	0.11	473 *	526.4				
	0.25	688 *	526.4				
	0.37	583 *	526.4				
	2.00	531 *	526.4				
	15.00	273 *	526.4				

* Mass flow rate scale from values for 3” break to 4” break by using the ratio of the break areas as a scaling factor (A4”/A3”) = 1.74.

Table 3.6-6 (Cont'd)

HIGH ENERGY LINE	BLOWDOWN [2] TIME AFTER BREAK (sec)	MASS FLOW RATE (lb/sec)	ENTHALPY (Btu/lb)[3]	ISOLATION VALVES	ISOLATION VALVE CLOSURE VALVE CLOSING TIME (sec)	SIGNAL DELAY TIME (sec)	TOTAL INTERVAL (sec)
RWCU pump discharge line	0.0	1,164	526.4	HV44-F001	10.0 max.	2.0	12.0 max.
at inlet to regenerative	0.2	947	526.4	HV44-F004	10.0 max.	2.0	12.0 max.
heat exchanger	0.3	904	526.4	HV44-F039[1]	-	-	-
(4 inch DCC-101)	0.35	1,051	526.4				
	1.0	682	526.4				
	1.5	503	526.4				
	2.0	471	526.4				
	16.0	223	526.4				
RWCU pump discharge line	0.00	1,164	203.9	HV44-F001	10.0 max.	2.0	12.0 max.
at inlet to nonregenerative	0.34	1,056	203.9	HV44-F004	10.0 max.	2.0	12.0 max.
heat exchanger(4 inch DCC-102)	0.59	868	203.9	HV44-F039[1]	-	-	-
	0.59	868	125.2				
	1.00	677	131.4				
	3.72	464	149.4				
	3.72	464	316.2				
	16.00	464	316.2				
	16.00	224	91.0				
	20.00	224	91.0				
HPCI steam supply line	0.0	1,470	1192.4	HV55-F002	12.0	1.0	13.0
at turbine inlet valve	0.24	1,045	1192.4	HV55-F003	12.0	1.0	13.0
(12 inch EBB-108)	0.36	280	1192.4				
	13.0	280	1192.4				
	14.0	0	1192.4				
HPCI steam supply line	0.0	2,940	1192.4	HV55-F002	12.0	1.0	13.0
in piping area	0.11	1,958	1192.4	HV55-F003	12.0	1.0	13.0
(12 inch EBB-108)	0.14	1,594	1192.4				
	0.22	266	1192.4				
	13.0	266	1192.4				
	14.0	0	1192.4				

Table 3.6-6 (Cont'd)

HIGH ENERGY LINE	BLOWDOWN [2] TIME AFTER BREAK (sec)	MASS FLOW RATE (lb/sec)	ENTHALPY (Btu/lb)[3]	ISOLATION VALVES	ISOLATION VALVE CLOSURE VALVE CLOSING TIME (sec)	SIGNAL DELAY TIME (sec)	TOTAL INTERVAL (sec)
HPCI steam supply line	0.0	2,940	1192.4	HV55-F002	12.0	1.0	13.0
in isolation valve compartment	0.135	1,272	1192.4	HV55-F003	12.0	1.0	13.0
(12 inch EBB-108)	0.23	902	1192.4				
	0.475	328	1192.4				
	13.0	328	1192.4				
	14.0	0	1192.4				
RCIC steam supply line	0.0	380	1192.4	HV49-F007	7.2	1.0	8.2
at turbine inlet valve	0.311	168	1192.4	HV49-F008	7.2	1.0	8.2
(6 inch EBB-109)	0.43	40	1192.4				
	7.2	40	1192.4				
	8.2	0	1192.4				
RCIC steam supply line	0.0	760	1192.4	HV49-F007	7.2	1.0	8.2
in upper pipe tunnel	0.13	382	1192.4	HV49-F008	7.2	1.0	8.2
(6 inch EBB-109)	0.26	85.4	1192.4				
	0.302	42	1192.4				
	7.2	42	1192.4				
	8.2	0	1192.4				

(1) Valve closure time is not applicable for HV44-F039 since it is a check valve. This valve prevents backflow of water from the feedwater lines into the RWCU equipment compartments in the event of a break.

(2) The blowdown table is based on original power level. Environmental effects from blowdown are addressed based 3527 MWt conditions in Table 3.6-7 and 3.6-9, which bounds the operation at MUR power level of 3515 MWt.

(3) Enthalpy of system is conservatively assumed for saturated steam at 1000 psig (0% moisture carryover) based on original design conditions.

Table 3.6-7

PRESSURE-TEMPERATURE TRANSIENT ANALYSIS RESULTS FOR [6]
HIGH ENERGY PIPE BREAKS OUTSIDE PRIMARY CONTAINMENT

(Unit 1)

COMPARTMENT[1]	PEAK[2] PRESSURE (psig)	TIME[5] AFTER BREAK (sec)	PEAK TEMPERATURE (°F)	TIME [5] AFTER BREAK (sec)
A. Main Steam Line Break in Main Steam Tunnel				
1. Main steam tunnel	10.18	3.0	321	0.60
2. Main steam tunnel vent stack (lower-region)	8.18	3.1	324	0.62
3. Main steam tunnel vent stack (mid-region)	5.32	3.1	325	0.64
4. Main steam tunnel vent stack (upper-region)	2.93	3.2	319	0.74
5. Main steam tunnel security plenum (U2 Only)	--	--	--	--
6. Main condenser area	0.56	0.23	179	5.8
7. Steam venting plenum	0.61	0.23	182	5.5
B. Main Steam Line Break in Main Condenser Area				
6. Main condenser area	2.33	4.2	208	4.95
7. Steam venting plenum	2.33	4.3	208	5.16
C. RWCU Suction Line Break in Penetration Room				
6. Nonregenerative heat exchanger room "A"	3.02	0.65	129[4]	0.51
7. Nonregenerative heat exchanger room "B"	3.02	0.64	128[4]	0.51
9. Regenerative heat exchanger room	2.92	0.65	127[4]	0.65
10. RWCU pump-room	2.91	0.38	105[4]	0.80
13. RWCU penetration room	2.92	0.40	202[4]	7.12

Table 3.6-7 (Cont'd)

(Unit 1)

COMPARTMENT[1]	PEAK[2] PRESSURE (psig)	TIME[5] AFTER BREAK (sec)	PEAK TEMPERATURE (°F)	TIME[5] AFTER BREAK (sec)
D. RWCU Pump Discharge Line Break in Pump-Room				
6. Nonregenerative heat exchanger room "A"	0.44[4]	0.78	115[4]	0.78
7. Nonregenerative heat exchanger room "B"	0.45[4]	0.76	115[4]	0.76
9. Regenerative heat exchanger room	0.44[4]	0.74	112[4]	0.29
10. RWCU pump-room (B, C only)	1.96[4]	0.33	222	3.07
10A.RWCU pump room (A only)	6.2	0.33	229	3.07
13. RWCU penetration room	0.44[4]	0.87	206[4]	15.00
E. RWCU Pump Discharge Line Break in Regenerative Heat Exchanger Room				
6. Nonregenerative heat exchanger room "A"	2.45[4]	0.92	113[4]	0.11
7. Nonregenerative heat exchanger room "B"	2.45[4]	0.92	112[4]	0.11
9. Regenerative heat exchanger room	2.42[4]	1.01	221	6.65
10. RWCU pump-room	1.78[4]	1.13	113[4]	0.43
13. RWCU penetration room	1.77[4]	1.16	215	15.95
F. RWCU Pump Discharge Line Break in Nonregenerative Heat Exchanger Room "A"				
6. Nonregenerative heat exchanger room "A"	1.56[4]	10.18	221	16.00
7. Nonregenerative heat exchanger room "B"	1.06[4]	10.82	111[4]	0.10
9. Regenerative heat exchanger room	0.50[4]	15.44	208[4]	16.13
10. RWCU pump-room	0.36[4]	16.23	107[4]	0.38
13. RWCU penetration room	0.36[4]	16.20	183[4]	16.43

Table 3.6-7 (Cont'd)

(Unit 1)

COMPARTMENT[1]	PEAK[2] PRESSURE (psig)	TIME[5] AFTER BREAK (sec)	PEAK TEMPERATURE (°F)	TIME[5] AFTER BREAK (sec)
G. RWCU Pump Discharge Line Break in Nonregenerative Heat Exchanger Room "B"				
6. Nonregenerative heat exchanger room "A"	1.07[4]	10.73	111[4]	0.10
7. Nonregenerative heat exchanger room "B"	2.03[4]	10.03	217[4]	16.00
9. Regenerative heat exchanger room	0.50[4]	15.55	208[4]	16.17
10. RWCU pump-room	0.36[4]	16.20	107[4]	0.40
13. RWCU penetration room	0.36[4]	16.23	183[4]	16.49
H. HPCI Steam Supply Line Break in HPCI Pump-Room				
17. HPCI pump-room	2.94	0.24	307	15.16
18. HPCI piping area	2.22	0.25	308	15.20
21. Isolation valve compartment	1.03	0.33	241	14.00
22. Steam venting tunnel	0.61	0.33	238	15.99
I. HPCI Steam Supply Line Break in HPCI Piping Area				
17. HPCI pump-room	2.54	0.14	264	15.51
18. HPCI piping area	6.64	0.13	295	0.13
21. Isolation valve compartment	1.52	0.21	241	15.71
22. Steam venting tunnel	0.91	0.21	235	12.88
J. HPCI Steam Supply Line Break in Isolation Valve Compartment				
21. Isolation valve compartment	1.51	0.18	273	16.55
22. Steam venting tunnel	0.98	0.18	270	15.93

Table 3.6-7 (Cont'd)

(Unit 1)

COMPARTMENT[1]	PEAK[2] PRESSURE (psig)	TIME[5] AFTER BREAK (sec)	PEAK TEMPERATURE (°F)	TIME[5] AFTER BREAK (sec)
K. RCIC Steam Supply Line Break in RCIC Pump-Room				
19. RCIC pump-room	2.94	0.27	229	11.17
20. RCIC upper pipe tunnel	2.56	0.29	218	11.15
21. Isolation valve compartment	0.50	0.38	129	13.99
22. Steam venting tunnel	0.50	0.38	128	0.38
L. RCIC Steam Supply Line Break in RCIC Upper Pipe Tunnel				
19. RCIC pump-room	2.68	0.16	153	0.16
20. RCIC upper pipe tunnel	5.77	0.03	306	7.04
21. Isolation valve compartment	0.50	0.22	142	13.99
22. Steam venting tunnel	0.50	0.22	135	11.12

Table 3.6-7 (Cont'd)

PRESSURE-TEMPERATURE TRANSIENT ANALYSIS RESULTS FOR
HIGH ENERGY PIPE BREAKS OUTSIDE PRIMARY CONTAINMENT

(Unit 2)

COMPARTMENT[1]	PEAK[2] PRESSURE (psig)	TIME[5] AFTER BREAK (sec)	PEAK TEMPERATURE (°F)[3]	TIME[5] AFTER BREAK (sec)
A. Main Steam Line Break in Main Steam Tunnel				
1. Main steam tunnel	11.39	3.1	320	0.60
2. Main steam tunnel vent stack (lower-region)	9.78	3.2	325	0.62
3. Main steam vent stack (security plenum)	7.81	3.2	325	0.67
4. Main steam tunnel vent stack (upper-region)	2.72	3.4	319	0.75
5. Main steam tunnel security plenum	10.30	3.2	320	0.60
6. Main condenser area	0.5	0.23	182	5.9
7. Steam venting plenum	0.54	0.23	187	5.6
B. Main Steam Line Break in Main Condenser Area				
6. Main condenser area	2.33	4.2	208	5.0
7. Steam venting plenum	2.33	4.3	208	5.2
C. RWCU Suction Line Break in Penetration Room				
6. Nonregenerative heat exchanger room "A"	3.02	0.65	129[4]	0.51
7. Nonregenerative heat exchanger room "B"	3.02	0.64	128[4]	0.51
9. Regenerative heat exchanger room	2.92	0.65	127[4]	0.65
10. RWCU pump-room	2.91	0.38	105[4]	0.80
13. RWCU penetration room	2.92	0.40	202[4]	7.12

Table 3.6-7 (Cont'd)

(Unit 2)

COMPARTMENT[1]	PEAK[2] PRESSURE (psig)	TIME[5] AFTER BREAK (sec)	PEAK TEMPERATURE (°F)[3]	TIME[5] AFTER BREAK (sec)
D. RWCU Pump Discharge Line Break in Pump-Room				
6. Nonregenerative heat exchanger room "A"	0.44[4]	0.78	115[4]	0.78
7. Nonregenerative heat exchanger room "B"	0.45[4]	0.76	115[4]	0.76
9. Regenerative heat exchanger room	0.44[4]	0.74	112[4]	0.29
10. RWCU pump-room(B, C only)	1.96[4]	0.33	222	3.07
10A.RWCU pump-room (A only)	6.2	0.33	229	3.07
13. RWCU penetration room	0.44[4]	0.87	206[4]	15.00
E. RWCU Pump Discharge Line Break in Regenerative Heat Exchanger Room				
6. Nonregenerative heat exchanger room "A"	2.4[4]	0.92	113[4]	0.11
7. Nonregenerative heat exchanger room "B"	2.45[4]	0.92	11[4]	0.11
9. Regenerative heat exchanger room	2.42[4]	1.01	221	6.65
10. RWCU pump-room	1.78[4]	1.13	113[4]	0.43
13. RWCU penetration room	1.77[4]	1.16	215	15.95
F. RWCU Pump Discharge Line Break in Nonregenerative Heat Exchanger Room "A"				
6. Nonregenerative heat exchanger room "A"	1.56[4]	10.18	221	16.00
7. Nonregenerative heat exchanger room "B"	1.06[4]	10.82	111[4]	0.10
9. Regenerative heat exchanger room	0.50[4]	15.44	208[4]	16.13
10. RWCU pump-room	0.36[4]	16.23	107[4]	0.38
13. RWCU penetration room	0.36[4]	16.20	183[4]	16.43

Table 3.6-7 (Cont'd)

(Unit 2)

COMPARTMENT[1]	PEAK[2] PRESSURE (psig)	TIME[5] AFTER BREAK (sec)	PEAK TEMPERATURE (°F)[3]	TIME[5] AFTER BREAK (sec)
G. RWCU Pump Discharge Line Break in Nonregenerative Heat Exchanger Room "B"				
6. Nonregenerative heat exchanger room "A"	1.11[4]	10.73	111[4]	0.10
7. Nonregenerative heat exchanger room "B"	2.11[4]	10.03	214[4]	16.00
9. Regenerative heat exchanger room	0.50[4]	15.55	206[4]	16.17
10. RWCU pump-room	0.38[4]	16.20	107[4]	0.40
13. RWCU penetration room	0.38[4]	16.23	184[4]	16.49
H. HPCI Steam Supply Line Break in HPCI Pump-Room				
17. HPCI pump-room	2 50	0.25	299	15.17
18. HPCI piping area	2.03	0.25	298	15.20
21. Isolation valve compartment	1.04	0.34	241	17.00
22. Steam venting tunnel	0.68	0.35	239	16.03
23. Security plenum	0.51	0.35	238	15.98
I. HPCI Steam Supply Line Break in HPCI Piping Area				
17. HPCI pump-room	2.79	0.14	243	0.14
18. HPCI piping area	2.79	0.14	299	4.74
21. Isolation valve compartment	1.39	0.22	271	17.00
22. Steam venting tunnel	0.97	0.22	260	16.04
23. Security plenum	0.72	0.22	260	15.97

Table 3.6-7 (Cont'd)

(Unit 2)

COMPARTMENT[(1)]	PEAK[(2)] PRESSURE (psig)	TIME[(5)] AFTER BREAK (sec)	PEAK TEMPERATURE (°F)[(3)]	TIME[(5)] AFTER BREAK (sec)
J. HPCI Steam Supply Line Break in Isolation Valve Compartment				
21. Isolation valve compartment	1.60	0.20	271	16.99
22. Steam venting tunnel	1.21	0.20	270	16.06
23. Security plenum	0.83	0.20	270	15.97
K. RCIC Steam Supply Line Break in RCIC Pump-Room				
19. RCIC pump-room	1.84	0.27	228	11.21
20. RCIC upper pipe tunnel	1.47	0.29	218	11.19
21. Isolation valve compartment	0.50	0.38	129	14.00
22. Steam venting tunnel	0.50	0.38	128	0.38
23. Security plenum	0.50	0.38	128	0.38
L. RCIC Steam Supply Line Break in RCIC Upper Pipe Tunnel				
19. RCIC pump-room	2.00	0.16	152	0.16
20. RCIC upper pipe tunnel	2.47	0.03	295	6.97
21. Isolation valve compartment	0.50	0.22	142	14.00
22. Steam venting tunnel	0.50	0.22	137	12.31
23. Security plenum	0.50	0.22	135	11.10

(1) Compartment numbers used in this table correspond to the compartment numbers used in the flow models (Figures 3.6-11, 3.6-12, 3.6-19, 3.6-23, 3.6-24, and 3.6-27).
(2) The compartment design pressures and the pressure-temperature transient analysis results are in Table 3.6-9.
(3) For Unit 2, design bulk temperatures may be less. Note also that temperatures in this area due to breaks elsewhere may be bounding.
(4) The value shown is based on original power level. It is bounded by another break in this compartment where the value shown is for the 3527 MWt power level.
(5) Time shown is based on original power level and was not recalculated for rerate since it is not used for any design basis evaluations.
(6) Values shown unless noted are based on power level of 3527 MWt. The values (except the main steam line valves) were established based on the original values and a multiplier. The multiplier was calculated based on a maximum pressure increase associated with a power level of 3527 MWt and its impact on the blowdown and subsequent impact to subcompartment pressures and temperatures.

Table 3.6-8

FEEDWATER PIPING STRESS LEVELS AND PIPE BREAK DATA
(PORTION INSIDE PRIMARY CONTAINMENT) (5)(6)

Node Point(1)	Node Type(2)	Stress By EQ. 10 (ksi)	Cumulative Usage Factor	Pipe Break Stress Limit 2.4 S_m (ksi)	Break Type(3)	Basis for Break Selection(4)
UNIT 1						
10	TTJ	77.05	0.0603	42.21	C	TE
45	SWP	91.10	0.5213	42.21	C	TE
55	TEE	104.58	0.4841	42.21	C&L	SFL
70	EL	69.14	0.1374	42.21	C&L	SFL
75	TTJ	76.74	0.6192	42.21	C	TE
100	TEE	100.79	0.3651	42.21	C&L	SFL
110	EL	72.19	0.1337	42.21	C&L	SFL
115	TTJ	79.85	0.6136	42.21	C	TE
170	EL	73.22	0.1316	42.21	C&L	SFL
180	TTJ	78.87	0.6103	42.21	C	TE
197	TTJ	74.17	0.0575	42.21	C	TE
UNIT 2						
10	TTJ	43.06	0.0531	42.21	C	TE
45	SWP	70.45	0.2216	42.21	C	TE
55	TEE	87.89	0.3960	42.21	C&L	SFL
115	TTJ	56.98	0.2482	42.21	C	TE
180	TTJ	61.18	0.2388	42.21	C	TE
197	TTJ	68.59	0.1942	42.21	C	TE
75	TTJ	59.97	0.2699	42.21	C	TE
100	TEE	102.14	0.8011	42.21	C&L	SFL

(1) Locations of the nodes listed in this table are shown in Figure 3.6-15.

(2) Node types are designated as follows:
- TEE - butt-welding tee
- TTJ - tapered transition joint
- EL - elbow
- SWP - sweepolet

(3) Break types are indicated as follows:
- C - circumferential
- L - longitudinal

(4) Symbols used to denote basis for break selection are as follows:
- TE - terminal end
- SFL - the stress and fatigue limits established in Section 3.6.2.1.1.3 are not met
- MBL - intermediate break location selected in order to satisfy the requirement for a minimum number of break locations

(5) The values provided above are based upon original licensed power conditions. These CUF values are conservatively based derived by using Sm for design temperatures instead of the load pair operating temperatures. For rerate power conditions, using the more realistic approach, the CUF values are found to be equal to or less than those in the table. No new pipe break is required for rerate power conditions. The maximum equation 10 stresses may be increased by up to 2% for the rerate condition.

(6) The information posted in this table was used for original System piping analysis. Refer to System ASME III piping analysis for current information.

Table 3.6-9

COMPARTMENT DESIGN PRESSURES AND PRESSURE-TEMPERATURE TRANSIENT ANALYSIS RESULTS

COMPARTMENT	UNIT 1 DESIGN PRESSURE(1) PSIG	UNIT 1 MAX TRANSIENT(2) PEAK PRESSURE PSIG	UNIT 2 DESIGN PRESSURE(1) PSIG	UNIT 2 MAX TRANSIENT(2) PEAK PRESSURE PSIG
Main Steam Tunnel	10.2	10.18	10.47	11.39
Main Steam Tunnel Vent Stack (Lower-Region)	8.2	8.18	8.99	9.78
Main Steam Tunnel Vent Stack (Mid-Region)	5.7	5.32	--	--
Main Steam Tunnel Vent Stack (Security Plenum)	--	--	6.97	7.81
Main Steam Tunnel Vent Stack (Upper-Region)	5.7	2.93	5.7	2.72
MST Security Plenum	--	--	2.14	2.33
Main Condenser Area	2.14	2.33	2.14	2.33
Steam Vent Plenum	2.14	2.33	10.47	10.30
HPCI Pump	2.7	2.94	2.7	2.79
HPCI Piping Area	6.1	6.64	6.1	2.79
RCIC Pump	2.8	2.94	2.8	2.00
RCIC Upper Piping Area	5.4	5.77	5.4	2.47
RHR, HPCI, RCIC Isolation Valve	1.41	1.52	1.47	1.60
HPCI, RCIC Steam Venting Tunnel	0.9	0.98	1.11	1.21
Security Plenum	- -	- -	1.11	0.83
Non-Regenerative Heat Exchanger	2.9	3.02	2.9	3.02
Non-Regenerative Heat Exchanger	2.9	3.02	2.9	3.02
Regenerative Heat Exchanger	2.9	2.92	2.9	2.92
RWCU Pump (A)	2.8	6.2	2.8	6.2
RWCU Pump (B and C)	2.8	2.91	2.8	2.91
RWCU Penetration	2.8	2.92	2.8	2.92

(1) In each compartment, the maximum pre-rerate peak transient pressure was used as a basis to establish the compartment design pressure. This is considered to be appropriate because of the conservatism in the analytical models used to calculate the mass and energy release rates.

(2) Maximum transient peak pressures are for power rerate conditions (3527 mwt). Although some peak pressures exceed listed design pressures, there are sufficient margin and conservatism in the existing design to accommodate the rerate condition. These pressures have been evaluated to be acceptable for each compartment.

Table 3.6-10

RWCU PIPING
STRESS LEVELS AND PIPE BREAK DATA
(PORTION INSIDE PRIMARY CONTAINMENT)[5][6]

Node Point[1]	Node Type[2]	Stress By EQ.10 (ksi)	Cumulative Usage Factor	Pipe Break Stress Limit 2.4 S_m (ksi)	Break Type[3]	Basis for Break Selection[4]
UNIT 1						
38	BW	28.735	0.0002	32.784	C	TE
77	TTJ	53.039	0.2145	32.784	C&L	SFL
79	TTJ	52.2	0.1883	32.784	C&L	SFL
82	TTJ	51.379	0.1666	32.784	C&L	SFL
90	TTJ	50.828	0.1497	32.784	C&L	SFL
103	TTJ	54.446	0.3216	32.784	C&L	SFL
115	SWP	45.461	0.0098	32.784	C	TE
UNIT 2						
39	STR	11.19	0.0	32.784	C	TE
103	TTJ	55.18	0.2643	32.784	C&L	SFL
115	SWP	45.36	0.0597	32.784	C	TE
47	TTJ	48.56	0.1348	32.784	C&L	SFL
53	TTJ	48.72	0.1394	32.784	C&L	SFL

(1) Locations of the nodes listed in this table are shown in Figure 3.6-17.

(2) Node types are designated as follows:
- TTJ - tapered transition joint
- BW - butt weld
- SWP - sweepolet
- STR - straight pipe

(3) Break types are indicated as follows:
- C - circumferential
- L - longitudinal

(4) Symbols used to denote the basis for break selection are as follows:
- TE - terminal end
- SFL - the stress and fatigue limits established in Section 3.6.2.1.1.3 are not met
- MBL - intermediate break location selected in order to satisfy the requirement for a minimum number of break locations

(5) The values provided above are based upon original licensed power conditions. These Cumulative Usage Factor (CUF) values are conservatively based derived by using Stress maximum (Sm) for design temperatures instead of the load pair operating temperatures. For rerate power conditions, using the more realistic approach, the CUF values are found to be equal to or less than those in the table. No new pipe break is required for rerate power conditions. The maximum equation 10 stresses may be increased by up to 2% for the rerate condition.

(6) The information posted in this table was used for the original System piping analysis. Refer to System ASME III piping analysis for current information.

Table 3.6-11

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Table 3.6-12

REACTOR VESSEL DRAIN PIPING
STRESS LEVELS AND PIPE BREAK DATA[(5)(6)]

Node Point[(1)]	Node Type[(2)]	Stress By EQ. 10 (ksi)	Cumulative Usage Factor	Pipe Break Stress Limit 2.4 S_m (ksi)	Break Type[(3)]	Basis for Break Selection[(4)]
UNIT 1						
553	RED	60.009	0.1068	33.936	C&L	SFL
643	TTJ	65.95	0.5875	33.936	C&L	SFL
645	TTJ	65.636	0.5678	33.936	C&L	SFL
670	WLD	49.61	0.0871	33.936	C&L	SFL
675	TTJ	56.538	0.1465	33.936	C&L	SFL
685	TTJ	56.273	0.1358	33.936	C&L	SFL
690	WLD	46.38	0.0573	33.936	C&L	SFL
720	RED	80.543	0.6964	33.936	C&L	SFL
725	TEE	61.08	0.2194	33.936	C	SFL
790	RED	57.937	0.1171	32.784	C	SFL
847	DMJ	54.883	0.1320	32.784	C	SFL
860	TTJ	39.605	0.0161	42.210	C	TE
866	RED	64.49	0.2543	32.784	C	SFL
875	TTJ	29.37	0.00	32.784	C	TE
917	TTJ	41.877	0.0237	32.784	C	SFL
925	TTJ	43.832	0.0311	32.784	C	SFL
935	TTJ	43.356	0.0289	32.784	C	SFL
UNIT 2						
553	RED	55.85	0.0979	32.78	C&L	SFL
643	TTJ	48.49	0.0555	32.78	C&L	SFL
645	TTJ	48.29	0.0527	32.78	C&L	SFL
670	WLD	45.62	0.0639	32.78	C&L	SFL
675	TTJ	42.75	0.0141	32.78	C&L	SFL
685	TTJ	42.83	0.0121	32.78	C&L	SFL
720	RED	65.23	0.7513	32.78	C&L	SFL
725	TEE	59.26	0.1430	32.78	C	SFL
860	TTJ	50.96	0.0535	42.210	C	TE
875	TTJ	50.74	0.1630	32.784	C	TE

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(1) Locations of the nodes listed in this table are shown in Figure 3.6-20.

(2) Node types are designated as follows:

- TEE - butt-welding tee
- TTJ - tapered transition joint
- RED - reducer
- DMJ - dissimilar metal joint
- WLD - weldolet

Table 3.6-12 (Cont'd)

(3) Break types are indicated as follows:

C	-	circumferential
L	-	longitudinal

(4) Symbols used to denote the basis for break selection are as follows:

TE	-	terminal end
SFL	-	the stress and fatigue limits established in Section 3.6.2.1.1.3 are not met
MBL	-	intermediate break location selected in order to satisfy the requirement for a minimum number of break locations

(5) The values provided above are based upon original licensed power conditions. These Cumulative Usage Factor (CUF) values are conservatively based derived by using Stress Maximum (Sm) for design temperatures instead of the load pair operating temperatures. For rerate power conditions, using the more realistic approach, the CUF values are found to be equal to or less than those in the table. No new pipe break is required for rerate power conditions. The maximum equation 10 stresses may be increased by up to 2% for the rerate condition.

(6) The information posted in this table was used for the original System analysis. Refer to System ASME III piping analysis for current information.

Table 3.6-13

HPCI STEAM SUPPLY PIPING STRESS LEVELS AND PIPE BREAK DATA
(PORTION INSIDE PRIMARY CONTAINMENT)(5)(6)

Node Point(1)	Node Type(2)	Stress By EQ. 10 (ksi)	Cumulative Usage Factor	Pipe Break Stress Limit 2.4 S_m (ksi)	Break Type(3)	Basis for Break Selection(4)
UNIT 1						
197	TTJ	47.5	0.0084	42.2	C	TE
200	EL	48.4	0.0018	42.2	C	MBL
225	EL	58.4	0.0061	42.2	C	MBL
358	STR	21.1	0.0002	42.2	C	TE
UNIT 2						
197	TTJ	53.86	0.0289	42.2	C	TE
358	STR	19.16	0.0004	42.2	C	TE

(1) Locations of the nodes listed in this table are shown in Figure 3.6-21.

(2) Node types are designated as follows:

TTJ - tapered transition joint
EL - elbow
STR - straight pipe

(3) Break types are indicated as follows:

C - circumferential
L - longitudinal

(4) Symbols used to denote the basis for break selection are as follows:

TE - terminal end
SFL - the stress and fatigue limits established in Section 3.6.2.1.1.3 are not met
MBL - intermediate break location selected in order to satisfy the requirement for a minimum number of break locations

(5) The values provided above are based upon original licensed power conditions. These Cumulative Usage Factor (CUF) values are conservatively based derived by using Stress maximu (Sm) for design temperatures instead of the load pair operating temperatures. For rerate power conditions, using the more realistic approach, the CUF values are found to be equal to or less than those in the table. No new pipe break is required for rerate power conditions. The maximum equation 10 stresses may be increased by up to 6% for the rerate condition.

(6) The information posted in this table was used for the original System piping analysis. Refer to System ASME III piping analysis for current information.

Table 3.6-14

HPCI STEAM SUPPLY PIPING STRESS LEVELS AND PIPE BREAK DATA
(PORTION OUTSIDE PRIMARY CONTAINMENT)[5][6]

NODE POINT[1]	NODE TYPE[2]	STRESS (ksi) EQ. 9	EQ. 10	TOTAL	PIPE BREAK STRESS LIMIT 0.8($1.2S_h+S_A$) (ksi)	BREAK TYPE[3]	BASIS FOR BREAK SELECTION[4]
UNIT 1							
45	BW	6.87	18.19	25.06	32.4	C	TE
55	TEE	7.52	15.59	23.11	32.4	C	MBL
65	EL	7.50	10.83	18.33	32.4	C	MBL
85	BW	7.02	5.49	12.51	32.4	C	TE
UNIT 2							
95	ST	9.62	12.39	22.01	32.4	C	TE
150	EL	10.72	11.03	21.75	32.4	C	TE

(1) Locations of the nodes listed in this table are shown in Figure 3.6-22.

(2) Node type are designated as follows:
- TTJ - tapered transition joint
- TEE - butt-welding tee
- EL - elbow
- BW - butt weld
- ST - straight pipe

(3) Break types are indicated as follows:
- C - circumferential
- L - longitudinal

(4) Symbols used to denote the basis for break selection are as follows:
- TE - terminal end
- SFL - stress and fatigue limits established in Section 3.6.2.1.1.3 are not met
- MBL - intermediate break location selected to satisfy the requirement for a minimum number of break locations

(5) The values provided above are based upon original licensed power conditions. These Cumulative Usage Factor (CUF) values are conservatively based derived by using Stress maximum (Sm) for design temperatures instead of the load pair operating temperatures. For rerate power conditions, using the more realistic approach, the CUF values are found to be equal to or less than those in the table. No new pipe break is required for rerate power conditions. The maximum equation 10 stresses may be increased by up to 6% for the rerate condition.

(6) The information posted in this table was used for the original System piping analysis. Refer to System ASME III piping analysis for current information.

Table 3.6-15

RCIC STEAM SUPPLY PIPING STRESS LEVELS AND PIPE BREAK DATA
(PORTION INSIDE PRIMARY CONTAINMENT)(5)(6)

Node Point(1)	Node Type(2)	Stress By EQ.10 (ksi)	Cumulative Usage Factor	Pipe Break Stress Limit 2.4 S_m (ksi)	Break Type(3)	Basis for Break Selection(4)
UNIT 1						
250	EL	35.62	0.0003	42.2	C	TE
320	EL	41.7	0.0008	42.2	C	MBL
350	EL	47.4	0.0013	42.2	C	MBL
391	TTJ	68.5	0.29	42.2	C	TE
UNIT 2						
250	EL	12.05	0.0000	42.2	C	TE
391	TTJ	62.60	0.1201	42.2	C	TE

(1) Location of the nodes listed in this table are shown in Figure 3.6-25.

(2) Node types are designated as follows:
- TTJ - tapered transition joint
- EL - elbow
- RED - reducer

(3) Break types are indicated as follows:
- C - circumferential
- L - longitudinal

(4) Symbols used to denote the basis for break selection are as follows:
- TE - terminal end
- SFL - the stress and fatigue limits established in Section 3.6.2.1.1.3 are not met
- MBL - intermediate break location selected in order to satisfy the requirement for a minimum number of break locations

(5) The values provided above are based upon original licensed power conditions. These Cumulative Usage Factor (CUF) values are conservatively based derived by using Stress maximum (Sm) for design temperatures instead of the load pair operating temperatures. For rerate power conditions, using the more realistic approach, the CUF values are found to be equal to or less than those in the table. No new pipe break is required for rerate power conditions. The maximum equation 10 stresses may be increased by up to 6% for the rerate condition.

(6) The information posted in this table was used for the original System piping analysis. Refer to System ASME III piping analysis for current information.

Table 3.6-16

RCIC STEAM SUPPLY PIPING STRESS LEVELS AND PIPE BREAK DATA
(PORTION OUTSIDE PRIMARY CONTAINMENT)[(5),(6)]

NODE POINT[(1)]	NODE TYPE[(2)]	STRESS (ksi) EQ. 9	EQ. 10	TOTAL	PIPE BREAK STRESS LIMIT $0.8(1.2S_h+S_A)$ (ksi)	BASIS FOR BREAK TYPE[(3)]	BREAK SELECTION[(4)]
UNIT 1							
40	BW	7.62	1.60	9.22	32.4	C	TE
100	EL	8.14	18.63	26.77	32.4	C	MBL
105	EL	7.66	16.84	24.50	32.4	C	MBL
117	STR	7.40	1.69	9.08	32.4	C	TE
UNIT 2							
30	TTJ	6706	927	7633	32400	C	TE
105	EL	6712	2618	9330	32400	C	TE

(1) Locations of the nodes listed in this table are shown in Figure 3.6-26.

(2) Node types are designated as follows:
- TTJ - tapered transition joint
- EL - elbow
- BW - butt weld
- STR - straight pipe

(3) Break types are indicated as follows:
- C - circumferential
- L - longitudinal

(4) Symbols used to denote the basis for break selection are as follows:
- TE - terminal end
- SFL - stress and fatigue limits established in Section 3.6.2.1.1.3 are not met
- MBL - intermediate break location selected to satisfy the requirement for a minimum number of break locations

(5) The values provided above are based upon original licensed power conditions. These cumulative usage factor (CUF) values are conservatively derived by using Stress maximum (Sm) for design temperatures instead of the load pair operating temperatures. For rerate power conditions, using the more realistic approach, the CUF values are found to be equal to or less than those in the table. No new pipe break is required for rerate power conditions. The maximum equation 10 stresses may be increased up to 6% for the rerate conditions.

(6) The information posted in this table was used for the original System piping analysis. Refer to System ASME III piping analysis for current information.

Table 3.6-17
MAIN STEAM DRAINAGE PIPING STRESS LEVELS AND PIPE BREAK DATA
(PORTION INSIDE PRIMARY CONTAINMENT)(5)(6)

Node Point(1)	Node Type(2)	Stress By EQ. 10 (ksi)	Cumulative Usage Factor	Pipe Break Stress Limit 2.4 S_m (ksi)	Break Type(3)	Basis for Break Selection(4)
UNIT 1						
10	SW	37.8	0.0196	42.2	C	TE
30	SW	39.12	0.0168	42.2	C	MBL
51	TEE	41.0	0.0075	42.2	C	MBL
65	SW	28.4	0.0143	42.2	C	MBL
90	SW	28.5	0.0124	42.2	C	TE
110	TTJ	13.24	0.0	42.2	C	TE
195	TEE	40.8	0.0014	42.2	C	MBL
220	SW	28.2	0.0141	42.2	C	MBL
240	SW	26.3	0.0113	42.2	C	TE
276	SW	39.5	0.0171	42.2	C	MBL
295	SW	37.64	0.0196	42.2	C	TE
UNIT 2						
10	SW	53.58	0.0402	42.2	C	TE
90	SW	38.49	0.0174	42.2	C	TE
110	TTJ	18.32	0.0006	42.2	C	TE
240	SW	38.93	0.0320	42.2	C	TE
295	SW	57.16	0.0582	42.2	C	TE

(1) Locations of the nodes listed in this table are shown in Figure 3.6-28.

(2) Node types are designated as follows:
- TEE - butt-welding tee
- TTJ - tapered transition joint
- SW - socket weld

(3) Break types are indicated as follows:
- C - circumferential
- L - longitudinal

(4) Symbols used to denote the basis for break selection are as follows:
- TE - terminal end
- SFL - the stress and fatigue limits established in Section 3.6.2.1.1.3 are not met
- MBL - intermediate break location selected in order to satisfy the requirement for a minimum number of break locations

(5) The values provided above are based upon original licensed power conditions. These Cumulative Usage Factor (CUF) values are conservatively based derived by using Stress maximum (Sm) for design temperatures instead of the load pair operating temperatures. For rerate power conditions, using the more realistic approach, the CUF values are found to be equal to or less than those in the table. No new pipe break is required for rerate power conditions. The maximum equation 10 stresses may be increased by up to 2% for the rerate condition.

(6) The information posted in this table was used for the original System piping analysis. Refer to System ASME III piping analysis for current information.

Table 3.6-18

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Table 3.6-19

RPV HEAD VENT PIPING
STRESS LEVELS AND PIPE BREAK DATA[5][6]

Node Point[1]	Node Type[2]	Stress By EQ. 10 (ksi)	Cumulative Usage Factor	Pipe Break Stress Limit 2.4 S_m (ksi)	Break Type[3]	Basis for Break Selection[4]
UNIT 1						
5	TTJ	47.61	0.0541	42.21	C	TE
10	CUR	46.95	0.0012	42.21	C	MBL
28	RED	54.22	0.3862	42.21	C	SFL
30	DMJ	59.49	0.1037	42.21	C	SFL
33	TTJ	28.92	0.0	42.21	C	TE
500	TEE	44.85	0.0157	42.21	C	MBL
620	SW	25.50	0.0672	42.21	C	TE
835	SW	54.69	0.0621	42.21	C	MBL
920	SW	51.97	0.0232	42.21	C	TE
292	SW	74.85	0.5044	42.21	C	SFL
UNIT 2						
5	TTJ	32.80	0.0032	42.21	C	TE
292	SW	74.85	0.5044	42.21	C	SFL
40	TTJ	21.09	0.00	33.94	C	TE
620	SW	19.03	0.0017	42.21	C	TE
910	SW	68.56	0.2355	42.21	C	TE

(1) Locations of the nodes are shown in Figure 3.6-30.

(2) Node types are designated as follows:
- TTJ - Tapered transition joint
- CUR - Butt-weld elbow
- RED - Reducer
- DMJ - Dissimilar metal joint
- TEE - Butt-welding tee
- SW - Socket weld

(3) Break types are indicated as follows:
- C - Circumferential

(4) Symbols used to denote the basis for break selection are as follows:
- TE - Terminal end
- MBL - Intermediate break location selected to satisfy the requirement for a minimum number of break locations.
- SFL - The stress and fatigue limits established in Section 3.6.2.1.1.3 are not met.

(5) The values provided above are based upon original licensed power conditions. These Cumulative Usage Factor (CUF) values are conservatively based derived by using Stress maximum (Sm) for design temperatures instead of the load pair operating temperatures. For rerate power conditions, using the more realistic approach, the CUF values are found to be equal to or less than those in the table. No new pipe break is required for rerate power conditions. The maximum equation 10 stresses may be increased by up to 2% for the rerate condition.

(6) The information posted in this table was used for the original System piping analysis. Refer to System ASME III piping analysis for current information.

Table 3.6-20

SLCS INJECTION PIPING
STRESS LEVELS AND PIPE BREAK DATA[5][6]

Node Point[1]	Node Type[2]	Stress By EQ. 10 (ksi)	Cumula-tive Usage Factor	Pipe Break Stress Limit 2.4 S_m (ksi)	Break Type[3]	Basis for Break Selection[4]
UNIT 1						
5	SW	57.25	0.1478	33.8	C	TE
15	SW	58.67	0.1784	33.8	C	SFL
20	SW	59.43	0.1887	33.8	C	SFL
35	SW	60.53	0.2049	33.8	C	SFL
40	SW	61.15	0.2208	33.8	C	SFL
42	SW	61.36	0.2285	33.8	C	SFL
43	SW	60.96	0.2204	33.8	C	SFL
70	SW	64.29	0.3164	33.8	C	SFL
75	SW	64.10	0.3139	33.8	C	SFL
76	SW	63.46	0.2940	33.8	C	TE
UNIT 2						
7	SW	40.31	0.0455	32.44	C	TE
76	SW	46.69	0.0559	32.44	C	TE

(1) Locations of the nodes are shown in Figure 3.6-31.

(2) Node type are designated as follows:
SW - Socket weld

(3) Break types are indicated as follows:
C - Circumferential

(4) Symbols used to denote the basis for break selection are as follows:
TE - Terminal end break
SFL - The stress and fatigue limits established in Section 3.6.2.1.1.3 are not met.

(5) The values provided above are based upon original licensed power conditions. These Cumulative Usage Factor (CUF) values are conservatively based derived by using Stress maximum (Sm) for design temperatures instead of the load pair operating temperatures. For rerate power conditions, using the more realistic approach, the CUF values are found to be equal to or less than those in the table. No new pipe break is required for rerate power conditions. The maximum equation 10 stresses may be increased by up to 2% for the rerate condition.

(6) The information posted in this table was used for the original System piping analysis. Refer to System ASME III piping analysis for current information.

Table 3.6-21

RHR SHUTDOWN COOLING PIPING
STRESS LEVELS AND PIPE BREAK DATA(7)(8)

Node Point(1)	Node Type(2)	Stress By EQ. 10 (ksi)	Cumulative Usage Factor	Pipe Break Stress Limit 2.4 S_m (ksi)	Break Type(3)	Basis for Break Selection(4)
UNIT 1						
107	BW	37.999	0.0193	32.79	C	TE
109	EL	60.727	0.3363	32.79	C&L	SFL
125	TTJ	60.924	0.976	32.79	C&L	SFL
142	TTJ	58.386	0.831	32.79	C&L	SFL
150	TTJ	60.330	0.8259	32.79	C	TE
288	TTJ	38.46	0.0242	32.79	C	TE
291	TTJ	37.659	0.0238	32.79	C	MBL
293	TTJ	40.264	0.0258	32.79	C	MBL
297	BW	32.920	0.0463	32.79	C	TE
325	TTJ	39.737	0.0269	32.79	C	TE
335	TTJ	41.565	0.0314	32.79	C	MBL
345	TTJ	45.696	0.0676	32.79	C	MBL
350	BW	34.127	0.0515	32.79	C	TE
UNIT 2						
107	TTJ	33.59	0.0141	32.79	C	TE
109(5)	EL	51.20	0.0415	32.79	C&L	SFL
125	TTJ	54.48	0.3439	32.79	C&L	SFL
142	TTJ	53.48	0.3065	32.79	C&L	SFL
150	TTJ	56.73	0.4447	32.79	C	TE
288	TTJ	35.98	0.0231	32.79	C	TE
297	TTJ	37.48	0.0396	32.79	C	TE
325	TTJ	38.16	0.0219	32.79	C	TE
350	BW	32.51	0.0319	32.79	C	TE
293(6)	TTJ	46.84	0.0527	32.79	C&L	SFL

(1) Locations of the nodes listed in this table are shown in Figure 3.6-32.

(2) Node types are designated as follows:

TTJ - tapered transition joint
EL - elbow
BW - butt weld

Table 3.6-21 (Cont'd)

(3) Break types are indicated as follows:

C	-	circumferential
L	-	longitudinal

(4) Symbols used to denote the basis for break selection are as follows:

TE	-	terminal end
SFL	-	the stress and fatigue limits established in Section 3.6.2.1.1.3 are not met
MBL	-	intermediate break location selected in order to satisfy the requirement for a minimum number of break locations

(5) EQ.13 = 38.37 ksi

(6) EQ.13 = 38.13 ksi

(7) The values provided above are based upon original licensed power conditions. These CUF values are conservatively based derived by using Sm for design temperatures instead of the load pair operating temperatures. For rerate power conditions, using the more realistic approach, the CUF values are found to be equal to or less than those in the table. No new pipe break is required for rerate power conditions. The maximum equation 10 stresses may be increased by up to 2% for the rerate condition.

(8) The information posted in this table was used for the original System piping analysis. Refer to System ASME III piping analysis for current information.

Table 3.6-22

LPCI INJECTION PIPING
STRESS LEVELS AND PIPE BREAK DATA[(5)(6)]

Node Point[(1)]	Node Type[(2)]	Stress By EQ. 10 (ksi)	Cumulative Usage Factor	Pipe Break Stress Limit 2.4 S_m (ksi)	Break Type[(3)]	Basis for Break Selection[(4)]
UNIT 1						
65	TTJ	30.87	0.0008	32.78	C	T
67	DMJ	56.97	0.12	32.78	C&L	SFL
90	EL	117.84	0.81	32.78	C&L	SFL
100	TTK	157.16	0.64	32.78	C&L	SFL
140	EL	156.47	0.80	32.78	C&L	SFL
145	STR	109.32	0.44	32.78	C&L	SFL
150	TTJ	154.95	0.62	32.78	C	TE
UNIT 2						
65	TTJ	37.48	0.0012	32.78	C	TE
90	EL	117.84	0.4709	32.78	C&L	SFL
100	TTJ	143.83	0.6160	32.78	C&L	SFL
140	EL	114.41	0.4580	32.78	C&L	SFL
150	TTJ	140.38	0.6493	32.78	C	TE

(1) Locations of the nodes listed in this table are shown in Figure 3.6-33.

(2) Node types are designated as follows:
- TTJ - tapered transition joint
- EL - elbow
- DMJ - dissimilar metal joint
- STR - straight pipe

(3) Break types are indicated as follows:
- C - circumferential
- L - longitudinal

(4) Symbols used to denote the basis for break selection are as follows:
- TE - terminal end
- SFL - the stress and fatigue limits established in Section 3.6.2.1.1.3 are not met
- MBL - intermediate break location selected in order to satisfy the requirement for a minimum number of break locations

(5) The values provided above are based upon original licensed power conditions. These Cumulative Usage Factor (CUF) values are conservatively based derived by using Stress maximum (Sm) for design temperatures instead of the load pair operating temperatures. For rerate power conditions, using the more realistic approach, the CUF values are found to be equal to or less than those in the table. No new pipe break is required for rerate power conditions. The maximum equation 10 stresses may be increased by up to 2% for the rerate condition.

(6) The information posted in this table was used for the original System piping analysis. Refer to System ASME III piping analysis for current information.

Table 3.6-23

CORE SPRAY INJECTION PIPING
STRESS LEVELS AND PIPE BREAK DATA(5)(6)

Node Point(1)	Node Type(2)	Stress By EQ. 10 (ksi)	Cumula-tive Usage Factor	Pipe Break Stress Limit 2.4 S_m (ksi)	Break Type(3)	Basis for Break Selection(4)
UNIT 1						
25	TTJ	30.19	0.0016	32.78	C	TE
30	DMJ	66.87	0.1365	32.78	C&L	SFL
63	STR	106.87	0.3789	32.78	C&L	SFL
65	EL	124.58	0.7034	32.78	C&L	SFL
72	RED	108.52	0.7573	32.78	C&L	SFL
73	STR	109.03	0.4141	32.78	C&L	SFL
75	TTJ	117.54	0.4955	32.78	C	TE
UNIT 2						
25	TTJ	30.49	0.0037	32.78	C	TE
75	TTJ	56.29	0.0733	32.78	C	TE

(1) Locations of the nodes listed in this table are shown in Figure 3.6-34.

(2) Node types are designated as follows:
- TTJ - tapered transition joint
- EL - elbow
- RED - reducer
- DMJ - dissimilar metal joint
- STR - straight pipe

(3) Break types are indicated as follows:
- C - circumferential
- L - longitudinal

(4) Symbols used to denote the basis for break selection are as follows:
- TE - terminal end
- SFL - the stress and fatigue limits established in Section 3.6.2.1.1.3 are not met
- MBL - intermediate break location selected in order to satisfy the requirement for a minimum number of break locations

(5) The values provided above are based upon original licensed power conditions. These Cumulative Usage Factor (CUF) values are conservatively based derived by using Stress maximum (Sm) for design temperatures instead of the load pair operating temperatures. For rerate power conditions, using the more realistic approach, the CUF values are found to be equal to or less than those in the table. No new pipe break is required for rerate power conditions. The maximum equation 10 stresses may be increased by up to 2% for the rerate condition.

(6) The information posted in this table was used for the original System piping analysis. Refer to System ASME III piping analysis for current information.

Table 3.6-24

RESTRAINT DATA USED IN VERIFICATION OF RECIRCULATION SYSTEM PIPE WHIP RESTRAINT DESIGN[(1)]

PIPE SIZE (in)	REST LOAD DIRECTION	C_2[(2)]	N[(2)]	LIMIT, Δ RESTRAINT[(2)]	INITIAL CLEARANCE (in)	EFFECTIVE CLEARANCE (in)	TOTAL CLEARANCE (in)
12	0°	27,733	0.24	6.129	4	1.941	5.941
12	90°	14,795	0.401	9.063	4	12.247	16.247
16	0°	109,265	0.24	6.278	4	1.934	5.934
6	90°	62,599	0.377	8.978	4	12.187	16.187
24	0°	102,228	0.24	8.222	4	1.984	5.984
24	90°	55,531	0.375	11.972	4	13.685	17.685
24	38°[(3)]	109,888	0.24	5.588	4	5.698	9.698
24	52°[(3)]	109,835	0.24	5.473	4	8.462	12.462

(1) The restraint data listed applies to one bar of a restraint.

(2) $F = C_2 (\Delta \text{ restraint})^N$
where F is the resistance force for one bar of a restraint and
where (Δ restraint) = (σ pipe) - (total clearance)

(3) Applies to restraint RCR 3 only.

Table 3.6-25

COMPARISON OF PDA AND NSC CODES

BREAK DESIGNATION[1]	RESTRAINT DESIGNATION[1]	NO. OF BARS		LOAD (kips)		RESTRAINT DEFLECTION(in)		FRACTION OF DESIGN RESTRAINT DEFLECTION (%)		PIPE DEFLECTION (in)	
		PDA	NSC	PDA	NSC	PDA	NSC	PDA	NSC	PDA	NSC
$RC1_J$	RCR1	5	5	803.2	788.3	6.57	7.926	79.93	96.4	17.72	15.58
$RC2_{LL}$	RCR1	5	5	766.4	458.4	14.99	7.495	125	62.6	35.83	24.52
$RC3_{LL}$	RCR2	6	6	747.0	639.7	2.27	3.73	27.65	45.35	17.16	20.11
$RC3_{LL}$	RCR2	6	6	796.6	780.3	10.22	10.54	57.8	59.6	41.48	43.0
$RC4_{LL}$	RCR3	5	5	846.0	838.4	7.64	8.05	92.95	97.98	18.87	16.43
$RC4_{LL}$	RCR3	8	8	1319.0	1073.9	5.43	4.62	99.23	76.85	23.28	17.25
$RC4C_V$	RCR3	8	8	1260.7	1275.0	4.49	5.58	80.37	99.89	22.56	18.73
$RC6A_V$	RCR3	8	8	928.5	722.5	1.22	1.77	22.46	31.7	23.68	95.39
$RC7_J$	RCR7	6	6	953.3	801.6	6.28	5.76	76.4	70.12	16.46	21.63
$RC8_{LL}$	RCR6	4 6	4 6	599.0 895.0	0 0	8.28 8.16	0 0	112.46 110.76	0 0	26.76 29.316	 8.39
$RC9C_V$	RCR6	4	4	575.8	520.16	4.16	5.53	50.63	67.33	13.2	14.56
$RC9_{LL}$	RCR8	6	6	830.2	546.8	11.408	6.815	95.29	56.9	36.612	26.24
RC11A	RCR8	6	6	818.3	493.6	10.98	5.99	91.72	50.07	31.404	23.71
RC13	RCR10	4	4	668.4	478.4	5.87	3.66	93.5	58.39	13.37	10.44
RC16	RCR11	4	4	687.4	518.4	6.59	4.38	105	69.86	15.37	10.22
$RC14C_V$	RCR20	8	8	285.0	309.6	2.83	5.88	46.3	95.92	15.45	13.96
$RC14_{LL}$	RCR20	8	8	116.3	129.9	0.96	3.36	10.5	37.1	22.13	23.56

[1] Break designations and restraint designations are shown on Figure 3.6-37.

3.7 SEISMIC DESIGN

All systems and equipment of the NSSS are defined as either seismic Category I or nonseismic Category I. The requirements for seismic Category I qualification are given in Section 3.2, along with a list of systems, components, and equipment which are so categorized. Seismic design requirements for nonseismic Category I items are also defined in Section 3.2. These items are classified as either Category II or Category IIA.

All systems, components, and equipment related to plant safety are designed to withstand SSE and OBE.

The SSE is that earthquake which is based upon an evaluation of the maximum earthquake potential considering the regional and local geology, and seismology and specific characteristics of local subsurface material. It is that earthquake which produces the maximum vibratory ground motion for which seismic Category I systems and components are designed to remain functional. These systems and components are those necessary to ensure:

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 radiological consequences of accidents which could result in potential offsite exposures, comparable to the dose limits of 10 CFR 50.67.

The OBE is that earthquake which, considering the regional and local geology, and seismology and specific characteristics of local subsurface material, could reasonably be expected to affect the plant site during the operating life of the plant. It is that earthquake which produces the vibratory ground motion for which those features of the nuclear power plant necessary for continued operation, without undue risk to the health and safety of the public, are designed to remain functional.

In addition to seismic loads, Category I structures, systems and components are also reviewed for hydrodynamic loads as described in Appendix 3A.

3.7.1 SEISMIC INPUT

3.7.1.1 Design Response Spectra

The site design response spectra used are shown in Figures 3.7-1 and 3.7-2. These spectra are for the horizontal components of the OBE and the SSE respectively. The response spectra for the OBE are scaled or normalized to a maximum horizontal ground acceleration of 7½% of gravity. The response spectra for the SSE are normalized to a maximum horizontal ground acceleration of 15% of gravity. The values for the vertical component of the design response spectra are 2/3 of the horizontal design response spectra described above. The response spectra are based on data developed from records of previous earthquake activity and represent an envelope of motion expected at a sound rock site from a nearby earthquake (Section 2.5.2).

Regulatory Guide 1.60 (December, 1973) "Design Response Spectra for Seismic Design of Nuclear Power Plants" was not used for the development of the spectra, because LGS Units 1 and 2 were docketed for construction permit review in March 1970, and the spectra were finalized in

1973. Further, a letter dated December 21, 1973 from J.M. Hendrie (NRC) to R.M. Collins (Bechtel) states that Regulatory Guide 1.60 is applicable only to the plants docketed for construction permit review after April 1, 1973.

3.7.1.2 Design Time History

A synthetic time history of motion was generated by modifying the actual records of the 1952 Taft earthquake according to the techniques proposed in Reference 3.7-1. This synthetic time history was then further modified to develop the time history shown in Figure 3.7-3, which corresponds to the design response spectra. The duration of the time history is 15 seconds. Figures 3.7-4 through 3.7-9 show a comparison of the time history response spectra and the design response spectra for 0.5%, 1%, 2%, 3%, 5%, and 7% damping values.

The spectra are computed at the frequency values as given in table 5-1 of Reference 3.7-2.

3.7.1.3 Critical Damping Values

3.7.1.3.1 Critical Damping Values (NSSS)

The damping values indicated in Table 3.7-1 are used in the response analysis of various structures and systems, and in preparation of floor response spectra used as forcing inputs for piping and equipment analysis or testing. It can be seen that the values given in Table 3.7-1 are somewhat less than those given in Regulatory Guide 1.61 (October 1973), "Damping Values for Seismic Design of Nuclear Power Plants". The calculated responses are, therefore, conservative. Alternative critical damping values for piping may be used as described in Section 3.7.1.3.3.

3.7.1.3.2 Critical Damping Values (Non-NSSS)

Critical damping values expressed as a percentage of critical damping and used for seismic Category I structures, equipment, and piping for both the OBE and SSE are given in Table 3.7-2. Alternative critical damping values for piping may be used as described in Section 3.7.1.3.3.

Regulatory Guide 1.61 is not used as a design basis as discussed in Section 1.8, except as discussed in Section 3.7.1.3.3. However, all the values shown in Table 3.7-2 are equivalent to or more conservative than those in Regulatory Guide 1.61 with the exception of the SSE value for welded steel structures. The damping value of 5% (PSAR table C.2.1) is based on information given in Reference 3.7-6. The 5% value has been used, with appropriate design margins, because the stress levels for SSE conditions are allowed to approach the yield point.

3.7.1.3.3 Alternative Critical Damping Values and Spectral Peak Broadening for Piping (NSSS and Non-NSSS)

Alternative critical damping values, as provided in ASME B&PV Code, Section III, Division 1 Code Case N-411 may be used. When used, the provisions listed below are applied unless otherwise approved by References 3.7-12 and 3.7-13. References 3.7-12 and 3.7-13 approved the use of Independent Support Motion (ISM) response spectra methodology in conjunction with the damping values specified in ASME Code Case N-411, as the basis for eliminating snubbers from 31 piping systems on Unit 1 and Unit 2.

- The code case damping is applied only to uniform (or envelope) response spectra loading analysis for seismic and seismiclike building filtered hydrodynamic loads and the annulus pressurization loading.

- The code case damping is applied to a spectral analysis load case in its entirety and is not mixed with other damping values within that one load case.

- Modal and direction combination of the three earthquake directions are combined in accordance with Regulatory Guide 1.92.

- Consideration of a sufficient number of modes such that the inclusion of additional modes would not result in more than a 10% increase in response.

- Assurance that the predicted piping displacements are such that adequate clearance exists with respect to adjacent components and equipment.

- Line mounted equipment is designed to withstand the increased pipe motion.

- The code case damping is not used for analyzing the dynamic response of piping systems incorporating supports designed to dissipate energy by yielding.

- The code case damping is not applied to piping analytical models that incorporate equipment with natural frequencies below 20 hertz.

- The code case damping is not applied to piping in which stress-corrosion cracking has occurred unless a case specific evaluation is reviewed by the NRC.

When time history or independent support motion response spectrum analysis is utilized, the following critical damping values are applied:

a. Time history analysis with 0.5% damping for OBE, 1% damping for SSE, or Regulatory Guide 1.61 damping for hydrodynamic loads.

b. Independent support motion response spectra analysis with Regulatory Guide 1.61 damping.

ASME Section III, Division I Code Case N-397, "Alternative Rules to the Spectral Broadening Procedures of N-1226.3 for Classes 1, 2 and 3 Piping, Section III, Division 1" may be selectively used on a load case basis. This code case is applicable to all spectral analysis load cases and all methodologies except it was not applied to a spectral analysis load case that utilized the independent support motion analysis methodology.

3.7.1.4 Supporting Media for Seismic Category I Structures

All seismic Category I structures are supported on sound rock or concrete backfill bearing on sound rock, except for some yard facilities such as valve pits and portions of electrical duct banks and underground piping which are supported on natural soil or fill (Section 2.5.4.10).

For the dynamic analysis of the rock-founded structures, soil-structure interaction is considered to be negligible due to the high stiffness of the rock. However, the floor response spectra developed for the reactor enclosure and the containment for equipment analysis are based on a model that considered the flexibility of the supporting medium. The modulus of elasticity, the shear-wave velocity, and the density of the supporting medium used in the analysis are 3.0×10^6 psi, 6000 fps, and 150 lbs/ft^3, respectively. Additional soil-structure interaction studies were performed for the containment structure and reactor and control enclosures to assess the sensitivity of the structural response to variations in the design basis rock modulus. Modal analyses have demonstrated that for a ±50% of the design elastic modulus range, variations in structural frequency do not exceed 10% for predominant modes. These results indicate that a reduction in rock modulus to 1.5×10^6 psi would not produce significant effects on the structural response. It is therefore concluded that the average dynamic elastic modulus value of 3.0×10^6 psi is adequate for design.

3.7.2 SEISMIC SYSTEM ANALYSIS

Seismic Category I structures and systems, and components of the NSSS that fall under the category of a seismic system, are discussed here. Seismic systems are analyzed for both OBE and SSE.

3.7.2.1 Seismic Analysis Method

3.7.2.1.1 Seismic Analysis Methods (NSSS)

Analysis of seismic Category I NSSS systems and components was accomplished, where applicable, using the response spectrum or time history approach. Either approach utilizes the natural period, mode shapes, and appropriate damping factors of the particular system. Certain pieces of equipment having very high natural frequencies may be analyzed statically. In some cases, dynamic testing of equipment may be used for seismic qualification.

A time history analysis involves the solution of the equations of the dynamic equilibrium (Section 3.7.2.1.1.1) by means of the methods discussed in Section 3.7.2.1.1.2. In this case, the duration of motion is of sufficient length to ensure that the maximum values of response are obtained.

A response spectrum analysis involves the solution of the equations of motion (Section 3.7.2.1.1.1) by the method discussed in Section 3.7.2.1.1.3.

3.7.2.1.1.1 The Equations of Dynamic Equilibrium

Assuming that the force due to damping is proportional to velocity, the dynamic equilibrium equations for a lumped-mass, distributed stiffness system are expressed in matrix form as:

$$[M]\{\ddot{u}(t)\} + [C]\{\dot{u}(t)\} + [K]\{u(t)\} = \{P(t)\} \qquad \text{(EQ. 3.7-1)}$$

where:

$u(t)$	=	time-dependent displacement of nonsupport points relative to the supports
$\dot{u}(t)$	=	time-dependent velocity of nonsupport points relative to the supports
$\ddot{u}(t)$	=	time-dependent acceleration of nonsupport points relative to the supports

[M]	=	diagonal matrix of lumped masses
[C]	=	damping matrix
[K]	=	stiffness matrix
P(t)	=	time-dependent inertial forces acting at nonsupport points

3.7.2.1.1.2 <u>Solution of the Equations of Motion by Mode - Superposition</u>

The first technique used for the solution of the equations of motion is the method of mode-superposition.

The set of homogeneous equations represented by the undamped free vibration of the system is:

$$[M]\{\ddot{u}(t)\} + [K]\{u(t)\} = \{0\} \quad \text{(EQ. 3.7-2)}$$

Since the free oscillations are assumed to be harmonic the displacements can be written as

$$\{u(t)\} = \{\phi\}\, e \quad \text{(EQ. 3.7-3)}$$

where:

$\{\phi\}$	=	column matrix of the amplitude of displacement {u}
ω	=	circular frequency of oscillation
t	=	time

Substituting Equation 3.7-3 and its derivatives into Equation 3.7-2 and noting that (e) is not necessarily zero for all values of (ωt) yields

$$[-\omega^2[M] + [K]]\{\phi\} = \{0\} \quad \text{(EQ. 3.7-4)}$$

Equation 3.7-4 is the classical algebraic eigenvalue problem, wherein the eigenvalues are the frequencies of vibrations, w_i, and the eigenvectors are the mode shapes, $\{\phi\}_i$.

For each frequency here is a corresponding solution vector $\{\phi\}_i$. It can be shown that the mode shape vectors are orthogonal with respect to the weighting matrix [K] in the n-dimensional vector space.

The mode shape vectors are also orthogonal with respect to the mass matrix [M].

The orthogonality of the mode shapes is used to effect a coordinate transformation of the displacements, velocities, and accelerations, so that the response in each mode is independent of the response of the system in any other mode. Thus, the problem becomes one of solving (n) independent differential equations rather than (n) simultaneous differential equations; and, since the system is linear, the principle of superposition holds, and the total response of the system oscillating simultaneously in (n) modes is determined by direct addition of the responses in the individual modes.

3.7.2.1.1.3 Analysis by Response Spectrum

As an alternative to the step-by-step mode-superposition method described in Section 3.7.2.1.1.2, the response spectrum method may be used. The response spectrum method is based on the fact that the modal responses can be expressed as a set of integral equations, rather than a set of differential equations. The advantage of this form of solution is that for a given ground motion, the only variables under the integral are the damping factor and the frequency. Thus, for a specified damping factor, it is possible to construct a curve which gives a maximum value of the integral as a function of frequency. This curve is called a response spectrum for the particular input motion and the specified damping factor. The integral has units of velocity, consequently the maximum of the integral is called the spectral velocity.

Using the calculated natural frequencies of vibration of the system, the maximum values of the modal responses are determined directly from the appropriate response spectrum. The modal maxima are then combined as discussed in Section 3.7.3.7.1.

The total seismic structural response is predicted by combining the response calculated from the two horizontal and the vertical analyses. When the response spectrum method is used, the methods for combining the loads from the three analyses is based on the method described in Section 3.7.2.6.

3.7.2.1.1.4 Support Displacements in Multisupported Structure

The methods described in Section 3.7.2.1.1.5 are used, where applicable, to account for the effects of relative anchor motion in the case of a multisupported structure.

3.7.2.1.1.5 Dynamic Analysis of Seismic Category I Systems and Components

The time history technique and the response spectrum technique were used as applicable for the dynamic analysis of seismic Category I NSSS systems and components which are sensitive to dynamic seismic events.

a. Dynamic Analysis of Piping Systems

Each pipeline is idealized as a mathematical model consisting of lumped masses connected by elastic members. The stiffness matrix for the piping system is determined using the elastic properties of the pipe. This includes the effects of torsional, bending, shear, and axial deformations, as well as change in stiffness due to curved members. Next the mode shapes and the undamped natural frequencies are obtained. The dynamic response of the system is calculated by using the response spectrum method of analysis. When the piping system is being anchored and supported at points with different excitations, the response spectrum analysis is performed using a response spectrum which envelopes all the response spectra of anchors and supports. Alternatively, except as restricted by Section 3.7.1.3.3, the multiple response spectra/independent support motion method of analysis may be used where distinct time histories or response spectra are applied to all piping system attachment points.

The relative modal displacement between anchors is determined from the dynamic analysis of the primary structures. The results of the relative anchor point displacement per mode are used for a static analysis to determine the secondary stresses due to relative anchor point displacements. The modal stresses are combined by the SRSS method to obtain the resultant secondary stresses.

b. Dynamic Analysis of Equipment

Equipment is idealized as a mathematical model consisting of lumped masses connected by elastic members or springs. The Table 3.9-6 series documents calculated and allowable loads for various loading combinations, including seismic loads, applied to major NSSS systems and components.

The response spectrum analysis is performed using the enveloped response spectrum of all attachment points. Alternatively, for equipment supported at two or more points located at different elevations in the same primary structure, the response spectrum for the most severe single support can be applied uniformly to all support points. As a second alternative, if appropriate, the response spectra at the elevation near the center of gravity of the equipment may be taken as the design spectra.

The relative modal displacement between supports is determined from the dynamic analysis of the structure. The relative support point displacements per mode are used for a static analysis to determine the secondary stresses due to support displacements. The modal stresses are combined by the SRSS method to obtain the resultant secondary stresses. Further details are given below.

c. Differential Seismic Movement of Interconnected Components

The procedure for considering differential displacements for equipment anchored and supported at points with different displacement excitations is as follows:

The relative displacements between the supporting points induces additional stresses in the equipment supported at these points. These stresses can be evaluated by performing a static analysis where each of the supporting points is displaced a prescribed amount. From the dynamic analysis of the complete structure, the time history of displacement at each supporting point is available. These displacements are used to calculate stresses by determining the peak modal responses. The stresses thus obtained for each natural mode are then superimposed for all modal displacements of the structure by the SRSS method.

In the static calculation of the stresses due to relative displacements in the response spectrum method, the maximum value of the modal displacement is used. Therefore, the mathematical model of the equipment is subjected to the maximum displacement at its supporting points obtained from the modal displacements. This procedure is repeated for the significant modes (modes contributing most to the total displacement response at the supporting point) of the structure. The total stresses due to relative displacement are obtained by combining the modal results using the SRSS method. Since the maximum displacements for different modes do not occur at the same time, the SRSS method is a realistic and practical method.

When a component is covered by the ASME B&PV Code, then the stresses due to relative displacement as obtained above are treated as secondary stresses.

3.7.2.1.1.6 Seismic Qualification by Testing

For certain seismic Category I equipment and components where dynamic testing is necessary to ensure functional integrity, test performance data and results reflect the following:

a. Performance data of equipment which, under the specified conditions, has been subjected to dynamic loads equal to or greater than those to be experienced under the specified seismic conditions

b. Test data from previously tested comparable equipment which, under similar conditions, has been subjected to dynamic loads equal to or greater than those specified

c. Actual testing of equipment in accordance with one of the methods described in Sections 3.9 and 3.10

Alternate test procedures that satisfy the requirements of these criteria are allowed, subject to review by the responsible engineer.

3.7.2.1.2 Seismic Analysis Methods (Non-NSSS)

In the analysis of seismic Category I structures, two distinct objectives must be satisfied:

a. Development of in-structure seismic response characteristics, where necessary, for use in the analysis and design of seismic Category I systems, equipment, and components

b. Determination of seismic force distribution within the various structures resulting from the design criteria free-field seismic input, for use in the design of seismic Category I structures

Two analytical procedures were employed to determine the seismic responses to the Category I structures. In general, a modal response spectrum analysis was used to compute the in-structure seismic responses, including nodal accelerations, nodal displacements, and member forces. Alternatively, a time history analysis procedure was used to generate the in-structure seismic responses discussed above. In addition, time history analysis was used to generate all floor response spectra. The mathematical idealization of the structural characteristics of the various seismic Category I structures was accomplished by a lumped-parameter beam-stick model. The general analytical methods and modeling techniques used in these analyses are in accordance with Reference 3.7-2. The seismic design criteria input is defined in terms of the OBE and SSE design response spectra (Section 3.7.1.1), the synthetic time history (Section 3.7.1.2), and the soil-structure interaction parameters (Section 3.7.2.4) used for development of floor response spectra for equipment assessment. Refer to Figures 3.7-10 through 3.7-19 for either a pictorial representation or an actual sketch of the mathematical models used. A complete description of the formulation of the mathematical models and their use is provided in Section 3.7.2.3.2.

3.7.2.2 Natural Frequencies and Response Loads

The natural frequencies of the primary containment, the reactor enclosure, and the control structure below 33 cps are shown in Tables 3.7-5 and 3.7-6 respectively. The significant mode shapes of the containment and the reactor enclosure and control structure are shown on Figures 3.7-20 through 3.7-30. The mode shapes for the primary containment are for the horizontal and vertical directions. The reactor enclosure and control structure mode shapes are for each of the three principal directions: east-west, north-south, and vertical.

Tables 3.7-7 through 3.7-16 show the response (i.e., displacements, accelerations, shear forces, bending moments, and axial forces) of the primary containment and the reactor enclosure and control structure for both OBE and SSE.

Response spectra at critical locations are shown on Figures 3.7-31 through 3.7-40. The curves are shown for each of the principal directions at the damping values shown (Section 3.7.2.15 for further discussion of damping values). Figures 3.7-41 and 3.7-42 represent the response spectra of the refueling area using soil-structure interaction.

3.7.2.3 Procedures Used for Modeling

3.7.2.3.1 Procedures Used for Modeling (NSSS)

3.7.2.3.1.1 Modeling Techniques for Seismic Category I Systems and Components

An important step in the seismic analysis of seismic Category I NSSS systems and components is the procedure used for modeling. The techniques currently being used are represented by lumped masses and a set of spring-dashpots idealizing both the inertial and stiffness properties of the system. The details of the mathematical models are determined by the complexity of the actual structures and the information required for the analysis.

3.7.2.3.1.2 Modeling of Reactor Pressure Vessel and Internals

The seismic loads on the RPV and internals are based on a dynamic analysis of an entire RPV enclosure complex, with the appropriate forcing function applied at ground level. For this analysis, the models shown in Figure 3.7-19 and the mathematical model of the enclosure are coupled together.

This mathematical model consists of lumped masses connected by elastic (linear) members. The stiffness properties of the model are determined using the elastic properties of the structural components. This includes the effects of both bending and shear. In order to facilitate hydrodynamic mass calculations, several mass points (fuel, shroud, vessel) are selected at the same elevation. The various lengths of CRD housings are grouped into the two representative lengths as shown in Figure 3.7-19. These lengths represent the longest and shortest housings in order to adequately represent the full range of frequency response of the housings. The high fundamental natural frequencies of the CRD housings result in very small seismic loads. Furthermore, the small frequency differences between the various housings, due to the length differences, result in negligible differences in dynamic response. Hence, the modeling of intermediate length members becomes unnecessary. Not included in the mathematical model are the stiffnesses of light components such as jet pumps, incore guide tubes and housings, spargers, and their supply headers. This is done to reduce the complexity of the dynamic model. To find seismic responses of these components, the floor response spectra generated from the system analysis are used.

The presence of fluid and other structural components (e.g., fuel within the RPV) introduces a dynamic coupling effect. Dynamic effects of water enclosed by the RPV are accounted for by introduction of a hydrodynamic mass matrix, which serves to link the acceleration terms of the equations of motion of points at the same elevation in concentric cylinders with a fluid entrapped in the annulus. The details of the hydrodynamic mass derivation are given in Reference 3.7-5. The seismic model of the RPV and internals has two generalized coordinates in the horizontal directions for each mass point considered in the analysis. The remaining generalized vertical coordinate is excluded because the vertical mode frequencies of RPV and internals are well above the significant horizontal mode frequencies. A separate vertical analysis is performed. The two rotational coordinates about each node point are excluded because the moment contribution from rotary inertia is negligible. Since all deflections are assumed to be within the elastic range, the rigidity of some components may be accounted for by equivalent linear springs.

The shroud support plate is loaded in its own plane during a seismic event and hence is extremely stiff, and, therefore, may be modeled as a rigid link in the translational direction. The shroud support legs and the local flexibilities of the vessel and shroud contribute to the rotational flexibilities and are modeled as an equivalent torsional spring.

3.7.2.3.2 Procedures Used for Modeling (Non-NSSS)

The mathematical models used for the analysis of all major Category I structures are fixed base, lumped-mass, elastic spring models, except for the primary containment, the reactor enclosure and control structure, where the lumped-mass seismic model considers a rigid mass resting on soil springs. The soil spring constants are determined from formulae given in table 3-2 of Reference 3.7-2. Two separate and independent analyses are done for the containment structure. The section properties for these two analyses are based on uncracked and cracked concrete sections, respectively. The same models are used both for the response spectrum and time history analyses. The mathematical models of the primary containment are shown in Figures 3.7-10 and 3.7-11, and those of the reactor enclosure and control structure are shown in Figures 3.7-14 and 3.7-15. The additional mathematical models used for development of floor response spectra, using soil-structure interaction for these structures, are shown in Figures 3.7-12, 3.7-13, 3.7-16, and 3.7-17.

For all models, the masses are located at elevations of mass concentrations, such as floors and roofs. However, in the case of the containment, a structure of continuous mass distribution, masses are lumped at approximate maximum intervals of 15 feet along the containment shell and reactor pedestal. These methods of mass distribution are in accordance with the procedures of section 3.2 of Reference 3.7-2. All equipment, components, and piping systems are lumped into the supporting structure except for the reactor vessel, which is incorporated into the lumped-parameter analysis of the containment structure to account for potential coupling. The detailed analysis of the NSSS is performed using a decoupled model as discussed in Section 3.7.2.3.1.

A more refined RPV model is being used in the DAR (Appendix 3A) to evaluate the anticipated effects of hydrodynamic loads in combination with seismic loads.

3.7.2.4 Soil-Structure Interaction

Since the seismic Category I structures are founded on competent bedrock, a soil spring approach to characterize soil-structure interaction is not used in the dynamic analysis. A simplified lumped-mass method using a fixed base model is used. However, for a more refined analysis of containment and reactor enclosure, the underlying foundation medium is considered to interact with the structure. The equivalent soil spring constant and damping coefficient are computed in accordance with the formulae of table 3-2 of Reference 3.7-2, and the analysis carried out by the methods discussed in appendix D of Reference 3.7-2. The resulting structure-foundation interaction coefficients are listed in Table 3.7-17.

3.7.2.5 Development of Floor Response Spectra

3.7.2.5.1 Floor Response Spectra (NSSS)

See Section 3.7.3.6.1.

3.7.2.5.2 Floor Response Spectra (Non-NSSS)

The time history method of analysis was used to develop the floor response spectra. A discussion of the technique of finding the nodal time history and then producing the spectrum may be found in Sections 4.2 and 5.2 of Ref 3.7-2.

3.7.2.6 Three Components of Earthquake Motion

3.7.2.6.1 NSSS

See Section 3.7.3.6.1.

3.7.2.6.2 Non-NSSS

The response spectrum method was used in seismic analysis of structures. Independent analyses are performed for the vertical and two horizontal (east-west and north-south) directions. Regulatory Guide 1.92, "Combining Responses and Spatial Components in Seismic Response Analysis" states that the design response value is obtained by taking the SRSS of the maximum co-directional responses caused by each of the three components of earthquake motion at a particular point of the structure. However, for design purposes, the response value used is the maximum value obtained by adding the response due to the vertical earthquake with the larger value of the response due to one of the horizontal earthquakes by the absolute sum method. A discussion of the adequacy of the conservatism provided by this two-component absolute summation technique as compared to the Regulatory Guide 1.92 requirement of a three-component SRSS procedure is provided below.

The general conditions for which the absolute summation of two resultants method is conservative may be demonstrated by considering the following:

$$R_H^{min} = fR_H^{max} \qquad \text{where } 0 \leq f \leq 1 \text{ where } f = R_H^{min}/R_H^{max}$$

and

$$R_V = CR_H^{max} \qquad 0 \leq C < \infty \text{ where } C = R_V/R_H^{max}$$

where:

R_H^{max} = Larger of the two seismic co-directional responses due to either of the horizontal earthquake components

R_H^{min} = Smaller seismic co-directional responses due to the other orthogonal horizontal earthquake component

R_V = Seismic co-directional response due to vertical earthquake

f = ratio of R_H^{min} to R_H^{max}

C = ratio of R_V to R_H^{max}.

Therefore SRSS of three components can be expressed as follows:

$$\sqrt{(R_H^{max})^2 + (R_H^{min})^2 = (R_V)^2}$$

$$\sqrt{(R_H^{max})^2 + f^2 \ (R_H^{max})^2 + C^2 \ (R_H^{max})^2}$$

$$\left(\sqrt{1 + f^2 + C^2} \quad R_H^{max} \right) \qquad \text{(EQ. 3.7-5)}$$

The absolute sum of two components can be expressed as follows:

$$R_H^{max} + R_V$$

$$R_H^{max} + CR_H^{max}$$

$$(\sqrt{(1 + C)^2}\ ) R_H^{max}$$

$$(\sqrt{1 + 2C + C^2}\ ) R_H^{max} \qquad \text{(EQ. 3.7-6)}$$

Equation 3.7-6 (absolute sum of two components) will be greater than or equal to Equation 3.7-5 (three-component SRSS) when:

$$\sqrt{1 + 2C + C^2} \geq \sqrt{1 + f^2 + C^2}$$

or

$$1 + 2C + C^2 \geq 1 + f^2 + C^2$$

or

$$2C \geq f^2$$

or $C \geq ½ f^2$ where $0 \leq f = \dfrac{R_H^{min}}{R_H^{max}} \leq 1$ (EQ. 3.7-7)

This relationship is shown in Figure 3.7-46.

The relative conservatism of the two-component absolute and the three-component SRSS summation techniques may be illustrated by considering the ratio of Equations 3.7-6 and 3.7-5. This ratio will be defined as y. Then:

$$y \quad = \quad \frac{\text{ABS of EQ. 3.7-6}}{\text{SRSS of EQ. 3.7-5}} \quad = \quad \frac{\left(\sqrt{1 + 2C + C^2}\right) R_H^{max}}{\left(\sqrt{1 + f^2 + C^2}\right) R_H^{max}}$$

$$y \quad = \quad \left(\frac{1 + 2C + C^2}{1 + f^2 + C^2}\right)^{1/2} \qquad \text{(EQ. 3.7-8)}$$

where:

$$0 \le f = \frac{R_H^{min}}{R_H^{max}} \le 1, \text{ as before}$$

This relationship is shown in Figure 3.7-47.

As shown above, the LGS two-component absolute summation method is conservative when the co-directional response due to the vertical excitation is equal to or greater than one-half the higher of the two horizontal responses ($C \ge ½$), regardless of the relationship between the two horizontal responses.

The minimum possible ratio between the two-component absolute summation method and the three-component SRSS procedure is equal to 0.707. This would occur only when the response due to the vertical excitation is zero ($C = 0$) and the two horizontal responses resulting from the two horizontal excitations are equal ($f = 1$). However, this case is unlikely to occur, and any other relationship of the various response components would produce a ratio larger than 0.707. The ratio between the two procedures, as shown in Figure 3.7-47 would be greater than one in most cases.

To further demonstrate the relationship of the seismic response components on the LGS structures, an evaluation has been performed for selected critical structural elements within the structures. Details of the evaluation are provided in the following paragraphs.

a. Containment Exterior Shell

For the structural design of the containment shell, consideration of two horizontal components is not necessary due to the axisymmetric nature of the shell. The maximum resulting loads from two horizontal earthquake components would not be coincident and would occur 90° apart on the circumference of the shell. Furthermore, when the resultant force from one horizontal component is maximum at a given location, the resultant force from the orthogonal horizontal component would be zero. This relationship corresponds to $f = 0$ in Figures 3.7-46 and 3.7-47. Therefore, the two-component absolute summation technique would produce a more conservative design.

b. Reactor Enclosure and Control Structure

Stress evaluations were performed for critical locations in the reactor enclosure and control structure. The northeast control structure and southwest reactor enclosure corner model locations are selected because of their sensitivity to large orthogonal responses due to out-of-plane seismic motion. The results obtained from the SSE seismic analysis are combined by the two-component ABS and three-component SRSS methods and are compared in Table 3.7-20. For 14 out of 20 wall locations evaluated, the seismic stress response comparison shows that the two-component ABS method is more conservative than the three-component SRSS method. In general, the two-component ABS method and the three-component SRSS method produce comparable results. Although the differences between the combined stress from the two methods are small, the values for some locations suggest that the two-component ABS method may be conservative when the seismic load is considered to act alone.

There are 6 wall locations where the two-component ABS method is less conservative. The critical wall location is the southwest corner of the reactor enclosure, at el 177', where the differential between ABS stresses and SRSS stresses equals 14% (Table 3.7-20). (This location is considered critical not only because of the difference between the ABS and SRSS stresses but also because of the level of the stresses.) However, when the seismic stresses are combined with the stresses due to other design loads, the effect of the differential between ABS stresses and SRSS stresses is reduced in proportion with the ratio of seismic wall load to total wall load. In addition, the critical wall (wall line "D" at el 177') is loaded to only 52% of its capacity (based on combined axial and bending stresses due to all design loads as shown in Figure 3A-432).

Therefore, when considering the reduced effect of ABS vs. SRSS in conjunction with the ample excess capacity of the wall, the use of the ABS method for calculating seismic stresses provides adequate conservatism when considered in combination with other loads in the design of the reactor enclosure and control structure.

3.7.2.7 Combination of Modal Responses

3.7.2.7.1 Combination of Modal Responses (NSSS)

See Section 3.7.3.7.1.

3.7.2.7.2 Combination of Modal Responses (Non-NSSS)

The modal responses (i.e., shears, moments, deflections, accelerations, and inertia forces) are combined by either the sum of the ABS method, or by the SRSS method with consideration of closely spaced modes. Two consecutive modes are defined as closely spaced when their frequencies differ from each other by 10% or less of the lower frequency. When the SRSS method is used, Regulatory Guide 1.92 shall be adopted for the combination of modal responses.

3.7.2.8 Interaction of Non-Category I Structures with Seismic Category I Structures

The turbine enclosure, auxiliary boiler enclosure, and the administration building are the only major non-Category I structures adjacent to seismic Category I structures. These non-Category I structures are designed for seismic loading in accordance with the UBC (Reference 3.7-3). In addition, the turbine enclosure was dynamically analyzed to ensure the capacity to withstand a SSE without collapsing on or impairing the integrity of the adjacent reactor and control structures. Similarly, the other non-Category I structures were analytically evaluated to ensure that they will not collapse on or otherwise impair the integrity of adjacent seismic Category I structures when subjected to the design seismic loads.

Structural separations have been provided to ensure that interaction between Category I and non-Category I structures does not occur. The minimum separation gap between the buildings is twice the relative displacement for the design seismic loads.

3.7.2.9 Effects of Parameter Variations on Floor Response Spectra

3.7.2.9.1 Effects of Parameter Variations on Floor Response Spectra (NSSS)

To account for potential variations in the primary structure frequencies due to uncertainties in material properties of the soil and structure, soil-structure interaction techniques, approximation in damping, and approximation in dynamic modeling, the computed floor response spectra are peak-broadened by ±15%. This is consistent with the requirements of Regulatory Guide 1.122, although this regulatory guide is not the design basis requirement for the LGS construction permit.

3.7.2.9.2 Effects of Parameter Variations on Floor Response Spectra (Non-NSSS)

To account for variations in the structural frequencies owing to uncertainties in the material properties of the structure and to approximations in the modeling techniques used in the seismic analysis, the computed floor response spectra are smoothed, and peaks associated with each of the structural frequencies are broadened. In lieu of making a parametric study considering changes in the material properties and other variables, the spectrum is broadened on either side of the peak value by 15% of the frequency at which the peaks occur.

3.7.2.10 Use of Constant Vertical Static Factors

Vertical seismic system multimass dynamic models are used to obtain vertical response loads for the seismic design of seismic Category I structures. Therefore, constant vertical static factors are not used to account for vertical response to earthquakes for the seismic design of Category I structures.

3.7.2.11 Methods Used to Account for Torsional Effects

Torsional effects for the reactor enclosure, diesel generator enclosure, spray pond pumphouse, and radwaste enclosure are accounted for as follows:

A static analysis is performed to account for torsion on these structures. The eccentricity is determined using the distance between the center of mass and the center of rigidity of the individual structure. The inertial force from the response spectrum analysis is applied at the center of mass. The resulting torsional moment is equal to the inertial force times the eccentricity. The shear forces due to the torsional moment are then distributed to the walls. The total seismic shear in a given structural element is equal to the sum of the absolute values of the shear due to translation and torsion. Torsional effects are negligible for the containment because of the symmetry of the structure about the vertical center line.

Dynamic analyses have been performed to verify the adequacy of the static-equivalent method described above. These dynamic analyses were performed for the reactor enclosure, diesel generator enclosure and spray pond pumphouse where structural eccentricities are larger than 5% of the larger building plan dimension.

Inertia forces, obtained from dynamic analysis, are multiplied by the mass eccentricity (distance between center of mass and center of rigidity) to obtain a torsional moment, which is distributed to structural walls for design assessment. This equivalent static method was found to produce conservative design torsional moments; more so than results obtained from a 3-D dynamic analysis which included structural eccentricities. The dynamic analysis was performed as follows:

a. 3-D horizontal stick models were constructed for the reactor enclosure, diesel generator enclosure, and spray pond pumphouse with structural masses at calculated eccentricities (Figures 3.7-48, 3.7-49, and 3.7-50). The direction that has the larger mass eccentricity was selected for dynamic response analysis.

b. Modal analysis was performed to determine the vibration frequencies of the structures with eccentricity accounted for. The vibration frequencies with eccentricity were compared to the vibration frequencies with zero eccentricity (Tables 3.7-21 through 3.7-23).

c. Time history analyses were performed for the east-west models of the spray pond pumphouse, diesel generator enclosure, and reactor enclosure.

Torsional moments and shear forces at the base slab were compared with the moments and shear forces of the original design obtained using the equivalent static approach (Tables 3.7-24 through 3.7-27) .

Torsional moment and shear data comparisons show that the original design values using the static approach are conservative.

3.7.2.12 Comparison of Responses

A comparison of the results of modal design response spectrum analysis and modal time history analysis for the containment is shown in Figures 3.7-4 through 3.7-9. The responses are calculated for both OBE and SSE. The uncracked containment model as described in Section 3.7.2.3.2 is used for the analyses. The responses due to synthetic time history input are lower than those due to response spectrum, because the sum of the absolute modal responses are used in the latter analysis, whereas the phase differences of the modal responses are considered in the time history analysis.

3.7.2.13 Methods for Seismic Analysis of Dams

Dams are not provided on LGS.

3.7.2.14 Determination of Seismic Category I Structure Overturning Moments

The overturning moment for seismic Category I structures is the absolute sum of the moments at the base of each stick of the mathematical model. For each stick, the moment at the base is determined by combining the modal overturning moments. The modal moments are combined by the methods described in Section 3.7.2.7.

The components of the earthquake motion used are the same as those discussed in Section 3.7.2.6. Section 3.8.5 discusses the factor of safety against overturning for several loadings, including seismic loads.

3.7.2.15 Analysis Procedure for Damping

3.7.2.15.1 Analysis Procedure for Damping (NSSS)

In a linear dynamic analysis, the procedure to be utilized to properly account for damping in different elements of a coupled system model is as follows:

a. The values for structural damping of the various structural elements of the model are first specified. Each value is referred to as the damping ratio (B_j) of a particular component which contributes to the complete stiffness of the system.

b. Perform a modal analysis of the linear system model. This will result in a modal matrix (ϕ) normalized so that:

$$\phi_i^T K \phi_i = W_i^2$$

where:

K	=	the stiffness matrix
W_i	=	circular natural frequency of mode i
ϕ_i^T	=	the transpose (ϕ) which is a column vector of (ϕ) corresponding to the mode shape of mode i

Matrix (ϕ) contains all translational and rotational coordinates.

c. Using the strain energy of the individual components as a weighting function, the following equation can be derived to obtain a suitable damping ratio (B_i) for the ith mode.

$$B_i \quad \frac{\sum_{j=1}^{N} [\phi_i^T \beta_j K_j \phi_i]}{W_i^2} \qquad \text{(EQ. 3.7-9)}$$

where:

N	=	Total number of structural elements
ϕ_i	=	Mode shape for mode i (ϕ_i^T as transpose)
β_i	=	Percent damping associated with element j

K_j = Stiffness matrix of element j

W_i = Circular natural frequency of mode i

3.7.2.15.2 Analysis Procedure for Damping (Non-NSSS)

The structures consist of reinforced concrete and welded/bolted structural steel. Damping values for these materials are shown in Table 3.7-2. However, in the seismic analysis of structures supported on rock, damping values of 2% and 5% are used for OBE and SSE respectively for welded/bolted structural steel, as well as for reinforced concrete. Therefore, the analysis by composite modal damping is not necessary.

For the time history analysis of containment and reactor enclosure where the soil-structure interaction is taken into account, the composite modal damping technique was used as described in appendix D of Reference 3.7-2.

3.7.3 SEISMIC SUBSYSTEM ANALYSIS

This section discusses the seismic analysis of subsystems, i.e., equipment, piping, Class 1E cable trays, and supports for seismic Category I HVAC ducts and cable trays.

3.7.3.1 Seismic Analysis Methods

3.7.3.1.1 Equipment

Seismic qualification of equipment was performed by using one of the following methods:

a. Analysis

b. Dynamic testing

c. Combination of analysis and dynamic testing

3.7.3.1.1.1 Analysis

For the purpose of analysis, equipment is idealized as a system of lumped masses and springs, for which frequencies and mode shapes are determined for vibration in the vertical direction and two orthogonal horizontal directions. For each direction of vibration, the spectral acceleration per mode is obtained from the appropriate response spectrum curve (i.e., corresponding to the location of the equipment) at the natural frequency of the equipment. Seismic loading, in terms of inertia forces, moments and shears, is determined for each direction using the response spectrum technique, summing the absolute values per mode. If the orientation of the equipment is not designated on the equipment location drawing, the horizontal seismic loading is taken as the maximum loading (worst case) obtained, using each horizontal direction of vibration and the appropriate horizontal response spectrum curve(s). If the frequencies of all equipment modes (determined by either analysis or testing) are above the frequency of the appropriate response spectrum curve at which the acceleration is constant in the rigid (high frequency) range, the seismic loading consists of static loading corresponding to that acceleration level. If the equipment damping is unknown, the response spectrum curve for 0.5% damping is used to arrive at a conservative seismic loading.

The damping value used for the OBE is increased for the SSE, where sufficient justification is established.

3.7.3.1.1.2 Dynamic Testing

In lieu of performing dynamic analysis, seismic adequacy may be established by providing dynamic test or previous dynamic environmental (performance) data which demonstrate that the equipment meets the seismic design criteria. The data include at least one of the following:

a. Recent test data acquired from dynamic tests of equipment

b. Dynamic test data from previously tested comparable equipment

c. Performance data from equipment which, during normal operating conditions, have been subjected to dynamic loads equal to or greater than those defined in Section 3.7.3.1.1.1

Typical test methods used are as follows:

a. Single-frequency sine beat test

b. Single-frequency dwell test

c. Multifrequency test

3.7.3.1.1.3 Combination of Analysis and Dynamic Testing

Certain equipment was qualified by a combination of analysis and dynamic testing. Experimental methods are used to aid in the formulation of the mathematical model for the equipment. Mode shapes and frequencies are determined experimentally and incorporated in the mathematical model of the equipment. The model is then subsequently analyzed by the procedure described in Section 3.7.3.1.1.1.

3.7.3.1.2 Piping Systems

Reference 3.7-4 describes the methods used for seismic analysis of piping systems. Reference 3.7-4 is followed on LGS with the following exceptions:

In seismic analysis the modal responses are combined by SRSS, and lower damping values than specified in Reference 3.7-4 are used. Alternative analytical methods and damping values may be utilized as described in Section 3.7.1.3.3.

See Section 3.7.3.7.2.

3.7.3.1.3 Class 1E Cable Trays

The cable trays are seismically qualified by the capacity evaluation method which consists of the following:

a. Calculation of the fundamental frequency of the cable tray based on the tray properties obtained from static tests

b. Seismic load computation based upon the tray frequency and the design spectra

c. Calculation of the tray allowable capacity

d. Evaluation of the tray capacity by interaction formula

3.7.3.1.4 Supports for Seismic Category I HVAC Ducts

The supports for HVAC ducts are analyzed by the response spectrum method (Reference 3.7-2).

3.7.3.1.5 Supports for Seismic Category I Electrical Raceway Systems

This section defines the procedures used for the design of the supports of electrical raceway systems, i.e., cable tray, conduit, and wireway gutter systems, subject to the seismic and other applicable loads. The raceway support system usually consists of raceways, horizontal and vertical support members, and lateral and longitudinal bracing members.

3.7.3.1.5.1 Loading Combinations

The adequacy of raceway systems to withstand seismic and other applicable static loads is determined according to the loading combinations and allowable responses given in Table 3.7-3. Load combinations and allowable stresses including the effects of hydrodynamic loads due to LOCA and SRV discharge are given in Table 3A-21.

3.7.3.1.5.2 Analytical Techniques

Either of two methods of analysis is used. Method 1 is a simplified method of analysis which determine the fundamental frequency of braced supports using two-dimensional analysis. Frequencies are determined in each of three principal directions. Then loads are determined by taking the spectral accelerations times the weight; and stresses are determined from static analysis. All members and connections are checked using stress criteria.

Method 2 uses a three-dimensional computer analysis and includes springs to represent joint stiffnesses. Response spectrum analyses are done to determine stresses and deformations. The number of stress cycles is determined by multiplying the time of maximum earthquake motion by the natural frequency of the system.

The allowable number of cycles is taken from Reference 3.7-8 for the joint rotations calculated. Only overhead connections are checked for fatigue because the test results (Reference 3.7-8, p. 7-19) demonstrate that failures occur only in overhead connections.

The basis for the design criteria and analysis Method 2 is the "Cable Tray and Conduit Raceway Test Program" (References 3.7-7 through 3.7-10).

3.7.3.1.5.3 Damping

Damping of 10% of the critical is used for the design of cable tray support systems; 7% damping for conduit and wireway gutter trapeze-type support systems; 5% damping for conduit and wireway gutter nontrapeze-type support systems. The recommended damping values for cable tray and conduit systems, developed from the test program, are shown in Figure 3.7-45. Wireway gutters were not tested: however, the manner in which they are constructed (with more bolted connections and more cables than conduit) provides more damping mechanisms than are present in conduit systems so that using the same damping value as in conduit systems is conservative.

3.7.3.1.5.4 Operating Basis Earthquake

The OBE is considered in the load combinations only for the overhead connections which are checked for fatigue. The OBE stresses are not checked during design for two reasons: first, raceway systems do not fail in a brittle or catastrophic mode as demonstrated by the test program in which such failures did not occur and the electrical systems were able to continue to function in all cases. Thus, there is no need to limit the OBE stresses to the low levels usually used to preclude such failures. Second, the OBE stresses will always be less than the SSE stresses as demonstrated below.

Based on Figure 3.7-45, a comparison of response spectra for corresponding damping values specified in Section 3.7.3.1.5.3 demonstrates that for all response spectra the OBE acceleration values are less than the corresponding SSE acceleration values (References 3.7-8 and 3.7-10). Thus, the OBE acceleration response and stresses are below the SSE acceleration response and stresses.

3.7.3.2 Determination of Number of Earthquake Cycles

3.7.3.2.1 Determination of Number of Earthquake Cycles (NSSS)

3.7.3.2.1.1 NSSS Piping

Fifty peak OBE cycles are postulated for fatigue evaluation.

3.7.3.2.1.2 Other NSSS Equipment and Components

To evaluate the number of cycles which exist within a given earthquake, a typical BWR enclosure/reactor dynamic model was excited by three different recorded time histories: May 18, 1940, El Centro NS component 29.4 sec; 1952, Taft N 69^{o} W component, 30 sec; and March 1957, Golden Gate S 80^{o} E component, 13.2 seconds. The modal response is truncated so that the response of three different frequency bandwidths could be studied: 0-10 Hz; 10-20 Hz; and 20-50 Hz. This is done to give a good approximation to the cyclic behavior expected from structures with different frequency content.

Enveloping the results from the three earthquakes and averaging the results from several different points of the dynamic model, the cyclic behavior as given in Table 3.7-18 was formed. A comparison has shown that the LGS design basis response spectrum is bounded by the spectra of the three earthquakes (Golden Gate, Taft, and El Centro) in the GE base study (Figure 3.7-43).

Independent of earthquake or component frequency, 99.5% of the stress reversals occur below 75% of the maximum stress level, and 95% of the reversals lie below 50% of the maximum stress level.

In summary, the cyclic behavior number of fatigue cycles of a component during an earthquake was found in the following manner:

a. The fundamental frequency and peak seismic loads are found by a standard seismic analysis.

b. The number of cycles which the component experiences are found from Table 3.7-18 according to the frequency range within which the fundamental frequency lies.

c. For fatigue evaluation, 0.5% (0.005) of these cycles are conservatively assumed to be at the peak load and 4.5% (0.045) at or above three quarter peak. The remainder of the cycles have negligible contribution to fatigue usage.

The SSE has the highest level of response. However, the encounter probability of the SSE is so small that it is not necessary to postulate the possibility of more than one SSE during the 40 year life of a plant. Fatigue evaluation due to the SSE is not necessary, since it is a faulted condition, and thus the evaluation is not required by ASME Section III.

The OBE is an upset condition, and therefore, must be included in fatigue evaluations according to ASME Section III. Investigation of seismic histories for many plants show that during a 40 year life, it is probable that five earthquakes with intensities of one-tenth of the SSE intensity, and one earthquake of approximately 20% of the proposed SSE intensity, will occur. To cover the combined effects of these earthquakes and the cumulative effects of even lesser earthquakes, ten peak OBE cycles are postulated for fatigue evaluation.

Table 3.7-19 shows the calculated number of fatigue cycles and the number of fatigue cycles used in design.

3.7.3.2.2 Determination of Number of Earthquake Cycles (Non-NSSS)

In general, the design of the equipment is not fatigue controlled. For equipment qualified by analysis, fatigue is not a controlling factor because the equipment is designed to remain below 90% of the yield strength of the material for the extreme loading condition. The number of stress cycles considered is 60 (5 OBE and 1 SSE events at 10 cycles each). Based on ASME Section III, appendix I criteria (figure I-9-1), this number of cycles will not result in a reduction of allowable stresses.

Any fatigue effects in tested equipment are accounted for by the duration of the test. Consequently, the number of cycles of the earthquake is considered.

In order to conduct a fatigue evaluation for nuclear Class I piping, the number of cycles for a given load set is obtained. This is done by considering ten maximum stress cycles per earthquake and 5 OBEs and 1 SSE to occur within the life of the plant. The results of the fatigue calculations for the most limiting BWR/4 component are shown in Table 3.7-4.

3.7.3.3 Procedures Used for Modeling

3.7.3.3.1 Procedures Used for Modeling (NSSS)

3.7.3.3.1.1 Modeling of Piping Systems

The continuous piping system is modeled as an assemblage of beams. The mass of each beam is lumped at the nodes connected by weightless elastic members representing the physical properties of each segment. The pipe lengths between mass points are no greater than the length which would have a natural frequency of 33 Hz, when calculated as a simply supported beam. All concentrated weights on the piping system such as main valves, relief valves, pumps, and motors are modeled as lumped masses. The torsional effects of the valve operators and other equipment with offset centers of gravity, with respect to the center line of the pipe, is included in the analytical model. Where the torsional effect is found to cause pipe stresses less than 500 psi, this effect is neglected.

3.7.3.3.1.2 Modeling of Equipment

For dynamic analysis, seismic Category I equipment is represented by lumped-mass systems which consist of discrete masses connected by weightless springs. The criteria used to lump masses are:

a. The number of modes of a dynamic system is controlled by the number of masses used. Therefore, the number of masses is chosen so that all significant modes are included. A mode is considered significant if the corresponding natural frequency is less than 33 Hz, and the stresses calculated from this mode are greater than 10% of the total stresses obtained from lower modes.

b. Mass is lumped at any point where a significant concentrated weight is located. Examples are: the motor in the analysis of pump motor stand; the impeller in the analysis of pump shaft; etc.

c. If the equipment has a free-end overhang span whose flexibility is significant compared to the center span, a mass is lumped at the overhang span.

d. When a mass is lumped between two supports, it is located at a point where the maximum displacement is expected to occur. This tends to conservatively lower the natural frequencies of the equipment, because the natural frequencies of the equipment are generally in the higher frequency range of the floor spectra. Similarly, in the case of live loads (mobile) and a variable support stiffness, the location of the load and the magnitude of support stiffness are chosen so as to yield the lowest frequency content for the system. This is to ensure conservative dynamic loads, since equipment frequencies are higher than the frequency at which the floor spectra peak occurs. If such is not the case, the model is adjusted to give more conservative results.

3.7.3.3.1.3 Field Location of Supports and Restraints

The field location of seismic supports and restraints for seismic Category I piping and piping systems components was selected to satisfy the following two conditions:

a. The location selected must furnish the required response to control strain within allowable limits.

b. Adequate structure strength for attachment of the components must be available.

The final location of seismic supports and restraints for seismic Category I piping, piping system components, and equipment (including the placement of snubbers), was checked against the drawings and instructions issued by the engineer. An additional examination of these supports and restraining devices was made to ensure that the location and characteristics of these supports and restraining devices are consistent with the dynamic and static analyses of the systems.

3.7.3.3.2 Procedures Used for Modeling (Non-NSSS)

Mathematical models which describe the mass and the stiffness properties of the equipment are used. The models define the dynamic behavior of the equipment within the frequency range of interest. The boundary conditions are modeled to reflect the actual mounting conditions. The equipment is represented by lumped-mass models. Massless elastic members are used to connect the masses.

Supports for HVAC ducts are modeled as two-dimensional, lumped- mass, plane frame models. The masses are lumped at the center of the ducts. The electrical raceway support system analytical techniques are discussed in Section 3.7.3.1.5.2. Equivalent structural properties for cable tray analysis were determined from the load-deflection tests (Reference 3.7-11).

Sections 2 and 3 of Reference 3.7-4 discuss the techniques and procedures used to model piping other than the buried type.

3.7.3.4 Basis of Selection of Frequencies

3.7.3.4.1 Basis of Selection of Frequencies (NSSS)

All frequencies in the range of 0.25-33 Hz are considered in the analysis and testing of structures, systems, and components. These frequencies cover the natural frequencies of most of the components and structures under consideration. If the fundamental frequency of a component is ≥33 Hz, it is treated as rigid and analyzed accordingly. Frequencies less than 0.25 Hz are not considered, as they represent very flexible structures and are not encountered in this plant.

The frequency range of between 0.25 Hz and 33 Hz covers the range of the broad-band response spectrum used in the design.

3.7.3.4.2 Basis of Selection of Frequencies (Non-NSSS)

The natural frequencies of components are calculated. Only those modes which have natural frequencies <33 Hz are considered in the dynamic analysis. If a component has a frequency ≥33 Hz, it is considered as rigid. If the natural frequency of the component falls within the broadened peak of the response spectrum curve, then it is designed to take the applied load.

3.7.3.5 Use of Equivalent Static Load Method of Analysis (Non-NSSS)

The equivalent static load method is used when the natural frequency of the equipment is not determined. If the equipment can be adequately represented by a single-degree-of-freedom system, then the applied inertia load is equal to the weight of the equipment times the peak value of the response spectrum curve. Seismic acceleration coefficients for multidegree of freedom systems, which may be in the resonance region of the amplified response spectra curves, are increased by 50% to account conservatively for the increased modal participation.

Appendix D of Reference 3.7-4 discusses the use of equivalent static load method of analysis as applicable to piping.

3.7.3.6 Three Components of Earthquake Motion

3.7.3.6.1 Three Components of Earthquake Motion (NSSS)

The simultaneous use of three components of earthquake motion was not a design basis requirement of the construction permit for the LGS plant. However, the NSSS systems and components are evaluated to the requirement of Regulatory Guide 1.92.

a. Response Spectrum Method

Response spectra generated by GE are developed considering three components of earthquake motion. The individual responses in each orthogonal direction are combined by SRSS of the colinear contribution due to the three directions of earthquake motion. These are used to predict the total response at each frequency.

b. Time history Method

When the time history method of analysis is used, one of the following options is used to obtain the peak value of any particular response of interest:

1. When maximum colinear contributions due to the three directions of earthquake motion are calculated separately, the total response is obtained as the SRSS combination of the colinear values.

2. When colinear time history responses from each of the three components of the earthquake motion are calculated individually by the step-by-step method and then combined algebraically at each time step, the maximum response is obtained as the peak value from the combined time solution.

3. When a response at each time step is calculated directly based on the simultaneous application of the three earthquake components, the maximum response is determined by scanning the combined time history solution.

The components of earthquake motion must be statistically independent for Options 2 and 3. Also, the time history method precludes the need to consider closely spaced modes.

3.7.3.6.2 Three Components of Earthquake Motion (Non-NSSS)

For equipment, cable trays, and supports for cable trays and HVAC ducts, the three spatial components of the earthquake are considered in the same manner as for structures (Section 3.7.2.6).

The criteria used for combining the results of horizontal and vertical seismic responses for piping systems are described in section 5.1 of Reference 3.7-4.

3.7.3.7 Combination of Modal Responses

3.7.3.7.1 Combination of Modal Responses (NSSS)

All piping and equipment analyzed or supplied by GE are evaluated to the requirements of Regulatory Guide 1.92.

When the response spectra method of modal analysis is used, all modes except the closely spaced modes (i.e., the difference between any two natural frequencies is equal to or less than 10%) are combined by the SRSS as described in Section 3.7.3.7.1.a. Closely spaced modes are combined by the double sum method with absolute sign as described in Section 3.7.3.7.1.b.

In the time history method of dynamic analysis, the vector sum at every time step is used to calculate the combined response. The use of the time history method precludes the need to consider modal spacing.

a. SRSS

The SRSS method is defined mathematically as:

$$R = \left[\sum_{i=1}^{n} (R_i)^2 \right]^{1/2} \qquad \text{(EQ. 3.7-10)}$$

where:

R = Combined response

R_i = Response in the (i) mode

n = Number of modes considered in the analysis

b. Procedure of Combining Closely Spaced Modal Response

This method is defined mathematically as:

$$R = \left[\sum_{k=1}^{N} \sum_{s=1}^{N} \left| R_k R_s \right| E_{ks} \right]^{1/2} \qquad \text{(EQ. 3.7-11)}$$

where:

R = representative maximum values of a particular response of a given element to a given component of excitation

R_k = peak value of the response of the element due to the kth mode

N = number of significant modes considered in the modal response combination

R_s = peak value of the response of the element attributed to the sth mode also:

$$E_{ks} = 1 + \left[\frac{W_k^1 - W_s^1}{\beta_k^1 w_k + \beta_s^1 w_s} \right]^2 \quad \text{(EQ. 3.7-12)}$$

in which:

$$w_k^1 = w_k (1 - \beta_k^2)^{1/2} \quad \text{and}$$

$$\beta_k^1 = \beta_k + \frac{2}{t_d w_k}$$

where:

w_k = modal frequency

β_k = damping ratio in the kth mode

t_d = duration of the earthquake.

3.7.3.7.2 Combination of Modal Responses (Non-NSSS)

The modal responses of equipment are combined by the SRSS method. The absolute values of two closely spaced modes are added first before combining with the other modes by the SRSS method. Two consecutive modes are defined as closely spaced when their frequencies differ from each other by 10% or less.

Procedures given in Regulatory Guide 1.92 for combining modal responses, when closely spaced modes are present, are not complied with in the seismic response spectra analysis for piping. All modal responses are combined by SRSS in the response spectra method of modal analysis for seismic loading (OBE and SSE). Seismic response spectra used in the piping analysis corresponds to conservative damping values of ½% for OBE and 1% for SSE. Alternative analytical methods and damping values may be utilized as described in Section 3.7.1.3.3.

3.7.3.8 Analytical Procedure for Piping

3.7.3.8.1 Analytical Procedure for Piping (NSSS)

The analytical procedures for piping analysis have been described in Section 3.7.2.1.1.5.a. Methods to include differential piping support movements at different support points are also described there.

3.7.3.8.2 Analytical Procedure for Piping (Non-NSSS)

The design criteria and the analytical procedures applicable to piping systems are as described in section 2 of Reference 3.7-4. Alternatively, except as restricted by Section 3.7.1.3.3, the multiple response spectra/independent support motion method of analysis may be used where distinct response spectra are applied to the piping system attachment points. The methods used to consider differential piping support movements at different support points are described in section 4 of Reference 3.7-4.

3.7.3.9 Multiple Supported Equipment Components with Distinct Inputs

3.7.3.9.1 Multiple Supported Equipment Components with Distinct Inputs (NSSS)

The procedure and criteria for analysis has been described in Section 3.7.2.1.1.5.b.

3.7.3.9.2 Multiple Supported Equipment Components with Distinct Inputs (Non-NSSS)

For cable trays and HVAC ducts whose supports have two or more distinct inputs, a response spectrum curve envelopes the curves at all support locations. When a piping system whose supports have two or more distinct inputs, a response spectrum curve may be used for analysis that envelopes the curves at all locations. Alternatively, except as restricted by Section 3.7.1.3.3, the multiple response spectra/independent support motion method of analysis may be used where distinct response spectra are applied to the piping system attachment points. Section 4 of Reference 3.7-4 discusses the methods used for the analysis of multiple supported piping due to differential seismic anchor movement.

3.7.3.10 Use of Constant Vertical Static Factors

3.7.3.10.1 Use of Constant Vertical Static Factors (NSSS)

Constant vertical static factors are not used for NSSS components.

3.7.3.10.2 Use of Constant Vertical Static Factors (Non-NSSS)

Constant vertical static factors are not used in the seismic design of subsystems.

3.7.3.11 Torsional Effects of Eccentric Masses

3.7.3.11.1 Torsional Effects of Eccentric Masses (NSSS)

Torsional effects of eccentric masses is discussed in Section 3.7.3.3.1.1.

3.7.3.11.2 Torsional Effects of Eccentric Masses (Non-NSSS)

The torsional effects of valves and other eccentric masses are considered in the seismic analysis of piping by the techniques discussed in section 3.2 of Reference 3.7-4.

3.7.3.12 Buried Seismic Category I Piping Systems and Tunnels (Non-NSSS)

Buried seismic Category I piping is analyzed and designed for seismic effects in accordance with section 6.0 of Reference 3.7-2. For portions of buried piping in soil, the soil response along the pipeline is determined in accordance with the procedure in Section 2.5.4.7.1.

The diesel oil storage tank structures and the ESW/RHRSW pipe tunnel (located beneath the diesel generator enclosure) are seismic Category I and are analyzed using an equivalent static load method.

Other tunnels at LGS are nonseismic Category I.

3.7.3.13 Interaction of other Piping with Seismic category I Piping

3.7.3.13.1 Interaction of other Piping with seismic Category I Piping (NSSS)

When other piping is attached to seismic Category I piping, the other piping is analytically simulated in a manner that does not significantly degrade the accuracy of the seismic Category I piping analysis. Furthermore, the other piping is designed to withstand the SSE without failing in a manner that would cause the seismic Category I piping to fail.

3.7.3.13.2 Interaction of Other Piping with seismic Category I Piping (Non-NSSS)

The techniques used to consider the interaction of seismic Category I piping with non-Category I piping are as follows:

Seismic boundary anchors are designed for the combined loads generated from both sides of a boundary anchor. The loads from the seismic Category I side are actual calculated loads, and the loads from the nonseismic Category I side are determined by one of the following:

a. The actual calculated seismic loads if the nonseismic side piping is dynamically analyzed for seismic events

b. The actual calculated loads if the nonseismic side piping is designed to a conservative simplified seismic design criteria (e.g., by simplified span methods such as those used for designed of small piping)

c. The loads determined by the plastic capability of the piping.

3.7.3.14 Seismic Analysis for Reactor Internals (NSSS)

The modeling of RPV internals is discussed in section 3.7.2.3.1.2. The damping values are given in Table 3.7-1.

3.7.3.15 Analysis Procedures for Damping

3.7.3.15.1 Analysis Procedures for Damping (NSSS)

Analysis procedures for damping are discussed in section 3.7.2.15.1.

3.7.3.15.2 Analysis Procedure for Damping (Non-NSSS)

If the equipment damping is unknown, the response spectrum curve for 0.5% damping is used to arrive at a conservative seismic loading. The damping values used for the OBE are increased for the SSE, where sufficient justification is established.

3.7.4 SEISMIC INSTRUMENTATION

3.7.4.1 Comparison with Regulatory Guide 1.12 (Rev 1)

The seismic instrumentation program complies with the intent of Regulatory Guide 1.12 (Rev. 1) and section 5 of ANSI N18.5-1974. The original analog system was replaced by a digital system. The digital system provides the function and information equivalent of that required in RG 1.12 and ANSI N18.5. The triaxial peak recording accelerographs are no longer a part of the seismic monitoring system. The digital system will perform the same functions as the accelerographs.

Regulatory Guide 1.12 paragraph C.1.b specifies the requirement for a containment foundation sensor. The sensor is physically located on the reactor enclosure basemat about 6 inches from primary containment foundation. This is an exception to RG 1.12; however, the location provides an equivalent response to primary containment foundation and meets the intent of RG 1.12. Further details for sensor locations is provided in 3.7.4.2.

3.7.4.2 Location and Description of Instrumentation

The following instrumentation is provided for Unit 1 only, as essentially the same response is expected at Unit 2.

a. Six triaxial time history accelerometers.

b. One response spectrum analyzer

c. A system control panel which includes seismic event visual and audible annunciators

d. Digital recorders and a playback unit

All instrument characteristics meet the requirements of section 5 of ANSI N18.5 (1974).

The operability of the seismic monitoring instrumentation ensures that sufficient capability is available to promptly determine the magnitude of a seismic event and evaluate the response of those features important to safety, This capability is required to permit comparison of the measured response to that used in the design basis for the unit.

3.7.4.2.1 Triaxial Time History Accelerometers

Triaxial time history accelerometers (T/A) produce a record of the time-varying acceleration at the sensor location. These data are recorded at the system control panel and may be used directly for analysis and comparison with reference information, and may be converted to response spectra form for spectral comparisons with design parameters.

Each T/A contains three accelerometers mounted in a mutually orthogonal array. All T/As have their principal axes oriented identically, with one horizontal axis parallel to the major horizontal axis assumed in the seismic analysis. T/As are located as shown in Table 3.7-28.

A recording system located in the control room is provided for multiple channel recording of the signals from the T/As mounted on items shown in Table 3.7-28.

3.7.4.2.2 Triaxial Peak Recording Accelerographs

A self contained T/A recording system is located in the Spray Pond Pumphouse. This T/A is not connected to the Control Room equipment. A separate portable computer is available to retrieve recorded data and transport to the Control Room. A single playback unit is located in the Control Room for playback of any of the recorder data.

3.7.4.2.3 Triaxial Seismic Switch (DELETED)

3.7.4.2.4 Response Spectrum Analyzer

The response spectrum analyzer is an electronic device which generates response spectra from a time-based complex waveform. The analyzer is a part of the computer driven unit. This data can be compared with the spectra generated from the mathematical model and used to make timely operating decisions.

3.7.4.2.5 System Control Panel

A panel located in the control room houses the recording, playback, and spectrum analysis units which are used in conjunction with the T/A sensors to produce a time history and frequency-amplitude record of the seismic event. The panel also contains signal conditioning and display equipment associated with the response spectrum analyzer and the system power supply unit.

3.7.4.3 Control Room Operator Notification

Activation of the Seismic Monitoring System causes an audible and visual annunciation in the control room to alert the plant operator that the T/A recording system has been activated.

Should the acceleration and response spectra exceed the design values or a loss of power condition arises, the system will activate an audible and visible alarm in the control room. The pumphouse time history accelerograph will operate independently and light its event indicator. However, it does not cause any audible or visual annunciation in the control room. The setpoint will be at a vertical acceleration level slightly higher than the expected background level.

The peak acceleration level and response spectra experienced on the sensors is available immediately following a seismic event. The level is obtained by reading the peak value from a chart recorder or the spectrum analyzer.

3.7.4.4 Comparison of Measured and Predicted Responses

Initial determination of the seismic event level is performed immediately after the event by comparing the measured response spectra from the containment base slab with the calculated OBE and SSE response spectra for the corresponding location. An outline of the order of actions to be taken after a seismic event is provided in Figure 3.7-44.

3.7.4.5 Controls

Applicability: The seismic monitoring instrumentation shall be operable at all times. This instrumentation is considered operable when it is capable of performing its specified functions and when all necessary attendant instrumentation, controls, electrical power, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its functions are also capable of performing their related support functions.

Actions: For requirements when one or more seismic monitoring instruments are inoperable, reference the TRM actions.

3.7.4.6 Surveillance Requirements

Each of the seismic monitoring instruments shall be demonstrated operable by the performance of the channel check, channel functional test and channel calibration at the frequencies shown and defined in Table 3.7-29.

Each of the seismic monitoring instruments which is accessible during power operation and which is actuated during a seismic event greater than or equal to 0.01g, and which does not self-reset, shall be restored to operable status within 24 hours and a channel calibration performed within 5 days following the seismic event. Data shall be retrieved from the actuated instruments and analyzed to determine the magnitude of the vibratory ground motion. A Special Report shall be prepared and submitted to the Nuclear Regulatory Commission pursuant to Specification 6.9.2 of the Technical Specifications within ten days describing the magnitude, frequency spectrum and resultant effect upon unit features important to safety.

3.7.5 REFERENCES

3.7-1 N.C, Tsai, "Spectrum Compatible Motions for Design Purposes", Journal of Engineering Mechanics Division, ASCE, Vol, 98, No. EM2, Proc, Paper 8807, pp. 345-356 (April 1972) .

3.7-2 "Seismic Analyses of Structures and Equipment for Nuclear Power Plants", BC-TOP-4A, Rev. 3, Bechtel Power Corporation, San Francisco, California, (November 1974).

3.7-3 "Uniform Building Code", by International Conference of Building Officials, Whittier, California, 1970 Edition.

3.7-4 "Seismic Analysis of Piping Systems" BP-TOP-1 Rev. 3. Bechtel Power Corporation, San Francisco, California, (January 1976).

3.7-5 L.K. Liu, "Seismic Analysis of the Boiling Water Reactor", Symposium on Seismic Analysis of Pressure Vessel and Piping Components, First National Congress on Pressure Vessel and Piping, San Francisco, California, (May 1971).

3.7-6 N.M. Newmark, "Design Criteria for Nuclear Reactors Subject to Earthquake Hazards", Proc IAEA Panel on Aseismic Design and Testing of Nuclear Facilities, Japan Earthquake Engineering Promotion Society, Tokyo, Japan, (1967).

3.7-7 "Development of Analysis and Design Techniques from Dynamic Testing of Electrical Raceway Support Systems", Technical Report, Bechtel Power Corporation, (July 1979).

3.7-8 Cable Tray and Conduit Raceway Seismic Test Program- Release 4", Test Report #1053-21.1-4, Volumes 1 and 2, ANCO Engineers, Inc., (December 15, 1978).

3.7-9 P.Y. Hatago and G.S. Reimer, "Dynamic Testing of Electrical Raceway Support Systems for Economical Nuclear Power Plant Installations", presented at the IEEE-PES, (February 4-9, 1979).

3.7-10 "Cable Tray and Conduit Raceway Seismic Test Program- Release 4", Addendum to Test Report #1053-21.1-4, Volume 3, ANCO Engineers, Inc., (May 1980).

3.7-11 "Cable Trays Seismic Qualification Report for the Limerick Generating Station Units 1 and 2", Specification 8031-E-49, P-W Industries, Inc., (September 18, 1975).

3.7-12 "Safety Evaluation Report Relating to the Use of ASME Code Case N-411 for Limerick Generating Station, Unit 2", R.M. Bernero (NRC), (March 27, 1987).

3.7-13 NRC Letter to PECO, dated April 10, 1991, Snubber Reduction Program, Limerick Generating Station, Unit 1 (TAC No. 80069).

Table 3.7-1

CRITICAL DAMPING VALUES FOR NSSS MATERIALS[1,2]

ITEM	CRITICAL DAMPING (%) OBE CONDITION	CRITICAL DAMPING (%) SSE CONDITION
Reinforced concrete structures	2.0	5.0
Welded structural assemblies (equipment and supports)	1.0	2.0
Bolted or riveted structural assemblies	2.0	3.0
Vital piping systems	0.5	1.0
Drywell (coupled)	2.0	5.0
RPV support skirt, shroud head, separator, and guide tubes	2.0	2.0
CRD housings	3.5	3.5
Fuel	7.0	7.0
Steel frame structures	2.0	3.0

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

(2) Alternative critical damping values for piping systems may be used as described in Section 3.7.1.3.3.

Table 3.7-2

CRITICAL DAMPING VALUES FOR NON-NSSS MATERIALS[1]

ITEM	CRITICAL DAMPING (%) OBE CONDITION	CRITICAL DAMPING (%) SSE CONDITION
Equipment	0.5	1
Piping systems	0.5	1
Welded steel structures	2	5
Bolted steel structures	3	7
Reinforced concrete structures (except as noted below for specific loading conditions in primary containment)	2	5
Primary containment (for those loading conditions in which DBA and seismic loadings are combined)	3	7

(1) Alternative critical damping values for piping systems may be used as described in Section 3.7.1.3.3

Table 3.7-3

LOAD COMBINATIONS AND ALLOWABLE RESPONSES FOR
ELECTRICAL RACEWAY SYSTEM

Equation	Condition	Load Combination[1]	Allowable Response[1]
1	Normal	D+L	F_s[4]
2[5]	Normal/ Severe	D+E	[2,4]
3	Abnormal/ Extreme	D+E'	[2,3,4]

---

[1] For notations, see Tables 3.8-2 and 3.8-9.

[2] The following equation is applicable for bending in overhead connections:

$$\frac{5n_{EQ}}{N_{OBE}} + \frac{n_{EQ}}{N_{SSE}} \leq 1.0$$

where:

n_{EQ} = Total number of load/stress cycles per earthquake.

N_{OBE} = Allowable number of load/stress cycles per OBE event.

N_{SSE} = Allowable number of load/stress cycles per SSE event.

[3] The following criteria are used for checking the members. In no case shall the allowable stress exceed 0.90 F_y in bending, 0.85 F_y in axial tension or compression, and 0.50 F_y in shear. Where the design is governed by requirements of stability (local or lateral buckling), the actual stress shall not exceed 1.5 F_s.

[4] Allowable shear and normal loads in connections are determined from the manufacturers' data or from code allowable stresses, whichever is applicable. The allowable values are increased 50% for load combination equation 3.

[5] Equation 2 applies only to connections for fatigue considerations.

Table 3.7-4

RESULTS OF FATIGUE CALCULATIONS FOR THE
MOST LIMITING BWR/4 COMPONENT

BWR/4 RPV FEEDWATER NOZZLE[1]

Loading	Fatigue Usage
10 OBE Cycles	0.006
All Others[2]	0.967
Total	0.973

[1] The most limiting calculation for the BWR/4 product line.

[2] All other fatigue contributions due to SRV, thermal operating transients, etc.

Table 3.7-5

NATURAL FREQUENCIES OF PRIMARY CONTAINMENT BELOW 33 Hz

MODE NO.	FREQUENCY (Hz) HORIZONTAL UNCRACKED	HORIZONTAL CRACKED	VERTICAL UNCRACKED	VERTICAL CRACKED
1	4.496	2.878	10.620	7.731
2	8.545	7.589	17.573	12.857
3	15.393	9.864	-	19.809
4	17.831	14.836	-	26.954
5	27.389	17.221	-	-
6	29.135	22.085	-	-
7	-	23.735	-	-
8	-	28.448	-	-
9	-	31.175	-	-

Table 3.7-6

NATURAL FREQUENCIES OF THE REACTOR ENCLOSURE AND CONTROL STRUCTURE BELOW 33 Hz

MODE NO.	FREQUENCY (Hz) VERTICAL	N-S	E-W
1	1.993	2.495	3.235
2	3.007	7.757	10.515
3	3.272	11.531	15.108
4	3.888	16.119	22.063
5	4.043	21.310	30.057
6	4.154	28.090	-
7	4.262	-	-
8	4.314	-	-
9	4.546	-	-
10	4.575	-	-
11	4.630	-	-
12	4.862	-	-
13	5.088	-	-
14	5.959	-	-
15	7.949	-	-
16	9.339	-	-
17	9.975	-	-
18	10.069	-	-
19	10.112	-	-
20	10.329	-	-
21	10.618	-	-
22	10.661	-	-
23	10.814	-	-
24	11.053	-	-
25	11.333	-	-
26	12.553	-	-
27	12.998	-	-
28	13.254	-	-
29	13.328	-	-
30	13.527	-	-
31	13.731	-	-
32	13.882	-	-
33	13.923	-	-
34	14.896	-	-
35	17.918	-	-
36	22.877	-	-
37	24.077	-	-
38	25.244	-	-
39	30.395	-	-

Table 3.7-7

PRIMARY CONTAINMENT SEISMIC INERTIA FORCES, DISPLACEMENTS, AND ACCELERATIONS (HORIZONTAL DIRECTION)

	INERTIA FORCES (kips)				DISPLACEMENTS (10^{-2} ft)				ACCELERATIONS (g)			
	UNCRACKED CONDITION		CRACKED CONDITION		UNCRACKED CONDITION		CRACKED CONDITION		UNCRACKED CONDITION		CRACKED CONDITION	
MASS POINT	OBE	SSE	OBE	SSE	OBE	SSE	OBE	SSE	OBE	SSE	OBE	SSE
1	141	230	179	269	1.738	2.663	4.330	6.125	0.735	1.193	0.928	1.390
2	669	1083	816	1216	1.650	2.525	4.038	5.750	0.685	1.108	0.835	1.245
3	1346	2165	1481	2169	1.513	2.313	3.625	5.163	0.610	0.980	0.670	0.981
4	1321	2098	1330	1913	1.338	2.050	3.113	4.425	0.509	0.808	0.511	0.736
5	1114	1749	1254	1854	1.189	1.813	2.713	3.850	0.428	0.671	0.481	0.771
6	971	1520	1105	1658	1.054	1.613	2.325	3.313	0.361	0.566	0.411	0.618
7	739	1151	913	1360	0.941	1.438	2.050	2.925	0.308	0.480	0.380	0.566
8	733	1160	868	1275	0.848	1.300	1.813	2.575	0.306	0.488	0.363	0.533
9	649	1038	771	1135	0.773	1.184	1.613	2.300	0.300	0.480	0.358	0.526
10	2251	3631	2624	3850	0.700	1.074	1.425	2.038	0.291	0.470	0.340	0.498
11	993	1614	1171	1730	0.551	0.849	1.061	1.513	0.258	0.419	0.305	0.450
12	1153	1895	1463	2226	0.405	0.625	0.713	1.019	0.219	0.360	0.279	0.424
13	1271	2156	1356	2110	0.263	0.409	0.378	0.543	0.186	0.315	0.199	0.309
14	1976	3431	1509	2410	0.160	0.251	0.148	0.215	0.150	0.260	0.114	0.183
15	200	333	179	264	1.713	2.788	3.850	5.488	0.714	1.186	0.636	0.941
16	326	536	281	413	1.588	2.425	3.238	4.600	0.573	0.940	0.493	0.724
17	293	473	290	421	1.313	2.013	2.663	3.788	0.469	0.758	0.465	0.675
18	234	375	251	364	1.159	1.775	2.363	3.363	0.409	0.655	0.439	0.635
19	233	376	240	353	0.915	1.400	1.838	2.613	0.346	0.563	0.359	0.526
20	188	306	238	366	0.534	0.823	0.989	1.413	0.266	0.435	0.339	0.523
21	280	469	355	565	0.379	0.586	0.615	0.885	0.244	0.409	0.310	0.493
22	384	661	411	664	0.244	0.380	0.310	0.449	0.200	0.345	0.215	0.346
23	560	876	445	644	3.325	5.075	5.425	7.713	1.249	2.113	1.073	1.550
24	866	1334	755	1081	2.713	4.150	4.588	6.525	0.991	1.526	0.864	1.238
25	896	1413	793	1143	2.150	3.288	3.788	5.388	0.816	1.286	0.721	1.040
26	764	1223	720	1045	1.634	2.513	3.063	4.350	0.628	1.005	0.591	0.859
27	-	-	-	-	1.413	2.163	2.738	3.900	-	-	-	-
28	-	-	-	-	1.159	1.775	2.363	3.363	-	-	-	-
29	-	-	-	-	1.011	1.550	2.088	2.963	-	-	-	-
30	-	-	-	-	0.709	1.088	1.400	2.000	-	-	-	-
31	-	-	-	-	0.160	0.251	0.148	0.215	-	-	-	-
32	-	-	-	-	3.213	4.900	5.275	7.488	-	-	-	-
33	-	-	-	-	1.663	2.563	4.100	5.828	-	-	-	-
34	-	-	-	-	1.216	1.863	2.450	3.482	-	-	-	-

Table 3.7-8

PRIMARY CONTAINMENT SEISMIC DISPLACEMENTS AND ACCELERATIONS
(VERTICAL DIRECTION)

	DISPLACEMENTS (10^{-2} ft)				ACCELERATIONS (g)			
	UNCRACKED CONDITION		CRACKED CONDITION		UNCRACKED CONDITION		CRACKED CONDITION	
MASS POINT	OBE	SSE	OBE	SSE	OBE	SSE	OBE	SSE
1	0.133	0.201	0.386	0.551	0.205	0.318	0.395	0.591
2	0.133	0.200	0.386	0.551	0.204	0.316	0.394	0.589
3	0.133	0.200	0.376	0.535	0.201	0.314	0.368	0.543
4	0.131	0.198	0.353	0.500	0.198	0.305	0.321	0.466
5	0.129	0.195	0.329	0.466	0.193	0.298	0.285	0.411
6	0.128	0.191	0.304	0.433	0.188	0.289	0.258	0.373
7	0.125	0.189	0.284	0.404	0.184	0.283	0.244	0.355
8	0.123	0.185	0.264	0.375	0.180	0.275	0.233	0.343
9	0.121	0.183	0.245	0.349	0.175	0.269	0.221	0.326
10	0.119	0.179	0.226	0.321	0.171	0.261	0.206	0.308
11	0.114	0.171	0.190	0.271	0.165	0.251	0.178	0.264
12	0.109	0.164	0.155	0.220	0.159	0.243	0.151	0.224
13	0.103	0.155	0.116	0.168	0.150	0.231	0.129	0.198
14	0.098	0.148	0.088	0.125	0.145	0.223	0.109	0.166
15	0.169	0.266	0.171	0.243	0.260	0.403	0.238	0.345
16	0.168	0.255	0.170	0.241	0.256	0.398	0.235	0.341
17	0.165	0.250	0.168	0.238	0.249	0.385	0.228	0.329
18	0.163	0.248	0.166	0.235	0.248	0.375	0.223	0.320
19	0.155	0.235	0.155	0.220	0.234	0.360	0.200	0.288
20	0.146	0.223	0.144	0.203	0.221	0.343	0.180	0.259
21	0.138	0.203	0.128	0.180	0.201	0.310	0.154	0.223
22	0.121	0.183	0.113	0.160	0.179	0.275	0.138	0.203
23	0.108	0.161	0.099	0.140	0.155	0.238	0.121	0.183
24	0.184	0.280	0.186	0.264	0.293	0.459	0.275	0.405
25	0.183	0.278	0.185	0.261	0.289	0.451	0.270	0.398
26	0.179	0.271	0.181	0.256	0.278	0.433	0.259	0.378
27	0.174	0.264	0.176	0.250	0.266	0.413	0.245	0.356
28	0.159	0.241	0.161	0.228	-	-	-	-
29	0.170	0.258	0.173	0.248	-	-	-	-
30	0.165	0.250	0.168	0.238	-	-	-	-

Table 3.7-9

PRIMARY CONTAINMENT SHEAR FORCES AND MOMENTS

JOINT NO.	MEMBER NO.	SHEAR FORCES (kips) UNCRACKED CONDITION OBE	SHEAR FORCES (kips) UNCRACKED CONDITION SSE	SHEAR FORCES (kips) CRACKED CONDITION OBE	SHEAR FORCES (kips) CRACKED CONDITION SSE	MOMENTS (10^3k-ft) UNCRACKED CONDITION OBE	MOMENTS (10^3k-ft) UNCRACKED CONDITION SSE	MOMENTS (10^3k-ft) CRACKED CONDITION OBE	MOMENTS (10^3k-ft) CRACKED CONDITION SSE
						0	0	0	0
	1	141	230	179	269				
33						0.9	1.5	1.1	1.8
	34	628	966	966	796				
2						2.0	3.1	2.3	3.3
	2	1,098	1,746	1,274	1,876				
3						16.3	25.9	18.8	27.6
	3	3,454	5,438	3,475	5,060				
4						70.1	110.5	73.3	106.9
	4	4,261	6,744	4,408	6,364				
5						125.5	198.1	130.5	189.6
	5	5,179	8,183	4,974	7,133				
6						184.3	291.1	189.1	273.3
	6	6,095	9,584	5,366	7,684				
7						237.9	375.9	239.8	344.6
	7	6,749	10,561	6,061	8,735				
8						300.9	474.6	291.1	416.8
	8	7,285	11,345	6,614	9,558				
9						359.1	565.3	338.9	477.1
	9	7,715	11,995	6,944	10,038				
10						418.6/ 420.5	657.5/ 654.6	374.4/ 371.6	534.1/ 530.4
	10	9,278	14,263	7,835	11,308				
11						546.4	850.9	453.4	646.0
	11	10,019	15,438	8,716	12,538				
12						680.4	1052.8	543.9	776.0
	12	11,085	17,163	9,455	13,500				
13						807.0	1260.0	641.5	914.6
	13	12,153	18,900	10,693	15,400				
14						919.0	1411.3	708.4	1008.5
15						0.0	0.0	0.0	0.0
	14	1,041	1,618	1,055	1,516				
16						20.5	31.9	20.8	29.9
	15	814	1,285	813	1,173				

Table 3.7-9 (Cont'd)

		SHEAR FORCES (kips)				MOMENTS (10^3k-ft)			
		UNCRACKED CONDITION		CRACKED CONDITION		UNCRACKED CONDITION		CRACKED CONDITION	
JOINT NO.	MEMBER NO.	OBE	SSE	OBE	SSE	OBE	SSE	OBE	SSE
17						36.6	57.1	36.9	53.0
	16	616	995	569	829				
18						41.8/ 14.5	65.4/ 29.0	41.9/ 23.8	60.1/ 34.6
	17	1,565	2,440	1,394	2,000				
29						26.8	41.8	30.9	44.6
	18	1,565	2,440	1,394	2,000				
19		1,754	2,743	1,621	2,329				
30						55.0/ 46.9	84.6/ 72.3	47.6/ 37.4	74.3/ 68.3
	20	431	700	1,353	1,978				
20						42.5	65.4	55.9	80.0
	21	485	765	1,441	2,076				
21						39.0	60.0	65.1	93.5
	22	685	1,085	1,630	2,351				
22						34.6	53.0	78.5	112.0
	23	1,041	1,688	2,041	3,014				
31						44.4	68.5	96.1	137.5
23						0.0	0.0	0.0	0.0
	25	560	876	445	644				
32						1.8	2.8	1.4	2.0
	26	159	268	139	209				
24						2.4	4.1	2.0	3.1
	27	425	680	338	490				
25						9.9	16.1	8.0	11.9
	28	1,168	1,805	996	1,426				
26						26.6	41.5	20.9	30.1
	29	1,889	2,950	1,676	2,408				
27						37.4	57.5	31.5	45.1
	30	1,899	2,950	1,676	2,408				
34						49.4	76.1	42.1	60.3
	31	1,899	2,950	1,676	2,408				
28						52.6	81.1	45.0	64.4
10						1.9	2.9	2.8	3.9
	32	968	1,484	708	1,009				
30						8.1	12.4	4.3	6.3
	33	95,713	146,875	79,175	112,675				

Table 3.7-10

PRIMARY CONTAINMENT AXIAL FORCES

AXIAL FORCE (kips)

MEMBER NO.	UNCRACKED CONDITION		CRACKED CONDITION	
	OBE	SSE	OBE	SSE
1	40	61	76	114
2	239	370	461	690
3	684	1063	1273	1889
4	1196	1854	2086	3063
5	1698	2626	2793	4064
6	2196	3393	3426	4953
7	2635	4064	3909	5615
8	3060	4711	4326	6186
9	3430	5273	4706	6735
10	4245	6513	5368	7680
11	4865	7450	5810	8304
12	5443	8319	6286	9003
13	5900	9003	6528	9345
14	73	113	66	96
15	219	340	240	291
16	374	579	341	495
17	1491	2310	1376	1998
18	1491	2310	1376	1998
19	1641	2539	1506	2181
20	2123	3269	1918	2755
21	2256	3474	2015	2889
22	2381	3661	2093	2944
23	2511	3854	2154	3071
24	118	185	110	164
25	331	519	311	459
26	429	670	401	590
27	988	1538	919	1341
28	988	1538	919	1341
29	988	1538	919	1341

Table 3.7-11

REACTOR ENCLOSURE AND CONTROL STRUCTURE E-W INERTIA FORCES, DISPLACEMENTS, AND ACCELERATIONS

	INERTIA FORCES (k)		DISPLACEMENTS (10^{-2} ft)		ACCELERATIONS (g)	
MASS POINT	OBE	SSE	OBE	SSE	OBE	SSE
1	8,840	14,393	0.412	0.637	0.184	0.300
2	6,681	10,574	0.736	1.140	0.250	0.396
3	10,605	16,776	0.957	1.470	0.290	0.459
4	1,446	2,269	1.260	1.940	0.314	0.493
5	11,955	18,851	1.450	2.240	0.341	0.538
6	1,660	2,600	1.650	2.540	0.332	0.520
7	11,794	18,564	1.830	2.810	0.350	0.551
8	1,718	2,689	2.060	3.180	0.336	0.526
9	21,634	33,907	2.160	3.330	0.334	0.524
10	2,729	4,250	2.290	3.520	0.338	0.527
11	25,796	40,472	2.440	3.760	0.395	0.620
12	17,795	27,783	3.020	4.650	0.658	1.027

Table 3.7-12

REACTOR ENCLOSURE AND CONTROL STRUCTURE N-S INERTIA FORCES, DISPLACEMENTS, AND ACCELERATIONS

	INERTIA FORCES (k)		DISPLACEMENTS (10^{-2} ft)		ACCELERATIONS (g)	
MASS POINT	OBE	SSE	OBE	SSE	OBE	SSE
1	8,354	13,645	0.419	0.647	0.174	0.284
2	6,791	10,735	0.932	1.440	0.302	0.475
3	11,026	17,361	1.300	2.000	0.302	0.475
4	1,475	2,311	1.800	2.760	0.320	0.502
5	12,320	19,355	2.120	3.270	0.351	0.552
6	1,809	2,841	2.470	3.800	0.361	0.568
7	12,469	19,661	2.770	4.270	0.370	0.584
8	1,892	2,945	3.210	4.940	0.370	0.576
9	26,060	40,829	3.410	5.250	0.403	0.631
10	2,996	4,638	3.740	5.750	0.371	0.575
11	29,068	45,167	4.140	6.370	0.445	0.692
12	20,473	31,545	5.510	8.470	0.757	1.166

Table 3.7-13

REACTOR ENCLOSURE AND CONTROL STRUCTURE VERTICAL INERTIA FORCES, DISPLACEMENTS, AND ACCELERATIONS

	INERTIA FORCES (k)		DISPLACEMENTS (10^{-2} ft)		ACCELERATIONS (g)	
MASS POINT	OBE	SSE	OBE	SSE	OBE	SSE
1	6,320	10,050	0.232	0.359	0.133	0.211
2	2,961	4,682	0.264	0.409	0.147	0.232
3	2,765	4,355	0.286	0.443	0.154	0.243
4	534	838	0.317	0.491	0.166	0.260
5	3,315	5,199	0.336	0.521	0.172	0.271
6	446	697	0.356	0.552	0.178	0.278
7	2,804	4,383	0.373	0.578	0.184	0.287
8	632	990	0.399	0.618	0.196	0.308
9	2,943	4,616	0.409	0.634	0.202	0.316
10	928	1,457	0.426	0.660	0.211	0.332
11	4,760	7,486	0.439	0.681	0.219	0.345
12	3,242	5,167	0.462	0.717	0.248	0.396
13	675	1,077	0.272	0.422	0.149	0.238
14	687	1,098	0.316	0.489	0.164	0.262
15	1,035	1,649	0.541	0.838	0.238	0.380
16	606	952	0.751	1.161	0.287	0.451
17	2,204	3,460	0.988	1.529	0.448	0.704
18	73	114	1.109	1.715	0.503	0.787
19	38	60	1.228	1.898	0.589	0.918
20	3,933	6,153	0.985	1.524	0.378	0.591
21	2,911	4,581	1.249	1.932	0.496	0.781
22	4,743	7,407	0.835	1.292	0.298	0.464
23	4,759	7,431	0.840	1.299	0.297	0.465
24	2,917	4,589	1.255	1.941	0.497	0.782
25	3,948	6,176	0.989	1.530	0.379	0.594
26	2,899	4,476	3.560	5.495	0.836	1.291
27	6,994	10,800	2.693	4.157	0.689	1.064
28	1,925	2,982	0.645	0.997	0.550	0.852
29	4,077	6,334	0.585	0.906	0.471	0.732
30	2,980	4,611	0.787	1.216	0.534	0.827
31	2,906	4,498	0.749	1.158	0.387	0.599
32	1,001	1,548	0.956	1.477	0.551	0.852
33	2,312	3,573	1.112	1.717	0.653	1.009
34	3,908	6,057	1.226	1.895	0.774	1.154
35	717	1,120	1.332	2.062	0.725	1.132
36	1,363	2,106	1.900	2.934	0.703	1.087
37	2,120	3,305	1.406	2.175	0.696	1.085
38	488	756	0.995	1.538	0.334	0.517
39	909	1,419	0.910	1.408	0.351	0.548
40	1,212	1,875	0.777	1.201	0.677	1.047
41	956	1,477	2.231	3.445	0.276	0.427
42	1,283	1,981	2.488	3.842	0.649	1.001
43	967	1,494	1.981	3.058	0.468	0.723

Table 3.7-13 (Cont'd)

MASS POINT	INERTIA FORCES (k) OBE	SSE	DISPLACEMENTS (10^{-2} ft) OBE	SSE	ACCELERATIONS (g) OBE	SSE
44	3,060	4,723	5.016	7.743	1.062	1.639
45	1,594	2,460	2.356	3.637	0.633	0.978
46	1,680	2,593	3.501	5.405	1.028	1.587
47	3,144	4,854	3.105	4.794	0.712	1.098
48	893	1,379	2.489	3.844	0.375	0.580
49	490	759	0.999	1.544	0.335	0.519
50	899	1,403	0.905	1.401	0.347	0.542
51	2,122	3,308	1.404	2.171	0.697	1.086
52	713	1,113	1.333	2.063	0.721	1.126
53	2,510	4,020	3.994	6.418	0.215	0.344
54	455	720	0.842	1.317	0.958	1.516

Table 3.7-14

REACTOR ENCLOSURE AND CONTROL STRUCTURE E-W SHEAR FORCES AND MOMENTS

		SHEAR FORCE (10^2 k)		MOMENTS (10^3 k-ft)	
JOINT NO.	MEMBER NO.	OBE	SSE	OBE	SSE
1				11,040	17,000
	1	834	1,288		
2				9,486	14,600
	2	779	1,201		
3				8,439	12,990
	3	706	1,091		
4				7,032	10,830
	4	694	1,073		
5				6,146	9,474
	5	659	1,018		
6				5,162	7,966
	6	655	1,012		
7				4,306	6,654
	7	608	941		
8				3,098	4,797
	8	596	922		
9				2,589	4,013
	9	423	656		
10				1,794	2,781
	10	400	621		
11				1,032	1,611
	11	178	278		
12				0	0

Table 3.7-15

REACTOR ENCLOSURE AND CONTROL STRUCTURE N-S SHEAR FORCES AND MOMENTS

JOINT NO.	MEMBER NO.	SHEAR FORCE (10^2 k) OBE	SHEAR FORCE (10^2 k) SSE	MOMENTS (10^3 k-ft) OBE	MOMENTS (10^3 k-ft) SSE
				10,430	16,060
	1	832	1,283		
2				9,106	14,020
	2	780	1,204		
3				8,176	12,580
	3	719	1,111		
4				6,835	10,520
	4	712	1,100		
5				6,009	9,255
	5	640	988		
6				5,041	7,766
	6	628	969		
7				4,198	6,469
	7	592	915		
8				3,130	4,823
	8	584	902		
9				2,699	4,164
	9	448	693		
10				1,916	2,951
	10	423	655		
11				1,187	1,830
	11	205	316		
12				0	0

Table 3.7-16

REACTOR ENCLOSURE AND CONTROL STRUCTURE AXIAL FORCES

MASS POINT	AXIAL FORCE (k) OBE	SSE
1	62,283	96,586
2	47,396	73,527
3	44,400	68,961
4	40,645	63,210
5	39,736	61,810
6	35,719	55,599
7	34,779	54,144
8	31,475	49,013
9	30,699	47,814
10	20,903	32,677
11	18,163	24,838
12	5,162	8,273
13	13,581	21,050
14	12,796	19,810
15	12,484	19,305
16	11,121	17,201
17	6,483	10,053
18	4,009	6,203
19	3,454	5,354

Table 3.7-17

STRUCTURE-FOUNDATION INTERACTION COEFFICIENTS

STRUCTURE	MOTION			EQUIVALENT SPRING CONSTANT	EQUIVALENT DAMPING COEFFICIENT
Primary containment	Translational			4.15×10^{7} k/ft	2.01×10^{5} k-sec/ft
	Rocking			8.12×10^{10} k-ft/rad	7.82×10^{7} k-ft-sec/rad
	Vertical			4.87×10^{7} k/ft	3.48×10^{5} k-sec/ft
Reactor enclosure and control structure	Translational:		E-W	8.17×10^{7} k/ft	8.98×10^{5} k-sec/ft
			N-S	8.55×10^{7} k/ft	8.79×10^{5} k-sec/ft
	Rocking:	E-W		2.04×10^{12} k-ft/rad	9.23×10^{9} k-ft-sec/rad
		N-S		6.22×10^{11} k-ft/rad	2.63×10^{9} k-ft-sec/rad
	Vertical			9.90×10^{7} k/ft	1.74×10^{6} k-sec/ft

Table 3.7-18

NUMBER OF DYNAMIC RESPONSE CYCLES EXPECTED DURING
A SEISMIC EVENT
FOR NSSS SYSTEMS AND COMPONENTS

	FREQUENCY BAND WIDTH (Hz)		
	0-10	10-20	20-50
Total number of seismic cycles	168	359	643
Number of seismic cycles (0.5% of total) between 75% and 100% of peak load	0.8	1.8	3.2
Number of seismic cycles (4.5% of total) between 50% and 75% of peak load	7.5	16.2	28.9

Table 3.7-19

FATIGUE EVALUATION DUE TO SEISMIC LOAD

	COMPONENT	CALCULATED NO. OF CYCLES AT PEAK STRESS	DESIGN NO. OF PEAK STRESS CYCLES PER OBE
1.	Reactor Pressure Vessel		
	a. Vessel	See Table 3.7-18	10
	b. Shroud support	See Table 3.7-18	10
	c. Skirt	See Table 3.7-18	10
2.	Seismic Category I Piping		
	a. Recirculation lines	[1]	10
	b. Steam lines	[1]	10

[1] Design number of peak stress cycles used in analysis

Table 3.7-20

REACTOR ENCLOSURE AND CONTROL STRUCTURE - SEISMIC RESPONSE
COMPARISON OF 2-COMPONENT ABS VERSUS 3-COMPONENT SRSS METHODS

	Combined Axial and Bending Stresses on Concrete Wall Section Due to:			Total Seismic Stress (KSF) (Axial and Bending Stress on Concrete Wall Section)		
Elevation	SSE Vertical Excitation (P) (A) (KSF)	SSE N-S Excitation (Mc) I N-S (KSF)	SSE E-W Excitation (Mc) I E-W (KSF)	3 Component SRSS (Regulatory Guide 1.92)	2 Component ABS (Section 3.7.2.6)	% Difference
Reactor Enclosure Wall – Southwest Corner						
352	12.27	23.4	10.2	28.3	35.7	+26
333	0.92	18.6	13.4	22.9	19.5	-15
313	0.88	26.4	19.5	32.8	27.3	-17
304	13.85	26.5	22.1	37.2	40.4	+9
283	14.19	37.1	30.6	50.1	51.3	+2
269	14.56	43.4	33.0	56.4	58.0	+3
253	14.92	51.7	39.7	66.9	66.6	-
239	16.11	59.4	45.7	76.7	75.5	-2
217	16.44	71.4	55.2	91.7	87.8	-4
201	15.18	57.2	55.9	81.4	72.4	-11
177	14.35	63.1	64.2	91.2	78.6	-14
Control Structure – Northeast Corner						
352	12.27	25.4	3.9	28.5	37.7	+32
333	0.92	20.1	5.1	20.8	21.0	+1
313	0.88	29.1	7.5	30.1	30.0	-
304	13.85	29.8	8.5	33.9	43.7	+29
283	14.19	41.8	11.7	45.7	56.0	+23
269	14.56	45.6	12.6	49.5	60.2	+22
253	14.92	54.3	15.2	58.3	69.2	+19
239	16.11	61.9	17.5	66.3	78.0	+18
217	16.44	74.4	21.2	79.1	90.8	+15
201	15.18	74.7	21.4	79.2	89.9	+14
177	14.35	73.7	24.6	79.0	88.1	+12

Table 3.7-21

SPRAY POND PUMPHOUSE[(1)]

FREQUENCIES WITH AND WITHOUT ECCENTRICITIES

GLOBAL DIRECTION: E-W

	VIBRATION FREQUENCIES (CPS)	
MODE NO.	3-D MODEL WITH ECCENTRICITY	ORIGINAL MODEL WITHOUT ECCENTRICITY
1	17.8	18.2
2	41.8	42.3

(1) Figure 3.7-48

Table 3.7-22

DIESEL GENERATOR ENCLOSURE[1]

FREQUENCIES WITH AND WITHOUT ECCENTRICITIES

GLOBAL DIRECTION: E-W

	VIBRATION FREQUENCIES (CPS)	
MODE	ORIGINAL MODEL WITHOUT ECCENTRICITY	3-D MODEL WITH ECCENTRICITY
1	9.33	8.84
2	22.59	20.13
3	48.05	37.50

(1) Figure 3.7-49

Table 3.7-23

REACTOR ENCLOSURE[1]

FREQUENCIES WITH AND WITHOUT
ECCENTRICITIES

GLOBAL DIRECTION: E-W

	VIBRATION FREQUENCIES (CPS)	
MODE NO.	3-D MODEL WITH ECCENTRICITY	ORIGINAL MODEL WITHOUT ECCENTRICITY
1	3.738	3.769
2	11.397	11.863
3	16.587	17.944
4	21.685	23.007
5	24.906	30.083

(1) Figure 3.7-50

Table 3.7-24

SPRAY POND PUMPHOUSE

COMPARISON OF TORSIONAL MOMENTS FROM
ORIGINAL DESIGN WITH VALUES OBTAINED
FROM 3-D MODEL WITH ECCENTRICITY

		TORSIONAL MOMENT (M_T, K-FT)	
DIRECTION	EARTHQUAKE	ORIGINAL DESIGN	3-D STICK MODEL
E-W	OBE	15,370	7,724
	DBE	27,620	14,230

Table 3.7-25

DIESEL GENERATOR ENCLOSURE

COMPARISON OF TORSIONAL MOMENTS FROM
ORIGINAL DESIGN WITH VALUES OBTAINED
FROM 3-D MODEL WITH ECCENTRICITY

		TORSIONAL MOMENT (M_T, K-FT)	
DIRECTION	EARTHQUAKE	ORIGINAL DESIGN	3-D STICK MODEL
E-W	OBE	96,804	74,940
	DBE	149,988	101,200

Table 3.7-26

REACTOR ENCLOSURE

COMPARISON OF TORSIONAL MOMENTS FROM
ORIGINAL DESIGN WITH VALUES OBTAINED
FROM 3-D MODEL WITH ECCENTRICITY

		TORSIONAL MOMENT (M_T, K-FT)	
DIRECTION	EARTHQUAKE	ORIGINAL DESIGN	3-D STICK MODEL
E-W	DBE (SSE)	$3.5X10^6$	$2.36X10^6$

Table 3.7-27

REACTOR ENCLOSURE

COMPARISON OF DESIGN SHEARS
AND SHEARS OBTAINED FROM 3-D MODEL
WITH ECCENTRICITY

SEISMIC EVENT - DBE

WALLS AT EL 177 FT (GROUND FLOOR)

WALL[1]	DESIGN SHEAR (KIPS)	SHEAR FROM 3-D MODEL WITH ECCENTRICITY (KIPS)
Line J	44,171	37,800
Line D	75,385	64,193
Line M_L	26,359	22,099

[1] Figure 3.7-50

TABLE 3.7-28

SEISMIC MONITORING INSTRUMENTATION

INSTRUMENTATION AND SENSOR LOCATIONS	MEASUREMENT RANGE	MINIMUM INSTRUMENTS OPERABLE
1. Triaxial Time-History Accelerographs (T/A's)		
a. Sensors		
1) XE-VA-102 Primary Containment Foundation Sensor, Located at Reactor Enclosure Basemat U/1 (Loc. 109-R15-177)	0 to 1 g	1
2) XE-VA-103 Containment Structure (Diaphragm Slab)	0 to 1 g	1
3) XE-VA-104 Reactor Enclosure Foundation (Loc. 111-R11-177)	0 to 1 g	1
4) XE-VA-105 Reactor Piping Support (Mn, Stm, Line 'D', El 313', in containment)	0 to 1 g	1
5) XE-VA-106 Outside Containment on Seismic Category I Equipment (RHR heat Exchanger, Loc. 102-R15-177)	0 to 1 g	1
6) XRSH-VA-107* Foundation of an Independent Seismic Category I Structure (Spray Pond Pump House, El 237')	0 to 1 g	1
b. Recorders (Panel 0oC693)		
1) XR-VA-102/103 for XE-VA-102	N/A	1
2) XR-VA-102/103 for XE-VA-103	N/A	1
3) XR-VA-104/105 for XE-VA-104	N/A	1
4) XR-VA-104/105 for XE-VA-105	N/A	1
5) XR-VA-106 for XE-VA-106	N/A	1
2. Triaxial Response Spectrum Analyzer (RSA); (Loc. Control Room)	1-33.5 Hz	1 **

* Includes sensor, trigger, recorder, and backup power supply.

** With reactor control room indication and annunciation.

Receives signal from playback unit fed with data from the Triaxial Accelerometers

TABLE 3.7-29

SEISMIC MONITORING INSTRUMENTATION SURVEILLANCE REOUIREMENTS

INSTRUMENTATION AND SENSOR LOCATIONS	CANNEL** CHECK	CHANNEL*** FUNCTIONAL TEST	CHANNEL**** CALIBRATION
1. Triaxial Time-History Accelerographs (T/A's)			
a. Sensors			
1) XE-VA-102 Primary Containment Foundation Sensor, Located at Reactor Enclosure Basemat U/1 (Loc. 109-R15-177)	N/A	SA	R
2) XE-VA-103 Containment Structure (Diaphragm Slab)	N/A	SA	R
3) XE-VA-104 Reactor Enclosure Foundation (Loc. 111-R11-177)	N/A	SA	R
4) XE-VA-105 Reactor Piping Support (Mn, Stm, Line 'D', El 313', in containment)	N/A	SA	R
5) XE-VA-106 Outside Containment on Seismic Category I Equipment, (RHR Heat Exchanger, Loc. 102-R15-177)	N/A	SA	R
6) XRSH-VA-107* Foundation of an Independent Seismic Category I Structure (Spray Pond Pump House, El 237')	N/A	SA	R
b. Recorders (Panel 0oC693)			
1) XR-VA-102/103 for XE-VA-102	N/A	SA	R
2) XR-VA-102/103 for XE-VA-103	N/A	SA	R
3) XR-VA-104/105 for XE-VA-104	N/A	SA	R
4) XR-VA-104/105 for XE-VA-105	N/A	SA	R
5) XR-VA-106 for XE-VA-106	N/A	SA	R
2. Triaxial Response Spectrum Analyzer (RSA)	N/A	SA	R

---

* Includes sensor, trigger, recorder, and backup power supply.

** channel check - the qualitative assessment of channel behavior during operation by observation. This determination shall include, where possible comparison of the channel indication and/or status with other indications and/or status derived from independent instrument channels measuring the same parameter.

TABLE 3.7-29 (Cont'd)

SEISMIC MONITORING INSTRUMENTATION SURVEILLANCE REOUIREMENTS

*** channel function test **-** shall be:

a) analog channels - the injection of a simulated signal into the channel as close to the sensor as practical to verify operability including alarm and/or trip functions and channel failure trips.

b) bistable channel - the injection of a simulated signal into the sensor to verify operability including alarm and/or trip functions.

The channel functional test may be performed by any series of sequential, overlapping or total channel steps such that the entire channel is tested.

******** channel calibration **-** the adjustment, as necessary, of the channel output such that it responds with the necessary range and accuracy to known values of the parameter which the channel monitors, The channel calibration shall encompass the entire channel including the sensor and alarm and/or trip functions, and shall include the channel functional test, The channel calibration may be performed by any series of sequential, overlapping or total channel steps such that the entire channel is calibrated.

Surveillance Frequency Notation

Notation	Frequency
N/A	Not Applicable
SA	At least once per 184 days
R (Refueling Interval)	At least once per 24 months (731 days)

3.8 DESIGN OF CATEGORY I STRUCTURES

3.8.1 CONCRETE CONTAINMENT

3.8.1.1 Description of the Containment

The primary containment is divided by a horizontal diaphragm slab into two major volumes: the drywell and the suppression chamber. The drywell encloses the reactor vessel, reactor recirculation system, and associated piping and valves. The suppression chamber stores a large volume of water.

The primary containment, shown on Figures 3.8-1 through 3.8-8, is in the form of a truncated cone over a cylindrical section, with the drywell being the upper conical section and the suppression chamber being the lower cylindrical section. These two sections comprise a structurally integrated, reinforced concrete pressure vessel, lined with welded steel plate and provided with a steel domed head for closure at the top of the drywell. Connection of the drywell head to the top of the drywell wall is shown on Figure 3.8-9. The diaphragm slab is a reinforced concrete slab structurally connected to the containment wall, as shown on Figure 3.8-10.

The primary containment is structurally separated from the surrounding reactor enclosure.

The concrete dimensions of the primary containment are as follows:

a. Inside diameter

   1. Suppression chamber - 88'-0"

   2. Base of drywell - 86'-4"

   3. Top of drywell - 36'-4½"

b. Height

   1. Suppression chamber - 52'-6"

   2. Drywell - 87'-9"

c. Thickness

   1. Base foundation slab - 8'-0"

   2. Containment wall - 6'-2"

3.8.1.1.1 Base Foundation Slab

The containment base foundation slab is a reinforced concrete mat, the top of which is lined with carbon steel plate.

3.8.1.1.1.1 Reinforcement

The base foundation slab is reinforced with #18, Grade 60 rebar at the top and bottom faces. The maximum rebar spacing is 18 inches. Shear reinforcement consists of #8 and #9 vertical and inclined ties. Cadweld splices are used for splicing all main reinforcing bars. Figure 3.8-11 shows the plan and section views of reinforcement.

3.8.1.1.1.2 Liner Plate and Anchorages

The steel liner plate is ¼ inch thick, and is anchored to the concrete slab by structural steel beams embedded in the concrete and welded to the plate. Figure 3.8-12 shows details of the liner plate and anchorages.

3.8.1.1.1.3 Reactor Pedestal and Suppression Chamber Column, Base Liner Anchorages

Figures 3.8-13 and 3.8-14 show the base foundation slab liner anchorages for the reactor pedestal and the suppression chamber columns, respectively. For the pedestal anchorage, Cadweld sleeves are welded to the top and bottom surfaces of the thickened base liner to permit anchoring of the pedestal vertical rebar into the base foundation slab. Metal studs are welded to the top and bottom surfaces of the thickened base liner in order to transfer radial and tangential shear forces from the pedestal to the base foundation slab. For the suppression chamber column anchorage, pipe caps are welded to the thickened base liner at the locations where the column anchor bolts penetrate the base liner, in order to ensure the leak-tight integrity of the base liner.

3.8.1.1.2 Containment Wall

The containment wall is constructed of reinforced concrete 6'-2" thick, and is lined with carbon steel plate on the inside surface.

3.8.1.1.2.1 Reinforcement

The containment wall is reinforced with #18, Grade 60 rebar at the inner and outer faces. The inner rebar curtain consists of two meridional layers and one hoop layer. The outer rebar curtain consists of one meridional layer, two hoop layers, and two helical layers. Radial shear reinforcement consists of #6 horizontal and inclined ties. Cadweld splices are used for splicing all main reinforcing bars. Figures 3.8-15 and 3.8-16 show sections and developed elevation views of the suppression chamber wall reinforcement and drywell wall reinforcement, respectively.

3.8.1.1.2.2 Liner Plate and Anchorages

The steel liner plate is ¼ inch thick, and is anchored to the concrete wall by vertical stiffeners, using structural tees spaced horizontally every 2 feet, or less. Horizontal plate stiffeners provide additional stiffening. Figures 3.8-17 and 3.8-18 show details of the liner plate and anchorages.

Loads from internal containment attachments, such as beam seats and pipe restraints, are transferred directly into the containment concrete wall. This is accomplished by thickening the liner plate, and attaching structural weldments that transfer any type of load to the concrete, without relying on the liner plate or its anchorages. Where internal containment attachment loads are large, the structural weldments penetrate the liner plate, rather than being welded to opposite sides of the liner plate. This eliminates the possibility of lamellar tearing. Section 3.8.1.1.2.5 contains a further description of internal containment attachments.

3.8.1.1.2.3 Penetrations

Services and communications between the inside and the outside of the containment are performed through penetrations. Basic penetration types include pipe penetrations, electrical penetrations, and access hatches (equipment hatches, personnel lock, suppression chamber access hatches, and CRD removal hatch). Each penetration consists of a pipe sleeve with an annular ring welded to it. The ring is embedded in the concrete wall, and provides an anchorage for the penetration to resist normal operating and accident loads. The pipe sleeve is also welded to the containment liner plate to provide a leak-tight penetration.

Meridional and hoop reinforcement is bent around typical penetrations. Additional local reinforcement in the hoop and diagonal directions is added at all large penetrations, as shown on Figures 3.8-19 and 3.8-20. Local thickening of the containment wall at penetrations is generally not required. Section 3.8.2.1 contains further discussion of penetrations.

a. Pipe Penetrations

Details of typical pipe penetrations are shown on Figure 3.8-21. There are two basic types of pipe penetrations. For piping systems containing high temperature fluids, a sleeved penetration is furnished, providing an air gap between the containment concrete wall and the hot pipe. This air gap is large enough to maintain the concrete temperature below 200°F in the penetration area. A flued head outside the containment connects the process pipe to the pipe sleeve. For piping systems containing low temperature fluid, a separate sleeve for the penetration is not furnished. For this type of penetration, the process pipe is welded directly to the two ends of the embedded pipe penetration.

b. Electrical Penetrations

Figure 3.8-22 shows a typical electrical penetration assembly used to extend electrical conductors through the containment. The penetrations are hermetically sealed, and provide for leak testing at design pressure.

c. Equipment Hatches and Personnel Lock

Two equipment hatches, with inside diameters of 12 feet, are furnished in the drywell wall. One of these equipment hatches includes a personnel lock. Figure 3.8-21 shows a detail of an equipment hatch. Figures 3.8-16 and 3.8-23 show details of reinforcement around the equipment hatches. Additional meridional, hoop, helical, and shear reinforcement is used to accommodate local stress concentrations at the opening. The containment wall is thickened at the equipment hatches to accommodate the additional rebars.

d. Suppression Chamber Access Hatches

Two access hatches, with internal diameters of 4'-4", are furnished in the suppression chamber wall, as shown on Figure 3.8-21. Figure 3.8-15 shows a detail of reinforcement around the suppression chamber access hatches. Additional local reinforcement in the meridional and diagonal directions is added as shown on this figure.

3.8.1.1.2.4 Drywell Head Assembly

The drywell head lower flange is anchored to the top of the drywell wall by rigid attachment to 108 meridional reinforcing bars in the inner curtain of the containment wall, as shown on Figure 3.8-9. The remainder of the drywell head assembly is discussed in Section 3.8.2.1.1.

3.8.1.1.2.5 Internal Containment Attachments

The principal items attached to the containment wall from the interior are the diaphragm slab, beam seats, pipe restraints, and the seismic truss.

a. Diaphragm Slab Embedments

   The diaphragm slab is attached to the containment wall by a structural weldment at the junction of the two components, as shown on Figure 3.8-10. Radial force and bending moment, carried by the diaphragm slab main reinforcement, are transferred to the containment wall by Cadwelding the diaphragm slab rebar to the top and bottom flanges of the structural weldment. The top and bottom flanges of the structural weldment are embedded in the containment concrete wall, and are anchored using structural steel anchors. Flexural shear in the diaphragm slab is transferred to the containment wall through the web of the structural weldment, which is welded to opposite sides of the thickened containment liner plate.

b. Beam Seat Embedments

   Beam seats are provided to support the drywell platforms. A typical beam seat embedment is shown on Figure 3.8-24.

c. Pipe Restraint Embedments

   Pipe restraints are provided to prevent pipe whip caused by rupture of high energy piping. Typical pipe restraint embedments are shown on Figure 3.8-25.

d. Seismic Truss Support Embedments

   The seismic truss provides lateral support for the reactor vessel and reactor shield. A typical seismic truss support embedment in the drywell wall is shown on Figure 3.8-26.

3.8.1.1.2.6 External Containment Attachments

There are no major external structural attachments to the primary containment wall, except brackets providing vertical support for some of the reactor enclosure floor beams. These floor beams support checkered plate blowout panels, and are small enough to not cause any vertical interaction between the containment structure and the reactor enclosure. In addition, the beam-to-bracket connections are sliding connections, preventing horizontal interaction between the containment structure and the reactor enclosure.

3.8.1.1.2.7 Steel Components Not Backed by Structural Concrete

Descriptions of steel portions of the primary containment that are not backed by concrete, such as the drywell head, equipment hatches, personnel lock, suppression chamber access hatches, CRD removal hatch, and piping and electrical penetrations, are given in Section 3.8.2.

3.8.1.2 Applicable Codes, Standards, and Specifications

The codes, standards, and specifications used in the design and construction of the primary containment are listed in Table 3.8-1.

Specifications were prepared to cover the areas related to the design and construction of the containment. These specifications were prepared by Bechtel specifically for this containment. These specifications emphasize important points of the industry standards for this containment, and reduce options such as would otherwise be permitted by the industry standards. Unless specifically noted otherwise, these specifications do not deviate from the applicable industry standards, and, as such, are not included in the UFSAR. These specifications cover the following areas:

a. Furnishing and delivering concrete

b. Forming, placing, finishing, and curing concrete

c. Furnishing, detailing, fabricating, delivering, and placing reinforcing steel

d. Splicing reinforcing bars

e. Furnishing, delivering, and erecting liner plate

Section 1.8 provides references to regulatory guides discussed in the UFSAR. Regulatory guides specific to this section are discussed in this section.

3.8.1.3 Loads and Loading Combinations

Table 3.8-2 lists the loading combinations used for the design and analysis of the containment. Loading combinations listed in ASME Section III, Division 2 were considered for the containment design. Table 3.8-21 identifies and explains differences between the loads listed in Table 3.8-2 and the ASME Code.

The containment is also analyzed and designed for hydrodynamic loads resulting from MSRV discharge and LOCA phenomena. For a definition of these loads and loading combinations, including hydrodynamic loads, refer to Reference 3.8-1 and Appendix 3A.

3.8.1.3.1 Dead Load

The dead load includes the weight of the structure and major equipment, plus any other permanent loads, such as soil or hydrostatic loads, or operating pressure.

3.8.1.3.2 Live Load

The live load includes those loads expected to be present when the plant is operating, such as movable equipment, piping, and cables.

3.8.1.3.3 Design Basis Accident Pressure Load

Transients resulting from the DBA LOCA are presented in Section 6.2.1, and serve as the basis for the containment internal design maximum pressure of 55 psig.

3.8.1.3.4 Thermal Loads

The thermal loads used in the design of the primary containment are shown in Table 3.8-3 for the operating and the postulated accident conditions.

Thermal effects anticipated at the time of the structural integrity test are not considered. The ambient air temperature during testing is close to that of construction; therefore, the effects of thermal gradient are negligible.

3.8.1.3.5 Wind and Tornado Loads

Wind and tornado loads are not considered because the containment is enclosed by the reactor enclosure.

3.8.1.3.6 Seismic Loads

Loads from the OBE result from a horizontal ground acceleration of 0.075 g and a vertical ground acceleration of 0.05 g, acting simultaneously.

Loads from the SSE result from a horizontal ground acceleration of 0.15 g and a vertical ground acceleration of 0.10 g, acting simultaneously.

3.8.1.3.7 External Pressure Load

The containment shell is designed to withstand an external pressure of 5 psi above the internal pressure.

3.8.1.3.8 Pipe Rupture Loads

The containment wall is designed to withstand the loads due to a postulated rupture of a 26 inch diameter main steam pipe, which produces the largest loads on the containment wall. These loads include the effects of jet impingement, pipe whip, and pipe reaction. An equivalent static load of 1000 kips is considered. This load includes an appropriate dynamic load factor to account for the dynamic nature of the load. Section 3.6 contains a detailed discussion of postulated pipe ruptures and their effects. For pipe whip and impact on restraints, the steel stress may exceed yield stress. For jet impingement, these stresses should not exceed yield stress. See Section 3.8.1.4 for further explanation.

3.8.1.3.9 Prestress Loads

The primary containment uses reinforced concrete; therefore, prestress loads have not been considered.

3.8.1.4 Design and Analysis Procedures

This section describes the procedures used for the design and analysis of the containment. For a description of the design and analysis procedures that consider the effects of hydrodynamic loads resulting from MSRV discharge and LOCA phenomena, refer to Reference 3.8-1 and Appendix 3A.

The analysis procedure consists of two parts. First, the uncracked forces, moments, and shears for both axisymmetric and nonaxisymmetric loads are determined. Axisymmetric loads are dead load, live load, design accident pressure load, vertical seismic load, and operating and design accident thermal loads. Nonaxisymmetric loads are horizontal seismic load and localized pipe rupture load. The second part of the procedure consists of taking into account the expected cracking of the concrete, and determining the concrete and reinforcing steel stresses and strains. The liner plate is not assumed to resist any load.

The 3D/SAP computer program, described in Section 3.8.7.1, is used to determine the uncracked forces, moments, and shears due to axisymmetric loads. The operating and design accident temperature gradients are computed using the ME 620 computer program, which is described in Section 3.8.7.9. For transient loads such as design accident pressure and thermal loads, the most critical combination of these loads is considered.

The seismic loads on the structure are determined by seismic analysis using the methods described in Section 3.7. The 3D/SAP program is used to analyze the containment for nonaxisymmetric loads due to missile and postulated pipe rupture, and for local construction loads.

The Concrete Element Cracking Analysis Program (CECAP), described in Section 3.8.7.3, is used to determine the extent of concrete cracking, and the concrete and rebar stresses and strains. The input data for the CECAP program consist of the uncracked forces, moments, and shears calculated by the 3D/SAP and seismic analysis programs. The CECAP program models a single element of unit height, unit width, and depth equal to the thickness of the wall or slab. The program assumes isotropic, linear-elastic material properties, and uses an iterative technique to obtain stresses, considering their redistribution due to cracking. The program determines the redistribution of thermal-stresses due to the relieving effect of concrete cracking.

3.8.1.4.1 Containment Wall

Figure 3.8-27 shows the 3D/SAP finite-element model used to analyze the containment wall for axisymmetric loads. A 10° wedge of the containment is modeled using solid finite-elements having isotropic, linear-elastic material properties. The model includes the containment wall, base foundation slab, diaphragm slab, reactor pedestal, and the foundation material. Boundary conditions are imposed on the analytical model by specifying nodal point forces or displacements. With reference to Figure 3.8-27, the nodal points lying along boundary A are allowed to move within the X-Z plane, and those along boundary B within the X-Y plane. Points along boundary C are prevented from moving in the radial direction, and points along boundary D are prevented from moving in the hoop direction. Nodal forces, moments, and shears are applied to boundaries E and F to account for reaction loads from the drywell head, and from the reactor vessel and reactor shield wall, respectively.

Figure 3.8-28 shows the 3D/SAP finite-element model used to analyze the drywell wall for nonaxisymmetric pipe rupture loads. A 180° half-model of the drywell wall, consisting of isotropic, linear-elastic, solid finite-elements, is used. With reference to Figure 3.8-28, the nodal points lying along boundary A are allowed to move within the X-Z plane. Points along boundary B are prevented from moving in the vertical and radial directions. Nodal forces, moments, and shears are applied to boundary C to account for reaction loads from the drywell head.

Tangential shears caused by seismic loads are totally resisted by helical reinforcing bars and concrete in compression. In calculating the reinforcing steel requirement, the helical reinforcement is designed to resist stresses due to design accident pressure and thermal loads, as well as tangential shears caused by seismic loads.

3.8.1.4.2 Base Foundation Slab

Figure 3.8-29 shows the 3D/SAP finite-element model used to analyze the base foundation slab. A 180° half-model of the base foundation slab, consisting of isotropic, linear-elastic solid finite-elements is used. The model includes the base foundation slab, a portion of the containment wall, and the foundation material. By virtue of continuous modeling of base slab and wall elements as finite elements in the 3D/SAP analysis, strain compatibility and stress equilibrium at the wall and slab boundary are satisfied. With reference to Figure 3.8-29, the nodal points lying along boundary A are allowed to move within the X-Z plane, and those along boundary B within the X-Y plane. Points along boundary C are prevented from moving in the radial direction. Axisymmetric forces, moments, and shears calculated using the 3D/SAP containment model, and seismically induced, tangential shears are applied to boundary D. The height of the model is chosen so that the overturning moment caused by the tangential shear is the same as the overturning moment determined by the seismic analysis. In order to be able to consider uplifting of the base foundation slab from its foundation, a thin layer of foundation material immediately beneath the foundation slab is considered separately from the remainder of the foundation material. If the computer output indicates tension in any of the elements in this thin layer of foundation material, the elements' modulus of elasticity is reduced to almost zero. Then a second computer run is made, and any additional uplift is identified. Further iterations and modifications of foundation material properties are made until the complete extent of uplift is determined.

3.8.1.4.3 Analysis of Areas Around Equipment Hatches

Figure 3.8-30 shows the 3D/SAP finite-element model used to analyze the areas of the containment wall around the equipment hatches. A 60° wedge of the containment wall is modeled using solid finite-elements having isotropic, linear-elastic material properties. To reduce the size of the analytical model, boundary A follows the equipment hatch's vertical plane of symmetry. The points delineating the outermost boundaries of the model are located at a sufficient distance from the opening so that the behavior of the model along the boundaries is compatible with that of the undisturbed shell. With reference to Figure 3.8-30, the nodal points lying along boundary A are allowed to move within the X-Z plane, and those lying along boundary B within the X-Y plane. Points along boundary C are prevented from moving in the hoop direction. Axisymmetric forces, moments, and shears calculated using the 3D/SAP containment model are applied to boundary D. Seismic loads calculated by the seismic analysis are applied locally to the elements. Seismically induced tangential shears around the equipment hatches are resisted by helical reinforcing bars and concrete in compression.

3.8.1.4.4 Liner Plate and Anchorages

The design and analysis of the liner plate and anchorages are performed in accordance with Reference 3.8-3.

3.8.1.5 Structural Acceptance Criteria

3.8.1.5.1 Reinforced Concrete

The containment wall, the diaphragm slab, and the reactor pedestal are designed for the factored load combinations listed in Table 3.8-2, in accordance with the strength method of ACI 318. The following allowable stresses are used:

a. Concrete

1. Compression - $0.85 f_c'$

2. Tension - not permitted

3. Radial shear - as specified in Chapter 11 of ACI 318 (1971)

4. Tangential shear - not permitted

b. Reinforcing Steel

1. Tension - 0.90 F

2. Compression - 0.90 F

The notations are defined as:

f_c' = specified compressive strength of concrete

F = specified yield strength of reinforcing steel

The calculated stresses are within the allowable limits.

3.8.1.5.2 Liner Plate and Anchorages

The structural acceptance criteria for the liner plate and anchorages are in accordance with Reference 3.8-3.

3.8.1.6 Materials, Quality Control, and Special Construction Techniques

3.8.1.6.1 Concrete Containment

The concrete and reinforcing steel materials for the containment are discussed in Section 3.8.6.

3.8.1.6.2 Liner Plate, Anchorages, and Attachments

a. Materials

Liner plate materials conform to the requirements of the standard specifications listed below:

Item	Specification
Liner plate (¼ inch thick)	ASTM A285, Grade A, Firebox Quality
Liner plate (>¼ inch thick)	ASME SA516, Grade 60 conforming to the requirements of ASME Section III, Subsection B, Article 12
Structural steel shapes, plates, and bars used for anchorages and attachments to the liner plate	ASTM A36

Pipe restraint attachments	ASTM A36 or ASTM A441

b. Welding

Liner plate and structural steel welding conforms to the applicable portions of Part UW, "Requirements for Unfired Pressure Vessels Fabricated by Welding" of ASME Section VIII. Welders and weld procedures are qualified in accordance with ASME Section IX. For seam welds of liner plate and attachments that penetrate the liner plate (penetrations), Paragraphs UW-26 through UW-38, except UW-35, apply in their entirety. In lieu of Paragraph UW-35, undercuts that do not exceed 1/32 inch and do not encroach on the required section thickness are in accordance with ASME Section III, Subsection NB(4424). The welding of liner plate butt welds and penetration attachment welds are performed by either the shielded metal arc, or the automatic, submerged arc process. The minimum number of individual weld layers for welds that must maintain leak-tightness is two. In addition, liner plate seam and penetration attachment welds are inspected in accordance with paragraph (d) below. Welding of liner anchorages and internal containment attachments to the liner plates are also performed in accordance with ASME Section VIII Paragraphs UW-26 through UW-38, and the visual inspection/acceptance of the nonpressure-retaining attachment weldments are in accordance with NCIG-01 (Reference 3.8-22).

c. Materials Testing

Liner plate material greater than ¼ inch thick, used with Cadweld connectors and other anchorages, is vacuum- degassed and ultrasonically tested, in accordance with the ASME Section III, Section A-N321-1, and conforms to the requirements of Article 12, Materials, of ASME Section III, Subsection B.

d. Nondestructive Examination of Liner Plate Seam Welds

Nondestructive examination of welds complies with Regulatory Guide 1.19, with the following alternate approaches:

Spot radiographic examination is performed for all radiographable liner plate seam welds. Radiography is performed in accordance with ASME Section V, Article 2. Personnel performing radiographic examinations are qualified in accordance with the Society for Nondestructive Testing's Recommended Practice No. SNT-TC-1A, Supplement A, Radiographic Testing Method, plus any additional requirements of the ASME Section V. Acceptance standards are in accordance with Paragraph UW-51 of ASME Section VIII, Division 1. Twelve inches of the first 10 feet of weld for each welder, and welding position are radiographed. Thereafter, one 12 inch long radiograph is taken for each welder and weld position, for each additional 50 foot increment of weld. A minimum of 2% of all liner seam welds are examined by radiography.

Where nonradiographable weld joints are used, the entire length of weld is magnetic particle examined. All magnetic particle examinations conform to the ASME Section V. Personnel performing magnetic particle examinations are qualified in accordance with SNT-TC-1A, plus any additional requirements of the ASME Section V. Acceptance standards are in accordance with the ASME Section VIII, Division 1, Appendix VI.

The vacuum box soap bubble test is performed on all accessible liner plate seam welds, except welds connecting thickened reinforcing liner plate to penetration nozzles with nominal diameter of 6 inches or less. A 5 psi minimum pressure differential is maintained for a minimum of 20 seconds. The leak detecting solution is continuously observed for bubbles that indicate leaks. If a leak is detected, the defective weld is repaired, and reinspected by vacuum box testing.

Welds that are inaccessible for vacuum box testing are 100% liquid penetrant tested. Liquid penetrant examinations conform to the ASME Section V. Personnel performing liquid penetrant examinations are qualified in accordance with SNT-TC-1A, plus any additional requirements of the ASME Section V. Acceptance standards conform to the ASME Section VIII, Division 1, Appendix VIII.

A leak chase channel system is provided only on liner plate seam welds which are inaccessible after the completion of construction, and those under water in the suppression chamber. Following installation, the leak chase system is pressurized with air to 72 psig. The pressure is monitored by valving off the air supply, and measuring any pressure decay with a pressure gauge. Any pressure decay in excess of the pressure gauge rated accuracy, observed within 15 minutes, is cause to reject that portion of the liner plate seam welds and the leak chase system. All leaks are repaired, and following repair, the affected portion of the leak chase system is retested.

e. Quality Control

Quality control requirements during construction are discussed in the document "Limerick Generating Station Units 1 and 2; Summary Description of the Quality Assurance Program for Design and Construction," referenced in FSAR Section 17.1.

f. Erection Tolerances

The erection tolerances for the liner plate are summarized as follows:

1. The slope of any 10 foot section of cylindrical liner plate, referred to as true vertical, does not exceed 1:180. The deviation from the theoretical slope of any 10 foot section of conical liner plate, measured within a vertical plane, does not exceed 1:120.

2. The cylindrical shell is plumb within 1/400 of the height, plus a 2 inch allowance for local "out-of- roundness." The vertical axis of the conical shell, as established at the top and bottom of the conical section, is plumb within 1/200 of the height.

3. The radial dimension to any point on the liner plate does not vary from the design radius by more than ±1 inch in the suppression chamber, or ±1½ inches in the drywell, except that the radial tolerance is ±2 inches for local "out-of-roundness." Local "out-of-roundness" tolerance is used where the maximum diameter minus the minimum diameter at the elevation does not exceed 4 inches. Radial measurements are taken at 24 locations, spaced equally around the containment at any elevation.

4. Plates joined by butt-welding are matched accurately, and retained in position during the welding operation. Misalignment in completed joints does not exceed the limitations of Paragraph UW-33 of ASME Section VIII, Division 1.

3.8.1.7 Testing and Inservice Surveillance Requirements

3.8.1.7.1 Preoperational Testing

3.8.1.7.1.1 Structural Acceptance Test

Structural and Pressure Integrity Tests were performed according to Article CC-6000 of the ASME Section III, Division 2, as applicable, after complete installation of all penetrations in the drywell and suppression chamber, and prior to initial fuel loading. The Structural and Pressure Integrity Tests complies with Regulatory Guide 1.18, with the following alternate approaches:

Paragraph C.1

A continuous increase in containment pressure, rather than incremental pressure increases, was used. This was considered justifiable because data observations at each pressure level were made rapidly by the computerized data acquisition system. "Rapidly" is defined as requiring a time interval for the data point sample sufficiently short so that the change in pressure during the observation would cause a change in structural response of less than 5% of the total anticipated change. Also, the rate of pressurization was limited to 3 psig/hr average to ensure that the structure responds to the pressure loading without any time lag. However, retesting for Unit 2 is performed at a rate of pressurization limited to 8 psig/hr average with adequate hold time at peak pressure to assure there is no time lag for the structural response. For a description of the structural integrity test for each unit, see the alternate approach to Paragraph C.12 in this section (3.8.1.7.1.1).

Measurements were recorded at atmospheric pressure, and at 5 psi increments of the pressurization and depressurization cycles. The data acquisition system allowed the measurements at all locations to be made and recorded simultaneously.

The pressure was held constant for at least 1 hour at the maximum test pressure, or for such time as was necessary for recording crack patterns.

Paragraph C.2

The number and distribution of measuring points for monitoring radial deflections were selected so that the as-built condition could be considered in the assessment of general shell response. In general, the locations of the measuring points for radial deflections were in agreement with figure B of the Regulatory Guide, with the exception of Point 1. Point 1 was provided at a distance of two times the wall thickness (12 feet) from the basemat. This variation was made to properly predict the containment behavior near the basemat to wall connection. If Point 1 was located at a height of three times the wall thickness (18 feet), it would be very close to Point 2 (suppression chamber wall midheight is 26 feet) and would not yield any additional behavior pattern of the containment.

Paragraph C.3

Radial deflections, but not tangential deflections, were recorded. The magnitude of the expected local tangential deformation under the test pressure conditions is negligibly small. The diametrical displacement of the hatch parallel to the tangential direction was measured and compared to the predicted displacement. This was in the location of largest predicted displacements for the hatch discontinuity and provided the most comprehensive check on structural integrity. The measurement of absolute tangential deflections at locations away from the immediate edge of the discontinuity was difficult because fixed reference points are difficult to define, making it impractical to attempt measurement of these small local deflections.

Paragraph C.9

The LGS containment is located inside and isolated from the reactor enclosure. The environmental conditions under which the test was conducted were controlled by the reactor enclosure internal environment. Therefore, limitations on testing during periods of extreme weather conditions, such as snow, heavy rain, and strong wind, had negligible effect on structural response and were not applicable.

Paragraph C.10

Rather than as suggested in the guide, if the test pressure dropped to, or below, the next lower pressure level due to an unexpected condition, the test would have continued without a restart at atmospheric pressure, unless the structural response deviated significantly from that expected. The LGS containment pressurization was not conducted using pressure increments. Therefore, the provisions of ASME Section III, Division 2, subsection CC-6252(b) were met, using a 3 psi pressure drop as a criterion for investigating deviation in structural response.

Paragraph C.12

LGS has provided a detailed description of the structural integrity test in the report prepared after the test.

The Structural and Pressure Integrity Test was conducted at 115% of the following design conditions:

a. A design pressure condition of 55 psig in both the drywell and suppression chamber

b. A design pressure condition of 55 psig in the drywell, and 25 psig in the suppression chamber

Additional pressure integrity tests may be made subsequently, during periods of plant shutdown.

The differential pressure test of the diaphragm slab described in item (b) above was accomplished by capping the downcomers above the diaphragm slab's upper surface.

The LGS Unit 1 test was performed in one test sequence which demonstrated the structural adequacy of the containment structure for both of the design conditions described above. However, for LGS Unit 2, the structural integrity test was conducted in two stages because the maximum differential pressure on the diaphragm slab could not be maintained during the initial pressurization of the containment structure. During the initial test, the structural adequacy of the containment structure for the design condition of 55 psig in both the drywell and the suppression chamber was satisfactorily demonstrated. Subsequent to the re-isolation of the drywell and the suppression chamber, the containment structure was retested at a rate of pressurization limited to 8 psig/hr average to 115% of the design condition "b" described above (drywell pressure equal to 63.25 psig and suppression chamber pressure equal to 28.75 psig).

Provisions are made to test the integrity of the primary containment system during the life of the plant.

The seals on the personnel airlock doors, hatches, and drywell head are capable of being tested for leakage at design pressure without pressurizing the drywell. In addition, provisions are made so that the space between the airlock doors can be pressurized to full drywell design pressure.

Electrical penetrations are testable at 115% of design pressure. Locations of test taps and seals allow electrical penetrations to be tested without entering or pressurizing the drywell or suppression chamber.

A more detailed description of the structural integrity test was provided in the report prepared after the initial test.

3.8.1.7.1.2 Leak Rate Testing

Preoperational leak rate testing is discussed in Section 6.2.6.

3.8.1.7.2 Inservice Leak Rate Testing

Inservice leak rate testing is discussed in Section 6.2.6.

3.8.2 ASME CLASS MC STEEL COMPONENTS OF THE CONTAINMENT

This section pertains to the ASME Class MC steel components of the concrete containment that form a portion of the containment pressure boundary, and are not backed by structural concrete. These components include the drywell head assembly, the equipment hatches, the personnel lock, the suppression chamber access hatches, the CRD removal hatch, and piping and electrical penetrations.

3.8.2.1 Description of the ASME Class MC Components

3.8.2.1.1 Drywell Head Assembly

The drywell head provides a removable closure at the top of the containment for reactor access during refueling operations. The drywell head assembly consists of a 2:1 hemi-ellipsoidal head and a cylindrical lower flange. The lower flange is supported on the top of the drywell wall as shown on Figure 3.8-9. The head is made of 1½ inch thick plate and is secured with eighty 2¾ inch diameter bolts at the 4 inch thick mating flange. The head-to-lower flange connection is made leak-tight by two replaceable gaskets. The space between the gaskets is provided with test connections to allow pneumatic testing from a remote location, outside the primary containment. The inside diameter of the drywell head at the mating flange is 37 feet 7½ inches. A double-gasketed manhole is provided in the drywell head. Figure 3.8-31 shows details of the drywell head assembly.

3.8.2.1.2 Equipment Hatches and Personnel Lock

Two 12 foot diameter equipment hatches are furnished in the drywell wall to permit the transfer of equipment and components into and out of the drywell. One hatch consists of a double-gasketed flange and a bolted dished door. The other hatch is furnished with a personnel lock welded to the removable door. The personnel lock is an 8'-7" diameter cylindrical pressure vessel, with inner and outer flat bulkheads. Interlocked doors, 2'-6" wide by 6' high, with double tongue-and-groove single element compression seals, are furnished in each bulkhead. A quick-acting, equalizing valve vents the personnel lock to the drywell to equalize the pressure in the two systems when the doors are opened and then closed. The two doors in the personnel lock are mechanically interlocked to prevent them from being opened simultaneously, and to ensure that one door is closed before the opposite door can be opened. The personnel lock has an ASME Code N-stamp. Figures 3.8-32 and 3.8-33 show details of the equipment hatch, and the equipment hatch with personnel lock, respectively.

3.8.2.1.3 Suppression Chamber Access Hatches

Two 4'-4" diameter access hatches are furnished in the suppression chamber wall to permit personnel access, and the transfer of equipment and components into and out of the suppression chamber. Each hatch consists of a double-gasketed flange and a bolted flat cover. Details of a suppression chamber access hatch are shown in Figure 3.8-21.

3.8.2.1.4 Control Rod Drive Removal Hatch

One 3 foot diameter CRD removal hatch is furnished in the drywell wall to permit transfer of the CRD assemblies into and out of the drywell. The hatch is furnished with a double-gasketed flange and a bolted flat cover. Figure 3.8-34 shows details of the CRD removal hatch.

3.8.2.1.5 Piping and Electrical Penetrations

A portion of each of the penetration sleeves extends beyond the containment wall, and is not backed by concrete. The entire length of any penetration sleeve, therefore, is considered an MC component, and, as such, is designed in accordance with ASME Section III, subsection B. Figures 3.8-21 and 3.8-22 show details of typical pipe and electrical penetrations, respectively.

3.8.2.2 Applicable Codes, Standards, and Specifications

The codes, standards, and specifications used in the design and construction of the containment are listed in Table 3.8-1.

Specifications are prepared to cover the areas related to the design and construction of the containment. These specifications are prepared by Bechtel specifically for this containment. These specifications emphasize important points of the industry standards for this containment, and reduce options that would otherwise be permitted by the industry standards. Unless specifically noted otherwise, these specifications do not deviate from the applicable industry standards, and as such are not included in the UFSAR. These specifications cover the following areas:

a. Furnishing and delivering concrete

b. Forming, placing, finishing, and curing concrete

c. Furnishing, detailing, fabricating, delivering, and placing reinforcing steel

d. Splicing reinforcing bars

e. Furnishing, delivering, and erecting liner plate

Section 1.8 provides references to regulatory guides discussed in the UFSAR. Regulatory guides specific to this section are discussed in this section.

3.8.2.3 Loads and Loading Combinations

Table 3.8-4 lists the loading combinations used for the design and analysis of the ASME Class MC components.

The ASME Class MC components are also assessed for Mark II hydrodynamic loads resulting from MSRV discharge and LOCA phenomena. For a definition of loads and loading combinations including hydrodynamic loads, refer to Reference 3.8-1 and Appendix 3A.

3.8.2.3.1 Dead and Live Load

For descriptions of dead and live load, see Sections 3.8.1.3.1 and 3.8.1.3.2, respectively.

3.8.2.3.2 Design Basis Accident Pressure Load

The drywell head is designed for a DBA internal pressure of 56 psi. The other MC components are designed for a DBA internal pressure of 62 psi.

3.8.2.3.3 External Pressure Load

The MC components are designed to withstand an external pressure of 5 psi above the containment internal pressure.

3.8.2.3.4 Thermal Loads

The operating and postulated design accident temperatures for the MC components are as follows:

	Temperature (°F)	
Condition	Drywell	Suppression Chamber
Operating	135 ①	95 (maximum)
Design Accident	340	220

Thermal cycles used in design are as follows:

a. Startup and shutdown - 500 cycles, 105°F range

b. Design Basis Accident - 1 cycle, 220°F range

3.8.2.3.5 Seismic Loads

The MC components are designed for acceleration values which were calculated using the methods described in Section 3.7.

The following seismic acceleration values are used for the design of the drywell head assembly:

a. OBE - 0.60 g horizontal, ±0.30 g vertical

b. SSE - 0.80 g horizontal, ±0.40 g vertical

The following seismic acceleration values are used for the design of the equipment hatch, personnel lock, and CRD hatch:

a. OBE - 0.40 g horizontal, ±0.20 g vertical

b. SSE - 0.45 g horizontal, ±0.25 g vertical

The following seismic acceleration values are used in the design of the suppression chamber hatches:

a. OBE - 0.20 g horizontal, ±0.15 g vertical

b. SSE - 0.30 g horizontal, ±0.20 g vertical

① This information is based on original design basis conditions. Further evaluation has validated the MC components for a drywell operating temperature of 150°F. The Stresses indicated in UFSAR Figures 3A-362 to 3A-380, Design Assessment Report for Containment Structures reflect current plant conditions.

3.8.2.3.6 Pipe Rupture Loads

The drywell head assembly is designed for a local pipe rupture load of 48,000 pounds, uniformly distributed over a circular area of 0.56 square foot, at any location on the drywell head. This load is due to the postulated rupture of the 6 inch diameter reactor head spray pipe, which produces the largest load on the drywell head. The head spray line does not exist for Unit 2 and has been removed from Unit 1. However, the analysis is still applicable because it envelopes loads from rupture of any other pipe in this region.

The equipment hatches are designed for a postulated pipe rupture load of 130,000 pounds, uniformly distributed over a circular area 12 feet in diameter.

The CRD removal hatch is designed for a postulated pipe rupture load of 160,000 pounds, uniformly distributed over a circular area 3 feet in diameter.

The loads on the equipment hatches and the CRD removal hatch are based on the rupture of a 28 inch diameter recirculation loop outlet pipe, which produces the largest load on the components.

The above values of static load include an appropriate dynamic load factor to account for the dynamic nature of the load. Section 3.6 contains a detailed discussion of postulated pipe ruptures and their effects, including pipe whip, jet impingement, environmental effects, flooding, and water spray.

3.8.2.3.7 Missile Impact Loads

As discussed in Section 3.5.1 and summarized in Table 3.5-7, missile impact is not considered credible for safety-related components inside the containment or the reactor enclosure.

3.8.2.4 Design and Analysis Procedures

This section describes the procedures used for the design and analysis of ASME Class MC steel components.

ASME Class MC components are also assessed for Mark II hydrodynamic loads resulting from MSRV discharge and LOCA phenomena. Assessment procedures that consider the effect of hydrodynamic loads are described in Appendix 3A.

3.8.2.4.1 Drywell Head Assembly

The analysis of the drywell head assembly uses the thin-shell computer program E0781, described in Section 3.8.7.7. This program calculates the stresses and displacements in thin-walled, elastic shells of revolution, when subjected to static edge, surface, and/or temperature loads with an arbitrary distribution over the surface of the shell.

The drywell head assembly is divided into two analytical models. Figure 3.8-35 shows the drywell head model and the lower flange model. Displacement compatibility of the two models at the mating flange surface is maintained in the analysis. Boundary conditions are imposed on the analytical models by specifying boundary forces or displacements. With reference to Figure 3.8-35, the translation and rotation of the top of the drywell wall are imposed as boundary conditions to boundary A. Boundary forces applied to boundary B are calculated in accordance with thin-shell theory.

See Section 3A.7.2.1.9.1 for discussion of analysis to reduce the drywell head bolt preload.

3.8.2.4.2 Access Hatches

Access hatches, including the equipment hatches, personnel lock, suppression chamber access hatches, and the CRD removal hatch, are designed as pressure-retaining components. The portions of the sleeves not backed by concrete are designed and analyzed according to the provisions of ASME Section III, Subsection B.

At the junction of the hatch cover to the flange on the sleeve, where local bending and secondary stresses occur, the computer program E0119, described in Section 3.8.7.7, is used for analysis. This program is also used for the analysis of the flat head covers.

3.8.2.4.3 Piping and Electrical Penetrations

For nuclear Class I flued head penetrations, the procedures used in design and analysis comply with ASME Section III, Subsection B.

For Class 1E electrical cable penetrations, the procedures used in design and analysis comply with ASME Section III, Subsection B.

3.8.2.5 Structural Acceptance Criteria

The structural acceptance criteria comply with Regulatory Guide 1.57, with the clarification that the Code Addenda in effect at the time of design rather than that cited in the guide were used. Table 3.8-4 lists the allowable stress criteria for the design and analysis of the ASME Class MC components.

3.8.2.6 Materials, Quality Control, and Special Construction Techniques

3.8.2.6.1 Materials

All carbon steel materials and stainless steel materials for the CRD hydraulic line penetrations conform to the requirements of ASME Section III, Subsection B. The drywell head assembly and other steel components are painted to protect against corrosion.

3.8.2.6.1.1 Drywell Head Assembly

Materials used in construction of the drywell head assembly conform to the following ASME specifications:

Item	Specification
Drywell head and lower flange	SA516, Grade 70, normalized
Bolts	SA320, Grade L43
Nuts	SA194, Grade 7

3.8.2.6.1.2 Access Hatches

Materials used in construction of the access hatches conform to the following ASME specifications:

Item	Specification
Sleeve and cover	SA516, Grade 60 or 70, normalized
Bolts	SA193, Grade B7
Nuts	SA194, Grade 7

3.8.2.6.1.3 Penetrations

Materials used in construction of piping and electrical penetrations conform to the following ASME specifications:

Item	Specification
Carbon steel sleeves	SA333, Grade 1 or SA516, Grade 60 or 70, normalized
Carbon steel caps for spare penetrations	SA234, Grade WPB
Stainless steel sleeves for CRD hydraulic line penetrations	SA312, Type 304L

Stainless steel fittings for CRD hydraulic line penetrations	SA182, Type 304

3.8.2.6.2 Welding

Welding conforms to the requirements of ASME Section III, Subsection NE-4000 except that all welding of the CRD hydraulic line penetrations conforms to the requirements of ASME Section III, Subsection NC. All pressure boundary welds are full penetration welds of double-welded, bevel-type. Welders and weld procedures are qualified in accordance with ASME Section IX.

Penetrations, access hatches, and the drywell head flange are postweld heat treated in accordance with ASME Section III, Article NE-4000. Penetrations are preassembled into the liner plate sections, and postweld heat treated as complete subassemblies.

3.8.2.6.3 Nondestructive Examination of Welds

All welds between penetrations and liner plate, access hatches and liner plate, and pressure-retaining welds not backed by concrete are examined in accordance with ASME Section III, Article NE-5000. Nondestructive examination of welds complies with Regulatory Guide 1.19, with the following alternate approaches:

Paragraph C.1.a

LGS uses an alternate approach with respect to radiographic examination of the first 10 feet of weld. For each welder and welding position, only 12 inches of the first 10 feet of weld is examined radiographically, instead of the entire 10 foot length. LGS conforms with all other guidelines for nondestructive examination of liner seam welds.

Paragraph C.4

LGS conforms with the intent, subject to the following interpretation: Nondestructive examinations are performed by employees of the liner plate Subcontractor. Bechtel, as licensee's agent, verifies that all personnel performing nondestructive testing are qualified in accordance with ASME Section V.

Paragraph C.7.a

LGS radiography acceptance standards are in accordance with UW-51 of the ASME Section VIII, Division I.

Paragraph C.7.b

Magnetic particle acceptance standards are in accordance with the ASME Section VIII, Division I, Appendix VI.

3.8.2.6.4 Quality Control

Quality control requirements during construction are discussed in the document "Limerick Generating Station Units 1 and 2; Summary Description of the Quality Assurance Program for Design and Construction," referenced in FSAR Section 17.1.

3.8.2.6.5 Erection Tolerances

The specified erection tolerances for ASME Class MC steel components of the containment are summarized as follows:

a. Suppression chamber penetrations are within 1 inch of their design elevations and circumferential locations.

b. Drywell penetrations are within 1 inch of their design circumferential locations. Critical penetrations, such as main steam, feedwater, core spray, etc, are within 1 inch of their design elevations. All other drywell penetrations vary from within 1 inch of design elevations for penetrations near the base of the drywell wall, to within 2 inches of design elevations for penetrations near the top of the drywell wall.

c. Alignments of penetrations are within 1° of the design alignments.

d. The average elevation of the drywell head lower flange is within 3 inches of the design elevation. The lower flange is within ½ inch of level.

3.8.2.7 Testing and Inservice Inspection Requirements

3.8.2.7.1 Preoperational Testing

3.8.2.7.1.1 Structural Acceptance Test

The drywell head assembly, equipment hatches, suppression chamber access hatches, CRD removal hatch, and piping and electrical penetrations are pneumatically tested to 1.15 times the design accident pressure during the containment structural acceptance test.

The personnel lock is pneumatically tested to 1.25 times the design pressure of 62 psig, following shop fabrication, or following field erection if final assembly of lock is performed in the field, to verify its structural integrity.

The CRD hydraulic line penetrations are hydrotested to 1.5 times the design pressure of 1250 psig following shop fabrication, in accordance with the ASME Section III, Subsection NC.

3.8.2.7.1.2 Leak Rate Testing

Leak-tightness of the containment pressure-retaining Class MC components is verified during the integrated leak rate test. Section 6.2.6 contains a description of the containment integrated leak rate test.

The personnel air lock is leak rate tested to 100% of the design accident pressure following shop fabrication, and following field erection. The maximum allowable leak rate for the personnel air lock shall be 5% of the allowable leak rate for the containment, L_a. L_a is 0.5% by weight of the air in the containment per 24 hours at test pressure, P_a (44.0 psig).

3.8.2.7.2 Inservice Leak Rate Testing

Inservice leak rate testing is discussed in Section 6.2.6.

3.8.3 CONTAINMENT INTERNAL STRUCTURES

3.8.3.1 Description of the Internal Structures

The functions of the containment internal structures include: support and shielding of the reactor vessel, support of piping and equipment, and formation of the pressure-suppression boundary. The containment internal structures are constructed of reinforced concrete and structural steel. The containment internal structures include the following:

a. Diaphragm slab

b. Reactor pedestal

c. Reactor shield wall

d. Suppression chamber columns

e. Drywell platforms

f. Seismic truss

Figures 3.8-1 through 3.8-8 show an overview of the containment including the internal structures.

3.8.3.1.1 Diaphragm Slab

The diaphragm slab serves as a barrier between the drywell and the suppression chamber. It is a reinforced concrete circular slab, with an outside diameter of 88 feet, and a thickness of 3'-6". Figure 3.8-36 shows details of the diaphragm slab reinforcement.

The diaphragm slab is supported by the reactor pedestal, the containment wall, and 12 steel columns. The connection of the diaphragm slab to the containment wall is shown on Figure 3.8-10. The diaphragm slab is penetrated by 87, 24 inch diameter downcomers. Additional reinforcement is furnished at downcomer penetrations. Section 6.2.1 contains a description of the downcomers.

A ¼ inch thick, carbon steel liner plate is provided on top of the diaphragm slab, and is anchored to it. The liner plate prevents bypass flow around the downcomers during a LOCA. Refer to Section 6.2.1, for a description of the bypass leakage requirements. Figure 3.8-37 shows the diaphragm slab liner plate and anchorage system.

3.8.3.1.2 Reactor Pedestal

The reactor pedestal is an 82 foot high, upright cylindrical reinforced concrete shell that rests on the containment base foundation slab, and supports the diaphragm slab, reactor vessel, and reactor shield wall, as well as drywell platforms, pipe restraints, and recirculation pumps. The connection of the reactor pedestal to the base foundation slab is shown on Figure 3.8-13. The reactor pedestal below the diaphragm slab has an inside diameter of 20'-1", and a wall thickness of 4'-9". The reactor pedestal above the diaphragm slab has an inside diameter of 20'-3", and a wall thickness of 4'-5". The thickness at the top of the pedestal increases to 5'-4" where it supports the reactor vessel and the reactor shield wall. Attachment of the reactor shield wall to the pedestal is described in Section 3.8.3.1.3. Attachment of the reactor vessel to the pedestal is accomplished through a ring girder, as described in Section 5.3.3.1.4.1. The ring girder is attached to the pedestal by 120, 3¼ inch diameter high strength anchor bolts, as shown on Figure 3.8-38.

Figures 3.8-39 and 3.8-40 show reinforcement details. Openings are provided in the reactor pedestal, to permit the flow of air and suppression pool water into and out of the pedestal cavity. Additional reinforcement is furnished at the openings. A ¼ inch thick carbon steel form plate is provided on the inside and outside surfaces of the reactor pedestal below the diaphragm slab. This plate acts as a concrete form during construction, and preserves the water quality of the suppression pool by preventing the leaching of chemicals from the reactor pedestal concrete into the suppression pool.

3.8.3.1.3 Reactor Shield Wall

The reactor shield wall is a 49 foot high upright cylindrical shell which rests on the top of the reactor pedestal, and provides primary radiation shielding, as well as support for pipe restraints and drywell platforms. The reactor shield wall is constructed of inner and outer carbon steel plates, with unreinforced concrete between the two plates. Figure 3.8-41 shows details of the reactor shield wall. The reactor shield wall has an inside diameter of 25'-7", and a wall thickness of 1'-9". The outer steel plate is 1½ inches thick, and is designed to withstand local loads transferred through pipe restraints and drywell platform attachments. The inner steel plate is ½ inch thick, and is designed to act with the outer plate to withstand local and nonlocal loads. The inner and outer plates are connected with shear ties spaced on approximately 5° centers in the hoop direction. The annular space between the inner and outer plates is filled with unreinforced high density concrete for radiation shielding. The reactor shield wall is connected to the top of the reactor pedestal by 48, 2 inch diameter high strength anchor bolts, as shown on Figure 3.8-38.

The seismic truss and reactor vessel stabilizer, which provide lateral support to the reactor vessel and reactor shield, are attached to the top of the reactor shield wall. Penetrations with hinged doors or removable plugs are provided in the reactor shield wall to accommodate piping connections to the reactor vessel, and to provide access for inservice inspection. The wall thicknesses of penetration sleeves are large enough to prevent local stress concentrations in the inner and outer plates.

3.8.3.1.4 Suppression Chamber Columns

Twelve hollow steel pipe columns are furnished to support the diaphragm slab. Each column is 52'-3" long, with an outside diameter of 42 inches, and a wall thickness of 1¼ inches. The columns are connected to the base foundation slab at the bottom, and to the diaphragm slab at the top with embedded anchor bolts. Figure 3.8-14 shows the connection to the base foundation slab; and Figure 3.8-42 shows the connection to the diaphragm slab.

3.8.3.1.5 Drywell Platforms

Platforms are furnished at six elevations in the drywell to provide access and support to electrical and mechanical components. The platforms consist of structural steel framing, with steel grating. Built-up box shapes are used for beams that must resist biaxial bending. Beams that span between the pedestal or reactor shield wall, and the containment wall are provided with sliding connections at one end. Figures 3.8-3 through 3.8-8 show details of the drywell platforms.

3.8.3.1.6 Seismic Truss and Reactor Vessel Stabilizer

The seismic truss and the reactor vessel stabilizer provide lateral support for the reactor vessel during earthquake and pipe rupture loading. The seismic truss horizontally spans the gap between the containment wall and the reactor shield wall; the reactor vessel stabilizer spans the gap between the reactor shield wall and the reactor vessel. The seismic truss is shaped like an eight-pointed star, and is fabricated of steel plates. Figure 3.8-43 shows details of the seismic truss. Figure 3.8-26 shows the connection of the seismic truss to the containment wall. This connection is designed to allow vertical and radial movement of the seismic truss relative to the containment wall, but to prevent tangential movement.

3.8.3.2 Applicable Codes, Standards, and Specifications

The codes, standards, and specifications used in the design and construction of the containment internal structures are listed in Table 3.8-1.

Specifications were prepared to cover the areas related to design and construction of the containment. These specifications were prepared by Bechtel specifically for this containment. These specifications emphasize important points of the industry standards for this containment, and reduce options that would otherwise be permitted by the industry standards. Unless specifically noted otherwise, these specifications do not deviate from the applicable industry standards, and as such are not included in the UFSAR. These specifications cover the following areas:

a. Furnishing and delivering concrete

b. Forming, placing, finishing, and curing concrete

c. Furnishing, detailing, fabricating, delivering, and placing reinforcing steel

d. Splicing reinforcing bars

e. Furnishing, detailing, fabrication, and delivering the suppression chamber columns, pedestal liner, structural steel for the diaphragm slab, and miscellaneous structural steel

f. Furnishing, delivering, and erecting liner plate

Section 1.8 provides references to regulatory guides discussed in the UFSAR. Regulatory guides specific to this section are discussed in this section.

3.8.3.3 Loads and Loading Combinations

Tables 3.8-2 and 3.8-5 through 3.8-8 lists the loading combinations used for the design and analysis of the containment internal structures. Tables 3.8-5 through 3.8-8 list only the most severe factored loading combinations used for miscellaneous internal steel components of the containment. These structures have been designed according to the working stress methods, except for the pipe restraints supported on the drywell platforms. Design of the pipe restraints and their box beam supports allows inelastic deformations due to postulated pipe rupture loads. However, there is no loss of restraint function due to pipe break load.

Apparent differences between the loads listed in Tables 3.8-5 through 3.8-8 and SRP 3.8.3, for the governing loading conditions, are identified and explained in Table 3.8-20.

The internal structures are also analyzed for hydrodynamic loads resulting from MSRV discharge and LOCA phenomena. For a definition of loads and loading combinations (including hydrodynamic loads), see Reference 3.8-1 and Appendix 3A.

3.8.3.3.1 Diaphragm Slab and Reactor Pedestal

Table 3.8-2 lists the loading combinations used for the design of the diaphragm slab and reactor pedestal. Descriptions of the loads are as follows:

a. Dead Load, Live Load, and Seismic Loads

   For a description of dead load, live load, and seismic loads, see Section 3.8.1.3.

b. DBA Pressure Load

   The diaphragm slab and the reactor pedestal are designed for the following pressures:

   1. Maximum pressure: 55 psig in the drywell and the suppression chamber

   2. Maximum differential pressure: 30 psig downward (55 psig in the drywell and 25 psig in the suppression chamber); 20 psig upward (55 psig in the suppression chamber and 35 psig in the drywell).

c. Thermal Loads

   The temperatures above and below the diaphragm slab for the operating and the postulated design accident conditions are shown in Table 3.8-3. The portions of the reactor pedestal above and below the diaphragm slab are designed for the drywell and suppression chamber maximum temperatures listed in Table 3.8-3.

   Thermal effects anticipated at the time of the structural acceptance test are insignificant, since the difference in temperatures inside and outside the containment during the test is small.

d. Pipe Rupture Loads

   The diaphragm slab and the reactor pedestal are designed to withstand the pipe rupture loads due to a postulated rupture of a 28 inch diameter recirculation loop pipe, which produces the largest loads on the structures. These loads include the effects of jet impingement, pipe whip, and pipe reaction. An equivalent static load of 1000 kips is considered. This load includes an appropriate dynamic load factor to account for the dynamic nature of the load. Section 3.6 contains a detailed discussion of postulated pipe ruptures and their effects.

3.8.3.3.2 Reactor Shield Wall

The reactor shield wall was designed as a steel member using AISC working stress methods. This wall is filled with a high density grout that has a main function of shielding. No composite action is assumed between the steel and the concrete. Table 3.8-5 lists the load combination used for the design of the reactor shield wall. The governing loading combination is the abnormal/extreme loading condition that combines the design basis accident loads with the maximum seismic loads. Descriptions of the loads are as follows:

a. Dead Load, Live Load, and Seismic Loads

For a description of dead load, live load, and seismic loads, see Section 3.8.1.3.

b. DBA Pressure Load

The reactor shield wall is designed for internal pressurization due to a postulated pipe rupture at the connection of the pipe to the reactor vessel nozzle safe end. The two following pressure conditions are considered:

1. Maximum unbalanced pressure: a pressure condition shortly after pipe break, which produces the largest lateral load on the reactor shield wall, resulting in a jet force of 1330 kips.

2. Maximum uniform pressure: a 94 psig internal pressure

c. Thermal Loads

The reactor shield wall is designed for the maximum drywell temperature listed in Table 3.8-3.

d. Pipe Rupture Loads

The reactor shield wall is designed to withstand the pipe rupture loads due to a postulated rupture of any high energy pipe penetrating the reactor shield wall, and connecting to the reactor vessel, such as the recirculation and feedwater pipes. These loads include the effects of jet impingement, pipe whip, and pipe reaction. Equivalent static loads are considered, which include an appropriate dynamic load factor to account for the dynamic nature of the load. Section 3.6 contains a detailed discussion of postulated pipe ruptures and their effects.

3.8.3.3.3 Suppression Chamber Columns

Table 3.8-6 lists the load combinations used for the design of the suppression chamber columns. The columns are designed to resist the reaction loads from the diaphragm slab, for the LOCA conditions. Section 3.8.3.3.1 includes a description of the diaphragm slab loads. Normal loading conditions are not listed because the abnormal loading conditions govern the design. Allowable stresses for the suppression chamber columns are in accordance with AISC (1970), 7th edition, for steel.

3.8.3.3.4 Pipe Whip Restraints/Drywell Platforms

The drywell platforms are designed using the AISC working stress design methods, except for the pipe whip restraints supported on the platforms. Design of the pipe whip restraints and their support structure allows inelastic deformations due to postulated pipe rupture loads. When considering elastic behavior, the design of pipe whip restraints is based on AISC working stress method with allowables up to 0.9 times the yield strength. For elasto-plastic behavior, the design is in accordance with the procedure outlined in Reference 3.8-6. However, there is no loss of restraint function due to pipe break loads. All safety-related items which the inelastic deformations may affect are evaluated to verify that no required safety function would be compromised.

Table 3.8-7 lists the load combinations used to design the pipe whip restraints and the drywell platforms. Design accident pressure loads are small in comparison with pipe restraint and jet force loads and therefore do not affect the designs of the drywell platforms. The thermal load has been included in the loading combination as given in Table 3.8-7. Seismic loads due to the dead weight of the platform members are insignificant when compared with pipe rupture loads and are ignored. The uniform design live load for the grating and framing beams is 200 lb/ft^2. The live load for the framing beams also includes the gravity load, thermal reaction load, and seismic SSE reaction load of all piping and equipment supported on the beams. This design procedure is also applicable to pipe whip restraints and their support structures outside of the containment structure.

3.8.3.3.5 Seismic Truss

The seismic truss is designed using the AISC working stress design methods. It is designed primarily for lateral seismic loads. However, it is also designed for jet impingement loads due to the postulated rupture of a 26 inch diameter main steam pipe. Design accident pressure and design live loads are negligible in comparison with the main steam pipe jet forces and therefore do not affect the design of the seismic truss. The expansion of structural members under thermal loadings is permitted. Therefore, thermal loads are negligible. Table 3.8-8 lists the governing load combination used to design the seismic truss.

3.8.3.4 Design and Analysis Procedures

This section describes the procedures used for design and analysis of the containment internal structures. All computer programs referenced are described in Section 3.8.7.

For a description of the design and analysis procedures that consider the effects of hydrodynamic loads resulting from MSRV discharge and LOCA phenomena, refer to Reference 3.8-1 and Appendix 3A.

3.8.3.4.1 Diaphragm Slab

The design and analysis procedures used for the diaphragm slab are similar to those used for the containment wall. Computer programs used in the analysis include 3D/SAP, CECAP, ME620, and seismic programs. See Section 3.8.1.4 for a detailed description of the analysis procedures.

Figure 3.8-44 shows the 3D/SAP finite-element model used to analyze the diaphragm slab for all loads, other than the seismic loads. A 15° wedge of the diaphragm slab is modeled using solid finite-elements having isotropic, linear-elastic material properties. The model includes the diaphragm slab, suppression chamber wall, reactor pedestal below the diaphragm slab, and a suppression chamber column. One vertical boundary plane goes through a suppression chamber column, and the other is halfway between two columns. Boundary conditions are imposed on the analytical model by specifying nodal point forces, or displacements. With reference to Figure 3.8-44, the nodal points lying along boundary A are allowed to move within the X-Z plane, and those along boundary B within the X-Y plane. Points along boundary C are prevented from moving in the hoop direction. Points along boundary D are prevented from moving in the radial direction, to account for the restraining effect of the inner portion of the diaphragm slab. Nodal forces, moments, and shears are applied to boundaries E and F, to account for reaction loads from the drywell wall and the reactor pedestal above the diaphragm slab, respectively.

Analytical techniques as described in Section 3.7 are used to analyze the diaphragm slab for seismic loads.

3.8.3.4.2 Diaphragm Slab Liner Plate and Anchorages

The design and analysis of the diaphragm slab liner plate and anchorages is in accordance with Reference 3.8-3.

3.8.3.4.3 Reactor Pedestal

The reactor pedestal is designed for axisymmetric loads using the FINEL computer program. Both concrete and reinforcing steel materials are included in the model. The operating and design accident temperature gradients are computed using the ME620 computer program. For transient loads such as design accident pressure and thermal loads, the most critical combination of these loads is considered. Figure 3.8-45 shows a vertical section through the FINEL model of the containment, used to analyze the reactor pedestal below the diaphragm slab. Points along boundary A are prevented from moving in the vertical direction, and points along boundary B are prevented from moving in the radial direction.

Figure 3.8-46 shows the FINEL model used to analyze the reactor pedestal above the diaphragm slab. The model includes the reactor pedestal above the diaphragm slab, and portions of the reactor vessel and the reactor shield wall. Local thermal effects at the top of the reactor pedestal due to heat input from the reactor vessel are determined by using the ME620 computer program. With reference to Figure 3.8-46, nodal points along boundary A are prevented from moving vertically and radially. Nodal forces, moments, and shears are applied to boundaries B and C to account for reaction loads from the reactor vessel and the reactor shield wall, respectively.

Nonaxisymmetric loads on the reactor pedestal include seismic loads, and reactor vessel and reactor shield reaction loads. Seismic forces, moments, and shears are calculated by the methods described in Section 3.7. Vertical forces, horizontal shears, and overturning moments at the base of the reactor shield wall are determined as described in Section 3.8.3.4.4. These loads are applied to the top of the reactor pedestal. Concrete and reinforcing steel stresses in the reactor pedestal due to the above loads are calculated using the design methods of ACI 307. ACI 307 includes equations for determining the neutral axis of reinforced concrete cylindrical shells subjected to axial force and overturning moment. The position of the neutral axis satisfies the equilibrium of internal stresses and external forces and moments.

Concrete and reinforcing steel stresses due to axisymmetric and nonaxisymmetric loads are combined to determine the total stress. Additional meridional, hoop, and shear reinforcement is provided at the top of the pedestal, as shown in Figure 3.8-38, to resist local loads on the pedestal from the reactor vessel and the reactor shield wall. Helical reinforcement is not needed in the reactor pedestal to resist seismically induced tangential shears. Meridional and hoop reinforcement is designed to carry the entire tangential shear by shear friction, using the design methods of ACI 318.

3.8.3.4.4 Reactor Shield Wall

The reactor shield wall is analyzed as an axisymmetric structure. The FINEL computer program is used for the axisymmetric loads, which include dead load and design accident thermal load. The temperature gradient across the thickness of the wall is determinedby using the ME620 computer program. For nonaxisymmetic loads, which include design accident pressure load, seismic load, and pipe rupture load, the ASHSD computer program is used. Figure 3.8-47 shows a vertical section through the model used for FINEL and ASHSD programs. Points along boundary A are prevented from moving vertically and radially. For nonaxisymmetric loads, boundary B, at the connection of the seismic truss to the containment wall, is prevented from moving radially. Total stresses in the reactor shield wall are determined by summing the axisymmetric and nonaxisymmetric stresses.

A second set of analyses of the reactor shield wall, as shown in Figure 3.8-48, was done by modeling a full 360° section of the wall, and by using the EASE finite-element computer program. This analysis ensures the integrity of the shield for loads equivalent to the DBA. The model includes one 64 inch diameter recirculation outlet penetration, and two adjacent 48 inch diameter recirculation inlet penetrations. Finer mesh sizes are used in the areas of the openings to obtain a good representation of the stress gradient. With reference to Figure 3.8-48, points along boundary A are prevented from moving vertically and radially. Boundary B is a free edge. No significant local stress concentrations are noted around the openings. The stiffening of the shell is provided by the thick walled penetration sleeves.

3.8.3.4.5 Suppression Chamber Columns

Axial force, shear, and moment in the columns due to axisymmetric loads, such as dead load, and design accident pressure and thermal loads, were determined using the FINEL computer program. Figure 3.8-45 shows the FINEL model of the containment used to analyze the suppression chamber columns. Boundary conditions are described in Section 3.8.3.4.3. Since the FINEL program can consider only axisymmetric structures, the 12 columns are modeled as an equivalent cylinder having the same axial stiffness as the total of the individual column stiffnesses. The axial force in the columns is calculated from the axial stress determined by the FINEL program. Shear and moment in the columns are calculated from relative displacements of the diaphragm slab and the base foundation slab determined by the FINEL program.

Axial force, shear, and moment in the columns due to seismic loads are determined using several methods. Axial force in the columns due to horizontal seismic load is determined using the ASHSD program. Figure 3.8-49 shows the model. Axisymmetric shell and solid finite-elements having isotropic, linear-elastic material properties are used. Nodal points lying along boundary A are prevented from moving vertically, and points along boundary B are prevented from moving radially. The load applied to the ASHSD model is the seismic horizontal shear and overturning moment for the containment calculated by the methods described in Section 3.7.

Shear and moment in the columns due to horizontal seismic load are determined using the analytical procedures described in Section 3.7. The lumped-mass model of the containment, including columns and downcomers, is shown in Figure 3.8-50.

Axial force in the columns due to vertical seismic load is determined by applying the vertical forces calculated from the containment seismic analysis, to the diaphragm slab at its connections to the containment wall and the reactor pedestal. The vertical force transmitted to the columns through the diaphragm slab is calculated considering the relative vertical stiffnesses of the containment wall, reactor pedestal, and columns.

The postulated rupture of a 28 inch diameter recirculation loop pipe produces a vertical jet impingement load on the top of the diaphragm slab, and, therefore, produces loads in the columns. Axial force, shear, and moment in the columns due to jet force are calculated by the CE 668 computer program. Figure 3.8-51 shows the 180° model of the diaphragm slab. A vertical jet force is applied along the axis of symmetry, and the reaction is calculated in the column adjacent to the applied load. Edges of the diaphragm slab along boundaries A and B are considered to have fixed supports.

The total axial force, shear, and moment in the columns for all load combinations are determined by summing the results of the separate analyses. Stability of the columns for the most critical load combinations is checked using the plastic design methods of Part 2 of the AISC "Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings."

3.8.3.4.6 Drywell Platforms

The drywell platforms are designed using conventional elastic design methods, in accordance with the AISC Specification, Part 1.

3.8.3.4.7 Seismic Truss

Seismic forces in the seismic truss are calculated using the methods described in Section 3.7. Axial force, shear force, and moment in the seismic truss due to postulated pipe rupture loads are calculated using moment distribution. Figure 3.8-52 shows the rigid frame model including boundary conditions.

3.8.3.5 Structural Acceptance Criteria

3.8.3.5.1 Reinforced Concrete

The allowable stresses for the reinforced concrete portions of the containment internal structures are the same as the allowable stresses for the reinforced concrete portions of the containment, as discussed in Section 3.8.1.5.1. The calculated stresses are within the allowable limits.

3.8.3.5.2 Diaphragm Slab Liner Plate and Anchorages

The structural acceptance criteria for the diaphragm slab liner plate and anchorages are in accordance with Reference 3.8-3.

3.8.3.5.3 Structural Steel

Structural steel portions of the containment internal structures include the reactor shield wall, suppression chamber columns, drywell platforms, and seismic truss. For normal loading conditions, the allowable stresses are in accordance with the AISC Specification. For extreme environmental and abnormal loading conditions, the allowable stresses are as follows:

a. Bending - 0.90 F_y

b. Axial tension or compression - 0.85 F_y, except that where allowable stress is governed by requirements of stability (local or lateral buckling), allowable stress does not exceed 1.5 F_s

c. Shear - 0.50 F_y

where:

F_s = allowable stress according to the AISC Specification, Part 1

F_y = specified yield strength of structural steel

The calculated stresses in all the structural steel elements are within the above allowable limits.

3.8.3.6 Materials, Quality Control, and Special Construction Techniques

The criteria of ACI 349, "Proposed ACI Standard: Code Requirements for Nuclear Safety-Related Concrete Structures," applicable to this section of the UFSAR are not used by LGS. This section discusses the alternate criteria used.

3.8.3.6.1 Concrete Containment Internal Structures

The concrete and reinforcing steel materials for the containment internal structures are discussed in Section 3.8.6.

3.8.3.6.2 Diaphragm Slab Liner Plate, Anchorages, and Attachments

a. Materials

Liner plate materials conform to the requirements of the following standard specifications:

Item	Specification
Liner plate (¼ inch thick)	ASTM A285, Grade A, Firebox Quality
Liner plate (>¼ inch thick)	ASME SA516, Grade 60 conforming to the requirements of the ASME Section III, Article NE-2000
Anchorages and attachments	ASTM A36 or ASTM A441

b. Welding

Welding requirements for the diaphragm slab liner plate and anchorages are the same as the welding requirements for the containment liner plate and anchorages. See Section 3.8.1.6.2 for a description of the welding requirements.

c. Nondestructive Examination of Liner Plate Seam Welds

Spot radiographic examination is performed for all radiographable liner plate seam welds. All nonradiographable liner plate seam welds are 100% magnetic particle examined, and 100% vacuum box soap bubble tested. Welds that are inaccessible for vacuum box testing are 100% liquid penetrant tested. Examination procedures, personnel qualification, and acceptance standards are as described in Section 3.8.1.6.2.

3.8.3.6.3 Reactor Shield Wall and Seismic Truss

a. Materials

Materials used in construction of the reactor shield wall and the seismic truss conform to the following standard specifications:

Item	Specification
Inner reactor shield plate, seismic truss, and pipe restraint	ASTM A36 or ASTM A516, Grade 70
Outer reactor shield plate	ASTM A516, Grade 70

b. Welding and Nondestructive Examination of Welds

All welding and welder qualification procedures are in accordance with the requirements of ASME Section IX.

c. Materials Testing

The 1½ inch thick outer plate of the reactor shield wall is vacuum-degassed in accordance with supplementary requirements S-1 of ASTM A20, and is ultrasonically tested in accordance with ASME Section III, Subsection NB-2532-1.

d. Erection Tolerances

1. Each of the two concentric cylinders of the reactor shield wall is plumb within 1:500 of the height, plus a 1 inch allowance for local "out-of-roundness."

2. The radial dimension to any point on the reactor shield plates does not vary by more than ±1 inch from the center line as established by the design.

3. The clear distance between the two steel plates does not vary more than ±½ inch from the theoretical distance at any point.

4. The penetration sleeve center line is within ±1 inch of the projected center line of the theoretical location of RPV nozzles.

5. The elevation of the top of the shield is within ±¼ inch of that shown on the design drawings.

6. Seismic truss members do not deviate from axial straightness by more than 1/1000 of axial length.

3.8.3.6.4 Suppression Chamber Columns

a. Materials

The column shafts, base plates, and top plates are fabricated of ASME SA516, Grade 70 material.

b. Welding

Weld procedures and qualifications conform to the provisions of ASME Section III, Subsection NE, Class MC. All welders are qualified in accordance with ASME Section IX.

c. Nondestructive Examination of Welds

Complete magnetic particle examinations are performed on the welds of the suppression chamber columns, and on the top plate welds (anchor bolts to top plate and column to top plate) in accordance with the requirements of the ASME Section III, Subsection NE-5000.

d. Fabrication Tolerances

The specified fabrication tolerances for the suppression chamber columns are as follows:

1. The outside diameter, based on circumferential measurements, does not deviate from the theoretical outside diameter by more than 0.5%.

2. Out-of-roundness, defined by the difference between the maximum and minimum diameters related to the theoretical diameter, is in accordance with the ASME Section VIII, Division 1, Paragraph UG-80.

3. The finished length does not differ from the theoretical length by more than ¼ inch.

4. The finished column shaft does not deviate from straightness by more than 1/8 inch in 1 foot, with a maximum for the full length of 1/1000 of the total length.

3.8.3.6.5 Drywell Platforms

a. Materials

Materials used in construction of the drywell platforms conform to the following standard specifications.

Item	Specification
Box beams and built-up wide flange beams	ASTM A441 and ASTM A588, Grade A
Structural shapes, plate,	ASTM A36

and bar

Connection bolts	ASTM A325

b. Welding

Welding is performed in accordance with the Structural Welding Code, AWS D1.1. Visual inspection and acceptance of structural welds is performed in accordance with NCIG-01 (Reference 3.8-22).

c. Nondestructive Examination of Welds

Welds in box beams and fabricated wide flange beams are subjected to magnetic particle examination in accordance with ASTM E109 (Reapproved 1971); the standards of acceptance are in accordance with paragraph 8.15 of AWS D1.1.

d. Erection Tolerances

Erection tolerances for the drywell platforms are in accordance with the AISC Specification.

3.8.3.6.6 Quality Control

Quality control requirements during construction are discussed in the document "Limerick Generating Station Units 1 and 2; Summary Description of the Quality Assurance Program for Design and Construction," referenced in FSAR Section 17.1.

3.8.3.7 Testing and Inservice Inspection Requirements

3.8.3.7.1 Preoperational Testing

3.8.3.7.1.1 Structural Acceptance Test

The diaphragm slab is tested to 1.15 times the design downward differential pressure. Section 3.8.1.7 contains a description of the structural acceptance test. Structural acceptance test results are available after testing is complete.

3.8.3.7.1.2 Leak Rate Testing

Preoperational leak rate testing is discussed in Section 6.2.6.

3.8.3.7.2 Inservice Leak Rate Testing

Inservice leak rate testing is discussed in Section 6.2.6.

3.8.4 OTHER SEISMIC CATEGORY I STRUCTURES

This section gives information on all seismic Category I structures, other than the primary containment and its internal structures. It also describes the turbine enclosure, which is a nonseismic Category I structure. The following structures are included in this section:

Seismic Category I, Safety-Related Structures

Secondary containment (Reactor enclosures and refueling area)
Control structure
Diesel generator enclosure
Spray pond pump structure
Spray pond
Miscellaneous structures

Seismic Category I, Nonsafety-Related Structure

Radwaste enclosure (including offgas portion)

Nonseismic Category I, Nonsafety-Related Structure

Turbine enclosure

The general arrangement of these structures is shown on drawing C-2.

3.8.4.1 Description of Structures

3.8.4.1.1 Secondary Containment

The reactor enclosures enclose the primary containments and, with the refueling area, provide secondary containment (Figures 1.2-2 through 1.2-16). The secondary containment houses the auxiliary systems of the NSSS, the spent fuel pool, the refueling facility, and equipment essential to the safe shutdown of the reactor. The secondary containment is structurally integral with the control structure described in Section 3.8.4.1.2.

The secondary containment, up to and including the roof slab, is of reinforced concrete construction. Exterior bearing walls are reinforced concrete, and are additionally designed as shear walls to resist lateral loads. The floors and roof are constructed of reinforced concrete, supported by steel beam and column framing systems. The concrete slabs are designed as diaphragms to transmit lateral loads to the shear walls. The structural steel beams and girders are supported by either structural steel columns, or reinforced concrete bearing walls. The steel columns are supported by base plates attached to the foundation. The reinforced concrete walls and floors meet structural, as well as radiation shielding, requirements. At certain locations, concrete block masonry walls are used to provide better access for erecting and installing equipment. The block walls also meet the structural and the radiation shielding requirements.

The refueling facility is located above the reactor enclosures. It consists of the spent fuel pool, the steam dryer and separator storage pool, the reactor well, the cask loading pit, the skimmer surge tank vaults, a 48 foot long refueling platform crane, and a 129 foot long reactor enclosure crane. The facility is supported by end bearing walls, and by two post-tensioned concrete girders with grouted tendons. The girders run east-west, and span over the primary containments without intermediate supports. Each girder spans approximately 162 feet, and is 6 feet wide. The depth is 46 feet at the supports, and is reduced to 26 feet at midspan, where the girders straddle the containments. The ends of the girders are supported by concrete pilasters. A gap between the bottom of the girders and the top of the containments ensures that loads from the refueling facility are not transferred to the containment. The details of the post-tensioned girders, including the tendon layout, are shown in Figure 3.8-53. The walls and slabs of the spent fuel pool, the cask loading pit, the reactor cavity, and the steam dryer and separator storage pool are lined on the inside with a stainless steel liner plate. The refueling facility meets the radiation shielding requirements.

The reactor enclosure crane consists of a main and an auxiliary hoist, with capacities of 125 tons and 15 tons, respectively. The crane is used during maintenance and refueling operations. It spans approximately 129 feet, and is 28 feet above the refueling floor. The crane is mounted on two 175 pound rails, supported by a pair of runway girders. The runway girders are supported by a series of built-up columns spaced at 27 foot centers, which in turn are supported by bearing walls. Figure 3.8-54 shows the details of the runway girders and the supporting columns. The reactor enclosure crane is discussed in Section 9.1.5.

The reactor enclosure is separated from the primary containment by a gap filled with compressible material. A gap is also provided at the interface of the secondary containment with the diesel generator, radwaste, and turbine enclosures.

3.8.4.1.2 Control Structure

The control structure, shown in drawings M-110, M-111, M-112, M-113, M-114, M-115, M-124, M-125, M-126, M-127, M-128, M-129, and M-130, is a reinforced concrete enclosure, structurally integrated with the secondary containment. The bearing walls are of reinforced concrete, and are additionally designed as shear walls to resist lateral loads. The floors and roof are constructed of reinforced concrete supported by steel beams. They are designed as diaphragms to transmit lateral loads to the shear walls. The beams span in the north-south direction and are supported at the ends by the bearing walls. The reinforced concrete walls and floors meet structural, as well as radiation shielding requirements. At certain locations, concrete block masonry walls are used to provide better access for erection and installation of equipment. The block walls also meet the structural and radiation shielding requirements.

The control structure is separated from the turbine enclosure by a seismic gap.

3.8.4.1.3 Diesel Generator Enclosure

The diesel generator enclosures, shown in drawings M-145 and M-146, house the standby diesel generators, which are essential for safe shutdown of the plant.

Concrete walls separate each diesel generator enclosure into four cells, one for each of the four diesel generators provided per unit. Each diesel generator unit is enclosed in its own concrete missile-protected cell. The walls between each generator unit are 2 feet thick and are 3 hour fire rated. All penetrations are small (2-3 inches), cast in concrete conduits for electrical cables. A concrete overhang on the south side of the enclosure serves as an air intake plenum. A concrete exhaust plenum is located on the north side of the enclosure roof.

The diesel generator enclosure is a reinforced concrete structure on wall foundations. The bearing walls are of reinforced concrete, and are additionally designed as shear walls to resist lateral loads. The floors and roof are constructed of reinforced concrete supported by steel beams. They are designed as diaphragms to transmit lateral loads to the shear walls. The north side of the enclosure bears on the pipe tunnel beneath. At certain locations, concrete block masonry walls are used to provide better access for erection and installation of equipment. The diesel generators are supported by the floors.

The diesel generator structures provide protection from missiles generated by internal rotating or pressurized mechanisms to prevent any credible missiles from damaging more than one engine. Any postulated missiles from a crankcase explosion are expected to be of low energy and incapable of penetrating the concrete barrier into any other cell. Failure of one or all of the starting air receivers by explosion is not expected to produce any credible missiles capable of penetrating the 2 foot thick cell walls. The diesel generator piping is designed to withstand pipe breaks and cracks.

3.8.4.1.4 Spray Pond Pump Structure

The spray pond pump structure, shown in drawings M-388, M-389 and M-390, contains the ESW and RHRSW pumps, auxiliary equipment, and related piping and valves.

The spray pond pump structure is a two-story reinforced concrete structure. The bearing walls are of reinforced concrete, and are additionally designed as shear walls to resist lateral loads. The operating floor and roof are constructed of reinforced concrete supported by steel beams. They are designed as diaphragms to transmit lateral loads to the shear walls. A mezzanine floor composed of grating over steel beams is provided to support the heating and ventilating equipment. An intermediate floor in the wing areas is provided to support valves and piping.

3.8.4.1.5 Spray Pond

The spray pond serves as the UHS for the plant. It is shown in Figures 3.8-55 through 3.8-57. The operation of the spray pond system is discussed in Section 9.2.6.

Spray pond dimensions are given in Table 9.2-18. The spray pond is designed so that normal operating water is retained in excavation only, i.e., not by constructed embankments.

An emergency spillway is provided at the north side of the pond. The only anticipated use of this spillway is either during a malfunction of the blowdown line, or during certain postulated conditions of heavy rainfall. The emergency spillway is designed to ensure that the maximum water level does not adversely affect the spray pond system, and to direct run-off water away from safety-related facilities in a controlled manner. The roadway surrounding the remainder of the spray pond provides a minimum freeboard of 4 feet.

The bottom and soil cut slopes of the spray pond are lined with a 12 inch thick layer of soil-bentonite lining. The soil-bentonite lining is covered with a 12 inch thick layer of soil. The rock-cut slopes are lined with shotcrete.

The stability of the subsurface materials, slopes, and lining at the spray pond site are discussed in Sections 2.5.4 and 2.5.5. Protection of slopes against waves is discussed in Section 2.5.5.

The spray network piping, which is located above the water, is supported by reinforced concrete columns. The columns are founded on bedrock or on concrete fill on top of bedrock.

3.8.4.1.6 Miscellaneous Structures

Subgrade pits, manholes, and tunnels which contain safety-related components are constructed of reinforced concrete. The locations of these miscellaneous structures are shown on Figure 3.8-58.

Safety-related piping, tanks, and electrical ducts, which are not located inside structures, are buried underground with adequate cover for missile protection. Additionally, soil erosion due to failure of nonseismic piping has also been considered. The integrity of safety-related seismic Category I buried pipe will not be impaired through soil erosion by a failure of one buried nonseismic Category I pipe. This conclusion was based on the following conditions.

a. All but approximately 170 feet of common trench has been constructed in rock, where erosion of the supporting medium would be insignificant.

b. The 170 feet of trench not constructed in rock is in Type 1 fill.

   The nonpressure pipes (gravity lines) in this section, such as the blowdown line and waste and storm lines, do not pose a significant erosion problem. The nonseismic Category I pressure lines in this section consist of a 36 inch Schuylkill River makeup water line and a 12 inch fire line (Figure 3.8-58).

   Failure of the nonseismic Category I pressure pipe may create progressive erosion in the Type 1 fill. It is anticipated that water under pressure would penetrate to the surface, creating a progressively enlarging crater. However, because the water will flow in the direction of least resistance, once the water penetrates to the surface, the crater will be enlarged at a relatively slow pace. The span capacity needed to support the weight of the safety-related pipes in the trench is conservatively estimated to be in excess of 30 feet, based on the maximum allowable spans given in the ASME code. A considerably long time would be required to erode a crater large enough to exceed this span capacity.

c. Instrumentation would give indication in the control room if a break occurred in the nonseismic Category I pressure pipe. Loss of flow from the makeup water line to the cooling tower would result in an alarm in the control room when low level is reached in the cooling tower basin. It is conservatively estimated that low level would be reached within 30 minutes.

   Low pressure in the 12 inch fire line, following a break, starts a fire pump that gives an alarm in the control room without a fire signal.

   Following an SSE, if either alarm described above is activated, personnel will investigate for evidence of a faulty condition in the pipelines described in (b) above and will initiate any necessary corrective action.

d. The procedures for operator response to a seismic event will include the requirement that, within two hours after an SSE, personnel will investigate for evidence of a faulty condition in the pipelines described in (b) above and will initiate any necessary corrective action.

3.8.4.1.7 Radwaste Enclosure

The radwaste enclosure is designed in accordance with seismic Category I criteria, even though: its integrity is not required to protect the RCBP, or to ensure the capability to safely shut down the reactor; and its failure would not result in potential offsite exposures comparable to the guideline exposures of 10CFR50.67. The radwaste enclosure, shown in drawings M-140 through M-144, houses systems for receiving, processing, and temporarily storing the radioactive waste products generated during the operation of the plant.

The radwaste enclosure, which includes the offgas enclosure, is a reinforced concrete structure. The bearing walls are of reinforced concrete, and are additionally designed as shear walls to resist lateral loads. The exterior walls are waterproofed, and are designed for hydrostatic effects as necessary. The floors and roof in the main portion of the radwaste enclosure are constructed of reinforced concrete supported by beam and column framing systems. They are designed as diaphragms to resist lateral loads. The columns are supported by base plates on the foundation. The floors and roof of the offgas portion of the radwaste enclosure are of reinforced concrete supported by steel beams and bearing walls. The reinforced concrete walls and floors meet structural, as well as radiation shielding requirements. At certain locations, concrete block masonry walls are used to provide better access for erection and installation of equipment. The block walls also meet the structural and radiation shielding requirements.

The radwaste enclosure is separated from the turbine enclosure and reactor enclosure by seismic gaps.

3.8.4.1.8 Turbine Enclosure

The turbine enclosure, which is shown in drawings M-110, M-111, M-112, M-113, M-114, M-115, M-124, M-125, M-126, M-127, M-128, M-129, and M-130, is divided into two units, separated by an expansion joint. It houses two inline turbine-generator units, and auxiliary equipment including condensers, condensate pumps, moisture separators, air ejectors, feedwater heaters, reactor feed pumps, ASDs for reactor recirculation pumps, interconnecting piping and valves, switchgear, and heating and ventilating equipment.

Three 110 ton overhead cranes are provided above the operating floor for servicing both turbine-generator units. Two reinforced concrete tunnels, one for each unit, are provided for the offgas pipelines at the foundation level, running from the area around the control structure to the radwaste enclosure.

The turbine enclosure rests on a reinforced concrete mat foundation. The superstructure is framed with structural steel and reinforced concrete. Rigid steel frames support the two turbine enclosure cranes. They also resist all transverse (north-south) lateral loads. Steel bracings resist longitudinal (east-west) lateral loads above the operating floor. Below this level, reinforced concrete shear walls transfer all lateral loads to the foundations.

Seismic separation gaps are provided at the interface of the turbine enclosure with the reactor, control, and radwaste enclosures.

The floors of the turbine enclosure are of reinforced concrete supported by structural steel beams. They are designed as diaphragms for lateral load transfer to the shear walls. The roof is built-up roofing on metal decking.

Exterior walls are covered by nonstructural precast reinforced concrete panels.

Interior walls required for radiation shielding or fire protection are constructed of reinforced concrete block. These walls are not used as elements of the load resistant system.

The seismic Category II turbine enclosure may undergo some plastic deformation under seismic loading resulting from the SSE, but the plastic deformation is limited to a ductility factor of 2. Those portions of the turbine enclosure which support the main steam lines are designed so that the main steam lines and their supports maintain their integrity under the seismic loading resulting from the SSE. Furthermore, the Turbine Enclosure will maintain its integrity to ensure that the MSIV Leakage Alternate Drain Pathway will be capable of performing its function as described in Section 6.7 during and following an SSE.

The turbine-generator units are supported on freestanding reinforced concrete pedestals. The mat foundations for the pedestals are founded on rock at the same levels as the basemat for the turbine enclosure. Separation joints are provided between the pedestals, and the turbine enclosure floors and walls, to prevent transfer of vibration to the enclosure. The operating floor of the turbine enclosure is supported on vibration damping pads at the top edge of the pedestal.

3.8.4.2 Applicable Codes, Standards, and Specifications

The codes, standards, and specifications used in the design, fabrication, and construction of seismic Category I structures are listed in Table 3.8-1.

3.8.4.3 Loads and Load Combinations

Tables 3.8-9, 3.8-10, and 3.8-11 list the loading combinations considered in the design seismic Category I structures (other than the primary containment).

The reactor enclosure and control structure are also assessed for Mark II hydrodynamic loads resulting from MSRV discharge and LOCA phenomena. For a definition of loads and loading combinations including hydrodynamic loads, refer to Reference 3.8-1 and Appendix 3A. Hydrodynamic loads have no significant effect on other seismic Category I structures described in Section 3.8.4.

3.8.4.3.1 Description of Loads

3.8.4.3.1.1 Normal Loads

Normal loads are loads which are encountered during normal plant operation and shutdown. They include dead loads, live loads, thermal loads due to operating temperature, and other permanent loads contributing stress, such as hydrostatic loads. Dead and live loads are described in Sections 3.8.1.3.1 and 3.8.1.3.2, respectively.

3.8.4.3.1.2 Severe Environmental Loads

Severe environmental loads are loads that could infrequently be encountered during the plant life. They include those loads induced by the OBE and the design wind. Loads due to OBE are discussed in Sections 3.7 and 3.8.1.3.6. Wind loads are discussed in Section 3.3.

3.8.4.3.1.3 Extreme Environmental Loads

Extreme environmental loads are loads which are credible, but which are highly improbable. They include those loads induced by the SSE and the design tornado. Loads due to the SSE are discussed in Sections 3.7 and 3.8.1.3.6. Tornado loads are discussed in Section 3.3.

3.8.4.3.1.4 Abnormal Loads

Abnormal loads are loads generated by a DBA. Abnormal plant conditions generated by a DBA include the postulated rupture of high energy piping. Loads induced by such an accident include elevated temperatures and pressures within or across compartments, and jet impingement and impact forces associated with such ruptures. Loads due to postulated rupture of piping are discussed in Section 3.6. Other loads that may be generated by a DBA include railroad accident blast loads and tornado loading (including missiles generated from both), and loading from aircraft impact, as described in Sections 2.2.3 and 3.5.1.6.

3.8.4.3.2 Load Combinations

Table 3.8-9 describes the load combinations applicable to the reactor enclosure and control structure. Table 3.8-10 contains the load combinations applicable to seismic Category I structures, other than the primary containment, reactor enclosure, and control structure. Load combinations considered in the design of miscellaneous structural components of seismic Category I structures are listed in Table 3.8-11. Applicable load combinations from Table 3.8-10 are used in the design of the turbine enclosure.

3.8.4.4 Design and Analysis Procedures

The structures are designed to maintain elastic behavior for the load combinations defined in Section 3.8.4.3. All reinforced concrete components of the structure are designed by the ultimate strength method, in accordance with ACI 318. Design of major reinforced and prestressed concrete structural components was completed prior to the formal adoption of ACI 349, and this standard was therefore not used for LGS. All structural steel components are designed by the working stress method, in accordance with the AISC Specification.

Roof and floor diaphragms transfer the horizontal loads to the shear walls, which then transfer the loads to the foundation. Standard analytical procedure is used to analyze the distribution of lateral loads from the diaphragms to the shear walls. Vertical loads are transferred to the foundation by bearing walls and steel columns.

Seismic analysis of the structures uses the techniques described in Section 3.7. The enclosures are analyzed dynamically.

Design of structures for missile and aircraft protection is covered in Section 3.5.3.

The design and analysis of the fuel pool girders of the reactor enclosures are in accordance with ACI 318.

Design and analysis of concrete block walls are by the working stress method, in accordance with the UBC.

The stainless steel liner plates, used on the inside of the spent fuel pool, the cask loading pit, the reactor well, and the steam dryer and separator storage pool, are designed to function as a leak-tight membrane and to facilitate decontamination. The liner plate is designed to accommodate stresses due to the combined effect of long-term shrinkage of the underlying structural concrete and temperature effects inside the pools.

The spray pond is basically a soil-structure. Its design is discussed in Sections 2.5.4 and 2.5.5.

The reactor enclosure and control structure are also assessed for Mark II hydrodynamic loads resulting from MSRV discharge and LOCA phenomena. Assessment procedures that consider the effect of hydrodynamic loads are described in Appendix 3A.

3.8.4.5 Structural Acceptance Criteria

3.8.4.5.1 Reinforced and Prestressed Concrete

The reinforced and prestressed concrete structural components are designed by the strength method, in accordance with ACI 318, for the loads and load combinations described in Section 3.8.4.3. The stresses resulting from the load combinations are within allowable limits.

3.8.4.5.2 Structural Steel

The structural steel components are designed by the working stress method in accordance with the AISC Specification for the loads and load combinations described in Section 3.8.4.3. The allowable stresses for different load combinations are indicated therein. The stresses developed in the structural components are within the allowable limits.

3.8.4.5.3 Concrete Masonry Block Walls

The design of masonry walls was performed in accordance with the criteria in the UBC. The masonry wall design was re-evaluated for conformance with the criteria in NRC IE Bulletin 80-11. A comparison of the LGS design criteria for masonry walls with the SEB criteria shows that the two are in good agreement except in the following two areas:

a. Under extreme environmental loading conditions, the LGS criteria (appendix I, part 1, Reference 3.8-21) lists the following combinations:

1. $D + L + T_a + H_a + R + P + E'$

2. $D + L + W' + T_o + H_o$

where:

D = Dead load of structure and equipment plus any other permanent loads

L = Live loads expected to be present when the plant is operating

E' = Design basis earthquake loads

W' = Tornado loads

T_o = Operating temperature loads

T_a = Accident temperature loads

H_o = Thermal operating condition pipe load

H_a = Thermal accident condition pipe load

R = Jet impingement load and/or pipe whip

P = Pressurization load due to HELB

The SEB criteria, under similar conditions, lists the following load combinations:

1. $D + L + T_o + R_o + E'$

2. $D + L + T_o + R_o + W_t$

3. $D + L + T_a + R_a + 1.5P_a$

4. $D + L + T_a + R_a + 1.25P_a + 1.0\ (Y_r + Y_j + Y_m) + 1.25E$

5. $D + L + T_a + R_a + 1.0P_a + 1.0\ (Y_r + Y_j + Y_m) + 1.0E'$

where:

R_o = H_o of the LGS criteria

E = OBE loads

W_t = W' of the LGS criteria

R_a = H_a of the LGS criteria

P_a = P of the LGS criteria

Y_r = Equivalent static load on the structure generated by the reaction on the broken high energy pipe during the postulated break, and including an appropriate dynamic load factor to account for the dynamic nature of the load.

Y_j = Jet impingement equivalent static load on a structure generated by the postulated break, and including an appropriate dynamic load factor to account for the dynamic nature of the load, = R of the LGS criteria.

Y_m = Missile impact equivalent static load on a structure generated by or during the postulated break, as from pipe whipping, and including an appropriate dynamic load factor to account for the dynamic nature of the load, = R of the LGS criteria.

Comparing the load combination under the two criteria, it is seen that combinations (1) and (2) of the LGS criteria are the same as combinations (v) and (ii) of the SEB criteria. Further, combination (v) of SEB is clearly more severe than combination (k) of SEB which is, therefore, not used in the LGS criteria. A study made on the masonry walls and their applicable load combinations indicated that load combination (v) of SEB will, in most cases, be more severe than combinations (iii) and (iv). For the cases where this was not true, the difference in loadings was not significant and additional calculations were made to ensure that the walls would be structurally adequate under SEB load combinations (iii) and (iv).

The results of the re-evaluation contained in Reference 3.8-21 are unaffected if the SEB load combinations (iii) and (iv) are considered in addition to the LGS load combinations in the re-evaluation of the masonry walls.

b. The LGS criteria permits the use of an increase factor for shear and bond of 1.67 which, when multiplied with the allowable stresses for normal operating conditions, gives that allowables for extreme environmental conditions. The increase factors under the SEB criteria are 1.5 for tension parallel to bed joint and shear in the reinforcement and 1.3 for tension normal to the bed joint and masonry shear.

All block walls at LGS contain steel tension reinforcement which takes all tension normal or parallel to the bed joint. Therefore the LGS allowable masonry tension stress normal or parallel to the bed joint was not used, i.e., zero psi was assumed for the reanalysis of block walls. Because the masonry tension strength at the bed joint was assumed to be zero, the increase factor stated in the question was also not used in the reanalysis of the block walls. Further, the factor 1.67 has also not been used for shear or bond calculations for the masonry wall re-evaluation; instead a factor of 1.0 has been used.

Because the code allowable stresses (Reference 3.8-23, chapter 10.1 of the commentary) are generally associated with a safety factor of 3, the 1.67 increase factor for stresses for masonry load combinations involving abnormal and/or extreme environmental conditions, provides a factor of safety against failure of 1.8 (3 divided by 1.67). The factor of safety of 1.8 is conservative and allows sufficient margin for abnormal and/or extreme conditions.

Reference 3.8-21 provides additional details and results of this evaluation.

3.8.4.6 Materials, Quality Control, and Special Construction Techniques

3.8.4.6.1 Reinforced Concrete, Masonry, and Prestressed Concrete

The reinforced concrete, masonry, and prestressed concrete materials are discussed in Section 3.8.6.

3.8.4.6.2 Structural Steel

3.8.4.6.2.1 Materials

The various structural steel components conform to the following ASTM Specifications:

Item	ASTM Specification
Beams, girders, and plates	A36 or A441
High strength bolts	A325 or A490
Anchor bolts	A36 or A307

3.8.4.6.2.2 Welding and Nondestructive Testing

All welding and nondestructive testing is performed in accordance with the AWS Structural Welding Code.

3.8.4.6.2.3 Fabrication

The fabrication of structural steel conforms to the AISC Specification.

3.8.4.6.3 Quality Control

Quality control requirements during construction are discussed in the document "Limerick Generating Station Units 1 and 2; Summary Description of the Quality Assurance Program for Design and Construction," referenced in FSAR Section 17.1.

3.8.4.6.4 Special Construction Techniques

No special construction techniques are involved in the construction of seismic Category I structures.

3.8.4.7 Testing and Inservice Inspection Requirements

Testing and inservice inspections are not required for seismic Category I structures (other than the primary and secondary containment). The operational test, to verify the inleakage requirements of SGTS in the reactor enclosure, is covered in Sections 6.5.3 and 9.4.2.

3.8.5 FOUNDATIONS

This section describes foundations for all seismic Category I structures, other than the primary containment and the spray pond. The primary containment foundation is described in Section 3.8.1. The spray pond is basically a soil-structure; its design is discussed in Sections 2.5.4 and 2.5.5. A description of the foundation for the turbine enclosure, which is a nonseismic Category I structure, is also included in this section.

3.8.5.1 Description of the Foundations

Reinforced concrete foundations resting on sound rock, or lean concrete on sound rock, are provided for the structures.

Bearing walls of the structures are rigidly connected to the foundation. Steel columns are attached to the foundation by base plates and anchor bolts. The bearing walls and the steel columns carry all the vertical loads; they also carry the vertical axial loads due to the overturning effect of lateral forces, from the structure to the foundation. Horizontal shears due to wind, tornado, and seismic loads are transferred to the shear walls by the roof and floor diaphragms. The shear walls transfer the horizontal shears to the foundation, and from there, to the foundation medium through friction and/or direct bearing. The sides of all structure foundations are keyed to the foundation rock by poured concrete. This helps transfer the horizontal shears to the foundation rock.

The foundation for each structure is separated from each adjacent foundation by a 1 inch minimum seismic gap, except along the east side of the radwaste enclosure foundation, which is supported by the reactor enclosure foundation (Figure 3.8-59).

Peripheral subterranean walls are designed to resist lateral pressures due to backfill and surcharge loads, in addition to dead loads, live loads, and seismic loads.

3.8.5.1.1 Reactor Enclosure and Control Structure

The foundation for the reactor enclosure and control structure is a single integral unit. The foundation plan and representative sections are shown in Figure 3.8-60. The reactor enclosure and control structure walls are supported on continuous wall footings. Columns are supported on spread footings. The wall footings and spread footings vary in dimension and reinforcing.

All of these elements of the foundation for the reactor enclosure and control structure are joined together by a continuous 3'-0" thick mat in the reactor structure area, and by a continuous 2'-0" thick mat in the control structure. The 3'-0" thick mat surrounds the primary containment basement, with a 1 inch seismic gap separating the two.

3.8.5.1.2 Diesel Generator Enclosure

Figure 3.8-61 shows the foundation plan and typical sections of the diesel generator enclosure. The wall foundations extend well below the diesel generator base slab to competent bedrock. The base slab is supported on fill, which for design purposes is considered to be both yielding and nonyielding. In addition, the base slab is designed to span between the wall foundations, without support from the fill.

3.8.5.1.3 Spray Pond Pump Structure

Figure 3.8-62 shows the foundation plan and typical sections of the spray pond pump structure. The walls are founded on competent bedrock. The walls do not have independent footings, but are connected monolithically by a 2'-0" thick slab at the foundation level.

3.8.5.1.4 Radwaste Enclosure

Figure 3.8-59 shows the foundation plan and typical sections of the radwaste enclosure. The walls of the radwaste enclosure are founded on competent bedrock and are supported on continuous wall footings. Columns are supported on spread footings. The wall footings and spread footings of the main portion of the radwaste enclosure are joined together by a 1'-10" thick slab at the foundation level.

The walls of the offgas portion of the radwaste enclosure are supported on continuous wall footings, connected by a 1'-6", or 2'-2" thick slab at the foundation level.

3.8.5.1.5 Turbine Enclosure

Figure 3.8-63 shows the foundation plan and typical sections of the turbine enclosure. The walls are supported on continuous wall footings. Columns are supported on spread footings. The wall footings and spread footings vary in dimension and reinforcing. All of these elements of the foundation are joined together by a continuous 1'-3" to 1'-6" thick mat.

3.8.5.1.6 Miscellaneous Structures

Figure 3.8-64 shows typical foundation details of some safety-related miscellaneous structures. These structures are founded on competent bedrock, compacted fill, or lean concrete which can safely withstand the superimposed loads. The walls of the subgrade pits, manholes, and trenches are connected monolithically with the floor slabs at the foundation level. The minimum thicknesses of the walls and the floor slabs are 1'-6" and 1'-0", respectively.

3.8.5.2 Applicable Codes, Standards, and Specifications

The codes, standards, and specifications used in the design, fabrication, and construction of structure foundations are listed in Table 3.8-1.

3.8.5.3 Loads and Load Combinations

The loads and load combinations used in the design of the primary containment foundation are described in Section 3.8.1.3. The loads and load combinations used in the design of other seismic Category I structure foundations are discussed in Section 3.8.4.3.

3.8.5.4 Design and Analysis Procedures

The foundations of all seismic Category I structures are designed in accordance with the same codes and standards as used in the design of the superstructure. The design and analysis procedures are discussed in Section 3.8.1.4 and Section 3.8.4.4 for the primary containment and other seismic Category I structures, respectively. The design loads and load combinations are discussed in Section 3.8.5.3.

The bearing walls and steel columns carry all the vertical loads from the structure to the foundation. The lateral loads are transferred to the shear walls by the roof and floor diaphragms, which then transmit them to the foundation. Determination of overturning moment due to seismic loads is discussed in Section 3.7.

Settlement of all seismic Category I structure foundations is considered negligible, as the foundations are supported by sound rock, or lean concrete on top of sound rock, except those of some yard facilities, such as valve pits, manholes, pipe trenches, electrical ducts, etc, which are supported on select fill and yard fill on dense residual fill, or on sound rock.

The sliding of seismic Category I structure foundations cannot occur because, as explained in Section 3.8.5.1, the sides of the foundations are keyed to the foundation rock by poured concrete.

Detailed description of the foundation rock and soil is contained in Section 2.5.4.

3.8.5.5 Structural Acceptance Criteria

The seismic Category I structure foundations are designed to meet the same structural acceptance criteria as the structures themselves. These criteria are discussed in Sections 3.8.1.5 and 3.8.4.5. In addition, for the load combinations described in Section 3.8.5.3, the minimum factor of safety against overturning is 1.5.

3.8.5.6 Materials, Quality Control, and Special Construction Techniques

The seismic Category I structure foundations are constructed of reinforced concrete. The reinforced concrete materials are discussed in Section 3.8.6. Cement for the ESW/RHRSW cross tie valve pits in the spray pond pump house yard is Type II. Concrete for these pits is commercial grade material dedicated for nuclear safety related use. The material characteristics identified in Section 3.8.6 were used as a basis for the dedication plan.

No special construction techniques are involved in the construction of these foundations.

3.8.5.7 Testing and Inservice Inspection Requirements

Foundation testing and inservice inspection are not considered necessary, since all seismic Category I structure foundations are supported on sound rock, or lean concrete on sound rock. The only testing carried out is as described in Section 3.8.1.7.

3.8.6 CONCRETE, REINFORCING, PRESTRESSED CONCRETE, AND MASONRY MATERIALS

Materials, workmanship, and quality control associated with the construction of reinforced concrete, prestressed concrete, and masonry features of seismic Category I structures are based on the codes, standards, recommendations, and specifications listed in Table 3.8-1. These documents are modified as required to suit the particular conditions associated with nuclear power plant design and construction while maintaining structural adequacy. The extent of application and principal exceptions are indicated herein, and as follows:

a. ACI 301-66

The provisions of ACI 301-66, "Specification for Structural Concrete for Buildings", are modified as follows:

1. The requirements of section 7.3.2 of ACI 318-71 apply in lieu of section 504(b).

2. The following requirements apply in lieu of section 804(b)2:

(b)2 Hot weather - Concrete deposited in hot weather shall have a placing temperature which shall not cause difficulty from loss of slump, flash set, or cold joints and not greater than 90°F.

3. Vertical construction joints are cleaned and roughened by waterblasting, sandblasting, or bush hammering after the concrete has reached its final set. Prior to receiving additional concrete, vertical construction joints are wetted in lieu of the use of neat cement grout as specified in section 805(c).

4. The following requirements apply in lieu of section 901(a) and the fifth sentence of section 902(d):

(a) Surface defects, such as form blowholes, honeycomb, etc., are repaired as soon as practicable, but no later than 28 days after form removal.

(b) Form tie holes on the structures below grade level and in concrete which is coated with a special protective coating are patched as soon as practicable, but no later than 28 days after form removal. Form tie holes in areas other than the above need not be patched. Form tie holes which are patched need not be cured or kept moist.

5. The following requirements apply in lieu of the requirements specified in section 1202(a) and (c) and 1405(d):

The requirements of ACI 306-66 apply during cold weather. In addition, when heated enclosures are used for curing and freezing protection for all thicknesses of concrete, the temperature at the surface of the concrete is maintained between 50°F and 70°F. Any housing, covering, or other protection remains in place and intact for at least 24 hours after the artificial heating is discontinued. The maximum allowable gradual drop in temperature throughout the first 24 hours after the end of the 7 day curing period is 30°F for concrete thicknesses greater than 36 inches.

6. The following requirements apply in lieu of the requirements of sections 1401(a), 1404(a), 1404(c), 1405(a), and 1405(c), respectively:

   (a) Concrete sections more than 3 feet in the least dimension are termed mass concrete.

   (b) The maximum working limit slump of the concrete is 3 inches, with an inadvertency margin of +2 inches, except for starter mixes, which have a maximum slump of 5 inches.

   (c) Concrete is placed in layers not more than 24 inches, with vibrator heads extending into the previously placed layer.

   (d) The minimum curing period is 7 days.

   (e) The concrete is either moist cured or liquid membrane cured for 7 days. Liquid membrane compounds are applied at the time that the free water on the surface has disappeared, and no water sheen is seen, but not so late that the liquid curing compound is absorbed into the surface pores of the concrete.

   (f) The following requirements apply in lieu of section 601(c):

   The surface of the concrete at all joints is thoroughly cleaned, and all laitance removed, except where a mechanical joint is used, in which case all laitance need not be removed. The use of a mechanical joint without the removal of laitance is permitted in areas not exposed to excessive moisture.

b. <u>ACI 318-71</u>

The provisions of ACI 318-71, "Building Code Requirements for Reinforced Concrete", are modified as follows:

1. Section 3.5.1(a) of the 1974 Supplement to ACI 318-71 applies in lieu of section 3.5.1(a) of ACI 318-71.

2. Starter mixes, defined as concrete with a 1 inch maximum size aggregate 3 inches and 5 inches, are used as an alternate to mortar in section 5.4.4 and grout in section 6.4.1.

3. The following requirements apply in place of the requirements specified in section 5.5.1:

   The concrete is moist cured or liquid membrane cured for 7 days. Liquid membrane compounds are applied at the time that the free water on the surface has disappeared and no water sheen is seen, but not so late that the liquid curing compound is absorbed into the surface pores of the concrete.

4. The following requirements apply in place of the requirements specified in section 6.3.2.2:

   The temperature of the liquid, gas or vapor shall not exceed 150°F except in local areas such as penetrations through walls and slabs where the temperature shall not exceed 200°F.

5. The following requirements apply in place of the requirements specified in section 6.3.2.4:

   Piping and fittings are tested in accordance with the requirements of the code governing that piping system (e.g., ASME B&PV Code, ANSI B31.1, state or local plumbing codes, etc.), as specified in other sections of the UFSAR.

   Whenever the piping system is not governed by such applicable codes or code cases, such systems are tested for leaks prior to concreting. The testing pressure above atmospheric pressure is 50% above the pressure to which the piping and fittings may be subjected in service; but the minimum testing pressure is not less than 150 psig. The pressure test is held for 4 hours, with no drop in pressure, except that which may be attributable to changes in ambient air temperatures; or the system is maintained at the hydrostatic test pressure for a minimum of 10 minutes, while all joints are visually examined for leakage.

6. The following requirements apply in place of the requirements specified in section 6.3.2.6:

   Piping systems not governed by applicable codes or code cases, carrying liquid, gas, or vapor, which is potentially explosive or injurious to health, are retested in accordance with item b.4 above, subsequent to the hardening of the concrete.

7. The following requirements apply in place of the requirements specified in section 6.3.2.7:

   Piping systems may be energized with water exceeding neither 50 psig, nor 90°F at any time, if approved by the responsible field engineer.

   Piping systems, including systems governed by piping system codes, may be energized above either 50 psig or 90°F, or energized with fluids other than water during the period beyond 7 days after the concrete placement, provided that the temperature of the fluid does not exceed 150°F, and the pressure does not exceed 200 psig. Piping systems may be energized prior to, and remain energized during, placement of concrete, provided that: the above temperature and pressure restrictions are applied, and if the pressure in the energized system drops during concrete placement, the lowest pressure reached becomes the limiting pressure until the 7 day postplacement time limit has elapsed. However, if the pressure drops further within 24 hours after completion of the concrete placement, the system can be re-energized only up to the limiting pressure at any time after the 24 hour period has elapsed.

8. Vertical construction joints are cleaned and roughened by waterblasting, sandblasting, or bush hammering after the concrete reaches its final set. Prior to receiving additional concrete, vertical construction joints are wetted, in lieu of the use of neat cement grout specified in section 6.4.1.

9. The following requirements apply in lieu of the second sentence of section 6.4.1:

   Where a joint is to be made, the surface of the concrete is thoroughly cleaned. All standing water and laitance are removed, except where a mechanical joint is used, in which case all laitance need not be removed. The use of a mechanical joint without removing the laitance is permitted only in areas not exposed to excessive moisture.

10. The following requirements apply in lieu of the first sentence of section 4.14:

    f'_c for 5000 psi design strength concrete shall be based on tests of 90 days. f'_c for all other concrete design strengths shall be based on 28 day tests.

11. The following requirements apply in lieu of the last sentence of section 4.3.1:

    Each strength test result shall be the average of two cylinders from the same sample tested at 28 days for concrete with a design strength of 4000 psi or less and at 90 days for concrete with a design strength of 5000 psi.

c. ACI 613-54

The provisions of ACI 613-54, "Recommended Practice for Selecting Proportions for Concrete", are adhered to, except that the following recommended laboratory tests, mentioned on pages 211-14 through 211-16, are not used:

1. Fineness of Portland-Cement by Air Permeability Apparatus - ASTM Designation C204

2. Specific Gravity of Hydraulic Cement - ASTM Designation C188

3. Percentage of Shale in Aggregate - AASHTO Designation T10.

4. Alkali Reactivity, Potential, of Cement-aggregate Combinations - ASTM Designation C227

5. Air Content (volumetric) of Freshly Mixed Concrete - ASTM Designation C173

6. Air Content of Fresh Concrete by Pressure Methods (Washington-type meter) - Bureau of Reclamation Concrete Manual, Designation 24

7. Air Content of Freshly Mixed Concrete - U.S. Army Corps of Engineers Handbook for Concrete and Cement, Designation CRD-C 41

8. Laboratory Concrete Mixing - Bureau of Reclamation Concrete Manual, Designation 28

9. Flow of Portland-cement Concrete by Use of the Flow Table - ASTM Designation C124

10. Compressive Strength of Concrete Using Portions of Beams Broken in Flexure (modified cube method) - ASTM Designation C116

11. Flexural Strength of Concrete (using beam with third-point loading) - ASTM Designation C78

12. Fundamental Transverse and Torsional Frequencies of Concrete Specimens - ASTM Designation C215

13. Hardened Concrete, Securing, Preparing, and Testing Specimens from, for Compressive and Flexural Strengths - ASTM Designation C42

14. Cement Content of Hardened Portland-cement Concrete - ASTM Designation C85

15. Volume Change of Cement, Mortar and Concrete - ASTM Designation C157

16. Absorption of Concrete - AASHTO Designation T25

d. <u>ACI 614-59</u>

The provisions of ACI 614-59, "Recommended Practice for Measuring, Mixing, Transporting, and Placing Concrete", are modified as follows:

1. In addition to the requirements of ACI 614-59, storage of concrete materials shall be in accordance with ACI 304-73.

2. The following requirement applies in lieu of the second sentence of part II, item 6:

   The fineness modulus does not vary more than 0.20 from the value assumed in selecting proportions for the concrete.

3. The following requirements apply in lieu of the fourth sentence of part V, item 2:

   Concrete is not allowed or caused to flow to a distance, within the mass, more than the amount required for proper consolidation, and in no case more than 5 feet from the point of deposition.

4. The following requirements apply in lieu of the requirements of part VI, item 6:

   Immediately before the concrete is placed, all horizontal surfaces are covered with a nominal ½ inch of flowable grout mix thoroughly broomed into horizontal surfaces. For congested areas where brooming is impossible, the grout mix is forced ahead of the concrete. If the congestion may cause segregation, the grout thickness is increased to a nominal 1 inch thickness. Starter mixes, as defined above in item b.2, may be used as an alternate to the ½ inch or 1 inch grout. Starter mixes may be used to cover the bottom mat of reinforcing steel to a nominal depth of 2 inches above the reinforcing steel, or 6 inches above existing concrete on construction joints where no reinforcing mat exists.

5. The following requirement applies in lieu of the requirement of the fourth sentence of part VI, item 7:

   The maximum working limit slump of the concrete is 3 inches, with an inadvertency margin of +2 inches, except for starter mixes, which have a maximum slump of 5 inches.

e. <u>ACI SP-2, Fifth Edition</u>

The provisions of ACI SP-2, "Manual of Concrete Inspection", are modified as follows:

1. All samples are taken in accordance with ASTM C172.

2. Samples for cylinder strength tests of central mixed concrete are taken at the point of discharge from the central mixer plant. Samples for cylinder strength tests of truck mixed concrete, and slump, air content, and concrete temperature tests for all concrete are taken at the point of discharge from the delivery truck or at the end of the pump line for pumped concrete.

3. In order to establish a correlation between concrete properties at the central mixer plant and at the end of the pump line, pumped concrete is sampled at both of these locations to determine slump, air content, and concrete temperature. These correlation samples are taken from the same batch of concrete that is sampled for concrete cylinder strength tests at the central mixer plant, and again from this same batch of concrete as it reaches the end of the pump line. The correlation samples are taken at the following frequencies:

(a) Samples for slump and concrete temperature are taken from the first batch of each class of concrete produced per day, and for every 200 cubic yards of each class of concrete produced thereafter.

(b) Samples for air content are taken from every 400 cubic yards of each class of concrete produced, with a minimum of once per day for each class of concrete produced.

4. Vertical construction joints are cleaned and roughened by waterblasting, sandblasting, or bush hammering after the concrete has reached its final set. Prior to receiving additional concrete, vertical construction joints are wetted in lieu of using neat cement grout.

3.8.6.1 Concrete and Concrete Materials

3.8.6.1.1 Concrete Material Qualifications

3.8.6.1.1.1 Cement

Cement is Type II, Type I after June 1983, Portland-cement conforming to ASTM C150 except for RPV pedestal repair where the cement used is an expansive cement Type E-1(K) complying with ASTM C845. The following initial user tests are performed to ascertain conformance with ASTM Specification C-150:

Test	Designation
Chemical Analysis of Hydraulic Cement	ASTM C114
Fineness of Portland-Cement by the Turbidimeter	ASTM C115
Autoclave Expansion of Portland-Cement	ASTM C151
Time of Setting of Hydraulic Cement by Gillmore Needles	ASTM C266
Compressive Strength of Hydraulic Cement Mortars	ASTM C109
Air Content of Hydraulic Cement Mortar	ASTM C185

In addition, these tests are repeated during construction to check storage environmental effects on cement characteristics. The tests supplement visual inspection of material storage procedures.

Certified copies of material test reports showing chemical composition of the cement and verification that the cement being furnished complies with requirements are furnished by the manufacturer for each batch or lot.

3.8.6.1.1.2 Normal Weight Aggregate

Fine and coarse aggregates conform to ASTM C33. The following initial user tests are performed to ascertain conformance with ASTM C33:

Test	Designation
Organic Impurities in Sand for Concrete	ASTM C40
Effect of Organic Impurities in Fine Aggregate on Strength of Mortar	ASTM C87
Soundness of Aggregates by Use of Sodium Sulfate or Magnesium Sulfate	ASTM C88
Materials Finer Than No. 200 Sieve in Mineral Aggregates by Washing	ASTM C117
Lightweight Pieces in Aggregate	ASTM C123
Specific Gravity and Absorption of Coarse Aggregate	ASTM C127
Specific Gravity and Absorption of	ASTM C128
Fine Aggregate	
Resistance to Abrasion of Small Size Coarse Aggregate by Use of the Los Angeles Machine	ASTM C131
Sieve or Screen Analysis of Fine and Coarse Aggregates	ASTM C136
Clay Lumps and Friable Particles in Aggregates	ASTM C142
Scratch Hardness of Coarse Aggregate Particles	ASTM C235
Potential Reactivity of Aggregates	ASTM C289
Petrographic Examination of Aggregates for Concrete	ASTM C295
Test for Flat and Elongated Particles in Coarse Aggregate (except that the acceptance criteria for the ratio of flatness and elongation shall be 4 in lieu of 3)	CRD C119

Coarse aggregate grading is in accordance with size Nos. 4, 57, and 67, as defined in ASTM C33. The quantity of flat and elongated particles in the coarse aggregate is limited to 15% by weight.

3.8.6.1.1.3 High Density Aggregates

The requirements specified for normal density aggregates apply for high density aggregates, except as noted below.

Fine and coarse aggregates have a minimum bulk specific gravity of 4.4, as determined by ASTM C127 and ASTM C128. The high density aggregates are combined to produce a workable mix which meets all shielding requirements.

Certified test reports are prepared by the supplier for each material shipment, attesting to aggregate conformance to cleanliness requirements when tested per ASTM C117, and specific gravity requirements when tested per ASTM C127 and ASTM C128.

Metal aggregate consists of commercial S-930 and S-780 chilled iron or steel shot. This shot conforms to the appropriate SAE gradation presented in table 1 of the SAE Handbook.

3.8.6.1.1.4 Mixing Water and Ice

Water and ice used in mixing concrete is free of injurious amounts of oil, acid, alkali, organic matter, or other deleterious substances, as determined by AASHTO T26. Such water and ice contains no injurious impurities that would cause either a change in the setting time of portland-cement of more than 25%, as determined in accordance with ASTM C266, or a reduction in compressive strength of mortar of more than 5%, compared with results obtained with distilled water, as determined in accordance with ASTM C109 (using 2 inch cube specimens).

3.8.6.1.1.5 Admixtures

Air entraining admixtures conform to ASTM C260. Water reducing and retarding admixtures conform to ASTM C494, Type A and/or D. Types A and D are used in accordance with the manufacturer's recommendations. Certificates stating conformance to the applicable ASTM specification are furnished with each shipment. The use of any other admixture is not permitted.

3.8.6.1.2 Concrete Mix Design

3.8.6.1.2.1 Concrete Properties

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

Property	Test Designation
Compressive strength	ASTM C39
Unit weight	ASTM C138
Slump	ASTM C143
Air content	ASTM C231

Samples for property testing are taken in accordance with ASTM C172.

The following additional properties of selected mix designs have been determined to ascertain material compatibility with design assumptions:

Property	Test Designation
Modulus of elasticity	ASTM C469
Poisson's ratio	ASTM C469
Thermal diffusivity	CRD C36
Thermal coefficient of expansion	CRD C39

3.8.6.1.2.2 Concrete Mix Proportions

Concrete mix designs are prepared in accordance with paragraphs 4.2.2.2 and 4.2.3 of ACI 318-71. Trial batches are prepared in accordance with ACI 613-54.

Four compressive strength cylinders are cast at each time of sampling for all normal density concrete. Two of these cylinders are tested at 7 days. The two remaining cylinders are tested at 28 days for concrete with a design strength of 4000 psi or less and at 90 days for concrete with a design strength of 5000 psi. For high density concrete, three compressive strength cylinders are cast at each time of sampling; one is tested at 7 days, and the other two are tested at 28 days.

Concrete design strengths for the various structures are in accordance with the following table:

Item	Strength (psi)
Fuel pool girders, critically stressed walls in the reactor enclosure	5000
Reactor enclosure, primary containment, control structure, spray pond pump structures, diesel generator enclosure, radwaste enclosure, cooling towers, turbine enclosure foundation and walls below el 217'	4000
All other structural concrete	3000
Mass concrete fill	2000

The total air content of the concrete, except for grout, varies with the concrete mix and aggregate size as follows:

Concrete mixes with a 1 inch maximum aggregate size (except for D-1 (5000 psi) concrete mix) have a total air content of not less than 4%, nor more than 7% of the concrete volume.

D-1 (5000 psi) concrete mix with a 1 inch maximum aggregate size has a total air content of not less than 4%, nor more than 6% of the concrete volume for onsite concrete, 7% for offsite concrete.

Concrete mixes with a maximum aggregate size of 1½ inches have a total air content of not less than 3%, nor more than 6% of the concrete volume.

Testing for air content is in accordance with ASTM C231.

In lieu of establishing limits on the water-cement ratio, the concrete is proportioned, mixed, and placed at specified slumps. The average slump at the point of placement is less than the "working limit", which is the maximum slump for estimating the quantity of mixing water used in the concrete. An "inadvertency margin" is the allowable deviation from the "working limit" for those occasional batches which may inadvertently exceed the "working limit." Job-site tests indicate that concrete with slumps at the "inadvertency margin" produce acceptable quality concrete.

3.8.6.1.3 Grout

3.8.6.1.3.1 Construction Grout

Construction grout for use at horizontal construction joints and similar applications is proportioned from the same materials as for concrete. Such grouts have a sand/cement ratio comparable to the concrete with which it is to be used. Grout strength is determined in accordance with ASTM C109.

3.8.6.1.3.2 Starter Mix

Starter mixes are used in applications, such as at the bottom of foundation slabs, and in lieu of construction grout. They are defined as concrete with a 1 inch maximum size aggregate, with a slump between 3 inches and 5 inches.

3.8.6.1.3.3 Nonshrink Grout

Nonshrink grout is prepared from proprietary materials such as Embeco 636 by Master Builders Company, or Five Star Grout by US Grout Corporation. Such grouts are proportioned in accordance with the manufacturer's recommendations, and are tested prior to use for expansion, compressive strength, and flow characteristics, with maximum water content recommended by the manufacturer.

3.8.6.1.4 Batching, Placing, Curing, and Protection

3.8.6.1.4.1 Storage

Storage of aggregates, cement, and admixtures is in accordance with the recommendations of ACI 614-59.

3.8.6.1.4.2 Batching, Mixing, and Delivering

Concrete for principal structures is provided as central mixed concrete from an onsite batch plant, or as truck mixed concrete from an offsite plant after June 1983. Concrete blockout grout, however, may be batched by volume and provided from a mortar mixer. Batch plant facilities are certified by the National Ready Mix Concrete Association. Measuring devices are calibrated at required intervals, and more frequently where deemed appropriate.

The measuring of materials, batching, mixing, and delivery of all concrete conforms to ASTM C94, except as otherwise noted.

3.8.6.1.4.3 Placing

Placing of concrete is in accordance with the recommendations of ACI 301-66, ACI 306-66, ACI 318-71, ACI 605-59, and ACI 614-59.

3.8.6.1.4.4 Consolidation

Consolidation of concrete is in accordance with the recommendations of ACI 614-59, and as specified below.

Concrete is placed with the aid of mechanical vibrating equipment, and supplemental hand spading and tamping. The vibrating equipment is of the internal type, with a minimum frequency of 7000 revolutions per minute.

Surface vibrators, or manual jitterbugs are permitted to consolidate concrete in slabs 8 inches thick or less.

3.8.6.1.4.5 Curing

Curing of concrete is in accordance with the recommendations of ACI 301-66, ACI 306-66, ACI 318-71, and ACI 605-59.

3.8.6.1.4.6 Hot and Cold Weather Concreting

Methods and means of placing and curing concrete in cold and hot weather comply with the recommended practices of ACI 306-66 and ACI 605-59, respectively.

3.8.6.1.5 Construction Testing of Concrete and Concrete Materials

A concrete and concrete materials testing laboratory operated by Bechtel QC personnel is established at the project site to monitor the quality of such work and materials, and to promptly report any deviations from specified conditions. Such testing personnel are qualified to meet the guidelines of Regulatory Guide 1.58. Qualifications and procedures in use by Bechtel QC personnel conform with Regulatory Guide 1.94. During operation, if repairs to concrete or concrete materials are required, the Company and contractor personnel performing NDE are trained, tested, qualified, or certified in accordance with a company procedure that meets applicable requirements of 10 CFR 50.55a and ASME Section XI with specific exceptions and clarifications as discussed in the QATR.

Production testing for concrete and concrete materials is as shown in Table 3.8-12. Production testing for concrete conforms with Regulatory Guide 1.94, with the following clarification.

For central mixed concrete, samples for concrete cylinder strength and unit weight tests are taken at the point of discharge from the central mixing plant. The concrete temperature is taken at the point of discharge from the central mixer when obtaining compressive strength samples. In addition, for each load of concrete represented by compressive strength samples, the concrete temperature is taken at the point of discharge from the delivery trucks, or at the end of the pump line for pumped concrete. For truck mixed concrete furnished from an offsite source, samples for concrete cylinder strength and unit weight tests, together with temperature readings, are taken at the point of discharge from the delivery truck or at the end of the pump line for pumped concrete. Materials that do not meet test requirements are not used in the construction.

Concrete cylinder test results are reviewed for compliance with section 4.3.3 of ACI 318-71, and are evaluated in accordance with ACI 214-65. Materials or portions thereof that do not meet the above criteria, but may inadvertently be used are handled as described in the document "Limerick Generating Station Units 1 and 2; Summary Description of the Quality Assurance Program for Design and Construction," referenced in FSAR Section 17.1.

3.8.6.2 Concrete Reinforcement Materials

3.8.6.2.1 Qualification

Reinforcing steel for concrete structures conforms to ASTM A615, Grade 60, including section S1 for bar sizes 14 and 18. Certified copies of material test reports indicating chemical composition, physical properties, and dimensional compliance are furnished by the manufacturer for each heat. Spiral reinforcing steel is plain wire conforming to ASTM A82.

Prior to installation at the job-site, all reinforcing steel is subjected to a testing program meeting the guidelines of Regulatory Guide 1.15. Any reinforcing steel which does not meet these requirements is not used in the construction.

Sleeves for reinforcing steel mechanical splices conform to ASTM A519 for Grades 1018 and 1026. Certified copies of material test reports indicating chemical composition and physical properties are furnished by the manufacturer for each sleeve lot.

3.8.6.2.2 Fabrication

3.8.6.2.2.1 Bending of Reinforcement

Hooks and bends are fabricated in accordance with ACI 318-71, section 7.1. Bars partially embedded in concrete are bent subject to the following conditions:

a. Bending Partially Embedded Reinforcement

Bending of reinforcement partially embedded in concrete is performed in accordance with the following description. The diameter of the bend measured on the inside of the bar, and the distance between the beginning of the bend and the existing concrete surface is not less than "D", shown below:

Bar Size	"D"
Nos. 3 through 8	6 bar diameters
Nos. 9, 10, and 11	8 bar diameters
Nos. 14S and 18S	10 bar diameters

Bar Nos. 3 to 5 inclusive may be bent once, and straightened once, cold.

Bar Nos. 6 to 9 inclusive may be bent once, and subsequently straightened. Heating is required for both bending and straightening, as described below.

However, when the bending does not exceed 30°, bar Nos. 6 to 9 *inclusive may be bent once, and straightened once, cold.

When only cold bending is applied, the distance between the beginning of the bend and the existing concrete may be decreased to a minimum of 0.5 "D" as shown above.

When heat is used, it is applied as uniformly as possible over a length of bar equal to 10 bar diameters, and is centered at the middle of the arc of the completed bend. The maximum bar temperature is between 1100°F and 1200°F, and is maintained at that level until bending (or straightening) is complete.

Temperature-measuring crayons, or a contact pyrometer is used to determine the temperature. Heat is applied with care so as to avoid damage to the concrete. Care is taken to prevent rapid quenching of heated bars.

The bars, which are heated for straightening and bending, are visually inspected to determine whether they are cracked, reduced in cross-section, or otherwise damaged. Any damaged portions are removed and replaced.

3.8.6.2.2.2 Splicing of Reinforcement

Lap Splices

In general, lapped splices are used for No. 11 and smaller bars. Such lap splices are in accordance with sections 7.5, 7.6, and 7.7 of ACI 318-71.

Mechanical Splices

Cadweld splices conform with Regulatory Guide 1.10, except for the alternate approaches to tensile testing frequency, and the procedure for substandard tensile test results, discussed below.

Cadweld splices are used most frequently for Nos. 14 and 18 main bars in the category I concrete structures. Cadweld splices are used most frequently for Nos. 14 and 18 main bars in other Category I concrete structures. Most completed splices are visually inspected for the presence of slag, excessive porosity, or voids in the filler metal, and for proper alignment and centering of the reinforcing bar. However, there are some instances where longitudinal centering of the splice sleeve on the splice ends is off center, and probing for voids in the filler metal cannot be performed due to interferences. In these instances, a test program simulating these conditions is performed. The test results conform with the tensile strength guidelines of Regulatory Guide 1.10.

For purposes of quality control, splices representing the work of each splicing crew are tensile tested for each position, bar size, and series (T or B). Bar-to-bar splices are designated as T-series splices. Bar-to-embedment splices are designated as B-series splices.

A crew is defined as the unique combination of persons preparing, assembling, and igniting the splice. Each unique combination of persons on a crew is considered as a separate crew. In the event that one or more members of the crew is unavailable for completing a production splice lot, or performing a sister splice, the remaining member or members of the crew complete the production splice lot, or perform the sister splice. For purposes of the tensile testing program, the remaining member or members are considered as comprising the crew.

A T-series "sister splice" consists of two 14 inch (+1 inch) bars spliced in sequence with, at the same location, under the same condition, and in an otherwise identical manner as the corresponding production splice. Straight sister splice samples are taken in lieu of production splice test samples for curved reinforcing bars, and for the splices in locations outside the primary containment where space limitations prohibit taking production splice test samples.

A B-series "sister splice" is made in the configuration shown on the design drawings, and is spliced in sequence with, at the same location, under the same condition, and in an otherwise identical manner as the corresponding production splice.

The number and frequency of Cadweld splice tensile tests performed by each crew for each position, bar size, and series are as follows:

a. One sister splice is tested, prior to the start of any production splicing. Qualification splices meet this requirement if performed by each member of the crew within the previous three calendar days, and the crew has made no production splices with visual defects or which fail tensile testing since making the qualification splices.

   Deviation from this criteria was allowed in some cases for the Cadweld splice installation. This deviating condition was documented in nonconformance report NCR#14244. The Cadweld splices installed without prior sister splice testing are acceptable per engineering review documented in the nonconformance report. The omission of sister splice testing prior to production splicing does not affect the performance safety-related structures because LGS meets the intent of Regulatory Guide 1.10. For the Cadweld installation without prior sister splice testing, the test frequency as stated in paragraph (b) below was implemented and completed satisfactorily. Therefore, the satisfactory completion of the testing frequency of paragraph (b) in conjunction with the crew qualification requirements, and the visual inspection requirement for production splices, combined to assure that the level of safety required for mechanical splices discussed in Regulatory Guide 1.10 is provided.

b. One test sample out of each lot of 25 consecutive production splices is tested, including the first group of 25.

   For T-series splices, each consecutive group of three such samples consists of one production splice and two sister splices. All test samples are sister splices for the B-series splices, and for splices of No. 18 bars to square Cadweld bars of the flange skirt assembly.

   Production splice test samples are selected at random by the inspector from any within the lot of 25 splices represented. Sister splice samples are made upon completion of the lot of 25 production splices. Horizontal sister splices are side or top filled to conform to the fill location of the majority of the splices.

   If fewer than a complete lot of 25 splices are made prior to the date of a concrete placement containing any of the splices in the incomplete lot, the test splices for that lot are made prior to the placement to determine the acceptability of the splices in that lot, and the lot is then treated as a complete lot.

Welded Splices

Whenever both lap and mechanical splices have been determined to be impractical, welded splices are used on a case-by-case approval basis. Such welding is performed by qualified welders using a procedure conforming to the basic recommendations of AWS D12.1.

3.8.6.2.2.3 Placing of Reinforcement

Reinforcement is securely tied with wire, and held in position by spacers, chairs, and other supports to maintain placement accuracy within the tolerances established for reinforcement protection, and the design requirements.

3.8.6.2.2.4 Spacing of Reinforcement

Spacing of reinforcement is in accordance with ACI 318-71.

3.8.6.2.2.5 Surface Condition of Reinforcement

Reinforcement surface conditions at the time of concrete placement are in compliance with section 7.2 of ACI 318-71.

3.8.6.2.3 Construction Testing of Concrete Reinforcement Materials

Inspection of reinforcement materials to ensure that bending, placing, splicing, spacing, and surface condition requirements are met is in accordance with Regulatory Guide 1.94, except as described in Section 3.8.6.2.2.2.

3.8.6.2.4 Formwork and Construction Joints

Formwork is constructed in accordance with the applicable provisions of ACI 347-68, so that the finished concrete surfaces do not exceed the tolerances of ACI 347-68.

Construction joints are made in accordance with ACI 301-66, ACI 318-71, and ACI 614-59. In addition, all horizontal construction joints are covered, immediately before the concrete is placed, with a nominal ½ inch of flowable grout mix, or a nominal ¾ inch of starter mix.

Concrete is placed in accordance with Regulatory Guide 1.55, except as discussed below.

Regulatory positions 2 and 3 of the Regulatory Guide state the presumed functional responsibilities of the "designer" and the "constructor." Under the designer's role, are listed the responsibilities for checking shop drawings and locations of construction joints. On this project, the former responsibility is fully delegated to the Bechtel field, although the design engineering office may check significant portions, and may advise the field accordingly. The responsibility for construction joint location is partly delegated to the field, in the sense that the field has to follow the guidelines set out in the design drawings and specifications prepared by the design engineering office.
Also, Regulatory Guide 1.55 references ACI 301-72 and ACI 305-72. However, concrete at LGS is placed in accordance with ACI 301-66 and ACI 605-59 respectively.

3.8.6.3 Prestressed Concrete

3.8.6.3.1 Post-Tensioning System for Prestressed Concrete

The post-tensioning system used for prestressed concrete is the VSL E5-55 post-tensioning system, described in the 1976 Edition of the Post-Tensioning Manual, published by the Post-Tensioning Institute.

3.8.6.3.2 Concrete for Prestressed Members

The concrete used for prestressed members is a 5000 psi concrete, designed and tested in accordance with Section 3.8.6.1.

3.8.6.3.3 Prestressing Steel

Prestressing steel consists of 7 wire, stress-relieved prestressing strands meeting the requirements of ASTM A416, and having a guaranteed minimum ultimate tensile strength of 270 ksi. Stress relaxation tests in accordance with ASTM E328 are also performed. The tests are performed for each 50 tons of a heat or fraction thereof. Any prestressing steel which does not meet the test requirements is not used in the construction.

3.8.6.3.4 Sheathing

Sheathing is spiral-wrapped, semirigid corrugated tubing, fabricated from a galvanized sheet, 24 gauge minimum split strip of lock-forming quality. The sheathing meets the requirements of ASTM A527, Coating Designation G90. It is not chemically treated, or oiled.

3.8.6.3.5 Anchorage Assembly

The anchorage assembly is the system of components consisting of the anchor head, wedge, bearing plate, and trumpet. The anchor head, wedge, and bearing plate materials meet the requirements of AISI 1026, AISI 86L20, and ASTM A537, respectively. After fabrication, bearing plates are galvanized in accordance with ASTM A123. Trumpets are fabricated from galvanized sheet meeting the requirements of ASTM A569.

3.8.6.3.6 Grout

The prestressing steel is bonded to the concrete by completely filling the entire void space between the sheathing and the tendon with grout. The grout consists of a mixture of Portland-cement, water, and admixture. The cement and water used to make the grout meet the testing requirements stipulated in Section 3.8.6.1. The grout is designed to have a minimum 7 day compressive strength of 2500 psi, in accordance with ASTM C109, and a minimum 28 day compressive strength of 3500 psi, in accordance with U.S. Army Corps of Engineers CRD-C589. Three samples for each test are taken each day grouting is performed for every 5 cubic yards, or fraction thereof produced.

3.8.6.3.7 Post-Tensioning System Performance Tests

Prior to actual post-tensioning operations, the post-tensioning system is subjected to static and dynamic tests in accordance with the recommendations of section 3.1.8 of the 1976 Edition of the Post-Tensioning Manual, published by the Post-Tensioning Institute.

3.8.6.4 Concrete Unit Masonry and Masonry Material

3.8.6.4.1 Material Qualification

3.8.6.4.1.1 Concrete Unit Masonry and Building Brick

Concrete unit masonry conforms to either ASTM C90, Type I, Grade N for hollow masonry units, or ASTM C145, Type I, Grade N for solid masonry units. Concrete building brick conforms to ASTM C55, Type I, Grade N. Certified test reports are prepared for each material.

3.8.6.4.1.2 Masonry Mortar

Masonry mortar conforms to ASTM C476, Type PL.

3.8.6.4.1.3 Masonry Grout

The three types of masonry grout described below have been used for filling cavities in masonry.

a. Chemtree Grout

Chemtree grout is a premixed grout manufactured by Chemtree Corporation. It has a minimum plastic unit weight of 206 lb/ft^3, and a minimum compressive strength of 1500 psi. Tests for unit weight and compressive strength are in accordance with Section 3.8.6.1.2.1.

Chemtree grout is used to fill the cavity and cells of a small number of multiple wythe walls in the radwaste enclosure and in the auxiliary bay portion of the turbine enclosure.

b. 2000 psi Masonry Grout

The second masonry grout used for filling cavities in masonry is a 2000 psi grout designed and tested in accordance with Section 3.8.6.1, as modified below:

1. Aggregate

Coarse and fine aggregate are combined as a single aggregate, graded to produce a workable mix that meets all shielding requirements.

2. User Tests

The following user tests are performed every 6 months during production:

Test	Designation
Potential Reactivity of Aggregates	ASTM C289
Specific Gravity and Absorption of Fine Aggregate	ASTM C128

Soundness of Aggregates by Use of Sodium Sulfate or Magnesium Sulfate	ASTM C88
Materials Finer than No. 200 Sieve in Mineral Aggregates by Washing	ASTM C117
Organic Impurities in Sand for Concrete	ASTM C40

3. Compressive Strength, Unit Weight, and Slump

Three compressive strength cylinders are cast. One cylinder is tested at 7 days, and two are tested at 28 days. The minimum compressive strength, determined at 28 days, is 2000 psi.

The minimum plastic unit weight for the masonry grout is 152 lb/ft^3.

The masonry grout is allowed a slump of 9 inches ±1 inch.

c. 2000 psi Masonry Grout Using High Density Aggregates

The third masonry grout used for filling cavities is a 2000 psi grout using high density and normal weight aggregates, designed and tested in accordance with Section 3.8.6.1, as modified below:

1. Aggregate

Normal weight and high density fine aggregate are used to produce a workable mix that meets all shielding requirements.

2. User tests

The user tests for normal weight fine aggregate are defined in Section 3.8.6.1.1.2. The user tests for high density fine aggregate are defined in Section 3.8.6.1.1.3

3. Compressive Strength, Unit Weight, and Slump

These requirements are the same as those discussed in Section 3.8.6.4.1.3.b.3

3.8.6.4.1.4 Concrete Infill

Concrete infill is used as an alternative to masonry grout for filling cavities in masonry. Concrete infill conforms to the requirements described in Section 3.8.6.1.

3.8.6.4.1.5 Steel Reinforcement

Steel reinforcement conforms to the requirements described in Section 3.8.6.2.

3.8.6.4.2 Construction Testing

Testing to monitor the quality of masonry work and masonry materials is in accordance with the program described in the document "Limerick Generating Station Units 1 and 2; Summary Description of the Quality Assurance Program for Design and Construction," referenced in FSAR Section 17.1.

Test frequency during the construction of concrete unit masonry is as shown in Table 3.8-12.

Materials that are inadvertently used in the construction, but do not meet test requirements, are handled as described in the document "Limerick Generating Station Units 1 and 2; Summary Description of the Quality Assurance Program for Design and Construction," referenced in FSAR Section 17.1.

3.8.7 COMPUTER PROGRAMS FOR STRUCTURAL ANALYSIS

This section describes the computer programs used for the structural analysis of all seismic Category I structures. Each program description includes a statement of the program's area of application, and a discussion of the modifications and assumptions made. In addition, the descriptions of the computer programs not in the public domain include a selection of the sample problems used to verify their solution accuracy.

3.8.7.1 3D/SAP (Finite-Element Analysis of Three-Dimensional Elastic Solids)

3.8.7.1.1 Application

Finite-element structural analysis program 3D/SAP is capable of performing static analyses of three-dimensional solid structures subjected to concentrated or distributed loadings, thermal expansion, and/or arbitrarily directed static body forces. It is a modern version of SAP (Reference 3.8-4) written as a general purpose structural analysis computer program.

3.8.7.1.2 Program Background

Program 3D/SAP was developed by the Control Data Corporation, and is in the public domain.

3.8.7.2 ASHSD (Axisymmetric Shell and Solid)

3.8.7.2.1 Application

ASHSD is a special purpose program which can be used in the elastic, static, or dynamic analysis of structural systems capable of being represented as axisymmetric shells and/or solids.

This program allows a useful study of the interaction between a typical nuclear containment structure, modeled as an axisymmetric shell, and the subsoil, modeled as an axisymmetric solid.

3.8.7.2.2 Program Background

This program is a refinement of the original ASHSD code developed at the University of California at Berkeley. The present program is highly modified for the special purpose of static and dynamic analysis of nuclear containment structures. The modified program has the following features:

a. The code has a shell finite-element which uses an interaction stiffness, allowing analysis of layered shells.

b. Since shell layers may be bonded or unbonded from each other, it is possible to describe concrete shells in their actual geometric form. For example, it is possible to describe liner plate, concrete, reinforcing steel, and post-tensioning steel in their real spatial locations.

c. Post-tension forces may be applied to the shell by subjecting only the unbonded post-tensioning elements to a pseudothermal loading.

d. Isotropic or orthotropic elastic constants are possible for both shell and solid elements. The orthotropic material properties may be used to describe the different stiffness of reinforcing steel in the hoop and meridional directions, for example.

e. Nonuniform thermal gradients through the wall thickness may be imposed.

f. Eigenvalues and eigenvectors may be computed by the program.

g. Three dynamic response routines are available in the program. They are:

    1. Arbitrary dynamic loading, or earthquake-based excitation using an uncoupled (modal) technique.

    2. Arbitrary dynamic loading, or earthquake-based excitation using a coupled (direct integration) technique.

    3. Response spectrum modal analysis for ABS and SRSS displacements and element stresses.

h. The coupled time history solution has the capability to allow an arbitrary damping matrix.

i. The stiffness and mass matrices may be obtained as punched output for input into other programs.

The version of this program currently used by Bechtel is maintained by the Control Data Corporation.

3.8.7.2.3 Sample Problems

This program is verified by comparing the computer results with hand-calculated solutions and published references. Three sample problems are presented as examples of verification.

a. Sample Problem A: Closed Cylinder Under Internal Pressure

This problem demonstrates the membrane state of stress in a closed cylinder subjected to a uniformly distributed internal pressure. Hand calculations are used to verify this aspect of the program.

The selected problem is a cylinder with closed ends subjected to internal pressure. Only one-half of the cylinder is required in the model because of symmetry. Furthermore, it is assumed that the closed ends are distant from the section being analyzed, and they are therefore excluded.

Two models of the cylinder are actually analyzed. One model uses the thin-shell elements, and the other uses the axisymmetric solid elements. These models are shown in Figures 3.8-65 and 3.8-66 with their key dimensions.

The problem parameters for both test cases are as follows:

Boundary Conditions

Node 1: Z displacement = 0
θ displacement = 0
Rotation in R-Z plane = 0
(free to move radially)

Node 16: θ displacement = 0
(free to move axially, radially, and to rotate about the θ axis)

Numerical Data

Material: concrete

Modulus of elasticity (E) = 4.031×10^6 psi

Thickness (t) = 36 inches

Radius (R) = 900 inches

Poisson's ratio (ν) = 0.17

Pressure (p) = 60 psi

Length (L) = 1800 inches

N = 27,000 lb/in (an equivalent node load applied at Node 16)

The theoretical values for the membrane force resultants are calculated to be pR/2 (= 27,000 lb/in) axial force, and pR (= 54,000 lb/in) for the circumferential force (hoop stress).

The results obtained from the ASHSD program are presented in Table 3.8-13, both for the thin-shell and the layered shell models. Analytical computations indicate maximum errors at Node 16 of 0.4% for the longitudinal force and 3.2% for the circumferential force.

b. Sample Problem B: Cylindrical Shell Subjected to Internal Pressure and Uniform Temperature Rise

This test example demonstrates the use of a combined static load and thermal load condition. A short circular cylindrical shell clamped at both ends is subjected to an internal pressure and a uniform temperature rise.

The theoretical solutions given in Reference 3.8-4 are used to verify this analysis.

The general arrangement of the cylinder is shown in Figure 3.8-67. Only one-half of the cylinder is used for the finite-element model because of symmetry. This is shown in Figure 3.8-68, with Node 1 located at the middle of the cylinder.

The problem parameters are as follows:

Boundary Conditions

At center of cylinder, Node 1:	Z displacement = 0 θ displacement = 0 Rotation in the R-Z plane = 0
At end of cylinder, Node 26:	R displacement = 0 Z displacement = 0 θ displacement = 0 (tangential) Rotation in the R-Z plane = 0

Numerical Data

Material: concrete

Modulus of elasticity (E) = 4.031×10^6 psi

Poisson's ratio (υ) = 0.17

Thermal coefficient of expansion (α) = 55×10^{-7} in/in/°F

Thickness (t) = 30 inches

Radius (R) = 600 inches

Length (L) = 1200 inches

Height (Z) = 600 inches

Pressure (p) = 60 psi

Temperature (T) = 150°F

R/t = 20

L/R = 2

The theoretical results are shown in Figure 3.8-69. These values are obtained by using the following equations from Reference 3.8-4:

Axial Moment: $M_x = 2\mu^2 D_x \left(\frac{PR^2}{Et} + R\alpha T\right)$ (EQ. 3.8-1)

where:

$$\mu^2 = \left[\frac{3(1-\upsilon^2)}{R^2 t^2}\right]$$

$$D_x = \frac{Et^3}{12(1-\upsilon^2)}$$

Normalized length: Ln = (Z/R)(L/2R)

Figure 3.8-69 compares the results obtained from the ASHSD program and the theoretical solution. The results of ASHSD agree well with those of the reference.

c. Sample Problem C: Asymmetric Bending of a Cylindrical Shell

This sample problem illustrates the use of higher harmonics for asymmetric loading cases. As a comparison to the computer output, results for this problem are taken from Reference 3.8-5.

The cylindrical shell analyzed is a short, wide cylinder as shown in Figure 3.8-70. The finite-element idealization of the cylinder and the pertinent data are illustrated in Figure 3.8-71. At each end of the cylinder, moments of the form $M = M_o \cos\eta\theta$ were input for harmonics η = 0, 2, 5, 20.

The problem parameters are:

Material: steel

Modulus of elasticity (E) = 29×10^6 psi

Thickness (t) = 1.25 inches

Radius (R) = 60.0 inches

Poisson's ratio (υ) = 0.3

Length (L) = 60.0 inches

L/R = 1

$R/t = 48$

$$M_o = \frac{Et^2}{100(1-\upsilon^2)} = 497939.56 \text{ in-lb/in} \quad \text{(EQ. 3.8-2)}$$

The comparison results are taken directly from Reference 3.8-5. Those results are plotted in Figure 3.8-72.

The comparison of the computer results to the reference results are shown in Figure 3.8-72. (Note that the longitudinal moments and radial displacements are expressed as nondimensional ratios).

The reference and computer program results show good agreement. This verifies the accuracy of the program for this type of analysis.

3.8.7.3 CECAP (Concrete Element Cracking Analysis Program)

3.8.7.3.1 Application

CECAP computes stresses in a concrete element under thermal and/or nonthermal (real) loads, considering effects of concrete cracking. The element represents a section of a concrete shell or slab, and may include two layers of reinforcing, transverse reinforcing, prestressing tendons, and a liner plate.

The program outputs stresses and strains at selected locations in the concrete, reinforcement, tendons, and liner plate, and resultant forces and moments for the composite concrete element.

3.8.7.3.2 Program Background

CECAP assumes linear stress-strain relationships for steel and for concrete in compression. Concrete is assumed to have no tensile strength. The solution is an iterative process, whereby tensile stresses found initially in concrete are relieved (by cracking) and redistributed in the element. The equilibrium of nonthermal loads is preserved. For thermal effects, the element is assumed free to expand inplane, but is fixed against rotation. The capability for expansion and cracking generally results in a reduction in thermal-stresses from the initial condition.

The version of this program currently used by Bechtel is maintained by the Control Data Corporation.

3.8.7.3.3 Sample Problems

Sample problems are analyzed by CECAP and compared with hand-calculated solutions. These sample problems consider a reinforced concrete beam as shown in Figure 3.8-73. The parameters for all sample problems are as follows:

Modulus of elasticity of concrete (E_c) = $3x10^6$ psi

Modulus of elasticity of reinforcing steel (E_s) = $30x10^6$ psi

Poisson's ratio for concrete (υ_c) = 0.22

Coefficient of thermal expansion of concrete
(α_c) = $6x10^{-6}$ in/in/°F

Temperature difference (ΔT) = 100°F

Coefficient of thermal expansion of reinforcing steel
(α_r) = α_c

Three sample problems are presented as examples of verification.

a. Sample Problem A: Beam with a Thermal Moment

The analysis of a reinforced concrete beam subjected to a linear thermal gradient is performed to test the redistribution of thermal-stresses due to the relieving effect of concrete cracking.

Figure 3.8-74 shows the reinforced concrete beam, and the corresponding CECAP concrete element used in the analysis. Boundary conditions, geometry, and applied loads are illustrated.

The following illustrates how thermal loads are treated in a cracked section analysis of a reinforced concrete beam. The main assumptions pertaining to thermal boundary conditions are:

1. The beam is allowed to freely expand axially.

2. There is no rotation of the initial thermal-stress slope.

The beam cross-section and initial thermal-stress distribution are shown in Figure 3.8-75. For ΔT = 100°F, the equivalent thermal moment and thermal-stresses in concrete and steel are:

Thermal moment (M) = $\Delta T \alpha_c E_c b t^2/12$ (EQ. 3.8-3)
= 3,175,000 in-lbs

Concrete stress (σ_c) = $\Delta T \alpha_c E_c/2$ (EQ. 3.8-4)
= 900 psi (compression)

Rebar stress (σ'_c) = $\dfrac{t/2-2}{t/2}\,\sigma_c$ (EQ. 3.8-5)
= 814 psi (tension)

where:

b = width of beam = 12 inches

t = depth of beam = 42 inches

E_c = modulus of elasticity = $3\text{x}10^6$ psi

α_c = coefficient of linear expansion for concrete
= $6\text{x}10^{-6}$ in/in/°F

The stress diagram used for the cracked section analysis with thermal loading is shown in Figure 3.8-76. The assumptions of free movement axially and constant thermal-stress slope are maintained by a lateral translation of the initial reference axis to a final cracked position.

From force equilibrium:

$$F_{rebar} + F_{concrete} = 0 \qquad \text{(EQ. 3.8-6)}$$

where:

$F_{rebar} = f_s$ = total stress in steel

$= 1.0\,(814 + \Delta\sigma_c)\,10$ psi

$F_{concrete} = f_c$ = total stress in concrete

$$= -\,900\left(\frac{42}{2}\right)\left(\frac{12}{2}\right) + \frac{\Delta\sigma_c\,(12)}{2}\left[21 + \left(\frac{900-\Delta\sigma_c}{900}\right)2\right]$$

Solving for $\Delta\sigma_c$:

$\Delta\sigma_c = 582$ psi

From above, the rebar and concrete stresses are:

$f_s = (814+582)10 = 13{,}970$ psi (tension)

$f_c = 900-582 = 318$ psi (compression)

The location of the cracked neutral axis is:

$$kd = x = \frac{900-582}{900}\,(21) = 7.42 \text{ in} \qquad \text{(EQ. 3.8-7)}$$

The self-relieved thermal moment is:

$$M_T = \frac{f_s A_s (d - x/3)}{12} = 43{,}690 \text{ in-lb} \qquad \text{(EQ. 3.8-8)}$$

The rebar and concrete stresses, self-relieved thermal moment, and neutral axis location obtained from the CECAP program are compared with the hand calculations in Table 3.8-14. It can be seen that the CECAP results compare favorably with the hand calculation.

b. Sample Problem B: Beam with a Real Moment

The analysis of a reinforced concrete beam subjected to a real moment tests the CECAP program for nonthermal moments.

Figure 3.8-77 shows the loading and geometry for the reinforced concrete beam, and the corresponding CECAP concrete element model.

The following illustrates the working stress analysis of reinforced concrete beams. The beam cross-section, stress block, and transformed sections are shown in Figure 3.8-78. The resultant forces and moment are:

$$C = f_c\,(kd)(b)/2 \qquad \text{(EQ. 3.8-9)}$$

$$T = A_s(f_s) \qquad \text{(EQ. 3.8-10)}$$

$$M = Cjd = Tjd \qquad \text{(EQ. 3.8-11)}$$

Equating the first moments of the compression and tension areas about the neutral axis of the transformed section,

$$kd(b)\left(\frac{kd}{2}\right) = nA_s\,(d - kd) \qquad \text{(EQ. 3.8-12)}$$

where:

$$n = E_s/E_c = 10.0$$

which yields:

$$(kd)^2 + 1.67kd - 66.67 = 0$$

Solving for kd:

$$kd = 7.37 \text{ in}$$

For an applied moment (m) of 3,175,000 in-lbs, the resultant forces are:

$$C = T = \frac{M}{jd} = \frac{3{,}175{,}000}{\left(40 - \frac{7.37}{3}\right)} = 84{,}570 \text{ lb} \qquad \text{(EQ. 3.8-13)}$$

Rebar and concrete stresses, respectively, are:

$$f_s = \frac{T}{A_s} = 84{,}570 \text{ psi (tension)} \qquad \text{(EQ. 3.8-14)}$$

$$f_c = \frac{2C}{kd(b)} = \frac{2(84{,}570)}{(7.37)(12)} \qquad \text{(EQ. 3.8-15)}$$

$$= 1193 \text{ psi (compression)}$$

Table 3.8-14 shows a comparison of rebar and concrete stresses, and neutral axis locations obtained from the CECAP program and hand calculations. The CECAP results are shown to compare to hand calculations within the force accuracy limits in the program.

c. Sample Problem C: Beam with a Real Moment and a Real Axial Load

This sample problem involves the analysis of a reinforced concrete beam subjected to both a real moment, and a real axial compressive load.

The loading and geometry for the reinforced concrete beam and corresponding CECAP model are illustrated in Figure 3.8-79.

The following illustrates the working stress analysis of reinforced concrete beams subjected to both moments and axial compressive loads. The beam cross-section and stress block are shown in Figure 3.8-80. The analysis uses the equations presented in Reference 3.8-7, simplified as follows:

$$(kd)^3 + 3\left[\frac{M}{N} - \frac{t}{2}\right](kd)^2 + \frac{6nA_s}{b}\left[d - \frac{t}{2} + \frac{M}{N}\right](kd) - \frac{6nA_s d}{b}\left[d - \frac{t}{2} + \frac{M}{N}\right] = 0 \qquad \text{(EQ. 3.8-16)}$$

$$f = \frac{N}{A_s}\left[\frac{M}{N} + \frac{kd}{3} - \frac{t}{2}\right] / \left[d - \frac{kd}{3}\right] \qquad \text{(EQ. 3.8-17)}$$

$$f = \frac{f_s kd}{n(d - kd)} \qquad \text{for } \frac{M}{N} \geq t/6 \qquad \text{(EQ. 3.8-18)}$$

where:

M = moment = 375,000 in-lbs

N = axial load = 101,000 lbs

t = depth of beam = 42 inches

b = width of beam = 12 inches

Equation 3.8-16 becomes:

$$kd^3 + 55.8kd^2 + 293kd - 11720 = 0$$

$$M/N = \frac{3,175,000}{101,000} = 31.4 \geq t/6 = \frac{42}{6} = 7$$

Solving the above by iteration for (kd) yields:

kd = 12.7 in

The resulting rebar and steel stresses are:

f_s = 41,320 psi (tension)

f_c = 1922 psi (compression)

The rebar and concrete stresses, and neutral axis location obtained from the CECAP program are compared with the hand calculations in Table 3.8-14. The results for the two solution methods agree very closely.

3.8.7.4 CE668 (Plate Bending Analysis)

3.8.7.4.1 Application

This program performs the linear-elastic analysis of a plate with arbitrary shape and support, stiffener beams, and elastic subgrade, under loads normal to the middle plane of the plate.

3.8.7.4.2 Sample Problems

Sample problems were analyzed by CE668, and compared with hand-calculated solutions.

a. Sample Problem A: Rectangular Plate with a Concentrated Load at the Center

The simply supported rectangular plate, shown in Figure 3.8-81, is subjected to a concentrated load of 300 pounds at its center. Only one-half of the plate is modeled by finite-elements, because of symmetry. The boundary conditions are zero-displacement with free normal rotation at the simply supported edges, and free displacement with zero normal rotation at the symmetry axis. The plate has isotropic structural properties.

The problem parameters are:

Poisson's ratio (υ) = 0.3

Modulus of elasticity (E) = $29\text{x}10^6$ psi

Plate thickness (h) = 0.5 in

Concentrated load (P) = 300 lb

Width of plate (a) = 10 inches

Length of plate (b) = 40 inches

The formulas for the deflections and moments are taken from Reference 3.8-8.

Deflection (at Center)

$$\omega = 0.01695 \frac{Pa^2}{D} \qquad \text{(EQ. 3.8-19)}$$

where:

$$D = \frac{Eh^3}{12(1-\upsilon^2)}$$

ω = 0.00153 in at Node 116

Moments

M_x: (for b >> a) (at x = 2, y = 0)

$$M_x = \frac{-P(1+\upsilon)}{8\pi} \ln \left[\frac{1 - \sin \frac{\pi x}{a}}{1 + \sin \frac{\pi x}{a}} \right] \quad \text{(EQ. 3.8-20)}$$

M_x = 20.92 in-lb at Node 113

M_y: (for b >> a) (at x = 6, y = 0)

$$M_y = -P\frac{(1+\upsilon)}{8\pi} \ln \left[\frac{1 - \sin \frac{\pi x}{a}}{1 + \sin \frac{\pi x}{a}} \right] \quad \text{(EQ. 3.8-21)}$$

M_y = 57.198 in-lb at Node 117

The hand-calculated values for deflections and moments are compared with the CE668 values in Table 3.8-15. The results for the two solution methods are in very close agreement, the largest difference being 1.55%.

b. Sample Problem B: Uniform Load on a Rectangular Plate with Various Edge Conditions

For this problem, the rectangular plate has one edge fixed, one edge free, and two edges simply supported, as shown in Figure 3.8-82. The plate is subjected to a uniformly distributed load (q) equal to 2.0 psi. Only one-half of the plate is modeled by finite-elements, because of symmetry. Boundary conditions are specified in accordance with the edge support conditions.

The problem parameters are as follows:

Poisson's ratio (υ) = 0.3

Modulus of elasticity (E) = 29×10^6 psi

Plate thickness (h) = 0.2 in

Uniformly distributed load (q) = 2.0 psi

Plate width (b) = 15 inches

Plate length (a) = 30 inches

The formulas for the deflections and moments were taken from Reference 3.8-8.

Deflection (at x = 15, y = 15)

$$\omega = 0.0582 \frac{qb^4}{D} \qquad \text{(EQ. 3.8-22)}$$

where:

$$D = \frac{Eh^3}{12(1-\upsilon^2)}$$

ω = 0.277 in at Node 11

Moments

M_x: (at x = 15, y = 15)

M_x = $0.0293\ qa^2$

= 52.74 in-lb at Node 11 (EQ. 3.8-23)

M_y: (at x = 15, y = 0)

M_y = $0.319\ qb^2$

= 143.55 in-lb at Node 121 (EQ. 3.8-24)

The hand-calculated values for deflections and moments are compared with the CE668 values in Table 3.8-15. The results for the two solution methods are in close agreement, the largest difference being 3.4%.

3.8.7.5 EASE (Elastic Analysis for Structural Engineering)

3.8.7.5.1 Application

EASE is a finite-element program which performs static analyses of two-dimensional and three-dimensional trusses and frames, plane elastic bodies, and plate and shell structures. The program library contains various types of elements, including truss, beam, plane stress and strain, plate bending, and shell elements. The EASE program accepts thermal loads, as well as pressure, gravity, and concentrated loads.

The program output includes joint displacements, beam forces and element stresses, and forces and moments.

3.8.7.5.2 Program Background

EASE was developed by the Engineering/Analysis Corporation, Redondo Beach, California, in 1969, and is in the public domain. The version currently used by Bechtel is maintained by the Control Data Corporation, Cybernet Service.

3.8.7.6 E0119

3.8.7.6.1 Application

This program analyzes a bolted flange. The program calculates allowable and actual bolt stresses, and longitudinal, radial, and tangential flange stresses.

3.8.7.6.2 Program Background

The symbols, terms, and analysis procedures used in this program are in accordance with ASME Section III, Appendix XI.

The version of this program currently used by Bechtel is maintained by the Control Data Corporation.

3.8.7.6.3 Sample Problems

Sample problems were analyzed by E0119 and compared with the solutions published in Reference 3.8-9.

The problem parameters for the two sample problems are as follows:

Design pressure = 400 psi

Design temperature = 500°F

Atmospheric temperature = 75°F

Poisson's ratio = 0.30

Corrosion allowance = 0

Gasket width = 0.75 inches

Effective gasket width = 0.306 inches

Gasket factor = 2.75

Gasket seating strength = 3700 psi

a. Sample Problem A: Welding Neck Flange

   Figure 3.8-83 details the dimensions of the welding neck flange. Table 3.8-16 compares the results obtained using E0119 with the results published in Reference 3.8-9.

b. Sample Problem B: Slip-on Flange

Figure 3.8-84 details the dimensions of the slip-on flange. Table 3.8-16 compares the results obtained using E0119 with the results published in Reference 3.8-9.

In both problems, the results obtained using E0119 compare favorably with the published results.

3.8.7.7 E0781 (Shells of Revolution Program)

3.8.7.7.1 Application

This program calculates the stresses and displacements in thin-walled elastic shells of revolution when subjected to static edge, surface, and/or temperature loads with arbitrary distribution over the surface of the shell. The geometry of the shell must be symmetric, but the shape of the median is arbitrary. It is possible to include up to three branch shells with the main shell in a single model. In addition, the shell wall may consist of different orthotropic materials, and the thickness of each layer and the elastic properties of each layer may vary along with the median.

3.8.7.7.2 Program Background

The Shells of Revolution Program is based on the analysis described in Reference 3.8-10.

The program numerically integrates the eight ordinary first-order differential equations of thin-shell theory. The equations are derived so that the eight variables chosen are those appearing on the boundaries of the axially symmetric shell; thus the entire problem is expressed in these fundamental variables.

The program has been altered so that a 4x4 force-displacement relation is used as a boundary condition, as an alternative to the usual procedure of specifying forces or displacements. This force-displacement relation describes the forces at the boundary in terms of displacements at the boundary, or the displacements at the boundary in terms of forces, or some compatible combination of the two. In this manner, it is possible to study the behavior of a large complex structure. It is also possible to introduce a "Spring Matrix" at the end of any part of the stress model. This matrix must be expressed in the form, Force = Spring Matrix · Displacement. In addition to the above changes, the program has been modified to increase the size of the problem that can be considered and to improve the accuracy of the solution.

3.8.7.7.3 Sample Problems

This program is verified by comparing the computer results with experimental measurements and published references. Two sample problems are presented as examples of verification.

a. Sample Problem A: Comparison of 2:1 Ellipsoidal and Torispherical Heads Subjected to an Internal Pressure Load

   This problem illustrates the program's ability to generate cylindrical, torispherical, and ellipsoidal shapes. The E0781 program's "solution" is compared to the experimental investigation of 2:1 ellipsoidal heads subjected to internal pressure discussed in Reference 3.8-11.

   The problem compares a 2:1 ellipsoidal head, to an equivalent torispherical head subjected to the same uniformly distributed internal pressure. An equivalent torisphere is defined as one having the same height above the tangent line as the ellipsoid, and a minimal L/b ratio (thus having the least possible discontinuity between the torus and the sphere). For the geometry shown in Figure 3.8-85:

$$(L\text{-}b) \sin \phi_o = A\text{-}r \qquad \text{(EQ. 3.8-25)}$$

$$(L\text{-}b) \cos \phi_o = L\text{-}B \qquad \text{(EQ. 3.8-26)}$$

Minimizing L/b using equations 3.8-25 and 3.8-26:

$$\tan \phi_o = B/A = 0.5019 \qquad \text{(EQ. 3.8-27)}$$

$\phi_o = 26.653°$

$$L/A = \frac{c \pm (c^2 - 2c)^{½}}{2} \qquad \text{(EQ. 3.8-28)}$$

where:

$c = B/A + A/B = 2.494$

$L = \frac{18.19}{2} [2.5 + (6.22 - 4.99)^{½} = 32.778$ in

$b = B [B/A - L/A] + A = 6.32$ in

and:

A = radius of cylinder = 18.19 inches

B = height of torisphere or ellipsoid above the base (tangent line) = 9.31 inches

L = major radius of torisphere

b = minor radius of torisphere

t = thickness of the shell

Segment lengths used are:

<u>cylinder</u>

$$(rt)^{½} = 2.37 \text{ inches} \qquad \text{(EQ. 3.8-29)}$$

where:

r = distance to pole = 18.16 inches

t = thickness of shell = 0.31 inches

<u>torisphere</u>

5° to 10° - 4 angular segments @ 1.25°

10° to 26.567° - 4 angular segments @ 4.13°

26.567° to 90° - 6 angular segments @ 10.57°

<u>ellipsoid</u>

5° to 10° - 2 4 angular segments @ 1.25°

10° to 30° - 4 angular segments @ 5°

30° to 90° - 6 angular segments @ 10°

Boundary Conditions:

It is assumed that at 5° from the pole, a membrane stress exists in both the ellipsoid and the torisphere:

$$Q = M\phi = 0 \qquad \text{(EQ. 3.8-30)}$$

$$N\phi = \frac{pr}{2 \sin \phi} \qquad \text{(EQ. 3.8-31)}$$

where:

r = distance to pole = 32.778 in

Q = transverse shear in ϕ direction, lb/in

$M\phi$ = moment resultant in ϕ direction, in-lb/in

$N\phi$ = membrane force in ϕ direction, lbs/in

Letting (p) = pressure = 680 psi, then for the torisphere:

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

If $N\phi$ = 11,144.5 lb/in, a preliminary run yields:

Q = 95.202 lb/in, so a new value for $N\phi$ for the torisphere is calculated:

$$\Delta N = \frac{\Delta Q}{\tan \phi} \qquad \text{(EQ. 3.8-32)}$$

$N\phi$ = 11,144.5 + ΔN = 10056.3 lb/in and an appropriate membrane state was generated.

For the ellipsoid:

$$r = \frac{A \sin \phi}{R} \qquad \text{(EQ. 3.8-33)}$$

where:

$R = [C_1 + (1 - C_1)\sin^2\phi]^{1/2}$

$C_1 = (B/A)^2 = 0.2519$

$R = 0.5075$

$N\phi = \frac{A \sin\phi}{R} \frac{P}{2\sin\phi} = 12{,}185.78$ lb/in

To better compare the heads, it seems desirable to have the longitudinal displacement at the center of the cylinder equal to zero ($\mu\phi = \phi$). Thus the problem is run twice; the first run yielding the radial displacement, W, required for zero-displacement at the center (W = 0.0966 in).

Start	W = 0.0966 in
	$N\phi$ = 10,056 lb/in
	$M\phi$ = N = 0
End	Q = N = M0 = 0
	$N\phi$ = 12,186 lb/in

Figure 3.8-86 shows the analytical model, with boundary conditions.

In order to check the results, the answers at the boundaries are examined first. It is assumed that there is a membrane state of stress at the boundaries. Therefore, at the edges Q and $M\phi$ must be approximately zero.

	Q (lb/in)	$M\phi$ (lb-in/in)
Start	-0.08636	0.0
End	-0.0009252	0.0001487

To satisfy equilibrium in the cylinder:

$N\phi \approx 0.5pr = 6169$ lb/in

Plots of the hoop force and longitudinal bending from E0781 results compare the ellipsoidal and torispherical heads. Even though the change in radii is minimized, the disturbance at the junction of the sphere and torus is considerable (Figure 3.8-87).

A comparison with the experimental ellipsoidal head shows a good correlation of stress values. Plots of $\nabla\phi$ and $\nabla\theta$ on the inside, outside, and meridian of the head are shown in Figures 3.8-88 through 3.8-92. Any deviations are caused by the changes in thickness, and by the experimental head's variation from a true 2:1 ellipsoidal head. These variations are shown in Figures 3.8-93 and 3.8-94.

b. Sample Problem B: Cylindrical Water Tank with Tapered Walls

This problem illustrates the program's capability to analyze a pressure load with one fixed boundary and one free boundary.

The problem used for this verification is "Shell of Variable Thickness," taken from Reference 3.8-12 (pp. 289-295).

The problem consists of a tapered shell filled with water. The shell has a radius of 9 feet, and is 12 feet high. The shell thickness varies from 11 inches at the bottom, to 3 inches at the top. These dimensions, and the location of the Z axis, are shown in Figure 3.8-95. The length of a segment is 18 inches $[(rt)^{1/2}]$, where r and t are the same as defined in Sample Problem A.

Since the pressure varies linearly, only two points are needed in the function generator in order to fully describe the function.

It is known that the pressure at the top of the tank, exposed to the atmosphere, is zero. Assuming that the density of the water is 62.5 lb/ft^3, the pressure at the bottom of the tank is:

$$p = \frac{(12\ ft)(62.5\ lb/ft^3)}{(144\ in^2/ft^2)} = 5.21\ psi$$

The boundary conditions for this problem are as follows:

W = displacement normal to surface

$U\phi$ = displacement component in ϕ direction

$\beta\phi$ = rotation of reference surface in ϕ direction

Q = transverse shear in ϕ direction

$N\phi$ = membrane force in ϕ direction

$M\phi$ = moment resultant in ϕ direction

Fixed at start $W = U\phi = \beta\phi = 0$

Free at end $Q = N\phi = M\phi = 0$

Table 3.8-17 lists the Program E0781 results, and compares them with the theoretical solutions from Reference 3.8-12 at two locations.

Program E0781 yields a maximum hoop force, $N\phi$, equal to 346.8 lb/in (4160 lb/ft), 54 inches from the base. This value differs from the theoretical solution of 4180 lb/ft by 0.48%.

Program E0781 yields a maximum moment at the base, M□, equal to -1539 in-lb/in (-1539 ft-lb/ft). This value differs from the theoretical solution of -1470 ft-lb/ft by 4.69%.

3.8.7.8 FINEL (Finite-Element Program for Cracking Analysis)

3.8.7.8.1 Application

This program performs a static analysis of stresses and strains in plane and axisymmetric structures by the finite-element method. The program computes the displacements of the corners of each element, and the stresses and strains within each element.

3.8.7.8.2 Program Background

In this program, the structure is idealized as an assemblage of two-dimensional finite-elements of triangular or quadrilateral shapes, having arbitrary material properties. Reinforcement of concrete materials is included by adjusting the element material properties. Bilinearity in compression, and bilinearity or cracking in tension are specially emphasized.

The version of this program currently used by Bechtel is maintained by Control Data Corporation.

3.8.7.8.3 Sample Problems

To verify this program, sample problems were analyzed by FINEL, and compared to experimental and/or hand-calculated solutions. Three sample problems are presented as examples of verification.

a. Sample Problem A: Simply Supported Beam with a Concentrated Load at the Center

The beam shown in Figure 3.8-96 is investigated experimentally and analytically. This investigation compares results obtained from the FINEL program with those obtained from References 3.8-13 and 3.8-14.

The finite-element mesh used in Reference 3.8-14, and in the FINEL analysis are shown in Figures 3.8-97 and 3.8-98, respectively. The FINEL analysis requires a finer mesh because it uses linear displacement elements. Reference 3.8-14 uses quadratic displacement elements.

The material properties of the concrete and reinforcing steel, and the loading history used in the FINEL analysis are given in Table 3.8-18.

The cracking patterns obtained from Reference 3.8-14 and FINEL are shown in Figure 3.8-99. The load-deflection curves from References 3.8-13 and 3.8-14 and the FINEL analysis are shown in Figure 3.8-100. The load-deflection curve obtained from the FINEL analysis shows very good agreement with the experimental results. The cracked region grows faster in the FINEL analysis, and more slowly in Reference 3.8-14, since the FINEL and Reference 3.8-14 load-deflection curves show different gradients (stiffnesses).

The results of analytical, experimental, and FINEL solutions are shown in Figure 3.8-100. The FINEL analysis agrees well with the experimental results, up to the point where the reinforcing steel in the beam yields. After the yield point, the FINEL analysis incorrectly calculated the effective stiffness of elements which have yielded. Therefore, the solution is not valid for further loadings. However, since all reinforcing steel remains elastic for the containment analysis, the FINEL program is verified for that application.

b. Sample Problem B: Axially Constrained Hollow Cylinder with a Distributed Pressure Loading

This problem investigates the response of an axially constrained hollow cylinder to internal pressure. A hand-calculated solution yields values of tangential, axial, and radial stresses at various radii from the center of the cylinder, which are then compared to the FINEL values.

The finite-element model is illustrated in Figure 3.8-101. Nodal points are free to move only in the radial direction, representing the conditions of axisymmetry and plane strain.

The problem parameters are as follows:

Poisson's ratio (υ) = 0.25

Modulus of elasticity (E) = 4.32×10^5 ksf

Number of nodal points = 22

Number of elements = 10

Internal pressure (p) = 1.0 ksf

From References 3.8-15 and 3.8-16, the following equations were used:

Hoop or tangential stress, T :

$$T = p \frac{a^2 (b^2 + r^2)}{r^2 (b^2 - a^2)} \qquad \text{(EQ. 3.8-34)}$$

Axial stress, T_2:

$$T_2 = \frac{2\upsilon P a^2}{b^2 - a^2} = \frac{P}{2} \frac{a^2}{b^2 - a^2} \qquad \text{(EQ. 3.8-35)}$$

Radial stress, T_R:

$$T_R = -p \frac{a^2 (b^2 - r^2)}{r^2 (b^2 - a^2)} \qquad \text{(EQ. 3.8-36)}$$

where:

a = 65.0 feet

b = 68.75 feet

p = 1.0 ksf

$a \leq r \leq b$

The results from FINEL for the hollow cylinder tangential, axial, and radial stresses are compared with the hand-calculated values in Table 3.8-19. The results are exactly the same, except for one value, where there is a 4.17% difference.

c. Sample Problem C: Axially Constrained Hollow Cylinder with a Linear Temperature Gradient

This problem evaluates the response of an axially constrained hollow cylinder, to a radially varying linear temperature gradient. The tangential, axial, and radial stresses are determined by hand calculations, and compared to the FINEL results.

Figure 3.8-102 illustrates the finite-element mesh. The conditions of axisymmetry and plane strain are imposed by using the axisymmetric quadrilateral element, and restraining all nodes against axial displacement.

The temperature profile is shown in Figure 3.8-103.

The problem parameters are as follows:

Poisson's ratio (υ) = 0.25

Modulus of elasticity (E) = 4.32×10^5 ksf

Coefficient of thermal expansion (α) = 6×10^{-6} ft/ft/°F

Number of nodal points = 22

Number of elements = 10

From References 3.8-16 and 3.8-17, the following equations are used:

Hoop or tangential stress, ∂_θ :

$$\partial_\theta = \frac{\alpha E}{1 - \upsilon} \frac{1}{r^2} \left[\frac{(r^2 + a^2)}{b^2 - a^2} \int_a^b Trdr + \int_a^r Tr' dr' - Tr^2 \right] \qquad \text{(EQ. 3.8-37)}$$

Axial stress, ∂_z :

$$\partial_z = \frac{\alpha E}{1-\upsilon} \left[\frac{2\upsilon}{b^2 - a^2} \int_a^b Trdr - T \right] \quad \text{(EQ. 3.8-38)}$$

Radial stress, ∂_R :

$$\partial_R = \frac{\alpha E}{1-\upsilon} \frac{1}{r^2} \left[\frac{(r^2 - a^2)}{b^2 - a^2} \int_a^b Trdr - \int_a^r Tr'dr' \right] \quad \text{(EQ. 3.8-39)}$$

where:

a	=	65.0 feet
b	=	68.75 feet
T	=	T(r) = temperature above reference, °F (reference temperature = 100°F)

Expressions for the temperature field are:

$T(r) = C_2r + C_1$ (EQ. 3.8-40)

$T(a) = 25 = C_1 + 65.0C_2$ (EQ. 3.8-41)

$T(b) = -25 = C_1 + 68.75C_2$ (EQ. 3.8-42)

Solving:

$$C_2 = \frac{-50}{68.75 - 65} = -13.33$$

$$C_1 = -25 - 68.75(-13.33) = 891.67$$

Then:

$$T(r) = -13.33r + 891.67$$

Evaluating the integral:

$$\int Trdr = \int (-13.33r + 891.67)rdr$$

$$= -\frac{13.33r^3}{3} + \frac{891.67r^2}{2} + C$$

$$= -4.44r^3 + 445.83r^2 + C$$

$$\int_a^b Trdr = -4.44(b^3 - a^3) + 445.83(b^2 - a^2)$$

$$\int_{s}^{b} Tr'dr' = -4.44(r^3 - a^3) + 445.83(r^2 - a^2)$$

The results from FINEL for the tangential, axial, and radial stresses are compared with the values obtained by hand calculations in Table 3.8-19. The results between the two methods of solution agree very closely.

3.8.7.9 ME620 (Transient Temperature Analysis of Plane and Axisymmetric Solids)

3.8.7.9.1 Application

The heat conduction program, ME620, is used to determine the temperature distribution as a function of time, within a plane or axisymmetric solid body subjected to step-function temperature or heat flux inputs. The program is also used for steady-state temperature analysis.

3.8.7.9.2 Program Background

The program utilizes a finite-element technique coupled with a step-by-step time integration procedure, as described in Reference 3.8-18.

The program was developed at the University of California, Berkeley, by Professor E. L. Wilson, and has been modified by Bechtel to incorporate the save and restart capabilities. The version of this program currently used by Bechtel is maintained by the Control Data Corporation.

3.8.7.9.3 Sample Problems

To verify this program, sample problems were analyzed by ME620 and compared with published data. Two sample problems are presented as examples of verification.

a. Sample Problem A: Heat Conduction in a Square Plate with One Edge Quenched

This problem tests the ability of the program to solve the temperature changes in a plane region subjected to conduction boundary conditions. The plate is brought to an equilibrium temperature, and one edge quenched while the other three edges are kept insulated.

A square plate is brought to equilibrium at a given initial temperature, T_o. Three edges are perfectly insulated, while a third edge is suddenly brought to a lower temperature, T_1. This quench is kept constant for the entire analysis. A temperature-time history is then obtained for the corner farthest from the quenched edge.

Figure 3.8-104 shows the actual plate arrangement, while Figure 3.8-105 shows a diagram of the finite-elements.

The problem parameters are as follows:

Nomenclature

L = length of longest heat flow path, inches

T_o = initial temperature of plate, °F

T_1 = quenching temperature of edge, °F

Data

The plate is 10 inches square

T_o = 100°F

T_1 = 0°F

Diffusivity α = 1.0 in^2/sec (chosen for convenience)

Time increment (ΔT) = 1 second for numerical solution

At any time, t, during the transient state, the time factor T (or characteristic time) is given by:

$$T = \alpha t/L^2$$

The time to reach steady-state is given when T = 1.0; hence the transient time is t = L^2/α = 100 seconds. The results derived from Reference 3.8-19 are plotted in Figure 3.8-106.

The temperature variation at point A is plotted in Figure 3.8-106 according to the results of ME620, and is compared with the theoretical transient change. The curves are seen to agree quite well. Deviations are due to the selected finite-element mesh size, and to the selected time step for the analysis.

b. Sample Problem B: Heat Conduction in a Surface Quenched Sphere

This problem tests the ability of ME620 to analyze the temperature distribution in an axisymmetric solid, with given temperature boundary conditions. The results of the program analysis are compared to a closed-form solution derived from Reference 3.8-20.

This problem considers a solid steel sphere (shown in Figure 3.8-107) which is brought to an equilibrium temperature, and suddenly quenched to a lower uniform temperature on its surface. The quenching environment is held at a constant temperature. A temperature-time history for three seconds is obtained from the program for all node points. The points used for the comparison are at a radius of 0.2 inches; only one time period is checked. The finite-element model is shown in Figure 3.8-108.

The problem parameters are as follows:

Nomenclature

L = length of the longest heat flow path (radius of sphere), inches

T_o = initial temperature of sphere, °F

T_1 = quenching temperature of outer surface, °F

Data

Radius of sphere (R) = 0.59 inches

T_o = 1472°F

T_1 = 68°F

Conductivity = 6.02×10^{-4} Btu/(in-sec-°F)

Diffusivity (α) = 0.0193 in^2/sec

Specific heat = 0.11 Btu/(lb-°F)

Density (p) = 0.284 lb/in^3

Time increment = 0.2 sec

At any time, t, during the transient state, the time factor T (or characteristic time) is given by:

$$T = \alpha t/L^2$$

The time to reach steady-state is given when T = 1.0; hence the transient time is $t = L^2/\alpha$ = 3.0 seconds.

The result from Reference 3.8-20 for the temperature at a radius of 0.2 inches and at time t = 1.8 seconds, is 933.8°F. The result obtained from ME620 is 923.4°F. The difference between these two values is 1.1%.

3.8.7.10 ANSYS

3.8.7.10.1 Application

The ANSYS engineering analysis system computer program is a large-scale general purpose computer program employing finite-element technology for the solution of several classes of engineering analysis problems. Analysis capabilities include static and dynamic; plastic, creep, and swelling; small and large deflections; steady-state and transient heat transfer; and steady-state fluid flow. A variety of finite-elements are available for use in the program. Structural loadings may be forces, displacements, pressures, temperatures, or response spectra. The program output includes joint displacements, element stresses, forces, and moments.

3.8.7.10.2 Program Background

ANSYS was developed by Swanson Analysis Systems, Inc, and is in the public domain. The version of the program currently used by Bechtel is maintained by the Control Data Corporation, Cybernet Service. Version 5.0 was used to evaluate the Spent Fuel Pool for increase storage capacity. This evaluation was performed by Gilbert/Commonwealth and Version 5.0 is maintained by Gilbert/Commonwealth.

3.8.7.11 IMAGES 3D

3.8.7.11.1 Application

The IMAGES 3D computer program is a three-dimensional general purpose finite element program for use on an IBM-compatible personal computer. Analysis capabilities include static, modal, and dynamic analysis of structures composed of frame, plates, shell and solid elements. Structural loadings may be forces, displacements, pressures, temperatures, response spectra, or time history. The program output includes joint displacements, element stresses, forces and moments. It also calculates natural frequencies and mode shapes.

3.8.7.11.2 Program Background

IMAGES 3D was developed by Celestial Software, which is a division of Robert L. Cloud & Associates. The program is in the public domain. Version 3.0 of the program is currently used by the licensee. The documentation package for IMAGES 3D includes a verification manual which documents a series of test problems that demonstrates that the results from IMAGES 3D are similar to those results of other public domain programs or classical solutions.

3.8.8 REFERENCES

3.8-1 General Electric Co., "Mark II Containment Dynamic Forcing Functions Information Report", (NEDO-21061) Rev 4.

3.8-2 INTENTIONALLY LEFT BLANK

3.8-3 Bechtel Power Corporation, "Containment Building Liner Plate Design Report," Topical Report BC-TOP-1, Rev 1, (December 1972).

3.8-4 E. L. Wilson, "SAP; A General Structural Engineering Program", Report No. UCSESM 70-20, Structures and Materials Research, Department of Civil Engineering, University of California at Berkeley, (September 1970).

3.8-5 H. Kraus, "Thin Elastic Shells", John Wiley, (1967), p. 136, (1967).

3.8-6 B. Budiansky and P. P. Radkowski, "Numerical Analysis of Unsymmetric Bending of Shells of Revolution", AIAA Journal, Vol. 1, No. 8, (August 1963).

3.8-7 O. W. Blodgett, "Design of Welded Structures", The James F. Lincoln Arc Welding Foundation, pp. 3.3-8 to 3.3-10, (June 1966).

3.8-8 Timoshenko, "Theory of Plates And Shells", 2nd Edition, McGraw-Hill, pp. 208-2211, (1959).

3.8-9 Bulletin 502, "Modern Flange Design", Gulf and Western Manufacturing Company.

3.8-10 A. Kalnin, "Analysis of Shells of Revolution Subjected to Symmetrical and Non-Symmetrical Loads", Journal of Applied Mechanics, Vol. 31, (September 1964).

3.8-11 J. M. Horowitz and R. Henschel, "Experimental Investigation of 2:1 Ellipsoidal Heads Subjected to Internal Pressure (Volume One)", Progress Report January 17, 1974 to April 16, 1976, Foster Wheeler Energy Corp., for Welding Research Council.

3.8-12 W. Flugge, "Stresses in Shells", Springer-Verlag, New York, (1973).

3.8-13 Beam-Column Connections in Reinforced Concrete", Structural Research Series No. 234, Civil Engineering Studies, University of Illinois, Urbana-Champaign, Illinois, (January 1962).

3.8-14 M. Suidan and W. C. Schnobrich, "Finite-Element Analysis of Reinforced Concrete", Journal of the Structural Division, ASCE, Vol. 99, No. ST10, pp. 2109-2122, (October 1973).

3.8-15 R. J. Roark, "Formulas for Stress and Strain", McGraw-Hill, Fourth Edition, p. 308, (1965).

3.8-16 Timoshenko and Goodier, "Theory of Elasticity", Second Edition, McGraw-Hill, pp. 412 and 60, (1951).

3.8-17 Manson, "Thermal-Stress and Low Cycle Fatigue", McGraw-Hill, pp. 28-29, (1960).

3.8-18 E. L. Wilson and R. E. Nickell, "Application of the Finite-Element Method to Heat Conduction Analysis", Journal of Nuclear Engineering and Design, (1966).

3.8-19 V. S. Arpaci, "Conduction Heat Transfer", Addition-Wesley, (1966).

3.8-20 Ingersol, Zobel, Ingersol, "Heat Conduction With Engineering, Geological, and Other Applications", Maple Press Co., Inc., York, Pennsylvania, p. 165, (1954).

3.8-21 "Report on the Re-evaluation of Concrete Masonry Walls for the Limerick Generating Station, Units 1 and 2", Bechtel Power Corporation, (April 1982).

3.8-22 "Visual Weld Acceptance Criteria for Structural Welding at Nuclear Power Plants, NCIG-01", Nuclear Construction Issues Group, Revision 2, (May 7, 1985).

3.8-23 "Building Code Requirements for Concrete Masonry Structures", ACI 531 and ACI 531-R, American Concrete Institute, Detroit, (1979).

Table 3.8-1

CODES, STANDARDS, RECOMMENDATIONS, AND SPECIFICATIONS USED IN DESIGN AND CONSTRUCTION OF SEISMIC CATEGORY I STRUCTURES(1)

DESIGNATION	TITLE	EDITION
American Concrete Institute (ACI)		
ACI 214	Recommended Practice for Evaluation of Compression Test Results of Field Concrete	1965
ACI 301	Specifications for Structural Concrete for Buildings	1966
ACI 306	Recommended Practice for Cold Weather Concreting	1966
ACI 307	Specification for the Design of Reinforced Concrete Chimneys	1959
ACI 315	Manual of Practice for Detailing Reinforced Concrete Structures	1965
ACI 318	Building Code Requirements for Reinforced Concrete	1971
ACI 347	Recommended Practice for Concrete Formwork	1968
ACI 605	Recommended Practice for Hot Weather Concreting	1959
ACI 613	Recommended Practice for Selecting Proportions for Concrete	1954
ACI 614	Recommended Practice for Measuring, Mixing, Transporting and Placing Concrete	1959
ACI SP-2	Manual of Concrete Inspection	5th Ed.
American Welding Society (AWS)		
AWS D1.0	Code for Welding in Building Construction	1969
AWS D1.1	Structural Welding Code(2)	1972, 1973 (Rev 1), 1974 (Rev 2), 1975, 1976 (Rev 1), 1977 (Rev 2)
AWS D12.1	Recommended Practice for Welding Reinforcing Steel, Metal Inserts, and Connections in Reinforced Concrete Construction	1961

Table 3.8-1 (Cont'd)

DESIGNATION	TITLE	EDITION
American Society of Mechanical Engineers (ASME)		
ASME	ASME Boiler and Pressure Vessel Code	
	a. Section II, Part C: Welding Rods, Electrodes, and Filler Metals	1971, 1974 with Addenda through 1976
	b. Section III, Subsection B: Requirements for Class B Vessels	1968, 1971, 1974 with Addenda through 1976
	c. Section III, Subsection NB: Class 1 Components	1977 with Addenda through Summer 1979
	d. Section III, Subsection NC: Class 2 Components	1968, 1971, 1974, 1977 with Addenda through Summer 1978
	e. Section III, Subsection NE: Class MC Components	1977 with Addenda through Winter 1978
	f. Section V, Nondestructive Examination	1968, 1971, 1974, 1977 with Addenda through Summer 1978
	g. Section VIII, Parts UW and UG, and Division 1	1968, 1971, 1974, 1977 with Addenda through Summer 1978
	h. Section IX, Welding and Brazing Qualification	1968, 1971, 1974 with Addenda through 1976
American Society for Testing and Materials (ASTM)[(3)]		
ASTM A20	General Requirements for Steel Plates for Pressure Vessels	1970, 1974a
ASTM A36	Structural Steel	1969, 1970, 1970a, 1974, 1975, 1977, 1979
ASTM A53	Welded and Seamless Steel Pipe	1970, 1972a, 1973, 1976
ASTM A82	Cold-Drawn Steel Wire for Concrete Reinforcement	1976

Table 3.8-1 (Cont'd)

DESIGNATION	TITLE	EDITION
ASTM A106	Seamless Carbon Steel Pipe for High Temperature Service	1970, 1972a, 1974
ASTM A108	Steel Bars, Carbon, Cold Finished, Standard Quality	1969, 1973
ASTM A123	Zinc (Hot-Galvanized) Coatings on Products Fabricated from Rolled, Pressed and Forged Steel Shapes, Plates, Bars and Strip	1973
ASTM A153	Zinc Coating (Hot-Dip) on Iron and Steel Hardware	1967
ASTM A167	Stainless and Heat-Resisting Chromium-Nickel Steel Plate, Sheet, and Strip	1974
ASTM A193	Alloy-Steel and Stainless Steel Bolting Materials for High Temperature Service	1977
ASTM A240	Heat-Resisting Chromium and Chromium-Nickel Stainless Steel Plate, Sheet, and Strip for Fusion Welded Unfired Pressure Vessels	1977
ASTM A242	High Strength, Low Alloy Structural Steel	1975
ASTM A262	Detecting Susceptibility to Intergranular Attack on Stainless Steels	1970
ASTM A276	Stainless and Heat-Resisting Steel Bars and Shapes	1976
ASTM A285	Pressure Vessel Plates, Carbon Steel, Low and Intermediate Tensile Strength	1969, 1972
ASTM A306	Carbon Steel Bars Subject to Mechanical Property Requirements	1964
ASTM A307	Carbon Steel Externally and Internally Threaded Standard Fasteners	1968
ASTM A312	Seamless and Welded Austenitic Stainless Steel Pipe	1974
ASTM A325	High Strength Bolts for Structural Steel Joints, Including Suitable Nuts and Plain Hardened Washers	1970a, 1971a, 1974
ASTM A386	Zinc Coating (Hot-Dip) on Assembled Steel Products	1967
ASTM A403	Wrought Austenitic Stainless Steel Piping Fittings	1973a

Table 3.8-1 (Cont'd)

DESIGNATION	TITLE	EDITION
ASTM A416	Uncoated Seven-Wire Stress-Relieved Strand for Prestressed Concrete	1974
ASTM A441	High Strength Low Alloy Structural Manganese Vanadium Steel	1970, 1970a, 1975
ASTM A449	Quenched and Tempered Steel Bolts and Studs	1976c
ASTM A479	Stainless and Heat-Resisting Steel Bars and Shapes for Use in Boilers and Other Pressure Vessels	1974, 1977
ASTM A480	Delivery of Flat-Rolled Stainless and Heat-Resisting Steel Plate, Sheet, and Strip	1974a, 1975
ASTM A490	Quenched and Tempered Alloy Steel Bolts for Structural Steel Joints	1975, 1976
ASTM A500	Cold-Formed Welded and Seamless Carbon Steel Structural Tubing in Rounds and Shapes	1974, 1976
ASTM A501	Hot-Formed Welded and Seamless Carbon Steel Structural Tubing	1976
ASTM A516	Pressure Vessel Plates, Carbon Steel for Moderate- and Lower-Temperature Service	1971, 1972, 1973, 1974, 1974a, 1975, 1976
ASTM A519	Seamless Carbon and Alloy Steel Mechanical Tubing	1973, 1977b, 1979
ASTM A527	Steel Sheet, Zinc-Coated (Galvanized) by the Hot-Dip Process	1971
ASTM A570	Hot-Rolled Carbon Steel Sheet and Strip, Structural Quality	1972
ASTM A575	Merchant Quality Hot-Rolled Carbon Steel Bars	1971, 1973
ASTM A606	Steel Sheet and Strip, Hot-Rolled and Cold-Rolled, High Strength, Low Alloy, with Improved Corrosion Resistance	1975
ASTM A537	Pressure Vessel Plates, Heat Treated, Carbon-Manganese-Silicon	1967, 1974a
ASTM A569	Steel, Carbon (0.15%, maximum). Hot-Rolled Sheet and Strip, Commercial Quality	1972
ASTM A588	High Strength Low Alloy Structural Steel with 50,000 psi Minimum Yield Point to 4 inches Thick	1980a

Table 3.8-1 (Cont'd)

DESIGNATION	TITLE	EDITION
ASTM A615	Deformed and Plain Billet Steel Bars for Concrete Reinforcement	1968, 1972, 1975
ASTM A668	Steel Forgings, Carbon and Alloy for General Industrial Use	1972
ASTM A675	Special Quality Hot-Rolled Carbon Steel Bars Subject to Mechanical Property Requirements	1972
ASTM B209	Aluminum-Alloy Sheet and Plate	1974, 1977
ASTM B308	Aluminum-Alloy Standard Structural Shapes, Rolled or Extruded	1973
ASTM C31	Making and Curing Concrete Test Specimens in the Field	1969
ASTM C33	Concrete Aggregates	1971a, 1978
ASTM C39	Compressive Strength of Molded Concrete Cylinders	1971, 1972
ASTM C40	Organic Impurities in Sand for Concrete	1966, 1973
ASTM C55	Concrete Building Brick	1975
ASTM C70	Surface Moisture in Fine Aggregate	1966, 1973
ASTM C87	Effect of Organic Impurities in Fine Aggregate on Strength of Mortar	1969
ASTM C88	Soundness of Aggregates by Use of Sodium Sulfate or Magnesium Sulfate	1971, 1971a, 1973, 1976
ASTM C90	Hollow Load-Bearing Concrete Masonry Units	1975
ASTM C94	Ready-Mixed Concrete	1969, 1972, 1978a
ASTM C109	Compressive Strength of Hydraulic Cement Mortars	1970T, 1970, 1973, 1977
ASTM C114	Chemical Analysis of Hydraulic Cement	1969, 1973, 1977
ASTM C115	Fineness of Portland-Cement by the Turbidimeter	1970, 1973, 1977, 1978a
ASTM C117	Materials Finer than No. 200 Sieve in Mineral Aggregates by Washing	1969, 1976

Table 3.8-1 (Cont'd)

DESIGNATION	TITLE	EDITION
ASTM C123	Lightweight Pieces in Aggregate	1969
ASTM C127	Specific Gravity and Absorption of Coarse Aggregate	1968, 1977
ASTM C128	Specific Gravity and Absorption of Fine Aggregate	1968, 1973
ASTM C131	Resistance to Abrasion of Small Size Coarse Aggregate by Use of the Los Angeles Machine	1969, 1976
ASTM C136	Sieve or Screen Analysis of Fine and Coarse Aggregates	1971, 1976
ASTM C138	Unit Weight, Yield, and Air Content of Concrete	1971T, 1971, 1977
ASTM C140	Sampling and Testing of Concrete Masonry Units	1970
ASTM C142	Clay Lumps and Friable Particles in Aggregates	1971, 1978
ASTM C143	Slump of Portland-Cement Concrete	1971, 1974
ASTM C144	Standard Specification for Masonry Mortar	1966T, 1970
ASTM C145	Solid Load-Bearing Concrete Masonry Units	1975
ASTM C150	Portland-Cement	1968, 1970, 1971, 1972, 1973a, 1977, 1978a
ASTM C151	Autoclave Expansion of Portland-Cement	1971, 1977
ASTM C171	Standard Specification for Sheet Materials for Curing Concrete	1969
ASTM C172	Sampling Fresh Concrete	1971
ASTM C183	Sampling Hydraulic Cement	1971, 1973, 1976
ASTM C185	Air Content of Hydraulic Cement Mortar	1971, 1975
ASTM C186	Heat of Hydration of Hydraulic Cement	1968, 1977, 1978
ASTM C190	Tensile Strength of Hydraulic Cement Mortars	1970, 1972, 1977

Table 3.8-1 (Cont'd)

DESIGNATION	TITLE	EDITION
ASTM C192	Making and Curing Concrete Test Specimens in the Laboratory	1969, 1976
ASTM C207	Standard Specification for Hydrated Lime for Masonry Purposes	1976
ASTM C231	Air Content of Freshly Mixed Concrete by the Pressure Method	1971, 1972T, 1978
ASTM C233	Testing Air Entraining Admixtures for Concrete	1969, 1978
ASTM C235	Scratch Hardness of Coarse Aggregate Particles	1968
ASTM C260	Air Entraining Admixtures for Concrete	1969, 1977
ASTM C266	Time of Setting of Hydraulic Cement by Gillmore Needles	1971, 1977
ASTM C289	Potential Reactivity of Aggregates	1971
ASTM C295	Petrographic Examination of Aggregates for Concrete	1965
ASTM C451	False Set of Portland-Cement	1972, 1975
ASTM C452	Potential Expansion of Portland-Cement Mortars Exposed to Sulfate	1968, 1975
ASTM C469	Static Modulus of Elasticity and Poisson's Ratio of Concrete in Compression	1965
ASTM C476	Mortar and Grout for Reinforced Masonry	1971
ASTM C494	Chemical Admixtures for Concrete	1971
ASTM C566	Total Moisture Content of Aggregate by Drying	1967
ASTM C617	Capping Cylindrical Concrete Specimens	1971a, 1976
ASTM C845	Expansive Hydraulic Cement	1980
ASTM D75	Sampling Aggregates	1959, 1971
ASTM E109	Dry Powder Magnetic Particle Inspection	1963, 1971
ASTM E328	Stress-Relaxation Tests for Materials and Structures	1972

Table 3.8-1 (Cont'd)

DESIGNATION	TITLE	EDITION
ASTM E437	Industrial Wire Cloth and Screens (Square Opening Series)	1971
American Association of State Highway and Transportation Officials (AASHTO)		
AASHTO T26	Quality of Water to be Used in Concrete	1970
US Army Corps of Engineers		
CRD C36	Coefficient of Thermal Diffusivity	1948
CRD C39	Thermal Coefficient of Expansion	1955
CRD C79	Method of Test for Flow of Grout Mixtures	1958
CRD C119	Test for Flat and Elongated Particles in Coarse Aggregate	1953
CRD C589	Methods of Sampling and Testing Expensive Grouts	1970
American National Standards Institute (ANSI)		
ANSI N45.2.6	Qualifications of Inspection, Examination, and Testing of Personnel for the Construction Phase of Nuclear Power Plants	1973
American Institute of Steel Construction		
AISC	Specification for the Design, Fabrication and Erection of Structural Steel for Buildings	1969
	Supplement #1	1970
	Supplement #2	1971
	Supplement #3	1974
AISC	Code of Standard Practice for Steel Buildings and Bridges	1970
International Conference of Building Officials		
UBC	Uniform Building Code	1967, 1970

Table 3.8-1 (Cont'd)

DESIGNATION	TITLE	EDITION
American Iron and Steel Institute		
AISI	Specification for the Design of Cold-Formed Steel Structural Members	1968
Society of Automotive Engineers		
SAE J444a	Cast Shot Specification for Shot Peening or Blast Cleaning	1976
American Petroleum Institute		
API 5L	Specification for Line Pipe	1973
API 5LX	Specification for High Test Line Pipe	1973
American Waterworks Association		
AWWA M11	Steel Pipe Manual	1964
Regulations of the Commonwealth of Pennsylvania		
	Regulations for Boilers and Unfired Pressure Vessels	-

(1) For exceptions to the above codes, standards, recommendations and specifications, see Section 3.8.6.
(2) The contractor or fabricator may also qualify its welding procedures and welders in accordance with the latest edition of AWS D1.1 code at the time of qualification.
(3) Principal editions used for design are listed; later editions may be used for specific plant modifications.

Table 3.8-2

LOAD COMBINATIONS FOR PRIMARY CONTAINMENT WALL,
DIAPHRAGM SLAB, AND REACTOR PEDESTAL

The primary containment, diaphragm slab, and reactor pedestal are designed for the following load combinations:

1. Normal Operating Conditions

   a. $U=1.5D+1.8L+1.0T_o+1.25H_o$

   b. $U=1.25(D+L+E+H_o)+1.0T_o$

   c. $U=0.9D+1.25(E+H_o)+1.0T_o$

   d. $U=1.0D+1.0L+1.8E+1.0T_o+1.25H_o$

2. Design Accident and Extreme Environmental Conditions

   a. $U=1.05D+1.05L+1.0T_A+1.0H_A+1.0R+1.5P$

   b. $U=1.05D+1.05L+1.0T_A+1.0H_A+1.0R+1.25P+1.25E$

   c. $U=1.0D+1.0L+1.0T_A+1.0H_A 1.0R+1.0P+1.0E'$

   d. $U=0.95D+1.25E+1.0T_A+1.0H_A+1.0R$(1)

   e. $U=1.05D+1.0B+1.25P'+1.25E$

The pressure-retaining steel elements are designed for the following load combinations(2):

		Stress Limits
a.	$D+L+1.15P$	1.15 times ASME Section III, Class B for "Normal Operating Conditions"
b.	$D+L+T_A+P$	ASME Section III, Class B for "Normal Operating Conditions"
c.	$D+L+T_A+P+H_A+R+E$	ASME Section III, Summer 1970 Addenda, Figure N-414, "For Emergency Conditions"
d.	$D+L+T_A+P+H_A+R+E'$	ASME Section III, Summer 1970 Addenda, Figure N-414, "For Faulted Conditions"

Table 3.8-2 (Cont'd)

Definitions:

U	=	Required ultimate load capacity, as defined in ACI 318-71
D	=	Dead load of structure and major equipment, plus any other major permanent loads contributing stress, such as soil or hydrostatic loads or operating pressures
L	=	Live loads expected to be present when the plant is operating. For loading conditions that include vertical seismic, replace L by $L_p + L_o$, where:
		L_p = permanent live load to include minor piping, minor equipment and the weight of equipment in lay-down areas: to be accelerated vertically
		L_o = 100 psf (except 50 psf for grating and platforms): not to be accelerated vertically
T_o	=	Thermal effects due to temperature gradient through the wall, under operating conditions
T_A	=	Thermal effects due to temperature gradient through the wall, under accident conditions
P	=	Design basis accident pressure load
R	=	Steam/water jet forces or reactions resulting from the rupture of process piping
E	=	Load due to the OBE resulting from a horizontal ground acceleration of 0.075 g, and a vertical ground acceleration of 0.05 g
E'	=	Load due to the design basis earthquake resulting from a horizontal ground acceleration of 0.15 g, and a vertical ground acceleration of 0.10 g
B	=	Hydrostatic loading due to postaccident flooding of the primary containment to the level of the reactor core
P'	=	Pressure of the atmosphere in the primary containment, with the containment flooded to the level of the reactor core

Table 3.8-2 (Cont'd)

H_o	=	Force on the structure due to thermal expansion or contraction of hanger critical pipes
H_A	=	Force on the structure due to thermal expansion of pipes, under accident conditions
T	=	Thermal loads resulting from temperature increases under accident conditions.

---

(1) Where overturning forces cause net tension in the absence of live load

(2) Bending from jet forces are considered as primary stresses.

Table 3.8-3

PRIMARY CONTAINMENT DESIGN PRESSURE AND TEMPERATURE CRITERIA

PORTION OF CONTAINMENT	TIME AFTER START OF ACCIDENT	CONTAINMENT ATMOSPHERE PRESSURE (psig)	CONTAINMENT ATMOSPHERE TEMPERATURE (°F)
Drywell	0 sec[2]	0.75	135[3]
Suppression Chamber	0 sec[2]	0.75	95
Drywell	0-45 sec	55	340
Suppression Chamber	0-45 sec	55	175
Suppression Chamber[1]	0-45 sec	25	175
Drywell	45 sec-60 min	35	340
Suppression Chamber	45 sec-60 min	35	175
Drywell	1-3 hrs	35	340
Suppression Chamber	1-3 hrs	35	220
Drywell	3-6 hrs	35	320
Suppression Chamber	3-6 hrs	35	220
Drywell	6-24 hrs	20	220
Suppression Chamber	6-24 hrs	20	220
Drywell	1-4 days	20	200
Suppression Chamber	1-4 days	20	200
Drywell	4-30 days	10	175
Suppression Chamber	4-30 days	10	175

---

(1) Drywell/suppression chamber maximum differential pressure condition

(2) Just before start of accident (i.e., at operating condition)

(3) This information is based on original design basis conditions. Further evaluation has validated the primary containment design for an initial drywell containment atmosphere temperature of 150°F. The stress indicated in UFSAR Figures 3A-362 to 3A-380, Design Assessment Report for Containment Structures reflects current plant conditions.

Table 3.8-4

LOAD COMBINATIONS AND ALLOWABLE STRESSES FOR
ASME CLASS MC COMPONENTS[1]

The drywell head assembly, equipment hatches, personnel lock, suppression chamber access hatches, and CRD removal hatch are designed for the following loading combinations and allowable stresses:

	Stress Limits
D+L+1.15P	1.15 times ASME Section III, Class B for "Normal Operating Conditions"
$D+L+T_A+P$	ASME Section III, Class B for "Normal Operating Conditions"
$D+L+T_A+P+H_A+R+E$	ASME Section III, Summer 1970 Addenda, Figure N-414, "For Emergency Conditions"
$D+L+T_A+P+H_A+R+E'$	ASME Section III, Summer 1970 Addenda, Figure N-414, "For Faulted Conditions"

Piping and electrical penetrations are designed for the following load combinations and allowable stresses:

a. The loads used in the design are as follows:

   1. Moments and forces transmitted by the piping to the penetration due to thermal expansion, weight, earthquake (including inertial effects and anchor movements), and other dynamic loads
   2. Pressures
   3. Thermal transients
   4. Number of operating cycles
   5. Pipe failure loads for faulted condition

b. The loading combinations are specified in Section 3.9.

c. Stress limits specified in ASME Section III, Article NB-3220 are used as the design criteria for Class I flued heads for design, normal and upset, and emergency condition. The rules contained in ASME Section III, Appendix F are used in evaluating the faulted condition for Class I and II flued heads.

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(1) Definitions of symbols are given in Table 3.8-2.

Table 3.8-5

LOAD COMBINATION FOR THE REACTOR SHIELD WALL[1]

The reactor shield wall is designed for the following loading combination:

Condition

Abnormal/Extreme	$D+L+T_A+R+P+E'$

[1] Symbols are defined in Table 3.8-2.

Table 3.8-6

LOAD COMBINATIONS FOR THE SUPPRESSION CHAMBER COLUMNS[1]

The suppression chamber columns are designed for the following loading combinations:

Condition	
Abnormal	$1.05D+1.05L+1.0T_A+1.0R+1.5P$
Abnormal/Severe	$1.05D+1.05L+1.0T_A+1.0R+1.25P+1.25E$
Abnormal/Extreme	$1.0D+1.0L+1.0T_A+1.0R+1.0P+1.0E'$

(1) Symbols are defined in Table 3.8-2.

Table 3.8-7

LOAD COMBINATIONS FOR THE PIPE WHIP RESTRAINTS
AND DRYWELL PLATFORMS[1]

The pipe whip restraints and drywell platforms are designed for the following loading combinations:

<u>Condition</u>

Normal	D+L
Abnormal[2,3]	D+L+R+T

---

(1) Symbols are defined in Table 3.8-2, except as noted below.

(2) For the design of pipe whip restraints which also support other pipe support(s), either pipe support SSE reaction load or the pipe rupture load (R) is considered.

(3) Thermal loads can be neglected when it can be shown that they are secondary and self-limiting in nature and where the material is ductile.

Table 3.8-8

LOAD COMBINATION FOR THE SEISMIC TRUSS[1]

The seismic truss is designed for the following loading combination:

Condition

Abnormal/Extreme	D+R+E'

---

(1) Symbols are defined in Table 3.8-2.

Table 3.8-9

LOAD COMBINATIONS FOR THE REACTOR ENCLOSURE[6]

A. Reinforced Concrete

1. Normal Operating and Severe Environmental Conditions

a. $U=1.5D+1.8L+1.0T_o+1.25H_o$

b. $U=1.25(D+L+H_o+E)+1.0T_o$

c. $U=1.25(D+L+H_o+W)+1.0T_o$

d. $U=0.9D+1.25(H_o+E)+1.0T_o$[1]

e. $U=0.9D+1.25(H_o+W)+1.0T_o$[1]

f. $U=1.0D+1.0L+1.8E+1.0T_o+1.25H_o$[2]

2. Design Accident and Extreme Environmental Conditions

a. $U=1.05D+1.05L+1.0T_A+1.0H_A+1.0R+1.5P$

b. $U=1.05D+1.05L+1.0T_A+1.0H_A+1.0R+1.25P+1.25E$

c. $U=1.0D+1.0L+1.0T_A+1.0H_A+1.0R+1.0P+1.0E'$

d. $U=0.95D+1.25E+1.0T_A+1.0H_A+1.0R$[1]

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

f. $U=1.0D+1.0L+1.0W'+1.0T_o+1.0H_o$

g. $U=1.0D+1.0L+1.0B_o+1.0E'$

h. $U=1.0D+1.0L+1.0R_o$

B. Steel Structures

	1. Normal Operation and Severe Environmental Conditions	Stress Limits
a.	$D+L+T_o+H_o$	F_s
b.	$D+L+T_o+H_o+E$	$1.25F_s$
c.	$D+L+T_o+H_o+W$	$1.33F_s$
d.	$D+L+T_o+H_o+E$[2]	F_s

Table 3.8-9 (Cont'd)

2.	Design Accident and Extreme <u>Environmental Conditions</u>		Stress <u>Limits</u>
	a.	$D+L+R+T_o+H_o+P+E'$	See note (3)
	b.	$D+L+R+T_A+H_A+P+E'$	See note (3)
	c.	$D+L+A+T_o+H_o$	See note (3)
	d.	$D+L+T_o+H_o+W'^{(4)}$	See note (3)
	e.	$D+L+R_o$	

C. <u>Post-tensioned Concrete Fuel Pool Girders</u>

1.	Normal Operating Conditions <u>(Including Refueling)</u>		Stress <u>Limits</u>
	For elastic analysis by straight line theory:		
	a.	$D+L+T_o+F$	F_c
	b.	$D+L+E+T_o+F$	$1.25F_c$
	c.	$D+E+T_o+F^{(5)}$	$1.25F_c$
	d.	$D+E'+T_o+F^{(5)}$	$1.50F_c$
	For ultimate strength method:		
	e.	$U=1.5D+1.8L+1.0T_o$	
	f.	$U=1.25(D+L+E)+1.0T_o$	
2.	<u>Design Accident and Extreme Environmental Conditions</u>		
	a.	$U=1.05D+1.05L+1.0T_A$	
	b.	$U=1.05D+1.05L+1.25E+1.0T_A$	
	c.	$U=1.0D+1.0L+1.0E'+1.0T_o$	

<u>Definitions</u>:

W = Wind load

W' = Tornado wind load

Table 3.8-9 (Cont'd)

B_o	=	Hydrostatic loading due to flooding of the ECCS and RCIC compartments in the reactor enclosure, to suppression pool equilibrium water level
A	=	Hydrostatic load due to probable maximum flood
f_s	=	Calculated stress in structural steel
F_s	=	Allowable stress for structural steel
F_y	=	Yield strength of structural steel
F	=	Prestressing force
F_c	=	Allowable flexural concrete stress, by straight-line theory
R_o	=	Blast pressure due to railroad accident or pipeline explosion as discussed in Section 2.2.3.1.1 and defined in Reference 2.2-1. (The effect of ground shock due to blast load is a subject of separate investigation.)

For all other definitions, see Table 3.8-2

(1) Where overturning forces cause net tension in the absence of live load

(2) For structural elements carrying mainly earthquake forces

(3) The allowable stress in structural steel does not exceed 0.9 F_y in bending, 0.85 F_y in axial tension or compression, and 0.5 F_y in shear. Where F_s is governed by requirements of stability (local or lateral buckling), f_s does not exceed 1.5 F_s. Thermal loads can be neglected when it can be shown that they are secondary and self-limiting in nature and where the material is ductile.

(4) Where protection against tornado forces is required

(5) When effects of upward vertical acceleration under earthquake conditions are considered

(6) Unless they are shown, stress limits comply with the ACI codes and AISC specifications. Deviations shown are consistent with typical nuclear power plant design practices.

Table 3.8-10

LOAD COMBINATIONS APPLICABLE TO SEISMIC CATEGORY I STRUCTURES OTHER THAN PRIMARY CONTAINMENT AND REACTOR ENCLOSURE[5]

A. Reinforced Concrete

1. Normal Operating and Severe Environmental Conditions

   a. $U=1.5D+1.8L+1.0T_o+1.25H_o$

   b. $U=1.25(D+L+H_o+E)+1.0T_o$

   c. $U=1.25(D+L+H_o+W)+1.0T_o$

   d. $U=0.9D+1.25(H_o+E)+1.0T_o$[2]

   e. $U=0.9D+1.25(H_o+W)+1.0T_o$[2]

   For structural elements carrying mainly earthquake forces:

   f. $U=1.0D+1.0L+1.8E+1.0T_o+1.25H_o$

2. Design Accident and Extreme Environmental Conditions

   a. $U=1.05D+1.05L+1.25E+1.0T_A+1.0H_A+1.0R$

   b. $U=0.95D+1.25E+1.0T_A+1.0H_A+1.0R$[2]

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

   d. $U=1.0D+1.0L+1.0E'+1.0T_A+1.0H_A+1.0R$

   e. $U=1.0D+1.0L+1.0A+1.0T_o+1.25H_o$

   f. $U=1.0D+1.0L+1.0W'+1.0T_o+1.0H_o$

   g. $U=1.0D+1.0L+1.0R_o$

Table 3.8-10 (Cont'd)

B. Steel Structures

1.	Normal Operating and Severe Environmental Conditions	Stress Limits
a.	$D+L+T_o+H_o$	F_s
b.	$D+L+T_o+H_o+E$	$1.25F_s$
c.	$D+L+T_o+H_o+W$	$1.33F_s$
d.	$D+L+T_o+H_o+E^{(3)}$	F_s
2.	Design Accident and Extreme Environmental Conditions	Stress Limits
a.	$D+L+R+T_o+H_o+E'+P$	See note (4)
b.	$D+L+R+T_A+H_A+E'+P$	See note (4)
c.	$D+L+A+T_o+H_o$	See note (4)
d.	$D+L+T_o+H_o+W'$	See note (4)
e.	D+L+Ro	

(1) Symbols are defined in Table 3.8-9.

(2) Where overturning forces cause net tension in the absence of live load

(3) For structural elements carrying mainly earthquake forces

(4) The allowable stress in structural steel does not exceed 0.9 F_y in bending, 0.85 F_y in axial tension, and 0.5 F_y in shear. Where F_s is governed by requirements of stability (local or lateral buckling), f_s does not exceed 1.5 F_s. Thermal loads can be neglected when it can be shown that they are secondary and self-limiting in nature and where the material is ductile.

(5) Unless they are shown stress limits comply with the ACI codes and AISC specifications. Deviations shown are consistent with typical nuclear power plant design practices.

Table 3.8-11

LOAD COMBINATIONS APPLICABLE TO MISCELLANEOUS STRUCTURAL COMPONENTS OF SEISMIC CATEGORY I STRUCTURES[3]

A. Anchor Bolts

		Stress Limits
1.	Normal Operating and Severe Environmental Conditions	
a.	$D+L+H_o$	F_s
b.	$D+L+H_o+(E \text{ or } W)$	$1.25F_s$
2.	Design Accident and Extreme Environmental Conditions	
a.	$D+L+R+H_o+P+E'$	[4]
b.	$D+L+R+H_A+P+E'$	[4]

B. Masonry

		Stress Limits
1.	Normal Operating and Severe Environmental Conditions	
a.	D+L	1.0xUBC[2]
b.	D+L+E	1.25xUBC[2]
2.	Design Accident and Extreme Environmental Conditions	
a.	D+L+P	1.0xUBC[2]
b.	D+L+P+E'	1.33xUBC[2]

---

(1) Symbols are defined in Table 3.8-9.

(2) As specified in UBC, but without the customary increase in normal allowable working stress due to earthquake

(3) Stress limits shown are deviations from the ACI codes and AISC specifications; these values are consistent with typical nuclear power plant design practices.

(4) Strength method is used. Stress limit in steel shall not exceed 0.85 F_y for tension and 0.50 F_y for shear. The allowable stress in concrete shall be in accordance with ACI 318-71.

Table 3.8-12

MINIMUM TESTING FREQUENCY FOR CONCRETE, CONCRETE MATERIALS, CONCRETE MASONRY, AND MASONRY MATERIALS

MATERIAL	REQUIREMENT	TEST	FREQUENCY
Cement properties	Standard physical and chemical	ASTM C150	Once for each 5000 cubic yards of concrete production
Aggregate	Organic impurities	ASTM C40	Once per 8 hour shift during concrete production
	Soundness of aggregate	ASTM C88	Once for each 5000 cubic yards of concrete production
	Material finer than No. 200 sieve	ASTM C117	Once for each 5000 cubic yards of concrete production
	Lightweight pieces in aggregates	ASTM C123	Once for mix qualification
	Specific gravity and absorption	ASTM C127/C128[(1)]	Once for mix qualification
	Abrasion of coarse aggregate	ASTM C131	Once for each 5000 cubic yards of concrete production
	Gradation of coarse aggregate	ASTM C136	Once for each size of coarse aggregate during each 8 hour shift of production
	Gradation of fine aggregate	ASTM C136	Twice during each 8 hour shift of production
	Potential reactivity	ASTM C289	Once for each 5000 cubic yards of concrete production
	Petrographic examination	ASTM C295	Once for mix qualification
	Flat and elongated particles	CRD C119	Once per week on each size of coarse aggregate
Water and ice	Quality of water to be used in concrete (to meet the requirements herein)	AASHTO T26	Once every 3 months
Admixtures	Air entraining agent	ASTM C260	Upon delivery at job-site, prior to approval for use
	Water reducing and retarding agent(s)	ASTM C494	Upon delivery at job-site, prior to approval for use

Table 3.8-12 (Cont'd)

MATERIAL	REQUIREMENT	TEST	FREQUENCY
Concrete	Mixer uniformity	National Ready Mix Concrete Association (NRMCA)	Initially, and then once every 2 years
	Slump	ASTM C143	Once for every 35 cubic yards of each class of concrete produced for concrete used in seismic Category I structures. For all other structures, once for every 50 cubic yards of each class of concrete.
	Air content, temperature, unit weight	ASTM C231/C138	Once for each 100 cubic yards of each class of concrete produced
	Compressive strength	ASTM C31/C39	The specified number of cylinders to be tested for design strength at the date specified are cast for each 100 cubic yards of each class of concrete, or at least once per day for each class of concrete
Masonry units	Compressive strength, absorption, weight, and moisture content	ASTM C140	Test frequency is in accordance with ASTM C140
Mortar aggregate	Physical properties	ASTM C144	The first load and each tenth successive load of masonry aggregate is tested
Cement for mortar	Standard physical and chemical properties	ASTM C150	Initially, and thereafter for every 10,000 pounds of cement delivered
Hydrated lime	Physical and chemical properties	ASTM C207, Type S or SA	Initially, and thereafter for every 5000 pounds of lime delivered
Chemtree grout	Compressive strength	ASTM C39	Two cylinders cast each day that grout is placed
	Unit weight	ASTM C138	Three times, at approximately equal time intervals, each day that grout is placed
Mortar	Compressive strength	UBC 24.23	Once per shift per mixer

(1) ASTM C128 is performed more frequently for medium and high density grout/concrete and masonry grout

Table 3.8-13

MEMBRANE FORCE RESULTANTS FROM "ASHSD" PROGRAM

	THIN-SHELL		LAYERED SHELL	
NODE POINT	LONGITUDINAL FORCE (lb/in)	CIRCUMFERENTIAL FORCE (lb/in)	LONGITUDINAL FORCE (lb/in)	CIRCUMFERENTIAL FORCE (lb/in)
1[(1)]	27,000	54,004	27,000	54,004
2	27,000	54,005	27,000	54,005
3	27,000	54,008	27,000	54,008
4	27,000	54,012	27,000	54,012
5	27,000	54,015	27,000	54,015
6	27,000	54,012	27,000	54,012
7	27,001	53,999	27,001	53,999
8	27,001	53,968	27,001	53,968
9	27,001	53,912	27,001	53,912
10	27,000	53,829	27,000	53,829
11	26,999	53,731	26,999	53,731
12	26,997	53,654	26,997	53,654
13	26,994	53,674	26,994	53,674
14	26,989	53,912	26,989	53,912
15	26,984	54,532	26,984	54,532
16	27,111	55,724	27,111	55,724

(1) Node point 1 represents the center of the cylinder.

Table 3.8-14

COMPARISON OF RESULTS FROM "CECAP" PROGRAM AND HAND CALCULATIONS

Sample Problem A: Beam with a Thermal Moment

PARAMETER	CECAP	HAND CALCULATION	% ERROR
f_s	13,150 psi	13,790 psi	5.9
f_c	-331 psi	-318 psi	4.1
kd	7.55 in	7.42 in	1.8
M_T	43,760 in-lb/in	43,690 in-lb/in	0.2

Sample Problem B: Beam with a Real Moment

PARAMETER	CECAP	HAND CALCULATION	% ERROR
f_s	79,170 psi	84,570 psi	6.4
f_c	-1845 psi	-1913 psi	3.6
kd	7.6 in	7.4 in	2.7

Sample Problem C: Beam with a Real Moment and Real Compressive Load

PARAMETER	CECAP	HAND CALCULATION	% ERROR
f_s	41,620 psi	41,320 psi	0.7
f_c	-1908 psi	-1922 psi	0.7
kd	12.2 in	12.7 in	3.9

Table 3.8-15

COMPARISON OF RESULTS FROM "CE668" PROGRAM AND HAND CALCULATIONS

Sample Problem A: Rectangular Plate with a Concentrated Load at the Center

	PARAMETER	HAND CALCULATION	CE668
1.	Deflection (in)		
	@ Node 116	0.00153	0.00151
2.	Moments (in-lbs)		
	M_x @ Node 113	20.92	21.24
	M_y @ Node 117	57.198	56.377

Sample Problem B: Uniform Load on a Rectangular Plate

	PARAMETER	HAND CALCULATION	CE668
1.	Deflection (in)		
	@ Node 11	0.277	0.278
2.	Moments (in-lbs)		
	M_x @ Node 11	52.72	50.92
	M_y @ Node 121	143.55	142.28

Table 3.8-16

COMPARISON OF RESULTS FROM "E0119" PROGRAM AND PUBLISHED RESULTS

Sample Problem A: Welding Neck Flange

1. Design (Operating) Condition

STRESS COMPONENT	ALLOWABLE STRESS (psi)	ACTUAL STRESS (psi) REFERENCE E0119	3.8-9
Bolts	25,000	21,801	-
Longitudinal flange	26,250	22,856	22,865
Radial flange	17,500	10,981	10,982
Tangential flange	17,500	6,799	6,800

2. Bolt-up Condition

STRESS COMPONENT	ALLOWABLE STRESS (psi)	ACTUAL STRESS (psi) REFERENCE E0119	3.8-9
Bolts	25,000	6,077	-
Longitudinal flange	26,250	20,278	20,288
Radial flange	17,500	9,743	9,744
Tangential flange	17,500	6,032	6,033

Sample Problem B: Slip-on Flange

1. Design (Operating) Condition

STRESS COMPONENT	ALLOWABLE STRESS (psi)	ACTUAL STRESS (psi) REFERENCE E0119	3.8-9
Bolts	25,000	20,971	-
Longitudinal flange	26,250	21,160	21,163
Radial flange	17,500	11,128	11,128
Tangential flange	17,500	13,763	13,764

Table 3.8-16 (Cont'd)

2. Bolt-up Condition

STRESS COMPONENT	ALLOWABLE STRESS (psi)	ACTUAL STRESS (psi) REFERENCE E0119	ACTUAL STRESS (psi) REFERENCE 3.8-9
Bolts	25,000	5,671	-
Longitudinal flange	26,250	15,644	15,648
Radial flange	17,500	8,227	8,228
Tangential flange	17,500	10,175	10,177

Table 3.8-17

COMPARISON OF RESULTS FROM "E0781" PROGRAM AND PUBLISHED RESULTS[(1)]

PROGRAM E0781 RESULTS			PUBLISHED RESULTS
DISTANCE FROM BASE (in)	Nϕ (lb/in)	Mϕ (in-lb/in)	(Solution At Maximums)
0.0	5.919×10^{-6}	-1539.0	Mϕ = -1470 in-lb/in
6.0	21.15	-903.9	-
12.0	71.29	-440.5	-
18.0	134.0	-124.8	-
24.0	194.3	71.47	-
30.0	253.3	177.1	-
36.0	297.2	218.3	-
42.0	327.3	217.6	-
48.0	343.3	192.8	-
54.0	346.8	157.1	Nϕ = 348 lb/in
60.0	339.6	119.5	-
66.0	324.2	85.46	-
72.0	303.0	57.80	-
78.0	277.9	36.29	-
84.0	250.8	23.41	-
90.0	222.9	15.00	-
96.0	195.1	10.58	-
102.0	167.8	8.685	-
108.0	141.4	8.075	-
114.0	115.9	7.754	-
120.0	91.45	7.032	-
126.0	68.13	5.584	-
132.0	46.29	3.453	-
138.0	26.50	1.177	-
144.0	94.53	-1.481×10^{-3}	-

(1) Published results are from Reference 3.8-12.

Table 3.8-18

PARAMETERS ASSUMED FOR "FINEL" PROGRAM VERIFICATION

A. Material Properties - Concrete and Reinforcing Steel

PROPERTY	CONCRETE	STEEL
E	4.3×10^6 psi	29×10^6 psi
ν	0.15	0.29
T_{yield}	-4820 psi	±44,900 psi
E_{yield}	0.0	0.0
T_{crack}	+546 psi	-
E_{crack}	1.0	-
Shear stiffness reduction factor for once-cracked concrete	0.5	-

B. Loading History

LOAD, P (lb)	NUMBER OF CYCLES AT LOAD FOR CONVERGENCE
1	1
87,000	4
20,000	4
28,000	1
31,200	4
31,300	1[(1)]

(1) Reinforcing steel yielded

Table 3.8-19

COMPARISON OF RESULTS FROM "FINEL" PROGRAM AND HAND CALCULATIONS

Sample Problem B: Hollow Cylinder with Distributed Pressure Loading

		TANGENTIAL STRESS (ksf)		AXIAL STRESS (ksf)		RADIAL STRESS (ksf)	
ELEMENT	r (ft)	HAND CALCULATION	FINEL	HAND CALCULATION	FINEL	HAND CALCULATION	FINEL
1	65.19	17.79	17.79	4.212	4.212	-0.95	-0.95
2	65.56	17.69	17.69	4.212	4.212	-0.84	-0.84
3	65.94	17.58	17.58	4.212	4.212	-0.73	-0.73
4	66.31	17.48	17.48	4.212	4.212	-0.63	-0.63
5	66.69	17.38	17.38	4.212	4.212	-0.53	-0.53
6	67.06	17.28	17.28	4.212	4.212	-0.43	-0.43
7	67.44	17.18	17.18	4.212	4.212	-0.33	-0.33
8	67.81	17.08	17.08	4.212	4.212	-0.24	-0.23
9	68.19	16.99	16.99	4.212	4.212	-0.14	-0.14
10	68.56	16.89	16.89	4.212	4.212	0.05	-0.05

Sample Problem C: Hollow Cylinder with a Linear Temperature Gradient

		TANGENTIAL STRESS (ksf)		AXIAL STRESS (ksf)		RADIAL STRESS (ksf)	
ELEMENT	r (ft)	HAND CALCULATION	FINEL	HAND CALCULATION	FINEL	HAND CALCULATION	FINEL
1	65.19	-78.34	-78.33	-77.96	-77.96	-0.22	-0.23
2	65.56	-60.67	-60.66	-60.68	-60.68	-0.62	-0.62
3	65.94	-43.10	-43.09	-43.40	-43.40	-0.91	-0.91
4	66.31	-25.63	-25.62	-26.12	-26.12	-1.10	-1.10
5	66.69	- 8.26	- 8.25	- 8.84	- 8.84	-1.19	-1.19
6	67.06	9.01	9.02	8.44	8.44	-1.18	-1.18
7	67.44	26.19	26.20	25.72	25.72	-1.08	-1.08
8	67.81	43.27	43.28	43.00	43.00	-0.88	-0.88
9	68.19	60.26	60.27	60.28	60.28	-0.58	-0.59
10	68.56	77.16	77.17	77.56	77.56	-0.21	-0.21

Table 3.8-20

COMPARISON OF CONTAINMENT LOAD COMBINATIONS WITH SRP 3.8.3

LGS UFSAR TABLE	TYPE OF STRUCTURE	CONDITION	OMISSION	JUSTIFICATION
3.8-5 thru 3.8-8	Concrete and Steel internal structures of containment	All accident conditions	Thermal pipe (R_a)[1] were not included	Thermal pipe reactions are checked locally for structural adequacy. These loads are generally small compared to pipe loads and negligible in combination with other loads.
3.8-7	Drywell Platforms	1) Abnormal	1) Seismic not included	1) Seismic is small in comparison with pipe restraints and jet force loads which have been considered.
		2) Normal	2) Seismic not included	2) Combining seismic with dead and live loads gives higher allowable stresses and thus does not control the design.
3.8-7	Drywell Platforms	Normal Abnormal	1) T_a Neglected	1) Expansion of structural members under thermal loadings are permitted; therefore, thermal-stresses are negligible.
	Seismic Truss	Abnormal/ Extreme	2) P_a Neglected	2) The pressure differential along the supporting beams is negligible.
3.8-8	Seismic Truss	Abnormal/ Extreme	Live load (L) not included	Live loads are negligible in comparison with main steam pipe jet forces (R)[1]

[1] "R" as defined in the UFSAR is equivalent to the expression ($Y_r + Y_j + Y_m$) as defined in the SRP

Table 3.8-21

COMPARISON OF CONTAINMENT LOAD COMBINATIONS WITH ASME CODE, ARTICLE CC-3000

ASME LOAD	DIFFERENCES	ASME CATEGORY	JUSTIFICATION
F	Prestress not applicable	All	LGS has a reinforced concrete containment and no prestress loads exist.
W & W_t	Wind and tornado loads not considered	Construction Severe environmental Abnormal/Severe Environmental Extreme Environmental	Tornado or wind loadings do not apply. The primary containment is protected by the reactor enclosure.
T_t	Thermal not considered for structural integrity test.	Test	LGS design criteria addresses the structural integrity test. Pressure has been increased 15% over ASME. The ambient air temperature during testing is close to that of construction; therefore, the effects of thermal gradient are negligible.
E_o, P_a	Load Factors for E_o and P_a have been slightly reduced.	Abnormal Abnormal/Severe Environmental	Load factors for loading combination are based on probable occurrence. By introducing the hydrodynamic event, a slight load factor reduction for these loads (E_o, P_a) is justifiable.

3.9 MECHANICAL SYSTEMS AND COMPONENTS

The system and component design criteria and analyses in Section 3.9 are referenced to LGS operation at the originally licensed rated reactor power of 3293 MWt. The effects of increased power, pressure, and flow rates for power rerate conditions on the reactor vessel and internals, and main steam and recirculation piping were evaluated. The results demonstrate LGS compliance with appropriate design and licensing criteria at rerate conditions. These are documented in Ref. 3.9-23 and 3.9-24. Additional analyses were also performed which considered the use of GE13 and GE14 fuel; these analyses are documented in Reference 3.9-28 and Reference 3.9-34, respectively. The GE14 fuel is demonstrated to be bounded by the GE13 results. The reactor vessel components were evaluated for the effects of MUR power uprate in Reference 3.9-31. The results show that the reactor vessel components continue to comply with the structural requirements. The RPV internals were evaluated for loads associated with the MUR power uprate in Reference 3.9-32. The RPV internal components are demonstrated to be structurally qualified for operation in the MUR conditions. Reference 3.9-33 identifies new design basis values for Fuel Lift Margin and Control Rod GuideTube Lift Forces under MUR conditions. Subsequently, analyses were performed which considered the use of GNF2 fuel; these analyses aere documented in Reference 3.9-35. The GNF2 fuel is demonstrated to be bounded by the analyses in Reference 3.9-33.

3.9.1 SPECIAL TOPICS FOR MECHANICAL COMPONENTS

3.9.1.1 Design Transients

This section discusses the transients used in the design and fatigue analysis of the ASME Code Class 1 components, component supports, and reactor internals. The number of cycles or events for each transient based on available and projected plant operating data at the time of design is included. The design transients shown in this section are included in the design specifications for the components. Transients or combinations of transients are classified with respect to the component operating condition categories, identified as "normal," "upset," "emergency," "faulted," or "testing" in the ASME B&PV Code, if applicable. The first four operating conditions correspond to service levels A, B, C and D, respectively.

3.9.1.1.1 Control Rod Drive Transients

The normal and test service load cycles used for design purposes in the 40 year life of the CRD are as follows:

	Transient	Category	Cycles
a.	Reactor startup and shutdown	Normal/Upset(1)	120
b.	Vessel pressure tests	Normal/Upset	130
c.	Vessel overpressure	Normal/Upset	10
d.	Scram test plus startup scrams	Normal/Upset	300
e.	Operational scrams	Normal/Upset	300(2)
f.	Jog cycles	Normal/Upset	30,000(3)
g.	Shim/drive cycles	Normal/Upset	1,000(3)

(1) In Section 3.9.1.1, whenever a transient is categorized with two classifications, i.e., normal/upset, the most limiting of the two is considered in the design.
(2) 180 scram cycles constitute the design basis; however, 300 cycles are applied to the CRD for conservatism.
(3) 30,000 jog cycles and 1000 shim/drive cycles are applied because they impose mechanical loads on the CRD while contributing negligible thermal loads.

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

	Transient	Category	Cycles
h.	Scram with inoperative buffer	Normal/Upset	10
i.	Scram with stuck control blade	Normal/Upset	1
j.	OBE[3]	Upset	10
k.	SSE	Faulted	1

All ASME Class 1 components of the CRD have been analyzed according to ASME Section III. The capability of the CRD to withstand other emergency and faulted conditions is verified by test rather than analysis.

3.9.1.1.2 CRD Housing and Incore Housing Transients

Transients, classifications, and number of cycles considered in the design and fatigue analysis of the CRD housing and incore housing are as follows:

	Transient	Category	Cycles
a.	Normal startup and shutdown	Normal/Upset	120
b.	Vessel pressure tests	Normal/Upset	130
c.	Vessel overpressure tests	Normal/Upset	10
d.	Interruption of feedwater flow	Normal/Upset	80[1]
e.	Scram	Normal/Upset	200[2]
f.	OBE	Upset	10
g.	SSE	Faulted	1

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(1) The interruption of feedwater flow imposes thermal loads on the CRD housing while contributing negligible mechanical loads.

(2) 180 scram cycles constitute the design basis; however, 200 cycles are applied to the CRD housing for conservatism.

(3) The frequency of occurrence of this transient indicates the emergency category. However, for conservatism, the OBE was analyzed as an upset condition. Ten peak stress cycles are postulated.

CRD HOUSING ONLY

h.	Stuck rod scram	Normal/Upset	1
i.	Scram with no buffer	Normal/Upset	1

3.9.1.1.3 Hydraulic Control Unit Transients

	Transient	Category	Cycles
a.	Reactor startup and shutdown	Normal/Upset	120
b.	Scram tests	Normal/Upset	300
c.	Operational scrams	Normal/Upset	300
d.	Jog cycles	Normal/Upset	30,000
e.	Scram with stuck scram discharge valve	Emergency	1
f.	OBE	Upset	10
g.	SSE	Faulted	1

3.9.1.1.4 Core Support and Reactor Internals Transients

Cycles considered in the reactor internals design and fatigue analysis are listed in Table 3.9-2.

3.9.1.1.5 Main Steam System Transients

Transients considered in the main steam piping stress analysis are as follows:

	Transient	Category	Cycles
a.	Startup	Normal	120[(1)]
b.	Loss of feedwater pumps, isolation valves closed	Upset	10
c.	Scram	Upset	180
d.	Shutdown	Normal	111[(1)]
e.	Hydrostatic test	Test	3
f.	Design hydrotest	Test	130

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(1) In the design of NSSS piping systems, there are 9 transients not counted for the shutdown; 8 are due to SRV blowdown and 1 is due to automatic depressurization.

g.	OBE	Upset	50
h.	Turbine stop valve closure	Upset	120
i.	Relief valve lift (at 3 cycles per actuation)	Upset	34,200

3.9.1.1.6 Recirculation System Transients

Transients considered in the recirculation piping stress analysis are as follows:

	Transient	Category	Cycles
a.	Startup	Normal	120[(1)]
b.	Turbine roll and increase to power	Normal	120
c.	Loss of feedwater heater	Upset	10
d.	Partial feedwater heater bypass	Upset	70
e.	Scrams	Upset	180
f.	Shutdown	Normal	111[(1)]
g.	Loss of feedwater pumps, isolation valves closed	Upset	10
h.	Single SRV blowdown	Upset	8
i.	Design hydrotest	Test	130
j.	OBE	Upset	50

3.9.1.1.7 Reactor Assembly Transients

The reactor assembly includes the RPV, support skirt, shroud support, and shroud plate. The cycles listed in Table 3.9-2 are as specified in the reactor assembly design and fatigue analysis.

(1) In the design of NSSS piping systems, there are 9 transients not counted for the shutdown; 8 are due to SRV blowdown and 1 is due to automatic depressurization.

3.9.1.1.8 Main Steam Isolation Valve Transients

The MSIVs are designed for the following service conditions and thermal cycles:

	Transient	Category	Cycles
a.	Preop @100°F/hr	Normal/Upset	150
b.	Startup (heating @100°F/hr)	Normal/Upset	120
c.	Shutdown		
	1. Cooling cycles @100°F/hr, 540°F to 375°F	Normal/Upset	120
	2. Cooling cycles @270°F/hr, 375°F to 330°F	Normal/Upset	120
	3. Cooling cycles @100°F/hr, 330°F to 100°F	Normal/Upset	120
d.	Scram cooling cycles @100°F/hr	Normal/Upset	180
e.	Emergency and faulted transients		
	1. 546°F to 281°F in 15 seconds	Emergency/Faulted	1
	2. 546°F to 375°F in 3.3 minutes	Emergency/Faulted	1
	375°F to 281°F @300°F/hr	Emergency/Faulted	1
	3. 546°F to 375°F in 10 minutes	Emergency/Faulted	8
	375°F to 281°F @100°F/hr	Emergency/Faulted	8
	4. 546°F to 583°F in 2 seconds	Emergency/Faulted	1
	583°F to 538°F in 30 seconds	Emergency/Faulted	1
	538°F to 400°F @100°F/hr	Emergency/Faulted	1
	400°F to 546°F @100°F/hr	Emergency/Faulted	1
	5. 561°F to 500°F in 7 minutes	Emergency/Faulted	10
	500°F to 400°F @100°F/hr	Emergency/Faulted	10
	400°F to 546°F @100°F/hr	Emergency/Faulted	10

3.9.1.1.9 Main Steam Relief Valves Transients

The transients used in the analysis of the MSRVs are as follows:

	Transient	Category	Cycles
a.	Preop and inservice testing (100°F/hr)	Normal/Upset	150
b.	Startup (100°F/hr) and pressure increase (0 psig to 1000 psig)	Normal/Upset	120
c.	Shutdown (100°F/hr, pressure decrease to 0 psig, 270°F/hr between 375°F and 330°F)	Normal/Upset	120
d.	Scram	Normal/Upset	180
e.	System pressure and temperature decay is from 1000 psig and 546°F, to 35 psig and 281°F within 15 seconds.	Emergency/Faulted	1
f.	System temperature change is from 546°F to 375°F within 3.3 minutes, and from 375°F to 281°F at a rate of 300°F/hr. Pressure change is from 1000 psig to 35 psig.	Emergency/Faulted	1
g.	System temperature change is from 546°F to 375°F within 10 minutes, and from 375°F to 281°F, at a rate of 100°F/hr. Pressure change is from 1000 psig to 35 psig.	Emergency/Faulted	8
h.	System temperature change is from 546°F to 583°F within 2 seconds, from 583°F to 538°F within 30 seconds, and from 538°F to 400°F with return to 546°F at a rate of 100°F/hr. Pressure change is from 1000 psig to 1350 psig, thence to 240 psig, with return to 1000 psig.	Emergency/Faulted	1

i.	System temperature changes greater than 30°F, are from 561°F to 500°F within 7 minutes, and from 500°F to 400°F, with return-to-normal operating temperature of 546°F, at a rate of 100°F/hr. Pressure change is from 1000 psig to 1180 psig, to 240 psig, with return-to-normal operating pressure of 1000 psig.	Emergency/Faulted	10

In addition, for RPV, RPV internals and piping New Loads Adequacy Evaluation, at least 7700 SRV cycles are considered to account for the pool dynamic loads. These 7700 cycles are based on 1100 actuations (total for 40 years) of all SRVs times seven stress cycles per actuation. Further, 4700 actuations (total for 40 years) for the most frequently actuated SRV times three stress cycles per actuation (14,100 total cycles) are used in the analysis of the SRV downcomers and SRV discharge lines in the wetwell and in the main steam piping analysis. The 4700 actuations over the 40 year plant design life for the most frequently actuated SRV is based on the original two-stage Target Rock SRV design used for LGS. This SRV has a longer blowdown time, similar to the Crosby SRV design.

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

3.9.1.1.10 Recirculation Flow Control Valve Transients

Not applicable; LGS has no flow control valve.

3.9.1.1.11 Recirculation Pump Transients

The following transients are listed in the design specification as a requirement for design considerations. The vendor was required to submit a certification of compliance with certified design calculations which considered only a pressure transient (no thermal-stresses were required to be considered). Nozzle piping loads were considered in accordance with the following paragraph from the design specification:

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

	Transient	Category	Cycles
a.	Heatup and cooldown at 100°F/hr	Normal/Upset	30
b.	±29°F temperature changes	Normal/Upset	600
c.	±50°F temperature changes	Normal/Upset	200
d.	RPV pressure transients to 110% design pressure	Normal/Upset	30
e.	SRV blowdowns	Emergency	1
f.	Improper pump startup, 100°F to 546°F in 15 seconds	Emergency	1
g.	Cooling transient, 552°F to 281°F in 15 seconds	Faulted	2
h.	Hydrotest to 1300 psig	Test	130
i.	Hydrotest to 1670 psig	Test	3

3.9.1.1.12 Recirculation Gate Valve Transients

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

	Transient	Cycles
a.	50°F-575°F-50°F at 100°F/hr	300
b.	±29°F between limits of 50°F and 575°F, instantaneous	600
c.	±50°F between limits of 50°F and 546°F, instantaneous	200
d.	546°F to 375°F, over a 10 minute period	30
e.	546°F to 281°F, over a 15 second period	2
f.	130°F to 546°F, over a 15 second period	1
g.	110% design pressure at 575°F	1
h.	1300 psi at 100°F installed hydrostatic test	130
i.	1670 psi at 100°F installed hydrostatic test	3

3.9.1.2 Computer Programs Used in Analysis

Sections 3.9.1.2.1 through 3.9.1.2.5 discuss computer programs used in the analysis of specific NSSS components (computer programs were not used in the analysis of all components, thus not all components are listed). Non-NSSS programs are discussed in Section 3.9.1.2.6.

GE Programs

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

a. PIPST01
b. MASS
c. SNAP (MULTISHELL)
d. GASP
e. NOHEAT
f. FINITE
g. DYSEA
h. SHELL 5
i. HEATER
j. FAP 71
k. CREEP PLAST
l. ANSYS
m. SAP4
n. ANSI-7
o. NOZAR
p. TSFOR
q. PISYS
r. PDA
s. EZPYP
t. LION4
u. SIMOK
v. DISPL
w. WTNOZ
x. SPECA04
y. GEAPL01
z. POSUM
aa. FTFLG

Vendor Programs

The verification of the following two groups of vendor programs is assured by contractual requirements between GE and the vendors. In accordance with the requirements, the quality assurance procedure of these proprietary programs used in the design of N-stamped equipment and non-ASME code items is in full compliance with 10CFR50, Appendix B.

Pump Motor Vendor Programs

a. RTRMEC

Chicago Bridge and Iron Programs

a. 7-11-GENOZZ
b. 9-48-NAPALM
c. 1027
d. 846
e. 781-KALNINS
f. 979-ASFAST
g. 766-TEMAPR
h. 767-PRINCESS
i. 928-TGRV
j. 962-E09262A
k. 984
l. 992-GAS
m. 1037-DUNHAM'S
n. 1335
o. 1606&1657-HAP
p. 1634N

3.9.1.2.1 Reactor Vessel and Internals

3.9.1.2.1.1 Reactor Vessel

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

3.9.1.2.1.1.1 Chicago Bridge and Iron Program 7-11 - GENOZZ

The GENOZZ computer program is used to proportion barrel and double taper-type nozzles to comply with the specifications of ASME Section III, and contract documents. The program either designs such a configuration or analyzes the configuration input to comply to code. If the input configuration does not comply with the specifications, the program modifies the design and redesigns it to yield an acceptable result.

3.9.1.2.1.1.2 Chicago Bridge and Iron Program 9-48 - NAPALM

The basis for the Nozzle Analysis Program - All Loads Mechanical (NAPALM) is to analyze nozzles for mechanical loads and find the maximum stress intensity and location. The program provides analyses at each mechanical load point of application. The maximum stress intensity is calculated for both the inside and outside surfaces at each reference location. The program measures the maximum stress intensity for both the inside and outside surfaces of the nozzle, as well as their angular locations as measured from the 0° reference location. The principle stresses are also listed.

Stresses resulting from each component of loading (bending, axial, shear, and torsion) are listed, as well as the loadings which cause these stresses.

3.9.1.2.1.1.3 Chicago Bridge and Iron Program 1027

This program is a computerized version of the analysis method contained in Reference 3.9-1.

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

3.9.1.2.1.1.4 Chicago Bridge and Iron Program 846

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

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

3.9.1.2.1.1.5 Chicago Bridge and Iron Program 781 - KALNINS

The KALNINS thin-shell program is used to establish the shell influence coefficient, and to perform the detailed stress analysis of the vessel. The stresses and the deformations of the vessel can be computed for any combination of the following axisymmetric loading:

a. Preload condition

b. Internal pressure

c. Thermal load

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

The basic method of analysis was published by Professor Kalnins (Reference 3.9-2).

3.9.1.2.1.1.6 Chicago Bridge and Iron Program 979 - ASFAST

The ASFAST program performs the stress analysis of axisymmetric, bolted closure flanges between the head and cylindrical shell.

3.9.1.2.1.1.6.1 ANSYS Engineering Analysis System, Revision 5.6, ANSYS, Inc.

ANSYS Program performed the stress analysis of the reactor vessel head, flange & upper shell for reduced pass tensioning/detensioning of RPV studs connecting RPV head to RPV shell.

3.9.1.2.1.1.7 Chicago Bridge and Iron Program 766 - TEMAPR

This program reduces any arbitrary temperature gradient through the wall thickness to an equivalent linear gradient. The resulting equivalent gradient has the same average temperature, and the same temperature-moment as the given temperature gradient. The input consists of the wall thickness and actual temperature distribution. The output contains the average temperature and total gradient through the wall thickness. The program is written in FORTRAN IV.

3.9.1.2.1.1.8 Chicago Bridge and Iron Program 767 - PRINCESS

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

3.9.1.2.1.1.9 Chicago Bridge and Iron Program 928 - TGRV

The TGRV program is used to calculate temperature distributions in structures or vessels. Although it is primarily a program for solving the heat conduction equations, some provisions have been made for including radiation and convection effects at the surfaces of the vessel.

The TGRV program is a highly modified version of the TIGER heat transfer program, written about 1958 at Knolls Atomic Power Laboratory, by A.P. Bray.

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

TGRV calculates and outputs the steady-state or transient temperature distributions in a given structure, as a function of time. The program inputs are any odd-shaped structure which can be represented by a three-dimensional field, its geometry and physical properties, boundary conditions, and internal heat generation rates.

3.9.1.2.1.1.10 Chicago Bridge and Iron Program 962 - E0962A

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

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

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

3.9.1.2.1.1.11 Chicago Bridge and Iron Program 984

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

3.9.1.2.1.1.12 Chicago Bridge and Iron Program 992 GASP

The GASP program, originated by Professor E.L. Wilson of the University of California at Berkeley, uses the finite-element method to determine the stresses and displacements of plane or axisymmetric structures of arbitrary geometry, and is written in FORTRAN IV. See Reference 3.9-3, for a detailed account. GASP structures may have arbitrary geometry, and have linear or nonlinear material properties. The loadings may be thermal, mechanical, accelerational, or a combination of these.

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

3.9.1.2.1.1.13 Chicago Bridge and Iron Program 1037 - DUNHAM'S

DUNHAM'S program is a finite ring element stress analysis program. It determines the stresses and displacements of axisymmetric structures of arbitrary geometry subjected to either axisymmetric loads, or nonaxisymmetric loads represented by a Fourier series.

This program is similar to the GASP program (Chicago Bridge and Iron Program 992). The major differences are that DUNHAM'S can handle nonaxisymmetric loads (which requires that each node have three degrees of freedom), while the material properties for DUNHAM'S must be constant. As in GASP, the loadings may be thermal, mechanical, and accelerational.

3.9.1.2.1.1.14 Chicago Bridge and Iron Program 1335

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

3.9.1.2.1.1.15 Chicago Bridge and Iron Programs 1606 and 1657 - HAP

The HAP program is an axisymmetric nonlinear heat analysis program. It is a finite-element program, used to determine nodal temperatures in a two-dimensional or axisymmetric body subject to transient disturbances. Programs 1606 and 1657 are identical, except that 1606 has a larger storage area allocated, and can thus be used to solve larger problems. The model for program 1606 is compatible with Chicago Bridge and Iron stress programs 992 (GASP) and 1037 (DUNHAM'S).

3.9.1.2.1.1.16 Chicago Bridge and Iron Program 1634N

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

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

3.9.1.2.1.2 Reactor Internals

3.9.1.2.1.2.1 Core Plate Beam Buckling - PIPST01

PIPST01 is a computer program that calculates approximate core plate beam buckling capability. It uses the Rayleigh-Ritz energy method to determine the applied moment needed to begin yielding and then to buckle a given tee beam. The tee beam models a segment of a BWR/2-5 core plate with a stiffener beam. The pressure differential across the plate that would have created this moment is calculated for a given length of beam or size of core plate.

Generic dimension and material properties are all input by the user.

3.9.1.2.1.2.2 Other Programs

Other computer codes used for the analysis of the internal components are:

a. MASS
b. SNAP (MULTISHELL)
c. GASP
d. NOHEAT
e. FINITE
f. DYSEA
g. SHELL 5
h. HEATER
i. FAP 71
j. CREEP PLAST
k. ANSYS

Descriptions of these programs are given in Section 4.1.4.1.

3.9.1.2.2 Piping

3.9.1.2.2.1 Structural Analysis Program - SAP4

SAP4 is a general Structural Analysis Program for static and dynamic analysis of linear elastic complex structures. The finite-element displacement method is used to solve the displacements, and to compute the stresses of each element of the structure. The structure can be composed of unlimited numbers of three-dimensional truss, beam, plate, shell, solid, plate strain-plane stress, brick, thick shell, spring, or axisymmetric elements. The program can treat thermal and various forms of mechanical loading, as well as internal element loading. The dynamic analysis includes mode- superposition, time history, and response spectrum analyses. Earthquake loading, as well as time-varying pressure, can be treated. The program is very versatile and efficient in solving large and complex structural systems. The output contains displacements of each nodal point, as well as stresses at the surface of each element.

3.9.1.2.2.2 Component Analysis - ANSI-7

The ANSI-7 Computer Program determines stress and accumulative usage factors in accordance with NB-3600 of ASME Section III. The program performs stress analyses in accordance with the ASME sample problem, and has been verified by reproducing the results of the sample problem analysis.

3.9.1.2.2.3 Area Reinforcement - NOZAR

The computer program Nozzle Area Reinforcement Program (NOZAR) performs an analysis of the required reinforcement area for openings. The calculations performed by NOZAR are in accordance with the rules of the 1974 edition of ASME Section III.

3.9.1.2.2.4 Turbine Stop Valve Closure - TSFOR

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

3.9.1.2.2.5 Piping Analysis Program/PISYS

PISYS is a computer code for analyzing piping systems subjected to both static and dynamic piping loads. Stiffness matrices representing standard piping components are assembled by the program to form a finite-element model of a piping system. The piping elements are connected to each other via nodes called pipe joints. It is through these joints that the model interacts with the environment, and loading of the piping system becomes possible. PISYS is based on the linear-elastic analysis in which the resultant deformations, forces, moments, and accelerations at each joint are proportional to the loading and the superposition of loading is valid.

PISYS has a full range of static and dynamic load analysis options. Static analysis includes dead weight, uniformly distributed weight, thermal expansion, externally applied forces, moments, imposed displacements, and differential support movement (pseudostatic load case). Dynamic analysis includes mode shape extraction, response spectrum analysis, and time history analysis by modal combination or direct integration. In the response spectrum analysis, i.e., Uniform Support Motion Response Spectrum Analysis (USMA) or Independent Support Motion Response Spectrum Analysis (ISMA), the user may request modal response combination in accordance with Regulatory Guide 1.92. In the ground motion (uniform motion) or independent support time history analysis, the normal mode solution procedure is selected. In analysis involving time-varying nodal loads, the step-by-step direct integration method is used.

The PISYS program has been benchmarked against NRC piping models. The results are documented in a report to the Commission, "PISYS Analysis of NRC Benchmark Problems," NEDO-24210, August 1979, for mode shapes and USMA options. The ISMA option has been validated against NUREG/CR-1677, "Piping Benchmark Problems Dynamic Analysis Independent Support Motion Response Spectrum Method," published in August 1985.

3.9.1.2.2.6 Piping Dynamic Analysis Program - PDA

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

Shear deformation is also neglected. The pipe bending moment-deflection (or rotation) relation used for these locations is obtained from a static nonlinear cantilever beam analysis. Using moment-rotation relations, nonlinear equations of motion are formulated using energy considerations, and the equations are numerically integrated in small time steps to yield the time history of the pipe motion. Additional discussion of PDA is provided in Section 3.6.2.2.2.

3.9.1.2.2.7 Piping Analysis Program - EZPYP

EZPYP links the ANSI-7 and SAP programs together. The EZPYP program can be used to run several SAP cases by making user- specified changes to a basic SAP pipe model. By controlling files and SAP runs, the EZPYP program makes it possible to perform a complete piping analysis in one computer run.

3.9.1.2.2.8 Thermal Transient Program - LION4

The LION4 program is used to compute radial axialthermal gradients in piping. The program calculates a time history of ΔT_1, ΔT_2, T_a, and T_b (defined in ASME Section III, Class 1 piping analysis) for uniform and tapered pipe wall thickness.

3.9.1.2.2.9 Synthetic Time History Program - SIMOK

The SIMOK program provides a time history that is equivalent to an input response spectrum. The synthetic time history is used to generate a new spectrum that is plotted with the input spectrum, to verify that the time history and spectrum are equivalent. Synthetic time histories are used in a multiple input analysis of the piping.

3.9.1.2.2.10 Differential Displacement Program - DISPL

The DISPL program provides differential movements at each piping attachment point, based on building modal displacements.

3.9.1.2.2.11 WTNOZ Computer Program

WTNOZ is a time-share program for piping weight calculations.

3.9.1.2.3 Pumps and Motors

3.9.1.2.3.1 Recirculation Pumps

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

3.9.1.2.3.2 Core Spray Pumps and Motors

The RTRMEC computer program is used in the analysis of a motor design representative of (or very similar in mechanical construction to) the core spray pump motor.

RTRMEC calculates and displays the results of a mechanical analysis of a motor rotor assembly acted upon by external forces at any point along the shaft (rotating parts only). The shaft deflection analysis, including magnetic and centrifugal forces, was conducted. The calculation for the seismic condition assumes that the motor is operating, and that the seismic, magnetic, and centrifugal forces all act simultaneously and in phase on the rotor-shaft assembly. Note that the distributed motor assembly weight is lumped at the various stations. The shaft weight at a station is the sum of one-half the weight of the incremental shaft length just before the station, plus one-half the weight of the adjacent incremental shaft length just after the station. Bending and shear effects are accounted for in the calculations.

The FTFLG computer program was used to analyze the flange joints connecting the pump bowl castings. The description of this program is provided in Section 3.9.1.2.5.3.

3.9.1.2.4 Dynamic Loads Analysis

3.9.1.2.4.1 Acceleration Response Spectrum Program/SPECA04

The SPECA04 computer program generates acceleration response spectrum, consistent with Regulatory Guide 1.122 for an arbitrary input of time history of piecewise-linear accelerations, i.e., to compute maximum acceleration responses for a series of single-degree-of-freedom systems subjected to the same input. It can accept acceleration time histories from a random file.
It also has the capability of generating the broadened/enveloping spectra in conformance with Regulatory Guide 1.122 when the spectral points are generated equally spaced in a logarithmic scale axis of period/frequency. This program is also used in seismic and SRV transient analysis.

3.9.1.2.4.2 Forces and Moment Time Histories Program/GEAPL01

The GEAPL01 computer program converts distributed asymmetric pressure time histories over a given area into equivalent time- varying nodal forces and moments for use as input to perform dynamic analysis of a system. The overall resultant forces and moment time histories at specified points of resolution can also be obtained from GEAPL01.

3.9.1.2.5 Residual Heat Removal Heat Exchangers

3.9.1.2.5.1 Structural Analysis Program - SAP4

SAP4 is used to analyze the structural and functional integrity of the RHR heat exchangers. The description of this program is provided in Section 3.9.1.2.2.1.

3.9.1.2.5.2 Beam Element Data Processing Program/POSUM

POSUM is used to process SAP4 generated data. POSUM is a computer code designed to process SAP4 generated beam element data for pump or heat exchanger models. The purpose is to determine the load combination that would produce the maximum stress in a selected beam element. It is intended for use on RHR heat exchangers with four nozzles or core spray pumps with two nozzles.

3.9.1.2.5.3 Effects of Flange Joint Connections/FTFLG

The flange joints connecting the pump bowl castings are analyzed using FTFLG program. This program uses the local forces and moments determined by SAP4 to perform flat flange calculations in accordance with the rules set forth in ASME Appendix II and ASME Section III.

3.9.1.2.6 Seismic Category I Items Other than NSSS

A list of computer programs used in the non-NSSS system components is provided in Table 3.9-3. This list consists of computer programs developed and/or owned by Bechtel, and of computer programs that are recognized and widely used in industry.

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

3.9.1.2.6.1 ME101, Linear-Elastic Analysis

ME101 is a finite-element computer program that performs linear-elastic analysis of piping systems using standard beam theory techniques. The input data format is specifically designed for pipe stress engineering. ME101 performs a thorough check of the input prior to analysis. In addition, the program automatically modifies the geometry to improve the finite-element model.

The output may be used directly for piping design, for conformation to Code, and for other regulatory requirements. Both ASME Section III and ANSI B31.1 piping code editions are incorporated in ME101 to the extent of computing flexibility factors, stress intensification factors, and stresses.

ME101 performs static and dynamic load analysis of piping systems, effective weight calculations, and ASME Section III, Class 2 and 3, and ANSI B31.1 Code stress checks.

Static analysis considers one or more of the following: thermal expansion, dead weight, uniformly distributed loads, and externally applied forces, moments, imposed displacements and rotations, individual force loads, static seismic (uniform directional acceleration) loads, or seismic anchor movement analysis.

Dynamic analysis is based on the standard normal superposition techniques. The input excitation may be in the form of seismic response spectra or time-dependent loading functions. In the single or multiple response spectrum analysis, the user may request modal synthesis by SRSS method or by Regulatory Guide 1.92 closely spaced mode 10% (equation 4) method. ME101 can consider further differential damping for large and small pipe according to Regulatory Guide 1.61. Various methods of eigenvalue solution are available. Determinant search or subspace iteration considers all data points as mass points. In the time history analysis, the excitation may be in the form of arbitrary nodal forces, support displacements, rotations, or support accelerations that are not necessarily in phase.

ME101 checks stresses from design loads versus allowable stresses according to ASME/ANSI Code equations. The user may request design load checks for sustained loads, occasional loads, multimode thermal expansion and pipe break, except for time history load cases.

The ME101 restraint load summary report prints the support load results from several load cases together in the same report, except for time history load cases.

The general loading combinations capability for ME101 can combine the results of several load cases together, according to certain algebraic rules, to form a new load case. The new load case resulting from this may be used in stress comparisons or restraint load summaries, except for time history load cases. ME101 has the capability of saving load case results on a tape and using these results in late runs for stress checks, restraint load summary reports, and general loading combinations, except for time history load cases.

For piping configurations with optional node numbering, ME101 generates isometric plots. The user may obtain plots on ZETA or CALCOMP plotters on a Tektronix 4014 graphics terminal, or on a RMS-600 printer/plotter.

ME101 uses out-of-core techniques for both static and response spectra analysis and has no practical limitations to the number of equations or band width. However, the use of very large systems may become prohibitive due to cost of computation. The maximum number of mode shapes allowable for response spectra analysis is currently 125.

This program considers the zero period acceleration effect in seismic response analysis. It accepts coordinate and key-word data in English or metric units.

The ASME Benchmark Problem 1 demonstrates the solution for natural frequencies of a three-dimensional structure, as described in Reference 3.9-4.

Natural frequencies, in Hertz, from ME101 and Reference 3.9-4, are as follows:

Mode	Reference 3.9-4	ME101
1	110	112
2	117	116
3	134	138

A total of 26 test problems were used for the verification of the ME101 results. These verification problems have been compared against one of the following:

a. ME632, Computer Program, "Seismic Analysis of Piping Systems", VERB MODB, 1976 Bechtel International Corporation, San Francisco, CA.

b. "Pressure Vessel and Piping 1972 Computer Programs Verification", ASME.

c. Hand Calculations

d. EDS Superpipe, EDS Nuclear, San Francisco, CA.

e. NUPIPE-IIM, Nuclear Services Corporation Piping Analysis Program, Campbell, CA.

f. TPIPE, A Computer Program for Analysis of Piping Systems, PMB Systems Engineering, San Francisco, CA.

g. ADINA, A Computer Program, Massachusetts Institute of Technology, Boston, MA.

h. MSC/NASTRAN Program, McNeal Schwendler Corporation, Los Angeles, CA.

i. EASE2 Program, Engineering/Analysis Corporation, San Francisco, CA.

j. ANSYS, Swanson Analysis System, Inc., 1975, Elizabeth, PA.

The J1 version of ME101 also includes seven NRC benchmarked problems, as referenced in NUREG/CR-1677, dated August 1980.

ME-101 was also validated with the four benchmarked problems (using the multiple response spectrum/independent support motion method) that were transmitted from M. Hartzman (NRC) to G. Wang (Bechtel) on August 10, 1983. The results and comparisons of the problems using ME-101 were transmitted to M. Hartzman (NRC) from T.J. McDonald (Bechtel) on December 15, 1983.

The ME-101 MRS/ISM methodology was again validated with four multiple response spectrum problems, as referenced in NUREG/ CR-1677, Volume II, dated August 1985.

3.9.1.2.6.2 ME632, Piping System Analysis

Program Description

ME632 performs stress analyses of three-dimensional piping systems. The effects of thermal expansion, uniform load of the pipe, pipe contents and insulation, concentrated loads, movements of the piping system supports, and other external loads, such as wind and snow, may be considered. The input data format is specifically designed for pipe stress engineering, and the English system of units is used. A thorough checking of the input has been coordinated in the program.

The output may be used directly for piping design, and for conformation to code and other regulatory requirements. Piping codes, the ASME B&PV code, the B31.1 code, and the B31.3 code have been incorporated into the program to the extent of computing flexibility factors, stress intensification factors, and stresses.

A response spectrum analysis may be performed to analyze the effect of earthquake forces on the piping system; transient effects of water hammer, steam hammer, or other impulsive types of dynamic loading are also handled by the program. A plot of piping geometry and/or response spectrum curves may be obtained to verify the accuracy of the model.

Program Version and Computer

The current UNIVAC version of ME632 is being used by Bechtel.

Extent of Application

ME632 is a piping program developed by Bechtel. Its development began in 1970, and it is being continuously supported by Bechtel. It has been used by various Bechtel projects.

Test Problems

The ASME Benchmark Problem No. 1 demonstrates the solution for natural frequencies of a three-dimensional structure, as described in Reference 3.9-4.

The following table lists the natural frequencies from ME632 and Reference 3.9-4:

Natural Frequency Comparison, Hz

Mode No.	Reference 3.9-4	ME 632
1	110	111
2	117	116
3	134	137

3.9.1.2.6.3 ME912, Thermal-Stress

Program Description

Finite-difference representation of the heat diffusion equation is used for the pipe or component wall section in contact with fluid of specified temperature and flow rate time histories. The program is quasi-two-dimensional, so that reduction of severity of a given transient with distance from inlet is accounted for.

Thermal properties of water, and stainless and carbon steel are built in the program. Film transfer coefficients for water are computed by the program for each time step and pipe section. For other fluids such as steam, the program is used on a one-dimensional basis with user-supplied film coefficients. Sequential computations are done for pipe lengths of different diameters or wall thicknesses. Fluid outlet temperature data from one pipe length are stored for use as the inlet to the next pipe length downstream. Average temperature differences, T_a - T_b, are thus calculated for structural discontinuity.

Program Version and Computer

The ME912 program has been used by Bechtel on various Bechtel projects. A Univac 1110 computer is used to run the ME912 program.

Extent of Application

The ME912 program was developed from References 3.9-7, 3.9-8, and 3.9-9 by Bechtel. The ME912 program has been extensively used since 1975 for nuclear Class 1 component design on the Fast Flux Test Facility project.

Test Problem

For local gradients, the program has been compared with analytical flat plate data of Reference 3.9-8, and numerical results by in-house program ME643. The results are acceptable. Table 3.9-4 shows the comparison of ME912 with ME643 and analytical results from Reference 3.9-8. For axial variations of fluid and wall temperatures, the program agrees closely with the analytical solution of Reference 3.9-9.

The ME643 program was developed from References 3.9-11 and 3.9-12 by Bechtel.

The results of ME643 transient temperature responses on both inside and outside surfaces of a sample pipe are compared with Chart 36 of Reference 3.9-13, and plotted in Figure 3.9-1.

3.9.1.2.6.4 ME913, Nuclear Class 1 Piping Stress Analysis

Program Description

ME913 can determine stress intensity levels for Class 1 nuclear power piping components, equations 9 through 14 of subarticle NB-3650, "Analysis of Piping Components", ASME Section III.

Prior to using this program, the following information external to the program is required:

a. Piping configuration

b. Piping and piping component properties

c. Moment reactions due to:

   1. Thermal expansion loads

   2. Weight loads

   3. Dynamic loads

d. The thermal response of the piping system due to the specified transients is:

   ΔT_1, ΔT_2 and the (T_a-T_b) values for the key points during system life.

Program Version and Computer

The current ME913 version is being used by Bechtel. A Univac 1100 computer is used to run the ME913 program.

Extent of Application

ME913 is the revised and expanded version of the LOTEMP program, originally developed by Bechtel, and made available for use through the CDC 6600 computer. The LOTEMP program has been extensively used by the Bechtel Fast Flux Test Facility Systems Analysis Group since 1972, in the preliminary design of Fast Flux Test Facility Class 1 piping. The ME913 program has been used to analyze nuclear Class 1 piping for Bechtel nuclear power plant projects.

Test Problems

The Grand Gulf Project feedwater line was selected as a test problem. Hand calculations of a selected component in the piping system were performed in accordance with the sample problem (Reference 3.9-14). The results were compared with the computer output for code equations 9 through 14 in ME913. Table 3.9-5 shows the comparison between the ASME sample problem (Reference 3.9-14) and ME913 results.

3.9.1.2.6.5 NE452, Submerged Steam Line Reflood Analysis

Program Description

NE452 is used to analyze reflood transient in steam relief valve discharge lines that discharge to a water pool and are equipped with a vacuum relief valve. The code models the effect of the vacuum relief valve (considering both the air flow and the valve dynamics) on the line reflood phenomenon by calculating the dynamics of the slug motion of the reflooding water. The slug motion is affected by the steam condensation on the surface of the reflooding water and by the presence of noncondensable air admitted by the vacuum relief valve.

The output (predicted maximum reflood) can be used in relief line clearing transient analysis codes (such as NE805, discussed in Section 3.9.1.2.6.6) to determine pipe run forces. NE452, together with NE805, can be used for vacuum relief valve sizing.

Program Version and Computer

The current NE452 version is being used by Bechtel. A UNIVAC 1100 computer is used to run the NE452 program.

Extent of Application

Development of the NE452 program began in 1977 and is being continuously supported by Bechtel. It has been used for various Bechtel nuclear power plant projects.

Test Problems

The NE452 program was verified against PBAPS and Monticello ramshead and Mark I T-quencher test data. It has also been verified against Karlstein Mark II T-quencher and Caorso X-quencher test data. The predicted results agreed reasonably well with test data.

3.9.1.2.6.6 NE805, Relief Valve Clearing Analysis

Program Description

NE805 is a computer code that analyzes transients in discharge lines of changing cross-sectional area following a relief valve opening. The code predicts the time-dependent forces on the various pipe segments of the relief valve discharge line. It models the steam flow through the relief valve and the steam/air flows in the line. It also models the water flow in the submerged part of the line. The options for the exit device are a straight pipe, a ramshead, or a quencher model in the reservoir. The quencher model considers sequential uncovering of the quencher holes during air/water clearing. NE805 uses the method of characteristics and allows for heat transfer through the pipe wall. It calculates flow parameters, pressure, velocity, and density as functions of time and the distance along the discharge line. Using these calculated values, the code computes the dynamic forcing functions induced on various pipe segments of the relief valve discharge line.

The force output can be used directly for piping stress analysis in codes such as ME101, described in Section 3.9.1.2.6.1.

NE805 generates plots of flow parameter time histories and/or force time histories, an option specified by the user. The plots are obtained by CALCOMP 1036 plotter.

Program Version and Computer

The current UNIVAC version of NE805 is being used by Bechtel. A UNIVAC 1100 computer is used to run the NE805 program.

Extent of Application

NE805 was developed by Bechtel. Its development began in 1975 and is being continuously supported by Bechtel. It has been used by various Bechtel projects.

Test Problems

The NE805 program has been verified against Monticello Mark I T-quencher test, Karlstein Mark II T-quencher test, and Caorso X-quencher test. Comparison with test data was found to be reasonable.

3.9.1.2.6.7 ANSYS

Refer to Section 3.8.7.10

3.9.1.2.6.8 ME210 - Local Stress in Cylindrical Shells Due to External Loads

This standard presents a method of analyzing and determining local stresses in cylindrical shells due to external moments and forces acting on rigid attachments of circular or rectangular shape. This program is based on a paper "Local Stresses in Spherical and Cylindrical Shells Due to External Loadings" by Wichman, Hopper, and Mershon, published in Welding Research Council Bulletin No. 107, August 1965 and March 1979 Revision. Values from Bijlaard curves are obtained by interpolation procedures.

This program also calculates piping stress intensity due to internal pressures and moments in accordance with the pressure and moment stress calculations specified in equation 9 and equation 10 of ASME Section III, NB-3650. The local stress intensity and piping stress intensity are summed and printed out if the required information for piping stress calculation is specified in the input. If no information for piping stress calculation is given, only the local stresses including primary plus secondary stress intensity and primary membrane stress intensities are printed out.

3.9.1.2.6.9 ME602 - Spectra Merging and Simplified Seismic Analysis

ME602 performs the seismic analysis of small diameter piping systems (2 inch and under) using the modified response spectrum method described in BP-TOP-1, Revision 3. The program generates a set of tables of seismic spans, support reactions, and stresses for various pipe sizes.

This program performs response spectrum curve merging along with the calculation of the seismic span. The program can also be used independently for the sole purpose of merging spectrum curves and storing the combined spectrum data for ME101 analysis. A neutral plot file of the "RAW" or "COMBINED" spectrum curves can be generated for plotting on RMS, TEKTRONIX, CALCOMP, or any neutral file compatible plotter.

3.9.1.2.6.10 ME351 - Pipe Rupture Analysis Program

This program performs nonlinear elastic-plastic analysis of three-dimensional piping systems subjected to concentrated static or dynamic time history forcing functions. These forces may result from fluid jet thrust at the location of a postulated rupture of high energy piping. PIPERUP is an adaptation of the finite-element method to the specific requirements of pipe rupture analysis. Straight and curved beam (elbow) elements are used to mathematically represent the piping, and axial and rotational springs are used to represent restraints. The stiffness characteristics of piping and restraints can reflect elastic/linear strain hardening material properties, and gaps between piping and restraints can be modeled.

3.9.1.2.7 Computer Programs Used for Component Supports

A list of computer programs used in pipe support analysis is provided in Table 3.9-3. The list consists of computer programs developed and/or owned by Bechtel, and computer programs that are recognized and widely used in the industry.

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

3.9.1.2.7.1 CE050 - BOLTS

BOLTS is an interactive program for determining the loads on concrete expansion anchors used for baseplates with symmetrical bolt patterns. The program incorporates the effects of plate flexibility, bolt stiffness, and attachment size. BOLTS is particularly well suited for the evaluation of baseplate expansion anchors commonly used in pipe, conduit, HVAC, cable tray, and such other small equipment supports.

3.9.1.2.7.2 ME150 - FAPPS - Frame Analysis for Pipe Supports

ME150 is an interactive program for the analysis and design of pipe support frames. It has built-in standard frames and the capability to design any other nonstandard ones. It optimizes member sizes, welds, and embedments based on various user-specified design criteria.

3.9.1.2.7.3 ME225 - Anchor Plate

ME225 presents a method of designing plate-type piping anchors. It determines thickness of anchor plate, thickness of guide plate, weld joining the plate and process pipe, weld joining the supporting structure, and take out dimension of anchor plate and guide plate.

3.9.1.2.7.4 ME035 - BASEPLATE

ME035 analyzes baseplate-type structural supports. It assumes a flexible plate resting on a nonlinear foundation. It gives concrete stresses, bolt factors of safety, and weld forces. It can analyze baseplates with variable thickness with multiple columnlike attachments.

3.9.1.2.7.5 ME226 - PICLAMP

PICLAMP will design the components of six special cases of pipe support clamps. It computes the minimum required thickness at two critical sections of the clamp. It also calculates the stress in the clamp studs. It computes the stresses in the stanchion and baseplate, when applicable, and the minimum weld size based on the stress. It also computes certain clamp dimensions and the total weight of the clamp and its associated hardware.

3.9.1.2.7.6 ME425 - STAND

STAND will design and evaluate pipe support base plates with concrete anchor bolt assemblies. Plates can be of arbitrary geometry anchored with bolts that can be located in a random pattern.

3.9.1.2.7.7 ME120 - WELD

The program presents a method of determining fillet weld sizes for connecting structural members. It accepts five different types of structural shapes and analyzes for 2-weld to 16-weld configurations. It is based on the approach described in "Design of Welded Structures" by O.W. Blodgett and "Solutions to Design of Weldments" by O.W. Blodgett.

3.9.1.2.7.8 ME152 - SMAPPS

SMAPPS analyzes and designs commonly used standard frames for pipe support including associated welds and baseplates with anchors for AISC and ASME Section III, Subsection NF requirements, as well as project deflection/stiffness requirements.

SMAPPS provides the benefits of a structural frame analysis program and the simplicity of pre-engineering standards. SMAPPS provides margin factors for frame, welds, and baseplate with anchors that minimize the need for re-evaluation of pipe support due to load changes and as-built reconciliation.

3.9.1.2.7.9 CE-901 - ICES STRUDL II

STRUDL is a broad, extensive, and general program for solving problems in structural engineering.

3.9.1.2.7.10 ME153 - MAPPS

MAPPS is a miscellaneous application program for pipe supports which enables the user to access any or all of the following pipe support analysis computer programs within the same run:

- Uniform weld
- Nonuniform welded
- Beta angle
- Clip angle
- Bolt spacing
- Anchor plate
- Local effects
- Clamp (PI clamp)

3.9.1.3 Experimental Stress Analysis

When experimental stress analysis is used in lieu of analytical methods for seismic Category I ASME Code items, the applicable ASME Code requirements for experimental testing of the specific component are applied. If testing is required for seismic Category I non-ASME code items, consideration is given to size effects, dimensional tolerances, and material properties of the tested part to ensure that the test results are a conservative representation of the load-carrying capability of the item installed at LGS.

3.9.1.3.1 Experimental Stress Analysis of NSSS Seismic Category I Items

No experimental stress analysis methods are used.

3.9.1.3.2 Seismic Category I Items Other Than NSSS

No experimental stress analysis methods are used.

3.9.1.4 Considerations for the Evaluation of Faulted Conditions

All seismic Category I equipment is evaluated for the faulted loading conditions. However, emergency stress limits rather than faulted stress limits are used in many cases. The following paragraphs in this section show examples of the treatment of faulted conditions for the major components on a component-by- component basis. Additional discussion of faulted analysis can be found in Sections 3.9.3 and 3.9.5, and Table 3.9-6. These analyses are based on the power rerate analysis, and do not reflect the use of GE13 and GE14 fuel. The impact of GE13 fuel is documented in Reference 3.9-28. The impact of GE14 fuel is documented to be bounded by GE13 fuel in Reference 3.9-34. The impact of the MUR power uprate on GE14 fuel is evaluated in Reference 3.9-31 and Reference 3.9-32. Reference 3.9-33 identifies new design basis values for Fuel Lift Margin and Control Rod Guide Tube Lift Forces under MUR conditions. Additional analyses which consider the use of GNF2 fuel are documented in Reference 3.9-35. The GNF2 fuel is demonstrated to be bounded by the analyses in Reference 3.9-33.

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

Elastic-plastic analysis has not been used in evaluating LGS seismic Category I systems and components for compliance with service Level D Limits. The stress levels of these components are below the ASME allowable stress.

3.9.1.4.1 Control Rod Drive System Components

3.9.1.4.1.1 Control Rod Drives

The ASME Section III Code components of the CRD have been analyzed for faulted conditions shown in Section 3.9.1.1.1. The loading criteria, calculated, and allowable stresses for various operating conditions are summarized in Table 3.9-6(u).

The design adequacy of non-ASME code components of the CRD has been verified by analysis and extensive testing programs on component parts, specially instrumented prototype drives, and production drives. The testing included postulated abnormal events, as well as the service life cycle listed in Section 3.9.1.1.1. The following three types of tests were performed:

a. A test was conducted with the lower CRD flange oscillating with a 2 inch peak-to-peak displacement. No adverse effects were observed during the normal continuous drive-in or jog operation.

b. To simulate the seismic interaction, the core plate and top guide structures of the test vessel were displaced relative to the CRD housing centerline. The results showed no effect in CRD performance.

c. The test vessel fuel channels were deflected to simulate the seismic interactions. The test was performed with fuel channel deflections up to 1.5 inches, which are greater than the expected deflection values. Because the CRD and control rod were not permanently deformed, the drive operability was maintained.

The load criteria, calculated, and allowable stresses for various operating conditions is summarized in Table 3.9-6(v) including the results of the New Loads Adequacy Evaluation program.

3.9.1.4.1.2 Hydraulic Control Unit

The HCU was analyzed for the faulted condition. The seismic and hydrodynamic loads adequacy was demonstrated by test and analysis. The results show peak dynamic loads of 6.2 g (vertical) at the natural frequency of 8-12 Hz and 6 g (horizontal) at a natural frequency of 3-4 Hz. For comparison, the capability of the HCU to withstand loading is 15 g (vertical) at 8-12 Hz, 7 g (horizontal) at 3-4 Hz, and 9 g (horizontal z displacement) at 3-4 Hz.

In addition, a detailed analysis was performed to confirm that the test-mounting configuration is applicable to the LGS unique field installation.

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

3.9.1.4.1.3 CRD Housing

The SSE is classified as a faulted condition; however, in the CRD housing analysis, the SSE event is treated as an emergency condition. The calculated and allowable stresses at various loading conditions are given in Table 3.9-6(v).

3.9.1.4.2 Standard Reactor Internal Components

3.9.1.4.2.1 Control Rod Guide Tube

The maximum calculated stress on the control rod guide tube occurs in the base during a faulted condition. In accordance with ASME Section III, the faulted limit is 2.4 S_m, where S_m = 16,000 psi at 575°F. The analysis and limiting stresses are summarized in Table 3.9-6(aa).

3.9.1.4.2.2 Incore Housing

The maximum calculated stress on the incore housing occurs at the outer surface of the vessel penetration during a faulted condition. The maximum allowable stress for the elastic analysis used is 2.4 S_m (39,948 psi), which bounds the calculated stress as given in Table 3.9-6(ab).

3.9.1.4.2.3 Jet Pump

The elastic analysis for the jet pump faulted conditions shows that the maximum stress is due to diffuser impulse loading during a pipe rupture and blowdown. The maximum allowable for this condition, in accordance with ASME Section III, is 3.6 S_m (60,840 psi). Table 3.9-6(w) summarizes the results of the analysis.

3.9.1.4.2.4 Low Pressure Coolant Injection Coupling

The maximum stress during a faulted condition on the LPCI coupling occurs at the "Bellows" (a purchased component designed to GE requirements for 120 normal operating condition cycles and 10 SSE cycles). Table 3.9-6(y) shows that calculated stresses are within allowable limits.

3.9.1.4.2.5 Orificed Fuel Support

Due to its complex configuration, a series of vertical and horizontal load tests were performed on the orificed fuel support to verify the design. The results show that the seismic and hydrodynamic loading is below the allowable limit with an average safety margin of 1.21 for normal and upset and 1.26 for faulted conditions.

3.9.1.4.3 Reactor Pressure Vessel Assembly

The RPV assembly was evaluated using elastic analysis methods for faulted conditions. Table 3.9-6(f) lists the calculated and allowable stresses for the various loading combinations.

3.9.1.4.4 Core Support Structure

The evaluations for faulted conditions for the core support structure are discussed in Section 3.9.5. The calculated and allowable stresses are summarized in Table 3.9-6(b).

3.9.1.4.5 Main Steam Isolation, Recirculation Gate, and MSRVs

Tables 3.9-6(g), 3.9-6(h), and 3.9-6(j) provide a summary of the analysis for the MSRVs, MSIVs, and recirculation gate valves, respectively.

Standard design rules, as defined in applied codes, are utilized in analyzing pressure boundary components of Class 1 active valves. Conventional, elastic stress analysis is used to evaluate components not defined in the code. The code allowable stresses are applied to determine acceptability of structure under applicable loading conditions, including faulted condition.

3.9.1.4.6 Main Steam and Recirculation Piping

For main steam and recirculation system piping, elastic analysis methods are used to evaluate faulted loading conditions. The allowable stresses using elastic techniques are obtained from the ASME Section III, Appendix F, "Rules for Evaluation of Faulted Conditions" (these are above elastic limits). Additional information for the main steam, recirculation piping, and pipe-mounted equipment is contained in Tables 3.9-6(d) and 3.9-6(e), respectively.

3.9.1.4.7 NSSS Pumps, Heat Exchangers, and Turbines

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

3.9.1.4.8 Control Rod Drive Housing Supports

Design adequacy of the CRD housing supports is shown by comparing the static and dynamic loads to the original design loads. The comparison, summarized in Table 3.9-6(z), shows that the hydrodynamic loads combined with other loads are less than the design load capability of the CRD housing supports.

3.9.1.4.9 Fuel Storage Racks

All analyses related to the Fuel Storage Racks are provided in UFSAR Section 9.1.

3.9.1.4.10 Fuel Assembly (Including Channel)

GE BWR fuel assembly (including channel) design bases, analytical methods, and evaluation results, including those applicable to the faulted conditions are contained in References 3.9-16 and 3.9-17. Evaluations specific to the LGS fuel assemblies have been performed in accordance with the methodology presented in Reference 3.9-17. The resulting acceleration profiles and fuel lift gap are summarized in Table 3.9-6(x).

3.9.1.4.11 Refueling Equipment

Refueling and servicing equipment which is important to safety is classified under essential components, per the requirements of 10CFR50, Appendix A. This equipment and other equipment whose failure would degrade an essential component is defined in Section 9.1 and is classified as seismic Category I. These components are subjected to an elastic dynamic finite-element analysis to generate loadings. This analysis utilizes appropriate seismic floor response spectra, and combines loads at frequencies up to 33 Hz for seismic and up to 100 Hz for hydrodynamic loads, in three directions. Imposed stresses are generated and combined for normal, upset, and faulted conditions. Stresses are compared, depending on the specific safety class of the equipment, to allowables of Industrial Codes, ASME, ANSI or Industrial standards, or AISC. Loading conditions, acceptance criteria, calculated and allowable stresses are shown in Table 3.9-6(s).

3.9.1.4.12 Seismic Category I Items Other than NSSS

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

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

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

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

3.9.2 DYNAMIC TESTING AND ANALYSIS

3.9.2.1a <u>Piping Vibration, Thermal Expansion, and Dynamic Effects Testing for NSSS Piping</u>

The test program is divided into three phases: piping vibration, thermal expansion, and dynamic effects.

3.9.2.1a.1 <u>Piping Vibration</u>

3.9.2.1a.1.1 <u>Preoperational Vibration Testing of Recirculation Piping</u>

The purpose of the preoperational vibration test phase is to verify that operating vibrations in the recirculation piping are within acceptable limits. This phase of the test uses visual observation and manual measurements by hand-held vibrograph to supplement remote measurements. If, during steady-state operation, visual observation indicates that vibration is significant, measurements are made with a hand-held vibrograph. Visual observations, and manual and remote measurements are made during the following steady-state conditions:

a. Recirculation pumps minimum flow

b. Recirculation pumps at 50% of rated flow

c. Recirculation pumps at 75% of rated flow

d. Recirculation pumps at 100% of rated flow

3.9.2.1a.1.2 <u>Preoperational Vibration Testing of Small Attached Piping</u>

During visual observation of each of the above test conditions (a. through d.), special attention is given to small attached piping and instrument connections to ensure that they are not in resonance with the recirculation pump motors or flow-induced vibrations. If the operating vibration acceptance criteria are not met, corrective action, such as modification of supports, is taken.

3.9.2.1a.1.3 <u>Startup Vibration Testing of Main Steam Recirculation and RCIC Piping</u>

The purpose of this phase of the program is to verify that the main steam, recirculation, and RCIC piping are within acceptable limits. The main steam and recirculation piping are instrumented with transducers to measure temperature, thermal movement, and vibration deflections. Because of limited access due to high radiation levels, no visual observation is required during this phase of the test. Remote measurements are made during the following steady-state conditions:

a. Main steam flow at 25% of rated

b. Main steam flow at 50% of rated

c. Main steam flow at 75% of rated

d. Main steam flow at 100% of rated

3.9.2.1a.1.4 Operating Transient Loads on Main Steam and Recirculation Piping

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

a. Recirculation pump starts

b. Recirculation pump trip at 100% of rated flow

c. Turbine stop valve closure at 100% power

d. Manual discharge of each SRV at 1000 psig, and at planned transient tests that result in SRV discharge

3.9.2.1a.2 Thermal Expansion Testing of Main Steam and Recirculation Piping

The thermal expansion preoperational and startup testing program verifies that normal thermal movement occurs in the piping systems, and is performed through the use of potentiometer sensors. The main purpose of this program is to ensure the following:

a. The piping system is free to expand and move without unplanned obstruction or restraint in the x, y, and z directions, during system heatup and cooldown.

b. The piping system does "shake down" after a few thermal expansion cycles.

c. The piping system is working in a manner consistent with the assumption of the NSSS stress analysis

d. There is adequate agreement between calculated and measured displacements.

e. Thermal displacements are consistent and repeatable during heatup and cooldown of the NSSS systems.

Thermal expansion displacement limits are established prior to the start of piping testing. These are compared with the actual measured displacements to determine the acceptability of the actual motion. If the measured displacement does not vary by more than the specified tolerance from the acceptance limit, the piping system is responding in a manner consistent with predictions, and is therefore acceptable. Two levels of displacement limits are established to check the systems, as discussed in Section 3.9.2.1a.4.

3.9.2.1a.3 Dynamic Effects Testing of Main Steam and Recirculation Piping

To verify that snubbers adequately perform their intended function during plant operation, a dynamic testing program is planned, as part of the normal startup operation testing. The main purpose of this program is to ensure the following:

a. The vibration levels from the various dynamic loadings during transient and steady-state conditions are below the predetermined acceptable limits.

b. Due to underestimating the dynamic effects caused by cyclic loading during plant transient operations, long-term fatigue failure does not occur.

This dynamic testing is to account for the acoustic wave due to the SRV lifts (RV1), SRV load resulting from air clearing (RV2), and turbine stop valve closure load. The maximum stresses developed in the piping by the RV1, RV2, and turbine stop valve closure transients analysis are used as a basis for establishing criteria which assures proper functioning of the snubbers. If field measurements exceed criteria limits, this may indicate that the snubbers are not operating properly. If field measurements are within criteria limits, it is assumed that the snubbers are functioning properly. Sample production snubbers of each size (i.e. 10 kips, 20 kips, 50 kips, etc.) will also be qualified and tested for design and faulted condition loadings, prior to shipment to field. Snubbers will be tested to allow free piping movements at low velocity. During plant startup, the snubbers will be checked for improper settings and checked for any evidence of oil leak.

The criteria for vibration displacements is based on the assumed linear relationship between displacements, snubber loads and magnitudes of applied loads, for any function and response of the system. Thus the magnitudes of limits of displacements, snubber loads, and nozzle loads are all proportional. Maximum displacements (Level 1 limits) are established to prevent the maximum stress in the piping systems from exceeding the normal and upset primary stress limits, and/or the maximum snubber load from exceeding the maximum load to which the snubber has been tested.

Based on the above criteria, Level 1 displacement limits are established for all instrumented points in the piping system. These limits will be compared with the field measured piping displacements. Method of acceptance is as explained in Section 3.9.2.1a.4.

3.9.2.1a.4 <u>Test Evaluation and Acceptance Criteria for Main Steam and Recirculation Piping</u>

The piping response to test conditions is considered acceptable if the organization responsible for the stress report reviews the test results, and determines that the tests verify that the piping responded in a manner consistent with the predictions of the stress report, and/or that the tests verify that piping stresses are within code limits (ASME Section III, NB-3600). Acceptable deflection limits are determined after the completion of the piping systems stress analysis and are provided in the startup test specifications.

To ensure test data integrity and test safety, criteria have been established to facilitate assessment of the test while it is in progress. These criteria, designated Level 1 and Level 2, are described in the following paragraphs.

3.9.2.1a.4.1 <u>Level 1 Criteria</u>

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

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

3.9.2.1a.4.2 Level 2 Criteria

If the Level 2 criteria are satisfied for both steady-state and operating transient vibrations, there will be no fatigue damage to the piping system due to steady-state vibration, and all operating transient vibrations are bounded by the values in the stress report.

Exceeding the Level 2 specified pipe motion requires that the responsible piping design engineer be advised. Plant operating and startup testing plans would not necessarily be altered. Investigations of the measurements, criteria, and calculations used to generate the pipe motion limits would be initiated. An acceptable resolution must be reached by all appropriate and involved parties, including the responsible piping design engineer.

Detailed evaluation is needed to develop corrective action or to show that the measurements are acceptable. Depending on the nature of such resolution, the applicable tests may or may not be repeated.

3.9.2.1a.4.3 Acceptance Limits

For steady-state vibration, the piping break stress due to vibration only (neglecting pressure) will not exceed 10,000 psi for Level 1 criteria and 5,000 psi for Level 2 criteria. These limits are below the piping material fatigue endurance limits as defined in Design Fatigue Curves in appendix I of ASME code for 10^6 cycles.

For operating transient vibration, the piping bending stress (zero to peak) due to operating transient only will not exceed 1.2 S_m or pipe support loads will not exceed the Service Level D ratings for Level 1 criteria. The 1.2 S_m limit ensures that the total primary stress including pressure and dead weight will not exceed 1.8 S_m, the new Code Service Level B limit. Level 2 criteria are based on pipe stresses and support loads not to exceed design basis predictions. Design basis criteria require that operating transients stresses and loads not to exceed any of the Service Level B limits including primary stress limits, fatigue usage factor limits, and allowable loads on snubbers.

3.9.2.1a.5 Corrective Actions for Main Steam and Recirculation Piping

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

a. Installation Inspection: A walkdown of the piping and suspension is made to identify any obstruction or improperly operating suspension components. Snubbers are located close to the midpoint of the total travel range at the operating temperature. Hangers are in their operating range between the hot and cold settings. If vibration exceeds the criteria, the source of the excitation must be identified to determine if it is related to equipment failure. Action is taken to correct any discrepancies before repeating the test.

b. Instrumentation Inspection: The instrumentation installation and calibration is checked, and any discrepancies are corrected. Additional instrumentation is added, if necessary.

c. Repeat Test: If actions a. or b. identify discrepancies that could account for failure to meet Level 1 criteria, the test is repeated.

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

If the test measurements indicate failure to meet Level 2 criteria, the following corrective actions are taken after completion of the test:

a. Installation Inspection: A walkdown of the piping and suspension is made to identify any obstruction or improperly operating suspension components. Snubbers are located close to the midpoint of the total travel range at the operating temperature. Hangers are in their operating range between the hot and cold settings. If the vibration exceeds limits, the source of the vibration must be identified. Actions, such as suspension adjustment, are taken to correct any discrepancies.

b. Instrumentation Inspection: The instrumentation installation and calibration are checked, and any discrepancies are corrected.

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

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

3.9.2.1a.6 <u>Measurement Locations for Main Steam and Recirculation Piping</u>

Remote vibration measurements during initial startup will be made for each of the main steam lines and recirculation lines. The locations of the measurements will be described in the startup test specification.

During preoperational testing of recirculation piping, visual observation and manual measurements by hand-held vibrograph are made to supplement the remote measurements.

3.9.2.1b <u>Piping Vibration, Thermal Expansion, and Dynamic Effects Testing of Non-NSSS Piping</u>

The dynamic effects on all safety-related ASME Class 1, 2, and 3 piping systems, including their supports and restraints, are considered as required by NB-3622.3, NC-3622, and ND-3622 of ASME Section III. The structural and functional integrity of each piping system under postulated seismic events is verified by dynamic analysis only. Piping systems having significant anticipated transient loads, caused by main stop valve closure or relief valve discharge for example, are analyzed for time-dependent forces. In addition, piping steady-state vibration and dynamic transient tests will be performed as summarized below, to ensure that:

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

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

The power ascension tests performed for each non-NSSS piping system are provided in Table 3.9-7. The table also describes the systems as to ASME Code class, high energy or moderate energy piping designation, and seismic category.

The startup test program specifications describe in detail the piping that is instrumented for remote monitoring of vibrations and thermal expansion and the piping that is accessible for preoperational or startup walkdown testing by test personnel. The test criteria limit the permissible pipe vibratory stress to the allowable limits prescribed in the industry standard for startup testing of nuclear power systems, ANSI/ASME OM3.

The LGS startup testing program requires that the following conditions be demonstrated in accordance with Regulatory Guide 1.70:

a. Thermal expansion is free from significant and unacceptable restraint not accounted for in the design.

b. Piping vibration is within acceptable limits for long-term vibratory stress.

c. Dynamic transient response of the piping is compared to the design analysis expected values. If those values are exceeded, the test results are compared with the maximum that would be allowed under the ASME Code stress limits.

Cognizant design personnel familiar with the systems to be tested participate in the development of test procedures, and in the evaluation of the test results. The data acquired from the tests are compared with the anticipated results to determine the acceptability of the total system response. Refer to STP-17, STP-33 and STP-36 in Table 14.2-3 for these startup test program procedures.

Test specifications governing the scope of startup testing of BOP piping were prepared and were intended to be the repository for all primary information relating to the scope, objectives, methods, instrumentation, measurements, and criteria for evaluation of the test results. The BOP piping systems are categorized in terms of the following:

a. Test environment (hot deflection, steady-state vibration or dynamic transient response).

b. Test measurements (remotely monitored, visual or none required due to small expected response to test environment).

c. The appropriate testing phase (preoperational or power ascension).

Piping thermal expansion tests are performed for the safety-related piping systems with normal operating temperatures exceeding 300°F. Safety-related piping systems with normal operating temperatures less than 300°F do not have significant thermal expansion to warrant these tests. Engineering review of all seismic Category I piping systems, including their supports, and restraints or snubbers, was performed after completion of construction and prior to fuel load. This ensures that no restraint of normal thermal movement occurs due to interferences and obstructions, and that the support and restraints are in accordance with the design intent. For those systems receiving thermal expansion tests, pipe movements are monitored to ensure that no restraint of normal thermal movement occurs at locations other than at the designed restraint locations.

By monitoring the thermal movement, the thermal expansion test program verifies that the free thermal expansion of piping systems takes place at the snubbers. Performance of the snubbers designed for transient loads, such as those resulting from main stop valve closure or MSRV discharge, are verified by measuring the load in the snubber during the dynamic transient tests. The snubbers are qualified by dynamic testing for cyclic loading as described in Section 3.9.3.5.2.

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

3.9.2.1b.1 Piping Dynamic Transient Tests

During the power ascension, the piping dynamic transient tests identified in Table 3.9-7 are performed. The following modes are considered:

a. Main steam piping outside the containment for main steam turbine stop valve trip at 25%, 75% (Unit 1 only), and 100% power. Main Steam Turbine trip test for Unit 2 at 100% power will be performed during commercial operation of that Unit.

b. Main steam bypass piping for the turbine stop valve closure

c. MSRV discharge piping for the MSRV opening

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

e. Feedwater piping for reactor feed pump trip/coastdown

f. Feedwater heater drain piping for dump and drain valve actuation (Unit 1 only)

g. Moisture separator drain piping for flashing during normal operation and moisture separator depressurization (Unit 1 only).

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

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

The acceptance criterion for this test is that the measured response of the piping system, the snubber, and restraint loads are below the design values. When the measured response exceeds the calculated response, or restraint loads, detailed evaluation of the design will be made to determine the acceptability.3.9.2.1b.2 Piping Steady-State Vibration Testing

The piping systems identified in Table 3.9-7 are tested for steady-state vibration during preoperational test programs or during power ascension. The following operating modes are considered:

a. RHR pump operation

b. HPCI pump and turbine operation

c. RCIC pump and turbine operation

d. Core spray pump operation

e. Main steam

f. Feedwater

g. RWCU

In addition, during the system walkdown, upon initial startup or power escalation, any abnormal vibrations of other systems observed are reviewed and instrumented, if necessary, to determine the acceptability of such vibration.

All safety-related process piping systems and safety-related instrument lines are included in the vibration monitoring program (Table 3.9-7). A vibration monitoring test specification is prepared to categorize the requirements for the testing program. Safety-related systems are categorized as follows:

a. Systems or portions of systems having no flow within a significant portion of their lines; for those system, no testing is required.

b. Systems with flow:

   - Accessible lines (including attached instrument lines) monitored visually or with hand-held instruments.

   - Inaccessible lines other than instrumentation lines will be monitored by remote instrumentation.

   - Inaccessible instrument lines.

   The inside containment instrumentation lines listed below that are inaccessible for visual inspection during power ascension testing will be included in the vibration monitoring program:

   - RPV level indicator instrumentation lines

   - Main steam instrumentation lines used for monitoring main steam flow

   - RCIC steam supply instrument lines used for monitoring steam flow

   - CRD lines

- Main steam sample lines.

The vibration monitoring program for the above lines consists of walkdown by stress analysts experienced in vibration assessment prior to power ascension testing. The review is to include proper restraining of the lines near the vibration source, at elbows and bends, and at the concentrated masses. Prior to walkdown, the stress engineer will review the mode shapes and piping analytical responses to normal and upset conditions. Any lines found to be inadequately restrained, resulting in the potential for excessive vibration, will be evaluated on a case-by-case basis to assess the impact on plant safety that could result from operation of the lines. The evaluation will identify the mode(s) of operation of such lines that cause the excessive vibration. Corrective action to prevent excessive vibration will be taken prior to operating any lines in the operating mode(s) that cause excessive vibration.

Steady-state vibration is primarily induced by the flow in the pipe and the equipment motion. In general, the nature of the steady-state vibration is not known a priori. Therefore, design engineers with stress analysis experience and with familiarity with the subject piping system observe the lines during significant modes of system operation as shown in Items a. through g. above, and classify each line as acceptable, if the vibration is not significant, or as questionable, if the vibration is significant. The lines with questionable steady-state vibration are monitored as applicable by suitable instrumentation to determine the system response.

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

The acceptance criterion for the steady-state vibration tests is that the maximum measured amplitude of the piping vibration does not induce more stress in the pipe than the endurance limit of the material. By limiting the maximum stress in the pipe due to steady-state vibration below the allowable limits prescribed in the industry standard for startup testing of nuclear power systems (ANSI/ASME OM3), the steady-state vibration induced stress does not contribute to reducing piping fatigue life. These limits are based on the piping design fatigue curves of up to 10^6 cycles of vibration given in ASME Section III, Appendix I. To account for fatigue with higher cycles, the design fatigue strength of carbon steels will be reduced by applying a factor of 0.8 and further employing a safety factor of 1.3. Austenitic pipe steels design fatigue strength reduction factor will be 0.6, and is further reduced by employing a safety factor of 1.3. Piping stress indices (K_2C_2) and intensification factors (2i) as applicable to each particular system are also applied in accordance with the standard.

When required, additional restraints are provided to reduce the steady-state vibration, and to keep the stresses below the acceptance criteria levels. Table 3.9-7 provides a reference to the appropriate test descriptions in Chapter 14.

3.9.2.2 Dynamic Qualification of Safety-Related Mechanical Equipment

The qualification discussion in the following sections is generally divided into two types of equipment: NSSS equipment and non-NSSS equipment. The design criteria and qualification procedures for NSSS equipment includes the effects of both seismic and hydrodynamic loads. The non-NSSS equipment qualification discussions include only seismic design criteria and qualification procedures. Refer to Appendix 3A.6.8 and 3A.7.1.7 for discussion of non-NSSS equipment subject to hydrodynamic loads. Appendix 3A.6.7 and 3A.7.1.5 contain further discussion of NSSS equipment qualification.

Safety-related NSSS and Non-NSSS mechanical equipment were reviewed to SRP 3.10 Seismic Qualification Review Team (SQRT) requirements including IEEE 344-1975 and Reg. Guides 1.100 and 1.92. The SQRT re-assessment concluded that the seismic and dynamic qualification program meets the intent of IEEE 344-1975 and Reg. Guides 1.100 and 1.92. Refer to Section 3.10.2.

3.9.2.2a Dynamic Qualification of NSSS Safety-Related Mechanical Equipment

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

3.9.2.2a.1 Tests and Analysis Criteria and Methods

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

Analysis is also used to show there are no natural frequencies below 33 Hz for seismic loads and 100 Hz for hydrodynamic loads. If a natural frequency lower than these is discovered, dynamic tests may be conducted, and in conjunction with mathematical analysis, used to verify operability and structural integrity at the required dynamic input conditions.

When the equipment is qualified by dynamic testing, the response spectrum or time history of the attachment point is used in determining input motion.

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

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

3.9.2.2a.1.1 Random Vibration Input

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

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

b. The anticipated response of the equipment is adequately represented by no resonances, one resonance, or widely spaced resonances.

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

3.9.2.2a.1.2 Application of Input Motion

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

3.9.2.2a.1.3 Fixture Design

The fixture design simulates the actual service mounting, and causes no dynamic coupling to the equipment.

3.9.2.2a.1.4 Prototype Testing

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

3.9.2.2a.2 Dynamic Qualification of Specific NSSS Mechanical Components

The following sections discuss the testing or analytical qualification of NSSS equipment. Seismic and hydrodynamic qualification is also described in Sections 3.9.1.4, 3.9.3.1, and 3.9.3.2 and in Appendix 3A.6.7 and 3A.7.1.6.

3.9.2.2a.2.1 Jet Pumps

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

3.9.2.2a.2.2 CRD and CRD Housing

A dynamic analysis of the CRD housing (with enclosed CRD) was performed. The results of the analyses established the structural integrity of these components. Preliminary dynamic tests verified the operability of the CRD subjected to displacements resulting from a dynamic event. A simulation test, imposing a static bow in the fuel channels, was performed with the CRD functioning satisfactorily.

3.9.2.2a.2.3 Core Support (Fuel Support and Control Rod Guide Tube)

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

3.9.2.2a.2.4 Hydraulic Control Unit

The HCU was evaluated by comparing the RRS with the TRS at mounting points through test and analysis.

3.9.2.2a.2.5 Fuel Assembly Including Channels

GE BWR fuel channel design bases, analytical methods, and evaluation results, including seismic and hydrodynamic considerations, are contained in References 3.9-16 and 3.9-17.

3.9.2.2a.2.6 Recirculation Pump and Motor Assembly

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

Primary stresses due to horizontal and vertical dynamic forces were considered to act simultaneously, and as a conservative measure, they were added directly. Horizontal and vertical seismic forces were applied at mass centers, and equilibrium reactions were determined for motor and pump brackets.

3.9.2.2a.2.7 ECCS Pump and Motor Assembly

The dynamic qualification of each ECCS pump and motor assembly (as a unit), while operating under faulted conditions, was met in the form of a response spectrum analysis. The maximum specified vertical and horizontal accelerations were applied, simultaneously, in the worst case combination. The results of the analysis indicate that the pumps are capable of sustaining the applicable loadings without overstressing the pump components.

A similarly designed motor was dynamically qualified by a combination of static analysis and dynamic testing. The complete motor assembly was seismically qualified by dynamic testing, in accordance with IEEE 344 (1975). The qualification test program included demonstration of startup and shutdown capabilities, as well as no-load operability during dynamic loading conditions. Refer to Section 3.10.

3.9.2.2a.2.8 RCIC Pump Assembly

The RCIC pump assembly was analytically qualified by static analysis for seismic and hydrodynamic loadings as well as the design operating loads of pressure, temperature, and external piping loads. The results of this analysis confirm that the stresses are less than the allowable.

3.9.2.2a.2.9 RCIC Turbine Assembly

The RCIC turbine was qualified for seismic and hydrodynamic loads by a combination of static analysis and dynamic testing.

The turbine assembly and its components were considered to be supported as designed, and horizontal/vertical accelerations were applied to the mass centers of gravity. The static analysis indicated that the turbine assembly is capable of sustaining the design accelerations and loadings without overstressing any components.

The complete turbine assembly was qualified by dynamic testing, in accordance with IEEE 344 (1975). The qualification test program demonstrated startup, steady-state operability, and shutdown capabilities. Refer to Section 3.10.

3.9.2.2a.2.10 SLCS Pump and Motor Assembly

The SLCS positive displacement pump and motor assembly is mounted on a common baseplate and qualified by a combination of static analysis and single-frequency testing.

The SLCS pump and motor assembly was analytically qualified by static analysis for dynamic loading, as well as for the design operating loads of pressure, temperature, and external piping loads. The results of this analysis confirm that the stresses are less than the allowable.

3.9.2.2a.2.11 RHR Heat Exchangers

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

3.9.2.2a.2.12 SLCS Tank

The SLCS storage tank is a cylindrical tank, 9 feet in diameter and 12 feet high, bolted to the concrete floor. The SLCS tank was qualified by analysis for:

a. Stresses in the tank bearing plate

b. Bolt stresses

c. Sloshing loads imposed at the natural frequency of sloshing (0.58 Hz). The natural frequency of the tank is 58.8 Hz.

d. Minimum wall thickness

e. Buckling

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

3.9.2.2a.2.13 Main Steam Isolation Valves

The MSIVs are qualified by dynamic testing and analysis. The dynamic characteristics of the MSIVs were modeled in the main steam piping analysis. The resulting moments and stresses from the piping interaction were below the test proven valve capability. This assured the structural integrity. The operability of the valve is demonstrated by the closure test while it was subjected to the faulted dynamic loads.

3.9.2.2a.2.14 Main Steam Relief Valves

Due to the complexity of this structure and the performance requirements of the valve, the total assembly of the MSRV (including electrical and pneumatic devices) was dynamically tested at accelerations equal to or greater than the combined SSE and hydrodynamic loading of this plant.

3.9.2.2a.2.15 HPCI Turbine

The HPCI turbine was dynamically qualified by a combination of static analysis and dynamic testing.

The turbine assembly and its components were considered to be supported as designed, and horizontal/vertical accelerations applied to the mass centers of gravity. The results of the analysis indicate that the turbine assembly is capable of sustaining the design accelerations and loadings without overstressing any components.

The complete turbine assembly was qualified by dynamic testing, in accordance with IEEE 344 (1975). The qualification test program demonstrated startup, steady-state operability, and shutdown capabilities. Refer to Section 3.10.

3.9.2.2a.2.16 HPCI Pumps

The HPCI booster pump and main pump assembly is a split body type, mounted on one common baseplate. The pump assembly was analytically qualified by three-dimensional dynamic analysis using the response spectrum modal analysis technique. Results are obtained by using acceleration forces acting simultaneously in two directions, one vertical and one horizontal.

3.9.2.2b Dynamic Qualification of Non-NSSS Safety-Related Mechanical Equipment

All non-NSSS seismic Category I equipment is designed to withstand the simultaneous horizontal and vertical accelerations caused by the OBE and the SSE, as defined herein, in conjunction with other applicable loads. Seismic Category I non-NSSS mechanical equipment and supports located in the containment, reactor enclosure, and control structure are also subjected to dynamic loads due to LOCA and MSRV discharge hydrodynamic phenomena. Appendix 3A.5.6, 3A.6.8 and 3A.7.1.7 provide a summary of the load combinations, capability assessment criteria, and methodology used to qualify the equipment to withstand these additional hydrodynamic loads in combination with seismic and all other applicable loads. The functions of instrumentation and controls, or other parts necessary for the functional requirements of the equipment, are not impaired when the equipment is subjected to these loads.

The criteria for the seismic qualifications of non-NSSS mechanical and electrical equipment, with the exception of valves and valve operators (other than relief valves), are contained in a project specification. IEEE 344, "Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations," is used as a supplement to the project seismic specification in some of the individual equipment specifications. Both Standard IEEE 344 and the project specification address the requirements of the demonstration of the seismic adequacy of equipment by analysis and/or tests.

Seismic qualification of non-NSSS motor and air operated valves is addressed in Section 3.9.3.2b.2, and qualification of control valves is addressed in Section 3.10.

3.9.2.2b.1 Dynamic Qualification

Seismic Category I non-NSSS mechanical equipment is qualified by any one of the following methods:

a. Dynamic analysis

b. Testing

c. Combination of analysis and testing

d. Similarity to previously tested equipment.

A list of dynamic qualification package numbers for non-NSSS safety-related mechanical equipment is provided in Table 3.9-9. The qualification packages will be maintained by the licensee in a centrally located, readably auditable permanent file.

3.9.2.2b.2 Criteria

The criteria for dynamic qualification of the equipment are given below (Refer to Appendix 3A.6.8 for the criteria for the equipment subjected to hydrodynamic loads in addition to seismic loads).

3.9.2.2b.2.1 Response Spectrum Curves

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

3.9.2.2b.2.2 Load Combinations and Allowable Stress Limits

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

a. OBE Conditions

This includes gravity loads and operating loads (or DBA loads, if applicable), including associated temperatures and pressures combined by absolute sums, with the dynamic seismic loading of the OBE.

Allowable stresses in the structural steel portions may be increased to 125% of the allowable working stress limits as set forth in ASME Section III, or other applicable industrial codes. The customary increase in normal allowable working stress due to an earthquake is used if, according to the appropriate code, it is less than 25%. Resulting deflections, misalignment or binding of parts, or effects on electrical performance (for example, contact bounce) do not prevent operation of the mechanical equipment during or after the seismic disturbance.

b. SSE Conditions

This includes gravity loads and operating loads (or DBA loads, if applicable), including associated temperatures and pressures combined by absolute sums with the dynamic seismic loading of the SSE. Allowable stresses in the structural portions may be increased to 150% of allowable working stress limits (the appropriate codes listed in a.) but may not exceed 0.9 F_y in bending, 0.85 F_y for axial tension, and 0.5 F_y in shear, where (F_y) equals the material minimum yield stress at the design temperature. For equipment designed by the maximum shear stress theory, the difference between the maximum and minimum principal stresses does not exceed 0.9 F_y. The resulting deflections, misalignment, or binding of parts, or effects on electrical performance (for example, contact bounce) do not prevent operation of the mechanical equipment during or after the seismic disturbance.

3.9.2.2b.2.3 Prevention of Overturning and Sliding

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

3.9.2.2b.2.4 Dynamic Testing

For equipment qualified by testing, seismic adequacy was established by dynamically testing the equipment in accordance with the project specifications. The equipment was tested with its mountings simulating the actual installation conditions.

3.9.2.2b.2.5 Combined Analysis and Test

Some equipment is qualified by a combination of analysis and testing procedures.

An analysis is conducted on the overall assembly to determine its stress level and the transmissibility of motion from the base of the equipment to the critical components. The critical components are removed from the assembly and subjected to a simulation of the environment on a test table.

Testing methods are also used to aid the formulation of the mathematical model for any piece of equipment. Mode shapes and frequencies are determined experimentally and incorporated into a mathematical model of the equipment.

3.9.2.2b.2.6 Criteria for the Diesel Fuel Oil Storage Tanks

These tanks are buried below-grade under a cover of 9 feet of earth.

Tanks and tank supports are designed to withstand Cooper E80 loading, applied above 9 feet of saturated overburden. Tank walls and ends do not deflect more than 3% maximum under the most unfavorable loading conditions.

The diesel fuel oil storage tanks conform to the requirements of the UL 58 Code.

Tanks and their supports are designated seismic Category I, and are designed to resist the increased earth pressure from the OBE and the SSE.

Tanks are designed to withstand external pressure resulting from being buried in the ground when the tanks are empty. Uplift forces on buried tanks are resisted by the weight of the empty tank and the foundation mat, plus 9 feet of overburden.

The tanks were analyzed using a mathematical model considering soil/structure interaction.

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

The major reactor internal components are subjected to extensive testing, coupled with dynamic system analyses, to properly describe the flow-induced vibration phenomena resulting from normal reactor operation, and from anticipated operational transients.

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

a. Dynamic analysis of major components and subassemblies is performed to identify natural vibration modes and frequencies. The analysis models used for seismic Category I structures are similar to those outlined in Section 3.7.2.

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

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

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

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

The dynamic modal analysis also forms the basis for interpreting prototype plant preoperational and initial startup test results (Section 3.9.2.4 below). Modal stresses are calculated, and relationships are obtained between sensor response amplitudes and peak component stresses, for each of the lower normal modes. The allowable amplitude in each mode is that which produces a peak stress amplitude of ±10,000 psi.

3.9.2.4 <u>Confirmatory Flow-Induced Vibration Testing of Reactor Internals</u>

The LGS reactor internals are tested in accordance with the provisions of Regulatory Guide 1.20 (Rev 2) for nonprototype Category I plants. The test procedure requires operating the recirculation system at the rated flow, with internals installed (less fuel), followed by inspection for evidence of vibration, wear, or loose parts. The test duration is sufficient to subject critical components to at least 10^6 cycles of vibration during two-loop and single-loop operation of the recirculation system. At the completion of the flow test, the vessel head and shroud head are removed. The vessel is drained, and the major components are inspected on a selected basis. The inspection covers all components which were examined on the prototype design, including the shroud, shroud head, core support structures, the jet pumps, and the peripheral CRD and incore guide tubes. Access is provided to the reactor lower plenum. The prototype reactor for LGS is the Browns Ferry-3 design docketed on July 31, 1968 (Reference 3.9-18).

Reactor internals for LGS are substantially the same as the internal design configurations which have been tested in prototype BWR/4 plants. Results of the prototype tests are presented in a Licensing Topical Report, Reference 3.9-18. This report also contains additional information on the confirmatory inspection program.

3.9.2.5 <u>Dynamic System Analysis of the Reactor Internals Under Faulted Conditions</u>

In order to assure that no significant dynamic amplification of load occurs as a result of the oscillatory nature of the blowdown forces (Figure 3.9-2), a comparison is made between the periods of the applied forces and the natural periods of the core support structures being acted upon by the applied forces. These periods are determined from a 12 node vertical dynamic model of the RPV and internals. In addition to the masses of the RPV and core support structures, allowance is made for the water inside the RPV.

The time-varying pressures are applied to the dynamic model of the reactor internals described above. Except for the nature and locations of the forcing functions and the dynamic model, the dynamic analysis method is identical to that described for seismic analysis, and is detailed in Section 3.7.2.1. These dynamic components are combined with other dynamic loads (including hydrodynamic and seismic) by the SRSS method. The resultant force is then combined with other steady-state and static loads on an ABS basis to determine the design load. The results of the dynamic analysis of the reactor internals are summarized in Table 3.9-6(b). These results are based on the power rerate analysis, and do not reflect the use of GE13 or GE14 fuel. The impact of GE13 fuel is documented in reference 3.9-28. The impact of GE14 fuel is documented to be bounded by GE13 fuel in Reference 3.9-24. The impact of the MUR power uprate is evaluated in Reference 3.9-31 and Reference 3.9-32. Reference 3.9-33 identifies new design basis values for Fuel Lift Margin and Control Rod Guide Tube Lift Forces under MUR conditions. Additional analyses which considered the use o GNF2 fuel are documented in Reference 3.9-35. The GNF2 fuel is demonstrated to be bonded by the analyses in Reference 3.9-33.

3.9.2.6 Correlations of Reactor Internals Vibration Tests with the Analytical Results

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

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

The vibration test data are supplemented by data from forced oscillation tests of reactor internal components, thereby providing the analysts with additional information concerning the dynamic behavior of the reactor internals.

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

3.9.3.1 Loading Combinations, Design Transients, and Stress Limits

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

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

Table 3.9-6 is the major part of this section; it presents the loading combination analytical methods (by reference or example) and also the calculated stress or other design values for the most critical areas of all ASME Class 1, 2, and 3 components supports and core support structures. These values (which are based on the power rerate analysis and do not reflect the use of GE13 or GE14 fuel) are also compared to applicable code allowables. The impact of GE13 fuel is documented in Reference 3.9-28. The impact of GE14 fuel is documented to be bounded by GE13 fuel in Reference 3.9-34. The impact of the MUR power uprate is evaluated in Reference 3.9-31 and

Reference 3.9-32. Reference 3.9-33 identifies new design basis values for Fuel Lift Margin and Control Rod Guide Tube Lift Forces under MUR conditions. Additional analyses which consider the use of GNF2 fuel are documented in Reference 3.9-35. The GNF2 fuel is demonstrated to be bounded by the analyses in Reference 3.9-33.

3.9.3.1.1 Plant Conditions

All events that the plant might credibly experience during a reactor year are evaluated to establish a design basis for plant equipment. These events are divided into four plant conditions.

The plant conditions described in the following paragraphs are based on event probability (i.e., frequency of occurrence) and correlated design conditions as defined in the ASME Section III.

3.9.3.1.1.1 Normal Condition

Normal conditions are any conditions in the course of system startup; operation in the design power range; normal hot standby (with condenser available); and system shutdown other than upset, emergency, faulted, or testing.

3.9.3.1.1.2 Upset Condition

Upset conditions are any deviations from normal conditions which are anticipated to occur often enough that the design should include a capability to withstand the conditions without operational impairment. The upset conditions include those transients which result from any single operator error or control malfunction, transients caused by a fault in a system component requiring its isolation from the system, and transients due to loss of load or power. Vibratory motions due to an OBE are conservatively treated as upset. Hot standby with the main condenser isolated is an upset condition.

3.9.3.1.1.3 Emergency Condition

Emergency conditions are those deviations from normal conditions which require shutdown to correct the conditions or to repair damage in the RCPB. These conditions have a low probability of occurrence, but are included to provide assurance that no gross loss of structural integrity results as a concomitant effect of any damage developed in the system. Emergency condition events include, but are not limited to, transients caused by one or more of the following: a multiple valve blowdown of the reactor vessel; loss of reactor coolant from a small break or crack which does not depressurize the reactor system, nor result in leakage beyond normal makeup system capacity, but which does require the safety functions of containment isolation and reactor shutdown; improper assembly of the core during refueling; or vibratory motions of an OBE in combination with associated system transients.

3.9.3.1.1.4 Faulted Condition

Faulted conditions are those combinations of conditions associated with extremely unlikely postulated events, with consequences such that the integrity and operability of the system may be so impaired that considerations of public health and safety are involved. Faulted conditions encompass events that are postulated because their consequences include the potential for releasing significant amounts of radioactive material. These postulated events are the most drastic that must be designed against, and thus represent limiting design bases. Faulted condition events include, but are not limited to, one or more of the following: a control rod-drop accident; a fuel handling accident; a main steam line break; a recirculation loop break; the combination of small break/large break accident, dynamic motion associated with an SSE and hydrodynamic, and a LOOP; or the SSE.

3.9.3.1.1.5 Correlation of Plant Conditions with Event Probability

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

PLANT CONDITIONS	EVENT ENCOUNTERED (Probability per reactor year)
Normal (planned)	1.0
Upset (moderate probability)	$1.0>P>10^{-2}$
Emergency (low probability)	$10^{-2}>P>10^{-4}$
Faulted (extremely low probability)	$10^{-4}>P>10^{-6}$

3.9.3.1.1.6 Regulatory Guide 1.48

Regulatory Guide 1.48 was issued after the design of this plant was established, and is therefore not used as a design basis requirement. However, the GE design basis was representative of good industry practices at the time of design, procurement, and manufacture, and is shown to be in general agreement with the guidelines of Regulatory Guide 1.48 through the use of the alternate approaches. For a comparison of NSSS compliance with Regulatory Guide 1.48, refer to Table 3.9-10. This comparison reflects general GE practice on BWR/4s and BWR/5s, and therefore is applicable to LGS.

The design limits and loading combinations for non-NSSS seismic Category I fluid systems components have been evaluated as being in conformance with Regulatory Guide 1.48, although in some cases the guidelines were not specifically a design basis.

Pump and valve operability assurance during and after design loading events, as discussed in footnotes 6 and 11 in section C of the regulatory guide, is discussed in Section 3.9.3.2b.

3.9.3.1.2 Reactor Pressure Vessel Assembly

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

The RPV, RPV support skirt, and shroud support are constructed in accordance with ASME Section III. The shroud support consists of the shroud support plate, and the shroud support cylinder and its legs. The RPV is an ASME Class 1 component. Complete stress reports on these components have been prepared in accordance with ASME requirements. Table 3.9-6(f) provides a summary of stress criteria load combinations, calculated stress, and available stresses. The stress analyses performed for the reactor vessel assembly, including the faulted condition, were completed using elastic methods. The stress load combinations and stress analyses for core support structures and other reactor internals are discussed in Section 3.9.5.

3.9.3.1.3 Main Steam Piping

The main steam piping discussed in this paragraph includes that piping extending from the reactor pressure vessel to the outboard MSIV. This piping is designed in accordance with the ASME Section III, Class 1, Subsection NB-3650. The load combinations and stress criteria for the main steam piping and pipe-mounted equipment are shown in Table 3.9-6(d).

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

3.9.3.1.4 Recirculation Loop Piping

The recirculation system piping which is bounded by the RPV nozzles is designed in accordance with the ASME Section III, Class 1, Subsection NB-3650. The load combinations and allowables for the recirculation piping and pipe-mounted equipment are shown in Table 3.9-6(e). The rules contained in ASME Section III, Appendix F are used in evaluating faulted loading conditions, independently of all other design and operating conditions. Stresses calculated on an elastic basis are evaluated in accordance with F-1360.

3.9.3.1.5 Recirculation System Valves

The recirculation system suction and discharge gate valves are designed in accordance with the ASME Section III, Class 1, Subsection NB-3600. Loading combinations and other stress analysis data are presented in Table 3.9-6(j).

3.9.3.1.6 Recirculation Pump

In the design of the recirculation pumps, the ASME Section VIII, Division 1, 1971 Edition (with latest addenda) is used as a guide in calculations made to determine the thickness of pressure-retaining parts and to size the pressure-retaining bolting. At the time the LGS recirculation pump was procured, ASME Section III design rules for pumps were under development. The New Loads Adequacy Evaluation has demonstrated that the design meets ASME Section III requirements.

The pump vendor's calculations for the design of the pressure- containing components include the determination of minimum wall thickness and allowable stress and pressures. Loading conditions and stress information of those calculations are shown in Table 3.9-6(i).

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

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

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

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

3.9.3.1.7 SLCS Tank

The ASME Code allowable stress limits for the faulted category are 1.2 S_m for general membrane, and 1.8 S_m for bending plus local membrane.

The SLCS tank is designed in accordance with ASME Section III. A summary of the design calculations and stress criteria used is shown in Table 3.9-6(m).

3.9.3.1.8 RHR Heat Exchangers

The RHR heat exchangers are designed in accordance with ASME Section III. The loading combinations and stress analyses for the RHR heat exchangers are given in Table 3.9-6(o).

3.9.3.1.9 RCIC Turbine

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

Operating conditions for the RCIC turbine include:

a. Surveillance Testing - Periodic operation with the reactor pressure at 1000 psia, nominal, and saturated temperature; and turbine exhaust pressure at 25 psia, peak, and saturated temperature.

b. Auto-Startup - 30 cycles per year with the reactor pressure at 1150 psia, nominal, and saturated temperature; and turbine exhaust pressure at 25 psia, peak, and saturated temperature.

Design conditions for the RCIC turbine include:

a. Turbine Inlet - 1250 psig at saturated temperature

b. Turbine Exhaust - 165 psig at saturated temperature

Table 3.9-6(q) contains a summary of the loading conditions, stress criteria, calculated stresses and allowable stresses of the RCIC turbine.

3.9.3.1.10 RCIC Pump

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

The RCIC pump is surveillance tested in conjunction with the RCIC turbine. An operational test is performed, in which the RCIC pump takes condensate from the CST, and discharges back to the CST at the design flow through a closed test loop.

Design conditions for the RCIC pump include:

a. Required NPSH - 20 feet

b. Total head - High speed 2800 feet
Low speed 525 feet

c. Pump capacity - 625 gpm

d. Normal ambient operating temperature - 60°F to 100°F

e. Normal plus upset conditions which control the pump design include:

1. Design pressure 1500 psig

2. Design temperature 40°F - 140°F

3. OBE(1) ½ of SSE

Table 3.9-6(r) contains a summary of the loading conditions, stress criteria, calculated and allowable stresses for the RCIC pump components.

3.9.3.1.11 ECCS Pumps

The RHR and core spray pumps are designed in accordance with ASME Section III. The stress criteria and calculated and allowable stresses are summarized in Table 3.9-6(n).

Design conditions for the RHR and core spray pumps are as follows:

	RHR	Core Spray
a. Design pressure		
1. Suction	220 psig	125 psig
2. Discharge	500 psig	500 psig
b. Design Temperature	40-360°F	40-212°F

3.9.3.1.12 SLCS Pump

The SLCS pumps are designed and fabricated following the requirements for an ASME Section III, Class 2 component.

The SLCS pumps and motors are functionally tested by pumping demineralized water through a closed test loop.

Design conditions for each SLCS pump include:

a. Flow rate 43 gpm

b. Available NPSH 12.9 psi @ 110°F

c. Maximum operating discharge pressure 1220 psig

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(1) Combine with hydrodynamic loads of Table 3.9-6.

d. Ambient conditions:

1. Temperature 65°F - 106°F

2. Relative Humidity 20% - 95%

e. Normal plus upset conditions which control the pump design include:

1. Design pressure 1400 psig

2. Design temperature 150°F

3. OBE ½ of SSE

f. Faulted or emergency conditions include:

1. Design pressure 1400 psig

2. Design temperature 150°F

3. SSE[(1)]
   horizontal 1.5 g
   vertical 0.14 g

A summary of loading combinations, stress criteria, and calculated and allowable stresses for the SLCS pump is contained in Table 3.9-6(I).

3.9.3.1.13 MSIVs and MSRVs

The MSIVs and MSRVs are designed in accordance with ASME Section III, NB-3500 for safety Class 1 components.

Load combination, analytical methods, calculated stresses, and allowable limits are shown for the MSRVs and MSIVs in Tables 3.9-6(g) and 3.9-6(h) respectively.

3.9.3.1.14 HPCI Turbine

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

Operating conditions for the HPCI turbine include:

a. Surveillance Testing - Periodic operation with the reactor pressure at 1000 psia, nominal, and saturated temperature; and the turbine exhaust pressure at 65 psia, peak, and saturated temperature.

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(1) Combine with hydrodynamic loads from Table 3.9-6.

b. Auto-Startup - 30 cycles per year with the reactor pressure at 1150 psia, nominal, and saturated temperature; and turbine exhaust pressure at 65 psia, peak, and saturated temperature.

Design conditions for the HPCI turbine include:

a. Turbine Inlet - 1250 psig at 575°F

b. Turbine Exhaust - 200 psig at 382°F

A summary of loading conditions, stress criteria, and calculated and allowable stresses for the HPCI turbine components is shown in Table 3.9-6(k).

3.9.3.1.15 HPCI Pump

The HPCI pump is designed and fabricated following the requirements for an ASME Section III, Class 2 component.

The HPCI pump is surveillance tested together with the HPCI turbine (Section 3.9.3.1.14). An operational test is performed in which the HPCI pump takes condensate from the CST, and discharges at the design flow rate back to the CST through a closed test loop.

Design conditions for the HPCI pump include:

			REACTOR CONDITION[2,3]		
			1	2	3
a.	Required NPSH (feet)		21	21	21
b.	Total head -	High speed (feet)	3030	850	700
		Low speed (feet)	525	525	525
c.	Constant flow rate (gpm)		5600	5600	3750

d. Normal ambient operating temperature - 60°F to 100°F

e. Normal plus upset conditions which control the pump design include:

1. Design pressure 1500 psig

2. Design temperature 40°F - 140°F

3. OBE[1] ½ of SSE

A summary of calculated stresses, allowable stresses, criteria and loading conditions for critical components is provided in Table 3.9-6(t).

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(1) Combine with hydrodynamic loads from Table 3.9-6.

(2) Reactor Conditions 1 and 2 refer to the range of reactor pressures within which the rated performance of the HPCI pump is guaranteed. This range is bounded by 1156 psia (saturation pressure, Condition 1) and 215 psia. Condition 3 refers to the 100 psig pump interlock point, and is the corresponding pump performance obtained from manufacturer's input.

(3) The HPCI design basis is that it must be capable of injecting design rated flow into the reactor vessel at a maximum reactor pressure equal to the lowest SRV nominal setpoint, plus the allowable setpoint tolerance. References 3.9-26 and 3.9-27 re-evaluated HPCI system operation at the maximum reactor pressure of 1205 psig (lowest SRV setpoint, 1170 psig, plus 3% setpoint tolerance). It was concluded that maintaining the currently rated turbine speed of 4190 rpm at 1205 psig would result in a maximum rated flow rate of 5400 gpm. Reference 3.9-27 performed a conservative analysis that demonstrated a reduced flow rate of 5000 gpm between reactor pressures of 1182 and 1205 psig is adequate for all design basis requirements of the HPCI system.

3.9.3.1.16 RWCU System

The requirements of ASME Section III, Class 3 components are used as guidelines in evaluating the RWCU system pump and heat exchangers. The loading combinations, stress criteria, calculated and allowable stresses are summarized in Tables 3.9-6(p) and 3.9-6(c), respectively.

3.9.3.1.17 Non-NSSS ASME Code Constructed Items

The design loading combinations, categorized by plant operating conditions, identified as normal, upset, emergency, or faulted for Non-NSSS ASME code constructed items, are presented in Table 3.9-11.

The design criteria and stress limits associated with each of the plant operating conditions for each type of ASME code constructed items are presented in Tables 3.9-12 through 3.9-18.

The component operating condition is the same as the plant operating condition, except that the emergency or faulted plant condition is considered normal for a pump or valve whose function must be ensured during an emergency or faulted plant condition.

3.9.3.2a NSSS Pump And Valve Operability Assurance

The NSSS active pumps and valves are listed in Table 3.9-19.

Active mechanical equipment classified as seismic Category I is designed to function during the life of the plant, under postulated plant conditions. Equipment with faulted condition functional requirements includes "active" pumps and valves in fluid systems such as the ECCS. ("active" equipment performs a mechanical motion to accomplish a safety function.)

Operability is ensured by satisfying the requirements of the following programs. Safety-related valves are qualified by prototype testing and/or analysis; and safety-related active pumps are qualified by analysis and/or testing with suitable stress limits and nozzle loads. The content of these programs is detailed below.

3.9.3.2a.1 ECCS Pumps

All active pumps are qualified for operability by first being subjected to rigid tests, before and after installation in the plant. The in-shop tests include hydrostatic tests of pressure-retaining parts at 125% of the design pressure; seal leakage tests; and performance tests, while the pump is operated with flow, to determine total developed head, minimum and maximum head, and NPSH requirements. Bearing temperatures (except water-cooled bearings) and vibration levels are also monitored during these operating tests. Both are below specified limits. After the pump is installed in the plant, it undergoes cold hydrotests, functional tests, and the required periodic inservice inspection and operation. These tests demonstrate reliability of the pump for the design life of the plant.

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

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

In order to avoid damage during the faulted plant condition, the stresses caused by the combination of normal operating loads, SSE, and dynamic system loads are limited to the material elastic limit, as indicated in Section 3.9.3.1 and Table 3.9-6. A three-dimensional finite-element model of the pump/motor and supports was developed using the response spectrum method of dynamic analysis. The average membrane stress (δ_m) for the faulted condition loads is maintained at 1.2 S_m, or approximately 0.75 δ_y (δ_y = yield stress); the maximum stress in local fibers (δ_m plus bending stress (δ_b)) is limited to 1.8 S_m, or approximately 1.1 δ_y.

The maximum dynamic nozzle loads were also considered in an analysis of the pump supports to ensure that a system misalignment will not occur.

A dynamic analysis was performed to determine the magnitude of the seismic loading from the applicable floor response spectra. An analysis was made to ensure that faulted condition nozzle loads and dynamic accelerations would not impair the operability of the pumps during or following the faulted conditions. Pump components having a natural frequency above 33 Hz are considered rigid. This frequency is considered sufficiently high to avoid problems with amplification between the component and structure for all seismic areas.

If the natural frequency is below 33 Hz, an analysis is performed to determine the amplified input accelerations necessary to perform the static analysis. The adjusted accelerations are determined using conservatisms contained in the horizontal and the vertical accelerations used for rigid structures. The static analysis is performed using the adjusted accelerations and must meet the same stress limit criterion stated in Table 3.9-6.

These analyses, with the conservative loads stated and with the restrictive stress limits of Table 3.9-6 as allowables, assure that critical parts of the pump are not damaged during the faulted condition, and that, therefore, the reliability of the pump for postfaulted condition operation will not be impaired by a hydrodynamic or seismic event.

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

Active pump/motor rotor combinations are designed to rotate at a constant speed under all conditions. Motors are designed to withstand short periods of severe overload. The high rotary inertia in the operating pump rotor and the nature of the random shot-duration loading characteristics of the seismic and hydrodynamic events prevents the rotor from becoming seized. The dynamic loadings cause a slight increase, if any, in the torque (i.e., motor current) necessary to drive the pump at constant design speed. Therefore the pump does not shut down during the faulted condition but operates at the design speed despite the faulted loads.

The functional ability of the active pumps after a faulted condition is assured, since only normal operating loads and steady-state nozzle loads exist at this time. For the active pumps, the faulted condition is greater than the normal condition due to seismic and hydrodynamic loads on the equipment and the increase in nozzle loads from the SSE on the connecting pipe. The SSE event is infrequent and of short duration compared to the design life of the equipment. Because it is demonstrated that the pumps would not be damaged during the faulted condition, the postfaulted condition operating loads are less than the normal plant operating limits. This is assured by requiring that the imposed nozzle loads (steady-state loads) for normal conditions and postfaulted conditions are limited by the normal condition nozzle loads. The postfaulted condition ability of the pumps to function under these applied loads is proven during the normal operating plant conditions.

3.9.3.2a.2 SLCS Pump and Motor Assemblies and RCIC Pump Assembly

These equipment assemblies are small, compact, rigid assemblies, with natural frequencies greater than 100 Hz. With this fact verified, each equipment assembly is dynamically qualified by static analysis only. This static qualification verifies operability under seismic and hydrodynamic conditions, and ensures that structural loading stresses are within code limitations.

3.9.3.2a.3 RCIC Turbine

Refer to Section 3.9.2.2a.2.9 and Table 3.9-6(q).

3.9.3.2a.4 ECCS Motors

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

The type test was performed on a 1250 hp vertical motor, in accordance with IEEE 323 (1971). First normal operation during the design life was simulated; then the motor was subjected to a number of seismic events; and then subjected to the abnormal environmental conditions possible during and after a LOCA. The type test plan was as follows:

a. Thermal aging of the motor electrical insulation system (which is a part of the stator only) was based on extrapolation, in accordance with the temperature life of the characteristic curve from IEEE 275 (1966) for the insulation type used on the ECCS motors. The amount of aging equals the total estimated operation days of maximum insulation surface temperature.

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

c. The normal induced current vibration effect on the insulation system was simulated by a 1.5 g horizontal vibration acceleration for one hour at current frequency.

d. Motor bearings were selected and their operating life established on the basis of bearing manufacturer's tests and operating data using the loads calculated to act on the bearings.

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

f. Dynamic loading aging and testing were performed on a biaxial test table in accordance with IEEE 344 (1971). During this type test, the shake table was activated, simulating the maximum design limit of the SSE and hydrodynamic loads with motor starts and operation combination, as may possibly occur during the life of the plant. Refer to Section 3.10.

g. An environmental test simulating a LOCA condition was performed for 100 days with the test motor fully loaded, simulating pump operation. The test consists of startup and six hours of operation at 212°F ambient temperature, and a 100% steam environment. After 1 hour standstill in the same environment, another startup and operation of the test motor was followed by sufficient operation at high humidity and temperature. This was based on extrapolation in accordance with the temperature life characteristic curve from IEEE 275 (1966) for the insulation type used on the ECCS motors.

3.9.3.2a.5 <u>HPCI Pump</u>

Operability of the HPCI pump assembly is demonstrated by a combination of analytical stress calculations, pump manufacturer's operating experience, and testing. The stress definitions and the allowable stress criteria are based on the ASME Section III. The code is directly applicable to the stamped pressure boundary components of the pump.

The witnessed hydrostatic and performance tests, as performed at the pump manufacturer's plant, demonstrate that the pump as designed meets the ASME Code requirements and the parameters of the design specification.

A dynamic analysis was performed by the pump manufacturer. A static analysis was performed for dead weight and nozzle loads. Dynamic analysis was used for the SSE loading condition.

Rotating parts are verified by analysis to ensure no contact with the stationary parts, except at engineered wear points, therefore, continued operation during and after an SSE is assured because the calculated stresses and deformations are less than the prescribed limits.

3.9.3.2a.6 NSSS Valves

3.9.3.2a.6.1 Class 1 Active Valves

The Class 1 active valves are the MSIVs and the MSRVs and SLCS valves. Each of these valves is designed to perform its mechanical motion in conjunction with a design basis accident. Dynamic qualification for operability is unique for each valve type; therefore, each method of qualification is detailed individually below.

3.9.3.2a.6.1.1 Main Steam Isolation Valve

The MSIVs were evaluated for operability during a dynamic loading event by analysis and test as follows:

a. The valve body was designed in accordance with ASME Section III, Class 1, which limits deformations in the operating area of the valve body to those within the elastic limit of the material by limiting pressure and pipe reaction input loads (including seismic and hydrodynamic), thereby ensuring no interference with valve operability.

b. The complete valve topworks, including the bonnet and a simulated stem of a similar design MSIV, was dynamically tested for upset loads and faulted load. The test sample was subjected to 30 seconds of bi-directional random frequency input for each test event. There were a total of 10 upset test events and 2 faulted test events. Valve closure was simulated half way through the testing duration during each event. After the complete test program, there was no significant change to the valve closure rate.

To ensure that design limits were not exceeded for both piping input loads and actuator dynamic loads, the MSIV was mathematically modeled in the main steam line system analysis. The valve dynamic characteristics were modeled in the overall steam line analysis. Pipe anchors and restraints are applied as required to limit pipe system resonance frequencies and amplified accelerations to within acceptable limits for the MSIVs. Details concerning the analysis of these valves is given in Table 3.9-6(h).

The MSIV operability during LOCA conditions is demonstrated as defined in report APED-5750 (March 1969). The test specimen is a 20 inch valve representative in design to the MSIVs. Operability during seismic acceleration is addressed in Section 3.9.2.2a.

3.9.3.2a.6.1.2 Main Steam Relief Valves

The MSRVs were evaluated for operability during seismic and hydrodynamic events. Structural integrity of the configuration during a seismic event was demonstrated by both code analysis and dynamic testing.

a. Each valve was designed for maximum moments which may be imposed on it when installed in service. These moments result from the dead weight, hydrodynamic, and seismic loading of both the valve and the connecting pipe, thermal expansion of the connecting pipe, and reaction forces from valve discharge.

b. Dynamic tests were performed on the LGS MSRV configuration. The tests validated the design static analysis and determined that the equipment remains functional during application of the specified "g" loads.

A mathematical model of this valve was included in the overall main steam line system analysis, along with the MSIVs, to ensure that design limits were not exceeded.

The MSRV analytical qualification results are shown in Table 3.9-6(g).

3.9.3.2a.6.1.3 Explosive Valves

The SLCS explosive valves are qualified to IEEE 344 (1975). The generic qualification test demonstrated the absence of natural frequencies below 35 Hz and the ability to remain operable after the application of horizontal dynamic loading equivalent to 6.5 g and a vertical dynamic loading equivalent to 4.5 g at 33 Hz. In addition, analysis shows that there are no natural frequencies below 100 Hz.

3.9.3.2a.6.2 Class 2 and 3 Active Valves

3.9.3.2a.6.2.1 Gate/Globe Valves

Class 2 active gate/globe valves provided by GE include five RHR valves, one HPCI valve, one RCIC valve, and three core spray valves; all valves are motor-operated. There are no Class 3 active valves in the GE scope of supply. The gate/globe valves are generically qualified by testing valves that are typical of the valves supplied by GE. Operability is ensured by testing under the maximum capability static load which envelopes the static design basis load. These tests ensure operability during and after the design basis load. The actuators are qualified to IEEE 382 (1972), up to levels that exceed the design loadings.

3.9.3.2a.6.2.2 Check Valves

GE provides one swing-check valve and one stop-check valve which are Class 2 active in each of the HPCI and RCIC systems. In addition, GE provides six Class 2 check valves for the RHR system and two Class 2 check valves for the core spray system. Operability of check valves is assured by performing design calculations and by providing sufficient structural margins so that movement of the disc/hinge pin is not impaired under any loading conditions. There are no Class 3 active valves in the GE scope of supply.

3.9.3.2b Non-NSSS Pump and Valve Operability Assurance

3.9.3.2b.1 Pumps

The following pumps are active pumps:

a. Diesel oil transfer pumps

b. RHRSW pumps

c. ESW pumps

d. Control room chilled water pumps

e. Safeguard piping fill pumps.

The safeguard piping fill pumps are Class 2. All the remaining pumps are Class 3.

Safety-related active pumps are subjected to stringent tests both prior to and after installation in the plant. The in-shop tests include: hydrostatic tests of pressure-retaining parts to 150% of the design pressure; seal leakage tests at the same pressure used in the hydrostatic tests; and performance tests which are conducted while the pump is operated with flow to determine total developed head, minimum and maximum head, NPSH requirements, and other pump/motor properties. Bearing temperatures and vibration levels are also monitored during these operating tests. Both are shown to be within the limits specified by the manufacturer. After the pump is installed at the plant, it undergoes startup tests and required inservice inspection and operation.

In addition to these tests, the active pumps are qualified for operation during and after a faulted condition. That is, safety-related active pumps are qualified for operability during an SSE condition by assuring that the pump will not be damaged during the seismic event, and that the pump will continue operating despite the SSE loads. (Refer to 3A.7.1.7 for pumps subjected to hydrodynamic loads.)

In order to meet the first criterion, that the pump will not be damaged during the seismic event, the pump manufacturer is required to determine by test or dynamic analysis whether the lowest natural frequency of the pump is greater than 33 Hz. The pump, when having a natural frequency above 33 Hz, is considered essentially rigid. This frequency is considered sufficiently high to avoid problems with amplification between the component and structure for all seismic areas. A static shaft deflection analysis of the rotor is performed with the conservative SSE accelerations of 1.5 times the applicable floor acceleration. The deflections determined from the static shaft analysis are compared to the allowable rotor clearances.

If the lowest natural frequency is found to be below 33 hertz, the equipment is considered flexible. If flexible, the equipment is analyzed using the response spectrum modal analysis technique. The frequencies and mode shapes are determined in the vertical and horizontal directions. The loads due to the excitation of each mode and the loads due to the accelerations in the three orthogonal directions are added using the square root of the sum of the squares method. Coupling effects are included in the mathematical model.

In order to avoid damage to the pumps during the faulted plant condition, the stresses caused by the combination of normal operating loads, SSE, and dynamic system loads are limited. The maximum seismic nozzle loads are also considered in an analysis of the pump supports to assure that a system misalignment cannot occur. Performance of these analyses, based upon conservative loads and restrictive stress limits, assures that critical parts of the pump will not be damaged during the faulted condition and, therefore, that the reliability of the pump for postfaulted condition operation will not be impaired by the seismic events.

The second criterion necessary to assure operability is that the pump will function throughout the SSE. The pump/motor combination is designed to rotate at a constant speed under all conditions unless the rotor becomes completely seized, i.e., with no rotation. Typically, the rotor can be seized five full seconds before a circuit breaker, used to prevent damage to the motor, opens to stop the pump. However, the high rotary inertia in the operating pump rotor and the nature of the random, short duration loading characteristics of the seismic event prevent the rotor from becoming seized. Actually, the seismic loadings cause only a slight increase, if any, in the torque, i.e., motor current, necessary to drive the pump at the constant design speed. Therefore, the pump will not shut down during the SSE and will operate at the design speed despite the SSE loads.

To complete the seismic qualification procedures, the pump motor and all appurtenances vital to the operation of the pump are independently qualified for operation during the maximum seismic event as discussed in Section 3.10.

From this regimen, it is concluded that the safety-related pump/motor assemblies will not be damaged and will continue to operate under SSE loadings and, therefore, will perform their intended functions. These proposed requirements take into account the complex characteristics of the pump and are sufficient to demonstrate and assure the seismic operability of the active pumps.

The functional ability of active pumps after a faulted condition is assured since only operating loads and steady-state nozzle loads exist. Since it is demonstrated that the pumps would not be damaged during the faulted condition, the postfaulted operating loads will be limited to the normal plant operating loads. This is assured by requiring that the imposed nozzle loads (steady-state loads) for normal conditions and postfaulted conditions are limited by the magnitudes of the normal condition nozzle loads. The postfaulted ability of the pumps to function under these applied loads is proved during the normal operating plant conditions for active pumps.

3.9.3.2b.2 Valves

The active valves are tabulated in Reference 3.9-25. Those active valves which are supplied by the NSSS vendor are identified. See Section 3.9.3.2a for a discussion of operability assurance of these valves.

Safety-related active valves are subjected to a series of stringent tests prior to service, and during the plant life. Before installation, the following tests are performed: the shell hydrostatic test, in accordance with ASME Section III requirements; back-seat and main-seat leakage tests; the disc hydrostatic test; functional tests which verify that the valve opens and closes within the specified time limits; and the operability qualification of motor operators for the environmental conditions over the installed life (i.e., aging, radiation, accident environment simulation, etc.), in accordance with IEEE 382 (1972). After installation, cold hydrostatic tests, functional tests (in accordance with the requirements of Chapter 14), and periodic inservice operation (in accordance with the requirements of Chapter 16) are performed to verify and ensure the functional ability of the valve.

The valves are designed using either stress analyses, or the pressure-containing minimum wall thickness requirements. For all active valves with extended topworks, an analysis is also performed for static-equivalent SSE loads applied at the extended structure's center of gravity. The maximum stress limits allowed in the analyses demonstrate structural integrity, and are equal to the limits recommended by the ASME for the particular ASME class of valve analyzed. Limits for each of the loading combinations are presented in Tables 3.9-13 and 3.9-18.

In addition to these tests and analyses, a representative valve of each design type is tested to verify operability during a simulated seismic event by demonstrating operational capabilities within the specified limits. The qualification testing procedures are described below.

The valve is mounted in a manner that conservatively represents typical valve installations. The valve unit includes the actuator and all appurtenances normally attached to the valve in service. The operability of the valve during an SSE is demonstrated by satisfying the following criteria. (Refer to Appendix 3A.7.1.7 for valves subjected to hydrodynamic loads.)

a. All the active valves with topworks are basically designed to have a first natural frequency greater than 33 Hz. This may be shown by suitable testing or analysis. Regardless of value, the first natural frequency of the topworks is modeled into the piping analysis for determination of maximum accelerations.

b. While in the shop and installed in a suitable test rig, the extended topworks of the valve are subjected to a statically applied equivalent seismic load. The load (the specified g-force times the weight of the topworks) is applied at the center of gravity of the topworks, in the direction of the weakest axis of the yoke. The design pressure of the valve is simultaneously applied to the valve during the static load tests.

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

d. Motor operators are qualified as operable during and after the SSE prior to their installation on the valve, as discussed in Section 3.10.2.2.

The equivalent seismic static load, which is used for the static valve qualification, is the maximum load which the valve is designed to withstand. The piping designer must maintain the motor operator accelerations to these levels.

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

The above factory testing program applies only to valves with overhanging structures, e.g., the motor operator or air actuator assembly. The testing is conducted on a representative number of valves. According to the size and pressure rating, valves from each of the primary safety-related design types, e.g., motor- operated gate valves, are tested. Valves that cover the range of sizes in service are qualified by tests, and the results are used to qualify all valves within the intermediate range of sizes, as shown in Table 3.9-20. Stress analyses are used to support the interpolation. Because of their simple characteristics, check and other compact valves are not adversely affected by seismic acceleration. Check valves have no extended structures to distort the valves and cause malfunctions. Check valve discs are designed to allow sufficient clearance around the disc to prevent binding or interference due to distortions from nozzle or other imposed loads. They are qualified by a combination of the following factory tests and analysis:

a. Stress analysis of critical areas and parts for SSE loads in accordance with the allowable specified in Tables 3.9-13 and 3.9-18

b. In-shop hydrostatic test

c. In-shop seat leakage test

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

A study was also performed that considered a feedwater pipe rupture outside containment to assure that the feedwater isolation check valves can perform their function following a postulated pipe break of the feedwater line outside containment. Analysis of a full circumferential break showed that the check valve inside containment (one of 3 feedwater PCIV's that are check valves) would close rapidly because of flow reversal in the pipe between the reactor and break location just outside containment. This results in a pressure surge of 2780 psia between the valve and reactor. It was determined that the piping and valves could adequately sustain this pressure.

To determine the capability of the check valve seat to withstand the initial impact caused by rapid closure and to sustain the pressure surge that follows, an estimate of check valve seat energy absorption capability was made for a postulated feedwater pipe break accident occurring upstream of the containment isolation valves. The basis for the seat stress calculation was for a valve disk closing velocity of 100 rad/sec and a pressure break in the pipe of 2780 psia.

The valve seat was assumed to consist of an assembly of six discrete, bilinear, elastic-plastic elements. The analysis assumed that the disc kinetic energy at impact equals work done in terms of seat under load, or area under the seat load-displacement curve. Valve seat yield strength was based on its being stressed to 50% of yield at a design pressure of 2132 psi. The load-displacement curve was constructed using Roark's stiffness equations for an annular plate loaded at the inner radius and fixed at the outer radius. Failure was assumed to occur at a ductility ratio of 30.

The analysis indicated that all of the seat elements reach yield, but none reach ultimate strain or fail. Effects not included that are believed to make the analysis conservative are:

a. No credit was taken for disc deformation and energy absorption

b. No credit was taken for hinge deformation and energy absorption

c. No credit was taken for valve body deformation and energy absorption

d. Hinge friction was omitted

e. No strain hardening or rate of strain effects in the seat were included.

The water hammer effects on the closed seat were determined. It was shown that the natural frequency of the combined valve seat stiffness and disc mass was much larger than the frequency of the pressure pulse, so that the effective pressure that the seat must withstand is only the peak pressure in the water hammer surge. The seat is able to withstand this maximum pressure.

Results of this analysis show that, although the conditions of a hypothetical pipe rupture on the feedwater check valves are severe, the valves should remain together at impact and are capable of withstanding 2800 psi. The valve seat should yield at disc impact but not fail, and the water hammer pressure pulse following closure will not cause failure.

An additional confirmatory analysis was also conducted which used a simplified model of single-phase, liquid flow from the reactor vessel to the pipe break, with the check valve disk being closed by drag forces.

Results of this simplified analysis indicate that the check valve disk closes in approximately 70 msec, with a closing velocity at impact of approximately 65 rad/sec. The peak pressure at the closed disk is estimated to be 2157 psi for a finite valve closure time of 70 msec. The precise pressure at valve closure cannot be predicted rigorously by the simplified method used in this study.

The results of this study confirmed the validity of the original design analysis and are consistent with the results of the analysis done for SSES and in particular for the Atwood Morrill check valves.

Operability testing is also not performed for relief valves. Due to the particular, simple characteristics of these SRVs, they are qualified by a combination of the following tests and analyses:

a. Stress analysis, including seismic loads where applicable

b. In-shop hydrostatic test

c. In-shop seat leakage test

d. Performance tests

e. Periodic in situ valve inspection as applicable and periodic valve removal, refurbishment, performance testing and reinstallation

The above testing and analysis is sufficient to ensure the functional capability of the valve.

During a seismic event, it is anticipated that the seismic acceleration imposed upon the valve may cause it to open momentarily and discharge under system conditions that otherwise would not result in valve opening, but this is considered to be of no real safety or other consequence.

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

3.9.3.3 Design and Installation of Pressure Relief Devices

3.9.3.3.1 Main Steam Relief Valves (NSSS)

Lifting of an MSRV results in a transient that produces momentary unbalanced forces acting on the discharge piping system from the time the MSRV opens, until a steady discharge flow from the RPV to the suppression pool is established. This period includes clearing the water slug from the end of the discharge piping submerged in the suppression pool. Pressure waves traveling through the discharge piping following the relatively rapid opening of the MSRV cause the MSRV discharge piping to vibrate. This in turn produces forces that act on the main steam piping.

The analysis of the relief valve discharge transient consists of a step-wise time history solution of the fluid flow equation, to generate a time history of the fluid properties at numerous locations along the pipe. The fluid transient properties are calculated based on the maximum set pressure specified in the steam system specification, and the value of ASME flow rating increased by a factor to account for the conservative method of establishing the rating. Simultaneous discharge of all valves is assumed in the analysis, because simultaneous discharge is considered to induce maximum stress in the piping. Reaction loads on the pipe are determined at each location corresponding to the position of an elbow. These loads are composed of pressure times area, momentum change, and fluid friction terms. Figure 3.9-3 shows a set of fluid property and pipe section load transients typical of those produced by relief valve discharge.

The method of analysis applied to determine piping system response to MSRV operation is time history integration. The forces are applied at locations on the piping system where the fluid flow changes direction, thus causing momentary reactions. The resulting loads on the MSRV, the main steam line, and the discharge piping are combined with loads due to other effects, as specified in Section 3.9.3.1. The code stress limits corresponding to load combination classifications of normal, upset, emergency, and faulted, are applied to the main steam lines and MSRV discharge piping.

In addition, a series of water discharge tests of SRVs used in BWRs was conducted to demonstrate the operational adequacy of the SRV and SRV discharge piping integrity for expected operating conditions for transients and accidents. The tests were performed to satisfy requirements of NUREG-0737, Item II.D.1.

The tests were run at the Wyle Laboratories test facility in Huntsville, Alabama. The facility included a steam and water supply system, the test SRV mounted on a representative steam line, and a representative SRV discharge line routed to a pool of water. Opening and closing of the SRVs were monitored. Fluid conditions and flows were measured, as were strains, accelerations, temperatures, and pressures in the SRV and associated piping.

The water discharge test conditions simulated the alternate shutdown cooling condition, which is an operating condition which is considered in the design evaluation of many BWR plants. The results show that all of the tested SRVs opened and closed on command for all water tests. The measured SRV discharge line loads for water discharge were significantly less than those for the high pressure steam discharge condition for which the piping is designed. LGS specific analyses have been performed (Reference 3.9-22) to establish RPV temperature and pressure conditions where initiation of alternate shutdown cooling will result in acceptable SRV discharge line loads. Alternate shutdown cooling is manually initiated only.

The tests and analyses as described in NEDE-24988-P (Reference 3.9-15) verify the adequacy of SRV operation and the integrity of the SRV discharge piping under expected liquid discharge conditions, and satisfy requirements of NUREG-0737, Item II.D.1.

A basis for concluding that the test results presented in NEDE-24988-P on SRV testing are applicable to LGS is described in the following paragraphs:

a. The SRV discharge piping configuration at LGS uses a T-quencher at the discharge pipe exit. The average length of the 14 SRV discharge line is 132 feet of 12 inch diameter pipe, and the submergence length in the suppression pool is approximately 18'-6". The SRV test program used a ramshead at the discharge pipe exit, a pipe length of 112 feet, a diameter of 10 inches, and a submergence length of approximately 13 feet. Loads on valve internals in the LGS configuration are within acceptable limits for the following reasons:

1. No dynamic mechanical load originating at the T-quencher is transmitted to the valve in the LGS configuration because there is at least one anchor point between the valve and the T-quencher.

2. The length of the first segment of piping downstream of the SRV in the test facility was selected to result in the test program having a bounding dynamic mechanical load on the valve. The LGS SRV discharge line piping configuration differs from the test facility in that the first line segment does not terminate in a 90° elbow, and the pipe size increases in the first segment. An assessment of the LGS configuration has confirmed that the mechanical loads imposed on the LGS valves by the low pressure water flow are enveloped by the high pressure steam loads.

3. Dynamic hydraulic loads (back pressure) are experienced by the valve internals in the LGS configuration. The back pressure loads may be either transient back pressures occurring during valve actuation or steady-state back pressures occurring during steady-state flow following valve actuation.

   The key parameters affecting the transient back pressures are the fluid pressure upstream of the valve, the valve opening time, the fluid inertia in the submerged SRV discharge line 0, and the SRV discharge line air volume. Transient back pressures increase with higher upstream pressure, shorter valve opening times, greater line submergence, and smaller SRV discharge line air volume. An evaluation of these differences has confirmed that the test facility and the LGS configuration have comparable back pressures. The maximum transient back pressure occurs with high pressure steam flow conditions. The transient back pressure for the alternate shutdown cooling mode of operation is always much less than the design for steam flow conditions because of the lower upstream pressure and the longer valve opening time.

   The steady-state back pressure in the test program was maximized by using an orifice plate in the SRV discharge line above the water level and before the ramshead. The orifice was sized to produce a back pressure greater than that calculated for any of the LGS SRV discharge lines.

An additional consideration in the selection of the ramshead for the test facility was to allow more direct measurement of the thrust load in the final pipe segment. The use of a T-quencher in the test program would have required quencher supports that would unnecessarily obscure accurate measurement of the pipe thrust loads.

The differences in the line configuration between the LGS plant and the test program as discussed above result in loads on the LGS valve internals that are within acceptable limits.

b. The LGS SRV discharge lines are supported by a combination of snubbers, rigid supports, and spring hangers. These supports were designed to accommodate combinations of loads resulting from piping, dead weight, thermal conditions, seismic and suppression pool hydrodynamic events, and a high pressure steam discharge transient. Each SRV discharge line at LGS has 2 to 5 spring hangers, all of which are located in the drywell. The test facility configuration utilized no spring hangers.

The dynamic load effects on the piping and supports of the test facility due to the water discharge events (the alternate shutdown cooling mode) were found to be significantly lower than corresponding loads resulting from the high pressure steam discharge event. As stated in Reference 3.9-15, this finding is considered generic to all BWRs because the test facility was designed to be prototypical of the features pertinent to this issue. Furthermore, assessment of a typical LGS SRV discharge line configuration has confirmed the applicability of the generic statement to LGS. LGS specific analysis for acceptable alternate shutdown cooling initiating conditions has been performed (Reference 3.9-22).

During the water discharge transient, there will be significantly lower dynamic loads acting on the snubbers and rigid supports than during the steam discharge transient. This will more than offset the small increase in the dead load on these supports due to the weight of the water during the alternate shutdown cooling mode of operation. Therefore, design adequacy of the snubbers and rigid supports is assured because they are designed for the larger steam discharge transient loads.

The design adequacy of the spring hangers with respect to the increased dead load due to the weight of the water during the liquid discharge transient has been addressed. As was discussed with respect to snubbers and rigid supports, the dynamic loads resulting from liquid discharge during the alternate shutdown cooling mode of operation are significantly lower than those from the high pressure steam discharge. The spring hangers have been reviewed for the deflections resulting from the steam discharge dynamic event and were found to be acceptable. In addition, the spring hangers have been evaluated for the increased dead load due to a water-filled condition. Both the spring hangers and piping stresses were acceptable. Furthermore, the effects of the water dead weight load does not affect the ability of SRVs to open to establish the alternate shutdown cooling path because the loads occur in the SRV discharge line only after valve opening.

c. The purpose of the SRV test program was to demonstrate that the SRV will open and reclose under all expected flow conditions. The expected valve operating conditions were determined through the use of analyses of accidents and anticipated operational occurrences referenced in Regulatory Guide 1.70 (Rev 2). Single failures were applied to these analyses so that the dynamic forces on the SRVs would be maximized. Test pressures were the highest predicted by conventional safety analysis procedures. The BWROG, in their enclosure to the September 17, 1980 letter from D.B. Waters (BWROG) to R.H. Vollmer (NRC), identified 13 events that may result in liquid or two-phase SRV inlet flow that would maximize the dynamic forces on the SRV. These events were identified by evaluating the initial events described in Regulatory Guide 1.70 (Rev 2), with and without the additional conservatism of a single active component failure or operator error postulated in the event sequence. It was concluded from this evaluation that the alternate shutdown cooling mode is the only expected event that will result in liquid at the valve inlet. Consequently, this was the event simulated in the SRV test program. This conclusion and the test results applicable to LGS are discussed below.

The 13 events and the plant specific features that mitigate these events are summarized in Table 3.9-32. Of these 13 events, only nine are applicable to LGS because of its design and specific plant configuration. Four events (5, 6, 10 and 13) are not applicable to LGS for the reasons listed below:

1. Events 5 and 10 are not applicable because LGS does not have a high pressure core spray system.

2. Event 6 is not applicable because LGS does not have RCIC head sprays.

3. Event 13 is not applicable because large breaks will not be isolated at LGS.

For the nine remaining events, the LGS specific features, such as trip logic, power supplies, instrument line configuration, alarms and operator actions, have been compared to the base case analysis presented in the BWROG submittal of September 17, 1980. The comparison has demonstrated that, in each case, the base case analysis is applicable to LGS because the base case analysis does not include any plant features that are not already present in the LGS design. For these events, Table 3.9-32 demonstrates that the LGS specific features are included in the base case analysis presented in the BWROG submittal of September 17, 1980. It is seen from Table 3.9-32 that all plant features assumed in the event evaluation are also existing features in the LGS plant. All features included in this base case analysis are similar to plant features in the LGS design. Furthermore, the time available for operator action is expected to be longer in the LGS plant than in the base case analysis for each case where operator action is required due to the conservative nature of the base case analysis.

Event 7, the alternate shutdown cooling mode of operation, is the only expected event that will result in liquid or two-phase fluid at the SRV inlet. Consequently, this event was simulated in the BWR SRV test program. In LGS, the event involves flow of subcooled water (approximately 31°F subcooled) at a pressure of approximately 156 psig. The SRV inlet fluid conditions tested in the BWROG SRV test program, as documented in Reference 3.9-15, are 15°F to 50°F subcooled liquid at 20 psig to 250 psig. These fluid conditions envelope the conditions expected to occur at LGS in the alternate shutdown cooling mode of operation.

As discussed above, the BWROG evaluated transients including single active failures that would maximize the dynamic forces on the SRVs. As a result of this evaluation, the alternate shutdown cooling mode is the only expected event involving liquid or two-phase flow. Consequently, this event was tested in the BWR SRV test program. The fluid conditions and flow conditions tested in the BWROG test program conservatively envelope the LGS plant specific fluid conditions expected for the alternate shutdown cooling mode of operation (Reference 3.9-22).

d. The flow coefficient, C_v, for the Target Rock SRV used in LGS was determined in the generic SRV test program (Reference 3.9-15). The average flow coefficient calculated from the test results for the Target Rock valve is reported in table 5.2-1 of Reference 3.9-15. This test value has been used by the licensee to confirm that the liquid discharge flow capacity of the LGS SRVs will be sufficient to remove core decay heat when injected into the RPV in the alternate shutdown cooling mode. The C_v of the valve determined in the SRV test demonstrates that the LGS SRVs are capable of returning sufficient flow to the suppression pool to accommodate injection by the RHR or core spray pump.

If it were necessary to place the LGS plant in the alternate shutdown cooling mode, the operator would ensure that adequate core cooling was being provided by monitoring the following parameters: RHR or core spray flow rate, reactor vessel pressure, and reactor vessel temperature.

The flow coefficient for the Target Rock valve reported in Reference 3.9-15 was determined from the SRV flow rate when the valve inlet was pressurized to approximately 250 psig. The valve flow rate was measured with the supply line flow venturi upstream of the steam chest. The C_v for the valve was calculated using the nominal measured pressure differential between the valve inlet (steam chest) and 3 feet downstream of the valve and the corresponding measured flow rate. Furthermore, the test conditions and test configuration were representative of LGS plant conditions for the alternate shutdown cooling mode, e.g., pressure upstream of the valve, fluid temperature, friction losses, and liquid flow rate. Therefore, the reported C_v values are appropriate for application to the LGS plant.

3.9.3.3.2 Design and Installation Details for Mounting of Pressure Relief Devices in ASME Class 1, 2, and 3 Systems (Non-NSSS)

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

a. Open Discharge

There are no open discharge pressure-relieving devices with limited runs of discharge piping mounted on ASME Code Class 1, 2, and 3 systems.

b. Closed Discharge

A closed discharge system is characterized by piping between the valve and a tank, or some other terminal end. Under steady-state conditions, there are no net unbalanced forces. The initial transient response and resulting stresses are determined by using either a time history computer solution, or a conservative equivalent static solution. In calculating initial transient forces, pressure and momentum terms are included. Water slug effects are also considered.

Time history dynamic analysis is performed for the discharge piping and its supports. The effect of the loading on the header is also considered. The design load combinations for a given transient are shown in Table 3.9-11, and the design criteria and stress limits are shown in Tables 3.9-12 and 3.9-16.

3.9.3.4 Component Supports Furnished with the NSSS

3.9.3.4.1 Piping

Hangers are designed in accordance with ANSI B31.7. In general, the load combinations for the various operating conditions correspond to those used to design the supported pipe. Design transient cyclic data are not applicable to hangers because no fatigue evaluation is necessary to meet the code requirements. All hangers are designed, fabricated, and assembled so that they cannot become disengaged by the movement of the supported pipe or equipment after they are installed. The design load on hangers is the load caused by dead weight. The hangers are calibrated to ensure that they support the design load at both their hot and cold load settings. Hangers provide a specified down travel and up travel in excess of the specified thermal movement. Visual inspection and acceptance of pipe supports are performed in accordance with NCIG-01 requirements (Reference 3.9-10).

For pipe supports, reactions produced by primary and secondary pipe loads are categorized as primary. The primary and secondary loads are summed and compared to the load rating to ensure that the rating is not exceeded. Because no distinction is made between primary and secondary loads, and load rated components are designed to primary limits or qualified by testing, the supports meet primary stress criteria for primary and secondary loads combined.

Required load capacity and snubber location for NSSS piping systems are determined by GE as a part of the NSSS piping system design and analysis scope. However, design, installation and inspection of snubbers are included in the non-NSSS scope (Section 3.9.3.5).

The entire piping system, including valves and the suspension system between anchor points, is mathematically modeled for complete structural analysis. In the mathematical model, the snubbers are modeled as springs with a given stiffness depending on the snubber size. The analysis determines the forces and moments acting on each component and the forces acting on the snubbers due to all dynamic loading conditions defined in the piping design specification. The design load on snubbers includes those loads caused by seismic forces (OBE and SSE), system anchor movements, and reaction forces caused by relief valve discharge, turbine stop valve closure, and other hydrodynamic forces (SRV, LOCA, annulus pressurization).

The assessment of all affected piping including their supports and structural modifications necessitated by reconciliation of the suppression pool hydrodynamic loads have been completed.

The snubber location and loading direction are decided by estimation so that the stresses in the piping system have acceptable values. The snubber locations and direction are refined by performing the computer analysis on the piping system as described above.

The spring constant required by the suspension design specification for a given load capacity snubber is compared against the spring constant used in the piping system model. If the spring constants are not in agreement, they are brought into agreement, and the system analysis is redone to confirm the snubber loads.

If the stiffness of the backup structure for the snubber is not large compared to that of the snubbers, the reduced effective snubber stiffness (spring constant) is used in the analysis to account for backup structure flexibility.

Snubber design is discussed in Section 3.9.3.5.2.

3.9.3.4.2 NSSS Floor-Mounted Equipment (Pumps, Heat Exchangers, and RCIC and HPCI Turbines)

The ECCS pumps, RCIC and SLCS pumps, RHR heat exchanger, and RCIC and HPCI turbines are analyzed to verify the adequacy of their support structure under various plant operating conditions. In all cases, the stress loads in the critical support areas are within ASME Code allowables. The assessment of all affected equipment including their supports and structural modifications necessitated by reconciliation of the suppression pool hydrodynamic loads have been completed. The loading conditions, stress criteria, and allowable and calculated stresses in the critical support areas are summarized in Tables 3.9-6(k), 3.9-6(l), 3.9-6(m), 3.9-6(n), 3.9-6(o), 3.9-6(q), 3.9-6(r), and 3.9-6(t).

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

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

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

a. Simulate actual mounting conditions

b. Simulate all static and dynamic loadings on the pump

c. Monitor pump operability during testing

d. Normal operation of the pump during and after the test indicates that the supports are adequate; any deflection or deformation of the pump supports that precludes the operability of the pump is not accepted.

e. Supports are inspected for structural integrity after the test; any cracking or permanent deformation is not accepted.

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

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

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

c. For emergency and faulted plant conditions, the deformations must not exceed the values permitted by the designer to ensure operability of the pumps.

3.9.3.4.4 RPV Support Skirt

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

An analysis of RPV support skirt buckling for faulted conditions shows that the support skirt has the capability to meet ASME Section III, paragraph F-1370(c), faulted condition limits of 0.67 times the critical buckling strength of the support at temperature. The faulted condition analyzed included the compressive loads due to the design basis maximum earthquake, the overturning moments and shears due to the jet reaction load resulting from a severed pipe, and the compressive effects on the support skirt due to the thermal and pressure expansion of the reactor vessel.

Subsequently, based on currently defined faulted condition loads, the maximum compressive stress in the support skirt including axial and bending loads in less than the faulted condition allowable of appendix F (paragraph F-1325) determined by the methods of ASME Section III, NB 3133.6. (Axial loads include weight, fuel interaction, seismic SSE, and the maximum of condensation oscillation, chugging and vent clearing due to a LOCA. Bending loads include seismic SSE and jet reaction, jet impingement, and annulus pressurization due to a LOCA.) The loading criteria, stress criteria, calculated and allowable stresses are summarized in Table 3.9-6(f).

3.9.3.4.5 Bolting Stress Limits (NSSS)

3.9.3.4.5.1 Component Support Bolting

The support bolting of the RWCU pump that is not essential to safety is designed for the effects of pipe load and SSE load to the requirements of ASME Section III, Appendix XVII. The stress limits of 0.41 S_y for tension and 0.15 S_y for shear are used.

For RCIC/SLCS pumps and RCIC turbine, the equipment-to-base plate bolting satisfies the following design criteria: For normal and upset conditions, 1.0S is used for primary membrane and 1.5S for primary membrane plus bending, where (S) is the allowable stress limit from ASME Section III, Appendix I, table I-7.3. For emergency and faulted conditions, stresses shall be less than 1.2 times the allowable limits for "normal and upset" given above.

There are no flange-type connections in pipe mounted component supports.

3.9.3.4.5.2 Piping Supports and Pipe-Mounted Equipment (Valves and Pump) Supports

The hanger type supports (including clamps) and their bolting are designed in accordance with the requirements of ANSI B31.7. The allowable stress limit for the bolting is equal to or less than the yield strength of the bolt material at temperature.

3.9.3.5 Component Supports Not Furnished with the NSSS

3.9.3.5.1 Design Basis

ASME Section III, Subsection NF, is used for the design and installation of the CRD piping supports and TIP piping supports. For the remainder of the non-NSSS portion of the LGS design and installation, Subsection NF is not used. The codes used instead are ANSI B31.7 for nuclear class piping and ANSI B31.1 for non-nuclear class piping. Visual inspection and acceptance of non-Subsection NF pipe supports are performed in accordance with NCIG-01 requirements (Reference 3.9-10). For a graphical definition of jurisdictional boundaries between pipe supports and supporting structures, refer to Figures 3.9-9 and 3.9-10.

The design loading combinations for supports for ASME Class 1, 2 and 3 components, categorized with respect to plant operating conditions identified as normal, upset, emergency, and faulted are given in Table 3.9-21. This table also provides the stress limits for each plant operating condition.

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

3.9.3.5.2 Snubbers

Snubbers are used in seismic Category I systems. The load ratings of the snubbers are appropriate for the design conditions and load combinations.

3.9.3.5.2.1 Analytical Methods

The methodology used for the stress analysis of seismic Category I, 2½ inches and larger piping systems is as follows:

a. For systems designed to seismic and hydrodynamic loads, the flexibility of the pipe supports are considered in the piping stress analysis. A stiffness tolerance criteria is used to facilitate support design and installation.

b. For systems designed to seismic loads, only the supports are considered as rigid members in the piping stress analysis model and are designed such that their fundamental frequencies in the direction of the applied load is within the rigid range of the seismic response spectra.

3.9.3.5.2.2 Snubber Design Specification

Snubbers for LGS are used to arrest shock due to seismic and other dynamic transient events. Under such applications, the snubbers will be subjected to a limited number of load cycles. Snubbers are not designed for vibration control. Therefore, no fatigue evaluation has been performed.

The purchase specification of new shock suppressors (snubbers) covers the following criteria for supplier's performance qualification tests and load tests. End clearance and lost motion are not considered in the piping stress analysis. Instead, linear average snubber stiffness is used in combination with that of the snubber support structure.

Mechanical Snubbers

a. The friction resistance of the suppressor to normal pipe movement shall be a maximum of 1% of the service level A rated load of the unit or 5 lb, whichever is greater.

b. The suppressor shall limit the acceleration of the pipe to a maximum of 0.02 g when subjected to any load up to the normal rated load.

   Based on experience from other plants, the activation threshold has been shown to be above the thermal growth rate of the piping systems.

c. The total lost movement at the suppressor shall not exceed ±0.040 inches due to any applied dynamic cycle load from 3-33 cps up to the rated load at the unit.

d. The suppressor shall be designed for an exposure to a temperature of room temperature (65° to 70°F) prior to initial startup and 200°F during continuous operations and to a radiation dose of 6.4×10^7 rads during the life of the plant. Functioning of the shock suppressor under 340°F temperature for a short duration under rated load shall be demonstrated.

Hydraulic Snubbers

a. The friction resistance of the snubbers rated at 20 kips or greater shall have a friction resistance to normal movement of less than or equal to one percent of the service level A rated load. Snubbers rated at less than 20 kips shall have a friction resistance to normal movement of less than or equal to two percent of the rated load.

b. Activation of the snubber shall be defined as the velocity at which the snubber begins to support load, and restricts movement to the maximum bleed rate of the snubber. The activation velocity may be referred to as the lock-up velocity or activation velocity interchangeably. These requirements shall be satisfied under both tension and compression. The activation velocity of new snubbers at room temperature (65 to 75°F) shall be greater than or equal to 4.72 IPM and less than or equal to 14.17 IPM for all sizes.

c. The bleed rate shall be defined as that velocity at which the snubber will move under constant rated loads after the activation velocity has been reached and the control valves have closed. These requirements shall be satisfied under both tension and compression. The bleed velocity for the new snubbers at room temperature (65 to 75°F) shall be greater than or equal to 0.47 IPM and less than or equal to 4.72 IPM for all sizes.

d. The snubber shall be designed to withstand the normal environmental conditions inside the drywell of -0.5 to 2.0 psig, 160°F (based on Drywell Air Cooling System Design Bases described in Section 9.4.5.2), 40 to 90% relative humidity, and a radiation dose of 6.4×10^7 rads during the life of the plant. This bounding radiaton dose may be reduced by component or model specific evaluations of normal and design basis accident radiation environments.

3.9.3.5.2.3 Snubber Performance Test

A production test is required to be performed on each unit.

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

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

c. On units which allow movement after the initial suppression of load, determine that the maximum allowable acceleration (mechanical snubbers) or velocity (hydraulic snubbers) is not exceeded. This requirement must be met in both tension and compression at room temperature.

d. Measure and record lost motion of the snubber mechanism (mechanical snubbers only).

Qualification tests are to be performed on randomly selected production models. These tests are used to demonstrate the required load performance (load rating) and specified displacement when subjected to dynamic load cycling. Also included in these tests are low temperature, high temperature, humidity, radiation and faulted load conditions.

Preinstallation, installation and postinstallation inspections of snubbers are performed before a preoperational test. Additional inspections are required if more than 6 months have elapsed between the last inspection and initial system power operation (Section 3.9.3.5.2.4).

Steady state vibration conditions will be identified during the preoperational test program. Snubbers have not been used to control steady state vibration. If the snubbers are used to correct such conditions, they will be evaluated for acceptability under those conditions.

In addition, the snubber inservice inspection program ensures that any potential malfunction due to fatigue-type failure will be detected.

3.9.3.5.2.4 Snubber Preservice Examination

Preservice examination of snubbers should be performed after installation, but not more than six months prior to initial system preoperational testing.

The mechanical snubber examination is described as follows. The objective is to verify adequate preservice examination to mechanical snubbers on all safety-related systems. The prerequisites of all preinstallation, installation, and postinstallation inspections have been performed on mechanical snubbers by designated inspection organizations. Verify through document review that all inspection activities have been completed, verified, and signed. Reviews will be made by systems and additional visual inspections will be made if original inspections are performed more than 6 months prior to initial power operation of the system.

The preservice examination should as a minimum verify the following:

a. There are no visible signs of damage or impaired operability as a result of storage, handling or installation.

b. The snubber location, orientation, position setting, and configuration (attachments, extensions, etc.) are according to design drawings and specifications.

c. Snubbers are not seized, frozen, or jammed.

d. Adequate swing clearance is provided to allow snubber movement.

e. Structural connections such as pins, fasteners, and other connecting hardware such as lock nuts, tabs, wire, and cotter pins are installed correctly.

If the period between the initial preservice examination and initial power operation exceeds 6 months, re-examination of items a. and d. shall be performed. Snubbers which are installed incorrectly or otherwise fail to meet the above requirements must be repaired or replaced and re-examined in accordance with the above criteria.

3.9.3.5.3 Struts

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

For pipe supports, reactions produced by primary and secondary pipe loads are categorized as primary. The primary and secondary loads are summed and compared to the load rating to ensure that the rating is not exceeded. Because no distinction is made between primary and secondary loads, and load rated components are designed to primary limits or qualified by testing, the supports meet primary stress criteria for primary and secondary loads combined.

3.9.3.5.4 Bolting Stress Limits (Non-NSSS)

The bolting used in pipe support components is designed to an allowable stress equal to or less than the yield strength of the bolt material at temperature.

For flanged connections, the bolt allowables used in the piping are ASME Section III, 1979 Summer Addenda, Sections NB, NC and ND for Class 1, 2, and 3, respectively.

3.9.4 CONTROL ROD DRIVE SYSTEM

The discussion in this Section includes the CRDM, the HCU, the condensate supply system, and the scram discharge volume, and extends to the coupling interface with the control rods.

3.9.4.1 Descriptive Information on CRD System

Descriptive information on the CRD system is contained in Section 4.6.

3.9.4.2 Applicable CRD System Design Specifications

The CRD system is designed to meet the functional design criteria as outlined in Section 4.6, and consists of the following:

a. Locking piston CRD

b. HCU

c. Hydraulic power supply (pumps)

d. Interconnecting piping

e. Flow and pressure and isolation valves

f. Instrumentation and electrical controls

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

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

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

3.9.4.3 Design Loads, Stress Limits, and Allowable Deformation

The ASME Code components of the CRD system are evaluated analytically, and the design loading conditions, stress criteria, calculated stresses, and allowable stresses are summarized in Tables 3.9-6(u) and 3.9-6(v). For the noncode components, experimental testing is used to determine the CRD performance under all possible conditions, as described in Section 3.9.4.4.

Deformation is not a limiting factor in the analysis of the CRD components based on the results of the numerous tests performed on the drive.

3.9.4.3.1 CRD Housing Supports

The CRD housing support system functions are described in Section 4.6.1.3.

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

The CRD housing supports are designed as seismic Category I equipment. Loading conditions and examples of stress analysis results and limits are given in Table 3.9-6(z).

3.9.4.4 CRD Performance Assurance Program

The CRD test program consists of the following tests:

a. Development tests

b. Factory quality control tests

c. 5 year maintenance life tests

d. 1.5x design life tests

e. Operational tests

f. Acceptance tests

g. Surveillance tests

All of the above tests except c. and d. are discussed in Section 4.6.3. Tests c. and d. are discussed below:

Test c. - 5 Year Maintenance Life Tests

Four CRDs are normally picked at random from the production stock each year, and subjected to various tests under simulated reactor conditions and 1/6 of the cycles specified in Section 3.9.1.1.

Upon completion of the test program, CRDs must meet, or surpass, the minimum specified requirements.

Test d. - 1.5x Design Life Tests

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

Two CRDs underwent such testing in 1976. Upon completion of the test program, these CRDs met or surpassed the minimum specified requirements.

3.9.5 REACTOR PRESSURE VESSEL INTERNALS

This section identifies and discusses the structural and functional integrity of the major RPV internals.

3.9.5.1 Design Arrangements

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

Core Support Structures

Shroud

Shroud support

Core support plate and holddown bolts

Top guide (including bolts and keepers)

CRD housing

Fuel supports

Control rod guide tubes

Reactor Internals

*Jet pump assemblies and instrumentation

*Feedwater spargers

Vessel head spray nozzle (Removed-Unit 1. Never installed-Unit 2)

Differential pressure and liquid control lines

Incore flux monitor guide tubes

*Initial startup neutron sources

*Surveillance sample holders

Core spray lines and spargers

*Incore instrument housings

LPCI coupling

*Steam dryer

*Shroud head and steam separator assembly

*Guide rods

CRD thermal sleeves

* = Nonsafety class component

A general assembly drawing of the important reactor components is shown in Figure 3.9-4.

The floodable inner volume of the RPV can be seen in Figure 3.9-5. This is the volume inside the core shroud up to the level of the jet pump suction inlet.

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

The LPCI couplings incorporate sleeves to allow free thermal expansion.

3.9.5.1.1 Core Support Structures

The core support structures consist of those items listed in Section 3.9.5.1. These structures form partitions within the reactor vessel, to sustain pressure differentials across the partitions, to direct the flow of the coolant water, and to laterally locate and support the fuel assemblies. Figure 3.9-5 shows the reactor vessel internal flow paths.

3.9.5.1.1.1 Shroud

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

3.9.5.1.1.2 Shroud Support

The shroud support is designed to support the shroud, and to support and locate the jet pumps. The shroud support provides an annular baffle between the RPV and the shroud. The jet pump discharge diffusers penetrate the shroud support to introduce the coolant to the inlet plenum below the core.

3.9.5.1.1.3 Shroud Head and Steam Separator Assembly

This component is not a core support structure. It is discussed here to describe coolant flow paths in the RPV. The shroud head and steam separator assembly is bolted to the top of the shroud, forming the top of the core discharge plenum. This plenum provides a mixing chamber for the steam-water mixture before it enters the steam separators. Individual stainless steel axial flow steam separators are attached to the top of standpipes that are welded into the shroud head. The steam separators have no moving parts. In each separator, the steam-water mixture rising through the standpipe passes vanes that impart a spin that establishes a vortex, separating the water from the steam. The separated water flows from the lower portion of the steam separator into the downcomer annulus.

3.9.5.1.1.4 Core Support Plate

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

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

3.9.5.1.1.5 Top Guide

The top guide is formed by a series of stainless steel beams joined at right angles to form square openings, and fastened to a peripheral rim. Each opening provides lateral support and guidance for 4 fuel assemblies, or in the case of peripheral fuel, for less than 4 fuel assemblies. Sockets are provided in the bottom of the beam intersections to anchor the incore flux monitors and startup neutron sources. The rim of the top guide rests on a ledge between the upper and central portions of the shroud. The top guide has alignment pins that engage and bear against slots in the shroud which are used to correctly position the assembly before it is secured. Lateral restraint is provided by wedge blocks between the top guide and the shroud wall.

3.9.5.1.1.6 Fuel Supports

The fuel supports, shown in Figure 3.9-6 are of two basic types; peripheral supports, and four-lobed orificed fuel supports. The peripheral fuel support is located at the outer edge of the active core, and is not adjacent to control rods. Each peripheral fuel support holds one fuel assembly, and contains a single orifice assembly designed to ensure proper coolant flow to the peripheral fuel assembly. Each four-lobed orificed fuel support holds four fuel assemblies, and is provided with four orifice plates to ensure proper coolant flow distribution to each rod-controlled fuel assembly. The four-lobed orificed fuel supports rest in the top of the control rod guide tubes, which are supported laterally by the core plate. The control rods pass through slots in the center of the four-lobed orificed fuel support. A control rod and the four adjacent fuel assemblies represent a core cell (Section 4.1.2).

3.9.5.1.1.7 Control Rod Guide Tubes

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

3.9.5.1.1.8 Jet Pump Assemblies

The jet pump assemblies are not core support structures, but are discussed here to describe coolant flow paths in the vessel. The jet pump assemblies are located in two semicircular groups in the downcomer annulus, between the core shroud and the reactor vessel wall. The design and performance of the jet pumps are covered in detail in References 3.9-19 and 3.9-20. Each stainless steel jet pump consists of driving nozzles, a suction inlet, a throat or mixing section, and a diffuser (Figure 3.9-7). The driving nozzle, suction inlet, and throat are joined together as a removable unit, and the diffuser is permanently installed. High pressure water from the recirculation pumps is supplied to each pair of jet pumps through a riser pipe welded to the recirculation inlet nozzle thermal sleeve. A riser brace consists of cantilever beams welded to a riser pipe and to pads on the reactor vessel wall.

The nozzle entry section is connected to the riser by a metal-to-metal, spherical-to-conical seal joint. Firm contact is maintained by a holddown clamp. The throat section is supported laterally by a bracket attached to the riser. There is a slip-fit joint between the throat and diffuser. Some jet pumps have been equipped with a clamp on this slip-fit joint to dampen vibration forces. The diffuser is a gradual conical section, changing to a straight cylindrical section at the lower end.

The licensee will reduce the preload on the beams from 30 kips to 25 kips in accordance with GE recommendations. This increases the expected life of the beams to 19-40 years. Inservice inspection of the jet pump holddown beam will be performed to detect cracking. Inspection frequencies will be based on a lead-plant experience and GE testing, and will be such that any crack initiation will be detected prior to beam failure.

3.9.5.1.1.9 Steam Dryers

The steam dryer assembly is not a core support structure. It is discussed here to describe coolant flow paths in the vessel. The steam dryers remove moisture from the wet steam leaving the steam separators. The extracted moisture flows down the dryer vanes to the collecting troughs, then flows through tubes and into the downcomer annulus. A skirt extends from the bottom of the dryer vane housing to the steam separator standpipe, below the water level. This skirt forms a seal between the wet steam plenum and the dry steam flowing from the top of the dryers to the steam outlet nozzles.

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

3.9.5.1.1.10 Feedwater Spargers

The feedwater nozzle and sparger design follows the resolution presented in Reference 3.9-6. These components are not core support structures. They are discussed here to describe flow paths in the vessel. The feedwater spargers are stainless steel headers located in the mixing plenum above the downcomer annulus. A separate sparger is fitted to each feedwater nozzle, and is shaped to conform to the curvature of the vessel wall. Sparger end brackets are pinned to vessel brackets to support the spargers. Feedwater flow enters the center of the spargers, and is discharged radially inward, mixing the cooler feedwater with the downcomer flow from the steam separators and steam dryer, before it contacts the vessel wall.

The feedwater also serves to condense the steam in the region above the downcomer annulus, and to subcool the water flowing to the jet pumps and recirculation pumps.

3.9.5.1.1.11 Core Spray Lines

This component is not a core support structure. It is discussed here because the core spray lines are the means for directing flow to the core spray nozzles, which distribute coolant during accident conditions.

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

3.9.5.1.1.12 Head Cooling Spray Nozzle (Removed)

This component is not a core support structure.

The head cooling spray nozzle (component B11-D072) was mounted on a short length of pipe and a flange, which was bolted to a mating flange (RPV Nozzle N6A) on the reactor vessel head. The piping supplying coolant to the nozzle has been disconnected, partially removed and the remainder abandoned in place in Unit 1 and was never installed in Unit 2. The head cooling spray nozzle in Unit 1 was removed with the discontinued piping and never installed in Unit 2. RPV Nozzle N6A is empty and blind flanged in both units.

3.9.5.1.1.13 Differential Pressure Line

This component is not a core support structure. It is discussed here to describe the coolant flow paths in the reactor vessel. The differential pressure line senses the differential pressure across the core support plate (Section 7.7). This line enters the reactor vessel at a point below the core shroud, as two concentric pipes. In the lower plenum, the two pipes separate. The inner pipe terminates inside the lower shroud support, with a capped, perforated length below the core support plate. This section of pipe was formerly utilized as the liquid control sprager but now is only used to sense the pressure below the core support plate. The outer pipe terminates open-ended immediately above the core support plate, and senses the differential pressure across the core support plate and the fuel support assemblies.

3.9.5.1.1.14 Incore Flux Monitor Guide Tubes

This component is not a core support structure, but is discussed here to describe the coolant flow paths in the reactor vessel. Incore flux monitor guide tubes provide a means of positioning fixed detectors in the core, as well as providing a path for calibration monitors (TIP system).

The incore flux monitor guide tubes extend from the top of the incore flux monitor housing (Section 5.3) in the lower plenum, to the top of the core support plate. The power range detectors for the PRNM System units, and the dry tubes for the SRM and IRM detectors are inserted through the guide tubes. A latticework of clamps, tie bars, and spacers give lateral support and rigidity to the guide tubes. The bolts and clamps are welded, after assembly, to prevent loosening during reactor operation.

3.9.5.1.1.15 Surveillance Sample Holders

This component is not a core support structure. It is discussed here to describe the coolant flow paths in the reactor vessel. The surveillance sample holders are welded baskets containing impact and tensile specimen capsules (Section 5.3). The baskets hang from the brackets that are attached to the inside wall of the reactor vessel, and extend to mid-height of the active core. The radial positions are chosen to expose the specimens to the same environment and maximum neutron fluxes experienced by the reactor vessel itself, while avoiding jet pump removal interference or damage.

3.9.5.1.1.16 Low Pressure Coolant Injection Lines

This component is not a core support structure, but is discussed here to describe the coolant flow paths in the reactor vessel. Four LPCI lines penetrate the core shroud through separate LPCI nozzles. Coolant is discharged inside the core shroud.

3.9.5.1.1.17 Startup Neutron Sources

The startup neutron sources are held in place by spring pressure between the top of the core support and the bottom of the top guide. For Unit 1, each source consists of two irradiated antimony rods within a single beryllium cylinder. Both the antimony and the beryllium are encased in stainless steel tubes. For Unit 2, californium is used; it is also encased in stainless steel tubes. The design provides for a sufficient source of neutrons present in the core to ensure that the core neutron flux is continuously detectable by installed neutron monitors and to ensure that significant changes in core reactivity are readily detectable by installed neutron flux instrumentation.

3.9.5.2 Design Loading Conditions

3.9.5.2.1 Events to be Evaluated

Examination of the spectrum of conditions that the safety design basis must satisfy by core support structure and ESF components reveals four significant faulted events:

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

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

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

d. SSE/relief valve discharge: SRV discharge in combination with SSE.

Analysis of other conditions existing during normal operation, abnormal operational transients, and accidents shows that the loads affecting the core support structures and other ESF reactor internals are less severe than these three postulated events. The faulted conditions for the RPV internals are discussed in Section 3.9.1.4. Loading combination and analysis for the RPV internals are discussed in Section 3.9.3.1, and Tables 3.9-2 and 3.9-6. These results are based on the power rerate analysis and do not reflect the use of GE13 or GE14 fuel; the impact of GE13 fuel is documented in Reference 3.9-28. The impact of GE14 fuel is documented to be bounded by GE13 fuel in Reference 3.9-34. The impact of the MUR power uprate is evaluated in Reference 3.9-31 and Reference 3.9-32. Reference 3.9-33 identifies new design basis values for Fuel Lift Margin and Control Rod Guide Tube Lift Forces under MUR conditions. Additional analyses which consider the use of GNF2 fuel are documented in Reference 3.9-35. The GNF2 fuel is demonstrated to be bounded by the analyses in Reference 3.9-33.

3.9.5.2.2 Pressure Differential During Rapid Depressurization

A digital computer code is used to analyze the transient conditions within the reactor vessel following the recirculation line break accident and the steam line break accident. The analytical model of the vessel consists of nine nodes, connected to the necessary adjoining nodes by flow paths having the required resistance and inertial characteristics. The program solves the energy and mass conservation equations for each node, giving the depressurization rates and pressure in the various regions of the reactor. Figure 3.9-8 shows the nine reactor nodes. The computer code used is the GE Short-Term Thermal-Hydraulic Model, described in Reference 3.9-21. This model is approved for use in ECCS conformance evaluation under 10CFR50, Appendix K. In order to adequately describe the blowdown pressure effect on the individual assembly components, three features are included in the model that are not applicable to the ECCS analysis and are, therefore, not described in Reference 3.9-21. These additional features are discussed below:

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

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

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

3.9.5.2.3 Recirculation Line and Steam Line Break

3.9.5.2.3.1 Accident Definition

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

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

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

3.9.5.2.3.2 Effects of Initial Reactor Power and Core Flow

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

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

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

3.9.5.2.4 Seismic and Hydrodynamic Loads

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

3.9.5.3 Design Bases

3.9.5.3.1 Safety Design Bases

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

a. Core support structures are arranged to provide a floodable volume, in which the core can be adequately cooled in the event of a breach in the nuclear system process barrier, external to the reactor vessel.

b. Deformation is limited to ensure that the control rods and the core standby cooling systems can perform their safety functions.

c. Mechanical design of applicable structures ensures that safety design bases a. and b., above, are satisfied so that the safe shutdown of the plant and removal of decay heat are not impaired.

3.9.5.3.2 Power Generation Design Bases

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

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

b. They are arranged to facilitate refueling operations.

c. They are designed to facilitate inspection.

3.9.5.3.3 Design Loading Categories

Loading combinations for the core support structures are shown in Table 3.9-26. The basis for determining faulted loads on the reactor internals is shown for seismic and hydrodynamic loads in Section 3.7, and for pipe rupture loads in Sections 3.9.5.2.3 and 3.9.5.3.4. Table 3.9-6(b) gives analytical methods and allowable and calculated stresses for typical core support structures and reactor internal components.

Stress intensity and other design limits are discussed in Section 3.9.5.3.5. The core support structures which are fabricated as part of the RPV assembly are discussed in Section 3.9.1.3.

LGS reactor internals were designed and procured prior to the issuance of ASME Section III, Subsection NG. However, an earlier draft of the ASME Code was used as a guide in the design of the reactor internals. These criteria are presented in this section and were used in lieu of Subsection NG. Subsequent to the issuance of Subsection NG, comparisons were made to ensure that the pre-NG design meets the equivalent level of safety as presented by Subsection NG.

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

3.9.5.3.3.1 Reactor Core Support and Internals Structural Margin Evaluations

The analyses of conditions found during inspections of reactor internal and core support structures use approved industry codes and standards as described in the LGS Inservice Inspection Program, references 5.2-10 and 5.2-11. These analyses are performed as described in the above referenced Sections except that limit load, linear elastic (LEFM), and elastic-plastic (EFPM) fracture mechanic methods may be used as discussed in Tables 3.9-23, 3.9-24, 3.9-28, and 3.9-30.

Neutron fluence is evaluated when the irradiation induced changes in the material fracture toughness properties are judged to be significant. These material properties include yield and ultimate tensile strengths, uniform elongation and upper-shelf Charpy energy. The trends in these properties as a function of fluence level are reviewed to determine a fluence value above which the use of LEFM or EPFM techniques would be necessary and to determine the appropriate flaw growth rate to be used in the structural margin analyses. The fluence calculations use the methodology discussed in Section 4.3.2.8.

a. The design loads for the LGS Unit 1 and Unit 2 core shroud horizontal welds H1 through H8 have been calculated and are documented in reference 3.9-29. The loads and their combinations are based on power rerate and new loads design adequacy evaluations as discussed in reference 3.9-23. The effects of additional loads from GE13 fuel design (reference 3.9-28), fuel lift loads, and increased core flow (3% noise) beyond power rerate and the new loads programs are included in the updated core shroud loads and analysis. The impact of GE14 fuel is documented to be bounded by GE13 fuel in Reference 3.9-34. Reference 3.9-33 identifies new design basis values for Fuel Lift Margin and Control Rod Guide Tube Lift Forces under MUR conditions. Additional analyses which consider the use of GNF2 fuel are documented in Reference 3.9-35. The GNF2 fuel is demonstrated to be bounded by the analyses in Reference 3.9-33.

Reference 3.9-30 was prepared in response to Generic Letter (GL) 94-03, Intergranular Stress Corrosion Cracking of Core Shrouds in Boiling Water Reactors, which required a plant specific safety assessment supporting continued operation of core shrouds. The report provides accident loading information applicable to LGS Unit 1 and Unit 2, as well as the safety assessment for postulated through-wall flaws at core shroud horizontal welds H1 through H7. The analysis considers both normal plant operations and limiting abnormal operational occurrences. It provides information on the plant response i.e., control rod insertion and ECCS injection, to postulated Main Steam Line Break (MSLB) and Recirculation Line Break (RLB), including a coincident Safe Shutdown Earthquake (SSE). The overall conclusion concerning plant safety is that the core shroud is an extremely flaw tolerant core support structure such that the probability is a very small that flawed welds would result in shroud separation under any desgn basis operational event.

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

3.9.5.3.4 Response of Internals Due to Inside Steam Break Accident

The maximum pressure loads acting on the reactor internal components result from an inside steam line break; on some components the loads are maximum when operating at the minimum power associated with the maximum core flow. This is substantiated by the analytical comparison of liquid versus steam breaks, and by the investigation of the effects of core power and core flow.

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

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

Elastic displacement is considered in the design of reactor internal components in which deflection can affect control rod insertability. Plastic deformation will not occur in any permanent core support structure component of the reactor vessel. Radiation-induced deformation can occur in the fuel channel over the core life. These effects are considered in control rod insertability tests. No fatigue analysis is required under the faulted conditions due to the low encounter frequency of faulted events and the low number of cycles. The forcing functions applicable to the reactor internals are discussed in Section 3.9.2.5. The stress, deformation, and fatigue limits are given in Table 3.9-6(b).

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

The stress, deformation, and fatigue limits are given in Table 3.9-6(f).

3.9.6 Inservice Testing Of Pumps And Valves

Inservice testing of pumps and valves is accomplished in accordance with the requirements of 10CFR50.55a, using the date of commercial operation for determining test intervals.

The Preservice Testing Program included provisions for design and access to enable the operational readiness testing of pumps and valves, and was required to comply, as a minimum, with the 1971 Edition of Section XI of the ASME B&PV Code, including the winter of 1972 Addenda (this being in effect 6 months prior to the LGS construction permit date of June 1974). That publication did not include requirements for preservice testing of pumps and valves to ensure operational readiness. The requirements for inservice testing of pumps and valves were added as Subsections IWV and IWP to the ASME B&PV Code, Section XI, Summer 1973 Addenda, effective December 30, 1973. The Preservice Testing Program for assessing operational readiness of pumps and valves was conducted, however, to the extent practical within design limitations, to comply with the 1980 Edition of ASME Section XI, with addenda through Winter 1981.

Inservice testing of pumps and valves to ensure operational readiness for the first 120-month interval was performed in accordance with the requirements of 10CFR50.55a and met, to the extent practical within design limitations, the requirements of the Code in effect 12 months prior to the date of commercial operation. The first 120-month interval for Unit 2 was required to comply with the 1986 Edition of ASME Section XI. As permitted by 10CFR50.55a, the Unit 1 IST Program was updated to comply with the 1986 Edition of ASME Section XI. Subsequently, NRC authorized a one-time extension of the first 120-month interval for Unit 1, resulting in both Units being on concurrent intervals.

During the second and successive 120-month intervals, inservice testing of pumps and valves to ensure operational readiness shall be performed in accordance with the requirements and limitations specified in 10CFR50.55a. Detailed information regarding current Code requirements, component selection, testing requirements, Code Class, and deviations from referenced Code requirements is provided in Reference 3.9-25.

3.9.7 REFERENCES

3.9-1 Welding Research Council, "Bulletin F107," (December 1965).

3.9-2 A. Kalnins, "Analysis of Shells of Revolution Subjected to Symmetrical and Nonsymmetrical Loads," Journal of Applied Mechanics, 31, pp. 467-476, (September 1964).

3.9-3 E.L. Wilson, "A Digital Computer Program for the Finite-Element Analysis of Solids with Nonlinear Material Properties, Technical Memorandum No. 23," Aerojet General Corporation Sacramento, California, (July 1965).

3.9-4 "Pressure Vessel and Piping 1972, Computer Programs Verification," ASME.

3.9-5 "Stress Report for the Target Rock 6x10 Relief Valve," Model # 7567F, 4th Edition, McGraw-Hill, (1965).

3.9-6 "Boiling Water Reactor Feedwater Nozzle/Sparger Final Report," NEDE-21821, GE, (March 1978).

3.9-7 T.K. Tung, and C.Y. Chern, "DELTAT, a Quasi-Two- Dimensional Program for Pipe Thermal Transients," ASME PVP-36, (June 1979).

3.9-8 D.R. McNeil, and J.E. Brock, "Charts for Transient Temperatures in Pipes", Heating/Piping/Air Conditioning, pp. 107-119, (November 1979).

3.9-9 H.S. Carslaw, and J.C. Jaeger, "Conduction of Heat in Solids", Oxford University Press, pp. 392-394, (1959).

3.9-10 "Visual Weld Acceptance Criteria for Structural Welding at Nuclear Power Plants, NCIG-01", Nuclear Construction Issues Group, Revision 2, (May 7, 1985).

3.9-11 E. Wilson, and S.R. Nickell, "Application of the Finite-Element Method to Heat Conduction Analysis", Nuclear Engineering and Design, 4, (1966).

3.9-12 E. Wilson, "Structural Analysis of Axisymmetric Solids", AIAA Journal, 3, (112), (December 1965).

3.9-13 P.J. Schneider, "Temperature Response Charts", John Wiley and Sons, Inc., (1963).

3.9-14 "Sample Analysis of a Class I Piping System", prepared by the Working Group on Piping (SGD, ScIII) of the ASME BP&V Code, (December 1971).

3.9-15 "Analysis of Generic BWR Safety/Relief Valve Operability Test Results", NEDE-24988-P, GE, (October 1981).

3.9-16 "BWR Fuel Channel Mechanical Design and Deflection", NEDE-21354-P, GE, (September 1976).

3.9-17 "BWR Fuel Assembly Evaluation of Combined Safe Shutdown Earthquake (SSE) and Loss-of-Coolant Accident (LOCA) Loadings," NEDE-21175-3-P, GE, (July 1982).

3.9-18 "Assessment of Reactor Internals Vibration in BWR/4 and BWR/5 Plant", NEDE-24057-P (Class 1II) and NEDO-24057 (Class 1), GE, (November 1977).

3.9-19 "Design and Performance of GE BWR Jet Pumps", APED-5460, GE, Atomic Power Equipment Department, (July 1968).

3.9-20 H.H. Moen, "Testing of Improved Jet Pumps for the BWR/6 Nuclear System", NEDO-10602, GE, Atomic Power Equipment Department, (June 1972).

3.9-21 "Analytical Model for Loss-of-Coolant Accident in Accordance with 10CFR50, Appendix K", NEDE-20566, Proprietary Document, GE.

3.9-22 "Safety/Relief Valve Pipe Forces During Water Discharge for the Limerick Generating Station", MDE-113-0986, GE, (September 1986).

3.9-23 "Reactor Pressure Vessel Power Rerate Stress Report Reconciliation for Limerick Units 1 and 2," NEDC-32234, GE-NE, December 1993.

3.9-24 "Limerick Generating Station Units 1 and 2 Power Rerate Evaluation of Main Steam and Recirculation Piping Systems, "GE-NE-123-E014-019, GE-NE, July 1993.

3.9-25 Program Document ML-008, Limerick Generating Station Units 1 and 2, "IST Program Plan" and "IST Basis Document."

3.9-26 "Limerick Generating Station, Units 1 and 2, SRV Setpoint Tolerance Relaxation Licensing Report," NEDC-32645P (December 1998).

3.9-27 "Limerick Generation Station, Units 1 and 2, Analysis of Reduced HPCI Flow," GENE-L12-00860-00-01, DRF L12-00860-00 (December 1998).

3.9-28 "Assessment of Reactor Internals due to GE13 Fuel Change," GENE-B13-02005-01, Revision 0, March 1999).

3.9-29 General Electric Nuclear Energy Report, "Core Shroud Loads at Horizontal Welds H1 through H8", B13-02066-00-20 Lt1, dated September 19, 2000, is issued as controlled document DWG, B11-D001-S-005.

3.9-30 General Electric Nuclear Energy Report, "Fermi 2 Hope Creek Unit 1, Limerick Units 1 and 2, Peach Bottom Units 2 and 3, and Susquehanna Units 1 and 2 Shroud Safety Assessment," GENE-523-A094-0995, Dated July 1999, shall be issued as Controlled Document B11-D001-S-011.

3.9-31 Design Analysis LEAM-MUR-0061, "Exelon Nuclear Limerick Units 1 and 2 Nuclear Stations Thermal Power Optimization (TPO) (3517 MWt) Task T0302: Reactor Vessel Integrity Stress and Fatigue Evaluation," GE Hitachi Nuclear Energy, 0000-0097-3401-R0, Revision 0, Janauary 2010.

3.9-32 Design Analysis LEAM-MUR-0050, "Exelon Nuclear Limerick Units 1 and 2 Thermal Power Optimization Task T0303: RPV Internals Mechanical Evaluation," GE Hitachi Nuclear Energy, 0000-0107-2835-R0, Revision 0, December 2009.

3.9-33 Design Analysis LEAM-MUR-0040, "Exelon Nuclear Limerick Units 1 and 2 Nuclear Stations Thermal Power Optimization (TPO) (3517 MWt) Task T0304: Reactor Internal Pressure Differences and Fuel Lift Evaluation," GE Hitachi Nuclear Energy, 0000-0101-4297-R0, Revision 0, July 2009.

3.9-34 "GE14 Fuel Design Cycle-Independent Analyses for Limerick Generating Station Unit 1 and 2," GE-NE-L12-00884-00-01P, March 2001 (G-080-VC-00146).

3.9-35 "GNF2 Fuel Design Cycle-Independent Analyses for Exelon Limerick Generating Station Unit 1 and 2," NEDC-33627P, Rev. 1, March 2001 (G-080-VC-00446).

Table 3.9-1

APPLICABLE THERMAL TRANSIENTS

PIPELINE	INITIAL TEMPERATURE (°F)	FINAL TEMPERATURE (°F)	TIME	TEMPERATURE RATE (°F/hr)[15]	ΔTEMPERATURE (°F)
A. 130 CYCLES, CONDITION - TEST (PRE-STARTUP LEAK TEST)[1]					
Main Steam Line	70	100	30 min	60	30
Recirculation Suction	70	100	30 min	60	30
Recirculation Discharge	70	100	30 min	60	30
Bottom Drain	70	100	30 min	60	30
SLCS	70	100	30 min	60	30
	100	50	Step	10 min	50
	50	100	Step	duration	50
Core Spray	70	100	30 min	60	30
Feedwater	70	100	30 min	60	30
B. 120 CYCLES, CONDITION - NORMAL (STARTUP)[2]					
Main Steam Line	100	546	-	100	446
Recirculation Suction	100	546	-	100	446
	546	538	Step	-	8
	538	522	Step	-	16
Recirculation Discharge	538	522	Step	-	16
Bottom Drain	538	522	Step	-	16
SLCS	538	522	Step	-	16
Core Spray	100	406	-	100	306
only 10 cycles	406	50	Step	-	356
	50	406	Step	-	356
	406	546	-	100	140
Feedwater	100	546	-	100	446
	546	90	Step	-	456
	90	420	30 min	660	330

Table 3.9-1 (Cont'd)

PIPELINE	INITIAL TEMPERATURE (°F)	FINAL TEMPERATURE (°F)	TIME	TEMPERATURE RATE (°F/hr)[15]	ΔTEMPERATURE (°F)
C. 10,400 CYCLES, CONDITION - NORMAL (DAILY POWER REDUCTION AND ROD PATTERN CHANGE)[3]					
Main Steam Line	546	546	-	-	-
Recirculation Suction	522	522	-	-	-
Recirculation Discharge	522	522	-	-	-
Bottom Drain	522	522	-	-	-
SLCS	522	522	-	-	-
Core Spray	522	522	-	-	-
Feedwater	420	354	15 min	264	66
	354	420	15 min	264	66
Cleanup Return	435	435	-	-	-
D. 2000 CYCLES, CONDITION - NORMAL (WEEKLY POWER REDUCTION)[3]					
Main Steam Line	546	546	-	-	-
Recirculation Suction	522	522	-	-	-
Recirculation Discharge	522	522	-	-	-
Bottom Drain	522	522	-	-	-
SLCS	522	522	-	-	-
Core Spray	522	522	-	-	-
Feedwater	420	324	30 min	192	96
	324	420	30 min	192	96
Cleanup Return	435	435	-	-	-
E. 70 CYCLES ,CONDITION - UPSET (PARTIAL FEEDWATER HEATER BYPASS)[3]					
Main Steam Line	546	546	-	-	-
Recirculation Suction	522	512	2 min	300	10
	512	522	4 min	150	10
Recirculation Discharge	512	522	4 min	150	10
Bottom Drain	512	522	4 min	150	10
SLCS	512	522	4 min	150	10
Core Spray	512	522	4 min	150	10
Feedwater	420	265	1.5 min	6200	155
	265	420	3 min	3100	155

Table 3.9-1 (Cont'd)

PIPELINE	INITIAL TEMPERATURE (°F)	FINAL TEMPERATURE (°F)	TIME	TEMPERATURE RATE (°F/hr)[15]	ΔTEMPERATURE (°F)
F. 10 CYCLES, CONDITION - UPSET (TURBINE TRIP 100 PERCENT BYPASS)[3]					
Main Steam Line	546	546	-	-	-
Recirculation Suction	522	490	1.5 min	1,280	32
	490	522	4 min	480	32
Recirculation Discharge	490	522	4 min	480	32
Bottom Drain	490	522	4 min	480	32
SLCS	490	522	4 min	480	32
Core Spray	490	522	4 min	480	32
Feedwater	420	100	1.5 min	12,800	320
	100	420	4 min	4,800	320
G. 40 CYCLES, CONDITION - UPSET (SCRAM - T/G TRIP FEEDWATER ON - MSIV OPEN)[4]					
Main Steam Line	546	565	10 sec7000		19
	565	538	15 sec6500		27
	538	400	-	100	138
	400	546	-	100	146
Recirculation Suction	522	400	-	100	122
	400	546	-	100	146
	546	538	Step	-	18
	538	522	Step	-	16
Recirculation Discharge	538	522	Step	-	16
Bottom Drain	538	522	Step	-	16
SLCS	538	522	Step	-	16
Core Spray	538	522	Step	-	16
Feedwater	420	275	1 min	8700	145
	275	100	15 min	700	175
	100	250	Step	-	150
	250	420	30 min	340	170

Table 3.9-1 (Cont'd)

PIPELINE	INITIAL TEMPERATURE (°F)	FINAL TEMPERATURE (°F)	TIME	TEMPERATURE RATE (°F/hr)(15)	ΔTEMPERATURE (°F)
H. 140 CYCLES, CONDITION - UPSET (ALL OTHER SCRAMS)(5)					
Main steam line	546	538	15 sec	1920	8
	538	400	-	100	138
	400	546	-	100	146
Recirculation Suction	522	400	-	100	122
	400	546	-	100	146
	546	538	Step	-	18
	538	522	Step	-	16
Recirculation Discharge	538	522	Step	-	16
Bottom Drain	538	522	Step	-	16
SLCS	538	522	Step	-	16
Core Spray	538	522	Step	-	16
Feedwater	420	275	1 min	8700	145
	275	100	15 min	700	175
	100	250	Step	-	150
	250	420	30 min	340	170
I. CYCLES LISTED BELOW, CONDITION - NORMAL (RATED POWER)(3)					
Main Steam Line	546	546	-	-	-
Recirculation Suction	522	546	-	-	-
Recirculation Discharge	522	522	-	-	-
Bottom Drain	522	150	1 hr	372	372
240 Cycles	150	522	Step	NA	372
SLCS	522	60	Step	NA	462
10 Cycles	60	522	60 min	462	462
Core Spray	522	522	-	-	-
Feedwater	420	420	-	-	-
Cleanup Return	435	435	-	-	-

Table 3.9-1 (Cont'd)

PIPELINE	INITIAL TEMPERATURE (°F)	FINAL TEMPERATURE (°F)	TIME	TEMPERATURE RATE (°F/hr)[15]	ΔTEMPERATURE (°F)
J. 111 CYCLES, CONDITION - NORMAL (SHUTDOWN)[6]					
Main Steam Line	546	375	-	100	171
	375	330	10 min	270	45
	330	100	-	100	230
RHR Return	375	50	Step	15 sec	325
	50	300	Step	duration	250
	300	100	-	100	200
Recirculation Suction	522	546	Step	-	24
	546	375	-	100	171
	375	330	10 min	270	45
	330	100	-	100	230
Bottom Drain	330	100	-	100	230
SLCS	330	100	-	100	230
Core Spray	330	100	-	100	230
Recirculation Discharge	522	546	Step	-	24
	546	375	-	100	171
	375	300	Step	-	75
	300	260	10 min	240	40
	260	100	-	100	160
Feedwater	420	265	30 min	310	155
	265	420	Step	-	155
	420	546	-	100	126
	546[7]	100	-	100	446
K. 10 CYCLES, CONDITION - EMERGENCY (LOSS OF FEEDWATER PUMPS - MSIV CLOSE)[8]					
Main Steam Line	546	573	3 sec	32,400	27
	573	561	10 sec	4300	12
	561	538	3 min	560	23
	538	561	73 min	19	23
	561	500	7 min	520	61
	500	400	-	100	100
	400	546	-	100	146
Recirculation Suction	522	300	30 min	444	222
	300	546	-	100	246
	546	538	Step	-	8
	538	522	Step	-	16

Table 3.9-1 (Cont'd)

PIPELINE	INITIAL TEMPERATURE (°F)	FINAL TEMPERATURE (°F)	TIME	TEMPERATURE RATE (°F/hr)(15)	ΔTEMPERATURE (°F)
Recirculation Discharge	538	522	Step	-	16
Bottom Drain	522	300	3.7 min	3600	222
	300	500	23 min	523	200
	500	300	7 min	1720	200
	300	546	-	100	246
	546	538	Step	-	8
	538	522	Step	-	16
SLCS	538	522	Step	-	16
Core Spray	538	522	Step	-	16
Feedwater	420	546	Step	-	126
	546	40	Step	-	506
	40	546	23 min	1300	506
	546	40	Step	-	506
	40	546	51 min	600	506
	546	40	Step	-	506
	40	300	5 min	3120	260
	300	546	-	100	246
	546	100	Step	-	446
	100	250	Step	-	150
	250	420	30 min	340	170

L. 1 CYCLE, CONDITION - EMERGENCY (REACTOR OVERPRESSURE DELAYED SCRAM)(9)

Main Steam Line	546	583	2 sec	66,600	37
	583	538	30 sec	5,400	45
	538	400	-	100	138
Recirculation Suction	522	562	11 sec	13,100	40
	562	400	-	100	162
Recirculation Discharge	562	400	-	100	162
Bottom Drain	562	400	-	100	162
SLCS	562	400	-	100	162
Core Spray	562	400	-	100	162
Feedwater	420	276	1 min	8,640	144
	276	100	15 min	704	176

Table 3.9-1 (Cont'd)

PIPELINE	INITIAL TEMPERATURE (°F)	FINAL TEMPERATURE (°F)	TIME	TEMPERATURE RATE (°F/hr)[15]	ΔTEMPERATURE (°F)
M. 8 CYCLES, CONDITION - EMERGENCY (SINGLE SRV BLOWDOWN)[10]					
Main Steam Line	546	375	10 min	1026	171
	375	100	-	100	275
Recirculation Suction	522	375	10 min	882	147
	375	100	-	100	275
Recirculation Discharge	375	100	-	100	275
Bottom Drain	375	100	-	100	275
SLCS	375	100	-	100	275
Core Spray	375	100	-	100	275
Feedwater	420	276	1 min	8640	144
	276	100	15 min	704	176
N. 1 CYCLE, CONDITION - EMERGENCY (AUTOMATIC DEPRESSURIZATION)[11]					
Main Steam Line	546	375	3.3 min	3100	171
	522	375	3.3 min	2700	147
	375	281	-	300	94
Recirculation Discharge	375	281	-	300	94
Bottom Drain	375	281	-	300	94
SLCS	375	281	-	300	94
Core Spray	375	281	-	300	94
Feedwater	420	276	1 min	8640	144
	276	100	15 min	784	176
O. 1 CYCLE, CONDITION - EMERGENCY (IMPROPER START OF COLD RECIRCULATION LOOP)[3]					
Main Steam Line	546	546	-	-	-
Recirculation Suction	522	130	Step	26 sec	392
	130	522	Step	duration	392
Recirculation Discharge	522	130	Step	34 sec	392
	130	522	Step	duration	392
Bottom Drain	522	522	-	-	-
SLCS	522	522	-	-	-
Core Spray	522	268	Step	34 sec	254
	268	522	Step	duration	254
Feedwater	420	420	-	-	-

Table 3.9-1 (Cont'd)

PIPELINE	INITIAL TEMPERATURE (°F)	FINAL TEMPERATURE (°F)	TIME	TEMPERATURE RATE (°F/hr)(15)	ΔTEMPERATURE (°F)
P. 1 CYCLE, CONDITION - EMERGENCY (SUDDEN PUMP START IN COLD LOOP)(3)					
Main Steam Line	546	546	-	-	-
Recirculation Suction	522	522	-	-	-
Recirculation Discharge	522	130	Step	34 second	392
	130	522	Step	duration	392
Bottom Drain	522	350	Step	34 second	172
	350	522	Step	duration	172
SLCS	350	522	Step	34 second duration	172
Core Spray	522	522	-	-	-
Feedwater	420	420	-	-	-
Q. 1 CYCLE, CONDITION - EMERGENCY (IMPROPER START WITH RECIRCULATION PUMPS OFF)(12)					
Main Steam Line	100	546	-	100	446
Recirculation Suction	100	546	-	100	446
Recirculation Discharge	100	546	-	100	446
	546	130	Step	34 sec	416
	130	546	Step	duration	416
Bottom Drain	100	546	5 min	5352	446
SLCS	100	546	5 min	5352	446
Core Spray	100	546	-	100	446
Feedwater	90	546	-	100	456
	546	90	Step	-	456
	90	420	30 min	660	330
R. 1 CYCLE, CONDITION - FAULTED (PIPE RUPTURE AND BLOWDOWN)(13)					
Main Steam Line	546	281	15 sec	63,500	265
Recirculation Suction	522	281	15 sec	57,000	241
Recirculation Discharge	522	281	15 sec	57,000	241
	281	223	35 sec		58
	223	50	Step	90 sec	173
	50	130	Step	duration	80

Table 3.9-1 (Cont'd)

PIPELINE	INITIAL TEMPERATURE (°F)	FINAL TEMPERATURE (°F)	TIME	TEMPERATURE RATE (°F/hr)(15)	ΔTEMPERATURE (°F)
Bottom Drain	522	281	15 sec	57,000	241
	281	273	35 sec	822	8
	273	50	Step	90 sec	223
	50	130	Step	duration	80
SLCS	50	130	Step	90 sec duration	80
Core Spray	522	406	10 sec	41,700	116
	406	50	Step	90 sec	356
	50	130	Step	duration	80
Feedwater	420	281	15 sec	33,400	139

S. BECHTEL CRITERIA FOR BOP PIPING

1. ½ SSE Cycles (OBE)	Condition - Upset
Expected number of equivalent ½ SSE in life of pipe system	5
Average duration of strong motion vibration ½ SSE	15 sec
Average number of maximum seismic load cycles of pipe system for each ½ SSE	10
Total lifetime number of maximum seismic load cycles of piping system	50
2. SSE Cycles (Design Basis Earthquake)	Condition - Faulted
Expected number of equivalent SSE in life of pipe system	1
Average duration of strong motion vibration SSE	15 sec

Table 3.9-1 (Cont'd)

T. GENERAL ELECTRIC CRITERIA FOR NSSS PIPING

1.	½ SSE Cycles (OBE)	Condition - Upset
	Expected number of equivalent ½ SSE in life of pipe system	1
	Average duration of strong motion vibration ½ SSE	30 sec
	Average number of maximum seismic load cycles of pipe system for each ½ SSE	10
	Total lifetime number of maximum seismic load cycles of piping system	10
2.	SSE Cycles (Design Basis Earthquake)	Condition - Faulted
	Expected number of equivalent SSE in life of pipe system	1
	Average duration of strong motion vibration SSE	30 sec
	Average number of maximum seismic load cycles of pipe system for each SSE	1
	Total lifetime number of maximum seismic load cycles of piping system	1
3.	Turbine Stop Valve(14) Closure	Condition - Upset 120 cycles
4.	Relief Valve Lift Cycles(14) (at 3 cycles per actuation)	Condition - Upset 34,200 cycles

Table 3.9-1 (Cont'd)

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(1) After temperature is raised to 100°F, reactor pressure is increased to 1250 psig and then decreased to 0 psig.
(2) Reactor pressure increases from 0 to 1000 psig at rate of temperature increase.
(3) Reactor pressure remains at 1000 psig.
(4) Reactor pressure increases to 1125 psig all relief valves open. Pressure decreases to 240 psig and then increases to 1000 psig.
(5) Reactor pressure decreases to 240 psig and then increases to 1000 psig.
(6) Reactor pressure decreases from 1000 psig to 0 psig.
(7) 5 step changes to 100°F and back during cooldown.
(8) Reactor pressure increases to 1180 psig. All relief valves open. Pressure decreases to 1125 psig and relief valves close. RCIC initiates and pressure decreases to 875 psig. RCIC trips off on high level and pressure increases to 1125. One relief valve opens and then closes as pressure decreases at rate of 100°F/hr.
(9) Reactor pressure increases to 1350 psig. All relief valves and safety valves open. Pressure decreases to 240 psig.
(10) Reactor pressure decreases to 0 psig with one relief valve or safety valve open.
(11) Reactor pressure decreases with auto-blowdown relief valves open to 35 psig.
(12) Reactor pressure increases to 1000 psig as temperature increases.
(13) Reactor pressure decreases from 1000 psig to 35 psig in 15 seconds.
(14) Not applicable to recirculation piping due to negligible effect.
(15) Temperature rates are informational only and approximated.

Table 3.9-2

PLANT EVENTS

EVENT NO.		NO. OF CYCLES
	NORMAL, UPSET, AND TESTING CONDITIONS	
1.	Bolt-up[(1)]	123
2.	Design hydrostatic test	130
3.	Startup (100°F/hr heatup rate)[(2)]	120
4.	Daily reduction to 75% power[(1)]	10,000
5.	Weekly reduction to 50% power[(1)]	2,000
6.	Control rod pattern change[(1)]	400
7.	Loss of feedwater heaters	80
8.	OBE event at rated operating conditions	10/50[(3)]
9.	Scram:	
	a. Turbine-generator trip, feedwater on, isolation valves stay open	40
	b. Other scrams	140
10.	Reduction to 0% power, hot standby, shutdown (100°F/hr cooldown rate)[(2)]	111
11.	Unbolt	123
12.	Pre-op blowdown	10
13.	Loss of RWCU	240
14.	MSRV actuations	7700
15.	SLCS Operation	10

Table 3.9-2 (Cont'd)

EVENT NO.			NO. OF CYCLES
	EMERGENCY CONDITIONS		
16.	Scram:		
	a.	Reactor overpressure with delayed scram, feedwater stays on, isolation valves stay open	1(4)
	b.	Loss of feedwater pumps, isolation valves closed	5
	c.	Automatic Blowdown	1(4)
	d.	Single safety or relief valve blowdown	8
17.	Improper start of cold recirculation loop		1(4)
18.	Sudden start of pump in cold recirculation loop		1(4)
19.	Improper startup with reactor drain shut off		1(4)
	FAULTED CONDITION		
20.	SSE at rated operating conditions		1(4)
21.	Pipe rupture and blowdown		1(4)

(1) Applies to RPV only.
(2) Bulk average vessel coolant temperature change in any 1-hour period.
(3) An environmental fatigue calculation provides the basis for reduced OBE cycle limits for the following piping systems: RHR Return and Supply piping – 20 cycles; Recirculation Drain piping – 20 cycles; Core Spray piping – 40 cycles (Unit 1), 30 cycles (Unit 2); and Reactor Recirculation piping – 30 cycles. All other piping has a limit of 50 OBE cycles. These administrative transient cycle limits are imposed to meet License Renewal Commitment T04740.
(4) The annual encounter probability of the one cycle events is $<10^{-2}$ for emergency and $<10^{-4}$ for faulted events.

Table 3.9-3

LIST OF COMPUTER PROGRAMS USED FOR NON-NSSS MECHANICAL SYSTEMS, COMPONENTS, AND COMPONENT SUPPORTS

COMPUTER PROGRAM No.	NAME	DOCUMENT TRACEABILITY	SYSTEM USED
ME101	Linear Elastic Analysis of Piping	Bechtel	UNIVAC 1100 series, Unix Workstation
ME632	Piping System Analysis	Bechtel	Honeywell 6000, UNIVAC 1100 series
ME912	Thermal Stress Program	Bechtel	UNIVAC 1100 series
ME913	Nuclear Class 1 Piping Stress Analysis	Bechtel	UNIVAC 1100 series
CE798	ANSYS	Swanson Analysis System, Inc. Elizabeth, Penn.	UNIVAC 1100 series
NE452	Reflood Analysis	Bechtel	UNIVAC 1100 series
NE805	Relief Valve	Bechtel Clearing Analysis	UNIVAC 1100 series
ME210	Local Stresses in Cylindrical Shells Due to External Loadings	Bechtel	UNIVAC 1100 series
ME602	Spectra Merging and Simplified Seismic Analysis	Bechtel	UNIVAC 1100 series
ME351	Pipe Rupture Analysis Program	Control Data Corporation	CDC CYBER
ME150	Frame Analysis Program for Pipe Support	Bechtel	UNIVAC 1100 series, VAX/VMS, UNIX Workstation
ME152	Standard Frame Analysis Program for Pipe Support	Bechtel	VAX/VMS, UNIX Workstation

Table 3.9-3 (Cont'd)

COMPUTER PROGRAM No.	NAME	DOCUMENT TRACEABILITY	SYSTEM USED
ME035	Base-Plate Analysis Program	Bechtel	UNIVAC 1100 series, UNIX Workstation
CE050	Concrete Expansion Anchor Bolt Program	Bechtel	UNIVAC 1100 series
CE901	Frame Analysis Program	Bechtel	UNIVAC 1100 series
ME225	Anchor Plate Program	Bechtel	UNIVAC 1100 series
ME226	Pipe Clamp Program	Bechtel	UNIVAC 1100 series
ME120	Weld Program	Bechtel	UNIVAC 1100 series
ME425	Strength Design of Pipe Support Anchor Bolt	Bechtel	UNIVAC 1100 series
ME153	Miscellaneous Application Program for Pipe Supports	Bechtel	UNIVAC 1100 series, VAX/VMS, UNIX Workstation
NUPIPE-SWPC	Linear Elastic Analysis of Piping	Stone and Webster	PC Workstation

Table 3.9-4

COMPARISON OF ME912 WITH ME643 AND ANALYTICAL RESULTS

CASE	PROGRAM	TEMPERATURE GRADIENTS[1] ΔT_1	ΔT_2	$T_a - T_b$[1]
450°F to 553°F Step	ME643	79.0	38.0	24.0
3 Inch Schedule 160, Stainless	ME912	79.7	40.6	24.3
Thicknesses 1.50:1	Reference 3.9-8	82.0	41.0	-
408°F to 100°F Step	ME643	136.2	40.1	83.0
12 Inch Schedule 80, Carbon	ME912	134.4	41.9	81.6
Steel Thicknesses 1.69:1	Reference 3.9-8	139.0	43.0	-

(1) Defined in the ASME B&PV Code, Section III, Subsection NB-3650.

Table 3.9-5

COMPARISON BETWEEN SAMPLE PROBLEM AND
COMPUTER PROGRAM ME913 RESULTS[1]

	ME 913	Sample Problem[2]
Equation 9	20,810 psi	20,825 psi
Equation 10	65,567 psi	65,596 psi
Equation 11	128,950 psi	128,920 psi
Equation 12	39,536 psi	39,564 psi
Equation 13	23,152 psi	23,155 psi
Total Usage Factor	0.3439	0.3699

(1) Comparison made for Butt-Welding Tee, Location 10.
(2) See Reference 3.9-14.

Table 3.9-6

LOADING COMBINATIONS, STRESS LIMITS, AND ALLOWABLE STRESSES

The following is a list of the tables that give the design loading combinations, allowable stresses, and calculated stresses for the major mechanical safety-related components in the plant and referenced in Section 3.9.

Table 3.9-6 (Cont'd)

3.9-6(s)	Reactor Refueling and Servicing Equipment
3.9-6(t)	HPCI Pump
3.9-6(u)	Control Rod Drive
3.9-6(v)	CRD Housing
3.9-6(w)	Jet Pumps
3.9-6(x)	Fuel Assembly (Including Channel)
3.9-6(y)	LPCI Coupling
3.9-6(z)	RPV Support Equipment; CRD Housing Support
3.9-6(aa)	Control Rod Guide Tube
3.9-6(ab)	Incore Housing

Table 3.9-6(a)

LOAD COMBINATION AND ACCEPTANCE CRITERIA FOR
ASME CLASS 1, 2, AND 3 NSSS PIPING, EQUIPMENT, AND SUPPORTS

LOAD COMBINATION	DESIGN BASIS	EVALUATION BASIS	SERVICE LEVEL
N + $SRV_{(ALL)}$	Upset	Upset	(B)
N + OBE	Upset	Upset	(B)
N + OBE + $SRV_{(ALL)}$	Emergency	Upset	(B)
N + SSE + $SRV_{(ALL)}$	Faulted	Faulted	(D)[1]
N + SBA + SRV	Emergency	Emergency	(C)[1]
N + SBA + $SRV_{(ADS)}$	Emergency	Emergency	(C)[1]
N + SBA/IBA + OBE + $SRV_{(ADS)}$	Faulted	Faulted	(D)[1]
N + SBA/IBA + SSE + $SRV_{(ADS)}$	Faulted	Faulted	(D)[1]
N + LOCA[2] + SSE	Faulted	Faulted	(D)[1]

LOAD DEFINITION LEGEND

N	-	Normal loads (e.g., weight, pressure, temperature, etc)
OBE	-	Operational basis earthquake loads
SSE	-	Safe shutdown earthquake loads
SRV	-	Safety/relief valve discharge induced loads from two adjacent valves (one valve actuated when adjacent valve is cycling)
SRV_{ALL}	-	Loads induced by actuation of all 14 safety/relief valves that activate within milliseconds of each other (e.g., turbine trip operational transient)

Table 3.9-6(a) (Cont'd)

SRV_{ADS}	-	Loads induced by the actuation of all 5 safety/relief valves associated with automatic depressurization system that actuate within milliseconds of each other during the postulated small or intermediate size pipe rupture.
LOCA	-	Loss-of-coolant accident associated with the postulated pipe rupture of large pipes (e.g., main steam, feedwater, recirculation piping)
$LOCA_1$	-	Pool-swell drag/fallback loads on piping and components located between the main vent discharge outlet and the suppression pool water upper surface
$LOCA_2$	-	Pool-swell impact loads on piping and components located above the suppression pool water upper surface
$LOCA_3$	-	Oscillating pressure induced loads on structures and equipment during condensation oscillation
$LOCA_4$	-	Oscillating pressure induced loads on structures and equipment during chugging
$LOCA_5$	-	Building motion induced loads from main vent air clearing
$LOCA_6$	-	Vertical and horizontal loads on main vent piping
$LOCA_7$	-	Annulus pressurization loads
SBA	-	Abnormal transients associated with a small break accident
IBA	-	Abnormal transients associated with an intermediate break accident.

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(1) All ASME Class 1, 2 and 3 piping that are required to function for safe shutdown under the postulated events are designed to meet the requirements described in NEDO-21985.

(2) The most limiting case of load combinations among $LOCA_1$ through $LOCA_7$.

Table 3.9-6(b)

REACTOR INTERNALS AND ASSOCIATED EQUIPMENT

ASME SECTION III, SUBSECTION NG PRIMARY STRESS LIMIT CRITERIA	LOAD CASE NUMBER[1]	PRIMARY STRESS TYPE	ALLOWABLE STRESS (psi)	MAXIMUM CALCULATED STRESS[3] (psi)
TOP GUIDE - HIGHEST STRESSED BEAM				
MATERIAL 304 S.S.				
A. NORMAL AND UPSET CONDITION:				
$P_m \le S_m$ $S_m = 16{,}900$ @ 550 ^{o} F	Normal and Upset Condition Loads: 1. Normal loads 2. Upset pressure 3. OBE 4. SRV	Primary membrane	16,900	1,889
$P_L + P_b \le 1.5\ S_m$ $1.5\ S_m = 25{,}350$ @ 550 ^{o} F		Primary membrane plus bending	25,350	17,735
B. EMERGENCY CONDITION:				
$P_m \le 1.5\ S_m$ $1.5\ S_m = 25{,}350$ @ 550 ^{o} F	Emergency Condition Loads: 1. Normal loads 2. Upset pressure 3. Chugging 4. SRV_1	Primary membrane	25,350	326
$P_L + P_b$ 2.25 S_m $2.25\ S_m = 38{,}025$ @ 550 ^{o} F[2]		Primary membrane plus bending	38,025	12,192

Table 3.9-6(b) (Cont'd)

ASME SECTION III, SUBSECTION NG PRIMARY STRESS LIMIT CRITERIA	LOAD CASE NUMBER(1)	PRIMARY STRESS TYPE	MAXIMUM ALLOWABLE STRESS (psi)	CALCULATED STRESS(3) (psi)
C. FAULTED CONDITION:				
$P_m \leq 2.4\ S_m$ $2.4\ S_m = 40{,}560$ @ 550 °F	Faulted Condition Loads: 1. Normal loads 2. Accident pressure 3. SSE 4. Jet reaction 5. Delta P	Primary membrane	40,560	3,383
$P_L + P_b \leq 3.0\ S_m$ $3.0\ S_m = 50{,}700$ @ 550 °F(2)	1. Normal loads 2. Accident pressure 3. SSE 4. SRV_1 5. Chugging	Primary membrane plus bending	50,700	34,412

D. MAXIMUM CUMULATIVE USAGE FACTOR: 0.901 at beam slot location

Table 3.9-6(b) (Cont'd)

ASME SECTION III, SUBSECTION NG PRIMARY STRESS LIMIT CRITERIA	LOAD CASE NUMBER[(1)]	PRIMARY STRESS TYPE	MAXIMUM ALLOWABLE STRESS (psi)	CALCULATED STRESS[(3)] (psi)
CORE PLATE (LIGAMENT IN TOP PLATE)				
MATERIAL: 304 S.S.				
A. NORMAL AND UPSET CONDITION				
$P_m \leq S_m$ $S_m = 16{,}900$ @ 550° F	Normal and Upset Condition Loads: 1. Normal loads 2. Upset pressure 3. OBE 4. SRV	Primary membrane	16,900	8,580
$P_L + P_b \leq 1.5\ S_m$ $1.5\ S_m = 25{,}350$ @ 550°F		Primary membrane plus bending	25,350	15,270
B. EMERGENCY CONDITION:				
$P_m \leq 1.5\ S_m$ $1.5\ S_m = 25{,}350$ @ 550°F	Emergency Condition Loads: 1. Normal loads 2. Upset pressure 3. Chugging 4. SRV_{ADS}	Primary membrane	25,350	14,300
$P_L + P_b \leq 2.25\ S_m$ $2.25\ S_m = 38{,}030$ @ 550°F[(2)]		Primary membrane plus bending	38,030	7,050

Table 3.9-6(b) (Cont'd)

ASME SECTION III, SUBSECTION NG PRIMARY STRESS LIMIT CRITERIA	LOAD CASE NUMBER(1)	PRIMARY STRESS TYPE	MAXIMUM ALLOWABLE STRESS (psi)	CALCULATED STRESS(3) (psi)
C. FAULTED CONDITION:				
$P_m \leq 2.4\ S_m$	Faulted Condition Loads:	Primary membrane	40,560	15,650
$2.4\ S_m = 40,560$ @ 550^o F	1. Normal loads 2. Accident pressure 3. Jet reaction 4. SSE 5. Delta P			
$P_L + P_b \leq 3\ S_m$		Primary membrane plus bending	50,700	25,650
$3\ S_m = 50,700$ @ 550^o F(2)				
D. MAXIMUM CUMULATIVE USAGE FACTOR: 0.257 at Core plate stud				

Table 3.9-6(b) (Cont'd)

ASME SECTION III, SUBSECTION NG PRIMARY STRESS LIMIT CRITERIA	LOAD CASE NUMBER[1]	PRIMARY STRESS TYPE	MAXIMUM ALLOWABLE STRESS (psi)	CALCULATED STRESS[3] (psi)
DIFFERENTIAL PRESSURE AND LIQUID CONTROL LINES				
MATERIAL: 304 S.S.				
A. NORMAL AND UPSET CONDITION:				
$P_m \leq S_m$	Normal and Upset Condition Loads:			
S_m = 16,950 @ 550°F	1. OBE 2. SRV			
$P_L + P_b \leq 3\ S_m$		Primary membrane plus bending plus secondary membrane	49,200	8,654
$3\ S_m$ = 50,850 @ 550°F				
B. EMERGENCY CONDITION:				
$P_m \leq S_m$	Emergency Condition Loads:			
S_m = 16,950 @ 550°F	1. OBE 2. SRV			
$P_L + P_b \leq 2.25\ S_m$		Primary membrane plus bending	36,900	8,654
$2.25\ S_m$ = 37,120 @ 550°F[2]				
C. FAULTED CONDITION:				
$P_m \leq S_m$	Faulted Condition Loads:			
S_m = 16,950 @ 550°F	1. Annulus pressurization 2. SSE			
$P_L + P_b \leq 3.6\ S_m$		Primary membrane plus bending plus secondary membrane	59,040	15,106
$3.6\ S_m$ = 61,020 @ 550°F[2]				

(1) Load cases are defined in Table 3.9-6.
(2) Value of S_m or S_y is shown depending on the controlling criteria (e.g., 1.8 S_m or 1.5 S_y for B)
(3) The loads listed here are associated with the operating level of 3458 MWt. Per Reference 3.9-32, the loads on the differential pressure and liquid control lines are bounding at MUR power uprate condition; the loads on the top guide and core plate are increased but within the allowable limits. Increased loads due to the revised Fuel Lift Margin and CRGT lift forces under MUR conditions are addressed in Reference 3.9-33.

Table 3.9-6(c)

RWCU HEAT EXCHANGERS

REGENERATIVE RWCU HX

Part	Thickness Required (in)	Allowable Stress(psi)	Actual Thickness (in)
Shell	0.779	15900	0.875
Shell head	0.745	15900	0.760
Channel shell	0.780	15900	2.6875
Tube sheet	3.087	15900	3.25
Tubes	0.0427	11950	0.0524
Piping	0.195	15900	0.337
Channel cover	3.09	17500	3.25

NONREGENERATIVE RWCU HX

Part	Thickness Required (in)	Allowable Stress(psi)	Actual Thickness (in)
Shell	0.1171	15000	0.375
Shell head	0.117	17500	0.375
Channel shell	0.7814	15900	2.6815
Channel cover	3.09	17500	3.25
Tube sheet	3.087	13900	3.25
Tubes	0.0561	11950	0.0585
Channel piping	0.185	15900	0.337
Shell piping	0.0608	15900	0.280

Table 3.9-6(d)

ASME CODE CLASS 1 MAIN STEAM PIPING AND PIPE-MOUNTED EQUIPMENT - HIGHEST STRESS SUMMARY

(UNIT 1)

Acceptance Criteria	Limiting Stress Type	Calculated Stress or Usage Factor	Allowable Limits	Ratio Actual/ Allowable	Loading	Identification of Locations of Highest Stress Points - NODG Point Numbers
ASME Section III, NB-3600						
Design Condition: Eq. 9 ≤ 1.5 S_m	Primary	15,765	26,550	0.594	1. Pressure 2. Weight 3. OBE	Steam Line D Riser lug (009)
Service Levels A & B (Normal & Upset) Condition:						
Eq. 12 ≤ 3.0 S_m	Secondary	32,318	54,600	0.59	1. Thermal expansion	Steam Line C Sweepolet (059)
Service Levels A & B (Normal & Upset) Condition: Eq. 13 ≤ 3.0 S_m	Primary Plus Secondary (Except Thermal Expansion)	54,509	54,600	0.99	1. Pressure 2. Weight 3. OBE 4. Temperature discontinuity	Steam Line A Sweepolet (063)
Service Levels A & B (Normal & Upset) Conditions:						
Cumulative Usage Factor	N.A.	0.457	1.0		N/A	Steam Line A Sweepolet (063)
Service Level B (Upset) Condition: Eq. 9 ≤ 1.8 S_m & 1.5 S_y	Primary	22,453	32,760	0.69	1. Pressure 2. Weight 3. OBE 4. SRV (Acoustic wave)	Steam Line D Sweepolet (060)

Table 3.9-6(d) (Cont'd)

(Unit 1)

Acceptance Criteria	Limiting Stress Type	Calculated Stress or Usage Factor	Allowable Limits	Ratio Actual/ Allowable	Loading	Identification of Locations of Highest Stress Points - NODG Point Numbers
Service Level C (Emergency) Condition:					1. Pressure 2. Weight 3. OBE	Steam Line B Sweepolet (500)
Eq. 9 < 2.25 S_m & 1.8 S_y	Primary	22,050	40,950	0.54	4. Chugging	
Service Level D (Faulted) Condition:					1. Pressure 2. Weight 3. SSE	Steam Line B Sweepolet (500)
Eq. 9 < 3.0 S_m	Primary	48,775	54,600	0.89	4. Annulus pressurization	

Table 3.9-6(d) (Cont'd)

(Unit 1)

Component/ Load Type	Highest Calculated Load	Allowable Load	Ratio Calculated/ Allowable	Loading	Identification of Equipment with Highest Loads
Snubber					
Service Level B	44,988	50,000	0.900	OBE + SRV	Steam Line A Snubber - SA9
Service Level C	25,555	66,500	0.384	Chugging + SRV (acoustic wave)	Steam Line B Snubber - SB2
Service Level D	72,874	75,000	0.972	Annulus Pressurization + SSE	Steam Line A Snubber - SA9
Accelerations					
Horizontal Level D	6.314g	6.5g	0.9714	SSE + Condensation + SRV (acoustic wave)	Steam Line D SRV Inlet (086)
Vertical Level D	4.20g	6.0g	0.7	SSE + Condensation + SRV (acoustic wave)	Steam Line B SRV Inlet (075)

Table 3.9-6(d) (Cont'd)

ASME CODE CLASS 1 MAIN STEAM PIPING AND PIPE-MOUNTED EQUIPMENT - HIGHEST STRESS SUMMARY

(UNIT 2)

Acceptance Criteria	Limiting Stress Type	Calculated Stress or Usage Factor	Allowable Limits	Ratio Actual/ Allowable	Loading	Identification of Locations of Highest Stress Points - NODG Point Numbers
ASME Section III, NB-3650						
Design Condition: Eq. 9 $\leq$ 1.5 S_m	Primary	21,333	26,550	0.78	1. Pressure 2. Weight 3. OBE	Steam Line A Riser lug (063)
Service Levels A & B (Normal & Upset) Condition:						
Eq. 12 $\leq$ 3.0 S_m	Secondary	32,318	54,600	0.59	1. Thermal expansion	Steam Line C Sweepolet (059)
Service Levels A & B (Normal & Upset) Condition: Eq. 13 $\leq$ 3.0 S_m	Primary Plus Secondary (Except Thermal Expansion)	54,509	54,600	0.99	1. Pressure 2. Weight 3. OBE 4. Temperature discontinuity	Steam Line A Sweepolet (063)
Service Levels A & B (Normal & Upset) Conditions:						
Cumulative Usage Factor	N.A.	0.457	1.0		N/A	Steam Line A Sweepolet (063)

Table 3.9-6(d) (Cont'd)

(Unit 2)

Acceptance Criria	Limiting Stress Type	Calculated Stress or Usage Factor	Allowable Limits	Ratio Actual/ Allowable	Loading	Identification of Locations of Highest Stress Points - NODG Point Numbers
Service Level B (Upset) Condition:					1. Pressure 2. Weight 3. OBE	Steam Line D Sweepolet (060)
Eq. 9 $\leq$ 1.8 S_m & 1.5 S_y	Primary	22,453	32,760	0.69	4. SRV (Acoustic wave)	
Service Level C (Emergency) Condition:					1. Pressure 2. Weight 3. OBE	Steam Line B Sweepolet (500)
Eq. 9 < 2.25 S_m & 1.8 S_y	Primary	22,050	40,950	0.54	4. Chugging	
Service Level D (Faulted) Condition:					1. Pressure 2. Weight 3. SSE	Steam Line B Sweepolet (500)
Eq. 9 < 3.0 S_m	Primary	48,715	54,600	0.89	4. Annulus pressurization	

Table 3.9-6(d) (Cont'd)

(Unit 2)

Component/ Load Type	Highest Calculated Load	Allowable Load	Ratio Calculated/ Allowable	Loading	Identification of Equipment with Highest Loads
Snubber					
Service Level B	44,988	50,000	0.900	OBE + SRV	Steam Line A Snubber - SA9
Service Level C	25,555	66,500	0.384	Chugging + SRV (acoustic wave)	Steam Line B Snubber - SB2
Service Level D	72,874	75,000	0.972	Annulus Pressurization + SSE	Steam Line A Snubber - SA9
Accelerations					
Horizontal Level D	6.314g	6.5g	0.9714	SSE + Condensation + SRV (acoustic wave)	Steam Line D SRV Inlet (086)
Vertical Level D	4.20g	6.0g	0.7	SSE + Condensation + SRV (acoustic wave)	Steam Line B SRV Inlet (075)

Table 3.9-6(e)

ASME CODE CLASS 1 RECIRCULATION PIPING AND PIPE-MOUNTED EQUIPMENT - HIGHEST STRESS SUMMARY

(UNIT 1)

Acceptance Criteria	Limiting Stress Type	Calculated Stress or Usage Factor	Ratio Allowable Limits	Actual/ Allowable	Loading	Identification of Locations of Highest Stress Points - NODG Point Numbers[1]
ASME Section III, NB-3650						
Design Condition: Eq. 9 $\leq$ 1.5 S_m	Primary	17,248	20,588	0.84	1. Pressure 2. Weight 3. OBE	Node (028) Recirculation Loop B (Hanger Lugs)
Service Levels A & B (Normal & Upset) Condition: Eq. 12 $\leq$ 3.0 S_m	Secondary	17,745	51,750	0.34	1. Pressure 2. Weight 3. Thermal expansion 4. OBE 5. SRV (structural feedback)	Node (500) Recirculation Loop A (Elbow)
Service Levels A & B (Normal & Upset) Condition: Eq. 13 $\leq$ 3.0 S_m	Primary plus secondary (except thermal expansion)	41,677	51,750	0.81	1. Pressure 2. Weight 3. OBE 4. SRV (structural feedback)	Node (200) Recirculation Loop B (Sweepolet)
Service Levels A & B (Normal and Upset) Condition:						
Cumulative Usage Factor	N.A.	0.25	1.0	0.25		

Table 3.9-6(e) (Cont'd)

(Unit 1)

Acceptance Criteria	Limiting Stress Type	Calculated Stress or Usage Factor	Ratio Allowable Limits	Actual/ Allowable	Loading	Identification of Locations of Highest Stress Points - NODG Point Numbers(1)
Service Level B (Upset) Condition: Eq. 9 ≤ 1.8 S_m & 1.5 S_y	Primary	19,227	23,472	0.82	1. Pressure 2. Weight 3. OBE 4. SRV (structural feedback)	Node (016) Recirculation Loop B (Small Tee)
Service Level C (Emergency) Condition: Eq. 9 ≤ 2.25 S_m & 1.8 S_y	Primary	18,886	28,166	0.67	1. Pressure 2. Weight 3. Chugging 4. SRV (structural feedback)	Node (016) Recirculation Loop B (Small Tee)
Service Level D (Faulted) Condition: Eq. 9 ≤ 3.0 S_m	Primary	24,895	31,296	0.80	1. Pressure 2. Weight 3. SSE 4. Annulus pressurization	Node (016) Recirculation Loop B (Small Tee)

Table 3.9-6(e) (Cont'd)

(Unit 1)

Component/ Load Type	Highest Calculated Load	Allowable Load	Ratio Calculated/ Allowable	Loading	Identification of Equipment with Highest Loads
Snubber Level B	104,265 lb	120,000 lb	0.87	OBE + SRV	Recirculation Loop A (SA 2)
Snubber Level C	20,022 lb	66,500 lb	0.301	Chugging + SRV	Recirculation Loop B (SB 9)
Snubber Level D	71,583 lb	75,000 lb	0.954	Annulus Pressurization + SSE	Recirculation Loop B (SB 9)
Suction Valve Level B	428,263 in-lb	1,747,285 in-lb	0.25	1. Weight 2. Thermal expansion 3. OBE 4. SRV	Recirculation Loop A (suction valve)
Discharge Valve Level C	284,213 in-lb	1,747,285 in-lb	0.16	1. Weight 2. Thermal expansion 3. OBE 4. SRV	Recirculation Loop B (discharge valve)
Discharge Valve Level D	1,438,256 in-lb	1,747,285 in-lb	0.82	1. Weight 2. Thermal expansion 3. Annulus pressurization 4. SSE	Recirculation Loop B (discharge valve)

Table 3.9-6(e) (Cont'd)

ASME CODE CLASS 1 RECIRCULATION PIPING AND PIPE-MOUNTED EQUIPMENT - HIGHEST STRESS SUMMARY

(UNIT 2)

Acceptance Criteria	Limiting Stress Type	Calculated Stress or Usage Factor	Allowable Limits	Ratio Actual/ Allowable	Loading	Identification of Locations of Highest Stress Points - NODG Point Numbers[1]
ASME Section III, NB-3650						
Design Condition: Eq. 9 $\leq$ 1.5 S_m	Primary	18,549	25,875	0.72	1. Pressure 2. Weight 3. OBE	Node (028) Recirculation Loop A (Buttweld)
Service Levels A & B (Normal & Upset) Condition: Eq. 12 $\leq$ 3.0 S_m	Secondary	40,919	51,750	0.79	1. Pressure 2. Weight 3. Thermal expansion 4. OBE 5. SRV (structural feedback)	Node (256) Recirculation Loop A (Transition)
Service Levels A & B (Normal & Upset) Condition: Eq. 13 $\leq$ 3.0 S_m	Primary plus secondary (except thermal expansion)	41,667	51,750	0.81	1. Pressure 2. Weight 3. OBE 4. SRV (structural feedback)	Node (200) Recirculation Loop B (Sweepolet)
Service Levels A & B (Normal and Upset) Condition:						
Cumulative Usage Factor	N.A.	0.098	1.0	0.098		

Table 3.9-6(e) (Cont'd)

(Unit 2)

Acceptance Criteria	Limiting Stress Type	Calculated Stress or Usage Factor	Ratio Allowable Limits	Actual/ Allowable	Loading	Identification of Locations of Highest Stress Points - NODG Point Numbers[1]
Service Level B (Upset) Condition:					1. Pressure 2. Weight 3. OBE	Node (016) Recirculation Loop B
Eq. 9 ≤ 1.8 S_m & 1.5 S_y	Primary	19,227	23,472	0.82	4. SRV (structural feedback)	Small Tee)
Service Level C (Emergency) Condition:					1. Pressure 2. Weight 3. Chugging	Node (016) Recirculation Loop B
Eq. 9 ≤ 2.25 S_m & 1.8 S_y	Primary	18,886	28,166	0.67	4. SRV (structural feedback)	(Small Tee)
Service Level D (Faulted) Condition:					1. Pressure 2. Weight 3. SSE	Node (016) Recirculation Loop B
Eq. 9 ≤ 3.0 S_m	Primary	24,895	31,296	0.80	4. Annulus pressurization	(Small Tee)

Table 3.9-6(e) (Cont'd)

(Unit 2)

Component/ Load Type	Highest Calculated Load	Allowable Load	Ratio Calculated/ Allowable	Loading	Identification of Equipment with Highest Loads
Snubber Level B	52,023 lb	120,000 lb	0.433	OBE + SRV	Recirculation Loop B (SB 2)
Snubber Level C	24,339 lb	159,600 lb	0.153	Chugging + SRV	Recirculation Loop B (SB 2)
Snubber Level D	86,272 lb	180,000 lb	0.479	Annulus Pressurization + SSE	Recirculation Loop B (SB 2)
Discharge Valve Level B	265,030 in-lb	1,747,285 in-lb	0.152	1. Weight 2. Thermal expansion 3. OBE 4. SRV	Recirculation Loop B (discharge valve)
Suction Valve Level C	250,422 in-lb	1,747,285 in-lb	0.143	1. Weight 2. Thermal expansion 3. OBE 4. SRV	Recirculation Loop B (suction valve)
Suction Valve Level D	535,499 in-lb	1,747,285 in-lb	0.307	1. Weight 2. Thermal expansion 3. Annulus pressurization 4. SSE	Recirculation Loop B (suction valve)

(1) Refer to Figure 3.6-4 for the identification of node point numbers.

Table 3.9-6(f)

RPV AND SHROUD SUPPORT ASSEMBLY

ASME SECTION III, SUBSECTION NB PRIMARY STRESS LIMIT CRITERIA	LOAD CASE NUMBER[(1)]	PRIMARY STRESS TYPE	MAXIMUM ALLOWABLE STRESS (psi)	CALCULATED STRESS[(4)] (psi)
<u>VESSEL SUPPORT SKIRT</u>				
MATERIAL: SA516, Grade 70				
A. NORMAL AND UPSET CONDITION:				
$P_m \leq S_m$ S_m = 19,150 @ 575°F	Normal and Upset Condition Loads: 1. Normal loads 2. Upset pressure 3. OBE 4. SRV	Primary membrane	19,150	14,723
$P_L + P_b \leq 1.5\ S_m$ 1.5 S_m = 28,725 @ 575°F		Primary membrane plus bending	28,725	20,640
B. EMERGENCY CONDITION:				
$P_m \leq S_y$ S_y = 28,425 @ 546°F	Emergency Condition Loads: 1. Normal loads 2. Upset pressure 3. OBE 4. Chugging 5. SRV_{ADS}	Primary membrane	29,425	20,565
$P_L + P_b \leq 1.5\ S_y$ 1.5 S_y = 44,137 @ 546°F[(2)]		Primary membrane plus bending	44,150	29,377

Table 3.9-6(f) (Cont'd)

ASME SECTION III, SUBSECTION NB PRIMARY STRESS LIMIT CRITERIA	LOAD CASE NUMBER[1]	PRIMARY STRESS TYPE	MAXIMUM ALLOWABLE STRESS (psi)	CALCULATED STRESS[4] (psi)
C. FAULTED CONDITION:				
$P_m \leq 2.4\ S_m$ $2.4\ S_m = 45{,}960$ @ 575°F	Faulted Condition Loads: 1. Normal loads 2. Accident pressure 3. Jet reaction 4. VC 5. SSE	Primary membrane	45,960	30,436
$P_m + P_b \leq 3.6\ S_m$ $3.6\ S_m = 68{,}940$ @ 575°F[2]	1. Normal loads 2. Accident pressure 3. Chugging 4. SRV_{ADS} 5. SSE	Primary membrane plus bending	68,940	45,044

D. MAXIMUM CUMULATIVE USAGE FACTOR: 0.83 at Skirt base junction

Table 3.9-6(f) (Cont'd)

ASME SECTION III, SUBSECTION NB PRIMARY STRESS LIMIT CRITERIA	LOAD CASE NUMBER[1]	PRIMARY STRESS TYPE	MAXIMUM ALLOWABLE STRESS (psi)	CALCULATED STRESS[4] (psi)
<u>SHROUD SUPPORT</u>				
MATERIAL: SB168 Inconel				
A. NORMAL AND UPSET CONDITION:				
$P_m \leq S_m$	Normal and Upset Condition Loads:	Primary membrane	23,300	21,700
S_m = 23,300 @ 575°F	1. Dead weight 2. OBE			
$P_L + P_b \leq 1.5\ S_m$		Primary membrane plus bending	34,950	33,640
1.5 S_m = 34,950 @ 575°F				
B. EMERGENCY CONDITION:				
$P_m \leq S_y$	Emergency Condition Loads:	Primary membrane	28,120	25,440
S_y = 28,120 @ 575°F	1. Dead weight 2. SSE 3. Jet reaction			
$P_L + P_b \leq 1.5\ S_y$		Primary membrane plus bending	42,180	40,360
1.5 S_y = 42,180 @ 575°F[2]				

Table 3.9-6(f) (Cont'd)

ASME SECTION III, SUBSECTION NB PRIMARY STRESS LIMIT CRITERIA	LOAD CASE NUMBER[1]	PRIMARY STRESS TYPE	MAXIMUM ALLOWABLE STRESS (psi)	CALCULATED STRESS[4] (psi)
C. FAULTED CONDITION:				
$P_m \leq S_y$ S_y = 28,120 @ 575°F	Faulted Condition Loads: 1. Dead weight 2. SSE 3. Jet reaction	Primary membrane	28,120	25,440
$P_L + P_b \leq 1.5\ S_y$ 1.5 S_y = 42,180 @ 575?F[2]		Primary membrane plus bending	42,180	40,360

D. MAXIMUM CUMULATIVE USAGE
FACTOR: 0.373 at point 24 in shroud support plate

Table 3.9-6(f) (Cont'd)

ASME SECTION III, SUBSECTION NB PRIMARY STRESS LIMIT CRITERIA	LOAD CASE NUMBER[1]	PRIMARY STRESS TYPE	MAXIMUM ALLOWABLE STRESS (psi)	CALCULATED STRESS[4] (psi)
RPV FEEDWATER NOZZLE				
MATERIAL: SA508 Class 1 Safe-end				
A. NORMAL AND UPSET CONDITION				
$P_m \leq S_m$ S_m = 17,700 @ 575°F	Normal and Upset Condition Loads: 1. Normal loads 2. Upset pressure 3. OBE 4. SRV	Primary membrane	17,700	16,220
$P_L + P_y \leq 1.5\ S_m$ 1.5 S_m = 26,550 @ 575°F		Primary membrane plus bending	26,550	22,930
B. EMERGENCY CONDITION				
$P_m \leq S_y$ S_y = 25,900 @ 594°F	Emergency Condition Loads: 1. Normal loads 2. Upset pressure 3. Chugging 4. SRV_{ADS}	Primary membrane	25,900	21,420
$P_L + P_b \leq 1.5\ S_y$ 1.5 S_y = 38,900 @ 594°F[2]		Primary membrane plus bending	38,850	22,400

Table 3.9-6(f) (Cont'd)

ASME SECTION III, SUBSECTION NB PRIMARY STRESS LIMIT CRITERIA	LOAD CASE NUMBER[(1)]	PRIMARY STRESS TYPE	MAXIMUM ALLOWABLE STRESS (psi)	CALCULATED STRESS[(4)] (psi)
C. FAULTED CONDITION				
$P_m \leq 2.4\ S_m$ $2.4\ S_m = 42{,}480$ @ 575°F	Faulted Condition Loads: 1. Normal loads 2. Accident pressure 3. Chugging 4. SSE 5. SRV_{ADS}	Primary membrane	42,480	23,210
$P_L + P_b \leq 1.5\ S_y$ $1.5\ S_y = 38{,}900$ @ 594°F[(2)]		Primary membrane plus bending	38,850	33,740
D. MAXIMUM CUMULATIVE USAGE FACTOR: 0.9957 at Safe-end				

Table 3.9-6(f) (Cont'd)

ASME SECTION III, SUBSECTION NB PRIMARY STRESS LIMIT CRITERIA	LOAD CASE NUMBER[1]	PRIMARY STRESS TYPE	MAXIMUM ALLOWABLE STRESS (psi)	CALCULATED STRESS[4] (psi)
CRD PENETRATION (Stub Tube)				
MATERIAL: SB167 - Inconel				
A. NORMAL AND UPSET CONDITION:				
$P_m \leq S_m$ S = 20,000 @ 575°F	Normal and Upset Condition Loads: 1. Normal leads 2. Upset pressure 3. OBE 4. SRV	Primary membrane	20,000	5,005
$P_L + P_b \leq 1.5\ S_m$ $1.5\ S_m$ = 30,000 @ 575°F		Primary membrane plus bending	30,000	28,200
B. EMERGENCY CONDITION:				
$P_m \leq S_y$ S_y = 24,100 @ 575°F	Emergency Condition Loads: 1. Normal loads 2. Upset pressure 3. Chugging 4. SRV_{ADS}	Primary membrane	24,100	6,755
$P_L + P_b \leq 1.5\ S_y$ $1.5\ S_y$ = 36,150 @ 575°F[2]		Primary membrane plus bending	36,150	30,260

Table 3.9-6(f) (Cont'd)

ASME SECTION III, SUBSECTION NB PRIMARY STRESS LIMIT CRITERIA	LOAD CASE NUMBER[(1)]	PRIMARY STRESS TYPE	MAXIMUM ALLOWABLE STRESS (psi)	CALCULATED STRESS (psi)
C. FAULTED CONDITION:				
$P \leq 2.4\ S_m$ $2.4\ S_m = 48{,}000$ @ 575°F	Faulted Condition Loads: 1. Normal loads 2. Accident pressure 3. Jet reaction 4. Scram 5. SSE	Primary membrane	48,000	7,287
$P_L + P_b \leq 3.6\ S_m$ $3.6\ S_m = 72{,}000$ @ 575°F[(2)]		Primary membrane plus bending	72,000	30,260
D. MAXIMUM CUMULATIVE USAGE FACTOR: 0.153 at Stub tube				

(1) Load cases are defined in Table 3.9-6.
(2) Value of S_m or S_y is shown depending on the controlling criteria (e.g., 1.8 S_m or 1.5 S_y for B).
(3) Maximum calculated values are based on design loads because they are greater than new loads.
(4) The loads here correspond to the operating power level of 3458 MWt. Per Reference 3.9-31, the loads for the vessel support skirt, feedwater nozzle and the CRD penetration are not changed as a result of MUR power uprate. Per Reference 3.9-32, the loads on the shroud support are bounding at MUR power uprate condition.

Table 3.9-6(g)

MAIN STEAM SAFETY/RELIEF VALVES (PILOT-OPERATED)
(ASME SECTION III, 1968, Including Addenda through Summer 1970)

Topic	Method of Analysis	Target Rock 9867F Analysis	Allowable Value	Calculated
1. Body inlet and outlet flange stresses <u>Note, Topics 1 and 2:</u> Design Pressures: P_d = 1250 psig (inlet) P_b = 500 psig (outlet) Analyses include applied moments of M = 409,000 in-lb (inlet) and M = 372,000 in-lb (outlet) Actual tested capability (including accelerations and moments) is as described in Topic No. 11. The analyses also include consideration of seismic, operational, and flow reaction forces. Allowable vs. tested capabilities are provided in Topic No. 12	$S_H = \frac{fM_o}{Lg^2B} + \frac{PB}{4g_o} < 1.5\ S_m$ $S_R = \frac{(4te/3+1)M_o}{Lt^2B} < 1.5\ S_m$ $S_T = \frac{YM_o}{t^2B} - Z\ S_R < 1.5\ S_m$ where: S_H = Longitudinal "hub" wall stress, psi S_R = Radial "flange" stress, psi S_T = Tangential "flange" stress, psi Body Material: A105 Grade II S_m = 19,400 psi (500°F, equivalent inlet and outlet temperature)	P_D (Target Rock) = P (codes)	1.5 S_m = 29,100 psi	<u>Inlet:</u> S_H = 27,656 psi S_R = 15,295 psi S_T = 14,257 psi <u>Outlet:</u> S_H = 15,591 psi S_R = 23,614 psi S_T = 557 psi

Table 3.9-6(g) (Cont'd)

Topic	Method of Analysis	Target Rock 9867F Analysis	Allowable Value	Calculated
2. Inlet and outlet stud area requirements	Total cross-sectional area excees the greater of: $Am_1 = \frac{Wm_1}{S_b}$, or $Am_2 = \frac{Wm_2}{S_a}$, where: Am_1 = total required bolt (stud) area for operating condition Am_2 = total required bolt (stud) area for gasket seating	$Am_1 = \frac{Wm_1}{S_b}$ $Am_2 = \frac{Wm_2}{S_a}$ * Bolting Material: SA193 Grade B7 *Where AM (required minimum) is the greater of Am_1 and Am_2; and A_b (actual bolt area) must exceed Am.	<u>Inlet</u>: Am_1 ($>Am_2$) = 8.15 in^2 <u>Outlet:</u> Am = 5.39 in^2	<u>Inlet:</u> A_b (actual area) = 13.85in^2 <u>Outlet:</u> A_b = 9.68 in^2
3. Body Wall thickness	1. Valve Wall Thickness Criterion: $t_{min.} < t_A$ where: $t_{min.}$ = minimum calculated thickness requirement, including corrosion allowance. t_A = Actual wall thickness (NOTE: This $t_{min.}$ is t_m per notation of the codes.)	Section at inlet: $t_{RQD} < t_{ACT}$ Section at middle of body $t_{RQD_c} < t_{ACT}$ Section at outlet: $t_{RQD} < t_{ACT}$ Section at neck: $t_{RQD} < t_{ACT}$	t_{RQD} = 0.7 in Actual thickness greater than t_m at the section under consideration.	t_{ACT} = 1.098 in. t_{ACT_c} = 0.805 in. t_{ACT} = 1.359 in. t_{ACT} = 0.953 in.

Table 3.9-6(g) (Cont'd)

Topic	Method of Analysis	Target Rock 9867F Analysis	Allowable Value	Calculated
3. (Cont'd)	Cycle Rating: Thermal $I_t = \Sigma \frac{Nri}{Ni}$	$I_t = \Sigma \frac{Nri}{Ni}$ (i = 1, 2 & 3)	I_t (max.) $\leq$ 1.0	I_t = 0.518
	Fatigue Na > 2,000 cycles, as based on S_a, where S_a is defined as the larger of $S_{p_1} = (2/3)Q_p + \frac{P_{eb}}{2} + Q_{T_2} + 1.3Q_{T_1}$ or $S_{P_2} = 0.4\ Q_p + \frac{K}{2}(P_{eb} + 2Q_{T_2})$ where: Sp_1 = Fatigue stress intensity at inside surface of crotch psi Sp_2 = Fatigue stress intensity outside surface of crotch psi	Na ≥ 2,000 cycles as based on $S_{A_2} = S_{p_3}$ (>S_p), where S_A (Target Rock) = S_a (codes) Uses same notation as codes	Na ≥ 2,000 cycles	Na (based on S_{p_2}) = 1.5×10^5 cycles ∴ satisfies criterion

Table 3.9-6(g) (Cont'd)

Topic	Method of Analysis	Target Rock 7567F Analysis	Allowable Value	Calculated
4. Body to pilot base Flange Stresses (body side)	$S_{H_1} = \frac{PB_1}{4g_1} \pm \frac{6M_H}{\pi B_1 g_1^2}$ (longitude hub stress adjacent to flange) $S_{H_2} = \left(\frac{Q}{\pi B_1 t} + P\right)(Z + Y)$ $+ \frac{Et\theta_B}{B_1}$ $+ \frac{0.075\ PB_1}{g_1} \pm \frac{1.8\ M_H}{\pi B_1 g_1^2}$ (circumferential stress in hub adjacent to flange) $S_R = \frac{6(M_p + M_s)}{t^2(\pi C\text{-}nD)}$ (@ Bolt circle) $S_R = \left(\frac{Q}{\pi B_1 t} + P\right) \pm \frac{6M_s}{\pi B_1 t^2}$ (adjacent to hub) $ST = \left(\frac{Q}{\pi B_1 t} + P\right)Z$ $\pm \left(\frac{Et\theta_B}{B_1} + \frac{1.8M_S}{\pi B_1 t^2}\right)$	$S_H < 1.5\ S_m$ $S_R < 1.5\ S_m$ $S_T < 1.5\ S_m$ P_{FD} (Target Rock) = P (codes) Material: SA-105 Grade II S_m = 19,400 psi (@ 500°F)	1.5 S_m = 29,100 psi	S_H = 19,590 psi S_R = 11,851 psi S_T = 6,364 psi
Bonnet Flange Stresses (Bonnet side)	Base side flange stresses less than body side flange stresses (Both sides see identical loads. The base side is thicker than the body side at all points.)	Same as Topic 4 Analysis	1.5 S_m = 29,000 psi	S_H 19,590 psi S_R 11,851 psi S_T 6,364 psi

Table 3.9-6(g) (Cont'd)

Topic	Method of Analysis	Target Rock 9867F Analysis	Allowable Value	Calculated
6. Base of pilot body stud area requirements	Total cross-sectional area shall exceed: $Am_1 = \frac{Wm_1}{S_b}$ where: Am_1 = Total required bolt (stud) area.	$Am_1 = \frac{Wm_1}{S_b}$ Bolting Material: SA193 Grade B16	Am_1 = 7.10 in^2	A_b (actual) = 9.04 in^2
7. Pilot body wall thickness	Body Wall Per ASME Section VIIII $T_m = \frac{P R_i}{S_m - 0.5 P}$	$t_m < t_a$ Material: SB-166 Grade 600 Sm = 23,300 psi (@500°F)	Bonnet Wall t_m = .085 in	Bonnet Wall t_a = 3.22 in.
10. Main disc stress	Using Roark's formulas for stress and strain, 4th edition, page 250 $S_{max} = \frac{\beta Wa^2}{t_o^2}$ where: $\beta = 1.63$ w = applied load a = radius of disc t_o = thickness at center	$S_{max} < S_m$ Material: SA182 S_m = 13,000 psi (@ 500°F)	S_m = 13,600 psi 1.55m = 19,500 psi	S_{max} = 18,180 psi
11. Seismic Capability: Stress analysis uses $F_{vertical}$ = (mass of value) • (2.0 g) and $F_{horizontal}$ = (mass of value) • (3.0 g), with concurrent 409,000 in-lb and 372,000 in-lb applied at the inlet and outlet, respectively. Valve operability has been verified by test, with applied moments of 800,000 in-lb and 600,000 in-lb at the inlet and outlet, respectively, and at equivalent acceleration levels of 6.5 g horizontal and 6.0 g vertical. Tests were per IEEE 344 (1975).				
12. Valve Loads: For comparison of calculated loadings vs. seismic capability see Table 3.9-6(e).				

Table 3.9-6(h)

MAIN STEAM ISOLATION VALVE

(UNIT 1)

CRITERIA	METHOD OF ANALYSIS	ALLOWABLE STRESS, (psi) MINIMUM THICKNESS (in), OR MINIMUM AREA (in^2)	CALCULATED STRESS, (psi) ACTUAL THICKNESS (in), OR MINIMUM AREA (in^2)
Design of Pressure-Retaining Parts	All references are made to ASME Code for Pumps and Valves for Nuclear Power, dated November 1968. Reference the same code for explanation of the symbols used.		
Body Minimum Wall Thickness	Reference Article 452.1b(2), Nonstandard Pressure - Rated Value. Table NB 451.4 For design condition of 1,250 psig and 575^{o}F. The primary service rating = 655 based on a core diameter of 23 in t_m = 1.925 in (including a corrosion allowance of 0.12 in).	1.925 in	1.9375 in
Body Shape Rules	Reference Article 452.2, Body Shape Rules		
Radius of Crotch	Reference Article 452.2a(1), Radius of Crotch. Criterion: $r_2 \geq 0.3\ t_m$; r_2 = 1.0 in, t_m = 1.925 (0.3 · 1.925) = 0.578 <1.0; criterion satisfied.	0.578 in	1.0 in
Out of Roundness	Reference Article 452.2e. Since no ovality was built into the valve body, the requirements of this article are satisfied.	Not applicable	Not applicable
Flat wall Limitation	Reference Article 452.2g, Flat Wall Limitation. Since no flat sections were built into the valve body design, the requirements of this article are satisfied.	Not applicable	Not applicable
Primary Crotch Stress Due to Internal Pressure	Reference Article 452.3 Criterion: $P_m = (A_f/A_m + 0.5)\ P_s < S_m$		
	where A_f = 504 in^2, A_m = 58 in^2, P_s = 1,375 psig, P_m = 12,650 psi, S_m = 19,400 psi; since $S_m > P_m$ criterion satisfied.	19,400 psi	12,650 psi
Valve Body Secondary Stress	Reference Article 452.4		
Primary Plus Secondary Stress Due to Internal Pressure	Reference Article 452.4a $Q_p = C_p\ (r_i/t_e + 0.5)\ P_s\ C_a$		
	where C_p = 3, r_i = 11.625 in, P_s = 1,375 psi, t_e = 2.75 for wye-type valve C_a = 1.33 ---> Q_p = 25,965 psi		

Table 3.9-6(h) (Cont'd)

(Unit 1)

CRITERIA	METHOD OF ANALYSIS	ALLOWABLE STRESS, (psi) MINIMUM THICKNESS (in), OR MINIMUM AREA (in^2)	CALCULATED STRESS, (psi) ACTUAL THICKNESS (in), OR MINIMUM AREA (in^2)
Secondary Stress Due to Pipe Reaction	Reference Article 452.4b, figures 452.4b(3), 452.4b(4), 452.4b(5)		
Direct or Axial Load Effect	$P_{ed} = F_dS/G_d$ where S = 30,750, F_d = 30 in^2, G_d = 183 in^2	19,400 psi	5,040 psi
	--->P_{ed} = 5,040 psi		
Bending Load Effect	$P_{eb} = C_b\ F_bS/G_b$ where S = 30,750, F_b = 340 in^3,		
	i.d. = 23.25 in, r_i = 11.625, t_e = 2.75, $\bar{r}$ = 13.90 in as t_e/r = .197 > .19 --->C_b = 1		
	$G_b = I/(r_i + t_e)$ where I = 15,028 in^4, r_i = 11.625 in,		
	t_e = 2.75 in --->G_b = 1052 in^3		
	--->P_{eb} = 9,940 psi	19,940 psi	9,940 psi
Torsion Load Effect	Reference Article 452.4b		
	$P_{et} = 2\ F_bS/G_t$ where F_b = 340 in^3, S = 30,750 psi		
	G_t = 2,162 in^3	19,400 psi	9,670 psi
	P_{et} = 9,670 psi		

Table 3.9-6(h) (Cont'd)

(Unit 1)

CRITERIA	METHOD OF ANALYSIS	ALLOWABLE STRESS, (psi) MINIMUM THICKNESS (in), OR MINIMUM AREA (in^2)	CALCULATED STRESS, (psi) ACTUAL THICKNESS (in), OR MINIMUM AREA (in^2)
Thermal Secondary Stress at Crotch Region	Reference Article 452.4C, figures 452.4C(4), 452.4C(3), 452.4C(5)		
	$Q_T = Q_{T_1} + Q_{T_2}$		
	where $Te_1 = 3$ in, $Q_{T_1} = 1,100$ psi		
	$Q_{T_2} = C_6 C_2 \Delta T_2$ where $C_2 = .21$, $C_6 = 220$, and $T_2 = 5.6$		
	$Q_{T_2} = 260$ psi, $Q_T = 1,360$ psi		
	Criterion: $S_N = Q_P + P_e = 2Q_T \le 3 S_m$		
	where $Q_p = 25,965$, $P_e = 9,940$, $Q_T = 1360$	58,200 psi	38,625 psi
	as 38,625 $\le$ 58,200; criterion satisfied		
Normal Duty Valve Fatigue Requirements	Reference Article 452.5, figure 452.5(a) Criterion $N_a \ge 2,000$ cycles		
	$S_{p_1} = 2/3\ Q_p + P_{eb}/2 + Q_{T_1} + 1.3\ Q_{T_2}$,		
	$S_{p_2} = 0.4\ Q_p + (K/2)\ (P_{eb} + 2Q_{T_2})$		
	where $Q_p = 25,965$, $P_{eb} = 9,940$ K-2,		
	$Q_{T_1} = 1,160$, $Q_{T_2} = 260$ psi		
	-->$S_{p_1} = 23,970$, $S_{p_2} = 20,845$, S_a equal to the larger of		
	S_{p_1} and S_{p_2} ---> $S_a = 23,970$ psi --->		
	$N_a = 55,000 \ge 2,000$; criterion satisfied		

Table 3.9-6(h) (Cont'd)

(Unit 1)

CRITERIA	METHOD OF ANALYSIS	ALLOWABLE STRESS, (psi) MINIMUM THICKNESS (in), OR MINIMUM AREA (in^2)	CALCULATED STRESS, (psi) ACTUAL THICKNESS (in), OR MINIMUM AREA (in^2)
Cyclic Loading Requirements at Valve Crotch	Reference Article 454 Thermal Transients Not Excluded by Code Criterion: ?(N_{ri}/N_i) < 1 Calculate the fatigue usage factor (I_t) as follows: S_n Max = $Q_p + P_{eb} + C_6 (C_3 + C_4) \Delta T_f$ max ---> S_n max = 105,810 psi for ΔT_{fi} = 90, N_{ri} = 120, N_i = 2,700 N_{ri}/N_i = 0.044 for ΔT_{fi} = 122, N_{ri} = 10, N_i = 1,600 N_{ri}/N_i = 0.006 for ΔT_{fi} = 342, N_{ri} = 8, N_i = 55 N_{ri}/N_i = 0.143 -->I_t = ?(N_{ri}/N_i) = 0.196 < 1; criterion satisfied		

Table 3.9-6(h) (Cont'd)

(Unit 1)

CRITERIA	METHOD OF ANALYSIS	ALLOWABLE STRESS, (psi) MINIMUM THICKNESS (in), OR MINIMUM AREA (in^2)	CALCULATED STRESS, (psi) ACTUAL THICKNESS (in), OR MINIMUM AREA (in^2)
Disk Design Calculation	From Reference 3.9-5, Pages 198, 200, 201 Disk design conditions, P_S = 1,250 psi at 575°F, S_m = 17,800 psi at 600°F		
	Case No. 13: $S_t = \frac{3W}{4mt^2 (a^2-b^2)} [a^4 (3m + 1) + b^4 (m - 1) -$ $4m\ a^2b^2 - 4(m + 1)\ a^2b^2\ (1n(a/b))]$ where W = 1,250 psi, m = $\frac{10}{3}$, t = 5.625 in, a = 10.75 in, b = 1.75 in, $S_{t_{13}}$ = 10,354 psi		
	Case No. 14: $S = \frac{3W}{2?\ mt^2} [\frac{2a^2 (m + 1)}{a^2 - b^2} 1n(a/b) + (m - 1)]$ where W = 59,044 lb_f, t = 5.625 in, m = $\frac{10}{3}$, a = 10.75 in, b = 1.75 in, $S_{t_{14}}$ = 4,943 psi	17,800 psi	15,297 psi
	Case No. 21: $S_r = \frac{3W}{4t^2} [\frac{4a^4(m + 1)\ 1n(a/b)\ a^4(m + 3) + b^4(m - 1) + 4a^2b^2}{a^2(m + 1) + b^2(m - 1)}]$ where W = 1,250, m = 10/3, t = 3.125 in, a = 10.75 in, b = 7.25 in --> $S_{r_{21}}$ = 5760 psi		
	Case No. 22: $S_r = \frac{3W}{2t^2} [\frac{2a^2(m + 1)\ 1n(a/b) + a^2(m - 1) - b^2(m - 1)}{a^2(m + 1) + b^2(m - 1)}]$ where W = 1,250, m = 10/3, t = 3,125, a = 10.75, b = 7.25 --> $S_{r_{22}}$ = 10,740 psi		
	Total stress = $S_{r_{21}} + S_{r_{22}}$ = 16,500 psi, allowable stress = 17,800 psi	17,800 psi	16,500 psi

Table 3.9-6(h) (Cont'd)

(Unit 1)

CRITERIA	METHOD OF ANALYSIS	ALLOWABLE STRESS, (psi) MINIMUM THICKNESS (in), OR MINIMUM AREA (in^2)	CALCULATED STRESS, (psi) ACTUAL THICKNESS (in), OR MINIMUM AREA (in^2)
Tensile Stress at Thread Relief Valve Stem	Valve open $S_A = F/A_t$ where F = 31,586 lbs, A_t = 1,956 in^2, S_{max} = 16,148 psi		
	Valve Closed F = 46,342 lbs S_{max} = 23,692 psi	30,600 psi	23,692 psi
Bonnet Design Calculations Including Seismic Accelerations For SSE	Paragraph UG-34c(2) on ASME Code Section VIII		
Minimum Thickness	$P_{fd} = P + P_{eg}$, $P_{eg} = \frac{16M}{\pi G^3} + \frac{4F}{\pi G^2}$ where M = 1,292,000 in-lbs, F = 53,739 lbs, G = 24.75 in P_{eg} = 546 psi, P_{fd} = 1,796 psi $t = d\left(\frac{CR}{1.5S} + \frac{1.78\ W\ hg}{1.5\ S d^3}\right)^{1/2}$ where C = 0.3, P_{fd} = 1,459 psi, S = 17,800 psi, hg = 2.625 in, W = 1,120,000 lbs, d = 24.75 in -->t = 4.503 in, t = 4.503 + 0.120 = 4.623 in (corrosion allowance is 0.120 in)	4.623 in	5.344 in
Reinforcement	Reference paragraph I-704.41(c) of USAS B31-7 to account for the opening for stem in the bonnet Required reinforcement d x t x 0.5 = $(d_3t_3 = d_4t_4)/2$ d_3 = 1.875, t_4 = 2.223, t_3 = 2.875, d_4 = 3 Reinforcement = 6.030 in^2 required = 6.6126 in^2 available	6.030 in^2	6.6126 in^2
Bonnet Studs Design Calculation	Reference Article E-1000 Bolt used 20 pieces of 2.652 in^2/bolts Total bolt area = 53.04 in^2		
Normal Operation	1. Pressure stress at Operating Condition $S_1 = W_{m1}/A_b$ = 21,116 psi where W_{m1} = 1,120,000 lbs and A_b = 53.04 in^2	27,700 psi	21,116 psi

Table 3.9-6(h) (Cont'd)

(Unit 1)

CRITERIA	METHOD OF ANALYSIS	ALLOWABLE STRESS, (psi) MINIMUM THICKNESS (in), OR MINIMUM AREA (in^2)	CALCULATED STRESS, (psi) ACTUAL THICKNESS (in), OR MINIMUM AREA (in^2)
	2. Gasket load at ambient condition with no internal pressure	35,000 psi	2,019 psi
	$S_2 = W_{m2}/A_b$ = 2,019 psi where W_{m2} = 107,065 lb_f and		
	A_b = 53.04 in^2		
	Maximum tensile stress = 21,116 psi		
	Thermal-stress is assumed negligible because the coefficients of thermal expansion of bonnet place and stud are the same.		
Body Flange Design Calculations	Reference paragraph I-704.5.1 of USAS B 317 Total flange moment under operating conditions		
	$M_O = M_D + M_G + M_T$		
	$M_D = H_D\ h_D$, $H_D = 0.785\ B^2P$, $h_D = R + 0.591$		
	where B = 21.75, P = 1,796 psi --> H_D = 667,290 lb_f, h_D = 2.813 in, M_D = 1,877,000 in-lbs		
	$M_G = H_G\ h_g$, $H_G = W - H$, $h_G = \frac{C-G}{2}$		
	where W is the higher of W_{m1} and W_{m2}		
	W_{m1} = 1,120,000 lbs		
	W_{m2} = 107,065 lbs		
	H_G = 256,392 lbs, h_G = 2.625 in -->M_G = 673,000 in-lbs		
	$M_T = H_T\ h_T$		
	H_T = 196,775 lbs, h_T = 3.375 in, M_T = 664,000 in-lb_f		
	M_o = 3,214,000 in-lbs		
	Total flange moment under gasket seating condition		
	$M_o = W(C-G)/2$, $W = S_a(A_m + A_b)/2$		
	where C = 30 in, A_b = 53.04 in^2, G = 24.75 in,		
	A_m = 32.857 in^2, s_a = 35,000 psi at 100°F		
	--->W = 1,503,193 lbs ---> M_o = 3,945,895 in-lbs		

Table 3.9-6(h) (Cont'd)

(Unit 1)

CRITERIA	METHOD OF ANALYSIS	ALLOWABLE STRESS, (psi) MINIMUM THICKNESS (in), OR MINIMUM AREA (in^2)	CALCULATED STRESS, (psi) ACTUAL THICKNESS (in), OR MINIMUM AREA (in^2)
Longitudinal Hub Stress	$S_H = fM_o/((LS_1)^2B) + PB/(4g_o) = 22,317$ psi $< 1.5\ S_{fo} = 26,700$ psi	26,700 psi	22,317 psi
Radial Stress	Reference UA-51 (1), Equation (7) of Section VIII of ASME B&PV Code, 1971 Edition		
	$S_R = \frac{(1.33\ t_e+1)M_o}{Lt^2B} = 13,132$ psi $< 1.5\ S_{fo} = 26,700$ psi	26,700 psi	13,132 psi
Tangential	$S_T = \frac{(YM_o)}{t^2B} - ZS_R = 7,563$ psi $< 1.5\ S_{fo} = 26,700$		
	where Y = 4.5, t = 4.125 in, z = 2.4, B = 21.75 in	26,700 psi	7,563 psi

Table 3.9-6(h) (Cont'd)

MAIN STEAM ISOLATION VALVE

(UNIT 2)

CRITERIA	METHOD OF ANALYSIS	ALLOWABLE STRESS, (psi) MINIMUM THICKNESS (in), OR MINIMUM AREA (in^2)	CALCULATED STRESS, (psi) ACTUAL THICKNESS (in), OR MINIMUM AREA (in^2)
Design of Pressure-Retaining Parts	All references are made to ASME Code for Pumps and Valves for Nuclear Power, dated November 1968. Reference the same code for explanation of the symbols used.		
Body Minimum Wall Thickness	Reference Article 452.1b(2), Nonstandard Pressure - Rated Value. Table NB 451.4 For design condition of 1,250 psig and 575°F. The primary service rating = 655 based on a core diameter of 23 in t_m = 1.925 in (including a corrosion allowance of 0.12 in).	1.925 in	1.9375 in
Body Shape Rules	Reference Article 452.2, Body Shape Rules		
Radius of Crotch	Reference Article 452.2a(1), Radius of Crotch. Criterion: $r_2 \geq 0.3\ t_m$; r_2 = 1.0 in, t_m = 1.925 (0.3 · 1.925) = 0.578 <1.0; criterion satisfied.	0.578 in	1.0 in
Out of Roundness	Reference Article 452.2e. Since no ovality was built into the valve body, the requirements of this article are satisfied.	Not applicable	Not applicable
Flat wall Limitation	Reference Article 452.2g, Flat Wall Limitation. Since no flat sections were built into the valve body design, the requirements of this article are satisfied.	Not applicable	Not applicable
Primary Crotch Stress Due to Internal Pressure	Reference Article 452.3 Criterion: $P_m = (A_f/A_m + 0.5)\ P_s < S_m$		
	where A_f = 504 in^2, A_m = 58 in^2, P_s = 1,375 psig, P_m = 12,650 psi, S_m = 19,400 psi; since $S_m > P_m$, criterion satisfied.	19,400 psi	12,650 psi
Valve Body Secondary Stress	Reference Article 452.4		
Primary Plus Secondary Stress Due to Internal Pressure	Reference Article 452.4a $Q_p = C_p\ (r_i/t_e + 0.5)\ P_s\ C_a$		
	where C_p = 3, r_i = 11.625 in, P_s = 1,375 psi, t_e = 2.75 for wye-type valve C_a = 1.33 ---> Q_p = 25,965 psi		

Table 3.9-6(h) (Cont'd)

(Unit 2)

CRITERIA	METHOD OF ANALYSIS	ALLOWABLE STRESS, (psi) MINIMUM THICKNESS (in), OR MINIMUM AREA (in^2)	CALCULATED STRESS, (psi) ACTUAL THICKNESS (in), OR MINIMUM AREA (in^2)
Secondary Stress Due to Pipe Reaction	Reference Article 452.4b, figures 452.4b(3), 452.4b(4), 452.4b(5)		
Direct or Axial Load Effect	$P_{ed} = F_d S/G_d$ where $S = 30,750$, $F_d = 30\ in^2$, $G_d = 183\ in^2$	19,400 psi	5,040 psi
	$\text{--->} P_{ed} = 5,040$ psi		
Bending Load Effect	$P_{eb} = C_b\ F_b S/G_b$ where $S = 30,750$, $F_b = 340\ in^3$,		
	i.d. = 23.25 in, $r_i = 11.625$, $t_e = 2.75$, $\bar{r} = 13.90$ in as $t_e/r = .197 > .19 \text{--->} C_b = 1$		
	$G_b = I/(r_i + t_e)$ where $I = 15,028\ in^4$, $r_i = 11.625$ in,		
	$t_e = 2.75$ in $\text{--->} G_b = 1052\ in^3$		
	$\text{--->} P_{eb} = 9,940$ psi	19,940 psi	9,940 psi
Torsion Load Effect	Reference Article 452.4b		
	$P_{et} = 2\ F_b S/G_t$ where $F_b = 340\ in^3$, $S = 30,750$ psi		
	$G_t = 2,162\ in^3$	19,400 psi	9,670 psi
	$P_{et} = 9,670$ psi		

Table 3.9-6(h) (Cont'd)

(Unit 2)

CRITERIA	METHOD OF ANALYSIS	ALLOWABLE STRESS, (psi) MINIMUM THICKNESS (in), OR MINIMUM AREA (in^2)	CALCULATED STRESS, (psi) ACTUAL THICKNESS (in), OR MINIMUM AREA (in^2)
Thermal Secondary Stress at Crotch Region	Reference Article 452.4C, figures 452.4C(4), 452.4C(3), 452.4C(5) $Q_T = Q_{T_1} + Q_{T_2}$ where Te_1 = 3 in, Q_{T_1} = 1,100 psi $Q_{T_2} = C_6 C_2 \Delta T_2$ where C_2 = .21, C_6 = 220, and T_2 = 5.6 Q_{T_2} = 260 psi, Q_T = 1,360 psi Criterion: $S_N = Q_{P_2} + P_e = 2Q_T \leq 3\ S_m$ where Q_p = 25,965, P_e = 9,940, Q_T = 1360	58,200 psi	38,625 psi
	as 38,625 $\leq$ 58,200; criterion satisfied		
<u>Normal Duty Valve Fatigue Requirements</u>	Reference Article 452.5, figure 452.5(a) Criterion $N_a \geq$ 2,000 cycles $S_{p_1} = 2/3\ Q_p + P_{eb_2}/2 + Q_{T_1} + 1.3\ Q_T$, $S_{p_2} = 0.4\ Q_p + (K/2)_2\ (P_{eb} + 2Q_T)$ where Q_p = 25,965, P_{eb} = 9,940 K-2, Q_{T_1} = 1,160, Q_{T_2} = 260 psi -->S_{p_1} = 23,970, S_{p_2} = 20,845, S_a equal to the larger of S_{p_1} and S_{p_2} ---> S_a = 23,970 psi ---> N_a = 55,000 $\geq$ 2,000; criterion satisfied		

Table 3.9-6(h) (Cont'd)

(Unit 2)

CRITERIA	METHOD OF ANALYSIS	ALLOWABLE STRESS, (psi) MINIMUM THICKNESS (in), OR MINIMUM AREA (in^2)	CALCULATED STRESS, (psi) ACTUAL THICKNESS (in), OR MINIMUM AREA (in^2)
Cyclic Loading Requirements at Valve Crotch	Reference Article 454 Thermal Transients Not Excluded by Code Criterion: ?(N_{ri}/N_i) < 1 Calculate the fatigue usage factor (I_t) as follows: S_n Max = $Q_p + P_{eb} + C_6 (C_3 + C_4) \Delta T_f$ max ---> S_n max = 105,810 psi for ΔT_{fi} = 90, N_{ri} = 120, N_i = 2,700 N_{ri}/N_i = 0.044 for ΔT_{fi} = 122, N_{ri} = 10, N_i = 1,600 N_{ri}/N_i = 0.006 for ΔT_{fi} = 342, N_{ri} = 8, N_i = 55 N_{ri}/N_i = 0.143 -->I_t = ?(N_{ri}/N_i) = 0.196 < 1; criterion satisfied		

Table 3.9-6(h) (Cont'd)

(Unit 2)

CRITERIA	METHOD OF ANALYSIS	ALLOWABLE STRESS, (psi) MINIMUM THICKNESS (in), OR MINIMUM AREA (in^2)	CALCULATED STRESS, (psi) ACTUAL THICKNESS (in), OR MINIMUM AREA (in^2)
Disk Design Calculation	From Reference 3.9-5, Pages 198, 200, 201 Disk design conditions, P_S = 1,250 psi at 575°F, S_m = 17,800 psi at 600°F		
	Case No. 13: $S_t = \frac{3W}{4mt^2(a^2-b^2)}[a^4(3m+1) + b^4(m-1) - 4m\,a^2b^2 - 4(m+1)\,a^2b^2\,(1n(a/b))]$		
	where W = 1,250 psi, m = $\frac{10}{3}$, t = 5.625 in,		
	a = 10.75 in, b = 1.75 in, $S_{t_{13}}$ = 10,354 psi		
	Case No. 14: $S = \frac{3W}{2?\,mt^2}\left[\frac{2a^2(m+1)}{a^2-b^2}1n(a/b) + (m-1)\right]$		
	where W = 59,044 lb_f, t = 5.625 in, m = $\frac{10}{3}$,		
	a = 10.75 in, b = 1.75 in, $S_{t_{14}}$ = 4,943 psi	17,800 psi	15,297 psi
	Case No. 21: $S_r = \frac{3W}{4t^2}\left[\frac{4a^4(m+1)\,1n(a/b)\,a^4(m+3) + b^4(m-1) + 4a^2b^2}{a^2(m+1) + b^2(m-1)}\right]$		
	where W = 1,250, m = 10/3, t = 3.125 in, a = 10.75 in, b = 7.25 in		
	--> $S_{r_{21}}$ = 5760 psi		
	Case No. 22: $S_r = \frac{3W}{2t^2}\left[\frac{2a^2(m+1)\,1n(a/b) + a^2(m-1) - b^2(m-1)}{a^2(m+1) + b^2(m-1)}\right]$		
	where W = 1,250, m = 10/3, t = 3,125, a = 10.75, b = 7.25		
	--> $S_{r_{22}}$ = 10,740 psi		
	Total stress = $S_{r_{21}} + S_{r_{22}}$ = 16,500 psi, allowable stress = 17,800 psi	17,800 psi	16,500 psi

Table 3.9-6(h) (Cont'd)

(Unit 2)

CRITERIA	METHOD OF ANALYSIS	ALLOWABLE STRESS, (psi) MINIMUM THICKNESS (in), OR MINIMUM AREA (in^2)	CALCULATED STRESS, (psi) ACTUAL THICKNESS (in), OR MINIMUM AREA (in^2)
Tensile Stress at Thread Relief Valve Stem	Valve open S = F/A_t where F = 31,586 lbs, A_t = 1,956 in^2, S_{max} = 16,148 psi		
	Valve Closed F = 46,342 lbs S_{max} = 23,692 psi	30,600 psi	23,692 psi
Bonnet Design Calculations Including Seismic Accelerations For SSE	Paragraph UG-34c(2) on ASME Code Section VIII		
Minimum Thickness	$P_{fd} = P + P_{eg}$, $P_{eg} = \frac{16M}{\pi G^3} + \frac{4F}{\pi G^2}$ where M = 1,292,000 in-lbs, F = 53,739 lbs, G = 24.75 in P_{eg} = 546 psi, P_{fd} = 1,796 psi $t = d\left(\frac{CR}{1.5S} + \frac{1.78\ W\ hg}{1.5\ S d^3}\right)^{½}$ where C = 0.3, P_{fd} = 1,459 psi, S = 17,800 psi, hg = 2.625 in, W = 1,120,000 lbs, d = 24.75 in -->t = 4.503 in, t = 4.503 + 0.120 = 4.623 in (corrosion allowance is 0.120 in)	4.623 in	5.344 in
Reinforcement	Reference paragraph I-704.41(c) of USAS B31-7 to account for the opening for stem in the bonnet Required reinforcement d x t x 0.5 = $(d_3t_3 = d_4t_4)/2$ d_3 = 1.875, t_4 = 2.223, t_3 = 2.875, d_4 = 3 Reinforcement = 6.030 in^2 required = 6.6126 in^2 available	6.030 in^2	6.6126 in^2
Bonnet Studs Design Calculation	Reference Article E-1000 Bolt used 20 pieces of 2.652 in^2/bolts Total bolt area = 53.04 in^2		
Normal Operation	1. Pressure stress at Operating Condition $S_1 = W_{m1}/A_b$ = 21,936 psi where W_{m1} = 1,163,504 lbs and A_b = 53.04 in^2	27,700 psi	21,936 psi

Table 3.9-6(h) (Cont'd)

(Unit 2)

CRITERIA	METHOD OF ANALYSIS	ALLOWABLE STRESS, (psi) MINIMUM THICKNESS (in), OR MINIMUM AREA (in^2)	CALCULATED STRESS, (psi) ACTUAL THICKNESS (in), OR MINIMUM AREA (in^2)
	2. Gasket load at ambient condition with no internal pressure	35,000 psi	2,019 psi
	$S_2 = W_{m2}/A_b$ = 2,019 psi where W_{m2} = 107,065 lb_f and		
	A_b = 53.04 in^2		
	Maximum tensile stress = 21,116 psi		
	Thermal-stress is assumed negligible because the coefficients of thermal expansion of bonnet place and stud are the same.		
Body Flange Design Calculations	Reference paragraph I-704.5.1 of USAS B 317 Total flange moment under operating conditions		
	$M_O = M_D + M_G + M_T$		
	$M_D = H_D\ h_D$, $H_D = 0.785\ B^2P$, h_D = R + 0.591		
	where B = 21.75, P = 1,865 psi --> H_D = 692,927 lb_f, h_D = 2.813 in, M_D = 1,949,204 in-lbs		
	$M_G = H_G\ h_g$, H_G = W - H, $h_G = \frac{C-G}{2}$		
	where W is the higher of W_{m1} and W_{m2}		
	W_{m1} = 1,163,504 lbs		
	W_{m2} = 107,065 lbs		
	H_G = 266,242 lbs, h_G = 2.625 in -->M_G = 698,885 in-lbs		
	$M_T = H_T\ h_T$		
	H_T = 204,335 lbs, h_T = 3.375 in, M_T = 689,631 in-lb_f		
	M_o = 3,337,720 in-lbs		
	Total flange moment under gasket seating condition		
	M_o = W(C-G)/2, W = $S_a(A_m + A_b)/2$		
	where C = 30 in, A_b = 53.04 in^2, G = 24.75 in,		
	A_m = 32.857 in^2, s_a = 35,000 psi at 100°F		
	--->W = 1,503,193 lbs ---> M_o = 3,945,895 in-lbs		

Table 3.9-6(h) (Cont'd)

(Unit 2)

CRITERIA	METHOD OF ANALYSIS	ALLOWABLE STRESS, (psi) MINIMUM THICKNESS (in), OR MINIMUM AREA (in^2)	CALCULATED STRESS, (psi) ACTUAL THICKNESS (in), OR MINIMUM AREA (in^2)
Longitudinal Hub Stress	$S_H = fM_o/((LS_1)^2B) + PB/(4g_o) = 23{,}031$ psi $< 1.5\ S_{fo} = 27{,}900$ psi	27,900 psi	23,031 psi
Radial Stress	Reference UA-51 (1), Equation (7) of Section VIII of ASME B&PV Code, 1971 Edition		
	$S_R = \frac{(1.33\ t_e+1)M_o}{Lt^2B} = 13{,}637$ psi $< 1\ S_{fo} = 18{,}600$ psi	18,600 psi	13,637 psi
Tangential	$S_T = \frac{(YM_o)}{t^2B} - ZS_R = 7{,}855$ psi $< 1\ S_{f_o} = 18{,}600$		
	where Y = 4.5, t = 4.125 in, z = 2.4, B = 21.75 in	18,600 psi	7,855 psi

Table 3.9-6(i)

RECIRCULATION PUMP

STRESS OR CRITERIA	METHOD OF ANALYSIS	ANALYTICAL RESULTS	ALLOWABLE ACTUAL THICKNESS
1. Casing Minimum Wall Thickness	$t = \frac{PR}{SE - 0.6P} + C$		
A. Loads: Normal and upset condition Design pressure and temperature	where: t = minimum required thickness, in P = design pressure, psig R = maximum internal radius, in S = allowable working stress, psi E = joint efficiency C = corrosion allowance, in	t = 2.69 in	S_{allow} = 15,075 psi t_{act} = 3.00 in
B. Primary membrane stress limit Allowable working stress per ASME Section III, Class 1			
2. Casing Cover Minimum Thickness			
A. Loads: Normal and upset condition Design pressure and temperature	Bending Stress $S_B \text{ max.} = \frac{KQa^2}{h^2}$	S_B max = 5950 psi	$S_{B_{allow}}$ = 15,075 psi t_{act} = 7 in

Table 3.9-6(i) (Cont'd)

CRITERIA	METHOD OF ANALYSIS	ANALYTICAL RESULTS	ALLOWABLE STRESS OR ACTUAL THICKNESS
B. Primary membrane stress limit Allowable working stress per Section III, Class 1	where: Q = design pressure a = outer radius b = inner radius $\frac{a}{b} = K = 0.711$ h = thickness		
	Shear Stress S_S max = F/A_S	S_S max = 3380 psi	$S_{S_{allow}}$ = 8750 psi
	where: F = force A_S = area		t_{act} = 3.5 in
3. Pump Discharge Nozzle Stress (Pressure Pending, Axial and Torsional)	Pressure $P_P = \frac{SEt}{R + 0.6t}$	P_P = 1594 psi P_{eb} = 18,500 psi	1.5 S_m = 28,837 psi (S_m = 19,225 psi)
	where: S = allowable stress E = joint efficiency t = thickness R = inside radius P_P = pressure load		
A. Loads: Normal and upset condition	Bending	P_{ed} = 7670 psi P_{et} = 16,000 psi	
Design pressure and temperature piping reaction during normal operation	$P_{eb} = \frac{C_b F_b S}{G_b}$		

Table 3.9-6(i) (Cont'd)

CRITERIA	METHOD OF ANALYSIS	ANALYTICAL RESULTS	ALLOWABLE STRESS OR ACTUAL THICKNESS
B. Combined Stress limit: 1.5 S_m per ASME Code for Pumps and Valves for Nuclear Power Class I	Axial $P_{ed} = \frac{F_d S}{G_d}$ Torsional $P_{et} = \frac{2F_b S}{G_t}$		
4. Cover and Seal Flange Bolt Areas A. Loads: Normal and upset condition Design pressure and temperature Design gasket load B. Bolting Stress Limit Allowable working stress per ASME Section III, Class C	Bolting loads, areas and stresses shall be calculated in accordance with "Rules for bolted Flange Connections" - ASME Section VIII, Paragraph UA-49.	Cover Flange Bolts S_{act} = 19,050 psi A_m = 90.2 in^2 Seal Flange Bolts S_{act} = 18,000 psi A_m = 9.85 in^2	S_{allow} = 20,000 psi A_{act} = 101.0 in^2 S_{allow} = 20,000 psi A_{act} = 11.1 in^2

Table 3.9-6(i) (Cont'd)

CRITERIA	METHOD OF ANALYSIS	ANALYTICAL RESULTS	ALLOWABLE STRESS OR ACTUAL THICKNESS
5. Cover Clamp Flange Thickness A. Loads: Normal and upset condition Design pressure and temperature Design gasket load Design bolting load B. Tangential Flange Stress Limit Allowable working stress per ASME Section III, Class C	Flange thickness and stress shall be calculated in accordance with "Rules for Bolted Flange Connections" - ASME Section VIII, Paragraph UA-49.	Flange Thickness and Stress t = 8.9 in	t_{act} = 9.25 in S_{allow} = 17,500 psi
6. Seal Compartment Wall Thickness A. Loads: Normal and upset condition Design pressure and temperature B. Primary membrane stress limit Allowable working stress per ASME Section III, Class C	$t = \frac{PR}{SE - 0.6P} + C$ where: t = minimum required thickness, in P = design pressure, psig R = maximum internal radius, in S = allowable working stress, psi E = joint efficiency C = corrosion allowance, in	t = 0.741	S_m = 15,075 psi t_{act} = 1.375 in

Table 3.9-6(i) (Cont'd)

CRITERIA	METHOD OF ANALYSIS	ANALYTICAL RESULTS	ALLOWABLE STRESS OR ACTUAL THICKNESS
7. Stresses Due to Seismic Loads A. Loads: Operation pressure and temperature SSE horizontal seismic force = 2.27 g SSE vertical seismic force = 1.39 g B. Combined Stress Limit: Yield stress per ASME Section VIII	The flooded pump-motor assembly shall be analyzed as a free body supported by constant support hangers from the pump brackets. Horizontal and vertical seismic forces shall be applied at mass center of assembly and equilibrium reactions shall be determined for the motor and pump brackets. Load, shear, and moment diagrams shall be constructed using live loads, dead loads, and calculated snubber reactions. Combined bending tension and shear stresses shall be determined for each major component of the assembly including motor support barrel, bolting and pump casing. The maximum combined tensile stress in the cover bolting shall be calculated using tensile stresses determined from loading diagram plus tensile stress from operating pressure.	Motor Bolt Tensile Stress S_{act} = 22,471 psi Pump Cover Bolt Tensile stress S_{act} = 19,417 psi Motor Support Barrel Combined Stress S_{act} = 3307 psi	 S_{allow} = 30,800 psi S_{allow} = 32,000 psi S_{allow} = 22,400 psi

Table 3.9-6(j)

REACTOR RECIRCULATION SYSTEM GATE VALVES, STRUCTURAL AND MECHANICAL LOADING CRITERIA

(UNIT 1)

COMPONENT/ LOADS/ DESIGN	DESIGN PROCEDURE[1]	ALLOWABLE DESIGN VALUE[2]	ACTUAL DESIGN VALUE[2]
I. 28" DISCHARGE VALVE			
Body and Bonnet			
Loads: Design pressure, design temperature, pipe reaction, thermal effects	System requirement not specified	1525 psi 575?F	1525 psi 575?F
Pressure rating, psi	Used ASME Section III Figure NB-3545.1-2	p_r = 800	p_r = 800
Minimum wall thickness, inches	Used ASME Section Para NB-3542	$t_m \geq$ 2.1164 in	t_m = 2.25 minimum
Primary membrane stress, psi	Used ASME Section III Para NB-3545.1	$P_m \leq S_m$ (500°F) = 19,600 psi	P_m = 11,068 psi
Secondary stress due to pipe reaction	Used ASME Section III Para NB-3545.2 (b)(1) (S = 30,000 psi)	$P_e \leq 1.5\ S_m$ (500°F) 1.5 (16,800) = 25,200 psi	P_{ed} = 5,580 psi P_{eb} = 12,702 psi P_{et} = 12,277 psi $P_e = P_{eb}$=12,702 psi
Primary plus secondary stress due to internal pressure	Used ASME Section III Para NB-3545.2(a)(1)	$S_n \leq 3\ S_m$ (500°F) = 58,800 psi	Q_p = 24,284 psi
Thermal secondary stress	Used ASME Section III Para NB-3545.2(c)	$S_n \leq 3\ S_m$ (500°F) = 58,800 psi	Q_T = 5,409 psi
Sum of primary plus secondary stress	Used ASME Section III Para NB-3545.2	$S_n \leq 3\ S_m$ (500°F) = 58,800 psi	$S_n = Q_p + P_e + 2Q_T$ = 47,804 psi

Table 3.9-6(j) (Cont'd)

(Unit 1)

COMPONENT/ LOADS/ DESIGN	DESIGN PROCEDURE[1]	ALLOWABLE DESIGN VALUE[2]	ACTUAL DESIGN VALUE[2]
Fatigue requirements	Used ASME Section III Para NB-3545.3	$N_a \geq 2{,}000$ cycles	$N_a > 10^5$ cycles
Cyclic rating	Used ASME Section III Para NB-3550	$I_t \leq 1$	$I_t = 0.00387$
Body-to-Bonnet Bolting			
Loads: Design pressure and temperature, gasket loads, seismic, hydrodynamic	Used ASME Section III Para NB-3647.1		
Bolt area	Used ASME Section III Para NB-3647.1	$A_b \geq 44.41$ in^2 ($S_b \leq 27{,}975$ psi)	$A_b = 55.86$ in^2 $S_b = 23{,}437$ psi
Body flange stresses	Used ASME Section III Para NB-3647.1		
Operating condition	Used ASME Section III Para NB-3647.1	$S_H \leq 1.5\ S_m$ (575°F) = 28,838 psi $S_R \leq\ S_m$ (575°F) = 19,225 psi $S_T \leq\ S_m$ (575°F) = 19,225 psi	$S_H = 27{,}854$ psi $S_R = 8{,}220$ psi $S_T = 9{,}270$ psi
Gasket seating condition	Used ASME Section III Para NB-3647.1	$S_H \leq 1.5\ S_m$ (100°F) = 30,000 psi $S_R \leq 1.5\ S_m$ (100°F) = 30,000 psi $S_T \leq 1.5\ S_m$ (100°F) = 30,000 psi	$S_H = 29{,}981$ psi $S_R = 11{,}671$ psi $S_T = 12{,}972$ psi
Bonnet flange			
Operating condition	Used ASME Section III Para NB-3647.1	$S_H \leq 1.5\ S_m$ (575°F) = 28,838 psi $S_R \leq\ S_m$ (575°F) = 19,225 psi $S_T \leq\ S_m$ (575°F) = 19,225 psi	$S_H = 27{,}854$ psi $S_R = 8{,}220$ psi $S_T = 9{,}270$ psi
Gasket seating condition	Used ASME Section III Para NB-3647.1	$S_H \leq 1.5\ S_m$ (100°F) = 30,000 psi $S_R \leq 1.5\ S_m$ (100°F) = 30,000 psi $S_T \leq 1.5\ S_m$ (100°F) = 30,000 psi	$S_H = 29{,}981$ psi $S_R = 11{,}671$ psi $S_T = 12{,}972$ psi

Table 3.9-6(j) (Cont'd)

(Unit 1)

COMPONENT/ LOADS/ DESIGN	DESIGN PROCEDURE[1]	ALLOWABLE DESIGN VALUE[2]	ACTUAL DESIGN VALUE[2]
Stresses in Stem			
Not applicable valve is passive			
Disc Analysis			
Loads: Maximum differential pressure			
Maximum stress in the disc	Calculate maximum stress according to table 10 of Reference 3.9-5	S_{max} < 1.5 S_m (500°F) = 28,500	S_{max} = 22,885 psi
Yoke and Yoke Connections			
Loads: Seismic and hydrodynamic	Calculate stresses in the yoke and yoke connections to acceptable structural analysis methods.		

Table 3.9-6(j) (Cont'd)

(Unit 1)

COMPONENT/ LOADS/ DESIGN	DESIGN PROCEDURE[1]	ALLOWABLE DESIGN VALUE[2]	ACTUAL DESIGN VALUE[2]
Stress in yoke legs bolts		$S_{max} \leq S_m$ = 28,800 psi (500°F)	S_{max} = 9,767 psi
Stress of yoke legs		$S_{max} \leq S_m$ = 19,400 psi (500°F)	S_{max} = 13,949 psi
Stress of yoke-bonnet connection		$S_{max} \leq S_m$ = 19,225 psi (575°F)	S_{max} = 15,650 psi
II. 28" SUCTION VALVE			
Body and Bonnet			
Loads: Design pressure, design temperature, pipe reaction thermal effects	System requirement not specified	1,275 psi 575°F	1,275 psi 575?F
Pressure rating, psi	Used ASME Section III Figure NB-3545.1-2	p_r = 668	p_r = 668
Minimum wall thickness, inches	Used ASME Section III Para NB-3542	$t_m \geq$ 1.7724 in	t_m = 2.0 minimum
Primary membrane stress, psi	Used ASME Section III Para NB-3545.1	$P_m \leq S_m$ (500°F) = 19,600 psi	P_m = 9,275 psi
Secondary Stress due to pipe reaction	Used ASME Section III Para NB-3545.2(b) (1) (S = 30,000 psi)	$P_e < 1.5\ S_m$ (500°F) 1.5 (16,800) = 25,200 psi	P_{ed} = 5,318 psi P_{eb} = 11,980 psi P_{et} = 11,575 psi P_e = 11,980 psi
Primary plus secondary stress due to internal pressure	Used ASME Section III Para NB-3545.2(a)(1)	$S_n \leq 3\ S_m$ (500°F) - 58,800 psi	Q_p = 20,580 psi
Thermal secondary stress	Used ASME Section III Para NB-3545.2	$S_n \leq 3\ S_m$ (500°F) - 58,800 psi	Q_t = 5,489 psi

Table 3.9-6(j) (Cont'd)

(Unit 1)

COMPONENT/ LOADS/ DESIGN	DESIGN PROCEDURE[1]	ALLOWABLE DESIGN VALUE[2]	ACTUAL DESIGN VALUE[2]
Sum of primary plus secondary stress	Used ASME Section III Para NB-3545.2	$S_n \leq 3\ S_m$ (500°F) = 58,800	$S_n = Q_p + P_e + 2Q_t$ = 43,538 psi
Fatigue Requirements	Used ASME Section III Para NB-3545.3	$N_a \geq$ 2,000 cycles	N_a = >10^6 cycles
Cyclic rating	Used ASME Section III Para NB-3550	$I_t \leq 1$	I_t = 0.00274
<u>Body-to-Bonnet Bolting</u>			
Loads: Design pressure and temperature, gasket loads, seismic, and hydrodynamic	Used ASME Section III Para NB-3647.1		
Bolt area	Used ASME Section III Para NB-3647.1	$A_b \geq$ 37.53 in^2 $S_b \leq$ 27,975 psi (575°F)	A_b = 55.86 in^2 S_b = 19,470 psi
Body flange stresses	Used ASME Section III Para NB-3647.1		
Operating condition	Used ASME Section III Para NB-3647.1	$S_H \leq 1.5\ S_m$(575°F) = 28,838 psi $S_R \leq\ S_m$(575°F) = 19,225 psi $S_T \leq\ S_m$(575°F) = 19,225 psi	S_H = 24,456 psi S_R = 6,539 psi S_T = 8,718 psi
Gasket seating condition	Used ASME Section III Para NB-3647.1	$S_H \leq 1.5\ S_m$(100°F) = 30,000 psi $S_R \leq 1.5\ S_m$(100°F) = 30,000 psi $S_T \leq 1.5\ S_m$(100°F) = 30,000 psi	S_H = 28,945 psi S_R = 10,253 psi S_T = 13,619 psi
Bonnet flange			
Operating condition	Used ASME Section III Para NB-3647.1	$S_H \leq 1.5\ S_m$(575°F) = 28,838 psi $S_R \leq\ S_m$(575°F) = 19,225 psi $S_T \leq\ S_m$(575°F) = 19,225 psi	S_H = 24,456 psi S_R = 6,539 psi S_T = 8,718 psi

Table 3.9-6(j) (Cont'd)

(Unit 1)

COMPONENT/ LOADS/ DESIGN	DESIGN PROCEDURE[1]	ALLOWABLE DESIGN VALUE[2]	ACTUAL DESIGN VALUE[2]
Gasket seating condition	Used ASME Section III Para NB-3647.1	$S_H \leq 1.5\ S_m(100°F) = 30,000$ psi $S_R \leq 1.5\ S_m(100°F) = 30,000$ psi $S_T \leq 1.5\ S_m(100°F) = 30,000$ psi	$S_H = 28,945$ psi $S_R = 10,253$ psi $S_T = 13,619$ psi
Stress in Stem			
Not applicable valve is passive			
Disc Analysis			
Loads: Maximum differential pressure			
Maximum stress in the disc	Calculate maximum stress according to table 10 of Reference 3.9-5	$S_{max} \leq 1.5\ S_m(500°F) = 28,500$ psi	$S_{max} = 19,418$ psi

Table 3.9-6(j) (Cont'd)

(Unit 1)

COMPONENT/ LOADS/ DESIGN	DESIGN PROCEDURE[1]	ALLOWABLE DESIGN VALUE[2]	ACTUAL DESIGN VALUE[2]
Yoke and Yoke Connections			
Loads: Seismic and hydrodynamic	Calculate stresses in the yoke and yoke connections to acceptable structural analysis methods		
Stress in yoke legs bolts		$S_{max} \leq S_m$ (500°F) = 28,800 psi	S_{max} = 5,247 psi
Stress at yoke legs		$S_{max} \leq S_m$ (500°F) = 19,400 psi	S_{max} = 7,372 psi
Stress at yoke-bonnet connection		$S_{max} \leq S_m$ (575°F) = 19,225 psi	S_{max} = 8,248 psi

Table 3.9-6(j) (Cont'd)

**REACTOR RECIRCULATION SYSTEM GATE VALVES,
STRUCTURAL AND MECHANICAL LOADING CRITERIA**

(Unit 2)

COMPONENT/ LOADS/ DESIGN	DESIGN PROCEDURE[1]	REQUIRED DESIGN VALUE[2]	ACTUAL DESIGN VALUE[2]
III. 28" DISCHARGE VALVE			
Body and Bonnet			
Loads: Design pressure, design temperature, pipe reaction, thermal effects	System requirement not specified	1525 psi 575°F	1525 psi 575°F
Pressure rating, psi	Used ASME Section III Figure NB-3545.1-2	p_r = 800	p_r = 800
Minimum wall thickness, inches	Used ASME Section III Para NB-3542	$t_m \geq 2.1164$ in	t_m = 2.25 minimum
Primary membrane stress, psi	Used ASME Section III Para NB-3545.1	$P_m \leq S_m$ (500°F) = 19,600 psi	P_m = 11,068 psi
Secondary stress due to pipe reaction	Used ASME Section III Para NB-3545.2 (b)(1) (S = 30,000 psi)	$P_e \leq 1.5\ S_m$ (500°F) 1.5 (16,800) = 25,200 psi	P_{ed} = 5,580 psi P_{eb} = 12,702 psi P_{et} = 12,277 psi $P_e = P_{eb}$ = 12,702 psi
Primary plus secondary stress due to internal pressure	Used ASME Section III Para NB-3545.2(a)(1)	$S_n \leq 3\ S_m$ (500°F) = 58,800 psi	Q_p = 24,284 psi
Thermal secondary stress	Used ASME Section III Para NB-3545.2(c)	$S_n \leq 3\ S_m$ (500°F) = 58,800 psi	Q_T = 5,409 psi
Sum of primary plus secondary stress	Used ASME Section III Para NB-3545.2	$S_n \leq 3\ S_m$ (500°F) = 58,800 psi	$S_n = Q_p + P_e + 2Q_T$ = 47,804 psi

Table 3.9-6(j) (Cont'd)

(Unit 2)

COMPONENT/ LOADS/ DESIGN	DESIGN PROCEDURE[1]	REQUIRED DESIGN VALUE[2]	ACTUAL DESIGN VALUE[2]
Fatigue requirements	Used ASME Section III Para NB-3545.3	$N_a \geq 2,000$ cycles	$N_a > 10^5$ cycles
Cyclic rating	Used ASME Section III Para NB-3550	$I_t \leq 1$	$I_t = 0.00387$
Body-to-Bonnet Bolting			
Loads: Design pressure and temperature gasket loads, stem operational load, seismic load (SSE)	Used ASME Section III Para NB-3647.1		
Bolt area	Used ASME Section III Para NB-3647.1	$A_b \geq 44.41$ in^2 ($S_b \leq 27,975$ psi (575°F)	$A_b = 55.86$ in^2 $S_b = 22,619$ psi
Body flange stresses	Used ASME Section III Para NB-3647.1		
Operating condition	Used ASME Section III Para NB-3647.1	$S_H \leq 1.5\ S_m$ (500°F) = 28,838 psi $S_R \leq 1.5\ S_m$ (500°F) = 19,225 psi S_T ? 1.5 S_m (500°F) = 19,225 psi	S_H = 27,049 psi S_R = 7,934 psi S_T = 8,946 psi
Gasket seating condition	Used ASME Section III Para NB-3647.1	$S_H \leq 1.5\ S_m$ (100°F) = 30,000 psi $S_R \leq 1.5\ S_m$ (100°F) = 30,000 psi $S_T \leq 1.5\ S_m$ (100°F) = 30,000 psi	S_H = 29,981 psi S_R = 11,671 psi S_T = 12,972 psi
Bonnet flange			
Operating condition	Used ASME Section III Para NB-3647.1	$S_H \leq 1.5\ S_m$ (500°F) = 28,838 psi $S_R \leq 1.5\ S_m$ (500°F) = 19,225 psi $S_T \leq 1.5\ S_m$ (500°F) = 19,225 psi	S_H = 27,049 psi S_R = 7,934 psi S_T = 8,946 psi
Gasket seating condition	Used ASME Section III Para NB-3647.1	$S_H \leq 1.5\ S_m$ (100°F) = 30,000 psi $S_R \leq 1.5\ S_m$ (100°F) = 30,000 psi $S_T \leq 1.5\ S_m$ (100°F) = 30,000 psi	S_H = 29,981 psi S_R = 11,671 psi S_T = 12,972 psi

Table 3.9-6(j) (Cont'd)

(Unit 2)

COMPONENT/ LOADS/ DESIGN	DESIGN PROCEDURE	REQUIRED DESIGN VALUE(2)	ACTUAL DESIGN VALUE(2)
Stresses in Stem			
Not applicable, valve is passive			
Disc Analysis			
Loads: Maximum differential pressure			
Maximum stress in the disc	Calculate maximum stress according to table 10 of Reference 3.9-5	S_{max} < 1.5 S_m (500°F) = 28,500	S_{max} = 22,885 psi
Yoke and Yoke Connections			
Loads: Stem operational load	Calculate stresses in the yoke and yoke connections to acceptable structural analysis methods.		
Tensile stress in yoke legs bolts		$S_{max} \leq S_m$ = 28,800 psi (500°F)	S_{max} = 3,716 psi
Stress of yoke legs		$S_{max} \leq$ 1.5 S_m = 19,400 psi (500°F)	S_{max} = 5,145 psi
Stress of yoke-bonnet connection		$S_{max} \leq S_m$ = 19,225 psi (575°F)	S_{max} = 5,741 psi

Table 3.9-6(j) (Cont'd)

(Unit 2)

COMPONENT/ LOADS/ DESIGN	DESIGN PROCEDURE[(1)]	REQUIRED DESIGN VALUE[(2)]	ACTUAL DESIGN VALUE[(2)]
IV. 28" SUCTION VALVE			
Body and Bonnet			
Loads: Design pressure, design temperature pipe reaction thermal effects	System requirement not specified	1,275 psi 575°F	1,275 psi 575°F
Pressure rating, psi	Used ASME Section III Figure NB-3545.1-2	$p_r = 668$	$p_r = 668$
Minimum wall thickness, inches	Used ASME Section III Para NB-3542	$t_m \geq 1.7724$ in	$t_m = 2.0$ minimum
Primary membrane stress, psi	Used ASME Section III Para NB-3545.1	$P_m \leq S_m$ (500°F) = 19,600 psi	$P_m = 9,275$ psi
Secondary stress due to pipe reaction	Used ASME Section III Para NB-3545.2(b)(1) (S = 30,000 psi)	$P_e < 1.5\ S_m$ (500°F) 1.5 (16,800) = 25,200 psi	$P_{ed} = 5,318$ psi $P_{eb} = 11,980$ psi $P_{et} = 11,575$ psi $P_e = 11,980$ psi
Primary plus secondary stress due to internal pressure	Used ASME Section III Para NB-3545.2(a)(1)	$S_n \leq 3\ S_m$ (500°F) = 58,800	$Q_p = 20,580$ psi
Thermal secondary stress	Used ASME Section III Para NB-3545.2	$S_n \leq 3\ S_m$ (500°F) = 58,800	$Q_t = 5,489$ psi

Table 3.9-6(j) (Cont'd)

(Unit 2)

COMPONENT/ LOADS/ DESIGN	DESIGN PROCEDURE[1]	REQUIRED DESIGN VALUE[2]	ACTUAL DESIGN VALUE[2]
Sum of primary plus secondary stress	Used ASME Section III Para NB-3545.2	$S_n \le 3\ S_m$ (500?F) = 58,800	$S_n = Q_p + P_e + 2Q_t$ = 43,538 psi
Fatigue Requirements	Used ASME Section III Para NB-3545.3	$N_a \ge 2{,}000$ cycles	$N_a > 10^6$ cycles
Cyclic rating	Used ASME Section III Para NB-3550	$I_t \le 1$	$I_t = 0.00274$
<u>Body-to-Bonnet Bolting</u>			
Loads: Design pressure and temperature, gasket loads, stem operational load, seismic load (SSE)	Used ASME Section III Para NB-3647.1		
Bolt area	Used ASME Section III Para NB-3647.1	$A_b \ge 37.53$ in^2 $S_b \le 27{,}975$ psi (575°F)	$A_b = 55.86$ in^2 $S_b = 19{,}283$ psi
Body flange stresses	Used ASME Section III Para NB-3647.1		
Operating condition	Used ASME Section III Para NB-3647.1	$S_H \le 1.5\ S_m$(575°F) = 28,838 psi $S_R \le 1.5\ S_m$(575°F) = 19,225 psi $S_T \le 1.5\ S_m$(575°F) = 19,225 psi	S_H = 24,264 psi S_R = 6,476 psi S_T = 8,364 psi
Gasket seating condition	Used ASME Section III Para NB-3647.1	$S_H \le 1.5\ S_m$(100°F) = 30,000 psi $S_R \le 1.5\ S_m$(100°F) = 30,000 psi $S_T \le 1.5\ S_m$(100°F) = 30,000 psi	S_H = 28,945 psi S_R = 10,253 psi S_T = 13,619 psi
Bonnet flange			
Operating condition	Used ASME Section III Para NB-3647.1	$S_H \le 1.5\ S_m$(575°F) = 28,838 psi $S_R \le 1.5\ S_m$(575°F) = 19,225 psi $S_T \le 1.5\ S_m$(575°F) = 19,225 psi	S_H = 24,264 psi S_R = 6,476 psi S_T = 8,634 psi

Table 3.9-6(j) (Cont'd)

(Unit 2)

COMPONENT/ LOADS/ DESIGN	DESIGN PROCEDURE[1]	REQUIRED DESIGN VALUE[2]	ACTUAL DESIGN VALUE[2]
Gasket seating condition	Used ASME Section III Para NB-3647.1	$S_H \leq 1.5\ S_m(100°F) = 30,000$ psi $S_R \leq 1.5\ S_m(100°F) = 30,000$ psi $S_T \leq 1.5\ S_m(100°F) = 30,000$ psi	$S_H = 28,945$ psi $S_R = 10,253$ psi $S_T = 13,619$ psi
Stress in Stem			
Not applicable, valve is passive			
Disc Analysis			
Loads: Maximum differential pressure			
Maximum stress in the disc	Calculate maximum stress according to table 10 of Reference 3.9-5	$S_{max} \leq 1.5\ S_m(500°F) = 28,500$ psi	$S_{max} = 19,418$ psi
Yoke and Yoke Connections			
Loads: Stem operational load	Calculate stresses in the yoke and yoke connections to acceptable structural analysis methods		

Table 3.9-6(j) (Cont'd)

(Unit 2)

COMPONENT/ LOADS/ DESIGN	DESIGN PROCEDURE[1]	REQUIRED DESIGN VALUE[2]	ACTUAL DESIGN VALUE[2]
Tensile stress in yoke legs bolts		$S_{max} \leq S_m$ (500°F) = 28,800 psi	S_{max} = 3,877 psi
Stress at yoke legs		$S_{max} \leq 1.5\ S_m$ (500°F) = 19,400 psi	S_{max} = 5,378 psi
Stress at yoke-bonnet connection		$S_{max} \leq S_m$ (575°F) = 19,225 psi	S_{max} = 6,004 psi

(1) ASME Section III refers to 1968 Edition of ASME B&PV Code.

(2) Terms used are defined as follows:

A_b	=	actual total cross-sectional area of bolts at root of thread or section of least diameter under stress, in^2
I_t	=	thermal cyclic index for a particular valve application
N_a	=	permissible number of complete startup/shutdown cycles at 100°F/hour fluid temperature change rate
p_r	=	primary pressure rating, lb
P_e	=	largest value among P_{eb}, P_{ed}, P_{et}, psi
P_{eb}	=	secondary stress in crotch region of valve body caused by bending of connected standard pipe, psi
P_{ed}	=	Secondary stress in crotch region of valve body caused by direct or axial load imposed by connected standard piping, psi
P_{et}	=	secondary stress in crotch region of valve body caused by twisting of connected standard pipe, psi
P_m	=	general primary-membrane stress intensity at crotch region, psi
Q_p	=	sum of primary-plus-secondary stresses at crotch resulting from internal pressure, psi
Q_T	=	thermal-stress in crotch region resulting from 100°F/hour fluid temperature change rate, psi
S_b	=	allowable bolt stress at design temperature, psi
S_B	=	bending stress, psi
S_H	=	calculated longitudinal stress in hub, psi
S_m	=	design stress intensity, psi
S_{max}	=	maximum stress, psi
S_n	=	sum of primary-plus-secondary stress intensities at crotch region resulting from 100?F/hour temperature change rate, psi
S_R	=	calculated radial stress in flange, psi
S_s	=	shear stress due to operator torque, psi
S_t	=	stress due to operator thrust, psi
S_T	=	tangential stress in flange, psi
t_m	=	minimum body wall thickness, in

Table 3.9-6(k)

HPCI TURBINE

CRITERIA/COMPONENT	LOADING CONDITION	STRESS TYPE	ALLOWABLE STRESS (PSI)	CALCULATED STRESS (PSI)
PRESSURE BOUNDARY CASTINGS				
Allowable stresses based on ASME Section III. Pressure boundary castings for Type SA216-WCB				
	For normal condition: Design pressure Design temperature Inlet and exhaust nozzle loads			
1. Stop Valve (100% radiograph)		General membrane	8,975	17,500
2. Turbine inlet (high pressure)		General membrane	6,550	14,000
3. Turbine wheel case (low pressure)		General membrane	6,000	14,000
	For upset, Emergency, or faulted condition: Design pressure Design temperature Controlling combination of OBE, SSE, SRV, & LOCA Inlet and exhaust nozzle loads			
1. Stop valve (100% radiograph)		General membrane	17,700	19,250
2. Turbine inlet (high pressure)		General membrane	15,250	15,400
3. Turbine wheel case (low pressure)		General membrane	13,690	15,400
PRESSURE BOUNDARY BOLTING				
Allowable stresses are based on ASME Section III, for pressure boundary bolting for Type SA193-B7				
	For normal condition: Design pressure Design temperature Inlet and exhaust nozzle loads			

Table 3.9-6(k) (Cont'd)

CRITERIA/COMPONENT	LOADING CONDITION	STRESS TYPE	ALLOWABLE STRESS (PSI)	CALCULATED STRESS (PSI)
PRESSURE-CONTAINING BOLTS				
Stop valve		Tensile	17,600	25,000
Turbine flange		Tensile	18,290	25,000
	For upset emergency, or faulted condition: Design pressure Design temperature Controlling combination of OBE, SSE, SRV, & LOCA Inlet and exhaust nozzle loads			
PRESSURE-CONTAINING BOLTS				
Stop valve		Tensile	17,950	25,000
Turbine flange		Tensile	18,655	25,000
	For faulted condition: Design pressure Design temperature Combination (ABS) of SSE, SRV, & LOCA Inlet and exhaust nozzle loads			
Turbine shaft stress		Bending	12,850 psi	45,000 psi
Turbine shaft deflection	-		0.032 inch	0.125 inch
Thrust bearing load		Force	4,600 lb_f	5,600 lb_f
Journal bearing load		Force	5,020 lb_f	19,500 lb_f
Pedestal bolts, coupling end		Tension	28,740 lb	31,100 lb
Taper pins, coupling end		Shear	39,110 lb	42,050 lb
Guide block weld, governor end		Shear	43,550 lb	43,650 lb

Table 3.9-6(l)

SLCS PUMP

Criteria/Loading		Component	Limiting Stress Type	Allowable Stress (psi)	Calculated Stress (psi)
Based on ASME Section III.					
Pressure Boundary Parts:					
1. Fluid Cylinder - SA182-F304,	S_y = 30,000 psi				
2. Discharge valve stop, stuffing box and cylinder head extension SA479-304,	S_y = 30,000 psi				
3. Discharge valve cover, cylinder head and stuffing box flange plate, SA285, Grade C	S_y = 30,000 psi				
4. Stuffing box gland, ASTM A461, Grade 630	S_y = 90,000 psi				
5. Studs, SA193-B7,	S_y = 105,000 psi				
6. Dowel pins[(2)] alignment, SAE4140,	S_y = 117,000 psi				
7. Studs, cylinder tie, SA193-B7,	S_A = 25,000 psi				
8. Pump holddown bolts, SAE Grade 1	T_A = 15,000 psi Q_A = 12,000 psi				
9. Power frame, foot area, cast iron,	S_A = 15,000 psi				
10. Motor holddown bolts, SAE Grade 1	T_A = 15,000 psi Q_A = 12,000 psi				
11. Motor frame foot area, cast iron,	S_A = 15,000 psi				

Table 3.9-6(I) (Cont'd)

Criteria/Loading	Component	Limiting Stress Type	Allowable Stress (psi)	Calculated Stress (psi)
Normal and Upset Condition Loads:				
1. Design pressure	1. Fluid Cylinder	General membrane	17,800	[3]
2. Design temperature	2. Discharge valve stop	General membrane	17,800	
3. Operating basis earthquake	3. Cylinder head extension	General membrane	17,800	
4. Nozzle loads[1]	4. Discharge valve cover	General membrane	17,800	
5. Thermal expansion	5. Cylinder head	General membrane	17,800	
6. SRV	6. Stuffing box flange plate	General membrane	17,800	
7. Dead weight	7. Stuffing box gland	General membrane	35,000	
	8. Cylinder head studs	Tensile	25,000	
	9. Stuffing box studs	Tensile	25,000	
Emergency or Faulted Condition:				
1. Design pressure	1. Fluid cylinder	General membrane	21,360	4,450
2. Design temperature	2. Discharge valve stop	General membrane	21,360	13,600
3. Weight of structure	3. Cylinder head extension	General membrane	21,360	13,600
4. Thermal expansion	4. Discharge valve cover	General membrane	21,360	8,150
5. Safe shutdown earthquake	5. Cylinder head	General membrane	21,360	8,150
6. LOCA	6. Stuffing box flange plate	General membrane	21,360	10,390
7. Nozzle loads	7. Stuffing box gland	General membrane	42,000	11,420
	8. Cylinder head studs	Tensile	25,000	18,820
	9. Dowel pins[2]	Shear only[2]	23,400	19,400
	10. Studs, cylinder tie	Tensile[2]	25,000	24,750
	11. Pump holddown bolts	Shear	12,000	7,560
	12. Pump holddown bolts	Tensile	15,000	9,950
	13. Power frame-foot area	Shear	15,000	1,850
	14. Power frame-foot area	Tensile	15,000	11,390
	15. Motor holddown bolts	Shear	12,000	3,470
	16. Motor holddown bolts	Tensile	15,000	5,660
	17. Motor frame-foot area	Shear	15,000	2,550
	18. Motor frame-foot area	Tensile	15,000	4,125

Table 3.9-6(I) (Cont'd)

Criteria/Loading	Component	Allowable Loads [5] (lb, ft-lb)	Calculated Loads (lb,ft-lb)
Normal and Upset Condition Loads:			
1. Design pressure	Suction	F_o = 730	F_i = 571
2. Design temperature	Nozzle	M_o = 282	M_i = 218
3. Weight of structure			
4. Thermal expansion	Discharge	F_o = 350	F_i = 138
5. Operating basis earthquake	Nozzle	M_o = 69	M_i = 38
6. SRV			
Emergency or Faulted Condition Loads:			
1. Design pressure	Suction	F_o = 730	F_i = 620
2. Design temperature	Nozzle	M_o = 282	M_i = 228
3. Weight of structure			
4. Thermal expansion	Discharge	F_o = 350	F_i = 170
5. Safe shutdown earthquake	Nozzle	M_o = 67	M_i = 46
6. LOCA			

___

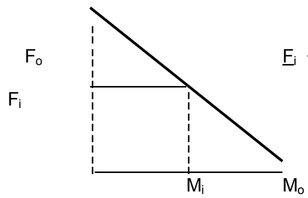
(1) Nozzle loads produce shear loads only.
(2) Dowel pins take all shear.
(3) Calculated stresses for emergency or faulted condition are less than the allowable stresses for the normal and upset condition loads, therefore the normal and upset condition is not evaluated.
(4) Operability: The sum of the plunger and rod assembly, pounds mass times 1.75, acceleration is much less than the thrust loads encountered during normal operating conditions. Therefore, the loads during the faulted condition have no significant effect on pump operability.

Table 3.9-6(l) (Cont'd)

(5) Allowable nozzle load criteria:

Units: Forces - lb
Moments - ft-lb

The allowable combinations of forces and moments are as follows:



$$\frac{F_i}{F_o} + \frac{M_i}{M_o} \le 1$$

<u>where</u>:

F_i = Largest absolute value of the three actual external orthogonal forces (F_x, F_y, F_z) that may be imposed by the interface pipe.

M_i = Largest absolute value of the three actual external orthogonal moments (M_x, M_y, M_z) permitted from the interface pipe when they are combined simultaneously for a specific condition.

F_o = Allowable value of F_i when all moments are zero.

M_o = Allowable value of M_i when all forces are zero.

Table 3.9-6(m)

SLCS TANK

(UNIT 1)

CRITERIA	METHOD OF ANALYSIS	ALLOWABLE STRESS OR MIN. THICKNESS REQ'D OR LOAD		ACTUAL STRESS OR MIN. THICKNESS REQ'D OR LOAD
1. Shell Thickness				
Loads: Normal and Upset	Brownell and Young			
Design Pressure and Temperature where:	"Process Equipment Design" $t = \frac{PR}{SE-0.6P}$	0.01542 in		0.1875 in
	t = min req'd. thickness, in P = design pressure, psig R = maximum internal radius, in S = allowable working stress, psi E = joint efficiency			
Stress Limit	Dynamic Analysis	25,800 psi		1200 psi
2. Nozzle Loads				
Loads: Normal and Upset Design pressure and temperature	The maximum moments due to pipe reaction and maximum forces shall not exceed the allowable limits.	Design Upset Emergency	25,800 psi 28,380 psi 30,960 psi	
Overflow Nozzle and Discharge Nozzle				Less than faulted
Loads: Faulted Dead weight, thermal expansion, and SSE	The maximum moments due to pipe reaction and maximum forces shall not exceed the allowable limits.	Faulted	41,280 psi	
Overflow nozzle Discharge nozzle				9556 psi 4962 psi

Table 3.9-6(m) (Cont'd)

(Unit 1)

CRITERIA	METHOD OF ANALYSIS	ALLOWABLE STRESS OR MIN. THICKNESS REQ'D OR LOAD	ACTUAL STRESS OR MIN. THICKNESS REQ'D OR LOAD
3. Anchor Bolts	API - 650 (ASME Section III)	18,750 psi (tensile) 15,000 psi (shear)	11,770 psi (tensile) 8,962 psi (shear)
4. Dynamic Loads	Equivalent static	Horizontal 1.5 g	0.578 g
SSE SRV LOCA		Vertical 0.14 g	0.898 g
Chugging	Dynamic Analysis	Load - See RRS curves	1,200 psi

Table 3.9-6(m) (Cont'd)

SLCS TANK

(UNIT 2)

CRITERIA	METHOD OF ANALYSIS	ALLOWABLE STRESS OR MIN. THICKNESS REQ'D OR LOAD		ACTUAL STRESS OR MIN. THICKNESS REQ'D OR LOAD
1. Shell Thickness				
Loads: Normal and Upset	Brownell and Young			
Design Pressure and Temperature where:	"Process Equipment Design" $t = \frac{PR}{SE-0.6P}$	0.01542 in		0.1875 in
	t = min req'd. thickness, in P = design pressure, psig R = maximum internal radius, in S = allowable working stress, psi E = joint efficiency			
Stress Limit	Dynamic analysis	25,800 psi		1200 psi
2. Nozzle Loads				
Loads: Normal and Upset Design pressure and temperature	The maximum moments due to pipe reaction and maximum forces shall not exceed the allowable limits.	Allowables: Design Upset Emergency	 25,800 psi 28,380 psi 30,960 psi	
Overflow Nozzle and Discharge Nozzle				5556 psi[(1)] 6357 psi
Loads: Faulted Dead weight, thermal expansion, and SSE	The maximum moments due to pipe reaction and maximum forces shall not exceed the allowable limits			
Overflow nozzle Discharge nozzle		Faulted Faulted	41,280 psi 41,280 psi	9556 psi[(1)] 6357 psi

Table 3.9-6(m) (Cont'd)

(Unit 2)

CRITERIA	METHOD OF ANALYSIS	ALLOWABLE STRESS OR MIN. THICKNESS REQ'D OR LOAD	ACTUAL STRESS OR MIN. THICKNESS REQ'D OR LOAD
3. Anchor Bolts	ASME Section III	18,750 psi (tensile) 15,000 psi (shear)	11,770 psi tensile 8,962 psi shear
4. Dynamic Loads	Equivalent static	Horizontal 1.5 g	0.578 g
SSE SRV LOCA		Vertical 0.14 g	0.898 g
Chugging	Dynamic Analysis	28,500 psi	1200 psi

(1) Since the actual stresses resulting from the analysis using the faulted loads are so low, it is not necessary to reanalyze for actual stresses from upset and emergency conditions.

Table 3.9-6(n)

ECCS PUMPS

LOCATION	LOADING CONDITION	CRITERIA	CALCULATED STRESS (psi)	ALLOWABLE STRESS (psi)
RESIDUAL HEAT REMOVAL PUMP				
Stuffing Box Pipe	FAULTED CONDITION Design Pressure Static Loads Dynamic Loads	ASME Section III	11,718	26,250
Discharge Elbow	FAULTED CONDITION Design Pressure Static Loads Dynamic Loads	ASME Section III	16,172	20,000
Nozzle Shell Intersection	FAULTED CONDITION Design Pressure Static Loads Dynamic Loads	ASME Section III	15,661	28,875
Motor Stand	FAULTED CONDITION Static Loads Dynamic Loads	Bolting Loads and Stresses per ASME Section III, Subsection NF	15,374 (Tensile) 3,058 (Compressive)	22,800 19,326
Motor Bolting	FAULTED CONDITION Static Loads Dynamic Loads	Bolting Loads and Stresses per ASME Section III, Subsection NF	18,755 (Tensile) 4,623 (Shear)	62,500 25,833

Table 3.9-6(n) (Cont'd)

LOCATION	LOADING CONDITION	CRITERIA	CALCULATED STRESS (psi)	ALLOWABLE STRESS (psi)
CORE SPRAY PUMP				
Stuffing Box Pipe	FAULTED CONDITION Design Pressure Static Loads Dynamic Loads	ASME Section III	12,629	26,250
Discharge Elbow	FAULTED CONDITION Design Pressure Static Loads Dynamic Loads	ASME Section III	18,097	20,000
Nozzle Shell Intersection	FAULTED CONDITION Design Pressure Static Loads Dynamic Loads	ASME Section III	28,352	34,650
Motor Stand	FAULTED CONDITION Static Loads Dynamic Loads	Bolting Loads and Stresses per ASME Section III, Subsection NF	11,626 (Tensile) 2,031 (Compressive)	22,800 19,352
Motor Bolting	FAULTED CONDITION Static Loads Dynamic Loads	Bolting Loads and Stresses per ASME Section III, Subsection NF	7,203 (Tensile) 2,622 (Shear)	62,500 25,833

Table 3.9-6(o)

RHR HEAT EXCHANGER

(UNIT 1)

CRITERIA	METHOD OF ANALYSIS	ALLOWABLE STRESS OR MINIMUM THICKNESS REQUIRED	ACTUAL STRESS OR MINIMUM THICKNESS REQUIRED
1. Closure Bolting	Bolting loads and stresses calculated per "Rules for Bolted Flange Connections" ASME Section VIII, Appendix II		
Loads: Normal and upset			
Design pressure and temperature			
Design gasket load			
Bolting Stress Limit	a. Shell to tube sheet bolts	39,375 psi	33,599 psi
Allowable working stress per ASME Section VIII	b. Channel cover bolts	39,375 psi	31,797 psi
2. Wall Thickness	Shell side ASME Section III Class 2 and TEMA Class C		
Loads: Normal and upset			
Design pressure and temperature	Tube side ASME Section VIII-Div. I and TEMA Class C		
Stress Limit			
ASME Section III	a. Shell	0.7915 in	0.8125 in
	b. Shell cover	1.0214 in	1.0625 in (minimum)
	c. Channel	1.1372 in	1.1825 in
	d. Tubes	0.047 in*	0.049 in
	e. Channel cover	6.856 in	6.875 in
	f. Tube sheet	5.7119 in	5.75 in
	g. Shell at nozzles	1.1372 in	1.1875 in

* Minimum wall thickness based on general corrosion.

Table 3.9-6(o) (Cont'd)

(Unit 1)

CRITERIA	METHOD OF ANALYSIS	ALLOWABLE STRESS OR MINIMUM THICKNESS REQUIRED	ACTUAL STRESS OR MINIMUM THICKNESS REQUIRED
3. Nozzle Loads		See (a) and (b) below	See (c) below
Design pressure and temperature			
Dead weight, thermal expansion and SSE	Primary local membrane stress less than 1.8 ASME Section VIII allowable.		

a) Maximum allowable piping loads for faulted conditions (including SSE) shall not exceed the following relationship for each nozzle:

$$\frac{F_{ix}}{F_{ox}} + \frac{F_{iy}}{F_{oy}} + \frac{F_{iz}}{F_{oz}} \leq 1 \text{ and } \frac{M_{ix}}{M_{ox}} + \frac{M_{iy}}{M_{oy}} + \frac{M_{iz}}{M_{oz}} \leq 1$$

where F_i (lbs) is the maximum piping load imposed on each nozzle in the x, y and z direction and M_i (lb-ft) is the maximum moment imposed on each nozzle in the x, y and z directions.

b) Allowable design basis limits (forces F_{ox}, F_{oy}, F_{oz} and Moments M_{ox}, M_{oy}, M_{oz}) for nozzles N1, N2, N3, and N4:

	N1 (Channel inlet)	N2 (Channel outlet)	N3 (Shell inlet)	N4 (Shell outlet)
F_{ox}	11,395 lb	11,395 lb	44,143 lb	26,692 lb
F_{oy}	25,621 lb	25,621 lb	19,630 lb	26,692 lb
F_{oz}	25,621 lb	25,621 lb	44,143 lb	11,871 lb
M_{ox}	65,926 lb-ft	65,926 lb-ft	21,839 lb-ft	12,230 lb-ft
M_{oy}	8,537 lb-ft	8,537 lb-ft	113,596 lb-ft	12,230 lb-ft
M_{oz}	8,537 lb-ft	8,537 lb-ft	21,839 lb-ft	68,680 lb-ft

Note: The calculated loads in (c) below that exceed these allowable loads have been evaluated and are acceptable.

Table 3.9-6(o) (Cont'd)

(Unit 1)

CRITERIA		METHOD OF ANALYSIS		ALLOWABLE STRESS OR MINIMUM THICKNESS REQUIRED	ACTUAL[2] STRESS OR MINIMUM THICKNESS REQUIRED
c)[1]					
F_{ox}	6,367 lb	6,822 lb	6,638 lb	1,101 lb	
F_{oy}	4,442 lb	8,034 lb	10,370 lb	2,526 lb	
F_{oz}	4,836 lb	1,818 lb	11,327 lb	8,178 lb	
M_{ox}	6,141 lb-ft	9,669 lb-ft	37,996 lb-ft	7,900 lb-ft	
M_{oy}	28,863 lb-ft	1,168 lb-ft	16,782 lb-ft	4,195 lb-ft	
M_{oz}	10,670 lb-ft	52,242 lb-ft	16,966 lb-ft	1,863 lb-ft	
4. Support Brackets and Attachment Welds		Stress allowables per ASME Section III Subsection NT (Upset Condition).			
Loads: Faulted		a. Lower bracket welds			
Design pressure and temperature, dead weight, nozzle loads, SSE.		Bending stress		12,863 psi	10,907 psi
		Shear stress		7,074 psi	6,496 psi
		b. Upper Bracket welds			
		Bending stress		14,438 psi	6,069 psi
		Shear stress		7,074 psi	3,631 psi
5. Anchor Bolts		Stress allowable per ASME III, Subsection NF XVIII			
Loads: Faulted		Lower support bolting			
Design pressure and temperature, dead weight, nozzle loads, SSE, SRV.		Interaction criteria		52,500 psi	33,048 psi

Table 3.9-6(o) (Cont'd)

(Unit 1)

CRITERIA	METHOD OF ANALYSIS	ALLOWABLE STRESS OR MINIMUM THICKNESS REQUIRED	ACTUAL[2] STRESS OR MINIMUM THICKNESS REQUIRED
6. Shell Adjacent to Support Brackets	Shell stress allowables per ASME Section III Subsection NC (Upset Condition).		
Loads: Faulted Design pressure and temperature, dead weight, nozzle loads, SSE.	a. Maximum principal stress adjacent to upper support	28,875 psi	17,068 psi
	b. Maximum principal stress adjacent to lower support	28,875 psi	20,929 psi
7. Shell Away from Discontinuities	Stress allowable per ASME Section III Subsection NC (Upset Condition)		
Loads: Faulted Design pressure and temperature, dead weight, nozzle loads, SSE.	Principal stress	19,250 psi	15,662 psi

Table 3.9-6(o) (Cont'd)

RHR HEAT EXCHANGER

(UNIT 2)

CRITERIA	METHOD OF ANALYSIS	ALLOWABLE STRESS OR MINIMUM THICKNESS REQUIRED	ACTUAL STRESS OR MINIMUM THICKNESS REQUIRED
1. Closure Bolting Loads: Normal and upset Design pressure and temperature Design gasket load	Bolting loads and stresses calculated per "Rules for Bolted Flange Connections" ASME Section VIII, Appendix II		
Bolting Stress Limit	a. Shell to tube sheet bolts	39,375 psi	31,973 psi
Allowable working stress per ASME Section VIII	b. Channel cover bolts	39,375 psi	30,721 psi
2. Wall Thickness Loads: Normal and upset	Shell side ASME Section III Class 2 and TEMA Class C		
Design pressure and temperature	Tube side ASME Section III Class 3 and TEMA Class C		
Stress Limit			
ASME Section III	a. Shell	0.7915 in	0.8125 in
	b. Shell cover	1.0214 in	1.0625 in (minimum)
	c. Channel	1.1372 in	1.1825 in
	d. Tubes	0.047 in	0.049 in
	e. Channel cover	6.856 in	6.875 in
	f. Tube sheet	5.7119 in	5.75 in

Table 3.9-6(o) (Cont'd)

(Unit 2)

CRITERIA	METHOD OF ANALYSIS	ALLOWABLE STRESS OR MINIMUM THICKNESS REQUIRED	ACTUAL STRESS OR MINIMUM THICKNESS REQUIRED
3. Nozzle Loads		See (a) and (b) below	See (c) below
Design pressure and temperature			
Dead weight, thermal expansion and SSE	Primary local membrane stress less than 1.8 ASME Section VIII allowable.		

a) Maximum allowable piping loads for faulted conditions (including SSE) shall not exceed the following relationship for each nozzle:

$$\frac{F_{ix}}{F_{ox}} + \frac{F_{iy}}{F_{oy}} + \frac{F_{iz}}{F_{oz}} \leq 1 \quad \text{and} \quad \frac{M_{ix}}{M_{ox}} + \frac{M_{iy}}{M_{oy}} + \frac{M_{iz}}{M_{oz}} \leq 1$$

where F_i (lbs) is the maximum piping load imposed on each nozzle in the x, y and z direction and M_i (lb-ft) is the maximum moment imposed on each nozzle in the x, y and z directions.

b) Allowable design basis limits (forces F_{ox}, F_{oy}, F_{oz} and Moments M_{ox}, M_{oy}, M_{oz}) for nozzles N1, N2, N3, and N4:

	N1 (Channel inlet)	N2 (Channel outlet)	N3 (Shell inlet)	N4 (Shell outlet)
F_{ox}	11,395 lb	11,395 lb	44,143 lb	26,692 lb
F_{oy}	25,621 lb	25,621 lb	19,630 lb	26,692 lb
F_{oz}	25,621 lb	25,621 lb	44,143 lb	11,871 lb
M_{ox}	65,926 lb-ft	65,926 lb-ft	21,839 lb-ft	12,230 lb-ft
M_{oy}	8,537 lb-ft	8,537 lb-ft	113,596 lb-ft	12,230 lb-ft
M_{oz}	8,537 lb-ft	8,537 lb-ft	21,839 lb-ft	68,680 lb-ft

Note: The calculated loads in (c) below that exceed these allowable loads have been evaluated and are acceptable.

Table 3.9-6(o) (Cont'd)

(Unit 2)

CRITERIA	METHOD OF ANALYSIS	ALLOWABLE STRESS OR MINIMUM THICKNESS REQUIRED	ACTUAL STRESS OR MINIMUM THICKNESS REQUIRED

c)(1)

RHR HEAT EXCHANGER NO. 2AE205

COMP. (FAULTED)	NOZZLE NOS. N1 (Channel Inlet)	N2 (Channel Outlet)	N3 (Shell Inlet)	N4 (Shell Outlet)
F_x (lbs)	4,253	4,792	7,079	2,384
F_y (lbs)	2,626	3,452	7,940	6,250
F_z (lbs)	2,144	1,861	13,677	4,417
M_x (ft-lbs)	10,895	12,011	21,948	43,883
M_y (ft-lbs)	16,393	1,461	33,124	7,688
M_z (ft-lbs)	23,153	7,357	31,163	3,242

RHR HEAT EXCHANGER NO. 2BE205

COMP. (FAULTED)	NOZZLE NOS. N1 (Channel Inlet)	N2 (Channel Outlet)	N3 (Shell Inlet)	N4 (Shell Outlet)
F_x (lbs)	6,720	8,469	5,568	2,870
F_y (lbs)	3,254	7,950	9,440	3,369
F_z (lbs)	404	1,867	10,762	3,977
M_x (ft-lbs)	3,521	9,418	24,813	14,848
M_y (ft-lbs)	81	693	21,090	11,111
M_z (ft-lbs)	11,116	49,060	25,455	2,521

Table 3.9-6(o) (Cont'd)

(Unit 2)

CRITERIA	METHOD OF ANALYSIS	ALLOWABLE STRESS OR MINIMUM THICKNESS REQUIRED	ACTUAL[2] STRESS OR MINIMUM THICKNESS REQUIRED
4. Support Brackets and Attachment Welds	Stress allowables per ASME Section III Subsection NF (Upset Condition).		
Loads: Faulted	a. Lower bracket welds		
Design pressure and temperature, dead weight, nozzle loads, SSE.	Bending stress	14,149 psi	10,107 psi
	Shear stress	7,074 psi	6,200 psi
	b. Upper Bracket welds		
	Bending stress	14,149 psi	3,766 psi
	Shear stress	7,074 psi	2,183 psi
5. Anchor Bolts	Stress allowable per ASME III, Subsection NF		
Loads: Faulted	Lower support bolting		
Design pressure and temperature, dead weight, nozzle loads, SSE, SRV.	Interaction criteria	52,500 psi	25,427 psi
6. Shell Adjacent to Support Brackets	Shell stress allowables per ASME Section III Subsection NC (Upset Condition).		
Loads: Faulted			
Design pressure and temperature, dead weight, nozzle loads, SSE.	a. Maximum principal stress adjacent to upper support	28,875 psi	18,083 psi
	b. Maximum principal stress adjacent to lower support	28,875 psi	19,311 psi

Table 3.9-6(o) (Cont'd)

(Unit 2)

CRITERIA	METHOD OF ANALYSIS	ALLOWABLE STRESS OR MINIMUM THICKNESS REQUIRED	ACTUAL[3] STRESS OR MINIMUM THICKNESS REQUIRED
7. Shell Away from Discontinuities	Stress allowable per ASME Section III Subsection NC (Upset Condition)		
Loads: Faulted			
	Principal stress	19,250 psi	12,080 psi
Design pressure and temperature, dead weight, nozzle loads, SSE.			

(1) The actual nozzle loads are provided by AE and are used to calculate the actual stresses.

(2) The "actual" stresses tabulated were calculated using the "maximum" nozzle loads that the RHR Heat Exchanger can withstand without exceeding the allowable stress.

(3) The "actual" stresses tabulated were calculated using the "actual" nozzle loads supplied by the AE for the Limerick 2 RHR Heat Exchanger.

Table 3.9-6(p)

RWCU PUMP

Following is a summary of the design calculations on the RWCU pump (B, C only):

Pump Part[1]	CALCULATED STRESS (psi)	ALLOWABLE STRESS (psi)
Casing wall	10,820	12,814
Cover bolting	20,000	25,000
Pedestal bolt (shear)	18,015	44,000
Motor Part[2]		
Motor foot bolts (shear)	174	60,000
Pump pedestal bolt (shear)	194	60,000
Foundation bolting	230	60,000

Following is a summary of the design calculations on the “A” RWCU pumps:

Part(1)	Calc. Stress (psi)	Allowable Stress (psi)
Pump Suction Nozzle	12,774 (U1) 1,582 (U2)	15,000
Pump Discharge Nozzle	12,824 (U1) 1,546 (U2)	17,500
Motor Case Outlet Nozzle	5,995 (U1) 5,995 (U2)	17,500
Motor Case Inlet Nozzle	5,997 (U1) 5,997 (U2)	17,500
Pump Support Flange Bolts (shear)	3,437 (U1) 3,325 (U2)	25,833 (U1) 11,800 (U2)
Pump Case/Motor Case Studs	23,229 (U1) 23,229 (U2)	25,000

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(1) ASME Code calculations.
(2) Non-ASME Code calculations.

Table 3.9-6(q)

RCIC TURBINE

(UNIT 1)

CRITERIA/LOADING	COMPONENT	LIMITING STRESS TYPE	ALLOWABLE STRESS (PSI)	CALCULATED STRESS (PSI)
The highest stressed sections of the various components of the RCIC turbine assembly are identified. Allowable stresses are based on ASME Section III, for:				
Pressure Boundary Castings SA216-WCB				
Pressure Boundary Boltings, SA193-B7				
Alignment Dowel Pins: AISI 4037, Rc 28-35				
Normal Condition Loads:				
1. Design pressure	Castings:			
2. Design Temperature	1) Stop valve[3]	General membrane	17,500	[1]
3. Inlet Nozzle Loads	2) Governor valve[3]	General membrane	17,500	
4. Exhaust Nozzle Loads	3) Turbine inlet	Local bending	21,000	
	4) Turbine case	Local bending	21,000	
	Pressure-containing bolts:	Tensile	25,000	
	Structure alignment pins:	Shear	61,100	
Upset, Emergency or Faulted Condition:				
1. Design Pressure	Castings:			
2. Design Temperature	1) Stop valve[3]	General membrane	19,250	14,160
3. Controlling Combinations of SSE, SRV, and LOCA	2) Governor valve[3]	General membrane	19,250	15,300
	3) Turbine inlet	Local bending	23,100	15,300
4. Inlet nozzle loads	4) Turbine case	Local bending	23,100	18,000
5. Exhaust Nozzle loads	Pressure-Containing Bolts	Tensile	25,000	20,100
	Structure Alignment pins	Shear	61,100	51,600

Table 3.9-6(q) (Cont'd)

(Unit 1)

CRITERIA/LOADING	COMPONENT	ALLOWABLE LOAD CRITERIA[4]	CALCULATED CRITERIA[4]
Nozzle Load Definition:			
Turbine vendor has defined allowable nozzle loads for the turbine assembly. The above calculated stresses assume these allowable nozzle loads have been satisfied.			
Normal Condition Loads:			
1. Design pressure 2. Design temperature 3. Weight of structure 4. Thermal expansion	Inlet Nozzle	$F = \frac{2620-M}{3}$	F = 421 lb M = 814 ft-lb
	Exhaust Nozzle	$F = \frac{6000-M}{3}$	F = 1,226 lb M = 2,043 ft-lb
Upset, Emergency, and Faulted Condition Loads:			
1. Design pressure 2. Design temperature 3. Weight of structure 4. Thermal expansion 5. Controlling combination of SSE, SRV, and LOCA	Inlet Nozzle	$F = \frac{7500-M}{3.75}$	F = 1,156 lb M = 1,432 ft-lb
	Exhaust Nozzle	$F = \frac{8500-M}{0.34}$, but less than 7000 lb	F = 2,440 lb M = 6,253 ft-lb

Table 3.9-6(q) (Cont'd)

RCIC TURBINE

(UNIT 2)

CRITERIA/LOADING	COMPONENT	LIMITING STRESS TYPE	ALLOWABLE STRESS (PSI)	CALCULATED STRESS (PSI)
The highest stressed sections of the various components of the RCIC turbine assembly are identified. Allowable stresses are based on ASME Section III, for:				
Pressure Boundary Castings SA216-WCB				
Pressure Boundary Boltings, SA193-B7				
Alignment Dowel Pins: AISI 4037, Rc 28-35				
Normal Condition Loads:				
1. Design pressure	Castings:			
2. Design Temperature	1) Stop valve[3]	General membrane	17,500	[1]
3. Inlet Nozzle Loads	2) Governor valve[3]	General membrane	17,500	
4. Exhaust Nozzle Loads	3) Turbine inlet	Local bending	21,000	
	4) Turbine case	Local bending	21,000	
	Pressure-containing bolts:	Tensile	25,000	
	Structure alignment pins:	Shear	61,100	
Upset, Emergency or Faulted Condition:				
1. Design Pressure	Castings:			
2. Design Temperature	1) Stop valve[3]	General membrane	19,250	14,160
3. Controlling Combinations of	2) Governor valve[3]	General membrane	19,250	15,300
SSE, SRV, and LOCA	3) Turbine inlet	Local bending	23,100	15,300
4. Inlet nozzle loads	4) Turbine case	Local bending	23,100	18,000
5. Exhaust Nozzle loads	Pressure-Containing Bolts	Tensile	25,000	20,100
	Structure Alignment pins	Shear	61,100	51,600

Table 3.9-6(q) (Cont'd)

(Unit 2)

CRITERIA/LOADING	COMPONENT	ALLOWABLE LOAD CRITERIA[4]	CALCULATED CRITERIA[4]
Nozzle Load Definition:			
Turbine vendor and GE have defined allowable nozzle loads for the turbine assembly. The above calculated stresses assume these allowable nozzle loads have been satisfied.			
Normal Condition Loads:			
1. Design pressure 2. Design temperature 3. Weight of structure 4. Thermal expansion	Inlet Nozzle	$F = \frac{2620-M}{3}$	F = 563 lb M = 575 ft-lb
	Exhaust Nozzle	$F = \frac{6000-M}{3}$	F = 324 lb M = 1,748 ft-lb
Upset, Emergency, and Faulted Condition Loads:			
1. Design pressure 2. Design temperature 3. Weight of structure	Inlet Nozzle	$F = \frac{7500-M}{3.75}$	F = 1,518 lb M = 1,137 ft-lb
4. Thermal expansion 5. Controlling combination of SSE, SRV, and LOCA	Exhaust Nozzle	$F = \frac{8500-M}{0.34}$, but less than 7000 lb	F = 1,016 lb M = 3,030 ft-lb

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(1) Calculated stresses for the faulted condition are lower than the allowable stresses for the normal condition, therefore the normal condition is not evaluated.

(2) Operability: Analysis indicated that shaft deflection with faulted loads is 0.014 inch; which is fully acceptable; and maximum bearing load with faulted condition is 80% of allowable.

(3) 100% radiograph

(4) F = resultant force (lb); M = resultant moment (ft-lb).

Table 3.9-6(r)

RCIC PUMP

CRITERIA/LOADING		COMPONENT	LIMITING STRESS TYPE	ALLOWABLE STRESS (PSI)	CALCULATED STRESS (PSI)
Pressure boundary stress limits of the various components for the RCIC pump assembly are based on ASME Section III, for pressure boundary parts at 140°F.					
1. Forged barrel, SA105 Grade II	S_y = 36,000 psi				
2. End cover plates, SA105 Grade II	S_y = 36,000 psi				
3. Nozzle connections, SA105 Grade II	S_y = 36,000 psi				
4. Aligning pin, SA105 Grade II	S_y = 36,000 psi				
5. Closure bolting, SA193-87	S_y = 105,000 psi				
6. Pump holddown bolting, SA449	S_y = 81,000 psi				
7. Taper pins, SA108 Grade B1112	S_y = 75,000 psi				
Normal and Upset Condition Loads:					
1. Design pressure		1. Forged barrel	General membrane[1]		
2. Design temperature		2. Nozzle reinforcement	Tensile shear		
3. Operating basis earthquake		3. Alignment pin	Tensile		
4. Suction nozzle loads		4. Taper pins			
5. Discharge nozzle loads		5. Pump holddown bolts			
6. Thermal expansion					
7. SRV					
8. Dead weight					
Emergency or Faulted Condition Loads:					
1. Design pressure		1. Forged barrel	General membrane	17,500	7,792
2. Design temperature		2. Nozzle reinforcement at barrel	General membrane	26,250	8,680
3. Safe shutdown earthquake					
4. Suction nozzle loads		3. Alignment pin	Shear	18,000	2,465
5. Discharge nozzle loads		4. Taper pins (bearing housing)		15,000	2,520
6. Thermal expansion					
7. LOCA		5. Pump holddown bolts	Tension	48,000	37,196
8. Dead weight					

Table 3.9-6(r) (Cont'd)

CRITERIA/LOADING	COMPONENT	ALLOWABLE LOADS[4] (lb, ft-lb)	CALCULATED LOADS (lb, ft-lb)
Normal and Upset Condition Loads:			
1. Design pressure	Suction	F_o = 1940	F_i = 1173
2. Design temperature	Nozzle	M_o = 2950[3]	M_i = 2555
3. Weight of structure			
4. Thermal expansion	Discharge	F_o = 3715	F_i = 913
5. Operating basis earthquake	Nozzle	M_o = 4330	M_i = 1193
6. SRV			
Emergency or Faulted Condition Loads:			
1. Design pressure	Suction	F_o = 2325	F_i = 1033
2. Design temperature	Nozzle	M_o = 2950	M_i = 2438
3. Weight of structure			
4. Thermal expansion	Discharge	F_o = 4450	F_i = 1061
5. Safe shutdown earthquake	Nozzle	M_o = 5200	M_i = 1751
6. LOCA			

---

(1) Calculated stresses for emergency or faulted condition are less than the allowable for normal plus upset condition.

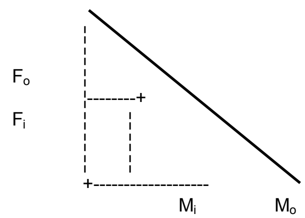
(2) Operability static analysis for emergency or faulted condition shows that the maximum shaft deflection is 0.0044 inch with a 0.0055 inch allowable, shaft stresses are 5602 psi with 32,000 psi allowable, drive end bearing loads are 45 lb with 7670 lb allowable, and thrust end bearing loads are 1462 lb with 17,200 lb allowable.

(3) This allowable moment was determined by analytical qualification which exceeds the Code-determined allowable.

Table 3.9-6(r) (Cont'd)

(4) Allowable nozzle load criteria:
Units: Forces - lb
Moments - ft-lb

The allowable combinations of forces and moments are as follows:



$$\frac{F_i}{F_o} + \frac{M_i}{M_o} \le 1$$

where:

F_i	=	Largest absolute value of the three (F_x, F_y, F_z) that may be imposed by actual external orthogonal forces the interface pipe.
M_i	=	Largest absolute value of the three actual external orthogonal moments (M_x, M_y, M_z) permitted from the interface pipe when they are combined simultaneously for a specific condition.
F_o	=	Allowable value of F_i when all moments are zero.
M_o	=	Allowable value of M_i when all forces are zero.

Table 3.9-6(s)

REACTOR REFUELING AND SERVICING EQUIPMENT

The information on this page has been deleted.

Table 3.9-6(s) (Cont'd)

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Table 3.9-6(s) (Cont'd)

ACCEPTANCE CRITERIA	LOADING	PRIMARY STRESS TYPE	ALLOWABLE STRESS (psi)	CALCULATED STRESS (psi)
FUEL PREPARATION MACHINE				
The allowable axial plus bending loads stresses are based on manual of steel construction, 1980, 8th edition				
F_y = 35,000 psi				
F_u = 38,000 psi				
For normal condition:[2] S_{limit} = 0.66 F_y	For normal condition: 1. Static	Axial load plus bending	23,100	4,441
For emergency condition:[2] S_{limit} = 0.88 F_y	For emergency condition: 1. Static 2. OBE 3. LOCA 4. SRV	Axial load plus bending	30,800	27,619
For faulted condition:[2] S_{limit} = 0.88 F_y	For faulted condition: 1. Static 2. SSE 3. LOCA 4. SRV	Axial load plus bending	30,800	30,118

Table 3.9-6(s) (Cont'd)

ACCEPTANCE CRITERIA	LOADING	PRIMARY STRESS TYPE	ALLOWABLE STRESS (psi)	CALCULATED STRESS (psi)
REFUELING PLATFORMS				
The allowable axial plus bending loads stresses are based on ASME Section III, Subsection NA				
For type				
S_u = 58,000 psi				
S_y = 36,000 psi				
For normal condition: $S_{limit} = S_m = 0.66\ S_y$	For normal condition: 1. Static loads	Axial load plus bending	23,760	6,366 [(3)]
For upset condition: $S_{limit} = 0.9\ S_y$	For upset condition: 1. Static 2. OBE 3. LOCA 4. SRV	Axial load plus bending	31,680	31,552 [(3)]
For faulted condition: $S_{limit} = 0.7\ S_u$	For faulted condition: 1. Static 2. SSE 3. LOCA 4. SRV	Axial load plus bending	40,600	32,455 [(3)]

[(1)] Pins in shear are limiting factor for horizontal loads applied to all rack castings. Allowable S_s assumed @ 50% allowable stress.
[(2)] The allowable stresses are shape dependent; therefore, these factors apply only at the location of the calculated stress.
[(3)] Calculated stresses shown on this table are based on conservative set of spectra, and pre-upgrade conditions. Calculation LS-0266 demonstrates on a comparative basis that these stresses bound the stresses in the upgraded configuration.

Table 3.9-6(t)

HPCI PUMP

LOCATION[1]	LOADING CONDITION	CRITERIA[2]	CALCULATED STRESS (psi)	ALLOWABLE STRESS (psi)
PRESSURE BOUNDARY PARTS				
Closure Bolting (Main)	Emergency/Faulted Condition: 1. Design pressure 2. Design temperature 3. Seismic loads 4. Nozzle loads 5. SRV 6. LOCA	The allowable stresses are based on normal and upset condition in accordance with ASME Section III for boundary parts at 140°F	19,950	25,000
Closure Bolting (Booster)			17,400	25,000
Casing Wall Thickness (Main)			12,050	14,000
Casing Wall Thickness (Booster)			3,650	14,000
NON-PRESSURE BOUNDARY PARTS				
Pump Bolts (Booster) (Tensile)	Emergency/Faulted Condition: 1. Design pressure 2. Design temperature 3. Seismic loads 4. Nozzle loads	F_o, M_o	20,860	30,000
		Actual	15,929	25,000
Pump Bolts (Main) (Tensile)		F_o, M_o	29,042	30,000
		Actual	14,813	25,000
Dowel Pins (Booster) (Shear)		F_o, M_o	30,880	42,000
		Actual	21,498	33,600
Dowel Pins (Main) (Shear)		F_o, M_o	38,488	42,000
		Actual	23,451	33,600

(1) Eight anchor bolts, each carries the stresses for both units mounted on a common base-plate.

(2) The allowable stress values for bolts are 0.5 S_u, and 0.4 S_u for pins where F_o, M_o criteria are used. For actual nozzle loads, the stress limits are based on ASME Section III at 140°F.

Table 3.9-6(u)

CONTROL ROD DRIVE (INDICATOR TUBE)

CRITERIA	LOADING	PRIMARY STRESS TYPE	ALLOWABLE STRESS (psi)	CALCULATED STRESS[2] (psi)
Allowable primary membrane stress plus bending stress plus bending is based on ASME Section III for type 316 stainless steel @ 250°F S_m = 20,000 psi				
For normal and upset condition: S_{allow} = 50,000 psi	For normal and upset condition: 1. Normal loads[1]	Primary membrane plus bending	50,000	45,500
For emergency condition: S_{allow} = 50,000 psi	For emergency condition: 1. Dynamic P 2. OBE 3. SRV	primary membrane plus bending	50,000	45,500
For faulted condition: S_{allow} = 59,900 psi	For faulted condition: 1. Dynamic P 2. SSE 3. LOCA 4. SRV	Primary membrane plus bending	59,900	46,600

[1] Normal loads include pressure, temperature, weight and mechanical loads.
[2] The loads listed here correspond to the operating power level of 3458 MWt. Per Reference 3.9-31, the loads are not changed for the MUR power uprate conditions.

Table 3.9-6(v)

CRD HOUSING[1]

CRITERIA	LOADING	PRIMARY STRESS TYPE	ALLOWABLE STRESS (psi)	CALCULATED STRESS[2] (psi)
Primary Stress Limit - The allowable primary membrane stress is based on ASME Section III, for Class 1 vessels, stainless steel.				
For normal and upset condition: $S_{limit} = 1.0\ S_m$ = 16,660 psi @ 575?F	Normal and upset condition loads: 1. Design pressure 2. Stuck rod scram loads 3. OBE, with housing lateral support installed 4. SRV 5. Hydraulic line loads	Maximum membrane stress intensity occurs at the tube-to-tube weld near the center of the housing for normal, upset, emergency, and faulted conditions.	16,660	11,900
For faulted conditions:	Faulted conditions loads: 1. Design pressure 2. Stuck rod scram loads 3. SSE, with housing lateral support installed 4. Annulus pressurization 5. LOCA 6. Hydraulic line loads		59,760	25,850

[1] Analyzed to emergency conditions limits
[2] The loads listed here correspond to the operating power level of 3458 MWt. Per Reference 3.9-32, the loads are bounding for the MUR power uprate condition.

Table 3.9-6(w)

JET PUMPS

Criteria	Loading Combinations	Stress Type	Allowable Stress (psi)	Calculated Stress[1] (psi)
Primary membrane plus bending stress based on ASME Section III				
For service levels A and B (normal and upset) condition: For type 304SS @ 550°F S_m = 16,900 psi S_{limit} = 3.0 S_m psi	Normal and Upset Condition Loads: 1. Pressure 2. Weight 3. Clamping force 4. OBE 5. Vibration force 6. Thermal loads 7. SRV_{ALL}	Primary membrane plus bending plus secondary	50,700	18,185
For service level C (emergency) condition: For type 304SS @ 550°F S_m = 16,900 psi S_{limit} = 2.25 S_m psi	Emergency Condition Loads: 1. Pressure 2. Weight 3. Clamping force 4. Chugging 5. SRV_{ALL}	Primary membrane plus bending	38,025	10,702
For service level D (faulted) condition: For type 304SS @ 550°F S_m = 16,900 psi S_{limit} = 3.6 S_m psi	Faulted Condition Loads: 1. Pressure (internal) 2. Pressure (external) 3. Weight 4. Clamping force 5. Shock wave loads 6. SSE 7. Annulus pressurization 8. Jet reaction	Primary membrane plus bending	60,840	46,788

[1] The loads listed here correspond to the operating power level of 3458 MWt. Per Reference 3.9-32, the loads are bounding for the MUR power uprate condition.

Table 3.9-6(x)

FUEL ASSEMBLY (INCLUDING CHANNEL)[1][2][3]

Acceptance Criteria	Loading	Primary Load Type	Calculated Peak Acceleration	Evaluation Basis Acceleration
Acceleration Envelope	Horizontal Direction: Peak Pressure SSE Annulus Pressurization	Horizontal Acceleration	2.6 g	[1]
	Vertical Direction: Peak Pressure SSE Safety/Relief Valve Chugging	Vertical Accelerations	2.6 g[4]	[1]

(1) Evaluation basis accelerations and evaluations are contained in NEDE-21175-3-P.

(2) The calculated maximum fuel assembly gap opening for the most limiting load combination is <0.01 inch based on the methodology contained in NEDE-21175-3-P.

(3) The fatigue analysis indicates that the fuel assembly has adequate fatigue capability to withstand the loadings resulting from multiple SRV actuations and the OBE + SRV event.

(4) This value is determined using the methodology contained in NEDE-21175-3-P.

Table 3.9-6(y)

HIGHEST STRESSED REGION ON THE LPCI COUPLING (ATTACHMENT RING)

Criteria	Loading Combinations	Stress Type	Allowable Stress (psi)	Calculated Stress[1] (psi)
Primary membrane plus bending stress based on ASME Section III NG-3000 for type CF3				
For service levels A and B (normal & upset) condition: S_{limit} = 1.5 S_m psi S_m = 16,900 psi @ 550°F	Normal and Upset Condition Loads: 1. Normal loads 2. Upset pressure 3. OBE 4. SRV	Primary membrane + bending	25,350	12,000
For service level C (emergency) condition: S_{limit} = 2.25 S_m S_m = 16,900 psi @ 550°F	Emergency Condition Loads: 1. Normal loads 2. Emergency pressure 3. Chugging 4. SRV	Primary membrane + bending	38,025	21,100
For service level D (faulted) condition: S_{limit} = 3.6 S_m S_m = 16,900 psi @ 550°F	Faulted Condition Loads: 1. Normal loads 2. Faulted pressure 3. Annulus pressurization 4. SSE	Primary membrane + bending	60,840	28,000

[1] The loads listed here correspond to the operating power level of 3458 MWt. Per Reference 3.9-32, the loads are bounding for the MUR power uprate condition.

Table 3.9-6(z)

REACTOR VESSEL SUPPORT EQUIPMENT:
CRD HOUSING SUPPORT

CRITERIA	LOADING	LOCATION	ALLOWABLE STRESS (psi)	CALCULATED STRESS (psi)
Primary Stress Limit				
AISC specification for the design, fabrication, and erection of structural steel for buildings	Faulted condition loads 1. Deadweight 2. Impact force from failure of a CRD housing	Beams (top chord)	33,000 33,000	f_a = 12,200 f_b = 16,500
		Beams (bottom chord)	33,000 33,000	f_a = 10,300 f_b = 11,700
For normal and upset condition: $f_a = 0.60\ f_y$ (tension) $f_b = 0.60\ f_y$ (bending) $f_v = 0.40\ f_y$ (shear) For faulted conditions: f_a limit = 1.5 f_a (tension) f_b limit = 1.5 f_b (bonding) f_v limit = 1.5 f_v (shear) f_y = material yield strength	(Deadweights and earthquake loads are very small compared to jet forces.)	Grid structure	41,500 27,500	f_b = 40,700 f_v = 11,100

Table 3.9-6(aa)

CONTROL ROD GUIDE TUBE

Criteria	Loading	Primary Stress Type	Allowable Stress (psi)	Calculated Stress[1] (psi)
Control Rod Guide Tube Primary stress limit - The allowable primary membrane stress plus bending stress is based on ASME Section III for type 304 stainless steel tubing.		Maximum stress occurs in the guide tube base		
For normal and upset conditions: 1.5 S_m = 1.5x16,000 psi = 24,000 psi	Upset condition loads 1. dead weight 2. external pressure 3. lateral flow impingement 4. OBE + SRV_{max} 5. scram	$P_m + P_b$	24,000	14,820
For emergency condition: 2.25 S_m = 36,000 psi	Emergency condition loads: 1. dead weight 2. external pressure 3. lateral flow impingement 4. SRV_{ADS} + CHUG	$P_m + P_b$	36,000	20,920
For faulted condition: 2.4 S_m = 38,400 psi	Faulted condition loads: 1. dead weight 2. external pressure 3. lateral flow impingement 4. SSE + SRV_{LSPA} + BCO	$P_m + P_b$	38,400	33,230

(1) The loads are calculated in Reference 3.9-32.

Table 3.9-6(ab)

INCORE HOUSING

Criteria	Loading	Primary Stress Type	Allowable Stress (psi)	Calculated Stress[1] (psi)
Primary Stress Limit - The allowable primary membrane stress is based on ASME Section III for Class 1 vessels for type 304 stainless steel.				
For normal, upset and emergency condition: $S_{limit} = 1.0\ S_m$ $= 16,660$ psi at 575°F	Service Level C (Emergency) condition loads 1. Design pressure 2. Design basis earthquake 3. SRV	Maximum membrane stress intensity occurs at the outer surface of the vessel penetration.	16,660	13,850
For faulted condition: $S_{limit} = 2.4\ S_m$	Faulted condition loads: 1. Faulted pressure 2. LOCA 3. SRV 4. SSE	Maximum membrane stress intensity occurs at the outer surface of the vessel penetration.	39,984	21,225

(1) The loads listed here correspond to the operating power level of 3458 MWt. Per Reference 3.9-32, the loads are bounding for the MUR power uprate condition.

Table 3.9-7

NON-NSSS PIPING SYSTEMS POWER ASCENSION TESTING

PIPING SYSTEM	CODE(S)/ SC/HE ME (1)	TEMP>200°F	THERMAL EXPANSION TEST (2)	DYNAMIC TRANSIENT TEST (3)	STEADY-STATE VIBRATION TEST (4)	REMARKS
Main steam and main steam relief	ASME III-2, B 31.1; SC I, SC II; HE	yes	yes	yes	yes	Main stop valve closure and SRV opening transients
Extraction steam	B 31.1; SC II; HE	yes	N/A(5)	N/A	N/A	
Condensate storage and transfer	B 31.1; SC II; ME	no	N/A	N/A	N/A	
Feedwater	ASME III-1,2, B 31.1; SC I, SC II; HE	yes	yes	yes	yes	Power ascension test for safety-related piping portion only
Air removal and seal steam	B 31.1; SC II, HE	yes	N/A	N/A	N/A	
Service water	B 31.1; SC IIA; ME	no	N/A	N/A	N/A	A portion of the system has 200°F < T < 300°F.
Condensate	B 31.1; SC II; HE	yes	N/A	N/A	N/A	
Clarified water	B 31.1; SC II; ME	no	N/A	N/A	N/A	
Fuel and diesel oil storage and transfer	B 31.1; SC I, SC II; ME	no	N/A	N/A	N/A	Emergency diesel exhaust has T>300°F and thermal expansion test performed.
RHRSW	ASME III-3, B 31.1; SC I, SC IIA, SC II; ME	no	N/A	N/A	N/A	

Table 3.9-7 (Cont'd)

PIPING SYSTEM	CODE(S)/ SC/HE ME (1)	TEMP>200°F	THERMAL EXPANSION TEST (2)	DYNAMIC TRANSIENT TEST (3)	STEADY-STATE VIBRATION TEST (4)	REMARKS
ESW	ASME III-3, B 31.1; SC I, SC IIA; ME	no	N/A	N/A	N/A	
Auxiliary steam	B 31.1; SC II; HE	yes	N/A	N/A	N/A	
Lube oil	B 31.1; SC II; ME	no				
Fire protection	SC II, SC IIA; ME	no	N/A	N/A	N/A	
Process sampling	ASME III-1,2,3, B 31.1; SC I, SC II; ME	no	N/A	N/A	N/A	
Chlorination	B 31.1; SC II; ME	no	N/A	N/A	N/A	
Compressed air	ASME III-2,3, B 31.1; SC I, SC II; ME	no	N/A	N/A	N/A	
Instrument gas	ASME III-2,3, B 31.1; SC I, SC II; ME	no	N/A	N/A	N/A	
TECW	B 31.1; SC II; ME	no	N/A	N/A	N/A	
Circulating water	B 31.1; SC II; ME	no	N/A	N/A	N/A	
Demineralized water makeup	B 31.1; SC II; ME	no	N/A	N/A	N/A	
Safeguard piping fill	ASME III-2, B 31.1; SC I, SC IIA; ME	no	N/A	N/A	N/A	
RECW	ASME III-2, 3, B 31.1; SC I, SC IIA; ME	no	N/A	N/A	N/A	
MSIV-LCS	ASME III-1,2, B 31.1; SC I, SC II; HE	yes	no	N/A	N/A	Abandoned

Table 3.9-7 (Cont'd)

PIPING SYSTEM	CODE(S)/ SC/HE ME [1]	TEMP>200°F	THERMAL EXPANSION TEST [2]	DYNAMIC TRANSIENT TEST [3]	STEADY-STATE VIBRATION TEST [4]	REMARKS
CSCWS	B 31.1; SC I; ME	no	N/A	N/A	N/A	
HPCI	ASME III-1,2, B 31.1; SC I, SC IIA, SC II; HE, ME	yes	yes	yes	yes	Steady-state vibration for steam supply and turbine exhaust Dynamic transient for turbine stop valve closure.
RCIC	ASME III-1,2, B 31.1; SC I; HE, ME	yes	yes	N/A	yes	Steady-state vibration for RCIC steam supply and turbine exhaust
Plant heating steam	B 31.1; SC II, SC IIA; HE	yes	N/A	N/A	N/A	
RWCU	ASME III-1,2,3; SC I, SC II, SC IIA; HE, ME	yes	yes	N/A	yes	Steady-state vibration for RWCU line inside containment
RHR	ASME III-1,2,3; SC I; HE, ME	yes	yes	N/A	yes	Majority of the system has normal operating temperature less than 300°F. Thermal expansion tests are done for SC I systems with T>300°F. Steady-state vibration for inside containment piping and RHR pump discharge

Table 3.9-7 (Cont'd)

PIPING SYSTEM	CODE(S)/ SC/HE ME (1)	TEMP>200?F	THERMAL EXPANSION TEST (2)	DYNAMIC TRANSIENT TEST (3)	STEADY-STATE VIBRATION TEST (4)	REMARKS
Condensate filter demineralizer	B 31.1; SC II; ME	no	N/A	N/A	N/A	
CRD hydraulic	ASME III-2, B 31.1; SC I, SC II, SC IIA; ME	no	N/A	N/A	N/A	
SLCS	ASME III-1,2, B 31.1; SC I, SC IIA; HE, ME	no (See remarks)	N/A (See remarks)	N/A	N/A	Only a small portion of the line near RPV has temperature >200°F.
Core spray	ASME III-1,2; SC I; HE, ME	yes	yes	N/A	yes	Steady-state vibration for core spray pump discharge
FPCC	ASME III-2,3, B 31.1; SC I, SC II, SC IIA; ME	no	N/A	N/A	N/A	
CAC	ASME III-2, B 31.1; SC I, SC IIA; ME	no	N/A	N/A	N/A	No safety-related piping with T>300°F
Solid radwaste	B 31.1; SC II; ME	no	N/A	N/A	N/A	
Liquid radwaste	B 31.1; SC II; ME	no	N/A	N/A	N/A	
Gaseous radwaste	B 31.1; SC II; ME	yes	N/A	N/A	N/A	
DCWS	B 31.1; SC II; ME	no	N/A	N/A	N/A	
Generator H_2 cooling and CO_2 purge	B 31.1; SC II; ME	no	N/A	N/A	N/A	

Table 3.9-7 (Cont'd)

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(1) Code(s): ASME III B&PV Code, -1, -2 or -3: Denotes nuclear Class 1, 2, or 3 piping;
B 31.1: Denotes ANSI B 31.1, Code for Pressure Piping;
SC I, II, or IIA: Denotes seismic Category I, II, or IIA;
HE: Denotes high energy piping system, i.e., pressure ≥275 psi or temperature ≥200°F during normal plant operation;
ME: Denotes moderate energy piping system.

(2) Thermal expansion test for the indicated systems corresponds to test description STP-17 in Table 14.2-3.

(3) Dynamic transient test for the indicated systems corresponds to test description STP-36 in Table 14.2-3. Main steam turbine trip test for Unit 2 (ref. Startup Test STP-36) at 100% power level will be performed during commercial operation of that Unit.

(4) Steady-state vibration tests for the indicated systems corresponds to test description STP-33 Table 14.2-3.

Instrument lines connected to process pipes on which steady-state vibration testing is performed are evaluated on the following basis;
a) for accessible lines; visually monitored
b) for inaccessible lines; instrument lines are monitored, inspected, and measured in accordance with startup test specification for BOP piping as committed in SSER-2.

(5) N/A: Denotes not applicable. Test is not performed for the following reason:
a) For thermal expansion tests: The system is not safety-related or the normal operating temperature < 300°F;
b) For dynamic transient test: The system is not safety-related or does not experience any significant transients;
c) For steady-state vibration tests: The system is not safety-related or no significant vibration is expected.

Table 3.9-8

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Table 3.9-9

DYNAMIC QUALIFICATION SUMMARY
NON-NSSS SAFETY-RELATED MECHANICAL EQUIPMENT

EQUIPMENT DESCRIPTION	ITEM NO.	SUPPLIER	DYNAMIC QUALIFICATION PACKAGE NO.
ESW pumps	8031-M-12	Byron Jackson	D-2
RHRSW pumps	8031-M-12	Byron Jackson	D-3
Diesel generators	8031-M-71	Colt Industries	D-7
Diesel oil transfer pumps	8031-M-79	Crane Dening	D-8
Diesel fuel oil storage tanks	8031-C-28	Buffalo Tank Div., Bethlehem Steel	D-209
Centrifugal fans	8031-M54A	Buffalo Forge	D-55
Drywell Sumps	8031-M-43A	Process Equip-ment Co.	D-5
Control room chilled water pumps	8031-M-58	Ingersoll-Rand	D-139
Control room chiller	8031-M-57A	Carrier Corp	D-138
Reactor enclosure crane	8031-M-16	Harnischfeger	D-50
Fuel pool skimmer surge tanks	8031-C-45	Pittsburgh-Des Moines Steel	D-172
RHR pump suction strainers	NE-265	ABB-Combustion Engineering	D-69
RCIC pump suction strainers	8031-M-162	Newark Wire Cloth	D-69

Table 3.9-9 (Cont'd)

EQUIPMENT DESCRIPTION	ITEM NO.	SUPPLIER	DYNAMIC QUALIFICATION PACKAGE NO.
HPCI pump suction strainers	8031-M-162	Newark Wire Cloth	D-69
Core spray pump suction strainers	NE-265	ABB-Combustion Engineering	D-69
Safeguard piping fill pumps	8031-M-164	Hayward Tyler Pump Co.	D-73
Primary containment vacuum breakers	8031-M-81	Anderson Greenwood Co.	D-135
Nuclear safety and relief valves	8031-M-204B	Crosby	D-74 D-154
		Lonergan	D-65 D-66
	8031-M-204C	Crosby	D-197
RHR HX vacuum relief valve	8031-M-204B	Crosby	D-67
RHR HX relief valve	8031-M-204B	Crosby	D-68
Pressure relief valves		Lonergan	D-31
CREFAS and RERS filter assembly	8031-M-72A	CIV	D-140
SGTS filters	8031-M-56	American Air Filter	D-137

Table 3.9-9 (Cont'd)

EQUIPMENT DESCRIPTION	ITEM NO.	SUPPLIER	DYNAMIC QUALIFICATION PACKAGE NO.
Containment hydrogen recombiners, recombiner power supply panels, recombiner control panels	8031-M-40	Atomics International	D-75
Rupture discs	8031-M-106	Continental Disc Corp.	D-97
MSIV and MSRV accumulator tanks	8031-M-170	Western piping and Engineering	D-157
Diesel oil transfer pump strainers	8031-M-36	Zurn Industries	D-4
Spray pond nozzles	8031-M-112	Spray Engineering Co.	D-9
Spray pond nozzle junction boxes	8031-M-112	Spray Engineering Co.	D-10
Drywell HVAC pressure relief valves	8031-M-123	American Air Filter	D-11
HPCI/RCIC Exhaust steam traps	8031-M-90A	Yarway	D-71
Nuclear wye strainers	8031-M-92AA	Western Piping and Engineering	D-76
MSRV Vacuum relief valves	8031-M-81	Anderson Greenwood Co.	D-136
HVAC Isolation valves	8031-M-70	Allis Chalmers	D-141 D-145
Back pressure steam isolation dampers	8031-M-113	American Warming and Ventilation Co.	D-142

Table 3.9-9 (Cont'd)

EQUIPMENT DESCRIPTION	ITEM NO.	SUPPLIER	DYNAMIC QUALIFICATION PACKAGE NO.
Volume control balancing dampers	8031-M-113	American Warming and Ventilation, Co.	D-148
Fire dampers	8031-M-113	American Warming and Ventilation, Co.	D-150
Slide gate dampers	8031-M-113	American Warming and Ventilation, Co.	D-151
SGTS heaters	8031-M-56	Industrial Engineering and Equipment Co.	D-184
Fuel oil and lube oil filter	8031-M-71	Colt Industries	D-218
Back flood check valves	8031-M-178	Zurn Industries	D-122
HVAC control panels	8031-M-66	Alison	D-162
HVAC control panels (Groups 1-10)	8031-M-66	MCC Powers	D-57 D-187 D-188 D-189 D-190 D-191 D-192 D-193 D-215 D-216
Drywell coolers	8031-M-123	American Air Filter	D-30
Spray pond pumphouse fan cabinets	8031-M-123	American Air Filter	D-48
Reactor enclosure fan cabinets	8031-M-123	American Air Filter	D-62

Table 3.9-9 (Cont'd)

EQUIPMENT DESCRIPTION	ITEM NO.	SUPPLIER	DYNAMIC QUALIFICATION PACKAGE NO.
Control enclosure fan cabinets	8031-M-123	American Air Filter	D-64
Diesel generator enclosure exhaust fans	8031-M-69C	Joy Mfg. Co.	D-6
Control enclosure recirculation fans	8031-M-69C	Joy Mfg. Co.	D-181
Reactor enclosure recirculation fans	8031-M-69C	Joy Mfg. Co.	D-63
Electro-hydraulic operated fan isolation and flow control dampers	8031-M-113	American Warming and Ventilation, Inc.	D-149
Pneumatic operated dampers	8031-M-113	American Warming and Ventilation, Inc.	D-152
Gravity back draft dampers	8031-M-113	American Warming and Ventilation, Inc.	D-153

Table 3.9-10

NSSS COMPARISON WITH REGULATORY GUIDE 1.48

	REGULATORY GUIDE 1.48[(1)]			LGS[(1)]				
COMPONENT	PLANT CONDITION[(2)]	LOADING COMBINATION 1/	DESIGN LIMIT	REGULATORY GUIDE PARAGRAPH	LOADING COMBINATION[(c)]	CODE ALLOWABLE STRESSES	ASME SECTION III REFERENCE	HOW LGS COMPARES WITH REGULATORY GUIDE 1.48
Class 1 Vessels	U	(NPC or UPC) + 0.5 SSE	NB-3223[2/]	1.a	(NPC or UPC), 0.5 SSE	3.0 S_m (includes) secondary stresses)	NB-3223	GE reflects industry position
	E	EPC	NB-3224[2/]	1.b	EPC, 0.5 SSE + transient	1.8$_m$	NB-3224 NB-3225	
	F	NPC + SSE +DSL	NB-3225[2/]	1.c	NPC + SSE + DSL	App. F - Section III		
Class 1 Piping	U	(NPC or UPC) + 0.5 SSE	NB-3654[2/]	1.a	(NPC or UPC), 0.5 SSE	3.0 S_m (includes secondary stresses	NB-3654	GE reflects industry position
	E	EPC	NB-3655[2/]	1.b	EPC, 0.5 SSE + transient	2.25 S_m	NB-3655	
	F	NPC + SSE + DSL	NB-3656[2/]	1.c	NPC + SSE + DSL	3.0 S_m	NB-3656	
Class 1 Pumps (inactive)	U	(NPC or UPC) + 0.5 SSE	NB-3223[5/1/]	2.a	(NPC or UPC), 0.5 SSE	1.63 S_m	NB-3223	GE reflects industry position
	E	EPC	NB-3224[1/]	2.b	EPC, 0.5 SSE + transient	1.8 S_m	NB-3224	
	F	NPC + SSE + DSL	NB-3225[1/]	2.c	NPC + SSE + DSL	App. F - Section III	NB-3225	
Class 1 Pumps (active)	U	(NPC or UPC) + 0.5 SSE	NB-3222[5/6/][7 8]	4.a	(NPC or UPC), 0.5 SSE	Not applicable	Not applicable	Not applicable
	E	EPC	NB-3222[5/6/][7/8/]	4.a	EPC			
	F[7/8/]	NPC + SSE + DSL	NB-3222[5/6/]	4.a	NPC + SSE + DSL			
Class 1 Valves (inactive) Designed by analysis.	U	(NPC or UPC) + 0.5 SSE	NB-3223[5/4/]	2.a	(NPC or UPC), 0.5 SSE	Not applicable	Not applicable	Not applicable
	E	EPC	NB-3224[4/]	2.b	EPC			
	F	NPC + SSE + DSL	NB-3225[2/4/]	2.c	NPC + SSE + DSL			

Table 3.9-10 (Cont'd)

COMPONENT	PLANT CONDITION[2]	LOADING COMBINATION 1/	DESIGN LIMIT	REGULATORY GUIDE PARAGRAPH	LOADING COMBINATION[c]	CODE ALLOWABLE STRESSES	ASME SECTION III REFERENCE	HOW LGS COMPARES WITH REGULATORY GUIDE 1.48
	REGULATORY GUIDE 1.48[1]				LGS[1]			
Class 1 Valves (inactive)	U	(NPC or UPC) + 0.5 SSE	1.1 Pr	3.a	(NPC or UPC), 0.5 SSE	1.1 P_r	NB-3525	GE reflects industry position
Designated by either standard or alternative	E	EPC	1.2 Pr	3.b	EPC, 0.5 SSE + transient	1.2 P_r	NB-3526	
design rules.	F	NPC + SSE + DSL	1.5 Pr	3.c	NPC + SSE + DSL	1.5 P_r	NB-3527	
Class 1 Valves (active)	U	(NPC or UPC) + 0.5 SSE	NB-3222[5/6/7/8/]	4.a	(NPC or UPC), 0.5 SSE	Not applicable	Not applicable	Not applicable
Designed by analysis.	E	EPC	NB-3222[5/6/7/8/]	4.a	EPC			
	F	NPC + SSE + DSL	NB-3222[5/6/7/8/]	4.a	NPC + SSE + DSL			
Class 1 Valves (active)	U	(NPC or UPC) + 0.5 SSE	1.0 Pr[6/]	5.a	(NPC or UPC), 0.5 SSE	1.0 P_r[a]	NB-3525	GE reflects industry position
Designed by standard or alternative design	E	EPC	1.0 Pr[6/]	5.a	EPC, 0.5 SSE + transient	1.0 P_r[a]	NB-3626	
rules.	F	NPC + SSE + DSL	1.0 Pr[6/]	5.a	NPC + SSE + DSL	1.0 P_r[a]	NB-3527	
Class 2 & 3 Vessels (Division 1) of	U	(NPC or UPC) + 0.5 SSE	1.1 g[9/]	6.a	(NPC or UPC), 0.5 SSE	σ_m = 1.1 S[b]	Code Case 1607	Faulted condition: NRC more conservative. GE reflects industry position.
ASME Section VIII	E	EPC	1.1 g[9/]	6.a	EPC, 0.5 SSE + transient		NC/ NB 3221.1(b)	
	F	NPC + SSE + DSL	1.5 g[9/]	6.b	NPC + SSE + DSL	?$_m$ = 2.0 S[b]		
Class 2 Vessels (Division 2) of	U	(NPC or UPC) + 0.5 SSE	NB-3223[2/]	7.a	(NPC or UPC), 0.5 SSE	Not applicable	Not applicable	Not applicable
ASME Section VIII	E	EPC	NB-3224[2/]	7.b	EPC			
	F	NPC + SSE + DSL	NB-3225[2/]	7.c	NPC + SSE + DSL			

Table 3.9-10 (Cont'd)

COMPONENT	PLANT CONDITION[2]	REGULATORY GUIDE 1.48[1]: LOADING COMBINATION 1/	REGULATORY GUIDE 1.48[1]: DESIGN LIMIT	REGULATORY GUIDE 1.48[1]: REGULATORY GUIDE PARAGRAPH	LGS[1]: LOADING COMBINATION[c]	LGS[1]: CODE ALLOWABLE STRESSES	LGS[1]: ASME SECTION III REFERENCE	HOW LGS COMPARES WITH REGULATORY GUIDE 1.48
Class 2 & 3 Piping	U	(NPC or UPC) + 0.5 SSE (b)(1)	NC-3611.1[10/] (b)(4)(c)	8.a	(NPC or UPC), 0.5 SSE	1.2 S_h	NC/ND 3611.3(b)	NRC more conservative. GE reflects industry position.
	E	EPC (b)(1)	NC-3611.1[10/] (b)(4)(c)	8.a	EPC, 0.5 SSE + transient	1.8 S_h	NC/ND 3611.3(c) (4)(b)	
	F	NPC + SSE + DSL (b)(4)(c) (b)(2)	NC-3611.1[10/]	8.b	NPC + SSE + DSL	2.4 S_h	Code case 1606	
Class 2 & 3 Pumps (inactive)	U	(NPC or UPC) + 0.5 SSE	$\sigma_m \le 1.1 S \ge \frac{\sigma_m + \sigma_b}{1.5}$	9.a	(NPC or UPC), 0.5 SSE	Not applicable	Not applicable	Not applicable
	E	EPC	$\sigma_m \le 1.1 S \ge \frac{\sigma_m + \sigma_b}{1.5}$	9.a	EPC			
	F	NPC + SSE + DSL	$\sigma_m \le 1.2 S \ge \frac{\sigma_m + \sigma_b}{1.5}$	9.b	NPC + SSE + DSL			
Class 2 & 3 Pumps (active)	U	(NPC or UPC) + 0.5 SSE	$\sigma_m \le 1.0 S \ge$[11/] $\frac{\sigma_m + \sigma_b}{1.5}$	10.a	(NPC or UPC), 0.5 SSE	$\sigma_m = 1.1 S$[(b)(d)]	Code case 1636	GE reflects industry position
	E	EPC	$\sigma_m \le 1.0 S \ge$[11/] $\frac{\sigma_m + \sigma_b}{1.5}$	10.a	EPC, 0.5 SSE + transient		NC/ND 3423	
	F	NPC + SSE + DSL	$\sigma_m \le 1.0 S \ge$[11/] $\frac{\sigma_m + \sigma_b}{1.5}$	10.a	NPC + SSE + DSL	$\sigma_m = 1.2 S$[(b)(d)]		
Class 2 & 3 Valves (inactive)	U	NPC or UPC) + 0.5 SSE	1.1 P_r	11.a	(NPC or UPC), 0.5 SSE	$\sigma_m = 1.1 S$[(b)]	Code case 1635	Equally conservative
	E	EPC	1.1 P_r	11.a	EPC, 0.5 SSE + transient			
	F	NPC + SSE + DSL	1.2 P_r	11.b	NPC + SSE + DSL	$\sigma_m = 1.2 S$[(b)]	NC/ND 3621	

Table 3.9-10 (Cont'd)

COMPONENT	PLANT CONDITION[2]	LOADING COMBINATION 1/	DESIGN LIMIT	REGULATORY GUIDE PARAGRAPH	LOADING COMBINATION[c]	CODE ALLOWABLE STRESSES	ASME SECTION III REFERENCE	HOWLGSCOMPARES WITH REGULATORYGUIDE 1.48
	REGULATORY GUIDE 1.48[1]				LGS[1]			
Class 2 & 3 Valves (active)	U	(NPC or UPC) + 0.5 SS	1.0 P_r 11/	12.a	(NPC or UPC), 0.5 SSE	σ_m = 1.1 S[a](b)	Code case 1635	Equally conservative. One valve, E41-F005, does not meet this LGS alternate position.
	E	EPC	1.0 P_r 11/	12.a	EPC, 0.5 SSE + transient			
	F	NPC + SSE + DSL	1.0 P_r 11/	12.a	NPC + SSE + DSL	σ_m = 1.2 S[a](b)	NC/ND 362	Structural integrity under its peak transient conditions was justified under applicable provisions of B31.1 (1967). This valve fully meets the load combination and acceptance criteria of Table 3.9-6.

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(1) Numerical indicators (e.g. 1/) in the Regulatory Guide portion of the table correspond to footnotes of Regulatory Guide 1.48.
Alphabetical indicators in the LGS portion of table (or comparative column) correspond to the following:

(a) In addition to compliance with the design limits specified, assurance of operability under all design loading combinations shall be in accordance with Section 3.9.3.2.

(b) The design limit for local membrane stress intensity or primary membrane plus primary bending stress intensity is 150% of that allowed for general membrane (except as limited to 2.45 for inactive components under faulted condition).

(c) When selecting plant events for evaluation, the choice of the events to be included in each plant condition is selected based on the probability of occurrence of the particular load combination. The combination of loads are those identified in Table 3.9-2.

UPC = Upset Plant Conditions
NPC = Normal Plant Conditions
EPC = Emergency Plant Conditions
DSL = Dynamic System Loading
SSE = Safe Shutdown Earthquake

(d) Inactive limits may be used since operability will be demonstrated in accordance with Section 3.9.3.2.

(2) U = Upset
E = Emergency
F = Faulted

Table 3.9-11

DESIGN LOADING COMBINATIONS FOR ASME CODE
CLASS 1, 2, AND 3 NON-NSSS COMPONENTS[2]

CONDITION		DESIGN LOADING COMBINATIONS[1]
Design		PD
Normal		PD + DW
Upset	(a)	$PO + DW + (OBE^2 + SRV_x^2)^{1/2}$
	(b)	$PO + DW + (RVC^2 + OBE^2)^{1/2}$
	(c)	$PO + DW + FV$
	(d)	$PO + DW + OBE + RVO$
Emergency	(a)	$PO + DW + (OBE^2 + SRV^2_{ADS} + SBA^2)^{1/2}$
	(b)	$PO + DW + (FV^2 + OBE^2)^{1/2}$
Faulted	(a)	$PO + DW + (OBE^2 + SRV^2_{ADS} + IBA^2)^{1/2}$
	(b)	$PO + DW + (SSE^2 + SRV^2_{ADS} + IBA^2)^{1/2}$
	(c)	$PO + DW + (SSE^2 + DBA^2)^{1/2}$

where:

- PD = design pressure
- PO = operating pressure
- DW = dead weight
- OBE = operating basis earthquake (inertia portion)
- SSE = safe shutdown earthquake (inertia portion)
- SRV_x = loads due to SRV blow, axisymmetric or asymmetric
- SRV_{ADS} = loads due to automatic depressurization SRV blow, axisymmetric
- SBA = small break accident
- IBA = intermediate break accident
- DBA = design basis accident
- FV = transient response of the piping system associated with fast valve closure (transients associated with valve closure times less than 5 seconds are considered)
- RVC = transient response of the piping system associated with relief valve opening in a closed system
- RVO = sustained load or response of the piping system associated with relief valve opening in an open system or last segment of the closed system with steady-state load

Table 3.9-11 (Cont'd)

SBA, IBA, and DBA include all event induced loads, as applicable, such as chugging, condensation oscillation, pool swell, drag loads, annulus pressurization, etc.

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(1) As required by the appropriate subsection (ie, NB, NC, or ND, of ASME Section III, Division I), other loads, such as thermal transient, thermal gradients, and anchor point displacement portion of the OBE or SRV, are considered in addition to the primary stress-producing loads listed.

(2) Table 3.9-6 lists the load combinations for ASME Code Class 1, 2, and 3 NSSS piping, equipment, and supports.

Table 3.9-12

DESIGN CRITERIA FOR ASME CLASS 1 NON-NSSS PIPING

CONDITION	SERVICE LEVEL	APPLICABLE CODE PARAGRAPH[(1)(2)]	PRIMARY STRESS LIMITS
Design	-	NB-3221 and NB-3652	1.5 S_m
Normal	A	NB-3222 and NB-3653	1.5 S_m
Upset	B	NB-3223 and NB-3654	1.8 S_m and 1.5 S_y
Emergency	C	NB-3224 and NB-3655	2.25 S_m and 1.85 S_y
Faulted	D	NB-3225 and NB-3656	3.0 S_m

(1) As specified by ASME Section III, 1977 Edition through Summer 1979 Addenda.
(2) Functional capability of essential piping is assured in accordance with NEDO-21985, September 1978.

Table 3.9-13

DESIGN CRITERIA FOR NON-NSSS ASME CODE CLASS 1 VALVES

CONDITION	STRESS LIMITS
Design	NB-3521[1]
Normal and upset	NB-3200 or NB-3500[1] (Standard Design Rules)
Emergency[2]	NB-3526[3]
Faulted[2]	NB-3527[3]

(1) As specified by ASME Section III, 1971 through Winter 1972 Addenda.
(2) Where valve function must be ensured (active valve) during the emergency or faulted conditions, the specified emergency or faulted condition for the plant shall be considered the normal condition for the valve.
(3) As specified by ASME Section III, 1971, through Winter 1973 Addenda.

Table 3.9-14

DESIGN CRITERIA FOR NON-NSSS ASME CODE CLASS 2 AND 3 VESSELS DESIGNED TO NC-3300 AND ND-3300

CONDITION	STRESS LIMITS[1]
Design and normal	The vessel shall conform to the requirements of NC-3300 and ND-3300.
Upset, emergency, and faulted	The vessel shall conform to the requirements of ASME Code Case 1607-1.

[1] As specified by ASME Section III, 1971 through Winter 1972 Addenda.

Table 3.9-15

DESIGN CRITERIA FOR NON-NSSS ASME CODE CLASS 2 VESSELS
DESIGNED TO ALTERNATE RULES OF NC-3200

CONDITION	STRESS LIMITS[1][2]
Design and normal	The vessel shall conform to the requirements of NC-3200.
Upset[3]	$P_e \leq 3\ S_m$
	$P_m \leq 1.1\ S_m$
	$(P_m \text{ or } P_L) + P_b \leq 1.65\ S_m$
Emergency	$P_m \leq$ greater of $1.2\ S_m$ or $1.0\ S_y$
	$(P_m \text{ or } P_L) + P_b \leq$ greater of
	$1.8\ S_m$ or $1.5\ S_y$
Faulted[4]	$P_m \leq 2.0\ S_m$
	$(P_m \text{ or } P_L) + P_b \leq 2.4\ S_m$

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(1) Definition of symbols:

P_m = General primary membrane stress intensity. This stress intensity is derived from the average value across the solid section under consideration. Excludes discontinuities and concentrations. Produced only by pressure and other mechanical loads.

P_L = Local primary membrane stress intensity. Same as P_m except that discontinuities are considered.

P_b = Primary bending stress intensity. Component of primary stress intensity proportional to distance from centroid of solid section. Excludes discontinuities and concentrations. Produced only by pressure and other mechanical loads.

P_e = Secondary stress intensity range. Developed by constraint of adjacent parts or by self-constraint of a structure. Considers discontinuities but not concentrations. Produced by mechanical loads and by thermal expansion.

Table 3.9-15 (Cont'd)

S_m = Design stress intensity value, ASME Section III, Appendix I, Table I-1.0.

S_y = Yield strength value, ASME Section III, Appendix I, Table I-2.0.

(2) These limits do not take into account either local or general buckling that might occur in thin-wall vessels. Such buckling shall be considered for upset conditions, but need not be considered for emergency or faulted conditions unless required by the design specification.

(3) Fatigue analysis requirements of NC-3219 and Appendix XIV are considered.

(4) As an alternative to satisfying these limits, the faulted condition stress limits of Appendix F may be applied provided that a complete analysis in accordance with NC-3211.1(c) is performed.

Table 3.9-16

DESIGN CRITERIA FOR NON-NSSS ASME CLASS 2 AND 3 PIPING

CONDITION	APPLICABLE CODE PARAGRAPH[(1)(2)]	PRIMARY STRESS LIMITS
Design:		
Sustained Loads	NC, ND 3652.1	1.0 S_h
Occasional Loads	NC, ND 3652.2	1.2 S_h
Normal & Upset	NC, ND 3652.2 & 3611	1.2 S_h
Emergency	NC, ND 3611	1.8 S_h
Faulted	Code Case 1606	2.4 S_h

(1) As specified by ASME Code Section III, 1971 through Winter 1972 Addenda except the following:

Nuclear Class 2 and 3 flanges are analyzed in accordance with ASME Section III 1977 edition through 1979 Summer Addenda.

(2) Functional capability of essential piping is assured in accordance with NEDO-21985, September 1978.

Table 3.9-17

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Table 3.9-18

SEISMIC CATEGORY I SYSTEM SNUBBER DESIGN INFORMATION

DESIGN CRITERIA FOR NON-NSSS ASME CODE CLASS 2 AND 3 VALVES

CONDITION	STRESS LIMITS[(2)]
Design and normal	The valve shall conform to the requirements of Section III, Paragraphs NC-3500 and ND-3500
Upset, emergency[(1)], and faulted[(1)]	The valve shall conform to the requirements of ASME Code Case 1635-1

(1) Where valve function must be ensured (active valve) during the emergency or faulted condition, the specified emergency or faulted conditions for the plant shall be considered as the normal condition for the valve.

(2) As specified by ASME Section III, 1971 through Winter 1972 Addenda.

Table 3.9-19

SEISMIC CATEGORY I ACTIVE PUMPS AND VALVES
(GE SCOPE OF SUPPLY)

COMPONENT NAME	IDENTIFICATION NUMBER	GE MPL NUMBER
MSIVs	M41-F022(A,B,C,D) M41-F028(A,B,C,D)	B21-F022(A,B,C,D) B21-F028(A,B,C,D)
MSRVs	M41-F013(A,B,C,D E,F,G,H,J,K,L, M,N,S)	B21-F013(A,B,C,D, E,F,G,H,J,K,L, M,N,S)
CRD Vent and Drain Globe Valves	M47-F010 M47-F011 M47-F180 M47-F181	C11-F010 C11-F011 C11-F180 C11-F181
SLCS Pump	1AP208 1BP208 1CP208(1)	C41-C001A C41-C001B C41-C001C(1)
SLCS Relief Valves	M48-F029(A,B,C)	C41-F029(A,B,C)
SLCS Explosive Valves	M48-F004(A,B,C)	C41-F004(A,B,C)
RHR Pump	1AP202 1BP202 1CP202 1DP202	E11-C002A E11-C002B E11-C002C E11-C002D
Core Spray Pump	1AP206 1BP206 1CP206 1DP206	E21-C001A E21-C001B E21-C001C E21-C001D
RCIC Pump	10P203	E51-C001
RCIC Turbine	10S212	E51-C002
HPCI Pump	10P204	E41-C001
HPCI Turbine	10S211	E41-C002

Table 3.9-19 (Cont'd)

COMPONENT NAME	IDENTIFICATION NUMBER	GE MPL NUMBER
RHR System		
Globe Valves	M51-F015(A,B)	E11-F015(A,B)
Gate Valves	M51-F016(A,B)	E11-F016(A,B)
Gate Valves	M51-F017(A,B,C,D)	E11-F017(A,B,C,D)
Gate Valves	M51-F021(A,B)	E11-F021(A,B)
Globe Valves	M51-F027(A,B)	E11-F027(A,B)
Testable Check Valves	M51-F041(A,B,C,D)	E11-F041(A,B,C,D)
Testable Check Valves	M51-F050(A,B)	E11-F050(A,B)
Core Spray		
Gate Valves	M52-F001(A,B,C,D)	E21-F001(A,B,C,D)
Gate Valve	M52-F005	E21-F005
Testable Check Valves	M52-F006(A,B)	E21-F006(A,B)
Gate Valve	M52-F037	E21-F037
HPCI		
Swing-Check	M55-F005	E41-F005
Globe Valve	M55-F012	E41-F012
Stop-Check Valve	M55-F021	E41-F021
RCIC		
Globe Stop-Check	M49-F001	E51-F001
Swing-Check	M49-F014	E51-F014
Globe Valve	M49-F019	E51-F019

(1) This SLCS pump is not within the GE scope of supply.

Table 3.9-20

VALVE QUALIFICATION TEST RANGE
(NON-NSSS SCOPE OF SUPPLY)

QUALIFICATION VALID FOR OTHER VALVES (in)[(1)]

Valve Size Tested	½	1	1 ½	2	3	4	6	8	10	12	14	16	18	20	22	24	26	28	30	36
½	X	X																		
1	X	X	X																	
1½		X	X	X																
2			X	X	X															
3				X	X	X														
4					X	X	X													
6						X	X	X												
8							X	X	X	X										
10								X	X	X	X									
12								X	X	X	X									
14									X	X	X	X	X	X						
16										X	X	X	X	X	X					
18											X	X	X	X	X	X				
20											X	X	X	X	X	X	X	X		
22												X	X	X	X	X	X	X	X	
24													X	X	X	X	X	X	X	
26														X	X	X	X	X	X	X
28														X	X	X	X	X	X	X
30															X	X	X	X	X	X
36																	X	X	X	X

(1) Test data acquired for a qualified valve may be used to qualify valves of the same type that fall within the range of sizes permitted by this table, provided geometric similarity is maintained and supporting stress calculations are provided. If the qualified valve is larger than 36 inch nominal diameter, extrapolation may be made to valves whose nominal size does not vary more than 25% from that of the qualified valve.

Table 3.9-21

DESIGN LOADING COMBINATIONS FOR SUPPORTS FOR ASME CODE CLASS 1, 2, AND 3 COMPONENTS

CONDITION STRESS[2][3]	DESIGN LOADING COMBINATIONS[1]	ALLOWABLE
Hydrostatic Test	a) HTDW	0.8 S_y
Normal and Upset	a) $DW + TH + (OBE^2 + SRV_x^2)^{1/2}$	S_h
	b) $DW + TH + (RVC^2 + OBE^2)^{1/2}$	
	c) DW + TH + FV	
	d) DW + TH + OBE + RVO	
Emergency	a) $DW + TH + (OBE^2 + SRV^2_{ADS} + SBA^2)^{1/2}$	1.8 S_h
	b) $DW + TH + (OBE^2 + FV^2)^{1/2}$	
Faulted	a) $DW + TH + (SSE^2 + SRV^2_{ADS} + IBA^2)^{1/2}$	0.9 S_y
	b) $DW + TH + (OBE^2 + SRV^2_{ADS} + IBA^2)^{1/2}$	
	c) $DW + TH + (SSE^2 + DBA^2)^{1/2}$	

where:

HTDW	=	piping dead weight due to hydrostatic test
TH	=	reaction of the support due to thermal expansion of the pipe
S_y	=	yield stress
S_h	=	allowable stress per ANSI B31.1

See Table 3.9-11 for additional nomenclature.

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(1) Loads due to OBE, SSE, SRV_x, SRV_{ADS}, SBA, IBA, and DBA include both the inertia portion and the anchor motion portion when the response spectra method is used. The loads from the inertia portion and anchor motion are combined by the SRSS method.

(2) The allowable stress shall be limited to 2/3 of the critical buckling stress.

(3) Snubbers, compensating starts, and struts comply with all the requirements of ASME Section III, Subsection NF; they are not commercially available to meet the requirements of ANSI B31.1.

Table 3.9-22

FATIGUE LIMIT
(FOR SAFETY CLASS REACTOR INTERNAL STRUCTURES ONLY)

Summation of fatigue damage usage with design and operation loads following Miner hypotheses [(1)]

CUMULATIVE DAMAGE IN FATIGUE	LIMIT FOR SERVICE LEVELS A AND B (NORMAL AND UPSET) DESIGN CONDITIONS
Design fatigue cycle usage from analysis using the method of ASME Code	$\leq$1.0

(1) M.A. Miner, "Cumulative Damage in Fatigue," Journal of Applied Mechanics, 12, (67), pp. A159-164, (September 1945).

Table 3.9-23

CORE SUPPORT STRUCTURES
STRESS CATEGORIES AND LIMITS OF STRESS INTENSITY FOR SERVICE LEVELS A AND B (NORMAL AND UPSET) CONDITIONS

STRESS CATEGORY	PRIMARY STRESSES: MEMBRANE P_m (4)(7)(8)	PRIMARY STRESSES: BENDING P_B (4)(7)(8)	SECONDARY STRESSES: MEMBRANE AND BENDING SECONDARY, Q (2)(4)(6)	PEAK STRESSES: PEAK, F (2)(4)(6)

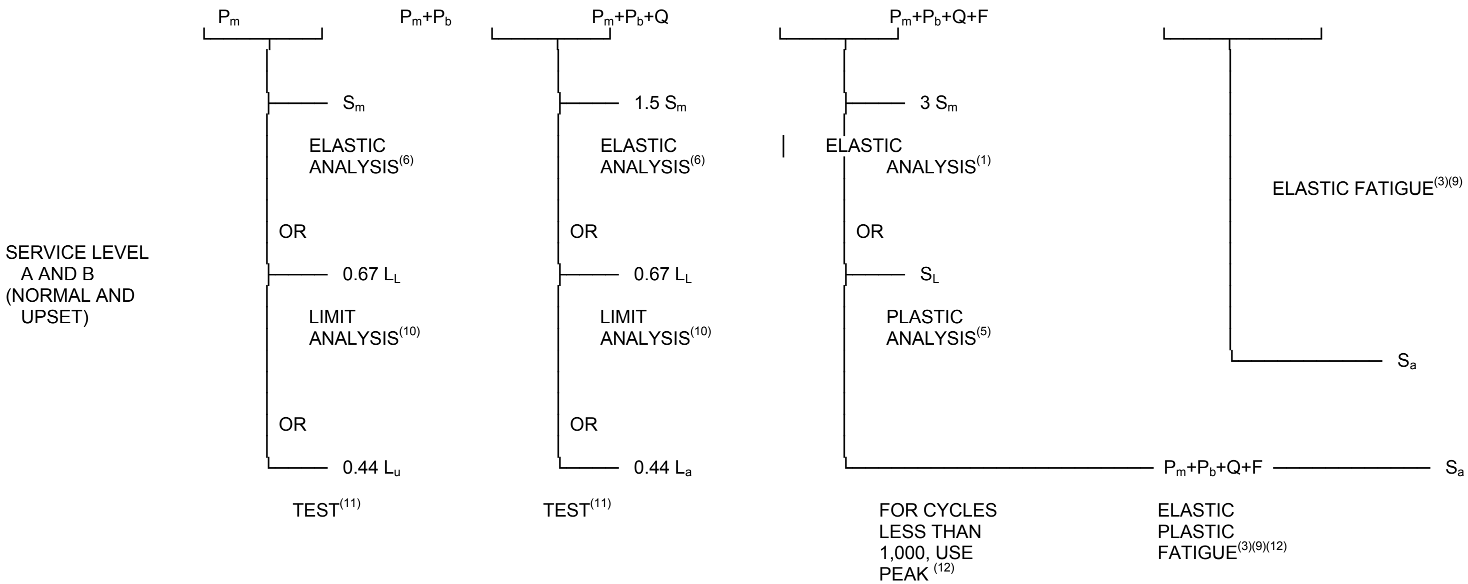

Table 3.9-23 (Cont'd)

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(1) This limitation applies to the range of stress intensity. When the secondary stress is due to a temperature excursion at the point at which the stresses are being analyzed, the value of S_m shall be taken as the average of the S_m values tabulated in tables I-1.1, I-1.2, and I-1.3 of ASME Section III for the highest and the lowest temperature of the metal during the transient. When part of the secondary stress is due to mechanical load, the value of S_m shall be taken as the S_m value for the highest temperature of the metal during transient.

(2) The stresses in Category Q are those parts of the total stress which are produced by thermal gradients, structural discontinuities, etc, and do not include primary stresses which may also exist at the same point. It should be noted, however, that a detailed stress analysis frequently gives the combination of primary and secondary stresses directly and, when appropriate, this calculated value represents the total of $P_m + P_b + Q$ and not Q alone. Similarly, if the stress in Category F is produced by a stress concentration, the quantity F is the additional stress produced by the notch, over and above the nominal stress. For example, if a plate has a nominal stress intensity, $P_m = S$, $P_b = O$, $Q = O$ and a notch with a stress concentration K is introduced, then $F = P_m (K-1)$ and the peak stress intensity equals $P_m + P_m (K-1) = KP_m$.

(3) S_a is obtained from the fatigue curves, figures I-9.1 and I-9.2 of ASME Section III. The allowable stress intensity for the full range of fluctuation is 2 S_a.

(4) The symbols P_m, P_b, Q, and F do not represent single quantities, but rather sets of six quantities representing the six stress components σ_t, σ_1, σ_r, γ_{tl}, γ_{lr}, γ_{rt}.

(5) S_L denotes the structural action of shakedown load as defined in paragraph NB 3213.18 of ASME Section III calculated on a plastic basis as applied to a specific location on the structure.

(6) The triaxial stresses represent the algebraic sum of the three primary principal stresses ($\sigma_1 + \sigma_2 + \sigma_3$) for the combination of stress components. Where uniform tension loading is present, triaxial stresses are limited to 4 S_m.

(7) For configurations where compressive stresses occur, the stress limits shall be revised to take into account critical buckling stresses (see paragraph NB-3211(c) of ASME Section III). For external pressure, the permissible "equivalent static" external pressure shall be as specified by the rules of paragraph NB-3133 of ASME Section III. Where dynamic pressures are involved, the permissible external pressure shall be limited to 25% of the dynamic instability pressure.

(8) When loads are transiently applied, consideration should be given to the use of dynamic load amplification, and possible change in modulus of elasticity.

(9) In the fatigue data curves, where the number of operating cycles are less than 10, use the S_a value for 10 cycles; where the number of operating cycles are greater than 10^6, use the S_a value for 10^6 cycles.

Table 3.9-23 (Cont'd)

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(10) L_L is the lower bound limit load with yield point equal to 1.5 S_m (where S_m is the tabulated value of allowable stress at temperature as contained in ASME Section III). The "lower bound limit load" is here defined as that produced from the analysis of an ideally plastic (nonstrain hardening) material where deformations increase with no further increase in applied load. The lower bound load is one in which the material everywhere satisfies equilibrium and nowhere exceeds the defined material yield strength using either a shear theory or a strain energy of distortion theory to relate multiaxial yielding to the uniaxial case.

(11) For service levels A and B (normal and upset) conditions, the limits on primary membrane plus primary bending need not be satisfied in a component if it can be shown from the test of a prototype or model that the specified loads (dynamic or static-equivalent) do not exceed L_u, where L_u is the ultimate load or the maximum load to load combination used in the test. In using this method, account shall be taken of the size effect and dimensional tolerances which may exist between the actual part and the test part, or parts, as well as differences which may exist in the ultimate strength or other governing material properties of the actual part and the tested part to assure that the loads obtained from the test are a conservative representation of the load-carrying capability of the actual component under the postulated loading for service level A and B (normal and upset) conditions.

(12) The allowable value for the maximum range of this stress intensity is 3 S_m except for cyclic events which occur less than 1000 time during the design life of the plant. For this exception, in lieu of meeting the 3 S_m limit, an elastic-plastic fatigue analysis in accordance with ASME Section III may be performed to demonstrate that the cumulative fatigue usage attributable to the combination of these low events, plus all other cyclic events, does not exceed a fatigue usage value of 1.0.

Table 3.9-24

CORE SUPPORT STRUCTURES
STRESS CATEGORIES AND LIMITS OF STRESS INTENSITY FOR SERVICE LEVEL C (EMERGENCY) CONDITIONS

STRESS CATEGORY	PRIMARY STRESSES: MEMBRANE P_m [1][2][10]		PRIMARY STRESSES: BENDING P_B [1][2][10]		SECONDARY STRESSES: MEMBRANE AND BENDING SECONDARY, Q	PEAK STRESSES: PEAK F
	P_m		P_m+P_B			
SERVICE LEVEL D (FAULT)[9]	1.5 S_m	ELASTIC ANALYSIS[3]	2.25 S_m	ELASTIC ANALYSIS[3]	EVALUATION NOT REQUIRED	EVALUATION NOT REQUIRED
	OR		OR			
	L_L	LIMIT ANALYSIS[4]	L_L	LIMIT ANALYSIS[4]		
	OR		OR			
	1.5 S_n	PLASTIC ANALYSIS[6]	2.25 S_m	PLASTIC ANALYSIS[5][6]		
	OR		OR			
	0.6 L_e	TESTS[7]	0.5 S_u	[5]		
	OR		OR			
	S_E	STRESS RATIO ANALYSIS[8]	0.6 L_e	TESTS[7]		
			OR			
			KS_E	STRESS RATIO ANALYSIS[8]		

Table 3.9-24 (Cont'd)

(1) The symbols P_m, P_b, Q, and F do not represent single quantities, but rather sets of six quantities representing the six stress components σ_t, σ_l, σ_r, γ_{tl}, γ_{lr}, γ_{rt}.

(2) For configurations where compressive stresses occur, stress limits shall be revised to take into account critical buckling stresses. For external pressure, the permissible "equivalent static" external pressure shall be taken as 150% of that permitted by the rules of paragraph NB-3133 of ASME Section III. Where dynamic pressures are involved, the permissible external pressure shall satisfy the preceding requirements or be limited to 50% of the dynamic instability pressure.

(3) The triaxial stresses represent the algebraic sum of the three primary principal stresses ($\sigma_1 + \sigma_2 + \sigma_3$) for the combination of stress components. Where uniform tension loading is present, triaxial stresses should be limited to 6 S_m.

(4) L_L is the lower bound limit load with yield point equal to 1.5 S_m (where S_m is the tabulated value of allowable stress at temperature as contained in ASME Section III). The "lower bound limit load" is here defined as that produced from the analysis of an ideally plastic (nonstrain hardening) material where deformations increase with no further increase in applied load. The lower bound load is one in which the material everywhere satisfies equilibrium and nowhere exceeds the defined material yield strength using either a shear theory or a strain energy of distortion theory to relate multiaxial yielding to the uniaxial case.

(5) S_u is the ultimate strength at temperature. Multiaxial effects on ultimate strength shall be considered.

(6) This plastic analysis uses an elastic-plastic evaluated nominal primary stress. Strain hardening of the material may be used for the actual monotonic stress-strain curve at the temperature of loading or any approximation to the actual stress-strain curve which everywhere has a lower stress for the same strain as the actual monotonic curve may be used. Either the shear or strain energy of distortion flow rule shall be used to account for multiaxial effects.

(7) For service level C (emergency) conditions, the stress limits need not be satisfied if it can be shown from the test of a prototype or model that the specified loads (dynamic or static-equivalent) do not exceed 60% of L_e, where L_e is the ultimate load or the maximum load or load combination used in the test. In using this method, account shall be taken of the size effect and dimensional tolerances which may exist between the actual part and the tested part or parts as well as differences which may exist in the ultimate strength or other governing material properties of the actual part and the tested parts to assure that the loads obtained from the test are a conservative representation of the load-carrying capability of the actual component under postulated loading for service level C (emergency) conditions.

(8) Stress ratio is a method of plastic analysis which uses the stress ratio combinations (combination of stresses that consider the ratio of the actual stress to the allowable plastic or elastic stress) to compute the maximum load a strain hardening material can carry. K is defined as the section factor; $S_e \leq 2\ S_m$ for primary membrane loading.

(9) Where deformation is of concern in a component, the deformation shall be limited to two-thirds the value given for service level C (emergency) conditions in the Design Specification.

(10) When loads are transiently applied, consideration should be given to the use of dynamic load amplification and possible change in modulus of elasticity.

Table 3.9-25

MAXIMUM PRESSURE DIFFERENTIALS ACROSS REACTOR VESSEL INTERNALS DURING A STEAM LINE BREAK

Reactor Component	Pressure Differential(psid)[3] Case 1[1]	Case 2[2]
Core Plate and Guide Tube	24.5	24.5
Shroud Support Ring and Lower Shroud	48	49.0
Upper Shroud	26.5	28.5
Average Channel Wall (Bottom)	13.1	10.6
Top Guide	2.4	3.5

(1) Reactor initially at 1.02 of 110% original power, 110% recirculation flow

(2) Reactor initially at 20% rated steam flow, 110% recirculation flow

(3) Values taken from Reference 3.9-28

Table 3.9-26

CORE SUPPORT STRUCTURE DESIGN LOADING
CONDITIONS AND COMBINATIONS

OPERATING CONDITION AND STRESS LIMITS	SERVICE LEVEL	DESIGN LOADINGS CONDITIONS AND COMBINATIONS
Normal and Upset	A and B	$N + A_D$ and $N + U$
Emergency	C	N and R, or other conditions which have a 40 year encounter probability from 10^{-1} to 10^{-3}
Fault	D	N and A_m and $\overline{R}$ or other conditions which have a 40 year encounter probability from 10^{-3} to 10^{-6}

where:

N	=	service level A (normal) loads
U	=	service level B (upset) loads excluding earthquake
A_D	=	½SSE including any associated transients.
A_m	=	SSE
R	=	automatic blowdown or equivalent auxiliary pipe rupture loading (pipe rupture loadings are not directly considered on piping itself because this is handled by a failure mode analysis)
$\overline{R}$	=	primary loadings which result from rupture of a main steam line or a recirculation line

Table 3.9-27

DEFORMATION LIMIT
(FOR REACTOR INTERNAL STRUCTURES ONLY)

EITHER ONE OF (NOT BOTH)	GENERAL LIMIT
a. $\left[\dfrac{\text{Permissible deformation, DP}}{\text{Analyzed deformation causing loss of function, DL}}\right]$	$\leq \dfrac{0.9}{SF_{min}}$
b. $\left[\dfrac{\text{Permissible deformation, DP}}{\text{Experiment deformation causing loss of function, DE}}\right]^{(1)}$	$\leq \dfrac{1.0}{SF_{min}}$

where:

DP = permissible deformation under stated conditions of service levels A, B, C or D (normal, upset, emergency or faulted)

DL = analyzed deformation which could cause a system loss of function[2]

DE = experimentally determined deformation which could cause a system loss of function

SF_{min} = minimum safety factor

(1) Equation b is not used unless supporting data are provided to the NRC by GE.

(2) "Loss of function" can only be defined quite generally until attention is focused on the component of interest. In cases of interest, where deformation limits can affect the function of equipment and components, they are specifically delineated. From a practical viewpoint, it is convenient to interchange some deformation condition at which function is assured with the loss of function condition if the required safety margins from the functioning conditions can be achieved. Therefore, it is often unnecessary to determine the actual loss of function condition because this interchange procedure produces conservative and safe designs. Examples where deformation limits apply are: CRD alignment and clearances for proper insertion, core support deformation causing fuel disarrangement or excess leakage of any component.

Table 3.9-28

PRIMARY STRESS LIMIT
(FOR SAFETY CLASS REACTOR INTERNAL STRUCTURES ONLY)

ANY ONE OF (NO MORE THAN ONE REQUIRED)		GENERAL LIMIT
a.	$\left[\dfrac{\text{Elastic evaluated primary stresses, PE}}{\text{Permissible primary stresses, PN}}\right]$	$\leq \dfrac{2.25}{SF_{min}}$
b.	$\left[\dfrac{\text{Permissible load, LP}}{\text{Largest lower bound limit load, CL}}\right]$	$\leq \dfrac{1.5}{SF_{min}}$
c.	$\left[\dfrac{\text{Elastic evaluated primary stresses, PE}}{\text{Conventional ultimate strength at temperature, US}}\right]$	$\leq \dfrac{0.75}{SF_{min}}$
d.	$\left[\dfrac{\text{Elastic - plastic evaluated nominal primary stress, EP}}{\text{Conventional ultimate strength at temperature, US}}\right]$	$\leq \dfrac{0.9}{SF_{min}}$
e.	$\left[\dfrac{\text{Permissible load, LP}}{\text{Plastic instability load, PL}}\right]^{(1)}$	$\leq \dfrac{0.9}{SF_{min}}$
f.	$\left[\dfrac{\text{Permissible load, LP}}{\text{Ultimate load from fracture analysis, UF}}\right]^{(1)}$	$\leq \dfrac{0.9}{SF_{min}}$
g.	$\left[\dfrac{\text{Permissible load, LP}}{\text{Ultimate load or loss of function load from test, LE}}\right]^{(1)}$	$\leq \dfrac{1.0}{SF_{min}}$

Table 3.9-28 (Cont'd)

where:

PE	=	primary stresses evaluated on an elastic basis. The effective membrane stresses are to be averaged through the load-carrying section of interest. The simplest average bending, shear or torsion stress distribution which will support the external loading will be added to the membrane stresses at the section of interest.
PN	=	permissible primary stress levels under service levels A or B (normal or upset) conditions under ASME Section III.
LP	=	permissible load under stated conditions of service levels A, B, C, or D (emergency or faulted).
CL	=	lower limit load with yield point equal to 1.5 S_m where S_m is the tabulated value of allowable stress at temperature of the ASME Section III Code or its equivalent. The "lower bound limit load" is here defined as that produced from the analysis of an ideally plastic nonstrain hardening material where deformations increase with no further increase in applied load. The lower bound load is one in which the material everywhere satisfies equilibrium and nowhere exceeds the defined material yield strength using either a shear theory or a strain energy of distortion theory to relate multiaxial yield to the uniaxial case.
US	=	conventional ultimate strength at temperature or loading which would cause a system malfunction, whichever is more limiting.
EP	=	elastic-plastic evaluated nominal primary stress. Strain hardening of the material may be used for the actual monotonic stress-strain curve at the temperature of loading or any approximation to the actual stress-strain curve which everywhere has a lower stress for the same strain as the actual monotonic curve may be used. Either the shear or strain energy of distortion flow rule may be used.
PL	=	plastic instability load. The "plastic instability load" is defined here as the load at which any load- bearing section begins to diminish its cross-sectional area at a faster rate than the strain hardening can accommodate the loss in area. This type analysis requires a true stress-true strain curve or a close approximation based on monotonic loading at the temperature of loading.

Table 3.9-28 (Cont'd)

UF	=	ultimate load from fracture analyses. For components which involve sharp discontinuities (local theoretical stress concentration <3) the use of a "fracture mechanics" analysis where applicable utilizing measurements of plane strain fracture toughness may be applied to compute fracture loads. Correction for finite plastic zones and thickness effects as well as gross yielding may be necessary. The methods of linear-elastic stress analysis may be used in the fracture analysis where its use is clearly conservative or supported by experimental evidence. Examples where "fracture mechanics" may be applied are for fillet welds or end of fatigue life crack propagation.
LE	=	ultimate load or loss of function load as determined from experiment. In using this method, account shall be taken of the dimensional tolerances which may exist between the actual part and the tested part or parts as well as differences which may exist in the ultimate tensile strength of the actual part and the tested parts. The guide to be used in each of these areas is that the experimentally determined load shall use adjusted values to account for material property and dimension variations, each of which has no greater probability than 0.1 of being exceeded in the actual part.
SF_{min}	=	minimum safety factor

[(1)] Equations e., f., and g. are not used unless supporting data are provided to the NRC by GE.

Table 3.9-29

BUCKLING STABILITY LIMIT
(FOR SAFETY CLASS REACTOR INTERNAL STRUCTURES ONLY)

ANY ONE OF (NO MORE THAN ONE REQUIRED)		GENERAL LIMIT
a.	$\left[\dfrac{\text{Permissible load, LP}}{\text{Service level A (normal) permissible load, PN}}\right]$	$\leq \dfrac{2.25}{SF_{min}}$
b.	$\left[\dfrac{\text{Permissible load, LP}}{\text{Stability analysis load, SL}}\right]$	$\leq \dfrac{0.9}{SF_{min}}$
c.	$\left[\dfrac{\text{Permissible load, LP}}{\text{Ultimate buckling collapse load from test, SE}}\right]^{(1)}$	$\leq \dfrac{1.0}{SF_{min}}$

where:

LP = permissible load under stated conditions of service levels A, B, C or D (normal, upset emergency or faulted)

PN = applicable service level A (normal) permissible load

SL = stability analysis load. The ideal buckling analysis is often sensitive to otherwise minor deviations from ideal geometry and boundary conditions. These effects shall be accounted for in the analysis of the buckling stability loads. Examples of this are ovality in externally pressurized shells or eccentricity on column members.

SE = ultimate buckling collapse load as determined from experiment. In using this method, account shall be taken of the dimensional tolerances which may exist between the actual part and the tested part. The guide to be used in each of these areas is that the experimentally determined load shall be adjusted to account for material property and dimension variations, each of which has no greater probability than 0.1 of being exceeded in the actual part.

SF_{min} = minimum safety factor

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(1) Equation c. is not used unless supporting data are provided to the NRC by GE.

Table 3.9-30

CORE SUPPORT STRUCTURES
STRESS CATEGORIES AND LIMITS OF STRESS INTENSITY FOR SERVICE LEVEL D (FAULT) CONDITIONS

	PRIMARY STRESSES				SECONDARY STRESSES	PEAK STRESSES
STRESS CATEGORY	MEMBRANE P_m[1][2][10]		BENDING P_B[1][2][10]		MEMBRANE AND BENDING SECONDARY, Q	PEAK F
	P_m		P_m+P_B			
	2.4 S_m	ELASTIC ANALYSIS[3]	3.0 S_m	ELASTIC ANALYSIS[3]	EVALUATION NOT REQUIRED	EVALUATION NOT REQUIRED
	OR		OR			
	0.75 S_u	[5][10]	1.33 L_L	LIMIT ANALYSIS[4]		
	OR		OR			
	1.33 I_L	LIMIT ANALYSIS[4][11]	0.75 S_u	PLASTIC ANALYSIS[5][6]		
	OR		OR			
	0.67 S_u	PLASTIC ANALYSIS[5][6][11]	0.8 L_F	TESTS[7]		
	OR		OR			
	0.8 L_F	TESTS[7][11]	KS_F	STRESS-RATIO ANALYSIS[8]		
	OR					
	S_F	STRESS-RATIO ANALYSIS[8]				

Table 3.9-30 (Cont'd)

(1) The symbols P_m, P_b, Q, and F do not represent single quantities, but rather sets of six quantities representing the six stress components δ_t, δ_l, δ_r, γ_{tl}, γ_{tr}, γ_{rl}.

(2) When loads are transiently applied, consideration should be given to the use of dynamic load amplification and possible changes in modulus of elasticity.

(3) For configurations where compressive stresses occur, stress limits take into account critical buckling stresses. For external pressure the permissible "equivalent static" external pressure shall be taken as 2.5 times that given by the rules of paragraph NB-3133 of ASME Section III. Where dynamic pressures are involved, the permissible external pressure satisfies the preceding requirements or be limited to 75% of the dynamic instability pressure.

(4) L_L is the lower bound limit load with yield point equal to 1.5 S_m (where S_m is the tabulated value of allowable stress at temperature as contained in ASME Section III). The "lower bound limit load" is defined as that produced from the analysis of an ideally plastic (nonstrain hardening) material where deformations increase with no further increase in applied load. The lower bound load is one in which the material everywhere satisfies equilibrium and nowhere exceeds the defined material yield strength using either a shear theory or a strain energy of distortion theory to relate multiaxial yielding to the uniaxial case.

(5) S_u is the ultimate strength at temperature. Multiaxial effects on ultimate strength are considered.

(6) This plastic analysis uses an elastic-plastic evaluated nominal primary stress. Strain hardening of the material may be used for the actual monotonic stress-strain curve at the temperature of loading or any approximation to the actual stress-strain curve which everywhere has a lower stress for the same strain as the actual monotonic curve may be used. Either the maximum stress or strain energy of distortion flow rule shall be used to account for multiaxial effects.

(7) For service level D (fault) conditions, the stress limits need not be satisfied if it can be shown from the test of a prototype or model that the specified loads (dynamic or static-equivalent) do not exceed 80% of L_F, where L_F is the ultimate load or load combination used in the test. In using this method, account is taken of the size effect and dimensional tolerances, as well as differences which may exist in the ultimate strength or other governing material properties of the actual part and the tested parts, to assure that the loads obtained from the test are a conservative representation of the load-carrying capability of the actual component under postulated loading for service level D (fault) condition.

(8) Stress ratio is a method of plastic analysis which uses the stress ratio combinations (combination of stresses that consider the ratio of the actual stress to the allowable plastic or elastic stress) to compute the maximum load a strain hardening material can carry. K is defined as the section factor; S_f is the lesser of 2.4 S_m or 0.75 S_u for primary membrane loading.

(9) Where deformation is of concern in a component, the deformation is limited to 80% of the value given for service level D (fault) conditions in the Design Specifications.

(10) 0.7 S_u per ASME Section III Appendix F.

(11) Same as ASME Section III Appendix F.

(12) 3.6 S_m per ASME Section III Appendix F.

Table 3.9-31

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Table 3.9-32

SRV TEST PROGRAM

	EVENTS [1][2]												
PLANT FEATURES	1	2	3	4	5	6	7	8	9	10	11	12	13
High Water Level 7 Alarm	X, S		X, S	X, S	X, NA				X, S	X, NA	X, S	X, S	X, NA
High Drywell Pressure Alarm													
FW Level 8 Trip	X, S	X, S											
RCIC Level 8 Trip			X, S	X, S	X, NA				X, S	X, NA	X, S		X, NA
HPCS Level 8 Trip				X, NA	X, NA				X, NA	X, NA			X, NA
HPCI Level 8 Trip			X, S	X, S					X, S		X, S		X, NA
HPCI/S and RCIS Initiation on Low Water Level	X, S	X, S	X, S	X, S	X, NA	X, NA		X, S	X, S				X, NA
HPCI/S Initiation of High Drywell Pressure			X, S	X, S					X, S	X, NA	X, S	X, S	X, NA
RCIC Initiation on High Drywell Pressure													X, NA
Low Pressure ECCS Initiation on High Drywell Pressure												X, S	X, NA
Low Pressure ECCS Initiation on Low Water Level													X, NA
FW Pumps Trip on Low Suction Pressure	X, S												
HPCI Trip on High Back pressure			X, S								X, S		
RCIC Trip on High Back pressure				X, S					X, S				
Turbine Trip on Vessel High Level	X, S	X, S											
MSIVs Closure on Low Turbine Inlet Pressure	X, S	X, S					X, S						
MSIVs Closure on High Steam Flow		X, S					X, S						

Table 3.9-32 (Cont'd)

PLANT FEATURES	EVENTS [1][2] 1	2	3	4	5	6	7	8	9	10	11	12	13
MSIVs Closure on High Steam Tunnel Temperature								X, S					
MSIV Closure on High Radiation								X, S					
Reactor Scram on Turbine Trip	X, S	X, S											
Reactor Scram on Nuetron Flux Monitor		X, S											
Reactor Scram on MSIVs Closure		X, S											
Reactor Scram on High Radiation								X, S					
Reactor Scram on High Drywell Pressure									X, S	X, NA	X, S	X, S	X, NA
Reactor Scram on Low Water Level													X, NA
Reactor Isolation on Low Water Level													X, NA

(1) Events

1 FW Cont. Failure, FW L8 Trip Failure
2 Pressure Regulator Failure
3 Transient HPCI, HPCI L8 Trip Failure
4 Transient RCIC, RCIC L8 Trip Failure
5 Transient HPCS, HPCS L8 Trip Failure
6 Transient RCIC Hd. Spr.
7 Alternate Shutdown Cooling, Shutdown Suction Unavailable
8 Main Steam Line Break - Outside Containment
9 SBA, RCIC, RCIC L8 Trip Failure
10 SBA, HPCS, HPCS L8 Trip Failure
11 SBA, HPCI, HPCI L8 Trip Failure
12 SBA, Depressurization & ECCS Overfill, Operator Error
13 LBA, ECCS Overfill Break Isolation

(2) X - Feature considered in Base Case Analysis
S - Feature in Plant Specific Design
NA - Not Applicable

3.10 QUALIFICATION OF SEISMIC CATEGORY I INSTRUMENTATION AND ELECTRICAL EQUIPMENT

The dynamic qualification criteria applicable to the seismic Category I instruments, electrical equipment, and their supports are provided in this section. The methods and procedures used to qualify them are also discussed. Seismic Category I instruments, equipment, and supporting structures are identified in Table 3.2-1.

3.10.1 DYNAMIC QUALIFICATION CRITERIA

The seismic Category I instruments and electrical equipment are designed to withstand the effects of the SSE defined in Section 3.7 and the hydrodynamic loads discussed in Section 3.9 and Appendix 3A.4 through 3A.7 without functional impairment.

The qualification discussion covered in the following sections is generally divided into two types of equipment: NSSS equipment and non-NSSS equipment.

The following design criteria and qualification procedures for NSSS equipment include the effects of both seismic and hydrodynamic loads. The non-NSSS equipment qualification discussions include only seismic design criteria and procedures. Refer to Appendix 3A.6.8 and 3A.7.1.7 for non-NSSS equipment subjected to hydrodynamic loads. Appendix 3A.6.7 and 3A.7.1.6 contain further discussion of NSSS equipment qualification.

3.10.1.1 Dynamic Loading Design Criteria (NSSS Equipment)

The criterion used in the design and subsequent qualification of all Class 1E instruments and electrical equipment supplied by GE is as follows: "The Class 1E equipment shall be capable of performing all safety-related functions during (1) normal plant operation, (2) anticipated transients, (3) DBAs, and (4) postaccident operation while being subjected to, and after the cessation of, the accelerations resulting from the SSE and hydrodynamic loads at the point of attachment of the equipment to the building or supporting structure."

The criteria for each of the devices used in the Class 1E systems depend on the use in a given system; for example, a relay in one system may have as its safety function to de-energize and open its contacts within a certain time, while in another system it must energize and close its contacts. Since GE supplies many devices for many applications, the approach taken was to test the device in the worst case configuration. In this way, the capability of protective action initiation and the proper operation of safety-related circuits is assured.

From the basic input ground motion data, a series of response curves at various structure elevations is developed after the building layout is completed. Standard requirement levels that meet or exceed the maximum expected unique plant information are included in the purchase specifications for seismic Category I equipment. Suppliers of equipment such as batteries and racks, instrument racks, control consoles, etc., are required to submit test data, operating experience and/or calculations to substantiate that their components, systems, etc will not suffer loss of function during or after dynamic loadings. The magnitude and frequency content of the loadings which each component will experience are determined by its specific location within the plant. All Class 1E equipment will be evaluated for the capability of performing its safety function during and after dynamic loading combinations given in Table 3.9-6.

3.10.1.2 Dynamic Loading Design Criteria (Non-NSSS Equipment)

The seismic Category I instruments and electrical equipment is the equipment that is designed to maintain its functional capability and/or to maintain the pressure boundary integrity during and after an SSE and at least 5 OBEs. (Refer to Appendix 3A.6.8 for criteria for equipment subjected to hydrodynamic loads.)

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

a. OBE Conditions:

The load combinations include gravity loads and operation loads (or LOCA loads, if applicable) including associated temperatures and pressures, combined with the seismic loading of an OBE.

Stresses in the structural steel portions may be increased to 125% of the allowable working stress limits accepted as good practice as set forth in the appropriate design standards; that is, AISC Manual of Steel Construction, ASME B&PV Code, ANSI B31.1 and B31.7 codes for pressure piping, or other equivalent industrial codes. The resulting deflections do not prevent continuous normal operation of the equipment during and after the seismic disturbance.

b. SSE Conditions:

The load combinations include gravity loads and operating loads (or LOCA loads, if applicable) including associated temperatures and pressures combined with the seismic loading of the SSE. Stresses in the structural portions may be increased to 150% of code allowable working stress limits but are not to exceed 0.9 F_y in bending, 0.85 F_y for tension, and 0.5 F_y in shear, where (F_y) equals the material yield stress at the design temperature. The resulting deflections will not prevent the operation of the equipment during and after the seismic disturbance.

The performance requirements of the seismic Category I items and their respective supports are structural as well as functional. Where applicable, the structural requirements are in accordance with AISC "Specifications for the Design, Fabrication, and Erection of Structural Steel for Buildings," adopted February 12, 1969, or similar codes applicable for other construction materials.

The structural requirements for electrical and instrumentation equipment and systems that are required to maintain pressure boundary integrity are in accordance with the ASME Section III.

3.10.2 METHODS AND PROCEDURES FOR QUALIFYING ELECTRICAL EQUIPMENT AND INSTRUMENTATION

Seismic Category I instruments and electrical equipment are qualified according to the criteria discussed in Section 3.10.1 by the methods and procedures described in this section. The qualification methods and procedures are discussed in two parts: NSSS and non-NSSS equipment; both of which were re-assessed to SRP 3.10 Seismic Qualification Review Team (SQRT) requirements including IEEE 344-1975, and Reg. Guides 1.10 and 1.92. The SQRT re-assessment concluded that the seismic and dynamic qualification program meets the intent of IEEE 344-1975 and Reg. Guides 1.100 and 1.92.

3.10.2.1 Methods and Procedures for Qualifying NSSS Electrical Equipment and Instruments (Excluding Motors and Valve-Mounted Equipment)

3.10.2.1.1 Methods of Showing NSSS Equipment Compliance with IEEE 344 (1975) and Regulatory Guide 1.100

Originally, NSSS equipment were qualified to IEEE 344 (1971), which was the plant commitment, and as such did not demonstrate compliance with Regulatory Guide 1.100 and other SQRT criteria. However, a re-assessment of all NSSS equipment qualifications to NEDE-24788 "SQRT Technical Approach" ensures an adequate degree of equipment conformance to IEEE 344-1975 and Regulatory Guides 1.100 and 1.92 requirements which represents an acceptable basis for qualifying the equipment.

GE-supplied Class 1E equipment meets the requirement that the dynamic qualification should demonstrate the capability to perform the required function during and after the dynamic event. Both analysis and testing were used, but most equipment was tested. Analysis was primarily used to determine the adequacy of mechanical strength (mounting bolts, etc) after operating capability was established by testing.

a. Analysis

GE-supplied Class 1E equipment performing primarily a mechanical safety function (pressure boundary devices, etc) was analyzed, since the passive nature of its critical safety role usually made testing impractical. Analytical methods sanctioned by IEEE 344 (1971) were used in such cases and were re-evaluated to IEEE 334 (1975) criteria with satisfactory results. Table 3.10-1 shows which items were qualified by analysis.

b. Testing

GE-supplied Class 1E equipment having primarily an active electrical safety function was tested in compliance with IEEE 344 (1971), section 3.2 and were re-evaluated to IEEE 344 (1975) criteria with satisfactory results.

Available documentation verifies that the dynamic qualification of GE-supplied Class 1E equipment is in accordance with the requirements of IEEE 344 (1975).

3.10.2.1.2 Testing Procedures for Qualifying NSSS Electrical Equipment and Instruments (Excluding Motors and Valve-Mounted Equipment)

The test procedure requires that the device be mounted on the table of the vibration machine in a manner similar to how it is normally installed. The device is tested in the operating states as if it were performing its Class 1E functions and these states are monitored before, during, and after the test to assure proper function and absence of spurious function. In the example of a relay, both energized and de-energized states and normally open and normally closed contact configurations are tested if the relay is used in those configurations in its Class 1E functions.

The dynamic excitation is a single-frequency test in which the applied vibration is a sinusoidal table motion at a fixed peak acceleration and a discrete frequency at any given time. The vibratory excitation is applied in three orthogonal axes individually, with the axes chosen as those coincident with the most probable mounting configuration.

The first step is to search for resonances in each axis. This is done because resonances cause amplification of the input vibration and are the most likely cause of malfunction. The resonance search is usually run at low acceleration levels (0.2 g) to avoid damaging the test sample in case a severe resonance is encountered. The resonance search is run in accordance with IEEE 344; if the device is large enough, the vibrations are monitored by accelerometers placed at critical locations, from which resonances are determined by comparing the acceleration level with that at the table of the vibration machine. If the devices are either too small for an accelerometer, have their critical parts in an inaccessible location, or have critical parts that would be adversely affected by the mounting of an accelerometer, the vibrations are monitored at the closest location.

Following the frequency scan and resonance determination, the devices are tested to determine their malfunction limit. This test is a necessary adjunct to the assembly test, as shown later. The malfunction limit test is run at each resonant frequency as determined by the frequency scan. In this test, the acceleration level is gradually increased until either the device malfunctions or the limit of the vibration machine is reached. If no resonances are detected (as is usually the case), the device is considered to be rigid (all parts move in unison), and the malfunction limit is therefore independent of frequency. To achieve maximum acceleration from the vibration machine, rigid devices are malfunction tested at the upper test frequency because that allows the maximum acceleration to be obtained from deflection-limited machines.

The summary of the tests on the devices used in Class 1E applications given in Table 3.10-1 includes the "qualification" limit for each device tested.

The above procedures are required of purchased devices, as well as of those made by GE. Vendor test results are reviewed, and, if unacceptable, the tests are repeated either by GE or by the vendor. If the vendor tests are adequate, the device is considered qualified to the limits of the test.

3.10.2.1.3 Qualification of Valve-Mounted Equipment

The piping analysis establishes the response spectra, the power spectral density function or time history characteristics, and develops horizontal and vertical accelerations for the pipe-mounted equipment. Class 1E MOV actuators are qualified in accordance with IEEE 382 (1972).

The SRV, including the electrical components mounted on the valve, is subjected to a dynamic test. This testing is described in Sections 3.9.2.2a.2.14 and 3.9.3.1.13.

3.10.2.1.4 Qualification of NSSS Motors

The seismic qualification of the ECCS motors is discussed in Section 3.9.2.2a.2.7 in conjunction with the ECCS pump and motor assembly. The seismic qualification of the SLCS pump motor is discussed in Section 3.9.2.2a.2.10 in conjunction with the SLC pump motor assembly.

3.10.2.2 Methods and Procedures for Qualifying Non-NSSS Instruments and Electrical Equipment

In regard to compliance with Regulatory Guide 1.100, the analysis and testing for the seismic qualification of non-NSSS Class 1E instruments and electrical equipment required to function during and after an SSE are in compliance with IEEE 344 (1971) for components purchased before issuance of IEEE 344 (1975) and are in compliance with IEEE 344 (1975) for components purchased after its issuance. In addition, all non-NSSS instruments and controls were re-evaluated to IEEE 344 (1975) and Reg. Guide 1.100 requirements (SQRT requirements) and found satisfactory.

Pipe-mounted instrumentation is qualified by analysis and/or testing to the acceleration levels allowed for piping systems. These levels include gravity and operation loading, as well as loading that is due to seismic excitation. Passive instruments that must only maintain mechanical and pressure boundary integrity are tested to the acceleration levels of the response spectra for the area in which they are to be installed.

Seismic Category I equipment is shown to be capable of withstanding the horizontal and vertical accelerations of five OBEs and one SSE by one of the following methods: dynamic analysis; dynamic testing; or a combination of dynamic analysis and testing. (Refer to Appendix 3A.7.1.7 for qualification methods for equipment subjected to hydrodynamic loads.)

3.10.2.2.1 Dynamic Analysis

For the analysis, equipment is idealized as a system of lumped masses and springs for which frequencies and mode shapes are determined for vibration in the vertical direction and two orthogonal horizontal directions. For each direction of vibration, the spectral accelerations per mode are obtained from the appropriate spectrum response curve corresponding to the location and damping of the equipment. Seismic loading in terms of inertia forces, moments, and shears is determined for each direction using the spectrum response method summing the absolute values per mode. If the orientation of the equipment is not designated, the horizontal seismic loading is taken as the maximum loading (worst case) obtained using each horizontal direction of vibration and the appropriate horizontal spectrum response curve(s). If the frequencies of all equipment modes (determined by either analysis or testing) are greater than the frequency of the appropriate spectrum response curve at which the acceleration is constant in the rigid (high frequency) range, the seismic loading consists of the static loading corresponding to that acceleration level.

If the equipment damping is unknown, the following values shall be used:

a. OBE - ½% damping

b. SSE - 1% damping

In lieu of determining the vibrational frequencies of equipment, the seismic loading of structurally simple equipment (that can be adequately represented as a single mass and spring) consists of a static load corresponding to 1.5 times the peak acceleration of the appropriate response spectrum curve. Total seismic loading consists of both the vertical seismic loading and the maximum horizontal seismic loading applied to the equipment simultaneously.

Where equipment must meet IEEE 344 (1975), the dynamic analysis is in accordance with section 5 of IEEE 344 (1975). Equipment qualified by analysis to IEEE 344 (1971) were re-evaluated to IEEE 344 (1975) criteria and found satisfactory.

3.10.2.2.2 Dynamic Tests

In lieu of performing a dynamic analysis, seismic adequacy is established by providing dynamic test or previous dynamic environmental (performance) data that demonstrate that the equipment meets the seismic design criteria as defined in this section. The previous data include at least one of the following:

a. Recent test data acquired from dynamic tests of equipment

b. Dynamic test data from previously tested comparable equipment

c. Performance data from equipment that, during normal operating conditions, is subjected to dynamic loads equal to or greater than those defined in this section

3.10.3 METHODS AND PROCEDURES OF ANALYSIS OR TESTING OF SUPPORTS OF ELECTRICAL EQUIPMENT AND INSTRUMENTATION

3.10.3.1 Dynamic Analysis Testing Procedures And Restraint Measures For NSSS Equipment Supports (Other Than Motors And Valve-Mounted Equipment)

Some GE-supplied Class 1E devices are qualified by analysis only (Table 3.10-1). Analysis is used for passive mechanical devices and sometimes is used in combination with testing for larger assemblies containing safeguard devices. For instance, a test may be run to determine if there are natural frequencies in the equipment within the critical frequency range. If the equipment is determined to be free of natural frequencies, then it is assumed to be rigid and a static analysis is performed. If the equipment has natural frequencies in the critical frequency range, then calculations of transmissibility are performed and responses to varying input accelerations are determined to see whether Class 1E devices mounted in the assembly would operate without malfunctioning. In general, the testing of Class 1E equipment is accomplished using the following procedure.

Assemblies (i.e., control panels) containing devices that have had dynamic loading malfunction limits established are tested by mounting the assembly on the table of a vibration machine, in a manner similar to that in which it is to be mounted when in use, and vibration testing the assembly by running a low level resonance search. As with the devices, the assemblies are tested in the three major orthogonal axes. The resonance search is run in the same manner as described for devices. If resonances are present, the transmissibility between the input and the locations of each Class 1E device is determined by measuring the accelerations at each device location and calculating the magnification between it and the input. Once known, the transmissibilities could be used analytically to determine the response at any Class 1E device location for any given input. It is assumed that the transmissibilities are linear as a function of acceleration, even though they actually decrease as acceleration is increased; therefore, this is a conservative assumption. As long as the device input accelerations are determined to be below their malfunction limits, the assembly is considered a rigid body with a transmissibility equal to 1, so that a device mounted on it is limited directly by the assembly input acceleration.

Since control panels and racks constitute the majority of seismic Category I electric assemblies supplied by GE, the qualification testing of these is discussed in more detail. There are basically four generic panel types: vertical board; instrument rack; local rack; and NEMA-12 enclosures. One or more of each type is tested using the above procedures.

Figures 3.10-1 through 3.10-4 illustrate the four basic panel types referenced above and show typical accelerometer locations. The status of the dynamic tests on the Class 1E panels supplied by GE for LGS is summarized in Table 3.10-2.

The full acceleration level tests described above disclose that most of the panel types have more than adequate mechanical strength and that a given panel design acceptability is just a function of its amplification factor and the malfunction levels of the devices mounted in it. Subsequent panels are therefore tested at lower acceleration levels, and the transmissibilities are measured to the various devices as described above. By dividing the devices' malfunction levels by the panel transmissibility between the device and the panel input, the panel qualification level can be determined. Several high level tests have been run on selected generic panel designs to assure the conservatism in using the transmissibility analysis described.

3.10.3.2 Non-NSSS Equipment Supports

Analyses or tests are performed for all supports of electrical equipment and instruments, such as switchgear, battery racks, instrument racks, control consoles, cabinets, and panels to ensure their structural capability to withstand seismic excitation.

The following bases are used in the seismic design and analysis of seismic Category I instrument tubing supports:

a. All instrument tubing supports are qualified by analysis, using the response spectrum method described in Section 3.10.2.2.1.

b. Analysis and design of seismic restraint measures for instrument tubing supports are based on combined limiting values for static load, span length, and computed seismic response.

c. Maximum stress is limited to 90% of minimum yield stress.

d. The seismic Category I instrument tubing systems are supported so that the allowable stresses permitted by ASME Section III are not exceeded when the tubing is subjected to the loads specified in Section 3.9 for Class 2 and 3 piping.

For field-mounted instruments, the following are applicable. (Refer to Appendix 3A.7.1.7 for instrument supports subjected to hydrodynamic loads.)

a. The mounting structures for seismic Category I instruments have a fundamental frequency of 33 Hz or higher (rigid range), which corresponds to the maximum floor acceleration. Therefore the ZPA for the installation is applicable.

b. The stress level in the mounting structure does not exceed the material allowable stress when the mounting structure is subjected to the maximum acceleration level for its location, in combination with other design loads.

3.10.4 OPERATING LICENSE REVIEW

3.10.4.1 NSSS Equipment

3.10.4.1.1 NSSS Control and Electrical Equipment (Other Than Motors and Valve-Mounted Equipment)

The qualification test plans and results for safety-related panels and control equipment within the NSSS scope of supply are maintained as follows:

a. Proprietary documents will be maintained by GE in a centrally located, readily auditable, permanent file.

b. Nonproprietary summary documents will be maintained by the licensee in a centrally located, readily auditable, permanent file.

If equipment fails to pass the tests, it is rejected. In some cases, equipment that fails one test is modified or repaired to meet the performance requirements and is retested. If the retested equipment passes the latter test, it may be used in a Class 1E application.

Table 3.10-1 lists the NSSS control devices by item number and vendor. The table also gives the corresponding acceleration levels for the devices used in Class 1E applications. The acceleration level shown in the right columns of Table 3.10-1 is the acceleration at which either the device malfunctioned or the limit of the vibration machine was reached.

3.10.4.1.2 NSSS Motors

Qualification test results for the ECCS motors are discussed in Section 3.9.2.2a.2.7 in conjunction with the ECCS pump and motor assembly. Qualification test results for the SLCS motor are discussed in Section 3.9.2.2a.2.10 in conjunction with the SLCS pump motor assembly.

3.10.4.1.3 Valve-Mounted Equipment

The SRVs, including the electrical components mounted on the valves, are subjected to dynamic tests. The results of these tests are discussed in Sections 3.9.2.2a.2.14 and 3.9.3.1.13, and are maintained in the same manner as discussed in Section 3.10.4.1.1.

3.10.4.2 Non-NSSS Equipment

A list of dynamic qualification packages for non-NSSS safety-related instruments and electrical equipment is given in Table 3.10-3. The qualification packages will be maintained by the licensee in a centrally located, readily auditable permanent file.

Table 3.10-1

NSSS ESSENTIAL ELECTRICAL COMPONENTS AND INSTRUMENTS (CLASS 1E) *

DESCRIPTION				SEISMIC QUALIFICATION			
					SEISMIC QUALIFICATION[3] (g)		
ITEM NO.	NAME	VENDOR	QUANTITY	OTHERS OF SAME TYPE IN SIMILAR AREA	F-B	S-S	V
System Title - Reactor							
B11-D193	Power range detector[2]	GE	43	-	4.5	4.5	4.5
System Title - Nuclear Boiler							
B21-K613	Power supply	Elma Eng.	2	-	7.0	7.0	5.5
B21-N004	Temp. element & thermowell[1]	PYCO	14	-			
B21-N010	Temp. element	PYCO	19	N011-N014,N016,N017	5.0	5.0	5.0
B21-N027	Level transmitter	Rosemount	1	-	10.0	5.8	1.8
B21-N040	Temp. element (thermowell)	Rosemount	3	N057	S.A.	S.A.	S.A.
B21-N064	Temp. element & thermowell[1]	Calif. Alloy	1	-			
B21-N075	Press. transmitter	Rosemount	4	-	10.0	5.8	1.8
B21-N076	Press. transmitter	Rosemount	4		10.0	5.8	1.8
B21-N078	Press. transmitter	Rosemount	24	N090,N094	10.0	5.8	1.8
B21-N080	Level transmitter	Rosemount	38	N081,N085-N089, N091,N095,N097	10.0	5.8	1.8
B21-N402	Level transmitter	Rosemount	4	-	9.4	9.4	7.2
B21-N403	Press. transmitter	Rosemount	4	-	9.4	9.4	7.2
B21-N600	Temp. switch	Riley Inst.	18	N603,N605-N608	4.5	4.5	4.5
B21-N675	Press. indicator switch	Rosemount	86	N676,N678-N681,N684, N686-N695,N693A,E,B,F, N697	15.0	15.0	15.0
B21-N693	Trip unit	Rosemount	4	N698	15.0	15.0	15.0
B21-R005	Diff. press. indicator	Barton	1	-	5.0	10.0	10.0
System Title - Reactor Recirculation							
B32-N014	Flow transmitter	Rosemount	8	N024	10.0	5.8	1.8
B32-N015	Diff. press. transmitter	Rosemount	2	-	10.0	5.8	1.8
B32-N023	Temp. element	Rosemount	2	-	S.A.	S.A.	S.A.

* NOTE: Table for historical purposes only.

Table 3.10-1 (Cont'd)

NSSS ESSENTIAL ELECTRICAL COMPONENTS AND INSTRUMENTS (CLASS 1E) *

DESCRIPTION				SEISMIC QUALIFICATION			
					SEISMIC QUALIFICATION(3) (g)		
ITEM NO.	NAME	VENDOR	QUANTITY	OTHERS OF SAME TYPE IN SIMILAR AREA	F-B	S-S	V
System Title - CRD HCU							
C11-N012	Level transmitter	Gould	4	-	5.0	5.0	10.2
C11-N013	Level switch	Magnetrol	4	-	4.6	4.6	3.6
C11-N013 (E-H)	Level switch	Magnetrol	4	-	4.1	4.1	9.5
C11-N601	Level indicator switch	Rosemount	4	-	15.0	15.0	15.0
System Title - Feedwater Control							
C32-N003	Diff. press. transmitter	Rosemount	8	N004	10.0	5.8	1.8
C32-N005	Press. transmitter	Rosemount	2	N008	10.0	5.8	1.8
C32-N017	Level transmitter	Statham	1	-	10.0	5.8	1.8
System Title - Standby Liquid							
C41-N003	Temp. switch	Fenwell	1	-	S.A.	S.A.	S.A.
C41-N004	Press. transmitter	Rosemount	3	-	9.4	9.4	7.2
C41-N006	Temp. element(1)	Fenwell	1	-			
C41-N010	Level transmitter	Gould	6	-	5.0	5.0	10.2
C41-N610	Level switch	GE	2	-	2.4	2.4	4.0
System Title - Neutron Monitoring							
C51-K002	Voltage preamplifier	GE	8	-	≥10.0	≥10.0	≥10.0
C51-K601	Intermediate range monitor	GE	8	-	8.5	8.5	8.5
C51-K605	Power range neutron monitor	GE	1		3.8	3.8	3.0
C51-N002	Detector	GE	8	-	S.A.	S.A.	S.A.
System Title - Remote Shutdown							
C61-K001	Sq Root converter	GE	1	-	9.0	9.0	13.0
C61-K002	Dc-ac inverter		1	-	5.0	3.0	8.5
C61-K005	Power supply	GE	2	K010	2.5	2.5	2.5
C61-N001	Flow transmitter	Rosemount	2	N010	10.0	5.8	1.8
C61-N006	Press. transmitter	Rosemount	1	-	10.0	5.8	1.8
C61-R001	Flow indicator controller	Bailey Meter	1	-	7.5	6.5	2.0

* NOTE: Table for historical purposes only.

Table 3.10-1 (Cont'd)

NSSS ESSENTIAL ELECTRICAL COMPONENTS AND INSTRUMENTS (CLASS 1E) *

DESCRIPTION				SEISMIC QUALIFICATION			
					SEISMIC QUALIFICATION(3) (g)		
ITEM NO.	NAME	VENDOR	QUANTITY	OTHERS OF SAME TYPE IN SIMILAR AREA	F-B	S-S	V
System Title - RPS							
C71-N050	Press. transmitter	Rosemount	4	-	10.0	5.8	1.8
C71-N052	Press. transmitter	Rosemount	4	-	10.0	5.8	1.8
System Title - PRMS							
D12-K603	Radiation Monitor	GE	4	-	3.0	3.0	3.0
D12-K609	Indicator/trip unit	GE	8	K610	3.0	3.0	3.0
D12-N006	Detector (steam line)	GE	4	-	8.0	8.0	8.0
D12-N010	Detector (sensor & converter)	GE	8	N011	15.0	15.0	15.0
System Title - RHR							
E11-N001	Cond. element	Balsbaugh	2	-	S.A.	S.A.	S.A.
E11-N004	Temp. element	Calif Alloy	7	N005,N027,N032	S.A.	S.A.	S.A.
E11-N008	Level transmitter	Barton	2	-	9.1	10.6	6.0
E11-N009	Temp. element	Calif Alloy	6	N029,N030	5.0	5.0	5.0
E11-N013	Flow transmitter	Rosemount	7	N015, N060	10.0	5.8	1.8
E11-N026	Press. transmitter	Rosemount	7	N028,N053	10.0	5.8	1.8
E11-N052	Transmitter	Rosemount	8	N258	10.0	5.8	1.8
E11-N055	Press. transmitter	Rosemount	8	N056	10.0	5.8	1.8
E11-N057	Press. transmitter	Rosemount	1	-	10.0	5.8	1.8
E11-N600	Temp. switch	Riley Inst.	4	N601	4.5	4.5	4.5
E11-N652	Trip unit	Rosemount	29	N655, N656	15.0	15.0	15.0
System Title - Core Spray							
E21-K601	Dc-ac inverter		4	-	15.0	10.0	7.0
E21-K602	Power supply	Elma Eng.	8	-	7.0	7.0	5.5
E21-K605	Ac/dc power supply	GE/Sola	1	-	5.5	5.5	5.5
E21-N003	Flow transmitter	Rosemount	2	-	3.0	3.0	3.0
E21-N051	Diff. press. transmitter	Rosemount	2	-	10.0	5.8	1.8
E21-N054	Press. transmitter	Rosemount	2	-	10.0	5.8	1.8
E21-N055	Press. transmitter	Rosemount	4	-	10.0	5.8	1.8
E21-N056	Diff. press. transmitter	Rosemount	1	-	3.0	3.0	3.0
E21-N651	Trip unit	Rosemount	12	N655	15.0	15.0	15.0

* NOTE: Table for historical purposes only

Table 3.10-1 (Cont'd)

NSSS ESSENTIAL ELECTRICAL COMPONENTS AND INSTRUMENTS (CLASS 1E) *

DESCRIPTION				SEISMIC QUALIFICATION			
					SEISMIC QUALIFICATION(3) (g)		
ITEM NO.	NAME	VENDOR	QUANTITY	OTHERS OF SAME TYPE IN SIMILAR AREA	F-B	S-S	V
System Title - MSIV-LCS							
E32-K601	Power supply	Elma Eng.	2	K602	7.0	7.0	5.5
E32-N006	Flow meter	S&K Inst.	4	-	3.55	3.55	4.87
E32-N050	Press. transmitter	Rosemount	8	N055,N058,N060,N061	10.0	5.8	1.8
E32-N051	Press. transmitter	Rosemount	5	N056	10.0	5.8	1.8
E32-N053	Flow transmitter	S&K Inst.	4	-	3.0	3.0	2.0
E32-N054	Diff. press. transmitter	Rosemount	2	N059	3.0	3.0	3.0
E32-N600	Time delay switch	Eagle Signal	13	N601,N602,N604	2.5	2.5	2.5
E32-N650	Press. indicator switch	Rosemount	20	N651,N653-N661	15.0	15.0	15.0
E32-R601	Millivolt to current converter	Bailey Meter	4	-	8.0	8.0	8.0
E32-R653	Flow indicator	GE	14	R654-R656, R658-R661	18.0	18.0	7.0
System Title - HPCI							
E41-K600	Power supply	GE	1	-	5.5	5.5	5.5
E41-K601	Sq root converter	Bailey Meter	1	-	9.0	9.0	13.0
E41-K603	Dc-ac inverter		1	-	15.0	10.0	7.0
E41-N008	Flow transmitter	Rosemount	1	-	10.0	5.8	1.8
E41-N013	Press. transmitter	Rosemount	2	N052	10.0	5.8	1.8
E41-N014	Level switch	Robert Shaw	1	-	4.6	4.6	3.6
E41-N024	Temp. element	Calif. Alloy/ PYCO	15	N025,N028-N030	5.0	5.0	5.0
E41-N050	Press. transmitter	Rosemount	5	N055B, F, N056	10.0	5.8	1.8
E41-N051	Flow transmitter	Rosemount	3	N061	10.0	5.8	1.8
E41-N053	Press. transmitter	Rosemount	1	-	10.0	5.8	1.8
E41-N055 (D,H)	Press. transmitter	Rosemount	2	-	10.0	5.8	1.8
E41-N057	Press. transmitter	Rosemount	6	N058	10.0	5.8	1.8
E41-N062	Level transmitter	Gould	2	-	7.0	7.0	8.0
E41-N600	Temp. switch	Riley Inst.	13	N601-N603	4.5	4.5	4.5
E41-N650	Trip unit	Rosemount	23	N651-N653, N655-N662	15.0	15.0	15.0
E41-R600	Flow indicator controller	Bailey Meter	1	-	7.5	6.5	20.0
System Title - RCIC							
E51-K600	Power supply	GE	1	-	2.5	2.5	2.5
E51-K601	Sq root convertor	Bailey Meter	1	-	9.0	9.0	13.0
E51-K603	Dc-ac inverter		1	-	15.0	10.0	7.0
E51-N003	Flow transmitter	Rosemount	4	N051,N057	10.0	5.8	1.8
E51-N007	Press. transmitter	Rosemount	1	-	10.0	5.8	1.8

* NOTE: Table for historical purposes only

Table 3.10-1 (Cont'd) *

DESCRIPTION				SEISMIC QUALIFICATION			
					SEISMIC QUALIFICATION[3] (g)		
ITEM NO.	NAME	VENDOR	QUANTITY	OTHERS OF SAME TYPE IN SIMILAR AREA	F-B	S-S	V
E51-N010	Level switch	Magnetrol	1	-	4.6	4.6	3.6
E51-N011	Temp. element	Calif. Alloy	15	N021-N023,N025	5.0	5.0	5.0
E51-N035	Level transmitter	Rosemount	2	-	3.0	3.0	3.0
E51-N050	Press. transmitter	Rosemount	2	N052	3.0	3.0	3.0
E51-N053	Press. transmitter	Rosemount	1	-	10.0	5.8	1.8
E51-N055	Press. transmitter	Rosemount	10	N056,N058	10.0	5.8	1.8
E51-N600	Temp. switch	Riley Inst.	11	N603	4.5	4.5	4.5
E51-N650	Press. indicator switch	Rosemount	21	N635,N651-N653,N655-N660	15.0	15.0	15.0
E51-R600	Flow indicator controller	Bailey Meter	1	-	7.5	6.5	20.0
System Title - RWCU							
G31-K602	Sq root converter	Bailey Meter	6	K603,K605	9.0	9.0	13.0
G31-K604	Five input summer	Bailey Meter	2	-	9.0	9.0	13.0
G31-N012	Flow transmitter	Rosemount	6	N036,N041	10.0	5.8	1.8
G31-N016	Temp. element	Calif Alloy	36	N022,N023	5.0	5.0	5.0
G31-N600	Temp. switch	Riley Inst.	24	N602	4.5	4.5	4.5
G31-N603	Diff. flow switch	Bailey Meter	2	-	7.5	8.5	20.0
G31-R616	Cycle timer	Eagle Signal	2	-	2.5	2.5	2.5

(1) Classified as pressure integrity or passive instrument
(2) Qualified by analysis
(3) S.A. = Stress Analysis

* NOTE: Table for historical purposes only

Table 3.10-2

SEISMIC QUALIFICATION TEST SUMMARY
NSSS SAFEGUARD CONTROL PANELS, LOCAL PANELS, AND RACKS

PANEL	DESCRIPTION	TYPE	CLASS 1E EQUIPMENT DESCRIPTION	COMMENTS
H12-P601	Reactor and containment cooling and isolation	Vertical board	Control switches, GE/MAC instruments, recorders	Seismic test complete
H12-P602	Reactor water cleanup and recirculation control	Bench board	Control switches	Qualification by similarity[1]
H12-P603	Reactor control	Bench board	Switches (range, push button, control)	Seismic test complete
H12-P606	Radiation monitor instrument Panel A	Instrument rack	Startup neutron monitoring electronics, radiation monitor trips units	Qualification by similarity
H12-P608	Power range neutron monitor	Instrument rack	APRM electronics, RBM electronics, Two-Out-Of-Four Logic Modules, power supplies, isolators, interface panels	Seismic test complete
H12-P609	Reactor protection system Division 1 & 2 logic	Vertical board	Relays, contactor, temperature monitor, trip units, switches, power supply	Seismic test complete
H12-P611	Reactor protection system Division 3 & 4 logic	Vertical board	Relays, contactor, temperature monitor, trip units, switches, power supply	Qualification by similarity
H12-P613	NSSS process instrument	Vertical board	Relays, trip limits, power supply	Qualification by similarity
H12-P614	NSSS temperature recorder	Vertical board	Relays	Seismic test complete
H12-P617	Division 1 RHR relay	Vertical board	Relays, trip units	Qualification by similarity
H12-P618	Division 2 RHR relay	Vertical board	Relays trip units, switches, power supply	Seismic test complete
H12-P620	HPCI relay	Vertical board	Relays, temperature monitor, switches	Seismic test complete

Table 3.10-2 (Cont'd)

PANEL	DESCRIPTION	TYPE	CLASS 1E EQUIPMENT DESCRIPTION	COMMENTS
H12-P621	RCIC relay	Vertical board	Relays, temperature monitor, switches	Qualification by similarity
H12-P622	Inboard valve relay	Vertical board	Relays	Qualification by similarity
H12-P623	Outboard valve relay	Vertical board	Relays	Qualification by similarity
H12-P626	ADS	Vertical board	Switches, indicators	Qualification by similarity
H12-P628	Prompt relief trip and ADS Division 1 relay	Vertical board	Relays, control switches	Qualification by similarity
H12-P631	ADS Division 3 relay	Vertical board	Relays, control switches	Qualification by similarity
H12-P633	Radiation monitor instrument Panel B	Instrument rack	Startup neutron monitoring electronics, radiation monitor trip units	Qualification by similarity
H12-P640	Division 3 RHR relay	Vertical board	Control switches, trip units, relays, temp. monitors, power supply	Qualification by similarity
H12-P641	Division 4 RHR relay	Vertical board	Control switches, trip units, temperature monitors, power supply	Qualification by similarity
H12-P647	HPCI	Vertical board	Controller, switches, power supply	Qualification by similarity
H12-P648	RCIC	Vertical board	Switches, power supply	Qualification by similarity
H12-P661	Safeguard systems A,B,C,D	Vertical board	Switches (non-NSSS supply)	Seismic test complete

Table 3.10-2 (Cont'd)

PANEL	DESCRIPTION	TYPE	CLASS 1E EQUIPMENT DESCRIPTION	COMMENTS
H12-P787	Termination cabinet	Cabinet	Cables	Qualification by similarity
H12-P788	Termination cabinet	Cabinet	Cables	Qualification by similarity
H12-P789	Termination cabinet	Cabinet	Cables	Qualification by similarity
H12-P790	Termination cabinet	Cabinet	Cables	Qualification by similarity
H12-P791	Termination cabinet	2 Bay cabinet	Cables	Qualification by similarity
H12-P792	Termination cabinet	2 bay cabinet	Cables	Qualification by similarity
H12-P793	Termination cabinet	2 bay cabinet	Cables	Qualification by similarity
H23-P001	Core spray system, Panel A	Local panel	Pressure transmitters	Seismic test complete
H23-P002	Reactor water cleanup	Local panel	Pressure transmitters, indicators, temperature control	Seismic test complete
H23-P004	Reactor vessel level & pressure, Panel A	Local panel	Pressure transmitters and indicators, level indicator, switches	Qualification by similarity
H23-P005	Reactor vessel level & pressure, Panel B	Local panel	Pressure transmitters	Qualification by similarity
H23-P006	Recirc pump, Panel A	Local panel	Pressure transmitters	Qualification by similarity
H23-P009	Jet pump, Panel A	Local panel	Pressure transmitters	Seismic test complete
H23-P010	Jet pump, Panel B	Local panel	Pressure transmitters	Qualification by similarity
H23-P014	HPCI system, Panel B	Local panel	Pressure transmitters, indicators	Qualification by similarity
H23-P015	Main steam flow, Panel B	Local panel	Pressure transmitters	Qualification by similarity
H23-P016	HPCI leak detection, Panel A	Local panel	Pressure transmitters	Qualification by similarity
H23-P017	RCIC, Panel A	Local panel	Pressure transmitters	Qualification by similarity
H23-P018	RHR System, Panel A	Local panel	Pressure transmitters	Qualification by similarity

Table 3.10-2 (Cont'd)

PANEL	DESCRIPTION	TYPE	CLASS 1E EQUIPMENT DESCRIPTION	COMMENTS
H23-P019	CS system, Panel B	Local panel	Pressure transmitters	Qualification by similarity
H23-P022	Recirc pump, Panel B	Local panel	Pressure transmitters	Qualification by similarity
H23-P025	Main steam flow, Panel D	Local panel	Pressure transmitters	Qualification by similarity
H23-P026	Reactor vessel level & pressure, Panel D	Local panel	Pressure transmitters	Qualification by similarity
H23-P027	Reactor vessel level & pressure, Panel C	Local panel	Switches, pressure transmitters	Qualification by similarity
H23-P030	SRM & IRM preamp A-D	NEMA - 12 enclosure	SRM-IRM preamplifiers	Seismic test complete
H23-P031	SRM & IRM preamp A-D	NEMA - 12 enclosure	SRM-IRM preamplifiers	Qualification by similarity
H23-P032	SRM & IRM preamp A-D	NEMA - 12 enclosure	SRM-IRM preamplifiers	Qualification by similarity
H23-P033	SRM & IRM preamp A-D	NEMA - 12 enclosure	SRM-IRM preamplifiers	Qualification by similarity
H23-P034	HPCI system, Panel A	Local panel	Pressure transmitters	Qualification by similarity
H23-P035	RCIC leak detection, Panel A	Local panel	Pressure transmitters	Qualification by similarity
H23-P036	HPCI leak detection, Panel B	Local panel	Pressure transmitters	Qualification by similarity
H23-P037	RCIC system, Panel B	Local panel	Pressure transmitters	Qualification by similarity
H23-P038	RCIC leak detection, Panel B	Local panel	Pressure transmitters	Qualification by similarity
H23-P041	Main steam flow, Panel C	Local panel	Pressure transmitters	Qualification by similarity
H23-P042	Main steam flow, Panel D	Local panel	Pressure transmitters	Qualification by similarity
H23-P073	MSIV LCS Division 1	Local panel	Pressure transmitters	Qualification by similarity
H23-P074	MSIV LCS Division 2	Local panel	Pressure transmitters	Qualification by similarity
H23-P075	RHR A/ADS, Panel A	Local panel	Pressure transmitters, indicators	Qualification by similarity

Table 3.10-2 (Cont'd)

PANEL	DESCRIPTION	TYPE	CLASS 1E EQUIPMENT DESCRIPTION	COMMENTS
H23-P076	RHR B/ADS, Panel C	Local panel	Pressure transmitters, indicators	Qualification by similarity
H23-P077	RHR C/ADS, Panel A	Local panel	Pressure transmitters, indicators	Qualification by similarity
H23-P078	RHR D/ADS, Panel C	Local panel	Pressure transmitters, indicators	Qualification by similarity
C61-P001	Remote shutdown	Local panel	Switches, square root converter, power supply	Qualification by similarity

(1) Qualification by similarity - Panel is structurally similar and in some cases identical to a panel that was tested.

Table 3.10-3

DYNAMIC QUALIFICATION TEST SUMMARY
NON-NSSS SAFETY-RELATED INSTRUMENTS AND ELECTRICAL EQUIPMENT

ITEM NUMBER	EQUIPMENT DESCRIPTION	SUPPLIER	DYNAMIC QUALIFICATION PACKAGE NO.
8031-E-7	Medium-voltage metal clad switchgear (4.16 kV)	Brown Boveri	D-14
8031-E-7	4.16 kV ATWS switchgear	Brown Boveri	D-167
8031-E-8B	4 kV Induction motors	GE	D-2 D-3
8031-E-10	Load centers	Brown Boveri	D-15
8031-E-11	480 V motor control centers	Cutler-Hammer	D-12 D-16
8031-E-11	Lighting panels	Cutler-Hammer	D-46
8031-E-13	Batteries and racks	C&D Batteries Div.	D-17 D-143
8031-E-14	250 V dc motor control centers	Westinghouse	D-18
8031-E-16	Dc distribution panels & fuse boxes	B-K Electrical Products Div.	D-13 D-19 D-70
8031-E-17	Battery chargers	C&D Batteries Div.	D-20
8031-E-33D	Heat tracing panels	Thermon	D-26
8031-E-37	Ac power dry-type transformers	Square D	D-45
8031-E-40	Primary containment electrical penetrations	Conax Corp.	D-27
8031-E-51	In-line plug connectors	Litton	D-44

Table 3.10-3 (Cont'd)

ITEM NUMBER	EQUIPMENT DESCRIPTION	SUPPLIER	DYNAMIC QUALIFICATION PACKAGE NO.
8031-M-66	HVAC control instruments	Various	D-144
8031-M-66	HVAC flow switches	FCI	D-155
8031-M-66	HVAC RTDs	Minco	D-156
8031-M-66	HVAC panel-mounted instruments	Various	D-159
8031-M-66	HVAC humidity transmitter/ sensor	American Instruments	D-160
8031-M-66	HVAC solenoid valves	ASCO	D-161
8031-M-66	HVAC flow elements	Annubar	D-163
8031-M-66	HVAC flow element air stations	Air Monitor	D-164
8031-M-66	Duct-mounted RTDs	Minco	D-182
8031-M-66	HVAC ITE time delay relay	Brown Boveri	D-211
8031-M-203	Radiation monitors, consisting of:	General Atomic	
	Item 1: Control room radiation monitors (4 each)		D-196
	Item 2: Control room emergency fresh air radiation monitors (2 each)		D-196
	Item 3: Primary containment post-LOCA radiation monitors (4 each)		D-58
8031-M-203	Single pen strip-chart recorders	General Atomic	D-200

Table 3.10-3 (Cont'd)

ITEM NUMBER	EQUIPMENT DESCRIPTION	SUPPLIER	DYNAMIC QUALIFICATION PACKAGE NO.
8031-M-203	Radiation monitoring control pen	General Atomic	D-201
8031-M-203	Communication and isolation devices	General Atomic	D-203
8031-M-206	Electronic transmitters	Rosemount	D-1
8031-M-212A	Flow elements venturi	BIF	D-214
8031-M-224	Pressure switches	Mercoid Corp.	D-28 D-49
8031-M-230	Differential pressure instruments	ITT-Barton	D-51 D-52
8031-M-231	Excess flow check valves	Marotta	D-134
8031-M-235	Containment gas sampling and analyzing system	Comsip-Delphi	D-59
8031-M-238	Atmospheric chlorine detectors	Nuclear Logistics Inc.	D-185
8031-M241	Nuclear pressure regulators	Target Rock Corp.	D-72
8031-M-242	Nuclear butterfly control valves	Fisher Controls	D-60
8031-M-243	Quality assured pressure gauges	Dresser Industries	D-32 D-53
8031-M-245	Quality assured solenoid valves	Target Rock Corp.	D-54
8031-M-245C	Solenoid valves	Valcor	D-212
8031-M-250A	Nuclear control valves	Masoneilan	D-56
8031-M-253	Wide range accident monitoring system	General Atomic	D-210

Table 3.10-3 (Cont'd)

ITEM NUMBER	EQUIPMENT DESCRIPTION	SUPPLIER	DYNAMIC QUALIFICATION PACKAGE NO.
8031-M-263	Suppression pool temperature monitoring system	WestronicsSimmonds-precision	D-178
8031-M-266	Class 1E pressure switches	ITT-Barton	D-180
8031-M-267	Class 1E RTDs	Weed	D-179

3.11 ENVIRONMENTAL DESIGN OF ELECTRICAL EQUIPMENT

This description provides a summary of the information addressed and substantiated in the separate Environmental Qualification Report (Section 1.1). The EQR identifies and documents the licensee's program for the environmental qualification of safety-related electric equipment installed in the LGS, in accordance with 10CFR50.49.

3.11.1 ENVIRONMENTAL DESIGN CRITERIA FOR ELECTRICAL EQUIPMENT

All safety-related equipment must be capable of performing its safety function and/or remaining in a safe mode under all conditions postulated to occur during its installed life. This requirement is embodied in GDC 1, 2, 4, and 23 of 10CFR50, Appendix A, in Criterion III and Criterion XI of 10CFR50, Appendix B, 10CFR50.55a(h), which incorporates by reference IEEE 279 (1971), "Criteria for Protection Systems for Nuclear Power Generating Stations," and in 10CFR50.49.

The NRC has issued definitive criteria in NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment", which contains the following criteria:

a. Category I, for plants whose construction permit SERs were issued after July 1, 1974, incorporates and supplements IEEE 323 (1974).

b. Category II, for plants whose construction permit SERs were issued before July 1, 1974, incorporates and supplements IEEE 323 (1971) unless the operating license applicant's record indicates that IEEE 323 (1974) is to be used, in which case Category I criteria are applicable.

The LGS construction permit SER was issued in June 1974, therefore, NUREG-0588 Category II criteria are applicable.

In February 1980 and June 1982, the NRC requested that the licensee perform a review of their environmental qualification program for safety-related equipment to identify the degree to which the program complied with the regulatory criteria. The EQR has been developed to provide detailed information on the LGS EQ program. The following sections provide a summary of the information contained in the EQR including mild environment qualification.

3.11.2 EQUIPMENT REQUIRING ENVIRONMENTAL QUALIFICATION

The list of equipment that is included in the environmental qualification program was established by considering those systems which are required to mitigate the consequences of a LOCA or HELB. This list also includes certain postaccident monitoring equipment. This list specifically includes the equipment required to achieve or support (1) emergency reactor shutdown, (2) containment isolation (3) reactor core cooling, (4) containment heat removal, (5) core residual heat removal, and (6) prevention of significant release of radioactive material to the environment.

Subsequent to identifying the equipment requiring qualification, equipment locations are identified using design drawings and are verified, where practicable, by field inspection. Equipment locations are identified by architectural room numbers with defined boundaries.

3.11.3 ENVIRONMENTAL SERVICE CONDITIONS

Environmental conditions have been determined for normal, abnormal, and accident conditions.

All areas inside containment and rooms containing high energy lines or post-LOCA recirculatory fluid lines outside of containment are considered to be harsh environments.

3.11.3.1 Environmental Conditions During Normal Plant Operation

Redundant plant HVAC systems are designed to maintain the temperature and humidity within the normal limits which are shown in the EQR. Section 9.4 describes the HVAC system.

The TID for normal operation for 40 years of equipment life have been calculated assuming a 100% load factor and rated power. The doses are based on the design radiation source terms of the radiation sources within each plant area.

Aging effects on all equipment are considered in the qualification program to conform to the requirements of section 4 of NUREG-0588. Components susceptible to aging effects are identified, and refurbishment and/or replacement is incorporated into the LGS Preventive Maintenance/Surveillance Program. Known susceptibility to aging degradation, results of inspections and manufacturer's recommendations are factored into the Maintenance/Surveillance Program.

Effects of known normal vibratory loads on equipment are considered in the EQ program when significant.

3.11.3.2 Accident Environmental Conditions

Operability duration requirements have been determined based on the length of time the equipment must maintain its ability to perform its safety function.

The primary containment time-dependent pressure and temperature profiles for the spectrum of postulated LOCAs and main steam line breaks have been generated using NRC approved methodology.

Temperature and pressure conditions resulting from a HELB outside containment have been determined using plant specific profiles. These profiles bound accident environments caused by other events. Sections 3.6 and 9.4 describe the analyses used in generating these profiles. Additional information is contained in the EQR.

Post-LOCA radiation doses inside primary and secondary containments were calculated in accordance with NUREG-0737, item II.B.2. The source terms are consistent with those specified in NUREG-0588 and NUREG-0737. Additional information is contained in Section 1.13.

Dynamic qualification of equipment due to seismic and hydrodynamic loads from SRV and LOCA is addressed in Section 3.10 and Appendix 3A.7.1.7.

The potential for submergence of equipment inside and outside containment is identified. Identified equipment is either qualified for submergence or analyzed to ensure both that it completes its function prior to submergence and that failure after submergence is acceptable. If these conditions cannot be met, the equipment will be relocated above flood level.

LGS has the capability of water spray actuation to mitigate the effects of a LOCA. Although no credit for spray actuation has been taken in determining temperature/pressure conditions inside containment, equipment inside containment is evaluated for the effects of spray.

Synergistic effects, where known, are considered. Specifically, where a supplier has identified synergisms or where the licensee is aware of synergistic effects for a particular component, it has been addressed. Appropriate documentation is included in the qualification file.

Margins appropriate to account for unquantified uncertainties in the effects of production variations and/or inaccuracies in test instruments are included in the qualification program.

3.11.4 QUALIFICATION TESTING AND ANALYSIS OF EQUIPMENT

All qualification testing and analysis of safety-related electrical equipment is performed according to the appropriate NUREG-0588 guidelines. Sequential testing and analysis or a combination thereof is performed for all electrical equipment. Analyses, performed on a case-by-case basis, have been conducted using approved methodologies to provide adequate justification.

3.11.5 METHODOLOGY FOR EVALUATING ENVIRONMENTAL QUALIFICATION TO SERVICE CONDITIONS

A comparison of environmental qualification to equipment service conditions is performed for all equipment located in harsh environmental zones. The equipment is evaluated for the 40 year normal environment and accident environments resulting from the spectrum of LOCAs and HELBs inside and outside containment. A point-by-point comparison is made between service condition parameters and qualified levels.

In addition, reviews and analyses were completed to ensure that the equipment tested was either identical or similar to the installed plant equipment.

3.11.6 MAINTENANCE/SURVEILLANCE PROGRAM

A Maintenance/Surveillance Program has been developed to encompass vendor prescribed maintenance procedures and periodic inspections of equipment to ensure that degradation is not occurring sooner than predicted. The program requires replacement of subcomponents (e.g., seals, gaskets, etc) at predetermined intervals to maintain the designated life. In addition, the program encompasses procedures to ensure that the environment is maintained relatively clean to avoid the possible adverse effects of dust.

3.11.7 REPLACEMENT PARTS PROGRAM

Safety-related equipment and spare and replacement parts are being ordered to meet or exceed the original specifications. For replacement parts being procured from the original specification and which are identical to the originally supplied equipment, a certificate of conformance is considered sufficient documentation to support qualification. However, if identical replacement parts are not available, environmental qualification for the new replacement parts will be demonstrated.

3.11.8 MILD ENVIRONMENT QUALIFICATION

Mild environment qualification is not included in the EQ program. A mild environment is an environment that would at no time be significantly more severe than the environment that would occur during normal plant operation, including anticipated operational occurrences.

3.12 CONTROL OF HEAVY LOADS

3.12.1 Introduction/Licensing Background

Site Licensing documentation is discussed in Section 1.12 (Subsection A-36) of the LGS UFSAR. Section 1.12 states "requirements for heavy loads, NUREG-0612, were addressed in the initial operating license review." Section 1.12 mentions a submittal dated August 13, 1984, which includes Bechtel Revision 3 of the LGS Overhead Handling Systems Review Final Report. Limerick Generating Station, Limerick Units 1 and 2, Overhead Handling Systems Review Final Report, Bechtel Revision 6/PECO Revision 1 (SDOC M-038-00008), August 1989 is incorporated in the LGS UFSAR as reference 9.1-1. The report summarizes the site correspondence to and from the NRC with respect to GL 81-07 and GL 85-11. Licensee submittal dated May 10, 1996, in response to NRC Bulletin 96-02 determines that at the time, NUREG-0612, Phase I commitments were being effectively implemented at LGS.

3.12.2 Safety Basis

UFSAR Section 1.12 (Subsection A-36) provides documentation of the site's compliance with Phase I of NUREG-0612 and Overhead Handling Systems Review Final Report (Reference 9.1-1) provides documentation that the site utilizes a single failure proof crane for RPVH/specified lifts and performed load drop analysis that demonstrates the fuel in the RPV will not be damaged and by providing electrical or mechanical interlocks on the crane to prevent travel of crane over spent fuel pool.

3.12.3 Scope of Heavy Load Handling Systems

The scope of load handling systems is documented in tabular form in M-038-00008 (Reference 9.1-1).

3.12.4 Control of Heavy Loads Program

3.12.4.1 Commitments in Response to NUREG-0612, Phase I Elements

The station is committed to NUREG-0612, Phase I elements as summarized in section 1.12 (Subsection A-36) of the LGS UFSAR. Detailed information is documented in M-038-00008 (Reference 9.1-1).

3.12.4.2 Reactor Pressure Vessel Head (RPVH) Lifting Procedures

The LGS UFSAR documents a single failure proof crane to be utilized to perform RPV head lifts in UFSAR Section 9.1.5 and is also documented further in reference 9.1- 1 (SDOC M-038-00008). In addition, the strong back/carousel used as a special lifting device for this evolution is single failure proof per UFSAR Section 9.1.4.2.5.8.

This safety basis (single failure proof lift for RPVH) is reflected in procedures M-041-200 (Reactor Pressure Vessel Disassembly) and M-041-400 (Reactor Pressure Vessel Reassembly).

3.12.4.3 Single Failure Proof Cranes for Spent Fuel Casks

The spent fuel cask will be equipped with lifting lugs and yoke(s) which are single failure proof and compatible with the single failure proof reactor enclosure crane and main hook, thus precluding a cask-drop due to a single failure (UFSAR section 15.7.5). Refer to Section 9.1.5 for a description of the reactor enclosure crane and the interlocks that prevent moving the crane over the fuel pool in the absence of specific action by the crane operator to allow such movement.

3.12.5 Safety Evaluation

The basis for the site's conclusion that the existing heavy loads program is compliant with the NUREG-0612, Phase 1 requirements is documented in UFSAR Section 1.12 under the heading of "Implementation and Status Summary." This section states, "the requirements for heavy loads, NUREG-0612, were addressed in the initial operating license submittal and Unit I Operating License NPF-27 License Condition 2.C(19)." This demonstrates that the LGS heavy loads program is compliant with the appropriate regulations, codes, and standards.

Reference 9.1 1 documents, that in the event of postulated load drops, the consequences are acceptable, as demonstrated by load drop analyses. Restrictions on load height, weight, medium under the load and safe paths are reflected in plant procedures.

Reference 9.1-1 documents that when using single failure proof cranes or equivalent to perform heavy loads lifts, the risk of a load drop is extremely unlikely and acceptably low.

Detailed review of load handling systems as documented in Reference 9.1-1 (SDOC M-038-00008) provides sufficient documentation (along with controls required by station rigging procedures) that heavy load lifts are done safely.