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* The listed drawings are included as "General References" only; i.e., refer to the drawings to obtain additional detail or to obtain background information. These drawings are not part of the UFSAR. They are controlled by the Controlled Documents Program.

CHAPTER 3.0 - DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

3.1 CONFORMANCE WITH NRC GENERAL DESIGN CRITERIA

3.1.1 Summary Description

This subsection contains an evaluation of the design bases of the LaSalle County Station (LSCS) as measured against the NRC General Design Criteria for Nuclear Power Plants, Appendix A of 10 CFR 50. The General Design Criteria are divided into six groups and total 55 in number.

For each of the 55 criteria, a specific assessment of the plant design is made and a complete list of references is included to identify where detailed information pertinent to each criterion is treated in the UFSAR.

Based on the content herein, the Applicant concludes that the LaSalle County Station fully satisfies and is in compliance with the General Design Criteria.

3.1.2 Criterion Conformance

3.1.2.1 Group I - Overall Requirements

3.1.2.1.1 Evaluation Against Criterion 1 - Quality Standards and Records

The total quality assurance program is described in Chapter 17.0 and consists of Topical Report CE-1A.

The detailed quality assurance program developed by the Applicant satisfies the requirements of Criterion 1.

3.1.2.1.2 Evaluation Against Criterion 2 - Design Bases for Protection Against Natural Phenomena

Structures and equipment important to plant safety are protected from or designed to withstand all appropriate natural phenomena at the plant site. Design is based on the most severe phenomena probable with special consideration for the uncertainty in prediction. Detailed discussions of the phenomena themselves, and how they are applied to the structures and equipment, are found in the following sections:

a. Meteorology, Section 2.3;

b. Hydrology, Section 2.4;

c. Geology and Seismology, Section 2.5;

d. Classification of Structures, Components, and Systems, Section 3.2;

e. Wind and Tornado Design Criteria, Section 3.3;

f. Water Level Design Criteria, Section 3.4;

g. Missile Protection Criteria, Section 3.5;

h. Seismic Design, Section 3.7;

i. Design of Seismic Category I Structures, Section 3.8;

j. Mechanical Systems and Components, Section 3.9;

k. Seismic Qualification of Seismic Category I Instrumentation and Electrical Equipment, Section 3.10; and

l. Environmental Design of Mechanical and Electrical Equipment, Section 3.11.

The design of the plant thus meets the requirements of Criterion 2.

3.1.2.1.3 Evaluation Against Criterion 3 - Fire Protection

Fires in the plant are prevented or mitigated by the use of noncombustible and heat resistant materials such as metal cabinets, metal wireways, and high melting point insulation wherever practicable.

Cabling is suitably rated and cable tray loading is designed to minimize internal heat buildup. Cable trays are suitably separated to avoid the loss of redundant channels of protective cabling should fires occur. The arrangement of equipment in protection channels assigned to separate cabinets provides physical separation and minimizes the effects of a possible fire.

Combustible supplies such as logs, records, manuals, etc., are limited in such areas as the control room to amounts required for current operation, thus minimizing the effect of a fire or explosion.

The plant fire protection system includes the following provisions:

a. automatic fire detection equipment in those areas where fire danger is greatest, and

b. extinguishing services which include automatic actuation with manual override as well as manually-operated fire extinguishers.

For further discussion, see the following chapters, sections, or subsections:

a. Construction Materials, Appendix E;

b. Instrumentation and Controls, Chapter 7.0;

c. Electric Power, Chapter 8.0;

d. Fire Protection System, Subsection 9.5.1;

e. Conduct of Operations, Chapter 13.0; and

f. Fire Hazards Analysis, Appendix H.

The design of the fire protection system thus meets the requirements of Criterion 3.

3.1.2.1.4 ~~Evaluation Against Criterion 4 - Environmental and Missile Design Bases~~

Structures and equipment important to safety are designed for compatibility with operation, maintenance, and testing conditions, including the postulated accident conditions. Also, equipment used to mitigate the consequences of accidents is either designed to be compatible with, or protected against, the effects of these accidents. Design requirements have been established for the amount of time such equipment must survive the extreme environmental conditions following a loss-of-coolant accident.

Further discussion of these design considerations is found in the following chapters or sections:

a. Classification of Structures, Components, and Systems, Section 3.2;

b. Wind and Tornado Design Criteria, Section 3.3;

c. Water Level Design Criteria, Section 3.4;

d. Missile Protection Criteria, Section 3.5;

e. Criteria for Protection Against Dynamic Effects Associated with the Postulated Rupture of Piping, Section 3.6;

f. Seismic Design, Section 3.7;

g Design of Seismic Category I Structures, Section 3.8;

h. Mechanical Systems and Components, Section 3.9;

i. Seismic Qualification of Seismic Category I Instrumentation and Electrical Equipment, Section 3.10;

j. Environmental Design of Mechanical and Electrical Equipment, Section 3.11;

k. Integrity of Reactor Coolant Pressure Boundary, Section 5.2;

l. Engineered Safety Features, Chapter 6.0;

m. Instrumentation and Controls, Chapter 7.0; and

n. Electric Power, Chapter 8.0.

The design of the plant thus meets the requirements of Criterion 4.

3.1.2.1.5 ~~Evaluation Against Criterion 5 - Sharing of Structures, Systems, and Components~~

No safety-related systems, structures, or components are shared unless such sharing has been evaluated to ensure that there will be no significant adverse impact on safety functions.

For a discussion of shared systems, structures or components, see the following chapters, sections, or subsections:

a. CSCS Equipment Cooling Water System, Sections 9.2 and 9.5;

b. Fire Protection System, Subsection 9.5.1;

c. HVAC Systems, Section 9.4;

d. Station Vent Stack, Section 3.8;

e. Solid and Liquid Radwaste System, Chapter 11.0;

f. Standby A-C Supply, Chapter 8.0;

g. Control Room, Section 1.2;

h. Combustible Gas Control System, Subsection 6.2.5;

i. Standby Gas Treatment System, Subsection 6.5.1;

j. Reactor Building Crane, Subsection 9.1.4; and

k. New Fuel Storage Vault, Subsection 9.1.1.

l. The design of the plant thus meets the requirements of Criterion 5.

3.1.2.2 ~~Group II - Protection by Multiple Fission Product Barriers~~

3.1.2.2.1 ~~Evaluation Against Criterion 10 - Reactor Design~~

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 fuel is designed to provide high integrity over a complete range of power levels, including transient conditions. The core is sized with a sufficient heat transfer area and coolant flow to ensure that there is no fuel damage under normal conditions or anticipated operational occurrences.

The reactor protection system is designed to monitor certain reactor parameters, sense abnormalities, and to scram the reactor thereby preventing fuel damage when trip points are exceeded. Scram trip setpoints are selected on operating experience and by the safety design basis. There is no case in which the scram trip setpoints allow the core to exceed the thermal hydraulic safety limits. Power for the reactor protection system is supplied by an independent high inertia a-c motor generator set. Alternate electric power is available to the reactor protection system buses.

An analysis and evaluation has been made of the effects upon core fuel following adverse plant operating conditions. The results of abnormal operational transients are presented in Chapter 15.0 and show that the thermal hydraulic safety design bases are satisfied, thereby assuring adequate fuel protection.

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

For further discussion, see the following chapter, sections, or subsections:

a. Principal Design Criteria, Subsection 1.2.1;

b. General Plant Description, Section 1.2;

c. Fuel System Design, Section 4.2;

d. Nuclear Design, Section 4.3;

e. Thermal and Hydraulic Design, Section 4.4;

f. Control Rod Drive Housing Supports, Section 4.6;

g. Reactor Recirculation System, Subsection 5.4.1;

h. Reactor Core Isolation Cooling System, Subsection 5.4.6;

i. Residual Heat Removal System, Subsection 5.4.7;

j. Reactor Protection System, Section 7.2; and

k. Accident Analysis, Chapter 15.0.

The design of the reactor thus meets the requirements of Criterion 10.

3.1.2.2.2 Evaluation Against Criterion 11 - Reactor Inherent Protection

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, and

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; 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 boiling water reactor has an inherently large moderator-to-Doppler coefficient ratio which permits the use of coolant flow rate for load following.

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

a. the use of coolant flow as opposed to control rods for load following,

b. the inherent self-flattening of the radial power distribution,

c. the ease of control, and

d. the spatial xenon stability.

The reactor is designed so that the moderator temperature coefficient is small and positive in the cold condition; however, the overall power reactivity coefficient is negative. Typically, the power coefficient at full power is about -0.04 $\Delta k/k/\Delta P/P$ at the beginning of life and about -0.03 $\Delta k/k/\Delta P/P$ at 10,000 MWd/T.

These values are well within the range required for adequate damping of power and spatial xenon disturbances.

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

For further discussion, see the following sections and subsections:

a. Principal Design Criteria, Subsection 1.2.1;

b. Nuclear Design, Section 4.3;

c. Thermal and Hydraulic Design, Section 4.4; and

d. Nuclear System Stability Analysis, Section 4.4.

The design of the reactor thus meets the requirements of Criterion 11.

3.1.2.2.3 ~~Evaluation Against Criterion 12 - Suppression of Reactor Power Oscillations~~

The Oscillation Power Range Monitor (OPRM) is designed to ensure that power oscillations which can result in conditions exceeding acceptable fuel design limits are either not possible or can be reliably and readily detected and suppressed. Reliability is enhanced by using highly redundant OPRM cells providing input to a safety grade trip system. The instrumentation systems provided for this purpose are discussed in Subsection 3.1.2.2.4, which follows.

Analytical fuel safety limit compliance is demonstrated for all expected modes of thermal-hydraulic neutron flux oscillations as discussed in Section 4.4.4.6.5.

The power reactivity coefficient is the composite simultaneous effect of the fuel temperature or Doppler coefficient, moderator void coefficient, and moderator temperature coefficient to the change in power level. It is negative and well within the range required for adequate damping of power and spatial xenon disturbances. Analytical studies indicate that for large boiling water reactors, underdamped, unacceptable power distribution behavior could only be expected to occur with power coefficients more positive than about –0.01 Δk/k/ΔP/P. Operating experience has shown large boiling water reactors to be inherently stable against xenon induced power instability. The large negative operating coefficients provide:

a. good load following with well-damped behavior and little undershoot or overshoot in the heat transfer response,

b. load following with recirculation flow control, and

c. strong damping of spatial power disturbances.

The reactor protection system design provides protection from excessive fuel cladding temperatures and protects the reactor coolant pressure boundary from excessive pressures which threaten the integrity of the system. Local abnomalities 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.

For further discussion see the following chapter, sections, or subsections:

a. Principal Design Criteria, Subsection 1.2.1;

b. Reactivity Control System, Sections 4.2 and 4.6;

c. Nuclear Design, Section 4.3;

d. Thermal and Hydraulic Design, Section 4.4;

e. Thermal Hydraulic Stability Analysis, Section 4.4;

f. Overpressurization Protection, Subsection 5.2.2;

g. Reactor Protection System, Section 7.2;

h. Reactor Manual Control System, Subsection 7.7.2; and

i. Accident Analysis, Chapter 15.0.

The design of the reactor thus meets the requirements of Criterion 12.

3.1.2.2.4 ~~Evaluation Against Criterion 13 - Instrumentation and Control~~

The fission process is monitored and controlled for all conditions from source range through power operating range. The intermediate and power ranges of the neutron monitoring system detect core conditions that threaten the overall integrity of the fuel barrier due to excess power generation and provide a signal to the reactor protection system. Fission detectors, located in the core, are used for neutron detection. The detectors are located to provide optimum monitoring in the intermediate and power ranges.

The intermediate range monitors (IRM) measure neutron flux from the upper range of the source range monitors (SRM) to the lower portion of the power range monitor (PRM) subsystems. The IRM's are capable of generating a trip signal to scram the reactor.

The local power range monitor (LPRM) subsystem consists of fission chambers located throughout the core, the signal conditioning equipment, and trip functions. LPRM signals are also used to block rod withdrawal and to generate the necessary trip signal for reactor scram.

The oscillation power range monitor (OPRM) subsystem takes signals from LPRM and detects reactor core instabilities using Period-, Amplitude-, and Rate of Growth-based algorithms. If instabilities are detected, the OPRM provides an alarm in the control room (based on period-based algorithm only). If the oscillations grow so as to potentially threaten the fuel safety limit, OPRM initiates an automatic suppression system trip to scram the reactor through the reactor protection system.

The reactor protection system protects the fuel cladding and the nuclear process barrier by monitoring plant parameters and causing a reactor scram when predetermined setpoints are exceeded. Separation of the scram and normal rod control function prevents failures in the reactor manual control circuitry from affecting the scram circuitry.

To provide protection against the consequences of accidents involving the release of radioactive materials from the fuel and reactor coolant pressure boundary, the containment and reactor vessel isolation control system initiates automatic isolation of appropriate pipelines whenever monitored variables exceed preselected operational limits.

Nuclear system leakage limits are established so that appropriate action can be taken to ensure the integrity of the reactor coolant pressure boundary. Nuclear system leakage rates within containment are classified as identified and unidentified, which corresponds respectively to the flow to the drywell equipment drain and floor drain sumps. The permissible total leakage rate limit to these sumps is based upon the makeup capabilities of various reactor component systems. Leakage flow into these sumps is determined by monitoring the rate of sump level increase which is correlated with the flow rate. The unidentified leakage rate as established in Subsection 5.2.5 is less than the value that has been conservatively calculated to be a minimum leakage from a crack large enough to propagate rapidly, but which still allows time for identification and corrective action before integrity of the process barrier is threatened.

The process radiation monitoring system monitors radiation levels of various processes and provides trip signals to the reactor protection system and containment and reactor vessel isolation control system whenever preestablished limits are exceeded.

As noted previously, adequate instrumentation has been provided to monitor system variables in the reactor core, reactor coolant pressure boundary, and reactor containment. Appropriate controls have been provided to maintain the variables in the operating range and to initiate the necessary corrective action in the event of an abnormal operational occurrence or accident.

For further discussion, see the following sections and subsections:

a. Principal Design Criteria, Subsection 1.2.1;

b. Reactivity Control System, Section 4.6;

c. Reactor Coolant Pressure Boundary Leakage Detection System, Subsection 5.2.5;

d. Main Steamline Isolation System, Subsection 5.4.5;

e. Containment Systems, Section 6.2;

f. Reactor Protection System, Section 7.2;

g. Primary Containment and Reactor Vessel Isolation Control System, Subsection 7.3.2;

h. Neutron Monitoring System, in Section 7.6.3;

i. Reactor Vessel - Instrumentation and Control, Subsection 7.7.1;

j. Process Computer System, Subsection 7.7.7;

k. Reactor Manual Control System, Section 7.7; and

l. Recirculation Flow Control System, Subsection 7.7.3:

These instrumentation and controls meet the requirements of Criterion 13.

3.1.2.2.5 Evaluation Against Criterion 14 - Reactor Coolant Pressure Boundary

The piping and pressure containing components within the reactor coolant pressure boundary up to and including the outer isolation valve(s) are designed, fabricated, erected, and tested to provide a high degree of integrity throughout the plant lifetime. Chapter 3.0 classifies systems and components within the reactor coolant pressure boundary as Quality Group A. The design requirements and codes and standards applied to this Quality Group ensure a quality product in keeping with the safety functions to be performed.

In order to minimize the possibility of brittle fracture within the reactor coolant pressure boundary, the fracture toughness properties and the operating temperature of ferritic materials are controlled to ensure adequate toughness. Subsection 5.2.3 describes the methods utilized to control toughness properties. Materials are impact tested in accordance with the ASME Boiler and Pressure Vessel Code, Section III.

Where reactor coolant pressure boundary piping penetrates the containment, the fracture toughness temperature requirements of the reactor coolant pressure boundary materials apply.

Piping and pressure containing components parts of the reactor coolant pressure boundary are assembled and erected by welding unless applicable codes permit flanged or screwed joints. Assembly is per ANSI B31.7 and ASME Section III, and erection is per ASME Section III. All welding procedures, welders, and welding machine operators are qualified in accordance with the requirements of Section IX of the ASME Boiler and Pressure Vessel Code for the materials to be welded. Qualification records, including the results of procedure and performance qualifications tests and identification symbols, assigned to each welder are maintained.

Section 5.2 contains the detailed material and examination requirements for the piping and equipment of the reactor coolant pressure boundary prior to and after its assembly and erection. Leakage testing and surveillance is accomplished as described in the evaluation against Criterion 30 of the General Design Criteria.

The design, fabrication, erection, and testing of the reactor coolant pressure boundary assure an extremely low probability of failure or abnormal leakage, thus satisfying the requirements of Criterion 14.

For further discussion, see the following chapters, sections, or subsections:

a. Principal Design Criteria, Subsection 1.2.1;

b. Design Criteria - Structures, Components, Equipment and Systems, Chapter 3.0;

c. Overpressurization Protection, Subsection 5.2.2;

d. Reactor Vessel, Section 5.3; and Reactor Vessel Internals, Subsection 3.9.5.

e. Reactor Recirculation Pumps, Subsection 5.4.1;

f. Reactor Vessel - Instrumentation and Control, Subsection 7.7.1;

g. Accident Analysis, Chapter 15.0; and

h. Quality Assurance Program, Chapter 17.0.

The reactor coolant pressure boundary thus meets the requirements of Criterion 14.

3.1.2.2.6 Evaluation Against Criterion 15 - Reactor Coolant System Design

The reactor coolant system consists of the reactor vessel and appurtenances, the reactor recirculation system, the nuclear system pressure relief system, the main steamlines, the reactor core isolation cooling system, the reactor water cleanup system, the residual heat removal system, and the nuclear system leak detection system. These systems are designed, fabricated, erected, and tested to stringent quality requirements and appropriate codes and standards which assure high integrity of the reactor coolant pressure boundary throughout the plant lifetime. The reactor coolant system is designed and fabricated to meet the requirements of the codes and standards discussed in Section 3.2.

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

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

The application of appropriate codes and standards and high quality requirements to the reactor coolant system and the design features of its associated auxiliary, control, and protection systems assure that the requirements of Criterion 15 are satisfied.

For further discussion, see the following chapters, sections, or subsections:

a. Principal Design Criteria, Subsection 1.2.1;

b. Design Criteria - Structure, Components, Equipment and Systems, Chapter 3.0;

c. Reactor Coolant System and Connected Systems, Chapter 5.0;

d. Reactor Protection System - Instrumentation and Control, Section 7.2; and

e. Accident Analysis, Chapter 15.0.

The reactor coolant system thus meets the requirements of Criterion 15.

3.1.2.2.7 ~~Evaluation Against Criterion 16 - Containment Design~~

The containment system consists of the following major components:

a. ~~Primary Containment~~

The primary containment is a steel-lined post-tensioned concrete pressure-suppression system of the over and under configuration. The drywell is located directly above the suppression chamber in the form of a frustum of a cone. The suppression pool chamber is cylindrical and separated from the drywell by a reinforced concrete slab which also functions as the drywell floor.

b. ~~Secondary Containment~~

A reactor building encloses the reactor and the primary containment. The structure provides secondary containment when the primary containment is in service and provides primary containment when the primary containment is open, as during refueling or maintenance. The reactor building houses the refueling and reactor servicing equipment and the new and spent fuel storage facilities. The principal purpose of the secondary containment is to confine the leakage of airborne radioactive materials from the primary containment and to provide a means for a controlled, elevated release to the atmosphere.

Primary containment temperature and pressure following an accident are limited by using the residual heat removal system to cool the suppression pool water.

For further discussion, see the following chapter, sections, or subsections:

a. General Plant Description, Section 1.2;

b. Design of Containment Structure, Section 3.8;

c. Containment Systems, Section 6.2; and

d. Accident Analysis, Chapter 15.0.

The design of the complete containment system meets the requirements of Criterion 16.

3.1.2.2.8 Evaluation Against Criterion 17 - Electrical Power Systems

Each unit of the station has two separate diesel-driven power sources and one common diesel-driven power source to provide electric power to three independent and redundant trains of engineered safety features. Each unit also has separate battery power sources to provide power to the separate and redundant vital d-c loads.

The offsite electric power system connections to the station are designed to provide a diversity of reliable power sources which are physically and electrically isolated so that any single failure can affect only one source of supply and does not propagate to alternate sources.

The station's auxiliary electric power system is designed to provide electrical isolation and physical separation of the redundant power supplies for station requirements which are important to nuclear safety. Means are provided for rapid location and isolation of system faults. Each diverse power source (diesel-generator and offsite) up to the point of connection to the engineered safety features system power buses, is physically and electrically independent. Redundant loads important to plant safety are split between redundant and independent engineered safety features system switchgear groups. A detailed discussion of these systems is presented in Chapter 8.0. The engineered safety features electrical systems are designed in accordance with IEEE Standards 279-1971 and 308-1971.

For further discussion, refer to the following chapters, sections or subsections:

a. Instrumentation and controls, Chapter 7.0;

b. Plant Electric Power, Chapter 8.0; and

c. Diesel-Generator Auxiliary Systems, Subsection 9.5.5, 9.5.6, 9.5.7 and 9.5.8.

The design of the electric power system thus meets the requirements of Criterion 17.

3.1.2.2.9 ~~Evaluation Against Criterion 18 - Inspection and Testing of Electric Power Systems~~

Provisions are provided in the design of offsite and onsite power systems for the inspection and testing of appropriate areas of the systems. Periodic tests are made of major portions of the power systems under conditions simulating the design conditions.

The engineered safety features systems are tested in accordance with NRC General Design Criteria (GDC) 18 and 21 to ensure that the systems can operate as designed and are available to function properly in the unlikely event of an accident. The Class IE power systems important to safety meet the testability requirements of GDC 18. Although GDC 18 does not require testing during normal operation, testing is performed in accordance with the following general program:

a. Prior to initial plant operation, a complete system test, which includes all actuation devices, circuits, electrical protective relays, and related instrumentation is conducted.

b. Subsequent to initial startup and during each regularly scheduled refueling outage a complete system test is conducted.

c. During normal operation with the unit in service, the majority of the ESF system components, analog, logic, and actuation circuitry are fully tested and the remaining components are partially tested.

d. During normal operation, the operability of all testable final actuation devices of the ESF systems are tested by manual initiation from the control room.

The following guidelines describe the testing circuitry and procedures for previous item c:

a. The test procedures must not involve the potential for damage to any plant equipment.

b. The test procedures must not expose the plant to an increased potential for accidental tripping.

c. The provisions for on-line testing must not compromise the ESF systems actuation circuits to the extent that their reliability is degraded.

Periodic testing of engineered safety features electrical auxiliary power equipment is made. Whenever one of the components of an ESF system requires maintenance, the necessary correction is made, the component is retested, and the channel or system of which the faulty component was a part is retested to confirm that the channel or system has been restored to serviceable condition as a result of the maintenance.

To ensure the operational readiness of each diesel generator, tests and inspections are conducted periodically. Each diesel generator is started and loaded for a period of time long enough to bring all the components of the system into equilibrium temperature conditions. Should one of the components require maintenance, the necessary corrections are made and the component is retested.

The station batteries and other equipment associated with the d-c system are serviced and tested periodically. Typical battery tests are specific gravity and voltage of the pilot cell, temperature of adjacent cells, and overall battery voltage. Periodically, each battery is subjected to a rated load discharge test.

Electric power systems important to safety are designed such that wiring, insulation, connections, and switchboards can be periodically inspected to verify their condition. In many cases, these items can be observed by removing a panel cover or housing. Clean, straightforward wiring is dictated on drawings which serves as an aid in inspecting these items.

For further discussion refer to the following chapters:

a. Plant Electric Power, Chapter 8.0;

b. Initial Test Program, Chapter 14.0 of the FSAR;

The testing of the electric power system thus meets the requirements of Criteria 18.

3.1.2.2.10 ~~Evaluation Against Criterion 19 - Control Room~~

The control room contains the following equipment: controls and necessary surveillance equipment for operation of the plant functions, such as the reactor and its auxiliary systems, engineered safety features, turbine generator, steam and power conversion systems, and station electrical distribution boards.

The control room is located in a Seismic Category I structure. Safe occupancy of the control room during abnormal conditions is provided for in the design. Adequate shielding is provided to maintain tolerable radiation levels in the control room in the event of a design-basis accident for the duration of the accident.

The control room HVAC system has redundant equipment and provides radiation detectors, ionization detectors, and ammonia detectors with appropriate alarms and interlocks. Provision is made for the control room air to be recirculated through charcoal filters. Provision is made to pass outdoor makeup air through HEPA and impregnated charcoal filters before introduction to the control room system.

The control room is continuously occupied by qualified operating personnel under all operating and accident conditions. In the unlikely event that the control room must be vacated and access is restricted, instrumentation and controls are provided outside the control room which can be utilized to safely perform a hot shutdown and a subsequent cold shutdown of the reactor.

For further discussion, see the following chapter, sections, or subsections:

a. General Plant Description, Section 1.2;

b. Control Building Design, Subsection 3.8.4;

c. Instrumentation and Controls, Chapter 7.0;

d. Shutdown from Outside Control Room, Subsection 7.4.4;

e. Heating, Ventilating, and Air Conditioning, Subsection 9.4.1 and Section 6.4;

f. Fire Protection, Subsection 9.5.1;

g. Ensuring that Occupation Radiation Exposures Are As Low As Reasonably Achievable (ALARA), Section 12.1;

h. Radiation Sources, Section 12.2; and

i. Seismic Category I Equipment, Subsection 3.2.1.

The design of the control room thus meets the requirements of Criterion 19.

3.1.2.3 ~~Group III - Protection and Reactivity Control System~~

3.1.2.3.1 ~~Evaluation Against Criterion 20 - Protection System Functions~~

The reactor protection system is designed to provide timely protection against the onset and consequences of conditions that threaten the integrity of the fuel barrier and the reactor coolant pressure boundary barrier. Fuel damage is prevented by initiation of an automatic reactor shutdown if monitored nuclear system variables exceed preestablished limits of anticipated operational occurrences. Scram trip settings are selected and verified to be far enough above or below operating levels to provide proper protection but not be subject to spurious scrams. The reactor protection system includes the motor-generator power system, sensors, relays, bypass circuitry, and switches that signal the control rod system to scram and shut down the reactor. The scrams initiated by neutron monitoring system variables, nuclear system high pressure, turbine stop valve closure, turbine control valve fast closure, and reactor vessel low-water level prevent fuel damage following abnormal operational transients. Specifically, these process parameters initiate a scram in time to prevent the core from exceeding thermal-hydraulic safety limits during abnormal operational transients. Response by the reactor protection system is prompt and the total scram time is short. Control rod scram motion starts in about 200 milliseconds after the high flux setpoint is exceeded.

A fully withdrawn control rod traverses 90% of its full stroke in sufficient time to ensure that acceptable fuel design limits are not exceeded.

In addition to the reactor protection system which provides for automatic shutdown of the reactor to prevent fuel damage, protection systems are provided to sense accident conditions and initiate automatically the operation of other systems and components important to safety. Systems such as the emergency core cooling system are initiated automatically to limit the extent of fuel damage following a loss-of-coolant accident. Other systems automatically isolate the reactor vessel or the containment to prevent the release of significant amounts of radioactive materials from the fuel and the reactor coolant pressure boundary. The controls and instrumentation for the emergency core cooling systems and the isolation systems are initiated automatically when monitored variables exceed preselected operational limits.

For further discussion, see the following chapter, sections, or subsections:

a. Principal Design Criteria, Subsection 1.2.1;

b. Reactivity Control System, Section 4.6;

c. Control Rod Drive System Design, Subsection 4.6.1.1;

d. Overpressurization Protection, Subsection 5.2.2;

e. Main Steamline Isolation System, Subsection 5.4.5;

f. Emergency Core Cooling System, Section 6.3;

g. Reactor Protection System, Section 7.2;

h. Primary Containment and Reactor Vessel Isolation Control System, Subsection 7.3.2;

i. Emergency Core Cooling Systems - Instrumentation and Control, Subsection 7.3.1;

j. Neutron Monitoring System, Subsection 7.6.3;

k. Process Radiation Monitoring System, Subsection 7.7.14;

l. Leak Detection System, Section 7.6; and

m. Accident Analysis, Chapter 15.0.

The design of the protection system thus meets the requirements of Criterion 20.

3.1.2.3.2 ~~Evaluation Against Criterion 21 - Protection System Reliability and Testability~~

Reactor protection system design ensures that, through redundancy, each channel has sufficient reliability to fulfill the single-failure criterion. No single component failure, intentional bypass maintenance operation, calibration operation, or test to verify operational availability can impair the ability of the system to perform its intended safety function. Additionally, the system design ensures that when a scram trip point is exceeded there is a high scram probability. However, should a scram not occur, other monitored components can scram the reactor if their trip points are exceeded. 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 reactor protection system includes design features that permit inservice testing. This ensures the functional reliability of the system should the reactor variable exceed the corrective action setpoint.

The reactor protection system initiates an automatic reactor shutdown if the monitored plant variables exceed preestablished limits. This system is arranged as two separately powered trip systems. Each trip system has two trip channels. An automatic or manual trip in either or both trip channels constitutes a trip system condition. A scram results when both trip systems have tripped. This logic scheme is a one-out-of-two taken twice arrangement. The reactor protection system can be tested during reactor operation. Manual scram testing is performed by operating one of the four manual scram controls. Two manual scram controls are associated with each trip system, one in each trip channel. Operating one manual scram control tests one trip channel and one trip system. The total test verifies the ability to deenergize the scram pilot valve solenoids. Indicating lights verify that the actuator contacts have opened. This capability for a thorough testing program significantly increases reliability.

Control rod drive operability can be tested during normal reactor operation. Drive position indicator and incore neutron detectors are used to verify control rod movement. Each control rod can be withdrawn one notch and then reinserted to the original position without significantly perturbing the nuclear system. One control rod is tested at a time. Control rod mechanism overdrive demonstrates rod-to-drive coupling integrity. Hydraulic supply subsystem pressures can be observed on control room instrumentation. More importantly, the hydraulic control unit scram accumulator and the scram discharge volume level are continuously monitored.

The main steamline isolation valves may be tested during full reactor operation. Individually, they can be fully closed without affecting the reactor operation. Provisions are made to evaluate valve stem leakage during reactor shutdown. During refueling operation, valve leakage rates can be determined.

Residual heat removal system testing can be performed during normal operation. Main system pumps can be evaluated by taking suction from the suppression pool and discharging through test lines back to the suppression pool. The low-pressure coolant injection mode can be tested after reactor shutdown.

Each active component of the emergency core cooling system is designed to be operable for test purposes during normal operation of the nuclear system.

The high functional reliability, redundancy, and inservice testability of the protection system satisfy the requirements specified in Criterion 21. For further discussion, see the following chapter, sections, or subsections:

a. Principal Design Criteria, Subsection 1.2.1;

b. Reactivity Control System, Section 4.6;

c. Main Steamline Isolation System, Subsection 5.4.5;

d. Residual Heat Removal System, Subsection 5.4.7;

e. Containment Systems, Section 6.2;

f. Emergency Core Cooling Systems, Section 6.3;

g. Reactor Protection System, Section 7.2;

h. Primary Containment and Reactor Vessel Isolation Control System, Subsection 7.3.2;

i. Emergency Core Cooling Systems - Instrumentation and Control, Subsection 7.3.1;

j. Neutron Monitoring System, Subsection 7.6.3;

k. Process Radiation Monitoring, Subsections 7.7.14 and Section 11.5;

l. Leak Detection System, Section 7.6; and

m. Accident Analysis, Chapter 15.0.

The design of the protection system thus meets the requirements of Criterion 21.

3.1.2.3.3 ~~Evaluation Against Criterion 22 - Protection System Independence~~

The components of protection systems are designed so that the mechanical and thermal environment resulting from any emergency situation in which the components are required to function does not interfere with the operation of that function. Wiring for the reactor protection system outside of the control room enclosures is run in rigid metallic wireways. No other wiring is run in these wireways. The wires from duplicate sensors on a common process tap are run in separate wireways. The system sensors are electrically and physically separated. Only one trip actuator logic circuit from each trip system may be run in the same wireway.

The reactor protection system is designed to permit maintenance and diagnostic work while the reactor is operating without restricting the plant operation or hindering the output of its safety functions. The flexibility in design afforded the protection system allows operational system testing by the use of an independent trip channel for each trip logic input. When an essential monitored variable exceeds its scram trip point, it is sensed by at least two independent sensors in each trip system. An intentional bypass, maintenance operation, calibration operation, or test results in a single channel trip. This leaves at least two trip channels per monitored variable capable of initiating a scram. At that time, only one trip channel in each trip system must trip to initiate a scram. Thus, the arrangement of two trip channels per trip system ensures that a scram occurs as a monitored variable exceeds its scram setting.

For further discussion, see the following chapter, sections, or subsections:

a. Principal Design Criteria, Subsection 1.2.1;

b. Reactivity Control System, Subsection 4.2.3;

c. Main Steamline Isolation System, Subsection 5.4.5;

d. Residual Heat Removal System, Subsection 5.4.7;

e. Emergency Core Cooling Systems, Section 6.3;

f. Reactor Protection System, Section 7.2;

g. Primary Containment and Reactor Vessel Isolation Control System, Subsection 7.3.2;

h. Emergency Core Cooling System - Instrumentation and Control, Subsection 7.3.11;

i. Neutron Monitoring System, Subsection 7.6.3;

j. Process Radiation Monitoring, Subsection 7.7.14;

k. Leak Detection System, Section 7.6; and

l. Accident Analysis, Chapter 15.0.

The protection system thus meets the requirements of Criterion 22.

3.1.2.3.4 ~~Evaluation Against Criterion 23 - Protection System Failure Modes~~

The reactor protection system is designed to fail to a safe condition. Use of an independent trip channel for each trip logic allows the system to sustain any trip channel failure without preventing other sensors monitoring the same variable from initiating a scram. A single sensor or trip channel failure causes a channel trip. Only one trip channel in each trip system must be actuated to initiate a scram. Intentional bypass, maintenance operation, calibration operation, or test results in a single channel trip. A failure of any one reactor protection system input or subsystem component produces a trip in one of two channels. This condition is insufficient to produce a reactor scram, but the system is ready to perform its protective function upon another trip.

The environmental conditions in which the instrumentation and equipment of the reactor 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.

For further discussion, see the following chapter, sections, or subsections:

a. Principal Design Criteria, Subsection 1.2.1;

b. Emergency Core Cooling Systems, Section 6.3;

c. Reactor Protection System, Section 7.2;

d. Primary Containment and Reactor Vessel Isolation Control System, Subsection 7.3.2;

e. Neutron Monitoring System, Subsection 7.6.3;

f. Leak Detection System Instrumentation and Control, Section 7.6; and

g. Electric Power Systems; Chapter 8.0.

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

3.1.2.3.5 ~~Evaluation Against Criterion 24 - Separation of Protection and Control Systems~~

There is separation between the reactor protection system and the process control systems. Sensors, trip channels, and trip logics of the reactor protection system are not used directly for automatic control of process systems. Therefore, failure in the controls and instrumentation of process systems cannot induce failure of any portion of the protection system. High scram reliability is designed into the reactor protection system and hydraulic control unit for the control rod drive. The scram signal and mode of operation overrides all other signals.

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

Process radiation monitoring is provided on process liquid and reactor gaseous exhaust gaslines that may serve as discharge routes for radioactive materials. Four instrumentation channels are used on reactor building exhaust plenum monitoring to prevent an inadvertent scram and isolation as a result of instrumentation malfunctions. The output trip signals from each channel are combined in such a way that two channels must signal high radiation to initiate scram and main steamline isolation.

For further discussion, see the following sections and subsections:

a. Principal Design Criteria, Subsection 1.2.1;

b. Reactivity Control System, Section 4.6;

c. Emergency Core Cooling System, Section 6.3;

d. Reactor Protection System, Section 7.2;

e. Primary Containment and Reactor Vessel Isolation Control System, Subsection 7.3.2;

f. Emergency Core Cooling System - Instrumentation and Control, Subsection 7.3.1;

g. Neutron Monitoring System, Subsection 7.6.3;

h. Process Radiation Monitoring, Subsection 7.7.14;

i. Leak Detection System Instrumentation and Control, Section 7.6; and

j. Reactor Manual Control System, Subsection 7.7.2.

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

3.1.2.3.6 ~~Evaluation Against Criterion 25 - Protection System Requirements for Reactivity Control Malfunctions~~

The reactor protection system provides protection against the onset and consequences of conditions that threaten the integrity of the fuel barrier and the reactor coolant pressure boundary. Any monitored variable which exceeds the scram setpoint initiates an automatic scram and does not impair the remaining variables from being monitored, and if one channel fails the remaining portions of the reactor protection system continue to function.

The reactor manual control system is designed so that no single failure can negate the effectiveness of a reactor scram. The circuitry for the reactor manual control system is completely independent of the circuitry controlling the scram valves. This separation of the scram and normal rod control functions prevents failures in the reactor manual control circuitry from affecting the scram circuitry. Because each control rod is controlled as an individual unit, a failure that results in energizing any of the insert or withdraw solenoid valves can effect 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 are considered to be when the reactor is just subcritical and an out of sequence rod is continuously withdrawn. The rod worth minimizer would normally prevent the withdrawal of out-of-sequence 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.

For further discussuion, see the following chapter, sections, or subsections:

a. Principal Design Criteria, Subsection 1.2.1;

b. Reactivity Control System, Section 4.6;

c. Nuclear Design, Section 4.3;

d. Thermal and Hydraulic Design, Section 4.4;

e. Reactor Protection System, Section 7.2;

f. Reactor Manual Control System, Subsection 7.7.2; and

g. Accident Analysis, Chapter 15.0.

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

3.1.2.3.7 ~~Evaluation Against Criterion 26 - Reactivity Control System Redundancy and Capability~~

Two independent reactivity control systems utilizing different design principles are provided. The normal method of reactivity control employs control rod assemblies which contain boron-carbide (B_4C) powder and/or hafnium metal. Control of reactivity is operationally provided by a combination of these movable control rods, burnable poisons, and the reactor coolant recirculation system flow. These systems accommodate fuel burnup, load changes, and long-term reactivity changes.

Reactor shutdown by the control rod drive system is sufficiently rapid to prevent 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. 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. Two sources of scram energy (accumulator pressure and reactor vessel pressure) provide needed scram performance over the entire range of reactor pressure, i.e., from operating conditions to cold shutdown.

The design of the rod worth minimizer system includes appropriate margin for malfunctions such as stuck rods in the event that they do occur. Control rod withdrawal sequences and patterns are selected prior to operation to achieve optimum core performance, and, simultaneously, low individual rod worths. The operating procedures to accomplish such patterns are supplemented by blocking of rod withdrawals that do not conform to the sequence utilized in the RWM system. An additional safety 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. In the event that a reactor scram is necessary, the unlikely occurrences of a limited number of stuck rods (within the available amount of shutdown margin discussed above) will not hinder the capability of the control rod system to render the core subcritical.

A standby liquid control system containing 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 determined to permit this capability accounts for the reactivity effects of xenon decay, eliminating steam voids, change in water density due to the reduction in water temperature, Doppler effect in uranium, changing neutron leakage from boiling to cold, and changing rod worth as boron affects neutron migration length. An additional margin of -0.05 ΔK is provided.

For further discussion, see the following sections and subsections:

a. Safety Design Criteria, Subsection 1.2.1.1;

b. Reactivity Control System, Section 4.6;

c. Standby Liquid Control System - Instrumentation and Control, Section 7.4;

d. Reactor Manual Control System, Subsection 7.7.2; and

e. Process Computer System, Subsection 7.7.7.

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

3.1.2.3.8 ~~Evaluation Against Criterion 27 - Combined Reactivity Control Systems Capability~~

There is no credible event applicable to the BWR which requires combined capability of the control rod system and poison additions by the standby liquid control system. The primary reactivity control system for the BWR during postulated accident conditions is the control rod system. Abnormalities are sensed and, if protection system limits are reached, corrective action is initiated through automatic insertion of control rods. 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 rod drive system. Response by the reactor protection system is prompt and the total scram time is short.

In operating the reactor there is a spectrum of possible control rod worths, depending on the reactor state and on the control rod pattern chosen for operation. Control rod withdrawal sequences and patterns are selected to achieve optimum core performance and low individual rod worths. The rod worth minimizer program prevents rod withdrawal other than by the preselected rod withdrawal pattern. The rod worth minimizer function assists the operator with an effective backup control rod monitoring routine that enforces adherence to established startup, shutdown, and low power level operations. As a result of this carefully planned procedure, prompt shutdown of the reactor can be achieved with scram insertion of less 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 an inoperable rod.

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: 1) fuel temperature or Doppler coefficient; 2) moderator void coefficient; and 3) moderator temperature coefficient. The overall power reactivity coefficient is negative and provides a strong negative reactivity feedback under severe power transient conditions.

For further discussion, see the following chapter, sections, or subsections:

a. Safety Design Criteria, Subsection 1.2.1.1;

b. Reactivity Control System, Section 4.6;

c. Nuclear Design, Section 4.3;

d. Thermal and Hydraulic Design, Section 4.4;

e. Reactor Protection System, Section 7.2;

f. Reactor Manual Control System, Subsection 7.7.2;

g. Process Computer System, Subsection 7.7.7; and

h. Accident Analysis, Chapter 15.0.

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

3.1.2.3.9 Evaluation Against Criterion 28 - Reactivity Limits

Control rod withdrawal sequences and patterns are selected to achieve optimum core performance and low individual rod worths. The rod worth minimizer program prevents withdrawal other than by the preselected rod withdrawal pattern. The rod worth minimizer function assists the operator with an effective backup control rod monitoring routine that enforces adherence to established startup, shutdown, and low power level operations control rod procedures.

The control rod mechanical design incorporates a hydraulic velocity limiter in the control rod which prevents rapid rod ejection. This engineered safeguard protects against a high reactivity insertion rate by limiting the control rod dropout velocity to less than 5 feet per second. Normal rod movement is limited to 6-inch increments and the rod withdrawal rate is limited through the hydraulic valve to nominally 3 inches per second.

The accident analysis (Chapter 15.0) evaluates the postulated reactivity accidents as well as abnormal operational transients in detail. Analyses are included for rod dropout, steamline 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 result in damage to the reactor coolant pressure boundary. In addition, the integrity of the core, its support structures and other reactor pressure vessel internals are maintained so that the capability to cool the core is not impaired for any of the postulated reactivity accidents described in the accident analysis.

For further discussion, see the following chapters, sections, or subsections:

a. Safety Design Criteria, Subsection 1.2.1.1;

b. Design Criteria - Structures, Components, Equipment and Systems, Chapter 3.0;

c. Reactor Core Support Structures and Internals Mechanical Design, Section 5.3 and Subsection 3.9.5;

d. Reactivity Control System, Section 4.6;

e. Nuclear Design, Section 4.3;

f. Control Rod Drive Housing Supports, Subsection 4.6.1;

g. Overpressurization Protection, Subsection 5.2.2;

h. Reactor Vessel and Appurtenances, Section 5.3 and Subsection 3.9.5;

i. Main Steamline Flow Restrictor, Subsection 5.4.4;

j. Main Steamline Isolation Valves, Subsection 5.4.5;

k. Process Computer System, Subsection 7.7.7; and

l. Accident Analysis, Chapter 15.0.

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

3.1.2.3.10 ~~Evaluation Against Criterion 29 - Protection Against Anticipated Operational Occurrences~~

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 detail in Criteria 21, 22, 23, 24, and 26.

An extremely high reliability of timely response to anticipated operational occurrences is maintained by a thorough program of inservice testing and surveillance. Active components are tested or removed from service for maintenance during reactor operation without compromising the protection or reactivity control functions even in the event of a subsequent single failure. Components important to safety are tested during normal reactor operation. Functional testing and calibration schedules are developed using available failure rate data, reliability analyses, and operating experience. These schedules represent an optimization of protection and reactivity control system reliability by considering, on one hand, the failure probabilities of individual components and, on the other hand, the reliability effects during individual component testing on the portion of the system not undergoing test. The capability for inservice testing ensures the high functional reliability of protection and reactivity control systems should a reactor variable exceed the corrective action setpoint.

For further discussion, see the following chapter, sections, or subsections:

a. Safety Design Criteria, Subsection 1.2.1.1;

b. Reactivity Control System, Section 4.6;

c. Main Steamline Isolation Valves, Subsection 5.4.5;

d. Residual Heat Removal System, Subsection 5.4.7;

e. Containment Systems, Section 6.2;

f. Emergency Core Cooling Systems, Section 6.3;

g. Reactor Protection System, Section 7.2;

h. Primary Containment and Reactor Vessel Isolation Control System, Subsection 7.3.2;

i. Emergency Core Cooling System - Instrumentation and Control, Subsection 7.3.1;

j. Neutron Monitoring System, Subsection 7.6.3;

k. Process Radiation Monitoring, Subsection 7.7.14;

l. Leak Detection System Instrumentation and Control, Section 7.6; and

m. Accident Analysis, Chapter 15.0.

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

3.1.2.4 ~~Group IV - Fluid Systems~~

3.1.2.4.1 ~~Evaluation Against Criterion 30 - Quality of Reactor Coolant Pressure Boundary~~

By utilizing conservative design practices and detailed quality control procedures, the pressure-retaining components of the reactor coolant pressure boundary are designed and fabricated to retain their integrity during normal and postulated accident conditions. Accordingly, components which comprise the reactor coolant pressure boundary are designed, fabricated, erected, and tested in accordance with recognized industry codes and standards listed in Chapter 5.0. Further, product and process quality planning is provided as described in Chapter 17.0 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 aspects of the reactor coolant pressure boundary, further discussion on this subject is treated in the response to Criterion 14.

Means are provided for detecting reactor coolant leakage. The leak detection system consists of sensors and instruments to detect, annunciate, and in some cases, isolate the reactor coolant pressure boundary from potential hazardous leaks before predetermined limits are exceeded. Small leaks are detected by temperature and/or pressure changes, increased frequency of sump pump operation, and by measuring fission product concentration. In addition to these means of detection, large leaks are detected by changes in flow rates in process lines and changes in reactor water level. The allowable leakage rates have been based on the predicted and experimentally determined behavior of cracks in pipes, the ability to make up coolant system leakage, the normally expected background leakage due to equipment design, and the detection capability of the various sensors and instruments. The total leakage rate limit is established so that, in the absence of normal a-c power with loss of feedwater supply, makeup capabilities are provided by the RCIC system. While the leak detection system provides protection from small leaks, the emergency core cooling system network provides protection for the complete range of discharges from ruptured pipes. Thus, protection is provided for the full spectrum of possible discharges.

For further discussion, see the following chapters, sections, or subsections:

a. Principal Design Criteria, Subsection 1.2.1;

b. Design Criteria - Structure, Components, Equipment and Systems, Chapter 3.0;

c. Overpressurization Protection, Subsection 5.2.2;

d. Reactor Coolant Pressure Boundary Leakage Detection System, Subsection 5.2.5;

e. Reactor Vessel and Appurtenances, Section 5.3 and Subsection 3.9.5;

f. Reactor Recirculation System, Section 5.2;

g. Reactor Vessel - Instrumentation and Control, Subsection 7.7.1;

h. Leak Detection System Instrumentation and Control, Section 7.6; and

i. Quality Assurance, Chapter 17.0.

The reactor coolant pressure boundary and its leak detection system are designed to meet the requirements of Criterion 30.

3.1.2.4.2 ~~Evaluation Against Criterion 31 - Fracture Prevention of Reactor Coolant Pressure Boundary~~

Brittle fracture control of pressure-retaining ferritic materials is provided to ensure protection against nonductile fracture. To minimize the possibility of brittle fracture failure of the reactor pressure vessel, the reactor pressure vessel was designed to the 1968 Edition of Section III of ASME Code and Addenda to and including Summer 1970 (except Paragraph N-355). An alternate method of compliance with the intent of Appendix G is presented in Subsection 5.2.3.

The nil-ductility transition (NDT) temperature represents the temperature below which ferritic steel breaks in a brittle rather than ductile manner. The NDT temperature increases as a function of neutron exposure at integrated neutron exposures greater than about 1 x 10^{17} nvt with neutron energies in excess of 1 MeV.

The reactor assembly design provides an annular space from the outermost fuel assemblies to the inner surface of the reactor vessel that serves to attenuate the fast neutron flux incident upon the reactor vessel wall. This annular volume contains the core shroud, jet pump assemblies, and reactor coolant. Assuming plant operation at rated power and 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.

For further discussion, see the following chapters and sections:

a. Design Criteria - Structures, Components, Equipment and Systems, Chapter 3.0;

b. Material Considerations, Subsection 5.2.3; and

c. Reactor Vessel and Appurtenances, Section 5.3 and Subsection 3.9.5.

The reactor coolant pressure boundary is designed, maintained and tested to ensure that the boundary will behave in a nonbrittle manner throughout the life of the plant. Therefore, the reactor coolant pressure boundary is in conformance with Criterion 31.

3.1.2.4.3 ~~Evaluation Against Criterion 32 - Inspection of Reactor Coolant Pressure Boundary~~

The reactor pressure vessel design and engineering effort include provisions for inservice inspection. Rotating doors in the sacrificial shield and removal panels in the insulation provide access for examination of the vessel and its appurtenances. Also, removable insulation is provided on the recirculation system and on the main steam and feedwater systems extending out to and including the first isolation valve outside the primary containment. Inspection of the reactor coolant pressure boundary is in accordance with the ASME Boiler and Pressure Vessel Code Section XI. Section 5.2 defines the inservice inspection plan, access provisions, and areas of restricted access.

Vessel material surveillance samples will be located within the reactor pressure vessel. The program will include specimens of the base metal, weld metal, and heat affected zone metal.

For further discussion, consult the following chapters and sections:

a. Design of Structures, Components, Equipment, and Systems, Chapter 3.0;

b. Reactor Coolant Pressure Boundary Leakage Detection System, Subsection 5.2.5;

c. Inservice Inspection, Subsection 5.2.4;

d. Reactor Vessel and Appurtenances, Section 5.3 and Subsection 3.9.5;

e. Reactor Recirculation System, Section 5.4.

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

3.1.2.4.4 ~~Evaluation Against Criterion 33 - Reactor Coolant Makeup~~

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 reactor coolant pressure boundary from potential hazardous leaks before predetermined limits are exceeded. Small leaks are detected by temperature and pressure changes, by increased frequency of sump pump operation, and by measuring fission product concentration. In addition to these means of detection, large leaks are detected by changes in flow rates in process lines, and changes in reactor water level. The allowable leakage rates have been based on predicted and experimentally determined behavior of cracks in pipes, the ability to make up coolant system leakage, the normally expected background leakage due to equipment design, and the detection capability of the various sensors and instruments. The total leakage rate limit is established so that, in the absence of normal a-c 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 emergency core cooling system provides protection for the complete range of discharges from ruptured pipes. Thus, protection is provided for the full spectrum of possible discharges to the extent that fuel cladding temperature limits are not exceeded.

For further discussion, see the following sections and subsections:

a. Reactor Coolant Pressure Boundary Leakage Detection System, Subsection 5.2.5;

b. Reactor Core Isolation Cooling System, Subsecton 5.4.6;

c. Emergency Core Cooling System, Section 6.3; and

d. Reactor Vessel - Instrumentation and Control, Section 7.6.

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

3.1.2.4.5 ~~Evaluation Against Criterion 34 - Residual Heat Removal~~

The residual heat removal (RHR) system provides the means to remove decay heat and residual heat from the nuclear system so that refueling and nuclear system servicing can be performed.

The major equipment of the RHRS consists of heat exchangers, main system pumps, and service water pumps. The equipment is connected by associated valves and piping, and the controls and instrumentation are provided for proper system operation. The main system pumps are sized on the basis of the flow required during the low-pressure coolant injection (LPCI) mode of operation, which is the mode requiring the maximum flow rate. The heat exchangers are sized on the basis of the required duty for the shutdown cooling function, which is the mode requiring the maximum heat exchanger area.

Two loops, each consisting of a heat exchanger, main system pump, and associated piping, are located in separate protected areas.

For further discussion, see the following chapters, sections, or subsections:

a. Residual Heat Removal System, Subsection 5.4.7;

b. Emergency Core Cooling System - Instrumentation and Control, Subsection 7.3.1;

c. Auxiliary Power System, Chapter 8.0;

d. Standy A-C Power Supply and Distribution, Chapter 8.0;

e. CSCS Equipment Cooling Water System, Subsection 9.2.1; and

f. Accident Analysis, Chapter 15.0.

The residual heat removal system is adequate to remove residual heat from the reactor core to ensure that fuel and reactor coolant pressure boundary design limits are not exceeded. Redundant onsite electric power systems are provided. The design of the residual heat removal system, including its power supply, meets the requirements of Criterion 34.

3.1.2.4.6 Evaluation Against Criterion 35 - Emergency Core Cooling System

The emergency core cooling system (ECCS) consists of the following: 1) high-pressure core spray system (HPCS), 2) automatic depressurization system (ADS), 3) low-pressure core spray (LPCS) system, and 4) low-pressure coolant injection (LPCI) (an operating mode of the RHRS). The emergency core cooling systems are designed to limit fuel cladding temperature over the complete spectrum of possible break sizes in the reactor coolant pressure boundary including a complete and sudden circumferential rupture of the largest pipe connected to the reactor vessel.

The HPCS system consists of a single motor-driven pump, system piping, valves, and controls and instrumentation. The HPCS system is provided to ensure that the reactor core is adequately cooled to prevent excessive fuel cladding temperatures for breaks in the nuclear system which do not result in rapid depressurization of the reactor vessel. The HPCS continues to operate when reactor vessel pressure is below the pressure at which LPCI operation or LPCS operation maintains core cooling. Two sources of water are available from either the condensate storage tank or the suppression pool.

The automatic depressurization system functions to reduce the reactor pressure so that flow from LPCI and the LPCS enters the reactor vessel in time to cool the core and prevent excessive fuel cladding temperature. The automatic depressurization system uses seven of the nuclear system pressure relief valves to relieve the high-pressure steam to the suppression pool.

The low-pressure core spray system consists of: a motor-driven pump, system piping and valves, and controls and instrumentation.

In case of low water level in the reactor vessel or high pressure in the drywell, the LPCS 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.

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

The LPCI system starts from the same signals which initiate the LPCS system and operates independently to achieve the same objective by flooding the reactor vessel.

Results of the performance of the emergency core cooling systems for the entire spectrum of liquid line breaks are discussed in Section 6.3. Peak cladding temperatures are well below the 2200° F design basis.

Also provided in Section 6.3 is an analysis to show that the emergency core cooling systems conform to 10 CFR 50 Appendix K criteria. This analysis shows complete compliance with the final acceptance criteria with the following results:

a. Peak cladding temperatures are well below the 2200° F NRC acceptability limit.

b. The amount of fuel cladding reacting with steam is nearly an order of magnitude below the 1% acceptability limit.

c. The cladding temperature transient is terminated while core geometry is still amenable to cooling.

d. The core temperature is reduced and the decay heat can be removed for an extended period of time.

The redundancy and capability of the onsite electrical power systems for the emergency core cooling systems are represented in the evaluation against Criterion 34.

For further discussion, see the following chapters, sections, or subsections:

a. Residual Heat Removal System, Subsection 5.4.7;

b. Emergency Core Cooling Systems, Section 6.3;

c. Emergency Core Cooling Systems - Instrumentation and Control, Subsection 7.3.1;

d. Auxiliary Power Systems, Chapter 8.0;

e. Standby A-C Power Supply and Distribution, Section 8.3;

f. CSCS Equipment Cooling Water System, Chapter 9.0; and

g. Accident Analysis, Chapter 15.0.

The emergency core cooling systems provided are adequate to prevent fuel and cladding damage which could interfere with effective core cooling and to limit cladding metal-water reaction to a negligible amount. The design of the emergency core cooling system, including their power supply, meets the requirements of Criterion 35.

3.1.2.4.7 ~~Evaluation Against Criterion 36 - Inspection of Emergency Core Cooling System~~

The emergency core cooling systems discussed in Criterion 35 include inservice inspection considerations. The spray spargers within the vessel are accessible for inspection during each refueling outage. Access doors in the sacrificial shield and removal panels in the vessel insulation provide access for examination of nozzles. Removable insulation is provided on the emergency core cooling systems piping out to and including the check valve inside the primary containment.

Inspection of the emergency core cooling systems is in accordance with the intent of Section XI of the ASME Code. Subsection 5.2.4 defines the inservice inspection plan, access provisions, and area of restricted access.

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 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 which are part of the reactor coolant pressure boundary are designed to specifications for inservice inspection to detect defects which might affect the cooling performance. Particular attention will be given to the reactor vessel nozzles and core spray spargers.

For further discussion, see the following sections:

a. Reactor Core Support Structures and Internals Mechanical Design, Sections 3.9 and 4.2;

b. Inservice Inspection Program, Subsection 5.2.4;

c. Reactor Vessel and Appurtenances, Section 5.3 and Subsection 3.9.5; and

d. Emergency Core Cooling Systems, Section 6.3.

The design of the reactor vessel and internals for inservice inspection, and the plant testing and inspection program ensures that the requirements of Criterion 36 are met.

3.1.2.4.8 ~~Evaluation Against Criterion 37 - Testing of Emergency Core Cooling System~~

The emergency core cooling system consists of the high-pressure core spray (HPCS) system, auto depressurization system (ADS), low-pressure coolant injection (LPCI) mode of the RHR system and low-pressure core spray (LPCS) 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 leaktight integrity of its components.

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

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

The emergency core cooling system is subjected to tests to verify the performance of the full operational sequence that brings each system into operation. The operation of the associated cooling water systems is discussed in the evaluation of Criterion 46.

For further discussion, see the following chapters, sections, or subsections:

a. Overpressurization, Subsection 5.2.2;

b. Emergency Core Cooling System Inspection and Testing, Section 6.3;

c. Emergency Core Cooling System - Instrumentation and Control, Subsection 7.3.1;

d. Standby A-C Power System, Chapter 8.0; and

e. Technical Specifications.

It is concluded that the requirements of Criterion 37 are met.

3.1.2.4.9 ~~Evaluation Against Criterion 38 - Containment Heat Removal~~

The primary containment heat removal function is accomplished by the containment cooling mode of the residual heat removal (RHR) system. In the event of a loss-of-coolant accident, within the drywell, the pressure suppression system rapidly condenses the steam to prevent overpressure of the containment. With the RHR in the suppression pool cooling subsystem of the containment cooling mode, water is pumped from the suppression pool through the RHR heat exchangers and back to the pool. In the containment spray subsystem of the containment cooling mode, water is pumped through the RHR heat exchangers and back through spray headers in the drywell and suppression chamber. Cooling systems remove heat from the reactor core, the drywell, and from the water in the suppression pool during accident conditions, and thus provide continuous cooling of the drywell.

Either or both RHR heat exchangers can be manually activated to remove energy from the suppression pool. The redundancy and capability of the offsite and onsite electrical power systems for the residual heat removal system is presented in the evaluation against Criterion 34.

The pressure suppression system is capable of rapid drywell pressure and temperature reduction following a loss-of-coolant accident so that design limits are not exceeded. Redundant onsite electrical power systems ensure that system safety functions can be accomplished.

For further discussion, see the following chapters, sections, or subsections:

a. Residual Heat Removal System, Subsection 5.4.7;

b. Containment Systems, Section 6.2;

c. Emergency Core Cooling Systems, Section 6.3;

d. Emergency Core Cooling Systems - Instrumentation and Control, Subsection 7.3.1;

e. Auxiliary Power System, Chapter 8.0;

f. Standby A-C Power Supply and Distribution, Chapter 8.0;

g. CSCS Equipment Cooling Water System, Chapter 9.0; and

h. Accident Analysis, Chapter 15.0.

The design of the containment heat removal system meets the requirements of Criterion 38.

3.1.2.4.10 ~~Evaluation Against Criterion 39 - Inspection of Containment Heat Removal System~~

The containment heat removal function is accomplished by the suppression pool cooling mode of the residual heat removal (RHR) system. During plant operations, the pumps, valves, heat exchangers, piping, instrumentation, wiring and other components located outside the primary containment can be visually inspected. Appropriate periodic inspection of the return lines and spray nozzle header inside the suppression chamber is also possible, when the plant is shut down and the suppression chamber is open for access.

Also, provisions are made to facilitate periodic inspection of the components of the containment spray mode of the RHR system. Although not required to accomplish the containment heat removal function, this mode of the RHR can be used as an alternate means of reducing the temperature in the drywell following a LOCA. Again, all components located outside the containment can be inspected during normal plant operation, and the spray headers inside the drywell can be inspected when the plant is shut down and the drywell is open for access.

For further discussion, see the following sections or subsections:

a. Residual Heat Removal System, Subsection 5.4.7;

b. Containment Systems, Section 6.2;

c. Emergency Core Cooling Systems, Section 6.3;

d. Emergency Core Cooling Systems - Instrumentation and Control, Subsection 7.3.1.

Thus, the containment heat removal system is designed to permit periodic inspection of major components. This design meets the requirements of Criterion 39.

3.1.2.4.11 ~~Evaluation Against Criterion 40 - Testing of Containment Heat Removal System~~

The containment heat removal function is accomplished by the suppression pool cooling mode of the residual heat removal (RHR) system.

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

The pumps and valves of the RHR are operated periodically to verify operability. The suppression pool cooling mode is not automatically initiated, but operation of the components is periodically verified. For further discussion see the following sections or subsections:

a. Residual Heat Removal System, Subsection 5.4.7;
b. Containment Systems, Section 6.2;
c. Emergency Core Cooling Systems, Section 6.3;
d. Emergency Core Cooling Systems Instrumentation and Control, Subsection 7.3.1.

The operation of associated cooling water systems is discussed in the response to Criterion 46. It is concluded that the requirements of Criterion 40 are met.

3.1.2.4.12 ~~Evaluation Against Criterion 41 - Containment Atmosphere Cleanup~~

As described in other sections of this UFSAR (Section 9.4), ventilation and refrigeration systems are provided in the drywell and reactor building areas to maintain suitable temperature conditions and to provide thorough mixing of the atmospheres during normal operation.

The standby gas treatment system (SGTS) is utilized during abnormal conditions to maintain the reactor building at a negative pressure and to filter the exhaust air for removal of potential fission products.

The SGTS also functions as a backup to the hydrogen recombiner and can filter air purged from the primary containment, post-LOCA, after the containment pressure has dropped below 2 psig.

A separate drywell and suppression chamber purge system is provided to clean up the drywell and suppression chamber atmospheres prior to entry of personnel for normal operation.

A combustible gas control system is provided to control the concentration of combustible gas in the primary containment following a LOCA. The containment atmosphere is continuously monitored for combustible gas concentration, and the control system can be manually operated as required. The hydrogen recombining function of the hydrogen recombiners is abandoned in place.

For further discussion, see the following chapters, sections, or subsections:

a. General Plant Description, Section 1.2;

b. Containment Functional Design, Subsection 6.2.1;

c. Containment Air Cleanup, Subsection 6.2.3;

d. ESF Filter Systems, Subsection 6.5.1;

e. Instrumentation and Controls, Chapter 7.0;

f. Electric Power System, Chapter 8.0;

g. HVAC, Section 9.4;

h. Gaseous Waste Systems, Section 11.3;

i. Process and Effluent Monitoring, Section 11.5; and

j. Accident Analysis, Chapter 15.0.

The previously described systems meet the requirements of Criterion 41.

3.1.2.4.13 ~~Evaluation Against Criterion 42 - Inspection of Containment Atmosphere Cleanup System~~

With the exception of ductwork and fans located in the drywell, all equipment of the ventilation, purge, and cleanup systems, and the combustible gas control system can be inspected during normal plant operation.

The reactor building ventilation system is operated continuously during plant operation and is monitored for satisfactory operation.

For further discussion, see the following chapters, sections, or subsections:

a. General Plant Description, Section 1.2;

b. Containment Functional Design, Subsection 6.2.1;

c. Containment Air Cleanup, Subsection 6.2.3;

d. ESF Filter Systems, Subsection 6.5.1;

e. Instrumentation and Controls, Chapter 7.0;

f. Electric Power, Chapter 8.0;

g. HVAC, Section 9.4;

h. Gaseous Waste System, Section 11.3;

i. Process and Effluent Monitoring, Section 11.5; and

j. Accident Analysis, Chapter 15.0.

The design of these systems therefore meets the requirement of Criterion 42.

3.1.2.4.14 ~~Evaluation Against Criterion 43 - Testing of Containment Atmosphere Cleanup Systems~~

This requirement is discussed under evaluation against Criterion 42. As detailed previously, the systems meet the requirements of Criteria 42 and 43.

The same references apply to those given in response to Criterion 42.

3.1.2.4.15 ~~Evaluation Against Criterion 44 - Cooling Water~~

The system provided to transfer heat from items of safety-related importance to the ultimate heat sink is the core standby cooling system - equipment cooling water system (CSCS-ECWS).

Redundancy and reliability of the cooling water supply from the ultimate heat sink is provided by three water pipelines from the lake screen house to each unit. Each unit's CSCS-ECWS consists of three independent divisions, each of which is provided with its own pumps and strainers. Each division of the CSCS-ECWS cools only essential loads of the same division. Any two divisions provide the required LOCA cooling capacity to the minimum required essential loads. The CSCS-ECWS is operable either from offsite power or from onsite emergency diesel generators.

Redundancy, isolation capability and separation are provided such that no single failure will prevent safe shutdown of both units.

For further discussion, see the following sections:

a. General Plant Description, Section 1.2;

b. CSCS Pond Flume Failure Analysis, Section 2.5.5.2.5;

c. Design of Seismic Category I Structures, Section 3.8; and

d. Water Systems, Section 9.2.

The design of this system thus meets the requirements of Criterion 44.

3.1.2.4.16 ~~Evaluation Against Criterion 45 - Inspection of Cooling Water System~~

All important components are located in accessible locations to facilitate periodic inspection during normal plant operation. Suitable manholes, handholes, inspection ports, or other design and layout features are provided for this purpose. Additionally, the shad net located across the UHS pond is frequently inspected/maintained and station procedures are in place for its inspection following a seismic event.

For further discussion, see the following chapter and sections:

a. General Plant Description, Section 1.2;

b. CSCS Pond Flume, Section 2.5.5.2.5;

c. Water Systems, Section 9.2; and

d. Initial Tests and Operation, Chapter 14.0 of the FSAR.

These features meet the requirements of Criterion 45.

3.1.2.4.17 Evaluation Against Criterion 46 - Testing of Cooling Water System

Isolation provisions have been made to permit hydrostatic testing of all portions of the CSCS-ECWS system. The CSCS-ECWS is designed to be operationally tested during any mode of plant operation without loss of capability to supply cooling water to essential loads. This testing includes transfer between the normal offsite power supplies and the onsite emergency diesel-generator power supplies. Two of the three CSCS-ECWS divisions are in service during a normal plant shutdown.

For further discussion, see the following chapters and sections:

a. General Plant Description, Section 1.2;

b. CSCS Pond Flume, Section 2.5.5.2.5;

c. Water Systems, Section 9.2;

d. Initial Tests and Operation, Chapter 14.0 of the FSAR; and

e. Technical Specifications.

The system design thus meets the requirements of Criterion 46.

3.1.2.5 Group V - Reactor Containment

3.1.2.5.1 Evaluation Against Criterion 50 - Containment Design Basis

The primary containment structure, including access openings and penetrations, is designed to withstand the peak accident pressure and temperatures that could occur during the postulated design-basis loss-of-coolant accident. In addition to incorporating appropriate safety factors into this design, considerable allowances are also included for energy addition from sources which may have been included in the postulated accident.

Further discussion of the containment design is given in the following chapter, sections, or subsections:

a. Criteria for Protection Against Dynamic Effects Associated with the Postulated Rupture of Piping, Section 3.6;

b. Seismic Design, Section 3.7;

c. Design of Containment Structure, Section 3.8;

d. Containment Functional Design, in Subsection 6.2.1;

e. Containment Heat Removal System, Subsection 6.2.2; and

f. Accident Analysis, Chapter 15.0.

The containment design thus meets the requirements of Criterion 50.

3.1.2.5.2 ~~Evaluation Against Criterion 51 - Fracture Prevention of Containment Pressure Boundary~~

The containment vessel material is tested for ductility at a temperature of 30° F below the minimum service temperature ensuring an adequately low transition temperature. Furthermore, provision is made to maintain the containment temperature at a suitable level during shutdown of the unit during cold weather.

The preoperational test program and the quality assurance program ensure the integrity of the containment and its ability to function under all normal operating and accident conditions.

Further details are given in the following chapter or subsections:

a. Containment Liner and Other Steel Elements Serving Pressure Vessel Functions, Section 5 of Appendix E;

b. Testing and Inservice Surveillance Requirements, Subsection 3.8.1; and

c. Quality Assurance, Chapter 17.0.

The containment pressure boundary thus meets the requirements of Criterion 51.

3.1.2.5.3 ~~Evaluation Against Criterion 52 - Capability of Leak Rate Testing~~

The containment system is designed and constructed and the necessary equipment is provided to permit periodic integrated leak rate tests during the plant lifetime. The testing program will be conducted in accordance with Appendix J to 10 CFR 50. Further discussion is given in Subsection 6.2.6.

The containment system thus meets the requirements of Criterion 52.

3.1.2.5.4 ~~Evaluation Against Criterion 53 - Provisions for Containment Testing and Inspection~~

A surveillance program exists whereby all penetrations are inspected and pressure tested at periodic intervals. This program consists of a leak rate testing program which is discussed in Subsection 6.2.6 and a tendon surveillance program which is discussed in Subsection 3.8.1.7. There are no penetrations with resilient seals or expansion bellows.

The containment system thus meets the requirements of Criterion 53.

3.1.2.5.5 ~~Evaluation Against Criterion 54 - Piping Systems Penetrating Containment~~

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

The engineered safety features actuation system test circuitry provides the means for testing isolation valve operability.

Conformance with Criterion 54 is further discussed in Subsections 3.1.2.5.6, 3.1.2.5.7, and 3.1.2.5.8.

The piping systems penetrating containment thus meet the requirements of Criterion 54.

3.1.2.5.6 ~~Evaluation Against Criterion 55 - Reactor Coolant Pressure Boundary Penetrating Containment~~

The reactor coolant pressure boundary, as defined in 10 CFR 50.2(V), consists of the reactor pressure vessel, pressure retaining appurtenances attached to the vessel, valves and pipes which extend from the reactor pressure vessel up to and including the outermost isolation valve. The lines of the reactor coolant pressure boundary which penetrate the primary containment have suitable isolation valves capable of isolating the primary containment thereby precluding any significant release of radioactivity. Similarly for lines which do not penetrate the primary containment but which form a portion of the reactor coolant pressure boundary, the design ensures that isolation from the reactor coolant pressure boundary can be achieved.

Further details are given in the following chapters, section, and subsection;

a. Integrity of Reactor Coolant Pressure Boundary, Section 5.2;

b. Containment Isolation Systems, Subsection 6.2.4;

c. Instrumentation and Controls, Chapter 7.0 and Appendix B;

d. Accident Analyses, Chapter 15.0; and

e. Technical Specifications.

The design of the isolation systems thus meets the requirements of Criterion 55.

3.1.2.5.7 ~~Evaluation Against Criterion 56 - Primary Containment Isolation~~

In accordance with Criterion 56, lines which penetrate the primary containment and communicate with the containment interior have two isolation valves: one inside the containment and the other outside the containment.

Further details are given in the following chapters and subsection:

a. Containment Isolation Systems, Subsection 6.2.4;

b. Instrumentation and Controls, Chapter 7.0 and Appendix B;

c. Technical Specifications.

The design of the containment isolation system thus meets the requirements of Criterion 56.

3.1.2.5.8 ~~Evaluation Against Criterion 57 - Closed System Isolation Valves~~

Each line that penetrates the primary containment and is not connected to the containment atmosphere nor part of the reactor coolant pressure boundary has at least one isolation valve located outside the containment near the penetration.

Details demonstrating conformance with Criterion 57 are provided in Subsection 6.2.4.

The design of the isolation valves thus meets the requirements of Criterion 57.

3.1.2.6 ~~Group VI - Fuel and Radioactivity Control~~

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

Waste handling systems have been incorporated in the plant design for processing and/or retention of radioactive wastes from normal plant operations to ensure that the effluent releases to the environment are as low as reasonably achievable.

The plant is also designed with provisions to prevent radioactivity releases during accidents from exceeding the limits of 10 CFR 100.

The principal gaseous effluents from the plant during normal operation are the noncondensable gases from the air ejectors. These gases are processed through a recombiner and temperature treated prior to passage into a 30-minute holdup line.

The holdup line permits decay of short-lived radioactive fission products before the gases enter the charcoal adsorbers. The effluent from this system is continuously monitored and controlled, and the system will be shut down and isolated in the event of abnormally high radiation levels.

Ventilation air from the various plant areas is continuously monitored and controlled and will be exhausted through HEPA and charcoal filters as required on a selective basis.

In the event of an accident, noncondensable gases are contained within the primary containment. The reactor building air is continuously processed through the SGTS. Exhaust from the SGTS is monitored and released in a controlled manner through HEPA and charcoal filters.

Liquid radioactive wastes are collected in waste collector tanks, treated on a batch basis through demineralizers or a vendor system using state-of-the-art technology, and then either returned to the plant systems or released in a controlled manner to the environment. All discharges to the environment are routed through a monitoring station that continuously monitors and records the activity of the waste and provides automatic isolation and an alarm to the operator in the unlikely event of high activity level.

Solid wastes including spent resins, filter sludges, filter cartridges, evaporator bottoms and contaminated tools, equipment, and clothing are collected, packaged, and shipped offsite in approved shipping containers.

These solid wastes may be stored temporarily onsite (e.g. IRSF) prior to offsite shipment.

For further discussion, see the following chapter, sections, or subsections:

a. General Plant Description, Section 1.2;

b. RCPB Leakage Detection System, Subsection 5.2.5;

c. Containment Functional Design, Subsection 6.2.1;

d. Air conditioning, Heating, Cooling, and Ventilation Systems, Section 9.4;

e. Liquid Waste Systems, Section 11.2;

f. Gaseous Waste Systems, Section 11.3;

g. Process and Effluent Radiological Monitoring Systems, Section 11.5;

h. Solid Waste System, Section 11.4;

i. Radiation Protection, Chapter 12.0; and

j. Accident Analyses, Chapter 15.0.

The design of the waste disposal systems meets the requirements of Criterion 60.

3.1.2.6.2 ~~Evaluation Against Criterion 61 - Fuel Storage. Handling, and Radioactivity Control~~

3.1.2.6.2.1 ~~New Fuel Storage~~

New fuel may be placed in dry storage in the new fuel storage vault which is located inside the reactor building. The storage vault within the reactor building provides adequate shielding for radiation protection. Storage racks preclude accidental criticality. (See evaluation against Criterion 62.)

3.1.2.6.2.2 ~~Spent Fuel Handling and Storage~~

Irradiated fuel is also stored in the reactor building. Fuel pool water is circulated through the fuel pool cooling, filtering, and demineralizing system (FPCF/D) to maintain fuel pool water temperature, purity, and water clarity. Storage racks preclude accidental criticality. (See evaluation against Criterion 62.)

3.1.2.6.2.3 ~~Radioactive Waste Systems~~

Radioactive liquids, gases, and solids, produced as a result of reactor operation, are collected, processed, and prepared for disposal by the necessary equipment provided within the radioactive waste systems. Liquid radwaste is classified as high conductivity, low conductivity, chemical, or laundry waste, to provide the most effective treatment. Liquid wastes are also decanted, leaving a residue which is accumulated for disposal as solid radwaste. Cement-solidified and dry solid radwaste is packaged in DOT approved drums or other DOT approved containers. Gaseous radwaste is monitored, filtered, processed, recorded, and controlled so that persons outside the controlled area receive doses which are below those allowed by applicable regulations.

Accessible portions of the reactor and radwaste buildings have sufficient shielding to maintain dose rates within the limits set forth in 10 CFR 20 and 10 CFR 50. The radwaste building is designed to preclude accidental release of radioactive materials to the environs.

The radwaste systems, which are used almost continuously, do not require regularly scheduled testing, with the exception of the discharge isolation valves, which are tested periodically.

For further discussion, see the following chapters or sections:

a. Residual Heat Removal System, Subsection 5.4.7;

b. Containment Systems, Section 6.2;

c. New Fuel Storage, Subsection 9.1.1;

d. Spent Fuel Storage, Subsection 9.1.2;

e. Fuel Pool Cooling and Cleanup System, Subsection 9.1.3;

f. Heating, Ventilating, and Air Conditioning System, Section 9.4;

g. Radioactive Waste Systems, Chapter 11.0; and

h. Radiation Protection, Chapter 12.0.

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

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

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 fuel assembly positions. The new and spent fuel racks are Seismic Category I components.

New fuel is placed in dry storage in the top-loaded new fuel storage vault. This vault contains a drain to prevent the accumulation of water. The new fuel storage vault racks are designed to prevent an accidental critical array, even in the event the vault becomes flooded or subjected to seismic loadings. The center-to-center new fuel assembly spacing limits the effective multiplication factor of the array to not more than 0.90 for new dry fuel. K_{eff} will not exceed 0.95 if the new fuel is flooded.

Spent fuel is stored under water 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 is maintained at a subcritical multiplication factor K_{eff} of less than 0.95 under normal and abnormal conditions.

Abnormal conditions may result from an earthquake, accidental dropping of equipment, or damage caused by the horizontal movement of fuel handling equipment without first disengaging the fuel from the hoisting equipment.

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

For further discussion, see the following sections:

a. Refueling Interlocks, Subsection 7.7.13;

b. New Fuel Storage Racks, Subsection 9.1.1; and

c. Spent Fuel Storage Racks, Subsection 9.1.2.

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

3.1.2.6.4 ~~Evaluation Against Criterion 63 - Monitoring Fuel and Waste Storage~~

Appropriate systems have been provided to meet the requirements of this criterion. A malfunction of the fuel pool cooling and cleanup system which could result in 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 and high/low level in the fuel storage pool and skimmer surge tanks. System temperature is also continuously monitored in the control room. The area radiation monitoring system monitors radioactivity in this area and initiates an alarm on abnormal radiation.

For further discussion, see the following sections:

a. Area Radiation Monitoring System, Section 7.7;

b. Fuel Storage and Handling, Section 9.1;

c. Liquid Radwaste System, Section 11.2;

d. Gaseous Radwaste System, Section 11.3; and

e. Solid Radwaste System, Section 11.4.

Area radiation, tank, and sump levels are monitored and alarmed to give indication of conditions which may result in excessive radiation levels to radioactive waste system areas. These systems satisfy the requirements of Criterion 63.

3.1.2.6.5 ~~Evaluation Against Criterion 64 - Monitoring Radioactivity Releases~~

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

a. gaseous releases from the station vent stack,

b. liquid discharge to the lake blowdown line,

c. turbine building ventilation,

d. radwaste building ventilation,

e. off-gas building ventilation,

f. reactor building ventilation,

g. control room ventilation, and

h. auxiliary equipment room ventilation.

In addition, the primary containment atmosphere is monitored, and onsite and offsite monitors are provided.

For further discussion of the means and equipment used for monitoring radioactivity releases, see the following chapter or sections:

a. Integrity of Reactor Coolant Pressure Boundary, Section 5.2;

b. Process Radiation Monitoring System, Subsection 7.6.1;

c. Site Environs Radiation Monitoring System, Sections 7.6 and 7.7; and

d. Radioactive Waste Management, Chapter 11.0.

The design meets the requirements of Criterion 64.

3.2 CLASSIFICATION OF STRUCTURES, COMPONENTS, AND SYSTEMS

Certain structures, components, and systems of the nuclear plant are considered important to nuclear safety because they perform safety actions required to prevent or 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 function they perform. In addition, design requirements are placed upon such equipment to ensure the proper performance of safety actions when required.

3.2.1 Seismic Classification

Plant structures, systems, and components important to safety are designed to withstand the effects of a safe shutdown earthquake (SSE) and remain functional, if they are required to ensure:

a. the integrity of the reactor coolant pressure boundary,

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

c. the capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures in excess of the guideline exposures of 10 CFR 100.

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

All Seismic Category I structures, systems, and components are analyzed under the loading conditions of the SSE and operating-basis earthquake (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 are based on levels of material stress or load factors, whichever is applicable, and yield margins of safety appropriate for each earthquake. The margin of safety provided for such structures, components, equipment, and systems ensures that their design functions are not jeopardized. For further details of seismic design criteria refer to:

a. mechanical, in Subsection 3.7.3;

b. electrical, in Section 3.10;

c. structural, in Subsection 3.7.2; and

d. instrumentation and controls, in Section 3.10.

3.2.2 System Quality Group Classifications

System quality group classifications have been determined for each water, steam, or radioactive waste-containing component of those applicable fluid systems which are relied upon to:

a. prevent or mitigate the consequences of accidents and malfunctions originating within the reactor coolant pressure boundary,

b. permit shutdown of the reactor and maintain it in the safe shutdown condition, and

c. contain radioactive material in large quantity or concentration.

A tabulation of quality group classification for each structure, system, and component is shown in Table 3.2-1 under the heading, "Quality Group Classification." Figures 3.2-1 and 3.2-2 are diagrams which depict the relative locations of these structures, systems, and components along with their quality group classification.

The implementation of the code requirements outlined in Tables 3.2-1, 3.2-2, 3.2-3, and 3.2-4 for fluid system components is discussed in Sections 3.7 and 3.9.

A boiling water reactor has a number of structures, systems, and components in the power conversion or other portions of the facility which have no direct safety function, but which may be connected to, or influenced by, the equipment within the nuclear safety-related classifications defined previously. Such structures, systems, and components are designated as "other."

The design requirements for equipment classified as "other" 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. When these are not available, the designer relies on accepted industry or engineering practice.

Structures, systems, and components whose safety functions require conformance to the quality assurance requirement of 10 CFR 50, Appendix B, are summarized in Table 3.2-1 under the heading, "Quality Assurance Requirements." The quality assurance program is described in Chapter 17.0.

TABLE 3.2-1
(SHEET 1 OF 31)

SEE END OF DOCUMENT FOR DISCREPANCIES

STRUCTURES, EQUIPMENT, AND COMPONENT CLASSIFICATIONS

PRINCIPAL COMPONENT (1)	(3) LOCATION	SEISMIC (5) CATEGORY	QUALITY (4a) GROUP CLASSIFICATION	QUALITY (4b) ASSURANCE REQUIREMENT	(4c) ELECTRICAL CLASSIFICATION	(2) PURCHASE DATE	COMMENTS
I. Reactor System							
1. Reactor vessel	PC	I	A	I	NA	11-70/4-71	See 5.2.1-1
2. Reactor vessel support skirt	PC	I	NA	I	NA	11-70/4-71	
3. Reactor vessel appurtenances, pressure retaining portions	PC	I	A	I	NA	7-74	
4. CRD housing supports	PC	I	NA	I	NA	11-71	(15)
5. Reactor internal structures, engineered safety features	PC	I	NA	I	NA	12-72	
6. Core support structures	PC	I	NA	I	NA	11-73	
7. Other internal structures (i.e., dryers, separators)	PC	II	NA	II	NA	6-71	(15,28)
8. Control rods	PC	I	NA	I	NA	1-71	(15)
9. Control rod drives	PC	I	NA	I	NA	9-71	
10. Power range detector hardware	PC	I	B	I	NA	6-74	(15)
11. Fuel assemblies	PC	I	NA	I	NA	5-70	(15)
12. Reactor vessel stabilizer	PC	I	NA	I	NA	2-72	
13. Reactor vessel insulation	PC	II	NA	II	NA	3-76	
II. Nuclear Boiler System							
1. Instrumentation condensing chambers	PC	I	B	I	NA	8-75	(10)
2. SRV and MSIV air accumulators	PC,RB	I	C	I	NA	9-74	(16)
3. Piping, main steam within the reactor coolant pressure boundary (RCPB)	PC,RB	I	A	I	NA	8-74	
4. Piping, feedwater within the RCPB	PC,RB	I	A	I	NA	9-74	(9)
5. Piping, feedwater within outermost isolation valve	RB	I	B	I	NA	9-74	(9)

TABLE 3.2-1
(SHEET 2 OF 31)

PRINCIPAL COMPONENT (1)	(3) LOCATION	SEISMIC (5) CATEGORY	QUALITY (4a) GROUP CLASSIFICATION	QUALITY (4b) ASSURANCE REQUIREMENT	(4c) ELECTRICAL CLASSIFICATION	(2) PURCHASE DATE	COMMENTS
6. Main Steam SRV (includes ADS)	PC	I	A	I	1E	9-74	
7. Piping, SRV discharge	PC	I	C	I	NA	9-74	
through quencher	PC	I	C	I	NA	9-74	
8. Valves, main steam isolation valves (MSIV)	PC,RB	I	A	I	1E	4-71/2-72	
9. Valves, feedwater valves within RCPB	PC,RB	I	A	I	1E	12-73	(9)
10. Valves, feedwater valves between RCPB and outermost isolation valve	RB	I	B	I	1E	12-73	(9)
11. Valves, other valves on branch lines within the RCPB	PC,RB	I	A	I	1E	12-73	(9) (10)
13. Cable with a safety function	PC,RB,A	I	NA	I	1E	10-75	
14. Electrical modules with a safety function	PC,RB,A	I	NA	I	1E	----	(15)
15. Instrument modules with a safety function	PC,RB	I	NA	I	1E	----	(15)
III. Recirculation System (Includes primary coolant sampling system)							
1. Piping	PC	I	A	I	NA	11-71	(32)
2. Pumps	PC	I	A	I	NA	5-71	
3. Valves, excluding sample lines isolation valves	PC	I	A	I	NON 1E	6-71	(19)
4. Motor, pump	PC	Special	NA	I	NON 1E	11-71	(19) (15)
5. Electrical and instrument modules with a safety funct.	PC,RB,A	I	NA	I	1E	----	(15)
6. Cable with a safety funct.	PC,RB,A	I	NA	I	1E	10-75	
7. M/B set	RB	II	NA	II	NON 1E		
8. Sample line isolation valve	PC,RB	I	B	I	1E		

TABLE 3.2-1
(SHEET 3 OF 31)

PRINCIPAL COMPONENT (1)	(3) LOCATION	SEISMIC (5) CATEGORY	QUALITY (4a) GROUP CLASSIFICATION	QUALITY (4b) ASSURANCE REQUIREMENT	(4c) ELECTRICAL CLASSIFICATION	(2) PURCHASE DATE	COMMENTS
IV. CRD Hydraulic System							
1. Valves; scram discharge volume lines	RB	I	B	I	1E	12-73	(9)
2. Valves; insert and withdraw lines	RB	I	B	I	NA	3-76	(6)
3. Valves, other	RB	II	D	II	NON 1E	12-73	
4. Piping, scram discharge volume lines	RB	I	B	I	NA	9-74	(23)
5. Piping, insert and withdraw lines	PC,RB	I	B	I	NA	3-76	
6. Piping, other	RB	II	D	II	NA	9-74	
7. CRD pumps, filter	RB	II	D	II	NA	2-71	(15)
8. CRD strainer	RB	II	D	II	NA	7-71	(15)
9. SDV level switches	RB	I	D	I	1E		
10. Electrical and instrumentation modules without safety function	RB	II	NA	II	NON 1E		
11. Hydraulic control unit and shutoff valves	RB	I	D	I	NON 1E	10-71	(13) (15)
12. Pump Motor	RB	II	NA	II	NON 1E	4-71	
13. Cable with safety function	RB,A	I	NA	I	1E	10-75	
14. Electrical and instrumentation modules with safety function	RB,A	II	NA	I	1E	---	
V. Standby Liquid Control System							
1. Standby liquid control storage tank	RB	I	B	l	NA	3-74	
2. Pumps	RB	I	B	I	NA	7-71	
3. Pump motor	RB	I	NA	I	NON 1E	7-71	
4. Valves, explosive	RB	I	A	I	NON 1E	2-72	
5. Valves, isolation and within primary containment	RB,PC	I	A	I	NA	---	
6. Valves, beyond isolation valves	RB	I	B	I	NON 1E	12-73	
7. Piping, within isolation valves	PC,RB	I	A	I	NA	9-74	

TABLE 3.2-1
(SHEET 4 OF 31)

PRINCIPAL COMPONENT (1)	(3) LOCATION	SEISMIC (5) CATEGORY	QUALITY (4a) GROUP CLASSIFICATION	QUALITY (4b) ASSURANCE REQUIREMENT	(4c) ELECTRICAL CLASSIFICATION	(2) PURCHASE DATE	COMMENTS
8. Piping, beyond isolation valves	RB	I	B	I	NA	9-74	
9. Electrical and instrument modules	RB	I	NA	I	NON 1E	---	
10. Cable	RB,A	II	NA	I	NON 1E	10-75	(15)
11. Serv. System valves and piping	RB	II	D	II	NON 1E	6-73	
VI. Neutron Monitoring System							
1. Piping, TIP	PC,RB	I	B	I	NA	1-74	
2. Valve, isolation, TIP subsystem	PC,RB	I	B	I	NA	1-74	
3. Electrical modules, IRM and APRM	RB	I	NA	I	1E	1-74	(15)
4. Cable, IRM and APRM	RB,A	II	NA	I	1E	5-75	
5. LPRM, incore detector assemblies	PC	I	A	I	1E	---	
VII. Reactor Protection System							
1. Electrical and instrument modules	T,PC,RB,A	I	NA	I	1E	2-74	
2. Cables	T,PC,RB,A	I	NA	I	1E	10-75	
VIII. Process Radiation Monitors							
1. Electrical and instrument modules main steam line and reactor building ventilation monitors	RB,A	I	NA	I	1E	5-74	(15)
2. Cable, main steamline and reactor building ventilation monitors	RB,A	I	NA	I	1E	10-75	
3. Electrical and instrument modules for process liquid, process ventilation, air ejector and off-gas radiation monitoring systems	A,T,RB	II	NA	II	NON 1E	7-74	(15)
IX. Residual Heat Removal (RHR) System							
1. Heat exchangers, primary side	RB	I	B	I	NA	5-71	(35)
2. Heat exchangers, secondary side	RB	I	C	I	NA	5-71	(36)
3. Piping, connected to RCPB within outermost isolation valves	PC,RB	I	A	I	NA	9-74	(10)

TABLE 3.2-1
(SHEET 5 OF 31)

PRINCIPAL COMPONENT (1)	(3) LOCATION	SEISMIC (5) CATEGORY	QUALITY (4a) GROUP CLASSIFICATION	QUALITY (4b) ASSURANCE REQUIREMENT	(4c) ELECTRICAL CLASSIFICATION	(2) PURCHASE DATE	COMMENTS
4. Piping, excluding that connected to the RCPB	RB	I	B	I	NA	9-74	
5. Pumps, RHR and water leg	RB	I	B	I	NA	9-70,6-74	
6. Pump motors	RB	I	NA	I	1E	10-70,6-74	(15)
7. Valves; RCP boundary isolation LPCI and shutdown lines	PC,RB	I	A	I	1E	12-73	
8. Valves; isolation, other	PC,RB	I	B	I	1E	12-73	
9. Valves, beyond isolation valves	RB	I	B	I	1E	12-73	
10. Electrical and instrument modules with safety funct.	RB,A	I	NA	I	1E	----	(15)
11. Cable, with safety function	RB,A	I	NA	I	1E	10-75	
X. Low-Pressure Core Spray System LPCS							
1. Piping, within outermost isolation valves	PC,RB	I	A	I	NA	9-74	(10)
2. Piping, beyond outermost isolation valves	RB	I	B	I	NA	9-74	
3. Pumps, LPCS and water leg	RB	I	B	I	NA	9-70,6-74	
4. Pump motors	RB	I	NA	I	1E	10-70,6-74	(15)
5. Valves, RCP boundary isolation valves within contain.	PC,RB	I	A	I	1E	12-73	
6. Valves, beyond outermost isolation valves	RB	I	B	I	E	12-73	
7. Electrical and instrument modules with a safety funct.	RB,A	I	NA	I	1E	5-75	(15)
8. Cable, with a safety function			NA	I	E	--	
XI. High-Pressure Core Spray System, HPCS							
1. Piping, within outermost isolation valves	PC,RB	I	A	I	NA	9-74	(10)
2. Piping, beyond outermost isolation valves	RB	I	B	I	NA	9-74	
3. Piping, return test line to condensate storage tank beyond reactor building	O,RB	II	D	II	NA	9-74	(33)

TABLE 3.2-1
(SHEET 6 OF 31)

PRINCIPAL COMPONENT (1)	(3) LOCATION	SEISMIC (5) CATEGORY	QUALITY (4a) GROUP CLASSIFICATION	QUALITY (4b) ASSURANCE REQUIREMENT	(4c) ELECTRICAL CLASSIFICATION	(2) PURCHASE DATE	COMMENTS
4. Pumps, HPCS and water leg	RB	I	B	I	NA	1-71, 6-74	
5. Valves, RCPB within outermost valve	PC,RB	I	A	I	1E	12-73	
6. Valves, excluding those within RCPB	RB	I	B	I	1E	12-73	
7. Electrical and instrument modules with a safety funct.	RB,A	I	NA	I			(15)
8. Cable, with a safety funct.	RB,A	I	NA	I	1E	10-75	
9. Motors	RB	I	NA	I	1E	1-71, 6-74	(15)
XII. Reactor Core Isolation Cooling System RCIC							
1. Piping, connected to RCPB within outermost isolation valves	PC,RB	I	A	I	NA	9-74	(10)
2. Piping, beyond outermost isolation valves	RB	I	B	I	NA	9-74	
3. Piping, return test line to condensate storage tank beyond reactor building	0	II	D	II	NA	9-74	
4. Vacuum pump discharge line from vacuum pump to containment isolation valves	RB	II	D	II	NA	9-74	
5. Pumps, RCIC and water leg	RB	I	B	I	NA	1-71, 6-74	
6. Valves, RCP boundary isolation & valves within contain.	PC,RB	I	A	I	1E	12-73	
7. Valves, return test line to condensate storage	RB	I	B	I	1E	12-73	
8. Valves, other	RB	I	B	I	1E	12-73	
9. Turbine	RB	I	E	--	NA	10-70	(11,15,17,28)
10. Electrical and instrument modules with a safety funct.	RB,A	I	N/A	I	1E	----	(15)
11. Cable, with a safety function	--	I	N/A	I	1E	10-75	
12. Water leg pump motor	RB	I	N/A	I	1E	6-74	(15)
13. Piping, within outermost isolation values excluding that connected to the RCPB	PC, RB	I	B	I	NA	9-74	

TABLE 3.2-1
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PRINCIPAL COMPONENT (1)	(3) LOCATION	SEISMIC (5) CATEGORY	QUALITY (4a) GROUP CLASSIFICATION	QUALITY (4b) ASSURANCE REQUIREMENT	(4c) ELECTRICAL CLASSIFICATION	(2) PURCHASE DATE	COMMENTS
XIII. FUEL SERVICE EQUIPMENT							
1. Fuel preparation machine	RB	I	N/A	I	N/A	10-72	(15)
2. General purpose grapple	RB	I	N/A	I	N/A	1-73	(15)
XIV. REACTOR VESSEL SERVICE EQUIPMENT							
1. Steamline plugs	RB	I	N/A	I	N/A	4-71	(15)
2. Dryer and separator sling	RB	I	N/A	I	N/A	4-75	(15)
3. Head strongback	RB	I	N/A	I	N/A	11-75	(15)
XV. IN-VESSEL SERVICE EQUIPMENT							
1. Control rod grapple	RB	I	N/A	I	N/A	8-75	(15)
XVI. REFUELING EQUIPMENT							
1. Refueling platform	RB	I	N/A	I	Non 1E	11-73	
2. Refueling bellows, reactor cavity	PC	I	N/A	II	N/A	1-76	
3. New fuel inspection stand	RB	II	N/A	II	N/A	1-73	(15)
XVII. STORAGE EQUIPMENT							
1. Fuel storage racks	Unit 1 RB	I	C	I	N/A	11-91	
	Unit 2 RB	I	C	I	N/A	1-86	
2. Defective fuel storage container	Unit 1 RB	I	N/A	I	N/A	12-75	(15)
	Unit 2 RB	l	C	I	N/A	1-86	
3. Spent fuel pool, dryer/ sep. pool, Rx well	Unit 2 RB	I	N/A	I	N/A	---	
XVIII. Radwaste System							
1. Tanks, atmospheric	RW,T	II	D	II	NA	12-73	
2. Heat exchangers	RW,T	II	D	II	NA	6-73	
3. Piping, other	RB,T,RW	II	D	II	NA	9-74	
4. Pumps	RB,RW,T, A	II	D	II	NON 1E	7-73	

TABLE 3.2-1
(SHEET 8 OF 31)

PRINCIPAL COMPONENT (1)	(3) LOCATION	SEISMIC (5) CATEGORY	QUALITY (4a) GROUP CLASSIFICATION	QUALITY (4b) ASSURANCE REQUIREMENT	(4c) ELECTRICAL CLASSIFICATION	(2) PURCHASE DATE	COMMENTS
5. Valves, flow control and filter system	RW,T	II	D	II	NON 1E	6-73	
6. Valves, other	RB,RW,T, A	II	D	II	NON 1E	6-73	
XIX. Reactor Water Cleanup System							
1. Vessels: filter/ demineralizer	RB	II	C	II	NA	7-71	
2. Heat exchangers	RB	II	C	II	NA	7-71	(14)
3. Piping, within RCPB outermost valve	RB,PC	I	A	I	NA	9-74	
4. Piping, within outermost isolation valves	RB	I	B	I	NA	9-74	(14,10)
5. Piping, other	RB	II	C	II	NA	9-74	(14,10)
6. Pumps/motors	RB	II	C	II	NON 1E	7-71	(14)
7. Valves, within RCPB	RB,PC	I	A	I	IE	12-73	
8. Valves, within containment boundary	RB	I	B	I	IE	12-73	
9. Valves, beyond outermost isolation valves	RB	II	C	II	NON 1E	12-73	
XX. Fuel Pool Cooling and Cleanup System							
1. Pumps, cooling	RB	II	C	II	NON 1E	6-74	
2. Heat exchangers	RB	II	C	II	NA	12-73	
3. Filter demineralizer vessels	T	II	C	II	NA	7-73	
4. Pumps, holding	T	II	C	II	NON 1E	7-73	
5. Precoat facility	T	II	D	II	NON 1E	7-73	
6. Piping	RB,T,A	II	C, D	II	NA	9-74	(18)
7. Valves	RB,T	II	C, D	II	NA	12-73	
8. Pumps, emergency makeup	A	I	C	I	1E	7-73	
9. Valves, emergency makeup	A,RB	I	C	I	NA	12-73	
10. Piping, emergency makeup	A,RB	I	C	I	NA	12-73	

TABLE 3.2-1
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PRINCIPAL COMPONENT (1)	(3) LOCATION	SEISMIC (5) CATEGORY	QUALITY (4a) GROUP CLASSIFICATION	QUALITY (4b) ASSURANCE REQUIREMENT	(4c) ELECTRICAL CLASSIFICATION	(2) PURCHASE DATE	COMMENTS
XXI. Control Room Panels							
1. Electrical panels with safety function	A	I	NA	I	1E	5-74,4-75	(15)
2. Cable with safety function	A	I	NA	I	1E	10-75	
XXII. Local Panels							
1. Electrical panels with a safety function	A,RB	I	NA	I	1E	4-74	(15)
2. Cable, with a safety function	A,RB	I	NA	I	1E	10-75	
3. Remote shutdown panel	A	I	NA	I	1E	10-74	
XXIII. Off-Gas System (2)							
1. Atmospheric glycol tanks	F	II	D	II	NA	10-71	
2. Heat exchangers	F,T	II	D	II	NA	10-74	
3. Piping and valves (downstream of steam jet air ejectors)	T,F,O	II	D	II	NON 1E	9-74	
4. Piping and valves (up to and including air ejector)	T	II	D	II	NON 1E	9-74	
5. Valves	T,F	II	D	II	NON 1E		
6. Steam jet air ejectors	T	II	D	II	NA	2-72	
7. Charcoal vessels	F	II	D	II	NA	10-71	
8. Recombiners	T	II	D	II	NA	10-71	
9. Filters	F	II	D	II	NA	10-71	
10. Afterfilter	F	II	D	II	NA	10-71	
11. Reheater	F	II	--	II	NON 1E	1-72	
12. Flow Elements/Transmitters 1&2 N62-N010 and 1&2 N62-N032	F	II	D	II	NON 1E	10-82	(30)

TABLE 3.2-1
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PRINCIPAL COMPONENT (1)	(3) LOCATION	SEISMIC (5) CATEGORY	QUALITY (4a) GROUP CLASSIFICATION	QUALITY (4b) ASSURANCE REQUIREMENT	(4c) ELECTRICAL CLASSIFICATION	(2) PURCHASE DATE	COMMENTS
XXIV. Service Water System							
1. Piping	RB,O,L,A,T	II	D	II	NA	9-74	
2. Strainers	L	II	D	II	NA	7-73	
3. Pumps	L	II	D	II	NA	7-73	
4. Pump motors	L	II	--	II	NON 1E	7-73	
5. Valves	T,O,L,A,RB	II	D	II	NA	6-73	
6. Electrical & instrument Modules	RB, L,A	II	--	II	NON 1E	--	
7. Cable	RB,O,L,A,T	II	--	II	NON 1E	10-75	
XXV. Drywell Pneumatic, Instrument Air, and Service Air System							
1. MSIV and SRV accumulators	PC,RB	I	C	I	NA	9-74	(16)
2. Piping in lines between accumulators and MSIV's and SRV's and to N_2 bottles	PC,RB	I	C	I	NA	9-74	(16)
3. Valves in lines supporting MSIV and SRV function	PC,RB	I	C	I	NA	12-73	
4. Valves, containment isolation	RB	I	B	I	1E	12-73	
5. Piping, within outermost isolation valve	RB	I	B	I	NA	9-74	
6. Piping, other	PC,RB,L,A,T,RW,F,O	II	D	II	NA	9-74	
7. Valves, other	PC,RB,L,A,T,RW,F,O	II	D	II	NA	9-74	
8. Compressors	RB,L	II	D	II	NON 1E		
9. Nitrogen bottles supplying ADS valves	RB	I			NA		
10. Drywell Pneumatic non-ADS Supply Regulator	RB	II	D	II	NA	11-89	

TABLE 3.2-1
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PRINCIPAL COMPONENT (1)	(3) LOCATION	SEISMIC (5) CATEGORY	QUALITY (4a) GROUP CLASSIFICATION	QUALITY (4b) ASSURANCE REQUIREMENT	(4c) ELECTRICAL CLASSIFICATION	(2) PURCHASE DATE	COMMENTS
XXVI. CSCS Equipment Cooling Water System							
1. Intake water tunnel	L	I	NA	I	NA	--	
2. Piping	O,A, L	I	C	I	NA	9-74	(31)
3 Valves	A, L	I	C	I	1E	12-73	(31)
4 Strainers	A	I	C	I	NA	11-73	
5. RHR Service Water Fuel Pool Emergency Makeup Pumps	A	I	C	I	NA	6-74	
6. Pump Motor	A	I	NA	I	1E	6-74	
7. UHS Instr. and Controls	L, RB	I	C	I	NA	9-74	
XXVII. Diesel Generator System							
1. Day tanks	A	I	C	I	NA	11-73	(21)
2. Piping; fuel oil system, diesel serv. water system, starting air system, downstream of and including the compressed isolation check valve, and intake and exhaust system	A	I	C	I	NA	9-74	(21)
3. Valves, fuel oil system and diesel service water system	A	I	C	I	1E	12-73	
4. Pumps, fuel oil system and Divisions 1 and 2 diesel service water system	A	I	C	I	NA	6-74	
5. Pump motors, fuel oil system and Divisions 1 and 2 diesel service water system	A	I	NA	I	1E	6-74	
6. Diesel generators (Divisions 1 and 2 only)	A	I	NA	I	1E	1-74	

TABLE 3.2-1
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PRINCIPAL COMPONENT (1)	(3) LOCATION	SEISMIC (5) CATEGORY	QUALITY (4a) GROUP CLASSIFICATION	QUALITY (4b) ASSURANCE REQUIREMENT	(4c) ELECTRICAL CLASSIFICATION	(2) PURCHASE DATE	COMMENTS
7. Electrical modules with safety function (Division 1 and 2 only)	A	I	NA	I	1E	1-74	(15)
8. Cable, with safety funct.	A	I	NA	I	1E	10-75	
9. Diesel fuel storage tanks	A	I	C	I	NA	12-73	
10. Diesel starting air receiver tanks (Division 1 and 2 only)	A	I	C	I	NA	1-74	
11. Starting air system equipment, piping, and valves upstream of compressor isolation check valve	A	I	D	I	NA	9-74	
12. Diesel exhaust silencer (Divisions 1 and 2 only)	A	I	NA	I	NA	1-74	(21)
13. Diesel intake filter (Divisions 1 and 2 only)	A	I	NA	I	NA	1-74	(21)
14. Division 3 diesel Service water pump	A	I	C	I	NA	6-71	
15. Division 3 diesel service water pump motor	A	I	NA	I	1E	6-71	
16. Division 3 diesel generators	A	I	NA	I	1E	6-71	
17. Division 3 electrical modules with safety function	A	I	NA	I	1E	1-73	(15)
18. Division 3 diesel starting air receiver tanks	A	I	C	I	NA	6-71	
19. Division 3 diesel exhaust silencer	A	I	NA	I	NA	6-71	(21)
20. Division 3 diesel intake filter	A	I	NA	I	NA	6-71	(21)

TABLE 3.2-1
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PRINCIPAL COMPONENT (1)	(3) LOCATION	SEISMIC (5) CATEGORY	QUALITY (4a) GROUP CLASSIFICATION	QUALITY (4b) ASSURANCE REQUIREMENT	(4c) ELECTRICAL CLASSIFICATION	(2) PURCHASE DATE	COMMENTS
XXVIII. Combustible Control System							
1 Piping	RB	I	B	I	NA	9-74	
2. Valves	RB	I	B	I	1E	12-74	
3. Gas control unit on skid	--	I	B	I	1E	11-76	
4. Electrical modules with a safety function	RB,A	1	NA	I	1E	11-76	(15)
5. Cables, with a safety function	RB,A	I	NA	I	1E	10-75	
XXIX. Standby Gas Treatment System							
1. a. Piping and valves (downstream of filter unit)	RB	I	C	I	1E	9-74	
b. Piping and valves (upstream of filter unit)	RB	I	D	II	1E	9-74	
2. SGTS equipment train (includes filters)	RB	I	NA	I	1E	4-76	
3. Electrical/mechanical modules, with a safety function	RB,A	I	NA	I	1E	4-76	
4. Cable, with a safety function	RB,A	I	NA	I	1E	10-75	
5. Instr. and Controls	RB,A	I	D	I	1E	9-74	
XXX. Primary Containment Ventilation and Ventilation Water System							
1. All components, except containment isolation valves and penetration piping	PC,RB	II	NA	II	NON 1E	--	
2. Valves, containment isolation	PC,RB	I	B	I	1E	12-73	
3. Piping, penetration	PC,RB	I	B	I	NA	9-74	

TABLE 3.2-1
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PRINCIPAL COMPONENT (1)	(3) LOCATION	SEISMIC (5) CATEGORY	QUALITY (4a) GROUP CLASSIFICATION	QUALITY (4b) ASSURANCE REQUIREMENT	(4c) ELECTRICAL CLASSIFICATION	(2) PURCHASE DATE	COMMENTS
XXXI. Power Conversion System							
1. Main steam piping between outermost isolation valves up to turbine stop valves	RB,T,A	I	D+	II	NA	9-74	(7,28)
2. Main steam branch piping to first valve closed or capable of automatic actuation	T,RB,A	I	D+	II	NA	9-74	(7,28)
3. Main steam branch piping after the first closed valve or valve capable of automatic actuation	T	II	D	II	NA	9-74	
4. Main turbine bypass piping up to bypass valve	T	I	D+	II	NA	9-74	(7,28)
5. First valve that is either normally closed or capable of automatic closure in branch piping connected to main steam and turbine bypass piping	T,RB	I	D+	II	1E	6-73	(7,28)
6. Turbine stop valves, turbine control valves, and turbine bypass valves	T	II	D	II	NON 1E	--	
7. Main steam leads from turbine control valve to turbine casing	NA	II	NA	II	NA	--	No piping
8. Feedwater and condensate system beyond containment isolation valve	RB,T,A	II	D	II	NON 1E	9-74	(9,38)

TABLE 3.2-1
(SHEET 15 OF 31)

PRINCIPAL COMPONENT (1)	(3) LOCATION	SEISMIC (5) CATEGORY	QUALITY (4a) GROUP CLASSIFICATION	QUALITY (4b) ASSURANCE REQUIREMENT	(4c) ELECTRICAL CLASSIFICATION	(2) PURCHASE DATE	COMMENTS
XXXII. Cycled Condensate Storage and Transfer System							
1. Condensate storage tank	O	II	D	II	NA	--	(12)
2. Piping, suction line to HPCS	O,RB	I	D	I	NA	9-74	(33)
3. Piping, suction line to RCIC	O, RB	I	B	I	NA	9-74	
4. Piping, other	O,RB,T,A, RW	II	D	II	NA	9-73	
5. Valves and other components	O,RB,T,A, RW		D	II	NA	6-73	
6. Instrument and Controls	T,A	II	D	II	NON 1E	9-74	
XXXIII. Class 1E Onsite Power Systems							
1. Diesel generator and directly associated auxiliaries	A	I	NA	I	1E	--	
2. 4160 volt switchgear and associated protective relays	A	I	NA	I	1E	--	
3. 480 volt unit substations (switchgear and supply transformers)	A	I	NA	I	1E	--	
4. 480 volt motor control centers including 120 volt AC instrument bus distribution equipment	A,RB	I	NA	I	1E	--	
5. Instrumentation, control and power cables (including cable splices and terminal blocks)	A,RB,PC	I	NA	I	1E	--	
6. Conduit supports and cable trays and their supports	A,RB,PC	I	NA	I	NON 1E	--	(25)
7. Conduits	A,RB,PC	I	NA	II	NON 1E	--	(25)
8. Control panels	A	I	NA	I	1E	--	
9. Containment electrical penetration assemblies	PC	I	NA	I	1E	--	
10. Other cable penetrations (fire stops)	A,RB,PC	II	NA	II	NON 1E	--	

TABLE 3.2-1
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PRINCIPAL COMPONENT (1)	(3) LOCATION	SEISMIC (5) CATEGORY	QUALITY (4a) GROUP CLASSIFICATION	QUALITY (4b) ASSURANCE REQUIREMENT	(4c) ELECTRICAL CLASSIFICATION	(2) PURCHASE DATE	COMMENTS
XXXIV. Class 1E DC Power Systems							
1. 125 and 250 volt batteries, battery chargers, and distribution equipment (including any protective relays)	A	I	NA	I	1E	--	
2. Cables	A,RB,PC	I	NA	I	1E	--	
3. Conduit supports and cable trays and their supports	A,RB,PC	I	NA	I	NON 1E	--	(25)
4. Conduits	A,RB,PC	I	NA	II	NON 1E	--	(25)
5. Battery racks	A	I	NA	I	NON 1E	--	
6. Control panels	A	I	NA	I	NON 1E	--	
XXXV. Miscellaneous Components							
1. Meteorological Monitoring	O	II	NA	II	NON 1E		
XXXVI. Reactor Building Closed Cooling Water System							
1. Pumps and heat exchangers	RB	II	D	II	NA	6-73	
2. Valves, containment isolation & containment penetration piping	PC,RB	I	B	I	1E	12-73	
3. Piping, other	PC,RB,A	II	D	II	NA	9-74	
4. Valves, other	PC,RB,A	II	D	II	NA	6-73	

TABLE 3.2-1
(SHEET 17 OF 31)

PRINCIPAL COMPONENT (1)	(3) LOCATION	SEISMIC (5) CATEGORY	QUALITY (4a) GROUP CLASSIFICATION	QUALITY (4b) ASSURANCE REQUIREMENT	(4c) ELECTRICAL CLASSIFICATION	(2) PURCHASE DATE	COMMENTS
XXXVII. Equipment and Floor Drainage System							
1. Sumps	RB,T,RW, A,PC	II	D	II	NA		
2. Pumps	RB,T,RW, A		D	II	NA	10-75	
3. Piping containment isolation	RB	I	B	I	NA	9-74	
4. Valves containment isolation	RB	I	B	I	1E	12-73	
5. Cable, with a safety function	--	II	NA	I	1E		
6. Piping, other	RB,T,RW, A,PC	II	D	II	NA	9-74	
7. Valves, other	RB,T,RW, A,PC	II	D	II	NA	6-73	
XXXVIII. HVAC Systems							
1. Control Room HVAC System							
a. Refrigeration units	A	I	NA	I	1E	1-76	
b. Fans and motors	A	I	NA	I	1E	5-76	
c. Cooling coils	A	I	NA	I	NA	5-76	
d. Refrigerant piping and accessories	A	I	NA	I	NA	2-76	
e. Ductwork and accessories	A	I	NA	I	NA	2-76	
f. Elec. & instrument with a safety function	A	I	NA	I	1E		
g. Filters	A	I	NA	I	NA		

TABLE 3.2-1
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PRINCIPAL COMPONENT (1)	(3) LOCATION	SEISMIC (5) CATEGORY	QUALITY (4a) GROUP CLASSIFICATION	QUALITY (4b) ASSURANCE REQUIREMENT	(4c) ELECTRICAL CLASSIFICATION	(2) PURCHASE DATE	COMMENTS
2. Auxiliary Electric Equipment Room HVAC System							
a. Refrigeration units	A	I	NA	I	1E	1-76	
b. Fans and motors	A	I	NA	I	1E	5-76	
c. Cooling coils	A	I	NA	I	NA	5-76	
d. Refrigerant piping and accessories	A	I	NA	I	NA	2-76	
e. Ductwork and accessories	A	I	NA	I	NA	2-76	
f. Elec. & instrument with a safety function	A	I	NA	I	1E		
g. Filters	A	I	NA	I	NA		
3. Diesel Generator Room Vent System							
All components	A	I	NA	I	1E	5-76, 2-76	
4. Essential Switchgear Room Ventilation System							
All components	A	I	NA	I	1E	5-76, 2-76	
5. CSCS Equipment Area Cooling System							
Fan motor	RB	I	NA	I	1E	5-76	
Damper actuator	RB	I	NA	I	1E	10-76	
Control switch	RB	I	NA	I	1E	10-76	
Temperature switch	RB	I	NA	I	1E	10-76	
Diff. pressure indicator	RB	I	NA	II	NON-1E	10-76	(26)
Temperature element	RB	I	NA	I, II	NON-1E, 1E	10-76	(26)
Temperature indicating controller	RB	I	NA	I, II	NON-1E, 1E	10-76	(26)
Temperature controller	RB	I	NA	I	1E	10-76	
6. Reactor Building Vent System							
Secondary containment isolation dampers	A	I	NA	I	1E	8-76	
Main steam airflow check dampers	RB	I	NA	I	NIE		
Exhaust air duct pressure relief damper	A	I	NA	I	NA		
Exhaust air duct excess flow check damper	A	I	NA	I	NA		

TABLE 3.2-1
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PRINCIPAL COMPONENT (1)	(3) LOCATION	SEISMIC (5) CATEGORY	QUALITY (4a) GROUP CLASSIFICATION	QUALITY (4b) ASSURANCE REQUIREMENT	(4c) ELECTRICAL CLASSIFICATION	(2) PURCHASE DATE	COMMENTS
7. Primary Containment Purge System							
Primary containment isolation valves	RB	I		I	1E	--	
Secondary containment isolation valves	A	I		I	1E	--	
XXXIX. Area Radiation Monitoring System							
1. All components	RW,T,A,RB	II	NA	II	NON 1E	--	(15)
XL. Leak Detection System							
1. Temperature element	PC,RB,T	I	NA	I	1E	--	(15)
2. Temperature switch	PC,RB,T	I	NA	I	1E	--	(15)
3. Differential temperature switch	PC,RB,T	I	NA	I	1E	--	(15)
4. Differential flow switch	PC,RB	I	NA	I	1E	--	(15)
5. Pressure switch	PC,RB	I	NA	I	1E	--	(15)
6. Differential pressure switch	PC,RB	I	NA	I	1E	--	(15)
7. Differential flow summer	RB	I	NA	I	1E	--	(15)
8. Reactor building floor drain sumps	RB	II	NA	II	NA	--	
9. Reactor building floor drain pumps and piping	RB	II	NA	II	NA	--	
10. Digital Recorders		I	NA	I	1E	--	(15)
XLI. Fire Protection System							
1. Water spray deluge systems	-	II	NA	II	NA	7-75	
2. Sprinkler systems	-	II	NA	II	NA	7-75	
3. Carbon dioxide systems	-	II	NA	II	NA	7-76	
4. Portable and wheeled extinguishers	-	II	NA	II	NA	--	
5. Halon system	-	II	NA	II	NA	7-76	

TABLE 3.2-1
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PRINCIPAL COMPONENT (1)	(3) LOCATION	SEISMIC (5) CATEGORY	QUALITY (4a) GROUP CLASSIFICATION	QUALITY (4b) ASSURANCE REQUIREMENT	(4c) ELECTRICAL CLASSIFICATION	(2) PURCHASE DATE	COMMENTS
XLII. Civil Structures							
1. Reactor building	RB	I	NA	I	NA		(22)
2. Lake screen house	L	Note 27	NA	II	NA		(22)
3. Radwaste building	RW	II, Note 34	NA	II	NA		(22)
4. Auxiliary building	A	I	NA	I	NA		(22)
5. Turbine building	T	II, Note 34	NA	II	NA		(22)
6. Off-gas filter building	F	II , Note 34	NA	II	NA		(22)
7. Steam tunnel	A	I	NA	I	NA		(22)
8. River screen house	O	II	NA	II	NA		(22)
9. Diesel-generator building	RB	I	NA	I	NA		(22)
10. Auxiliary Spillway	O	NA	NA	II	NA		(22)
11. Cooling Lake Embankment	O	II	NA	II	NA		(22, 37)
12. Submerged CSCS Pond (Ultimate Heat Sink)	O	I	NA	I	NA		(22, 33)
13. Biological Shield	PC	I	NA	I	NA		(22)
14. Primary Containment	PC	I	NA	I	NA		
a. Vacuum breaker piping	PC/RB	I	B	I	NA	9-74	
b. Vacuum breaker valves	PC/RB	I	B	I	1E		
c. Maintenance butterfly valves	PC/RB	I	B	I	NA		
d. Suppression vent downcomers	PC	I	NA	I	NA	----	
15. Interim Radwaste Storage Facility	O	II	N/A	II	N/A		
XLIII. Deleted							

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PRINCIPAL COMPONENT (1)	(3) LOCATION	SEISMIC (5) CATEGORY	QUALITY (4a) GROUP CLASSIFICATION	QUALITY (4b) ASSURANCE REQUIREMENT	(4c) ELECTRICAL CLASSIFICATION	(2) PURCHASE DATE	COMMENTS
XLIV. Clean Condensate System							
1. Condensate storage tank	O	II	D	II	NA	12-73	
2. Transfer pumps and	T	II	D	II			
motors	T	II	D	II	NON 1E	11-72	
3. Piping, within outermost isolation valve	RB	I	B	I	NA	9-74	
4. Piping, other	RC,RB,A, T,RW,F	II	D	II	NA	9-74	
5. Valves, isolation	RB	I	B	I	NA	12-73	
6. Valves, other	PC,RB,S, T,RW,F	II	D	II	NA	9-74	
XLV. Containment Monitoring							
1. Piping, within containment pressure boundary and/or with post-LOCA function	RB	I	B	I	NA	9-74	Note (26)
2. Piping, outside containment pressure boundary and with no post-LOCA function	RB	II	D	II	NA		
3. Valves, within containment pressure boundary and/or with post-LOCA function	RB	I	B	I	1E		
4. Valves, outside containment pressure boundary and with no post-LOCA function	RB	II	D	II	NON 1E		

TABLE 3.2-1
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PRINCIPAL COMPONENT (1)	(3) LOCATION	SEISMIC (5) CATEGORY	QUALITY (4a) GROUP CLASSIFICATION	QUALITY (4b) ASSURANCE REQUIREMENT	(4c) ELECTRICAL CLASSIFICATION	(2) PURCHASE DATE	COMMENTS
5. Electrical and instrumentation modules with post-LOCA function	RB	I	NA	I	1E	1-80	
6. Electrical and instrumentation modules with no post-LOCA function	RB	II	NA	II	NON 1E	7-77	
XLVI. Alternate Rod Insertion/ MSIV Level 1 Closure System							
1. Electrical and instrumentation modules	RB,A	I	NA	I	1E	3-85	
2. Cables	RB,A	I	NA	I	1E	3-85	
XLVII. Electrical Penetration Pressurization System (Unit 2 Only)							
1. Piping and valves between Electrical Penetration and the outboard isolation valve.		I	B	I	NA	1-78	(29)
2. Piping and valves upstream of out board isolation valve.		II	D	II	NA	---	
XLVIII. Hydrogen Water Chemistry (HWC) System							
1. Hydrogen Storage Equipment	0	I	D	II	Non 1E	8-95	
2. Nitrogen Storage Equipment	0	II	D	II	Non 1E	8-95	
3. Oxygen Storage Equipment	0	II	D	II	Non 1E	8-95	
4. Outdoor Piping	0	II	D	II	NA	8-95	
5. Hydrogen Injection Equipment	T	II	D	II	Non 1E	8-95	
6. Oxygen/Air Injection Equipment	T	II	D	II	Non 1E	8-95	

TABLE 3.2-1
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EQUIPMENT CLASSIFICATION COMMENTS

(1) A module is an assembly of interconnected components which constitute an identifiable device or piece of equipment. For example, electrical modules include sensors (including electromechanical), power supplies, and signal processors; and mechanical modules include filters, strainers, and flow (element) assemblies/ orifices.

(2) Purchase order dates (month/year) are given for equipment as a basis for determining certain applicable codes on Tables 3.2-2, 3.2-3, and 3.2-4. Where two dates are given and indicated with a slash between them (e.g., 9-70/5-71) the first date corresponds to Unit 1 and the second date corresponds to Unit 2. Where two dates are given with a comma between (e.g., 9-70, 5-71), multiple purchase orders apply.

(3) PC = within primary containment

RB = within reactor building

O = outdoors onsite

L = lake screen house

A = auxiliary building

T = turbine building

RW = radwaste building

F = off-gas filter building

-- = all buildings except O, L

(4) a. Quality group classification per Tables 3.2-2, 3.2-3, and 3.2-4. Group "E" components are special engineered components in accordance with the codes and standards specified in the notes and comments for this Table.

b. I - The equipment meets the quality assurance requirements of 10 CFR 50, Appendix B.

II - The equipment is not required to meet the quality assurance requirements of 10 CFR 50, Appendix B.

c. 1E – Electrical equipment that meets the quality assurance standards of NRC guidelines and IEEE Standard 323-1971. Non-1E Electrical equipment that is not required to meet 1E requirements. NA - not applicable because the equipment is not electrical.

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(5) I - The equipment is designed in accordance with the seismic requirements for the SSE.

II - The seismic requirements for the SSE are not applicable to the equipment.

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

(7) The main steam lines between the outermost containment isolation valve up to the turbine stop valve, the main turbine bypass lines up to the turbine bypass valve, and all branch lines (2-1/2 inch nominal size and larger) connected to these portions of the main steam and turbine bypass lines up to the first valve capable of timely actuation are classified as D+. These sections of pipes meet all of the pressure integrity requirements of code practice for steam power plants plus the following additional requirements:

a. All longitudinal and circumferential butt weld joints are radiographed (or ultrasonically tested to equivalent standards). Where size or configuration does not permit effective volumetric examination, magnetic particle or liquid penetrant examination is substituted. Examination procedures and acceptance standards are at least equivalent to those specified as supplementary types of examination, in ANSI B31.1 Code.

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 as supplementary types of examinations, in ANSI B31.1 Code.

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.

(8) The first valve capable of timely actuation in branch lines connected to the main steamlines between the outermost containment isolation valve and turbine stop valve and the first valve in branch lines connected to turbine bypass valve meets all the pressure integrity requirements of code practice for steam power plants plus the following additional requirements:

a. Pressure-retaining components of all cast parts of valves of a size and configuration for which volumetric examination methods are effective are radiographed. Ultrasonic examination to equivalent standards may be used as an alternate to radiographic methods. If size or configuration do not permit effective volumetric examination, magnetic particle or liquid penetrant methods may be substituted. Examination procedures and acceptance standards are at least equivalent to those specified as supplementary types of examination. Paragraph 136.4.3 in ANSI B31.1 Code.

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

(9) The outermost valve of the three isolation valves in the feedwater lines and the isolation valve in the branch line consisting of the reactor water cleanup return is a motor-operated valve of high leaktight integrity. The check valves inside containment in the feedwater line are the swing type. The check valves outside containment in the feedwater line are the non-slam type.

The classification of the feedwater lines and the RWCU return line from the RCPV valve to the isolation valves is Quality Group B.

(10) a. Lines equivalent to a 3/4-inch or smaller liquid line which are part of the reactor coolant pressure boundary are Quality Group B, ASME III, Class 2, and Seismic Category I.

b. All instrument lines, which are connected to the reactor coolant pressure boundary are Quality Group B, ASME III, Class 2 from the outer isolation valve or the process shutoff valve (root valve) to the instrument rack shutoff valve or welded coupling for instrument racks supplied by General Electric (Figure 3.2-2).

c. All other instrument and sample lines:

(1) Instrument and sample lines up to and through the root valve are of the same classification as the system to which they are attached.

(2) Instrument and sample lines beyond the root valve, if used to actuate a safety system, are of the same classification as the system to which they are attached up to and through the instrument isolation valve or welded coupling for instrument racks supplied by General Electric.

(3) Instrument and sample lines beyond the root valve, if not used to actuate a safety system, are Quality Group D and B31.1.0.

(4) Instrument and sample lines beyond the instrument rack isolation valve or welded coupling for G.E. supplied racks are Quality Group D. Safety system sensing and sample lines are Seismically supported on the instrument rack.

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d. ASME/ANSI Code-Case 78 (included in ASME Boiler and Pressure Vessel Code) is applied to lines 3/4-inch and smaller classified as Quality Group A or B.

(11) The RCIC turbines are categorized as machinery. To ensure that the turbine is fabricated to the standards commensurate with their safety and performance requirements, General Electric has established specific design requirements for this component, which are as follows:

a. All welding is qualified in accordance with Section IX, ASME Boiler and Pressure Vessel Code.

b. All pressure containing castings and fabrications are hydro-tested to 1.5 x design pressure.

c. All high pressure castings are radiographed according to:

ASTM	E-94	
	E-142	20% coverage, minimum
	E-71, 186, or 280	severity level 3

d. As-cast surfaces are magnetic particle or liquid penetrant tested according to ASME, Section III. Paragraph NB-2575 or NB-2576.

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

f. Butt-welds are radiographed according to ASME Section III, NB-2573, and magnetic particle or liquid penetrant tested according to ASME Section III, NB-2575, or NB-2576.

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(SHEET 27 OF 31)

g. Records of major repairs received and maintained.

h. Record system and traceability according to ASME Section III, Code, Paragraphs NA-4442.1 and NB-2151.

i. Control and identification according to ASME Section III, Code, Paragraphs NA-4442.1 and NB-2151.

j. Procedures conform to ASME Section III, NB-5520.

k. Inspection personnel are qualified according to ASME Section III, IX-400.

(12) Cycled condensate storage tanks are Quality Group D. The cycled condensate storage tanks are designed, fabricated, and tested to meet the intent of ANSI B96.1. In addition, the non-destructive examination (NDE) requirement for the tank requires 1) 100% surface examination of nozzle welds, and 2) volume examination of the shell weld joints in accordance with ANSI B96.1.

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

Although the hydraulic control unit, as a unit, is field installed and connected to process piping, many of its internal parts differ markedly from process piping components because of the more complex functions they must provide.

Thus, although the codes and standards invoked by Groups 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). The HCU shutoff (isolation) valves are Quality Group B.

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The design and construction specifications for the HCU do invoke such codes and standards as can be reasonably applied to individual parts in developing required quality levels, but these codes and standards are supplemented with additional requirements for these parts and for the remaining parts and details. For example, 1) all welds are penetrant tested (PT), 2) all socket welds are inspected for gaps between pipe and socket bottom, 3) all welding is performed by qualified welders, and 4) all work is done per written procedures. Quality Group D is generally applicable because the codes and standards invoked by that group contain clauses which permit the use of manufacturer's standards and proven design techniques which are not explicitly defined within the codes of Quality Group A, B, or C, This is supplemented by the QC techniques described above.

(14) Reactor Water Cleanup

A high leaktight integrity isolation valve is provided in the reactor water cleanup discharge line connecting to the feedwater header outside of the containment. This valve is remote manually operated from the control room.

(15) No principal industrial code is applicable.

(16) Pneumatic systems associated with actuation of safety/related valves to accomplish safety functions (e.g., main steam isolation valves, main steam safety/relief valves) are classified Quality Group C. This classification is intended to apply to components such as the air piping, fittings, and accumulators (refer to Figure 3.2-1). This classification does not apply to components of the system such as air control valves, air check valves, and cylinder (or diaphragm)

air actuators. These components are classified as "special equipment" and are selected based on engineering reviews, operating experience and testing as being the most suitable for the application. Such equipment is required to be qualified to demonstrate operability during normal and emergency ambient conditions. Components normally furnished with the process valve (e.g., air control valves, air actuators) are performance tested with the valve as part of its acceptance test procedure. Group C classification has not been applied to these components due to the nonavailability of the equipment with "N" symbol stamp and due to the inappropriate restrictions (e.g., materials, minimum allowable wall thickness) imposed by the code on the equipment in the relatively low-pressure, low temperature air service. The special equipment designation for the previously described components is based on considerations consistent with those of Comment 15.

(17) RCIC turbine steam exhaust line is Quality Group B except that hydrostatic testing of this portion of the line is not required.

(18) The following piping in the Fuel Pool Cooling & Cleanup (FC) is classified as Seismic Category I:

TABLE 3.2-1
(SHEET 28a OF 31)

Piping which provides a flow path from the fuel pool skimmer surge tanks to the RHR system and back to the fuel pool up to and including the isolation valves, which provide the pressure boundary for this mode of operation.

Lines 1(2)FC11C & 1(2)FC14B associated with primary containment penetrations M-65 and M-59.

The drain piping from the reactor well to primary containment penetration M-65.

Line 2FCD1A (Unit 2 only) associated with penetration M-65 is Quality Group B. In addition, lines 1(2)FC11C & 1(2)FC14B associated with primary containment penetrations M-65 and M-59 are Quality Group B.

TABLE 3.2-1
(SHEET 29 OF 31)

(19) Special (engineered design-quality) requirements (motors, pumps, tanks, and equipment).

The engineering QC requirements for the specified equipment has been procured/designed to the horizontal and vertical seismic values. This equipment is capable of withstanding inertial forces equal to the weight multiplied by the seismic coefficient as applied to each member and to the system as a whole.

The QA/QC requirements are as required by either:

a. ASME B&PVC Section III, Appendix XI or equivalent; equipment ordered prior to January 1, 1970 apply QC plan "in effect"' based on purchase order requirements.

b. A QA plan/program at least equivalent to that required by QA.

(20) The unprocessed radwaste piping will meet Group D requirements and the following supplementary requirements:

a. Piping

For sizes over 4 inches nominal, random radiography of 20% of the joints was performed on girth and longitudinal butt-welds. Sockets and fillet welds in sizes over 4-inch nominal will be given random magnetic particle and liquid penetrant examination on 20% of the joints.

b. Pumps and valves

Welds in pumps and valves of pipe size over 4-inch was given random magnetic particle or liquid penetrant examination. Random examination is defined as examination of the linear dimension of a weld in a pump or valve with piping connecting over 10-inch nominal size or as examination of all of the welding in 20% of the pump and valves with piping connecting 10-inch nominal or less.

TABLE 3.2-1
(SHEET 30 OF 31)

(21) Quality group classification requirements do not apply to piping and components supplied by the diesel engine manufacturer as an integral part of the diesel-generator unit. In this case, the manufacturer's standards are used with the intent that the piping or component is to function as reliably as possible.

(22) Civil structures were used in missile analyses as barriers. No individual missile barriers other than civil structures were credited.

(23) Includes Scram Discharge Volume Accumulators.

(24) Expendables and Consumables are purchased per original specification and stored under controlled conditions.

(25) Includes raceway installations containing Class 1E cables and other raceway installations required to meet Seismic Category I requirements (those whose failure during a seismic event may result in damage to any Class 1E or other safety-related system or component.

(26) Subsystems required for post-LOCA monitoring include containment hydrogen monitoring, containment pressure monitoring, containment temperature monitoring, suppression pool water level monitoring, suppression pool water temperature monitoring, and containment high-range radiation monitoring. Subsystems not required for post-LOCA monitoring include containment humidity monitoring, containment particulate monitoring (leak detection), containment continuous particulate, noble gas, and iodine monitoring.

(27) Concrete portions of the building, including the portions that contain Category I systems, are designed to withstand Seismic Category I loads and future modifications to these concrete portions of the building, if ever required, shall also be designed to withstand Category I loads. Appropriate QA procedures will be followed during future modifications to assure the original integrity of the Category I LSCS structures within this building.

(28) The system or component is considered regulatory related and will come under the control of the Quality Assurance Program for future modifications and repairs.

TABLE 3.2-1
(SHEET 31 OF 31)

(29) Quality group classification requirements do not apply to piping and components supplied by the electrical penetration manufacturer as an integral part of the electrical penetration assembly. In this case, manufacturers' standards are used with the intent that the piping or the component is to function as reliably as possible.

(30) The Flow Element/Transmitter and the Mating Flange meet the requirements of Reg. Guide 1.143, Revision 1.

(31) The classification for pipe line 0RH01AA-S4" and valve 0E12-F300 shall be Seismic Category I, Quality Group Classification D, Quality Assurance Requirement I, and Electrical Classification N/A.

(32) The 3/4" Process Sample Line is Quality Group B.

(33) Cooling lake embankment is designed to withstand the effects of the OBE.

(34) The shear walls for the Reactor Building, Auxiliary Building, Turbine Building, Radwaste Building, Diesel Generator Buildings, and Off-gas Filter Building are all interconnected. All of these shear walls have been considered to act together to resist lateral loads applied to these buildings. Therefore, the shear walls for these buildings are seismic Category I.

(35) Primary (Shell) Side of the Residual Heat Removal (RHR) Heat Exchangers were originally designed & fabricated to the requirements of the ASME Boiler & Pressure Vessel Code Section III, 1968 Edition with Addenda through Summer 1970 for Class ‘A’ Components. These heat exchangers were later reclassified to ASME Section III, Class ‘C’ via GE Purchase Order 205-AM358 (Rev. 1) dated 10-07-80 & FDI #127-57434, and Manufacturers Nameplate restamped. (Reference SEAG 99-000861)

(36) Secondary (Tube) Side of the RHR Heat Exchangers were originally designed & fabricated to the requirements of the ASME Boiler & Pressure Vessel Code section III, 1968 Edition with Addenda through Summer 1970 for Class ‘C’ Components. This section of the ASME Code applied the requirements of Section VIII Division 1 to the heat exchanger except for its over-pressure protection devices. (Ref. SEAG 99-000861)

(37) Cooling lake embankment is designed to withstand the effects of the OBE.

(38) Replacement Low Pressure Feedwater Heaters 1CB03AA, 1CB03AB and 1CB03AC were designed and fabricated to the requirements of ASME Boiler & Pressure Vessel Code Section VIII DIV.1, 2004 with Addenda through A2006.

TABLE 3.2-2

CODE REQUIREMENTS FOR COMPONENTS AND SYSTEMS
ORDERED PRIOR TO JULY 1, 1971

	QUALITY GROUP CLASSIFICATION			
	A**	B	C	D
Pressure Vessels +	ASME Boiler and Pressure Vessel Code, Section III, Class A - 1968 Addenda through Summer 1970.	ASME Boiler and Pressure Vessel Code, Section III, Class C - 1968 Addenda through Summer 1970.	ASME Boiler and Pressure Vessel Codes, Section VIII, Div. 1 - 1968 Addenda through Summer 1970.	ASME Boiler and Pressure Vessel Codes, Section VIII, Div. 1 - 1968 Addenda through Summer 1970.
Piping**	ANSI B31.7 Nuclear Power Piping, Class I - 1969.	ANSI B31.7 Nuclear Power Piping, Class II - 1969.	ANSI B31.7 Nuclear Power Piping, Class III - 1969.	ANSI B31.1.0 Code for Pressure Piping - 1967. Addendum - 1969.
Pumps and Valves**	ASME Code for Pumps and Valves for Nuclear Power, Class I - 1968 Draft Addenda March 1970.	ASME Codes for Pumps and Valves for Nuclear Power, Class II - 1968 Draft Addenda March 1970.	ASME Code for Pumps and Valves for Nuclear Power, Class III - 1968 Draft Addenda March 1970.	ANSI B31.1.0 Code for Pressure Piping* - 1967.
Low - Pressure Tanks	---	---	American Petroleum Institute, Recommended Rules for Design and Construction of Large Welded Low - Pressure Storage Tanks, API 620 1963 edition.	American Petroleum Institute, Recommended Rules for Design and Construction of Large Welded Low - Pressure Storage Tanks, API 620 1963 edition.
Atmospheric Storage Tanks	---	American Waterworks Association, Standard for Steel Tanks, Standpipes, Reservoirs and Elevated Tanks for Water Storage, AWWA-D100 1967 edition; or Welded Steel Tanks for Oil Storage, API-650 1964 edition. ++	American Waterworks Association, Standard for Steel Tanks, Standpipes, Reservoirs and Elevated Tanks for Water Storage, AWWA-D100 1967 edition; or Welded Steel Tanks for Oil Storage, API-650 1964 edition.	American Waterworks Association, Standard for Steel Tanks, Standpipes, Reservoirs and Elevated Tanks for Water Storage, AWWA-D100 1967 edition; or Welded Steel Tanks for Oil Storage, API-650 1964 edition.
Heat Exchangers	ASME Boiler and Pressure Vessel Code, Section III, Class A - 1968 Addenda through Summer 1970.	ASME Boiler and Pressure Vessel Code, Section III, Class C,1968 Addenda through Summer 1970, and Tubular Exchanger Manufacturers Association (TEMA) Class C.	ASME Boiler and Pressure Vessel Code, Section VIII, Div. 1, 1968 Addenda through Summer 1970, and Tubular Exchanger Manufacturers Association (TEMA) Class C.	ASME Boiler and Pressure Vessel Code, Section Div. 1, 1968 Addenda through Summer 1970, and Tubular Exchanger Manufacturers Association (TEMA) Class C.

* Pumps operating above 150 psi and 212° F ASME Section VIII, Division 1 of the Boiler and Pressure Vessel Code shall be used as a guide for calculating the thickness of pressure retaining parts and in sizing cover bolting; below 150 psi and 212° F manufacturer's standards for service intended will be used.

** Group A nuclear piping, pumps and valves will meet the provisions of ASME Boiler and Pressure Vessel Code, Section III, Summer Addenda 1969, Paragraph N-153.

+ RPV and Containment Vessel excluded, refer to Section 5.3 for RPV code application and to Section 6.2 for containment.

++ Supplementary NDE - 100% volumetric examination of the side wall for plates over 3/16-inch thick and 100% surface examination of welds for plates 3/16-inch thick or less. Also, 100% surface examination for side-to-bottom welds.

TABLE 3.2-3

CODE REQUIREMENTS FOR COMPONENTS AND SYSTEMS

ORDERED AFTER JULY 1, 1971

	QUALITY GROUP CLASSIFICATION			
	A	B	C	D
Pressure Vessels*	ASME Boiler and Pressure Vessel Code, Section III - 1971, Class 1.	ASME Boiler and Pressure Vessel Code, Section III - 1971, Class 2.	ASME Boiler and Pressure Vessel Code, Section III - 1971, Class 3.	ASME Boiler and Pressure Vessel Code, Section VIII, Div. 1 - 1968 Addenda through Winter 1970.
Piping	ASME Boiler and Pressure Vessel Code, Section III - 1971, Class 1.	ASME Boiler and Pressure Vessel Code, Section III - 1971, Class 2.	ASME Boiler and Pressure Vessel Code, Section III - 1971, Class 3.	ANSI B31.1.0 - 1967, Code for Pressure Piping. Addendum B31.1.0a - 1969.
Pumps and Valves	ASME Boiler and Pressure Vessel Code, Section III - 1971, Class 1.	ASME Boiler and Pressure Vessel Code, Section III - 1971, Class 2.	ASME Boiler and Pressure Vessel Code, Section III - 1971, Class 3.	ANSI B31.1.0 - 1967. Code for Pressure Piping. Addendum B31.1.0a - 1969.**
Low-Pressure Tanks	-	-	American Petroleum Institute, Recommended Rules for Design and Construction of Large Welded Low-Pressure Storage Tanks, API 620 1963 edition.	American Petroleum Institute, Recommended Rules for Design and Construction of Large Welded Low-Pressure Storage Tanks, API 620 1963 edition.
Atmospheric Storage Tanks	-	American Waterworks Association, Standard for Steel Tanks, Standpipes, Reservoirs and Elevated Tanks for Water Storage, AWWA-D100 1967 edition; or Welded Steel Tanks for Oil Storage, API-650 1964 edition. +	American Waterworks Association, Standard for Steel Tanks, Standpipes, Reservoirs and Elevated Tanks for Water Storage, AWWA-D100 1967 edition; or Welded Steel Tanks for Oil Storage, API-650 1964 edition.	American Waterworks Association, Standard for Steel Tanks, Standpipes, Reservoirs and Elevated Tanks for Water Storage, AWWA-D100 1967 edition; or Welded Steel Tanks for Oil Storage, API-650 1964 edition.
Heat Exchangers	ASME Boiler and Pressure Vessel Code, Section III - 1971, Class 1.	ASME Boiler and Pressure Vessel Code, Section III, - 1971, Class 2.	ASME Boiler and Pressure Vessel Code, Section III, - 1971, Class 3.	ASME Boiler and Pressure Vessel Code, Section VIII, Div. 1, 1968. Addenda through Winter 1970, and Tubular Exchanger Manufacturers Association (TEMA) Class C.++

* RPV and Containment Vessel excluded. Refer to Section 5.3 for RPV code application and to Section 6.2 for containment.

** For pumps operating above 150 psi and 212° F ASME Section VIII, Division 1, shall be used as a guide for calculating thickness of pressure retaining parts and in sizing cover bolting; below 150 psi and 212° F manufacturer's standards for service intended will be used.

\+ Supplementary NDE - 100% volumetric examination of the side wall for plates over 3/16-inch thick and 100% surface examination of welds for plates 3/16-inch thick or less. Also, 100% surface examination for side-to-bottom welds.

++ Temporary repairs have been made to the steam inlet nozzles of the 1CB02AA/AB/AC and 2CB02AA/AB/AC low pressure feedwater heaters. The repair configuration deviates from the acceptable types of welded nozzle configurations identified in the ASME Boiler and Pressure Vessel Code Section VIII. The temporary repairs were performed during L1R10 (1CB02AA/AB/AC), L2R10 (2CB02AC), and L2R11 (2CB02AA/AB/AC) and will remain in place no longer than two refuel cycles (L1R12 for Unit 1, L2R12 for Unit 2 repairs performed during L2R10, and L2R12 for Unit 2 repairs performed during L2R11).

TABLE 3.2-4

CODE REQUIREMENTS FOR COMPONENTS AND SYSTEMS ORDERED AFTER JULY 1, 1974

	QUALITY GROUP CLASSIFICATION			
	A	B	C	D
Pressure Vessels*	ASME Boiler and Pressure Vessel Code, Section III - 1974, Class 1.	ASME Boiler and Pressure Vessel Code, Section III - 1974, Class 2.	ASME Boiler and Pressure Vessel Code, Section III - 1974, Class 3.	ASME Boiler and Pressure Vessel Code, Section VIII, Div. 1 - 1974.
Piping	ASME Boiler and Pressure Vessel Code, Section III - 1974, Class 1.	ASME Boiler and Pressure Vessel Code, Section III - 1974, Class 2.	ASME Boiler and Pressure Vessel Code, Section III - 1974, Class 3.	ANSI B31.1 - 1973, Code for Pressure Piping.
Pumps and Valves	ASME Boiler and Pressure Vessel Code, Section III - 1974, Class 1.	ASME Boiler and Pressure Vessel Code, Section III - 1974, Class 2.	ASME Boiler and Pressure Vessel Code, Section III - 1974, Class 3.	ANSI B31.1 - 1973, Code for Pressure Piping.**
Low-Pressure Tanks	-	-	American Petroleum Institute, Recommended Rules for Design and Construction of Large Welded Low-Pressure Storage Tanks, API 620 1963 edition.	American Petroleum Institute, Recommended Rules for Design and Construction of Large Welded Low-Pressure Storage Tanks, API 620 1963 edition.
Atmospheric Storage Tanks	-	American Waterworks Association, Standard for Steel Tanks, Standpipes, Reservoirs and Elevated Tanks for Water Storage, AWWA-D100 1967 edition; or Welded Steel Tanks for Oil Storage, API-650 1964 edition. +	American Waterworks Association, Standard for Steel Tanks, Standpipes, Reservoirs and Elevated Tanks for Water Storage, AWWA-D100 1967 edition; or Welded Steel Tanks for Oil Storage, API-650 1964 edition.	American Waterworks Association, Standard for Steel Tanks, Standpipes, Reservoirs and Elevated Tanks for Water Storage, AWWA-D100 1967 edition; or Welded Steel Tanks for Oil Storage, API-650 1964 edition.
Heat Exchangers	ASME Boiler and Pressure Vessel Code, Section III, - 1974, Class 1.	ASME Boiler and Pressure Vessel Code, Section III, - 1974, Class 2.	ASME Boiler and Pressure Vessel Code, Section III, - 1974, Class 3.	ASME Boiler and Pressure Vessel Code, Section VIII, Div. 1, 1974, and Tubular Exchanger Manufacturers Association (TEMA) Class C.

* RPV and Containment Vessel excluded. Refer to Section 5.3 for RPV code application and to Section 6.2 for containment.

** For pumps operating above 150 psi and 212° F ASME Section VIII, Division 1, shall be used as a guide for calculating thickness of pressure retaining parts and in sizing cover bolting; below 150 psi and 212° F manufacturer's standards for service intended will be used.

+ Supplementary NDE - 100% volumetric examination of the side wall for plates over 3/16-inch thick and 100% surface examination of welds for plates 3/16-inch thick or less. Also, 100% surface examination for side-to-bottom welds.

3.3 WIND AND TORNADO LOADINGS

3.3.1 Wind Loadings

3.3.1.1 Design Wind Velocity

A design wind velocity of 90 mph based upon a 100-year recurrence interval is used in the design of Seismic Category I structures at the LaSalle County Station (LSCS).

3.3.1.1.1 Basis for Wind Velocity Selection

The wind velocity and recurrence interval specified in Subsection 3.3.1.1 are obtained from wind distribution charts and probability studies (Reference 1) as they pertain to the LSCS site. Figure 3.3-1 is a reproduction of the pertinent wind distribution chart.

3.3.1.1.2 Vertical Velocity Distribution and Gust Factors

The variation of wind velocity with height follows the $1/7^{th}$ power equation, as stated in References 2 and 3, and is referenced herein as V_z.

The gust factor applied to the velocity, V_z, varies linearly from 1.1 at grade (Reference 3) to 1.0 at 400 feet above grade elevation.

3.3.1.2 Determination of Applied Forces

The design wind velocity, V, which is the product of V_z and the gust factor specified in Subsection 3.3.1.1.2 is translated into an equivalent dynamic pressure according to the provisions outlined in ASCE Paper 3269 (Reference 3). Table 3.3-1 gives the calculated values.

The dynamic wind pressures are converted to an equivalent static force, P, by considering appropriate drag coefficients.

For flat-topped buildings, the drag coefficients considered are 0.9 for windward pressure, 0.5 for leeward suction, and 0.8 for side walls and roof suction (Reference 3).

Table 3.3-2 gives the design values of the force P as a function of height above grade.

3.3.2 Tornado Loadings

3.3.2.1 Applicable Design Parameters

The following are the tornado design parameters (Reference 4):

a. a maximum rotational velocity of 300 mph,

b. a translational velocity of 60 mph,

c. an external pressure drop of 3 psi at the vortex within a 3-second interval, and

d. a radius of maximum wind speed of 227 feet.

The characteristics and spectrum of tornado-generated missiles are found in Subsection 3.5.1.4.

3.3.2.2 Determination of Forces on Structures

3.3.2.2.1 Transformation of Tornado Winds into Effective Pressures

All tornado-wind pressure and differential pressure effects are considered static in application, since the natural period of the building structure and its exposed elements is short compared with the rise in time of applied design pressure.

The tornado model incorporates the design parameters of Subsection 3.3.2.1. The variation of the differential pressure and tangential plus translational velocity as a function of the distance from the center of the tornado, is shown graphically in Figure 3.3-2 as obtained from Reference 5.

The tornado velocity is converted into an equivalent static pressure according to the procedures outlined in ASCE Paper 3269 (Reference 3) considering no reduction in velocity with height and a gust factor of 1.0. The drag coefficients outlined in Subsection 3.3.1.2 are used to determine tornado wind loading.

The dynamic pressure due to wind velocity alone is shown in Figure 3.3-3. Figure 3.3-4 shows the resultant static surface pressure when the pressure drop components and dynamic wind components are combined for rectangular flat-topped structures.

3.3.2.2.2 Venting of the Structure

Venting of concrete structures is not relied upon to reduce differential pressure loadings. However, all siding and roof decking in the reactor building superstructure is designed and detailed to blow off when the tornado approaches the station, and the bare frame is designed to resist tornado forces.

3.3.2.2.3 Tornado-Generated Missiles

The characteristics and spectrum of tornado-generated missiles are found in Subsection 3.5.1.4.

The procedures used for designing for the impactive dynamic effects of a point load resulting from tornado-generated missiles are found in Subsection 3.5.3.

3.3.2.2.4 Tornado Loading Combinations

Refer to Tables 3.8-3, 3.8-6, and 3.8-8 through 3.8-11 for the load factors and load combinations associated with tornado loading. In designing for the postulated tornado, the structure under consideration is placed in various locations of the pressure field to determine the maximum critical effects of shear, overturning moment, and torsional moment on the structure.

The effective tornado load at each point under consideration, on Seismic Category I structures above grade, along the radius is found based on combining the components of the tornado load in the following manner:

$$W_t = W_w + W_p \qquad \text{(Figure 3.3-4)}$$

where:

W_t = total tornado load
W_w = tornado wind load
W_p = tornado differential pressure load.

The tornado missile is also considered individually and in combination with the tornado wind load. Since the pressure drop reduces the resultant wind load effects, it is conservatively omitted from the missile load combination. Thus the following equations are considered:

$$W_t = W_m$$
$$W_t = W_w + W_m$$

where:

W_t and W_w as previously defined
W_m = tornado missile load.

The blowoff loads for siding and decking in the reactor building superstructure are considered as follows:

$W_t = W_a$

where:

W_a = effects of approaching tornado immediately prior to the blow off of siding and decking.

This pressure field is indicated in Figure 3.3-4. In addition, it is conservatively considered that the tornado missile can impact the structure at any location in the pressure field, windward, leeward or at sides and the roof with the full design missile velocity.

3.3.2.3 Ability of Seismic Category I Structures to Perform Despite Failure of Structures Not Designed for Tornado Loads

The structures whose failure could affect Seismic Category I structures are the turbine building superstructure and the Lake Screen House. A detailed analysis is made for the turbine building superstructure for the tornado loads specified in Subsection 3.3.2.2.

The turbine building superstructure is designed to withstand the tornado loads on the exposed structural frame so that collapse is prevented. The turbine room siding and roof decking is designed to blow off in an approaching tornado, to ensure venting of the structure.

A detailed discussion of the structural integrity of the non-Seismic Category I Lake Screen House and its effect on the Seismic Category I service water intake structure is presented in Subsection 3.8.4.1.7.2. The Lake Screen House is designed to withstand the effects of a tornado to the extent that it will not collapse in such a way as to block the service water intake structure.

All other non-Seismic Category I structures are separated from Seismic Category I structures by a distance greater than their height. Thus, the integrity of all Seismic Category I structures is ensured.

3.3.3 References

1. H. C. S. Thom, "New Distributions of the Extreme Winds in the United States," Journal of the Structural Division, ASCE, 94 (ST7): pp. 1787-1801, July 1968.

2. Ibid.: pp. 1797-1800.

3. "Task Committee on Wind Forces, Committee on Loads and Stresses, Wind Forces on Structures, Final Report," Paper No. 3269, Transactions, ASCE, Vol. 26: pp. 1142-1167, 1961.

4. J. A. Dunlap and K. Wiedner, "Nuclear Power Plant Tornado Design Considerations," Journal of the Power Division, ASCE, 94 (PO2): 409, March 1971.

5. John D. Stevenson, "Engineering and Marketing Guide to Tornado, Missile, Jet Thrust and Pipe Whip Effects on Equipment and Structures," Report prepared for: Nuclear Structural Associates, 176 Thornberry Drive, Pittsburgh, Pa. 15235, Appendix B: 1-3.

TABLE 3.3-1

DYNAMIC WIND PRESSURE, q, FOR SEISMIC CATEGORY I STRUCTURES

$$q = .002558V^2$$

HEIGHT ABOVE GRADE (ft)	DYNAMIC WIND PRESSURE q (1b/ft^2)
0 - 50	26.4
50 - 100	30.7
100 – 150	35.7
150 – 200	37.8

TABLE 3.3-2

EQUIVALENT STATIC FORCE, P, FOR SEISMIC CATEGORY I STRUCTURES

HEIGHT ABOVE GRADE (ft)	WINDWARD PRESSURE ($1b/ft^2$)	LEEWARD SUCTION ($1b/ft^2$)	TOTAL DESIGN FORCE ($1b/ft^2$)	SIDE WALLS AND ROOF SUCTION ($1b/ft^2$)
0 - 50	24.0	13.0	37	21.3
50 – 100	28.0	15.0	43	24.9
100 – 150	32.0	18.0	50	28.4
150 – 200	34.0	19.0	53	30.2

3.4 WATER LEVEL (FLOOD) DESIGN

3.4.1 Flood Protection

3.4.1.1 Flood Sources

Exterior Floods

The LaSalle County external flood control efforts are directed towards three goals. They are:

a. to prevent flood damage from the lake due to the probable maximum flood (PMF) corresponding to a probable maximum precipitation (PMP) with an antecedent standard project storm;

b. to control potential flooding resulting from a local PMP at the plant site; and

c. to prevent plant flooding resulting from the Illinois River's PMF.

The first of these goals relates to a PMF level of 704 feet 4 inches mean sea level (Subsection 2.4.8.2.5) on the lake. The maximum wind wave runup at the plant that could result from this condition is 1 foot 4 inches above the PMF level (Subsection 2.4.8.2.8). This gives the maximum estimated wave runup level of 705 feet 8 inches. As discussed in Subsection 2.4.1.1 the plant grade is over 4 feet higher than this level. Thus the safety-related functions are not affected by a lake PMF.

Due to the local intense PMP, water buildup will not exceed an elevation of 710.41 feet as discussed in Subsection 2.4.2.3. This is lower than the plant floor elevation of 710 feet 6 inches and, therefore, does not provide any threat of flooding.

The Illinois River's PMF level is more than 180 feet below the plant grade and does not affect the safety-related systems and structures (Subsection 2.4.3).

Interior Floods

The LaSalle County internal flood control efforts are directed towards the three sources that could conceivably introduce large amounts of water into the plant below grade areas. The sources result from:

a. a failure of the following water lines connected directly to the lake which would permit the entering of lake water into the plant via gravity:

1. circulating water system,

2. CSCS equipment cooling water system, and

3. service water system.

b. a failure in a water line directly connected to the lake which, if the pump were to remain on, would result in water flowing into the plant. This could occur to the lines with the following pumps:

1. circulating water pumps,

2. diesel-generator cooling water pumps,

3. RHR service water pumps,

4. service water pumps, and

5. fuel pool emergency makeup pumps

c. a failure of the suppression pool.

3.4.1.2 Safety-Related Systems

The list of safety-related systems and components is found in Table 3.2-1. The systems and components located below grade level in Seismic Category I structures which are flood protected are shown in Figures 3.4-1 and 3.4-2. Their location can be found in the general arrangement drawings of Section 1.2.

3.4.1.3 Description of Structures

The structures that house safety-related equipment are the reactor, auxiliary, diesel-generator, and the lake screen house buildings. These structures all have reinforced concrete walls below grade level. The reactor, auxiliary, and diesel-generator buildings, shown in Figures 3.4-1 and 3.4-2 below grade, are all connected with the turbine, solid radwaste, and service buildings and act as a unit in resisting exterior floods. The only exterior personnel or equipment access to these buildings is at grade level or above. All pipes penetrating the exterior walls are provided with watertight penetration sleeves. Also pumps and drains are provided throughout the Seismic Category I structures to provide protection.

3.4.1.4 Flood Protection Measures

Exterior Flood Protection Measures

In addition to the flood protection measures described in Subsection 3.4.1.1, additional protection is provided by means of waterproofing and waterstops. All exterior walls to grade level are sealed with a waterproof membrane, and all exterior construction joints are sealed with waterstops to grade level.

Interior Flood Protection Measures

The interior flood control program consists of the erection of floodwalls and bulkhead doors to keep uncontrollable gravity floodwater sources contained, an alarm and indication system for key sumps, and the formulation of abnormal (flooding) operating procedures for the station operators. The floodwalls and bulkhead doors are depicted in Figures 3.4-1 and 3.4-2.

a. Gravity Flooding

Flood protection against a gravity-fed failure of a lake-connected water line is afforded by a watertight floodwall referred to as the Condenser Pit as shown in Fgures 3.4-1 and 3.4-2. The floodwall extends to elevation 701 feet and surrounds the Condenser water box and the associated Circulating Water and Service Water piping. Flooding with gravity-fed lake water within the condenser pits will be contained within these non-critical areas, preventing flooding of the plant.

Lake-connected lines not within the confines of the Condenser Flood-Protected Zone include:

1) 120" Ice-melting line and MOV

The MOV for each Unit's ice melting line is installed in a compartment that is watertight against groundwater only, and is otherwise open to the Turbine Building floor elevation 663'. The ice melting line within the compartment has been seismically analyzed and shown to meet the crack exclusion criteria of Standard Review Plan (SRP) 3.6.2.

2) 48" Service Water Return Piping Standpipes (1 per unit)

The Service Water System enters the plant at an elevation above maximum lake level and supplies cooling water to heat exchangers located above maximum lake level. Four Service Water return standpipes are below lake level. However, two of the standpipes (one per unit) are located outside of the flood-protected zone. The top portion of each standpipe above elevation 692' has a normal operating pressure of less than 10 psig, and therefore has not be analyzed for postulated pipe cracks, based on Appendix J. The standpipe portions between elevation 692 feet and 663 feet have been seismically analyzed and shown to meet the crack exclusion criteria of SRP 3.6.2.

3) 36" Circulating Water manway (1 per unit)

Two of four Circulating Water manways and associated piping are located outside of the Condenser Pit. The manway piping, with access covers raised above elevation 701 feet, have been seismically analyzed and shown to meet the crack exclusion criteria of SRP 3.6.2.

4) Circulating Water Dewatering Lines

All of the 4", 6", 8", and 12" Dewatering System pipelines are outside of the flood protection zone. The dewatering lines have normally closed suction and discharge isolation valves. The portions of these lines between the circulating waterpipe and the suction/discharge isolation valve have been seismically analyzed and have been shown to meet the crack exclusion criteria of SRP 6.3.2. When the dewatering pumps are in service, the portion of the circulating water piping that is being dewatered is isolated from the source of the lake water. Therefore, flooding due to a crack in the dewatering lines is not a concern.

5) Amertap System piping (2 1/2" x 3")

All of each Unit's Amertap System piping is outside of the Condenser Pit's flood protection zone. However, there are isolation valves inside the condenser pit that can be closed to isolate a leak. The time available to isolate a leak in the Amertap lines is more than 48 hours, which is more than adequate for Operator actions to occur.

The outfalls from the CSCS equipment cooling water system rise above lake level between the reactor building and the lake. Back siphoning from the lake is, therefore, impossible in the event of a piping failure.

Flooding due to the failure of piping associated with the Diesel-Generator Cooling Water pumps, RHR Service Water pumps, or Fuel Pool Emergency Makeup pumps, would be contained within the respective flood zones of Flood Control Area I shown in Figures 3.4-1 and 3.4-2. These areas, except for both Division III CSCS pump rooms, are surrounded by watertight walls to elevation 701', and have zone-dedicated floor drains and sumps. The Div. III CSCS-ECWS rooms are watertight to only elevation 697' because of VY System ventilation ductwork connecting the pump room to the associated Div. III switchgear room. Flooding of the Division III CSCS-ECWS Pump Rooms have been assessed by evaluation of the room-enclosed pipelines against the SRP 3.6.2 crack exclusion criteria, concluding that failures are non-credible.

b. Pump Induced Flooding

Water level alarms are positioned throughout the plant in key sumps to detect pump induced flooding. Subsequent operator action in accordance with abnormal (flooding) procedures is required to shut down the offending pumps mentioned in Subsection 3.4.1.1 and close the necessary valves, thereby isolating the source of floodwater.

c. Flooding Due to a Suppression Pool Rupture

To prevent flooding of the RHR pump areas of the reactor building, watertight walls are erected up to an elevation of 686 feet 7 inches. Also, watertight doors are installed and the sumps are separated. Thus, these areas will be protected if the highly unlikely event of a suppression pool rupture occurs. These flood control areas are depicted in Figures 3.4-1 and 3.4-2.

3.4.2 Analysis Procedures

The portions of the structures below the elevation of 700 feet are analyzed and designed for the hydrostatic head from a flood at the water table elevation superimposed on other Seismic Category I loadings. The walls in the lake screen house that are exposed to the cooling lake are designed for hydrodynamic forces as well. The loading combinations used in the design of the lake screen house are listed in Table 3.8-10.

The hydrodynamic wave forces on the exposed lake screen house walls are estimated using the method outlined in Section 7.32 of the "Shore Protection Manual," Volume II (U.S. Army Coastal Engineering Research Center, Department of the Army, Corps of Engineers, 1973). The two conditions considered for estimating the wind, wave forces are:

a. Maximum (1%) wave due to 40 mph overland wind over the PMF pool elevation of 704.3 feet.

b. Maximum (1%) wave due to 50 mph overland wind over the normal pool elevation of 700.0 feet.

The maximum (1%) wave heights and periods corresponding to the conditions 1 and 2 are 1.6 feet and 1.75 seconds, and 2.17 feet and 1.95 seconds, respectively at the lake screen house retaining walls. The depth of water at these walls varies between 26 feet and 30 feet for the normal pool and the PMF pool, respectively. This depth is much greater than 1.5 times the maximum wave height and hence the Miche-Rundgren method stated in the "Shore Protection Manual" is adopted for calculating the nonbreaking wave forces. The wind, wave forces on the exposed lake screen house walls are shown in Figure 3.4-3.

The hydrodynamic loads as a result of the seismic forces are designed for by using the methods described in Subsection 2.5.4.10.1.3.2.2.

3.5 MISSILE PROTECTION

Where possible, all Seismic Category I structures, equipment, or station nuclear safety-related systems are protected from missiles generated by internal rotating or pressurized equipment through basic station component arrangement such that, if equipment failure should occur, the missile does not cause the failure of the Seismic Category I structure of other nuclear safety-related systems. Where it is impossible to provide protection through station layout, suitable physical barriers are provided when required to isolate the missile or to shield the critical system or component. In addition, redundant Seismic Category I components are suitably protected such that one missile cannot simultaneously damage a critical system component and its backup system, or vice versa. Table 3.2-1 provides a tabulation of safety-related structures, systems, and components, along with their applicable Seismic Category, Quality Group Classification, and location.

3.5.1 Missile Selection and Description

3.5.1.1 Internally Generated Missiles (Outside Containment)

Essential equipment in the reactor, auxiliary, and the diesel-generator buildings is protected to the extent practicable from the effects of postulated missiles either by barriers or, in the case of redundant systems and components, by physical separation. Rotating equipment which has the potential for being subjected to an overspeed condition in excess of design limitations is considered as a potential source of missiles and is isolated from other components to the extent practicable by physical separation or barriers.

3.5.1.2 Internally Generated Missiles (Inside Containment)

Missile protection is provided within the containment for the following two general sources of postulated missiles:

a. Rotating component failures.

b. Pressurized component failures.

The principal design bases are that missiles generated within the reactor containment during normal operation, maintenance, testing and postulated loss-of-coolant accident, shall not cause loss of function of any redundant engineered safety feature. Engineered safety features separation and redundancy have been provided as the primary protection against missiles. A tabulation of all safety-related structures, systems and components inside the containment, their location, seismic category and quality group classification is given in Table 3.2-1.

3.5.1.2.1 Rotating Component Failure Missiles - Selection and Evaluation

The most significant pieces of rotating equipment in the primary containment are the recirculation pump and motor which, in the event of a major recirculation line break on the pump suction side, can theoretically reach overspeed beyond practical design limitations (Subsection 5.4.1.4). Since the pump to motor shaft will shear in the postulated event, only the recirculation pump impeller missiles need be considered.

Studies indicate that missiles from impeller fragments will not penetrate the pump case; however, impeller missiles may be ejected from the open end of the broken suction pipe (Subsection 3.5.1.7). If all missiles are permitted to be ejected from the open end of the broken pipe, only the largest of such missiles at a few specific locations have the potential to cause significant damage within the primary containment. The probability for significant damage is demonstrated to be acceptably low, and potentially damaging impeller missile ejection from the broken pipe is minimized by effective placement of pipe whip restraints to the extent necessary that the offset of the two ends of the broken pipe is controlled. Thus, the blowdown forces from both ends of the broken pipe oppose each other, and the missile dissipates the energy acquired during its passage through the broken pipe. Upon leaving the break, the missile has insufficient energy to cause damage.

The above discussion demonstrates that the probability of significant damage from recirculation pump or motor missiles is so low that no protection in addition to the pipe restraints referred to is recommended.

3.5.1.2.2 Pressurized Component Failure Missiles - Selection and Evaluation

Pressurized components within the primary containment capable of producing missiles have been reviewed. Although piping failures can result in significant dynamic effects if permitted to whip, they do not form missiles per se since the whipping section remains attached to the remainder of the pipe. Since Section 3.6 addresses the dynamic effects associated with pipe breaks, pipes are not included here as potential internal missiles.

Since pressurized gas containers are considered as a potential source of missiles, they are located such that no credible single failure causing release of energy in the building environment will generate missiles to impair the functioning of the redundant safety-related equipment. Details of the design of pressurized gas containers are given in Subsection 9.3.1.

The only remaining pressurized components potentially capable of producing missiles are considered to be:

a. Valve bonnets (large and small)

b. Valve stems

c. Thermowells

d. Retaining bolts

e. CRD mechanisms

f. Flange connections to the RPV.

The above components are designed to strict nuclear standards and it is expected that the failure probabilities are less than that of the piping. Pipe failure rates have been estimated (Reference 5) to be approximately 1×10^{-5} failures per year. Because the failure rates for the above components are less or probably of the same magnitude or less the above missiles are considered as statistically significant missiles and their strike probabilities and damage potential must be examined.

The parameters required to determine missile penetration for each credible missile listed above are discussed below. Missiles are generally characterized by size, weight, origin, impact area velocity, and impact energy. Missiles may also be classified by the potential energy source which serves as the driving force: stored strain energy, contained fluid energy (jet propelled and piston type missiles). The methods used to determine missile characteristics are presented.

Acceleration of a failed valve bonnet or thermowell is produced by a "jet" of escaping fluid, whereas the acceleration of a valve stem or CRD mechanism is caused by piston type action. Both of these "contained fluid energy" type missiles are briefly described below. Stored strain energy type missiles such as retaining bolts are also included.

Piston Type Missile. The velocity of a piston type missile (e.g., valve stem and CRD mechanism) is calculated by assuming that the work done will be converted into the kinetic energy of the missile, with no losses of energy due to friction, air resistance, etc. Work is the integral of force times displacement, while the kinetic energy of the missile is one-half the product of missile mass times the square of the missile velocity. Assuming the force constant (and equivalent to PA) and equating the kinetic energy to the work done results in a missile velocity given by the expression (Reference 7).

$$V = \left[\frac{2\,P\,A_0\,\ell}{w/g} \right]^{1/2} \qquad (3.5\text{-}1)$$

where

P = Pressure acting on area A_0 (lb/ft^2)

A_0 = Area of missile under pressure (ft^2)

l = Displacement of length of "piston" stroke (ft)

W = Weight of the missile (lb)

V = Velocity of the missile (ft/sec)

g = Acceleration of gravity (ft/sec^2)

Design of the containment and the piping system has considered the possibility of missiles being generated from the failure of pressurized components such as valve bonnets, valve stems, and instrumentation thermowells. Missile protection is accomplished through basic plant arrangement such that the flight of the missile is away from the containment vessel. The arrangement of plant components takes the possibility of missile generation into account, even though such missiles may not have sufficient energy to penetrate the containment. Equipment associated with engineered safety systems is segregated so that failure of one cannot cause the failure of another or that component failure resulting in a need for engineered safeguard systems will not render the redundant system inoperable. The control rod drive mechanisms are located under the reactor vessel and are surrounded by reinforced concrete walls and floor to provide protection from missiles.

Jet-Propelled Missiles. Jet-propelled missiles (valve bonnets and thermowells) are missiles propelled by fluid escaping from a pressurized system in which there is essentially no lateral constraint on the fluid. Thus, the escaping jet of fluid will not only impinge on the missile during the period of missile acceleration, but will also flow around and past the missiles. The velocity of such a missile is estimated by employing the jet property solution of Moody (Reference 8) for saturated steam blowdowns.

The work of Reference 8 was directed toward the prediction of blowdown thrust and jet forces on stationary targets; however, by making a few simplifying assumptions and applying the principle of momentum, this work can be applied to the determination of velocity-displacement relationships for jet propelled missiles. The specific assumptions are: (1) the asymptotic properties of the jet exist over the entire region of travel of the missile; (2) the missile is completely surrounded by the fluid jet during its time of flight. Applying these assumptions and the principles of momentum to the relative velocity of the jet and the missile, the following expression results relating the missile displacement and velocity:

$$\frac{y}{(W\ /\ A)} = V_\infty \left[\ell n \left(\frac{1}{1 - v\ /\ u_\infty} \right) - \frac{V}{u_\infty} \right] \quad (3.5\text{-}3)$$

where

y = distance traveled by the missile from the break (ft),

W = missile weight (lb),

A = frontal area of missile (ft^2),

u_∞ = asymptotic velocity of jet (ft/sec),

V_∞ = asymptotic specific volume of jet (ft^3/lb), and

V = velocity of missile (ft/sec).

The above expression assumes that the water and steam velocities are equal (i.e., unity velocity ratio) in the case of a saturated water blowdown. The jet asymptotic velocity, u , and the jet asymptotic specific volume are determined by the methods described by the previous Moody reference. The corresponding velocity-displacement relationships for missiles resulting from saturated water and saturated steam blowdowns are present in Figure 3.5-6. The ordinate is the missile velocity and the abscissa is the displacement parameter Y:

where

$$Y = \frac{y}{(W\ /\ A)} \quad (3.5\text{-}4)$$

Included in Figure 3.5-6 is the influence of different values of the friction parameter, f*, defined by:

$$f^* = \left(\frac{\bar{f}\ 1}{D} \right)_p \bullet \left(\frac{A_E}{A_p} \right)^2 \quad (3.5\text{-}5)$$

where:

$$\left(\frac{\bar{f}\ 1}{D} \right)_p$$

= equivalent loss coefficient between the broken pressurized component and fluid reservoir (dimensionless),

A_E = area of break (ft^2), and

A_p = area of pressurized component between break and fluid reservoir (ft^2) (assumes $A_p \geq A_E$).

As illustrated in Figure 3.5-6, the effect of friction on the velocity-displacement relationship is reasonably small, and it can be conservatively assumed that the most extreme friction condition persists; $f^* \approx 100$ for the case of saturated water blowdown and $f^* \approx 0$ for the case of saturated steam blowdown.

Stored Strain Energy Missiles. Assuming all the strain energy of a retaining bolt which fails is converted to kinetic energy, the velocity is calculated from the following formulas (Reference 7):

$$V = \left(\frac{gE}{W}\right)^{1/2} \varepsilon \qquad (3.5\text{-}6)$$

and

$$V = \left(\frac{g}{EW}\right)^{1/2} \sigma \qquad (3.5\text{-}7)$$

where

V = missile velocity (ft/sec),

E = modulus of elasticity (lb/ft^2),

W = specific weight of missile (lb/ft^3),

ε = ultimate strain in the bolt before failure (in./in.), and

σ = ultimate stress in the bolt before failure (lb/ft^2).

The failure is assumed in the location that produces the most energetic missile. These equations provide a conservation analysis of missile energy because the ultimate tensile stress (σ or ε) for the material is used, resulting in a larger amount of energy than would actually be present at fracture, and all strain energy is converted to kinetic energy with no consideration for energy losses due to friction, relaxation, or air resistance.

3.5.1.2.3 Valve Missile Protection Inside Containment

The impact of possible missiles from all small and large valves inside the containment, that could strike the containment liner, was evaluated using the following procedures:

a. The missile velocities were calculated using the equations stated in Subsection 3.5.1.2.2.

b. The missile impact on containment liner was evaluated in accordance with ASME Code Section III Division 2, Subsection CC-3900.

It was determined that the penetration potential of missiles from only one valve (1(2)B21F011A-Feedwater Isolation Valve) could exceed the limit permitted by Reference 14. A missile barrier is provided for this valve (1(2)B21F011A-Feedwater Isolation Valve) to protect the containment liner.

All small and large valves inside the containment were also reviewed as to their potential for damage to safety-related equipment and components inside the containment. The principal design basis, as stated in Subsection 3.5.1.2, is that missiles generated within the reactor containment shall not cause loss of function of any redundant engineered safety feature.

Upon review of the safety-related equipment inside containment, it was determined that the following valves would need missile protection:

1(2)E22-F005
1(2)B33-F067A
1(2)B33-F067B

Missile barriers, to prevent these valves from damaging other components, are provided for these valves.

3.5.1.3 Turbine Missiles

With the replacement of the Low Pressure (LP) rotors, all the turbine rotors are of the monoblock design. The monoblock rotors have very low stress level. Missile generation due to turbine failure is generally postulated to be caused by turbine overspeed. General Electric has established that the speed capability of these rotors is considerably higher than the maximum attainable speed of these turbine-generator units. Consequently, the probability of missiles being generated is statistically insignificant. (References 15-19).

3.5.1.4 Missiles Generated by Natural Phenomena

Tornado-generated missiles used in the design of Seismic Category I structures are as follows:

MISSILE	PHYSICAL PROPERTIES	IMPACT VELOCITY (mph)
Wood Plank	4 in. x 12 in. x 12 ft	225
Automobile Wt. 4000 lbs	20 ft^2 front area	50

The maximum height reached by the automobile is 25 feet above the grade elevation. Wood plank is postulated to reach the height of the structure.

3.5.1.5 Missiles Generated by Events Near the Site

As described in Subsection 2.2.3, only accidents from empty gasoline barges which have correct air-mix ratios for explosions, the 2-2000 gallon above grade fuel tanks onsite, and highway trucks carrying explosives may lead to credible explosions. The energy of the explosions and their distance from the plant site are the determining factors for site proximity missiles. These energies and distances of the credible explosions at the plant site are discussed in Subsection 2.2.3 and are insignificant for the LSCS site. In addition, the Seismic Category I structures are designed to withstand the tornado-generated missiles. No adverse effects due to missiles generated from explosions will occur to the plant.

3.5.1.6 Aircraft Hazards

The airports and the airways in the region of the station are described in Subsection 2.2.2.5. That information indicates that:

a. There are no federal airways or airport approaches passing within 2 miles of the station. The closest airway corridor is 3 miles away from the station.

b. There are no commercial airports existing within 10 miles of the site and there is only one private airstrip within 5 miles.

c. The projected landing and take-off operations out of those airports located within 10 miles of the site are far less than 500 d^2 per year, where d is the distance in miles. The projected operations per year for airports located outside of 10 miles is less than 1000 d^2 per year.

d. The only military facility within 10 miles of the site is the Illinois Army Reserve National Guard Training Facility. It is located approximately 1 mile northwest of LSCS cooling lake. There are no airstrips at the Training Facility.

Hence, the probability of radiological consequences due to aircraft hazards to the station is within the guidelines of 10 CFR 100.

3.5.1.7 Recirculation Pump Overspeed Analysis

A generic analysis of recirculation pump overspeed (Reference 6) for the complete spectrum of breaks in piping on the discharge side of the recirculation pump shows that no overspeed condition exists. In the unlikely event of a completely offset guillotine suction break, a potential overspeed may be calculated, however, further considerations support the conclusion that this calculated overspeed condition would not realistically create an unsafe condition. As a result, there is no need for protective equipment on the recirculation pumps.

A generic upper boundary probabilistic analysis of the effect of recirculation pump missiles on primary containment equipment in a typical BWR-5 MARK II nuclear power plant has been added as Attachment 3.B to Chapter 3.0, to comply with the intent of Regulatory Guide 1.46. This analysis indicates that no damage results to primary containment, nor to any major piping system inside containment, nor to any inboard main steam isolation valve. Absence of damage is due to the fact that trajectories of postulated missiles do not intersect these systems. The presence of pipe restraints on the recirculation system adds approximately 33% additional protection on a probabilistic basis as indicated in Table I of Attachment 3.B.

If more conservative break location assumptions are postulated (i.e., breaks occur at all fittings and at all equipment piping circumferential welds), the relative probability of impact and perforation, or destructive damage to critical targets is shown in Table I of Attachment 3.B. The probability that a secondary (ricochet) impact on a vital line or a main stream isolation valve, given the expulsion of a missile is PvX0.031 as shown in Table III of Attachment 3.B. For any particular line within containment, the total probability of damaging impact is of the order of 10^{-5} as shown in Table III of Attachment 3.B.

3.5.2 Systems to be Protected

3.5.2.1 Missile Protection Design Philosophy

Systems that are protected from missiles are listed in Section 3.2.

For internally generated missiles, protection is provided through basic station component arrangement such that, if equipment failure should occur, the missile does not cause the failure of the Seismic Category I structure of any other nuclear safety-related system. Where it is impossible to provide protection through station layout, suitable physical barriers whose function is either to isolate the missile or to shield the critical system or component are provided when required. Refer to Figures 3.5-3 through 3.5-5 for the location of missile barriers. In addition, redundant Seismic Category I components are suitably protected such that one missile cannot simultaneously damage a critical component and its backup system.

3.5.2.2 Structures Designed to Withstand Missile Effects

Seismic Category I structures are designed to withstand postulated external or internal missiles which may impact them. The following is a list of the structures designed to withstand missile effects, and the missiles that each structure has been designed for:

STRUCTURE	MISSILE	GENERAL NOTES
Reactor building enclosure (including equipment access building)	tornado-generated missiles from outside the building	see Subsection 3.5.1.4 for list
Auxiliary building enclosure	same as above and pipe whip missiles when restraints are not provided	same as above
diesel-generator building enclosure	tornado-generated missiles	same as above
lake screen house	same as above	same as above
Interior walls designed to withstand missiles	pipe whip missiles also missiles described in Subsections 3.5.1 and 3.5.2	see Subsections 3.5.1 and 3.5.2

The reactor building superstructure's metal siding and roof deck are not designed to withstand tornadoes. However components which directly affect the ultimate safe shutdown are located either under the protection of reinforced concrete or underground.

Seismic Category I structures not tabulated above are not affected by any missiles because of basic station component arrangement and the use of whip restraints.

3.5.3 Barrier Design Procedures

Two types of structural response to missile impact have been investigated:

a. the penetration resistance of a structure and potential for secondary missiles by spalling, and

b. the stability of the panels.

Generally, all projectiles are considered as impacting instantaneously with a very short time rise relative to the natural period of the impacting structure.

Two types of barriers are designed to resist missile impact:

a. Steel Plate Barriers - The thickness of steel plate required to resist the impacting missile is calculated using the Stanford Formula (References 9 and 11). The overall structural response, including structural stability and deformations, is investigated using concepts and methods presented in Reference 12.

b. Reinforced Concrete Barriers - The concrete thickness required to resist the impacting missile is calculated using the modified Petry Formula (References 10 and 13). Concrete barriers are designed such that the missile penetrates no more that two-thirds of the thickness of the barrier; thus spalling is prevented (Reference 10). The deformation and stability of structural panels is investigated using methods presented in Reference 13. Reference 13 presents an equation of motion suitable for estimating the time required for penetration. To determine the capacity for the barriers to absorb energy the deflection due to missile impact is determined by integrating the equation of motion, or by using a simplified expression adapted from the equation of motion. This is compared with the maximum allowable flexural deflection. The concepts used in Reference 9 are comparable to those of Reference 8.

Composite steel and concrete barriers are not utilized for missile protection.

3.5.4 References

1. Deleted

2. Deleted

3. Deleted

4. Deleted

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14. ASME Code Section III; Division 2, Subsection CC-3900 - Design Criteria for Impulse Loading and Missile Impact.

15. Letter of L. G. Knutson, Manager Engineering Services - GE Industrial and Power Systems to R. H. Mirochna; dated August 17,1990.

16. Letter of M. L. McCleish, Customer Service Manager - GE Industrial and Power Systems to Keith Colvert; dated August 23, 1991; "CECo - LaSalle Missile Analysis Safety Evaluation Report”.

17. Letter of M. L. McCleish (G. E.) to R. H. Mirochna dated October 29, 1990; "LaSalle L. P. Rotor Replacement Project Action Item Update".

18. Letter of L. G. Knutson (G. E.) to R. H. Mirochna dated December 13, 1990; "Rotor Missile Analysis".

19. Letter of L. G. Knutson (G. E.) to R. H. Mirochna dated February 20, 1991, Issue #2.

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

This section describes the measures that have been used to ensure that the reactor vessel and all essential equipment within the primary containment, including components of the reactor coolant pressure boundary, engineered safety features, and equipment supports, are adequately protected against the loss-of-coolant accident (LOCA) dynamic effects.

Protection measures taken to protect essential equipment and components against the dynamic effects of pipe rupture outside of the containment are covered in Appendix C.

The plant is designed with appropriate protection against the consequences of a LOCA. Specifically, protection includes: an emergency core cooling system to protect the core from the thermal-hydraulic consequences of a LOCA; a containment system to protect the public from the radiological consequences of a LOCA; a system of piping restraints to limit the effects of a pipe rupture; physical separation of equipment and piping; protective shields and physical constraints to limit propagation of damage from the dynamic effects (i.e., blowdown jet forces and pipe whip) associated with a LOCA.

The design provisions and corresponding criteria for the emergency core cooling and containment systems are covered in Chapter 6.0.

As it applies to a postulated pipe rupture, a loss-of-coolant accident includes those postulated accidents that result from the loss of reactor coolant at a rate in excess of the normal makeup system from breaks up to and including a break equivalent in size to the double-ended rupture of the largest pipe in the reactor coolant system. In addition, a LOCA is considered to be the combination of any single pipe break (as defined previously) and subsequent pipe and/or equipment failure that occurs as a direct consequence of the first failure, and which may occur simultaneously with a safe shutdown earthquake (SSE) with or without a concomitant loss of offsite power. For consistency with the NRC General Design Criteria for Nuclear Power Plants, a single random active component failure must be assumed to occur in any of the heat removal systems (reactor or containment), the reactor protection system, or the secondary containment atmosphere control systems, whichever is most restrictive. Hence, such a single random active component failure is assumed to occur during or following a LOCA.

3.6.1 Postulated Piping Failures in Fluid Systems Inside of Containment

3.6.1.1 Design Basis

In the analysis of the effects of a pipe rupture inside the containment, all affected structures were assumed to be safety-related and therefore all large whipping pipes were restrained.

During the initial stage of plant design, the criteria for postulating pipe breaks used locations at pipe fittings; then later, whenever possible, the stress criteria were added as explained in MEB 3-1.

All systems or components surrounding the postulated break or affected by its occurrence were protected. This ensures that no systems or components within the primary containment which are required for safe shutdown or are important to plant safety are susceptible to the consequences of high energy piping failures.

3.6.1.1.1 Core Cooling Requirements

The designed ECCS capability can be maintained provided that dynamic effects consequences do not exceed the following break area, break combination, and maintenance of minimum core cooling requirements.

3.6.1.1.1.1 Maximum Allowable Break Areas

a. For breaks involving recirculation piping, the total effective area of all broken pipes, including the effective area of the recirculation line break, do not exceed the total effective area of the design basis double-ended recirculation line break. By limiting the total area of all broken pipes involving recirculation loops to an area less than or equal to that of the design-basis accident (DBA) (circumferential break of recirculation loop), no accident could be more severe than the DBA.

b. For breaks not involving recirculation piping, the effects are much less severe than recirculation line breaks. Hence, the total break area can be allowed to be larger than the recirculation breaks. Therefore, the total break area shall not exceed the sum of one feedwater header pipe area, one steam line (upstream of flow limiter) pipe area, and one core spray pipe area.

3.6.1.1.1.2 Break Combinations

In addition to the pipe break area restrictions, breaks involving one recirculation loop shall not result in loss of function or damage to the other recirculation loop or loss of coolant from the other loop in excess of that which would result from a break of the attached cleanup connection on the suction side of the loop.

3.6.1.1.1.3 Required Cooling Systems

To ensure compliance with Appendix A of 10 CFR 50, General Design Criteria for Nuclear Power Plants, the following cooling system requirements must be met after an additional single active safety system failure:

a. For breaks not involving recirculation piping, at least two LPCI pumps or one core spray system shall be available for core cooling.

b. For breaks involving recirculation piping, at least one core spray line and two LPCI pumps, or two core spray lines, shall be available for core cooling.

c. For a LOCA with a total effective break area less than 0.7 ft^2, either the HPCS or ADS shall be available for reactor depressurization.

d. For liquid breaks such as cleanup suction or combination of liquid and steam breaks whose total break area is less than 0.7 ft^2 in which the ADS system is required for depressurization, at least (n-1) ADS valves must be available (n = total number of ADS valves).

e. For breaks less than the equivalent flow area of one open ADS valve, at least (n-1) ADS valves must be available. However, the required number of ADS valves will be one less for each additional steam break area equivalent to the area of one open ADS valve.

3.6.1.1.2 Containment System Integrity

The following was considered in addressing the LOCA dynamic effects with respect to containment system integrity:

a. Leaktightness of the containment fission product barrier shall be assured throughout any LOCA.

b. For those lines which penetrate the containment and are normally closed during operation, the inboard isolation valve is located as close as practical to the reactor pressure vessel. This arrangement reduces the length of pipe subject to a pipe break.

c. For those lines which penetrate the containment and are open during normal operation, the outboard isolation valve is located as close as practical to the containment, with surrounding equipment located so as to preclude the possibility that a single event can cause rupture of the reactor coolant pressure boundary piping anywhere from the containment to and including this isolation valve.

3.6.1.1.3 Design Limits for Piping and Components

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 which 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 when the consequences result in direct damage causing the loss of integrity of the primary containment or causing loss of required shutdown core cooling systems.

Components such as vessel safe ends and valves which are part of the broken piping system and do not serve a safety function, or whose failure would not further escalate the consequences of the accident, need not be designed to meet code imposed limits for essential components under faulted loading.

If these components are required for safe shutdown, or serve a safety function to protect the structural integrity of an essential component, limits to meet the Code requirements for faulted conditions and limits to ensure operability are met.

3.6.1.2 Description

The high energy systems identified in the design are listed in Table 3.6-1 and the particular lines within these systems subjected to analysis of postulated breaks are shown in Figures 3.6-1 through 3.6-16e. The limiting break locations are listed in Tables 3.6-2 through 3.6-5.

There are cases within the LSCS design where high energy lines have been enclosed in structures or compartments. All such situations have been analyzed to determine the effects of pressurization resulting from a line break within the structure or compartment. This was done by considering a line break within the compartment and determining the resulting pressure within the compartment and the adjoining ones. The resulting pressure differentials across the walls were then checked against the design values to determine the design margins. A detailed description of the method of analysis and the subcompartments that are affected is provided in Subsection 6.2.1.2 for breaks inside the primary containment. The pressure differentials determined for the worst cases for a selected compartment, along with the compartment location of the break, are given in Section 6.2.

The possibility of a high energy line break affecting a safety-related component in the same or an adjourning compartment was also investigated. The peak pressure and temperature, along with the predicted relative humidity for each room, are given in Section 6.2. The environmental effects of high energy line break outside the containment are enveloped by the parameters given in Section 3.11.

3.6.1.3 Safety Evaluation

The analysis of postulated line breaks and the resulting addition of restraint features into the design have ensured that failure in any single high energy line in the plant will not result in unacceptable damage to any other system or component.

3.6.2 Determination of Break Locations and Dynamic Effects Associated with the Postulated Rupture of Piping

This section describes the design basis for locating postulated breaks inside of the containment and procedures used to determine the blowdown and impingement loads associated with these postulated breaks.

Systems in which Design Basis Piping Break Occur

High energy piping systems are defined as those systems, or portions of systems, which during "normal plant conditions" are either in operation or maintained pressurized under conditions where either or both of the following are met:

a. maximum temperature exceeds 200° F, or

b. maximum pressure exceeds 275 psig.

Normal plant operation (per ASME Section III Paragraph NB-3113) includes startup, operation in the design power range, normal hot standby, and system shutdown. Normal hot standby is a normally attained zero power plant operating state (as opposed to a hot standby initiated by a plant upset condition) where both feedwater and main condenser are available and in use.

Moderate energy piping systems are those systems, or portions of systems, which 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 temperature is 200° F or less, and

b. maximum pressure is 275 psig or less.

High energy systems may be classified as moderate energy if the total time that either of the previous conditions are met is less than either of the following:

a. one percent of the normal operating life span of the plant, or

b. two percent of the time period required to accomplish its system design function.

The following high energy piping systems (or portions of systems) within and outside of the containment are considered as potential initiators of a pipe break and have been analyzed for dynamic effects damage potential:

a. all piping which is part of the reactor coolant pressure boundary and subject to reactor pressure continuously during station operation,

b. all piping which is beyond the second isolation valve but which is subject to reactor pressure continuously during station operation, and

c. in addition to piping under items a and b, all other piping systems or portions of piping systems considered high energy systems.

Systems in which one of the following conditions exists were not considered as an initiator of pipe break:

a. piping which never or only infrequently (i.e., during test operations) is subject to reactor pressure;

b. piping which is classified as moderate energy piping; and

c. piping where the internal energy level associated with the whipping pipe is insufficient to impair the safety function of any structure, system, or component to an unacceptable level. The energy level in the whipping pipe is considered insufficient to rupture an impacted pipe when it is of equal or greater nominal pipe size and equal or heavier wall thickness. The internal fluid energy level associated with the pipe break reaction takes into account any line restrictions (e.g., flow limiters) between the pressure source and break location, and the effects of either a single-ended or double-ended flow condition.

Initial pipe break events are not assumed to occur in pump and valve bodies because of their greater wall thickness.

Consideration of Other Systems

While none of the following systems are needed during or following a LOCA, some dynamic effects must be considered because a "non-safety-class" system or component failure could initiate or escalate the LOCA. The following subsystems and components are in this category, however they are not required for the safe shutdown of the reactor nor are they required for the limitation of the offsite release in the event of a LOCA:

a. reactor water cleanup system,

b. CRD return lines,

c. reactor head spray,

d. steam to RHR heat exchanger and RCIC turbine,

e. RHR shutdown suction and return piping, and

f. CRD insert lines.

3.6.2.1 Criteria Used to Define Break and Crack Location and Configuration

The following definitions are utilized for piping run terminology:

Main Run - piping interconnecting terminal ends. All branch lines from the main run are considered branch runs, with the exception of the following:

a. free ended branch lines throughout which there is no significant restraint to thermal expansion are considered part of the main run, and

b. all ASME, Section III, Class 1 branch lines which are included with the main run piping in the code stress analysis computer mathematical model are considered part of the main run.

Piping Run - a main or branch run.

Terminal End - piping originating at structure or components (such as vessel and equipment nozzles and structural piping anchors) that act as rigid constraint to the thermal expansion. Typically, the anchors assumed for the piping code stress analysis are considered terminal ends. The branch connection to the main run is one of the terminal ends of a branch run, except where the branch run was classified as part of a main run as defined above.

3.6.2.1.1 Break Locations in ASME Section III Class 1 Piping Runs

Postulated pipe break locations are selected in accordance with the intent of Regulatory Guide 1.46, USNRC Branch Technical Position APCSB 3.1, Appendix B, and as expanded in NRC Branch Technical Position MEB 3.1. For ASME Section III, Class 1 piping systems, the postulated break locations are as follows:

a. The terminal ends of the pressurized portions of the run. (Terminal ends are extremities of piping runs that connect to structures, equipment, or pipe anchors that are assumed to act as rigid constraints to free thermal expansion of piping. A branch connection to a main piping run is a terminal end for a branch run, except when the branch and the main run is modeled as a common piping system during the piping stress analysis.)

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

   1. If the stress range calculated using Equation (10) of the Code exceeds $2.4.S_m$ but is not greater than 3 S_m, no

breaks will be postulated unless the cumulative usage factor exceeds 0.1.

2. The stress ranges, as calculated by Equations (12) or (13) of the Code, exceed 2.4 S_m or if the cumulative usage factor exceeds 0.1 when Equation (10) exceeds 3 S_m.

c. In the event that two or more intermediate locations cannot be determined by stress or usage factor limits, a total of two intermediate locations shall be identified on a reasonable basis for each piping run or branch run. (Reasonable basis shall be one or more of the following:

1. Fitting locations.

2. 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 were used. A break at each end of a fitting may be classified as two discrete break locations where the stress analysis is sufficiently detailed to differentiate stresses at each postulated break.)

Break locations required by the criteria in 3.6.2.1.1(c) above and in 3.6.2.1.2-3 are termed arbitrary intermediate breaks (AIB's). In a July 18, 1986 letter from E. G. Adensam (Director, BWR Project Directorate No. 3, Division of BWR Licensing NRC) to D. L.. Farrar (Director of Nuclear Licensing, Commonwealth Edison) the NRC eliminated the requirement to provide mechanical pipe rupture protection against AIB's. The staff's approval to eliminate AIB's is for pipe rupture protection purposes only. The elimination of AIB's is not to be utilized to eliminate any areas of harsh environments, or to reduce the severity of environmental conditions in those areas, that have been previously included in the LaSalle equipment environmental qualification program.

Conformance to the above pipe break criteria is demonstrated in Figures 3.6-1, 3.6-1a, 3.6-2, and 3.6-2a, and Table 3.6-8.

3.6.2.1.2 <u>Break Locations in ASME Section III Class 2 and 3 Piping Runs</u>

Breaks were postulated to occur at the following locations in each ASME Section III Class 2 and 3 piping run:

a. at the terminal ends of the pressurized portions of the pipe run,

b. at intermediate locations determined by one of the following selection rules:

1. at each location of potentially high stress or fatigue, such as pipe fittings (elbows, tees, reducers, etc.), valves, flanges and welded attachments; or

2. at all locations where the stress, S, exceeds 0.8 $(1.2\ S_h + S_A)$.

where:

S = stresses under the combination of loadings associated with the normal and upset plant condition loadings, as calculated from the sum of equations (9) and (10) in Subarticle NC-3600 of the ASME Boiler and Pressure Vessel Code, Section III;

S_h = basic material allowable stress at maximum (hot) temperature from the allowable stress tables in Appendix I of the ASME Boiler and Pressure Vessel Code, Section III; and

S_A = allowable stress range for expansion stresses, as defined in Subarticle NC-3600 of the ASME Boiler and Pressure Vessel Code, Section III.

3. For those cases without at least two intermediate locations where S exceeds $0.8(1.2\ S_h + S_A)$, two separated locations were chosen based upon highest local stress. Where the piping run has only one change of direction, a minimum of one intermediate break was postulated.

   The pattern of postulated intermediate break locations was determined separately for the normal plant condition load combination which results in the highest level of stress. (See note under Subsection 3.6.2.1.1(c)).

3.6.2.1.3 <u>Break Locations in Other Piping Runs</u>

Breaks were postulated to occur at the following locations in each piping run which was not ASME Section III Class 1, 2, or 3 piping:

a. at the terminal ends of the pressurized portions of each run, and

b. at each intermediate location of potentially high stress or fatigue, such as pipe fittings (elbows, tees, reducers, etc.) valves, flanges and welded attachments.

3.6.2.1.4 Break Configuration

High energy piping systems are analyzed for appropriate break configurations.

3.6.2.1.4.1 High Energy Piping Systems

The following types of breaks were postulated in high energy piping systems:

a. No breaks were postulated in piping having a nominal diameter less than or equal to 1 inch.

b. Longitudinal and circumferential breaks were postulated in piping having a nominal diameter greater than 1 inch. Breaks were arbitrarily postulated at pipe fittings and at points of maximum constraint without the benefit of detailed stress analysis. In pipe runs exceeding twenty times the pipe diameter at least two break locations were postulated based on stress considerations and at least one intermediate break considered for pipe run of less than twenty pipe diameters but greater than three.

Except where limited by structural design features, a circumferential break results in pipe severance with full separation. The break was assumed perpendicular to the longitudinal axis of the pipe at the break location. The fluid discharge coefficient at the break was determined from analytical or experimental work. In the absence of this data, the discharge coefficient was assumed to be 1.0.

A longitudinal break results in an axial split without severance. The split was assumed to be orientated nonselectively at any point about the circumference of the pipe. For design purposes, the longitudinal break was assumed to be rectangular in shape, with an area equal to the largest piping cross-sectional flow area at the point of break, a length equal to twice the piping internal diameter at that cross section, and to have a discharge coefficient of 1.0. Any other values used for the area, diameter, or discharge coefficient associated with a longitudinal break were verified by test data that defined the limiting break geometry.

For purpose of analysis, circumferential and longitudinal breaks are assumed to reach full size within 1 millisecond after break initiation, unless otherwise analytically or experimentally substantiated.

3.6.2.1.4.2 Moderate Energy Piping Systems

Inside the primary containment no breaks were postulated to occur in moderate energy piping systems and therefore no break configuration was assumed.

3.6.2.1.5 Containment Penetrations

Information concerning containment penetrations is discussed in Subsection 3.8.1.1.3.5. Table 3.8-1 and Figures 3.8-15 and 3.8-18 through 3.8-21 provide penetration configuration and location.

3.6.2.2 Analytical Methods Used to Define Forcing Functions and Response Models

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. Unsteady loads result from depressurization wave propagation, which causes the various sections of pipe to be loaded with time dependent forces. Steady blowdown thrust loads are equivalent to a corresponding thrust applied normally to the plane of the break and opposite to fluid blowdown velocity. These loads can be computed for each section of piping system, and corresponding external restraints can be provided if it is necessary to limit the movement of the piping system. A detailed discussion of the analytical methods employed to compute these blowdown loads for LSCS are given in References 1 and 2.

3.6.2.2.1 Dynamic Analysis

3.6.2.2.1.1 Design-Basis Breaks

Table 3.6-1 indicates the number of design-basis breaks along with their rupture orientation which were postulated inside the containment (for pipe rupture outside containment, see Appendix C). Inside the containment, 102 specially designed pipe whip restraints are utilized in controlling the dynamic effects associated with the postulated rupture of piping. Figures 3.6-1 through 3.6-16e illustrate the location of postulated breaks and pipe rupture restraints.

3.6.2.2.1.2 Forcing Function Used for Pipe Whip Dynamic Analysis

Two independent dynamic analyses have been utilized inside the containment. The first dynamic analysis was performed on the reactor recirculation system. The other dynamic analysis was performed on the remaining systems listed in Table 3.6-1. The forcing functions associated with each analysis are different and are depicted in Figures 3.6-17 and 3.6-18.

3.6.2.2.1.2.1 Forcing Function Associated with Pipe Whip Dynamic Analysis on the Reactor Recirculation System

Several types of time dependent loads may be applied, only one of which can be used for any given analysis. The forcing function used (for blowdown thrust) was a three-step function to conform with pipe dynamic analysis (PDA) input (Reference 3).

3.6.2.2.1.2.2 Forcing Function Associated with Pipe Whip Dynamic Analysis Performed on High Energy Systems Inside Containments Excluding Reactor Recirculation

The forcing function is shown in Figure 3.6-18. This is a step function depicting a steady-state blowdown force to conservatively represent the magnitude of the jet reaction. The pipe break is assumed to occur instantaneously.

For circumferential breaks, the force has the following magnitude:

$$PA \leq F \leq 1.26PA$$

where:

P = operating pressure

A = break area.

For longitudinal breaks, an instantaneously applied steady force equal to 1.26 PA was used.

3.6.2.2.2 Method for the Dynamic Analysis of Pipe Whip

The analytical approaches used in the two dynamic analyses which determine the response of a pipe subjected to the thrust force occurring after a pipe break are described in this section.

The dynamic analyses of pipe whip are performed by both General Electric and Sargent & Lundy. The reactor recirculation system is analyzed by GE (Method I) and the remaining high energy lines inside the containment are analyzed by S&L (Method II).

The method used by GE in performing the pipe whip analysis is described in Subsection 3.6.2.2.2.1. The method used by S&L is described in Subsection 3.6.2.2.2.2. The essential differences between the two methods can be summarized as follows:

(a) Restraint material characteristics utilized by GE include strain hardening effects. Sargent & Lundy assumed elastic; perfectly plastic behavior.

(b) Pipe material characteristics utilized by GE include strain hardening effects. Sargent & Lundy assumed the rigid, perfectly plastic, moment rotation law.

The usage of different analysis methods for different piping systems is a result of the division of responsibility between General Electric and Sargent & Lundy rather than an indication of limitations on applicability of either method.

Method I has been verified by testing and is reported in detail in a proprietary GE document No. NEDE 10811, “Pipe Restraint Testing Program Conducted in Conjunction with the Design of the Enrico Fermi Plant Unit 2.” In addition, an independent verification of the General Electric PDA Computer Code was performed by Nuclear Services Corporation (NSC). The results of this verification are summarized in Subsection 3.6.4.1.1.1 of the GESSAR.

3.6.2.2.2.1 Method I (Reactor Recirculation System)

A generic representation of the pipe in any given analysis is shown in Figure 3.6-19. In certain specific geometries, the stiffness of the piping segment located between A and B will be such that the slope of the pipe length, BD, at B, will always be zero. When this is the case, the pipe may be treated as built-in at B, as indicated in Figure 3.6-20. Other geometries may permit the slope of BD at B to vary considerably from zero. In this case, the pipe may be considered to have a fixed, simple support, (pinned end) at B, as seen in Figure 3.6-20. Therefore, in this analysis, a pipe may be assumed to have either a built-in end at B or a fixed, simple support at B.

The PDA program also has the capability of analyzing a case where the pipe is assumed to have both ends supported (Figure 3.6-21). To analyze this case with the computer model as described in the preceding paragraph, two simplifications are required. First, an equivalent point mass is assumed to exist at D instead of pipe length DE. The inertia characteristics of this mass, as it rotates about point B, are calculated to be identical to those of pipe length DE, as it rotates about point E. Secondly, from the bending moment - angular deflection relationship for pipe length DE, an equivalent resisting force can be calculated for any deflection for the case of a built-in end. 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 3.6-21.

When the thrust force is applied to the end of the pipe, angular acceleration will occur about point B (Figure 3.6-20). As the pipe moves, a resisting bending moment will be created and will then reduce the net angular acceleration. This net angular acceleration will also be reduced by the application of a restraining force at C. When the resisting moments about B exceed the applied thrust moment, angular deceleration will occur. The kinetic energy will be absorbed by the deflection of the restraining device and the bending of the pipe.

It is assumed that all deflections of the pipe are due to pipe rotation about points B and C. Based on this assumption, the pipe length BC and CD are linear as shown in Figure 3.6-22.

The restraining device is assumed to be composed of two components acting in series, i.e., the restraint itself, and the structure to which the restraint is attached. Both parts of the restraining device will deflect under load. The restraint behaves as dictated by an experimentally or analytically determined force-deflection relationship. The structure deflects as a simple, linear spring of any given spring constant. Given that the restraint and structure are connected in series, the deflection of the restraining device under any load can be determined. Normally there is a clearance between the pipe and the restraint.

Upon contacting the restraint, pipe length CD (Figure 3.6-20) may move relative to pipe length BC. If this occurs, pipe length CD is assumed to act as a pipe built in at C with a force applied at the free end D. While point D may move relative to point C, point C may also be moving relative to B.

The PDA program uses Lagrangian equations of motion to balance energy and calculate velocities and displacements over small increments of time which are summed up, processed, and printed on the output as peak dynamic forces between restraint and structure, restraint and pipe deflections, time to restraint impact, and time to equilibrium.

3.6.2.2.2.2 Method II (High Energy System Inside Containment Excluding Reactor Recirculation)

Pipe whip restraints provide clearance for thermal expansion during normal operation. If a break occurs, the restraints or anchors nearest the break are designed to prevent unlimited movement at the point of break (pipe whip). The simplified models shown in Figure 3.6-23 were used to represent the local region near the break and to calculate the displacement of the pipe and the restraint. These calculated displacements were then used to estimate strains in the pipe and restraint.

A rigid-perfectly plastic-moment rotation law is assumed for the pipe. The restraint and structure resistances R and R_S are also assumed rigid-perfectly plastic.

3.6.2.2.2.3 Stages of Motion

All references to points and lengths in this section can be found in Figure 3.6-23.

At the start of motion the pipe is assumed fixed at point A. Physically point A is an anchor, restraint, or elbow. In general, a hinge will form at some point B and outboard pipe segment BD will rotate as a rigid body until contact with the restraint is made at point C.

During the next stage of motion the hinge at B must move in order to satisfy the requirement that shear at a plastic hinge is zero. At the same time a hinge will form at the restraint (point C) if the yield moment M_O is exceeded. Initially at contact, the force exerted on the pipe by the restraint is R, the restraint resistance. This force will remain constant as long as the restraint continues to deform.

If the structure resistance is $R_S < R$, at some point restraint deformation will stop while structure deformation (motion of point E) continues. The force on the pipe (and attached mass M) is the R_S. In any event, the moving hinge B will reach the fixed support at A before motion stops at C. In the final stage of motion hinges may exist at A and C until motion stops.

3.6.2.2.2.3.1 First Stage of Motion

The initial location of the hinge at B is determined by locating the point of zero shear and is given by:

$$L_S = 1.5\left[1 + \left(1 + \frac{8M_tF}{3mM_O}\right)^{1/2}\right]\frac{M_O}{F} \qquad (3.6\text{-}1)$$

where:

M_t = tip mass (lbm),

F = blowdown force (lb),

m = mass of pipe/inch,

M_O = plastic moment of pipe (in.-lbm), and

Ls = location in inches.

3.6.2.2.2.3.2 Second Stage of Motion (Moving Hinge)

Case 1. No hinge at restraint (Figure 3.6-24).

After integrating, with respect to time, the equations for conservation of linear and angular momentum are:

$$P_1 t = I_1 \omega - C_1 \qquad (3.6\text{-}2)$$

$$P_2 t = I_2 \omega - C_2 \qquad (3.6\text{-}3)$$

where:

C1 and C2 are constants and are determined at

t = 0,

t = time of motion from B to present location,

$I_1 = 1/2\ m\ L^2 + M_S(L-L_2) + M_tL$,

$I_2 = (1/6)mL^2\ (3\ L_2-L) + M_t\ L_2L$,

$P_1 = F - R$,

$P = F\ L_2 - M_O$, and

$\omega = \theta$ (radians/second).

From equations (3.6-2) and (3.6-3)

$$\omega=\frac{P_1 t+C_1}{I_1} \qquad (3.6\text{-}4)$$

$$t=\frac{C_1 I_2 - C_2 I_1}{P_2 I_1 - P_1 I_2} \qquad (3.6\text{-}5)$$

Equations (3.6-4) and (3.6-5) describe the second stage of motion.

Case 2. Hinge at restraint (Figure 3.6-24).

For conservation of linear and angular moments of the segments:

$$\theta = C_S/(L-L_2)^2 \qquad (3.6\text{-}6)$$

$$P_S t + C_5 = M_{12}C_3/(L-L_2)^2 + M_{11}W \qquad (3.6\text{-}7)$$

$$P_1 t + C_4 = I_3 C_3/(L-L_2)^2 + M_{12}W \qquad (3.6\text{-}8)$$

where:

C3, C4 and C5 are constants and are determined at

$I_3 = 1/2m(L+L_2) + M_S + M_t$,

$M_{11} = (1/3)mL_2^3 + M_t\, L_2\, 2$, and

$M_{12} = 1/2mL_2^2 + M_t L_2$.

From Equations (3.6-7) and (3.6-8):

$$t=\frac{C_3\left(M_{12}2-M_{11}I_3\right)/\left(L-L_2\right)^2-\left(C_5 M_{12}-C_4 M_{11}\right)}{\left(P_2 M_{12}-P_1 M_{11}\right)} \qquad (3.6\text{-}9)$$

$$W=\frac{P_2 t+C_5-C_5-C_3 M_{12}/\left(L-L_2\right)^2}{M_{11}} \qquad (3.6\text{-}10)$$

Equations (3.6-6), (3.6-9), and (3.6-10) describe the second stage of motion for hinge at restraint.

3.6.2.2.2.3.3 <u>Third Stage of Motion (Hinge at Support)</u>

From summation of moment about two hinges (at support and restraint) one gets:

$$K_{11}\,\ddot{\theta}_1 + K_{12}\,\ddot{\theta}_2 = FL - RL_1 - M_O \qquad (3.6\text{-}11)$$

$$K_{12}\ddot{\theta}_1 + K_{22}\ddot{\theta}_2 = FL_2 - M_O \qquad (3.6\text{-}12)$$

where:

$K_{11} = (1/3)mL^3 + M_t L^2 + M_S L_1^2$,

$K_{12} = (1/2)mL_2^2 + (L - L_2)/ 3 + M_t LL_2$ and

$K_{22} = (1/3)mL_2^3 + M_t L_2^2$

Equations (3.6-11) and (3.6-12) describe motion in the third stage.

3.6.2.2.3 Summary of Dynamic Analyses for Postulated Rupture of Piping Inside Containment

The results of the dynamic analyses are presented in Tables 3.6-2 through 3.6-8. These tables present the most significant breaks. The breaks are shown in Figures 3.6-1 through 3.6-16e. Every postulated break and restraint shown in Figures 3.6-1 through 3.6-16e has been analyzed; in each case the restraints are capable of performing their intended safety function.

Table 3.6-2 presents the most significant breaks separated into loads and deflections for the reactor recirculation system. The remaining high energy systems inside the containment are shown in Tables 3.6-3 through 3.6-5 and contain the following information:

a. line number,

b. restraint number,

c. break location,

d. blowdown load,

e. break type, and

f. subsystem number.

Tables 3.9-5 and 3.9-6 contain the following information concerning main steam and recirculation loop piping:

a. time dependent loading on restraint,

b. tip displacement,

c. allowable tip displacement based on 50% in pipe,

d. restraint deflection,

e. maximum strain in restraint, and

f. allowable strain in restraint.

There are three types of time dependent loadings described in Tables 3.6-3 through 3.6-5. The three types of loadings are shown in Figure 3.6-25. The first type of loading occurs when two plastic hinges are created in the whipping pipe. The second type of loading occurs when one plastic hinge is created in the whipping pipe. The third type of loading occurs when no plastic hinges are created in the whipping pipe.

3.6.2.3 Dynamic Analysis Methods Used to Verify Integrity and Operability

This subsection describes the design basis and loading for both pipes and restraints for the prevention or escalation of dynamic effects associated with a pipe rupture. A summary of protective measures is also included within this subsection.

3.6.2.3.1 Load Combinations and Associated Design Limits - Pipes

The following operating conditions and design loads were used in the pipe break analysis. Reactor coolant pressure boundary stress limits corresponding to these loads were determined in accordance with ASME Code, Section III Class 1:

Operating Condition	Loads
Design	1. Design Pressure
	2. Design Temperature
	3. Deadweight
	4. OBE
	5. Relief Valve Forces (steam piping only)
Normal and Upset	1. Thermal Expansion
	2. OBE Free End Displacement
	3. Operating Pressure
	4. Deadweight
	5. OBE Inertial Effects
	6. Cyclic Pressure and Temperature
Emergency	1. Deadweight
	2. OBE Inertial Effects
	3. Peak Pressure
	4. Relief Valve Forces (steam piping only)
Faulted	1. Deadweight
	2. SSE Inertial Effects
	3. Peak Pressure
	4. Relief Valve Forces (steam piping only)

3.6.2.3.2 Load Combinations and Associated Design Limits - Restraints

The following defines the functions of the pipe whip restraint as to types, design and loading. Both Sargent & Lundy and General Electric design restraints are discussed.

3.6.2.3.2.1 Definition of Pipe Whip Restraint Function

Pipe whip restraints are differentiated from piping supports and are designed to function and carry load for an extremely low probability gross failure in the reactor coolant pressure boundary (RCPB). The RCPB piping integrity does not depend on the pipe whip restraints during normal, upset, emergency, or faulted conditions as defined in Paragraph NB 3113, Section III of the 1974 ASME Boiler Pressure Vessel Code, but relies on piping supports to maintain acceptable piping design stress values and piping integrity.

The pipe whip restraints (i.e., those devices which serve only to control the movement of a ruptured pipe following gross failure) are subjected to a once in a lifetime loading. Local pipe and restraint deformations which occur upon impact do not further affect the integrity of the RCPB. For the purpose of design, the pipe break event is considered to be a faulted condition and the pipe, its restraints, and the structure to which the restraint is attached were analyzed accordingly.

Piping is no longer considered to be a part of the RCPB following the pipe break. Plastic deformation in the pipe is considered as a potential energy absorber. Limits of strain can be imposed which are similar to strain levels allowed in restraint plastic members. Piping systems were designed so that plastic instability does not occur in the pipe at the design dynamic and static loads and deformations when the consequences would have resulted in direct damage causing loss of integrity of the primary containment or causing loss of required shutdown core cooling systems.

3.6.2.3.2.2 Types of Pipe Whip Restraint Components

In order to establish a design basis relating to material selection, fabrication inspection, installation quality assurance, and applicable design limits, three types of restraint hardware are defined.

In addition, the structural and civil components must be considered as a separate type.

Type - Restraint Energy Absorption Members - Members that are under the influence of impacting pipes (pipe whip) will absorb energy by significant plastic deformations (e.g., U-bolts, rods, bars).

Type II - Restraint Connecting Members - Those components which form a direct link between the restraint plastic members and the structure (e.g., devices, brackets, pipe).

Type III - Restraint Connecting Member Structural Attachments - Those fasteners which provide the method of securing the restraint connecting members to the structure (e.g., weld attachment).

Type IV - Structural and Civil Components - Steel and concrete structures which ultimately must carry the restraint load (e.g., sacrificial shield, truss).

3.6.2.3.2.3 Restraint Loading Basis

For the purpose of design, the pipe restraint components as classified previously, Types I-IV, were designed for the following dynamic loads:

a. blowdown thrust of the pipe section which impacts the restraint, and

b. dynamic inertia loads of the moving pipe section which is accelerated by the blowdown thrust and impacts the restraint.

3.6.2.3.2.4 Restraint Design Requirements

Objectives specific to restraint design are as follows:

a. the restraints in no way increase the reactor coolant pressure boundary stresses by their presence during any mode of reactor operation or condition, and

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

3.6.2.3.2.5 Design Basis for Sargent & Lundy Designed Restraints of Reactor Coolant Pressure Boundary

3.6.2.3.2.5.1 Type I

a. Materials - all materials which were used to absorb energy through significant plastic deformation conform to:

    1. ASME - Section III, Subsection NB, B&PV Code for Class I Components; or

    2. ASTM - Specifications with consideration for brittle fracture control, or

    3. ASME - Section III, Subsection NF, B&PV Code when applicable.

b. Inspection - inspection and identification of material conforms to:

1. ASME - Section III, Subsection NB, B&PV Code for Class I Components (Section V Nondestructive Examination Methods), or

2. ASTM - Specifications procedures including volumetric and surface inspection, or

3. ASME - Section III, Subsection NF, B&PV Code when applicable.

c. Design Limits

1. Design Local Strain - the permanent strain in metallic ductile materials was limited to 50 percent of the minimum actual uniform elongation based on restraint material tests, or

2. Design Steady-State Load - the maximum restraint load was limited to 80 percent of the minimum calculated static ultimate restraint strength at the drywell design temperature.

3. Dynamic Material Mechanical Properties - the material selected exhibits tensile impact properties which are not less than:

a) 70 percent of the static percent elongation, and

b) 80 percent of the statically determined minimum total energy absorption.

3.6.2.3.2.5.2 Type II

a. Materials selection conforms to:

1. ASTM specifications including consideration for brittle fracture control, or

2. ASME - Section III, Subsection NF, B&PV Code when applicable.

b. Inspection conforms to:

1. ASME/ASTM requirements or process qualification and finished part surface inspection per ASTM methods, or

2. ASME - Section III Subsection NF, B&PV Code when applicable.

c. Design Limits are based on the following stress limits:

1. Primary stresses are limited to the higher of:

a) 70% of S_u where S_u = minimum ultimate strength by tests or ASTM specification,

b) $S_y + 1/3 (S_u - S_y)$ where S_y = minimum yield strength by tests or ASTM specification, or

2. Recommended stress limits per ASME Section III - Subsection NF or faulted conditions when applicable.

3.6.2.3.2.5.3 Type III

3.6.2.3.2.5.3.1 Fasteners

a. Materials - fastener material conforms to ASTM, ASME, or MIL requirements,

b. Inspection - all fasteners are inspected or certified per applicable ASTM, ASME, or MIL specifications, and

c. Design Limits - same as Type II.

3.6.2.3.2.5.3.2 Welds

a. Materials - weld materials for attachments to carbon steel structures are per AWS/ASME specification per:

1. AWS A5.1, A5.5 or ASME SFA 5.1, low hydrogen electrode for metal arc welding, or

2. AWS A5.18, or #70S2, filler metal for gas metal arc welding (GMAW) or gas tungsten are welding (GTAW).

b. Inspection - liquid penetrant surface inspection are performed per:

1. ASTM Specifications E165,

2. ASME Section VIII, Appendix VIII, B&PV Code, and

3. acceptance standards are per paragraph NB 5350 ASME Section III, B&PV Code.

c. Design Limits - are based on the following stress limits:

the maximum primary weld stress intensity (two times maximum shear stress) is limited to three times AWS or AISC building allowable weld shear stress.

d. Procedures - procedures and welders are qualified per:

1. AWS Code for welding in building structures D10-69, or

2. ASME - Section IX, B&PV Code, and

3. in addition, weld qualifications include Charpy-V Notch impact testing to ensure ductile behavior.

3.6.2.3.2.6 Design Basis for General Electric Recirculation Loop Pipe Whip Restraints

The restraint design used on this plant is of the type designed for a number of GE BWR 4 and 5 product line recirculation systems. The restraint uses a moderately low clearance design with a frame attached to a support and either carbon steel wire ropes or stainless steel bars restraining the pipe.

The analytical methods used in the design are similar to those used on Fermi-2 and Duane Arnold recirculation piping. They have, however, been upgraded by applying the latest force-deflection data available on wire rope and using GE's Code (Reference 3) 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 (that is, parallel to the restraint base).

The criteria used to determine the adequacy of the restraint load-carrying capacity are as follows:

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

b. The design limit used for the analysis of the stainless steel bar and the carbon steel restraint frame was 50 percent of the minimum uniform ultimate tensile elongation.

3.6.2.3.3 Protective Measures

3.6.2.3.3.1 Protection and Analyses Guidelines

Protection against the dynamic affects of a LOCA is provided in the form of pipe whip restraints, equipment shields, and physical separation of piping, equipment, and instrumentation. The precise method used in choosing the kind of protection depends on other limitations placed on the designer, such as accessibility, maintenance, and proximity to other pipes. The following are examples of present designs intended to better protect safety-related equipment from the consequences of the pipe breaks:

a. The following lines were analyzed for restraint against pipe whip inside the containment and drywell:

    1. main steamlines,

    2. feedwater lines,

    3. RHR lines upstream of the check valve to the recirculation loop,

    4. head spray line upstream of the check valve, and

    5. RCIC steamline to the RCIC turbine.

b. High energy lines outside of the containment are analyzed for restraint against pipe whip and are covered in Appendix C.

c. Safety/relief valves and the RCIC steamline are so located and restrained that a pipe failure will not prevent depressurization.

d. Barriers are provided to preserve the independence of the LPCS and LPCI systems when they are so located that a pipe failure could prevent the low-pressure water injection from occurring.

Dynamic effects associated with the LOCA do not compromise the integrity of the containment and drywell. In most cases, restraint of the potentially hazardous pipes will be utilized.

The consequences of dynamic effects external to the containment were considered to ensure that no external pipe break, in addition to an SSE, loss of offsite a-c power, and a single active safety system failure can result in any of the following:

a. failure to insert control rods,

b. failure to isolate the reactor coolant pressure boundary, and

c. failure to meet the core cooling system requirement of Subsection 3.6.1.1.2.

Dynamic effects as a result of pipe break between a normally closed inboard isolation valve and the containment were analyzed and determined to be inconsequential.

Valves which are normally closed and are not signalled to be open were assumed to be closed.

Impacted active equipment (e.g., valves and instruments) are considered unable to perform their intended functions unless loads are shown to be within allowable limits. Impacted passive equipment (pipes, restraints, and structures) are considered capable of continuing to perform their intended functions since the analysis shows that the resulting strain levels do not exceed defined limits.

The internal fluid energy level associated with the pipe break reaction takes into account any line restrictions (e.g., flow limiter) between the pressure source and break location, and the effects of either single-ended or double-ended flow conditions as applicable. The energy level in a whipping pipe is considered as insufficient to rupture an impacted pipe if it is of equal or greater nominal pipe size and equal or heavier wall thickness.

Protection of the reactor pressure vessel from the surface impact effects of a pipe whip need not be considered because the impact energy is insufficient to cause loss of the functional integrity of the vessel.

In calculating the pipe reaction, jet impingement, containment pressure loads, and break areas for core cooling, full credit was taken for any line restriction and line friction between the break and the pressure reservoir. The following represent typical restrictions to flow which are specifically considered:

a. jet pump nozzles;

b. core spray nozzles (inside internals shroud);

c. feedwater sparger; and

d. steamline flow restricter.

For the purpose of predicting the pipe rupture forces associated with the reactor blowdown, the local line pressures are those normally associated with the reactor operating at 105% of rated power and with a vessel dome pressure of 1045 psig.

3.6.2.3.3.2 Equipment Shields for Isolation

Equipment shields are provided in order to isolate the portion of the equipment in an accident and prevent it from causing a further chain accident. These shields are designed to withstand the rupture forces from piping, jets, and equipment, and will segregate the redundant systems.

3.6.2.3.3.3 Pipe Whip Impact Shields

Pipe whip and impact shields are designed to withstand the impact forces arising from the whipping action.

The design has considered elasto-plastic behavior of structures and shields using the loading criteria defined herein.

3.6.2.3.3.4 Jet Impingement Shields

Jet impingement shields are provided to limit the consequence of rupture of the piping and are designed to withstand the resultant jet forces, using the codes specified in Section 3.8.

3.6.2.3.3.5 Separation

Maintaining the independence of redundant safety systems and components is achieved in most cases by separating the redundant components so that no single postulated event can prevent the safety-related function from occurring. This is achieved by the following:

a. physical separation of source and target,

b. routing of cables so that different penetrations and paths are utilized to ensure that one event will not preclude both the primary and backup components from fulfilling their design function,

c. deflection utilized to redirect a jet spray from an essential component,

d. utilization of intermediate components and structure to intercept and defray forces, and

e. location of duplicate instrument lines to ensure that one cause will not preclude each of the redundant systems from fulfilling their design function.

3.6.2.3.3.6 Typical Pipe Whip Restraints

The typical pipe whip restraint configuration utilized on the reactor recirculation piping system is shown in Figure 3.6-26. In this type of restraint, steel bars are also used for energy absorbers in tension only. A diagram of a restraint frame is shown in Figure 3.6-26a.

There are four types of pipe whip restraints used on the remaining high energy systems inside the containment which are shown in Figure 3.6-27. The specific type used at a specific location depends on the surrounding geometry of the structure and the most probable direction of loading.

The first type of pipe whip restraint (Figure 3.6-27) is the basic three bar restraint. The two outer bars are comprised of necked down steel plates designed to yield and carry load only in tension. The middle bar is referred to as the compression post.

When the most probable direction of loading is to load the compression post, the second type of pipe whip restraint is utilized (Figure 3.6-27). This restraint is like the first type except that there is crushable material on the end of the compression post which is used as an energy absorbing device.

Where space does not permit using type 1 or 2, type 3 (Figure 3.6-27) is utilized. In this type of restraint the two outer bars of the type 1 restraint are replaced with necked down bolts designed to yield and act in tension only. The compression post is replaced by a steel plate.

The fourth type of pipe whip restraint (Figure 3.6-27) is used when the surrounding structural environment does not permit the use of types 1, 2, or 3, and no plastic hinges are formed in the whipping pipe.

3.6.2.4 Guard Pipe Assembly Design Criteria

Guard pipe assemblies were not used on the LSCS Mark II containment.

3.6.2.5 Material to be Submitted for Operating License Review

3.6.2.5.1 Implementation of Criteria for Defining Pipe Break Locations and Configurations

The implementation of the criteria as defined in Subsection 3.6.2.1 has been adhered to in the analysis of pipe rupture. The postulated rupture orientation and number of design-basis breaks used in the pipe rupture analysis inside the containment can be found in Table 3.6-1.

3.6.2.5.2 Implementation of Criteria Dealing with Special Features

The use of special protective devices, pipe whip restraints, have been used on LSCS and are in conformance with Subsection 3.6.2.3.2. Location of restraints on high energy piping systems within the containment can be found in Figures 3.6-1 through 3.6-16. They are of the type shown in Figures 3.6-26 and 3.6-27.

3.6.2.5.3 Acceptability of Analysis

The postulation of high energy line break locations and the analysis of resulting jet thrust, impingement and pipe whip effects have conservatively identified all areas where restraints or other protection devices are required to protect safety-related systems and components.

3.6.2.5.4 Design Adequacy

All safety-related systems and components have been protected from the dynamic effects of pipe whip and are assumed to function under normal operating conditions. The pipe restraint requirements as defined in Subsections 3.6.2.3.2.4 and 3.6.2.3.2.5 are met.

3.6.2.5.5 Implementation of Criteria

The criteria pertinent to the design of piping restraints is detailed in Subsections 3.6.2.3.2, 3.6.2.3.2.6, and 3.6.2.3.3, along with design limits for stress levels, operating conditions, and material properties. Typical final design configurations are shown in Figures 3.6-26 and 3.6-27. In most applications, installation of restraints will not pose any impediment to inservice inspection operation.

In a few instances however, a split-ring design will be utilized to facilitate disassembly and partial removal of the restraint device for inspection purposes.

Information concerning the arrangement of access openings is provided in Subsections 3.8.1.1.3.5.4 and 3.8.2.1.

3.6.3 References

1. GE Report NEDE 10313 - "PDA - Pipe Dynamic Analysis Program for Pipe Rupture Movement." (Proprietary Filing)

2. GE Spec No. 22A4046 - "Design Report Recirculation System Pipe Whip Restraint for BWR/4, 218, 251 Mark I and Mark II Product Line Plant." (Sections 1, 2 and 3 only).

3. "PDA, Pipe Dynamic Analysis, A Description of the Computer Code," General Electric Company, (NEDO 10813) (GE Class II document on file with USAEC).

TABLE 3.6-1

POSTULATED RUPTURE ORIENTATION AND NUMBER OF DESIGN-BASIS BREAKS USED IN PIPE RUPTURE ANALYSIS INSIDE CONTAINMENT

	NUMBER OF BREAKS	
SYSTEM	CIRCUMFERENTIAL	LONGITUDINAL
Main Steam	30	12
Feedwater	30	11
Reactor Recirculation	12	2
Residual Heat Removal	13	4
Reactor Core Isolation Cooling	7	2
Reactor Water Cleanup	6	0
High-Pressure Core Spray	1	0
Low-Pressure Core Spray	2	1
Standby Liquid Control	2	0
TOTAL	103	32

TABLE 3.6-2
(SHEET 1 of 3)

RECIRCULATION PIPING SYSTEM OPERATING STRESSES AT BREAK LOCATIONS*

BREAK ID**	JOINT NO.***	STRESS RATIO PER: EQ. (10) $\frac{Sn}{3Sm}$	EQ. (12) $\frac{Sc}{3Sm}$	EQ. (13) $\frac{S}{3Sm}$	USAGE FACTOR	BREAK TYPE	BREAK BASIS PARA. NUMBER
A. Recirculation Loop A							
S1	1I	1.01	0.13	0.59	0.0	Circ.	Terminal End 3.6.2.1.1.(a)
S^{6}_{c}, S^{6}_{L}	9I	1.36	0.42	0.70	0.39	Circ., long.	3.6.2.1.1.(b)(2)
D^{6}_{c}, D^{6}_{L}	36J	1.52	0.17	0.89	0.80	Circ., long.	3.6.2.1.1.(b)(2)
F2	60I	1.04	0.22	0.68	0.02	Circ.	3.6.2.1.1.(c)
F8	52I	1.36	0.15	0.73	0.63	Circ.	3.6.2.1.1.(b)(2)
D9	41I	0.42	0.02	0.31	0.0	Circ.	3.6.2.1.1.(c)
F19	68I	1.57	0.45	0.73	0.91	Circ.	3.6.2.1.1.(b)(2)
F25	76I	1.13	0.30	0.68	0.04	Circ.	3.6.2.1.1.(c)

*The value of $2.4S_m$ for Recirculation Piping Material is 39,960 psi.
** For Break ID see Figure 3.6-1.
*** For joint no., see Figure 3.6-1a.

TABLE 3.6-2*
(SHEET 2 of 3)

		STRESS RATIO PER;					
		EQ. (10)	EQ. (12)	EQ. (13)			
BREAK ID**	JOINT NO.***	Sn / 3Sm	Sc / 3Sm	S / 3Sm	USAGE FACTOR	BREAK TYPE	BREAK BASIS PARA. NUMBER
F6	64J	0.78	0.13	0.54	0.0	Circ.	Terminal End 3.6.2.1.1 (a)
F12	56J	0.90	0.23	0.54	0.0	Circ.	3.6.2.1.1 (a)
F17	48J	1.00	0.16	0.59	0.01	Circ.	3.6.2.1.1 (a)
F23	72J	0.84	0.16	0.56	0.0	Circ.	3.6.2.1.1 (a)
F29	80J	0.81	0.18	0.55	0.0	Circ.	3.6.2.1.1 (a)
B. Recirculation Loop B							
S1	1I	1.01	0.16	0.54	0.0	Circ.	Terminal End 3.6.2.1.1 (a)
D^6_c,D^6_L	36J	1.55	0.17	0.88	0.95	Circ. , long.	3.6.2.1.1 (b) (2)
F2	60I	1.13	0.33	0.69	0.03	Circ.	3.6.2.1.1(c)
F8	52I	1.49	0.33	0.75	0.72	Circ.	3.6.2.1.1(b) (2)

** For Break ID, see Figure 3.6-2.
*** For joint ID, see Figure 3.6-2a.

TABLE 3.6-2
(SHEET 3 of 3)

STRESS RATIO PER;

BREAK ID	JOINT NO.	EQ. (10) Sn / 3Sm	EQ. (12) Sc / 3Sm	EQ. (13) S / 3Sm	USAGE FACTOR	BREAK TYPE	BREAK BASIS PARA. NUMBER
D9	41I	0.42	0.02	0.31	0.0	Circ.	3.6.2.1.1(c)
F19	68I	1.46	0.42	0.71	0.39	Circ.	3.6.2.1.1(b)(2)
F25	76I	1.17	0.30	0.67	0.05	Circ.	3.6.2.1.1(c)
F6	64J	0.82	0.20	0.54	0.0	Circ.	Terminal End 3.6.2.1.1(a)
F12	56J	0.89	0.20	0.54	0.0	Circ.	3.6.2.1.1(a)
F17	48J	1.09	0.23	0.62	0.01	Circ.	3.6.2.1.1(a)
F23	72J	0.84	0.10	0.55	0.0	Circ.	3.6.2.1.1(a)
F29	80J	0.81	0.18	0.55	0.0	Circ.	3.6.2.1.1(a)

TABLE 3.6-3
(SHEET 1 OF 2)

MAIN STEAM SYSTEM - RESULTS OF DYNAMIC ANALYSIS FOR POSTULATED PIPE RUPTURE

Subsystem Number	Line Number	Restraint Number	Break Location	Blowdown Load Kips	Implingment Load	Break Type *
MS-02	1MS01AB-26	R127	C200	321.43	None Applicable	C
	2MS01AB-26	R60	C203A	48.2		L
		R62	C203	321.43		C
		R62	C204	432.7		L
		R64	C207	321.43		C
		R64	C208	321.43		L
		R65	C208	321.43		C
MS-01	1MS01AA-26	R128	C209	321.4		C
	2MS01AA-26	R68	C211	321.4		C
		R69	C212A	427.1		C
		R69	C212	459.2		L
		R71	C213	321.4		L
		R71	C214	321.4		C
		R72	C214	321.4		L
		R72	C214	227.3		C
		R72	C213	321.4		C
MS-03	1MS01AC-26	R129	C200	321.4		C
	2MS01AC-26	R76	C203A	48.2		L
		R78	C203	321.4		C
		R78	C204	450.0		L
		R80	C207	321.4		C
		R80	C208	321.4		L
		R81	C208	321.4		C
MS-04	1MS01AD-26	R130	C209	321.4		C
	2MS01AD-26	R84	C212A	427.0		C
		R84	C212	459.2		L
		R85	C211	321.4		C
		R87	C213	321.4		L
		R87	C214	321.4		C
		R88	C214	321.4		L
		R88	C214	227.3		C
		R88	C213	321.4		C

* C - Circumferential
L - Longitudinal
** - Unit 2 only

TABLE 3.6-3
(SHEET 2 OF 2)

Subsystem Number	Line Number	Restraint Number	Break Location	Blowdown Load Kips	Implingment Load	Break Type *
MS-02	1MS06A-10	R40	C112	50.1	None Applicable	C
(RCIC)	2MS06A-10	R41	C114	50.1		C
	1RI01A-10	R46	C121	54.3		C
	2RI01A-10	R46	C121	54.3		L
		R47	C122	59.3		C
		R48	C123	50.0		C
		R49	C126	50.0		C
		R50 **	C126	74.4		L
		R51	C128	35.4		C
MS-25	1MS14A-3	SR08	C7	6.4		C
	2MS14A-3	SR12	C19	1.93		C
	1MS14AA-2	SR03	C12	2.39		C
	2MS14AA-2	SR07	C6	2.2		C
	1MS14AB-2	SR04	C13	2.42		C
	2MS14AB-2	SR06	C5	2.2		C
	1MS14AC-2	SR01	C10	2.42		C
	2MS14AC-2	SR10	C9	2.2		C
	1MS14AD-2	SR02	C11	2.39		C
	2MS14AD-2	SR09	C8	2.2		C
	1RI09A-2	SR05	C4	2.2		C
	2RI09A-2	SR11	C53	1.65		C

* C - Circumferential
L - Longitudinal
** - Unit 2 only

TABLE 3.6-4
(SHEET 1 of 2)

FEEDWATER SYSTEM - RESULTS OF DYNAMIC ANALYSIS FOR POSTULATED PIPE RUPTURES

SUBSYSTEM	LINE NO.	RESTRAINT NUMBER	BREAK LOCATION	BREAK TYPE*	BLOWDOWN LOAD KIPS
FW-01	1FW02EC-12	R89	C215	C	75.7
	2FW02EC-12	R91	C217	C	110.5
	1FW02EB-12	R94	C220	C	80.4
	2FW02EB-12	R95	C221	C	78.6
		R96	C222	C	88.5
		R97	C228	L	175.8
	1FW02EA-12	R98	C224	C	93.
	2FW02EA-12	R100	C226	C	110.6
		R101	C227	C	96.1
		R101	C227A	L	147.8
	1FW02DA-18	R102	C223	L	175.8
	2FW02DA-18	R102	C228	L	175.8
		R103	C229	L	175.8
	1FW02CA-24	R105	C230	C	183.3
	2FW02CA-24	R105	C231	C	230.1
		R105	C231B	C	230.1
		R131	C227	L	175.8
		R131	C229	L	175.8
		R131	C230A	C	237.8
	1FW02FA-24	R106	C232	C	198.5
	2FW02FA-24	R106	C233	C	183.5
		R106	C233A	C	183.5
FW-02	1FW02ED-12	R108	C215	C	82.7
	2FW02ED-12	R110	C217	C	110.5
	1FW02EE-12	R113	C220	C	80.4
	2FW02EE-12	R113	C220	L	121.0
		R114	C221	C	78.6
		R115	C222	C	88.5
		R116	C228	L	175.8
	1FW02EF-12	R117	C224	C	93.0
		R119	C226	C	110.6
		R120	C227	C	95.9
		R120	C227A	L	147.8

* C - Circumferential
L - Longitudinal

TABLE 3.6-4
(SHEET 2 of 2)

FEEDWATER SYSTEM - RESULTS OF DYNAMIC ANALYSIS FOR POSTULATED PIPE RUPTURES

SUBSYSTEM	LINE NO.	RESTRAINT NUMBER	BREAK LOCATION	BREAK TYPE*	BLOWDOWN LOAD KIPS
FW-02 (Cont'd)	1FW02DB-18	R121	C223	L	175.8
	2FW02DB-18	R121	C228	L	175.8
		R122	C229	L	175.8
	1FW02CB-24	R124	C230	C	183.3
	2FW02CB-24	R124	C231	C	230.1
		R124	C231B	C	230.1
	1FW02FB-24	R125	C232	C	198.5
	2FW02FB-24	R125	C233	C	183.5
		R125	C232	L	287.4
		R132	C227	L	175.8
		R132	C229	L	175.8
		R132	C230A	C	237.8

*C - Circumferential
L - Longitudinal

TABLE 3.6-5
(SHEET 1 of 2)

RESULTS OF DYNAMIC ANALYSIS FOR POSTULATED PIPE RUPTURE
(Reactor Recirculation, Residual Heat Removal, Reactor Water Cleanup,
Reactor Core Isolation, Control Rod Drive, Standby Liquid Control,
High-Pressure Core Spray, and Low-Pressure Core Spray)

SUBSYSTEM	LINE NO.	RESTRAINT NUMBER	BREAK LOCATION	BREAK TYPE*	BLOWDOWN LOAD KIPS
RH-01	1RH03DA-12	R2	C69	L	108.2
	2RH03DA-12	R2	C70	C	102.
	1RH04A-20	R4	C28	C	42.9
	2RH04A-20	R4	C28A	L	327.8
	1RH04B-20	R5	C28A	C	273.2
	2RH04B-20				
	1RR07AA-12	R17	C67	C	3.5
	2RR07AA-12				
	1RH03CA-12	R18	C69	C	101.0
	2RH03CA-12				
	1RR01AA-24	R18A	CB18A	C	382.5
	2RR01AA-24				
	1RR02AA-24	R20A	CB20A	C	385.
	2RR02AA-24				
RH-02	1RR07AB-12	R19	C72	C	.64
	1RH03DB-12	R20	C73	C	101.0
	1RR07AB-12	R133	C73	C	106.0
	2RR07AB-12				
	1RR01AB-24	R18B	C18B	C	383.
	2RR01AB-24				
	1RR02AB-24	R20B	C20B	C	385.
	2RR02AB-24				
RH-04	1RH40BA-12	R25	C81	C	101.
	2RH40BA-12	R25	C82	L	120.1
		R26	C83	C	101.
		R26	C84	C	101.
		R26	C84	L	108.

* C - Circumferential
L - Longitudinal

TABLE 3.6-5
(SHEET 2 of 2)

RESULTS OF DYNAMIC ANALYSIS FOR POSTULATED PIPE RUPTURE
(Reactor Recirculation, Residual Heat Removal, Reactor Water Cleanup,
Reactor Core Isolation, Control Rod Drive, Standby Liquid Control,
High-Pressure Core Spray, and Low-Pressure Core Spray)

SUBSYSTEM	LINE NO.	RESTRAINT NUMBER	BREAK LOCATION	BREAK TYPE*	BLOWDOWN LOAD KIPS
RH-05	1RH40BB-12	R29	C85	C	101.
	2RH40BB-12	R29	C86	L	120.1
		R30	C87	C	101.
		R31	C88	C	101.
RH-06	1RH53B-12	R32	C89	C	101.
	2RH53B-12	R33	C92	C	76.4
		R34	C91	C	101.
		R34	C92	C	101.
		R34	C92	L	108.
RH-07	1RHB4AA-2	SR29	C2	C	2.59
	2RHB4AA-2	SR30	C3	C	2.35
RH-71	1RHB4AB-2	SR13	C25	C	2.59
	2RHB4AB-2	SR14	C35	C	2.35
RR-01	1RT01B-6	R10	C45	C	24.96
	2RT01B-6	R14	C59	C	16.72
		R15	C61	C	24.96
	1RT17A-4	R35	C95	C	10.83
	2RT17A-4	R37	C99	L	12.9
		R37	C100	C	10.83
SC-02	1SC02B-1½	SR15	C1	C	1.86
	2SC02B-1½	SR16	C2	C	1.86
HP-01	1HP02B-12	R24	C79	C	1.6
	2HP02B-12				
LP-01	1LP02B-12	R21	C75	C	101.0
	2LP02B-12	R21	C76	L	121.1
		R23	C76	C	101.1

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* C - Circumferential
L - Longitudinal

TABLE 3.6-6

MATERIAL PROPERTIES OF PIPE WHIP RESTRAINTS

COMPONENT	MATERIAL	STRESS/STRAIN PROPERTIES	MINIMUM ULTIMATE RESTRAINT LOADS	STRAIN ε MAXIMUM CALCULATED	STRAIN ε DESIGN LIMIT
Embedment plates	A588 Gr. 50	$\sigma \geq 50$ ksi static		.00164	.00172
Tension legs and bolts	A588 Gr. 50	$\sigma \geq 50$ ksi, $u \geq .18$ static σ Dynamic ≥ 57.51 ksi		.07 to .08	$\leq .09 \cong .5\varepsilon$
Compression legs	A36	$\sigma \geq 36$ ksi	$P_U = P_B =$ 5/3xFxDIFxA		
	A588 Gr. 50	$\sigma \geq 50$ ksi			
Bearing plates	A36 A588 Gr. 50	$\sigma \geq 36$ ksi $\sigma \geq 50$ ksi		.00118 .00164	.00124 .00172
Honeycomb		$\sigma = 6$ ksi, $u \geq .70$		.50000	.50000
Facing plates	A572 Gr. 50	$\sigma \geq 50$ ksi		.00164	.00172
Ring plate	A588 Gr. 50	$\sigma \geq 50$ ksi, $u \geq .18$		.00164	.00172
Welds	E 70 xx	$\sigma \geq 60$ ksi			

TABLE 3.6-7
(SHEET 1 OF 2)

SUMMARY OF RESTRAINT DATA FOR RECIRCULATION PIPING

									STRAIN OR LOAD			
MATERIAL			YIELD AND ULTIMATE STRESS		ULTIMATE STRAIN		MINIMUM ULTIMATE LOAD		CALCULATED		DESIGN	
PIPE SIZE (in)	FRAME	WIRE ROPE	FRAME	WIRE ROPE	FRAME	WIRE ROPE	FRAME AT ULTIMATE STRAIN (kips)	WIRE ROPE (kips)	FRAME	WIRE ROPE	FRAME	WIRE ROPE
24.00	ASTM A36	Carbon steel (plow steel)	σ_y = 36,000 psi σ_u = 58,000 psi	See Note 2.	Design strain equals 50% of ultimate strain. ε_u = 20%	See Note 2.	1229	788	5.74%	709	10%	709
16.00	ASTM A36	Carbon steel (plow steel)	σ_y = 36,000 psi σ_u = 58,000 psi	See Note 2.	Design strain equals 50% of ultimate strain. ε_u = 20%	See Note 2.	731	572	7.41%	490	10%	515
12.75	ASTM A36	Carbon steel (plow steel)	σ_y = 36,000 psi σ_u = 58,000 psi	See Note 2.	Design strain equals 50% of ultimate strain. ε_u = 20%	See Note 2.	644	300	0.94%	253	10%	270

TABLE 3.6-7
(SHEET 2 OF 2)

SUMMARY OF RESTRAINT DATA FOR RECIRCULATION PIPING

NOTES

1. Dynamic stress-strain data were not used in the analysis. All calculations are based on static stress-strain data, which is more conservative.

2. Wire rope design criteria are specified with the breaking strength, therefore the stress-strain data are not applicable.

3. The strain data are applicable to the frame; the load data are applicable to the wire rope.
   a. Design strain for the frame is equal to 50% of ultimate strain.
   b. Design load for the wire rope is equal to 90% of minimum ultimate load.

TABLE 3.6-8
(SHEET 1 OF 6)

CLASS 1, 2, and 3 PIPING STRESS INTENSITY RANGE AND USAGE FACTORS

SUBSYSTEM	RESTRAINT	BREAK	BREAK TYPE *	NODE NUMBER	2.4 S_m	ASME STRESSES PSI Eq. 10	Eq. 12	Eq. 13	USAGE FACTOR	REMARKS
FW-01	R-89	C-215	C	10A	47340	36056			.006	
	R-91	C-217	C	20A	47340	25356			.004	
	R-94	C-220	C	55A	47340	46937			.017	
	R-95	C-221	C	65B	47340	24184			.003	
	R-96	C-222	C	65A	47340	24508			.003	
	R-97	C-228	L	45	47340	61827	34126	31506	.289	Works L/R-102
	R-98	C-224	C	105A	47340	40599			.012	
	R-100	C-226	C	115A	47340	61240	43287	20422	.039	
	R-101	C-227	C	120B	47340	79306	58581	26242	.152	
	R-101	C-227A	L	120A	47340	63257	43947	24479	.042	
	R-102	C-223	L	45	47340	61827	34126	31506	.289	
	R-102	C-228	L	45	47340	61827	34126	31506	.289	
	R-103	C-229	L	95	47340	71946	39435	34338	.338	
	R-105	C-230	C	125A	47340	27178			.011	
	R-105	C-231	C	130B	47340	29651			.014	
	R-105	C-231B	C	136	47340	50981	3424	39066	.151	
	R-106	C-232	C	140B	47340	22100			.014	
	R-106	C-233	C	140A	47340	25021			.017	
	R-106	C-233A	C	137	47340	45698			.083	
	R-131	C-227	L	120B	47340	79306	58581	26242	.152	
	R-131	C-229	L	95	47340	71946	39435	34338	.338	
	R-131	C-230A	C	125B	47340	28028			.011	
FW-02	R-108	C-215	C	10A	47340	34394			.006	
	R-110	C-217	C	20A	47340	27220			.004	
	R-113	C-220	C	55A	47340	53901	27293	27371	.018	
	R-113	C-220	L							
	R-114	C-221	C	65B	47340	46039			.005	
	R-115	C-222	C	65A	47340	44424			.004	
	R-116	C-228	L	45	47340	83420	34211	47669	.313	Works W/R-121
	R-117	C-224	C	105A	47340	47223			.015	
	R-119	C-226	C	115A	47340	64704	43036	24823	.039	
	R-120	C-227	C	120B	47340	87435	58538	34832	.163	
	R-120	C-227A	L	120A	47340	64446	44473	25671	.042	
	R-121	C-233	L	45	47340	83420	34211	47669	.313	
	R-121	C-228	L							

* C Indicates Circumferential
L Indicates Longitudinal

TABLE 3.6-8
(SHEET 2 OF 6)

SUBSYSTEM	RESTRAINT	BREAK	BREAK TYPE *	NODE NUMBER	2.4 S_m	ASME STRESSES PSI Eq. 10	Eq. 12	Eq. 13	USAGE FACTOR	REMARKS
FW-02 (Cont'd)	R-122	C-229	L	95	47340	82014	39750	41582	.339	
	R-124	C-230	C	125A	47340	32726			.012	
	R-124	C-231	C	130B	47340	34875			.013	
		C-231B	C	136	47340	59623	3444	43540	.017	
	R-125	C-232	C	140B	4730	29712			.014	
	R-125	C-233	C	140A	47340	27718			.107	
	R-125	C-233A	C	137	47340	51347	3233	37428	.090	
	R-132	C-230A	C	125B	47340	36778			.012	
	R-132	C-227	L	120B	47340	87435	58538	34382	.163	
	R-132	C-229	L	95	47340	82014	39750	41582	.399	
MS-01	R-68	C-211	C	275B	40802	16984			.015	
	R-69	C-212A	C	35A	42480	39292			.015	
	R-69	C-212	L	35B	42480	42271			.016	
	R-71	C-214	C	25A	42480	41974			.014	
	R-71	C-213	L	25B	42480	43099	20423	31085	.015	
	R-72	C-213	C	25B	42480	43099	20423	31085	.015	
	R-72	C-214	C	25A	42480	41974			.014	
	R-72	C-214	L							
	R-128	C-209	C	280B	42480	40819			.015	
MS-02	R-60	C-203A	L	180T	44480	44488	6715	46316	.03	
	R-62	C-203	C	40B	42480	41315			.017	
	R-62	C-204	L	40A	42480	42449			.017	
	R-64	C-207	C	25A	42480	41189			.041	
	R-64	C-208	L	25B	42480	42271			.015	
	R-65	C-208	C	25B	42480	42271			.015	
	R-127	C-200	C	405B	42480	39397			.012	
MS-03	R-76	C-203A	L	110T	43680	47989	10028	49585	.034	
	R-78	C-203	C	40B	42480	38997			.015	
	R-78	C-204	L	40A	42480	40860			.016	
	R-80	C-207	C	25A	42480	40449			.012	
	R-80	C-208	L	25B	42480	41587			.013	
	R-81	C-208	C	25B	42480	41587			.013	
	R-129	C-200	C	405B	42480	40512			.015	

* C Indicates Circumferential
L Indicates Longitudinal

TABLE 3.6-8
(SHEET 3 OF 6)

SUBSYSTEM	RESTRAINT	BREAK	BREAK TYPE *	NODE NUMBER	2.4 S_m	ASME STRESSES PSI Eq. 10	Eq. 12	Eq. 13	USAGE FACTOR	REMARKS
MS-04	R-84	C-212A	C	35A	42480	42208			.017	
		C-212	L	35B	42480	39703			.016	
	R-85	C-211	C	275B	42480	41814			.017	
	R-87	C-214	C	25A	42420	40294			.013	
	R-87	C-213	L	25B	42480	43510	20018	32898	.015	
	R-88	C-213	C	25B	42480	43510	20018	32898	.015	
	R-88	C-214	C	25A	42480	40294			.013	
	R-88	C-214	L							
	R-130	C-209	C	280B	42480	41178			.014	
MS-02 (RCIC)	R-40	C-112	C	470B	42552	40970			.011	
	R-41	C-114	C	465B	42552	54998	10501	45328	.119	
	R-46	C-121	C	445A	42552	41831			.007	
	R-46	C-121	L							
	R-47	C-122	C	430B	42552	35498			.005	
	R-48	C-123	C	430A	42552	29646			.005	
	R-49	C-126	C	425A	42552	29379			.005	
	R-50	C-126	L	425A	42552	29379			.005	
	R-51	C-128	C	420A	42552	30242			.005	
HP-01	R-24	C-79	C	16B	42480	26496			.000	
LP-01	R-21	C-75	C	15A	42480	36817			.000	
	R-21	C-76	L							
	R-23	C-76	C	15B	42480	35261			.003	
RH-01	R-2	C-70	C	45A	43368	60076	239	37740	.067	
	R-2	C-69	L	45B	40512	24507			.001	
	R-4	C-28	C	320B	40512	49813			.015	
	R-4	C-28A	L	320A	43368	73501	19016	40691	.230	
	R-5	C-28A	C	320A	43368	73501	19016	40691	.230	
	R-17	C-67	C	65B	40512	26545			.003	
	R-18	C-69	C	45B	40512	24507			.001	
	R-18A	C-B18A	C	165B	40008	31069			.009	
	R-20A	C-B20A	C	355B	40008	30216			.000	
RH-02	R-19	C-72	C	A64	40512	40960			.056	
	R-20	C-73	C	65A	40512	20041			.001	
	R-133	C-73	C							
	R-18B	C-18B	C	165B	40008	20096			.000	
	R-20B	C-20B	C	330B	40008	30831			.088	

* C Indicates Circumferential
L Indicates Longitudinal

TABLE 3.6-8
(SHEET 4 OF 6)

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*C Indicates Circumferential
L Indicates Longitudinal

TABLE 3.6-8
(SHEET 5 OF 6)

SUBSYSTEM	RESTRAINT	BREAK	BREAK TYPE *	NODE NUMBER	2.4 S_m	ASME STRESSES PSI Eq. 10	Eq. 12	Eq. 13	USAGE FACTOR	REMARKS
RH-04	R-25	C-81	C	95B	42480	28220			.012	
	R-25	C-82	L	95A	42480	24958			.006	
	R-26	C-83	C	85B	42480	23108			.003	
	R-26	C-84	L	80B	42480	21970			.003	
	R-26	C-84	C							
RH-05	R-29	C-85	C	20A	42480	27822			.000	
	R-29	C-86	L	20B	42480	24212			.000	
	R-30	C-87	C	30A	42480	34859			.000	
	R-31	C-88	C	45	42480	23685			.000	
RH-06	R-32	C-89	C	20A	42480	31533			.000	
	R-33	C-92	C	30B	42480	25994			.000	
	R-34	C-91	C	30A	42480	32727			.000	
	R-34	C-92	C	30B	42480	25994			.000	
	R-34	C-92	L							
RR-01	R-10	C-45	C	145A	42552	22925			.000	
	R-14	C-59	C	5	42552	41025			.144	
	R-15	C-61	C	20B	42552	28023			.002	
	R-35	C-95	C	245A	42552	23303			.007	
	R-36	C-97	C	255A	42552	16666			.004	
	R-37	C-100	C	270B	42552	27670			.008	
MS-25	SR-01	C-10	C	305	36900	30822			.043	
	SR-02	C-11	C	245	36900	32608			.053	
	SR-03	C-12	C	195	36900	29186			.037	
	SR-04	C-13	C	140	36900	31394			.053	
	SR-05	C-4	C	40	36900	23890			.002	
	SR-06	C-5	C	50	36900	39415			.009	
	SR-07	C-6	C	65	36900	17625			.001	
	SR-08	C-7	C	5	36900	34017			.014	
	SR-09	C-8	C	80	36900	37555			.010	
	SR-10	C-9	C	95	36900	53045	32429	26897	.034	
	SR-11	C-53	C	530	36900	37965			.038	
	SR-12	C-9	C	95	36900	37555			.010	
RH-70	SR-29	C2	C	35	48000	57782			.736	
	SR-30	C1	C	5	48000	57696			.705	
	SR-30	C3	C	70	48000	59594			.755	

* C Indicates Circumferential
L Indicates Longitudinal

TABLE 3.6-8
(SHEET 6 OF 6)

SUBSYSTEM	RESTRAINT	BREAK	BREAK TYPE*	NODE NUMBER	2.4 S_m	ASME STRESSES PSI Eq. 10	Eq. 12	Eq. 13	USAGE FACTOR	REMARKS
RH-71	SR-13	C25	C	25	48000	57860			.727	
	SR-14	C20	C	20	48000	57447			.689	
		C35	C	35	48000	39981			.137	
SC-02	SR-15	C1	C	20	40560	7258			.000	
	SR-16	C2	C	82	40560	22008			.000	

* C Indicates Circumferential
L Indicates Longitudinal

3.7 SEISMIC DESIGN

Safety-related structures, systems, and components that are designed to remain functional in the event of a safe shutdown earthquake (SSE) are designated as Seismic Category I. Seismic Category I items are analyzed and designed through the use of appropriate methods of dynamic analysis as described in the following subsections.

3.7.1 Seismic Input

3.7.1.1 Design Response Spectra

The site response spectra which are defined at the free field foundation level for the SSE and the operating basis earthquake (OBE) are presented in Subsection 2.5.2 and are shown in Figures 2.5-39 and 2.5-40. The maximum horizontal ground acceleration at the free field foundation level, corresponding to above site response spectra, is 20% gravity for SSE and 10% gravity for OBE. Vertical response spectra used are 2/3 of the horizontal response spectra. Earthquake history, site geology, and seismology are discussed in Section 2.5.

3.7.1.2 Design Time History

In the design of the station, time-history response analyses are used to determine the seismic environment in which internal equipment systems and components must be designed to function. The site response spectra cannot be used directly as the seismic load in the time-history analysis; rather, equivalent time-history forcing functions are used as the seismic load.

Spectrum compatible time history is obtained by modifying an actual earthquake time-history record in such a way that its response spectrum matches closely with the given OBE spectrum. The matching of the response spectrum is done such that the points which are higher are suppressed first. To suppress the response spectrum, the selected time-history motion is passed through a two parameter frequency-suppression filter. The first parameter is a damping parameter that mainly controls the amount of suppression at the given period, and the second parameter controls the band width of suppression. These two parameters are adjusted such that the desired suppression effect is obtained at a given period. After that, raising of response spectrum at required periods is done by adding sine waves of appropriate amplitude and phase lag (Reference 1). Figures 3.7-1 and 3.7-2 illustrate the horizontal synthetic time histories in both N-S and E-W directions. These two synthetic time histories are statistically independent. The vertical synthetic time history is taken from the horizontal E-W synthetic time history with a 1/3 overall reduction in acceleration.

Modified 1940 El Centro earthquake records for N-S and E-W components are used for these compatible time-history forcing functions. Compatibility is verified by generating response spectra for 2% and 5% damping ratios as shown in Figures 3.7-3 through 3.7-6. In generating these spectra, 72 period intervals from 0.02 to 2.0 seconds are considered. The period intervals at which the response spectra are calculated are as follows:

Period Range (sec)	Increment (sec)
0.02 - 0.1	0.005
0.1 - 0.4	0.01
0.4 - 0.5	0.02
0.5 - 1.0	0.05
1.0 - 2.0	0.1

3.7.1.3 Critical Damping Values

Viscous damping is used to simulate energy dissipation in the dynamic models. Damping values (expressed as a percentage of critical damping) which are used are listed in Table 3.7-1.

Some piping systems have been analyzed using the damping values contained in ASME Code Case N-411. The Code Case will be used in new piping and equipment dynamic analyses, and in reanalyses for support reconciliation work, support optimization and piping evaluation.

LaSalle County Station, Units 1 & 2, received NRC approval to use ASME Code Case N-411 in a NRC letter from E.G. Adensam (Director, BWR Project Directorate No. 3, Division of BWR Licensing) to D. L. Farrar (Director of Nuclear Licensing-CECo) dated April 1, 1986. The NRC clarified restrictions on the use of ASME Code Case N-411 in a letter from E. G. Adensam to D. L. Farrar dated July 18, 1986.

ASME Code Case N-411 allows the use of the alternative damping values recommended by the Pressure Vessel Research Council in Welding Research Council Bulletin 300. The following paragraphs, extracted from the NRC letters dated April 1, 1986 and July 18, 1986, describe the restrictions on the use of Code Case N-411.

Code Case N-411 is a conditionally acceptable Code Case by the staff. Its use has been approved by the staff for specific plant applications, pending a revision of Regulatory Guide 1.61, if the following information and commitments have been provided:

1) The application of the Code Case shall be limited to piping systems analyzed by the response spectrum method only.

2) The alternate damping criteria of the Code Case shall be used in their entirety in any given analysis. The mixed application of the Code Case and Regulatory Guide 1.61 is not permitted.

3) Due to the increased flexibility of the system, the user shall check all recalculated displacements to verify there is adequate clearance between the piping system and adjacent structures, components, and equipment, and to verify the ability of mounted equipment to withstand the increased motion.

4) The user shall clearly indicate whether the Code Case will be used for new analyses, for reconciliation work, or for support optimization.

A Commonwealth Edison letter from M. S. Turback to H. R. Denton dated February 11, 1986 made the necessary commitments, identified above, regarding the application of Code Case N-411. It also stated that the Code Case will be used in new piping and equipment dynamic analyses, and in reanalyses for support reconciliation work, support optimization, and piping evaluation.

By letter dated April 1, 1986, the staff approved the use of ASME Code Case N-411 for LaSalle. In that letter, it is stated that Commonwealth Edison agreed to review any analysis using Code Case N-411 damping on the recirculation system with the staff prior to using the results for snubber reduction purposes. The staff required that commitment from Commonwealth Edison because the recirculation pump is modelled in that piping system and the damping values identified in the Code Case are applicable only to piping. The damping values specified in Regulatory Guide 1.61 should be used for equipment. However, the Code Case damping values may be used to analyze piping systems in which such equipment is included in the stress analysis model, provided the analysis results show that the equipment has no responses below 20 hertz. Furthermore, the Code Case damping values are applicable only to piping for which current seismic spectra and analysis methods are used. This includes model and direction combination, of all three earthquake directions, in accordance with Regulatory Guide 1.92, use of enveloped response spectra, and consideration of a sufficient number of modes such that inclusion of additional modes does not result in more than a 10% increase in response, i.e., do not omit consideration of the load contribution of piping dynamic modes with natural frequencies above 33 hertz.

Based on the above, the staff concludes it is acceptable to use ASME Code Case N-411 for LaSalle County Station, Units 1 and 2 in the manner, and within the limitations, described above. Finally, this approval to use the Code Case damping values does not include application to piping analyses when energy absorbers are included in the stress analysis model

3.7.1.4 Supporting Media for Seismic Category I Structures

The description of the supporting media for Seismic Category I structures are presented and discussed in Section 2.5 and soil properties such as shear wave velocity, modulus of elasticity, and compression wave velocity density are given in Table 2.5-28 and Figure 2.5-55.

The following is a list of Seismic Category I structures with the embedment depth, the depth of soil between bedrock and foundation, the foundation width, and the structural height.

Structure	Embedment Depth (ft)	Depth of Soil (ft)	Foundation Width (ft) Direction N-S	Foundation Width (ft) Direction E-W	Structural Height Above Grade (ft)
Reactor Building	44	126	310	137	183
Auxiliary Building	54	116	310	73	133
Diesel-Generator Building	41	129	68	100	38

3.7.2 Seismic System Analysis

3.7.2.1 Seismic Analysis Methods

The calculation of the dynamic response of the nuclear power station complex subjected to an earthquake loading is divided into two broad categories. The first category is the analysis of the major buildings and structures which house and/or support Seismic Category I systems and components. The second is the analysis of Seismic Category I subsystems and components supported by Seismic Category I buildings or structures.

Major seismic systems such as buildings, the containment, and the reactor pressure vessel are modeled and analyzed. The motion of major structures, obtained from their analysis, is then used as the forcing function in the dynamic analysis of smaller Seismic Category I systems and components.

The analysis of models, which uses the response spectra, or an equivalent time-history motion obtained from the soil structure interaction analysis described in Subsection 3.7.2.4 is referred to as the system analysis. An analysis which uses the response spectra derived from the system analysis as the seismic load is referred to as a subsystem analysis.

3.7.2.1.1 Analysis of Building Structure Systems

To determine the exact dynamic forces acting on a structure, the accelerations (and, therefore, the displacements) of every mass particle must be evaluated. Since any real structure's mass is distributed over the spatial extent of the structure, an infinite number of coordinates is required to describe the motion of every mass particle when the structure is subjected to a dynamic load. Calculation of time-dependent displacements at every point in a complex structure is impossible, but the analysis can be simplified by the judicious selection of a limited number of displacement components or coordinates. In dynamic structural analysis, two different assumptions are used to specify the deflected shape of a structure. These are referred to as the lumped-mass approach and the distributed-coordinate approach. The lumped-mass approach is the most convenient and versatile method to use in analyzing the complex structural configurations which arise in a nuclear power station; this approach is used in the seismic analysis of station structures.

In the lumped-mass idealization, it is assumed that the entire mass of the structure is concentrated at a number of discrete points and the structural elements are assumed to have linear elastic properties. A six degree-of-freedom lumped mass is general, in the sense that the discrete mass possesses all possible degrees of freedom.

The computer program, DYNAS (Dynamic Analysis of Structures), is used to analyze Seismic Category I building structures. The description of this program is presented in Appendix F.

The time-history method of analysis is used to generate time-history motions which are used to generate response spectra for subsystem analysis.

3.7.2.1.1.1 Horizontal Seismic Analysis

The site response spectra presented in Subsection 3.7.1.1 are interpreted as one horizontal component of the earthquake. The effect of soil amplification is included in the response spectra, but the effect of soil-structure interaction is considered as described in Subsection 3.7.2.4 for the soil-supported structures.

Simultaneous action of several components of horizontal ground-motion are considered by analyzing the dynamic models for simultaneous excitations parallel to a model's two orthogonal axes. Analyses for simultaneous excitation parallel to a model's two orthogonal axes are accomplished by:

a. Response spectrum method of analysis - is used to generate the design forces as follows:

   1. analyze the model for X-excitation (N-S direction),

   2. analyze the model for Y-excitation (E-W direction), and

   3. combine the results from 1 and 2 as described in Subsection 3.7.2.6.

b. Time-history method of analysis - is used to generate response spectra for subsystem analysis as follows:

   1. analyze the model by X- and Y-excitation by applying two statistically independent time-histories simultaneously, and

   2. combine the responses algebraically at each time step.

3.7.2.1.1.1.1 <u>Shear Structure System</u>

The station building structures are complex systems, asymmetric in plan, with heavy concrete slabs at the various floor elevations. These slabs are interconnected with numerous concrete shear walls and/or heavy cross-braced steel members. The overall height dimensions are smaller than the plan dimensions. This low height-to-plan ratio indicates that under lateral loads, the predominant deformations of the long shear walls are shear deformations. Consequently, the relative rotations of the slabs about horizontal axes do not cause significant deformations; but, due to the asymmetrical mass-stiffness distribution, rotation of the slabs about a vertical axis occurs when this type of structure is subjected to lateral loads. Since the predominant deformation of this type of structure under horizontal seismic loading is a horizontal shear deformation of the walls, it is referred to as a shear structure system.

Figure 3.7-7 shows a simplified shear structure system and the x-y-z axis system where the z-axis is vertical and the x- and y-axes are parallel to the principal axes of the structure. The significant deformations of the structure under horizontal seismic excitation are described with three coordinates, Δ_x, Δ_y, and θ_z. These three degrees-of-freedom describe the motion of a concrete slab. Neglect of the θ_x, θ_y, and Δ_z degrees-of-freedom implies that the slab mass moves in a horizontal plane.

In discussing the shear structure system model, the words "model slab" are substituted for the words "lumped mass" because the mass of the actual structure is simulated in the model with slabs located at the elevations of the major floor slabs and roof of the structure. The mass of the walls between two floors is lumped to the floors to which they are connected.

The mass of equipment supported on slabs in the structure is included in the mass of the slabs. The actual slabs are considered to be infinitely rigid in their own planes. The rigid body motions of the model slabs consist of three degrees-of-freedom: horizontal translation in two perpendicular directions and rotation about the vertical axis. The model slabs are interconnected by weightless elastic springs which possess stiffness in the x- or y-direction and simulate the shear walls and vertical bracing in the structure. These springs are distributed horizontally on the model slabs to simulate the torsional stiffness interconnecting the two slabs. A typical plot of the walls is given in Figure 3.7-8. Each shear wall is identified by number.

Three coordinates are required to describe the motion of each model slab. Therefore, three mass parameters are determined for each model slab. These mass parameters for the i-th model slab are:

a. M_{xi}, associated with x-translation,

b. M_{yi}, associated with y-translation, and

c. $I_{\theta i}$, associated with rotation about vertical axis,

where the mass parameters associated with x-translation and y-translation are the same and are equal to the mass of the slab. The mass polar moment of inertia, I_θ, is about the vertical axis through the centroid of the slab.

To evaluate the stiffness of the structural components which interconnect slabs, the following assumptions are made:

a. all points on the same slab translate in the horizontal plane passing through the mass-center of the slab and the slab rotates only about the vertical axis, and

b. only in their longitudinal direction, the walls offer resistance to relative displacement between slabs.

When resisting lateral loads applied parallel to the long dimension, most walls act as short, deep beams; therefore, the contribution of shear to the deflection is considered in calculating the stiffness of a wall. The stiffness (K) of an individual wall is calculated using the following formulae:

$$K = \frac{1}{\Delta} \qquad (3.7\text{-}1)$$

$$\Delta = \frac{Fh}{GA} + \frac{h^3}{12EI} \qquad (3.7\text{-}2)$$

where:

Δ = deflection of the wall due to a unit force,

F = shear form factor,

A = cross-sectional area of the wall,

G = shear modulus of concrete,

h = height of the wall,

E = modulus of elasticity for concrete, and

I = moment of inertia of the wall.

The stiffness of steel framing which acts as springs is evaluated with conventional elastic frame and truss analysis computer programs STRESS and STRUDL II (Appendix F).

Dynamic analysis of the shear structure system is accomplished with the computer program, DYNAS. The input to DYNAS is prepared by using the program, SSANA (Spring Slab Analysis). The description and analytical details of programs SSANA and DYNAS are given in Appendix F.

3.7.2.1.1.1.2 Frame Structure System

In the shear structure system discussed in the previous subsection, the motion of the structure's mass is restricted to horizontal translations and rotation about the vertical axis. For many structural systems under dynamic loading, motions are not restricted to a horizontal plane, and all six possible degrees-of-freedom of the discrete masses are required to describe the dynamic behavior of the structure. Dynamic analysis of this type of structure is accomplished by the program DYNAS.

3.7.2.1.1.1.3 Combined Shear-Frame Structure System

The shear-type structures with three degrees-of-freedom for each slab mass (Subsection 3.7.2.1.1.1.1) and the frame-type structures with six degrees-of-freedom for each mass (Subsection 3.7.2.1.1.1.2) could both be present in a building system. The analysis of a coupled shear-frame structure is performed by the DYNAS program. Rigid or flexible frame members are used to connect the joints of the frame members to the slab centroids where interconnections exist.

Figure 3.7-9 shows the coupled shear-frame structure model used in the analysis. Joints 1 through 17 represent the shear structure system, (Subsection 3.7.2.1.1.1.1) and joints 18 through 51, the frame structure system, (Subsection 3.7.2.1.1.1.2). The two systems are connected by rigid members at different elevations.

3.7.2.1.1.2 Vertical Seismic Analysis

3.7.2.1.1.2.1 Vertical Seismic Loading

The seismic input is discussed in Subsection 3.7.1. The effect of soil structure interaction is considered as described in Subsection 3.7.2.4.

3.7.2.1.1.2.2 Modeling Technique

The dynamic behavior of a building in the vertical direction is a function of the wall axial stiffness, the floor system flexural stiffness, and the mass distribution. Figure 3.7-10 shows the plane frame model which simulates the building's dynamic characteristics in the vertical direction. The vertical members in the model simulate the axial stiffness of the walls and the horizontal members simulate the flexural stiffness of the floor systems. Although only two wall systems are shown in Figure 3.7-10, any number of wall systems can be incorporated in an analysis depending on the layout of the structure to be analyzed.

In the dynamic model, the masses can displace relative to one another with one degree-of-freedom in the vertical direction. The lumped-mass approach is used, and the wall masses are lumped equally to the nearest joint. In the floor system, part of the actual structure's mass moves with the wall, whereas part of the mass motion is amplified because of slab flexibility. Hence the floor mass is distributed between the wall joints and the slab joint shown at the center of a horizontal member (Figure 3.7-10).

Each floor system has several natural periods of vibration within the threshold of rigidity which must be considered in the vertical analysis. Several single degree-of-freedom systems are connected between the wall systems (Figure 3.7-11) and the mass-stiffness properties of these systems simulate the multiperiod characteristics of the complex floor system.

3.7.2.1.1.2.3 Analysis Procedure

As in the horizontal analysis, both response spectrum method of analysis and time-history analysis are performed on the vertical model as described above using DYNAS program. The forces obtained from the response spectrum method of analysis are used to determine seismic forces in different structural elements. The time-history analysis yields response spectra at wall and slab joints which are used as input for the design of various subsystems.

3.7.2.1.1.3 Differential Seismic Movement of Interconnected Supports

When a component is deformed due to the differential movement of floor or other major elements of a building, the deformed component is designed to remain capable of performing its Seismic Category I functions during and after such deformations. The effects of differential movements of interconnected components due to seismic disturbance are considered in the seismic analysis of piping systems and components as presented under Subsection 3.7.3.

3.7.2.2 Natural Frequencies and Response Loads

3.7.2.2.1 Horizontal Analysis

The periods, mode shapes, and the dynamic response of the lumped mass system (Figure 3.7-9) are computed. Table 3.7-2 presents the summary of the first 30 modal periods, the modal participation factors for x-direction excitation, and the modal participation factors for the y-direction excitation. Table 3.7-3 summarizes the displacements of slabs for OBE and SSE.

The shear force and the moment diagrams of the containment for OBE are shown in Reference 12. The forces and moments correspond to the global coordinate system as indicated by solid and dashed lines at the bottom of the diagrams.

The horizontal response spectra for the reactor-auxiliary building complex at elevations 694 feet 6 inches and 843 feet 6 inches, and for the reactor containment at elevation 786 feet 6 inches, are presented in Figures 3.7-14 through 3.7-25.

3.7.2.2.2 Vertical Analysis

The vertical model shown in Figure 3.7-10 has been analyzed by the DYNAS program. Table 3.7-4 lists the periods and the modal participation factors.

The main object of the vertical analysis is to generate response spectra for the design of Seismic Category I equipment located at different floor levels. However, the forces in the structure are also determined by response spectrum method of analysis. The slabs and shear walls of the reactor building, the auxiliary building, the reactor containment shield, and all other Seismic Category I structures are designed to withstand these forces due to vertical excitation.

The vertical response spectra for the reactor-auxiliary building complex at elevation 694 feet 6 inches, elevation 843 feet 6 inches, and for the reactor containment at elevation 786 feet 6 inches are presented in Figures 3.7-26 through 3.7-35.

3.7.2.3 Procedure Used for Modeling

3.7.2.3.1 Structural Modeling

3.7.2.3.1.1 Horizontal Analysis

Since each slab in the model has three degrees-of-freedom, three mass parameters are associated with each slab. The mass parameter associated with x-translation and y-translation is the same and equal to the mass of a slab; the mass parameter associated with θ_z is the mass polar moment of inertia of a slab about a vertical axis through its centroid.

The masses and mass polar moments of inertia for slabs are based on the mass distribution of the slab, equipment locations and equipment masses, and tributary wall masses at the wall locations.

A more detailed description of horizontal modeling is given in Subsection 3.7.2.1.1.1.

3.7.2.3.1.2 Vertical Analysis

In the dynamic model formulated for the vertical analysis, the masses can displace, relative to one another, with one degree-of-freedom in the vertical direction. The mass parameters are calculated in the following manner:

a. The masses are concentrated at joints (as shown in Figure 3.7-10) and interconnected by weightless linear springs which simulate the stiffness of the slabs or walls.

b. In general, the wall masses are lumped equally to the nearest joints.

c. For the slabs, it is assumed that one-third of the total slab mass is effective; the remaining mass of the slab is lumped with the wall mass at that elevation.

d. The mass of the reactor containment shield includes only the mass of concrete and contributory slab mass.

A more detailed description of vertical modeling is given in Subsection 3.7.2.1.1.2.

3.7.2.3.2 Modeling Techniques for Seismic Category I Structures, Systems, and Components

The modeling techniques for Seismic Category I structures, systems, and components are discussed in Subsection 3.7.3.14.6.

3.7.2.4 Soil-Structure Interaction

3.7.2.4.1 Horizontal Excitation

A finite element approach is used to account for the effect of the soil-structure interaction. The criteria used in modeling and the general procedure are the same as described in Report SL-3026 (Reference 2).

Strain dependent soil parameters used in the interaction study are presented in Table 2.5-28 and Figure 2.5-55. The SHAKE program described in Appendix F is used to obtain strain compatible shear modulus and damping values for each layer. The corresponding compatible rock motions are also obtained in each case. The two dimensional finite element soil model is shown in Figure 3.7-39. The DAPS program described in Appendix F is used to extract normalized modes from this model.

Using modal synthesis technique, the three-dimensional building model described in Subsection 3.7.2.1.1.1 is analyzed using the DYNAS program for the two postulated earthquake loadings, OBE and SSE. One discrete torsional soil spring and corresponding mass are included to account for possible torsional interaction due to nonsymmetric nature of the building complex. The effective soil column for calculating soil torsional stiffness is shown in Figure 3.7-40. The torsional spring constant for each layer is calculated as

$$k = G\,J/L \tag{3.7-3}$$

where: G = shear modulus obtained from SHAKE

L = thickness of the layer

J = torsional constant defined as

$J = (x^3y^3) / 3.6(x^2+y^2)$

where x and y are the sides of the equivalent rectangle for that layer. Finally, the total spring constant is calculated by adding the torsional stiffnesses of springs in series for all layers. The effective mass M of the soil participating in torsional vibration is taken as the mass of the top one-third of the soil column. A tabulation of the torsional modal properties for OBE and SSE is presented in Table 3.7-10.

A weighted constant modal damping for soil is used in modal synthesis. To obtain weighted constant soil damping, the damping for each layer from SHAKE is multiplied by the thickness of that layer. The sum for all the layers is divided by the total height of the soil profile. The factor obtained is the weighted constant soil damping.

A comparison of the resulting interaction spectra at the foundation of the structure and the free field input design spectra is presented in Figures 3.7-41 and 3.7-42 for OBE and SSE respectively. A typical comparison of the design response spectrum and the free field foundation spectrum in the interaction model is presented in Figure 3.7-43.

The horizontal design response spectra at relevant locations of the structure are generated using a fixed base model subjected to translational base excitation obtained from the interaction model. This detailed decoupled analysis is justified by comparing spectra generated at selected points of the structure using a soil-structure coupled model with the spectra generated using the decoupled model. This comparison is presented in Figures 3.7-44 through 3.7-51. The comparison shows that the effect of the foundation torsion and rocking is insignificant.

3.7.2.4.2 Vertical Excitation

For excitation in the vertical direction, a lumped mass stick model is used instead of the finite element model. The soil column directly under the foundation is considered effective for vertical excitation. Since the vertical excitation is predominantly due to compressional wave propagation, the strain dependent shear modulus curves used in horizontal excitation are modified in terms of axial strains and the equivalent compressional wave modulus G'. The shear strains γ are converted to axial strains ε by the relation.

$$\varepsilon = \frac{\gamma}{1+\nu} \qquad (3.7\text{-}4)$$

and the corresponding shear modulus G are converted to equivalent compressional wave modulus G' as

$$G' = G\,\frac{2(1-\nu)}{1-2\nu} \qquad (3.7\text{-}5)$$

where

ν = Poisson's ratio.

For strain dependent damping curves, the shear strains are modified to axial strains as described above, but the corresponding damping values are not changed. The SHAKE program is used to obtain soil properties and rock motion compatible with strains developed in vertical excitation.

The soil column below the foundation is now modeled as axial spring and mass system. Each layer is represented by an axial spring with its mass lumped at its two ends. The stiffness of each of these axial springs is the compressional stiffness of that layer given as (G'A/L) where A is the surface area of layer (same as building foundation base area) and L is the thickness of that layer. The DAPS program is used to extract normalized modes for this model.

Using modal synthesis technique, the vertical building model described in Subsection 3.7.2.1.1.2 is analyzed using the DYNAS program for the two postulated earthquake loadings OBE and SSE. The interaction spectra at the foundation of the building are compared with the free field input design spectra in Figures 3.7-52 and 3.7-53 for OBE and SSE, respectively.

3.7.2.5 Development of Floor Response Spectra

3.7.2.5.1 Introduction

When a structure is subjected to an earthquake, the base of a subsystem (or equipment) mounted on the floor slab or wall experiences the motion of the slab or wall. This motion may be significantly different from the input motion at the base of the structure. Therefore, the response spectra used in the analysis of the structure are not directly applicable to the analysis of subsystems mounted in the structure unless the subsystem element is modeled in the dynamic model of the structure. Also, unless the subsystem element is a rigid mass, rigidly connected to the slab or wall, the motion of the subsystem is different from the motion of the slab or wall, because the subsystem element is a flexible elastic system which responds dynamically to the motion of the slab. For these reasons, the motion experienced by a subsystem is the structure's base excitation modified as a function of the structure's characteristics, and the mode of attachment to the structure.

To establish explicit slab or wall motions, applicable to development of subsystem design criteria, time-history forcing functions are used to excite the building models used in the system analysis. Resulting time-history slab or wall motions are used to generate response spectra for the analysis of subsystems supported in the building.

3.7.2.5.2 Horizontal Response Spectra

Time-history analyses of each building system are performed on the horizontal seismic model as discussed in Subsection 3.7.2.1.1.1. The following general procedure is used to develop horizontal seismic subsystem input using the modified El Centro earthquake time-history forcing functions described in Subsection 3.7.1.2. The procedure is as follows:

a. The responses at each slab of interest are obtained by exciting the model simultaneously along x- and y-directions and the responses are combined algebraically at each time step.

b. A set of damping values is selected to generate response spectra based upon appropriate damping for typical subsystems present in the station.

c. Response spectra are generated on each slab which supports Seismic Category I subsystems or components. At least 50 periods from 0.02 to 2.0 seconds are used to develop each spectrum curve. The periods used include all modal periods of the system to evaluate the effect of resonance. Periods in between the system modal periods are considered to establish the shape of the spectra and to avoid missing any prominent peak.

d. The peaks of the spectra curves are widened by 10% on the period scale, to either side of the peak's period. This increase in the peak width accounts for the expected variation of structural properties, damping, and soil properties.

e. The final subsystem horizontal input consists of separate response spectra in the x- and y-directions. For the design and testing of subsystems where conditions necessitate the use of one horizontal spectrum, such a spectrum is obtained by using the square root of the sum of the squares of the accelerations given by the two orthogonal response spectra.

3.7.2.5.3 Vertical Response Spectra

The procedure for determining subsystem response spectra in the vertical direction is the same as that for the horizontal direction. However, in this case, response spectra are generated for uncoupled time-history motion in the vertical or z direction.

The vertical response spectra are generated along the wall and at the center of the slabs. These spectra are used in the design of the subsystems. The peaks of the spectra curves are widened by 20%, on the period scale, to either side of the peak's period. This increase in the peak width accounts for the expected variation of structural properties, damping, and soil properties.

3.7.2.6 Three Components of Earthquake Motion

Seismic response resulting from analysis of systems due to three components of earthquake motions are combined in the following manner:

$$R = \sqrt{R_x^2 + R_y^2 + R_z^2} \tag{3.7-6}$$

where:

R = design seismic response,

R_x = probable maximum seismic response due to horizontal earthquake motion along the x-axis,

R_y = probable maximum seismic response due to vertical earthquake motion along the y-axis, and

R_z = probable maximum seismic response due to horizontal earthquake motion along the z-axis.

R_x, R_y, and R_z are probable maximum, co-directional seismic responses of interest (strain, displacement, stress, moment, shear, etc.) due to earthquake excitations in x, y, and z directions, respectively.

3.7.2.7 Combination of Modal Responses

When a response spectrum method of analysis is used to analyze a system (BOP structures), the maximum response (displacements, accelerations, shears, and moments) in each mode is calculated independent of time, whereas actual modal responses are nearly independent functions of time and maximum responses in different modes do not occur simultaneously. Based on References 3 and 4, the final response can be computed as:

$$R = \left[\sum_{k=1}^{n} \sum_{l=1}^{n} \left(R_k \, R_l \, \xi_{kl} \right) \right]^{1/2}$$

where:

R_k = responses due to k-th mode

$$\xi_{k_\ell} = \left[1 + \left(\frac{\omega_k' - \omega_\ell'}{\beta_k' \omega_k + \beta_\ell' \omega_\ell} \right)^2 \right]^{-1}$$

$$\omega_k' = \omega_k \left[1 - (\beta_k')^2 \right]^{1/2}$$

$$\beta_k' = \beta_k + \frac{2}{t_d \, \omega_k}$$

where:

ω_k = modal frequency in the k-th mode,

β_k = modal damping in the k-th mode, and

t_d = duration of the earthquake.

3.7.2.8 Interaction of Non-Seismic Category I Structures with Seismic Category I Structures

When Seismic Category I and non-Seismic Category I structures are integrally connected, the non-Seismic Category I structure is included in the model when determining the forces on Seismic Category I structures. The non-Seismic Category I structure is designed under the criteria that ensures that a failure of any part of the non-Seismic Category I structure does not affect the seismic behavior or structural integrity of Seismic Category I structures or systems.

3.7.2.9 Effects of Parameter Variations on Floor Response Spectra

The effect of variations of structural properties, dampings, soil properties, soil structure interaction on floor response spectra, and time histories is discussed in Subsection 3.7.2.5.

3.7.2.10 Use of Constant Vertical Static Factors

In general, Seismic Category I structures, systems, and components are analyzed in the vertical direction using the methods specified in Subsection 3.7.2.1. No vertical static factors are used for structures.

3.7.2.11 Method Used to Account for Torsional Effects

The methods used to account for torsional effects are discussed in Subsection 3.7.2.1.

3.7.2.12 Comparison of Responses

The forces obtained from the response spectrum method of analysis are used in the design of structural components of the building. The floor response spectra are generated by time-history analysis. The comparison of responses obtained from the response spectra and time-history methods of analysis is presented in Table 3.7-5.

3.7.2.13 Methods for Seismic Analysis of Dams

This section is not applicable since there are no Seismic Category I dams in the LSCS site.

3.7.2.14 Determination of Seismic Category I Structure Overturning Moments

The Seismic Category I structure overturning moments are determined from the relation of the shear force of the structure and the height of the structure for each mode separately. The overturning moments for each mode are then combined as described in Subsection 3.7.2.7.

3.7.2.15 Analysis Procedure for Damping

In case of structures with components of different damping characteristics, there are two approximate techniques of computing composite modal damping values to lead to a normal mode solution. These are based on weighting the damping factors according to the mass or the stiffness of each element. The two formulations are:

$$\beta_j^m = \frac{\sum_{i=1}^{n} \{\phi_j\}^T \beta_i [M]_i \{\phi_j\}}{\{\phi_j\}^T [M] \{\phi_j\}} \qquad (3.7\text{-}7)$$

$$\beta_j^k = \frac{\sum_{i=1}^{n} \{\phi_j\}^T \beta_i [K]_i \{\phi_j\}}{\{\phi_j\}^T [K] \{\phi_j\}} \qquad (3.7\text{-}8)$$

where:

n	= total number of components,
β_j	= composite modal damping for mode j.
β_i	= critical modal damping associated with component i,
ϕ_j	= mode shape vector,
$[M]_i$,$[K]_i$	= subregion of mass or stiffness matrix associated with component i, and
[M] and [K]	= are the mass and stiffness matrices of the system.

In cases where the stiffness and mass matrices are both diagonal, both equations (3.7-7) and (3.7-8) would give identical results. In a complex structural system where the previous condition is not met, the two methods would give different results and it is not possible to project the superiority of one technique over the other. Since both methods provide rational approximate results, equation (3.7-7) is used in the analysis of fixed base dynamic models.

3.7.3 Seismic Subsystem Analysis

3.7.3.1 Seismic Analysis Methods

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 changes 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 anchored and/or supported at points with different excitations, the response spectrum analysis is performed using the enveloped response spectra of all response spectras which apply.

3.7.3.1.1 Differential Seismic Movements of Interconnected Supports

Systems that are supported at points which undergo certain displacements due to a seismic event are designed to remain capable of performing their Seismic Category I functions. The displacements, obtained from a time-history analysis of the supporting structure, cause moments and forces to be induced into the piping system. Since the resulting stresses are self-limiting, it is justified to place them in the secondary stress category. Therefore these stresses exhibit properties much like a thermal expansion stress and a static analyses is used to obtain them.

3.7.3.2 Determination of Number of Earthquake Cycles

3.7.3.2.1 Piping

The number of stress cycles caused by an earthquake in the piping subsystem fatigue analysis is ten. The number of 1/2 safe shutdown earthquakes (SSE's) in the life of the piping subsystem is five. The total number of earthquake maximum stress cycles in the piping subsystem fatigue analysis is then determined to be 50.

3.7.3.2.2 Equipment

Seismic Category 1 equipment that is qualified by test is tested for an equivalent of five Operating Basis Earthquake (OBE) events and one Safe Shutdown Earthquake (SSE) event. In accordance with IEEE-344-1975 paragraph 6.6.5, the duration of each OBE/SSE equivalent test is at least equal to the strong motion portion of the original time history used to obtain the Required Response Spectra (RRS) for the SSE. As discussed in section 3.7.1.2, modified EL Centro earthquake records were used to develop the RRS. The duration of this time history is 10 seconds.

3.7.3.3 Procedure Used for Modeling

3.7.3.3.1 Modeling of the Piping System

The continuous piping system is modeled as an assemblage of beams. The mass of each beam is lumped at nodes which are connected by weightless elastic members, representing the physical properties of each segment. The pipe lengths between mass points are not 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 center of gravity with respect to centerline of the pipe is included in the analytical model.

3.7.3.3.2 Field Location of Supports and Restraints

The field location of seismic supports and restraints for Seismic Category I piping and piping systems components is selected to satisfy the following two conditions:

a. The location selected must furnish the required response to control strain within allowable limits.

b. Adequate building 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, is checked against the drawings and instructions issued by the engineer. An additional examination of these supports and restraints devices is made to ensure that the location and characteristics of these supports and restraining devices are consistent with the dynamic and static analyses of the systems.

All Seismic Category I piping except for the main steamlines inside the containment, reactor recirculation system, CRD insert and withdraw lines, and the scram discharge header is designed by Sargent & Lundy, including location of supports and restraints. The field location of supports and restraints is done only for non-Seismic Category I piping, 2-inch nominal pipe size and under. Reactor Controls, Inc. had the responsibility for designing the support system for the CRD system.

3.7.3.4 Basis for Selection of Frequencies

The basis for the selection of forcing frequencies is presented in the seismic qualification criteria. All frequencies in the range of 1 to 33 Hz are considered in the analysis and testing of the components and their supporting structures.

3.7.3.5 Use of Equivalent Static Load Method of Analysis

No static load method is utilized in the seismic analyses of piping systems. However, in the seismic analyses of equipment the equivalent static load method is used if the equipment's fundamental natural period (FNP) is not known.

If the FNP is known, the static seismic coefficient is equal to 1.5 times the g level corresponding to the equipment FNP in the applicable response spectrum curves

(RSC). If the FNP is unknown, the static coefficient is equal to 1.5 times the peak g level in the applicable RSC.

The equivalent seismic static load is the product of the equipment mass and the static seismic load coefficient and is applied at the center of gravity.

3.7.3.6 Three Components of Earthquake Motion

Seismic responses resulting from analysis of subsystems due to three components of earthquake motions are combined in the same manner as the seismic response resulting from the analysis of building structures (Subsection 3.7.2.6).

3.7.3.7 Combination of Modal Responses

BOP Subsystems

When a response spectrum method of analysis is used to analyze a subsystem, the maximum response (displacements, accelerations, shears, and moments) in each mode is calculated independent of time; whereas, actual modal responses are nearly independent functions of time and maximum responses in different modes do not occur simultaneously. It has been shown that the probable maximum response is about equal to the square root of the sum of the squares of the modal maxima. This square root criterion is used in combining the modal responses in the response spectrum method of analysis.

The final response, R, is computed as the square root of the sum of the squares of individual modal responses, R . Thus

$$R = \left[\sum_{k=1}^{N} R_K^2 \right]^{1/2} \qquad (3.7\text{-}10a)$$

If the frequencies of the subsystem are well separated, the SRSS method (Equation 3.7-10a) gives acceptable results, however, where the structural periods are not well separated, the coupling between close modes may be considered based on References 3 and 4. The final response would then be computed as

$$R = \left[\sum_{s=1}^{N} \sum_{k=1}^{N} (R_k R_s \varepsilon_{ks}) \right]^{1/2} \qquad (3.7\text{-}10b)$$

However, a more conservative equation 3.7-10c will be used to combine the modal responses in response spectrum method of analysis.

$$R = \left[\sum_{k=1}^{N} \sum_{s=1}^{N} \left| R_k R_s \right| \varepsilon_{ks} \right]^{1/2} \qquad (3.7\text{-}10c)$$

where:

$$\varepsilon_{ks} = \left[1 + \left(\frac{\omega_k' - \omega_s'}{\beta_k' \omega_k + \beta_s' \omega_s} \right)^2 \right]^{-1} \qquad (3.7\text{-}10d)$$

in which

$$\omega_k' = \omega_k \left[1 - (\beta_k')^2 \right]^{1/2} \qquad (3.7\text{-}10e)$$

$$\beta_k' = \beta_k + \frac{2}{t_d \omega_k} \qquad (3.7\text{-}10f)$$

ω_k and β_k are the modal frequency and damping in the k^{th} mode, respectively and t_d is the duration of the earthquake.

As an alternative, the Ten Percent Method can be used for combination of modal responses in accordance with Section 1.2.2 of Regulatory Guide 1.92, Revision 1.

NSSS Subsystems

Modal responses are generated using both the time-history and the response-spectrum methods. When the time-history method is used for combining the effects of three-dimensional earthquakes, the vector sum at every step is used to calculate the maximum response. This method precludes the consideration of modal spacing.

When response-spectrum method of modal analysis is used, all modes except the closely spaced modes are combined by the square-root-of-the-sum-of-the-squares (SRSS) method. However, for closely spaced modes, all piping and equipment are evaluated to the requirements of the guide by using the double sum method:

$$R = \left[\sum_{k=1}^{N} \sum_{s=1}^{N} \left| R_k R_s \right| \varepsilon_{ks} \right]^{1/2}$$

where R is the representative maximum value of a particular response of a given element to a given component of excitation, R_k is the peak value of the response of the element due to the k(th) mode, and N is the number of significant modes considered in the modal response combination. In addition, R_s is the peak value of the response of the element attributed to s (th) mode. Also,

$$\varepsilon_{ks} = \left[1 + \left(\frac{\omega'_k - \omega'_s}{\beta'_k \omega_k + \beta'_s \omega_s} \right)^2 \right]^{-1}$$

in which

$$\omega'_k = \omega_k \left[1 - (\beta'_k)^2 \right]^{1/2} \qquad \beta'_k = \beta_k + \frac{2}{t_d \omega_k}$$

where ω_k and β_k are the modal frequency and the damping ratio in the k (th) mode, respectively, and t_d is the duration of the earthquake.

3.7.3.8 Analytical Procedures for Piping

3.7.3.8.1 Introduction

All Seismic Category I piping is analyzed for seismic effects by a dynamic response spectra analysis or a simplified analysis depending on the code class and the size of pipe.

3.7.3.8.2 General Electric's Simplified Analysis

For piping systems selected to be analyzed by simplified seismic analysis, the following range will be avoided to ensure that the periods of all piping spans are not in resonance with the predominant building and/or component periods:

$$0.7 \times \frac{T_b}{T_p} < 2.0 \qquad (3.7\text{-}11)$$

where

T_b = the predominant building and component period, and

T_p = the fundamental period of the selected span.

3.7.3.8.3 Dynamic Analysis

Each pipeline is idealized as a mathematical model consisting of lumped masses connected by elastic members. Appendages having significant dynamic effects on the piping system, such as motors attached to motor-operated valves, are included in the model. Using the elastic properties of the pipe, the stiffness matrix for the piping system is determined. This includes the effects of torsional, bending, shear, and axial deformations, as well as change in stiffness due to curved members. Next, the frequencies and mode shapes for all the significant modes of vibrations are calculated. After the frequency is determined for each mode, the corresponding horizontal and vertical spectral accelerations with appropriate damping are read from the appropriate response spectrum curves. For each mode, the inertia forces, moments, displacements and accelerations are determined due to each excitation direction (two horizontal and one vertical). The responses in each of these directions are established by either the square root of the double sum (SRDS) Method or the Ten-Percent method (Subsection 3.7.3.7) of all the modal responses in that direction. Stresses due to this excitation are determined using these values. The total response is determined by taking the square root of the sum of squares (SRSS) of the individual responses in the three directions. Horizontal and vertical earthquake excitations are assumed to occur simultaneously. All of the calculations outlined in this subsection are performed by using a computer program for the dynamic analysis of a three-dimensional piping system.

The relative displacement between anchors corresponding to the elevation of seismic supports and the reactor pressure vessel at the elevation of the nozzles is determined from the dynamic analysis of the structures and vessel. The results of the relative anchor-point displacement are used as input to the computer program for a static analysis to determine the additional stresses due to relative anchor-point displacements.

3.7.3.8.4 Allowable Stress

Allowable stresses in the piping caused by an earthquake are in accordance with Section III of the ASME Code. Allowable stresses in the earthquake restraint components, such as shock suppressors, are in accordance with any additional stress limits that may have been established by ASME Section III at the time the restraint components were purchased.

3.7.3.9 Multisupported Equipment Components

When the equipment or component is supported at points with different elevations, the envelope of these elevation response spectra is used for the seismic qualification of the equipment.

3.7.3.10 Use of Constant Vertical Static Factors

In general, Seismic Category I subsystems are analyzed in the vertical direction using the methods specified in Subsection 3.7.3.1. No vertical static factors are used for subsystems.

3.7.3.11 Torsional Effects of Eccentric Masses

All concentrated loads in the piping system, such as valves and valve operators are modeled as massless members with the mass of the components lumped at its center of gravity. A rigid member is modeled connecting the center of gravity to the piping so that the torsional effects of the eccentric masses are considered.

The stress produced at the pipe connection is given in the Computer output.

3.7.3.12 Buried Seismic Category I Piping Systems and Tunnels

During an earthquake, buried structures such as piping and tunnels respond to various seismic waves propagating through the surrounding soil as well as to the dynamic differential movements of the buildings to which the structures are connected. The various waves associated with earthquake motion are P (compression) waves, S (shear) waves, and Rayleigh waves. The stresses in the buried structure are governed by the velocity and angle of incidence of these traveling waves. However, the wave types and their directions during earthquake are very complex. For design purposes, one can derive simple expressions for upper bound stresses by making use of the published results of Newmark (Reference 6), Yeh (Reference 7), and Shah Chu (Reference 8). Here, the various effects are considered separately and then combined properly. The resulting stresses are further combined with other applicable design stresses.

In addition to shaking, an earthquake can also cause faulting. Faulting includes the direct shearing of rock or soil which may carry to the ground surface. Such shearing is generally limited to relatively narrow zones of seismically active faults which can be identified by surveys. (According to the survey reported in Subsection 2.5.3, there exists no previous history of faulting at the site.) From a structural viewpoint, landsliding, ground fissuring, and consolidation of backfill soil have similar effects on buried structures. In general, it is not feasible to design structures to restrain such major soil displacements. However, design measures can be taken to identify and avoid areas prone to such displacements.

In the section following, it is assumed that the soil does not lose its integrity due to the occurrence of an earthquake and thus only the effects of shaking are considered. It is further assumed that the soil is a homogeneous, linear elastic medium and that the buried structure can be treated according to the classical theory of bars and beams.

3.7.3.12.1 Seismic Stress Calculations

The calculation of the seismic stresses in buried structures is divided into three parts: long straight sections far from the ends, bends or elbows, and penetrations.

3.7.3.12.1.1 Long Straight Sections

For the portions of a long buried structure far from the ends and free of any external support other than the surrounding soil, it was assumed that the soil strain (both axial and curvature) is fully transferred to the structure.

According to Yeh (Reference 7), the maximum axial (σ_a) and bending (σ_b) stresses for the various seismic waves are given as follows:

a. Oblique shear wave propagating at some angle :

$\sigma_a = \pm E V_m/(2C_S)$ ($\theta = 45°$) (3.7-12)

$\sigma_b = \pm E R A_m/C_S^2$ ($\theta = 0°$) (3.7-13)

$\varepsilon_m = \pm V_m/(2C_S)$ (3.7-14)

b. Oblique compression wave propagating at angle :

$\sigma_a = \pm E V_m/C_P$ ($\theta = 0°$) (3.7-15)

$\sigma_b = \pm 0.3849 E R A_m/C_p^2$ ($\theta = 35°16'$) (3.7-16)

$\varepsilon_m = \pm V_m/C_p$ (3.7-17)

c. Rayleigh wave propagating in the direction of the pipe axis:

$\sigma_a = \pm E V_m/C_r$ (3.7-18)

$\sigma_b = \pm E R A_m/C_r^2$ (3.7-19)

$\varepsilon_m = \pm V_m/C_r$ (3.7-20)

where:

E = modulus of elasticity of pipe,

R = pipe outside radius,

V_m = maximum ground velocity,

A_m = maximum acceleration due to earthquake,

ε_m = maximum soil strain,

C_s = velocities of shear waves,

C_p = velocities of compression waves, and

C_r = velocities of Rayleigh waves.

These maximum stresses can then be combined, which, in effect, averages the stresses resulting from considering each of the waves separately.

In the above, it was assumed that no relative slippage occurs between structure and soil. This assumption is too conservative for structures whose length is less than the minimum required for the soil to develop the full friction force.

According to Reference 8, let L_m represent this minimum length and define:

$$l_m = \frac{\varepsilon_m AE}{f} \qquad (3.7\text{-}21)$$

where:

l_m = maximum slippage length,

A = pipe metal cross-sectional area, and

f = friction force along pipe axis per unit length.

When $L_m < 2\ l_m$ axial stresses will be proportionately less than those calculated assuming pipe strain equal to soil strain.

3.7.3.12.1.2 Bends

Buried bends can be analyzed by use of the beam on an elastic foundation analogy (Reference 9). The bending stress in the bend can be expressed by:

$$\sigma_b = i\frac{M}{Z} \qquad (3.7\text{-}22)$$

where:

i = stress intensification factor,

M = maximum bending moment, and

Z = section modulus.

In the determination of the bending moment, one needs to consider both the lengths and stiffnesses of the joining legs.

3.7.3.12.1.2.1 Sample Moment Calculation

Let a buried bend be comprised of a longitudinal leg whose axis is in the direction of the maximum soil strain and a transverse leg upon which a normal force of some magnitude dependent on soil movement and the stiffness of the structure. In Reference 8, it is shown that the bending moment is a function of the net movement δ of the structure at the bend. This movement is equal to the soil strain over the effective slippage length minus the amount of elastic deformation of the soil by the transverse leg. Assume that:

a. the longitudinal element is stiff and the transverse element is flexible,

b. the slippage length is given by l_m, and

c. the length of the flexible element is such that $l_1 \geq 3\pi/(4\lambda)$.

The deflection δ, shear force S and bending moment M are then given by:

$$\delta = \frac{\varepsilon_m l_m}{2} \qquad (3.7\text{-}23)$$

$$S = \frac{k\delta}{\lambda} \qquad (3.7\text{-}24)$$

$$M = \frac{k\delta}{2\lambda^2} \qquad (3.7\text{-}25)$$

where:

$$\lambda = \sqrt[4]{k/(4EI)} \qquad (3.7\text{-}26)$$

and k is the soil spring constant per unit length and I is the moment of inertia of the pipe.

If the length of the transverse leg is such that $l_1 < 3\pi/(4\lambda)$, the above values for S and M may be too conservative. Expressions for the stress resultants can then be obtained by considering the transverse leg to be a finite beam on an elastic foundation (Reference 9). Further refinement can be achieved by consideration of the shear load in the effective slippage length.

The expressions for the stress resultants in buried elements forming a tee is given in Reference 8.

3.7.3.12.1.3 Penetrations

Near the entry points into buildings, additional stresses are induced in the buried structure due to the differential movement between the building and the soil. There are two movements to consider, axial and lateral.

a. Axial Movement

This movement causes only an axial stress given by:

$$\sigma_a = \frac{P}{A} \qquad (3.7\text{-}27)$$

where:

$$P = \sqrt{2EAf\delta_x} \qquad (3.7\text{-}28)$$

and δ_x is the axial building/soil movement. In order for this expression to be applicable, the pipe must continue straight for a distance equal to at least P/f.

b. Lateral Movement

Bending, σ_b , and shear, τ , stresses are produced by lateral movement. One also needs to distinguish between a fixed and hinged penetration design. For a fixed design, the stresses are given by:

$$\sigma_b = \pm\left(\frac{k\,\delta_r}{2\lambda^2}\right)\frac{R}{I} \qquad (3.7\text{-}29)$$

$$\tau = \left(\frac{\alpha k}{\lambda A}\right)\delta_r \qquad (3.7\text{-}30)$$

where δ_r is the resultant lateral movement and α is a shape factor. They occur at the point of penetration.

For a hinged type of penetration design, the stresses can be determined from:

$$\sigma_b = \pm 0.1612\left(\frac{kR}{\lambda^2 I}\right)\delta_r \qquad (3.7\text{-}31)$$

$$\tau = \left(\frac{\alpha k}{2\lambda A}\right)\delta_r \qquad (3.7\text{-}32)$$

The maximum bending stress now occurs at a distance $\pi/(4\lambda)$ from the penetration while the shear stress is again maximum at the penetration.

3.7.3.13 Interaction of Other Piping with Seismic Category I Piping

The seismic induced effects of non-Seismic Category I piping systems on Seismic Category I piping are accounted for by including in the analysis of the Seismic Category I piping a length of the non-Seismic Category I systems, to the first anchor beyond the point where the change in category occurs.

3.7.3.14 Seismic Analysis for Reactor Internals

3.7.3.14.1 Introduction

The approach to the solution of the equations of dynamic equilibrium by the modal-superposition method involves two steps: the solution of the characteristic value problem represented by the free vibration response of the system, and the transformation to normal coordinates utilizing the mode shapes of the system. This procedure uncouples the equations of motion, so that the response of the system in each individual mode may be evaluated independently.

The stress, strain, and deformation criteria are described in Sections 3.8, 3.9, and 3.10.

3.7.3.14.2 The Equations of Dynamic Equilibrium

Assuming velocity proportional damping, the dynamic equilibrium equations for a lumped mass, distributed stiffness are expressed in matrix form as:

$$[M]\left\{\ddot{u}(t)\right\} + [C]\left\{\dot{u}(t)\right\} + [K]\{u(t)\}3 = \{P(t)\} \qquad (3.7\text{-}33)$$

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] = mass matrix,

[C] = damping matrix,

[K] = stiffness matrix, and

P(t) = time dependent inertial forces acting at non-support points.

The manner in which a distributed mass, distributed stiffness system is idealized into a lumped mass distributed stiffness system of the building is shown in Figure 3.7-9 along with a schematic representation of relative acceleration $\ddot{u}(t)$, support acceleration $\ddot{u}_s(t)$, and total acceleration $\ddot{u}_t(t)$.

3.7.3.14.3 Solution of the Equations of Motion by Mode Superposition

The technique used for the solution for the equations of motion is the method of modal superposition.

The set of homogeneous equations represented by the undamped free vibration of the system is $[M]\left\{\ddot{u}(t)\right\}+[K]\{u(t)\}=\{0\}$ (3.7-34)

Since the free oscillations are assumed to be harmonic, the displacements can be written as

$$\{u(t)\} = \{\Phi\}\, e^{i\omega t} \quad (3.7\text{-}35)$$

where $\{\Phi\}$ is a column matrix of the amplitude of displacements $\{u\}$, ω is the circular frequency of oscillation, and t is the time. Substituting Equation 3.7-35 and its derivatives in Equation 3.7-34 and noting that $e^{i\omega t}$ is not necessarily zero for all values of t yields

$$\left[-\omega^2\left[M\right]+\left[K\right]\right]\{\Phi\}=\{0\} \qquad (3.7\text{-}36)$$

Equation 3.7-36 is the classical algebraic eigenvalue problem wherein the eigenvalues are the frequencies of vibration ω_1 and the eigenvectors are the mode shapes, $\{\Phi\}_i$.

For each frequency ω_i there is a corresponding solution vector $\{\Phi\}$. 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 perform a coordinate transformation of the displacements, velocities, and accelerations such that the response in each mode is independent of the response of the system in any other mode. Thus, the problem is reduced to 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.

3.7.3.14.4 Analysis by Response Spectrum

As an alternative to the step-by-step mode superposition method described in Subsection 3.7.3.14.3, the response spectrum method is 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 as 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.

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. Probable maximum response is obtained by taking the square root of the sum of the squares (SRSS) of the modal response.

The calculated maximum responses due to one horizontal directional earthquake excitation are combined with the responses due to the vertical earthquake by the sum of the absolute values method. The maximum responses due to another perpendicular horizontal earthquake are also combined with the responses due to the vertical earthquake in the same manner. The larger of the two values is for design.

3.7.3.14.5 Support Displacement in Multisupported Structures

The preceding sections have discussed analysis procedures for forces and displacement induced by time dependent support accelerations. In a multisupported structure there are, in addition, time dependent support displacements which produce additional displacements at nonsupport points and pseudo-static forces at both support and nonsupport points. The total force vector due to both support accelerations and support displacement are given by:

$$\begin{Bmatrix} F(t) \\ F_s(t) \end{Bmatrix} = [\bar{K}] \begin{Bmatrix} u(t) \\ u_s(t) \end{Bmatrix} \quad (3.7\text{-}37)$$

where:

$F(t)$ = time dependent forces at nonsupport points,

$F_s(t)$ = time dependent forces at support points (reactions),

$u(t)$ = time dependent displacements at nonsupport points due to support accelerations,

$u_s(t)$ = time dependent displacements at support points, and

$[\bar{K}]$ = stiffness matrix of the free structure (i.e., a singular matrix and it is built up from the static stiffness coefficients of each element without the application of displacement boundary conditions).

Similarly, the total or absolute displacement of nonsupport points is given by:

$$\{U_t(t)\}=\{u(t)\}+[R]\{u_s(t)\} \quad (3.7\text{-}38)$$

where:

$\{U_t(t)\}$ = total displacement, and

$[R]$ = the transformation matrix which related displacement at nonsupport points due to unit displacements at support points.

3.7.3.14.6 Modeling Techniques for Seismic Category I Structures, Systems, and Components

An important step in the seismic analysis of Seismic Category I systems or structures is the procedure used for modeling. The techniques currently being used are represented by lumped masses and a set of spring dashpots idealizing both the inertia 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.

3.7.3.14.6.1 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 member, representing the physical properties of each segment. The pipe lengths between mass points will be 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 center of gravity with respect to centerline of the pipe is included in the analytical model. If the torsional effect is expected to cause pipe stresses less than 500 psi, this effect may be neglected.

3.7.3.14.6.2 Modeling of Equipment

For dynamic analysis, Seismic Category I equipment is represented by lumped mass systems which consist of discrete masses connected by weightless strings. The following criteria are used to lump masses:

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. The modes are considered as significant if the corresponding natural frequencies are less than 33 Hz and the stresses calculated from these modes 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. 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 such that the floor spectra peak is in the lower frequency range. If such is not the case, the model is adjusted to give more conservative results.

3.7.3.14.6.3 Modeling of Reactor Pressure Vessel and Internals

The seismic loads on the reactor pressure vessel (RPV) and internals are based on a dynamic analysis of an entire RPV-building complex with the appropriate forcing function supplied at ground level. The seismic model of the reactor pressure vessel and internals is given in Figure 3.7-36. The Reference 11 seismic analysis evaluated the impact of GE9 80 mil channels (see section 3.9.5.2.3 for applicability to ATRIUM-9B fuel).

This mathematical model consists of lumped masses connected by elastic (linear) members. Using the elastic properties of the structural components, the stiffness properties of the model are determined. 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 control rod drive housings are grouped into the two representative lengths shown. 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 light components such as jet pumps, incore guide tubes and housings, sparger, and their supply headers. This is done to reduce the complexity of the dynamic model. If the seismic responses of these components are needed, they can be determined after the system response has been found.

The presence of a 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 will serve 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 seismic model of the RPV and internals has two horizontal coordinates for each mass point considered in the analysis. The remaining translational coordinate (vertical) is excluded because the vertical frequencies of RPV and internals are well above the significant horizontal frequencies. Furthermore, all support structures and building and containment walls have a common centerline; hence the coupling effects are negligible. A separate generic and applicable vertical analysis is performed. Dynamic loads due to vertical motion are added to or subtracted from the static weight of components, whichever is the more conservative. The two rotational coordinates about each node point are excluded because of the moment contribution of rotary inertia from surrounding nodes. 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 may therefore 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.

3.7.3.14.6.4 Comparison of Responses

The comparison between the calculated seismic load in the RPV and internals are given in the text of references 17 and 18.

3.7.3.14.7 Dynamic Analysis of Seismic Category I Structures, Systems, and Components

Time-history techniques and the response spectrum technique are used as applicable for the dynamic analysis of Seismic Category I structures, systems, and components which are sensitive to dynamic seismic events.

3.7.3.14.7.1 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 effect of torsional, bending, shear, and axial deformations as well as changes in stiffness due to curved members. Next the mode shape 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 the response spectrum which is closest to and higher in elevation than the center of the mass of the piping system. The relative displacement between anchors is determined from the dynamic analysis of the structures. The results of the relative anchor point displacement are used for a static analysis to determine the additional stresses due to relative anchor point displacement.

A typical dynamic model of the recirculation piping and the steamline piping systems will be described in the stress report.

3.7.3.14.7.2 Dynamic Analysis of Equipment

Equipment are idealized as mathematical models consisting of lumped masses connected by elastic members or springs. Results for some selected large Seismic Category I equipment are given in Table 3.7-6.

Seismic loadings due to two orthogonal horizontal directions and the vertical are combined as detailed in Subsection 3.7.3.14.4.

When the equipment is supported at more than two points located at different elevations in the building, the response spectrum at the higher elevation is chosen as the design spectrum.

The relative displacement between supports is determined from the dynamic analysis of the structure. The relative support point displacement is used for a static analysis to determine the additional stresses due to support displacements.

3.7.3.14.7.3 Differential Seismic Movement of Interconnected Components – The Procedure

The procedure for considering differential displacements for equipment anchored and supported at points with different displacement excitation is as follows:

The relative displacement between the supporting point 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. The time history of stresses thus obtained as a superposition of all modal displacements of the structure at each instant of time.

In the static calculation of the stresses due to relative displacements in the response spectrum method, the maximum value of the model displacement is used. Therefore, the mathematical model of the equipment is subjected to a 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 is obtained by combining the modal results using the SRSS (square root of sum of the square) method. Since the maximum displacement 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 Boiler and Pressure Vessel Code, the stresses due to relative displacement as obtained above are treated as secondary stresses.

3.7.3.14.8 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 will 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 will be allowed, subject to review by the engineer.

3.7.3.14.8.1 Equipment Testing and Test Evaluation

Seismic Category I equipment which are difficult to represent in a mathematical model for calculations or which were required to demonstrate their ability to remain operating without changing the mode of their operation (such as a level switch which should not switch from "on" to "off" or visa versa during the earthquake) were subjected to actual vibration inputs on shake tables. These shake tests were performed by qualified laboratories for the equipment suppliers.

The seismic qualification in the laboratory generally followed the same procedures, which consisted of the following:

a. The equipment was mounted on the shake table in such a manner as to represent its installed condition.

b. Sine sweep tests were performed covering all practicable frequency ranges with constant or variable acceleration levels to determine the resonance frequencies of the equipment. This procedure enables the determination of the predominant mode periods by monitoring the output response.

c. With the predominant period thus obtained, it was used for the determination of the necessary acceleration levels which were obtained from the applicable floor response spectra developed for the applicable value of damping.

d. With acceleration levels and predominant frequencies obtained, the full level or endurance tests were performed to establish the capability of equipment to function during, and withstand the effects of, the accelerations corresponding to the resonance frequency. This is accomplished by one of the following methods.

   1. <u>Sine Dwell Test</u>

      This test utilizes a sine wave function with one of the equipment natural frequencies and the corresponding acceleration levels as input vibrations. The test duration is generally 30 seconds, during which time the behavior of the equipment is monitored and recorded.

   2. Sine Beat Tests

      A sine beat function with number of beats and cycles per beat corresponding to the equipment natural frequency and with predetermined acceleration level is used as input motion to test and record the behavior of the equipment tested. This approach simulates the actual conditions that the equipment would undergo during the actual specified earthquake. The behavior of the equipment was observed and recorded to ensure its capability to withstand the input vibrations.

In all cases the testing was performed for both horizontal and vertical vibrations separately.

3.7.3.14.8.2 Acceptance

Where analyses were performed, utilizing the seismic coefficients or floor response spectra curves, a detailed dynamic analysis of a lumped mass or a finite element model of the equipment was performed.

All calculations, test procedures, and results supplied by equipment manufacturers or their laboratories were reviewed and accepted by qualified specialist engineers prior to release of equipment for shipment.

3.7.3.14.9 Determination of Number of Earthquake Cycles

To evaluate the number of cycles which exist within a given earthquake, a typical boiling water reactor building-reactor dynamic model was excited by three different recorded time histories: May 18, 1940, El Centro NS component, 29.4 seconds; 1952, Taft, N69°W component, 30 seconds; and March 1957, Golden Gate S80°E component, 13.2 seconds. The modal response was truncated such that the response of three different frequency bandwidths could be studied, 0+ to 10 Hz, 10 to 20 Hz, and 20 to 50 Hz. This was 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 3.7-7 was formed.

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. This relationship is graphically shown in Figure 3.7-37.

In summary, the cyclic behavior number of fatigue cycles of a component during an earthquake is 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 in Table 3.7-7 according to the frequency range within which the fundamental frequency lies.

c. For fatigue evaluation, one-half percent (0.005) of these cycles are conservatively assumed to be at the peak load 4.5% (0.045) at three-quarter peak. The remainder of the cycles will have negligible contribution to fatigue usage.

The safe shutdown earthquake 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 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 in PSAR's of many plants shows that during a 40-year life, it is probable that five earthquakes with intensities one-tenth of the SSE intensity, and one earthquake approximately 20% of the proposed SSE intensity, will occur. Therefore, the probability of even an OBE is extremely low. To cover the combined effects of these earthquakes and the cumulative effects of even lesser earthquakes, one OBE intensity earthquake is postulated for fatigue evaluation.

Table 3.7-8 shows the calculated number of fatigue cycles and the number of fatigue cycles used in design.

3.7.3.15 Analysis Procedure for Damping

The analysis procedure for damping for Seismic Category I subsystems is the same as discussed in Subsection 3.7.2.15.

3.7.3.16 Cable Tray and Cable Tray Support System

The cable tray and cable tray support system is designed to withstand forces from dead load, live load, seismic loads, LOCA loads, and hydro-dynamic loads. The loads are combined per Section 5.3.2 of reference 5. For the design of cable trays, live load is defined as a 200 pound man at the midspan during construction.

For horizontal seismic analysis the cable pans and the support system are modeled as a multi-degree-of-freedom system with the mass of the cables plus pan lumped at the levels at which they are supported. The Figure 3.7-38 shows the typical models for a three layer hanger with one, two, and three diagonal members to carry load from horizontal excitation. For vertical seismic analysis the fundamental period of vibration is computed by using a simplified model of continuous beam. This assumption is found to be consistent with test models studied for this purpose. The response spectra obtained from the analysis of the building model are used in determining the seismic forces.

The horizontal and vertical seismic excitations are assumed to be acting simultaneously along the principal axes on the cable pan system. The design seismic responses, computed by taking the square root of the sum of the squares of the individual responses, are limited to 90% of yield and the design procedure is based on Reference 10.

It is observed that 90% of the contribution is due to a single mode and hence the effect of closely spaced modes is negligible.

3.7.3.17 Determination of Number of Safety/Relief Valve (SRV) Discharge Cycles

The number of cycles used in the fatigue analysis of the Class 1 BOP piping for SRV discharge, is calculated for each particular subsystem by multiplying the frequency corresponding to the highest peak in the enveloped SRV(ALL) building response spectra applicable to the subsystem by the duration of the building response. This number of cycles per event is then multiplied by the total number of valve lifts shown in Table 5-6 of the Dynamic Forcing Functions Report (DFFR) NEDO-21061, Rev. 3, to obtain a conservative number of cycles for fatigue analysis.

The cyclic loadings due to SRV (actuation) were considered in the fatigue analyses of the ASME Code Safety Class 1 NSSS piping and equipment where applicable. The thermal cycles due to SRV opening (i.e., acoustic wave) and all other transients considered in the fatigue analyses of the major piping systems and the reactor pressure vessel and internals are listed in Table 3.9-33. The number of dynamic cycles due to SRV opening (acoustic wave), SRV (structural feed back due to valve discharge piping air clearing) considered in the fatigue analyses of the major NSSS piping and equipment are given below.

a. Main Steam and Recirculation Piping

Dynamic Transients	Cycles/Events
SRV* (Acoustic wave)	8400 / 2800
SRV (Structural feed back)	15000 / 253

(*SRV [Acoustic wave] load cycles are not applicable to recirculation piping due to negligible effect.)

b. Main Steam and Recirculation Piping Mounted Valves

In compliance with the requirements of ASME Code Section III, Subsection NB-3500 the normal duty fatigue analyses of the Main Steam Isolation Valves, Main Steam Safety/Relief Valves and Reactor Recirculation Gate Valves mounted in the piping, considered all the applicable thermal cycles including thermal cycles due to SRV (valve opening i.e., acoustic wave) and are listed in Table 3.9-33. A factor of safety of 31.2 or greater on the code permissible cumulative usage factor (It ≤ 1) and a factor of safety of 37.5 or greater on the code permissible fatigue cycles (Na ≥ 2000 cycles) was obtained in the fatigue analyses calculations. The calculated permissible fatigue cycles were determined by entering the ASME Code design fatigue curve with the maximum code permissible stress. Also it should be noted that this code design fatigue curve exhibits a factor of safety of 20 reduction in cycles and a factor of 2 reduction in stress relative to the code crack initiation curve.

Although dynamic cycles were not considered a design basis at the time these valves were ordered, the above safety margins indicate that the consideration of additional dynamic cycles associated with OBE and SRV will not jeopardize the as-design fatigue life of this equipment.

c. Reactor Pressure Vessel and Internals

Dynamic Transients	Cycles/Events
OBE (operating basis earthquake)	10/1
SRV (acoustic wave)	N/A*
SRV (structural feed back)	19740**

*Not applicable due to the negligible effect.

**SRV (structural feedback) cycles are based on distributed amplitudes of SRV loads.)

3.7.4 Seismic Instrumentation

3.7.4.1 Comparison with NRC Regulatory Guide 1.12

The seismic instrumentation program provided for LaSalle County Station is compared with Regulatory Guide 1.12, "Instrumentation for Earthquakes," in Appendix B.

3.7.4.2 Location and Description of Instrumentation

3.7.4.2.1 Active Instruments

Seismic Monitoring Instrumentation Panel 0PA11J located in the Auxiliary Electric Equipment Room is provided with a central recorder for seismic data acquisition and storage, a personal computer (PC), monitor, and printer for display of recorded data, an annunciator instrument that provides local indicator lights and control room indication, and an uninteruptible power supply (UPS). The lights will indicate whether the system is triggered and whether the operating basis or safe shutdown maximum accelerations are exceeded in any one of the three orthogonal directions in the basement of the containment structure or at elevation 820 feet 6 inches of the containment structure. These directions coincide with the major axes of the analytical model used in the seismic analysis of the plant structure (Figure 3.7-8).

The central recorder is Seismic Monitoring Instrumentation Panel 0PA11J is connected to:

a. A seismic switch located at the containment foundation which alerts the operator when a predetermined acceleration for that location has been exceeded.

b. Four triaxial accelerometers, each of which measure the absolute acceleration as a function of time in three orthogonal directions. These directions coincide with the major axes of the analytical model of the structure. These accelerometers are placed at the following locations:

    1. in the free field, a concrete pad near the lake screen house;

    2. in the basement of the reactor building, elevation 673 feet 4 inches, along the E-W centerline;

    3. at elevation 820 feet 6 inches, near the top of the containment structure; and

4. in the auxiliary electric equipment room.

c. The free field sensor transmits tridirectional acceleration data to the central unit. The signal is fed to a trigger mechanism which senses when a very low "g" level threshold has been exceeded, and initiates the recording functions for all the accelerometers.

3.7.4.2.2 Passive Instruments

Four triaxial peak-recording accelerographs which measure the absolute peak acceleration in three orthogonal directions coinciding with the major axes of the analytical model of the structure. These accelerographs are placed at the following locations:

a. on the standby gas treatment system which is located in the reactor building at elevation 820 feet 6 inches;

b. on the RHR service water cooling line;

c. on a diesel generator control panel; and

d. on the main control board.

3.7.4.3 Control Room Operator Notification

Seismic Monitoring Instrumentation Panel 0PA11J is centrally located beneath the Main Control Room in the Auxiliary Electric Equipment Room. The seismic instrumentation is the source of operator information concerning the acknowledgment of an earthquake. An acceleration of 0.01g in any direction activates the seismic trigger which turns on the seismic monitoring instrumentation and lights up the seismic alarm lights at Panel 0PA11J.

Seismic data is stored in the central recorder at Panel 0PA11J and can be viewed and/or printed by means of the PC, monitor, and printer also located at Panel 0PA11J. The recorded seismic data can be used to facilitate the analysis of structural loads during a seismic event.

Observed values which exceed the OBE acceleration threshold setpoint are indicated by an alarm light on the SWP-300 annunciator instrument at Panel 0PA11J.

Further analysis is needed to authenticate structural loads and to evaluate observations via the structural response-seismic model. An observation which exceeds the SSE acceleration threshold is validated in a similar manner with the structural response-seismic model. When evaluated accelerations exceed SSE threshold values, the reactor is shut down. The alarm lights and the recorder data are available simultaneously with the seismic event.

3.7.4.4 Comparison of Measured and Predicted Responses

The computer program which evaluates the time-history data computes the maximum response accelerations at various points of the model. The observed response spectra for the reactor building foundation and the 820 foot 6 inch elevation can be compared with the computed response spectra. Agreement between the observed response spectra and the computed response spectra from the time-history inputs demonstrates the adequacy of the analytical model. The magnitude of actual forces at various structural positions can then be compared to design values to authenticate the capability of the plant to continue operation without undue risk to the health and safety of the public.

3.7.5 References

1. N. C. Tsai, "Spectrum Compatible Motions for Design Purposes," Journal of the Engineering Mechanics Division, Vol. 98 No. EM2, ASCE, April 1972.

2. "Seismic Soil-Structure Interaction Analysis of Nuclear Power Plants," SL-3026, Sargent & Lundy, Chicago, Illinois, May 9, 1973.

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4. A. K. Singh, S. L. Chu, and S. Singh, "Influence of Closely Spaced Modes in Response Spectrum Method of Analysis," ASCE Specialty Conference on Structural Design of Nuclear Plant Facilities, Chicago, Illinois, December 17-18, 1973.

5. LaSalle County Station, "Mark II - Design Assessment Report," Commonwealth Edison Company, Chicago, Illinois, February 1976.

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7. C. K. Yeh, "Seismic Analysis of Slender Buried Beams," Bulletin of the Seismological Society of America, Vol. 64, No. 5, pp. 1551-1562, October 1974.

8. H. H. Shah and S. L. Chu, "Seismic Analysis of Underground Structural Elements," Journal of the Power Division, Proceedings of the American Society of Civil Engineers, Vol. No. P01, pp. 53-62, July 1974.

9. M. Hetenyi, Beams on Elastic Foundation, The University of Michigan Press, Ann Arbor, Michigan, 1967.

10. "Specification for the Design of Cold-Form Steel Structural Members," American Iron and Steel Institute, 1968 Edition.

11. P. B. Shah, "Final Report of the Impact Evaluation of Using GE9 80-Mil Fuel Channel for the LaSalle Units 1 and 2", GE-NE-523-A191-1294, June, 1995.

12. LaSalle County Station Seismic Response Spectra Design Criteria DC-SE-02-LS, Revision 2, Exhibit 42 and 43.

13. General Electric Specification 22A6011AA, Revision 3.

14. General Electric Document Number NEDO-21061, Revision 4, November 1981

15. LaSalle Calculation Number 027830(EMD), Revision 0.

16. LaSalle County Station Seismic Response Spectra Design Criteria DC-SE-02-LS Revision 2, Exhibit 42 and 43.

17. General Electric Stress Report 22A5563 (Unit 1)

18. General Electric Stress Report 22A5472 (Unit 2)

TABLE 3.7-1

CRITICAL DAMPING RATIOS FOR DIFFERENT STRUCTURE OR COMPONENT

ITEM	PERCENT CRITICAL DAMPING OBE CONDITION	PERCENT CRITICAL DAMPING SSE CONDITION
Equipment and large-diameter piping systems, pipe diameter greater than 12 inches	1.0 *	2.0 *
Small-diameter piping systems, diameter equal to or less than 12 inches	1.0 *	2.0*
Welded steel structures	2.0	5.0
Bolted steel structures	5.0	10.0
Prestressed concrete structures	2.0	5.0
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*
Reactor pressure vessel, support skirt, shroud head, separator and guide tubes	2.0	2.0
Fuel	7.0	7.0
Control rod drive housings	3.5	3.5
Steel frame structures	2.0	3.0

Other values may be used if they are indicated to be reliable by experiment or study.

* Some piping systems have been analyzed using the PVRC recommended damping values contained in ASME Code Case N-411.

TABLE 3.7-2
(SHEET 1 OF 2)

PERIODS AND PARTICIPATION FACTORS *

(Horizontal Excitation)

MODE	PERIOD (seconds)	MODAL PARTICIPATION FACTOR X-EXCITATION	Y-EXCITATION
1	.7692	.75417	4.52134
2	.5965	.60951	- 6.79848
3	.3358	8.72109	.14934
4	.3160	.00756	-.00012
5	.2901	- .00586	.00025
6	.2712	.33018	- 58.78088
7	.2697	- .74822	- 21.10515
8	.2490	-.51167	16.62395
9	.2482	11.21937	.20653
10	.2459	-.01908	-.00578
11	.2417	- 17.98441	- .91669
12	.2338	.46054	44.35655
13	.2124	- 6.84213	- 91.09450
14	.2007	- 59.57472	18.11993
15	.1824	2.36382	67.13478
16	.1734	86.48803	1.59956
17	.1686	.74278	.07287
18	.1673	- 16.93375	- 14.93459
19	.1669	- 15.06947	8.04400
20	.1600	- 50.40051	- 5.32233
21	.1589	.06748	.00583
22	.1574	.64208	- 1.95423

TABLE 3.7-2
(SHEET 2 OF 2)

PERIODS AND PARTICIPATION FACTORS *

(Horizontal Excitation)

MODE	PERIOD (seconds)	MODAL PARTICIPATION FACTOR X-EXCITATION	Y-EXCITATION
23	.1572	-11.58795	-.72132
24	.1376	7.79043	- 1.92533
25	.1363	7.64445	.71109
26	.1266	-84.32710	7.78057
27	.1187	23.45847	21.61664
28	.1181	- 6.79507	16.52135
29	.1158	14.61769	4.36955
30	.1095	.02634	.00744

* This table presents historical information. See Design Criteria DC-SE-02-LS, Table 17, for the current horizontal period and participation factors.

Table 3.7-3

PROBABLE MAXIMUM DISPLACEMENTS *

(feet)

	OPERATING BASIS EARTHQUAKE		SAFE SHUTDOWN EARTHQUAKE	
	X-EXCITATION	Y-EXCITATION	X-EXCITATION	Y-EXCITATION
MASS	X-DISPLACEMENT	Y-DISPLACEMENT	X-DISPLACEMENT	Y-DISPLACEMENT
1	.00038	.00059	.00054	.00087
2	.00086	.00123	.00121	.00178
3	.00198	.00287	.00313	.00430
4	.00277	.00443	.00439	.00668
5	.00339	.00645	.00530	.00977
6	.00402	.00832	.00633	.01284
7	.00433	.00893	.00683	.01383
8	.00446	.00919	.00703	.01423
9	.08508	.09574	.05821	.16268
10	.00179	.00334	.00241	.00488
11	.00202	.00429	.00302	.00630
12	.07546	.10283	.07094	.23626
13	.00305	.00522	.00396	.00773
14	.00276	.00540	.00370	.00803
15	.00000	.00000	.00000	.00000
16	.00862	.05159	.01350	.05593
17	.00849	.02940	.01330	.03205

* This table presents historical information. See Design Criteria DC-SE-02-LS, Tables 18 and 18A, for the current probable maximum displacement values.

TABLE 3.7-4

PERIODS AND PARTICIPATION FACTOR *

(Vertical Excitation)

MODE	PERIOD (seconds)	PARTICIPATION FACTORS
1	.26411	.94
2	.21717	1.53
3	.20030	2.13
4	.14305	2.01
5	.11145	- 3.23
6	.08974	- 26.50
7	.07146	- 4.79
8	.06765	- 4.44
9	.06711	- 2.76
10	.06693	- 1.03
11	.06688	- 0.33
12	.06685	- 0.77
13	.06681	0.44
14	.06679	- 0.67
15	.06677	0.11
16	.06676	0.07
17	.06675	0.18
18	.06672	0.03
19	.06670	0.00
20	.06670	0.00
21	.06670	0.00
22	.06068	- 5.22
23	.05851	- 6.04
24	.05497	3.30
25	.04744	1.11
26	.04646	0.00
27	.04632	- 1.48
28	.04156	- 0.81
29	.03697	0.55
30	.03333	0.12

* This table presents historical information. See Design Criteria DC-SE-02-LS, Tables 21 and 22, for the current vertical period and participation factors.

TABLE 3.7-5

COMPARISON OF RESPONSE

ELEVATION	ACCELERATION (g) FOR RESPONSE SPECTRUM ANALYSIS				ACCELERATION (g) FOR TIME-HISTORY ANALYSIS				REMARKS
	N-S DIRECTION		E-W DIRECTION		N-S DIRECTION		E-W DIRECTION		
	OBE	SSE	OBE	SSE	OBE	SSE	OBE	SSE	
761 ft	0.227	0.282	0.202	0.368	0.296	0.322	0.245	0.345	Main floor
843 ft. 6 in.	0.217	0.340	0.324	0.387	0.248	0.346	0.319	0.408	Operating floor
755 ft.	0.154	0.380	0.350	0.412	0.174	0.344	0.342	0.387	Top of reactor pedestal

TABLE 3.7-6
(SHEET 1 OF 2)

COMPARISON OF CALCULATED SEISMIC LOADS TO DESIGN

SEISMIC LOADS OF SEISMIC CATEGORY I EQUIPMENT, SSE CONDITION

EQUIPMENT	NATURAL FREQUENCY (Hz)	SEISMIC LOADS	DESIGN LOADS	LOAD DIRECTION
HPCI Pump and Turbine	>33	0.37 g	1.5 g	NS (North-South)
	>33	0.30 g	-	EW (East-West)
	>33	0.15 g	-	V (Vertical)
RCIC Pump and Turbine	>33	0.37 g	1.5 g	NS
	>33	0.30 g	-	EW
	>33	0.15 g	-	V
SLC Tank	>33	0.37 g	1.75 g	NS
	>33	0.44 g	1.75 g	EW
	>33	0.62 g	1.75 g	V
Spent Fuel Rack	18.1	0.42 g	3.4 g	NS
Defective Fuel Rack	11.6	0.50 g	5.0 g	EW
New Fuel Rack	25.13	0.62 g	3.8 g	V
Dynamic Seismic Analysis Completed				
Refueling Platform	10	0.2 g	*	NS
Dynamic Seismic	11	1.5 g	*	EW
Analysis Completed	16	6.0 g	*	V

* Designed to meet the building floor response spectra plus a conservative safety factor.

TABLE 3.7-6
(SHEET 2 OF 2)

COMPARISON OF CALCULATED SEISMIC LOADS TO DESIGN

SEISMIC LOADS OF SEISMIC CATEGORY I EQUIPMENT, SSE CONDITION

EQUIPMENT	NATURAL FREQUENCY (Hz)	SEISMIC LOADS	DESIGN LOADS	LOAD DIRECTION
Fuel Prep Machine	0.8	0.40 g	0.61 g	NS
	0.8	0.43 g	Against Wall	EW
	0.8	0.20 g	Against Wall	V
RHR Hx.	36.7	0.30 g	1.5 g	NS
	36.7	0.30 g	1.5 g	EW
	36.7	0.32 g	0.5 g	V
HCU	2.0	2.0 g	11g **	NS
	2.75	2.0 g	14g **	EW
	10	6.0 g	18g **	V

** Accelerations determined by test.

TABLE 3.7-7

NUMBER OF DYNAMIC RESPONSE CYCLES EXPECTED
DURING A SEISMIC EVENT

Frequency Band, CPS	0^{+} -10	10-20	20-50
Number of Seismic Cycles	168	359	643

TABLE 3.7-8

FATIGUE EVALUATION DUE TO SEISMIC LOAD

COMPONENT	CALCULATED NUMBER OF CYCLES AT PEAK STRESS		DESIGN NUMBER OF OBE CYCLES AT PEAK STRESS
I. Reactor Pressure Vessel	UNIT 1	UNIT 2	
vessel shell	290	350	10
shroud support	600	13,500	10
skirt	5000	4,000	10
II. Seismic Category I Piping			
recirculation lines	60		60
steamlines	60		60

TABLE 3.7-9

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TABLE 3.7-10

SOLID TORSIONAL MODAL PROPERTIES

M	= 7.61×10^{10} kips-ft ^{2}
K (OBE)	= 0.1306×10^{12} kips-ft/rad
K (DBE)	= 0.0767×10^{12} kips-ft/rad
f (OBE)	= 7.4 rad/sec.
f (DBE)	= 5.7 rad/sec.

3.8 DESIGN OF CATEGORY I STRUCTURES

3.8.1 Concrete Containment

3.8.1.1 Description of the Containment

3.8.1.1.1 General

3.8.1.1.1.1 Primary Containment

The primary containment (Figures 3.8-1 and 3.8-2) consists of a steel dome head and posttensioned concrete wall standing on a base mat of reinforced concrete. The inner surface of the containment is lined with steel plate which acts as a leaktight membrane. The containment wall also serves as a support for the floor slabs of the reactor building and for the refueling pools. The floor slabs are resting on corbels that are formed as part of the concrete containment wall. The refueling pools are integrally connected to, and supported by, the concrete containment wall.

The suppression system is of the over-and-under configuration. The drywell, in the form of a frustum of a cone, is located directly above the suppression chamber. The suppression pool chamber is cylindrical and separated from the drywell by a reinforced concrete slab. The drywell is topped by an elliptical steel dome called the drywell head. The drywell atmosphere is vented into the suppression chamber through a series of downcomer pipes penetrating and supported by the drywell floor. A typical downcomer pipe is shown in Figure 3.8-3.

The drywell houses the reactor and the associated radioactive primary system. The drywell floor is rigidly connected to the containment wall after the application of prestress by means of the detail shown in Figure 3.8-4. The drywell head is bolted to a steel ring girder attached to the top of the concrete containment wall as shown in Figure 3.8-5. The base of the ring girder serves as the top anchor seat for the vertical posttensioning tendons while the top of the ring serves as anchor seat for the drywell head. In order to prevent flooding of the drywell during refueling, a bellows type seal is used to seal the space between the reactor vessel and the drywell. The bellows permit free relative vertical movement and offer some restraint to relative lateral displacement of the RPV and the primary containment vessel. Details of the bellows are shown in Figure 3.8-5.

The suppression chamber of the primary containment is in the shape of a cylinder. The foundation mat serves as the base of the suppression chamber.

Description of steel portions of the containment which are not backed by concrete such as the equipment hatch, drywell head, personnel lock, CRD hatch, suppression chamber access hatches, and electrical and pipe penetrations are given in Subsection 3.8.2.

3.8.1.1.1.2 Dimensions of the Primary Containment

The dimensions of the primary containment are as follows:

a. base slab thickness - 7 feet,

b. inside diameter of suppression pool cylinder wall - 86 feet 8 inches,

c. thickness of suppression pool cylinder - 4 feet,

d. height of suppression pool cylinder - 60 feet 3-1/2 inches,

e. height of drywell cone - 81 feet 1-1/4 inches,

f. inside diameter of top of drywell cone - 31 feet 8 inches, and

g. thickness of drywell wall - 6 feet.

3.8.1.1.1.3 Secondary Containment

The reactor building encloses the reactor and its primary containment. The structure provides secondary containment when the primary containment is in service, and will provide primary containment when the primary containment is open, as during refueling or maintenance. The reactor building houses the refueling and reactor servicing equipment and the new and spent fuel storage facilities. The principal purpose of the secondary containment is to confine the leakage of airborne radioactive materials from the primary containment and provide a means for a controlled, elevated release to the atmosphere. More information on the secondary containment is given in Subsection 3.8.4.1.1.

3.8.1.1.2 Base Foundation Slab

The base foundation slab is a reinforced concrete mat with a 7-foot nominal thickness. The mat is continuous under both the containment and the reactor building. The top of the base foundation slab, within the containment, is lined with a stainless steel liner plate that serves as the suppression pool floor.

3.8.1.1.2.1 Reinforcement

The mat is reinforced in two orthogonal directions. Figures 3.8-6 and 3.8-7 show top and bottom reinforcing plan views of the base slab.

3.8.1.1.2.2 Liner Plate and Anchorage

The liner plate is 1/4-inch thick stainless steel and is anchored by structural steel rolled sections embedded in the concrete and welded to the plate (Figure 3.8-8). The liner plate has been modified to provide additional capacity to resist hydrodynamic uplift loads by adding stiffeners. The stiffeners are made of 3/8 inch thick plates folded into channels with 6 inch webs and 3-3/4 inch flanges. For further details see DAR Sections 4.2, 5.2, and 7.1.

3.8.1.1.2.3 Pedestal and Suppression Chamber Column Base Liner Anchorages

Figure 3.8-9, sheets 1 and 2, show the base line anchorages for the reactor pedestal and suppression chamber columns respectively. Both anchorages consist of steel rods embedded in the base slab and welded to the underside of the liner plate. The vertical reinforcing steel of the pedestal and columns is cadwelded to the top of the liner plate, thus providing full continuity to the base slab.

3.8.1.1.3 Containment Wall

3.8.1.1.3.1 General

The containment wall varies from a 4-foot minimum thickness from the base slab elevation of 673 feet 4 inches to elevation 732 feet 8 inches; and a 6-foot thickness from elevation 732 feet 8 inches to elevation 815 feet 2-1/2 inches. Containment reinforcing consists of hoop and meridional reinforcing that is typically placed in each face of the containment wall. Prestressing tendons are arranged in the hoop and meridional directions.

3.8.1.1.3.2 Reinforcing Layout

Reinforcing consists primarily of #11 bars in both meridional and hoop directions and #7 bars for shear reinforcing. Figure 3.8-10 illustrates the reinforcing scheme.

3.8.1.1.3.3 Prestressing System and Layout

The wall of the primary containment is prestressed using the posttensioning BBRV system. This system utilizes parallel lay and unbonded type tendons, each composed of button-headed wires and end anchorage hardware. Figure 3.8-11 shows the typical tendon layout.

There are 188 horizontal tendons placed in a 240° system to the elevation of 792 feet and in a 360° system above the elevation of 792 feet. For the 240° system, three buttresses are equally spaced around the containment, and each horizontal tendon is anchored at buttresses 240° apart, bypassing the intermediate buttress (Figures

3.8-11 and 3.8-12). For the 360° system, two buttresses are on opposite sides of the containment, and each tendon starts and ends at the same buttress.

There are 120 meridional tendons used in the containment wall. These are anchored at the underside of the base slab as shown in Figure 3.8-13. One-half of the tendons terminate at midheight of the containment wall at the elevation of 786 feet 6 inches. One-quarter of the tendons anchor the drywell head support ring to the concrete at the elevation of 815 feet 2-1/2 inches. The remaining one-quarter of the tendons extend to elevation 841 feet 6 inches and elevation 821 feet 6 inches where they are anchored in recesses.

The tendons are placed inside conduits embedded in the concrete and are protected by corrosion preventive grease. All anchorages for the prestressing system are located outside the primary containment structure and are so designed, furnished and fabricated that the prestressing tendons can be installed after concrete work is complete.

The posttensioning of the tendons takes place after the entire containment and reactor building are constructed up to the operating floor level. The posttensioning sequence is essentially as described below.

The vertical tendons are tensioned first. The longer tendons running the full height of the containment are tensioned prior to the shorter ones. The force is applied from the tendon access tunnel using three equally spaced jacks, simultaneously stressing three tendons. The three jacks are then moved over three tendons and the tensioning pattern is continued until all tendons are tensioned.

The hoop tendons are tensioned next. Each horizontal tendon is tensioned from both ends simultaneously. The tensioning procedures use six jacks to tension an entire ring consisting of three tendons. Every third ring along the entire height of the containment is tensioned before returning to stress the intermediate tendons.

The LSCS containments are carefully checked for the partial pre-stressing stages. The effects of elastic shortening are accounted for in one of the following two ways:

a. by tensioning the first stressed tendons to a higher value than the last ones by the anticipated amount of elastic shortening; or

b. by reducing the tendon capacity considered in design, accounting for the effects of creep, shrinkage, and relaxation.

Friction tests are made on typical tendons of the containment structure prior to the stressing of hoop tendons. The friction factors thus established are used to calculate the tendon elongations. The measured tendon elongations that exceed ±10% of the calculated values, are investigated and corrected where necessary.

3.8.1.1.3.4 Liner Plate Detail and Anchorage

The 1/4-inch liner plate is attached to the containment walls by means of a 3 x 2 x 1/4-inch vertical angles spaced horizontally every 1 foot 3 inches. Additional horizontal stiffeners are provided such that the liner serves as formwork for the containment wall. Figure 3.8-14 shows typical liner plate anchorage.

3.8.1.1.3.5 Penetrations

3.8.1.1.3.5.1 General

Access to the interior of the containment structure is provided by a personnel lock, a control rod drive removal hatch, an equipment hatch located just above the drywell floor level, and two access hatches in the suppression pool at an elevation of 714 feet. The equipment and access hatches are not utilized during normal operation or at other times when containment is required. The containment structure is also penetrated by process pipe lines and electrical penetration assemblies.

To maintain the containment pressure boundary, containment penetration sleeves and head fittings have the following design characteristics:

a. capability to withstand peak transient temperatures,

b. capability to withstand without failure:

   1. static and dynamic effects of process pipe rupture at the outside face of the head fitting, which is caused by the impingement of the fluid from the largest local pipe or connection and/or other dynamic loadings, or process pipe maximum operating pressure applied in the annulus between the process pipe and the penetration sleeves.

c. capability to accommodate the thermal and mechanical stresses which may be encountered during all modes of operation without failure, and

d. capability to withstand piping design and operating pressures and temperatures.

The orientation of the containment penetrations is shown in Figure 3.8-15. The sizes and locations of the penetrations are listed in Table 3.8-1.

Typical penetration reinforcing is shown in Figure 3.8-16. The type, size, and location of the penetration, as well as any load that may be imposed by the penetration, determines whether any additional reinforcing is required. To provide for continuity, the tendons are deflected around the penetrations.

Typical tendon layouts around penetrations are shown in Figure 3.8-17. Local thickening of the containment around penetrations was not necessary.

3.8.1.1.3.5.2 Pipe Penetrations

Pipe penetrations are of the type shown in Figures 3.8-18, 3.8-19, and 3.8-20 for all process lines penetrating the containment. The pipe is welded directly to the head fitting which is welded to the sleeve. The sleeve is embedded into the concrete as it penetrates the containment. Air gaps are provided around all pipes. Insulation and cooling coils are provided around hot pipes to reduce thermal stress in the containment during normal operations. In addition to their function as a primary containment barrier, the penetrations serve as anchors to the pipes and are designed to carry the loads associated with a postulated pipe rupture. Thermal growth and movement is taken up in the piping system.

3.8.1.1.3.5.3 Electrical Penetrations

Canister-type electrical penetration assemblies are used to extend electrical conductors through the pressure boundary of the containment structure. Electrical penetrations are functionally grouped into low voltage power, low voltage control cable penetration assemblies, medium voltage power cable penetration assemblies, and shielded cable penetration assemblies. Figure 3.8-21 shows a typical electrical penetration assembly in place within the containment wall. Hermetic seals between each conductor and the metallic canister's end header plates are obtained by the use of high-strength, high-temperature glass or ceramics. An assembly is sized to be inserted in the 12-inch schedule 80 penetration nozzles which are furnished as part of the containment structure.

3.8.1.1.3.5.4 Equipment Hatch and Personnel Lock

Thickening of the shell around the equipment hatch and personnel lock is not necessary. Additional reinforcing as shown in Figure 3.8-16 is provided to account for the stress concentrations due to the openings. The tendons are deflected around the equipment hatch as shown in Figure 3.8-17 and in a similar way for the personnel lock.

3.8.1.1.3.5.5 Drywell Head

The drywell head ring plate assembly is anchored to the concrete containment wall by one-quarter (30) of the vertical tendons as shown in Figure 3.8-5. See Figure 3.8-10 for the reinforcing in that area.

3.8.1.1.3.6 <u>Internal Containment Attachments</u>

3.8.1.1.3.6.1 <u>Pipe Whip Restraint Embedment</u>

Pipe whip restraint embedments vary in orientation in the containment wall. Typical embedments are shown in Figure 3.8-22. Embedments are cast in the concrete wall of the containment. Leaktest chambers cover the weld joining the embedment insert plate to the liner.

3.8.1.1.3.6.2 <u>Reactor Stabilizer Embedments</u>

The reactor stabilizer structure located at elevation 803 feet 11 inches, supports the sacrificial shield and is keyed into embedments in the containment. Details for the stabilizer are shown in Figure 3.8-23.

3.8.1.1.3.6.3 <u>Beam Seats</u>

Beam seats are provided to support the structural steel framing inside the containment. A typical beam seat embedment is shown in Figure 3.8-24.

3.8.1.1.3.6.4 <u>Drywell Floor Embedment</u>

The drywell floor is attached to the containment wall by means of a steel plate at the junction of the two components. The steel plate is anchored to the containment by means of anchor bolts welded to the plate. The plate is anchored to the drywell floor by cadwelding the radial drywell floor reinforcing to the plate after posttensioning. Figure 3.8-4 illustrates this connection.

3.8.1.1.3.6.5 <u>Snubber Embedments</u>

The snubbers are pipe restraint dampening devices that provide support for pipes during seismic events. Some of these snubbers are attached to plates embedded in the containment wall. A typical snubber embedment detail is shown in Figure 3.8-24.

3.8.1.1.3.7 <u>External Containment Attachments</u>

3.8.1.1.3.7.1 <u>Floor Slab Corbel</u>

The reactor building floor slabs rest on reinforced concrete corbels that are a part of the containment shell. Reinforcing for a typical support corbel is featured in Figure 3.8-25.

3.8.1.1.3.7.2 Mechanical Equipment Hangers

Embedments are provided to support minor pieces of mechanical equipment on the outside wall of the containment. Standard embedment details are used to satisfy the variation in loads and anchorage situations. An example of a typical hanger embedment is found in Figure 3.8-24.

3.8.1.2 Applicable Codes, Standards, and Specifications

This section lists codes, specifications, standards of practice, and other accepted industry guidelines which are adopted, to the extent applicable, in the design and construction of the containment. The codes, standards, and specifications are listed and discussed in Table 3.8-2 and given a reference number. The reference numbers for the containment are:

a. 1 through 13;

b. 16 through 19; and

c. 21, 22, 24, 27 through 29 and 31.

For additional criteria to be used for the design of single angle HVAC duct hanger members, see Subsection 3.8.4.5.2.

The slenderness ratio for compression members in ceiling mounted supports for cable trays, conduits, and HVAC ductwork is limited to 300.

3.8.1.3 Loads and Loading Combinations

Table 3.8-3 lists the load definitions and loading combinations used in the original loading design of the containment shell and reactor building base slab. However, the final design incorporates additional loads associated with postulated dynamic forces due to safety/relief valve (SRV) discharge and LOCA (Reference 1).

Loading combination numbers 1 through 4 and 9 through 14 are in compliance with Article CC-3000 of the ASME B&PV Code, Section III, Division 2. Since a severe environmental condition is not considered under service load combinations, a 1.7 load factor for operating basis earthquake is used in load combination numbers 5 through 8 rather than the 1.5 load factor required by the ASME B&PV Code.

The containment has been subsequently analyzed and elevated for hydrodynamic loads resulting from safety/relief valve discharge and LOCA. The definition of loads and load combinations for this evaluation are given in the LSCS-Mark II Design Assessment Report (Reference 1).

The containment liner is designed for all loads and load combinations listed in Table 3.8-3 except that all load factors are 1.0, in accordance with Subsection CC-3000 of the ASME B&PV Code, Section III.

3.8.1.4 Design and Analysis Procedures

3.8.1.4.1 General

Throughout the analysis, the following areas are given special attention:

a. the intersection between the base slab and the suppression pool cylinder;

b. the intersection between the drywell floor, suppression pool cylinder, and drywell cone;

c. the stresses around the large penetrations;

d. the behavior of the base slab relative to the underlying foundation material;

e. the stresses due to transient temperature gradients in the liner plate and concrete;

f. the penetrations and points of concentrated loads;

g. the intersection of containment wall and buttress; and

h. the tendon anchorage zones.

The design and analysis procedure is in full compliance with the requirements of Article CC-3000 of the ASME B&PV Code, Section III, Division 2. Special emphasis has been placed upon the analyses of hydrodynamic loads imposed on major drywell and wetwell structures and components by various combinations of SRV discharges and postulated LOCA events in the LSCS-Mark II Design Assessment Report (Reference 1).

3.8.1.4.2 Containment

The containment is analyzed by a thin shell of revolution analysis. This analysis is performed using the programs SOR-III, KALSHEL, and DYNAX (Appendix F). The models for the three programs are essentially the same and are shown in Figure 3.8-26. SOR-III is used for all axisymmetric loads such as pressure, dead load, prestressing, and temperature. KALSHEL and DYNAX are used for nonaxisymmetric loads such as reaction from the pools at the operating floor and pipe break forces. In addition, the effect of the liner expansion is calculated using KALSHEL because of its capability to handle layered shells. As in all thin shell analyses, compatibility is solved and the loads are applied at the centerline of the shell wall, therefore, consideration is given in the loads and boundary conditions to account for this effect. The liner is assumed not to assist the containment in carrying loads.

The SOR-III model has three important boundaries. The first is the intersection of the base slab and containment wall. The stiffness of the base slab is modeled as an elastic boundary condition calculated from the CSEF-III (Appendix F) base slab model discussed in Subsection 3.8.1.4.3. With these stiffnesses and the free displacements of the base slab under common loading conditions, compatibility of the containment wall and base slab is solved.

Compatibility is solved at the intersection of the drywell floor and containment wall in a similar way except that the stiffnesses and free displacement are obtained from the SOR-III model of the drywell floor discussed in Subsection 3.8.3.4.1.

The third important boundary is the intersection of the fuel pools and containment wall. The radial and rotational restraints are represented by elastic springs at the containment top. The vertical restraint of the containment by the complete reactor building and the resulting forces and moments on the containment due to differential movement between the two buildings caused by posttensioning, temperature, pressure, creep, and shrinkage are calculated and superimposed in the appropriate load cases.

The containment base slab is integral with the reactor and auxiliary building base slabs, and therefore, the membrane shears and overturning moments due to earthquakes are obtained from the seismic model using DYNAS (Appendix F) which is discussed in Section 3.7. The moment and force plots of the containment shell for the dead load, posttension load, and pressure load are provided in Figures 3.8-27 to 3.8-29.

3.8.1.4.3 Base Slab

The base slab of the reactor building is analyzed by the finite element program SLSAP (Appendix F). Building walls are also included in the finite element model in order to account for the stiffening contributed by the walls. Foundation soil is represented by springs at the nodal points of the elements. The finite element model is extended to cover the base slab of the reactor building for Unit 1. The boundary conditions between the two units is so chosen to account for the influence of Unit 2.

The base slab is also analyzed by using the computer program CSEF-III. The results of the two methods of analysis are compared to verify the assumptions used in each technique.

3.8.1.4.4 Analysis of Areas Around Large Penetrations

To determine the local effects around large penetrations, such as the equipment hatch, main steam pipes, and personnel locks, the areas around these penetrations are analyzed by the finite element program, PLFEM-II (Appendix F). The element nodes lie along the centerline of the containment wall, thus the curvature of the wall is accounted for. The size of the model is so chosen that the boundary conditions are compatible with the containment analysis that neglects the presence of the penetrations within the containment wall, as described in Subsection 3.8.1.4.2.

Forces resulting from deflecting the prestressing tendons around the openings are appropriately included in the analysis.

3.8.1.4.5 Primary Containment Liner

Force in a typical liner panel prior to buckling of any panel is determined from the strain imposed on the liner by prestress, creep, shrinkage, and liner thermal strain restrained by the surrounding containment wall.

The liner anchorage system is analyzed using computer program LAFD (Appendix F) which calculates the force and deflection at the anchorage points. The following cases, considered to produce the worst possible loading conditions on the anchorage system, are included in the analysis:

a. Case I - an initial inward deflection of 1/16 inch;

b. Case II - lower yield bound and 15% decrease in plate thickness of buckled panel;

c. Case III - upper yield bound and 15% increase in plate thickness in stable liner panels; and

d. Case IV - anchor spacing doubled to simulate failed or missing anchor (zipper effect). This case considers the postbuckling strength of this panel to be zero.

The anchor is designed such that if failure were to occur, it would be in the anchor and not in the liner.

The factor of safety against failure is computed by dividing the total energy capacity of the anchor by the energy absorbed up to the point of equilibrium.

3.8.1.4.6 Thermal Analysis

The containment is analyzed for both steady-state and transient thermal gradients.

The steady-state gradients are applied to each design section along with any appropriate axial loads and moments that could be acting simultaneously with the temperature. These include the loads resulting from the self-restraining effect of the structure other than the moment due to the gradient. The stresses in the concrete and reinforcing then are found by using TEMCO (Appendix F) which takes into account the extent of cracking of the section.

The moment resulting from the thermal gradient is the only stress resultant that is permitted to change due to cracking. All other forces and moments are obtained from the various programs using the concrete as a homogeneous material of appropriate stiffness.

For the transient gradient, an equivalent linear gradient is found by summing moments about the centerline of the section. The section is analyzed for this equivalent gradient by the same procedure used for the steady-state gradients. The effect of the nonlinear, self-balancing temperature is added to the effects of the equivalent linear gradient to find the stress due to the transient temperature gradient.

3.8.1.4.7 Creep and Shrinkage Analysis

Effects of elastic shortening, creep, and shrinkage are included in computations for stress losses in tendons and strains imposed on the steel liner. For purposes of analysis, the parameters involved are assumed on the basis of published data and results of tests on concrete for other containments. These assumptions are verified with laboratory tests on the actual concrete used for construction.

3.8.1.4.8 Buttress Analysis

The buttresses anchoring the hoop prestressing tendons are analyzed as a plane strain problem. A horizontal cross section is modeled by the finite element program PLFEM-II using quadrilateral elements. The model is from the centerline of the buttress and extends around the containment until the boundary conditions are compatible with those of an undisturbed cylinder.

The increase in stiffness of the containment wall due to the buttress is also investigated.

3.8.1.5 Structural Acceptance Criteria

The acceptance criteria stated in this paragraph is in full compliance with Article CC-3000 of the ASME B&PV Code, Section III, Division 2. The margin of safety implied by the use of the code provision is best defined by the committee reports that led to that code.

3.8.1.5.1 Working Stress

The service load combinations in Table 3.8-3 are designed using the following allowable stresses:

a. Concrete

1. Compression:

a) membrane compression - 0.30 f 'c,

b) membrane plus flexural compression - 0.60 f 'c,

c) local compression - 0.75 f 'c, and

d) compression under the tendons' end anchor bearing plate - 0.60 f 'c[3] $\sqrt{\frac{A_2}{A_1}}$

2. Tension:

a) membrane tension - not permitted, and

b) flexural tension - ACI 318-63 (Chapter 16).

3. Shear:

a) radial shear - ACI-318-63 (Chapter 26).

b. Reinforcing Steel

1. tension - 0.40 fy, and

2. compression (load-carrying) - 0.40 fy.

c. Tendons

1. tension during prestressing - 0.80 fpu, and

2. tension immediately after anchoring - 0.70 fpu.

3.8.1.5.2 Yield Limit

The factored load combinations in Table 3.8-3 are designed using the yield limit criteria given in this paragraph. The yield strength of the structure is defined for this design as the upper limit of elastic behavior of the effective load-carrying material. The reinforcing steel stress-strain relationship is assumed to be linearly elastic. The concrete compressive stress-strain relationship is defined by a half parabola whose apex is the point where the strain is 0.002 and the stress is 0.85 f 'c. The strain in the reinforcing steel and concrete is assumed to be directly proportional to the distance from the neutral axis. The following allowable stresses are used:

a. Concrete

1. Compression:

a) membrane compression - 0.60 f 'c,

b) membrane plus flexural compression - 0.85 f 'c, and

c) local compression - 0.90 f 'c.

2. Tension:

a) membrane tension - not permitted, and

b) flexural tension - ACI 318-71 (Chapter 18).

3. Shear:

a) radial shear - ACI 318-71 (Chapter 11), and

b) tangential shear - principal tension ≤ $3(f'c)^{1/2}$.

b. Reinforcing Steel

1. Tension - 0.90 fy, and

2. Compression (load-carrying) - 0.90 fy.

The allowables are defined as:

f 'c = Specified minimum compressive strength of concrete.

A_1 = Maximum concrete surface area perpendicular to the tendon axis geometrically similar to and concentric with the contact area of the end anchor bearing plate, which does not overlap corresponding areas of adjacent tendon anchorages.

A = Total surface area of end-anchor bearing plate neglecting the loss of area from the tube.

fpu = Ultimate strength of prestressing steel.

fy = Minimum guaranteed reinforcing steel yields strength.

Interaction diagrams based on these acceptance criteria are generated using the computer program COLID-74 at the sections illustrated in Figure 3.8-30, sheet 1. On the interaction diagrams (Figure 3.8-30, sheets 2 through 21), load levels resulting from the analysis described in Subsection 3.8.1.4 are plotted for various significant loading combinations. For each combination, the variation in moment due to temperature changes is included. These diagrams clearly illustrate the relationship of containment capacity to the imposed loads.

Table 3.8-4 gives the allowables for the containment liner and liner anchorages for self-limiting loads. The allowable stresses and strains for the liner plate are limited to values that have been shown to provide leaktight vessels. The allowables for mechanical loads resulting from the safety/relief valve discharge are shown in the LSCS-MARK II DAR (Reference 1).

3.8.1.6 Materials, Quality Control, and Special Construction Techniques

Material and quality control requirements for containment elements that serve pressure vessel functions are listed in Appendix E of the UFSAR. The physical properties of these materials are listed in the appropriate section of Appendix E of the UFSAR.

3.8.1.7 Testing and Inservice Surveillance Requirements

3.8.1.7.1 Code Compliance Requirements

There are two basic structural tests that are performed to check the containment integrity; (1) structural acceptance tests, and (2) leak rate testing. In addition, inservice testing of the containment is performed to provide a continuing check on structural adequacy.

3.8.1.7.2 Preoperational Testing

3.8.1.7.2.1 Structural Acceptance Test

The structural acceptance test was performed after completion of the construction of the containment complete with liner, concrete structures, all electrical and piping penetrations, equipment hatch, and personnel lock.

The containment structure was instrumented for strain and deflection measurements in accordance with Article CC-6000 of the ASME B&PV Code, Section III, Division 2 (Figure 3.8-31).

The pressure inside the entire containment was increased to 1.15 times the design pressure in approximately four equal increments while maintaining a minimum temperature of 70°F. At each level, the pressure was held constant for at least 1 hour before the deflections and strains are recorded. Crack patterns for all cracks larger than .01 inch in width were mapped at atmospheric pressure both before and after the test and at the maximum pressure level achieved during the test. The crack mapping locations are designated in Figure 3.8-32. The containment was depressurized in the same order as the pressurization. The stress-changes and deflection measurements taken during the test were compared with the predicted values, given in Table 3.8-5, to verify the adequacy of the procedures used in the design. Unit 1 containment structure was successfully tested as a prototype in December 1978. Therefore, Unit 2 containment was considered a non-prototype and strain measurements were not taken during the test.

The containment structure is considered to have satisfied the structural acceptance test if the following requirements are met:

a. yielding of reinforcement does not develop as determined from analysis of crack width, strain gauge, and deflection data;

b. no visible signs of permanent damage to either the concrete structure or the steel liner are detected;

c. the deflection recovery 24 hours or less after complete depressurization is 80% or more; and

d. the measured maximum deflections at points of maximum predicted deflection do not exceed predicted values by more than 30%. This requirement is waived if the 24-hour recovery is greater than 90%.

3.8.1.7.2.2 Leak Rate Testing

Leak rate testing is discussed in Subsection 6.2.6.

3.8.1.7.3 Inservice Testing

3.8.1.7.3.1 Tendon Surveillance

The surveillance requirements for demonstrating the Containment structural integrity shall be in compliance with Section XI, Subsection IWL of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10 CFR 50.55a, as amended by relief granted in accordance with 10 CFR 50.55a(a)(3).

3.8.1.7.3.1.1 Lift-Off Tests

For the first three tests, at least 13 tendons, eight hoop and five vertical, are randomly but representatively selected for lift-off tests. The lift-off stress in the selected sample tendons is measured by calibrated jacks. The lift-off test is considered to be acceptable if the lift-off stress in the sample tendon is equal to or greater than the required design value. If the lift-off stress of any one tendon in the total sample population is less than the design value, two adjacent tendons, one on each side of the defective tendon are checked for lift-off stress. If both of these adjacent tendons are found acceptable, the single deficient is considered unique and acceptable. This single tendon is restored to the required level of integrity. If more than one deficient tendon out of the original sample population is unacceptable, a detailed evaluation is made to verify the integrity of the containment and to establish any evidence of abnormal degradation of the posttensioning system.

3.8.1.7.3.1.2 Wire Sample

In order to evaluate the physical conditions of the tendons, a wire from one tendon of each type, hoop, and vertical, is removed for testing and examination over the entire length.

Tensile tests are made on at least three samples cut from each removed wire, one at each end and one at midlength.

Failure to meet the guaranteed ultimate strengths of any one of the samples is considered as an indication of abnormal degradation and must be investigated.

3.8.1.7.3.1.3 Visual Examination

a. The exterior surface of the Containment should be visually examined to detect areas of large spall, severe scaling, D-cracking in an area of 25 square feet or more, other surface deterioration or disintegration, or grease leakage.

b. Tendon anchorage assembly hardware (such as bearing plates, stressing washers, shims, wedges, and button heads) of all tendons selected as described in Section XI, Subsection IWL of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10 CFR 50.55a, as amended by relief granted per 10 CFR 50.55a(a)(3) should be visually examined. The Containment on which only visual inspection is performed, the tendons selected should be visually inspected to the extent practical without dismantling load bearing components of the anchorage or removing grease caps.

c. Bottom grease caps of all vertical tendons should be visually inspected to detect grease leakage or grease cap deformations. Removal of grease cap is not necessary for inspection.

d. Concrete surrounding visually inspected tendon anchorages should also be checked visually for indication of abnormal material behavior.

In order to determine the corrosion inhibiting qualities of the casing filler, the grease is visually examined at the ends of each selected tendon. Laboratory samples are taken for analysis to determine the presence of impurities. The filler is examined to detect any presence of significant voids, free water, and chemical or physical properties.

3.8.1.7.3.1.4 Frequency

Complete tendon surveillances, which include lift off tests, wire sampling and visual examination have been performed for the 1st, 3rd, 5th, 10th, and 15th year of Unit 1 and 1st, 3rd, 5th, and 15th year of Unit 2.

There was no evidence of abnormal degradation of the containment tendons during the first three tests of the tendons. Therefore, the number of tendons checked for lift-off stress during subsequent tests have been reduced to a representative sample of 7 tendons, 3 vertical and 4 hoop.

For future surveillances commencing from 25th year on Unit 2 and 25th on Unit 1, a physical (i.e., Lift Off Test, wire sampling, and visual examination) and visual only will be alternated at 5 year intervals in accordance with Section XI, Subsection IWL of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50.55a, as amended by relief granted per 10 CFR 50.55a(a)(3). The 20 year inspection on Unit 1 and the 10 year inspection on Unit 2 are visual only.

One tendon from each group tested in the first surveillance is included in all the subsequent tests in order to correlate the observed data, develop and maintain a history of tendon behavior.

3.8.1.7.3.2 Leak Rate

Inservice leak rate testing is described in Subsection 6.2.6.

3.8.2 Steel Containment and ASME Class MC Component

This section pertains to the ASME Class MC components that are a part of the primary containment vessel that is described in Subsection 3.8.1. The MC components include the drywell head assembly, personnel lock, equipment hatches, CRD removal hatch, suppression chamber access hatches, and piping and electrical penetrations. The MC components are discussed in Subsection 3.8.1.1.3.5.

3.8.2.1 Description of the ASME Class MC Components

3.8.2.1.1 Personnel Access Lock

A personnel access lock is provided for access to the interior of the containment (refer to Figure 3.8-33).

The personnel lock consists of an interlocked double door, welded steel assembly. A quick-acting equalizing valve connects the personnel lock with the interior of the containment vessel to equalize the pressure in the two systems when the doors are opened and then closed.

The two doors in the personnel lock are interlocked to prevent both from being opened simultaneously and to ensure that one door is completely closed before the opposite door can be opened. An annunciator in the control room indicates the door operational status.

3.8.2.1.2 Equipment Hatch

An equipment hatch is provided for access to the containment during shutdown. The transfer of equipment and components through the containment wall is accomplished through this opening.

The equipment hatch, Figure 3.8-34, is fabricated from welded steel and then furnished with a double-gasketed flange and bolted dished door. The hatch barrel is welded to the containment liner.

Provisions are made to test pressurize the space between the double gaskets of the door flanges and the welded seam test channels at the liner joint. A shielding door is provided at the outside end of the hatch.

3.8.2.1.3 Drywell Head Assembly

A hemi-ellipsoidal steel plate head is designed to provide a removable containment closure for the top of the concrete containment vessel. The removable head provides reactor access for the refueling operation. The head is made of 1-3/8-inch plate and is secured with 3-inch diameter bolts at the 4-inch thick mating flange. The inside radius of the mating flange is 15 feet 8-11/16 inches. Figure 3.8-5 illustrates the head detail.

3.8.2.1.4 CRD Removal Hatch

The CRD removal hatch is located above the drywell floor through the containment wall. The hatch size is 24 inches ID with a 5/8-inch thick steel wall and a 2-inch thick hinged and bolted flat cover. The hatch is located to provide access to the control rod drive assemblies.

3.8.2.1.5 Suppression Chamber Access Hatches

Two suppression chamber access hatches penetrate the containment wall in the upper portion of the suppression chamber. The 36-inch ID hatches have 3/4-inch thick cylinders with 2-inch thick hinged and bolted flat head doors.

3.8.2.1.6 Penetrations

In addition to the design characteristics discussed in Subsection 3.8.1.1.3.5.1, the process piping penetrations have the capability to act as process pipe supports.

The arrangement of the containment penetrations is shown in Figure 3.8-15, and the sizes and locations are listed in Table 3.8-1.

3.8.2.1.6.1 Penetration Types

Process pipe penetrations which include mechanical (M) and instrument (I) penetrations, fall under any of three types. Figures 3.8-18 through 3.8-20 show the basic designs of the three different types. For all three penetration types, the penetration sleeve is anchored in the wall and extends just inside the containment wall liner. For Type I penetrations, the flued head and a section of the process pipe is one forged piece. For Type II, the flued head is forged and is welded to the process pipe by a full penetration weld. The head fitting for Type III penetrations is a flat plate attached to the process pipe by a full penetration weld.

At the time of design, the determination of the penetration type is made, based on the magnitude of the applicable loads.

Electrical penetrations are of the type discussed in Subsection 3.8.1.1.3.5.3 and illustrated in Figure 3.8-21.

3.8.2.1.6.2 Component Classification

The penetration sleeve in its entire length, when it passes through a Class MC containment vessel, is considered as an MC component and, as such, is designed in accordance with Subsection NE of the ASME B&PV Code, Section III.

The portion of the containment penetration assembly that consists of the head fitting is considered a Code Class 1 component if it is a Type I flued head (Figure 3.8-18), and a Code Class MC component if it is a Type II or III flued head (Figures 3.8-19 and 3.8-20).

3.8.2.2 Applicable Codes, Standards, and Specifications

A list of codes, standards, and specifications is found in Table 3.8-2 with reference numbers given for each item. Those items applicable to the steel pressure retaining components of the containment are as follows:

a. 12, and

b. 27 through 29.

3.8.2.3 Loads and Loading Combinations

The loads and loading combinations for process and instrument piping penetrations are given in Table 3.8-12. The loads and loading combinations for all other MC components are given in Table 3.8-6.

These loads and loading combinations conform to Article NE-3000 of the ASME Code.

3.8.2.4 Design and Analysis Procedures

3.8.2.4.1 Access Hatches and Electrical Penetrations

Access hatches, including the equipment hatch, personnel lock, CRD removal and suppression chamber access hatches, and the electrical penetrations are designed as pressure retaining components. The portions of the sleeves not backed by concrete are analyzed and designed according to the provisions of Section III, Subsection NE of the ASME B&PV Code.

At the junction of hatch covers to the flange on the sleeve, where local bending and secondary stresses occur, the computer program E0119 is used for analysis. This program is also used for the analysis of the flat head covers.

3.8.2.4.2 Drywell Head Assembly

The original design of the Drywell Head Assembly was performed by Chicago Bridge and Iron. The refueling head assembly is divided into two models, the lower flange and the head. The stress analysis for these two models was accomplished using the CB&I thin shell computer program EO781, which 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 reaction of the concrete vessel at its junction with the tendon ring has been evaluated and the resulting deflections and rotations of the reference surface at this point and the structural stiffness matrix of the concrete vessel are used as boundary conditions at the start of the lower flange model.

The Drywell Head Assembly and tendon ring has also been modeled using the finite analysis program ANSYS Revision 5.3, Second Release to calculate drywell head bolting requirements. The model used standard ANSYS elements to model the structure. The loads, loading combination, stress limits and displacement acceptance criteria from the original CB&I analysis were used in this ANSYS model.

3.8.2.4.3 Process Piping Penetrations

The entire penetration assembly, including sleeve, flued head, and attached portion of pipe, is designed for the loads described in Subsection 3.8.2.3 by the finite element computer program, PENAN. The boundary conditions for the finite element model are taken as fixed against all degrees of freedom at the outside face of the containment wall. Thermal gradient evaluations and the final penetration assembly stress analysis report, are also performed by PENAN.

3.8.2.5 Acceptance Criteria

3.8.2.5.1 Drywell Head Assembly, Access Hatches, and Pipe and Electrical Penetrations

The drywell head, access hatches, and electrical penetrations are designed as Class MC components according to Subsection NE of Section III of ASME 1971 B&PV Code, up to and including Winter 1971 addenda.

These components are designed for the loads and loading, combinations given in Table 3.8-6 for the allowables given below.

Load combinations 1, 2, 3, 4, 5, 8, and 9 in Table 3.8-6 are designed according to the allowable stresses specified in Paragraph NE-3322 (a) or (b).

Loading combinations 6, 7, 10, 11, and 12 in Table 3.8-6 are designed according to the allowable stresses specified in Paragraph NE-3322 (c) 1 or 2.

The primary difference between the Winter 1971 version of the code and the 1974 version is the increase of both the design pressure and the stress allowable values by 10%, which results in the same factor of safety for a vessel designed by either version of the code.

3.8.2.5.2 Process Piping Penetration Assemblies

The process piping penetrations (mechanical and instrumentation) are designed according to the Winter 1971 Addenda to the ASME B&PV Code, Section III, Division 1 and are checked for compliance with the 1974 edition of the code. In this subsection, reference is made to paragraphs in the 1974 edition to indicate applicable acceptance criteria.

3.8.2.5.2.1 Loading Conditions

Containment piping penetration sleeves and head fittings meet all stress limits associated with the worst loading combinations for Design, Normal, Upset, Emergency, Faulted, and Testing Component Conditions, in accordance with the requirements and provisions of Division 1 of the ASME Code, Section III.

3.8.2.5.2.2 Loading Combinations and Stress Limits for Penetration Sleeves and Head Fittings

The load components, loading combinations, and stress limits corresponding to each of the loading conditions stated in Subsection 3.8.2.5.2.1 are defined in Subsections 3.8.2.5.2.2.1 through 3.8.2.5.2.2.5, and are summarized in Reference 5 and Table 3.8-13.

3.8.2.5.2.2.1 Design Conditions

The penetration sleeves and head fittings are evaluated for the worst combination of Design Pressures and Temperatures, plus loads due to: Weight, Operating Basis Earthquake (OBE), Hydraulic Transients, Safety Relief Valve Discharge (SRV), Pool Swell, and Bubble Effects, as applicable (see Reference 5).

Under these loading combinations, the head fitting and penetration sleeves meet all applicable stress requirements set forth in Paragraph NE-3221 of the ASME Code, Section III (NB-3221 for Type I head fittings). These stress requirements are summarized in Table 3.8-13.

3.8.2.5.2.2.2 Normal and Upset Conditions

The penetration sleeves and head fittings are evaluated for the worst combination of maximum Operating Pressures and Temperatures, plus Thermal Transients, plus loads due to: Weight, Operating Basis Earthquake (OBE), Thermal Expansion, Relative Seismic Displacements, Hydraulic Transients, Safety Relief Valve Discharge (SRV), Pool Swell, and Bubble Effects, as applicable (see Reference 5).

Under these loading combinations, the head fittings and penetration sleeves meet all applicable stress requirements set forth in Paragraph NE-3222 of Section III, (Paragraphs NB-3222 and NB-3223 for Type I head fittings). These stress requirements are summarized in Table 3.8-13.

3.8.2.5.2.2.3 Emergency Conditions

The penetration sleeves and head fittings are evaluated for the worst combination of maximum Operating Pressures and Temperatures, plus loads due to: Weight, Safe Shutdown Earthquake (SSE), Hydraulic Transients, Safety Relief Valve Discharge (SRV), Pool Swell and Bubble Effects, as applicable (see Reference 5).

Under these loading combinations, the head fittings and penetration sleeves meet all applicable stress requirements set forth in Paragraph NB-3224 of Section III. These stress requirements are summarized in Table 3.8-13.

3.8.2.5.2.2.4 Faulted Conditions

The penetration sleeves and head fittings are evaluated for:

a. the maximum Operating Pressures and Temperatures plus the worst combination of the following loads, applied at the outer face of the head fitting:

   1. axial load equal to (2) (1.26) (P) (A), where P is the maximum operating pressure and A is the process pipe flow area,

   2. shear load equal to the axial load, above,

   3. bending moment equal to the limit bending moment capacity of the process pipe, and

   4. torsional moment equal to the elastic bending moment capacity of the process pipe.

b. The process pipe maximum Operating Pressure applied in the annulus between the process pipe and the penetration sleeve.

Under each of these loading cases, the head fittings and penetration sleeves meet all applicable stress requirements described in F-1324.1, F-1324.6, and Table F-1322.2-1 of Appendix F of the ASME Code, Section III, for System Inelastic - Component Elastic Analysis. These stress requirements are summarized in Table 3.8-13.

3.8.2.5.2.2.5 Testing Conditions

Type I penetration head fittings are evaluated for testing conditions and satisfy the requirements specified in Paragraphs NB-3226, NB-6222, and NB-6322 of the ASME Code, Section III. Class MC penetration sleeves and head fittings are evaluated for testing conditions and satisfy the requirements specified in Paragraphs NE-6222 and NE-6322 of the ASME Code, Section III.

3.8.2.6 Materials, Quality Control, and Special Construction Techniques

Material requirements for steel elements that serve pressure vessel functions are listed in Appendix E.6. The physical properties of these materials are listed in Table 3.8-7. The penetration components mentioned in Appendix E.6 fully comply with the materials Article NE2000 of the ASME Code Section III, Division 1. LaSalle County Station is in compliance with the fabrication and installation requirements of Article NE4000 of the Code, as well as with the provisions of Article NE5000, dealing with examination of components. Standard construction techniques are used in the fabrication and erection of MC components.

3.8.2.7 Testing and Inservice Surveillance Requirements

3.8.2.7.1 Structural Acceptance and Initial Leak Rate Tests

All MC components are tested for their structural acceptance and leak rate at the same time of the containment tests described in Subsection 3.8.1.7.2. Type B leak rate tests are performed on all hatches by pressurizing the plenum between the double gaskets.

In addition, the personnel access hatch is shop tested according to the following procedure:

a. Initial Soap Bubble Test - check for leaks with lock interior at 5 psig.

b. Overpressure Test - lock interior pressurized to 52 psig and held for 1 hour.

c. Second Soap Bubble Test - pressure is reduced to 45 psig and the soap bubble test performed.

d. Initial Leak Rate Test - with the pressure held at 45 psig, the maximum leakage did not exceed 0.5% of the volume of the airlock in 24 hours.

3.8.2.7.2 Inservice Surveillance

Periodic leak rate tests on the containment, including the MC components, are performed as described in Subsection 3.8.1.7.

3.8.3 Concrete and Structural Steel Internal Structures of the Containment

3.8.3.1 Description of Internal Structures

Internal structures of the containment support and shield the reactor pressure vessel, support recirculation pumps, support piping and auxiliary equipment, and form the pressure suppression system. The internal structures include the following:

a. drywell floor,

b. reactor stabilizer structure,

c. steam supply system supports,

d. reactor pedestal,

e. reactor shield,

f. platforms and galleries, and

g. downcomer vent bracing system.

Figure 3.8-1 gives an overall view of the containment, including these internal structures.

3.8.3.1.1 Drywell Floor

The drywell floor is a reinforced concrete circular slab, 3 feet thick, having an inner diameter of 29 feet 11 inches and an outer diameter of 82 feet 8 inches.

The drywell floor serves both as a pressure barrier between the drywell and suppression chamber and as the lateral support structure for the reactor pedestal and lateral and vertical support for the downcomers.

The Drywell Floor is implicitly included in the Reactor Coolant Leakage Detection System in that the floor can allow a holdup of a volume of primary boundary leakage water such that there would be a delay in the Floor Drain sump being able to detect the increases in leakage. The construction tolerances listed in section 3.8.3.2.1 insure that any leakage water held up will not cause a delay in detecting an increased leakrate in an overall reasonable time.

The drywell floor is rigidly connected to the containment wall. A full moment and shear connection is provided by dowels and shear lugs welded to the reinforced liner plate (Figure 3.8-4). Thermal expansion is considered in the containment design, and the resulting forces and moments on the floor are accommodated within the allowable stress limits.

The drywell floor is supported on the pedestal at its inside periphery, on a series of concrete columns and from the containment wall at the outside periphery of the slab. See Figure 3.8-35 for reinforcing details of the floor and figure 3.8-2 for the arrangement of the downcomers and columns.

3.8.3.1.2 Reactor Stabilizer Structure

This structure has the appearance of a truss and serves to laterally brace the top of the reactor shield wall to the containment wall (Figure 3.8-23). The structure consists of 12-inch diameter pipe members which are welded to the top plate of the shield wall at one end and keyed into the containment wall at the other end. This key transmits shear only in a direction tangential to the containment leaving the truss free to move both radially and vertically relative to the containment wall. The reactor pressure vessel is keyed into the top of the shield wall by means of the stabilizer lug and bracket described in Subsection 3.8.3.1.3.

3.8.3.1.3 Reactor Steam Supply System Support

The steam supply system piping and pumps are supported by various types of hangers which in turn are supported by the structural steel galleries or the shield or containment walls. A description of these supports is found in Subsection 5.4.14. In addition, the reactor vessel itself is supported laterally near the top by means of the stabilizer lug and bracket shown in Figure 3.8-36. The lug is an integral part of the pressure vessel wall, and the bracket is rigidly attached to the top plate of the shield wall. Only shear tangential to the vessel is transmitted by this lug and bracket arrangement allowing the vessel to grow freely both radially and vertically relative to the shield wall.

3.8.3.1.4 Reactor Pedestal

The reactor support pedestal, Figures 3.8-37 and 3.8-38 is an upright cylindrical reinforced concrete shell which rests on the containment base slab and stands approximately 82 feet in height. The shell has an inside diameter of 20 feet 3 inches and a thickness of 4 feet 10 inches. The lower portion of the pedestal cavity from the top of the base mat to elevation 699'-10" is filled with reinforced concrete. The concrete is integrally connected to the shell by 1/2-inch diameter welded studs spaced at 8.0 inches, center-to-center.

Two circular slabs are supported on the pedestal:

a. The cavity floor slab, which is located 53 feet 9 inches above the top of the base slab floor plate, is 3 feet 9 inches thick, and projects inward from the pedestal shell walls.

b. The second slab is the drywell floor slab which is described in Subsection 3.8.3.1.1.

The principle reactor anchorage is provided by two 3-1/4-inch and two 3-inch diameter bolts for Unit I and Unit II, respectively, at every 6° of circumference. The reactor shield is anchored by two 2-1/4-inch diameter bolts at every 6° of circumference (Figure 3.8-37, sheets 2 through 4). All concrete surfaces in the suppression chamber are lined with a 1/4-inch stainless steel liner plate, which includes the bottom of the cavity floor on the inside of the pedestal and extends to the drywell floor on the outside of the pedestal.

The liner plate is anchored to the concrete of the pedestal by means of embedded fasteners.

The pedestal cavity is vented to the annular space between the reactor vessel and the reactor shield through openings in the reactor pressure vessel support skirt. The cavity is also vented to the drywell through access openings in the reactor pedestal wall. The design of the reactor pedestal includes the internal pressure resulting from a recirculating line break in the annular space between the reactor vessel and sacrificial shield wall, and also the maximum external transient pressure due to any single high energy line break in the drywell.

3.8.3.1.5 Reactor Shield

The reactor shield, Figures 3.8-1 and 3.8-39, is a composite structural steel and plain concrete, open-ended cylindrical shell placed around the reactor pressure vessel (RPV). The function of the shield is to act as a radiation and heat barrier between the RPV and the drywell wall.

Features of the reactor shield are illustrated in Figure 3.8-39 and Figure 3.8-37, sheets 3 and 4. General features of the reactor shield are as follows:

a. The outside steel shell is constructed of 1-inch and 1-1/2-inch plates with 12-inch deep built-up stiffening rings and vertical stiffeners. A-588 grade 50 steel is used for the plate and the stiffeners. This shell is designed to carry all of the loads that might be imposed on the shield.

b. The inside steel shell is constructed of a 3/8-inch plate with WT 5 x 12.5 stiffening rings and vertical stiffeners. A 36 steel is used for the plate and the stiffeners. This comprises the insulation support and concrete forming system and is designed for the loads associated with the insulation and the concrete placement.

c. The 1 foot 9 inch plain concrete fill in between the two shells is needed only to satisfy the shielding requirements.

d. The shield structure is anchored to the concrete reactor support pedestal at the base, and the top of the shield is supported from the containment wall by the stabilizer structure. The anchorage detail of the shield base plate to the pedestal consists of the following:

    1. a 1-1/2-inch base plate anchored to the pedestal with two 2-1/4-inch diameter A193 Gr B7 bolts or equivalent at every 6° of circumference, and

    2. 3/4-inch thick plate stiffeners at every 3° of circumference welded to the shell and the base plate.

e. Major pipe penetrations are sealed by prefabricated steel door units which are designed to:

    1. provide the required radiation shielding, and

    2. resist transient pressure loadings within the shield annulus.

f. Although the primary purpose of the shield is for radiation shielding, the shield is also designed as a structural member to support equipment and piping-loads as well as to resist pipe rupture, pressure, thermal, and seismic loads.

3.8.3.1.6 Platforms, Galleries and Downcomer Bracings

The platforms and galleries inside the containment serve the dual function of providing access to electrical and mechanical components in addition to structural support for these items. The platforms and galleries consist of structural steel framing supported by the pedestal, containment, and shield walls with steel grating spanning between framing beams except for the grating around the reactor at Elevation 819'-8 ¾" which may be aluminum. Beams which span between the shield or pedestal and the containment wall are provided with connections at the containment that allow for free thermal expansion. Thus, no thermal axial loads are developed in the beams and no radial loads are imposed on the containment wall. For the downcomer bracings, refer to DAR Section 5.3.3 for a description.

3.8.3.2 Applicable Codes, Standards and Specifications

This subsection lists codes, standards of practice, and other accepted industry guidelines that are adopted, to the extent applicable, in the design and construction of the structures internal to the containment. To eliminate repetitious listing of the codes and standards for each structure, the codes and standards are listed and discussed in Table 3.8-2 and given a reference number. For each structure internal to the containment, the reference numbers are listed below.

3.8.3.2.1 Drywell Floor

a. 1 through 10;

b. 16 through 19;

c. 21, 22, and 24; and

d. 27 through 29.

3.8.3.2.2 Reactor Stabilizer Structure

a. 12, 16 through 19; and

b. 24, 27 through 29.

3.8.3.2.3 Reactor Steam Supply System Support

See Subsection 5.4.14 for the codes and standards applicable to the steam supply system hangers and supports.

3.8.3.2.4 Reactor Pedestal

a. 1 through 10;

b. 16 through 19;

c. 21, 22, and 24; and

d. 27 through 29.

3.8.3.2.5 Reactor Shield

a. 3 through 5, 8 through 10;

b. 12, 16 through 19, 24; and

c. 27 through 29.

3.8.3.2.6 Platforms, Galleries and Downcomer Bracings

a. 12, 16 through 19, and 24;

b. refer to DAR Section 5.3.3.4 for the downcomer bracing applicable codes.

3.8.3.3 Loads and Loading Combinations

The drywell floor and the reactor pedestal were initially designed using the loads, load combinations, and load factors listed and discussed in Tables 3.8-3 and 3.8-8 respectively. However, as required by the NRC, the final design was modified to incorporate SRV discharge and postulated LOCA loads as discussed in Reference 1. The loads and loading combinations for the drywell floor are in compliance with Article CC-3000 of the ASME B&PV Code, Section III, Division 2, to the extent discussed in Subsection 3.8.1.3.

The loads and loading combinations for the reactor pedestal comply with the sections of ACI-349, which are based on ACI-318.

The drywell floor and the reactor pedestal are designed for the following pressures:

a. full pressure: 45 psig in the drywell and the suppression chamber;

b. partial pressure large: 45 psig in the drywell and 20 psig in the suppression chamber; and

c. partial pressure small: 25 psig in the drywell and zero pressure in the suppression chamber.

Time-dependent pressure and temperature loads are applied simultaneously using the time-temperature and time-pressure curves shown in Section 6.2.

The structural steel elements of the internal structures, which include the reactor stabilizer structure, downcomer vent bracings, the reactor shield and the miscellaneous platforms and galleries are designed using the applicable loads and load combinations listed and discussed in Table 3.8-9. Appropriate impact factors have been applied to the calculated dynamic loads at the time of analysis of each structural steel element to account for load application time. The thermal loads associated with the reactor shield include temperature gradients under normal operating and accident conditions. The sacrificial shield is also designed for a differential pressure acting outward due to a pipe break in the annulus between the shield and the reactor vessel.

The reactor stabilizer structure is designed primarily for lateral seismic loads. However, all the loads associated with a support at the top of the sacrificial shield such as pressure and pipe whip loads are included in the design of the stabilizer structure.

Internal structures are designed for the reactions of all other structures or equipment that they may support, including the steam supply system hangers and supports. Also, the internal structures have been subsequently analyzed and elevated for the hydrodynamic loads resulting from safety/relief valve discharge and LOCA. The definition of loads and load combinations for this evaluation are given in the LSCS-Mark II Design Assessment Report (Reference 1).

The downcomer vent bracing was added following the analysis of the hydrodynamic loads acting upon the downcomers. Refer to DAR Section 5.3.3.1 and Figures 5.3.1 through 5.3.4 for downcomer bracing details.

3.8.3.4 Design and Analysis Procedures

The design and analysis of all internal structures are described in this subsection with the exception of the analysis for hydrodynamic loads due to safety/relief discharge and LOCA (Reference 1).

3.8.3.4.1 Drywell Floor

The analysis of the drywell floor is made using the computer programs SOR-II, TEMCO, and PLFEM-II which are described in Appendix F. The analysis performed is in full conformance with Article CC-3000 of the ASME B&PV Code, Section III, Division 2.

The axial forces, moments and shear forces obtained from the applicable load are combined for each critical design section, as stated in Subsection 3.8.3.3, and checked based upon the allowable stresses stated in Subsection 3.8.3.5. Throughout the analysis, the following areas of the drywell floor are given special attention:

a. the connections with the containment wall and with the reactor pedestal to the drywell floor are modeled using the appropriate boundary conditions to include the interconnected behavior of the containment, pedestal, and the supporting base slab structure;

b. the connections of the downcomers to the drywell floor; and

c. the effects due to the pipe penetrations in the drywell floor.

The method used to analyze the distribution of forces due to the downcomer whip loads is a finite element method using the program PLFEM-II.

The steady-state temperature gradients are applied to each design section along with any appropriate axial loads and moments that could be acting simultaneously with the temperature. This includes the loads resulting from the self-resisting effect of the structure other than the moment due to the gradient. The stresses in the concrete and reinforcing then are found by using TEMCO which takes into account the extent of cracking of the section.

The moment resulting from the thermal gradient is the only stress resultant that is permitted to change because of cracking. All other forces and moments are obtained from the various programs treating the concrete as a homogeneous material.

For the transient gradient, an equivalent linear gradient is found by summing moments about the centerline of the section. The section then is analyzed for this equivalent gradient by the same procedure used for the steady-state gradients. The effects of the nonlinear, self-balancing temperature is then added to the effects of the equivalent, linear gradient to find the stress caused by the transient temperature gradient.

Strains caused by creep, shrinkage, and elastic shortening are accounted for in the design. The methods and data used for analysis are based on the latest published literature in this field and the results of past experience and tests that have been done for other containments. During the construction of the drywell, laboratory tests on the concrete are used to verify the design data.

3.8.3.4.2 Reactor Stabilizer Structure

The stabilizer structure resists loads through truss action. It is analyzed as a plane frame with the sacrificial shield model (Subsection 3.8.3.4.5).

Loads are input on appropriate members and joints of the shield wall. Supports at the containment wall are modeled as pins that are free to displace radially and vertically.

3.8.3.4.3 Reactor Steam Supply System Supports

Design and analysis procedures for the steam supply system hangers and supports are found in Subsection 5.4.14.

3.8.3.4.4 Reactor Pedestal

The reactor support pedestal is analyzed by two methods. The first using a thin shell of revolution analysis, and the second using a solid revolution finite element method. The basic design is done with the solid of revolution finite element approach, and the thin shell approach is used to verify the results. The structural analysis program, DYNAX (Appendix F), is used in both methods. In addition, the reactor pedestal has been analyzed and evaluated for the hydrodynamic loads resulting from safety/relief valve discharge and LOCA. The definition of loads and load combinations for this evaluation are provided in Reference 1.

The top and the bottom boundaries are represented by elastic springs to account for the effects of the reactor vessel skirt and the sacrificial shield base plate at the top of the pedestal and for the base mat at the bottom. Also, both the drywell floor and the cavity floor are included in the model.

Thermal analysis is performed in a similar manner to the containment analysis as discussed in Subsection 3.8.1.4.6.

The cylindrical shell is reinforced (Figure 3.8-37) on both faces with meridional and hoop steel for moments and membrane forces. Additional meridional and hoop steel is provided where required for tangential shear. Radial ties are provided where required for radial shear.

The capacity of the section under combined loads is checked using the TEMCO computer program.

3.8.3.4.5 Reactor Shield

The reactor shield resists loads in the same manner as a hoop stiffened cylindrical shell. The pipe whip restraints are attached to the plates at the stiffeners, enabling the pipe whip forces to be rapidly distributed by shell action.

The reactor shield wall is analyzed by a general three-dimensional finite element program, SLSAP. See Appendix F for a description of SLSAP.

Due to the presence of openings in the shell, asymmetry exists and is the reason for choosing SLSAP over an axisymmetric analysis program.

Four types of elements are used, plate, beam, truss, and boundary elements, with openings approximated by elements with zero stiffness. Boundary elements are used at the top of the shield to represent the reactor stabilizer structure in the model (Subsection 3.8.3.4.2). The bottom boundary is assumed to be fixed to the reactor pedestal.

For each loading condition, all individual element stresses are output by SLSAP. A postprocessing computer program obtains a maximum stress envelope for all of the various load combinations.

The magnitude and direction of the design whip forces are furnished from the pipe whip restraint analysis as described in Subsection 3.6.2. For both normal operating and accident conditions, the temperature gradients across the shield and their corresponding axial temperature changes are calculated by applying the principles of heat transfer. The temperature gradients and axial temperature changes are input to the SLSAP model as loading conditions according to Subsection 3.8.3.3.

An analysis of the postulated pipe breaks in the region of the nozzle safe ends indicates that the reactor shield structure may be subjected to a combination of jet impingement forces and a transient internal pressure between the shield and the reactor. The shield structure is designed to resist these forces.

3.8.3.4.6 Miscellaneous Platforms and Galleries

The platforms and galleries are designed using conventional elastic design methods which conform to the AISC Specification. Refer to DAR Section 5.3.3.5 and Figures 5.3.3 and 5.3.4 for the downcomer bracing analysis details.

3.8.3.5 Structural Acceptance Criteria

3.8.3.5.1 Reinforced Concrete

Reinforced concrete is designed using working stress as well as yield limit criteria. The allowables for the two methods are given in the following subsections.

3.8.3.5.1.1 Working Stress

The service load combinations in Table 3.8-8 are designed using the following allowable stresses in conjunction with ACI 318-63 Part IV-A.

a. Concrete

   1. Compression

      a) membrane compression = 0.25 f'c, and

      b) membrane plus flexural compression - 0.45 f'c.

   2. Tension

      tension is not permitted.

   3. Shear

      ACI Code 318-63.

b. Reinforcing Steel

   1. Tension = 0.40 fy, and

   2. Compression (load-carrying) - 0.40 fy.

3.8.3.5.1.2 Yield Limit

The factored load combinations in Table 3.8-8 are designed using the yield limit criteria given in this paragraph. The yield strength of the structure is defined for this design as the upper limit of elastic behavior of the effective load-carrying material. The reinforcing steel stress-strain relationship is assumed to be linearly elastic. The concrete compressive stress-strain relationship is defined by a half parabola whose apex is the point where the strain is 0.002 and the stress is 0.85 f'c. The strain in the reinforcing steel and concrete is assumed to be directly proportional to the distance from the neutral axis. The following allowable stresses are used:

a. Concrete

1. Compression

a) membrane compression = 0.60 f'c, and

b) membrane plus flexural compression - 0.85 f'c.

2. Tension

ACI Code 318-71.

3. Shear

ACI Code 318-71

a) radial shear - ACI 318-71 (Chapter 11.0), and

b) tangential shear - principal tension ≤ $\sqrt{f'c}$

Reinforcing is designed to carry the entire tangential shear by shear friction, using a coefficient of friction of 1.0 such that the stresses in the reinforcement do not exceed the allowable stresses specified below:

b. Reinforcing Steel

1. Tension = 0.9 fy, and

2. Compression (load-carrying) = 0.9 fy.

The allowables are defined in Subsection 3.8.1.5.

3.8.3.5.2 Structural Steel

For the working stress load combinations specified in Subsection 3.8.3.3, the stresses in structural steel are limited to those specified in Part I of the 1969 AISC specification with no over stress allowed for earthquake loads. For the yield stress load combinations, steel stresses are limited to 1.6 times the AISC working stress allowables but no more than 0.95 fy ensuring a safety factor of 1.05 against yield. Under working stress conditions deflections are limited to L/200 for floor beams. In all loading cases, steel deformations are limited by the steel stresses to the elastic range.

3.8.3.6 Materials, Quality Control, and Special Construction Techniques

The construction materials and material quality control procedures for reinforced concrete structures internal to the containment conform to the standards set forth in Appendix E of the UFSAR. Structural steel standards are found in Appendix E of the UFSAR. The quality control program for the design and construction of the structures internal to the containment is described in detail in Chapter 17.0.

3.8.3.6.1 Drywell Floor

The drywell floor is comprised of 4500 psi strength concrete. The steel plates for the downcomers are SA-240, Type 304, Grade 60 steel.

3.8.3.6.2 Reactor Stabilizer Structure

The pipe members in the stabilizer truss are ASTM A53 Grade B steel. Plate material is ASTM A588 steel.

3.8.3.6.3 Steam Supply System Supports

The stabilizer bracket supporting the reactor vessel is fabricated primarily from ASTM A36 steel plate.

Other component support materials are discussed in Subsection 5.4.14.

3.8.3.6.4 Reactor Pedestal

Concrete of 4500 psi strength is used in the pedestal. The liner plate on the suppression pool face is SA-240, Type 304, stainless steel.

3.8.3.6.5 Reactor Shield Wall

Plates and structural steel shapes in the shield wall are ASTM A588 Grade 50 (or A572) steel and A36 steel. For more details see Subsection 3.8.3.1.5. The concrete fill is 4500 psi strength.

3.8.3.6.6 Platforms, Galleries and Downcomer Bracings

The structural steel shapes and plates used for beams and framing in the platforms and galleries are ASTM-A36 or A572 Grade 50. For the downcomer bracing material, see DAR Section 5.3.3.1.

3.8.3.6.7 Compliance With Specific Codes

The material and quality control requirements for internal containment structures fully comply with ACI 318-71 and AISC 1969. Quality control in general meets the requirements of ANSI N45.2.5.

3.8.3.7 Testing Inspection and Surveillance Requirements During Plant Construction Phase

The structures specified in Subsection 3.8.3 are visually inspected as part of the surveillance program. Structural steel members are examined for corrosion, excessive deformation, and warpage, and their bolted or welded connections are examined for tightness and soundness. The structural integrity of reinforced concrete members is evaluated by checking for spalling and excessive deformations. Representative samples of anchor bolts are tested for tightness.

A periodic leakage rate test of the drywell floor is performed at a reduced pressure of 1.5 psig per the Technical Specifications.

3.8.3.7.1 Drywell Floor Structural Integrity Test

The drywell floor structural integrity test is performed after completion of the construction of the containment with liner, concrete structures, all electrical and piping penetrations, equipment hatch and personnel lock. The pressure inside the drywell is increased to 25 psig and held for at least 1 hour.

3.8.3.7.2 Drywell To Wetwell Bypass Leakage Test

The preoperational high- and low-pressure bypass leakage integrity of the suppression chamber/drywell barrier will be determined by two separate bypass tests. Each test will be conducted at designated times during the construction/preoperational test periods respectively.

3.8.3.8 Testing and Surveillance Requirements During Plant Operation

A low-pressure bypass leakage test will be performed on the vacuum breaker valves periodically as part of the Drywell to Wetwell Bypass Leakage Test.

3.8.3.8.1 Drywell to Wetwell Bypass Leakage Surveillance Test (1.5 psid)

A low pressure (1.5 psid) bypass leakage test will be performed on the vacuum breaker valves periodically as a part of the surveillance test program.

3.8.3.8.2 <u>Drywell to Wetwell Bypass Leakage Surveillance Test (Conditional 5.0 psid)</u>

At the first refueling outage and during the next subsequent Type A Overall Integrated Containment Leakage Rate test, a 5.0 psid differential pressure test will be performed as part of the drywell to wetwell bypass leakage test. With consistently acceptable results from two tests, this 5.0 psid test will be discontinued.

3.8.4 <u>Other Seismic Category I Structures</u>

3.8.4.1 <u>Description of the Structures</u>

The Seismic Category I structures other than the containment and its internals are as follows:

a. containment enclosure building, reactor building;

b. auxiliary building;

c. fuel storage facilities;

d. control room;

e. diesel-generator building;

f. other Seismic Category I structures,

    1. station vent stack,

    2. main steam pipe tunnel,

    3. equipment access building,

    4. concrete masonry walls (see Attachment 3.C); and

g. non-Seismic Category I safety-related structures,

    1. solid radwaste building waste storage,

    2. lake screen house-intake structure, and

    3. Interim Radwaste Storage Facility (IRSF).

The general arrangement of these structures is found in Drawing Nos. M-4 through M-22.

3.8.4.1.1 Containment Enclosure Building, Reactor Building

The reactor building serves as the secondary containment to the primary containment described in Subsection 3.8.1 (Drawing Nos. M-INDEX and M-4 through M-12). It is located adjacent to the auxiliary building at the east side of the plant complex. It provides access for refueling the reactor and houses equipment essential to the safe shutdown of the reactor.

3.8.4.1.1.1 Basic Structure

The reactor building, up to and including the operating floor, is of reinforced concrete supported on a mat foundation that is common to the containment structure. Above the operating floor, structural steel framing supports the roof, siding, and overhead crane. Of the levels from the foundation to the operating floor, two are located below the final plant grade. The floors are supported by the shear walls, containment and pool walls, and by a beam and column framing system. The frames run in both directions with the exterior beams at each end supported by the shear walls or containment. Beam and slabs supported by the containment rest on elastomeric bearing pads which are designed to absorb containment thermal movement.

At the mat level, special diagonal flood walls are provided to isolate the residual heat removal and core spray pumps. These walls are designed to carry the flood loads from a pump failure and to prevent other areas of the reactor building from being flooded.

The exterior walls of the reactor building are designed to carry a negative pressure of 0.25 psig and will serve as the containment during shutdown when the primary containment vessel is open for refueling or maintenance. Normal access to the reactor building is through an air lock located in the equipment access building.

3.8.4.1.1.2 Operating Floor

The operating floor is the uppermost level in the reactor building and it is the floor from which the reactor is refueled (Drawing Nos. M-5 and M-6). The floor slab is an average of 1 foot 6 inches thick and it is connected directly to the pool wall.

The structural steel superstructure enclosing the floor provides unobstructed access to the floor for the overhead bridge crane. The superstructure consists of a conventional braced frame system with 6-foot (average) deep girders spanning the short building dimension (129 feet). The overhead crane runs on a built-up crane girder which is supported on the superstructure columns. Pipe members provide vertical bracing in the bays on each wall.

3.8.4.1.2 Auxiliary Building

The auxiliary building is located between the reactor building and the turbine building (Drawing Nos. M-7 through M-18).

The building is a reinforced concrete shear wall structure supported on a mat foundation which is continuous with the mats under the reactor and turbine buildings. Above the mat the auxiliary building is structurally integral with the reactor and turbine buildings. The lower levels of the auxiliary building are continuous, two-way slab and beam construction. The levels above the elevation of 710 feet 6 inches are steel framing with poured concrete over decking or concrete slabs.

The auxiliary building contains the main steam tunnel, turbine building access elevator, vent stack, HVAC equipment, laboratories, electrical equipment, 4160-volt and 6900-volt switchgear, 480-volt substation, battery rooms, instrument room, computer room, control room and offices, and facilities for shift operating personnel.

3.8.4.1.3 Fuel Storage Facilities

The refueling pools are located below the operating floor in the reactor building and include the spent fuel and dryer-separator pools (Drawing Nos. M-5 and M-6). The pools are integrally connected to, and supported by, the containment vessel and exterior reactor building walls. The spent fuel pool is 39 feet 1 inch deep. The walls adjacent to the spent fuel racks and the bottom slab are at least 6 feet thick. The reactor shield plugs consist of three 2-foot thick plugs for a total thickness of 6 feet. The dryer-separator pool is 25 feet 4 inches deep with 6-foot thick walls and a 3-foot 4-inch thick floor. The inside surfaces of the pools are lined with 1/4-inch thick stainless steel plate which serves as a leak tight barrier. This plate, along with stiffeners anchored in the concrete, also serves as the form when the walls are poured. The new fuel storage vault, which is between the refueling pools, is 19 feet 4 inches deep. Its walls are between 4 feet and 6 feet thick. The floor is 4 feet thick.

3.8.4.1.4 Control Room

The control room (Drawing No. M-7) is in the auxiliary building. It is located between the elevation of 768 feet and 786 feet 6 inches. It is also between column rows 12 and 18 and J and N. The control room is protected by a minimum of 2 feet of concrete shielding as detailed in Tables 6.4-1 and 6.4-2.

3.8.4.1.5 Diesel-Generator Building

The diesel-generator buildings are reinforced concrete diaphragm-shear-wall type structures. One is located north of the auxiliary building, the other south. Both are located east of the turbine building.

Drawing Nos. M-8 through M-11, M-13, and M-17 show the views of the structure. The buildings are supported on a mat foundation and each is divided into three cubicles separated by reinforced concrete shear walls. The lower floor level, elevation 674 feet, houses the diesel oil fuel tanks. Access to these tanks is provided through manholes in the floor slab above. The diesel generators are located on the upper floor, each in a separate room for fire protection as well as segregation requirements. Fire escapes are provided by means of access doors through the roof and an access door to the adjoining turbine building. Normal access to the generator rooms is through doors leading to the 663 feet and 710 feet 6 inches levels of the auxiliary building.

3.8.4.1.6 Other Seismic Category I Structures

3.8.4.1.6.1 Station Vent Stack

The plant ventilation stack (Drawing No. M-17) is located on the auxiliary building roof and serves as a single point of release for the reactor building, turbine building, and solid radwaste building ventilation as well as off-gas standby gas treatment and plant gland seal exhaust system. The top of the stack is at elevation 1080 feet.

3.8.4.1.6.2 Main Steam Chase

The reinforced concrete main steam chase connects the containment vessel to the main steam tunnel. It protects the piping from external missiles and protects the other Seismic Category I components in the reactor building from the effect of radioactive steam in the unlikely event of a pipe rupture inside the chase. The walls are designed for all effects of a postulated high energy pipe break outside the containment.

3.8.4.1.6.3 Equipment Access Building (Reactor Building Airlock for Railroad Car)

The equipment access building is located at the grade level on the east side of the reactor building, north of, and contiguous with, the off-gas building (Drawing Nos. M-9 and M-15). It provides controlled access to the reactor building for equipment, including a rail car, by means of an air lock equipped with inner bulkhead doors that are flood and missile proof. Effectively, the air lock is an extension of the reactor building envelope. The structure is Seismic Category I and consists of reinforced concrete walls and roof. The majority of the south wall of the equipment access building is common with the north wall of the off-gas building.

3.8.4.1.7 Other Non-Seismic Category I Safety-Related Structures

3.8.4.1.7.1 Solid Radwaste Building - Waste Storage

The function of the solid radwaste building is to collect, monitor, process, package, and provide temporary storage facilities for radioactive solid wastes prior to offsite shipment and permanent disposal. The entire radwaste building (Drawing Nos. M-8 and M-9) shell is poured concrete with the interior walls being made of concrete and concrete block. The structure is supported on a mat foundation and adjoins the turbine building on the turbine building's western side.

The IRSF is a non-seismic, non-Category I, non-safety-related structure. The structure is designed in accordance with Uniform Building Code requirements only. The entire IRSF radwaste storage area consists of a poured-in-place concrete shell supported on an independent mat foundation at grade.

3.8.4.1.7.2 Lake Screen House - Intake Structure

The lake screen house is the only non-Seismic Category I structure that houses Category I SSCs; the service water tunnel is located in the building and CSCS lines draw a suction on the tunnel.

The lake screen house (Drawing No. M-19) which carries the cooling water to the condensor is provided with a bar grill, traveling screens, circulating water pumps, station service water pumps and station fire protection pumps. It consists of a reinforced concrete box type structure from elevation 670 feet to ground floor elevation 714 feet 6 inches. Above 714 feet 6 inches to elevation 741 feet 6 inches, a steel frame is constructed with a precast concrete roof with insulation and built-up roofing.

The lake screen house is a non-Seismic Category I structure. However, concrete portions of the building, including the portions that contain the Category I systems, are designed to withstand Seismic Category I loads.

The wall at the east end of the lake screen house is a permanent concrete retaining wall. It is designed as a Seismic Category I structure. The steel sheet piling wall adjacent to this wall at the west end of the flume is also designed as a Seismic Category I structure.

3.8.4.2 Applicable Codes, Standards, and Specifications

The codes, standards, and specifications applicable to the design, fabrication, construction, testing, and inservice inspection of safety-related structures outside the containment are found in Table 3.8-2 and include the following reference numbers:

a. 1 through 10,

b. 12, and

c. 14 through 31.

The slenderness ratio for compression members in ceiling mounted supports for cable trays, conduits, and HVAC ductwork is limited to 300.

3.8.4.3 Loads and Loading Combinations

The list of loads and their definitions, and the loading combinations applicable to the design of Seismic Category I structures outside the containment, are found in Tables 3.8-9 and 3.8-10. The list of load categories where the types of loads are defined is also found in these tables.

In addition to their own dead loads including the weight of equipment, piping, cable pans, etc., floors are designed for conservative live loads resulting from the movement of the largest possible pieces of equipment. These live loads are patterned to produce the most critical loading effects for the slabs and beams. A snow load of up to 30 psf is used in the design of building roofs, depending upon the load combination. In addition to the snow load, roofs are designed for a negative pressure of 3 psi due to tornado suction and checked for the effects of probable maximum precipitation (Subsection 2.4.2). Floors and roofs are also checked for their ability to transmit shear loads through diaphragm action.

The live load on subgrade walls includes a surcharge load of 1000 psf for the construction condition. For lateral soil and water pressure on the walls see Section 2.5. The seismic shear forces on the shear walls are obtained from the seismic analysis described in Subsection 3.7.2.2.

Loadings on the spent fuel and dryer-separator pools include the effects of water set in motion by seismic accelerations and the thermal gradient resulting from the high temperature of the water in the pools.

The siding and roof deck on the reactor building superstructure are designed to blow off at a specified pressure, such that only the steel frame need be designed for the peak tornado pressures.

3.8.4.4 Design and Analysis Procedures

Conventional elastic techniques are used in the design and analysis of all structural components. All buildings are analyzed basically as shear wall structures, and all floors are checked for their ability to transmit shear forces through diaphragm action. Exterior walls are designed to resist a combination of vertical loads, bending moments, lateral shear and overturning moments associated with seismic forces (Section 3.7) and tornado loads (Section 3.3). Longitudinal and lateral shears are transferred to the mat through friction and keys. The floor slab or beam and column framing are modeled to approximate the actual structural behavior, and the boundary conditions are determined, where critical, by stiffness evaluation of the actual intersecting structural members at the points of intersection.

The computer programs STRESS and STRUDL-II (Appendix F) are used to analyze various lines of beam and column framing in the auxiliary building. Concrete columns are analyzed using PCAUC in this building. In the reactor building, because of the unique interaction between the framing and the containment structure and because of the nonsymmetrical nature of the framing throughout the building, complete frames from foundation to operating floor were modeled and analyzed using the computer program STRUDL II (Appendix F). The refueling pools are analyzed and designed using the finite element program SLSAP (Appendix F). The STAND system (Appendix F) is used to analyze and design beams and columns in the superstructure areas such as the reactor, auxiliary, and diesel-generator buildings.

All concrete structures, for both operating and design-basis loadings, have concrete strain limited to 0.003 with the exception of structures analyzed for the effects of a high energy line break outside the containment where yield line theory is used. For steel structures under operating and design-basis loadings, strains are limited to within the elastic range.

LSCS's design and analysis procedures comply with the sections of ACI-349 which are based on ACI-318.

3.8.4.5 Structural Acceptance Criteria

3.8.4.5.1 Reinforced Concrete

Calculations for concrete members located within the reactor building are based on the ultimate strength design provisions in ACI 318-71. For reinforced concrete structural components elsewhere, stresses and strains are limited to those specified in ACI 318-63. However, the design of these structures is verified using the ultimate strength design theory. For the strength design method load combinations, the margins of safety are contained in the capacity reduction (0) factors specified in the code. For load combinations 9, 13, and 14 yield line theory is used (Table 3.8-10). Members are designed to meet the serviceability requirements of the code and the deflection requirements of the manufacturers of special equipment.

LSCS complies with portions of ACI-349 which are based on ACI-318.

3.8.4.5.2 Structural Steel

The stresses and strains in structural steel are limited to those specified in the 1969 AISC Specifications, Part I, for the service load combinations of Subsection 3.8.4.3. The related margins of safety are as described in the commentary, Section 1.5, of the specifications. No overstresses are allowed for severe environmental loading combinations. For the design-basis load combinations the allowable stresses are increased to 1.6 times the AISC allowable but not more than 0.95 times the steel yield strength, ensuring a margin of safety of 1.05 against steel yielding. In both loading cases, steel deformations are limited since stresses are held within the elastic range. Thus, an additional margin of safety against failure is provided since no plastic deformations are allowed. In addition, deflections are checked and kept within the limits prescribed in the AISC specification.

For single angle HVAC duct hanger members, the flexural allowable stress, for the normal and severe environmental loading combinations, will be obtained from Equations 5.4.3(1) and 5.4.3(2) of the Australian Steel Code (No. 31, Table 3.8-2). The elastic lateral flexural torsional buckling stress, used in computing the bending allowable, will be obtained according to the formulas developed by Leigh and Lay (Reference 3). The flexural allowable stress (F_b) will be limited to:

$b_1/t < 65/\sqrt{F_Y}$ $\qquad F_b = 0.66F_Y$

$65/\sqrt{F_Y} < b_1/t < 76/\sqrt{F_Y}$ $\qquad F_b = 0.60F_Y$

$b_1/t > 76/\sqrt{F_Y}$ $\qquad Fb = 0.60Q_sF_Y$

where b_1 is the width of angle leg in compression, t is the thickness, Q_s is given by Equation (C2-1) or Equation (C2-2) of Appendix C to the AISC Specification and F_Y is the yield stress. The bending allowable stress for the design-basis load combinations will be 1.6 times the allowable for the normal and severe environmental load combinations, but not more than 0.95 times the steel yield strength. The unsupported length to thickness ratio (l/t) for single angle members will not exceed 690.

The above methodology was approved by the NRC in their letter dated August 11, 1986 (Reference 4).

3.8.4.6 Materials, Quality Control, and Special Construction Techniques

Construction materials conform to the standards set forth in Appendix (E.1, E.2, E.4, and E.7). Also in Appendix E the material quality control procedures for sampling and testing of materials are described. The quality control program for the design and construction of the Seismic Category I structures outside the containment is described in detail in Chapter 17.0.

The only construction technique that might be considered unusual is the use of cadwelding for splicing reinforcing bars greater than #11 in size (Appendix E.2.3). The material and quality control requirements for Seismic Category I structures fully comply with ACI 318-71 and AISC 1969, whichever is applicable. Quality control in general meets the requirements of ANSI N45.2.5. Reinforcing bars are not welded in these structures.

3.8.4.7 Testing and Inservice Surveillance Requirements

No preliminary structural integrity or performance tests will be conducted. However, rigorous inspection techniques and the quality control procedures described in Appendix E are adopted throughout construction.

3.8.5 Foundations

3.8.5.1 Descriptions of Foundations

3.8.5.1.1 Main Building Complex

The reactor, auxiliary, and diesel-generator buildings are all supported on a common reinforced concrete mat foundation. See Figure 3.8-40 for a plan view of the mat and Figure 3.8-41 for sections through the mat. The 5-foot to 7-foot thick mat rests on undisturbed soil (see Subsection 3.8.1.1.2 for further description of the reactor building base slab). The mat under the auxiliary building is also common to the turbine and heater bay buildings which are non-Seismic Category I. Because of this connection and the resulting seismic interaction between buildings, the entire main building complex is modeled as a unit for the seismic analysis (Subsection 3.7.2). For a typical construction joint in the mat foundation see Figure 3.8-42. Shear resulting from shear wall action is transferred to the mat through shear-friction. Lateral shears on vertical load carrying elements are transferred to the mat through keys (Figure 3.8-42).

The equipment access building at the northeast corner of the reactor building is attached to the reactor building wall.

3.8.5.1.2 Lake Screen House

The foundation for the lake screen house consists of a monolithic 4-foot thick concrete mat.

3.8.5.2 Applicable Codes, Standards, and Specifications

The codes, specifications, standards of practice, and other accepted industry guidelines which are adopted to the extent applicable in the design fabrication, testing, inservice inspection, and construction of the foundations for Seismic Category I structures are found in Table 3.8-2. To eliminate repetition, these codes, standards and specifications which are described and discussed in Table 3.8-2 are given a specification reference number. Listed below are the reference numbers for the foundations.

a. 1 through 10; and

b. 14, 20, 21, and 22.

3.8.5.3 Loads and Loading Combinations

The loads and loading combinations listed and discussed in Subsection 3.8.4.3 are applicable to the design of the foundations. Refer to Tables 3.8-9 and 3.8-10 for the load definitions and the list of load combinations that are considered in design.

3.8.5.4 Design and Analysis Procedures

3.8.5.4.1 General

The analysis of Seismic Category I foundations is done using conventional elastic techniques. Design is based on the ACI 318-71 code. All loads interior and exterior to the buildings are transferred to the base mats through elastic deformation of the slabs, supporting shear walls, and columns. The design and analysis of the foundation complies with the portions of the ACI-349 Code, which are based on ACI-318.

3.8.5.4.2 Main Building Complex

The design of the reactor building base slab is discussed in Subsection 3.8.1.4.3. The auxiliary and diesel-generator building base slabs are analyzed by the STRESS computer program (Appendix F). Building shear walls also are included in the model to account for their stiffening effect. The slab is modeled far enough into the adjoining reactor and turbine building base slabs to minimize the effect of these structures on the analysis results. Settlements are taken into account by the soil springs modeled at each joint (See Subsection 2.5.4.10 for further discussion on settlement).

Lateral loads are transmitted to the soil through friction between the concrete and soil. Overturning is resisted through the resulting nonuniform bearing pressures in the soil (Subsection 2.5.4.10).

3.8.5.4.3 Lake Screen House

Elastic techniques are used to proportion the lake screen house's mat. The lake screen house's concrete retaining wall is analyzed by the STRUDL-II computer program (Appendix F). The steel sheet piling wall is analyzed with the free end Rankine Method for the static and dynamic condition.

Lateral active soil pressures and seismic forces are resisted by passive soil resistance and, in the case of the pump structure, by nonuniform bearing on the natural till.

3.8.5.5 Structural Acceptance Criteria

3.8.5.5.1 Structural Members

The acceptance criteria for the reactor building base slab are specified in Subsection 3.8.1.5.

The auxiliary and diesel-generator buildings base mats are proportioned according to the criteria set forth in Subsection 3.8.4.5.

The intake flume sheet piling and structural steel also conform to the acceptance criteria of Subsection 3.8.4.5.

3.8.5.5.2 Stability

3.8.5.5.2.1 Main Building Complex

As described in Subsection 3.8.5.4 the base mats are modeled on elastic soil springs and overturning is resisted by unequal bearing pressure. The seismic load combinations are most critical, and the resulting minimum factor of safety against overturning is 1.5 of OBE and 1.1 for SSE.

The potential for sliding is investigated by assuming a coefficient of friction between concrete and soil of 0.487 and neglecting any passive resistance against subgrade walls. The earthquake forces, which are combined for different directions as described in Section 3.7, are most critical in checking sliding, and the resulting minimum factors of safety for the building complex are 4.8 for OBE and 3.3 for SSE.

Potential for flotation is investigated by considering only the dead weight of the structures in resisting the buoyant forces of the PMF (Section 3.4), and any contributions from reduced live loads or frictional resistance of soil against subgrade walls are neglected. The resulting factor of safety against flotation of Seismic Category I structures is a minimum of 2.0.

3.8.5.5.2.2 Lake Screen House

The factor of safety against flotation for the lake screen house is 1.5.

For the lake screen house concrete retaining wall, the earthquake forces described in Section 3.7 are most critical in checking sliding and overturning. The resulting minimum factors of safety are 1.68 for OBE and 1.18 for SSE.

3.8.5.6 Materials, Quality Control, and Special Construction Techniques

The materials, quality control, and special construction techniques for foundations conform to those set forth for Seismic Category I structures and are discussed in Subsection 3.8.4.6.

3.8.5.7 Testing and Inservice Surveillance Techniques

Rigorous inspection during construction in conjunction with the quality control procedures for the structural materials outlined in Appendix E will be carried out. Structural integrity and/or performance tests are specified in Subsection 3.8.1.7 for the containment base slab.

3.8.6 References

1. LaSalle County Station, "Mark II-Design Assessment Report LSCS-DAR)," Commonwealth Edison Company, Chicago, Illinois, September, 1982.

2. Military Standard, "Sampling Procedures and Tables for Inspection by Variable for Percent Defective," MIL-STD-414, June 11, 1957.

3. Leigh, J. M. and Lay, M. G., "The Design of Laterally Unsupported Angles," BHP Technical Bulletin 13(3), Melbourne, Australia, November 1969.

4. Letter from E. G. Andensam, Nuclear Regulatory Commission, to D. L. Farrar, Commonwealth Edison Company, dated August 11, 1986; Subject: Safety Evaluation on the Design of Single Angle Members for HVAC Hanger Frames for LaSalle County Station, Units 1 & 2.

5. Design Specification DS-PA-01-LS, Revision 2, Table 5.5.

TABLE 3.8-1
(SHEET 1 of 10)

CONTAINMENT PENETRATIONS

PENETRATION NUMBER	SIZE (in.)	WALL TH (in.)	ELEV. (ft.)	AZIMUTH (degrees)	SLOPE (in/ft.)	DEVIATION	DESCRIPTION
M-1	46.0	1.375	742.4896	0.0	0.125	3.6667	Main Steam (includes Drain Line and MSIV-LCS Line)
M-2	46.00	1.375	742.4844	0.0	0.12500	11.00	Main Steam (includes Drain Line and MSIV-LCS Line)
M-3	46.00	1.375	742.4844	0.0	0.1250	- 11.0	Main Steam (includes Drain Line and MSIV-LCS Line)
M-4	46.00	1.375	742.4896	0.0	0.1250	- 3.6667	Main Steam (includes Drain Line and MSIV-LCS Line)
M-5	44.00	1.500	748.6667	0.0	0.0	7.3333	Reactor Feed (includes connection to RWCU)
M-6	44.00	1.500	748.6667	0.0	0.0	- 7.3333	Reactor Feed (includes connection to RWCU)
M-7	34.00	0.875	738.1250	146.0	0.0	0.0	RHRS/Shutdown Suction
M-8	22.00	1.125	737.8750	15.0	0.0	10.7709	RHRS/Shutdown Return
M-9	22.00	1.125	737.8750	243.0	0.0	0.0	RHRS/Shutdown Return
M-10	22.00	1.125	774.2917	123.2	0.0	0.0	LP Core Spray
M-11	22.00	1.125	774.2917	236.8	0.0	0.0	HP Core Spray
M-12	22.00	1.125	770.9583	0.0	0.0	2.7916	RHRS/LPCI
M-13	22.00	1.125	770.9583	158.0	0.0	0.0	RHRS/LPCI
M-14	22.00	1.125	770.9583	0.0	0.0	- 2.7916	RHRS/LPCI
M-15	28.00	0.500	749.5000	146.0	0.0	0.0	Steam to RCIC System (includes RHR supply)
M-16	14.00	0.438	742.5000	155.0	0.0	0.0	Cooling Water Supply
M-17	14.00	0.438	789.0000	210.0	0.0	0.0	Cooling Water Return
M-18	26.00	1.125	772.5000	110.5	0.0	0.0	RHRS Containment Spray
M-19	26.00	1.125	766.2500	340.0	0.0	0.0	RHRS Containment Spray
M-20	36.00	1.375	749.0000	328.5	0.0	0.0	Vent to Drywell (Drywell Purge)

TABLE 3.8-1
(SHEET 2 of 10)

CONTAINMENT PENETRATIONS

PENETRATION NUMBER	SIZE (in.)	WALL TH (in.)	ELEV. (ft.)	AZIMUTH (degrees)	SLOPE (in/ft.)	DEVIATION	DESCRIPTION
M-21	36.0	1.375	810.5833	30.00	0.0	0.0	Vent From Drywell; Drywell Pressure; RPV Level; Backfull (U-1)
M-21	36.00	1.375	802.75	74.0	0.0	0.0	Vent from Drywell, Drywell Pressure (U-2)
M-22	22.00	0.375	738.50000	0.0	0.0	7.33333	Main Steam Drain
M-23	30.00	0.750	799.0000	280.0	0.0	0.0	Spare (U-1)
M-23	30.000	0.750	799.250	74.0	0.0	0.0	Combustible gas control Drywell Suction (U-2)
M-24	30.00	0.750	793.5000	143.0	0.0	0.0	Spare
M-25	20.00	0.375	774.0000	330.0	0.0	0.0	Chilled Water Supply
M-26	20.00	0.375	774.0000	330.0	0.0	9.5	Chilled Water Supply
M-27	20.00	0.375	774.0000	330.0	0.0	3.1667	Chilled Water Return
M-28	20.00	0.375	774.0000	330.0	0.0	6.3333	Chilled Water Return
M-29	14.00	0.438	746.0000	29.0	0.0	0.0	RCIC RPV Head Spray including RHR head spray.
M-30	22.00	0.250	776.0000	33.0	0.0	0.0	Reactor Clean-up
M-31	12.75	0.375	779.0000	203.5	0.0	- 2.75	Containment high Rad. Detector
M-32	26.00	1.000	788.5000	77.0	0.0	0.0	Containment high Rad. Detector
M-33	26.00	1.000	788.5000	90.0	0.0	0.0	Combustible Gas control Drywell Suction (U-1); Spare (U-2)
M-34	10.75	0.365	779.0000	203.5	0.0	0.0	Standby Liquid Control
M-35	8.625	0.250	747.5000	155.0	0.0	0.0	Spare
M-36	8.625	0.250	756.0000	197.0	0.0	0.0	Recirc. Loop Sampling
M-37	10.75	0.365	742.5000	164.0	0.0	0.0	Clean Condensate
M-38	10.75	0.365	742.5000	197.0	0.0	0.0	Service Air

TABLE 3.8-1
(SHEET 3 of 10)

CONTAINMENT PENETRATIONS

PENETRATION NUMBER	SIZE (in.)	WALL TH (in.)	ELEV. (ft.)	AZIMUTH (degrees)	SLOPE (in/ft.)	DEVIATION	DESCRIPTION
M-39	10.75	0.365	743.0000	90.0	0.0	- 2.0	Spare
M-40 (a-d)	3.5	0.318	770'-4 1/4" to 774'-5 3/8"	270±7'6" 90±7'6"	0.0	0.0	Control Rod Drive (CRD) Insert Lines
M-41 (a-d)	3.5	0.318	770'-4 1/4" to 774'-5 3/8"	270±7'6" 90±7'6"	0.0	0.0	CRD Withdraw Lines
M-42	1.900	0.200	739.8859	254.0	-7.6448	4.4996	TIP Drive
M-43	1.900	0.200	739.8859	262.0	-7.6448	2.0988	TIP Drive
M-44	1.900	0.200	739.8859	270.0	-7.6448	0.0	TIP Drive
M-45	1.900	0.200	739.8859	278.0	-7.6448	- 2.0988	TIP Drive
M-46	1.900	0.200	739.8859	286.0	-7.6448	- 4.6676	TIP Drive
M-47	10.75	0.365	739.8859	270.0	-7.6448	- 11.6667	TIP Drive Air Supply
M-48	14.00	1.250	747.5000	243.0	0.0	- 1.5	Spare
M-49	16.00	0.375	768.7500	100.0	0.0	0.0	Recird. Pump Control Valve Hydraulic Piping
M-50	16.00	0.375	768.7500	280.0	0.0	0.0	Recirc. Pump Control Valve Hydraulic Piping
M-51	16.00	0.844	738.5000	0.0	0.0	- 7.3333	Spare
M-52	16.00	0.844	809.2500	225.0	0.0	0.0	Spare (U-1)
M-52	16.00	0.844	809.250	135.0	0.0	0.0	RPV Level, Backfill (U-2)
M-53	14.00	1.250	799.0000	250.0	0.0	0.0	Combustible Gas Control Drywell Suction (U-1)
M-53	14.00	1.250	799.000	110.0	0.0	0.0	Combustible Gas Control Drywell Suction (U-2)
M-54	14.00	1.250	768.7500	80.0	0.0	0.0	Spare (U-1); Air Dryer Blowdown, Drywell Pneumatic Comp. Suction/Disch. (U-2)
M-55	12.75	0.844	775.0000	203.0	0.0	3.0	ADS Pneumatic Supply
M-56	12.75	0.844	788.5000	110.0	0.0	0.0	Reactor Water Level; spare

TABLE 3.8-1
(SHEET 4 of 10)

CONTAINMENT PENETRATIONS

PENETRATION NUMBER	SIZE (in.)	WALL TH (in.)	ELEV. (ft.)	AZIMUTH (degrees)	SLOPE (in/ft.)	DEVIATION	DESCRIPTION
M-57	10.75	0.844	742.0000	29.0	0.0	0.0	Spare
M-58							Deleted
M-59	10.75	0.844	808.5000	267.0	0.0	0.0	Clean Cond. to Refueling Bellows; RPV Level; Backfill (U-1)
M-59	10.75	0.844	808.500	93.0	0.0	0.0	Clean Cond. to Refueling Bellows (U-2)
M-60	10.75	1.000	755.000	112.0	0.0	0.0	Drywell Pneumatic Compressor Discharge (U-1)
M-60	10.75	1.000	755.0000	112.0	0.0	0.0	Pneumatic Feed to ADS Valve (U-2)
M-61	10.75	0.365	743.0000	90.0	0.0	4.00	ADS Pneumatic Supply (U-1); Spare (U-2)
M-62	10.75	0.365	743.000	90.0	0.0	1.00	Drywell Air Compressor Suction; Spare (U-2)
M-63	8.625	0.250	747.0000	97.0	0.0	0.0	Recirc. Pump Seal Inject Supply
M-64	8.625	0.250	747.0000	277.0	0.0	0.0	Recirc. Pump Seal Inject Supply
M-65	18.00	0.375	809.2500	279.0	0.0	0.0	Reactor Bulkhead Drain (U-1)
M-65	18.00	0.375	809.250	82.5	0.0	0.0	Reactor Bulkhead Drain; RPV Level, Backfill (U-2)
M-66	34.00	0.500	726.0000	328.5	0.0	0.0	Vent to Suppression Chamber
M-67	34.00	0.500	727.5000	84.0	0.0	0.0	Vent From Suppression Chamber
M-68	32.00	0.625	681.1667	152.0	0.0	0.0	LPCS Suction From Suppression Pool
M-69	32.00	0.625	681.1667	277.5	0.0	0.0	HPCS Suction From Suppression Pool
M-70	32.00	0.625	681.1667	32.0	0.0	0.0	RHR (LPCI) Suction from Suppression Pool; Suppression Pool Water Level
M-71	32.00	0.625	681.1667	215.0	0.0	0.0	RHR (LPCI) Suction from Suppression Pool; Suppression Pool Water Level
M-72	32.00	0.625	681.1667	250.0	0.0	0.0	RHR (LPCI) Suction From Suppression Pool
M-73	12.75	0.375	727.5000	45.0	0.0	0.0	RHRS Suppression Spray Ring

TABLE 3.8-1
(SHEET 5 of 10)

CONTAINMENT PENETRATIONS

PENETRATION NUMBER	SIZE (in.)	WALL TH (in.)	ELEV. (ft.)	AZIMUTH (degrees)	SLOPE (in/ft.)	DEVIATION	DESCRIPTION
M-74	12.75	0.375	730.5000	234.5	0.0	0.0	RHRS Suppression Spray Ring
M-75	16.00	0.375	688.5000	131.0	0.0	0.0	RCIC Pump Suction from Suppression Pool
M-76	18.00	0.375	704.5000	123.0	0.0	0.0	RCIC Turbine Exhaust
M-77	26.00	0.500	703.5000	42.0	0.0	0.0	LPCS Min. Flow Line; LPCS test line; RCIC full flow test return to supp. pool
M-78	26.00	0.500	704.5000	235.0	0.0	0.0	Spare
M-79	22.00	0.375	704.7500	163.0	0.0	0.0	RHR Min. Flow Line, RHR Test Line
M-80	10.75	0.365	701.0000	136.0	0.0	0.0	RCIC Minimum Flow Line
M-81	10.75	0.365	704.5000	127.0	0.0	0.0	RCIC Vacuum Pump Discharge
M-82	22.00	0.375	703.5000	250.0	0.0	0.0	HPCS Testline; HPCS Min. flow line
M-83	12.75	0.375	701.0000	147.0	0.0	0.0	LPCS Safety Relief Valve Discharge
M-84	20.00	0.375	701.0000	320.0	0.0	0.0	RHR Min. Flow Line, RHR Test Line
M-85	12.75	0.375	703.5000	26.0	0.0	0.0	RHR Safety Relief Valve Discharge
M-86	12.75	0.375	704.5000	240.0	0.0	0.0	RHR Safety Relief Valve Discharge
M-87	12.75	0.375	704.5000	245.0	0.0	0.0	RHR Safety Relief Valve Discharge
M-88	14.00	0.375	704.5000	36.0	0.0	0.0	RHR Safety Relief Valve Discharge & Hx Vent Line
M-89	14.00	0.375	704.5000	217.0	0.0	0.0	RHR Safety Relief Valve Discharge & Hx Vent Line
M-90	10.75	0.365	704.5000	207.5	0.0	0.0	RHR Safety Relief Valve Discharge
M-91	10.75	0.365	701.0000	156.0	0.0	0.0	RHR Safety Relief Valve Discharge
M-92	12.75	0.375	704.5000	132.0	0.0	0.0	RCIC Safety Relief Valve Dishcarge
M-93	12.75	0.375	701.0000	152.0	0.0	0.0	LPCS Safety Relief Valve Discharge

TABLE 3.8-1
(SHEET 6 of 10)

CONTAINMENT PENETRATIONS

PENETRATION NUMBER	SIZE (in.)	WALL TH (in.)	ELEV. (ft.)	AZIMUTH (degrees)	SLOPE (in/ft.)	DEVIATION	DESCRIPTION
M-94	10.75	0.365	704.5000	280.0	0.0	0.0	HPCS Safety Relief Valve Discharge
M-95	12.75	0.375	701.0000	30.0	0.0	0.0	Combustible Gas Discharge Pipe Capped (spare)
M-96	12.75	0.375	718.500	322.0	0.0	6.50	Drywell Equipment Drain
M-97	12.75	0.375	718.5000	322.0	0.0	0.0	Drywell Equipment Drain Cooling
M-98	12.75	0.375	718.5000	322.0	0.0	3.25	Drywell Floor Drain
M-99	12.75	0.375	704.5000	142.0	0.0	0.0	RHR Safety Relief Valves Discharge
M-100	12.75	0.375	727.5000	120.0	0.0	0.0	Suppression Chamber Oxygen (Unit 1) Spare (Unit 2)
M-101	12.75	0.375	704.5000	114.0	0.0	0.0	RCIC Turbine Exhaust Vacuum-Breaker Line; RCIC SRV Discharge
M-102	12.75	0.375	701.0000	212.0	0.0	0.0	Combustible Gas Discharge Pipe Capped (Spare)
M-103	24.00	0.375	725.5000	320.0	0.0	0.0	Suppression Vacuum Breaker Line
M-104	24.00	0.375	725.5000	207.0	0.0	0.0	Suppression Vacuum Breaker Line; Supp. Pool Water Level; Combustible Gas Control Return
M-105	24.00	0.375	723.0000	19.0	0.0	6.7788	Suppression Vacuum Breaker Line; Supp. Pool Water level
M-106	24.00	0.375	725.5000	179.0	0.0	- 24.2317	Suppression Vacuum Breaker Line; Combustible Gas Control Return
M-107	24.00	0.375	751.0000	342.0	0.0	- 22.6893	Suppression Vacuum Breaker Line
M-108	24.00	0.375	751.5000	225.25	0.0	- 14.2881	Suppression Vacuum Breaker Line
M-109	24.00	0.375	754.000	32.0	0.0	0.0	Suppression Vacuum Breaker Line
M-110	24.00	0.375	752.708	166.0	0.0	- 10.1702	Suppression Vacuum Breaker Line
E-1	18.00	0.938	772.000	147.0	0.0	1.75	Spare
E-2	18.00	0.938	771.7500	325.0	0.0	- 3.0	Medium Voltage Power
E-3	12.75	0.688	754.000	90.0	0.0	- 3.0	Low Voltage Power

TABLE 3.8-1
(SHEET 7 of 10)

CONTAINMENT PENETRATIONS

PENETRATION NUMBER	SIZE (in.)	WALL TH (in.)	ELEV. (ft.)	AZIMUTH (degrees)	SLOPE (in/ft.)	DEVIATION	DESCRIPTION
E-4	12.75	0.688	756.2500	147.0	0.0	- 2.5	Low Voltage Power
E-5	12.75	0.688	756.5000	235.0	0.0	3.0	Low Voltage Power
E-6	12.75	0.688	755.0000	265.0	0.0	0.0	Low Voltage Power
E-7	12.75	0.688	754.0000	90.0	0.0	0.0	Low Voltage Control
E-8	12.75	0.688	756.2500	147.0	0.0	0.5	Low Voltage Control
E-9	12.75	0.688	756.5000	235.0	0.0	0.0	Low Voltage Control
E-10	12.75	0.688	755.0000	270.0	0.0	3.0	Low Voltage Control
E-11	12.75	0.688	754.0000	38.0	0.0	1.5	Neutron Monitoring
E-12	12.75	0.688	756.2500	147.0	0.0	3.5	Neutron Monitoring
E-13	12.75	0.688	756.5000	235.0	0.0	6.0	Neutron Monitoring
E-14	12.75	0.688	754.0000	322.0	0.0	0.5	Neutron Monitoring
E-15	12.75	0.688	754.0000	38.0	0.0	- 1.5	Control Rod Pos. Indication
E-16	12.75	0.688	754.0000	177.0	0.0	- 11.75	Control Rod Pos. Indication
E-17	12.75	0.688	755.0000	260.0	0.0	0.0	Control Rod pos. Indication
E-18	12.75	0.688	754.0000	322.0	0.0	- 2.5	Control Rod pos. Indication
E-19	12.75	0.688	770.0000	12.0	0.0	3.0	Instr. & Low Voltage Shielded
E-20	12.75	0.688	774.0000	225.0	0.0	3.0	Instr. & Low Voltage Shielded
E-21	12.75	0.688	770.0000	12.0	0.0	0.0	Instrumentation
E-22	12.75	0.688	775.0000	140.0	0.0	- 4.5	Spare
E-23	12.75	0.688	775.0000	140.0	0.0	- 1.5	Unit 1 – Spare; Unit 2 – Instrumentation.

TABLE 3.8-1
(SHEET 8 of 10)

CONTAINMENT PENETRATIONS

PENETRATION NUMBER	SIZE (in.)	WALL TH (in.)	ELEV. (ft.)	AZIMUTH (degrees)	SLOPE (in/ft.)	DEVIATION	DESCRIPTION
E-24	18.00	0.938	772.0000	147.0	0.0	- 1.75	Medium Voltage Power
E-25	12.75	0.688	774.0000	225.0	0.0	- 3.00	Spare
E-26	12.75	0.688	774.0000	225.0	0.0	0.0	Unit 1 – Spare; Unit 2 – Instrumentation.
E-27	18.00	0.938	771.7500	325.0	0.0	0.500	Spare
I-1	10.75	0.594	794.8333	85.0	0.25	0.0	Spare
I-2	3.500	0.3180	773.750	63.0	0.25	0.0	Reactor Pressure & Level
I-3	3.500	0.3180	775.0000	161.0	- 1.00	0.0	Spare
I-4	10.75	0.594	794.8333	105.0	0.25	0.0	Reactor Pressure & Level; (Spare); Drywell Humidity Monitor; Backfill
I-5	10.75	0.594	791.8333	210.0	0.25	0.0	Reactor Pressure & Level; (Spare); Drywell Humidity Monitor; Backfill
I-6	3.500	0.318	788.5000	213.0	0.25	0.0	Reactor Pressure & Level
I-7	3.500	0.318	794.8333	318.0	0.25	0.0	Reactor Pressure & Level; Backfill
I-8	10.75	0.594	794.83333	17.0	0.25	0.0	Reactor Pressure & Level; (Spare); Backfill; Drywell Pressure; RPV Head Seal Leak Detection (U-1)
I-9	10.75	0.594	774.50	105.0	0.25	0.0	Reactor Pressure & Level; (Spare); ADS Accumulator Pressure
I-10	10.75	0.594	775.0000	203.0	0.25	0.0	Reactor Press. & Level; RCIC Steam Flow & Pressure; Spare
I-11	30.00	0.375	775.0000	155.0	0.25	0.0	Primary Containment Air Sampling System; Post Loca Containment Monitor
I-12	10.75	0.594	774.0000	320.0	0.25	0.0	Reactor Pressure & Level; (Spare); ADS Accumulator Pressure
I-13	3.500	0.318	757.0000	80.0	- .25	0.0	Drywell Pressure
I-14	10.75	0.594	754.0000	28.0	0.25	0.0	Spare
I-15	10.75	0.594	725.5000	41.0	0.25	0.0	Main Steam Flow; RWCU Flow
I-16	10.75	0.594	749.0000	88.0	0.25	0.0	RHR Line "A" Integrity, RCIC Steam Flow & Pressure; LPCS/RHR Diff. Pressure; (Spare)

TABLE 3.8-1
(SHEET 9 of 10)

CONTAINMENT PENETRATIONS

PENETRATION NUMBER	SIZE (in.)	WALL TH (in.)	ELEV. (ft.)	AZIMUTH (degrees)	SLOPE (in/ft.)	DEVIATION	DESCRIPTION
I-17	10.75	0.594	755.0000	94.0	0.25	0.0	Jet Pump Head Pressure; (Spare)
I-18	3.500	0.318	756.0000	315.0	- .25	0.0	Drywell Pressure
I-19	10.75	0.594	755.0000	103.5	0.25	0.0	Jet Pump Flow "A" Loop
I-20	10.75	0.594	755.0000	107.0	0.25	0.0	Jet Pump Flow "A" Loop
I-21	10.75	0.594	755.0000	120.0	0.25	0.0	Spare
I-22	18.75	0.594	725.5000	95.0	0.25	0.0	Recirc. pump "A" Seal Cav. Pressure, Recirc. Pump "B" Flow, Recirc. Pump "A" Diff. Pressure
I-23	10.75	0.594	725.5000	99.0	0.25	0.0	Spare; Recirc. Pump "A" Suction Pressure; Recirc. Pump "A" Flow RHR Shutdown Flow
I-24	10.75	0.594	748.0000	162.0	0.25	0.0	Spare
I-25	10.75	0.594	756.0000	201.0	0.25	0.0	RHR Line Integrity; Spare
I-26	3.500	0.318	756.0000	205.0	- .25	0.0	Drywell Pressure
I-27	10.75	0.594	725.5000	261.0	0.25	0.0	Recirc. Pump "A" Flow RHR Shutdown Flow Recirc. Pump "B" Seal Cav. Pressure
I-28	10.75	0.594	725.5000	265.0	0.25	0.0	Recirc. Pump "B" Suction Pressure, Recirc. Pump "B" Diff. Pressure, Recirc. Pump "B" Flow; PRV Bottom Head Drain
I-29	10.75	0.594	725.5000	277.0	0.25	0.0	Main Steam Flow; Core Diff. Preessure; RPV Bottom Head Drain
I-30	10.75	0.594	725.5000	315.0	0.25	0.0	Vessel/HPCS Diff. Pressure; MSIV Accumulator Pressure
I-31	10.75	0.594	747.0000	267.0	0.25	0.0	Jet Pump Flow "B" Loop
I-32	10.75	0.594	747.0000	264.0	0.25	0.0	Jet pump Flow "B" Loop
I-33	3.500	0.318	755.0000	97.0	- .25	0.0	Drywell Pressure
I-34	10.75	0.594	725.5000	38.0	0.25	0.0	Main Steam Flow (Spare)
I-35	3.500	0.318	725.5000	325.0	- 1.00	0.0	Post LOCA Containment Monitoring
I-36	3.500	0.318	720.0000	154.0	- 1.00	0.0	Post LOCA Containment Monitoring

TABLE 3.8-1
(SHEET 10 of 10)

CONTAINMENT PENETRATIONS

PENETRATION NUMBER	SIZE (in.)	WALL TH (in.)	ELEV. (ft.)	AZIMUTH (degrees)	SLOPE (in/ft.)	DEVIATION	DESCRIPTION
I-37	10.75	0.594	725.5000	281.0	0.25	0.0	Main Steam Flow (Spare)
I-38	3.500	0.318	720.0000	17.0	0.0	0.0	Suppression Chamber Air Temp.
I-39	3.500	0.318	720.0000	198.0	0.0	0.0	Suppression Chamber Air Temp.
I-40	3.500	0.318	701.4167	25.0	0.25	0.0	Suppression Pool Water Level
I-41	3.500	0.318	698.8333	25.0	0.25	0.0	Suppression Pool Water Level
I-42	3.500	0.318	701.4167	204.0	0.25	0.0	Suppression Pool Water Level
I-43	3.500	0.318	698.8333	204.0	0.25	0.0	Suppression Pool Water Level
I-44	3.500	0.318	683.0000	17.0	0.0	0.0	Suppression Pool Water Temp
I-45	3.500	0.318	725.5000	154.0	- 1.00	0.0	Suppression Chamber Air Sampling; Drywell Humidity Sampling; Post LOCA Containment Monitoring
I-46	3.500	0.318	683.0000	197.0	0.0	0.0	Suppression Pool Water Temp
I-47	3.500	0.318	730.0000	320.0	- 1.00	0.0	Post LOCA Containment Monitoring
I-48	3.500	0.318	701.4167	345.0	0.0	0.0	Suppression Pool Water Level
I-49	3.500	0.318	698.8333	345.0	0.0	0.0	Suppression Pool Water Level
I-50	3.500	0.438	770.3542	270.0	0.0	0.0	Post LOCA Containment Monitoring
M-111	120.0	0.75	745.25	117.0	0.0	0.0	Personnel Access Airlock Hatch
M-112	146.00	0.750	744.375	220.0	0.0	0.0	Equipment Hatch
M-113	36.0	0.75	714.00	29.0	0.0	0.0	Emergency Suppression Pool Personnel Access Hatch
M-114	36.0	0.75	714.00	208.0	0.0	0.0	Emergency Suppression Pool Personnel Access Hatch
M-115	24.0	0.625	741.75	38.0	0.0	0.0	Control Rod Drive (CRD) Removal Hatch

Notes:
1. The penetration elevation is at the center of the penetration.
2. Positive indicated slope (+) is away from the Drywell. Negative indicated slop (-) is towards the Drywell.
3. Azimuthal deviation along x-axis (feet): positive (+) is clockwise, negative (-) is counter-clockwise.

TABLE 3.8-2
(SHEET 1 of 5)

LIST OF SPECIFICATIONS, CODES AND STANDARDS

SPECIFICATION REFERENCE NUMBER	SPECIFICATION OR STANDARD DESIGNATION	TITLE	EDITION	REMARKS
1	ACI 318	Building Code Requirements for Reinforced Concrete	1963	
2	ACI 318	Building Code Requirements for Reinforced Concrete	1971	
3	ACI 214	Recommended Practice for Evaluation of Compression Test Results	1965	
4	ACI 301	Specifications for Structural Concrete for Buildings	1972	Exceptions are listed in Appendix E
5	ACI 306	Recommended Practice for Cold Weather Concreting	1966	Additions are listed in Appendix E
6	ACI 315	Manual of Standard Practice for Detailing Reinforced Concrete Structures	1957	
7	ACI 347	Recommended Practice for Concrete Formwork	1968	
8	ACI 605	Recommended Practice for Hot Weather Concreting	1959	Exceptions are listed in Appendix E

TABLE 3.8-2
(SHEET 2 of 5)

LIST OF SPECIFICATIONS, CODES AND STANDARDS

SPECIFICATION REFERENCE NUMBER	SPECIFICATION OR STANDARD DESIGNATION	TITLE	EDITION	REMARKS
9	ACI 211.1	Recommended Practice for Selecting Proportions for Concrete	1970	Normal and Heavyweight
10	ACI-304 -73	Recommended Practice for Measuring, Mixing, and Placing Concrete	1973	
11	ACI-ASCE	Tentative Recommendations for Concrete Members Pre-stressed with Unbonded Tendons (Committee 423)	1969	
12	AISC	Manual of Steel Construction	1969	
13	ANSI B31.1.0	Standard Code for Pressure Piping, Power Piping	1967	
14	ANSI A123.1	Standard Nomenclature for Steel Door and Steel Door Frames	1967	
15	AWS D1.0	Code for Welding in Building Construction	Addenda of March 1965	

TABLE 3.8-2
(SHEET 3 of 5)

LIST OF SPECIFICATIONS, CODES AND STANDARDS

SPECIFICATION REFERENCE NUMBER	SPECIFICATION OR STANDARD DESIGNATION	TITLE	EDITION	REMARKS
16	AWS A3.0	Definitions for Welding and Cutting	1969	
17	AWS A5.1	Mild Steel Arc-Welding Electrodes	1969	
18	AWS A6.1	Recommended Safe Practice for Inert-Gas Metal-Arc Welding	1966	
19	AWS D12.1	Recommended Practice for Welding Reinforcing Steel	1971	
20	CRSI	Manual of Standard Practice	1970	
21	CRSI	Recommended Practice for Placing Reinforcing Bars	1968	
22	AISI	Light Gage Cold-Formed Steel Design Manual	1962	
23	ASTM	Annual Books of ASTM Standards	1972	For applicable ASTM Standards see Appendix E
24	ASA B1.1	Unified Inch Screw Threads	1960	
25	ASA B18.2	Square and Hexagonal Bolts and Nuts	1960	

TABLE 3.8-2
(SHEET 4 of 5)

LIST OF SPECIFICATIONS, CODES AND STANDARDS

SPECIFICATION REFERENCE NUMBER	SPECIFICATION OR STANDARD DESIGNATION	TITLE	EDITION	REMARKS
27	ASME	ASME Boiler and Pressure Vessel Code, Section III and Section IX	Summer of 1972 Addenda	
28	ASME	1971 ASME Boiler & Pressure Vessel Code, Material Specifications, Section II	Summer of 1972 Addenda	
29	ASME	ASME Boiler and Pressure Vessel Code, Section XI, "In Service Inspection of Nuclear Reactor Coolant System"	1974 Edition Summer of 1975 Addenda	
30	API Spec No 620	Specification for Welded Steel Storage Tanks	February 1970	
31	Standard Assoc of Australia AS1250	The use of Steel in Structures	1981	

TABLE 3.8-2
(SHEET 5 of 5)

LIST OF SPECIFICATIONS, CODES AND STANDARDS

EXPLANATION OF ABBREVIATIONS

ACI American Concrete Institute

ASCE American Society of Civil Engineers

AISC American Institute of Steel Construction

PCI Prestressed Concrete Institute

AISI American Iron and Steel Institute
(ASA)

ANSI American National Standards Institute

AWS American Welding Society

ASTM American Society for Testing and Materials

CRSI Concrete Reinforcing Steel Institute

ASME American Society of Mechanical Engineers

API American Petroleum Institute

TABLE 3.8-3
(SHEET 1 of 2)

LOAD DEFINITIONS AND COMBINATIONS FOR

PRIMARY CONTAINMENT AND DRYWELL FLOOR

(For Definitions See Table 3.8-11)

LOAD FACTORS**																
LOADING CATEGORY	ITEM NO.	GENERAL							SEVERE ENVIRONMENTAL		ABNORMAL				EXTREME ENVIRONMENTAL	
		D	F	L	R_o	T_o	P_o	P_t	E_o	W	R_r	T_a	P_a	R_a	E_{ss}	W_t
Construction	1	1.0	1.0	1.0		1.0										
Test	2	1.0	1.0	1.0		1.0*		1.0								
Normal	3	1.0	1.0	1.0	1.0	1.0	1.0									
Normal	4	1.5	1.0	1.8	1.0	1.0										
Severe	5	1.25	1.0	1.0	1.0	1.0			1.7							
Environmental	6	1.25	1.0	1.0	1.0	1.0				1.7						
	7	.9	1.0		1.0	1.0			1.7							
	8	.9	1.0		1.0	1.0				1.7						
Extreme	9	1.0	1.0	1.0	1.0	1.0									1.0	
Environmental	10	1.0	1.0	1.0	1.0	1.0										1.0

* Temperature at time of test.

** If for any combination the effect of any load other than D reduces the total load, it is deleted from the combination. For load combinations with SRV and LOCA loads, see Reference 1.

TABLE 3.8-3
(SHEET 2 of 2)

LOAD DEFINITIONS AND COMBINATIONS FOR

PRIMARY CONTAINMENT AND DRYWELL FLOOR

(For Definitions See Table 3.8-11)

LOAD FACTORS**																
LOADING CATEGORY	ITEM NO.	GENERAL							SEVERE ENVIRONMENTAL		ABNORMAL				EXTREME ENVIRONMENTAL	
		D	F	L	R_o	T_o	P_o	P_t	E_o	W	R_r	T_a	P_a	R_a	E_{ss}	W_t
Abnormal	11	1.0	1.0	1.0								1.0	1.5	1.0		
Abnormal/Severe	12	1.0	1.0	1.0					1.25			1.0	1.25	1.0		
Environmental	13	1.0	1.0	1.0						1.25		1.0	1.25	1.0		
Abnormal/Extreme Environmental	14	1.0	1.0	1.0							1.0	1.0	1.0	1.0	1.0	

TABLE 3.8-4
(SHEET 1 of 2)

ALLOWABLE STRESSES AND STRAINS-CONTAINMENT LINER PLATE AND ANCHORAGES

For the construction loading category, the stress allowable for membrane and combined membrane plus bending is:

$f_{st} = f_{sc} = 0.67\ S_y$

For all normal loading categories, the strain allowables are:

sc = 0.002 in./in.)) st = 0.001 in./in.)	membrane
sc = 0.004 in./in.)) st = 0.002 in./in.)	combined membrane and bending

For all abnormal loading categories, the strain allowables are:

sc = 0.005 in./in.)) st = 0.003 in./in.)	membrane
sc = 0.014 in./in.)) st = 0.010 in./in.)	combined membrane and bending

The allowables are defined as:

f_{st} = allowable liner plate tensile stress, psi;
f_{sc} = allowable liner plane compressive stress, psi;
S_y = yield strength, psi;
sc = allowable liner plate compressive strain; and
st = allowable liner plate tensile strain.

The allowable forces and displacements for the liner plate anchors are limited to the following values: for all normal and severe environmental loading categories, the force/displacement allowables are:

a. the lesser of $F_a = 0.67\ F_y$ and $F_a = 0.33\ F_u$, for mechanical loads; and

TABLE 3.8-4
(SHEET 2 of 2)

ALLOWABLE STRESSES AND STRAINS-CONTAINMENT LINER PLATE AND ANCHORAGES

b. $\delta_a = 0.25\delta_u$, for displacement-limited loads.

For all extreme environmental loading categories, the force/ displacement allowables are:

a. the lesser of F_a = 0-9 F_y and $F_a = 0.5\ F_u$, for mechanical loads; and

b. $\delta_a = 0.25\delta_u$, for displacement-limited loads.

The allowables are defined as:

F_a = allowable liner anchor force capacity, lb;

F_u = liner anchor ultimate force capacity, lb;

F_y = liner anchor yield force capacity, lb;

δ_a = allowable displacement for liner anchors, in.; and

δ_u = ultimate displacement for liner anchors, in.

The liner allowables under safety/relief valve discharge loads are given in Reference 1.

TABLE 3.8-5
(SHEET 1 of 2)

PREDICTED RESPONSE READINGS FOR CONTAINMENT
UNDER 52 PSIG TEST PRESSURE*

METER	PREDICTED STRAIN (x 10^{-4} in./in.) UNIT 1	UNIT 2
BS 1	-0.607	0.183
BS 2	-0.129	0.802
BS 3	0.183	-0.129
BS 4	0.802	-0.607
CW 5	0.104	1.157
CW 6	1.157	0.104
CW 7	-0.052	0.682
CW 8	0.125	-0.052
CW 9	0.682	0.125
CW 10	0.132	0.132
CW 11	-0.140	-0.140
CW 12	0.320	1.450
CW 13	1.450	0.710
CW 15	1.450	1.450
CW 16	0.710	0.320
DF 18	-0.536	-0.536
DF 19	0.439	0.439
DF 20	1.312	1.312
DF 21	0.255	0.255
CW 22	0.743	0.743
CW 23	0.318	0.318
CW 25	0.318	0.033
CW 26	0.033	0.318
CW 27	0.764	0.764
CW 28	0.246	0.246
CW 30	0.317	0.317
CW 31	0.740	0.740
CW 32	0.393	0.644
CW 33	0.644	0.393
CW 34	-0.130	0.240
CW 35	0.240	0.395
CW 36	0.395	-0.130
CW 37	-0.137	-0.137
CW 38	0.397	0.397
BT 39	0.925	0.925
BT 40	0.249	-0.284
BT 41	-0.284	-0.301
BT 42	0.301	0.249
BT 43	-0.145	-0.145
BT 44	0.349	-0.349
BT 45	0.347	0.189
BT 46	0.189	0.347
BT 47	0.013	0.403
BT 48	0.169	0.169
BT 49	0.403	0.169
BT 50	0.169	0.013

* For instrument location see Figure 3.8-31.

TABLE 3.8-5
(SHEET 2 of 2)

PREDICTED RESPONSE READINGS FOR CONTAINMENT
UNDER 52 PSIG TEST PRESSURE*

METER	PREDICTED STRAIN (x 10^{-4} in./in.) UNIT 1	UNIT 2
EQ 51	1.377	1.377
EQ 52	0.743	0.743
EQ 53	1.305	-0.222
EQ 54	-0.222	1.305
EQ 55	1.249	1.249
EQ 56	0.590	0.590
EQ 57	0.660	0.660
EQ 58	0.201	0.201
EQ 59	1.610	1.610
EQ 60	0.559	0.715
EQ 61	0.715	0.559
EQ 62	0.172	0.172
EQ 63	0.696	0.696
EQ 64	0.028	0.028
EQ 65	0.314	0.314
EQ 66	0.746	0.314
EQ 67	0.314	0.746

	PREDICTED STRESS (KSF) UNIT 1	UNIT 2
CW 14	40.664	40.664
DF 17	30.400	30.400
CW 24	27.632	27.632
CW 29	23.496	23.496

	PREDICTED DISPLACEMENT (IN) UNIT 1	UNIT 2
V1 to V6	-0.009	-0.009
V7 to V12	0.187	0.187
D1 to D6	0.047	0.047
D7 to D12	0.081	0.081
D13 to D18	0.020	0.020
D19 to D24	0.038	0.038
D25 to D30	0.025	0.025
D31	0.048	0.048
D32	0.047	0.047
D33	0.036	0.036
D34	0.030	0.030
D35	0.025	0.025
D36	0.015	0.015
D37	0.021	0.021
D38	0.020	0.020
D39	0.024	0.024
D40	0.030	0.030
D41	0.025	0.025
D42	0.015	0.015

* For instrument location see Figure 3.8-31.

TABLE 3.8-6

LOAD DEFINITIONS AND COMBINATIONS FOR CLASS MC CONTAINMENT COMPONENTS

(OTHER THAN PIPING PRESENTATIONS)

(For Definitions See Table 3.8-11)

LOAD FACTORS															
LOADING CATEGORY	ITEM NO.	GENERAL							SEVERE ENVIRONMENTAL	ABNORMAL					EXTREME ENVIRONMENTAL
		D	L	R_o	T_o	P_o	P_p	P_t	E_o	R_r	T_a	P_a	R_a	M_a	E_{ss}
Construction	1	1.0	1.0		1.0										
Test	2	1.0	1.0		1.0*			1.0							
Normal	3	1.0	1.0	1.0	1.0	1.0	1.0								
Severe Environmental	4	1.0	1.0	1.0	1.0	1.0	1.0		1.0						
Abnormal	5	1.0	1.0				1.0				1.0	1.0	1.0		
	6	1.0	1.0				1.0			1.0	1.0	1.0	1.0		
Extreme Environmental	7	1.0	1.0	1.0	1.0	1.0									1.0
Abnormal/Severe Environmental	8	1.0	1.0				1.0		1.0		1.0	1.0	1.0		
	9	1.0	1.0						1.0		1.0		1.0	1.0	
	10	1.0	1.0				1.0		1.0	1.0	1.0	1.0	1.0		
Abnormal/Extreme Environmental	11	1.0	1.0				1.0			1.0	1.0	1.0	1.0		1.0
	12	1.0	1.0				1.0			1.0	1.0	1.0	1.0		1.0

* Temperature at time of test.

TABLE 3.8-7

PHYSICAL PROPERTIES FOR MATERIALS TO BE USED FOR PRESSURE PARTS OR ATTACHMENT TO PRESSURE PARTS MC COMPONENTS

MATERIAL SPECIFICATION	Su MINIMUM ULTIMATE TENSILE (ksi)	Sy MINIMUM YIELD AT THE AMBIENCE (ksi)	Sy MINIMUM YIELD AT 340°F (ksi)	Sm ASME CODE ALLOWABLE STRESS INTENSITY AT 340°F (ksi)	NOTES
Plate					
SA516 Gr 70	70	38	33.26	17.5	
SA516 Gr 60	60	32	27.94	15	
SA240 Tp 304	75	30	21.78	16.44	
Pipe					
SA106 Gr B	60	35	30.6	15	
SA333 Gr 6	60	35	30.6	15	
SA333 Gr 1	55	30	26.24	13.75	
SA312 Type 304	75	30	21.78	16.44	
SA376 Type 304	75	30	21.78	16.44	
Forgings and					
Fittings					
SA350 LF-1	60	30	26.24	15.0	
SA350 LF-2	70	36	31.96	17.5	
SA182 F304	70	30	21.78	16.44	
SA182 Gr-F316	75	30	18.22	18.28	
Bolting					
SA193 B7	115	95	83.98	23	Between 2.5 in. and 4 in. dia.
SA193 B7	125	105	93.06	25	Under 2.5 in. dia.
SA194 Gr 7*	---	---	---	---	
SA320 L43	125	105	94.14	25	4 in dia. and under

* No yield or tensile strength specified. Assume it is the same as an equivalent grade in SA-193-B7.

TABLE 3.8-8

LOAD DEFINITIONS AND COMBINATIONS FOR REACTOR PEDESTAL

(For Definitions See Table 3.8-11)

LOAD FACTORS**												
LOADING CATEGORY	ITEM NO.	GENERAL					SEVERE ENVIRONMENTAL	ABNORMAL				EXTREME ENVIRONMENTAL
		D	L	R_o	T_o	P_t	E_o	R_r	T_a	P_a	R_a	E_{ss}
Construction	1	1.0	1.0		1.0							
Test	2	1.0	1.0		1.0*	1.0						
Normal	3	.9	1.5	1.5	1.5							
	4	1.5	1.7									
Severe Environmental	5	.9	1.5	1.5	1.5		1.5					
	6	1.5	1.5	1.5	1.5		1.5					
	7	1.1	1.7				1.9					
Extreme Environmental	8	1.0	1.0	1.0	1.0							1.0
Abnormal	9	1.0	1.0						1.0	1.5	1.0	
Abnormal/Severe Environmental	10	1.0	1.0				1.25		1.0	1.25		
Abnormal/Extreme Environmental	11	1.0	1.0					1.0	1.0	1.0	1.0	1.0

* Temperature at time of test.

** If for any combination the effect of any load other than D reduces the total load, it is deleted from the combination. For load combinations with SRV and LOCA loads, see Reference 1.

TABLE 3.8-9
(SHEET 1 of 2)

SEISMIC CATEGORY I STRUCTURES
LOAD COMBINATION - STRUCTURAL STEEL ELASTIC DESIGN
(For Definition See Table 3.8-11)

LOADS*																
LOADING CONDITION	ITEM NO.	GENERAL					SEVERE ENVIRONMENTAL		ABNORMAL				EXTREME ENVIRONMENTAL			DESIGN STRESS
		D	L	S	R_o	T_o	E_o	W	R_r	T_a	P_a	R_a	E_{ss}	W_t	H'	
Construction	1	1.0	1.0	1.0												1.33 AISC Allowable
	2	1.0	1.0					1.0								1.33 AISC Allowable
Test	3	1.0	1.0	1.0												1.33 AISC Allowable
Normal	4	1.0	1.0	1.0	1.0											AISC Allowable
Severe Environmental	5	1.0	1.0		1.0		1.0									AISC Allowable
	6	1.0	1.0		1.0			1.0								AISC Allowable
Abnormal	7	1.0	1.0	1.0						1.0	1.0	1.0				Flexure: 1.6 AISC Allowable $\leq 0.95\ F_y$ Shear: 1.6 AISC Allowable $\leq 0.95\ F_y/\sqrt{3}$
Extreme	8	1.0	1.0		1.0	1.0							1.0			Flexure: 1.6 AISC Allowable $\leq 0.95\ F_y$ Shear: 1.6 AISC Allowable $\leq 0.95\ F_y/\sqrt{3}$

NOTE: In loading combination 1-6 the design stress is increased to 1.5 AISC Allowable when T_o is considered.
* For load combinations with SRV and LOCA loads, see Reference 1.

TABLE 3.8-9
(SHEET 2 of 2)

		LOADS*														
LOADING CONDITION	ITEM NO.	GENERAL					SEVERE ENVIRONMENTAL		ABNORMAL				EXTREME ENVIRONMENTAL			DESIGN STRESS
		D	L	S	R_o	T_o	E_o	W	R_r	T_a	P_a	R_a	E_{ss}	W_t	H'	
Environmental	9	1.0	1.0			1.0							1.0			Flexure: 1.6 AISC Allowable $\leq 0.95\ F_y$ Shear: 1.6 AISC Allowable $\leq 0.95\ F_y/\sqrt{3}$
	10	1.0	1.0			1.0									1.0	Flexure: 1.6 AISC Allowable $\leq 0.95\ F_y$ Shear: 1.6 AISC Allowable $\leq 0.95\ F_y/\sqrt{3}$
Abnormal/Severe Environmental	11	1.0	1.0				1.0		1.0	1.0	1.0	1.0				Flexure: 1.6 AISC Allowable $\leq 0.95\ F_y$ Shear: 1.6 AISC Allowable $\leq 0.95\ F_y/\sqrt{3}$
Abnormal/Extreme Environmental	12	1.0	1.0						1.0	1.0	1.0	1.0	1.0			Flexure: 1.6 AISC Allowable $\leq 0.95\ F_y$ Shear: 1.6 AISC Allowable $\leq 0.95\ F_y/\sqrt{3}$

NOTE: In loading combination 1-6 the design stress is increased to 1.5 AISC Allowable when T_o is considered.

* For load combinations with SRV and LOCA loads, see Reference 1.

TABLE 3.8-10

SEISMIC CATEGORY I STRUCTURE LOAD COMBINATION - REINFORCED CONCRETE STRUCTURES OTHER THAN CONTAINMENT

(For Definitions See Table 3.8-11)

LOADS*																
LOADING CONDITION	ITEM NO.	GENERAL					SEVERE ENVIRONMENTAL		ABNORMAL				EXTREME ENVIRONMENTAL			DESIGN STRENGTH
		D	L	F	R_o	T_o	E_o	W	R_r	T_a	P_a	R_a	E_{ss}	W_t	H'	
Construction	1	1.4	1.7	1.0												ACI 318-71
	2	1.05	1.25	1.0		1.05		1.3								ACI 318-71
	3	.9	1.3	1.0		1.3		1.3								ACI 318-71
Normal	4	1.4	1.7	1.0	1.3	1.3										ACI 318-71
Severe Environmental	5	1.05	1.3	1.0	1.05	1.05		1.3								ACI 318-71
	6	0.9	1.3	1.0	1.3	1.3		1.3								ACI 318-71
	7	1.05	1.3	1.0	1.05	1.05	1.4									ACI 318-71
	8	0.9	1.3	1.0	1.3	1.3	1.4									ACI 318-71
Abnormal	9	1.0	1.0	1.0						1.0	1.5	1.0				Yield Line Theory
	10	1.0	1.0	1.0	1.0	1.0							1.0			ACI 318-71
	11	1.0	1.0	1.0	1.0	1.0								1.0		ACI 318-71
	12	1.0	1.0	1.0	1.0	1.0									1.0	ACI 318-71
Abnormal/Severe Environmental	13	1.0	1.0	1.0			1.25		1.0	1.0	1.25	1.0				Yield Line Theory
Abnormal/Extreme Environmental	14	1.0	1.0	1.0					1.0	1.0	1.0	1.0	1.0			Yield Line Theory

* For load combinations with SRV and LOCA loads, see Reference 1.

TABLE 3.8-11
(SHEET 1 of 4)

DEFINITIONS OF STRUCTURAL TERMINOLOGY

LOADING CATEGORIES

Construction

All events and loads during structural construction including the various stages of prestressing, but excluding those during testing.

Testing

All events and loads applied during structural integrity tests and preoperational tests such as hydrostatic testing of equipment and the pressure tests. Each testing event is considered to be mutually exclusive of other testing events.

Normal

All events and loads that could reasonably be expected during the operation, shutdown, and normal maintenance of the power plant. The magnitude of these events and loads based on probability of one in the design life of the plant.

Severe Environmental

All loads due to infrequent site-related environmental events like operating basis earthquake and design wind.

Abnormal

All loads due to postulated accident events. They include pressure, temperature, pipe whip, jet impingement, and pipe reactions due to each rupture postulated for the design-basis accidents. This loading condition also includes plant-related nonenvironmental missiles. The loads from each postulated accident event are considered to be mutually exclusive of other postulated accidents.

Extreme Environmental

All loads due to site-related environmental events which are credible but highly improbable. These events include the safe shutdown earthquake, design-basis tornado, probable maximum flood, and the postulated site-related accidents not included in the abnormal loading category.

Abnormal/Severe Environmental

Loads due to the highly improbable simultaneous occurrence of abnormal and severe environmental loading categories. Only the specified combinations of these categories, are considered.

Abnormal/Extreme Environmental

Loads due to the extremely improbable simultaneous occurrence of the abnormal and extreme environmental loading conditions. Only the specified combinations of these conditions are considered.

LOAD DEFINITIONS

General Loads

D = Dead load of the structure or its related internal moments and forces including any permanent equipment, soil or hydrostatic pressure. Construction loading is indicated as dead load for the construction combination. Prestartup loads are also included.

F = loads resulting from the application of prestress.

L = Live loads or their related internal moments and forces including any movable equipment loads and other loads which vary in intensity and occurrence such as roof loads and crane loads. Appropriate impact factors are included for such moving loads as from trolleys and cranes.

S = Stability loading applicable to all steel framing systems.

T_o = Most critical transient or steady-state thermal load condition on the structure at normal operation or shutdown conditions. This also includes other thermal effects such as frictional loads due to expansion, unless otherwise indicated.

Pools	+152° F
Drywell	+135° F
Suppression Chamber	+90° F
External Reactor	
Building walls	+70° F

R_o = Pipe, cable pan, and duct reactions due to:

1. self-weight including contents,

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2. critical transient or steady state thermal condition at normal operating or shutdown conditions, and

3. effects of unbalanced pressure and thrust.

P_o = Normal expected operating pressure range,

Internal pressure on the containment = 2 psig

External pressure on the containment = 1 psig

Maximum internal pressure on the secondary containment = negative .25 inch of water

P_p = Operating differential peak pressure for process piping penetrations.

P_t = Containment test pressure = 52 psig. (For drywell floor test pressure, see Subsection 3.8.3)

<u>Severe Environmental Loads</u>

E_o = Seismic excitations from the operating basis earthquake (see Section 3.7). The seismic effects include loads from structure, equipment, piping, cable pans, dynamic soil, hydrodynamic pressures, snow and all other items that could be considered as inertial forces for seismic analysis.

W = Design wind velocity loads (see Section 3.3).

<u>Extreme Environmental Loads</u>

E_{ss} = Seismic excitation from the safe shutdown earthquake (see Section 3.7). The seismic effects include loads from structure, equipment, piping cable pans, dynamic soil, hydrodynamic pressures, and snow.

W_t = Design-basis tornado loads (see Section 3.3). These include effects of:

1. translational and rotational velocity pressure,

2. atmospheric pressure change,

3. tornado-generated missile impact effects (see Section 3.5), and

4. enclosure blow-off loads for reactor building superstructure.

H' = Force associated with maximum possible flood or seiche.

Abnormal Loads: For the design-basis accident under consideration.

P_a = Maximum differential pressure generated by a postulated pipe break including an appropriate margin to account for dynamic effects and the uncertainty in the calculation. This includes P_o for all other areas not affected by pipe breaks

internal P_a on the containment = 45 psig

external P_a on the containment = 5 psig

main steam tunnel P_a = 19.5 to 29.5 psig

internal pressure on the secondary containment = positive .25 psig

external pressure on the secondary containment = wind load, soil pressures hydrostatic pressures as applicable

T_a = Effects of thermal environment on the structure generated by a postulated pipe break. This includes T_a for all other areas not affected by the pipe break.

peak drywell	340° F
peak suppression chamber	275° F

R_a = Effects of thermal environment on the pipe and equipment reactions generated by a postulated pipe break. This includes R_o for all other areas not affected by the pipe break.

R_r = Effects on the structure generated by a postulated pipe break including appropriate dynamic load factors to account for the dynamic nature of the loads. These loads include:

1. reactions from pipe supports and whip restraint,

2. jet impingement, and

3. missile impact due to a postulated ruptured pipe.

TABLE 3.8-12

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TABLE 3.8-13
(SHEET 1 of 2)
ALLOWABLE STRESSES FOR
PENETRATION SLEEVES AND HEAD FITTINGS

STRESS CATEGORY		ALLOWABLE STRESS VALUES FOR EACH LOADING CONDITION (NOTE 1)			
		NORMAL AND UPSET	DESIGN (NOTE 3)	EMERGENCY (NOTE 3)	FAULTED (NOTES 3 & 4)
PRIMARY STRESSES	GENERAL MEMBRANE (P_m)	(Note 2)	S_m	The larger of $1.2S_m$, or S_y	The larger of $0.7S_u$, or $S_y + (S_u - S_y)/3$
	LOCAL MEMBRANE (P_L)	(Note 2)	$1.5S_m$	The larger of $1.8S_m$, or $1.5S_y$	The larger of $1.05S_u$, or $1.5S_y + (S_u - S_y)/2$
	MEMBRANE + BENDING ($P_L + P_B$)	(Note 2)	$1.5S_m$	The larger of $1.8S_m$, or $1.5S_y$	The larger of $1.05S_u$, or $1.5S_y + (S_u - S_y)/2$
SECONDARY STRESSES	EXPANSION STRESSES	$3S_m$			
	PRIMARY + SECONDARY ($P_L + P_B + P_e + Q$)	$3S_m$			
PEAK STRESSES	(F)	(Note 5)			

NOTES:

1. Values for S_m, S_y , and S_u , shall be temperature-dependent and taken from Section III Tables, as follows: S_y from Tables I-2.0; S_u from Tables I-3.0; S_m from Tables I-1.0 for non-MC components, and from Tables I-10.0 for MC components.

2. There are no specific limits established on the primary stresses that result from operating conditions.

3. Design, emergency, and faulted conditions do not require secondary and peak stress evaluation.

4. The specified stress limits for faulted conditions are applicable for system inelastic and component elastic evaluation.

5. Used in combination with all primary and secondary stresses for calculating alternating stresses (for fatigue evaluation).

3.9 MECHANICAL SYSTEMS AND COMPONENTS

3.9.1 Special Topics for Mechanical Components

3.9.1.1 Design Transients

This subsection describes the transients which were used to demonstrate the design of the ASME Code Class 1 core supports, reactor components, piping systems and mechanical equipment. The transients and combinations of transients are classified with respect to the component operating condition categories identified as "normal," "upset," "emergency," "faulted," or "test" in the ASME Boiler and Pressure Vessel Code, as applicable. (The first four operating condition categories correspond to Service Levels A, B, C, and D, respectively.)

3.9.1.1.1 Thermal Transients

The thermal transients used in the design and fatigue analysis of ASME Code Class 1, core supports, reactor components, piping systems and mechanical equipment are listed in Table 3.9-24. These thermal transients were derived from those established for the RPV nozzles and are tabulated for the major piping systems connected to the RPV. Other reactor coolant pressure boundary lines which connect to these major lines were assumed to have identical thermal transients.

3.9.1.1.2 Hydrodynamic Transients

The hydrodynamic transients associated with safety/relief valve (SRV) actuations and postulated LOCA events are described in the DFFR (Reference 4). These hydrodynamic transients were defined subsequent to design, procurement and delivery of the LSCS mechanical systems and components. The effects of the hydrodynamic transients on plant structures are presented in the LSCS-DAR (Reference 9).

The static and dynamic load effect on the mechanical systems and components resulting from the hydrodynamic transients were evaluated in a design assessment. This design assessment evaluation considered the dynamic response of the containment and structures located on the containment base mat and the response on mechanical systems and components resulting from structural excitations. The load combinations used for the design assessment evaluation are defined in Table 3.9-16. FSAR Tables 3.9-2 through 3.9-11, 3.9-18, 3.9-22 and 3.9-37 also listed the calculated stresses or other design values at the time of licensing. Modification, performed after licensing, that effects any calculated stress or design value will be evaluated against accepted allowables at the time of the modification. Results of these evaluations will be listed in the applicable stress report or analysis.

Descriptions of the hydrodynamic transient loads and the shield wall annulus pressurization loads are provided in the following paragraphs.

3.9.1.1.2.1 Safety/Relief Valve (SRV) Actuation Loads (Structural Excitations)

The actuation of SRV's causes pressure disturbances in the suppression pool water that produce oscillatory transient pressure forces on the suppression pool boundary. These pressure oscillations result in structural excitations which impart dynamic responses to the attached piping and equipment.

Safety/relief valve hydrodynamic transients and associated calculational procedures are further described in References 1 and 7. Structural response loads due to the following SRV actuations were evaluated:

a. actuation of all valves,

b. actuation of the automatic depressurization system (ADS) valves,

c. the actuation of three adjacent valves, and

d. actuation of a single valve (subsequent actuation), and

e. actuation of lowest setpoint group of valves.

Reference 7 describes the characteristics of the SRV load as a function of the piping configuration and the discharge device (rams head or quencher) located at the exit of the SRV line. The quencher device typically produces lower dynamic loads. The LSCS design assessment evaluation was conservatively based on loads calculated with a T-quencher device.

In response to NUREG 0519, Safety Evaluation Report, a safety/relief valve in-plant test on Unit 1 was conducted. The test results demonstrated the adequacy of existing design basis hydrodynamic loads resulting from safety/relief valve actuation. Details are included in the "LaSalle County I In-Plant S/RV Test Initial Evaluation Report", submitted to the NRC with a letter from C.W Schroeder to A. Schwencer, March 4, 1983.

3.9.1.1.2.2 Loss-of-Coolant Accident (LOCA) Loads

The postulated LOCA event gives rise to several hydrodynamic phenomena which cause transient pressure loads on the suppression pool boundary. These phenomena include main vent clearing, pool swell, condensation oscillation, and chugging. A brief description of each follows:

a Main Vent Clearing

Following a postulated LOCA, the drywell pressure increases due to blowdown of the reactor system. Pressurization of the drywell will cause the water initially in the vent system to be accelerated out through the vents. During this water expulsion process the resulting water jets cause impingement loads on local containment structures. However, this water clearing process has been found to produce insignificant structural response loads on the drywell piping and equipment.

b. Pool Swell

Following main vent clearing, an air/steam bubble forms at the vent exit. This causes a hydrostatic pressure increase in the pool water resulting in a loading condition on the pool boundaries. The steam condenses in the pool. However, the continued addition and expansion of the drywell air causes the pool volume to swell, resulting in the rise of the pool surface and associated drag and impact loads on surrounding structures. Reference 1 provides further details and calculational procedures.

c. Condensation Oscillation Loads

Evaluation of test results for the steam condensation cycle has revealed the occurrence of a dynamic load during high mass-flow of steam into the suppression pool. This low-pressure, symmetric, sinusoidal pressure fluctuation occurs over a low frequency range which acts on the pool boundary. These fluctuating pressures excite the structure producing low-frequency responses on the drywell piping systems and equipment. Reference 1 (Revision 3) provides further description and calculation procedures.

d. Chugging

The application of chugging loads is described in the "Mark II Phase I - 4T Tests Application Memorandum" submitted to the NRC in June 1976. This Application Memorandum was used for the LSCS Design Assessment Evaluation to expedite licensing review. Additional methods for the application of chugging loads are being developed (NEDO-24014, June 1977; NEDE-21669-P, February 1978). These new methods for the application of multivent chugging loads provide a realistic load definition, and the results of these new methods, when completed, will provide the final design-basis load definition.

3.9.1.1.3 Annulus Pressurization

Annulus pressurization refers to the loading on the shield wall and reactor vessel caused by a postulated pipe rupture at the reactor pressure vessel nozzle safe-end to pipe weld. The pipe rupture assumed is an instantaneous guillotine rupture which allows mass/energy release into the drywell and annular region between the biological shield wall and the reactor pressure vessel (RPV).

The mass and energy released during this postulated pipe rupture causes:

a. A rapid asymmetric decompression acoustic loading of the annular region between the vessel and shroud from the pipe break at or beyond the vessel nozzle safe-end weld.

b. A transient asymmetric differential pressure within the annular region between the biological shield wall and the reactor pressure vessel (annulus pressurization).

c. A jet stream release of the reactor pressure vessel inventory and the impact of the ruptured pipe against the pipe whip restraint attached to the biological shield wall.

The results of the mass and energy release evaluation are then used to produce a dynamic structural analysis (force-time history) of the RPV and shield wall. The force-time history output from the dynamic analysis is subsequently used to compute loads on the reactor components.

The postulated pipe rupture at the weld between recirculation or feedwater piping and the reactor nozzle safe-end leads to a high flow rate of water and steam mixture into the annulus between the RPV and the shield wall. Calculation of the mass/energy release is performed using the generic method for short-term mass releases. This method is described in Attachment 6.A to Chapter 6.0, where a sample calculation is also provided.

3.9.1.1.3.1 Acoustic Loads

Because the boiling water reactor (BWR) is a two-phase system that operates at or close to saturation pressure (1000 psi), the differential pressure across the reactor shroud is of short duration, and the BWR system is not subjected to a significant shock-type load with respect to structural supports. This short-duration acoustic load is confined to a bending moment and shear force on the reactor pressure vessel and reactor shroud support.

3.9.1.1.3.2 Pressure Loads

The pressure responses of the RPV-shield wall annulus for a recirculation suction line postulated rupture and a feedwater line postulated rupture were investigated using the RELAP4 computer code. An asymmetric model, using several nodes and flow paths, was developed for the analysis of the recirculation and feedwater line ruptures. Further description of these analytical models and detailed discussion of the analyses may be found in Section 6.2.

3.9.1.1.3.3 Jet Loads

Structural loads on the vessel and internals, jet thrust, jet impingement and pipe whip restraint loads were considered in conjunction with the above mentioned pressure loads. Jet thrust refers to vessel reaction force which results as the jet stream of liquid is released from the rupture. Jet impingement refers to the jet stream force which leaves the broken pipe and impacts the vessel. The pipe whip restraint load is the force which results when the energy absorbing pipe whip restraint restricts the pipe separation to less than one full pipe diameter. These jet loads are calculated as described in Reference 5.

3.9.1.1.3.4 Pool Slosh Loads

A representative analysis was performed on a 6 inch wetwell line having a submergence depth of 10 feet. The pool was modeled as a rigid annular container with a horizontal base excitation. The water is assumed to be inviscid, homogeneous, incompressible, and irrotational. It is treated as an ideal fluid with no sources or sinks and the velocity potential theory is applicable. The first ten sloshing modes were considered sufficient for an accurate analysis. The harmonious base motion is assumed to have the excitation of a frequency equal to the building effective fundamental frequency in the horizontal plane. The amplitude of this base excitation is the larger of the OBE or SSE basemat acceleration. The north-south and east-west excitations were assumed to be acting simultaneously and responses were assumed to be in phase with each other. The sloshing frequency is low compared to the pipe so that a static analysis is acceptable.

The pool slosh loads on the 6-inch pipe are shown below:

North-South Excitation

Depth(ft)	Radial Force(lb/ft)	Tangential Force(lb/ft)
-10	2.3864	-1.113
- 9	2.326	-1.221
- 8	2.2513	-1.339
- 7	2.1549	-1.4704
- 6	2.0273	-1.615
- 5	1.8521	-1.773
- 4	1.5999	-1.950
- 3	1.2162	-2.14699
- 2	0.5972	-2.3702
- 1	-.4948	-2.627
0	-2.534	-2.927

Loads from east-west excitation are lower. Total forces are:

North-South Radial	=	13.38 lb
North-South Tangential	=	20.553 lb
East-West Radial	=	9.9028 lb
East-West Tangential	=	15.2963 lb

Ignoring the restraints on the piping system, these loads produce a maximum stress of 454 psi. This is negligible with respect to the Equation 9 allowable of 18,000 psi.

Therefore, pool slosh loads can be seen to constitute a negligible load that need not be specifically included in the analysis of piping.

3.9.1.2 Computer Programs Used in Analysis

The following sections discuss computer programs used in the analysis of specific components. The GE computer programs are maintained either by General Electric or by outside computer program developers. In either case, the quality of the programs and the computer results are controlled. For each program, one or more individuals are assigned. Their duties are:

a. to keep abreast of the capability, the software contents and the theory of the program;

b. to run test cases and maintain the reliability of the program; and

c. to advise users on the proper usage of the program and the correct interpretation of computed results.

All necessary modifications are coordinated and verified by the responsible individuals. Thus, user's confusion over the changes is avoided and the high reliability of these programs is maintained.

3.9.1.2.1 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 CE Program - CHAT 12100 (Unit 1 only)

This program uses finite difference method to determine transient and steady-state temperature solutions for the reactor pressure vessel structures. The general heat balance equations used are those developed by Hellman, Halbetler and Babsov which permits the calculation of film coefficients, fluid flow variable properties and heat generation. Through use of this program, temperature distributions can be obtained in bodies having irregular geometrics and composed of different materials.

3.9.1.2.1.2 CE Program - SEAL-SHELL-2 (Unit 1 only)

This program is used for the stress analysis of reactor pressure vessel nozzles by representing the nozzle structure as a symmetrical model. This is accomplished by treating the vessel as a spherical shell of radius equal to 1.5 times the actual cylindrical vessel radius. Stresses obtained from the Seal-Shell solution are then adjusted for the actual cylindrical vessel loads and deformations.

3.9.1.2.1.3 CE Program - KALNINS (Unit 1 only)

This program is a thin elastic shell program for shells of revolution, developed by Dr. A. Kalnins of Lehigh University. The revisions and improvements have been made to yield the CE version of this program.

3.9.1.2.1.4 CB&I Program 711 "GENOZZ" (Unit 1 only)

The GENOZZ computer program is used to proportion barrel and double taper type nozzles to comply with the specifications of the ASME Code Section III and contract documents. The program will either design such a configuration or analyze the configuration input into it. If the input configuration will not comply with the specifications, the program will modify the design and redesign it to yield an acceptable result.

3.9.1.2.1.5 CB&I Program 943 - "NAPALM"

The basis for the program NAPALM, Nozzle Analysis Program--All Loads Mechanical, is to analyze nozzles for mechanical loads and find the maximum stress intensity and location. Specified locations are analyzed from the point of application for the mechanical loads. At each location the maximum stress intensity is calculated for both the inside and outside surfaces of the nozzle. The program gives the maximum stress intensity for both the inside and outside surfaces of the nozzle as well as its angular location. The principal stresses are also printed. The stresses resulting from each component of loading (bending, axial, shear, and torsion) are printed, as well as the loadings which caused these stresses.

3.9.1.2.1.6 CB&I Program 1027

This program is a computerized version of the analysis method contained in the Welding Research Council Bulletin No. 107, December 1965. The theory is based on Professor P. O. Bijlaard's experimental work.

Part of this program provides for the determination of the shell stress intensities (S) at each of four cardinal points at both the upper and lower shell plate surfaces (ordinarily considered outside and inside surfaces) around the perimeter of a loaded attachment on a cylindrical or spherical vessel. With each determination of S, the components of that are also determined. (2 normal stresses, σ_x and σ_y, and one shear stress τ). This program provides the same information as the manual calculation, and the input data is essentially the geometry of the vessel and attachment.

3.9.1.2.1.7 CB&I 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 the ASME Code, Section III.

In cases were the penetration has a counterbore, the thickness is determined so that the counterbore does not penetrate the outside surface of the head.

3.9.1.2.1.8 CB&I Program 781 - "KALNINS"

This program is a thin elastic shell program for shells of revolution. This program was developed by Dr. A. Kalnins of Lehigh University. Extensive revisions and improvements have been made by Dr. J. Endicott to yield the CB&I version of this program (Reference 8).

The KALNINS thin shell program (Program 781) is used to establish the shell influence coefficient and to perform detail stress analysis of the vessel. The stresses and the deformations of the vessel can be computed for any combination of the following axi-symmetric loading:

a. preload condition,

b. internal pressure, and

c. thermal load.

3.9.1.2.1.9 CB&I Program 979 - "ASFAST"

ASFAST Program (Program 979) performs the stress analysis of axisymmetric, bolted closure flanges between head and cylindrical shell.

3.9.1.2.1.10 CB&I Program 766 - "TEMAPR"

This program will reduce any arbitrary temperature gradient through the wall thickness to an equivalent linear gradient. The resulting equivalent gradient will have the same average temperature and the same temperature-moment as the given temperature distribution. Input consists of plate thickness and actual temperature distribution. The output contains the average temperature and total gradient through the wall thickness.

3.9.1.2.1.11 CB&I Program 767 - "PRINCESS"

The PRINCESS computer program calculates the maximum alternating stress amplitudes from a series of stress values by the method in Section III of the ASME Pressure Vessel Code.

3.9.1.2.1.12 CB&I 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 greatly modified version of the TIGER heat transfer program written about 1958 at Knolls Atomic Power Laboratory by A. P. Bray. There have been many versions of TIGER in existence, including TIGER II, TIGER II B, TIGER IV, and TIGER V, in addition to TGRV.

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 are used to solve the mesh network for every time interval. Included in the analysis are the three basic forms of heat transfer, conduction, radiation and convection, as well as internal heat generation.

Given any odd-shaped structure which can be represented by a three-dimensional field and its geometry, physical properties, boundary conditions, and internal heat generation rates, TGRV will calculate and give as output the steady-state or transient temperature distributions in the structure as a function of time.

3.9.1.2.1.13 CB&I 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 by 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 very 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.

3.9.1.2.1.14 CB&I Program 984

Program 984 is used to calculate the stress intensity of the 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 Section III of the ASME Code.

3.9.1.2.1.15 CB&I Program 992 - GASP

The GASP computer program, originated by Prof. 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.

This program determines the stresses and displacement of plane or axisymmetric structures using the finite element method. The structures may have arbitrary geometry and have linear or nonlinear material properties. The loadings may be thermal, mechanical, accelerational, or a combination of these.

The structure to be analyzed is broken up into a finite number of discrete elements or "finite elements" which are interconnected at finite number or "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.

3.9.1.2.1.16 CB&I Program 1037 - "DUNHAM'S"

DUNHAM'S program is a finite ring element stress analysis program. It will determine the stresses and displacement 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 (CB&I 992). The major differences are that DUNHAM'S uses constant material properties at each mode. As in GASP, the loadings may be thermal, mechanical, and accelerational.

3.9.1.2.1.17 CB&I Program 1335

This program models the baffle plate as a continuous circular plate for the purpose of computing stresses in the shroud support. The program allows the baffle plate to be included in CB&I Program 781 as two isotropic parts and an orthotropic portion at the middle (where the diffuser holes are located).

3.9.1.2.1.18 CB&I Programs 1606 and 1657 - "HAP"

The HAP program is an axisymmetric nonlinear heat analysis program. It is a finite-element program and is used to determine nodal temperatures in a two-dimensional or axisymmetric body subjected 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 CB&I stress programs 992 and 1037.

3.9.1.2.1.19 CB&I Program 1635

Program 1635 offers the following three features to aid the stress analyst in preparing a stress report:

a. Generates punched card input for program 767 (PRINCESS) from the stress output of program 781 (KALNINS).

b. Writes a stress table in a format such that it can be incorporated into a final stress report.

c. Has the option to remove through-wall thermal bending stress and report these results in a stress table similar to the one mentioned above.

3.9.1.2.1.20 CB&I Program 953

The program is a general purpose program, which does the following:

a. It prepares input cards for the thermal model.

b. It prepares the node and element cards for the finite element model.

c. It sets up the model in such a way that the nodal points in the TGRV model correspond to points in the finite element model. They have the same number so that there is no possibility of confusion in transferring temperature data from one program to the other.

d. It plots both the thermal model and stress model.

e. It treats the most general geometry that can be treated by the "GASP" program (i.e., a general axisymmetric body).

3.9.1.2.1.21 CB&I Program 955 "MESHPLOT"

This program plots input data used for finite element analysis. The program plots the finite element mesh in one of three ways: without labels, with node labels, or with element labels. The output consists of a listing and a plot. The listing gives all node points with their coordinates and all elements with their node points. The plot is a finite element model with the requested labels.

3.9.1.2.1.22 CB&I Program 1028

This program calculates the necessary form factors for the nodes of the model which simulates heat transfer by radiation. Inputs are shape and dimensions of the head-to-skirt knuckle junction. The program is limited to junctions with a toroidal knuckle part.

3.9.1.2.1.23 CB&I Program 1038

This program calculates the loads required to satisfy the compatibility between the shroud baffle plate and the jet pump adaptors in the RPV.

3.9.1.2.1.24 GE Program - DYSEA

This program is a General Electric proprietary program developed specifically to compute seismic and dynamic responses on the reactor pressure vessel structures, internals, and reactor pedestal and shield wall complex. It calculates the dynamic response of linear structural systems by either temporal model superposition or response spectrum method. Fluid-structure interaction effect in the reactor pressure vessel is taken into account by way of hydrodynamic mass.

The DYSEA program was based on the SAP-IV program (see Subsection 3.9.1.2.2.1) with added capability to handle the hydrodynamic mass effect. Structural stiffness and mass matrices are formulated similar to SAP-IV. Solution is obtained in time domain by calculating the dynamic response mode by mode. Time integration is performed by using Newmark's method. A response spectrum solution is also available as an option.

3.9.1.2.1.25 GE Program - SEISM

This program is a General Electric proprietary program developed to compute dynamic responses of non-linear structural systems. It also predicts responses of structural systems to dynamic disturbances including hydrodynamic and impact effects. The method used is known as the component element method which can account for both linear and non-linear structural behavior in the analysis. In this method, the structural system is modeled as an assemblage of elementary components; and, the dynamic equations of motion are integrated by using the finite difference method.

3.9.1.2.2 Piping

The computer programs used in the analysis of NSSS piping systems within GE's scope of supplies are identified and their use summarized in the following subsections.

3.9.1.2.2.1 Structural Analysis Program - SAP4 (Unit 1 only)

SAP is a general Structural Analysis Program for static and dynamic analysis of linear elastic complex structures. The finite element displacement method is used to solve for the displacements and compute the stresses of each element of the structure. The structure can be composed of unlimited number of three-dimensional truss, beam, plate, shell, solid, plate strain-plane stress, brick, thick shell, spring, 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 type of loading as well as time varying pressure can be treated. The program is very versatile and efficient in solving large and complex structural systems. The output contains displacements of each nodal point as well as stresses at the surface of each element.

3.9.1.2.2.2 Component Analysis/ANSI 7

The ANSI 7 Computer Program determines stress and accumulative usage factors in accordance with NB-3600 of ASME Boiler and Pressure Vessel Code Section III.

The program was written to perform stress analysis in accordance with the ASME sample problem, and has been verified by reproducing the results of the sample problem analysis.

3.9.1.2.2.3 Area Reinforcement/NOZARP (Unit 1 only)

The computer program NOZARP (Nozzle Area Reinforcement Program) performs an analysis of the required reinforcement area for openings. The calculations performed by NOZARP are in accordance with the rules of the 1974 edition of Section III, ASME Boiler and Pressure Vessel Code.

3.9.1.2.2.4 Dynamic Forcing Functions

3.9.1.2.2.4.1 Relief Valve Discharge Pipe Forces Computer Program/RVFOR

The relief valve discharge pipe connects the relief valve to the suppression pool. When the valve is opened, the transient fluid flow causes time dependent forces to develop in the pipe wall. This computer program computes the transient fluid mechanics and the resultant pipe forces using the method of characteristics.

3.9.1.2.2.4.2 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.

3.9.1.2.2.5.1 Integral Attachment/LUGSTR (Unit 1 only)

The computer program "LUGSTR" was prepared to evaluate the stress in the pipe walls that are produced by loads applied to the integral attachments. The program was prepared based on the Welding Research Council Bulletin 198.

3.9.1.2.2.5.2 Integral Attachment/Code Cases (Unit 2)

Code Case N-318-4 was utilized for shear lug welded attachments on the Main Steam system. Specifically these welded attachments are located in subsystems 2MS01, 2MS02, 2MS03, and 2MS04.

Code Case N-392-1 was utilized for stanchion welded attachments on the Main Steam system. Specifically these welded attachments are located in subsystem 2MS03.

3.9.1.2.2.6 Piping Dynamic Analysis Program/PDA

The pipe whip analysis was performed using the PDA computer program. 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. Similar to the popular plastic hinge concept, bending of the pipe is assumed to occur only at the fixed end and at the location supported by the restraint.

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

3.9.1.2.2.7 Piping Analysis Program/EZPYP (Unit 1 only)

EZPYP links the ANSI-7 and SAP program 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 gives the analyst the capability to perform a complete piping analysis in one computer run.

3.9.1.2.2.8 Thermal Transient Program/LION (Unit 1 only)

The LION program is used to compute radial and axialthermal gradients in piping. The program calculates a time history of ΔT_1 , ΔT_2, Ta, and Tb (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 (Unit 1 only)

The SIMOK program provides a time history that is equivalent to an input response spectra. The synthetic time history is used to generate a new spectra that is plotted with the input spectra to verify that the time history and spectra are equivalent. Synthetic time histories are used in multiple input analysis of the piping.

3.9.1.2.2.10 Differential Displacement Program/DISPL (Unit 1 only)

The DISPL program provides differential movements at each piping attachment point based on building model displacements.

3.9.1.2.2.11 WTNOZ Computer Program (Unit 1 only)

WTNOZ is a time-share program for piping weight calculations.

3.9.1.2.2.12 Piping Analysis Program/PISYS

PISYS is a computer code specialized for piping load calculations. It utilizes selected stiffness matrices representing standard piping components, which are assembled to form a finite element model of a piping system. The technique relies on dividing the pipe model into several discrete substructures, called pipe elements, which are connected to each other via nodes called pipe joints. It is through these joints that the model interacts with the environment and loading of the structure becomes possible. PISYS is based on the linear classical elasticity in which the resultant deformation and stresses are proportional to the loading and the superposition of loading is valid.

PISYS has a full range of static and dynamic analysis options which include: distributed weight, thermal expansion, differential support motion modal extraction, response spectra, and time history analysis by modal or direct integration. The PISYS program has been benchmarked against five Nuclear Regulatory Commission piping models for the option of response spectrum analysis and the results are documented in a report to the Commission, "PISYS Analysis of NRC Benchmark Problems", NEDO-24210, August 1979.

3.9.1.2.2.13 Piping Analysis Program/SUPERPIPE

SUPERPIPE is a comprehensive computer program developed by ABB Impell for the structural analysis and design checking of piping systems (with particular emphasis on power plant piping), under various types of loading including earthquake - induced oscillations and response spectra. Analysis may be carried out in several standard piping codes.

SUPERPIPE executes in distinct phases; namely specification of system geometry, static analysis, determination of dynamic characteristics, response spectrum analysis, and design checking against code requirements. Appropriate combinations of these phases may be executed during any specific computer run.

Mode shapes and frequencies are determined by the subspace iteration algorithm or by the Q-R method. Response spectra analysis may be performed, assuming either single or multiple support excitation, and the directional and modal results combined by any of the standard methods (SRRS, 10 percent method, grouping, etc). A library of stress intensification factors is also included in the program.

The output from SUPERPIPE includes a detailed summary of stresses and displacements. The results of the analyses can be saved permanently on problem data files and recalled for use in subsequent computer runs. The restart options include storage, recall, and modification of pipe geometry and support information. A code compliance summary based on any of the several standard piping codes built into the program in output. Nozzle and penetration summaries are also given. Output from SUPERPIPE may be stored on tape and used in a number of postprocessors and plotting routines.

The SUPERPIPE program has been benchmarked against the required Nuclear Regulatory Commission piping problems and accepted for piping dynamic analysis using the response spectrum method. This is documented in EDS Nuclear Report

01-0160-1187, "SUPERPIPE Verification To Benchmark Problems Contained in NUREG/CR-1677."

3.9.1.2.3 Computer Programs Used in Analyses of BOP Piping and Equipment

The PIPSYS or NUPIPE computer programs have been used by the A-E in the dynamic analyses to determine the structural and functional integrity of Seismic Category I systems and supports. A description of the PIPSYS program is provided in Appendix A of the LaSalle County Station - DAR (Reference 9).

The OPTPIPE computer program is a special purpose program which performs linear elastic static and dynamic analysis of three-dimensional piping systems arbitrarily oriented in space. The program can perform static analyses for dead weight, internal pressure, thermal effects, support displacements and externally applied loads. Dynamic analyses can be performed for earthquake loading represented by either an acceleration response spectrum (response spectrum approach) or a time history (time history approach using either the modal superposition or direct integration). The OPTPIPE program has been benchmarked against the required Nuclear Regulatory Commission piping problems and accepted for piping dynamic analysis using the response spectrum method in the NRC letter to Mr. L. D. Butterfield (Nuclear Licensing Manager, Commonwealth Edison Company), dated August 10, 1987 (Docket No. STN 50-454).

3.9.1.2.4 ECCS Pumps and Motors

An equivalent static computer analysis was performed on the ECCS pump motor rotor shafts. The model consisted of lumped masses simulating the distribution of mass in the system, connected by massless elastic members, simulating the distribution of shaft stiffnesses. The analysis was performed iteratively to obtain compatibility between the rotor displacements and the magnetic and centrifugal forces acting on the rotor.

All other analysis of specific motor components and pump components consisted of hand calculations.

3.9.1.2.5 RHR Heat Exchangers

Following are the computer programs used in dynamic and static analysis to determine structural and functional integrity of the RHR heat exchangers:

a. Support Load Seismic Analysis (SAP 4)

   This computer program computes the total loads at the upper and lower supports of the RHR heat exchanger. This computer program takes into account the heat exchanger flooded weight, seismic (either OBE or SSE) and the allowable nozzle loads and sets up the worst combination of these loads. By maximizing seismic loads together with nozzle loads, maximum conservative moments and forces at the upper and lower supports are calculated.

b. Stress Analysis of Supports (BILRD01)

This program performs a full stress analysis of the upper and lower supports of the RHR heat exchanger. The stresses in the supports (both upper and lower) caused by loads resulting from seismic and nozzle loads are computed in the Support Load Program (SAP 4) and are used as input values for this program. This program computes the membrane stresses on the shell of the heat exchanger by the use of the Bijlaard's analysis.

3.9.1.3 Experimental Stress Analysis

Experimental stress analysis methods have not been utilized. The analytical methods employed for Seismic Category I systems, components, equipment, and supports are based on those methods specified by ASME Section III.

3.9.1.4 Considerations for the Evaluation of the Faulted Condition

Only elastic methods as described in the ASME B&PV Code Section III have been used to evaluate stresses for the Seismic Category I components. Allowable stresses and deformations are based on those specified in the Code.

3.9.2 Dynamic System Analysis and Testing

3.9.2.1 Preoperational Vibration and Dynamic Effects Testing on NSSS Piping

Vibration amplitudes of the recirculation system induced by fluid flow and recirculation pump operation are instrumented and measured as a part of the preoperational test program. The measured amplitudes are compared with the allowable vibration amplitudes calculated by an analysis of the system.

The interaction between recirculation pump and the flow control valve was tested up to full loop flow on a generic basis in a reactor mockup test loop at full operating pressure and temperature before these components were installed in the plant. These tests included pump starts and pump trips, normal valve flow transient, maximum rate valve opening or closing transients caused by simulated gross malfunction of the control system, and a combination of pump trips and valve closure transients.

In-plant system demonstration and preoperational tests will be conducted to determine flow interactions with other components of the reactor and reactor coolant pressure boundary. These tests will be with cold water, and the maximum flow rate will be set by equipment cavitation limits or by equipment load limits. The tests include pump starts, pump trips, and valve transients. The recirculation piping system was visually inspected and instrumented to detect vibrations (Unit 1). The vibration measurement, visual inspection, and instrumentation stations will be located according to specified location criteria. If the vibration levels are beyond those allowed by the design stress levels, the pipe hangers and supports will be adjusted, relocated or redesigned until postadjustment tests show that the vibration levels have been reduced to within acceptable design limits.

A piping dynamic and thermal expansion verification testing program will be performed on the recirculation system. The following tests will be performed:

a. system thermal expansion,

b. vibration during operation of pump at maximum speed with system cold,

c. vibration during system startup,

d. vibration during a recirculation pump trip, and

e. shakedown of system.

Vibration of the main steam system piping is caused by either steam flow or valve operation. The effects of these causes are discussed in the following.

3.9.2.1.1 Steam Flow

Flow induced vibration of the main steam piping has been shown to be insignificant by a test program conducted in a prototype plant of the same configuration and flow rates. Therefore, testing or analysis for this condition is not conducted for each plant, since its effect has been found by test to be insignificant.

3.9.2.1.2 Turbine Stop Valve Closure

The effects of turbine stop valve closure are evaluated analytically by means of a dynamic analysis of the piping system. The piping is modeled as a lumped mass system. Forcing functions are applied at points of fluid momentum change, such as elbows. The forcing functions are described by fluid momentum equations, and the shock wave velocity. The results of this method of analysis are compared with results from actual test measurements in this plant.

3.9.2.1.3 Relief Valve Operation

The effects of relief valve operation on the main steam pipe are evaluated analytically by means of dynamic analysis of the main steam valve and discharge piping. The main steam and discharge system is modeled as a lumped mass system. Forcing functions are applied at points of momentum change in the system. The forcing functions are described by fluid momentum equations and the shock wave velocity. The results of this method of analysis are compared with results obtained from actual test measurements on Unit 1 in this Plant.

A piping dynamics testing program is to be performed on the main steam systems for Unit 1 only.

The following tests are planned:

a. system thermal expansion;

b. dynamic system response to relief valve operation, turbine stop valve closure, and main steam isolation valve closure; and

c. shakedown of system.

3.9.2.2 Seismic Qualification of Safety-Related Mechanical Equipment

This subsection describes the criteria (capability of many of the components so noted) for qualification of mechanical safety-related equipment and also describes 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, motor-powered pumps. Qualification testing is also discussed in Subsection 3.9.3.2. Electrical supporting equipment such as control consoles, cabinets, panels, and instruments and controls are discussed in Section 3.10.

3.9.2.2.1 Tests and Analysis Criteria and Methods

The ability of equipment to perform its Seismic Category I (safety-related) function during and after an earthquake was demonstrated by tests and/or analysis. Selection of testing, analysis or a combination of the two was determined by the type, size, shape, and complexity of the equipment being considered. When practical, the Seismic Category I operations were performed simultaneously with vibratory testing. Where this was not practical the operation and/or loads were simulated by mathematical analysis.

Equipment that is large, simple, and/or consumes large amounts of power was usually qualified by analysis or static bend test to show that the stresses and deflections are less than the allowables.

Other equipment was qualified by dynamic testing by mounting on a fixture that simulates the intended inservice mounting and causes no dynamic coupling to the equipment.

Equipment having an extended structure, such as a valve operator, was analyzed by applying static equivalent loads at the center of gravity of the extended structure. In cases where the equipment structural complexity made mathematical analysis impractical, a static bend test was used to determine spring constant and operational capability at maximum equivalent dynamic load conditions.

3.9.2.2.1.1 Random Vibration Input Tests

When random vibration input is used, the actual input motion enveloped the appropriate floor input motion at the individual modes. However, single frequency input, such as sine beats, was used provided one of the following conditions were met:

a. The characteristics of the required input motion was dominated by one frequency.

b. The anticipated response of the equipment was adequately represented by one mode.

c. The input has sufficient intensity and duration to excite all modes to the required magnitude, such that the testing response spectra enveloped the combined response spectra of the individual modes.

3.9.2.2.1.2 Application of Input Motion for Dynamic Tests

When dynamic tests were performed, the input motion was applied to one vertical and one horizontal axis simultaneously. However, when the equipment response along the vertical direction was not sensitive to the vibratory motion along the horizontal direction, and vice versa, the input motion was applied in one direction at a time. In the case of single frequency input, the time phasing of the inputs in the vertical and horizontal directions was such that a purely rectilinear resultant input was avoided.

3.9.2.2.1.3 Fixture Design

The fixture design simulated the actual service mounting and caused no dynamic coupling to the equipment.

3.9.2.2.1.4 Prototype Testing

Initial equipment tests were conducted on prototypes of the NSSS equipment installed in this plant. Later tests were conducted on the same models of equipment as was installed in the plant.

3.9.2.2.2 Seismic Qualification of Specific NSSS Mechanical Components

The following sections discuss the seismic qualification testing or analytical qualification of NSSS equipment.

3.9.2.2.2.1 Jet Pumps

A static analysis of the jet pumps was performed assuming 3.0g horizontal acceleration and 1.5g vertical. The stresses resulting from the analysis were below the design allowables. Static analysis with an appropriate amplification factor was used in lieu of dynamic analysis since the jet pump is a simple component with a natural frequency of slightly less than 33 hertz. A dynamic analysis of the jet pump was performed and the stresses determined from the analysis were below design allowables. The results are summarized in Table 3.9-22 of the FSAR. Modification that effect the jet pumps will be evaluated against accepted allowables at the time of the modification. The results will be listed in the stress report or analyses.

The impact on the jet pumps for the Extended Operating Domain (EOD) which includes the effects of Increased Core Flow (ICF) and Final Feedwater Temperature Reduction (FFWTR) was evaluated in Reference 16. The ICF causes an increase in reactor internals pressure difference which may impact the loads. The FFWTR changes the downcomer fluid density which will impact the later loads for a recirculation line break. The Reference 16 report concluded that stresses produced while operating with ICF and/or FFWTR are within the allowable design limits.

3.9.2.2.2.2 CRD and CRD Housing

The dynamic analysis of the fuel, core support, top guide and control rod drive housing (with contained control rod drive) indicates these components behave essentially in an elastic manner during the combined loadings.

The housing provides the basic structural member for the drive, so the dynamic load effects on the CRD are evaluated from a drive housing deflection standpoint. Restraints were provided to prevent flange motion, so the housing deflection was limited to a small midpoint bow. This bow was smaller than the clearance between drive and housing, and thus did not affect drive motion. Tests have been conducted on the drive with dynamic deflections of 2 inches peak to peak at the flange (at the natural frequency of drive and housing). There was no measurable effect.

Additional testing of the CRD has been conducted with static displacement of the core support and top guide equal to the maximum calculated dynamic deflection. The effect on scram performance was negligible.

Channel bow tests, with varying amounts of fixed channel deflection, indicate very little effect on scram with a bow greater than the calculated maximum dynamic deflection under combined loads.

Drive performance under dynamic deflections should give even greater margins than the above tests conducted with static deflections.

The impact on the CRD / CRD Housing for the Extended Operating Domain (EOD) which includes the effects of Increased Core Flow and Final Feedwater Temperature Reduction (FFWTR) was evaluated in Reference 16. The ICF causes an increase in reactor internals pressure difference which may impact the loads. The FFWTR changes the downcomer fluid density which will impact the lateral loads for a recirculation line break. The Reference 16 report concluded that stresses produced are while operating with ICF and/or FFWTR are within the allowable design limits.

The impact on the CRD penetrations and CRD housing for the introduction of 80 mil fuel channels has been evaluated in Reference 15. Reference 15 concluded that, based on available margin and detailed evaluation, the CRD penetrations and CRD housing components are acceptable with the revised loading due to introduction of 80 mil fuel channels.

3.9.2.2.2.3 CRD Guide Tube

No dynamic testing of the CRD guide tube has been conducted; however, a detailed analysis imposing dynamic effects due to dynamic events has shown that the maximum stresses developed during these events are much lower than the maximum allowed for the component material.

The impact on the CRD guide tube for the Extended Operating Domain (EOD) which includes the effects of Increased Core Flow and Final Feedwater Temperature Reduction (FFWTR) was evaluated in Reference 16. The ICF causes an increase in reactor internals pressure difference which may impact the loads. The FFWTR changes the downcomer fluid density which will impact the lateral loads for a recirculation line break. The Reference 16 report concluded that stresses produced while operating with ICF and/or FFWTR are within the allowable design limits.

3.9.2.2.2.4 Hydraulic Control Unit (HCU)

The hydraulic control piping was analyzed for the faulted condition. The maximum stress on the HCU frame was calculated to be below the maximum allowable for the SSE faulted condition once additional bracing had been added to the HCU frame. The total required response spectra (RRS) at HCU floor are enveloped by the test required spectra (TRS) for both the horizontal and vertical directions.

3.9.2.2.2.5 Fuel Channels

GE BWR fuel channel design bases, analytical methods and evaluation results including seismic considerations, are contained in References 6, 7 and 15.

Reference 16 evaluated the fuel channels and the fuel assembly for ICF operation considering the effects of loads under normal, upset, and faulted load combinations. The results of this evaluation found that the channel wall pressure gradients were within the allowable design limits.

3.9.2.2.2.6 Recirculation Pump and Motor Assembly

Calculations were made to assure that the recirculation pump and motor assembly are designed to withstand the specific static equivalent forces. The flooded assembly was analyzed as a free body supported by constant support hangers from the brackets on the motor mounting member with hydraulic snubbers attached to brackets located on the pump case and the top of the motor frame.

Primary stresses due to horizontal and vertical forces were considered to act simultaneously and are conservatively added directly. The horizontal and vertical seismic forces were applied at mass centers and equilibrium reactions determined for motor and pump brackets.

3.9.2.2.2.7 ECCS Pump and Motor Assembly

This section discusses the ECCS pump and motor assemblies. The qualification of these pump and motor assemblies as a unit while operating under dynamic conditions was provided in the form of a static analysis. The maximum specified vertical and horizontal response spectra were constantly applied simultaneously in the worst-case combination. The results of the analysis indicate that the pump is capable of sustaining the above loadings without overstressing the pump components.

The ECCS Pump and Motor Assemblies were qualified using dynamic analysis to address the suppression pool hydrodynamic loads (in addition to seismic loads) and concerns identified in Regulatory Guide 1.100 Revision 1 and IEEE 344-1975.

A similar design motor has been qualified via a combination of static analysis and dynamic testing. The complete motor assembly has been qualified via 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.

For static analysis on a similar design motor, the dynamic forces of each component or assembly are obtained by concentrating its mass at the center of gravity of the component or assembly, and multiplying by the seismic acceleration, (earthquake coefficient).

3.9.2.2.2.8 RCIC Pump Assembly

The RCIC pump construction is a barrel type on a large cross-section pedestal. Qualification by analysis was performed. The seismic design analysis is based on 4.15g horizontal and 3.45g vertical accelerations. Results are obtained by using acceleration forces acting simultaneously in two horizontal directions, and one vertical direction. The fundamental frequencies of the base and shaft were calculated to be greater than 50 hertz.

The RCIC pump assembly has been analytically qualified by static analysis for vibratory loading as well as the design operating loads for pressure, temperature, and external piping. The results of this analysis confirm that the stresses are within the allowable limits.

The RCIC Pump Assembly was requalified using static analysis to address the suppression pool hydrodynamic loads (in addition to seismic loads) and concerns identified in Regulatory Guide 1.100 Revision 1 and IEEE 344-1975.

3.9.2.2.2.9 RCIC Turbine Assembly

The RCIC turbine has been qualified via a combination of static analysis and dynamic testing. The turbine assembly consists of rigid masses interconnected with control levers and electronic control systems, necessitating final qualification via dynamic testing. A computer aided analysis, using SAP IV computer program has been employed to verify the structural integrity of the turbine oil piping. Static loading analysis has been employed to verify the adequacy of bolting under operating and seismic loading conditions. The complete turbine assembly of similar construction has been seismically qualified via 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 faulted loading conditions. Operability under normal load conditions can be assured by comparison of the operability of similar turbines in other operating plants.

The RCIC Turbine Assembly was requalified using a combination of test and dynamic analysis to address the suppression pool hydrodynamic loads (in addition to seismic loads) and concerns identified in Regulatory Guide 1.100 Revision 1 and IEEE 344-1975.

Requirements

The specification for qualification of the RCIC turbine and its accessories states that they shall be capable of withstanding the specified accelerations at all frequencies within the range of 0.1 hertz to 50 hertz. Proper performance may be demonstrated by the seller by tests, analysis, or a combination of both. If all natural frequencies of the turbine, the component parts, and the accessories are greater than 50 hertz (as defined by test and/or analysis), a static load analysis may be performed. The seismic forces of each component or assembly are obtained by concentrating its mass at the center of mass of the component or assembly, and multiplying by the acceleration coefficient. The magnitude of the acceleration coefficients are 1.5g in both horizontal and vertical directions. If component parts and/or accessories have natural frequencies below 50 hertz, these parts must be dynamically analyzed or tested, demonstrating satisfaction of the defined floor response spectra. If the equipment capability is demonstrated by test, the equipment must be subjected to simultaneous horizontal and vertical acceleration inputs of random wave-form motion for a minimum duration of 30 seconds. The random input must envelop the defined floor response spectra.

Test Qualification Results

The RCIC turbine assembly was subjected to a total of 12 vibratory tests (5 OBE, 1 SSE per biaxial orientation) with an accumulated test time of 480 seconds (8 minutes). Input to the equipment was random wave-form motion in two directions, one horizontal and the other vertical. Sine beats were superimposed at 1/3-octave intervals as necessary to envelop the required response spectra. The required response spectra enveloped all postulated dynamic loads including those from seismic and hydrodynamic transient by wide margin. A 200-psi, 1200-cfm air source was used as the operating medium for the turbine. A second turbine assembly was subjected to a total of 13 vibratory tests with an accumulated test time of approximately 400 seconds. Input to the equipment was biaxial multifrequency random motion.

Because beats were not needed to satisfy the required response spectra, the operative medium for the turbine was 200 psig saturated steam. The electronic governor system, the turbine hydraulic system with interconnecting piping and levers, and all turbine instrumentation were under actual operating conditions during both test programs. Nozzle loadings were simulated for the turbine inlet and exhaust piping.

Analytical Qualification Results

The rigid components of the RCIC turbine assembly have been analytically qualified via static analysis for vibratory loading, as well as the design operating loads of pressure, temperature, and external piping loads. The results of this analysis confirm that the stresses in all components are below allowable levels.

3.9.2.2.2.10 Standby Liquid Control Pump and Motor Assembly

The SLC positive displacement pump and motor mounted on a common base plate has been qualified by static analysis and testing.

The design analysis of the pump is based on 1.4g in each direction for upset and 2.0g for emergency. Results are obtained by using acceleration forces acting simultaneously in two directions, one vertical and one horizontal. The pump/motor/base assembly has been shown by static analysis to have a natural frequency of 48 hertz. The SLC pump and motor assembly has been analytically qualified by static analysis for dynamic loading as well as the design operating loads of pressure, temperature, and external piping loads. The results of this analysis confirm that the stresses are substantially less than the allowables.

The Standby Liquid Control Pump and Motor Assembly was requalified using static analysis to address the suppression pool hydrodynamic loads (in addition to seismic loads) and concerns identified in Regulatory Guide 1.100 Revision 1 and IEEE 344-1975.

3.9.2.2.2.11 RHR Heat Exchangers

A dynamic analysis is performed to verify that the RHR heat exchanger will withstand dynamic loads. Testing is an impractical method to verify the equipment adequacy when predictable seismic loads can be determined by dynamic and static analysis.

The heat exchanger, including its appurtenances and supports, is designed to withstand the static plus dynamic loading. The acceleration coefficients are applied at the center of gravity of the heat exchanger, assuming the heat exchanger to be flooded.

3.9.2.2.2.12 Standby Liquid Control Tank

The standby liquid control storage tank is a cylindrical tank 9 feet in diameter and 12 feet high bolted to the concrete floor. Stresses can be calculated readily by conventional methods. The magnitude of the earthquake coefficients are 1.75g in each horizontal and vertical direction. The standby liquid control tank has been qualified by analysis for:

a. stresses in the tank bearing plate,

b. bolt stresses,

c. sloshing loads imposed by earthquake natural frequency of sloshing = 0.58 Hertz – natural frequency of tank = 52.0 Hertz,

d. minimum wall thickness, and

e. buckling

3.9.2.2.2.13 Main Steam Isolation Valves

The main steam isolation valve with the actuator is modeled in the piping system analysis. The axial forces and moments at the body-bonnet centerline are predicted from the analysis for the worst combination of piping loads. These values did not exceed the maximum allowable values determined from a simplified valve/actuator analysis which uses the design g-coefficient of the static equivalent seismic load.

The main steam isolation valve structure has been evaluated by test to determine operability at SSE acceleration. A static load test completed on a representative configuration demonstrated operability of actuator assembly while subjected to a simulated dynamic event. Equivalent accelerations utilized during the tests were in excess of those calculated from the piping system analysis. The fundamental requirement of the MSIV following a safe shutdown earthquake is to close and remain closed after the event. Proper MSIV functioning was demonstrated by the dynamic tests.

The main steam isolation valves have been qualified to address the suppression pool hydrodynamic loads (in addition to seismic loads) and concerns identified in Regulatory Guide 1.100 Revision 1 and IEEE 344-1975.

3.9.2.2.2.14 Main Steam Safety/Relief Valves

Due to the complexity of this structure and the performance requirements of the valve, the total assembly of the safety/relief valve (including electrical, pneumatic devices) was dynamically tested. Satisfactory operation of the valves were demonstrated during and after the test. Tests and analysis satisfy operability criteria.

3.9.2.3 Dynamic Response of Reactor Internals Under Operational Flow Transients and Steady-State Conditions

The major reactor internal components within the vessel are subjected to extensive testing coupled with dynamic system analyses to properly describe the resulting flow-induced vibration phenomena incurred from normal reactor operation and from anticipated operational transients.

In general, the vibration forcing functions for operational flow transients and steady-state conditions were not predetermined by detailed analysis. Special analyses of the response signals, measured from reactor internals of similar designs were performed to predict amplitude and modal contributions. Parametric studies were performed by extrapolating the results from tests of internals and components of similar designs. This vibration prediction method is appropriate where standard hydrodynamic theory cannot be applied due to 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 was performed to identify natural vibration modes and frequencies.

The analysis models used for Seismic Category I structures were similar to those outlined in Subsection 3.7.2.

b. Data from previous plant vibration measurements were assembled and examined to identify predominant vibration response modes of major components. In general, response modes are similar but response amplitudes vary among BWR's of differing size and design.

c. Parameters were identified which are expected to influence vibration response amplitudes among the several referenced 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 for the various parameters are developed which, multiplied by response amplitudes, tend to minimize the statistical variability between plants. A correlation function was obtained for each major component and response mode.

e. Predicted vibration amplitudes for components of the prototype plant were obtained from these correlation functions, based on particular values of the parameters for that prototype plant. The predicted amplitude for each dominant response mode was stated in terms of a range, taking into account the degree of statistical variability in each of the correlations. The predicted mode and frequency were obtained from the dynamic analysis of Item a above.

This dynamic model analysis also forms the basis for interpretation of the prototype plant preoperational and initial startup test results. 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 was taken as that which produces a peak stress amplitude of ± 10,000 psi.

During L1R08, the inlet mixers for Unit 1 Jet Pumps 9 and 10 were replaced. The crud layer has not been established as quickly as anticipated, resulting in higher flows through these jet pumps. At 100% power and 105% core flow, the jet pump 9 and 10 flows result in vibration stresses up to 10,700 psi. When the Unit 1 jet pump stresses are at 10,700 psi, the fatigue usage factor is increased 0.001/year.

3.9.2.4 Preoperational Flow-Induced Vibration Testing of Reactor Internals

Reactor internals for LSCS Units 1 and 2 are substantially the same as the internals design configurations which have been tested in prototype BWR/4 plants. An exception is the jet pump adapter, which is a new BWR/5 design. A jet pump vibration measurement and inspection program will be implemented and the results will be compared with the results of the prototype plant, to verify the design of the jet pumps with respect to vibration.

Results will be made available for NRC review after completion of the tests.

LSCS Unit 1 utilizes an adapter joining the jet pump to the shroud support which is of a unique design. Analysis indicates that this adapter might introduce added flexibility to reduce the natural frequencies of LSCS Unit 1 jet pumps below those of the prototype (Tokai-2) jet pumps. Therefore, vibration instrumentation is provided in the LSCS Unit 1 reactor to evaluate this deviation from the prototype design configuration. Vibration sensors are installed on the jet pump riser braces, as in the prototype plant, and on the jet pump adapter. One jet pump pair will be instrumented. Data will be acquired during the preoperational flow test, and also at specific flow and power conditions during the startup tests. LSCS Unit 1 is designated as a non-prototype plant with reference to requirements of Regulatory Guide 1.20, only for this particular item.

After completion of flow testing, the Unit 1 vessel head and the shroud head is removed and the vessel will be drained. Access to the lower plenum will be provided by opening a manhole in the shroud support plate. Reactor internal structures and components, including those in the lower plenum region, will be given a close visual inspection to detect possible wear, cracking, loosening of bolts, and the presence of debris and loose parts.

The vibration instrumentation and test conditions for LSCS Unit 1 jet pumps are described above. Data is acquired during preoperational testing to provide an early assessment of vibration performance. Data obtained during startup testing reflects performance in actual long-term operating conditions.

The vibration data were recorded on magnetic tape and on strip charts with the chart records being the primary medium for data analysis and evaluation. Peak-to-peak amplitudes and dominant frequencies were read directly from the charts. A spectrum analyzer was also used to identify significant response frequencies.

The LSCS-1 vibration test results for the jet pumps were evaluated during an analytically derived acceptance criteria which relates vibration amplitudes at sensor locations to peak stresses elsewhere in the structure. A mode frequency analysis was performed, and maximum modal stresses were calculated on a normalized basis. The maximum allowable vibration amplitude at a sensor location is that which produces a sustained peak stress amplitude of ± 10,000 psi in a given mode.

During L1R08, the inlet mixers for Unit 1 Jet Pumps 9 and 10 were replaced. The crud layer has not been established as quickly as anticipated, resulting in higher flows through these jet pumps. At 100% power and 105% core flow, the jet pump 9 and 10 flows result in vibration stresses up to 10,700 psi. When the Unit 1 jet pump stresses are at 10,700 psi, the fatigue usage factor is increased 0.001/year.

LSCS Unit 2 reactor internals are to be tested in accordance with provisions of Regulatory Guide 1.20, Revision 2, for non-prototype plants. The test procedure requires operation of the recirculation system at or near 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 at least 10 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 will be removed, the vessel drained and major components inspected on a selected basis. The inspection will cover all components which are examined on the prototype design, including the shroud, shroud head, and core support structures, and jet pumps, and the peripheral control rod drive and incore guide tubes. Access will also be provided to the reactor lower plenum.

Results of the prototype tests are presented in GE Licensing Topical Report, NEDE-24057-P (Class III) and NEDO-24057 (Class I), "Assessment of Reactor Internals Vibration in BWR/4 and BWR/5 Plants," November 1977. This report also contains additional information on the jet pump vibration measurement and inspection programs performed in the Tokai-2 plant. It describes the confirmatory inspection program. Amendment 2 to NEDO-24057-2 and NEDE-24057-P transmitted to Mr. O. D. Parr (J. T. Quirk, GE letter MFN 170-79, June 26, 1979) provided a summary of the Tokai-2 results.

3.9.2.5 Dynamic System Analysis of Reactor Internals Under Faulted Conditions

3.9.2.5.1 Safety Evaluation

The preoperational and startup testing series are utilized to authenticate sequentially the adequacy of reactor components and subsystems to respond properly in abnormal and faulted conditions. Consult Chapter 14.0 of the FSAR for test abstracts and the general pattern of startup tests which confirm the integrity of these reactor systems and the reactor internals.

3.9.2.5.2 Evaluation Methods

To determine that the safety design bases are satisfied, responses of the reactor vessel internals to loads imposed during normal, upset, emergency, and faulted conditions were examined. The effects on the ability to insert control rods, cool the core, and flood the inner volume of the reactor vessel were determined.

3.9.2.5.2.1 Input for Safety Evaluation

The operating conditions that provide the basis for the design of the reactor internals to sustain normal, upset, emergency, and faulted conditions, as well as combinations of design loadings that were accounted for in design of the core support structure, are covered in Tables 3.9-16, 3.9-21, 3.9-22, and 3.9-23.

In addition each combination of operating loads was categorized with respect to either normal, upset, emergency, or faulted conditions as well as the associated design stress intensity or deformation limits.

The bases for the proposed design stress and deformation criteria are also specified in Chapter 3.0.

3.9.2.5.2.2 Events To Be Evaluated

Examination of the spectrum of conditions for which the safety design basis must be satisfied reveals three dominating faulted events:

a. Recirculation Line Break

   A break in a recirculation line between the reactor vessel and the recirculation pump suction.

b. Steamline Break Accident

   A break in one main steamline 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

   An SSE subjects the core support structures and reactor internals to significant forces as a result of ground motion.

Analysis of other conditions existing during normal operation, abnormal operational transients, and accidents shows that the loads affecting the core support structures and reactor internals are less severe than these three postulated events.

3.9.2.5.2.3 Pressure Differential During Rapid Depressurization

A digital computer code (Reference 1) was used to analyze the transient conditions within the reactor vessel following the recirculation line break accident and the steamline break accident. The analytical model of the vessel consists of nine nodes which are 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 to give the depressurization rates and pressure in the various regions of the reactor. Figure 3.9-7 shows the nine reactor nodes.

3.9.2.5.3 Recirculation Line and Steamline Break

3.9.2.5.3.l Accident Definition

Both a recirculation line break (the largest liquid break) and an inside steamline break (the largest steam break) were considered in determining the design-basis accident for the reactor internals. The recirculation line break was the same as the original design-basis loss-of-coolant accident described in Section 6.3 of the FSAR, Rev. 0, April 1984. A sudden, complete circumferential break was assumed to occur in one recirculation loop. The pressure differentials on the reactor internals and core support structures were in all cases lower than for the main steamline break.

The analysis of the steamline break assumed a sudden, complete circumferential break of one main steamline between the reactor vessel and the main steamline restrictor. This is not the same accident described in Chapter l5.0 which has greater potential radiological effects. A steamline break upstream of the flow restrictors produces a larger blowdown area and thus a faster depressurization rate than a break downstream of the restrictors. The larger blowdown area results in greater pressure differentials across the reactor assembly internal structures.

The steamline break accident produces significantly higher pressure differentials across the reactor assembly internal structures than does the recirculation line break. This results from the higher reactor depressurization rate associated with the steamline break. Therefore, the steamline break is the design-basis accident for internal pressure differentials.

The reactor loading from a recirculation line break can be effected by the core flow rate and the amount of subcooling in the coolant. The recirculation line break is also analyzed in Reference 16 due to the effects of Increase Core Flow and/or (ICF) Final Feedwater Temperature Reduction (FFWTR). The additional subcooling in the downcomer from FFWTR can lead to an increase in the flow induced loads. The increased core flow can increase the reactor internals pressure difference which may impact the loads.

3.9.2.5.3.2 Effects of Initial Reactor Power and Core Flow

For purposes of illustration, the maximum internal pressure loads can be considered to be composed of two parts: steady-state and transient pressure differentials. For a given plant the core flow and power are the two major factors 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 will be 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 reactor pressure vessel and thus the depressurization rate and the transient part of the maximum pressure load are increased. As a result, the total loads on some components are higher at low power.

To ensure that the calculated pressure differences bound those which could be expected if a steamline break should occur, an analysis was conducted at a low power-high recirculation flow condition in addition to the standard safety analysis condition (105% steam flow, rated recirculation flow). The power chosen for analysis was the minimum value permitted by the recirculation system controls at rated or greater recirculation flow. This condition maximizes those loads which are inversely proportional to power. It must be noted that this condition, while possible, is unlikely; first, because the reactor will generally operate at or near full power; second, because high core flow is neither required nor desirable at such a reduced power condition.

Table 3.9-1 summarizes the maximum pressure differentials. Condition 1 is the safety analysis condition; Condition 2 is the low power-high flow condition. Conditions 1a and 2a represent analysis done at 3323 MWt. Conditions 1b and 2b represent analysis done for 3489 MWt. Comparison of these values illustrates the statements made in the foregoing paragraphs.

Reference 16 documents evaluations that were performed to determine the bounding acoustic and flow-induced loads, reactor internal pressure difference loads, and fuel support loads for ICF and/or FFWTR operation. The additional subcooling in the downcomer resulting from FFWTR operation can lead to an increase in the flow-induced loads. The ICF operation will increase internal pressure differences across the reactor internals. The reactor internals determined to be most effected by ICF and FFWTR were core plate, shroud support, shroud, top guide, shroud head, steam dryer, control rod guide tube, control rod drive housing and jet pump. These and other components were evaluated by GE using the bounding pressure differential loads under normal, upset and faulted conditions. It was concluded that the stresses produced in these and other components are within the allowable design limits in the UFSAR or ASME Code, Section III.

The reactor internals were evaluated for the effects of Power Uprate to 3489 MWt and MELLL (Maximum Extended Load Line Limit) in Reference 21. The calculated stresses remain within the allowable design limits given in ASME Code, Section III.

The LaSalle Units 1 and 2 reactor internals were evaluated for the effects of GE14 fuel introduction in Reference 27. Introduction of GE14 fuel will result in increased pressure differentials across the reactor internal components for normal, upset, and faulted conditions. The calculated stresses remain within the allowable design limits given in ASME Code, Section III.

Basis:

1. GE 14 transition
2. Reference 27

3.9.2.5.3.3 Conclusions

It was concluded that the maximum pressure loads acting on the reactor internal components result from an inside steamline break, and on some components the loads are greatest with operation at the minimum power associated with the maximum core flow (Table 3.9-1 Condition 2). This has been 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, although possible, it is not probable that the reactor would be operating at the rather abnormal condition of minimum power and maximum core flow. More realistically, the reactor would be at or near a full power condition and thus the maximum pressure loads acting on the internal components would be as listed under Condition 1 in Table 3.9-1.

As discussed in section 3.9.2.5.3.2, the effects of the Extended Operating Domain (EOD) have also been evaluated for normal, upset and faulted conditions in

Reference 16. The Reference 16 evaluation concluded that the reactor internals stresses would be within the allowable design limits given in the ASME code, Section III.

As discussed in Section 3.9.2.5.3.2, the reactor internals were evaluated for the effects of Power Uprate to 3489 MWt and MELLL (Maximum Extended Load Line Limit) in Reference 21. The calculated stresses remain within the allowable design limits given in ASME Code, Section III.

As discussed in Section 3.9.2.5.3.2, the LaSalle Units 1 and 2 reactor internals were evaluated for the effects of the introduction of GE14 fuel in Reference 27. The calculated stresses remain within the allowable design limits given in ASME Code, Section III.

Basis:

3. GE14 transition
4. Reference 27

3.9.2.6 Correlation of Reactor Internals Vibration Tests with Analytical Results

Prior to initiation of the instrumented vibration test program for the prototype plant, extensive dynamic analyses of the reactor and internals were performed. The results of these analyses were used to generate the allowable vibration levels during the vibration test. The vibration data obtained during the tokai-2 test were analyzed in detail. The results of the data analysis, vibration amplitudes, natural frequencies, and mode shapes were then compared to those obtained from the theoretical analysis.

Such comparisons provided insight into the dynamic behavior of the reactor internals. The additional knowledge gained was utilized in the generation of the dynamic models for seismic and LOCA analyses for LSCS. The models used for this plant are the same as those used for the vibration analysis of the prototype plant.

The flow-vibration test data are supplemented by data from forced oscillation tests of reactor internal components to provide the analysts with additional information concerning the dynamic behavior of the reactor internals.

3.9.2.6.1 Analysis Methods Under LOCA Loadings

In order to ensure that no significant dynamic amplification of load occurs as a result of the oscillatory nature of the blowdown forces, a comparison was made of the periods of the applied forces and the natural periods of the core support structures being acted upon by the applied forces. These periods were determined from a comprehensive dynamic model of the RPV and internals with 27 degrees-of-freedom (Figure 3.9-1).

Only motion in the vertical direction was considered here; each structural member (between two mass points) can only have an axial load. Besides the real masses of the reactor pressure vessel (RPV) and core support structures, account was made for the water inside the RPV.

Time varying pressure was applied to the dynamic model of the reactor internals described previously. Except for the nature and locations of the forcing functions and this dynamic model, this dynamic analysis method is identical to that described for the analysis of the seismic and hydrodynamic excitation for suppression pool and annulus pressurization events.

Reference 15 discusses the seismic/LOCA evaluation that was performed for the use of 80 mil channels at LaSalle. Compared to the surrounding structures, (shroud, vessels, etc.), a 20% decrease in fuel channel thickness (100 mil channels to 80 mil channels) can be significant. The fuel is a significant mass in the horizontal RPV and internals mathematical model and is modeled as a separate element. A 20% decrease in channel thickness results in about a 10% decrease in the frequency of the first fuel mode. Since this is in the range of frequencies of high seismic excitation, the effect on the horizontal seismic response is significant. Also since the fuel is coupled to the shroud, vessel and shield wall, the seismic response of these structures is also affected. Since the frequency range of strong dynamic excitation for the LOCA (and SRV) is generally higher than the first fuel frequency, the effect on the dynamic responses due to these loads is not significant. (Reference 15)

Reference 15 also documents the fuel assembly liftoff during a combined seismic and LOCA event to be 0.22 inches, compared to 0.52 inch criteria for liftoff in Reference 7, Amendment 3. Reference 17 documents the results for ATRIUM-9B fuel to be 0.23 inches, compared to the 0.655 inch criteria also listed in this reference. Reference 28 documents the ATRIUM-10 liftoff on 0.24 inches compared to a criteria of 0.67 inches.

Reference 17 also evaluates the effects of ATRIUM-9B fuel on the seismic/LOCA response of the core. As discussed in Reference 15, the LOCA differential pressure and seismic lateral loads are the main contributors to the fuel channel stresses and deformations. Since the FANP ATRIUM-9B and ATRIUM-10 fuel have been shown to have an allowable LOCA differential pressure higher than required and a concurrent allowable seismic acceleration higher than that applied to the GE fuel, the FANP ATRIUM-9B and ATRIUM-10 meet the requirement for LaSalle specific loads.

3.9.3 ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures

In the stress analysis for Seismic Category I equipment which is part of the primary coolant pressure boundary, the reactor vessel was analyzed according to the requirements of the ASME Section III Code. The main steam piping system and the recirculation piping system were analyzed to comply with ASME Section III Code (NB-3600). In the stress analysis of other Seismic Category I equipment such as RHR heat exchangers and pumps, elastic analysis was used. The maximum allowable stresses were less than the yield stresses of the materials used and these are given in FSAR Tables 3.9-2 through 3.9-11, 3.9-18, 3.9-22, 3.9-25, and 3.9-37. FSAR Tables 3.9-2 through 3.9-11, 3.9-18, 3.9-22, 3.9-25, and 3.9-37 also listed the calculated stresses or other design values at the time of licensing. Modification, performed after licensing, that effects any calculated stress or design value will be evaluated against accepted allowables at the time of the modification. Results of these evaluations will be listed in the applicable stress report or analysis.

3.9.3.1 Loading Combinations and Stress Limits

ASME Code Class 2 and 3 components of fluid systems were constructed in accordance with Section III of the ASME Boiler and Pressure Vessel Code. Some components (piping, pumps, and valves) ordered prior to July 1971 were designed to other industry codes when the effective Section III was not applicable.

Functional capability has been assessed on safety-related subsystems and all subsystems meet the criteria stated in the DAR with the following exceptions:

a. Where elbows with $h \leq .25$ were failing the criteria, B_1 was set equal to 0 and B_2 equal to $.67C_2$ as per the Rodabaugh criteria.

b. Also, the piping where $Do/t>50$, B_2,B_{2b}, and B_{2r} were divided by (1.3-.006 D/t) (1.033-.00033T) for ferritic material, and (1.3-.006 D/t) for other materials.

The use of these additional criteria were sufficient to qualify those components failing the original criteria.

FSAR Tables 3.9-2 through 3.9-11, 3.9-18, 3.9-22, and 3.9-37 list the design loading combinations for the major components of each safety-related system.

3.9.3.1.1 Design Loading Combinations

The combination of design loadings is categorized with respect to plant conditions identified as normal, upset, emergency, or faulted as shown in FSAR Tables 3.9-2 through 3.9-11, 3.9-18, 3.9-22, and 3.9-37 for the major components.

This subsection delineates the criteria for selection and definition of design limits and loading combinations associated with normal operation, postulated accidents, and specified seismic events for the design of safety-related ASME code components, except containment components, which are discussed in Section 3.8.

This section also lists the major ASME Class 1, 2, and 3 pressure parts and associated equipment on a component-by-component basis and identifies the applicable loadings, calculation methods, calculated stresses, and allowable stresses. Seismic loads are discussed in Subsection 3.9.2.2 and Section 3.7.

FSAR Tables 3.9-2 through 3.9-11, 3.9-18, 3.9-22, and 3.9-37 listed the calculated stresses or other design values at the time of licensing. Modification, performed after licensing, that effects any calculated stress or design value will be evaluated against accepted allowables at the time of the modification. Results of these evaluations will be listed in the applicable stress report or analysis.

3.9.3.1.2 Design Stress Limits

3.9.3.1.2.1 Stress Level for Seismic Category I Components

Stress analyses were performed for the design basis to determine structural adequacy of pressure components under the operating conditions of normal, upset, emergency, or faulted, as applicable. The stress analyses were performed as appropriate during the design assessment evaluation.

Significant discontinuities such as nozzles, flanges, etc. were considered. In addition to the design calculations required by the ASME III code, stress analysis was performed by methods outlined in the code appendices or by other methods by reference to analogous codes or other published literature.

FSAR Tables 3.9-2 through 3.9-11, 3.9-18, 3.9-22, and 3.9-37 listed the calculated stresses or other design values at the time of licensing. Modification, performed after licensing, that effects any calculated stress or design value will be evaluated against accepted allowables at the time of the modification. Results of these evaluations will be listed in the applicable stress report or analysis.

3.9.3.1.2.2 Field Run Piping

Non-Seismic Category I pipe systems (Class C or D), 2-inch nominal pipe size and less, are field run and are identified in Table 3.2-1. Schematic routing and criteria were provided to the constructor to ensure proper design interface. All non-Seismic Category I piping is anchored where it interfaces with Seismic Category I piping or interfaces appropriately controlled by guides. All Seismic Category I pipe systems (Class A, B, and C) up to and including the isolation valves and piping hangers except the main steam, reactor recirculation system, and insert, withdraw, and scram discharge lines, are identified in Table 3.2-1.

Schematic routing and criteria were provided to the constructor to interface with wall and floor penetrations, shield walls, and equipment access, consistent with the overall plant design.

3.9.3.1.2.3 Stress Levels for ASME Code Class 2 and 3

For safety-related ASME Code Class 2 and 3 components, the design stress limits, allowable loads or required dimensions are listed in FSAR Tables 3.9-2 through 3.9-11, 3.9-18, 3.9-22, 3.9-34 through 3.9-37.

FSAR Tables 3.9-2 through 3.9-11, 3.9-18, 3.9-22, 3.9-25, and 3.9-37 also listed the calculated stresses or other design values at the time of licensing. Modification, performed after licensing, that effects any calculated stress or design value will be evaluated against accepted allowables at the time of the modification. Results of these evaluations will be listed in the applicable stress report or analysis.

Inelastic methods as permitted by ASME Section III for Class 1 components were not used for these components except for Primary Containment Penetrations M-49 and M-50. Non-linear analysis were performed in accordance with Appendix F of the ASME Boiler & Pressure Vessel Code 1974 and 1989 Editions to address thermal overpressurization in Primary Containment Penetrations M-49 andM-50 (Reactor Recirculation System Flow Control Valve Hydraulic Lines) in response to NRC Generic Letter 96-06.

3.9.3.1.2.3.1 Fatigue Evaluation of Downcomer and S/RV Discharge Piping in the Wetwell Air Volume

To evaluate the potential for steam bypass arising from fatigue failure due to high cyclic loadings acting on the downcomers and main steam safety relief valve (S/RV) discharge lines located in the wetwell air volume, a complete fatigue analysis has been performed in accordance with the applicable portions of the ASME Boiler and Pressure Vessel (B&PV) Code, Section III, Division 1, Subsection NB-3600. This evaluation is considered supplemental to and not a replacement for the original design basis for these lines as set forth in the DAR.

3.9.3.1.2.3.1.1 Loads and Load Combinations Used for Assessment

The S/RV piping systems and downcomers are subject to numerous dynamic and hydrodynamic loads from normal, upset, and LOCA related plant operating and accident conditions. For purposes of the fatigue evaluation, the following loads were included: (1) all significant thermal and pressure transients; (2) all cyclic effects due to the hydrodynamic loads including S/RV actuation, CO and chugging; and (3) seismic effects. A description of each of these loads and their combinations is provided in the appropriate DAR sections. The number of occurrences and duration of each load event is given in the DFFR.

3.9.3.1.2.3.1.2 Acceptance Criteria

The design rules, as set forth in the ASME B&PV Code, Section III, Division I, Subsection NB-3600 were utilized for the fatigue assessment. When required, allowables for fatigue stress evaluation were based on mill certification reports.

3.9.3.1.2.3.1.3 Method of Analysis

The S/RV discharge lines and downcomers were analyzed for the appropriate load combinations and their associated number of cycles. The combined stresses and corresponding equivalent stress cycles were computed to obtain the cumulative fatigue usage factors in accordance with the equations of Sub-section NB-3600, Section III, Division I of the ASME B&PV Code.

3.9.3.1.2.3.1.4 Results

The calculated cumulative fatigue usage factors were demonstrated to be all less than 1.0 for the portions of the S/RV discharge lines and downcomers located in the wetwell air volume.

3.9.3.1.2.4 Fatigue Analysis for ASME Code Class 1

For each node point in the subsystem, usage factors are calculated as described in NB-3222.4e(5). These factors are based on the alternating stress defined in NB-3653.3 and Equation 11 of NB-3653.2. The PIPSYS computer program used to find the worst thermal load set and to calculate S_{alt} and the usage factor. With this code, when one load set is eliminated, the moments and a new alternating stress and usage factor are computed. This process continues until all loads are eliminated. Mechanical loads such as earthquake and SRV discharge are also included in this process. All usage factors are then added and compared to applicable design criteria.

3.9.3.1.3 Plant Conditions

All events that the plant might credibly experience during a reactor year are evaluated to establish a design basis for plant equipment. These events are divided into four plant conditions.

The plant conditions described in the following paragraphs are based on event probability (i.e., frequency of occurrence) and correlated design conditions defined in the ASME Boiler and Pressure Vessel Code, Section III.

3.9.3.1.3.1 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.

3.9.3.1.3.2 Upset Condition

Any deviations from normal conditions anticipated to occur often enough that design should include a capability to withstand the conditions without operational impairment. The upset conditions include those transients which result from a single operator error or control malfunction, transients caused by a fault in a system component requiring its isolation from the system, and transients due to loss of load or power, vibratory motions due to an operating-basis earthquake are conservatively treated as upset. Hot standby with the main condenser isolated is an upset condition.

3.9.3.1.3.3 Emergency Condition

Those deviations from normal conditions which require shutdown for correction of the conditions or repair of damage in the RCPB. The conditions have a low probability of occurrence but are included to provide assurance that no gross loss of structural integrity will result as a concomitant effect of any damage developed in the system. Emergency condition events include, but are not limited to, transients caused by one of the following: 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 requires the safety functions of isolation of containment and reactor shutdown; improper assembly of the core during refueling, and vibratory motions of an OBE in combination with associated system transients.

3.9.3.1.3.4 Faulted Condition

Those combinations of conditions associated with extremely low probability, postulated events whose consequences are such that the integrity and operability of the system may be impaired to the extent that considerations of public health and safety are involved. Faulted conditions encompass events that are postulated because their consequences would include the potential for the release of significant amounts of radioactive material. These postulated events are the most drastic that must be designed against and thus represent limiting design bases. Faulted condition events include, but are not limited to, one of the following: a control rod drop accident, a fuel-handling accident, a main steamline break, a recirculation loop break, the combination of (small/large break accident dynamic motion associated with a safe shutdown earthquake and hydrodynamic loads plus a loss of offsite power, or the safe shutdown earthquake.

3.9.3.1.3.5 Correlation of Plant Conditions with Event Probability

The probability of an event occurring per reactor year 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)	$10^{-4} > P > 10^{-6}$

3.9.3.1.4 Safety Class Functional Criteria

For any normal or upset design condition event, Safety Class 1, 2, and 3 equipment shall be capable of accomplishing its safety functions as required by the event and shall incur no permanent changes that could deteriorate its ability to accomplish its safety functions as required by any subsequent design condition event.

For any emergency or faulted design condition event, Safety Class 1, 2, and 3 equipment shall be capable of accomplishing its safety functions as required by the event, but repairs could be required to ensure its ability to accomplish its safety functions as required by any subsequent design condition event.

3.9.3.2 Pump and Valve Operability Assurance

Active mechanical equipment classified as Seismic Category I are designed to perform their function during the life of the plant under postulated plant conditions. Equipment with faulted condition functional requirements include "active" (active equipment must perform a mechanical motion during the course of accomplishing a safety function) pumps and valves in fluid systems such as the residual heat removal system, and core spray systems.

Most NSSS Seismic Category I equipment was originally designed, qualified, purchased and installed to IEEE 344-1971 criteria. Some requalification to IEEE

344-1975 standards on certain Seismic Category I equipment has been accomplished by comparison with dynamic tests performed on functionally equivalent and similarly built NSSS equipment.

Most BOP Seismic Category equipment was originally designed, qualified, and installed to IEEE 344-1975 criteria.

Operability is assured by satisfying the requirements of the following programs. Safety-related valves are qualified by prototype testing and analysis, and safety-related pumps by analysis with suitable stress limits and nozzle loads. The content of these programs is detailed in the following paragraphs.

3.9.3.2.1 ECCS Pumps

All active pumps as listed in the LaSalle controlled computer database are qualified for operability by first being subjected to rigid tests both prior to installation in the plant and after installation in the plant. The in-shop tests include (1) hydrostatic tests of pressure-retaining parts to 125% of the design pressure times the ratio of material allowable stress at room temperature to the allowable stress value at the design temperature; (2) seal leakage tests; and (3) performance tests, while the pump is operated with flow, to determine total developed head, minimum and maximum head, net positive suction head (NPSH) requirements, and other pump/motor parameters. Also monitored during these operating tests are bearing temperatures (except water-cooled bearings) and vibration levels. Both have been shown to be below specified limits. After the pump is installed in the plant, it undergoes the cold hydro tests, functional tests, and the required periodic 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 have been analyzed for operability during a seismic condition by ensuring that (1) the pump will not be damaged during the seismic event, and (2) the pump will continue to operate after the event.

3.9.3.2.1.1 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 and dynamic system loads are limited to the material elastic limit. The average membrane stress (Σ) for the faulted condition load is maintained at 1.2S, or approximately $0.75\sigma_y$ (σ_y - yield stress). The maximum stress in local fibers (σ_m + bending stress (σ_b)) is limited to 1.8S, or approximately $1.1\sigma_y$.

The qualification of the pump and motor as an integral unit while operating under dynamic conditions is provided in the form of a static-earthquake-acceleration analysis. Under this criteria, the unit is considered to be supported as designed, and maximum specified vertical and horizontal accelerations are constantly applied simultaneously in the worst-case combination. The maximum allowable nozzle loads from the attached piping system were considered in an analysis of the pump support to assure that there would be no geometrical/dimensional deformation on the pump components.

3.9.3.2.1.2 Pump Operation During and Following Vibratory Loading

Active pump/motor rotor combinations are designed to rotate at a constant speed under all conditions. Motors are designed to withstand short periods of severe overload. The high rotary inertia in the operating pump rotor and the nature of the random, short-duration loading characteristics of dynamic events will prevent the rotor from becoming seized. In actuality, the loading will 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 a dynamic event and will operate at the design speed after the event.

The functional ability of the active pumps after a faulted condition is assured, since only normal operating loads and steady-state nozzle loads exist. For the active pumps, the faulted condition is greater than the normal condition only due to dynamic loads on the equipment itself.

Faulted events are infrequent and of relatively short duration compared to the design life of the equipment. Since it is demonstrated that the pumps would not be damaged during the faulted condition, the post-faulted condition operating loads will be no worse than the normal plant operating limits. This is assured by requiring that the imposed nozzle loads (steady-state loads) for normal conditions and post-faulted conditions are limited by the magnitudes of the normal condition nozzle loads. The postfaulted condition ability of the pumps to function under these applied loads is proven during the normal operating plant conditions for active pumps.

3.9.3.2.2 SLC Pump and Motor Assembly and RCIC Pump Assembly

These equipment assemblies are small, compact, rigid assemblies with natural frequencies well above 50 hertz. With this fact verified, each equipment assembly has been qualified via static analysis only. This static qualification verifies operability under dynamic conditions, and assures structural loading stresses within Code limitations.

3.9.3.2.3 RCIC Turbine Assembly

The RCIC turbine has been dynamically qualified for operability via a combination of static analysis and dynamic testing. For further information refer to Subsection 3.9.2.2.2.9 and FSAR Table 3.9-37. Modifications that effect the turbine will be evaluated against accepted allowables. The results will be listed in the latest stress analysis.

The qualification of the associated condensate and vacuum pump and motor as an integral unit while operating under dynamic conditions is provided in the form of the static-earthquake-acceleration analysis. Under this criteria the unit is considered to be supported as designed and maximum specified vertical and horizontal accelerations are applied simultaneously in the worst case combination. The maximum allowable nozzle loads from the attached piping system were considered in an analysis of the pump support to assure that there would be no geometrical/dimensional deformation on the pump components.

3.9.3.2.4 ECCS Motors

The analysis of the ECCS motors is performed by a computer program which consists of the static mechanical analysis of motor rotor assembly when acted upon by external forces including magnetic and centrifugal forces at any point along the shaft. The calculation for the seismic condition assumes that the motor is operating and the vibratory, magnetic and centrifugal forces all act simultaneously and in phase on the rotor shaft assembly. Other components of the motor, such as stator frame, lower-end shield, stator supports, base fasteners, top cap, and conduit bus, are checked for the combined effects including self-weight and operational loadings, and consideration of bending, shear, torsion, and direct bearing loads.

The analysis and tests that are used for qualification of ECCS pump motors were performed on an ECCS test motor of very similar mechanical construction.

The type test was performed on a 1250-hp vertical motor in accordance with IEEE 323-1974, first simulating normal operation during the design life, then the motor being subjected to a number of vibrating events, and then to the abnormal environmental condition possible during and after a loss-of-coolant accident (LOCA). The test plan for the type test 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 characteristic curve from IEEE 275-1966 for the insulation type used on the ECCS motors. The amount of aging equaled the total estimated operation days of maximum insulation surface temperature.

b. Radiation aging of the motor electrical insulation equals the maximum estimated integrated dose of gamma during normal and abnormal conditions.

c. The normal operation induced current vibration effect on the insulation system was simulated by 1.5g horizontal vibration acceleration at current frequency for 1 hour duration.

d. The deflection analysis performed on the rotor shaft to ensure adequate rotation clearance, was verified by static loading and deflection of the rotor for the type test motor.

e. Aging and testing on a similar motor was performed on a biaxial test table in accordance with IEEE 344-1975. During this type test, the shake table was activated simulating the vibration design limit of the safe shutdown earthquake with motor starts and operation conditions which may possibly occur during a plant life.

f. An environmental test simulating a LOCA condition with 100 days duration time was performed with the test motor fully loaded, simulating pump operation. At 212°F ambient temperature and 100% steam environment. Another startup and operation of the test motor after 1 hour standstill in the same environment was followed by sufficient operation at high humidity and temperature, based on extrapolation in accordance with the temperature life characteristic curve from IEEE 275-1966 for the insulation type used on the ECCS motors.

3.9.3.2.5 NSSS Valves

The Class 1 active valves are the main steam isolation valves, safety/relief valves, and the standby liquid control valves. These valves are designed to perform their mechanical motion in conjunction with a design base accident. Qualification for operability is unique for each valve type; therefore, each method of qualification is detailed individually in the following.

3.9.3.2.5.1 Main Steam Isolation Valve (MSIV)

The MSIV's are evaluated for operability during dynamic events by both analysis and test.

Analysis - The valve body is designed in accordance with the ASME Boiler and Pressure Vessel code, Section III, Class 1. The code limits deformation in the operating area of the valve body to be within the elastic limit of the material by limiting pressure and pipe reaction input loads (including dynamic) thereby assuring no interference with valve operability. In order to assure design limits are not exceeded for both piping input loads and actuator dynamic loads, the MSIV is mathematically modeled in the main steam line system analysis. The valves' actual input loads, amplified accelerations, and resource frequencies are determined based on site excitation input to the system as a part of the overall steamline analysis. Pipe anchors and restraints are employed as required to limit pipe system resonance frequencies and amplified accelerations to within acceptable limits for the MSIV's. The MSIV analytical qualification results are shown in FSAR Table 3.9-9. Modifications that effects these values will be evaluated against accepted allowables. Results will be listed in the latest stress report or analysis.

Test - A dynamic test was conducted on the MSIV actuator to assure operability at design dynamic loading requirements. A sine wave sweep test is used to determine resonance frequencies of the actuator assembly. A sine beat was used to excite the actuator assembly at all frequencies up to 50 hertz with special emphasis at the resonance frequency. Operability was then demonstrated at each frequency of the sine beat test which verifies that no significant change in valve closing rate resulted from the test. It was also demonstrated that the valve configuration had sufficient integrity to withstand the required simulated dynamic event without compromise of structure or electrical function.

Faulted conditions such as loss-of-coolant accident (LOCA) or downstream line break have been factored into the valve requirements. The LOCA does not affect valve closure as demonstrated by valve qualification test. The valve was also demonstrated to close following a line break by the "State Line Test." The main steam isolation valve operability during LOCA conditions was demonstrated as defined in the report APED-5750 (March 1969). The test specimen was a 20-inch valve of a design representative of the LSCS actuators.

3.9.3.2.5.2 Main Steam Safety/Relief Valves

The SRV's were mechanically qualified by test for operability during a dynamic event. Structural integrity of the configuration during a dynamic event was demonstrated by both code analysis and test.

Analysis - Valves were designed for maximum moments which may be imposed when installed in service for inlet and outlet conditions of 800,000 in.-lb and 600,000 in.-lb respectively. These moments are resultants due to dead weight, thermal expansion, plus dynamic loadings of valve and the connecting pipe. The safety/relief valve analytical qualification results are shown in FSAR Table 3.9-8. Modifications that effect these values will be evaluated against accepted allowables. Results will be listed in the latest stress report or analysis.

A mathematical model of the safety/relief valve is included in the main steamline system analysis to assure that the equipment design limits are not exceeded.

Test - A production safety/relief valve demonstrated operability during a dynamic qualification (shake table) test with moment and "g" loads applied greater than the specified equipment design limit loads. Tests included a resonance frequency search for natural frequencies up to and beyond 50 hertz. The test qualification results of the safety/relief valve are shown in FSAR Table 3.9-8.

3.9.3.2.5.3 Standby Liquid Control Valve (Explosive Valve)

The SLC explosive valves have been generically qualified to IEEE 344-1975 by the vendor. The explosive valves are qualified for operability by test firing the detonator under representative acceleration and environmental conditions. The generic qualification test demonstrated the absence of natural frequencies below 50 hertz, and the ability to remain operable after the application of (SSE) horizontal dynamic loading equivalent to 6.5g and vertical dynamic loading equivalent to 4.5g at 50 hertz.

3.9.3.3 Design and Installation Details for Mounting of Pressure Relief Devices

Safety valves and relief valves were analyzed in accordance with the ASME Section III Code, 1971 and Summer 1972 Addenda.

The method of analysis for safety valves and relief valves suitably accounted for the time-history of loads acting immediately following a valve opening (i.e., first few milliseconds). The fluid induced forcing functions were calculated for each safety valve and relief valve using one-dimensional equations for the conservation of mass, momentum, and energy.

The calculated forcing functions were applied at locations along the associated piping where a change in fluid flow direction occurs. Application of these forcing functions to the associated piping model constituted the dynamic time-history analysis referred to as a hydraulic transient analysis which calculated the dynamic response of the piping system to the forcing functions. Therefore, a dynamic amplification factor is inherently accounted for in the analyses.

It should be noted that the main steam relief valve piping going to the suppression pool has a column of water sitting in the pipe. The hydraulic transient analyses on this piping have accounted for blowing out this water column.

Hydraulic snubbers or strut-type restraints are used on all relief valve and safety valve piping to ensure that the stresses resulting from the loads produced by the sudden opening of a relief or safety valve when combined with stress due to other upset loads satisfy the ASME Section III code for upset conditions. Also, the analyses show that the loads applied to the flanges of the safety and relief valves do not exceed the maximum loads specified by the manufacturer.

3.9.3.4 Component Supports

3.9.3.4.1 Piping

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Piping supports were designed in accordance with Subsection NF of ASME Section III or ANSI B31.1 as appropriate. 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 piping supports as no fatigue evaluation is necessary to meet the Code requirements. All component supports are designed, fabricated, and assembled so that they cannot become disengaged by the movement of the supported pipe or equipment after they have been installed.

The design criteria and dynamic testing requirements for component supports were as follows:

1. Hangers

   the design load on hangers is the load caused by dead weight. The hangers are calibrated to ensure that they support the design load at both the hot and cold load settings. Hangers provide a specified down travel and up travel in excess of the specified thermal movement.

2. Snubbers

   The design load on snubbers includes those loads caused by seismic forces (operating basis earthquake and safe shutdown earthquake) system anchor movements, reaction forces caused by relief valve discharge, turbine stop valve closure, and all other dynamic loads.

   The snubbers were designed to be able to carry the load under normal, upset, emergency, and faulted loading conditions.

   The snubbers were also tested dynamically to ensure that they can perform as required:

   a. The snubber was subjected to either force or displacement that varies approximately sinusoidally.

   b. The frequency of the input motion or force was verified at 5 Hz increments to be within the specified range.

   c. The resulting relative displacements and corresponding loads across the working components is recorded.

d. The test was conducted with the snubber at various temperatures.

e. The peak load in both tension and compression was equal to or higher than the rated load.

f. The duration of the test at each frequency was specified.

The snubber was Type-Tested dynamically at a frequency within a specified frequency range and a minimum specified temperature for the faulted load. Test duration was specified. The snubber was also tested for various abnormal environment conditions.

Upon completion of the above abnormal environmental transient test, the snubber is tested dynamically at a frequency within a specified frequency range. The snubber operated normally during the dynamic test.

3. Struts

The design load on struts includes those loads caused by dead weight, thermal expansion, primary seismic forces, i.e., operating basis earthquake (OBE) and safe shutdown earthquake (SSE), system anchor displacements, reaction forces caused by relief valve discharge, turbine stop valve closure, and all other dynamic loads.

Balance of Plant

Piping supports were designed in accordance with ASME Section III or ANSI B31.1 as appropriate. 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 piping supports as no fatigue evaluation was necessary to meet the Code requirements. All component supports are designed, fabricated, and assembled so that they cannot become disengaged by the movement of the supported pipe or equipment after they have been installed.

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   The design load on snubbers included those loads caused by seismic forces (operating basis earthquake and safe shutdown earthquake) system anchor movements, reaction forces caused by relief valve discharge, turbine stop valve closure, and all other dynamic loads.

   The snubbers were designed to be able to carry the load under normal, upset, emergency, and faulted loading conditions.

3. Struts

   The design load on struts included those loads caused by dead weight, thermal expansion, primary seismic forces, i.e., operating basis earthquake (OBE) and safe shutdown earthquake (SSE), system anchor displacements, reaction forces caused by relief valve discharge, turbine stop valve closure, and all other dynamic loads.

3.9.3.4.2 Equipment

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1. ECCS Pumps

   The HPCS, LPCS, and RHR pumps have been tested in the shop and were tested as defined in Subsection 3.9.3.2. These tests prove the adequacy of the support structure for the pump assembly under operating conditions. Furthermore, the stress calculation summary provided in FSAR Tables 3.9-37, defines the stress level margins in the critical support areas which prove the adequacy of the equipment. Modifications performed after licensing that effect these stresses will be evaluated against accepted allowables at the time of the modification. Results of these evaluations will be listed in the applicable stress analysis.

2. RCIC Turbine

The RCIC turbine assembly has been tested, as defined in Subsection 3.9.3.2. These tests proved the adequacy of the support structure for the turbine assembly under actual operating conditions. Furthermore, the calculated stresses provided in FSAR Table 3.9-37 define the stress level margins in the critical support areas, which prove the adequacy of the equipment. Modifications performed after licensing that effect these stresses will be evaluated against accepted allowables at the time of the modification. Results of these evaluations will be listed in the applicable stress report or analysis.

Balance of Plant

ASME Class 1, 2, and 3 active components are either pumps or valves. Since valves are supported by piping and not tied to building structures, pipe design criteria govern.

Seismic Category I active pump supports were seismically qualified by testing when the pump supports along with the pump fulfill 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. the normal operation of the pump during and after the test indicates that the supports are adequate; and

e. supports were inspected for structural integrity after the test.

Seismic qualification of component supports by analysis was generally accomplished as follows:

a. Stresses at all support elements and parts such as pump holddown and baseplate holddown bolts, pump support pads, pump pedestal, and foundation were 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 do not exceed the values permitted by the designer based on the design verification tests that ensure the operability of the pump.

c. For emergency and faulted plant conditions, the deformations must not exceed the values permitted by the designer to ensure the operability of the pump. Elastic/plastic analysis can be performed when the deflections are above the elastic limits.

3.9.4 Control Rod Drive Systems (CRDS)

3.9.4.1 Descriptive Information of CRDS

Descriptive information on the control rod drive systems is given in Subsection 4.6.1. In particular, the design criteria are discussed in Subsection 4.6.1.1.1, description of the system is given in Subsection 4.6.1.1.2, and method of operation to evaluate adequacy of system is given in Subsection 4.6.2.

3.9.4.2 Applicable CRDS Design Specifications

The quality group classification, code classification, and standards applied in the design, fabrication, and construction of the CRDS are defined in Subsection 3.2.2. Regulatory guide conformance for this system is addressed in Appendix B.

The portions of the CRDS that form a part of the reactor coolant pressure boundary, as described in Subsection 4.6.1, have been analyzed in accordance with the requirements of ASME Section III, as required by the applicable code classification. The remainder of the CRDS has been analyzed to the requirements of a Group D system, as defined in Subsection 3.2.2.

3.9.4.3 Design Loads, Stress Limits, and Allowable Deformations

Design loading combinations are categorized with respect to plant conditions identified as normal, upset, emergency, or faulted as shown in FSAR Table 3.9-3 for the major CRDS components. Stress analysis was used to determine the structural adequacy of pressure-retaining components under the design loading combinations. These analyses utilized allowable stresses and deformation limits established by ASME Section III, and were performed by methods outlined in the Code. FSAR Table 3.9-3 also gives the calculated stress levels or maximum loadings at significant areas for the major system components at the time of licensing. Modifications made after the final licensing analysis will be compared to accepted allowables at the time of the modification. A design assessment calculation is conducted to include dynamic loads. Results will be recorded in the latest stress analysis.

3.9.4.4 CRDS Performance Assurance Program

Quality control (QC) of welding, heat treatment, dimensional tolerance, material verification, and other factory QC tests are used throughout the manufacturing process to ensure reliable performance of the mechanical reactivity control components. Acceptance tests include the following: (1) control rod absorber tube tests to verify integrity, (2) control rod drive mechanism tests, and (3) hydraulic control unit tests to authenticate operational performance.

After installation, all rods and drives are tested through their full stroke for operability. During operation, each time a control rod is withdrawn a notch, the operator can observe the incore monitor indications to verify that the control rod is following the drive mechanism. Hydraulic supply subsystem pressures can be observed from instrumentation in the control room. Scram accumulator pressures can be observed locally on the nitrogen pressure gauges.

Preinstallation specifications define acceptance criteria for characteristics such as seal leakage, friction, and scram performance under fixed test conditions. Normal and scram motions are authenticated in preoperational tests to illustrate proper installation.

A surveillance test is made following core alterations to demonstrate adequate shutdown margin. Also, routine rod withdrawal exercises are made by notch motions to authenticate operable control rods during power operations. Coupling and overtravel tests are also a part of rod exercising tests. Scram tests are conducted periodically every 120 days of operation, following maintenance potentially affecting scram times and at refueling outages to authenticate scram times within acceptable limits.

3.9.5 Reactor Pressure Vessel Internals

3.9.5.1 Design Arrangement

The core support structures and reactor vessel internals include (exclusive of fuel, control rods, and incore nuclear instrumentation) the following components:

a. Core Support Structures

    1. shroud;

    2. shroud support;

    3. core support and holddown bolts;

    4. top guide (including wedges, bolts, and keepers);

5. fuel support pieces; and

6. control rod guide tubes.

b. Reactor Internals

1. jet pump assemblies and instrumentation;

2. shroud head and steam separator assembly (including shroud head bolts);

3. steam dryers;

4. *feedwater spargers;

5. vessel head cooling spray nozzle;

6. differential pressure and liquid control line;

7. *incore flux monitor guide tubes and stabilizers;

8. *initial startup neutron sources;

9. *surveillance sample holders;

10. core spray lines;

11. core spray spargers (part of shroud); and

12. LPCI coupling tie on sparger.

(*non-safety-related)

A general assembly drawing of the important reactor components is shown in Figure 3.9-2.

The floodable inner volume of the reactor pressure vessel can be seen in Figure 3.9-3. It is the volume inside the core shroud up to the level of the jet pump suction inlet.

3.9.5.1.1 Core Support Structure

The core support structure consists of the shroud, shroud support, core support, fuel support pieces, control rod guide tubes, and top guide. This structure is used to form partitions within the reactor vessel, to sustain pressure differentials across the partitions, to direct the flow of the coolant water, and to locate laterally and support the fuel assemblies. Figure 3.9-3 shows the reactor vessel internal flow paths.

3.9.5.1.2 Core Shroud

The core shroud is 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 the shroud is characterized by three regions. The upper shroud surrounds the core discharge plenum, which is bounded by the shroud head on top and the top guide below. The central portion of the shroud surrounds the active fuel and forms the longest section of the shroud. This section is bounded at the bottom by the core support. The lower shroud, surrounding part of the lower plenum, is welded to the reactor pressure vessel shroud support.

3.9.5.1.3 Shroud Head and Steam Separator Assembly

This component is not a core support structure or safety class component. It is discussed here to describe coolant flow paths in the reactor pressure vessel.

The shroud head and steam separator assembly is bolted to the top of the upper shroud flange to form 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, shown in Figure 4.1-5, are welded 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 to establish a vortex separating the water from the steam. The separated water flows from the lower portion of the steam separator into the downcomer annulus.

3.9.5.1.4 Core Support Plate

The core support plate consists of a circular stainless steel plate with bored holes 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 neutron sources. The last two items are also supported vertically by the core support plate.

The entire assembly is bolted to a support ledge between the central and lower portions of the core shroud. Alignment pins that engage slots and that bear against the shroud are used to correctly position the assembly before it is secured.

3.9.5.1.5 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 four fuel assemblies or, in the case of peripheral fuel, one fuel assembly. Notches 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.

3.9.5.1.6 Fuel Support

The fuel supports, shown in Figure 3.9-4 are of two basic types; namely, 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 will support one fuel assembly and contains a single orifice assembly designed to ensure proper coolant flow to the fuel peripheral assembly. Each four-lobed orificed fuel support will support four fuel assemblies and is provided with 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 support 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.

3.9.5.1.7 Control Rod Guide Tubes

The control rod guide tubes, located inside the vessel, extend from the top of the control rod drive housings up through holes in the core support 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 control rod drive 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 control rod drive 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 control rod drive housing to hold the thermal sleeve in position.

3.9.5.1.8 Jet Pump Assemblies

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 pump is covered in detail in References 2 and 3. Each stainless steel jet pump consists of driving nozzles, suction inlet, throat or mixing section, and diffuser (see Figure 3.9-5). The driving nozzle, suction inlet, and throat comprise the inlet mixer assembly which is a removable unit; 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 four cantilever beams and cross members welded to the riser pipe and to four pads on the reactor wall.

The nozzle entry section is connected to the riser by a metal-to-metal, spherical-to-conical seal joint. Contact is maintained by a hold-down clamp. The throat section is supported laterally by a bracket attached to the riser. There is a slip-fit joint between the throat and diffuser. The diffuser is a gradual conical section changing to a straight cylindrical section at the lower end.

Due to damage repaired during L1R08, the following unique features are associated with Unit 1 jet pump 9. Two auxiliary wedges functionally replace the two restrainer bracket adjusting screws. Machining a new surface approximately 0.25 inches from the original damaged face restored the damaged surface of the restrainer bracket pad. A new inlet mixer, containing a wider and thicker wedge replaced the damaged inlet-mixer wedge. These actions maintain the three points of lateral support for the inlet-mixer specified in the original jet pump design.

The use of either one or two auxiliary wedges will provide adequate lateral support for any jet pump.

High levels of slip joint flow can result in damage to the inlet mixer wedge due to flow-induced vibration. A clamp installed at the slip joint will reduce these vibration loads.

During L2R10, replacement jet pump mixer assemblies were installed in all of the Unit 2 jet pumps. To minimize flow-induced vibration, the replacement mixers include a labyrinth seal design at the slip-joint fit between the mixer and the diffuser, and a wider inlet-mixer wedge at the restrainer bracket pad.

During L1R11, jet pump riser brace clamps were installed on Unit 1 jet pumps 5/6 and 9/10 to mitigate crack indications by structurally replacing the upper and lower riser brace yoke to riser pipe welds designated as RS-8 and RS-9.

3.9.5.1.9 Steam Dryers

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 into the downcomer annulus (see Figure 3.9-6). 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 would occur only under accident conditions, is restrained by steam dryer hold-down brackets attached to the reactor vessel top head.

3.9.5.1.10 Feedwater Spargers

The feedwater spargers, located in the mixing plenum above the downcomer annulus, are of the Improved Interference Fit design (also designated by NRC as Triple-Sleeve Spargers) which utilize three concentric thermal fluid zones. This design was chosen for its effectiveness in minimizing thermal cycling in the nozzle bore and blend radius.

The spargers distribute feedwater uniformly to the annular volume between the core shroud and the vessel wall by conducting feedwater through the innermost fluid cylinder of the thermal sleeve, to the sparger header, and through the converging discharge elbow nozzles on the top of the header. This provides subcooling for the jet pumps and helps to maintain a uniform core power distribution.

The LaSalle feedwater spargers are stainless steel. The six headers are served through six feedwater nozzles, each fitted with the triple-thermal sleeves. GE generic report NEDE-21821-02, provided to NRC in response to Generic Activity A-10, includes the technical justification for this type of sparger/thermal sleeve design as employed at LaSalle.

Visual inspection of the flow holes and welds in the sparger assembly is to be accomplished once every four refueling cycles. External UT examination of the nozzle blend radius is included in the formal LaSalle ISI Program. The cladding was removed around the feedwater nozzles of the LaSalle vessels and the base line inspections were made for zone 1 and 2 from the outside of the vessel. A penetrant examination was made on the inner surface of the blend radii. Results are tabulated in the baseline report.

3.9.5.1.11 Core Spray Lines

The core spray lines are the means for directing flow to the core spray nozzles inside the shroud which distribute coolant so that peak fuel cladding temperatures of 2200°F are not exceeded during accident conditions.

Two core spray lines enter the reactor vessel through the two core spray nozzles (see Section 5.4). 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 and pass 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 ends of the two spargers are supported by brackets designed to accommodate thermal expansion. The line routing and supports are designed to accommodate differential movement between the shroud and vessel. The other core spray line is identical except that it enters the opposite side of the vessel and the spargers are at a slightly different elevation inside the shroud. The correct spray distribution pattern is provided by a combination of distribution nozzles pointed radially inward and downward from the spargers.

3.9.5.1.12 Vessel Head Cooling Spray Nozzle

When reactor coolant is returned to the reactor vessel, part of the flow can be diverted to a spray nozzle in the reactor head. This spray maintains saturated conditions in the reactor vessel head volume by condensing steam being generated by the hot reactor vessel walls and internals. The spray also decreases thermal stratification in the reactor vessel coolant. This ensures that the water level in the reactor vessel can rise. The higher water level provides conduction cooling to more of the mass of metal of the reactor vessel and therefore limits thermal stress in the vessel during cooldown.

The vessel head cooling spray nozzle is flange mounted to a mating flange on the reactor vessel head nozzle.

3.9.5.1.13 Differential Pressure and Liquid Control Line

The differential pressure and liquid control line serves a dual function within the reactor vessel - to provide a path for the injection of the liquid control solution into the coolant stream and to sense the differential pressure across the core support plate (described in Section 5.3). 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 near the lower shroud with a perforated length below the core support plate. It is used to sense the pressure below the core support plate during normal operation and to inject liquid control solution if required. This location facilitates good mixing and dispersion. The inner pipe also reduces thermal shock to the vessel nozzle should the standby liquid control system be actuated. The outer pipe terminates immediately above the core support plate and senses the pressure in the region outside the fuel assemblies.

3.9.5.1.14 Incore Flux Monitor Guide Tubes

These tubes provide a means of positioning fixed detectors in the core as well as provide a path for calibration monitors (TIP system).

The incore flux monitor guide tubes extend from the top of the incore flux monitor housing (see Section 5.3) in the lower plenum to the top of the core support plate. The power range detectors for the power range monitoring units and the dry tubes for the source range monitoring and intermediate range monitoring (SRM/IRM) detectors are inserted through the guide tubes. A latticework of clamps, tie bars, and spacers gives lateral support and rigidity to the guide tubes. The bolts and clamps are welded, after assembly, to prevent loosening during reactor operation.

3.9.5.1.15 Surveillance Sample Holders

The surveillance sample holders are welded baskets containing impact and tensile specimen capsules. The baskets hang from the brackets that are attached to the inside wall of the reactor vessel and extend to midheight 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 interference or damage.

3.9.5.2 Design Loading Conditions

The spectrum of conditions for which the safety design basis must be satisfied for the three dominating faulted events is described in Subsection 3.9.2.5.2.2.

3.9.5.2.1 Pressure Differential During Rapid Depressurization

Pressure differentials produced during rapid depressurization are analyzed as shown in Subsection 3.9.2.5.2.3.

3.9.5.2.2 Recirculation Line and Steamline Break

3.9.5.2.2.1 Accident Definition

Both a recirculation line break and an inside steamline break were previously discussed in Subsection 3.9.2.5.3.

3.9.5.2.2.2 Response of Structures Within the Reactor Vessel to Pressure Differences

The maximum differential pressures are used, in combination with other structural loads, to determine the total loading on the various structures within the reactor. The structures are then evaluated to assess the extent of deformation and buckling instability, if any. Of particular interest are: (1) the responses of the guide tubes and the metal channels around the fuel bundles, and (2) the potential leakage around the jet pump joints.

The guide tube is evaluated for buckling instability caused by externally applied pressure. Two primary modes of failure have been analyzed and are described in Subsection 3.9.5.1.7. For a guide tube with minimum wall thickness and maximum allowed ovality, the pressure which causes yield stress is 93 psi compared to the service design pressure of 37.5 psi. The design pressure is in all cases greater than any pressure differential the guide tube will experience including accident conditions. The stress the guide tube would experience is given in Subsection 3.9.2. It is concluded that the guide tube will not fail under the assumed conditions.

The fuel channel load resulting from an internally applied pressure is evaluated, utilizing a fixed-beam analytical model under a uniform load. Tests to verify the applicability of the analytical model indicate that the model is conservative. A roller, at the top of the control rod, guides the blade as it is inserted. If the gap between channels is less than the diameter of the roller, the roller deflects the channel walls as it makes its way into the core. The friction force is a small percentage of the total force available to the control rod drives for overcoming such friction, and it is concluded that the main steamline break accident does not impede the insertability of the control rod.

Jet pump joints have been analyzed to evaluate the potential leakage from within the floodable inner volume of the reactor vessel during the recirculation line break and subsequent LPCI reflooding. Because the jet pump diffuser is welded to the shroud support, the only remaining source of leakage from the lower plenum to the downcomer annulus is the jet pump throat-to-diffuser joint. These joints for all jet pumps leak no more than a total of 225 gpm.

LPCI capacity is sized to accommodate 500 gpm leakage at these locations. It is concluded that the reactor vessel structures retain sufficient integrity during the recirculation line break accident to allow reflooding of the inner volume of the reactor vessel and in sufficient time to prevent significant increases in cladding temperature.

3.9.5.2.3 Dynamic Loads

The seismic and dynamic loads acting on the structures within the reactor vessel are based on analyses as described in Subsection 3.9.1.1 and Section 3.7.

Reactor vessel internals have been evaluated for their response to loads due to operation in ICF and/or FFWTR, the use of 80 mil channels, and the transition to FANP ATRIUM-9B and ATRIUM-10 fuel.

Reference 15 evaluated the effects of a 20% reduction in channel thickness. Reference 15 concluded that the load comparison for vessel internals was either acceptable or remained within design limits due to available margin, and therefore, was also acceptable. Reference 16 determined that the stresses produced on the reactor internals due to ICF and FFWTR would be within the allowable design limits. In Reference 17, the determination is made, with FANP input, that the ATRIUM-9B fuel assembly is similar to the GE9 fuel assembly with regard to its effect on the dynamic response of the 80 mil fuel channel and the change in fuel assembly mass has an insignificant effect on the RPV system. In Reference 20 and 28, the determination is made that an ATRIUM-9B and ATRIUM-10 with 100 mil channels also has an insignificant effect on the RPV and its internals.

3.9.5.2.4 Safety Evaluation

3.9.5.2.4.1 Evaluation Methods

To determine that the safety design bases are satisfied, responses of the reactor vessel internals to loads imposed during normal, upset, emergency, and faulted conditions are examined. The effects on the ability to insert control rods, cool the core, and to flood the inner volume of the reactor vessel are determined. The design assessment procedures are included in Subsection 3.9.1.1.

3.9.5.2.4.1.1 Input for Safety Evaluation

The operating conditions that provide the basis for the design of the reactor internals to sustain normal, upset, emergency, and faulted conditions, as well as load combinations that were used for the core support structures, are covered in Table 3.9-16.

In addition, each combination of operating loads is categorized with respect to either normal, upset, emergency, or faulted conditions as well as the associated design stress intensity or deformation limits.

3.9.5.3 Design Loading Categories

3.9.5.3.1 Stress, Deformation, and Fatigue Limits for Reactor Internals (Except Core Support Structure)

The stress deformation and fatigue criteria listed in Tables 3.9-17 through 3.9-20 were used or the criteria established in applicable codes and standards for similar equipment, by manufacturers' standards, or by empirical methods based on field experience and testing. For the quantity SF_{min} (minimum safety factor) appearing in those tables, the following values listed were used:

Design Condition	SF_{min}
Normal	2.25
Upset	2.25
Emergency	1.5
Fault	1.125*

* Alternate allowable limits for primary stress (based on ASME Code Section III) can be used for faulted condition evaluations.

3.9.5.3.2 Stress, Deformation, and Fatigue Limits for Core Support Structures

The stress, deformation, and fatigue criteria presented in Tables 3.9-21 through 3.9-23 were imposed for the original design basis and the design assessment evaluation. These criteria are supplemented, where applicable, by the criteria for the reactor internals in the previous paragraph, but in no case are the criteria presented in Tables 3.9-21 through 3.9-23 exceeded for core support structures.

3.9.5.4 Design Bases

3.9.5.4.1 Safety Design Bases

The reactor core support structures and internals shall meet the following safety design bases:

a. Arrangement provides a floodable volume in which the core can be adequately cooled thus limiting fuel damage.

b. Deformation is limited to ensure that the control rod movement is not impaired.

c. Mechanical design of applicable structures ensures that safety design bases (a) and (b) are satisfied so that the safe shutdown of the plant and removal of decay heat are not impaired.

3.9.5.4.2 Power Generation Design Bases

The reactor core support structures and internals were designed to the following power generation design bases:

a. They provide the proper coolant distribution during all anticipated normal operating conditions to allow power operation of the core without fuel damage.

b. They are arranged to facilitate refueling operations.

c. They are designed to facilitate planned maintenance and periodic inservice inspection.

3.9.5.4.3 Fuel Assembly Restraints

The fuel assembly structural design demonstrates sufficient dimensional stability and sufficient fuel rod support to maintain core geometry thus avoiding fuel damage for both planned operation and abnormal operational transients.

3.9.5.4.4 Material Selection

The material used for fabricating most of the reactor core support and reactor internal structures are solution heat-treated, unstabilized Type 304 austenitic stainless steel conforming to ASTM and ASME specifications. Weld procedures and welders were qualified in accordance with the intent of Section IX of the ASME Boiler and Pressure Vessel Code. Further controls for stainless steel welding are covered in Subsection 5.2.3.

All the materials of construction exposed to the reactor coolant are resistant to stress corrosion in the BWR coolant. Conservative corrosion allowances were provided for all exposed surfaces of carbon or low alloy steels.

Contaminants in the reactor coolant are controlled to very low limits by the reactor water quality specifications. No detrimental effects occur on any of the materials from allowable contaminant levels in the high purity reactor coolant. Radiolytic products in a BWR have no adverse effects on the construction materials.

3.9.5.4.5 Radiation Effects

Where feasible, the design is such that irradiation effects on the material properties are minimized. Where irradiation effects cannot be minimized, the design of the reactor vessel internals has provisions for replaceable components, or the design satisfies a set of stress and fatigue design limits that have been arrived at considering the effect of irradiation damage on the fracture toughness, ductility, and tensile properties of the materials.

3.9.5.4.6 Accident Conditions

Response analyses of the reactor structures show that deformations are sufficiently limited to allow both adequate control rod insertion and proper operation of the emergency core cooling system. Sufficient integrity of the structures was retained during accident conditions to allow successful reflooding of the reactor vessel inner volume. The analyses considered various loading combinations, including loads imposed by external forces. Thus, safety design bases were satisfied.

3.9.5.4.7 Inspection and Testing

Quality control methods were used during the fabrication and assembly of reactor vessel internals to ensure that the design specifications are met.

The reactor coolant system, which includes the core support structures and reactor internals, is thoroughly cleaned and flushed before fuel is loaded initially.

During the preoperational test program, (Chapter 14.0 of the FSAR) operational readiness tests are performed on various systems. In the course of these tests such reactor internals as the feedwater spargers, the core spray lines, the vessel head cooling spray nozzle, and the standby liquid control system line are functionally tested.

3.9.6 Preservice Inspection

The LSCS reactor coolant pressure boundary including the RPV's Class 1 piping, and all Class 1, 2, 3, and D+ pressure retaining components, defined according to ASME Boiler and Pressure Vessel Code Section III, were examined in accordance with ASME Section XI, 1974 Edition including the Summer 1975 Addenda. Components exempted from examination were those specified in ASME Section XI, IWB-1220 and IWC-1220. One hundred percent of the Class 1, 2, and 3 components were examined prior to initial startup, except as exempted above.

Examination categories for Class 1 and Class 2 components are as specified respectively in ASME Section XI, Tables IWB-2500 and IWC-2520. Class 1 examination methods, as specified in Table IWB-2600, use the ultrasonic method for volumetric examination and either the liquid penetrant or magnetic particle methods for surface examination. Class 2 examination methods, as specified in Table IWC-2600 use these same methods. All Class 3 examinations were visual.

Standards for evaluation of the examination were as follows:

a. Class 1 - As specified in ASME Section XI, IWA-3000 and IWB-3000

b. Class 2 - As specified in ASME Section XI, and D+IWA-3000 and IWC-3000

c. Class 3 - As specified in ASME Section XI, IWA-3000 and IWD-3000

The system pressure test for Class 1, 2, 3, and D+ components was conducted in accordance with ASME Section XI, IWA-5000 and ASME Section III, NB-6000, NC-6000, and ND-6000, respectively.

Records and inspection reports for Class 1, 2, 3, and D+ components were developed and maintained as specified by ASME Section XI, IWA-6000. Personnel performing the nondestructive examinations were qualified per procedures prepared in accordance with SNT-TC-1A, June 1975, for the applicable examination method.

Inservice Inspection

The design of the RPV shield wall and external equipment subject to Inservice Inspection was complete prior to the publication of the amendment to 10 CFR 50.55a which requires the upgrading of the code commitment. Inasmuch as the LSCS plant might be required to meet the requirements of future editions of Section XI, an attempt was made during design to allow more inspection access and added anticipated coverage. The analysis of inspection results for mechanical inspection devices and generally allowed piping examinations were to be upgraded to the requirements of Summer 1975 Addenda to ASME Section XI. 10 CFR 50.55a(g) (2) is applicable for LSCS preservice inspections.

The inservice examinations conducted during the first 120 months will comply with ASME Section XI, 1980 Edition including the Winter 1980 Addenda. The inservice examinations conducted during the second 120 month Inspection Interval will comply with the 1989 Edition of ASME Section XI, except in cases where relief has been granted by the NRC. The inservice examinations conducted during the third 120 month Inspection Interval will comply with the 2001 Edition through the 2003 addenda, including the December of 2003 Erratum of ASME Section XI, except in cases where relief has been granted by the NRC. The Class 1, 2, 3, and D+ pressure-retaining components (as defined in ASME Section III) including the RPV's, will be examined. LaSalle maintains an independent program to address concerns with Intergranular Stress Corrosion Cracking (IGSCC) of Austenitic Stainless Steel piping in accordance with BWRVIP-75 for normal water chemistry plants. (Reference 26) This program governs the examination methods, examination frequency, and sample expansion of those components that fall under the BWRVIP-75 requirements for IGSCC Categories B through G. The Risk-Informed Inservice Inspection Program subsumes all BRWVIP-75 Category A welds.

Examination Categories are those identified in ASME Section XI, 1980 Edition including the Winter 1980 Addenda, the 1989 Edition of ASME Section XI, or the 2001 Edition through the 2003 addenda, of ASME Section XI as applicable. Examination methods (surface or volumetric) are identical to those cited for the preservice examination. The standards for examination evaluations, personnel qualifications, and maintenance of records and reports are accomplished to the same code standards cited for the preservice inspection.

The system pressure tests for Class 1, 2, and 3 components shall be conducted in accordance with ASME Section XI, IWA-5000, IWB-5000, IWC-5000, and IWD-5000.

Inservice Testing of Pumps and Valves

The initial inservice test program for Unit 1 was submitted by letter of December 10, 1980 from L. O. DelGeorge to B. Youngblood as a part of the ISI Plan for LaSalle covering not only piping systems but also pump and valve testing as required by Sections IWV and IWP of ASME Section XI.

The Unit 1 Inservice Inspection Program for piping systems and vessels was resubmitted July 13, 1982 to upgrade the program to the 1980 Edition (1980 Winter Addenda) of ASME Section XI. The Unit 1 Inservice Test Program for pumps and valves was added to the ISI program via Edison letter of February 18, 1983 from C. W. Schroeder (CECo) to A. Schwencer (NRC). Revision 1 to this pump and valve inservice test program was forwarded to the NRC from C. M. Schroeder via letter of July 9, 1983. This revision included responsiveness to the NRC staff visit to the station on September 14 and 15, 1982.

The Inservice Inspection Program for Unit 2 had a similar first-round submittal but was updated to the 1980 Code (Winter 1980 Addenda) and then resubmitted via Edison letter of November 3, 1982 from C. W. Schroeder (CECo) to A. Schwencer (NRC). The Inservice Test Plan for Unit 2 pumps and valves was forwarded to the commission prior to commercial service of that unit. It is similar to the IST plan for Unit 1. The NRC issued an SER for Unit 1 and Unit 2 on August 16, 1988 (Reference 12). CECo re-submitted the Inservice Testing Program for both Units on November 16, 1988 (Reference 13).

Inservice Testing conducted during the second 120 month Testing Interval complied with the 1989 Edition of ASME Section XI, except in cases where relief had been granted by the NRC.

Inservice Testing conducted during the third 120 month Testing Interval will comply with ASME OM Code 2001 Edition through 2003 Addenda, Mandatory Appendix I, Mandatory Appendix II, and ASME OM Code Case OMN-1, Rev. 0, except in cases where relief has been granted by the NRC (Reference 29).

Motor Operated Valve (MOV) Testing

LaSalle has developed an MOV program, in response to NRC Generic Letter 89-10, "Safety Related Motor-Operated Valve Testing and Surveillance" and Generic Letter 96-05, “Periodic Verification of Design-Basis Capability of Safety-Related Motor Operated Valves.” This program includes a comprehensive testing program to verify valve operability and ensure that correct switch settings are established, maintained and monitored throughout the life of the plant. MOVs in safety related systems will be static tested with diagnostics and full dp testing, with diagnostics, will be done when practicable. (References 14, 22, 23, 24, and 25)

3.9.7 References

1. General Electric Company, "Analytical Model for Loss of Coolant Analysis in Accordance with 10 CFR 50 Appendix K," Proprietary Document, General Electric Company, NEDE-20566, November 1975.

2. "Design and Performance of GE BWR Jet Pumps," General Electric Company, Atomic Power Equipment Department, APED-5460, July 1968.

3. R. H. Moen, "Testing of Improved Jet Pumps for the BWR/6 Nuclear System," General Electric Co., Atomic Power Equipment Department, NEDO-10602, June 1972.

4. Mark II Containment Dynamic Forcing Functions Information Report (DFFR) NEDO/NEDE 21061, September 1975.

5. ANSI 176 (Draft), "Design Basis for Protection of Nuclear Power Plants Against Effects of Postulated Pipe Ruptures," January 1977.

6. "BWR Fuel Channel Mechanical Design and Deflection" NEDE-21354-P, September 1976.

7. "BWR/6 Fuel Assembly Evaluation of Combined Safe Shutdown Earthquake (SSE) and Loss of Coolant Accident (LOCA) Loadings," NEDE-21175-P, November 1976 and "Licensing Topical Report BWR Fuel Assembly Evaluation of Combined SSE and LOCA Loading", (Amendment No. 3) NEDE-21175-P, dated October, 1984.

8. A. Kalnins, Journal of Applied Mechanics, Vol. 31, pp. 467-476, September 1964.

9. LaSalle County Station, "Mark II - Design Assessment Report (LSCS-DAR)," Commonwealth Edison Company, Chicago, Illinois.

10. General Electric Company, "LaSalle County Station Unit 2 NSSS New Loads Design Adequacy Evaluation Final Summary Report," NEDO-30159, July 1983.

11. Deleted.

12. Safety Evaluation Report and Relief of the Inservice Testing Program for Pumps and Valves from P. C. Shemanski (NRC) to H. E. Bliss (CECo) dated August 16, 1988.

13. Inservice Testing Program for Pumps and Valves submitted by W. E. Morgan (CECo) to U.S. NRC Document Control Desk on November 16, 1988.

14. Response to Generic Letter 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance," dated September 28, 1990. (AIR 373-104-89-01000)

15. General Electric Report GE-NE-523-A191-1294 "Final Report of the Impact of Using GE9 80-Mil Fuel Channel for the LaSalle Unit 1 and 2", latest revision and supplements.

16. General Electric Report NEDC-31455, Extended Operating Domain and Equipment Out of Service for LaSalle County Nuclear Station Units 1 and 2 , Revision 2, March, 1990.

17. Mechanical & Structural Review, DG96-001813, ComEd Letter J. Minichiello D. Worthington, Review of Siemens Report EMF-96-122(P), October 18, 1996.

18. SPC Report, ANF-89-98 (P), "Generic Mechanical Design Criteria for BWR Fuel Designs," Revision 1, May 1991.

19. Cycle Specific Fuel Design Report (SPC only).

20. Calculation L-002517, Rev. 0 (08-31-99) Seismic Evaluation for Using ATRIUM-9B Fuel Bundles, S. Singh, Sargent & Lundy to D.A. Worthington, September 29, 1999.

21. Power Uprate Project Task 304, "Reactor Internal Pressure Differences," GE-NE-A1300384-41-01, Revision 0, July 1999.

22. Response to Generic Letter 96-05, "Periodic Verification of Design-Basis Capability of Safety-Related Motor Operated Valves" dated November 13, 1996.

23. Response to Generic Letter 96-05, "Periodic Verification of Design-Basis Capability of Safety-Related Motor Operated Valves" dated March 15, 1997.

24. Additional Information Regarding Generic Letter 96-05, "Periodic Verification of Design-Basis Capability of Safety-Related Motor Operated Valves" dated August 24, 1998.

25. Additional Information Regarding Generic Letter 96-05, "Periodic Verification of Design-Basis Capability of Safety-Related Motor Operated Valves" dated April 12, 1999.

26. BWR Vessel and Internals Project Technical Basis for Revisions to Generic Letter 88-01 Inspection Schedules (BWRVIP-75).

27. GE14 Fuel Design Cycle Independent Analyses for LaSalle Unit 1 and Unit 2, GE document GE-NE-0000-0026-4769-00, Revision 0 dated January 2005.

28. LaSalle ATRIUM-10 Seismic/LOCA Evaluation, DEG:01 : 150 dated September 25, 2001.

29. NRC SER associated with Technical Specification Amendments 185 (Unit 1) and 172 (Unit 2) for Technical Specification 5.5.7, dated October 12, 2007.

TABLE 3.9-1

PRESSURE DIFFERENTIALS ACROSS

REACTOR VESSEL INTERNALS - UNIT 1

RESULTS - FAULT CONDITION
Main Steamline Break

Reactor Internals and Core Support Structures - Reactor at Power

REACTOR COMPONENT	CONDITION 1 (PRESSURE DIFFERENTIAL psid)		CONDITION 2 (PRESSURE DIFFERENTIAL psid)	
	1.a	1.b	2.a	2.b
Core Plate and Guide Tube	24.1	24.6	29.1	25.2
Shroud Support Ring and Lower Shroud	45.7	47.8	45.3	45.4
Upper Shroud	25.2	24.6	25.4	26.6
Shroud Head	25.9	25.2	25.9	27.0
Shroud Head to Water Level, Irreversible Δp	27.8	27.3	27.2	28.7
Shroud Head to Water Level, Elevation Δp	1.3	1.4	1.9	2.0
Average Power Channel Box (Bulge)	15.2	16.1	14.4	12.5
Average Power Channel Box (Collapse)	None	None	None	None
Top Guide	2.0	2.2	2.5	2.4
Steam Dryer**	6.0	None	8.1	8.8

CONDITION 1.a: Power corresponding to 105% rated steam flow; 100% rated recirculation flow (100% power = 3323 MWt).

CONDITION 1.b: Power corresponding to 102% of uprated power: 105% rated core flow (100% uprated power = 3489 MWt).

CONDITION 2.a: 66.9 %, rated steam flow/thermal power: 110% rated recirculation flow. (100% power = 3323 MWt).

CONDITION 2.b: 51.4% of uprate power: 105% rated core flow. (100% uprated power = 3489 MWt).

Design Basis Event: Main steamline break inside containment for all DPs calculation except for steam dryer DP which is based on main steamline break outside of containment.

TABLE 3.9-4a

FUEL ASSEMBLY (INCLUDING CHANNEL) – UNIT 2 [4]

Acceptance Criteria [1]	Loading [2]	Primary Load Type	Calculated Peak Acceleration	Evaluation Basis [3] Acceleration
Acceleration Envelope	Horizontal Direction:	Horizontal Acceleration Profile	1.3 G	3.6 G
	1. Peak Pressure			
	2. Operation Basis Earthquake			
	3. Safety Relief Valve			
	4. Chugging			
	Vertical Direction:	Vertical Accelerations	4.2 G	12.0 G
	1. Peak Pressure			
	2. Safe Shutdown Earthquake			
	3. Safety Relief Valve			
	4. Condensation Oscillation			
NOTES:				
(1)	The fatigue analysis indicates that the fuel assembly has adequate fatigue capability to withstand the loading resulting from multiple SRV actuations and the OBE + SRV event.			
(2)	The calculated maximum fuel assembly gap opening for the most limiting load combination is 0.160 inch. This is less than the gap (0.52 inch) required to start the disengagement of the lower tie plate from the fuel support casting.			
(3)	Evaluation Basis Accelerations and Evaluations are contained in NEDE-21175-3-P. The evaluation basis acceleration envelope is defined by a coincident 8G vertical acceleration with the 3.6 G horizontal acceleration. The 3.6 G horizontal value is reduced linearly to zero as the corresponding vertical acceleration increases from 8 to 12 G's (Monte Carlo 84/50 mean estimate).			
(4)	The fuel system criteria for SPC fuel are documented in Reference 18. References 18 and 19 indicate that the fuel assembly structural components must not show yielding with the design limit load.			

TABLE 3.9-16
(SHEET 1 OF 9)

DESIGN LOADING CONDITIONS AND COMBINATIONS

OPERATING CONDITIONS AND STRESS LIMITS*	DESIGN LOADING CONDITIONS AND COMBINATIONS
Normal and Upset	N and A_D or N and U
Emergency	N and R or other conditions which have a 40-year encounter probability from 10^{-1} to 10^{-3}
Fault	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 = normal loads,

U = upset loads excluding earthquake,

A_D = safe shutdown earthquake/2 (SSE/2) including any associated transients,

A_m = safe shutdown earthquake (SSE) including any associated transients.

R = automatic blowdown or equivalent auxiliary pipe rupture loading including any associated transients - pipe rupture loadings are not directly considered on piping itself because this is handled by a failure mode analysis, and

$\overline{R}$ = primary loadings which result from rupture of a main steamline or a recirculation line.

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* The design stress, deformation, and fatigue limits are:

a. for RPV and appurtenances - ASME Section III;

b. for core support structures - Tables 3.9-21, 3.9-22, and 3. 9-23; and

c. for reactor internal structures - Tables 3.9-16, 3.9-17, 3.9-18, and 3.9-19, and 3.9-20.

TABLE 3.9-16
(SHEET 2 OF 9)

LOAD COMBINATIONS AND ACCEPTANCE CRITERIA

a. NSSS ASME Code Vessel/Internals and Piping Components

LOAD CASE[1]	N	SRV_X (4)	SRV ADS	OBE	SSE	SBA/IBA (3)	DBA (6,7)	ACCEPTANCE CRITERIA
1	X	X						Upset B
2	X	X		X				Upset B (5)
3	X	X			X			Faulted D (2)
4	X		X			X		Emergency C (2)
5	X		X	X		X		Faulted D (2)
6	X		X		X	X		Faulted D (2)
7	X				X		X	Emergency C (2)
8	X							Normal A
9	X			X				Upset B

Notes:
(1) See legend at the end of table for definition of terms.
(2) (a) For essential piping systems, faulted allowables are acceptable if functional capability is demonstrated. Essential systems are systems required to mitigate the consequences of the postulated events which cause the loading conditions.
(b) For the reactor vessel and internals, faulted allowables will be used; however, deformation and buckling will be evaluated in accordance with Tables 3.9-17, 3.9-18 and 3.9-19.
(3) SBA or IBA, whichever is greater.
(4) SVR_1, SRV_2, SRV_{LSPA}, SRV_{ALL} (whichever is controlling) will be used.
(5) Not considered in the fatigue evaluation.
(6) DBA includes $LOCA_1$ through $LOCA_7$.
(7) From rated power initial conditions.

TABLE 3.9-16
(SHEET 3 OF 9)

b. BOP PIPING

LOAD CASE		SERVICE LEVEL
1	$N + (OBE^2 + SRV^2_{ALL/ASY} + TR^2)^{1/2}$	B
2	$N + (OBE^2 + SRV^2_{ALL/SIN} + TR^2 + CO^2)^{1/2*}$	C**
3	$N + (OBE^2 + SRV^2_{ALL/ASY} + TR^2 + CO^2)^{1/2*}$	C**
4	$N + (SSE^2 + SRV^2_{ALL/SIN} + TR^2 + CO^2)^{1/2}$	C**
5	$N + (SSE^2 + SRV^2_{ALL/ASY} + TR^2 + CO^2)^{1/2*}$	C**
6	$N + (OBE^2 + SRV^2_{ALL/ASY} + TR^2 + CHUG^2)^{1/2}$	C**
7	$N + (SSE^2 + SRV^2_{ALL/ASY} + TR^2 + CHUG^2)^{1/2}$	C**
8	$N + (SSE^2 + AP^2)^{1/2}$	C**

Load combinations bounded by the above load cases are as follows:

BOUNDED LOAD COMBINATIONS	SERVICE LEVEL	BOUNDING CASE NO.
$N + (OBE^2 + SRV^2_{ALL/ASY} + TR^2)^{1/2}$	C	1
$N + (OBE^2 + TR^2)^{1/2}$	B	1
$N + (SSE + TR^2)^{1/2}$	C	3
N	A	1
$N + (SSE^2 + TR^2)^{1/2} + CO$	C	5
$N + (SSE^2 + SRV_{ALL/ASY}{}^2 + TR^2)^{1/2}$	C	7

where:

N = Normal Loads.
OBE = Operating Basis Earthquake.

*NUREG-0484, Rev. 1, permits the use of SRSS for the CO load. Although SRSS is the permissible method of combination of the CO load and may have been used in some cases, in general the more conservative method of combination by absolute sum was used.

TABLE 3.9-16
(SHEET 4 OF 9)

**Service level D if functional capability is not required.

$SRV_{ALL/ASY}$ = Envelope of All and Asymmetric Valve Discharges - Quencher Definition. This Load is used for ADS valve discharge loads as well as all valve discharge loads.

$SRV_{ALL/SIN}$ = Envelope of single and All Valve discharge - Quencher Definition

TR = Hydraulic Transient Load Where Applicable

CO = Condensation Oscillation

AP = Annulus Pressurization

TABLE 3.9-16
(SHEET 5 OF 9)

c. BOP EQUIPMENT

1. Nonfluid System Equipment

PLANT CONDITION	ACTIVE (ELASTIC DEFLECTION)	NONACTIVE (AND ACTIVE EXACT DEFLECTION)
Upset		
(Normal Operating Loads + $SRV_{ALL\text{-}TQ}$ + OBE Seismic)	σ_m $\leq 0.6\ S_y$ (D.M.) $\leq 0.3\ S_u$ (B.M.) σ_t $\leq 0.7\ S_y$ (D.M.) $\leq 0.4\ S_u$ (B.M.)	σ_m $\leq 0.6\ S_y$ (D.M.) $\leq 0.4\ S_u$ (B.M.) σ_t $\leq 0.9\ S_y$ (D.M.) $\leq 0.6\ S_u$ (B.M.)
Emergency		
1. N + $SRV_{ALL\text{-}TQ}$ 2. N + SSE + CO-2 + $SRV_{ADS\text{-}TQ}$ 3. N + Chugging + $SRV_{ADS\text{-}TQ}$ + SSE 4. N + SSE + AP 5. N + SSE + CO-1	σ_m $\leq 0.7\ S_Y$ (D.M.) $\leq 0.4\ S_u$ (B.M.) σ_t $\leq 0.95\ S_y$ (D.M.) $\leq 0.6\ S_u$ (B.M.)	σ_m $\leq 0.9\ S_y$ (D.M.) $\leq 0.6\ S_u$ (B.M.) σ_t $\leq 1.5\ S_y$ (D.M.) $\leq 0.9\ S_u$ (B.M.)

2. Active Fluid System Equipment

PLANT CONDITION	ASME CLASS 1	ASME CLASS 2 & 3
Upset .		
(SRV_{ALL} + OBE +Normal Operating Loads)	Per ASME Sec. III Same as Nonactive	Per ASME Sec. III Same as Nonactive
Emergency		
1. N + OBE + $SRV_{ALL\text{-}TQ}$ 2. N + SSE + CO-2 + $SRV_{ADS\text{-}TQ}$ 3. N + Chugging + $SRV_{ADS\text{-}TQ}$ + SSE 4. N + SSE + AP 5. N + SSE + CO-1	σ_m $\leq 1.00\ S_m$ σ_t $\leq 1.5\ S_m$	σ_m $\leq 1.00\ S_h$ σ_t $\leq 1.65\ S_h$

TABLE 3.9-16
(SHEET 6 OF 9)

3. HVAC Ducts and Supports

Design Basis for Both Inside and Outside Containment

The structural integrity of the safety-related HVAC ducts and supports for all applicable loading combinations is achieved by the following design rules:

a. Determining, and controlling if necessary by modifying the support structure, the frequencies of the duct-support assembly to avoid peak response.

b. Analyzing the supporting structures for all applicable loadings (in all directions including axial direction) and obtain the resultant stresses in members and connections.

c. All calculated loads at the interface between the support and the structural steel is transmitted to the Structural Dept. and will be used in checking the structural steel.

d. Selecting a set of design limits to be associated with applicable loading combinations (see below). These design limits will not permit the stresses to exceed the yielding stress. This will be strictly followed in designing the support members and connections; however, local yielding in the duct may be allowed on a case by case basis after additional studies.

Design Load Combinations

The following bounding load cases were used for reevaluation of HVAC ducts and supports. These load combinations are consistent with those used in other components.

Loading Combination	Stress Limit	Plant Condition
a. N + OBE (1% damping) + SRV $_{ALL-TQ}$ (1% damping)	0.9 S_y	Upset
b. N + SSE (2% damping) + C0-2 (2% damping) + SRV_{ADS-TQ} (2% damping)	1.2 S_y	Emergency

TABLE 3.9-16
(SHEET 7 OF 9)

Loading Combination (Cont'd)	Stress Limit	Plant Condition
c. N + SRV_{ADS-TQ} (2% damping) + Chugging (2% damping) + SSE (2% damping)	1.2 S_y	Emergency
d. N + SSE (2% damping) + AP (2% damping)	1.2 S_y	Emergency
e. N + SSE (2% damping) + CO-1 (2% damping)	1.2 S_y	Emergency

The seismic loads are combined with the pool dynamic loads by SRSS method except case (b) and (e) where CO is combined using absolute sum.

Note: SRV_{ALL-TQ} = envelope of SRV_{ALL-TQ} and $SRV_{ASYM-TQ}$
SRV_{ADS-TQ} = envelope of SRV_{ALL-TQ} and $SRV_{ASYM-TQ}$

Special Considerations

Axial restraints along the direction of the duct are placed, if necessary to provide longitudinal strength and rigidity.

The supports on each side of active HVAC components (such as dampers) are designed to assure that the loads used in qualifying these components will not be exceeded.

Where:

σ_m = Membrane stress
S_u = Ultimate stress corresponding temperature
σ_t = Membrane + bending stress
S_h, S_m = As defined by Section III
D.M. = Ductile material
S_y = Yield stress at corresponding temperature
B.M. = Brittle material

TABLE 3.9-16
(SHEET 8 OF 9)

LOAD DEFINITION LEGEND

Normal (N)	-	Normal and/or abnormal loads depending on acceptance criteria.
OBE	-	Operational basis earthquake loads.
SSE	-	Loads due to vibratory motion from safe shutdown earthquake loads.
SRV (1)	-	Safety/relief valve discharge induced loads from one valve's subsequent actuation.
SRV (2)	-	Safety/relief valve discharge induced loads from 2 adjacent valves.
SRV (ALL)	-	The loads induced by actuation of all safety/relief valves which activate within milliseconds of each other (e.g., turbine trip operational transient).
SRV (ADS)	-	The loads induced by the actuation of safety/relief valves associated with automatic depressurization system which actuate within milliseconds of each other during the postulated small or intermediate size pipe rupture.
SRV (LSPA)	-	Actuation of safety/relief valves - lowest setpoint actuation.
LOCA	-	The 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/fallout 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 acting on piping and components located above the suppression pool water upper surface.
$LOCA_3$	-	Oscillating pressure induced loads on submerged piping and components during condensation oscillations, i.e., chugging.
$LOCA_4$	-	Building motion induced loads from chugging (condensation oscillation).

TABLE 3.9-16
(SHEET 9 OF 9)

$LOCA_5$	-	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	-	Small break accident.
IBA	-	Intermediate break accident.

TABLE 3.9-17

DEFORMATION LIMIT

(For Reactor Internal Structures Only)

EITHER ONE OF (NOT BOTH)	GENERAL LIMIT
a. $\left[\dfrac{\text{Permissible deformation, DP}}{\text{Analyzed deformation causing loss of function, DE}}\right]$	$\leq \dfrac{0.9}{SF_{min}}$
b. $\left[\dfrac{\textit{Permissible deformation, DP}}{\textit{Experiment deformation causing loss of function, DE}}\right]$ *	$\leq \dfrac{0.9}{SF_{min}}$

where:

DP = permissible deformation under stated conditions of normal, upset, emergency, or fault;

DL = analyzed deformation which could cause a system loss of function;**

DE = experimentally determined deformation which could cause a system loss of function; and

SF_{min} = minimum safety factor

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* Equation b was not used because equation a criterion was met.

**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 will be specifically delineated. From a practical viewpoint, it is convenient to interchange some deformation condition at which function is assured with the loss of function condition if the required safety margins from the functioning conditions can be achieved. Therefore, it is often unnecessary to determine the actual loss of function condition because this interchange procedure produces conservative and safe designs. Examples where deformation limits apply are: control rod drive alignment and clearances for proper insertion, core support deformation causing fuel disarrangement, or excess leakage of any component.

TABLE 3.9-18
(SHEET 1 OF 3)

PRIMARY STRESS LIMIT

(For Reactor Internal Structures Only)

ANY ONE OF (NO MORE THAN ONE REQUIRED)	GENERAL LIMIT
a. $\left[\frac{\text{Elastic evaluated primary stresses, PE}}{\text{Permissible primary stresses, PN}}\right]$	$\leq \frac{2.25}{SF_{min}}$
b. $\left[\frac{\text{Permissible load, LP}}{\text{Largest lower bound limit load, CL}}\right]$	$\leq \frac{1.5}{SF_{min}}$
c. $\left[\frac{\text{Elastic evaluated primary stresses, PE}}{\text{Conventional ultimate strength at temperature, US}}\right]$	$\leq \frac{0.75}{SF_{min}}$
d. $\left[\frac{\text{Elastic – plastic evaluated nominal primary stress, EP}}{\text{Conventional ultimate strength at temperature, US}}\right]$	$\leq \frac{0.9}{SF_{min}}$
e. $\left[\frac{\text{Permissible load, LP}}{\text{Plastic instability load, PL}}\right]^{*}$	$\leq \frac{0.9}{SF_{min}}$
f. $\left[\frac{\text{Permissible load, LP}}{\text{Ultimate load from fracture analysis, UF}}\right]^{*}$	$\leq \frac{0.9}{SF_{min}}$
g. $\left[\frac{\text{Permissible load, LP}}{\text{Ultimate load or loss of function load from test, LE}}\right]^{*}$	$\leq \frac{1.0}{SF_{min}}$

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 normal or upset conditions under ASME Boiler and Pressure Vessel Code, Section III.

- Equations e, f, and g were not used because criteria "a, b, and c" were met.

TABLE 3.9-18
(SHEET 2 OF 3)

LP = permissible load under stated conditions of normal, upset, emergency, or fault.

CL = 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 of the ASME 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.

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.

TABLE 3.9-18
(SHEET 3 OF 3)

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.

TABLE 3.9-19

BUCKLING STABILITY LIMIT

(For Reactor Internal Structures Only)

ANY ONE OF (NO MORE THAN ONE REQUIRED)	GENERAL LIMIT
a. $\left[\dfrac{\text{Permissible load, LP}}{\text{Code normal event permissible load, PN}}\right]$	$\le \dfrac{2.25}{SF_{min}}$
b. $\left[\dfrac{\text{Permissible load, LP}}{\text{Stability analysis load, SL}}\right]$	$\le \dfrac{0.9}{SF_{min}}$
c. $\left[\dfrac{\text{Permissible load, LP}}{\text{Ultimate buckling collapse load from test, SE}}\right]^*$	$\le \dfrac{1.0}{SF_{min}}$

where:

LP = permissible load under stated conditions of normal, upset, emergency, or fault.

PN = applicable code normal event 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.

- Equation c was not used because criteria "a and b" were met.

TABLE 3.9-20

FATIGUE LIMITS

(For Reactor Internal Structures Only)

Summation of fatigue damage usage with design and operation loads following Miner hypotheses . . . *

ANY ONE OF (NO MORE THAN ONE REQUIRED)	LIMIT FOR NORMAL AND UPSET DESIGN CONDITIONS
a. Mean fatigue **, † cycle usage from analyses	≤ 0.05
b. Mean fatigue **, † cycle usage from test	≤ 0.33
c. Design fatigue cycle usage from analysis using the method of Table 3.9-21	≤ 1.0

* M.A. Miner, "Cumulative Damage in Fatigue," Journal Applied Mechanics, Vol. 12, Vol. 67, pp A159-A164, ASME September 1945.

** Fatigue failure is defined here as a 25% area reduction for a load carrying member which is required to function, or excess leakage, whichever is more limiting.

† Equations a and b were not used.

TABLE 3.9-21
(SHEET 1 OF 4)

CORE SUPPORT STRUCTURES STRESS CATEGORIES AND LIMITS OF

STRESS INTENSITY FOR NORMAL AND UPSET CONDITIONS

STRESS CATEGORY	PRIMARY STRESSES		SECONDARY STRESSES	PEAK STRESSES
	MEMBRANE, P_m (NOTES 4, 7, AND 8)	BENDING P_b (NOTES 4, 7, AND 8)	MEMBRANE & BENDING SECONDARY, Q (NOTES 2, 4, AND 6)	PEAK, F (NOTES 2, 4, AND 6)
NORMAL AND UPSET	P_m S_m ELASTIC ANALYSIS (NOTE 6) or $0.67L_L$ LIMIT ANALYSIS (NOTE 10) or $0.44L_u$ TEST (NOTE 11)	P_m+P_b $1.5S_m$ ELASTIC ANALYSIS (NOTE 6) or $0.67L_L$ LIMIT ANALYSIS (NOTE 10) or $0.44L_u$ TEST (NOTE 11)	P_m+P_b+Q $3S_m$ ELASTIC ANALYSIS (NOTE 1) or S_L PLASTIC ANALYSIS (NOTE 5) or FOR CYCLES LESS THAN 1000, USE PEAK (NOTE 12)	P_m+P_b+Q+F S_a ELASTIC FATIGUE (NOTES 3 AND 9) P_m+P_b+Q+F S_a ELASTIC PLASTIC FATIGUE (NOTES 3, 9 AND 12)

TABLE 3.9-21
(SHEET 2 OF 4)

NORMAL AND UPSET CONDITIONS

NOTE 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 Boiler and Pressure Vessel Code, Section III (ASME 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 the transient.

NOTE 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, $P_m = S$, $P_b = 0$, $Q = 0$, and a notch with a stress concentration K is introduced, then $F = P_m (K-1)$ and the peak stress intensity equals $P_m + P_m (K-1) = KP_m$.

NOTE 3 - S_a is obtained from the fatigue curves, Figures I-9.1 and I-9.2 of ASME III. The allowable stress intensity for the full range of fluctuation is 2 S_a.

NOTE 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 o_t, σ_l, σ_r, τ_{tl}, τ_{lr}, τ_{rt}

NOTE 5 - S_L denotes the structural action of shakedown load as defined in Paragraph NB-3213.18 of ASME III calculated on a plastic basis as applied to a specific location on the structure.

TABLE 3.9-21
(SHEET 3 OF 4)

NOTE 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.

NOTE 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 III). For external pressure, the permissible equivalent static external pressure shall be as specified by the rules of Paragraph NB-3133 of ASME III. Where dynamic pressures are involved, the permissible external pressure shall be limited to 25% of the dynamic instability pressure.

NOTE 8 - When loads are transiently applied, consideration should be given to the use of dynamic load amplification, and possible change in modulus of elasticity.

NOTE 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.

NOTE 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 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.

NOTE 11 - For 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 44% of L_u, where L_u is the ultimate load or the maximum load or load combination used in the test.

TABLE 3.9-21
(SHEET 3 OF 4)

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 normal and upset conditions.

NOTE 12 - The allowable value for the maximum range of this stress intensity is $3S_m$ except for cyclic events which occur less than 1000 times during the design life of the plant. For this exception, in lieu of meeting the $3S_m$ limit, an elastic-plastic fatigue analysis in accordance with ASME 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.

CORE SUPPORT STRUCTURES STRESS CATEGORIES AND LIMITS OF STRESS INTENSITY FOR EMERGENCY CONDITIONS

STRESS CATEGORY	PRIMARY STRESSES		SECONDARY STRESSES	PEAK STRESSES
	MEMBRANE, P_m (NOTES 1, 2, AND 10)	BENDING P_B (NOTES 1, 2, AND 10)	MEMBRANE & BENDING SECONDARY, Q	PEAK, F
EMERGENCY (NOTE 9)	P_m $1.5S_m$ — ELASTIC ANALYSIS (NOTE 3) OR L_L — LIMIT ANALYSIS (NOTE 4) OR $1.5S_m$ — PLASTIC ANALYSIS (NOTE 6) OR $0.6L_e$ — TEST (NOTE 7) OR S_E — STRESS-RATIO ANALYSIS (NOTE 8)	P_m+P_B $2.25S_m$ — ELASTIC ANALYSIS (NOTE 3) OR L_L — LIMIT ANALYSIS (NOTE 4) OR $2.25S_m$ — PLASTIC ANALYSIS (NOTE 5 AND 6) OR $0.5S_u$ — (NOTE 5) OR $0.6L_e$ — TEST (NOTE 7) OR KS_E — STRESS-RATIO ANALYSIS (NOTE 8)	EVALUATION NOT REQUIRED	EVALUATION NOT REQUIRED

EMERGENCY CONDITIONS

NOTE 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, τ_{tl}, τ_{lr}, τ_{rt}.

NOTE 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 Boiler and Pressure Vessel Code, Section III (ASME 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.

NOTE 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.

NOTE 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 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.

NOTE 5 - S_u is the ultimate strength at temperature. Multiaxial effects on ultimate strength shall be considered.

NOTE 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.

NOTE 7 - For 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 emergency conditions.

NOTE 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; $S_e \leq 2S_m$ for primary membrane loading.

NOTE 9 - Where deformation is of concern in a component, the deformation shall be limited to two-thirds the value given for emergency conditions in the design specification.

NOTE 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 3.9-23
(SHEET 1 OF 3)

CORE SUPPORT STRUCTURES STRESS CATEGORIES AND LIMITS
OF STRESS INTENSITY FOR FAULT CONDITIONS

STRESS CATEGORIES	PRIMARY STRESSES		SECONDARY STRESSES	PEAK STRESSES
	MEMBRANE, P_m (NOTES 1, 2, AND 3)	BENDING P_B (NOTES 1, 2, AND 3)	MEMBRANE & BENDING SECONDARY, Q	PEAK, F
FAULT (NOTE 9)	P_m $2.4\ S_m$ ELASTIC ANALYSIS OR $0.75S_u$ (NOTE 5) OR $1.33L_L$ LIMIT ANALYSIS (NOTE 4) OR $0.67S_u$ PLASTIC ANALYSIS (NOTES 5 AND 6) OR $0.8L_F$ TEST (NOTE 7) OR S_F STRESS-RATIO ANALYSIS (NOTE 8)	P_m+P_B $3.0S_m$ ELASTIC ANALYSIS (NOTE 10) OR $1.33L_L$ LIMIT ANALYSIS (NOTE 4) OR $0.75S_u$ PLASTIC ANALYSIS (NOTES 5 AND 6) OR $0.8L_F$ TESTS (NOTE 7) OR KS_F STRESS-RATIO ANALYSIS (NOTE 8)	EVALUATION NOT REQUIRED	EVALUATION NOT REQUIRED

TABLE 3.9-23
(SHEET 2 OF 3)

NOTE 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, τ_{lr}, and τ_{rt}.

NOTE 2 - When loads are transiently applied r consideration should be given to the use of dynamic load amplification and possible changes in modulus of elasticity.

NOTE 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 Boiler and Pressure Vessel Codes Section III (ASME III). Where dynamic pressures are involved, the permissible external pressure shall satisfy the preceding requirements or shall be limited to 75% of the dynamic instability pressure.

NOTE 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 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.

NOTE 5 - S_u is the ultimate strength at temperature. Multiaxial effects on ultimate strength shall be considered.

NOTE 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 curve may be used either the maximum shear stress or strain energy of distortion flow rule shall be used to account for multiaxial effects.

TABLE 3.9-23
(SHEET 3 OF 3)

NOTE 7 - For 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 shall be 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 fault condition.

NOTE 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.

NOTE 9 - Where deformation is of concern in a component, the deformation shall be limited to 80% of the value for fault conditions in the design specifications.

NOTE 10 - Alternate allowable limits for primary stress (based on ASME Code, Section III) can be used for faulted condition evaluations.

TABLE 3.9-24
(SHEET 1 OF 21)

APPLICABLE THERMAL TRANSIENTS

(Prestartup Hydro Test
130 Cycles Condition - Test)

PIPELINE	INITIAL TEMP. ° F	FINAL TEMP. ° F	TEMP. RATE ° F/HOUR	ΔTEMP. ° F
Main steamline	70	100	60	30
Head spray (RHR)	70	100	60	30
Feedwater	70	100	60	30
Recirculation suction	70	100	60	30
Recirculation discharge	70	100	60	30
Core spray	70	100	60	30
CRDHS return	70	100	60	30.
Standby liquid control	70	100	60	30
10-minute duration	100	50	Step	50
	50	100	Step	50
Bottom drain	70	100	60	30

Note: After temperature is raised to 100° F reactor pressure is increased to 1250 psig and then decreased to 0 psig.

TABLE 3.9-24
(SHEET 2 OF 21)

(Applicable Thermal Transients Startup
120 Cycles Condition - Normal)

PIPELINE	INITIAL TEMP. ° F	FINAL TEMP. ° F	TEMP. RATE ° F/HOUR	ΔTEMP. ° F
Main steamline	100	552	100	452
Head spray (RHR)	100	552	100	452
Feedwater	100	552	100	452
	552	50	Step	462
	50	420	660	330
Recirculation suction	100	552	100	452
	552	544	Step	8
	544	528	32	16
Recirculation discharge	100	552	100	452
	552	544	Step	8
	544	528	32	16
Core spray	100	552	100	452
	552	544	Step	8
	544	528	32	16
CRDHS return	100	50	Step	50
Standby liquid control	100	400	100	300
	400	544	Step	144
	544	528	32	16
Bottom drain	100	400	100	300
	400	544	Step	144
	544	528	32	16

TABLE 3.9-24
(SHEET 3 OF 21)

(Applicable Thermal Transients Daily Power Reduction
and Rod Pattern Change 10,400 Cycles Condition - Normal)

PIPELINE	INITIAL TEMP. ° F	FINAL TEMP. ° F	TEMP. RATE ° F/HOUR	ΔTEMP. ° F
Main steamline	552	552	No Change	
Head spray	552	552	No Change	
Feedwater	420	354	264	66
	354	420	264	66
Clean return	436	436	No Change	
Recirculation suction	528	528	No Change	
Recirculation discharge	528	528	No Change	
Core spray	528	528	No Change	
CRDHS return	50	50	No Change	
Standby liquid control	528	528	No Change	
Bottom drain	528	528	No Change	

Note: Reactor pressure remains at 1000 psig.

TABLE 3.9-24
(SHEET 4 OF 21)

(Applicable Thermal Transients Weekly Power Reduction
2000 Cycles Condition - Normal)

PIPELINE	INITIAL TEMP. ° F	FINAL TEMP. ° F	TEMP. RATE ° F/HOUR	ΔTEMP. ° F
Main steamline	552	552	No Change	
Head spray (RHR)	552	552	No Change	
Feedwater	420	324	192	96
	324	420	192	96
Cleanup return	436	436	No Change	
Recirculation suction	528	528	No Change	
Recirculation discharge	528	528	No Change	
Core spray	528	528	No Change	
CRDHS return	50	50	No Change	
Standby liquid control	528	528	No Change	
Bottom drain	528	528	No Change	

Note: Reactor pressure remains at 1000 psig.

TABLE 3.9-24
(SHEET 5 OF 21)

(Applicable Thermal Transients Turbine Trip
100% Bypass 10 Cycles Condition - Upset)

PIPELINE	INITIAL TEMP. ° F	FINAL TEMP. ° F	TEMP. RATE ° F/HOUR	ΔTEMP. ° F
Main steamline	552	552	No Change	
Head spray (RHR)	552	552	No Change	
Feedwater	420	100	Step	320
	100	420	4800	320
Recirculation suction	528	496	1280	32
	496	528	480	32
Core spray	528	496	1280	32
	496	528	480	32
CRDHS return	50	50	No Change	
Standby liquid control	528	496	1280	32
	496	528	480	32
Bottom drain	528	496	1280	32
	496	528	480	32

Note: Reactor pressure remains at 1000 psig.

TABLE 3.9-24
(SHEET 6 OF 21)

(Applicable Thermal Transients Feedwater Heater Loss
Partial Heater Bypass 70 Cycles Condition - Upset)

PIPELINE	INITIAL TEMP. ° F	FINAL TEMP. ° F	TEMP. RATE ° F/HOUR	ΔTEMP. ° F
Main steamline	552	552	0	0
Head spray (RHR)	552	552	0	0
Feedwater	420	265	6200	155
	265	420	3100	155
Recirculation suction	528	518	Step	10
	518	528	Step	10
Recirculation discharge	528	518	Step	10
	518	528	Step	10
Core spray	528	518	Step	10
	518	528	Step	10
CRDHS return	50	50	0	0
Standby liquid control	528	518	Step	10
	518	528	Step	10
Bottom drain	528	518	Step	10
	518	528	Step	10

Note: Reactor pressure remains at 1000 psig.

TABLE 3.9-24
(SHEET 7 OF 21)

(Applicable Thermal Transients Scram - 1-G Trip
Feedwater On - MSIV Open 40 Cycles Condition - Upset)

PIPELINE	INITIAL TEMP. ° F	FINAL TEMP. ° F	TEMP. RATE ° F/HOUR	ΔTEMP. ° F
Main steamline	552	565	4680	13
	565	538	6500	27
	538	400	100	138
	400	552	100	152
Head spray (RHR)	552	565	4680	13
	565	538	6500	27
	538	400	100	138
	400	552	100	152
Feedwater	420	275	8700	145
	275	100	700	175
	100	250	Step	150
	250	420	340	170
Recirculation suction	528	400	100	128
	400	552	100	152
	552	544	Step	8
	544	528	32	16
Recirculation discharge	528	400	100	128
	400	552	100	152
	552	544	Step	8
	544	528	32	16
Core spray	528	400	100	128
	400	552	100	152
	552	544	Step	8
	544	528	32	16
CRDHS return	50	50	No Change	
10 Cycles only	(50	528	Step	478
	(528	50	Step	478
Standby liquid control	528	250	200	278
	250	400	100	150
	400	544	Step	144
	544	528	32	16
Bottom drain	528	250	200	278
	250	400	100	150
	400	544	Step	144
	544	528	32	16

Note: Reactor pressure increases to 1125 psig all relief valves open; pressure decreases to 240 psig and then increases to 1000 psig.

TABLE 3.9-24
(SHEET 8 OF 21)

(Applicable Thermal Transients All Other Scrams
140 Cycles Condition - Upset)

PIPELINE	INITIAL TEMP. ° F	FINAL TEMP. ° F	TEMP. RATE ° F/HOUR	ΔTEMP. ° F
Main steamline	552	538	3360	14
	538	400	100	138
	400	552	100	152
Head spray (RHR)	552	538	3360	14
	538	400	100	138
	400	552	100	152
Feedwater	420	275	8700	145
	275	100	700	175
	100	250	Step	150
	250	420	340	170
Recirculation suction	528	400	100	128
	400	552	100	152
	552	544	Step	8
	544	528	32	16
Recirculation discharge	528	400	100	128
	400	552	100	152
	552	544	Step	8
	544	528	32	16
Core spray	528	400	100	128
	400	552	100	152
	552	544	Step	8
	544	528	32	16
CRDHS return	50	50	No Change	
30 Cycles only	(50	528	Step	478
	(528	50	Step	478
Standby liquid control	528	250	200	278
	250	400	100	150
	400	544	Step	144
	544	528	32	16
Bottom drain	528	250	200	278
	250	400	100	150
	400	544	Step	144
	544	528	32	16

Note: Reactor pressure decreases to 240 psig and then increases to 1000 psig.

TABLE 3.9-24
(SHEET 9 OF 21)

(Applicable Thermal Transients Rated Power
*See Below for Events Condition - Normal)

PIPELINE	INITIAL TEMP. ° F	FINAL TEMP. ° F	TEMP. RATE ° F/HOUR	ΔTEMP. ° F
Main steamline	552	552	No Change	0
Head spray (RHR)	552	552	No Change	0
Feedwater	420	420	No Change	0
Recirculation suction	528	528	No Change	0
Recirculation discharge	528	528	No Change	0
Core spray	528	528	No Change	0
Core spray high pressure				
*10 Cycles	528	40	Step	488
CRDHS return	50	50	No Change	0
Standby liquid control	528	60	-	468
*10 Cycles	60	528	462	468
Bottom drain	528	528	No Change	0

Note: Reactor pressure remains at 1000 psig.

TABLE 3.9-24
(SHEET 10 OF 21)

(Applicable Thermal Transients Reduction to 0% Power
111 Cycles Condition - Normal)

PIPELINE	INITIAL TEMP. ° F	FINAL TEMP. ° F	TEMP. RATE ° F/HOUR	ΔTEMP. ° F
Main steam	552	552	0	0
Head spray	552	552	0	0
Feedwater	420	265	310	155
Recirculation suction	528	552	32	24
Recirculation discharge	528	552	32	24
Core spray	528	552	32	24
CRDHS return	50	50	0	0
Standby liquid control	528	552	48	24
	552	400	200	152
Bottom drain	528	552	48	24
	552	400	200	152

Note: Reactor pressure remains at 1000 psig.

TABLE 3.9-24
(SHEET 11 OF 21)

(Applicable Thermal Transients Hot Standby
111 Cycles Condition - Normal)

PIPELINE	INITIAL TEMP. ° F	FINAL TEMP. ° F	TEMP. RATE ° F/HOUR	ΔTEMP. ° F
Main steam	552	552	0	0
Head spray	552	552	0	0
Feedwater	265	552	Step	287
	552	100	Step	452
	100	552	Step	452
Recirculation suction	552	552	0	0
Recirculation discharge	552	552	0	0
Core spray	552	552	0	0
CRDHS return	50	50	0	0
Standby liquid control	400	400	0	0
Bottom drain	400	400	0	0

Note: Reactor pressure remains at 1000 psig.

TABLE 3.9-24
(SHEET 12 OF 21)

(Applicable Thermal Transients Shutdown
111 Cycles Condition - Normal)

PIPELINE	INITIAL TEMP. ° F	FINAL TEMP. ° F	TEMP. RATE ° F/HOUR	ΔTEMP. ° F
Main steam	552	375	100	177
	375	330	270	45
	330	100	100	230
Head spray	552	375	100	177
	375	50) 15 sec	Step	325
	50	300)duration	Step	250
	300	100	100	200
Feedwater	552	100 5 cycles	100	452
	552	100) of 3 min	Step	452
	100	552) duration during 1st 2 hrs	Step	452
Recirculation suction	552	375	100	177
	375	330	270	45
	330	100	100	230
Recirculation discharge	552	375	100	177
	375	300	Step	75
	300	260	240	40
	260	100	100	260
Core spray	552	375	100	177
	375	330	270	45
	330	100	100	230
CRDHS return	50	50	0	0
Standby liquid control	400	375	100	25
	375	330	270	45
	330	100	100	230
Bottom drain	400	375	100	25
	375	330	270	45
	330	100	100	230

Note: Reactor pressure decreases from 1000 psig to 0 psig.

TABLE 3.9-24
(SHEET 13 OF 21)

(Applicable Thermal Transients Unbolt
123 Cycles Condition - Normal)

PIPELINE	INITIAL TEMP. ° F	FINAL TEMP. ° F	TEMP. RATE ° F/HOUR	ΔTEMP. ° F
Main steam	100	70	Step	30
Head spray	100	70	Step	30
Feedwater	100	70	Step	30
Recirculation suction	100	70	Step	30
Recirculation discharge	100	70	Step	30
Core spray	100	70	Step	30
CRDHS return	100	70	Step	30
Standby liquid control	100	70	Step	30
Bottom drain	100	70	Step	30

Note: Reactor pressure remains at 0 psig.

TABLE 3.9-24
(SHEET 14 OF 21)

(Applicable Thermal Transients Loss of Feedwater Pumps
MSIV Close 10 Cycles Condition - Upset

PIPELINE	INITIAL TEMP. ° F	FINAL TEMP. ° F	TEMP. RATE ° F/HOUR	ΔTEMP. ° F
Main steam	552	573	Step	21
	573	561	4320	12
	561	525	240	36
	525	573	480	48
	573	561	4320	12
	561	490	610	71
	490	573	625	83
	573	561	4320	12
	561	485	650	76
	485	400	100°/hr	85
	400	552	100°/hr	152
Head spray	561	40	Step	521
3 times cycle	40	525	Step	485
Feedwater	552	573	Step	21
	573	561	4320	12
	561	525	240	36
	525	573	480	48
	573	561	4320	12
	561	490	610	71
	490	573	625	83
	573	561	4320	12
	561	485	650	76
	485	400	100	85
	400	552	100	152
	552	100	Step	452
	100	250	Step	150
	250	420	340	170
Recirculation suction and discharge	528	525	20	3
	525	573	480	48
	573	561	4320	12
	561	490	610	71
	490	573	625	83
	573	561	4320	12
	561	485	650	76
	485	400	100°/hr	85
	400	552	100°/hr	152
	552	544	Step	8
	544	528	32	16

TABLE 3.9-24
(SHEET 15 OF 21)

(Applicable Thermal Transients Loss of Feedwater Pumps
MSIV Close 10 Cycles Condition - Upset)

PIPELINE	INITIAL TEMP. ° F	FINAL TEMP. ° F	TEMP. RATE ° F/HOUR	ΔTEMP. ° F
Core spray	528	40	Step	488
	40	528	Step	488
	528	40	Step	488
	40	528	Step	488
	528	40	Step	488
	40	528	Step	488
	528	400	100°/hr	128
	400	552	100°/hr	152
	552	544	Step	8
	544	528	32	16
CRDHS system	50	528	Step	478
	528	250	200°/hr	278
	250	400	100°/hr	150
	400	544	Step	144
	544	528	32	16
Standby liquid control	528	250	200	278
	250	400	100	150
	400	544	Step	144
	544	528	32	16
Bottom drain	528	250	200	278
	250	400	100	150
	400	544	Step	144
	544	528	32	16

Note: 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 and one relief valve opens and then closes as pressure decreases at rate of 100° F/hr.

TABLE 3.9-24
(SHEET 16 OF 21)

(Applicable Thermal Transients Reactor Overpressure
Delayed Scram 1 Cycle Condition - Emergency)

PIPELINE	INITIAL TEMP. ° F	FINAL TEMP. ° F	TEMP. RATE ° F/HOUR	ΔTEMP. ° F
Main steam	552	583		31
	583	538	5400	45
	538	400	100	138
	400	552	100	138
Head spray	552	583		31
	583	538	5400	45
	538	400	100	138
	400	552	100	138
Feedwater	420	276	8640	144
	276	100	704	176
	100	250	Step	150
	250	420	340	170
Recirculation suction	528	400	100	128
	400	552	100	152
	552	544	Step	8
	544	528	32	16
Recirculation discharge	528	400	100	128
	400	552	100	152
	552	544	Step	8
	544	528	32	16
Core spray	528	400	100	128
	400	552	100	152
	552	544	Step	8
	544	528	32	16
CRDHS return	50	50	0	0
Standby liquid control	528	250	200	278
	250	400	100	150
	400	544	Step	144
	544	528	32	16
Bottom drain	528	250	200	278
	250	400	100	150
	400	544	Step	144
	544	528	32	16

Note: Reactor pressure increases to 1350 psig. All relief valves and safety valves open. Pressure decreases to 240 psig.

TABLE 3.9-24
(SHEET 17 OF 21)

(Applicable Thermal Transients Single Safety or
Relief Valve Blowdown 8 Cycles Condition - Emergency)

PIPELINE	INITIAL TEMP. ° F	FINAL TEMP. ° F	TEMP. RATE ° F/HOUR	ΔTEMP. ° F
Main steam	552	375	1062	177
	375	100	100	275
Head spray	552	375	1062	177
	375	100	100	275
Feedwater	420	276	8640	144
	276	100	704	176
Recirculation suction	528	375	918	153
	375	100	100	275
Recirculation discharge	528	375	918	153
	375	100	100	275
Core spray	528	375	918	153
	375	100	100	275
CRDHS return	50	50	0	0
Standby liquid control	528	100	100	428
Bottom drain	528	100	100	428

Note: Reactor pressure decreases to 0 psig with one relief valve or safety valve open.

TABLE 3.9-24
(SHEET 18 OF 21)

(Applicable Thermal Transients Automatic Depressurization
1 Cycle Condition - Emergency)

PIPELINE	INITIAL TEMP. ° F	FINAL TEMP. ° F	TEMP. RATE ° F/HOUR	ΔTEMP. ° F
Main steam	552	375	3218	177
	375	281	300	94
Head spray	552	375	3218	177
	375	281	300	94
Feedwater	420	276	8640	144
	276	100	704	176
Recirculation suction	528	375	2780	153
	375	281	300	94
Recirculation discharge	528	375	2780	153
	375	281	300	94
Core spray	528	375	2780	153
	375	281	300	94
CRDHS return	50	50	0	0
Standby liquid control	528	375	2780	153
	375	281	300	94
	281	130	200	151
Bottom drain	528	375	2780	153
	375	281	300	94
	281	130	200	151

Note: Reactor pressure decreases with autodepressurization relief valves open to 35 psig.

TABLE 3.9-24
(SHEET 19 OF 21)

(Applicable Thermal Transients Improper Start of
Cold Recirculation Loop 1 Cycle Condition - Emergency)

PIPELINE	INITIAL TEMP. ° F	FINAL TEMP. ° F	TEMP. RATE ° F/HOUR	ΔTEMP. ° F
Main steam	552	552	0	0
Head spray	552	552	0	0
Feedwater	420	420	0	0
Recirculation suction	528	130) 26 sec	Step	398
	130	528) duration	Step	398
Recirculation discharge	528	528	0	0
Core spray	528	528	0	0
CRDHS return	50	50	0	0
Standby liquid control	528	528	0	0
Bottom drain	528	528	0	0

Note: Reactor pressure remains at 1000 psig.

TABLE 3.9-24
(SHEET 20 OF 21)

(Applicable Thermal Transients Sudden Pump Start
in Cold Loop 1 Cycle Condition - Emergency)

PIPELINE	INITIAL TEMP. ° F	FINAL TEMP. ° F	TEMP. RATE ° F/HOUR	ΔTEMP. ° F
Main steam	552	552	0	0
Head spray	552	552	0	0
Feedwater	420	420	0	0
Recirculation suction	528	528	0	0
Recirculation discharge	528	130) 34 sec	Step	398
	130	528) duration	Step	398
Core spray	528	528	0	0
CRDHS return	50	50	0	0
Standby liquid control	528	528	0	0
Bottom drain	528	528	0	0

Note: Reactor pressure remains at 1000 psig.

TABLE 3.9-24
(SHEET 21 OF 21)

(Applicable Thermal Transients Pipe Rupture and Blowdown 1 Cycle Condition - Faulted)

PIPELINE	INITIAL TEMP. ° F	FINAL TEMP. ° F	TEMP. RATE ° F/HOUR	ΔTEMP. ° F
Main steam	552	281	Step	271
Head spray	552	281	Step	271
Feedwater	420	281	Step	139
Recirculation suction	528	281	Step	247
Recirculation discharge	528	281	Step	247
Core spray	528	40	Step	488
	40	130	Step	90
CRDHS return	50	281	Step	231
Standby liquid control	528	281	Step	247
	281	223	Step	58
	223	50	Step	173
	50	130	Step	80

Note: Reactor pressure decreases from 1000 psig to 35 psig in 15 seconds.

3.10 SEISMIC QUALIFICATION OF SEISMIC CATEGORY I INSTRUMENTATION AND ELECTRICAL EQUIPMENT

In this section, the criteria, methods, and procedures used in the qualification of Seismic Category I instrumentation, electrical equipment, and their supports are described. Seismic Category I electrical power equipment is synonymous with Class 1E equipment as defined in IEEE 308-1971 and Seismic Category I electrical and electromechanical instrumentation is synonymous with Class 1E instrumentation as defined in IEEE 308-1974. All Seismic Category I instrumentation and electrical equipment utilized at LaSalle are identified in a controlled database. This database provides a list of the equipment contained in each system (including reactor protection, engineered safety feature, emergency power, and auxiliary safety-related systems), their location, manufacturer and model number, and references for the seismic and environmental qualification.

3.10.1 Seismic Qualification Criteria

The criteria used in the design and qualification of safety-related electrical control and instrumentation equipment are as follows:

a. The equipment must be capable of performing all safety-related functions during normal plant operation, anticipated transients, design basis accidents, and during post-accident operation while being subjected to, and after the cessation of the accelerations resulting from the postulated SSE, hydrodynamic and LOCA events.

b. Equipment possessing stationary (passive) safety functions (e.g., cable supports, instrument supports, and other components which do not perform a mechanical motion as part of their safety function) must maintain their structural integrity so that the operability of other safety-related equipment is not affected.

c. Equipment possessing nonstationary (active) safety functions (e.g., switches, motor-operators, and other equipment which perform a mechanical motion as part of their safety function) must be demonstrated operable during and after the postulated seismic event by analyses, testing, or a combination of both.

d. Cable tray and bus duct supports, regardless of function, must meet the requirements of Seismic Category I structures by dynamic analysis using the appropriate seismic response spectra. The analytical maximum values are obtained using the SRSS method for the stresses and reactions of all significant modes. Controlled cable tray loading was used throughout the design regardless of tray width or weight. Final cable tray

Hanger assessment was performed to assure that the resulting stresses from the actual loading do not exceed the allowable hanger stresses.

The specific criteria and the qualification method for each device depend on the particular location and use in a given system. Because many devices are used at different locations and for various applications, the devices are qualified for all possible usage applications; thus, assuring the capability of protective action initiation and proper operation.

To assure adequate qualification, the test and/or analysis requirements exceed the design requirements of the particular equipment. The complete range of hydrodynamic loads to which the equipment will be exposed, is included in the assessment of the impact of loads on the equipment. Input motions used in the qualification envelop the worst case combination of loads.

3.10.2 Methods and Procedures for Qualifying Electrical Equipment and Instrumentation

The qualification methods used to demonstrate that electrical equipment and instrumentation have the capability to perform their required function during and after the postulated safe shutdown earthquake are:

a. Dynamic analysis

b. Static analysis

c. Testing

d. Combination of test and analysis

The details of each method are described in the following subsections.

3.10.2.1 Dynamic Analysis

Flexible safety-related structures are typically qualified by dynamic analysis using response spectra in the three directions. If appropriate, time histories are also used in the analysis. The analysis is performed using a suitable finite element structural analysis program such as SAP. The structure is modeled for the program along with the properties of the structural members and the masses of the device.

The system stiffness and mass matrices are generated using standard techniques. A seismic analysis is performed using the following equations of motion and procedure to uncouple these equations:

The equations of motion in matrix form are as follows:

$$M\left(\ddot{X} + \ddot{Y}\right) + C\dot{\overline{X}} = KX = 0 \qquad (3.10\text{-}1)$$

where:

M	=	mass matrix, n x n (this includes the hydrodynamic mass),
X	=	column vector of displacement relative to ground (n x l),
C	=	damping matrix (n x n),
K	=	stiffness matrix (n x n),
$\ddot{Y}$	=	column vector of ground accelerations (n x l),
.	=	first derivative with respect to time, and
..	=	second derivative with respect to time.

It should be noted that where equipment contains fluid, a hydrodynamic mass coupling exists between real structural masses. This hydrodynamic mass appears as diagonal and off-diagonal terms in the mass matrix. The overall system stiffness matrix K is determined by either the matrix force method or the matrix displacement method. The resulting stiffness matrix is similar.

Removing the driving-point acceleration vector to the right side of equation 3.10-1, the equation reduces to the classical form:

$$\mathrm{M}\ddot{\mathrm{X}} + \mathrm{C}\dot{\overline{\mathrm{X}}} + \mathrm{KX} = -\mathrm{M}\ddot{\mathrm{Y}} \qquad (3.10\text{-}2)$$

In order to uncouple equation 3.10-2, we set:

$$\mathrm{X} = \Phi\, \mathrm{q} \qquad (3.10\text{-}3)$$

Equation 3.10-2 then becomes

$$\mathrm{M}\,\Phi\,\mathrm{q} + \mathrm{C}\,\Phi\,\mathrm{q} + \mathrm{K}\,\Phi\,\mathrm{q} = -\mathrm{M}\,\ddot{\mathrm{Y}} \qquad (3.10\text{-}4)$$

Premultiplying Equation 3.10-4 by ΦT, the transpose of Φ, the coordinate transformation described in Equation 3.10-4 is performed such that Φ is defined by the following orthogonality conditions:

$$\Phi^T M \Phi = I \qquad (3.10\text{-}5)$$

$$\Phi^T K \Phi = \left[\diagdown \omega^2 \searrow \right] \qquad (3.10\text{-}6)$$

Where I is an identity matrix (nxn) and $\left[\diagdown \omega^2 \searrow \right]$ is a diagonal matrix of the eigenvalues. Then 3.10-4 becomes

$$\Phi^T M \Phi \ddot{q} + \Phi^T C \Phi \dot{q} + \Phi^T K \Phi q = -\Phi^T \ddot{M} Y \qquad (3.10\text{-}7)$$

$$\ddot{q} + \Phi^T C \Phi q + \diagdown \omega^2 \searrow q = -\Phi^T \ddot{M} Y \qquad (3.10\text{-}8)$$

The above procedure for uncoupling the equation of motion by using the modal matrix of the undamped system assumes that damping in the system is small. It will further be assumed that the damping matrix C is such that $\Phi^T C \Phi$ is a diagonal matrix. The elements of this diagonal-matrix are the modal damping values.

With the above assumptions, Equation 3.10-8 may be written in the following uncoupled form:

$$\ddot{q}_i + 2 \beta_i \omega \dot{q}_{i} + \omega_i q_i = S_i \ddot{U}_g \qquad (3.10\text{-}9)$$

$$i = 1,2, \text{---} n$$

where:

$$X_i = \begin{bmatrix} x_{li} \\ x_{2i} \\ \bullet \\ \bullet \\ \bullet \\ \bullet \\ X_{ni} \end{bmatrix} \qquad \Phi_i = \begin{bmatrix} \Phi_{li} \\ \Phi_{2i} \\ \bullet \\ \bullet \\ \bullet \\ \bullet \\ \Phi_{ni} \end{bmatrix}$$

The maximum physical displacement for each mass is then taken to be the square root of the sums of the squares of each of the maximum displacement responses for each mode, i.e.,

$$(X)\max = \left[\sum_{j=1}^{n} x_{ij}^{2}\right]^{1/2}, i = 1,2,\ldots m$$

where: (X) maximum is the column vector of maximum displacements.

Similarly, the maximum load response for the i^{th} mode is found from

$$L_{ji} = \beta_j X_i$$

$$L_{ji} = \begin{bmatrix} L_{1i} \\ L_{2i} \\ \bullet \\ \bullet \\ \bullet \\ L_{mi} \end{bmatrix}$$

where:

β_j is the stress matrix for element j, j=l, ...m,

m = total number of elements,

β_i = damping ratio for the i^{th} mode expressed as percent of critical damping,

ω_i = i^{th} natural angular frequency of the system,

S_i = modal participation factor the i^{th} mode - Φ_i^t MD,

$\ddot{U}_g$ = ground or floor acceleration time history,

Φ_i^t = transpose of the I th mode shape, and

D = earthquake direction vector.

The response is calculated using the response spectra specified for the location of the input to the analytical model. The analytical procedure is described briefly in the following paragraphs.

The system of one-degree-of-freedom equations represented by Equation 3.10-8 or 3.10-9 can be solved by the response spectrum method. With this method, the maximum modal response for each natural frequency of interest is found from the applicable response spectra. Response spectrum curves are essentially plots of the maximum responses of single-degree-of-freedom systems described by Equation 3.10-9 with Si = 1.0 as a function of their natural frequencies.

Having found the maximum modal displacements q, i = 1...m, the maximum physical displacement for the i^{th} mode is given by:

$$X_i = \Phi_i S_i q_i$$

The maximum load response is taken to be the square root of the sums of the squares of each of the maximum responses for each mode, i.e.

$$(L_j)\max = \left[\sum_{i=1}^{n} L^2{}_{ji}\right]^{1/2} : j = 1,2,...m$$

Where (L) max is the column vector of maximum loads.

The accelerations for each mode are determined by multiplying the displacements vector for that mode (X_i) by the natural frequency ($_i$) of that mode.

$$A_i = X_i \omega_i^2$$

The maximum accelerations are then determined by

$$(A)\max = \left[\sum_{i=1}^{n} A_i^2\right]^{1/2}$$

When modes are separated by less than 10% of each other, the load and actuation responses of these modes are added absolutely.

The structure is also analyzed for static loads such as weight and the resulting total stresses are compared with the allowables as discussed in Section 3.9.

3.10.2.2 Static Analysis

When the structure is found to be rigid, it is analyzed for the maximum floor accelerations of the dynamic loads, the static loads such as weight, and all other applicable loads. If the structure is simple such as a pipe stand, it is analyzed by simple analytical techniques and using a hand calculation. If the structure is complex such as a control panel, it is analyzed by finite element methods using one of the computer programs described in Section 3.9. The stresses obtained from all the loads are compared with the allowables of the material.

3.10.2.3 Testing

Safety-related equipment having primary active functions are almost always qualified by testing. The exceptions for this are the components which can not be tested by he available test facilities and/or the components whose normal operating loads are much higher than the postulated dynamic loads. Examples of this exception are the emergency diesel engine/generator and large motors.

The test method appropriate to the component to be qualified is selected from the various test methods recommended by IEEE 344-1975. The most commonly used method is the biaxial phase incoherent random input motion. The input motion is typically adjusted at 1/3 octave intervals to envelop the required response spectra at the corresponding critical damping factor.

Single frequency and/or single axis tests are not widely used for qualification. In cases where existing test results from high frequency and/or single axis tests are used for qualification, suitable margin has been maintained between the required input motion and the test input motion to account for any modal participation and cross coupling.

Components which were originally qualified to IEEE 344-1971 requirements were reevaluated to the requirements of IEEE 344-1975. Devices that did not meet the intent of IEEE 344-1975 were tested.

For equipment with original design requirements for seismic qualifications to IEEE 344-1971, "like-for-like" replacements may be purchased to the original design requirements. New and replacement (except "like-for-like" replacements) components must meet the requirements of IEEE 344-1975 and Regulatory Guide 1.100, Revision 1, "Seismic Qualification of Electric Equipment for Nuclear Power Plants".

3.10.2.4 Combination of Test and Analysis

When safety-related components are located on flexible structures other than the building floor or wall, the support structures are analyzed to obtain the required input motion for the components as discussed in 3.10.2.1. This input motion is used for testing the components separately as discussed in 3.10.2.3.

3.10.3 Methods and Procedures for Analysis or Testing of Supports of Electrical Equipment and Instrumentation

Supports for Seismic Category I electrical equipment and instrumentation such as battery racks, instrument racks, control consoles, cabinets, panels, cable tray, etc., are qualified by test and/or analysis as described in Subsection 3.10.2. Where possible, the supports were qualified by test with the equipment installed and operable. Otherwise, a dummy mass was used to simulate equipment mass effects and dynamic coupling to the supports. Testing was done under the following conditions:

a. Simulate actual loading conditions, including any operating loads.

b. Monitor the operability of the equipment during testing. Normal operation of the equipment during and after the test indicated that the supports were adequate. Any excessive deflection or deformation of equipment supports which precluded the operability of the equipment negated acceptance of that equipment.

c. Supports were inspected for structural integrity after the test. Any crack or permanent deformation was not accepted.

In the case of analysis, the stresses at all support points and in such parts as motor holddown and baseplate holddown bolts, support pads, pedestals, foundations, etc., were checked against the allowables of the applicable codes.

Figures 3.10-1 through 3.10-4 illustrate the four basic panel types used to support electrical equipment and instrumentation and show typical accelerometer locations used during testing.

3.10.4 Qualification Results

The analytical methods and/or testing used in the seismic qualification of electrical equipment and instrumentation are in accordance with the requirements of IEEE 344-1975.

The extensive documentation filed (test reports, analyses, etc.) indicate that the equipment meet or exceed the required design criteria. The seismic qualification ensures that the equipment as designed will maintain their structural and functional integrity and therefore, their capability to perform the required function during and after the design basis seismic event.

TABLE 3.10-2
(SHEET 1 OF 4)

SEISMIC QUALIFICATION TEST SUMMARY FOR LASALLE CLASS 1E CONTROL PANELS AND LOCAL PANELS

PANEL	DESCRIPTION	TYPE	CLASS 1E EQUIPMENT DESCRIPTION	COMMENTS
H13-P601	Reactor core cooling system	Benchboard	SBM & CR 2940 switches, GEMAC instruments	Seismic test on similar type panel
H13-P602	Reactor water cleanup & recirculation	Benchboard	SBM & CR 2940 switches, GEMAC instruments	Seismic test on similar type panel
H13-P603	Reactor control	Benchboard	Mode switch, range switches	Seismic test completed
H13-P608	Power range neutron monitor	5-bay instrument panel	APRM, ICPS, RBM, Flow Comp, Morgan Power Supply	Seismic test completed
H13-P609	Reactor protection system Division 1&2 logic	Vertical board	HFA & HMA relays, CR 105 contractor	Identical to tested H13-P611 panel (tested)
H13-P611	Reactor protection system Division 3&4 logic	Vertical board	HFA & HMA relays, CR 105 contractor	Seismic test completed.
H13-P612	FW & recirculation instrument	2-bay instrument panel	GEMAC instruments	Seismic test completed.
H13-P613	NSS process instruments	2-bay instrument panel	GEMAC instruments	Seismic test completed.
H13-P618	Division 2 RHR relays	Vertical board	HFA & HMA relays	Seismic test on similar type panel
H13-P621	Reactor core isolation cooling relays	Vertical board	HFA & HMA relays	Seismic test on similar type panel
H13-P622	Inboard isolation valve relays	Vertical board	HFA & HMA relays	Seismic test on similar type panel
H13-P623	Outboard isolation valve relays	Vertical board	HFA & HMA relays	Seismic test on similar type panel
H13-P624	Area radiation common monitor	Vertical board	HFA & HMA relays	Seismic test on similar type panel
H13-P625	HP core spray relays	Vertical board	HFA & HMA relays	Seismic test on similar type panel
H13-P628	ADS Channel A relay vertical board	Vertical board	HFA & HMA relays	Seismic test on similar type panel
H13-P629	Division 1 LPCS & RHR relays	Vertical board	HFA & HMA relays	Seismic test completed

TABLE 3.10-2
(SHEET 2 OF 4)

SEISMIC QUALIFICATION TEST SUMMARY FOR LASALLE CLASS 1E CONTROL PANELS AND LOCAL PANELS

PANEL	DESCRIPTION	TYPE	CLASS 1E EQUIPMENT DESCRIPTION	COMMENTS
H13-P631	ADS channel B relay vertical board	Vertical board	HFA & HMA relays	Seismic test on similar type panel
H13-P632	Division 1 leak detection	Vertical board	Scam temperature monitor, GEMAC instruments, timer, CR 2940 HMA	Seismic test completed
H13-P635	Division 1 radiation monitoring	Instrument panel	IRM, SRM, LPRM, trip auxiliary unit	Seismic test on similar type panel
H13-P636	Division 2 radiation monitoring	Instrument panel	IRM, SRM, LPRM, trip auxiliary unit	Seismic test on similar type panel
H13-P642	Division 2 leak detection	Vertical board	Scam temperature monitor, GEMAC instruments, timer, CR 2940, HMA	Identical to H13-P632 panel (tested)
H13-P654	MSIV leakage control Division 1	Vertical board	CR 2940, Agastat Relay, Eagle System timer	Seismic test on similar type panel
H13-P655	MSIV leakage control Division 2	Vertical board	CR 2940, Agastat Relay, Eagle System timer	Seismic test on similar type panel
H22-P001	LPCS	Local Panel	Pressure switches	Seismic test completed
H22-P002	Reactor water cleanup	Local Panel	Pressure transmitters	Seismic test completed
H22-P004	Reactor vessel level and pressure –A	Local Panel	Pressure switches, level indicator/ transmitter	Seismic test on similar type panel
H22-P005	Reactor Vessel level and pressure – C	Local Panel	Pressure switches, level indicator/ transmitter	Seismic test on similar type panel
H22-P006	Recirculation A	Local Panel	Pressure Transmitter	Seismic test on a similar type panel
H22-P009	Jet pump B	Local Panel	Pressure Transmitter	Seismic test completed
H22-P010	Jet pump A	Local Panel	Pressure Transmitter	Identical to H22-P009 panel (tested)
H22-P015	Main steam flow A	Local Panel	Pressure Transmitter	Identical to H22-P025 panel (tested)
H22-P017	RCIC panel A	Local Panel	Pressure Transmitter, pressure switches	Seismic test on similar type panel
H22-P018	RHR panel A	Local Panel	Pressure switches	Seismic test on similar type panel
H22-P021	RHR panel B	Local Panel	Pressure switches	Seismic test on similar type panel

TABLE 3.10-2
(SHEET 3 OF 4)

SEISMIC QUALIFICATION TEST SUMMARY FOR LASALLE CLASS 1E CONTROL PANELS AND LOCAL PANELS

PANEL	DESCRIPTION	TYPE	CLASS 1E EQUIPMENT DESCRIPTION	COMMENTS
H22-P022	Recirculation B	Local panel	Pressure transmitter	Seismic test on similar type panel
H22-P024	HPCS	Local panel	Pressure transmitter, pressure switches	Seismic test on similar type panel
H22-P025	Main steam flow B	Local panel	Pressure switches	Seismic test completed
H22-P026	Reactor vessel level and pressure D	Local panel	Pressure switches, pressure transmitters	Seismic test on similar type panel
H22-P027	Reactor vessel level and pressure B	Local panel	Pressure switches, pressure transmitters	Seismic test on similar type panel
H22-P028	HPCS diesel-generator protection relay	Local panel	Relay	Seismic test to complete
H22-P029	RCIC panel B	Local panel	Pressure switches	Seismic test on similar type panel
H22-P030	SRM & IRM preamp A	NEMA 12 enclosures	SRM & IRM Preamplifiers	Seismic test completed
H22-P031	SRM & IURM preamp B	NEMA 12 enclosures	SRM & IRM Preamplifiers	Identical to H22-P030 enclosure (tested)
H22-P032	SRM & IRM Preamp C	NEMA 12 enclosures	SRM & IRM Preamplifiers	Identical to H22-P030 enclosure (tested)
H22-P033	SRM & IRM Preamp D	NEMA 12 enclosures	SRM & IRM Preamplifiers	Identical to H22-P030 enclosure (tested)
H22-P073	MSIV Leakage control Division 1	Local panel	Pressure transmitters	Seismic test on similar type panel
H22-P074	MSIV Leakage control Division 2	Local panel	Pressure transmitters	Seismic test on similar type panel

TABLE 3.10-2
(SHEET 4 OF 4)

SEISMIC QUALIFICATION TEST SUMMARY FOR LASALLE CLASS 1E CONTROL PANELS AND LOCAL PANELS

NOTES

Seismic tests on essential C&I panels fall into the following categories:

1. Panels not tested due to size limitations, e.g., H13-P601.
2. Seismic test completed – tests are run on panels essentially identical but possibly built for a different plant.
3. Tests on identical panels – when two panels are exact duplicates of one another. Test are run on only one panel (e.g., H13-P609) and H13-P0611 are identical; only H13-P0611 was tested).
4. Tests on similar panels – when panel size and configurations are similar but not necessarily identical, test results for a similar panel are used.
5. All panels (H12 and H22) are qualified to accelerations of 1.5g in the horizontal direction and 0.5g in the vertical direction.

TABLE 3.10-3

STANDARD ENCLOSURES

CURVE	ENCLOSURE	WIDTH	DEPTH	MODE OF FAILURE
C1	Instrument rack	24 in.	24 in.	
	Instrument rack	24 in.	30 in.	
	Vertical board	24 in.	24 in.	Side to side
	Vertical board	24 in.	30 in.	
	Benchboard	24 in.	48 in.	
	Benchboard	24 in.	54 in.	
C2	Instrument rack	30 in.	30 in.	
	Instrument rack	30 in.	24 in.	
	Instrument rack	48 in.	24 in.	
	Instrument rack	60 in.	24 in.	
	Instrument rack	72 in.	24 in.	Front to back
	Instrument rack	96 in.	24 in.	or
	Vertical board	36 in.	24 in.	Back to front
	Vertical board	48 in.	24 in.	
	Vertical board	60 in.	24 in.	
	Vertical board	72 in.	24 in.	
	Vertical board	96 in.	24 in.	
C3	Instrument rack	48 in.	30 in.	
	Instrument rack	60 in.	30 in.	
	Instrument rack	72 in.	30 in.	
	Instrument rack	96 in.	30 in.	Front to back
	Vertical board	36 in.	30 in.	or
	Vertical board	48 in.	30 in.	Back to front
	Vertical board	60 in.	30 in.	
	Vertical board	72 in.	30 in.	
	Vertical board	96 in.	30 in.	
C4	Console	96 in.	42 in.	Back to front
C5	Benchboard	48 in.	54 in.	Side to side
	Benchboard	48 in.	48 in.	
C6	Benchboard	72 in.	48 in.	
	Benchboard	96 in.	48 in.	Front to back
	Console	96 in.	48 in.	
C7	Benchboard	72 in.	54 in.	Back to front
	Benchboard	96 in.	54 in.	

TABLE 3.10-4
(SHEET 1 OF 3)

SEISMIC DESIGN VERIFICATION DATA SHEET

Cabinet Name: Area Radiation Monitor, 236 x 400

Applied horizontal acceleration	1.5g
Applied vertical acceleration	0.5g
Tension stress (maximum safe)	28,000 psi
Shear stress (maximum safe)	21,000 psi
Weight of cabinet	675 lb
Number of mounting bolts	4
Height of center of gravity	48 in.
Maximum allowable weight per bolt (From Curve No. C1 on page 8 of Seismic Design Guide, 225A4582)	830 lb/bolt
Maximum allowable cabinet weight 830 1b/bolt *4 bolts =	3,320 lb
Factor of Safety = Maximum Allowable Weight / Weight =	4.9

Cabinet Name: TIP Control, 236 x 401 (913)

Applied horizontal acceleration	1.5g
Applied vertical acceleration	0.5g
Tension stress (maximum safe)	28,000 psi
Shear stress (maximum safe)	21,000 psi
Weight of cabinet	755 lb
Number of mounting bolts	8
Height of center of gravity	50 in.
Maximum allowable weight per bolt (From Curve No. C3 on page 8 of Seismic Design Guide, 225A4582)	1,110 1b

TABLE 3.10-4
(SHEET 2 OF 3)

SEISMIC DESIGN VERIFICATION DATA SHEET

Maximum allowable cabinet weight 1,110 1b/bolt * 8 bolts =	8,880 lb
Factor of Safety = Maximum Allowable Weight / Weight =	11.7

Cabinet Name: Startup Neutron Monitor, 236 x 402 (936)

Applied horizontal acceleration	1.5g
Applied vertical acceleration	0.5g
Tension stress (maximum safe)	28,000 psi
Shear stress (maximum safe)	21,000 psi
Weight of cabinet	1,910 1b
Number of mounting bolts	12
Height of center of gravity	50 in.
Maximum allowable weight per bolt (From Curve No. C3 on page 8 of Seismic Design Guide, 225A4582)	1,110 lb/bolt
Maximum allowable cabinet weight 1,110 1b/bolt * 12 bolts =	13,320 lb
Factor of Safety = Maximum Allowable Weight / Weight =	11.9

Cabinet Name: Power Range Monitor, 236 x 403 (937)

Applied horizontal acceleration	1.5g
Applied vertical acceleration	0.5g
Tension stress (maximum safe)	28,000 psi
Shear stress (maximum safe)	21,000 psi
Weight of cabinet	4,345 lb
Number of mounting bolts	40
Height of center of gravity	46 in.

TABLE 3.10-4
(SHEET 3 OF 3)

SEISMIC DESIGN VERIFICATION DATA SHEET

Maximum allowable weight per bolt (From Curve No. C3 on page 8 of Seismic Design Guide, 225A4582)	1,210 lb/bolt
Maximum allowable cabinet weight 1,210 1b/bolt * 40 bolts =	48,400 lb
Factor of Safety = $\frac{\text{Maximum Allowable Weight}}{\text{Weight}}$ =	11.1

Cabinet Name: Rod Position Information System, 236 x 404 (927)

Applied horizontal acceleration	1.5g
Applied vertical acceleration	0.5g
Tension stress (maximum safe)	28,000 psi
Shear stress (maximum safe)	21,000 psi
Weight of cabinet	2,500 lb
Number of mounting bolts	20
Height of center of gravity	45 in.
Maximum allowable weight per bolt (From Curve No. C3 on page 8 of Seismic Design Guide, 225A4582)	1,225 lb/bolt
Maximum allowable cabinet weight 1,225 lb/bolt * 20 bolts =	24,500 lb
Factor of Safety = $\frac{\text{Maximum Allowable Weight}}{\text{Weight}}$ =	9.8

Conclusion

Review of the factor of safety of each standard cabinet indicates that the mounting bolts of each cabinet are capable of withstanding seismic disturbances as specified in the Seismic Design Guide.

3.11 ENVIRONMENTAL DESIGN OF MECHANICAL AND ELECTRICAL EQUIPMENT

This section provides the environmental conditions and design bases of the safety-related equipment and components and describes the qualification methods which ensure their acceptable performance in normal, abnormal, and postulated accident environmental conditions.

3.11.1 Equipment Identification and Environmental Conditions

The safety-related electrical equipment, components, and the active mechanical equipment utilized in the various systems of the plant that are required to function during and subsequent to any of the design-basis accidents are identified in the LaSalle controlled computer database. The specific information presented in this database is as follows:

a. Equipment Number: Provides the specific plant numbers of the equipment (for ease of reference and correlation with other documents and drawings).

b. Equipment Name: Provides the generic name of equipment.

c. Equipment Manufacturer: Identifies the manufacturer or vendor of the equipment.

d. Model Number: Provides the equipment catalog number.

e. EZ = Environmental Zone: Identifies the most adverse environment to which the equipment is exposed. The environmental zones (EZ) are designated as LHX, LNC, LCX for harsh, normal, and controlled exposure conditions for the particular equipment affected by the most limiting accident to which it is exposed according to definitions provided in Subsections 3.11.1.1, 3.11.1.2, and 3.11.1.3.

f. Location: Provides the building and elevation where the equipment is located. If the equipment is part of an equipment system, that system is also identified.

g. Environmental Reference: Provides the qualification reference for environmental parameters.

h. <u>Seismic References</u>: Provides the qualification reference for dynamic loads, including seismic and hydrodynamic loads if any.

i. <u>Comments</u>: This column identifies the status of qualification, replacement of equipment if under consideration, and any other relevant information.

The plant areas containing safety-related electrical equipment are divided into three types of zones based on the environmental conditions that occur within these areas as a result of various plant events. These zone classifications are the harsh environment, normal environment, and controlled environment and are further described and delineated in the following subsections.

3.11.1.1 <u>Harsh Environment (LOCA/HELB)</u>

Harsh environments are defined as those areas of the plant which experience environmental conditions resulting from a postulated loss-of-coolant accident (LOCA) inside containment, a high-energy line break (HELB), or an instrument line break outside containment.

There are ten areas of the plant that are affected by harsh environments. Each of these areas has been assigned a zone number unique to the environment contained therein as shown in Figure 3.11-1. Figures 3.11-2, 3.11-3, 3.11-4, and 3.11-6 define subzones within Zones H2, H4, H5, and H10 respectively. A general description of these zones is provided in the following:

a. Zone H1 - inside the reactor pressure vessel (RPV).

b. Zone H2 - inside primary containment (drywell).

c. Zone H3 - inside primary containment (wetwell).

d. Zone H4 - reactor building general area and refuel floor, excluding ECCS equipment cubicles; HELB local areas as defined below; and the lower two floors of the reactor building.

e. Zone H5 - HELB local areas, including main steam pipe tunnel, RCIC pipe chase, LPCS/RCIC equipment cubicle, RWCU equipment areas (pump, valve, and heat exchanger rooms including the corridor west of the heat exchanger valve rooms at elevation 786' and pipe chases), and lower two floor elevations of the reactor building outside the ECCS equipment cubicles.

f. Zone H6 - reactor building ECCS equipment cubicles for the HPCS and RHR systems.

g. Zone H7 - turbine building general areas outside the EQ Zone H8 (condenser cavity, heater bays, etc.) and DG Corridor on elevation 710’-6”.

h. Zone H8 - turbine building including areas such as the condenser cavity, heater bays and other limited access areas.

i. Zone H9 - auxiliary building, in the following areas:

1. Within 10 feet of the SGTS exhaust pipe, located on floor elevation 786 feet 6 inches and between columns/rows J and L and 15 and 16.

2. Within 15 feet of the charcoal filter beds in the emergency make-up filter units (OVC01SA,B), on floor elevation 802 feet, and between columns/rows N and R and around 13 and 17.

3. Within the plant discharge stack.

j. Zone H10 - auxiliary building, within 20 feet of the VQ system supply/return ducts, on floor elevation 786'6".

Each of these zones experiences normal service conditions and may experience accident or abnormal environmental conditions. Normal service conditions are defined as those that exist during routine plant operations. In some cases, these service conditions may extend over a range of values which could be expected to occur at various times during the operating life of the plant. Abnormal or bounding accident conditions are defined as those conditions that can occur as the result of postulated failures or accidents, such as LOCA and HELB. Because the bounding conditions usually present the most difficult challenge to equipment operation, they are discussed first. Note that the abnormal radiation exposure value represents the summation of the total integrated normal dose and the accident dose. The accident dose contribution is calculated from a source term consistent with Appendix B of NUREG-0588, except that the time dynamics of fluid distribution is accounted for and a 10% margin was added to the result in accordance with IEEE 323-1974, Section 6.3.1.5, to establish the required testing value.

It must be emphasized that in each case the abnormal and normal service environmental conditions represent conservative selections chosen to bound any other conditions that may occur in these zones. In certain instances, a more refined or more detailed analysis may be performed for specific components or groups of components to establish more realistic and representative environmental conditions than those ultra-conservative conditions specified for the environmental zone. Therefore, the environments enumerated on the Qualifications Summary Tables may not in all cases agree with the general zone environmental conditions identified for the equipment; however, unique calculations have been performed to justify these conditions and are part of the environmental qualification records. Future application of this approach may be utilized to obtain more realistic environmental conditions for evaluating or testing equipment that is currently not fully qualified or being subjected to additional evaluation.

3.11.1.1.1 Selection of Bounding Conditions

Zone H1

For this zone, the operating conditions inside the RPV, excluding radiation exposure, present the most limiting environmental conditions for equipment qualification. The radiation conditions represent the total integrated normal plus accident dose. These conditions are summarized in Table 3.11-3.

Zone H2

The bounding conditions for the drywell are determined from an envelope of the conditions resulting from a LOCA or HELB inside the containment. These conditions are presented in Table 3.11-4.

a. Pressure

As discussed in Section 6.2, the maximum pressure response inside the containment occurs as a result of the postulated instantaneous guillotine break of the reactor recirculation line (LOCA). This peak calculated pressure is 39.9 psig (Table 6.2-8a). The maximum pressure chosen for the bounding condition (45 psig) represents the containment design pressure and thus, is very conservative. Main steamline, feedwater, instrument line, and other high-energy line breaks were evaluated, and the resultant peak calculated pressures were bounded by the recirculation break case. The negative pressure (-2 psig) represents an ultraconservative value that exceeds that calculated for any accident condition. Any negative pressure would be of very short duration and should only be considered as a transient rather than a steady-state condition.

b. Temperature

As discussed in the analysis results presented in Section 6.2, the peak calculated drywell temperature occurs for the main steamline break case, and is 330° F (Figure 6.2-9). The bounding condition selection of 340° F represents a conservative value for equipment qualification.

c. Relative Humidity

An all-steam environment, followed by 100% relative humidity, represents a very conservative approach. However, certain areas will experience 100% relative humidity due to a reactor water cleanup system HELB. See Figure 3.11-1 and Table 3.11-6 for a listing of these areas.

d. Duration

The durations indicated for the bounding conditions were based on generic information supplied by General Electric in their interface documents. The General Electric specific analysis presented in Section 6.2 is bounded by this generic information. Therefore, the plant generic information is utilized rather than the design profiles presented in NUREG-0588 Appendix C.

e. Radiation

The integrated dose is based on a 40-year lifetime expected operating dose value plus an accident value computed on the basis of NUREG-0588 criteria plus approximately a 10% margin. An equivalent gamma dose for anticipated neutron fluence has been included in this dose, and the expected contribution from beta is also included.

f. Submergence

The maximum flood level in the drywell following a LOCA is determined by the projection of the suppression vent downcomers above the drywell floor. These vents project 6 inches above the floor elevation of 736 feet 7-1/2 inches. No Class 1E equipment is located within this submergence zone.

Zone H3

The bounding conditions for the wetwell (suppression chamber) are presented in Table 3.11-5. The analytical basis for these conditions is also presented in Section 6.2.

a. Pressure

The peak calculated suppression chamber pressure, 27.9 psig, as presented in Table 6.2-5a and Figure 6.2-5a is bounded by the selected value of 45 psig.

b. Temperature

The peak calculated suppression chamber temperature occurs on the long-term basis, rather than in the first few moments following a LOCA. The 200° F value selected as bounding in Tables 3.11-5 can be seen to be conservative when compared to the analytical results presented in Figure 6.2-7a.

c. Relative Humidity

The basis for the assumption of 100% relative humidity above the pool is based on examination of the normal and abnormal pool vapor conditions and requires no further comment.

d. Duration

The durations presented are based on the worst case analytical results as presented in Figures 6.2-5a and 6.2-7a.

e. Radiation

The same basis as that of Zone H2 applies here.

f. Submergence

The maximum normal pool level is elevation 700 feet 2 inches. During a LOCA or in various transient conditions, this level can increase to elevation 706 feet 9 inches; however, this level increase has no effect on any wetwell systems or components.

Zone H4

The maximum conditions in the general reactor building area result from HELB events in equipment cubicles venting to the general floor area. The most limiting case, selected from the analytical evaluation of all the HELB cases, was chosen for these bounding conditions. These conditions are listed in Table 3.11-6.

a. Pressure

No significant pressurization of the reactor building volume occurs during any postulated event. Therefore, atmospheric pressure was chosen as the upper bound.

b. Temperature

The temperatures associated with this zone actually occur at only one location in the reactor building, but were selected as the design basis since they bound the other locations.

c. Relative Humidity

In general, the 95% relative humidity represents a conservative limit above and beyond that which was actually determined to occur by analysis. However, certain areas experience 100% relative humidity due to a revised HELB analysis. See Figure 3.11-1 and Table 3.11-6 for these areas.

d. Duration

The durations were based on actual time histories determined from the bounding analysis.

e. Radiation

The same basis as that of Zone H2 applies here.

f. Submergence

No significant flooding of the reactor building occurs during HELB conditions. The drainage and sump systems prevent any water buildup, and the leak detection systems act to isolate the offending line by closing the associated isolation valves. (See Subsection 3.11.1.4 for additional discussion of flooding and submergence.)

Zone H5

The conditions for local areas subjected to HELB's are based on bounding conditions selected from GE interface documents compared with the analytical results in the specific line break cases. Table 3.11-7 lists these conditions.

a. Pressure

The initial transient pressure conditions were selected from the bounding analysis and represent the peak pressure in the main steam tunnel. Subsequent pressure values bound the analytical results.

b. Temperature

Although analysis shows transient peak pressures above atmospheric conditions, these occur in the range of a few seconds and, therefore, are not considered to have sufficient time to influence the ambient temperature in the local areas subject to an HELB. These areas quickly depressurize to atmospheric pressure. Therefore, the selection of the saturation temperature of 212° F for the first 6 hours represents a more conservative and bounding value than that obtained from the HELB analysis.

c. Relative Humidity

The saturated steam environment for 6 hours, 100% relative humidity for another 6 hours, and a 90% relative humidity for 100 days bound all analytical results.

d. Duration

As discussed in the previous sections, the durations bound the analytical results.

e. Radiation

The same basis as that of Zone H2 applies here.

f. Submergence

The same basis as that of Zone H4 applies here.

Zone H6

This zone encompasses the areas inside the ECCS equipment cubicles, except for the LPCS/RCIC cubicle, in the basement (floor elevation 673 feet 4 inches) and RHR heat exchanger rooms (floor elevation 710 feet 6 inches) of the reactor building. These areas are not subject to adverse pressure, temperature, humidity, or flooding conditions as the result of a LOCA or HELB. Each of these areas is provided with a dedicated, safety-related cubicle cooler which ensures that a controlled nonadverse environment is maintained at all times the equipment is required to operate. The maximum conditions were based on the normal operating and testing of the ECCS equipment, as well as operation during design-basis accidents.

Thus, for all conditions other than radiation, these areas would be considered to be controlled zones rather than harsh zones. However, under the LOCA conditions, the radiation dose in these areas is increased to a value which warrants the classification of the area as a harsh zone. Therefore, the abnormal and service conditions provided in Tables 3.11-8 and 3.11-17, respectively, are identical except for the radiation dose. The basis for these conditions is adequately described in the tables, with the exception of radiation. The basis for the radiation conditions is the same as that provided for Zone H4 (item e).

Zone H7

The turbine building general areas and DG corridor on elevation 710'-6" are subject to an HELB, although they contain very little safety-related equipment. Therefore, the bounding environment conditions (Table 3.11-9) are selected to be the same as that tabulated for Zone H5, with the exception of the radiation and submergence values. The basis for the bounding pressure, temperature, and humidity conditions and their duration remain the same as Zone H5. The radiation levels are based on a bounding calculation for the most severe radiation environment in that zone and do not represent in all cases the actual expected dose which is much lower. The basis for flooding is described in Subsection 3.11.1.4. No safety-related equipment is submerged as a result of the postulated exterior or interior floods.

Safety-related electric equipment located in EQ Zone H7 required to mitigate the consequences of the postulated HELB's are environmentally qualified for the postulated HELB environment.

The consequences of a LOCA do not result in adverse environmental conditions in the turbine building and receive no special consideration since they are bounded by the HELB or normal conditions.

Zone H8

The same basis as Zone H7 applies here also. Bounding environmental conditions are shown in Table 3.11-10.

Zone H9

The areas associated with this zone are not subject to adverse pressure, temperature, humidity, or flooding conditions as a result of an HELB or LOCA. The service conditions for these parameters, therefore, also form the bounding conditions for equipment qualification (Tables 3.11-11 and 3.11-20). However, the LOCA condition results in adverse radiation doses in these areas, since the equipment located in Zone H9 is involved with the processing of postaccident HVAC flows. Again, the basis for the radiation dose is the same as that described for Zone H4 (item e), except that an area-specific calculation has been performed that includes the postaccident HVAC system equipment in these areas.

Zone H10

The areas associated with this zone are not subject to adverse pressure or radiation conditions as a result of a LOCA within containment. The service conditions for these parameters also form the bounding conditions for equipment qualification (Tables 3.11-28 and 3.11-29). However, the LOCA condition results in adverse temperatures and humidity in these areas. The basis for the temperature and humidity (in Table 3.11-28) is found in an area-specific calculation perform to analyze the effects of a LOCA during Primary Containment Purging.

3.11.1.1.2 Service Conditions

The normal service conditions of the plant represent those conditions actually expected to occur during the normal operation of the plant.

Zone H1

The range of normal operating conditions inside the RPV are shown in Table 3.11-12.

Zone H2

Table 3.11-13 presents the range of normal operating conditions that are expected to occur inside the drywell.

Zone H3

Table 3.11-14 presents the range of normal operating conditions in the wetwell.

Zone H4

Table 3.11-15 presents the range of normal operating conditions expected in the reactor building general area.

Zone H5

Table 3.11-16 presents the bounding range of normal operating conditions expected in the HELB areas. Highest temperatures occur in the Subzone H5B, the RCIC pipe tunnel. The highest radiation exposure occurs in Subzone H5C, the main steam tunnel.

Zone H6

Table 3.11-17 presents the range of normal operating conditions expected in the RHR and HPCS equipment cubicles located in the two lower floors of the reactor building. It should be noted that the specified service environment is equivalent to the bounding environment specified in Table 3.11-8, except for the radiation dose.

Zone H7

Table 3.11-18 presents the range of normal operating conditions expected in the designated areas of the turbine building, as shown in Figure 3.11-1. In the case of radiation values, a bounding value was applied within the entire zone even though actual doses would be considerably lower.

Zone H8

Table 3.11-19 presents the range of normal operating conditions expected to occur in the designated areas of the turbine building, as shown in Figure 3.11-1. Again, the radiation values are bounding cases, not actual doses.

Zone H9

Table 3.11-20 presents the range of normal operating conditions expected to occur in the special areas of the auxiliary building as shown in Figure 3.11-1.

Zone H10

Table 3.11-29 presents the range of normal operating conditions expected to occur in the specific areas of the auxiliary building as shown in Figure 3.11-1 and Figure 3.11-6.

3.11.1.2 Normal Environment (Non-Safety-Related HVAC Systems)

Normal environments are those areas maintained at room, or ambient outdoor conditions by the non-safety-related HVAC system during routine plant operations. Abnormal conditions may occur in some of these areas due to loss of the HVAC system. The areas include auxiliary building and radwaste building.

The auxiliary building areas considered have been assigned a zone number and boundaries as defined in Figure 3.11-5. The bases for selecting the bounding conditions for each zone are as follows:

Zone N1 includes areas that are served by the air conditioning system to maintain environmental conditions suitable for personnel comfort and equipment operation. Since these areas do not contain any safety-related equipment and are occupied during normal plant operation, the environmental conditions resulting from loss of the air conditioning system were not evaluated. The normal environmental conditions are listed in Table 3.11-21.

Zone N2 consists of areas that are normally provided with ventilation to maintain the temperature range in conformance with the equipment temperature ratings. The time-dependent histogram of pressure, temperature, and relative humidity were determined using the historical engineering weather data (compiled by the Engineering Meteorology Section of the U.S. Air Force, Manual No. AFM 88-29, dated July 1, 1978) applicable to the plant site. The environmental conditions in the areas where safety-related equipment is installed were selected for the analysis since only the Class 1E equipment needs the qualification. Also evaluated were the transient conditions in the selected area during the loss of ventilation system and the consequences of ventilation openings that exist between the turbine building and the 4160V switchgear Bus 142X/242X area of the auxiliary building, should a postulated HELB occur that makes the turbine building harsh. The results are presented in Table 3.11-22.

Zone N3 consists of areas that are exposed to outdoor ambient conditions. These zone conditions are applicable to outdoor air intake plenums or shafts when the corresponding HVAC system is in operation. The histogram of the temperature and relative humidity was determined using the historical engineering weather data as referenced above for Zone N2 and are presented in Table 3.11-23.

The Radwaste building is served by non-safety related HVAC system to maintain environmental conditions suitable for personnel comfort and equipment operation. This building does not contain any safety-related equipment and is a mild environment.

3.11.1.3 Controlled Environment (Safety-Related HVAC Systems)

Controlled environments are defined as:

a. those areas of the plant housing safety-related equipment and served by redundant safety-related HVAC systems; and

b. those areas of the plant where the safety-related equipment is redundant and each redundant part is served by a separate safety-related HVAC system that maintains the temperature and humidity in accordance with the equipment operating requirements.

There are four areas of the plant that are considered as having a controlled environment. Each of these areas has been assigned a zone number and the boundaries are defined in Figure 3.11-5. A general description of these zones together with the bases for selecting the bounding conditions are as follows:

Zone C1 consists of the areas inside the main control room and auxiliary electric equipment room. The HVAC system serving these areas is designed to maintain the pressure, temperature, relative humidity, and radiation as listed in Tables 3.11-24 and 3.11-25 during normal and accident plant conditions.

Zone C2 includes the areas inside the essential switchgear rooms. The ventilation system serving these areas is designed to maintain the pressure, temperature, and relative humidity as listed in Table 3.11-26 during normal and accident plant conditions with the switchgears in use. The durations indicated were based on the engineering weather data (compiled by the Engineering Section of the U.S. Air Force, Manual No. AFM 88-29, dated July 1, 1978) applicable to the plant site.

Zone C3 is that area inside the diesel generator building including the RHR and the HPCS service water pump rooms and switchgear rooms. The minimum and normal conditions were determined based on the engineering weather data (USAF No. AFM 88-29) applicable to the plant site and when the equipment is not operating. The maximum condition was determined when the equipment is operating. The duration for the maximum conditions were conservatively based on the normal operating and testing of the equipment plus their operation during design basis accidents. Also, evaluated were the maximum steady state temperature in the Division 1 and 2 Core Standby Cooling Service (CSCS) pump rooms when the Diesel Generator (DG) cooling water pumps operate without the room's ventilation system operating. The environmental conditions are as listed in Table 3.11-27.

3.11.1.4 Evaluation for Flooding and Submergence

A detailed flooding evaluation for the entire plant has been performed, and is reported in Section 3.4. A summary of that evaluation is presented herein as follows.

3.11.1.4.1 Exterior Floods (See Section 3.4)

a. Sources

   1. Cooling lake - PMF corresponding to PMP with storm

   2. Local PMP

b. Results and Protective Measures

   1. PMF Lake Level with Wave Run-Up - 705 feet 8 inches. Plant is 4 feet higher than this level.

   2. PMP water buildup will not exceed elevation 710. 41 feet; whereas, plant floor elevation is 710 feet 6 inches.

c. Additional Protective Measures

   1. Waterproofing on all exterior walls to grade level.

   2. Waterstops on all exterior construction joints to grade.

3.11.1.4.2 Interior Floods (see Section 3.4)

a. Sources, Results and Protective Measures

1. Failure of water lines connected to lake

a. Circulating Water (inside Condenser Pit)

1) Floods noncritical Turbine Building area within Condenser Pit;

2) Watertight floodwall to contain flood water up to elevation 701' corresponding to maximum lake level during 1000 year-flood;
3) Water level alarms and duplex sump pumps are provided.
4) Auto trip of Circulating Water Pumps and operator actions to trip Service Water Pumps and to close valves to isolate source.

b. CSCS-ECWS in Pump Rooms

1) Watertight floodwalls to elevation 701' (except Division III as noted in Section 3.4.1.4.a) are provided between each room which house divisionally separated support equipment.
2) Flooding demonstrated to be non-credible in Division III CSCS-ECWS pump rooms based on SRP 3.6.2 crack exclusion criteria.
3) Water level alarms and duplex sump pumps are provided.

c. Service Water & Circulating Water (outside Condenser Pit)

1) Flooding from any of the five sources identified in Section 3.4.1.4.a has been demonstrated to be isolable or non-credible; based on SRP 3.6.2 crack exclusion criteria.

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2. Failure of Suppression Pool

   a) Suppression Pool level equalizes between inside and outside;
   b) Watertight floodwalls provided for each ECCS cubicle up to elevation 686'-7";
   c) Water level alarms, duplex sump pumps, and leak detection instrumentation, provided in each cubicle;
   d) Motor-operated valves are available in each cubicle for flood-related isolation.

b. <u>Additional Protective Measures</u>

1. Moderate energy line breaks (see Appendix J) leak detection, sump alarms, sump pumps, or other protective features advise the operator of leakage, and provide sufficient time to isolate the source by remote manual or automatic means.

2. Passive ECCS leaks - same as moderate energy breaks.

Also, flooding from moderate-energy line breaks (MELB) has been addressed in Appendix J, Section J.4.

It is also important to recognize that significant flooding does not result from an HELB outside of the primary containment, nor does flooding outside of the containment result from a LOCA inside the primary containment. Therefore, the flooding threat to any area is not directly connected to the accidents used to establish harsh environmental zones at the LaSalle County Station. This independence of harsh environments from flooding considerations is evident from previous discussion. This independence is important for environmental qualification considerations.

For the MELB, flooding may occur in certain cubicles. However, this flooding does not pose a threat because safety-related equipment located in possibly flooded cubicles has redundant safety-related equipment located in other cubicles that are not flooded. Therefore, flooding and the associated equipment submergence present no adverse environmental threat to LSCS.

3.11.2 Qualification Tests and Analyses

The safety-related equipment and components required to function during and subsequent to any of the postulated design accidents that degrade equipment capability (LOCA's, HELB's) are qualified to meet Category I or II requirements of NUREG-0588. The qualification is accomplished by either test, analysis, operating history, or a combination of these methods. Because many of these items are used in several systems and in different plant locations, they are qualified for the worst-case situation.

The typical approach used at LaSalle for the qualification of equipment potentially exposed to a harsh environment was as follows:

Evaluate the thermal and radiation life capability according to a planned 40-year service exposure to normal and abnormal conditions plus a worst accident exposure to those conditions. Use the Arrhenius methodology on weak link materials or if test data exist to derive a useful life estimate.

Complete qualification test profiles for NUREG-0588 Category II qualifications where information does not exist to represent certain effects such as humidity, mechanical or radiation aging, or thermal or pressure accident parameters.

Perform qualification test profiles for NUREG-0588 Category I qualification where equipment was unqualified and must be upgraded through modification, or must be replaced. If modified, the modified model was tested to Category I requirements.

Perform Category I qualification test, with seismic and mechanical testing in sequence, for new equipment obtained since May 1983.

It should be noted that the LaSalle EQ Program was executed concurrently with the seismic qualification (SQRT) program under NRC agreement, and that the appraisal review and acceptability sign-offs were made by the same group of AE engineers and Edison reviewers who had access to both sources of qualification records.

3.11.2.1 Qualification by Test

This method is widely used for equipment located in harsh environmental zones. Testing is done on actual equipment to simulate normal, abnormal, service, and accident conditions. While testing, the specimen is subjected to accelerated aging using Arrhenius methodology and other aging methods which are supported by type tests (e.g., 10° C rule). Synergistic effects are considered in the accelerated aging program where synergistic effects have been identified on materials that are included in the equipment being qualified. When size or other practical requirements limit or, preclude type tests, this part of the demonstration is completed by operating experience, qualification by analysis, and combined qualification. Partial type tests are augmented by tests of components where size, applications, time, or other test limitations preclude the use of a full type test.

3.11.2.2 Qualification Analysis

This method is widely used for equipment located in mild environmental zones.[1] When environmental parameters were not contained in the procurement specification, additional work was done to determine if qualification exists within Wyle Laboratories and S&L data banks for that particular organic component. If qualification documentation is available, it is reviewed to determine if the organic component qualified is identical to that to be qualified. If the former is enveloped by the latter, then an analysis to determine qualification life is performed using the existing data. If qualification does not exist within S&L data banks, a material degradation analysis is performed. Partial type tests on vital components of the equipment under qualification were provided in support of this method.

3.11.2.3 Operating Experience

Qualification of mechanical equipment using operating experience is used as a basis for environmental qualification should certificate of compliance from the vendor and analysis not be feasible.[2] This evaluation is done using similar equipment with a successful operating history in a service environment equal to or more severe than the environment for the equipment in question. The validity of operating experience as a means of qualification is determined from the type and amount of available supporting documentation, the service conditions, and equipment performance. As this approach qualifies the equipment for normal environments, additional material degradation analysis is performed to qualify the equipment for the DBE. Partial type tests on vital components of the equipment under qualification are provided in support of this method.

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[1]Qualification Analysis is also used for equipment located in harsh areas for which testing is precluded by physical size of the equipment or other limitations.

[2]This method is used for equipment for which testing is precluded by physical size of the equipment.

3.11.2.4 Combined Qualification

Combined qualification is used for any equipment which cannot be qualified through a full type test. Combined qualification is usually any combination of type test, previous operating experience, and analysis. Partial type tests with extrapolation or analysis, operating experience with extrapolation or analysis, and type test supplemented with tests of components and analysis are examples of the use of combined qualification. The diesel generator qualification program utilized a combined qualification technique.

3.11.3 Qualification Results

The results of the environmental qualification for each type of the safety-related equipment identified in the LaSalle controlled computer database are included in the extensive file of EQ binders created and maintained for LaSalle County Station. Because the only acceptable test result is successful qualification with a justified lifetime, equipment which does not pass the test was either retested or replaced by other qualified equipment. The environmental qualification file provides documentation of evaluations, analyses, and test results to show that safety-related equipment utilized at the station is qualified to perform intended functions for its qualified life.

3.11.4 Loss of Ventilation

3.11.4.1 Control Room Air-Conditioning and Ventilation System

Controls and electrical equipment necessary for safe plant shutdown are located in the control room. The control room is air-conditioned and shielded against radiation to enable safe and continued occupancy under accident conditions. Air-conditioning equipment and environmental components are designed to Seismic Category I requirements. The refrigeration equipment condensers are fabricated in accordance with the ASME Code. Redundant equipment is provided. Upon loss of offsite power, emergency power is automatically supplied from the onsite diesel-generator sets. No single failure results in loss of control room air-conditioning. Operability of the safety-related control and electrical equipment located in the control room will not be impaired and can continue to function in an accident environment. No special environmental design requirements for loss of ventilation or air-conditioning to essential safety-related electrical or instrumentation equipment is necessary because they are located in the control room.

The safe shutdown panel required in case of control room evacuation for fire is located below the control room at floor elevation 731' - 0". The HVAC system serving that location (auxiliary electric equipment room) has the same design requirements as that serving the control room.

3.11.4.2 Emergency Switchgear Room Ventilation System

Redundant emergency switchgear for safety-related equipment is located in separate rooms in the auxiliary building. The switchgear rooms are provided with ventilation systems to maintain the rooms at 104° F and approximately 40%

relative humidity under outdoor design conditions. However, the switchgear equipment is conservatively designed to function under conditions at 104° F and 90% relative humidity for extended periods.

The ventilation equipment for each switchgear room is designed to Seismic Category I requirements (Section 9.4), and each switchgear room ventilating system is supplied with onsite emergency power upon loss of offsite power. No single failure can result in loss of ventilation in any of the emergency switchgear rooms (see Chapter 7.0).

3.11.5 Estimated Chemical and Radiation Environment

The LaSalle County Station contains no special chemical environments that warrant investigation for their effects on safety-related equipment.

The radiation environment for the zones identified in Subsection 3.11.1, includes the 40 year normal operating dose and the design-basis accident dose in accordance with NUREG-0588. The normal dose was established using the highest calculated dose rate in the zone. The various subzones were identified based on consideration of shielding, distance, and other potential radiation sources. The accident doses were calculated with a time dependent leakage model which uses a double containment with a controlled exhaust. Regulatory Guide 1.3 sources are released to the drywell, the radioactivity slowly leaks to the secondary containment where it builds up for several days, and then decreases because the SGTS filtered exhaust and the radioactive decay remove it faster than the leakage supplies it. Additional accident sources, i.e., ECCS pump leakage and radioactive ECCS equipment and piping inside the secondary containment were also included in the total equipment qualification exposure.

The postaccident doses for the harsh zones outside the double containment, include the immersion dose from the SGTS exhaust cloud, and the unshielded direct and scattered radiation dose.

The equipment qualification dose used for each zone and subzone is the sum of the integrated normal and accident doses and includes at least a 10% margin. These doses are listed in Tables 3.11-3 through 3.11-29.

TABLE 3.11-3

HARSH ENVIRONMENT ZONE H1 - BOUNDING

ENVIRONMENTAL CONDITIONS INSIDE THE RPV

	NORMAL	DESIGN	MAXIMUM
Temperature (°F)	546	575	581
Pressure (psig)	1010	1250	1340
Relative Humidity	Saturated Steam/Water	Saturated Steam/Water	Saturated Steam/Water
Radiation	2.3×10^{10} rads (gamma integrated)* 7.9×10^{16} neutrons/cm^2 (neutron fluence)*		

NOTE: The bounding radiation dose ≥ (normal service radiation dose integrated over 40 years + accident dose + 10% margin on the accident dose per IEEE 323-1974, Section 6.3.1.5)

* Integrated over 40 years.

TABLE 3.11-4

HARSH ENVIRONMENT ZONE H2 - BOUNDING

ENVIRONMENTAL CONDITIONS INSIDE THE DRYWELL

Temperature (°F)	340	320	250	200
Pressure (psig)	-2 to 45	-2 to 45	0 to 25	0 to 20
Relative Humidity	Steam	Steam	100%	100%
Duration	0-3 hr	3-6 hr	6 hr to 1 day	1 day to 100 days
Radiation	2×10^8 rads gamma (integrated)			

NOTE: The bounding radiation dose ≥ (normal service radiation dose integrated over 40 years + accident dose + 10% margin on the accident dose per IEEE 323-1974, Section 6.3.1.5)

TABLE 3.11-5

HARSH ENVIRONMENT ZONE H3 - BOUNDING

ENVIRONMENTAL CONDITIONS INSIDE THE WETWELL

Temperature (°F)	200	200	150
Pressure (psig)	-2 to 45	0 to 25	0 to 20
Relative Humidity	100%	100%	100%
Duration	0 to 6 hr	6 hr to 1 day	1 day to 100 days
Radiation	2×10^8 rads gamma (integrated)		

NOTE: The bounding radiation dose ≥ (normal service radiation dose integrated over 40 years + accident dose + 10% margin on the accident dose per IEEE 323-1974, Section 6.3.1.5)

TABLE 3.11-6
(Sheet 1 of 3)

HARSH ENVIRONMENT ZONE H4 - BOUNDING
ENVIRONMENTAL CONDITIONS IN THE REACTOR BUILDING

A. HELB Accident Conditions:

Sub-Zone H4A/H4B/H4C/H4E (See Figure 3.11-3):

Temperature (°F)	Duration (Days)
145-134	0-1
134-128	1-3
128-115	3-13
115-111	13-38
111-110.5	38-100

Pressure: -0.25 inch W. G
Relative Humidity: 30-95% (all zones except H4E)
100% (Zone H4E)

Sub-Zone H4F (See Figure 3.11-3):

Temperature profile: See Figure 3.11-7
Pressure: -0.25 inch W. G
Relative Humidity: 100%

Sub-ZoneH4G (See Figure 3.11-3):

Temperature profile: See Figure 3-11-8
Pressure: -0.25 inch W. G
Relative Humidity: 100%

Sub-Zone H4H (See Figure 3.11-3):

Temperature profile: Short-term peak of 300°F for the first 60 seconds followed by 212°F and gradually decreasing thereafter.
Pressure: -0.25 inch W. G.
Relative Humidity: 100% RH

TABLE 3.11-6
(Sheet 2 of 3)

HARSH ENVIRONMENT ZONE H4 - BOUNDING ENVIRONMENTAL CONDITIONS IN THE REACTOR BUILDING

B. LOCA Environmental Conditions:

Sub-Zone H4A (See Figure 3.11-3 for Zone Boundary):

Temperature Profile: See Figure 3.11-9
Pressure: -0.25 inch W.G
Relative Humidity: 95% RH (Maximum)
Radiation (T.I.D): 1.0E07 rads gamma

Sub-Zone H4B (See Figure 3.11-3 for Zone Boundary):

Temperature Profile: See Figure 3.11-10
Pressure: -0.25 inch W.G
Relative Humidity: 100% RH (Maximum)
Radiation (T.I.D): 4.0E07 rads gamma

Sub-Zone H4C (See Figure 3.11-3 for Zone Boundary):

Temperature Profile: See Figure 3.11-10
Pressure: -0.25 inch W.G
Relative Humidity: 100% RH (Maximum)
Radiation (T.I.D): 2.0E08 rads gamma

Sub-Zone H4E, General Area on Elevation 820'6" (See Figure 3.11-3 for Zone Boundary):

Temperature Profile: See Figure 3.11-10
Pressure: -0.25 inch W.G
Relative Humidity: 100% RH (Maximum)
Radiation (T.I.D): 1.0E07 rads gamma

Sub-Zone H4E, General Area on Elevation 820'6" (See Figure 3.11-3 for Zone Boundary):

Temperature Profile: See Figure 3.11-11
Pressure: -0.25 inch W.G
Relative Humidity: 100% RH (Maximum)
Radiation (T.I.D): 1.0E07 rads gamma

TABLE 3.11-6
(Sheet 3 of 3)

HARSH ENVIRONMENT ZONE H4 - BOUNDING ENVIRONMENTAL CONDITIONS IN THE REACTOR BUILDING

Sub-Zone H4F (See Figure 3.11-3 for Zone Boundary):

Temperature Profile: See Figure 3.11-12
Pressure: -0.25 inch W.G
Relative Humidity: 95% RH (Maximum)
Radiation (T.I.D): 1.0E07 rads gamma

Sub-Zone H4G (See Figure 3.11-3 for Zone Boundary):

Temperature Profile: See Figure 3.11-13
Pressure: -0.25 inch W.G
Relative Humidity: 95% RH (Maximum)
Radiation (T.I.D): 1.0E07 rads gamma

Sub-Zone H4H, All Areas Except the aisle outside the south and north RWCU Sludge Pump Rooms at Elevation 807"-0" (See Figure 3.11-3 for Zone Boundary):

Temperature Profile: See Figure 3.11-14
Pressure: -0.25 inch W.G
Relative Humidity: 95% RH (Maximum)
Radiation (T.I.D): 1.0E07 rads gamma

Sub-Zone H4H, Aisle outside the south and north RWCU Sludge Pump Rooms at Elevation 807"-0" (See Figure 3.11-3 for Zone Boundary):

Temperature Profile: See Figure 3.11-15
Pressure: -0.25 inch W.G
Relative Humidity: 95% RH (Maximum)
Radiation (T.I.D): 1.0E07 rads gamma

Notes:
1. The bounding radiation dose (normal service radiation dose integrated over 40 years + accident dose + 10% margin on the accident dose per IEEE 323-1974, Section 6.3.1.5)

TABLE 3.11-7

HARSH ENVIRONMENT ZONE H5 - BOUNDING ENVIRONMENTAL CONDITIONS FOR HELB AREAS IN THE REACTOR BUILDING

Temperature (°F)	212	150	150
Pressure	7 in. W. G. *	7 in. W. G.	Atmospheric
Relative Humidity	Steam	100%	90%
Duration	0-6 hr	6-12 hr	12 hr to 100 days
Radiation	1×10^7 rads gamma (integrated)		

NOTE: The bounding radiation dose ≥ (normal service radiation dose integrated over 40 years + accident dose + 10% margin on the accident dose per IEEE 323-1974, Section 6.3.1.5)

* Transient peak pressure of 29.4 psig can occur within the first 7 seconds of the line break event. However, venting and subsequent depressurization to atmospheric pressure occurs very rapidly.

TABLE 3.11-8

HARSH ENVIRONMENT ZONE H6 - BOUNDING ENVIRONMENTAL

CONDITIONS INSIDE THE ECCS CUBICLES

(EXCLUDING LPCS/RCIC CUBICLE) IN THE REACTOR BUILDING

WHEN THE ECCS EQUIPMENT IS OPERATING

The maximum cubicle temperature is 148°F*, 15% relative humidity and at atmospheric pressure for the duration of 100 days. The total number of hours the cubicle is at 148°F will be ~22,110 hours (~921 days). The 100 days accident conditions are included.

Radiation: 1 x 10^7 rads gamma (integrated)

Pressure: 0 inch W. G.

* For accident conditions involving a maximum analyzed cooling water inlet temperature of 104 deg-F to the HPCS Room Cooler, the peak cubicle air temperature is 150 deg-F. This peak cooling water temperature occurs for a time period of less than 6 hours.

NOTE: The bounding radiation dose ≥ (normal service radiation dose integrated over 40 years + accident dose + 10% margin on the accident dose per IEEE 323-1974, Section 6.3.1.5).

TABLE 3.11-9

HARSH ENVIRONMENT ZONE H7

BOUNDING ENVIRONMENTAL CONDITIONS FOR THE

TURBINE BUILDING*

Temperature (°F)	212	150	150
Pressure	7 in. W. G.	7 in. W. G.	Atmospheric
Relative Humidity	Steam	100%	90%
Duration	0-6 hr	6-12 hr	12 hr to 100 days
Radiation	1×10^7 rads gamma (integrated)		

NOTE: The bounding radiation dose ≥ (normal service radiation dose integrated over 40 years + accident dose + 10% margin on the accident dose per IEEE 323-1974, Section 6.3.1.5).

* The DG corridor on elevation 710’-6” is also included in the Zone H7 boundary due to open pathways between turbine building and DG corridor.

TABLE 3.11-10

HARSH ENVIRONMENT ZONE H8

BOUNDING ENVIRONMENTAL CONDITIONS FOR THE

TURBINE BUILDING

Temperature(°F)	212	150	150
Pressure	7 in. W. G.**	7 in. W. G.	Atmospheric
Relative Humidity	Steam	100%	90%
Duration	0-6 hr	6-12 hr	12 hr to 100 days
Radiation	4×10^7 rads gamma (integrated)		

NOTE: The bounding radiation dose ≥ (normal service radiation dose integrated over 40 years + accident dose + 10% margin on the accident dose per IEEE 323-1974, Section 6.3.1.5).

** Transient peak pressure of up to 6.0 psig can occur within the first 10 seconds of the line break event. However, venting and subsequent depressurization to atmospheric pressure occurs very rapidly.

TABLE 3.11-11
(SHEET 1 OF 2)

HARSH ENVIRONMENT ZONE H9 - BOUNDING ENVIRONMENTAL

CONDITIONS IN SPECIAL AREAS OF THE AUXILIARY BUILDING

A. INSIDE CONDITIONS WITH THE VENTILATION SYSTEM OPERATING

TEMPERATURE (°F)	RELATIVE HUMIDITY (%)	DURATION NO. OF DAYS IN 40 YEARS PLANT LIFE
106	34	10
105	32	97
100	34	275
95	38	540
90	40	812
85	46	1136
80	50	1266
75	50	1167
70	52	1007
65	50	8300

Radiation: 1 x 10^5 rads gamma (integrated)

Pressure: 0 inch W. G.

TABLE 3.11-11
(SHEET 2 OF 2)

B. INSIDE CONDITIONS DURING LOSS OF NORMAL VENTILATION SYSTEM

	Time (Hr)	0	100	1350	2400
SUMMER	Temp. (°F)	106	94	86	82
	Rel. Hum. (%)	34	50	64	72
	Time (Hr)	0	24	1200	2400
WINTER	Temp. (°F)	65	60	36	30
	Rel. Hum. (%)	50	60	100	100

Radiation: 1×10^5 rads gamma (integrated)

Pressure: 0 inch WG

NOTES:

1. The bounding radiation dose ≥ (normal service radiation dose integrated over 40 years + accident dose + 10% margin on the accident dose per IEEE 323-1974, Section 6.3.1.5.)

2. Temperature and relative humidity change linearly from one step to the next.

TABLE 3.11-12

HARSH ENVIRONMENT ZONE H1 - SERVICE

CONDITIONS INSIDE THE RPV

	NORMAL	DESIGN	MAXIMUM
Temperature (°F)	546	575	581
Pressure (psig)	1010	1250	1340
Relative Humidity		- Saturated Steam/Water -	
Radiation			
Gamma (rads)*	2.3×10^{10}		
Neutrons (neutrons/cm^2)	7.9×10^{16}		

* Integrated over 40 years

TABLE 3.11-13

HARSH ENVIRONMENT ZONE H2 - SERVICE CONDITIONS INSIDE THE DRYWELL

SUBZONE H2A - GENERAL DRYWELL AREA:		
	NORMAL	MAXIMUM
Temperature (°F)	135	150
Pressure (psig)	-0.5 to 2	-
Relative Humidity	40 - 55%	90%
Duration	14,320**	***
Radiation	2 x 10^7 rads gamma (integrated)	
SUBZONE H2B - CRD AREA INSIDE RPV PEDSTAL		
	NORMAL	MAXIMUM
Temperature (°F)	103	175
Pressure (psig)	-0.5 to 2	
Relative Humidity	4 to 24%	
Duration*	14,610	
Radiation	2 x 10^7 rads gamma (integrated) +	

* Number of days in 40 year life

** The reactor is conservatively assumed to have 120 shutdowns (averaging two days each) in the 40 year normal operating life. During shutdown, the temperature is conservatively assumed to be 104°F.

*** The maximum temperature of 150°F occurs (a) in the RPV annular space where no safety-related equipment is located, (b) possibly during VP HVAC System switchover (assumed not to exceed a total of 10 days during the 40 year normal operating life), and (c) due to some unidentified system leakage (assumed not to exceed 40 days during the 40 year normal operating life because unit shutdown would be initiated if containment temperatures reach 150°F).

\+ Integrated over 40 years, and includes gamma equivalent dose to account for neutron fluence.

TABLE 3.11-14

HARSH ENVIRONMENT ZONE H3 - SERVICE

CONDITIONS INSIDE THE WETWELL

	NORMAL	MAXIMUM
Temperature (°F)	100	110
Pressure (psig)	-0.5 to 2	
Relative Humidity (%)	100	100
Radiation	2×10^7 rads gamma (integrated)	

TABLE 3.11-15

HARSH ENVIRONMENT ZONE H4 - SERVICE CONDITIONS

IN THE REACTOR BUILDLING (GENERAL ACCESSIBLE AREAS)

TEMPERATURE (°F)	HUMIDITY (%)	DURATION* (DAYS)	
		UNIT 1	UNIT 2
118	23	9	9
113	25	86	91
108	26	245	257
103	29	481	504
98	32	724	759
94	35	11,466	12,019

Radiation: 2×10^6 rads gamma (integrated)

Pressure: 0.25 inch W. G.

NOTE: Time/Temperature profile prior to November 6, 1986 can be found on Amendment 64, Section 3.11 of the FSAR.

* Based on remaining plant life:
[40 years - (revised setpoint date - start up date)]
Start up of Unit 1 - June 21, 1982, Unit 2 - March 10, 1984
Revised setpoint date - November 6, 1986

TABLE 3.11-16
(SHEET 1 OF 4)

HARSH ENVIRONMENT ZONE H5 - SERVICE CONDITIONS
IN THE REACTOR BUILDING AREAS SUBJECT TO HELB

SUBZONE H5A - LPCS/RCIC CUBICLE:

1. When the ECCS equipment is operating:

   The maximum cubicle temperature is 148°F, 15% relative humidity and at atmospheric pressure. The duration of normal operating and testing of the ECCS equipment is approximately 41.06 hours per month or a total of 17,544 hours (731 days) for Unit 1 and 18,384 hours (766 days) for Unit 2 within the remaining years plant life.

TEMPERATURE	HUMIDITY	DURATION* (DAYS)	
(°F)	(%)	UNIT 1	UNIT 2
124	20	9	9
119	21	82	86
114	22	232	243
109	25	454	476
109	27	680	713
100	29	10,823	11,346

Radiation: 5 x 10^5 rads gamma (integrated)

Pressure: -0.25 inch W. G.

NOTE: Time/Temperature profile prior to November 6, 1986 can be found on Amendment 64, Section 3.11 of the FSAR.

* Based on remaining plant life:
[40 years - (revised setpoint date - start up date)]
Start up of Unit 1 - June 21, 1982, Unit 2 - March 10, 1984
Revised setpoint date - November 6, 1986

TABLE 3.11-16
(SHEET 2 OF 4)

SUBZONE H5B - RCIC PIPE TUNNEL:

TEMPERATURE (°F)	HUMIDITY (%)	DURATION (DAYS)	
		UNIT 1	UNIT 2
160	8	9	9
155	8.5	86	91
150	9	245	257
145	8.5	481	504
140	10	724	759
136	10.5	11,466	12,019

Radiation: 5×10^5 rads gamma (integrated)

Pressure: -0.25 inch W. G.

SUBZONE H5C - MAIN STEAM TUNNEL:

TEMPERATURE (°F)	HUMIDITY (%)	DURATION* (DAYS)	
		UNIT 1	UNIT 2
130	16	9	9
125	18	86	91
120	19	245	257
115	20	481	504
110	22	724	759
106	25	11,466	12,019

Radiation: 6×10^6 rads gamma (integrated)

Pressure: -0.25 inch W. G.

NOTE: Time/Temperature profile prior to November 6, 1986 can be found on Amendment 64, Section 3.11 of the FSAR.

* Based on remaining plant life:
[40 years - (revised setpoint date - start up date)]
Start up of Unit 1 - June 21, 1982, Unit 2 - March 10, 1984
Revised setpoint date - November 6, 1986

TABLE 3.11-16
(SHEET 3 OF 4)

SUBZONE H5D - RWCU EQUIPMENT AREAS:

TEMPERATURE	HUMIDITY	DURATION* (DAYS)	
(°F)	(%)	UNIT 1	UNIT 2
131	16	9	9
126	17	86	91
121	18	245	257
116	20	481	504
111	23	724	759
107	25	11,466	12,019

Radiation: 5×10^6 rads gamma (integrated)

Pressure: -0.25 inch W. G.

NOTE: Time/Temperature profile prior to November 6, 1986 can be found on Amendment 64, Section 3.11 of the FSAR.

* Based on remaining plant life:
[40 years - (revised setpoint date - start up date)]
Start up of Unit 1 - June 21, 1982, Unit 2 - March 10, 1984
Revised setpoint date - November 6, 1986

TABLE 3.11-16
(SHEET 4 OF 4)

SUBZONE H5E - BASEMENT AND UPPER BASEMENT OUTSIDE THE ECCS EQUIPMENT CUBICLES:

TEMPERATURE (°F)	HUMIDITY (%)	DURATION(DAYS)	
		UNIT 1	UNIT 2
118	23	9	9
113	25	86	91
108	26	245	257
103	29	481	504
98	32	724	759
94	35	11,466	12,019

Radiation: 5×10^5 rads gamma (integrated)

Pressure: -0.25 inch W. G.

NOTE: Time/Temperature profile prior to November 6, 1986 can be found on Amendment 64, Section 3.11 of the FSAR.

* Based on remaining plant life:
[40 years - (revised setpoint date - start up date)]
Start up of Unit 1 - June 21, 1982, Unit 2 - March 10, 1984
Revised setpoint date - November 6, 1986

TABLE 3.11-17

HARSH ENVIRONMENT ZONE H6 - SERVICE CONDITIONS
INSIDE THE ECCS CUBICLES
(EXCLUDING LPCS/RCIC CUBICLE) IN THE REACTOR BUILDING

A. WHEN THE ECCS EQUIPMENT IS OPERATING

The maximum cubicle temperature is 148°F, 15% relative humidity and at atmospheric pressure. The duration of normal operating and testing of the ECCS equipment is approximately 41.06 hours per month or a total of 17,544 hours (731 days) for Unit 1 and 18,384 hours (766 days) for Unit 2 within the remaining years plant life.

B. WHEN THE ECCS EQUIIPMENT IS NOT OPERATING

TEMPERATURE (°F)	HUMIDITY (%)	DURATION(DAYS)*	
		UNIT 1	UNIT 2
123	20	9	9
118	21	82	86
113	23	232	243
108	25	454	476
103	27	680	713
99	29	10,823	11,346

Radiation: 5×10^5 rads gamma (integrated)

Pressure: -0.25 inch W. G.

NOTE: Time/Temperature profile prior to November 6, 1986 can be found on Amendment 64, Section 3.11 of the FSAR.

* Based on remaining plant life:
[40 years - (revised setpoint date - start up date)]
Start up of Unit 1 - June 21, 1982, Unit 2 - March 10, 1984
Revised setpoint date - November 6, 1986

TABLE 3.11-18
(SHEET 1 OF 2)

HARSH ENVIRONMENT ZONE H7 - SERVICE CONDITIONS

INSIDE THE TURBINE BUILDING

A. INSIDE CONDITIONS WITH THE VENTILATION SYSTEM OPERATING

TEMPERATURE (°F)	RELATIVE HUMIDITY (%)	DURATION NO. OF DAYS IN 40 YEARS PLANT LIFE
102	47	3
100	46	57
98	45	197
96	45	234
94	45	355
92	45	3288
87	40	1167
83	39	9309

Radiation: 6×10^6 rads gamma (integrated)

Pressure: 0 inch W. G.

TABLE 3.11-18
(SHEET 2 OF 2)

HARSH ENVIRONMENT ZONE H7 - SERVICE CONDITIONS

INSIDE THE TURBINE BUILDING

B. INSIDE CONDITIONS DURING LOSS OF VENTILATION SYSTEM AFTER LOSS OF OFFSITE POWER

SUMMER	Time (Hr)	0	112	250	475	1389	2400	
	Temp. (°F)	102	130	150	150	118	98	
	Rel. Hum. (%)	47	22	14	14	30	54	

WINTER	Time (Hr)	0	112	250	417	1111	1667	2400
	Temp. (°F)	83	100	120	120	80	62	48
	Rel. Hum. (%)	39	16	14	14	43	80	100

Radiation: 6×10^6 rads gamma (integrated)

Pressure: 0 inch W. G.

NOTE:

Temperature and relative humidity change linearly from one time step to the next.

TABLE 3.11-19
(SHEET 1 OF 2)

HARSH ENVIRONMENT ZONE H8 - SERVICE CONDITIONS

INSIDE THE TURBINE BUILDING

A. INSIDE CONDITIONS WITH THE VENTILATION SYSTEM OPERATING

TEMPERATURE (°F)	RELATIVE HUMIDITY (%)	DURATION NO. OF DAYS IN 40 YEARS PLANT LIFE
121	28	3
119	27	57
117	27	197
115	27	234
113	26	355
111	26	3288
106	22	1167
102	21	9309

Radiation: 3×10^7 rads gamma (integrated)

Pressure: 0 inch W. G.

TABLE 3.11-19
(SHEET 2 OF 2)

B. INSIDE CONDITIONS DURING LOSS OF VENTILATION SYSTEM AFTER LOSS OF OFFSITE POWER

SUMMER	Time (Hr)	0	24	84	500	1389	2400	
	Temp. (°F)	121	225	225	180	128	105	
	Rel. Hum. (%)	28	2.5	2.5	7	23	45	
WINTER	Time (Hr)	0	24	84	417	915	1667	2400
	Temp. (°F)	102	212	212	160	108	72	56
	Rel. Hum. (%)	21	1.5	1.5	5	18	55	95

Radiation: 3×10^7 rads gamma (integrated)
Pressure: 0 inch W. G.

NOTE:

Temperature and relative humidity change linearly from one time step to the next.

TABLE 3.11-20
(SHEET 1 OF 2)

HARSH ENVIRONMENT ZONE H9 - SERVICE CONDITIONS

IN SPECIAL AREAS OF THE AUXILIARY BUILDING

A. INSIDE CONDITIONS WITH THE NORMAL VENTILATION SYSTEM OPERATING

TEMPERATURE (°F)	RELATIVE HUMIDITY (%)	DURATION NO. OF DAYS IN 40 YEARS PLANT LIFE
106	34	10
105	32	97
100	34	275
95	38	540
90	40	812
85	46	1136
80	50	1266
75	50	1167
70	52	1007
65	50	8300

Radiation: 1×10^3 rads gamma (integrated)

Pressure: 0 inch W. G.

TABLE 3.11-20
(SHEET 2 OF 2)

B. INSIDE CONDITIONS DURING LOSS OF NORMAL VENTILATION SYSTEM

SUMMER	Time (Hr)	0	100	1350	2400
	Temp. (°F)	106	94	86	82
	Rel. Hum. (%)	34	50	64	72

WINTER	Time (Hr)	0	24	1200	2400
	Temp. (°F)	65	60	36	30
	Rel. Hum. (%)	50	60	100	100

Radiation: 1×10^3 rads gamma (integrated)
Pressure Range: 0 inch W. G.

NOTE:

Temperature and relative humidity change linearly from one step to the next.

TABLE 3.11-21

NORMAL ENVIRONMENT ZONE N1 - SERVICE CONDITIONS

IN THE AUXILIARY BUILDING

	MINIMUM	NORMAL	MAXIMUM
Temperature (°F)	65	75	85
Pressure (Inches W. G.)	0	0	0
Relative Humidity (%)	35	45	55
Radiation	1×10^4 rads gamma (integrated) [1]		

[1] The total integrated radiation dose (40 – year normal operation plus one–year accident) for the area within the 10’ 0” radius of the air locks into the Reactor building is 6.40×10^4 rads gamma.. The one–year post–accident dose is 5.38×10^4 rads gamma within the 10’ 0” radius of the air locks due to shine from the reactor building through the air lock door.

TABLE 3.11-22
(SHEET 1 OF 2)

NORMAL ENVIRONMENT ZONE N2 - SERVICE CONDITIONS

IN THE AUXILIARY BUILDING

A. INSIDE CONDITIONS WITH THE VENTILATION SYSTEM OPERATING

TEMPERATURE (°F)	RELATIVE HUMIDITY (%)	DURATION NO. OF DAYS IN 40 YEARS PLANT LIFE
106	34	10
105	32	97
100	34	275
95	38	540
90	40	812
85	46	1136
80	50	1266
75	50	1167
70	52	1007
65	50	8300

Radiation: 1 x 10^4 rads gamma (integrated) [(2)]

Pressure: 0 inch W. G.

TABLE 3.11-22
(SHEET 2 OF 2)

B. INSIDE CONDITIONS DURING LOSS OF VENTILATION SYSTEM

1) HVAC Equipment Area At Floor El. - 786 feet 6 inches

Summer	Time (Hr)	0	100	1350	2400
	Temp. (°F)	106	94	86	82
	Rel. Hum. (%)	34	50	64	72
Winter	Time (Hr)	0	24	1200	2400
	Temp. (°F)	65	60	36	30
	Rel. Hum. (%)	50	60	100	100

2) 4160-V Switchgear Bus 142X/242X Area

Summer	Time (Hr)	0	150	390	900	2400	
	Temp. (°F)	106	120	134	134	122	
	Rel. Hum. (%)	34	24	16	16	22	
Winter	Time (Hr)	0	24	150	360	840	2400
	Temp. (°F)	65	84	106	122	122	104
	Rel. Hum. (%)	50	28	14	9	9	15

NOTES:

1. The pressure range inside the zone N2 area is 0 in. W. G.

2. The total integrated radiation dose (40 – year normal operation plus one–year accident) for the area within the 10' 0" radius of the air locks into the Reactor building is 6.5 x 10^4 rads gamma. The one–year post–accident dose is 5.38 x 10^4 rads gamma within the 10' 0" radius of the air locks due to shine from the reactor building through the air lock door.

3. Temperature and relative humidity change linearly from one step to the next.

4. 4160V Switchgear Bus 142X/242X Area on elevation 731' and General Accessible area on elevation 749' of the auxiliary building will be exposed to short-term transient conditions due to a high energy line break in the turbine building. For the Bus 142X/242X area, the short-term conditions are 100% RH humidity and 130°F temperature. The temperature decreases to normal ambient conditions in approximately 30 minutes. The temperature in the general accessible area on elevation 749' of the auxiliary building is not affected. However, the short-term humidity will be 100% RH.

TABLE 3.11-23

NORMAL ENVIRONMENT ZONE N3 - SERVICE CONDITIONS

IN AMBIENT EXPOSED AREAS

TEMPERATURE (°F)	RELATIVE HUMIDITY (%)	DURATION NO. OF DAYS IN 40 YEARS PLANT LIFE	DURATION OF COINCIDENT SATURATION (100% RH)
100	45	107	3
90	48	816	27
80	58	1949	93
70	73	2435	93
60	80	1978	143
50	80	1898	50
40	80	2660	35
30	75	1602	32
20	55	659	40
10	50	340	27
0	40	134	17
-10	20	22	3
-20	10	10	0

Radiation: 1×10^4 rads
Pressure: Atmospheric

Note: Duration of coincident saturation is the tabulation of days per 40 year plant life for estimated occurrences of 100% relative humidity for each temperature listed

TABLE 3.11-24

CONTROLLED ENVIRONMENT ZONE C1A - CONDITIONS INSIDE THE MAIN CONTROL ROOM (4)

	MINIMUM	NORMAL	MAXIMUM
Temperature (°F)	50	(1)	104
Pressure (Inches W. G.)	(2)	(2)	+3.0 (2)
Relative humidity (%)	2.6 (3)	(3)	(3)
Radiation		1×10^3 rads gamma (integrated)	
Duration (days/40 yr. Life)		(4)	

Notes:

(1) This zone is served by the safety related/redundant HVAC system which maintains the environment between 65°F and 85°F.

(2) During normal plant operations, the zone will be positively pressurized with respect to adjacent areas. In the emergency filtration mode, the pressure will be ≥ 1/8 inch water gauge. Maximum expected pressure during purge mode of operation is 3 inches of water.

(3) There is no Relative Humidity control system. Relative Humidity is expected to range between 20% and 50%. During winter months, a Relative Humidity of 2.6% may occur. A maximum humidity of 50% in the zone ensures that the resultant humidity of the air entering the recirculation filter remains below the value of 70% assumed in the filter testing program. The maximum design humidity to which equipment in the zone may be subjected is 90%.

(4) The four variables in this Table are independent variables. The number of hours the system may be operating at minimum, normal, and maximum conditions is difficult to accurately define because the parameters vary with outside weather conditions and vary with internal system configuration changes.

TABLE 3.11-25

CONTROLLED ENVIRONMENT ZONE C1B - CONDITIONS INSIDE

THE AUXILIARY ELECTRIC EQUIPMENT ROOM (4)

	MINIMUM	NORMAL	MAXIMUM
Temperature (°F)	50	(1)	104
Pressure (Inches W. G.)	(2)	(2)	+3.0 (2)
Relative humidity (%)	2.6 (3)	(3)	(3)
Radiation		1 x 10^3 rads gamma (integrated)	
Duration (days/40 yr. Life)		(4)	

Notes:

(1) This zone is served by the safety related/redundant HVAC system which maintains the environment between 65°F and 85°F.

(2) During normal plant operations, the zone will be positively pressurized with respect to adjacent areas. In the emergency filtration mode, the pressure will be ≥ 1/8 inch water gauge. Maximum expected pressure during purge mode of operation is 3 inches of water.

(3) There is no Relative Humidity control system. Relative Humidity is expected to range between 20% and 50%. During winter months, a Relative Humidity of 2.6% may occur. A maximum humidity of 50% in the zone ensures that the resultant humidity of the air entering the recirculation filter remains below the value of 70% assumed in the filter testing program. The maximum design humidity to which equipment in the zone may be subjected is 90%.

(4) The four variables in this Table are independent variables. The number of hours the system may be operating at minimum, normal, and maximum conditions is difficult to accurately define because the parameters vary with outside weather conditions and vary with internal system configuration changes.

TABLE 3.11-26

CONTROLLED ENVIRONMENT ZO NE C2 - CONDITIONS INSIDE

THE ESSENTIAL SWITCHGEAR ROOMS

TEMPERATURE (°F)	RELATIVE HUMIDITY (%)	DURATION NO. OF DAYS IN 40 YEARS PLANT LIFE
104	36	10
103	34	99
98	36	282
93	40	554
88	43	833
83	49	1164
78	53	1294
73	55	584
70	57	9890

Radiation: 1 x 10^3 rads gamma (integrated)

Pressure: 1.08 inch W. G.

TABLE 3.11-27
(SHEET 1 OF 2)

CONTROLLED ENVIRONMENT ZONE C3 - CONDITIONS INSIDE

THE DIESEL GENERATOR ROOMS, RHR SERVICE WATER

PUMP ROOMS, HPCS SWITCHGEAR ROOMS AND

HPCS DG COOLING WATER PUMP ROOMS

A. WHEN THE DIESEL GENERATOR, DG COOLING WATER AND RHR-WS PUMPS, AND SWITCHGEAR ARE OPERATING

TEMPERATURE (°F)	RELATIVE HUMIDITY (%)	DURATION NO. OF DAYS IN 40 YEARS PLANT LIFE
119	21	10
114	25	94
109	26	52
104	29	50
99	30	49
94	34	38
89	37	53
84	38	20
80	38	92

Radiation: 1 x 10^3 rads gamma (integrated)

Pressure: 1.165 inch W. G.

TABLE 3.11-27
(SHEET 2 OF 2)

B. WHEN THE DIESEL GENERATOR, PUMPS AND SWITCHGEAR ARE NOT OPERATING

TEMPERATURE (°F)	RELATIVE HUMIDITY (%)	DURATION NO. OF DAYS IN 40 YEARS PLANT LIFE
109	32	5
107	32	147
105	32	215
105	31	334
103	31	3189
98	29	1147
94	28	9215

Radiation: 1 x 10^3 rads gamma (integrated)

Pressure: +0.76 inch W. G.

C. Conditions in this zone during maintenance or plant shutdown could approach the following abnormal ranges:

Humidity (%): 10 - 90
Temperature (°F): 50 – 119

D. Conditions inside the Division 1 & 2 CSCS Pump Rooms when the DG Cooling Water Pumps operate without the ventilation system (Note: this condition will occur should the DG Cooling Water Pumps Operate without the RHR-WS Pumps operating).

The maximum steady-state temperature in the Division 1 & 2 CSCS pump rooms when the DG cooling water pumps operate but the room's ventilation system does not operate is 139.2°F.

TABLE 3.11-28
(SHEET 1 OF 2)

HARSH ENVIRONMENT ZONE H10-SERVICE CONDITIONS
IN THE AUXILIARY BUILDING

A. INSIDE CONDITIONS WITH THE VENTILATION SYSTEM OPERATING

TEMPERATURE (° F)	RELATIVE HUMIDITY (%)	DURATION NO. OF DAYS IN 40 YEARS PLANT LIFE
106	34	10
105	32	97
100	34	275
95	38	540
90	40	812
85	46	1136
80	50	1266
75	50	1167
70	52	1007
65	50	8300
Radiation: 1 x 10^4 rads gamma (integrated)		
Pressure: 0 to –0.4 inch W.G.		

TABLE 3.11-28
(SHEET 2 OF 2)

B. INSIDE CONDITIONS DURING LOSS OF VENTILATION SYSTEM

1) HVAC Equipment Area At Floor El. – 786 feet 6 inches

SUMMER	Time (Hr)	0	100	1350	2400
	Temp (°F)	106	94	86	82
	Rel. Hum. (%)	34	50	64	72
WINTER	Time (Hr)	0	24	1200	2400
	Temp (°F)	65	60	36	30
	Rel. Hum. (%)	50	60	100	100

NOTES:

1. The pressure range inside the zone H10 area is 0 to -0.4 in. W.G.

2. Radiation: 1 x 10^4 rads gamma (integrated).

3. Temperature and relative humidity change linearly from one time step to the next.

TABLE 3.11-29

HARSH ENVIRONMENT ZONE H10
BOUNDING ENVIRONMENTAL CONDITIONS IN AUXILIARY BUILDING
AREAS SUBJECT TO VQ DUCT RUPTURE DURING LOCA

Temperature (°F)	200	138	115	112
Duration	0-11 sec	11 sec – 1 hr	1-24 hr	24 hr to 100 days

Pressure	.715 psig Maximum*
Relative Humidity	100% Maximum*
Radiation	1 x 10^4 rads gamma (integrated)

NOTE: The bounding radiation dose $\geq$ (normal service radiation dose integrated over 40 years + accident dose + 10% margin on the accident dose per IEE 323-1974, Section 6.3.1.5)

- For specific pressure and humidity values and duration refer to basic calculation.

ATTACHMENT 3.B

A GENERIC UPPER BOUNDARY PROBABILISTIC ANALYSIS OF THE EFFECT OF RECIRCULATION PUMP MISSILES ON CONTAINMENT AND EQUIPMENT IN A TYPICAL BWR 5 MARK II GENERAL ELECTRIC BOILING WATER REACTOR NUCLEAR POWER PLANT

1. SCOPE

1.1 This report documents the probabilistic studies performed to evaluate the consequences of recirculation pump overspeed leading to potential pump impeller missile generation following a postulated design basis loss-of-coolant accident (DBA-LOCA) for a typical GE BWR 5, Mark II-containment nuclear power plant.

1.2 This report covers the following:

a. Criteria for determining postulated break locations in recirculation piping

b Probability that a postulated recirculation pump missile would perforate the primary containment or another major piping system in the containment or damage an inboard main steam isolation valve

c. Effect of pipe impact restraints installed at all bottom horizontal-to-vertical recirculation system elbows. (See Section 11 for definitions.)

2. CONCLUSIONS

2.1 Application of break location criteria in compliance with the intent of USAEC Regulatory Guide 1.46 (Paragraph 3.1 of this document) indicates no damage to primary containment, any major piping system, or to an inboard main steam isolation valve. Absence of damage is due to the fact that trajectories of postulated missiles do not intersect these systems.

2.2 If more conservative break location assumptions are made (i.e., postulated breaks occur at all fittings and at equipment-piping circumferential welds), the relative probability of impact and perforation of, or destructive damage to critical targets is shown in Table I.

3. BREAK LOCATION CRITERIA

3.1 <u>USAEC Criteria (Regulatory Guide 1.46)</u>

3.1.1 USAEC requirements for postulations of the most probable break locations, where the consequences of the dynamic effects of pipe breaks must be considered, are summarized below for the class 1 austenitic stainless steel recirculation piping systems.

a. Assume a break location at all terminal ends of the piping run or branch.

TABLE I

PROBABILITY OF DAMAGE RESULTING FROM POSTULTED BREAKS
(per LOCA)

	With Recirc. System Pipe Impact Restraints (Group 2B) (2)	Without Recirc. System Pipe Impact Restraints (Group 2A) (2)
Loss of Containment Integrity		
USAEC Regulatory Guide 1.46	(1)	(1)
Pipe Break Criteria	None	None
Postulated Breaks per Paragraph 3.2	4.2×10^{-6}	5.6×10^{-6}
Perforation of , or Damage to, Vital Equipment	(1)	(1)
USAEC Regulatory Guide 1.46 Pipe Break Criteria	None	None
Postulated Breaks per Paragraph 3.2		
-Main Steam Line	0.01×10^{-3}	0.01×10^{-3}
-Feedwater Line	1.07×10^{-3}	1.43×10^{-3}
-High Press Core Spray Line	0.04×10^{-3}	0.06×10^{-3}
-Low Press Core Spray Line	0.04×10^{-3}	0.06×10^{-3}
-Low Press Core Injection Line	0.74×10^{-3}	0.94×10^{-3}
-Reactor Core Isolation Cooling Line	0.34×10^{-3}	0.66×10^{-3}
-Main Steam Isolation Valve	0.14×10^{-3}	0.20×10^{-3}
Total	2.38×10^{-3}	3.36×10^{-3}

(1) Per Reference 1.
(2) See Section 4 for definition of groups.

b. Assume a break location at all intermediate locations between terminal ends where the maximum stress range exceeds the 2.4 S as calculated between any two load sets (including zero load set) according to Article NB-3600 of ASME Code, Section III, for upset plant conditions and an Operating Basis Earthquake (OBE) event transient, at locations where the stress range is calculated using Equation (10) of NB-3600, or the cumulative usage factor exceeds 0.1.

3.1.2 If two or more intermediate locations cannot be deter- mined by stress or usage factor limits, a total of two inter- mediate circumferential locations shall be identified on a reasonable basis[1] for each piping run or branch run. 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 will be used.

3.1.3 Break locations identified using USAEC criteria are shown in Figure 1. These locations are based on a representative ASME Code, Section III, piping analysis for a plant using standard plant envelope seismic spectra for the BWR 5 recirculation piping system in a Mark II containment.

3.2 Break Locations at all Fittings and Equipment Circumferential Welds

3.2.1 Break locations identified using these criteria are shown in Figure 1. A significantly higher number of breaks are postulated using these criteria than are postulated using the USAEC criteria. The smaller number of breaks postulated for USAEC criteria results from low thermal-expansion and temperature gradient stresses during operation.

4. GROUPS-PROBABILISTIC MODELS

4.1 The following analysis presents four probabilistic models for evaluation of a typical BWR 5-Mark II plant. Drawing 761E195, Primary Containment, shows the drywell arrangement of the largest BWR 5-Mark II standard plant.

4.2 The four probabilistic models are made up of two groups, each of which is divided into two subgroups. In Group 1, analysis is based on the USAEC pipe break criteria; in Group 2, analysis is based on postulated breaks per Paragraph 3.2. Subgroup A is without pipe impact restraints, such as pipe whip restraints or energy absorbing material, under the bottom horizontal-to-vertical elbows of the

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[1]Reasonable basis shall be one or more of the following:

1. Fitting locations
2. Highest stress or usage factor locations used.

recirculation system to prevent pipes from impacting the concrete floor. (Elbows are at breaks RA/B4C, RA/B5A, and RA/B9A; see Section 5.) Subgroup B has pipe impact restraints installed to limit downward movement of piping to no more than 1 foot.

This restraint movement results in pipe alignment as described in Reference 1 and provides a safe[2] condition for breaks RB3C, RA/B4A, RA/B9C, and RA/B11A (Section 5). Break RB3A could be considered safe, [2] but, to be conservative, it is considered a break that has the potential of expelling a high speed missile.

5. BREAK LOCATIONS

5.1 <u>Definitions</u>

5.1.1 RA/BlA means break RAlA on the "A" loop and break RBlA on the "B" loop.

5.1.2 For the purpose of this report, the "B" loop is defined as the recirculation loop having an RHR pipe connection in the suction line (postulated breaks RB3A, RB3B, and RB3L).

5.2 <u>Break Locations Per USAEC Regulatory Guide 1.46 Criteria (Group 1)</u>

5.2.1 USAEC Regulatory Guide 1.46 breaks are identified on Figure 1. They are RA/BlA, RA/BRl, RA/BR2, RA/BH3, RA/BH4 and RA/BH5 at the terminal ends of the piping runs and RA/B2A, RA/BB, RA/B10L, RA/BH1, RA/BH2, RA/BH3, RA/BH4, and RA/BHS. All of these breaks are safe. (See Reference 1.)

5.3 Break Locations at all Fittings and Equipment Circumferential Welds (Group 2)

5.3.1 The location of possible breaks is limited, for all practical purposes, to those places identified on Figure 1. The piping geometry and pipe whip restraint locations are such that breaks at RA/BlA, RA/B2A, RB3L[3], RA/B4C, RA/B5A, RA/B6A, RA/B6C, RA/B7A, RA8A, RA8B, RA9A, RA9B, RA/B10L[3], RA/BllL[3], RA/B B[3], RA/BHl [3], RA/BH2[3], RA/BH3[3], RA/BH4[3], RA/BH5[3], RA/BRl [3] RA/BR2[3], RA/BR3[3], RA/BR4[3], and RA/BR5[3] are safe per Reference 1. Breaks RA/B8B, RA/B9A, RA/B9B, RA/B9C, RA/BllA, and RA/BllC are safe because a break in the discharge line beyond the Flow Control Valve will not, under normal running condition, produce destructive overspeed of the recirculation pump impeller (Reference 2). Also per Reference 2, no longitudinal break will produce destructive overspeed of the recirculation pump impeller regardless of location, meaning that they all are

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[2]See definition for "safe", Section 11.

[3]Not a design basis LOCA break

safe. The foregoing are basic assumptions for this probabilistic analysis and limit the scope of this analysis to considerations of the effects of missiles leaving the recirculation piping at break locations RA/C2C[4], RB3A[4], RB3C[4], RA/B4A[4] and RB8A. The difference between the piping runs of General Electric Mark I and Mark II containments has resulted in the addition of breaks RA/B8B and RA/B9B and in the reclassification of break RB8A from safe to unsafe due to lower main steam penetrations in the Mark II containment. Reference 1 is based on a General Electric BWR 4-Mark I nuclear power plant.

5.3.2 The possible occurrences that could result from missiles being formed and expelled from one of the preceding breaks are: penetration of the primary containment, penetration of a main steam line, penetration of a feedwater line, destruction of an inboard main steam line isolation valve operator, and penetration of an ECCS line. The probability of each of these occurrences is estimated in the event of a design basis LOCA occurring at each of the break locations from which a missile could be expelled. Since these occurrences have widely different consequences, they are evaluated separately. In each case, the probability of the occurrence is given.

6. ESCAPE PROBABILITY

6.1 Given that a design basis LOCA occurs at one of the 34 Group 2 break locations (without an asterisk) shown on Figure 1, the probability of the break occurring at one specific location in loop A or B is approximately 0.03 (1/34). Destructive overspeed and breakup of the recirculation pump impeller may break up the impeller in just about any manner, with various possible sizes of missiles being formed. However, the most likely breakup is believed to be with five main blade segments, five main shroud segments, and numerous smaller pieces. Of these, the large shroud segments are the only ones having sufficient mass to achieve enough energy to cause significant damage. The size and shape of these shroud segments are such that it would be very difficult for one to leave the pump and enter the piping system; i.e., the missile must leave the pump with its longitudinal axis aligned within a few degrees of a specific orientation. There is no apparent "funnelling" effect in the geometry at the pump-pipe interface. Assigning a conservative probability of 0.10 to this occurrence, and assuming (in the worst case) that each of the five possible large shroud segments would exist and would have an independent opportunity to escape the pump, results in a total probability of 0.4 (=1-0.9) for missiles of sufficient energy to enter the piping system, given a design basis LOCA break. For each specific location, the probability of this occurrence is

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[4]These break locations are on the suction side of the recirculating piping. In Reference 1, it is stated that these breaks would not open sufficiently to allow missiles to escape. Subsequent analysis indicates that these breaks may open sufficiently to expel missiles.

0.012, (0.03 x 0.4) given a design basis break at any of the 34 break locations in the recirculation loops.

7. PROBABILITY OF CONTAINMENT PENETRATION

7.1 It is necessary to examine each specific break location to estimate the probability of an escaping missile hitting the containment on a primary impact with a high enough angle of incidence to penetrate (at least partially). With a break occurring at break RB8A, the velocity of the missile is sufficiently low to avoid penetration of the primary containment regardless of impact angles. With a break occurring at break RA/B2C, RB3A, RB3C or RA/B4A, the impact angle is so small (less than 30°) that the missile ricochets without penetration.

7.2 The probability of a secondary impact on the steel containment is calculated as follows:

7.2.1 It is estimated that a missile has a:

a. 0.012 probability of being created and expelled at any one of the break locations for group 2.

b. 0.25 probability of clearing structures without loss of a major part of its energy.

c. 0.50 probability of staying clear of nonvital lines and equipment without loss of a major part of its energy.

d. 0.75 probability of impacting a wall, the ceiling, or the floor with high loss of energy (0.25 probability of low or no loss of energy if a wall, ceiling, or floor is impacted).

e. 0.25 probability that the surface mentioned in (d) above is the steel containment or liner.

f. 0.50 probability that the lost energy is sufficient for perforation if the missile had enough energy to begin with to perforate. (Half of the high-loss impacts will ricochet.)

g. 0.01 probability that the impact will be on a point or an edge lined up for penetration. (The missile will have already impacted another surface and would therefore be rotating at random, making it highly unlikely that a penetration would occur.)

h. 1.00 probability that the missile will clear vital lines at primary or secondary impacts.

7.2.2 All above probabilities are mutually exclusive. The probability that a ricocheting missile will escape from one specific break and be lined up for perforation is therefore 0.14×10^{-5}.

7.2.3 The probability must be multiplied by the number of breaks where the missile might have enough initial energy to penetrate the steel containment had the impact been at a sufficiently large impact angle. For Group 1-A this number is 4 (RA/B2C, RB3A and RB3C). For Group 1-B, this number is 3 (RA/B2C and RB3A).

7.2.4 The probability for loss of containment integrity for Group 2-A is 0.56×10^{-5}. For Group 2-B it is 0.42×10^{-5}. (See Table I.)

7.2.5 The preceding probabilities are based on the assumption that after a damaging primary impact, a missile does not have sufficient energy (velocity) to penetrate the containment.

8. PROBABILITY OF PERFORATION OF A PIPE

8.1 Perforation of a pipe depends on pipe wall thickness, impacting missile velocity, shape and material strength of missile, and material strength and stiffness of target. The formula used in this report to calculate minimum material thickness required to resist penetration was developed by the Stanford Research Institute (SRI). The plate used by SRI was held very rigidly and the missiles used were made of tool steel (Reference 1). The recirculation missile is made of relatively soft stainless steel and the pipes are flexible relative to the SRI target plate; this makes the calculated minimum material thickness of the pipe walls very conservative.

8.2 Impacts are divided into three categories (Paragraph 8.3) as follows:

a. Impact Category 1 with a .5 probability of perforation

b. Impact Category 2 with a .05 probability of perforation

c. Impact Category 3 with a 0.00 probability of perforation

These categories are conservative to begin with and will be used in a conservative manner.

8.3 The shape of the impacting missile is such that at least 50 percent of all impacts will be grazing (Category 1 impact). In the cases where the wall thickness of the pipe is about the same as the minimum required steel plate thickness calculated using the SRI formula, it is estimated that 90 percent of the remaining impacts will hit the pipe with a flat surface with no penetration. The probability of this Category 2 impact is .5 x (1.0 - .9) or 0.05. In the cases where the wall thickness of the pipe is thicker than the wall thickness calculated using the SRI formula, no penetration will take place and we have a Category 3 impact.

9. PROBABILITY OF DAMAGE TO VITAL EQUIPMENT DUE TO A PRIMARY IMPACT

9.1 To evaluate the effect of escaping missiles making first impact on piping and other objects in the containment, it is necessary to examine the geometry at each specific location where a break might occur (Table II).

9.2 The cone of dispersion from a break at location RB8A or RB8B does not include any vital pipe lines. Therefore, a missile ejected from either one of these breaks is of no consequence as far as possible direct impact on vital piping or equipment is concerned.

9.3 At location RA2C, an escaping missile could have a direct hit on part of the feedwater line or part of the LPCI line. The probability of hitting the feedwater line is estimated to be 0.05 with a 0.5 probability of perforation if hit. The estimated probability of impacting the LPCI line is 0.05 with a 0.5 probability of perforation if hit.

9.4 At location RB2C, an escaping missile could hit a part of a feedwater line or part of an LPCI line. The probabilities of hit and perforation for the feedwater line are 0.05 and 0.5 respectively; for the LPCI line, the probabilities are 0.03 and 0.5.

9.5 At locations RB3A and RB3C, a part of the feedwater line, a part of the LPCI LINE, and a part of the RCIC line might be in the way of the postulated missile. The probabilities of hit and perforation respectively for the feedwater line are 0.05 and 0.5; for the LPCI line they are 0.03 and 0.5; and for the RCIC line they are 0.05 and 0.5.

9.6 The geometry at breaks RA/B4A is different from the other suction line breaks due to expected deflection of the suction pipe below the break and the reduced velocity due to passage through a 90° elbow. The cone of dispersion from RA4A would include a section of main steam line, feedwater line, and LPCI line. The estimated probability of a Category 3 impact on a main steam line is 0.15 with no perforation if hit. The estimated probability of a Category 3 impact on the feedwater line is 0.10

TABLE II

EFFECT OF MISSIBLES ON MAJOR PIPING THROUGH PRIMARY IMPACTS*

Location Of Break	Thickness of Rigid Steel Plate 90°Impact (Inch)	Missile Velocity (Ft/sec)	Probability of Damage from Direct Impact (0.012 x Probability of Hit x Probability of Perforation) (X 10)$^{-3}$						
			MSL	FW	HPCS	LPCS	LPCI	RCIC	MSIV
RA 2C*	1.10	206	-	0.30	-	-	0.30	-	-
4A	.37	80	0.00	0.00	-	-	0.00	-	-
RB 2C*	1.10	206	-	0.30	-	-	0.18	-	-
3A*	.84	160	-	0.30	-	-	0.18	0.30	-
3C	.84	160	-	0.30	-	-	0.18	0.30	-
4A	.37	80	-	0.00	-	-	0.00	0.00	-
8A*	.42	88	-	-	-	-	-	-	-
Total Group 2A	-	-	0.00	1.20	0.00	0.00	0.84	0.60	0.00
Total Group 2B	-	-	0.00	0.90	0.00	0.00	0.66	0.30	0.00

* Given a design basis LOCA at any of 34 locations in the recirculation piping system. The 0.012 probability of the creation and escape of a high energy missile is included.

* Only break locations with an asterisk can expel a missile if the lower elbows of the recirculation system are prevented from impacting the floor (subgroup B breaks).

with a 0.00 probability or perforation if hit. There is a 0.05 probability of hitting the LPCI line with a 0.00 probability of perforation from a Category 3 impact.

9.7 A missile expelled from RB4A might hit a feedwater line, the LPCI line, or the RCIC line. The probabilities for hit and perforation for the feedwater line are 0.05 and 0.00 respectively; for the LPCI line .03 and .00; and 0.05 and 0.00 for the RCIC line.

9.8 Due to the low location of the main steam isolation valve and the pipe whip restraints at break location RB8B, no missile will make a direct hit on an MSIV.

10. PROBABILITY OF DAMAGE TO VITAL EQUIPMENT DUE TO A SECONDARY IMPACT

10.1 In addition to damage by direct impact, there is also a possibility of damage to vital equipment from missiles that might ricochet (Table III). It is very difficult to estimate the probability of secondary impact realistically, but it is possible to estimate a maximum limit based on conservative assumptions.

10.2 The drywell is heavily occupied by structures, platforms, ladders and nonvital equipment. It is estimated that a missile, not hitting a vital line, will have a probability of:

a. Less than 0.25 to clear structures without loss of the major part of its energy.

b. Less than 0.50 to stay clear of nonvital equipment without loss of a major part of its energy.

c. Less than 0.25 not to transfer major part of its energy to a wall, the ceiling, or the floor of the drywell.

10.3 The probability that the missile will ricochet and then impact a vital line is thus less than 0.25 x 0.50 x 0.25 = 0.031 since these values are mutually exclusive.

10.4 The total probability of a secondary impact on a vital line is the product of the probabilities that the missile will not be stopped by a primary impact on a vital line and the probability that the missile will clear structures and the building without loss of a major part of its energy. The probability that the missile will not be stopped on a primary impact on a vital line, Pv, is the probability that the missile will either miss the vital lines completely or will have a grazing, primary impact on one of the primary lines in the cone of dispersion. (The missile can only have one primary impact.) This probability is 1.0 minus the sum of the products of probabilities for impacts of vital lines (Paragraph 9) and for grazing impact (0.5). The probability that the missile will not be stopped by structures or

TABLE III

EFFECT OF SECONDARY IMPACT (RICOCHET) ON MAJOR PIPING SYSTEMS

Location of Break	Stiff Plate Thickness @ 90° Impact (Inches)	Missile Velocity (ft/sec)	Prop. Of Secondary Impact on Vital Line	No. Of Lines or Valves(2) In Range/Impact Category: MS Line	FW Line	HPCS Line	LPCS Line	LPCI Line	RCIC Line	MSI Valve	Total No. Of Lines and Valves	Prob. Of Hitting a Line (1.0/No. of Lines)	Total Probability of Secondary Impact (3) with Penetration of Pipe (.012 x Prob of Ricochet With Impact x prob of hitting line x impact category prob x No. Of Lines) ($x10^{-5}$): MS	FW	HPCS	LPCS	LPCI	RCIC	MSIV
RRA 2C(1)	1.10	206	.031 x .95	2/2	3/1	0/1	0/1	2/1	0/1	4/1	11	.091	0.32	4.83	-	-	3.22	-	6.43
4A	.37	80	.031 x .85	2/3	3/3	0/3	0/3	2/3	0/3	4/1	11	.091	0.00	0.00	-	-	0.00	-	5.76
RB 2C (1)	1.10	206	.031 x .96	2/2	3/1	1/1	1/1	1/1	0/1	0/1	9	.112	0.40	6.00	2.00	2.00	2.00	2.00	-
3A(1)	.84	160	.031 x .94	2/3	3/1	1/1	1/1	1/1	1/1	0/1	9	.112	0.00	5.88	1.96	1.96	1.96	1.96	-
3C	.84	160	.031 x .94	2/3	3/1	1/1	1/1	1/1	1/1	0/1	9	.112	0.00	5.88	1.96	1.96	1.96	1.96	-
4A	.37	80	.031 x .94	2/3	3/3	1/3	1/3	1/3	1/3	0/1	9	.112	0.00	0.00	0.00	0.00	0.00	0.00	-
8A(1)	.42	88	.031 x 1.0	4/3	2/3	0/3	0/3	0/3	0/3	4/1	10	.100	0.00	0.00	-	-	-	-	7.44
Total Group 2-A													0.72	22.59	5.92	5.92	9.14	5.92	19.63
Total Group 2-B													0.72	16.71	3.96	3.96	7.18	3.96	13.87

(1) Subgroup B Breaks

(2) This is an inventory of the specific items of vital equipment that could be reached by a ricocheting missile.

(3) Given a design basis LOCA, but including a probability of 0.012 that a large missile will be expelled from each specific break.

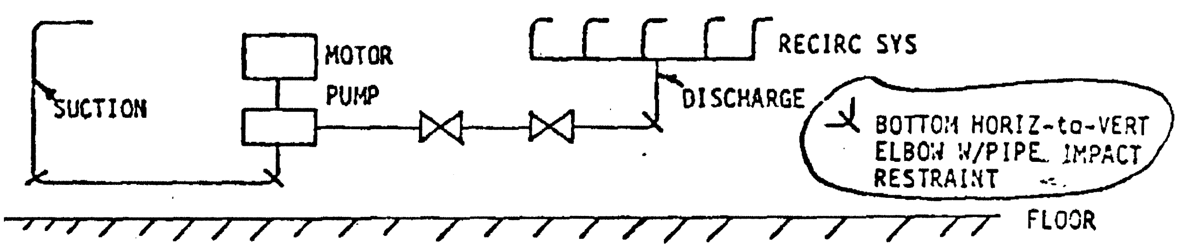
the building is computed in Paragraph 10.3 to be 0.031. Thus, the total probability of a secondary impact on a vital line, giving the expulsion of a missile, is Pv x 0.031 (Table III). The probability of a ricochet and impact on a vital line or a main steam isolation valve is estimated to be of equal value.

10.5 The total probability of penetration of a vital line is the sum of the probability of damage from a primary impact and the probability of damage from a secondary impact if these two probabilities can be assumed to be mutually exclusive; that is, if it is assumed that a missile cannot damage a line on primary impact and still have sufficient energy to damage another line; this seems to be a very reasonable assumption. The combined probabilities are listed in Table I.

11. DEFINITIONS

a. Pipe Impact Restraint is a device to limit pipe motion to 12 inches.

b. Bottom Horizontal-to-Vertical Elbow



c. Safe. A break is considered safe if the postulated missile

1. Is contained within the piping system

2. May leave the piping system at a low velocity insufficient to perforate the containment or an essential piping system

3. Will impact a nonessential target which does not escalate the consequences of the accident

d. Unsafe. Is used in the meaning of Not Safe

12. REFERENCES

1. J.L. Grimaldi, "Analysis of Recirculation Pump Overspeed in a Typical GE BWR, "General Electric Report NEDO 10677, October, 1972.

2. Letter, J. L. Grimaldi to G. A. Esswein, of March 10, 1975.

ATTACHMENT 3.C

CATEGORY I CONCRETE MASONRY WALLS

3.C.1 Description

Concrete masonry walls in Category I structures are used as non- load bearing walls and are not included as part of shear wall system for the Category I structures. Masonry walls are relied upon only as interior partition walls and are separated from the floor above by a gap.

Masonry walls in Category I structures are constructed as single or multi-wythe grouted or solid block walls with full mortar bedding of the units using running bond construction. No cavity wall construction is allowed. Wythes are bonded together with continuous solid or grouted masonry header courses, with full mortar collar joints and by continuous truss type reinforcement which overlaps the adjacent wythes every second course.

No major piping or equipment is attached to the Category I masonry walls. Attachments which are allowed include small bore piping, instrument lines, conduits, junction boxes, etc., and are made with expansion anchors or with through bolt plate assembly.

3.C.2 Applicable Codes, Standards and Specifications

Masonry walls are designed to conform to the National Concrete Masonry Association's "Specifications for the Design and Construction of Load Bearing Masonry," 1979, which is in general agreement with the Uniform Building Code - 1979 and ACI 531-79.

3.C.3 Loads and Load Combinations

Masonry walls in Category I structures are designed for loads and load combinations as given in Table 3.C-1. These walls are not subjected to design loads, such as wind, tornado, missile, pipe whip, and jet impingement loads.

3.C.4 Analysis Method and Design of Concrete Masonry Walls

Masonry walls are designed using working stress principles and are analyzed based on conventional elastic methods. Design is made using the nominal masonry unit size. Horizontal joint reinforcement is ignored in the flexural design of the masonry walls, except for a few walls where the reinforcement is considered in the design.

Dynamic lateral loads are determined by an equivalent static method using the expression:

$$W_D = g_w\, W_W + g_a\, W_a$$

W_D = Dynamic lateral load

W_W = Weight of concrete masonry wall

W_a = Uniform or concentrated attachment load on the wall

g_w = Wall acceleration using appropriate damping values discussed below.

g_a = Peak acceleration for attachment loads using appropriate damping values discussed below.

The natural frequency of concrete masonry walls has been determined using standard expressions for single degree of freedom systems and using the section properties of the wall based on the nominal masonry unit sizes. Frequency calculations have been based on moment of inertia of an uncracked section because applied moments are always less than the moment capacities of uncracked sections.

The walls have been assumed as simply supported spanning horizontally or vertically or as horizontal cantilevers, as applicable. Steel columns embedded in the wall provide lateral support for out-of-plane loads for horizontally spanning walls.

The safety related concrete masonry walls at LaSalle County have been designed using damping values of 4% for OBE and SRV load combinations, and 7% for OBE or SSE with SRV and LOCA load combinations. The response spectra used in the analysis has been based upon the floor at the bottom of the wall. CECo has voluntarily reassessed the concrete masonry walls using the following more stringent criteria:

A. Wall Accelerations

1. Damping values of 2% for OBE and SRV load combinations and 4% for OBE or SSE with SRV and LOCA load combinations were used.

2. The following frequency range to account for any material variation and other uncertainties affecting the response of the wall was used:

Solid/Grouted Units	0.9f - 1.1f
Hollow Units	0.8f - 1.0f

where

f	=	frequency determined per requirements of Section 6.3 based on $E_M = 1000\ f'_M$
E_M	=	Modulus of elasticity for concrete masonry
f'_M	=	Masonry compressive strength. A value of 1350 psi is used for LaSalle County Station

3. The design "g" value was determined by reading the largest value within the frequency range from the response spectra curves for each floor elevation at the top and bottom of the wall elevations and using the average of the two maximum values.

4. The design value of "g" was increased by 1.05 to account for participation of higher modes when wall frequency was less than 33 cps.

B. Accelerations for Attachment Loads

1. Damping values of 2% for OBE and 4% for SSE load combinations were used for attachment loads.

2. The peak "g" value at each floor elevations corresponding to top and bottom of the wall elevations was used to determine the design "g" value by taking the average of the two "g" values.

The design moments have been obtained considering a 12 inch wide beam strip. The walls have been assumed as simply supported or horizontally cantilevered, as applicable, with due consideration to the boundary conditions.

Structural steel columns have been used to provide lateral support for the masonry walls for out-of-plane loads, thereby creating horizontally spanning simply supported conditions.

The structural steel columns are not subject to any load in the vertical direction as the top connections of the columns have been provided with vertical slotted holes.

No overstress factor has been used in the design of masonry walls for load combinations containing OBE seismic loads which is in compliance with SEB Interim Criteria, Rev. 1, July 1981.
All concrete masonry walls have been designed for out-of-plane seismic loadings. Vertical seismic acceleration is less than 1.0g for all of these walls, thus causing no net tension on the wall.

The applied moment due to out-of-plane dynamic loads and attachment loads is always less than the uncracked moment capacity of the masonry wall. As such, moment of inertia of the uncracked section has been used for frequency calculations and hence, the dynamic response of the wall.

The local pull-out effect due to an attachment load has been considered in the design. This design condition is not critical for the structural integrity of the wall.

The masonry walls have been analyzed as rigid for their in-plane behavior. In-plane loads have not been calculated for each wall.

Out-of-plane drift effects due to relative displacement of one floor with respect to the other are not imposed on the masonry walls at the LaSalle County Station for the following reasons:

A. There is a 1" gap between the top of the walls and the underside of the floor diaphragms above

B. The top connections of the masonry lateral support steel columns are pinned connections.

In-plane drift effects have been evaluated for the masonry walls with the following conditions:

A. As mentioned earlier in Section 2.0, masonry walls are not part of the primary vertical or lateral load resisting system. They are non-load bearing, interior partition walls.

B. In-plane interstory drift is an imposed displacement on a masonry wall, and the resultant in-plane load is, therefore, a function of the in-plane shear stiffness of the masonry wall.

The in-plane stiffness is unpredictable, therefore a strain criteria, rather than stress criteria, is more reliable for evaluating drift effects. Shear strain values of 0.001 under SSE conditions and 0.0006 (= 0.001/1.67) under OBE conditions have been selected as acceptable maximum allowable values for the safety related concrete masonry walls at LaSalle County. The SSE allowable strain corresponds to initiation of cracking in masonry, and not the failure of the wall. Therefore, the criteria is conservative.

The maximum shear strain in safety related masonry walls at LaSalle County under SSE conditions is 0.0004"/" which corresponds to the maximum strain in the reinforced concrete shear walls. This strain is significantly less than 0.001"/".

3.C.5 Structural Acceptance Criteria

Masonry walls are designed to conform to the requirements of the National Concrete Masonry Association "Specification for the Design and Construction of Load Bearing Concrete Masonry," 1979. No overstress factor of 1.67 is used. Allowable stresses for inspected workmanship are used for masonry wall design.

Five hollow–block masonry walls in the plant do not meet the above NCMA allowables for the Extreme Environmental and Abnormal/Extreme Environmental load cases. These walls are qualified using allowables determined from testing performed at the Clinton Power Station. This testing determined a modulus of rupture of a hollow-block masonry wall (f_r = 250 psi for horizontally spanning walls, f_r = 125 psi for vertically spanning walls). The allowable flexural stress is calculated using this modulus of rupture decreased by a factor of safety of 2.5. This methodology is used on walls 1WAR673-006, 1WAR710-005, 1WAR786-003, 1WAD736-005 and 2WAR694-004.

The same allowable (i.e., the Clinton test values and a factor of safety of 2.5) is used for the masonry walls of the Unit 1 and 2 VR Exhaust Plenums for the Abnormal, Abnormal/Severe Environmental and Abnormal/Extreme Environmental load cases. The walls of the Unit 1 and 2 VR Exhaust Plenums for which this allowable is used are 1WAA786-008, 1WAA786-009, 1WAA786-011, 2WAA786-012, 2WAA786-013, 2WAA786-020, 1WAA796-001, 1WAA796-002, 2WAA796-001 and 2WAA796-002.

The wall support steel for the VR Exhaust plenum walls for the Abnormal, Abnormal/Severe Environmental and Abnormal/Extreme Environmental load cases are designed to AISC allowables stresses increased by a factor of 1.6. In cases where this allowable can not be met, and the section in question can fully develop its plastic moment, these member are qualified using a maximum ductility ratio of 10.

3.C.6 Materials

The following concrete masonry materials are used:

a.	Hollow Concrete Masonry Blocks:	ASTM C90, Grade N-1
b.	Solid Concrete Masonry Blocks:	ASTM C140, Grade N-1
c.	Grouted Masonry Blocks:	Hollow blocks as per ASTM C90 Grade N-1 and grout to conform to ASTM C476
d.	Mortar:	ASTM C270, Type M
e.	Reinforcement for Concrete Masonry:	Truss reinforcement, ASTM A82 with fy = 65 ksi

TABLE 3.C-1

LOAD COMBINATION TABLE FOR CATEGORY / CONCRETE MASONRY WALLS

LOAD CATEGORY	D	L	P_O	P_{HELB}	E_O	E_{SS}	SRV	LOCA	ALLOWABLE STRESS
				LOAD FACTORS					
Normal	1.0	1.0	1.0				1.0		NCMA*
Severe Environmental	1.0	1.0	1.0		1.0		1.0		NCMA
Abnormal	1.0	1.0		1.0					NCMA x 1.67***
	1.0	1.0					1.25	1.25	
Extreme Environmental	1.0	1.0	1.0			1.0	1.0		NCMA x 1.67***
Abnormal/Severe Environmental	1.0	1.0		1.0	1.1		1.1	1.1	NCMA x 1.67**,***
Abnormal/Extreme Environmental	1.0	1.0		1.0		1.0	1.0	1.0	NCMA x 1.67**,***

Load Symbols are defined as follows:

- D = Dead load of masonry wall including attachment load
- L = Live load
- P_o = Operating pressure differential across a masonry wall
- P_{HELB} = Short-term differential pressurization load on the VR plenum masonry walls resulting from a HELB in the Main Steam Tunnel and a non-instantaneous closure of the protection dampers
- E_o = Operating Basis Earthquake (OBE)
- E_{ss} = Safe Shutdown Earthquake (SSE)
- SRV = Building filtered loads associated with safety/relief valve discharge, where applicable
- LOCA = Building filtered loads associated with loss-of-coolant accident, where applicable

Notes:

* National Concrete Masonry Association "Specification for the Design and Construction of Load Bearing Concrete

** Earthquake and HELB components can be combined using the SRSS method.

*** Certain walls have increased allowables, see Section 3.C.5 for details